Chapter 9: Auxiliary and Emergency Systems

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CHAPTER 9
AUXILIARY AND EMERGENCY SYSTEMS

The Auxiliary and Emergency Systems are supporting systems required to insure the safe operation or servicing of the Reactor Coolant System (RCS) (described in Chapter 4).

In some cases the dependable operation of several systems is required to protect the RCS by controlling system conditions within specified operating limits. Certain systems are required to operate under emergency conditions.

The systems considered in this Section are:

- Chemical and Volume Control System

  This system provides for boric acid injection, chemical additions for corrosion control, reactor coolant cleanup and degasification, reactor coolant makeup, reprocessing of water letdown from the Reactor Coolant System, and reactor coolant pump seal water injection.

- Auxiliary Coolant System

  This system provides for transferring heat from reactor plant coolant to the Service Water System and consists of the following three systems:

  a. The Residual Heat Removal (RHR) System removes the residual heat from the core and reduces the temperature of the RCS during the second phase of plant cooldown.

  b. The Spent Fuel Pool Cooling System removes the heat generated by spent fuel elements stored in the spent fuel pool.

  c. The Component Cooling System removes heat from the RCS, via the RHR System, during plant shutdown, cools the letdown flow to the Chemical and Volume Control System during power operation, and provides cooling to dissipate waste heat from various reactor plant components and the boric acid and waste evaporators.

- Sampling System

  This system provides the equipment necessary to obtain liquid and gaseous samples from the reactor plant systems.

- Facility Service Systems

  These systems include Fire Protection, Service Water, and Auxiliary Ventilation Systems.

- Fuel Handling System

  This system provides for handling fuel assemblies, Rod Cluster Control Assemblies (RCCAs) and material irradiation specimens.

- Equipment and System Decontamination Processes

  These procedures provide for the decontamination of equipment, tools, and personnel.

9.1 GENERAL DESIGN CRITERIA

Criteria, which are specific to one of the auxiliary or emergency systems are listed and discussed in the appropriate system design basis subsection below. Criteria which apply primarily to other systems (and are discussed in other Sections) are also listed and cross-referenced below because details of closely related systems and equipment are given in this Section.

9.1.1 Related Criteria

9.1.1.1 Reactivity Control Systems Malfunction

Criterion: The reactor protection systems shall be capable of protecting against any single malfunction of the reactivity control system, such as unplanned continuous withdrawal (not ejection or dropout) of a control rod, by limiting reactivity transients to avoid exceeding acceptable fuel damage limits (GDC 31).

As described in Chapter 7 and justified in Chapter 14, The Reactor Protection Systems are designed to limit reactivity transients to DNBR > 1.30 due to any single malfunction in the deboration controls.

9.1.1.2 Engineered Safety Features Performance Capability

Criterion: Engineered Safety Features such as the emergency core cooling system and the containment heat removal system shall provide sufficient performance capability to accommodate the failure of any single active component without resulting in undue risk to the health and safety of the public (GDC 41).

Each of the auxiliary cooling systems which serves an emergency function provides sufficient capability in the emergency mode to accommodate any single failure of an active component and still function in a manner to avoid undue risk to the health and safety of the plant personnel and the public.

9.1.1.3 Containment Heat Removal Systems

Criterion: Where an active heat removal system is needed under accident conditions to prevent exceeding containment design pressure this system shall perform its required function, assuming failure of any single active component (GDC 52).

Each of the auxiliary cooling systems, which serve as emergency function to prevent exceeding containment design pressure, provides sufficient capability in the emergency mode to accommodate any single failure of an active component and still perform its required function.

9.2 CHEMICAL AND VOLUME CONTROL SYSTEM

The Chemical and Volume Control System:

1. adjusts the concentration of chemical neutron absorber for chemical reactivity control;
2. maintains the proper water inventory in the RCS;
3. provides the required seal water flow for the reactor coolant pump shaft seals;
4. processes reactor coolant letdown for reuse of boric acid;
5. maintains the proper concentration of corrosion inhibiting chemicals in the reactor coolant; and
6. keeps the reactor coolant fission product and corrosion product activities to within design levels.

The system is also used to fill and hydrostatically test the RCS.

During normal operation, therefore, this system has provisions for supplying:

1. Hydrogen to the volume control tank
2. Nitrogen as required for purging the volume control tank
3. Hydrazine or pH control chemical, as required, via the chemical mixing tank to the charging pumps suction.

9.2.1 Design Bases

9.2.1.1 Redundancy of Reactivity Control

Criterion: Two independent reactivity control systems, preferably of different principles, shall be provided (GDC 27).

In addition to the reactivity control achieved by the Rod Control System described in Chapter 7, reactivity control is provided by the Chemical and Volume Control System which regulates the concentration of boric acid solution neutron absorber in the Reactor Coolant System. The system is designed to prevent uncontrolled or inadvertent reactivity changes, which might cause system parameters to exceed design limits.

9.2.1.2 Reactivity Hot Shutdown Capability

Criterion: The reactivity control system provided shall be capable of making and holding the core sub-critical from any hot standby or hot operating condition (GDC 28).

The reactivity control systems provided are capable of making and holding the core sub-critical from any hot standby or hot operating condition, including conditions resulting from power changes. The maximum excess reactivity expected for the core occurs for the cold, clean condition at the beginning of life of the core. The full-length RCCAs are divided into two categories comprising control and shutdown groups.

The control group used in combination with boric acid as a chemical shim provides control of the reactivity changes of the core throughout the life of the core at power conditions. This group of RCCAs is used to compensate for short term reactivity changes at power such as those produced due to variations in reactor power requirements or in coolant temperature. The chemical shim control is used to compensate for the more slowly occurring changes in reactivity throughout core life such as those due to fuel depletion, fission product buildup and decay, and the xenon transient associated with power level changes.

9.2.1.3 Reactivity Shutdown Capability

Criterion: One of the reactivity control systems provided shall be capable of making the core sub-critical under any anticipated operating condition (including anticipated operational transients) sufficiently fast to prevent exceeding acceptable fuel damage limits. Shutdown margin should assure sub-criticality with the most reactive control rod fully withdrawn (GDC 29).

The reactor core, together with the reactor control and protection system is designed so that the minimum allowable DNBR is no less than 1.30 and there is no fuel melting during normal operation, including anticipated transients.

The shutdown groups are provided to supplement the control group of RCCAs to make the reactor at least one percent sub-critical ($k(eff) \leq 0.99$) following trip from any credible operating condition to the hot, zero power condition assuming the most reactive RCCA remains in the fully withdrawn position.

Sufficient shutdown capability is also provided to maintain the core sub-critical, with the most reactive rod assumed to be fully withdrawn, for the most severe anticipated cooldown transient associated with a single active failure, e.g., accidental opening of a steam bypass or relief valve. This is achieved with combination of control rods and automatic boron addition via the Safety Injection (SI) System.

9.2.1.4 Reactivity Hold-Down Capability

Criterion: The reactivity control systems provided shall be capable of making the core sub-critical under credible accident conditions with appropriate margins for contingencies and limiting any subsequent return to power such that there will be no undue risk to the health and safety of the public (GDC 30).

Normal reactivity shutdown capability is provided by control rods with boric acid injection used for xenon transients and for plant cooldown. When the plant is at power, the quantity of boric acid available will exceed that quantity required to establish the cold shutdown boron concentration. This quantity will always exceed the quantity of boric acid required to bring the reactor to hot shutdown and to compensate for subsequent xenon decay.

The normal supply of boric acid is maintained in the boric acid storage tanks. The boric acid solution is transferred from the boric acid tanks to the suction of the charging pumps by either boric acid transfer pump. The charging pumps then inject the boric acid solution to the reactor coolant. Any charging pump and any boric acid transfer pump can be energized from diesel generator power. The Refueling Water Storage Tank (RWST) provides a backup supply of borated water that can be aligned to the suction of the charging pumps. The RWST normally has adequate quantity of borated water to place the plant in the cold shutdown condition if necessary. Either boric acid supply to any charging pump will provide sufficient negative reactivity to the reactor coolant to compensate for xenon decay.

Boric acid could be injected to shutdown the reactor independent of the RCCAs, which normally serve this function in the short term. The quantity of acid maintained to establish the cold shutdown boron concentration is more than that needed to initially shutdown the reactor without using RCCAs.

On the basis of the above, the injection of boric acid is shown to afford backup reactivity shutdown capability and provides adequate long term hold-down capability necessary to compensate for xenon transients and for plant cooldown.

9.2.1.5 Codes and Classifications

All pressure retaining components (or compartments of components) which are exposed to reactor coolant, comply with the following codes:

1. System pressure vessels - ASME Boiler and Pressure Vessel Code, Section III, Class C.
2. System valves, fittings, and piping - USAS B31.1, including nuclear code cases.

System component code requirements are tabulated in Table 9.2-1.

The tube and shell sides on the regenerative heat exchanger and the tube side of the excess letdown heat exchanger are designed to ASME Section III, Class C. This designation is based on the following considerations:

1. each exchanger is connected to the RCS by lines equal to or less than 2 inches; and
2. each is located inside the Reactor Containment Vessel.

Analyses show that the accident associated with a 2-inch line break does not result in clad damage or failure. Additionally, previously contaminated reactor coolant escaping from the RCS during such an accident is confined to the Reactor Containment Vessel and no public hazard results.

9.2.2 System Design and Operation

The Chemical and Volume Control System (CVCS), shown in Figure 9.2-1 through Figure 9.2-5, provides a means for injection of the neutron control chemical in the form of boric acid solution, chemical additions for corrosion control, and reactor coolant cleanup, degasification and deboration. This system also adds makeup water to the RCS, reprocesses water letdown from the RCS, and provides seal water injection to the reactor coolant pump seals. Design seal injection to each reactor coolant pump is 8 gpm with 5 gpm leaking through the labyrinth seal into the reactor coolant system. The seal is designed to leak 3 gpm or less back to the CVCS. Materials in contact with the reactor coolant are austenitic stainless steel or equivalent corrosion resistant materials.

System components whose design pressure and temperature are less than the RCS design limits are provided with overpressure protective devices.

System discharges from overpressure protective devices (safety valves) and system leakage are directed to closed systems. Effluents removed from such closed systems are monitored and discharged under controlled conditions.

During plant operation, reactor coolant flows through the letdown line from a loop cold-leg on the suction side of the reactor coolant pump and, after processing is returned to the cold-leg of the loop on the discharge side of the pump via a charging line and through the in leakage in the reactor coolant pump seals. An excess letdown line is also provided for removing coolant from the Reactor Coolant System.

Each of the connections to the RCS has an isolation valve located close to the loop piping. In addition, a check valve is located downstream of each charging line isolation valve. Reactor coolant entering the CVCS flows through the shell side of the regenerative heat exchanger where its temperature is reduced. The coolant then flows through letdown orifices, which reduce the coolant pressure. The cooled, low-pressure water leaves the Reactor Containment Vessel and enters the Auxiliary Building where it undergoes a second temperature reduction in the tube side of the letdown heat exchanger followed by a second pressure reduction by the low-pressure letdown valve. After passing through the letdown filter and one of the mixed bed demineralizers, where ionic impurities are removed, coolant flows through the reactor coolant filters and enters the volume control tank through a spray nozzle.

Hydrogen is automatically supplied, as determined by pressure control, to the vapor space in the volume control tank, which is predominantly hydrogen and water vapor. The hydrogen within this tank is supplied to the reactor coolant for maintaining a low oxygen concentration. If required, fission gases can be removed from the system by venting the volume control tank to the Waste Disposal System.

From the volume control tank the coolant flows to the charging pumps which raise the pressure above that in the RCS. The coolant then enters the containment, passes through the tube side of the regenerative heat exchanger, and is returned to the RCS.

The cation bed demineralizer, located downstream of the mixed bed demineralizers, is used intermittently to control cesium activity in the coolant and also to remove excess lithium, which is formed from B^{10} (n, α) Li^7 reaction.

Boric acid is dissolved in hot water in the batching tank to a concentration of nominal 8.0 percent by weight. The lower portion of the batching tank is jacketed to permit heating of the batching tank solution with low-pressure steam. A transfer pump is used to transfer the batch to the boric acid tanks. Small quantities of boric acid solution are metered from the discharge of an operating transfer pump for blending with reactor makeup water as makeup for normal leakage or for increasing the reactor coolant boron concentration during normal operation. Electric heaters maintain the temperature of the boric acid tanks solution high enough to prevent precipitation. The boric acid piping is heat traced to prevent precipitation.

Excess liquid effluents containing boric acid flow from the RCS through the letdown line and are collected in the holdup tanks. As liquid enters the holdup tanks, the nitrogen cover gas is displaced to the gas decay tanks in the Waste Disposal System through the waste vent header. The concentration of boric acid in the holdup tanks varies throughout core life from the refueling concentration to essentially zero at the end of the core cycle. A recirculation pump is provided to transfer liquid from one holdup tank to another and to recirculate the contents of individual holdup tanks.

Liquid effluent in the holdup tanks is processed as a batch operation. This liquid is pumped through the evaporation feed ion exchangers, which primarily remove lithium hydroxide (Li7OH), and fission products such as long-lived cesium. It then flows through the ion exchanger filter and into the gas stripper/boric acid evaporator package.

The dissolved gases are removed from the liquid by the gas stripper section and are vented to the Waste Disposal System. The liquid effluent from the gas stripper section then enters the evaporator section. The distillate produced in the boric acid evaporator leaves the evaporator condenser and is pumped through a condensate cooler where the distillate is cooled to the operating temperature of the two-evaporator condensate demineralizers. If it is required, evaporator carry over is removed by one of the two-evaporator condensate demineralizers, and then the condensate flows through the condensate filter and accumulates in one of two monitor tanks. The dilute boric acid solution originally in the boric acid evaporator remains as the bottoms of the distillation process and is concentrated to nominal weight 8.0 percent boric acid.

Subsequent handling of the condensate is dependent on the results of sample analysis. Discharge from the monitor tanks is recycled through the evaporator condensate demineralizers, returned to the holdup tanks for reprocessing in the evaporator train or discharged to the environment with the condenser circulating water, when within the allowable activity concentration as discussed in Chapter 11. If the sample analysis of the monitor tank contents indicates that it may be discharged safely to the environment, at least two valves must be opened to provide a discharge path. As the effluent leaves, it is continuously monitored by the Waste Disposal System liquid effluent monitor. If an unexpected increase in radioactivity is sensed, one of the valves in the discharge line to the service water discharge header closes automatically and an alarm sounds in the control room.

Boric acid evaporator bottoms are discharged through a concentrate filter to the concentrates holding tank. Solution collected in the concentrates holding tank is sampled and then transferred to the boric acid tanks if analysis indicates that it meets specifications for use as boric acid makeup. Otherwise the solution is pumped to the holdup tanks for reprocessing by the evaporator train.

The concentrated solution can also be pumped from the evaporator to the Waste Disposal System to be placed in containers. These containers can then be stored at the plant site for ultimate shipment off-site for disposal.

The deborating demineralizers can be used intermittently to remove boron from the reactor coolant near the end of the core life. When the deborating demineralizers are in operation, the letdown stream passes from the mixed bed demineralizers, through a deborating demineralizer and into the volume control tank after passing through the reactor coolant filter.

During plant cooldown, when the residual heat removal loop is operating and the letdown orifices are not in service, a flow path is provided to remove corrosion impurities and fission products. A portion of the flow leaving the residual heat exchangers passes through the letdown heat exchanger, letdown filter, mixed bed demineralizers, reactor coolant filter, and volume control tank. The fluid is then pumped, via the charging pump, through the tube side of the regenerative heat exchanger into the RCS. Flow may also bypass the regenerative heat exchanger and letdown orifices, by way of a direct connection from the 1A RHR heat exchanger to the letdown heat exchanger.

9.2.2.1 Expected Operating Conditions

Table 9.2-2 and Table 9.2-3 list the system performance requirements, and data for individual system components. Reactor coolant equilibrium activities are given in Appendix D.

9.2.2.2 Reactor Coolant Activity Concentration

The parameters used in the calculation of the reactor coolant fission product inventory, including pertinent information concerning the expected coolant cleanup flow rate and demineralizer effectiveness, are presented in Appendix D. In these calculations, 1 percent defects are assumed to be present in the fuel rods at initial core loading and are uniformly distributed throughout the core. The fission product escape rate coefficients are therefore based upon an average fuel temperature.

The fission product activity in the reactor coolant during operation with small cladding pinholes or cracks in 1 percent of the fuel rods is computed using the following differential equations:

For parent nuclides in the coolant,

$$\frac{dN_{w_i}}{dt} = D\upsilon_i N_{c_i}\left(\lambda_i + R\eta_i + \frac{B'}{B_o - tB'}\right)N_{w_i}$$

For daughter nuclides in the coolant,

$$\frac{dN_{w_j}}{dt} = D\upsilon_j N_{c_j} - \left(\lambda_j + R\eta_j + \frac{B'}{B_o - tB'}\right)N_{w_j} + \lambda_i N_{w_i}$$

where:

N = population of nuclide

D = fraction of fuel rods having defective cladding

R = purification flow, coolant system volumes per sec.

B_o = initial boron concentration, ppm

B´ = boron concentration reduction rate by feed and bleed, ppm per sec.

η = removal efficiency of purification cycle for nuclide

λ = radioactive decay constant

υ = escape rate coefficient for diffusion into coolant

Subscript C refers to core

Subscript w refers to coolant

Subscript i refers to parent nuclide

Subscript j refers to daughter nuclide

Tritium is produced in the reactor from ternary fission in the fuel, irradiation of boron in the burnable poison rods and irradiation of boron, lithium, and deuterium in the coolant. The deuterium contribution is less than 0.1 curie per year and may be neglected. The parameters used in the calculation of tritium production rate are also presented in Appendix D.

9.2.2.3 Reactor Makeup Control

The reactor makeup control consists of a group of instruments arranged to provide a manually pre-selected makeup water composition to the charging pump suction header or the volume control tank. The makeup control functions are to maintain desired operating fluid inventory in the volume control tank and to adjust reactor coolant boron concentration for reactivity and shim control.

Makeup for normal plant leakage is regulated by the reactor makeup control, which is set by the operator to blend water from the reactor makeup water tank with concentrated boric acid to match the reactor coolant boron concentration.

The makeup system also provides concentrated boric acid or reactor makeup water to either increase or decrease the boric acid concentration in the RCS. To maintain the reactor coolant volume constant, an equal amount of reactor coolant at existing reactor coolant boric acid concentration is letdown to the holdup tanks. Should the letdown line be out of service during operation, the excess letdown would be available which would aid in providing sufficient volume in the pressurizer to accept the amount of boric acid necessary for cold shutdown.

Makeup water to the RCS is provided by the CVCS from the following sources:

1. The reactor makeup water tanks, which provide water for dilution when the reactor coolant boron concentration is to be reduced.
2. The boric acid tanks, which supply concentrated boric acid solution when reactor coolant boron concentration is to be increased.
3. The refueling water storage tank (RWST), which supplies borated water for emergency makeup.
4. The chemical mixing tank, which is used to inject small quantities of solution when additions of hydrazine or pH control chemical are necessary.

The reactor makeup control is operated from the control room by manually pre-selecting the desired makeup composition to the charging pump suction header or the volume control tank in order to adjust the reactor coolant boron concentration for reactivity control. Makeup is provided to maintain the desired operating fluid inventory in the RCS. The operator can stop the makeup operation at any time in any operating mode. The reactor makeup water supply and boric acid transfer pumps are normally lined up for automatic operation as required by the makeup controller.

A portion of the high-pressure charging flow is injected into the reactor coolant pumps between the pump impeller and the shaft seal so that the seals and the lower radial bearing are not exposed to high temperature reactor coolant. The injection flow splits and part becomes the shaft seal leak-off flow and the remainder enters the Reactor Coolant System through a labyrinth seal on the pump shaft. The shaft seal leak-off flow cools the lower radial bearing, passes through the seals, is cooled in the seal water heat exchanger, filtered, and returned to the volume control tank.

Seal water inleakage to the RCS requires a continuous letdown of reactor coolant to maintain the desired inventory. In addition, bleed and feed of reactor coolant are required for removal of impurities and adjustment of boric acid in the reactor coolant.

9.2.2.4 Automatic Makeup

The “automatic makeup” mode of operation of the reactor makeup control provides boric acid solution preset by the operator to match the boron concentration in the RCS. The automatic makeup compensates for minor leakage of reactor coolant without causing significant changes in the coolant boron concentration.

Under normal plant operating conditions, the mode selector switch is set in the “Automatic Makeup” position. A preset low level signal (17 percent) from the volume control tank level controller causes the automatic makeup control action to open the makeup stop valve to the charging pump suction, modulate the concentrated boric acid control valve, open the reactor makeup water control valve and to switch the boric acid transfer pumps to high speed operation. The flow controllers then blend the makeup stream according to the preset concentration. Makeup addition to the charging pump suction header causes the water level in the volume control tank to rise. At a preset high level set-point (27 percent), the makeup is stopped; the reactor makeup water control valve closes, the concentrated boric acid control valve returns to its normal position (open), the makeup stop valve to charging pump suction closes, and the boric acid transfer pumps are returned to their previous mode of operation.

9.2.2.5 Dilution

The “dilute” mode of operation permits the addition of a pre-selected quantity of reactor makeup water at pre-selected flow rate to the RCS. The operator sets the mode selector switch to “dilute,” the reactor makeup water flow controller set point to the desired flow rate, and the reactor makeup water batch integrator to the desired quantity. Upon manual start of the system the makeup stop valve to the volume control tank opens, and the reactor makeup water control valve opens. Makeup water is added to the volume control tank and then goes to the charging pump suction header. Excessive rise of the volume control tank water level is prevented by automatic actuation (by the tank level controller) of a three-way diversion valve, which routes the reactor coolant letdown flow to the holdup tanks. When the preset quantity of reactor makeup water has been added, the batch integrator causes the reactor makeup water control valve and makeup stop valve to close.

9.2.2.6 Alternate Dilute

The “alternate dilute” mode is similar to the dilute mode except the dilution water, after passing through the blender, splits and a portion flows directly to the charging pump suction and a portion flows into the volume control tank via the spray nozzle and then flows to the charging pump suction. The operator sets the mode selector switch to “alternate dilute,” the reactor makeup water flow controller set point to the desired flow rate, the reactor makeup water batch integrator to the desired quantity, and actuates the makeup start. The start signal causes the makeup control to open the makeup stop valve to the volume control tank, the makeup stop valve to the charging pump suction header, and the reactor makeup control valve. Reactor makeup water is simultaneously added to the volume control tank and to the charging pump suction header. This mode is used for load follow and permits the dilution water to follow the initial xenon transient and simultaneously dilute the volume control tank. Excessive water level in the volume control tank is prevented by automatic actuation of the volume control tank level controller which routes the reactor coolant letdown flow to the holdup tanks. When the preset quantity of reactor makeup water has been added, the batch integrator causes the reactor makeup water control valve and the reactor makeup stop valves to close. This operation may be stopped manually by actuating the makeup stop.

9.2.2.7 Boration

The “borate” mode of operation permits the addition of a pre-selected quantity of concentrated boric acid solution at a pre-selected flow rate to the RCS. The operator sets the mode selector switch to “borate,” the concentrated boric acid flow controller set point to the desired flow rate, and the concentrated boric acid batch integrater to the desired quantity. Upon manual start of the system the makeup stop valve to the charging pumps opens, the concentrated boric acid control valve modulates, the boric acid transfer pumps start, if not already running, and the concentrated boric acid is added to the charging pump suction header. The total quantity added in most cases is so small that it has only a minor effect on the volume control tank level. When the preset quantity of concentrated boric acid solution has been added, the batch integrater causes the boric acid control valve and the makeup stop valve to the charging pump suction to return to their normal position, and stops the Boric Acid Transfer Pumps.

The capability to add boron to the reactor coolant is such that it imposes no limitation on the rate of cooldown of the reactor upon shutdown. The maximum rates of boration and the equivalent coolant cooldown rates are given in Table 9.2-2. One set of values is given for the addition of boric acid from a boric acid tank with one transfer and one charging pump operating. The other set assumed the use of refueling water but with two-of-the-three charging pumps operating. The rates are based on full operating temperature and on the end of the core life when the moderator temperature coefficient is most negative.

By manual action of the operator, the boric acid transfer pumps can discharge directly to the charging pump suction and bypass the blender and volume control tank.

9.2.2.8 Alarm Functions

The reactor makeup control is provided with alarm functions to call attention to the following conditions:

- Deviation of reactor makeup water flow rate from the control set point by ±5 gpm.
- Deviation of concentrated boric acid flow rate from the control set point by ± 0.2 gpm.
- Low level (makeup initiation point) in the volume control tank if the level decreases to the low level makeup initiation set point.

9.2.2.9 Charging Pump Control

Three positive-displacement variable-speed-drive-charging pumps are used to supply charging flow to the RCS.

The speed of each pump can be controlled manually or automatically. During normal operation, charging pumps are operated as necessary to maintain inventory in the reactor coolant system. During load changes, the pressurizer level set point is varied automatically to compensate partially for the expansion or contraction of the reactor coolant associated with the T_{avg} changes. Automatic control of charging pump speed does not change rapidly with pressurizer level variations due to the reset action of the pressurizer level controller.

If the pressurizer level increases, the speed of the pump decreases; likewise, if the level decreases, the speed increases. If the charging pump on automatic control reaches the high speed limit, an alarm is actuated and a second pump is manually started. The speed of the second pump is manually regulated. If the speed of the charging pump on automatic control does not decrease and the second charging pump is operating at maximum speed, the third charging pump can be started and its speed manually regulated. If the speed of the charging pump on automatic control decreases to its minimum value, an alarm is actuated and the speed of the pumps on manual control is reduced.

A selector switch is used to choose one-of-two pressurizer water level channel signals for input to a single level controller, which in turn controls charging pump speed.

The charging pump cannot overpressurize the system because:

1. The spray system in the pressurizer can suppress maximum charging flow.
2. Assuming a spray failure (single failure), the power operated relief valves (PORVs) can handle more than the maximum charging flow (2 channels).
3. Assuming three failures (spray and two PORVs), the code relief valves can handle more than the maximum charging flow.

In addition, there are many redundant and diverse alarms to bring the operator's attention to the situation (high pressurizer pressure, high pressurizer water level, pressurizer level deviation, etc.). There is also an alarm on reaching the high or low limits imposed on the automatic control signal to the pump controller. The operator would then go to manual control of the charging pumps.

To ensure that the charging pump flow is always sufficient to meet the RCP seal water requirements, the pump has a control stop which can be preset and does not permit pump flow lower than the specified minimum. This control stop is adjustable to permit higher minimum flow limits to be set if mechanical seal leakage increases during plant life.

9.2.2.10 Components

A summary of principal component data is given in Table 9.2-3.

9.2.2.11 Regenerative Heat Exchanger

The regenerative heat exchanger is designed to recover the heat from the letdown stream by re-heating the charging stream during normal operation. This exchanger also limits the temperature rise, which occurs at the letdown orifices during periods when letdown flow exceeds charging flow by a greater margin than at normal letdown conditions.

The letdown stream flows through the shell of the regenerative heat exchanger and the charging stream flows through the tubes to place the lower pressure requirements on the shell. The unit is a three shell, multiple tube pass heat exchanger made of austenitic stainless steel, and is of all-welded construction.

9.2.2.12 Letdown Orifices

One of the three letdown orifices controls flow of the letdown stream during normal operation and reduces the pressure to a value compatible with the letdown heat exchanger design. Either of two letdown orifices, each 40 gpm, is used to pass normal letdown flow. The third orifice, 80 gpm, is designed to be used for maximum purification flow at normal RCS operating pressure and can pass twice the normal letdown flow. The orifices are placed in and taken out of service by remote manual operation of their respective isolation valves. One or both of the standby orifices may be used in parallel with the normally operating orifice in order to bring the letdown flow up to normal when the RCS pressure driving force is below normal. This arrangement provides a full standby capacity for control of letdown flow. Each orifice consists of bored pipe made of austenitic stainless steel.

9.2.2.13 Letdown Heat Exchanger

The letdown heat exchanger cools the letdown stream to the operating temperature of the mixed bed demineralizers. Reactor coolant flows through the tube side of the exchanger while component-cooling water flows through the shell. The letdown stream outlet temperature is automatically controlled by a temperature control valve in the component cooling water outlet stream. The unit is a multiple-pass tube-and-shell heat exchanger. All surfaces in contact with the reactor coolant are austenitic stainless steel, and the shell is carbon steel. Since the maximum operating temperature for the components in question is considerably less than 200°F (maximum operating temperatures are 125°F for the letdown heat exchanger shell, 130°F for the charging pump, and 100°F for the monitor tank pump), a design temperature of 200°F provides adequate margin. An increase in design temperature to 250°F would not provide any significant changes in equipment design and would not affect reactor safety.

If a significant leak develops in the letdown heat exchanger, such that the charging system is unable to maintain pressurizer water level, the letdown line isolation valves near the reactor coolant loop will trip closed. The excess letdown path can then be placed in service while maintenance is performed on the letdown heat exchanger.

9.2.2.14 Mixed Bed Demineralizers

Two flushable mixed bed demineralizers maintain reactor coolant purity. A lithium-7 (or H+ form) cation resin and a hydroxyl form anion resin are initially charged into the demineralizers. Both forms of resin remove fission and corrosion products, and, in addition, the reactor coolant causes the anion resin to be converted to the borate form. The resin bed is designed to reduce the concentration of ionic isotopes in the purification stream.

Each demineralizer is sized to accommodate the normal letdown flow. One demineralizer serves as a standby unit for use if the operating demineralizer becomes exhausted during operation.

The demineralizer vessels are made of austenitic stainless steel, and are provided with suitable connections to facilitate resin replacement when required. The vessels are equipped with a resin retention screen. Each demineralizer has sufficient capacity, after operation for one core cycle with 1 percent defective fuel rods, to reduce the activity of the reactor coolant to refueling concentration.

9.2.2.15 Cation Bed Demineralizer

A flushable cation resin bed in the hydrogen form is located downstream of the mixed bed demineralizers and is used intermittently to control the concentration of lithium-7 which builds up in the coolant from the B^{10} (n, α) Li^7 reaction. The demineralizer also has sufficient capacity to maintain the cesium-137 concentration in the coolant below 1.0 μCi/cc with 1 percent defective fuel. The demineralizer would be used intermittently to control cesium.

The demineralizer is made of austenitic stainless steel and is provided with suitable connections to facilitate resin replacement when required. The vessel is equipped with a resin retention screen.

9.2.2.16 Deborating Demineralizers

When required, two anion demineralizers remove boric acid from the RCS fluid. The demineralizers are provided for use near the end of a core cycle, but can be used at any time. Hydroxyl-form ion-exchange resin is used to reduce RCS boron concentration. Facilities are provided for regeneration. When regeneration is no longer feasible the resin is flushed to the spent resin-shipping cask.

Each demineralizer can remove the quantity of boric acid that must be removed from the RCS to maintain full power operation near the end of core life without the use of the holdup tanks or evaporators.

9.2.2.17 Resin Fill Tank

The resin fill tank is used to charge fresh resin to the demineralizers. The line from the conical bottom of the tank is fitted with a dump valve and may be connected to any one of the demineralizer fill lines. The demineralized water and resin slurry can be sluiced into the demineralizer by opening the dump valve. The tank, designed to hold approximately one-third the resin volume of one mixed bed demineralizer, is made of austenitic stainless steel.

9.2.2.18 Reactor Coolant Filter

This filter collects resin fines and particulates larger than 25 microns from the letdown stream. The vessel is made of austenitic stainless steel, and is provided with connections for draining and venting. Design flow capacity of the filter is equal to the maximum purification flow rate. Disposable synthetic bag filter elements are used.

9.2.2.19 Volume Control Tank

The volume control tank (VCT) collects the excess water, released from zero power to full power that is not accommodated by the pressurizer. It also receives the excess coolant release caused by the deadband in the reactor control temperature instrumentation. Overpressure of hydrogen gas is maintained in the volume control tank to control the hydrogen concentration in the reactor coolant at 25 to 50 cc per kg of water (standard conditions).

A spray nozzle is located inside the tank on the VCT inlet line coming from the reactor coolant filter. This spray nozzle provides intimate contact to equilibrate the gas and liquid phases. A remotely operated vent valve discharging to the Waste Disposal System permits removal of gaseous fission products which are stripped from the reactor coolant and collected in this tank. Post-accident the VCT vents gaseous fission products to containment. The VCT also acts as a head tank for the charging pumps and a reservoir for the leakage from the reactor coolant pump controlled-leakage seal. The tank is constructed of austenitic stainless steel.

9.2.2.20 Charging Pumps

Three charging pumps inject coolant into the RCS. The pumps are the variable-speed positive-displacement type, and all parts in contact with the reactor coolant are fabricated of austenitic stainless steel and other material of adequate corrosion resistance. Special low-chloride packing is used in the pump glands. These pumps have mechanical packing followed by a leakoff to collect reactor coolant before it can leak to the outside atmosphere. Pump leakage is piped to the RHR sump pump pit. The pump design precludes the possibility of lubricating oil contaminating the charging flow, and the integral discharge valves act as check valves.

Each charging pump has sufficient capacity (60 gpm) to compensate for normal letdown purification flow (40 gpm) and No. 1 seal leakage (normally 3 gpm/pump) and, thus, maintain the proper reactor coolant total inventory, temperature and pressure. Each pump is designed to provide rated flow against a pressure equal to the sum of the RCS maximum pressure (existing when the pressurizer PORV is operating) and the piping, valve and equipment pressure losses of the charging system at the design charging flows.

A suction stabilizer and pulsation dampener have been installed on the inlet and discharge piping of each of the three charging pumps. The installation significantly reduces vibration and stress levels on the charging pumps.

One of the three charging pumps can be used to hydrotest the RCS. A small motor can be directly coupled to the pump for hydrostatic test purposes. A design change removed the small motor, and disconnected the power and control cables; however, the hydrotest capability still exists.

9.2.2.21 Chemical Mixing Tank

The primary use of the chemical mixing tank is in the preparation of pH control chemical solutions and hydrazine for oxygen scavenging.

The capacity of the chemical mixing tank is determined by the quantity of 35 percent hydrazine solution necessary to increase the concentration in the reactor coolant by 10 ppm. This capacity is more than sufficient to prepare the solution of pH control chemical for the RCS. The chemical mixing tank is made of austenitic stainless steel.

The mixing tank is provided with an orifice located in the reactor makeup water inlet line to limit the flow rate through the tank as the solution is flushed to the charging pump suction. The orifice is designed to pass the tank volume within 2.5 minutes by the reactor makeup water pressure.

9.2.2.22 Demineralizer Letdown Pre-Filter

The filters, a normal and a bypass, collect particulates larger than 15 microns from the letdown stream before it enters the demineralizers. The vessels are made of austenitic stainless steel, and are provided with connections for draining and venting. Design flow capacity of each filter is equal to the maximum purification flow rate. Disposable filter elements are used.

9.2.2.23 Excess Letdown Heat Exchanger

The excess letdown heat exchanger cools reactor coolant letdown flow at a rate equal to the nominal injection rate through the reactor coolant pump labyrinth seal, if letdown through the normal letdown path is blocked. The unit is designed to reduce the letdown stream temperature from the cold-leg temperature to 195°F. The letdown stream flows through the tube side and component cooling water is circulated through the shell side. All surfaces in contact with reactor coolant are austenitic stainless steel and the shell is carbon steel. All tube joints are welded. Flanges have been installed in the excess letdown line to the heat exchanger to permit tube bundle removal.

9.2.2.24 Seal Water Heat Exchanger

The seal water heat exchanger removes heat from the reactor coolant pump seal water returning to the volume control tank and reactor coolant discharge from the excess letdown heat exchanger. Reactor coolant flows through the tubes and component cooling water is circulated through the shell side. The tubes are welded to the tube sheet to prevent leakage in either direction, which would result in undesirable contamination of the reactor coolant or component cooling water. All surfaces in contact with reactor coolant are austenitic stainless steel and the shell is carbon steel.

The unit is designed to cool the excess letdown flow and the seal water flow to the temperature normally maintained in the volume control tank if all the reactor coolant pump seals are leaking at the maximum design leakage rate.

9.2.2.25 Seal Water Filter

The filter collects particulates larger than 25 microns from the reactor coolant pump seal water return and from the excess letdown heat exchanger flow. The filter is designed to pass the sum of the excess letdown flow and the maximum design leakage from the reactor coolant pump controlled-leakage seals. The vessel is constructed of austenitic stainless steel and is provided with connections for draining and venting. Disposable synthetic filter cartridges are used.

9.2.2.26 Seal Water Injection Filters

Two filters are provided in parallel, each sized for the injection flow. They collect particulates from the water supplied to the reactor coolant pump seal. The vessel is constructed of stainless steel and the filter elements are disposable cartridges.

9.2.2.27 Boric Acid Filter

The boric acid filter collects particulates larger than 25 microns from the boric acid solution being pumped to the charging pump suction line. The filter is designed to pass the design flow of two boric acid transfer pumps operating simultaneously and at high speed. The vessel is constructed of austenitic stainless steel and the filter elements are disposable synthetic cartridges. Provisions are available for venting and draining the filter.

9.2.2.28 Boric Acid Tanks

The boric acid tank capacities are sized to store sufficient boric acid solution for refueling plus enough boric acid solution for cold shutdown shortly after full power operation is achieved.

The concentration of boric acid solution in storage is maintained between 7.5 and 8.5 percent by weight (13,000 to 15,000 ppm boron) at a temperature of at least 125°F. Periodic manual sampling is performed and corrective action is taken, if necessary, to ensure that these limits are maintained.

As a consequence, measured amounts of boric acid solution can be delivered to the reactor coolant to control the chemical poison concentration. The combination overflow and breather vent connection has a water loop seal to minimize vapor discharge during storage of the solution. The tanks are constructed of austenitic stainless steel.

9.2.2.29 Boric Acid Tank Heaters

Two 100 percent capacity electric immersion heaters located near the bottom of each boric acid tank are designed to maintain the temperature of the boric acid solution at nominal 135°F with an ambient air temperature of 40°F. Thus ensuring a temperature in excess of the solubility limit (for 8.5 percent weight boric acid solution, crystallization occurs at 105°F). The temperature is monitored and is alarmed (high and low temperature alarms) in the Control Room. The heaters are sheathed in austenitic stainless steel.

9.2.2.30 Batching Tank

The batching tank is sized to hold 6 days makeup supply of nominal 8.0 percent boric acid solution for the boric acid tank. The basis for makeup is reactor coolant leakage of ½ gpm at beginning of core life. The tank is also used for solution storage. A local sampling point is provided for verifying the solution concentration prior to transferring it to the boric acid tank.

A tank manway is provided, with a removable screen to prevent entry of foreign material. In addition, the tank is provided with an agitator to improve mixing during batching operation. The tank is constructed of austenitic stainless steel and is not used to handle radioactive substances. The tank is provided with a steam jacket for heating the boric acid solution to 135°F.

The source of heat for the steam-jacketed boric acid batching tank is the process steam.

The boric acid batching tank is not required under post-accident conditions, since the boric acid tanks or the refueling water storage tank are used to supply the borated water required for safe shutdown.

9.2.2.31 Boric Acid Transfer Pumps

Two centrifugal pumps with two-speed motors are used to circulate or transfer boric acid. The pumps circulate boric acid solution through the boric acid tanks and inject boric acid into the charging pump suction header.

Normally one pump is used for boric acid batching and transfer. Both pumps are used for boric acid injection. The design capacity of each pump is equal to the normal letdown flow rate. The design head at high speed is sufficient, considering line and valve losses, to deliver rated flow to the charging pump suction header when volume control tank pressure is at the maximum operating value (relief valve setting). All parts in contact with the solutions are austenitic stainless steel and other adequately corrosion-resistant materials.

The enclosures around the Boric Acid Transfer Pumps are heated to prevent boric acid solidification. Each pump is heated by two cartridge heaters installed in stainless steel blocks mounted to the base of the pump. The suction piping within the enclosure is heated by fin strip heaters located on the sides of the piping. The electrical configuration maintains train separation of the heaters for each pump.

The transfer pumps are operated either automatically or manually from the main control room. The reactor makeup control operates both pumps automatically at high speed when boric acid solution is required for makeup or boration.

9.2.2.32 Boric Acid Blender

The boric acid blender enhances thorough mixing of boric acid solution and reactor makeup water from the reactor makeup supply circuit. The blender consists of a conventional pipe elbow fitted with a perforated tube insert. All material is austenitic stainless steel. The blender decreases the pipe length required to homogenize the mixture.

9.2.2.33 Recycle Process

9.2.2.33.1 Holdup Tanks

Three holdup tanks contain radioactive liquid, which enters from the letdown line. The liquid is released from the RCS during startup, shutdown, and load changes and from boron dilution to compensate for burnup. The contents of one tank may be processed by the gas stripper/boric acid evaporator package while another tank is available as a standby. The total liquid storage capacity of the three holdup tanks is designed to meet the liquid dilution required to return to power from cold shutdown when the unit is at approximately 90 percent of core life. The tanks are constructed of austenitic stainless steel.

The tanks utilize the gas decay tanks, with nitrogen as a backup, to prevent oxygen contamination.

9.2.2.34 Holdup Tank Recirculation Pump

The holdup tank recirculation pump is a centrifugal type used to mix the contents of a holdup tank or transfer the contents of one holdup tank to another. The pump is sized to transfer the contents of one tank to another in less than two hours. The wetted surface of this pump is constructed of austenitic stainless steel.

9.2.2.35 Gas Stripper Feed Pumps

Two canned rotor centrifugal gas stripper feed pumps supply feed from a holdup tank to the gas stripper/evaporator train. The capacity of each pump is equal to twice the gas stripper/evaporator capacity. The non-operating pump is a standby and is available for operation in the event the operating pump malfunctions. These centrifugal pumps are constructed of austenitic stainless steel.

9.2.2.36 Evaporator Feed Ion Exchangers

Three flushable evaporator feed ion exchangers remove cation contamination from the holdup tank effluent. Experiments performed by Westinghouse indicate that the decontamination factor for cesium (see Appendix D) is conservative. One ion exchanger has sufficient capacity to supply the gas stripper/evaporator train. These ion exchangers may be operated in parallel or series. Each vessel is constructed of austenitic stainless steel and contains a resin retention screen.

9.2.2.37 Ion Exchanger Filters

These filters collect resin fines and particulates from the evaporator feed ion exchanger. The vessels are made of austenitic stainless steel and are provided with connections for draining and venting. Disposable synthetic filter cartridges are used. The design flow capacity is equal to the boric acid evaporator flow rate.

9.2.2.38 Boric Acid Evaporator/Gas Stripper Package

One boric acid evaporator/gas stripper package is provided to remove radioactive gases and concentrate boric acid for reuse in the Reactor Coolant System.

Liquid effluent from the holdup tanks is preheated and then passed through the gas stripper column where dissolved and entrained gases are removed. The feed stream leaving the column enters the evaporator where the water and boric acid is concentrated in the evaporator shell to a nominal 8.0 percent weight boric acid solution and then pumped out. All liberated noncondensible gases flow to the vent header.

The evaporator/gas stripper package consists of a feed pre-heater, vent condenser, stripping column, evaporator, absorption tower, evaporator condenser, distillate cooler and the following pumps: two evaporator concentrates pumps and two distillate pumps. The package also includes valves, piping and associated component and process instrumentation.

All evaporator/gas stripper package equipment in contact with the process fluid is constructed of austenitic stainless steel.

9.2.2.39 Evaporator Condensate Demineralizers

Two demineralizers remove the radioactive contaminants carried over with the evaporator condensate. When the resin is exhausted, it is flushed to the Waste Disposal System. Normally one demineralizer is used as needed, with one available as a standby. The demineralizer vessels are constructed of austenitic stainless steel.

9.2.2.40 Condensate Filter

A filter collects resin fines and particulates larger than 25 microns from the boric acid evaporator condensate stream. The required flow capacity of the filter is based on the boric acid evaporator flow rate. The vessel is made of austenitic stainless steel and is provided with a connection for draining and venting. Disposable synthetic filter elements are used. The design flow capacity of the filter is equal to the boric acid evaporator flow rate.

9.2.2.41 Monitor Tanks

The monitor tanks permit continuous operation of the evaporator train. When one tank is filled, the contents are analyzed and either reprocessed, discharged to the Waste Disposal System, or pumped to one of the reactor makeup water tanks. The monitor tanks can also be filled by water from the makeup water treatment system. These tanks are constructed of austenitic stainless steel.

9.2.2.42 Monitor Tank Pumps

Two monitor tank pumps discharge water from the monitor tanks. Each pump is designed to empty a monitor tank in 1.5 hours. The pumps are constructed of austenitic stainless steel.

9.2.2.43 Reactor Makeup Water Tanks

Two reactor makeup-water tanks contain liquid supplied from the makeup water demineralizers. The monitor tanks, after sampling has shown the liquid to be of proper quality and acceptable radioactivity level, could be transferred to the reactor makeup tank, if desired. These stainless steel tanks serve as a source of RCS makeup water.

9.2.2.44 Reactor Makeup Water Pumps

Two reactor makeup water pumps serve as a supply source for the Reactor Coolant Makeup System. These austenitic stainless steel pumps take suction from the reactor makeup water tanks.

9.2.2.45 Concentrates Filter

A disposable synthetic cartridge type filter removes particulates larger than 25 microns from the evaporator concentrates. Design flow capacity of the filter is equal to the boric acid evaporator concentrates transfer pump capacity. The vessel is made of austenitic stainless steel.

9.2.2.46 Concentrates Holding Tank

The concentrates holding tank is sized to hold one batch of concentrates from operation of the evaporator. The tank is supplied with an electrical heater, which prevents boric acid precipitation and is constructed of austenitic stainless steel.

9.2.2.47 Concentrates Holding-Tank Transfer Pumps

Two holding-tank transfer pumps discharge boric acid to the boric acid tanks or to the holdup tanks for recycling.

9.2.2.48 Electrical Heat Tracing

Electrical heat tracing is installed under the insulation on all piping, valves, line-mounted instrumentation, and components normally containing concentrated boric acid solution. The heat tracing compensates for heat loss due to cooling and prevents boric acid precipitation. The heat tracing system was designed in accordance with the following criteria:

1. 100 percent redundant and separate heat tracing systems are provided, with the exceptions that the concentrates holding tank is equipped with a single immersion heater, and the boric acid batching tank is steam heated.
2. The heat tracing system is designed to maintain the fluid temperature between 125°F and 135°F with an ambient air temperature of 40°F.
3. Each redundant heat tracing system is supplied from a separate power source connected to the redundant emergency diesel generators.
4. Normally, only one heat tracing system is energized. Failure of the energized heat tracing system is annunciated in the Control Room. An automatic transfer of control from the primary to the standby heat tracing system is achieved by energizing the redundant system by different settings in the two separate control thermostats, such that on failure of the primary system, the standby system will pick-up automatically without reaching the low temperature alarm setting.
5. The boric acid tanks are equipped with individual means of heating by immersion heaters supplied, as in (3) above.

The lines and components of the CVCS which are provided with heat tracing or heater enclosures are shown in Figure 9.2-1 through Figure 9.2-5.

The boric acid tanks, boric acid batching tank, and the concentrates holding tank are provided with individual means of heating and need not be electrically heat traced.

Redundant electrical heat tracing is installed on all sections of the CVCS normally containing boric acid solution, to provide standby capacity if the operating section malfunctions. The power supply for the redundant lines of heat tracing is connected to the diesel-powered buses to ensure continuous operation during a prolonged outage of normal power supplies.

The combination of electrical heat tracing and insulation maintains the temperature of the piping and contents at 125°F to 135°F with an ambient air temperature of 40°F. Separate thermostatic controls are provided for each of the duplicate sets of heat tracing to maintain the temperature within the specified control band. A high/low alarm is provided in the Control Room to warn of failure to maintain the normal temperature control band for the piping and equipment containing concentrated boric acid solution. Transfer of control between the redundant heat tracing is an automatic operation. Any single failure of a heat tracing line, heater controller or alarm will not result in a reduction of temperature below the point where precipitation of concentrated boric acid might occur.

9.2.2.49 Valves

Valves for radioactive service that perform a modulating function are equipped with two sets of packing and an intermediate leakoff connection that discharges to the Waste Disposal System. Manual and motor operated valves larger than 2 inches for radioactive service, with fluids at temperatures above 212°F, also have an intermediate leakoff connection that discharges to the Waste Disposal System. Globe valves are installed with flow over the seats when such an arrangement reduces the possibility of leakage. Basic material of construction is stainless steel for all valves except the batching tank steam jacket valves, which are carbon steel.

Isolation valves are provided at all connections to the RCS. Lines with flow into the reactor containment also have check valves inside the containment to prevent reverse flow from the containment.

Relief valves are provided for lines and components that might be pressurized above design pressure by improper operation or component malfunction. Pressure relief for the tube side of the regenerative heat exchanger, is provided by the check valve, which bypasses the charging line isolation valve.

9.2.2.50 Piping

All CVCS piping handling radioactive liquid is austenitic stainless steel. All piping joints and connections are welded, except where flanges are required to facilitate equipment removal for maintenance.

9.2.3 System Design Evaluation

9.2.3.1 Availability and Reliability

A high degree of functional reliability is assured in this system by providing standby components and by assuring fail-safe response to the most probable mode of failure. Special provisions include duplicate heat tracing (with alarm protection) of lines, valves, and components normally containing concentrated boric acid.

The system has three high-pressure charging pumps, each capable of supplying the normal reactor coolant pump seal and makeup flow.

The electrical equipment of the CVCS is arranged so that multiple items receive their power from various 480-V buses as described in Chapter 8.

The two boric acid transfer pumps are powered from separate 480-V buses. One charging pump and one boric acid transfer pump are capable of meeting cold shutdown requirements shortly after full-power operation. In cases of loss of a-c power, any charging pump and boric acid transfer pump can be energized from diesel generator power.

9.2.3.2 Control of Tritium

The CVCS is used to control the concentration of tritium in the RCS. Essentially all of the tritium is in the chemical combination with oxygen as a form of water. Therefore, any leakage of coolant to the containment atmosphere carries tritium in the same proportion as it exists in the coolant. Thus, the level of tritium in the containment atmosphere, when it is sealed from outside air ventilation, is a function of tritium level in the reactor coolant, the cooling water temperature at the cooling coils, which determines the dew point temperature of the air, and the presence of leakage other than reactor coolant as a source of moisture in the containment air.

There are two major considerations with regard to the presence of tritium:

1. Possible plant personnel hazard during access to the containment. Leakage of reactor coolant during operation with a closed containment causes an accumulation of tritium in the containment atmosphere. It is desirable to limit the accumulation to allow containment access.
2. Possible public hazard due to release of tritium to the plant environment.

Neither of these considerations is limiting in this plant.

The concentration of tritium in the reactor coolant is maintained at a level, which precludes personnel hazard during access to the containment. This is achieved by discharging the condensate from the boric acid recovery process via the plant circulating water discharge.

The uncertainties associated with estimating the amounts of tritium generated are discussed in Appendix D.

Periodic determinations of tritium concentrations will be made by liquid scintillation counting of condensed water vapor from the containment.

9.2.3.3 Leakage Prevention

Quality control of the material and the installation of the CVCS valves and piping, which are designated for radioactive service, is provided in order to essentially eliminate leakage to the atmosphere. The components designated for radioactive service are provided with welded connections to prevent leakage to the atmosphere. However, flanged connections are provided in each charging pump suction and discharge, on each boric acid pump suction and discharge, on the relief valves inlet and outlet, on three-way valves, on the flow meters and elsewhere where necessary for maintenance.

The positive displacement charging pumps stuffing boxes are provided with leakoffs to collect reactor coolant before it can leak to the atmosphere. All valves which are larger than 2 inches and which are designated for radioactive service at an operating fluid temperature above 212°F are provided with a stuffing box and lantern leakoff connections. Leakage to the atmosphere is essentially zero for these valves. All control valves are either provided with stuffing box and leakoff connections or are totally enclosed. Leakage to the atmosphere is essentially zero for these valves.

Diaphragm valves are provided where the operating pressure and the operating temperature permit the use of these valves. Leakage to the atmosphere is essentially zero for these valves.

9.2.3.4 Incident Control

The letdown line and the reactor coolant pumps seal water return lines penetrate the reactor containment. The letdown line contains two air-operated valves inside the reactor containment upstream of the regenerative heat exchanger. Three parallel air-operated orifice block valves inside the reactor containment and an air-operated valve outside the reactor containment are automatically closed by the containment isolation signal.

The reactor coolant pump’s seal water return line contains one motor-operated isolation valve outside and one inside the reactor containment which are automatically closed by the containment isolation signal.

The two seal-water injection lines to the reactor coolant pumps and the charging line are inflow lines penetrating the reactor containment. Each line contains two check valves inside the reactor containment to provide isolation of the reactor containment if a break occurs in these lines outside the reactor containment.

9.2.3.5 Malfunction Analysis

To evaluate system safety, failures or malfunctions were assumed concurrent with a LOCA and the consequences analyzed and presented in Table 9.2-4. As a result of this evaluation, it is concluded that proper consideration has been given to plant safety in the design of the system.

If a rupture were to take place between the reactor coolant loop and the first isolation valve or check valve, this incident would lead to an uncontrolled loss of reactor coolant. The analysis of LOCAs is discussed in Chapter 14.

Should a rupture occur in the CVCS outside the containment, or at any point beyond the first check valve or remotely operated isolation valve, actuation of the valve would limit the release of coolant and assure continued functioning of the normal means of heat dissipation from the core. For the general case of rupture outside the containment, the largest source of radioactive fluid subject to release is the contents of the volume control tank. The consequences of such a release are considered in Chapter 14.

When the reactor is sub-critical; i.e., during cold or hot shutdown, refueling, and approach to criticality, the relative reactivity status (neutron source multiplication) is continuously monitored and indicated by fission chambers counters and count rate indicators. Any appreciable increase in the neutron source multiplication, including that caused by the maximum physical boron dilution rate, is slow enough to give ample time to start a corrective action (boron dilution stop and/or emergency boron injection) to prevent the core from becoming critical. See Chapter 14 for a more complete discussion of boron dilution accidents.

At least three separate and independent flow paths are available for reactor coolant boration; i.e., the charging line or through the two reactor coolant pump labyrinths. The malfunction or failure of one component will not result in the inability to borate the RCS. An alternate flow path is always available for emergency boration of the reactor coolant. As backup to the boration system, the operator can align the RWST outlet to the suction of the charging pumps.

Concentrated boric acid can be injected into the Reactor Coolant System by means of the charging pumps through two flow paths:

1. normal charging line (30 gpm), or
2. seal water supply lines to the two reactor coolant pumps while bypassing seal injection filters (8 gpm per pump 3 gpm of which leaks back into the CVCS and 5 gpm of which passes into the RCS).

Each flow path is provided with a flow indicator.

Suction to the charging pumps can be delivered through three flow paths:

1. the blender and flow control valve,
2. a local manual valve path, or
3. an emergency boration path through the motor operated valve.

Each flow path is provided with a flow meter.

A letdown or charging line break would be indicated by excessive auto-makeup to the volume control tank. A break in the charging line or upstream of the letdown orifices in the letdown line would also result in an increase in charging pump speed and a possible hi-speed alarm, depending on the break size. If the break size is such that the charging/letdown system is unable to maintain pressurizer water level, the letdown line isolation valves near the reactor coolant loop would be tripped closed by a low level signal from the pressurizer level instrumentation (redundant valves and level channels are provided to insure letdown isolation in the event of a single active failure).

In the event that the letdown line must be isolated, except in an emergency, the charging line is also isolated to avoid thermal shocking of the charging line penetrations into the RCS. If the charging line must be isolated, the letdown line is also isolated to avoid flashing of the letdown stream as the pressure of the high temperature flow is reduced.

With the letdown and charging lines isolated, the RCS can be borated via the reactor coolant pump labyrinth seals by allowing the pressurizer water level to increase. If letdown of reactor coolant is necessary, the excess letdown line is capable of letting down a flow equivalent to the total labyrinth seal in-leakage from both reactor coolant pumps. As the system is cooled down, makeup water to maintain pressurizer level would be provided through the labyrinth seals. Therefore, the normal charging and letdown paths are not required to go to cold shutdown condition.

The minimum rate of injection of boric acid solution into the RCS is 10 gpm from the labyrinth seal leakage through each reactor coolant pump. With this injection rate and charging not in service, the time necessary to borate the system sufficiently for a cold shutdown is approximately five hours at EOL. Normal charging capability is 40 gpm (30 gpm through the charging line plus 10 gpm through the reactor coolant pump labyrinth seals), which can borate the Reactor Coolant System to the concentration necessary for a cold shutdown in approximately 1.5 hours. With the charging and letdown lines out of service, the length of time necessary to bring the reactor to cold shutdown conditions is increased by the three additional hours necessary to reach the appropriate boron concentration as discussed above.

Concentrated boric acid is normally injected into the RCS by means of the charging pumps which take suction from the boric acid tanks via the boric acid transfer pumps. Each operation is considered in turn:

1. Concentrated boric acid can be delivered to the suction of the charging pumps using the following paths:
   a. Through the blender and flow control valve. For this operation the operator may read the flow meter and the boric acid tank level indicators.
   b. Through a local manual valve path. For this operation the operator may read the flow meter and the boric acid tank level indicators.
   c. In the event that neither flow path “a” nor “b” is available, the operator would use the emergency boration path through the motor operated valve. For this emergency operation the operator may read the emergency path flow meter and the boric acid tank level.
2. The charging pumps can deliver concentrated boric acid into the RCS via the following paths:
   a. Normal charging line with flow meter.
   b. Seal water supply lines to the two reactor coolant pumps while bypassing the seal injection filters. If either path had to be used, local and Control Room flow indicators would indicate flow.
3. The SI pumps can also take a suction from the RWST and provide borated water to the RCS, when the RCS pressure is less than SI pump shutoff head. The quantity of boric acid stored in the RWST is sufficient to achieve Cold Shutdown at any time during core life.

On loss of seal injection water to the reactor coolant pump seals, seal water flow can be re-established by manually starting a standby-charging pump. Even if the seal water injection flow is not immediately re-established, the plant can continue to operate temporarily. The thermal barrier cooler has sufficient capacity to cool the reactor coolant flow which will pass through the thermal barrier cooler and seal package, as long as seal leakoff flow is > 2.5 gpm. If seal leakoff flow is 2.5 gpm when normal seal injection flow is lost, there will be one to two hours of slowly increasing temperatures before the pump’s operating limits are reached.

With >2.5 gpm seal leakoff flow from each pump’s #1 seal, long term operation is possible, but in order to protect the reactor coolant pump seals from prolonged flow of unfiltered reactor coolant it is recommended that this condition be only temporary. The effect of continuous reactor coolant pump operation without injection water would be to possibly cause clogging of the #1 seal with the introduction of unfiltered water.

The thermal barrier-cooling coil is a complete backup to seal injection for cooling the reactor coolant pump bearings and seals and no overheating would result from continued operation without seal injection water, provided the pump's #1 seal leakoff flow rate is > 2.5 gpm.

9.2.3.6 Galvanic Corrosion

The only type of materials which are in contact with each other in borated water are stainless steels, Inconel, Stellite valve materials and Zircaloy fuel element cladding. These materials have been shown (Reference 1) to exhibit only an insignificant degree of galvanic corrosion when coupled to each other.

For example, the galvanic corrosion of Inconel versus 304 stainless steel resulting from high temperature tests (575°F) in lithiated, boric acid solution was found to be less than 20.9 mg/dm^5 for the test period of nine days. Further galvanic corrosion would be trivial since the cell currents at the conclusion of the tests were approaching polarization. Zircaloy versus 304 stainless steel was shown to polarize in 180°F lithiated, boric acid solution in less than eight days with a total galvanic attack of 3.0 mg/dm^5. Stellite versus 304 stainless steel was polarized in seven days at 575°F in lithiated boric acid solution. The total galvanic corrosion for this couple was 0.97 mg/dm^5.

As can be seen from the tests, the effects of galvanic corrosion are insignificant to systems containing borated water.

9.2.3.7 Fuel Element Failure Detection

Fuel element failure detection is achieved by monitoring the letdown flow, using channel R-9 of the Area Radiation Monitoring System (see Section 11.2). This channel consists of a fixed position, gamma sensitive GM tube detector. The radiation level is indicated locally outside the Letdown Heat Exchanger Room and remotely in the Control Room where it is recorded. A high radiation alarm is displayed on the radiation monitoring panels in the Control Room and locally. A remotely operated, long half-life radiation check source is provided. The source strength is sufficient to produce an upscale meter indication above background. The range of the channel is 0.1 mr/hr to 100 r/hr.

Delay time for the monitor in detecting fuel element failure ranges from approximately one minute to approximately three minutes, depending on the letdown flow rate. At the maximum letdown flow rate (80 gpm), approximately 1.5 minutes will pass before the reactor coolant, contaminated by fission product release from the failed fuel element, will reach the monitor. At normal letdown rate (40 gpm), approximately three minutes will pass before the contaminated flow reaches the detection area.

The monitor will detect the release of failed fuel element fission products against a background of:

- N-16 source (assuming a sixty-second decay) -- 4.5E+1 mr/hr
- Nominal corrosion product sources -- 1.1E+1 mr/hr
- Previous fuel element defects -- Determined during operation

Fuel failure severity can be determined by relating any increase in radiation detected to a corresponding increase in either rod gap release of general fuel element defects:

- 0.01 Percent fuel element defects -- 7.2E+1 mr/hr
- 1 Rod gap release -- 3.2E+2 mr/hr

9.2 References

1. WCAP 1844, "The Galvanic Behavior of Materials in Reactor Coolants," D. G. Sammarone, August 1961

Table 9.2-1
Chemical and Volume Control System Code Requirements

Regenerative heat exchanger	ASME III*, Class C
Letdown heat exchanger	ASME III, Class C, tube side ASME VIII**, shell side
Mixed bed demineralizers	ASME III, Class C
Reactor coolant filter	ASME III, Class C
Demineralizer letdown pre-filter	ASME III, Class C
Volume control tank	ASME III. Class C
Seal-water heat exchanger	ASME III, Class C tube side ASME VIII, shell side
Excess letdown heat exchanger	ASME III, Class C, tube side ASME VIII, shell side
Chemical mixing tank	ASME VIII (No code stamp required)
Deborating demineralizers	ASME III, Class C
Cation bed demineralizer	ASME III, Class C
Seal-water injection filters	ASME III, Class C
Holdup tanks	API - 620***
Boric acid filter	ASME III, Class C
Evaporator condensate demineralizers	ASME III, Class C
Ion exchanger filters	ASME III, Class C
Condensate filter	ASME III, Class C
Piping and valves	USAS B31.1****

* ASME III - American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section III, Nuclear Vessels.

** ASME VIII - American Society of Mechanical Engineers, Boilers and Pressure Vessel Code, Section VIII.

*** American Petroleum Institute Code.

**** USAS B31.1 - Code for Pressure Piping and special nuclear cases where applicable.

Table 9.2-2
Chemical and Volume Control System Performance Requirements

Plant design life, years	40
Seal water supply flow rate, normal, gpm	16
Seal water return flow rate, normal, gpm	6
Normal letdown flow rate, gpm	40
Maximum letdown (purification) flow rate, gpm	80
Normal charging pump flow, gpm	46
Normal charging line flow, gpm	30
Maximum rate of boration with one transfer and one charging pump (at EOL, 10 ppm), ppm/min	15.5
Equivalent cooldown rate to above rate of boration,°F/min	6.0
Maximum rate of boron dilution with two charging pumps (at refueling), ppm/min	3.3
Two charging pump rate of boration, using refueling water (at EOL), ppm/min	8.6
Equivalent cooldown rate to above rate of boration, °F/min	2.7
Design temperature of reactor coolant entering system at full power, °F	544.5
Design temperature of coolant return to RCS at full power, °F	487.6
Normal coolant discharge temperature to holdup tanks, °F	127.0
Amount of boric acid tank solution (bounding value) at 13000 ppm boron concentration to meet cold shutdown boron concentration requirements shortly after full power operation is achieved, including consideration for one stuck rod, gallons	4300

Table 9.2-3
Principal Component Data Summary

	Quantity	Heat Transfer BTU/hr	Letdown Flow lb/hr	Letdown ΔT °F	Design Pressure psig	Design Temperature °F
Heat Exchangers					Shell/tube	Shell/tube
Regenerative	1	5.46E+6	19,760	254.5	2485/2735	650/650
Letdown	1	10.2E+6	19,760	166.4	150/600	200/400
Seal water	1	1.19E+6	79,940	23	150/150	250/250
Excess letdown	2	1.88E+6	4940	357	150/2485	200/650

	Quantity	Type	Capacity Each gpm	Head ft	Design Pressure psig	Design Temperature °F
Pumps						
Charging	3	Pos. displ.	60.5	2385 psi	3000	200
Boric acid transfer	2	Centrifugal	40	235	150	250
Holdup tank recirc	1	Centrifugal	500	100	150	200
Reactor makeup water	2	Centrifugal	95	245	150	200
Monitor tank	2	Centrifugal	100	150	150	200
Concentrates holding tank transfer	2	Canned	40	150	150	200
Gas stripper feed	2	Canned	15	250	150	200

Table 9.2-3
Principal Component Data Summary

	Quantity	Type	Volume Each, gal	Design Pressure psig	Design Temperature °F
Tanks					
Volume control	1	Vertical	220 ft^3	75Int/15Ext	250
Boric acid	2	Vertical	4000	Atmos.	250
Chemical mixing	1	Vertical	5.0	150	200
Batching	1	Jacket	400	Atmos.	300
Holdup	3	Vertical	4400 ft^3	14	200
Reactor makeup water	2	Vertical	40,000	Atmos.	125
Concentrates holding	1	Vertical	900	Atmos.	250
Monitor	2	Vertical	7500	Atmos.	150

	Quantity	Type	Resin Volume ft^3	Head ft	Design Pressure psig	Design Temperature °F
Demineralizers						
Mixed bed	2	Flushable	30	80	200	250
Cation bed	1	Flushable	12	40	200	250
Evaporator feed ion exchanger	3	Flushable	12	15	200	250
Evaporator condensate	2	Fixed	12	15	200	250
Deborating	2	Fixed	30	40	200	250

Table 9.2-4

Malfunction Analysis of Chemical and Volume Control System

Component	Failure	Comments and Consequences
1. Letdown line	Rupture in the line inside the reactor containment	The remote air-operated valve located near the main coolant loop is closed on low pressurizer level to prevent supplementary loss of coolant through the letdown line rupture. The containment isolation valve in the letdown line outside the reactor containment and also the orifice block valves are automatically closed by the containment isolation signal initiated by the concurrent LOCA. The closure of the isolation valve outside the reactor containment prevents any leakage of the reactor containment atmosphere to the outside atmosphere.
2. Charging line	See above	The check valves located near the main coolant loops prevent supplementary loss of coolant through the line rupture. The remote operated valve located upstream of the check valve in the defective line also may be closed to isolate the Reactor Coolant System from the rupture. The check valves located at the boundary of the reactor containment prevent any leakage of the reactor containment atmosphere outside the reactor containment.
3. Seal-water return line	See above	The motor-operated isolation valve located outside the containment is manually closed or is automatically closed by the containment isolation signal initiated by the concurrent LOCA. The closure of that valve prevents any leakage of the reactor containment atmosphere outside the reactor containment.

Figure 9.2-1 Chemical and Volume Control System - Flow Diagram (Sheet 1 of 4)

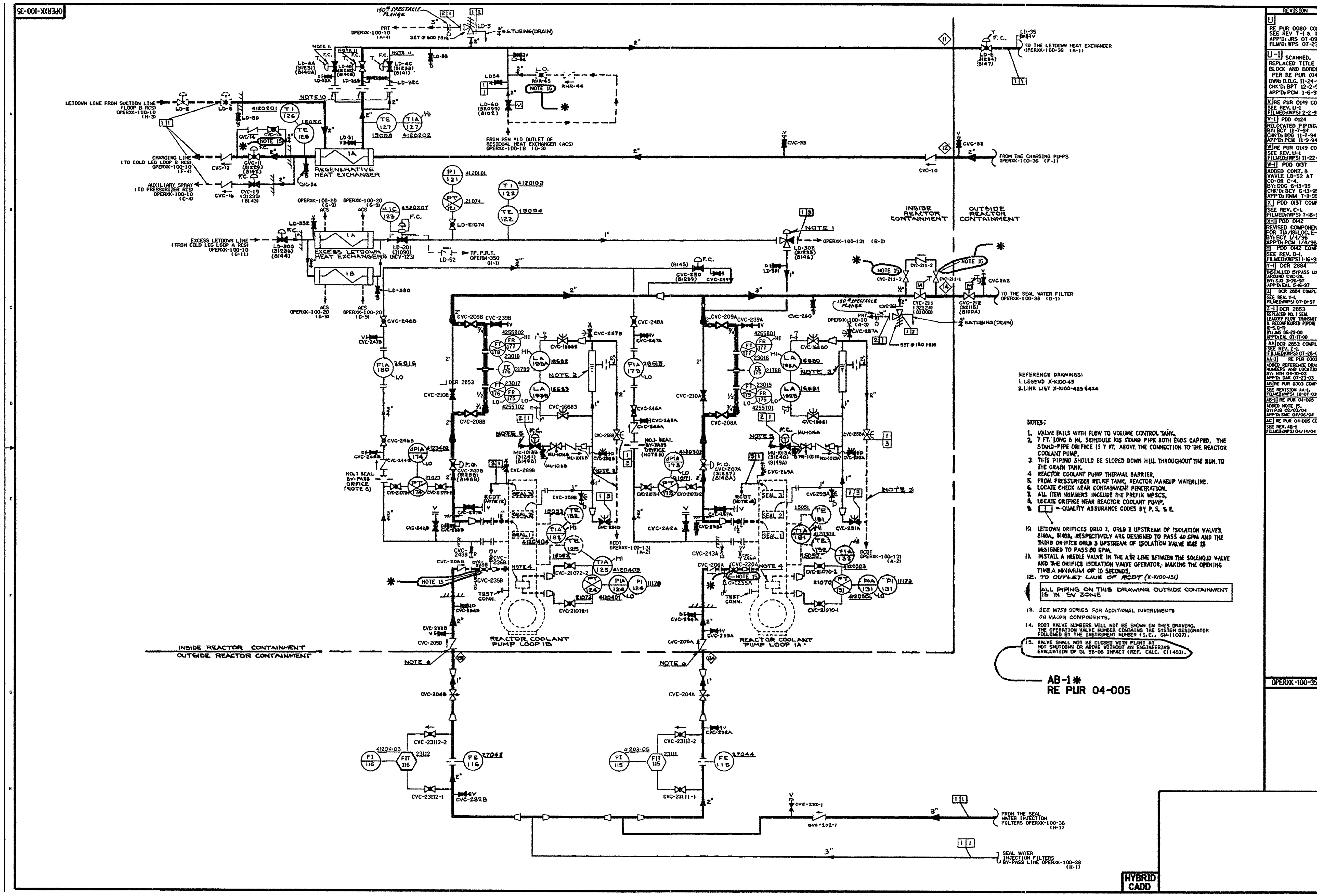


Figure 9.2-2 Chemical and Volume Control System - Flow Diagram (Sheet 2 of 4)

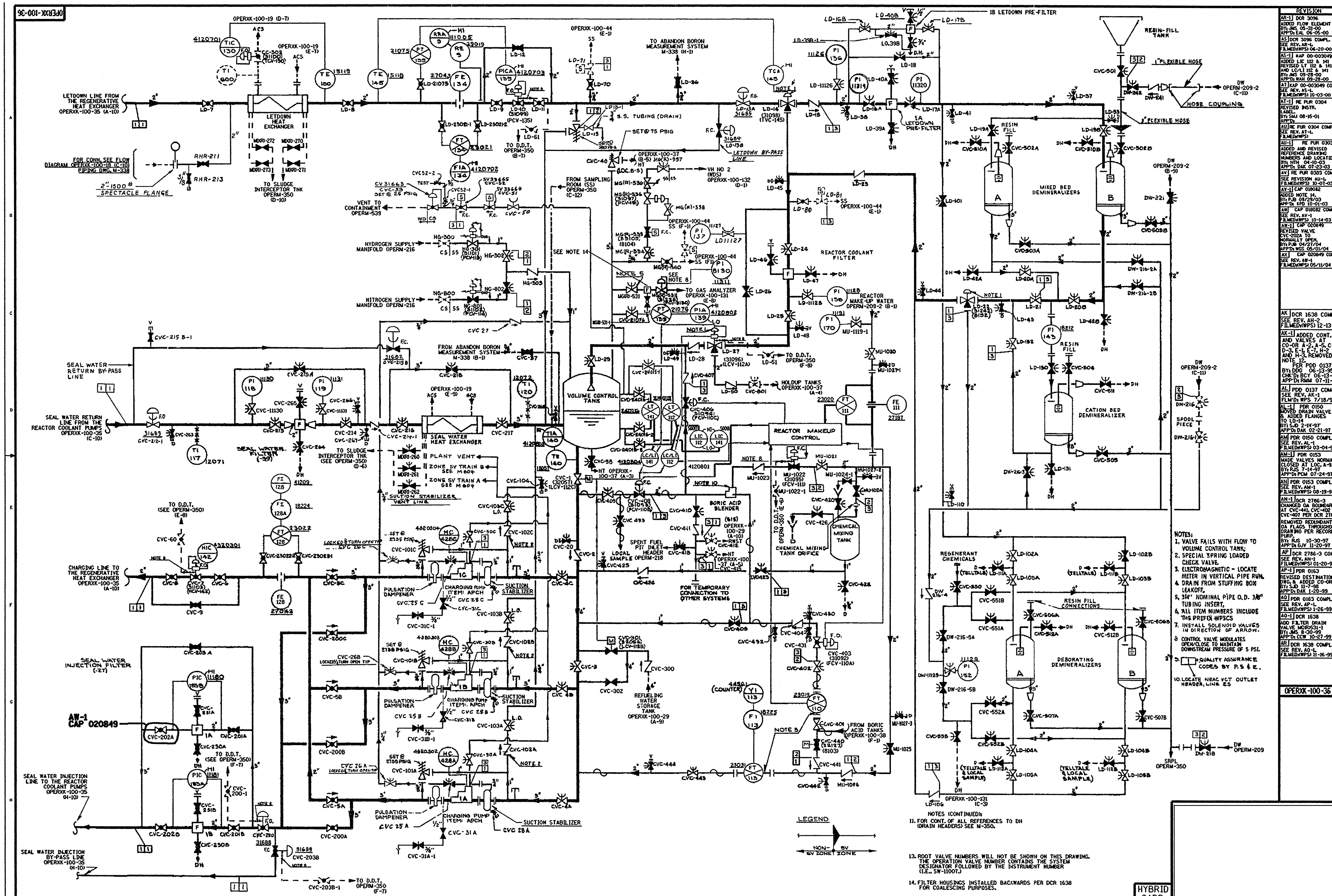


Figure 9.2-3 Chemical and Volume Control System - Flow Diagram (Sheet 3 of 4)

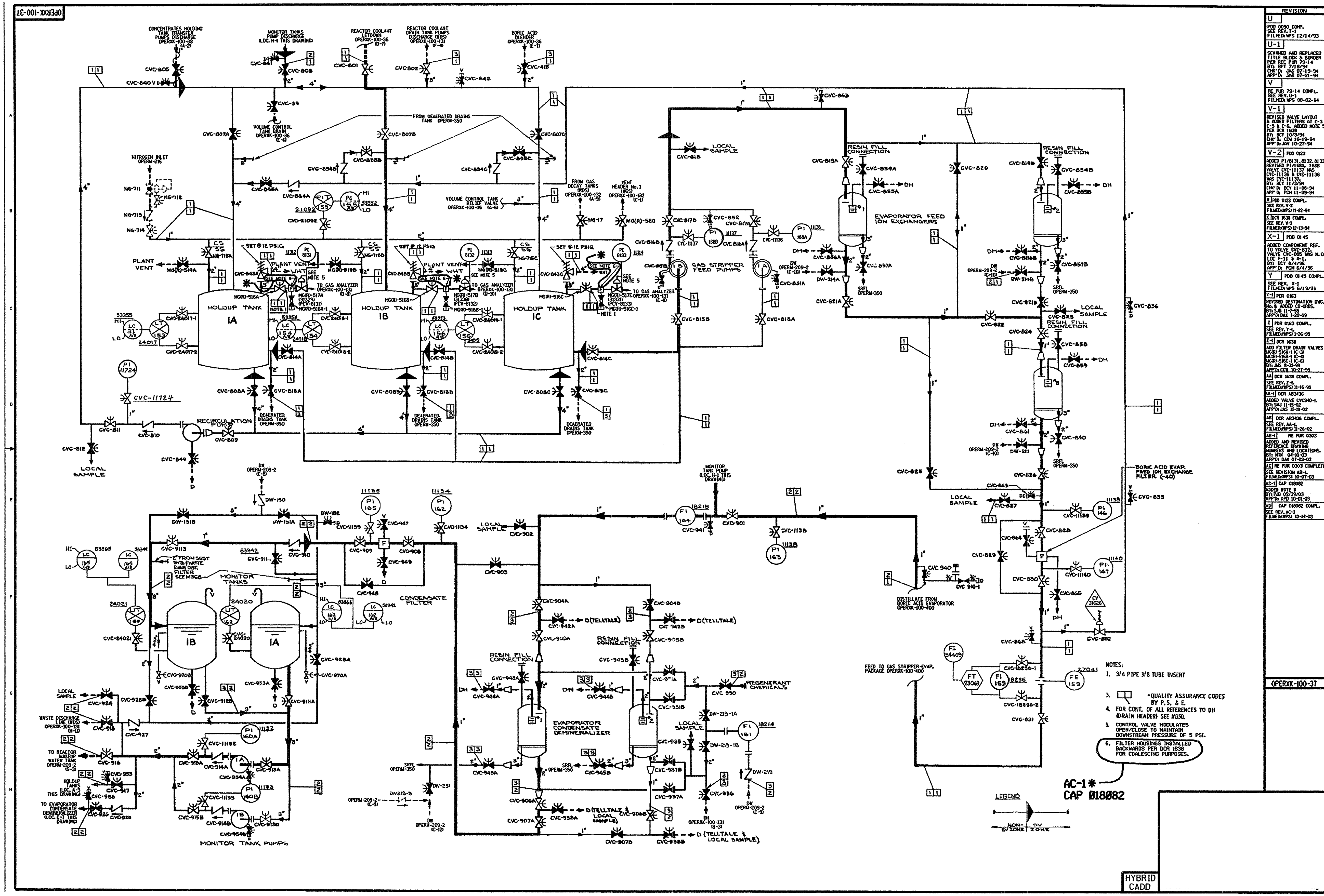


Figure 9.2-4 Chemical and Volume Control System - Flow Diagram (Sheet 4 of 4)

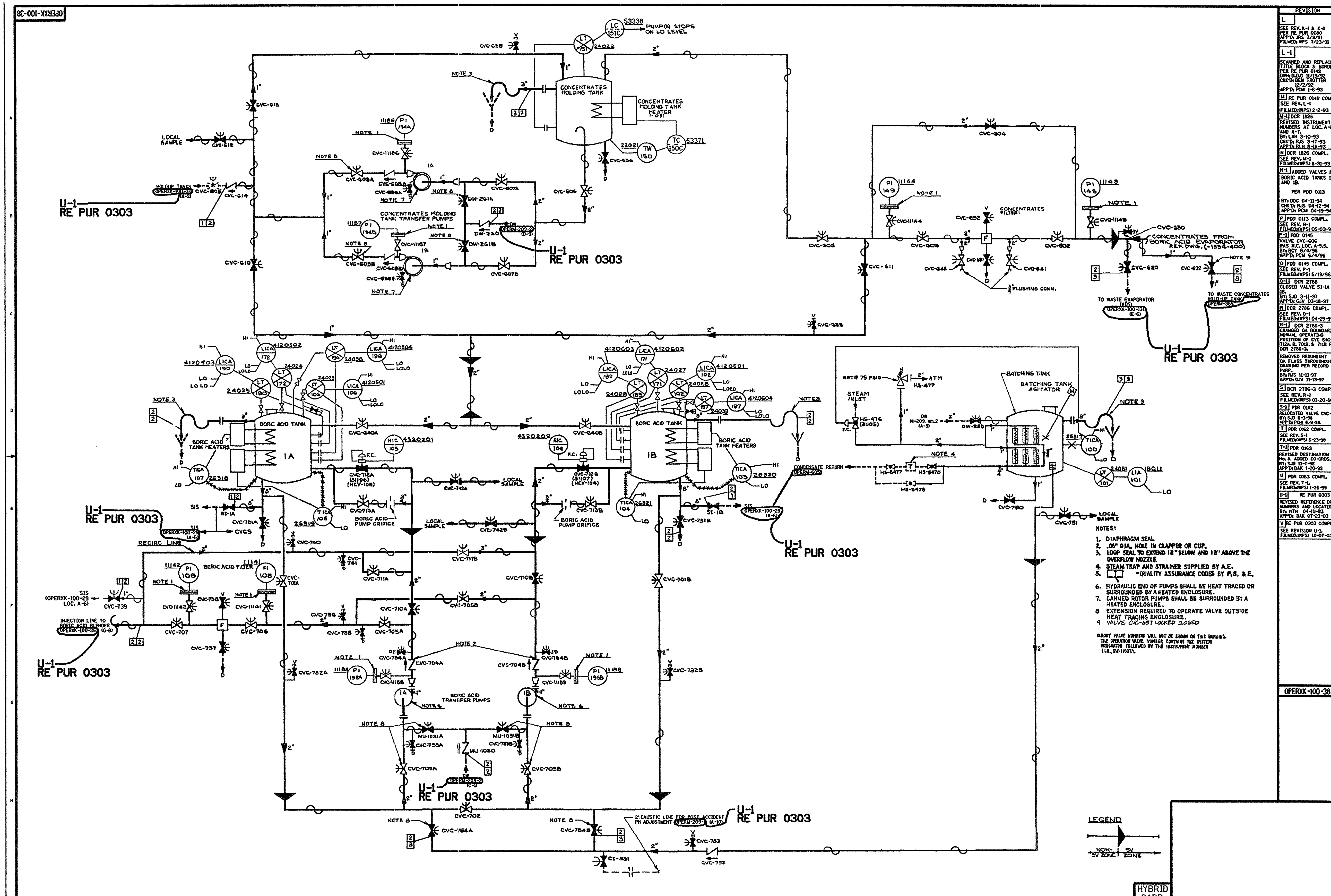
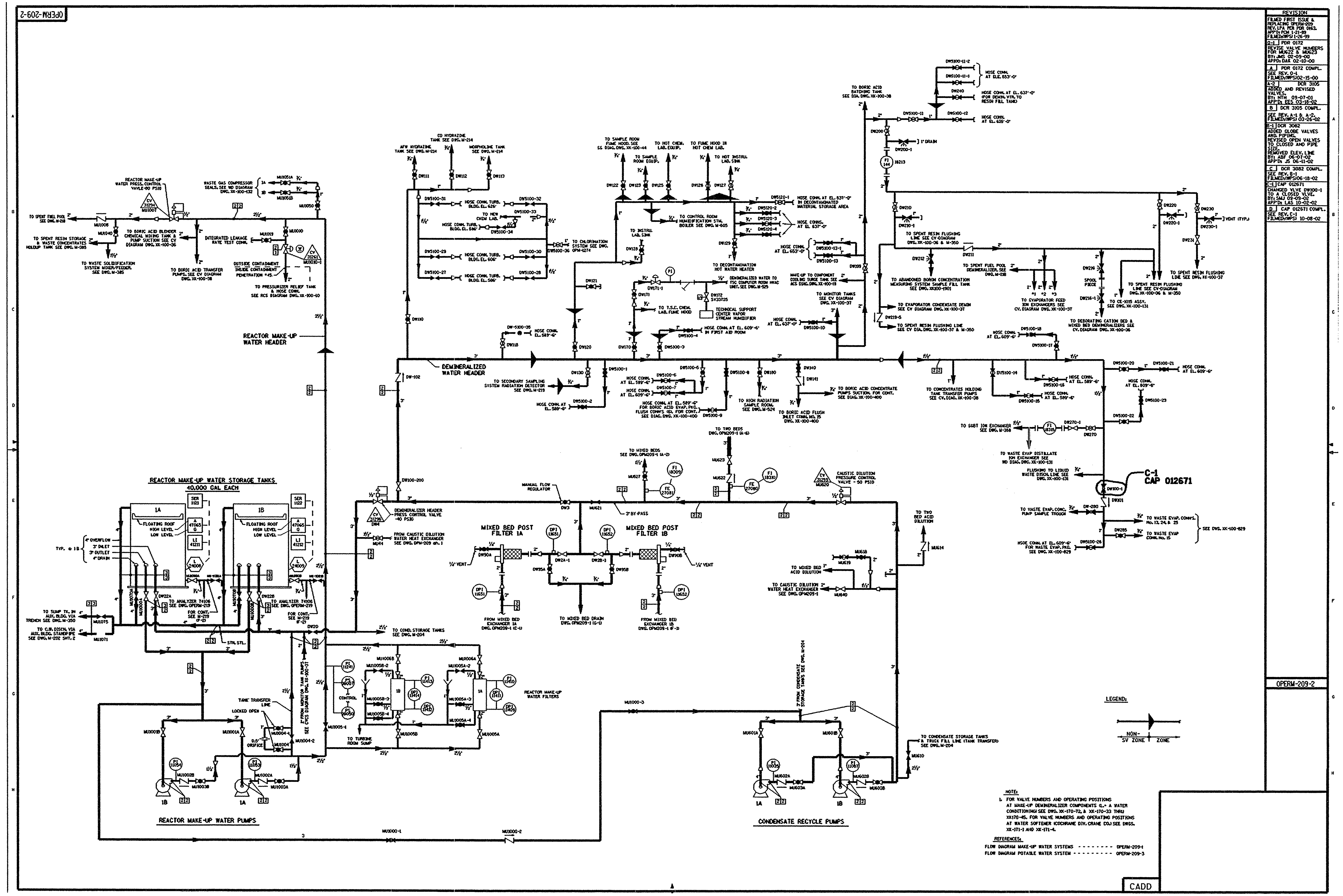


Figure 9.2-5 Make-Up and Demineralized Water System Flow Diagram



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9.3 AUXILIARY COOLANT SYSTEM

9.3.1 Design Basis

The Auxiliary Coolant System consists of three systems the Component Cooling System, the RHR System, and the Spent Fuel Pool Cooling System as shown in Figure 9.3-1 through Figure 9.3-4.

9.3.1.1 Component Cooling System

The Component Cooling System is designed to:

- remove residual and sensible heat from the RCS, via the RHR System, during plant shutdown;
- cool the letdown flow to the CVCS during power operation; and
- to provide cooling to dissipate waste heat from various RCS components.

Since the heat is transferred from the component cooling water to the service water, the Component Cooling System serves as an intermediate system between the Reactor Coolant and the Service Water Systems and insures that any leakage of radioactive fluid from the components being cooled is contained within the plant.

Active system components, which are relied upon to perform the cooling function are redundant.

The system design provides for detection of radioactivity entering the system from reactor coolant sources.

9.3.1.2 Residual Heat Removal System

The RHR System is designed to remove residual and sensible heat from the core and reduce the temperature of the RCS during the second phase of plant cooldown. During the first phase of cooldown, the temperature of the RCS is reduced by transferring heat from the RCS to the Steam and Power conversion System.

The RHR System is placed in operation approximately four hours after reactor shutdown when the pressure and temperature of the RCS are ≤ 425 psig and < 400°F, respectively. Under normal operating conditions, assuming a reactor core power of 1772 MWt, the RHR System will reduce the temperature of the reactor coolant to 140°F within twenty-two hours following reactor shutdown. An evaluation at 1772 MWt has shown that the RHR System is capable of cooling the RCS within Technical Specification shutdown requirements, with reduced service water and component cooling water flows and increased service water temperature.

As a secondary function, the RHR System is used to transfer refueling water between the RWST and the refueling cavity at the beginning and end of refueling operations.

In addition, portions of the system are utilized as parts of the Emergency Core Cooling System and the Containment Spray System. These functions are discussed in Chapter 6.

The RHR System provides sufficient capability in the emergency operational mode to accommodate any single active or passive failure and still function in a manner to avoid risk to the health and safety of the public.

The system design precludes any significant reduction in the overall design reactor shutdown margin when the system is brought into operation for RHR or for emergency core cooling by recirculation.

System components, whose design pressure and temperature are less than the RCS design limits, are provided with overpressure protective devices and redundant isolation means.

9.3.1.3 Spent Fuel Pool Cooling System

The Spent Fuel Pool Cooling System has been designed to remove decay heat from spent fuel stored in the pool. The system is currently analyzed to ensure heat removal capability equivalent to the decay heat generated by 1205 fuel assemblies (Reference 5). The original spent fuel racks were replaced in the north and south pools and racks were added to the north end of the fuel transfer canal, permitting a larger number of fuel assemblies to be stored in the pool (see Section 9.5). The expansion to 1205 assemblies was necessitated by not having an off-site fuel storage facility available.

The system is designed to maintain pool water temperature at 150°F when 1205 fuel assemblies are stored in the pool. With only a freshly discharged 1/3 core in the pool, the pool water will be maintained at a maximum of 140°F during normal conditions through control of cooling water flow.

The Spent Fuel Pool Cooling System will maintain the fuel pools at a maximum of 140°F during normal refueling and maintain a maximum temperature of 150°F following an entire core offload. The design temperature of the spent fuel pool cooling loop piping and valves is 120°F; however, these components have been analyzed to 400°F. A limit of 150°F is chosen to be consistent with other temperatures in the system.

Since there is a large capacity for heat absorption in the spent fuel pool, active system components are not redundant. Alternate cooling capability can be made available under anticipated malfunctions or failures.

In addition to the cooling mode, the system contains provisions for full flow filtering and for demineralization of a portion of the main circulation flow.

The spent fuel pool can not be drained as a result of component failure due to valving and piping arrangements.

9.3.1.4 Codes and Classifications

All piping and components of the Auxiliary Coolant System are designed to the applicable codes listed in Table 9.3-4. Carbon steel is used in the Component Cooling System since the component cooling water contains a corrosion inhibitor. Austenitic stainless steel is used throughout the RHR and the Spent Fuel Pool Cooling Systems.

9.3.2 System Design and Operation

9.3.2.1 Component Cooling System

The Component Cooling System consists of two component cooling pumps, two component cooling heat exchangers, a component cooling surge tank, cooling lines to the various components being cooled, and associated piping, valves, and instrumentation. Component cooling piping to the various components is arranged in parallel flow circuits.

During normal operation, the component cooling pumps and heat exchangers will be operated as needed to accommodate the heat removal loads. One pump and one heat exchanger can provide 100 percent heat removal capability during normal operation. Two pumps and two heat exchangers are utilized to remove the residual and sensible heat during plant shutdown. If one of the pumps or one of the heat exchangers is not operative, safe shutdown of the plant is not affected; however, the time for cooldown is extended.

Component cooling is provided for the following heat sources:

- Residual heat exchangers (Auxiliary Coolant System, ACS)
- Reactor coolant pumps (RCS)
- Letdown heat exchanger (Chemical and Volume Control System, CVCS)
- Excess letdown heat exchanger (CVCS)
- Seal water heat exchanger (CVCS)
- Boric acid evaporator (CVCS)
- Sample heat exchangers (Sampling System)
- Waste evaporator condenser (Waste Disposal System, WDS)
- Waste gas compressors (WDS)
- RHR pumps (ACS)
- Safety Injection pumps (SIS)
- Containment spray pumps (Containment Vessel Internal Spray System)

At the reactor coolant pump, component-cooling water removes heat from the bearing oil and the thermal barrier. Since the heat is transferred from the component cooling water to the service water, the component cooling loop serves as an intermediate system between the Reactor Coolant and Service Water Cooling Systems. This double barrier arrangement reduces the probability of leakage of high pressure, radioactive reactor coolant to the Service Water System.

The component cooling surge tank accommodates expansion, contraction and in-leakage of water, and ensures a continuous component cooling water supply until a leaking cooling line can be isolated. A radiation monitor in the component cooling header annunciates in the control room in the unlikely event that the radiation level reaches a preset level above the normal back ground.

Makeup water is taken from the demineralized water header to the component cooling surge tank. Emergency makeup is from the Service Water System.

The operation of the system is monitored with the following instrumentation:

- Temperature detectors in the common supply line to the component cooling pumps and outlet lines of the component cooling heat exchangers.
- A pressure detector on the line between the component cooling pumps and the component cooling heat exchangers
- A flow indicator in the component cooling return header
- A radiation monitor on the outlet header from the component heat exchangers
- A level indicator on the surge tank.

9.3.2.2 Residual Heat Removal System

The RHR System consists of two residual heat exchangers, two residual heat removal pumps, piping and the necessary valves and instrumentation. During plant shutdown to remove core decay heat, reactor coolant flows from the RCS to the residual heat removal pumps, through the tube side of the residual heat exchangers and back to the RCS. The heat loads are transferred by the residual heat exchangers to the component cooling water.

During plant shutdown the cooldown rate of the reactor is controlled by regulating the reactor coolant flow through the tube side of the residual heat exchangers. A bypass line and flow control valve (operated in manual or automatic) around the residual heat exchangers are used to maintain a constant flow through the RHR System.

During the fuel and control rod replacement operation the required number of RHR pumps and heat exchangers to be maintained in service is determined by the core decay heat.

After refueling the RHR pumps are used to drain the reactor-refueling cavity by pumping water from the Reactor Coolant System hot leg connection to the RWST, prior to head replacement. When the reactor head is in place, the normal RHR flow is re-established.

Redundant, remotely operated valving in the RHR System inlet lines is provided to isolate the system from the RCS. Two valves in series are provided with each valve interlocked (designed to meet the intent of IEEE-279) to prevent valve opening unless the primary system pressure is below the RHR System design pressure (see NRC, SER in Reference 1).

Overpressure in the system is prevented by two relief valves (RHR33 and RHR33-1) in the inlet piping which discharge to the Pressurizer Relief Tank and the Containment Sump B, respectively. A locked open valve (RHR45) in the line around LD-60 allows LD-5 to be used for discharge relief.

9.3.2.3 Spent Fuel Pool Cooling System

The Spent Fuel Pool Cooling System removes decay heat from fuel stored in the pool. The system is required to handle the heat generated by 1/3 of a freshly discharged core. However, the system is analyzed to remove the decay heat load of a full core offload during refueling (Reference 5 and Reference 6). Under abnormal conditions, which have been chosen as the design basis, the system will remove the heat generated by 1205 spent fuel assemblies, with one core freshly unloaded.

The Spent Fuel Pool Cooling System consists of two half-capacity pumps, a heat exchanger, two half-capacity filters, a demineralizer with pre- and post-filters, and associated piping, valves, and instrumentation. The spent fuel pool pumps draw water from the pool, circulate it through the filters and heat exchanger, and return it to the pool. Redundancy of this equipment is not required because of the large heat capacity of the spent fuel pool. If one of the two pumps should fail, the remaining pump could easily handle the heat load with 1/3 of a freshly discharged core in the pool. If a pump should fail while 1205 spent fuel assemblies are stored in the pool, sufficient time exists to either repair the failed pump or to connect a temporary pump in the system. Heat exchanger failure is not considered to be likely; however, tube plugging is a short-term operation and can be accomplished before a significant increase in pool temperature occurs. Alternate cooling provisions are available from the RHR heat exchanger through existing piping and valving arrangements. Since this alternate cooling would only be required when a freshly discharged core is stored in the spent fuel pool (i.e., no fuel in the reactor); it does not reduce safety features equipment availability.

The clarity and purity of the spent fuel pool water is maintained by passing the full cooling flow through the two half-capacity filters. Each filter is capable of removing particulates down to 15 microns in size. In addition, approximately 10 percent of the cooling flow is passed through a purification loop consisting of a pre-filter, demineralizer, and post-filter, and then is returned to the spent fuel pool.

The spent fuel pool pump suction lines are located well above the fuel assemblies and a system failure cannot result in loss of pool water. The return lines enter the pool above the top of the fuel assemblies and the lines contain check valves at the point of entry into the pool shielding concrete. Thus, line failure outside of the spent fuel pool cannot cause a loss of pool water due to siphon action.

Since the full spent fuel pool cooling flow is drawn from the pool surface through adjustable skimmer boxes, the Spent Fuel Pool Cooling System also serves as a skimming system, since the full cooling flow is filtered.

The refueling water purification pump is used to circulate water from the RWST through the purification loop of the Spent Fuel Pool Cooling System and back to the tank. This provides water cleanup capability following refueling.

During refueling, the refueling cavity water may be cleaned by rotating two spectacle flanges between the RHR System and the Spent Fuel Pool Cooling System. Water then is taken from the refueling cavity, via the reactor vessel and the reactor coolant loop hot leg, through the residual heat removal pump and heat exchanger. A portion then passes through the spent fuel pool filters and into the spent fuel pool. Water returns to the refueling cavity from the spent fuel pool via the fuel transfer tube.

Boron addition can be made to the spent fuel pool by use of the boric acid makeup portion of the CVCS.

9.3.2.4 Component Cooling System Components

Design parameters for the Component Cooling System components are presented in Table 9.3-1.

9.3.2.5 Component Cooling Heat Exchangers

Two component cooling heat exchangers are of the shell and horizontal straight tube type. Service water circulates through the tubes while component-cooling water circulates through the shell side. Each Component Cooling Heat Exchanger is designed to remove one-half of the heat load occurring during plant shutdown. The heat load during normal full power operation requires only one heat exchanger with the other serving as a standby unit. The tubes are Admiralty metal and the shells are carbon steel.

9.3.2.6 Component Cooling Pumps

The two component cooling pumps, which circulate component cooling water through the Component Cooling System, are horizontal, centrifugal units. The pump casings are made from corrosion erosion resistant material. The material thickness is dictated by the ability to withstand mechanical damage and the casings are thus substantially over designed from a stress level standpoint. Each Component Cooling Pump is designed to provide one-half the flow required during the plant cooldown operation. Normal full power operation requires the use of only one pump with the other serving as a standby unit.

9.3.2.7 Component Cooling Surge Tank

The component cooling surge tank accommodates changes in component cooling water volume and is constructed of carbon steel. In addition to piping connections, the tank has a capped connection at the top for the addition of the chemical corrosion inhibitor to the component cooling water. The surge tank accommodates surges resulting from thermal expansion and contraction, and also collects any leakage into the system from the components being cooled. In the event of a leaking cooling line, the component cooling surge tank will provide a supply of coolant until the leak has been isolated.

9.3.2.8 Component Cooling Valves

The valves used in the Component Cooling System are constructed of carbon steel with bronze or stainless steel trim. Since the component cooling water is not normally radioactive, special valve features (such as special leakoff line to the Waste Disposal System) to prevent leakage to the atmosphere are not provided.

Self-actuated spring loaded relief valves are provided for lines and components that could be pressurized to their design pressure by improper operation or malfunction.

A limit stop has been added to flow control valve CC-302, to limit the maximum CCW flow through the letdown heat exchanger. The purpose of the limit stop is to prevent the CCW pumps from entering a runout condition.

9.3.2.9 Component Cooling Piping

All Component Cooling System piping is carbon steel with welded joints and connections except at certain components where flanged connections are used to facilitate maintenance.

9.3.2.10 Residual Heat Removal System Components

Design parameters for the RHR System components are presented in Table 9.3-2.

9.3.2.11 Residual Heat Exchangers

The two residual heat exchangers are of the shell and U-tube type with the tubes welded to the tube sheet. Each of the residual heat exchangers is designed to remove one-half of the total loop heat load during plant cooldown. Reactor coolant circulates through the tubes, while component-cooling water circulates through the shell side. The tubes and other surfaces in contact with reactor coolant are austenitic stainless steel and the shell is carbon steel.

9.3.2.12 Residual Heat Removal Pumps

The two RHR pumps are vertical, centrifugal units with special seals to prevent reactor coolant leakage to the atmosphere. Each pump is sized to provide one-half of the maximum loop flow requirement during plant cooldown. All pump parts in contact with reactor coolant are austenitic stainless steel or equivalent corrosion resistant material.

9.3.2.13 Residual Heat Removal Valves

The valves used in the RHR System are constructed of austenitic stainless steel or equivalent corrosion resistant material. Manual stop valves are provided to isolate equipment for maintenance. Throttle valves are provided for remote and manual control of the residual heat exchanger tube-side flow. Check valves prevent reverse flow through the RHR pumps.

Valves that perform a modulating function or are larger than 2½ inches are equipped with two sets of packing and an intermediate leakoff connection that discharges to the Waste Disposal System.

Manually operated valves have backseats to facilitate re-packing and to limit the stem leakage when the valves are open.

9.3.2.14 Residual Heat Removal Piping

All RHR System piping is austenitic stainless steel. The piping is welded, with flanged connections at some components for ease of maintenance.

9.3.2.15 Spent Fuel Pool Cooling System Components

Design parameters for the Spent Fuel Pool Cooling System components are presented in Table 9.3-3. In addition to the design service water temperature condition of 66°F, the Spent Fuel Pool Cooling System has been analyzed and determined to be acceptable for elevated service water temperatures of up to 80°F (see Reference 5 and Reference 6).

9.3.2.16 Spent Fuel Pool Heat Exchanger

The spent fuel pool heat exchanger is of the shell and horizontal U-tube type with tubes welded into the tube sheet. Spent fuel pool water circulates through the tube side and is cooled by the service water flowing through the shell side. The shell material is carbon steel and the tube and head material is austenitic stainless steel.

9.3.2.17 Spent Fuel Pool Pumps

Two single-stage horizontal, end suction, centrifugal 50 percent capacity pumps circulate spent fuel pool water for cooling and purification. Pump operation is manually controlled from the Control Room. All wetted surfaces of the pumps are austenitic stainless steel.

9.3.2.18 Spent Fuel Pool Filters

Two 50 percent capacity filters remove particulate matter down to 15 microns from the total spent fuel pool cooling flow. The filter vessel is constructed of austenitic stainless steel.

9.3.2.19 Spent Fuel Pool Demineralizer

The spent fuel pool demineralizer is sized to pass approximately 60 gpm of the circulation flow and to provide adequate purification of the fuel pool for unrestricted access to the working area.

9.3.2.20 Spent Fuel Pool Valves

Manual stop valves are used to isolate equipment and lines. Flow control is provided for the purification loop. A self-contained constant flow control valve maintains a constant purification flow. Valves in contact with the spent fuel pool water are austenitic stainless steel.

9.3.2.21 Spent Fuel Pool Piping

All piping in contact with the spent fuel pool water is austenitic stainless steel. The piping is welded except where flanges are provided to facilitate maintenance.

9.3.3 System Evaluation

9.3.3.1 Availability and Reliability

9.3.3.1.1 Component Cooling System

The Component Cooling System provides cooling for the reactor coolant pumps and the excess letdown heat exchanger. To ensure cooling availability most of the piping, valves, and instrumentation are located outside the primary concrete shield of an elevation well above the anticipated post-accident water level in the bottom of the containment. The exceptions to this are the component cooling lines near the reactor coolant pumps. However, these lines could be isolated post-accident. The entire component cooling equipment inside containment is missile protected. Also, the radiation shielding provided by the reactor coolant pump vaults allows maintenance and inspections of the component cooling and other equipment in this area to be performed during power operation.

The Component Cooling System is an open system outside containment because the component cooling surge tank overflow line is connected to the waste holdup tank, which is vented to the Auxiliary Building vent exhaust ductwork. The double barrier concept required for containment integrity is maintained by:

1. meeting the nine criteria of ANSI/ANS 56.2 - 1984 for a closed system inside containment, and

2. the CC System motor operated valves outside containment CC-601A, CC-601B, CC-612A, CC-612B and CC-653 for penetrations 32N, 32E, 33N, 33E and 40, respectively, and the common header valve CC-600 for Penetration 39.

Outside the containment, the RHR pumps, the residual heat exchangers, the component cooling pumps and heat exchangers, and associated valves, piping and instrumentation can be maintained and inspected during power operation. System design provides for the replacement of one pump or one heat exchanger while the other units are in service.

Several of the components in the Component Cooling System are fabricated from carbon steel. The component cooling water contains a corrosion inhibitor to protect the carbon steel. Welded joints and connections are used except where flanged closures are employed to facilitate maintenance. The entire system is seismic Class I design. The components are designed to the codes given in Table 9.3-4. In addition, the components are not subjected to any high pressures (see Table 9.3-1) or stresses; hence, a rupture or failure of the system is very unlikely. Reactor coolant pump thermal barrier and connections are constructed to RCS design pressure and temperature requirements.

During the recirculation phase following a LOCA, one of the two component cooling water pumps delivers flow to the shell side of the residual heat exchangers.

9.3.3.2 Residual Heat Removal System

Two pumps and two heat exchangers are utilized to remove residual and sensible heat during plant cooldown. If one of the pumps and/or one of the heat exchangers is not operative, safe operation of the plant is not affected; however, the time for cooldown is extended. The function of this equipment following a LOCA is discussed in Section 6.5. The entire system is seismic Class I design. The components are designed to the codes given in Table 9.3-4.

9.3.3.3 Spent Fuel Pool Cooling System

This manually controlled system may be safely shut down for reasonable time periods for maintenance or replacement of malfunctioning components. The components are designed to the codes given in Table 9.3-4.

9.3.3.4 Leakage Provisions

9.3.3.4.1 Component Cooling System

Welded construction is used where possible throughout the Component Cooling System piping, valves and equipment to minimize the possibility of leakage.

The component cooling water could become contaminated with radioactive water due to a leak in any heat exchanger tube in the Chemical and Volume Control, the Sampling, or the

Auxiliary Coolant Systems, or a leak in the cooling coil for the reactor coolant pump thermal barrier.

Tube or coil leaks in components being cooled would be detected during normal plant operation by the leak detection system described in Section 6.5. Such leaks are detected by a radiation monitor located on the discharge header downstream of the component cooling water heat exchangers.

Leakage from the Component Cooling System can be detected by a falling level in the component cooling surge tank. The rate of water level decrease and the area of the water surface in the tank permit determination of the leakage rate.

The component, which is leaking, can be located by sequential isolation or inspection of equipment in the system. If the leak is in one of the component cooling water heat exchangers, the affected heat exchanger could be isolated and repaired. During normal operation the leaking exchanger could be left in service with leakage out of the Component Cooling System up to the capacity of the makeup line to the system from the primary water treatment plant.

Should a large tube-side to shell-side leak develop in a residual heat exchanger, the water level in the component cooling surge tank would rise and the operator would be alerted by a high water alarm. If the leaking residual heat exchanger fills the surge tank, the surge tank will overflow to the Auxiliary Building waste holdup tank through the surge tank overflow line.

The severance of a cooling line serving an individual reactor coolant pump cooler would result in substantial leakage of component cooling water. However, this piping is small as compared to the remainder of the component cooling system. Therefore, the water stored in the surge tank after a low-level alarm, together with makeup flow, provides ample time for the closure of the valves external to the components in the Component Cooling System.

The relief valves on the component cooling water header downstream from each of the reactor coolant pumps are designed with a capacity equal to the maximum rate at which reactor coolant can enter the Component Cooling System from a severance-type break of the reactor coolant pump thermal barrier cooling coil. The valve set pressure (150 psig) equals the design pressure of the component cooling piping.

The relief valves on the cooling water lines downstream from the sample, excess letdown, seal-water, letdown, and residual heat exchangers are sized to relieve the volumetric expansion occurring if the exchanger shell side is isolated when cold and high temperature coolant flows through the tube side. The set pressure equals the design pressure of the shell side of the heat exchangers.

In the remote event of a tube rupture in any one of the Component Cooling System heat exchangers, the overflow line will pass the maximum flow of water entering the surge tank to the waste holdup tank. During a heat exchanger tube rupture event, the pressure at the system low point (RHR pump vault) will be 173.4 psig, which is 15.6 percent over the system design pressure of 150 psig. This 15.6 percent over the system design pressure is within the 16 percent limit allowed by Section VIII of the ASME Boiler and Pressure Vessel Code.

9.3.3.5 Residual Heat Removal System

Welded construction is used where possible throughout the RHR System piping, valves, and equipment to minimize the possibility of leakage. During reactor operation, all equipment of the RHR System is idle and the associated isolation valves are closed. During the LOCA condition, water from the containment sump is recirculated through the exterior piping system. To obtain the possible total radiation dose to the public due to leakage from this system, the potential leaks have been evaluated and discussed in Chapter 6.

Each of the two RHR pumps is located in a shielded compartment with a floor drain. The leakage drains to the residual heat removal pump pit sump and is then pumped to the waste holdup tank by the sump pumps. Post-accident the collected leakage can be pumped into the deaerated drains tank. Each drain line has a remotely operated valve which automatically closes on high level within the RHR pump compartment which would indicate either massive failure, or inability of the sump pumps to handle the leakage. Thus, the remaining pump pit is protected from flooding by a failure in the other loop of the RHR System. Two 60-gpm sump pumps are provided and each is capable of handling the flow, which results from the failure of a RHR pump seal.

The sump has a high level alarm which will cause an alarm in the main Control Room on high water level. Each of the lines from the containment sump B to the individual RHR pumps has two remotely operated isolation valves in series.

9.3.3.6 Spent Fuel Pool Cooling System

If leaking fuel assemblies are transferred into the spent fuel pool, a small quantity of fission products may enter the pool water. The purification loop is provided to remove these fission products in order to maintain low water radioactivity levels.

The probability of inadvertently draining water from the spent fuel pool is essentially zero. Since the pump suction connections extend no more than 2 feet below normal water level, there is no possibility of inadvertently draining pool water below that level. Also, the pool water return lines enter the pool above the top of the fuel assemblies and contain check valves at the point of entry into the pool shielding concrete. Therefore, draining the pool below the level of the fuel by a siphon effect is not possible.

The results of the natural circulation analysis indicate that sufficient margin exists to bulk fluid boiling following a loss of forced flow with a full core offload (Reference 5). Both temperature and level indicators in the pool would alert the operator to a loss of cooling. Local and remote alarms are provided. This allows the operator to take corrective measures to restore cooling capability to the spent fuel pool-cooling loop. Complete loss of heat removal capability is not considered credible since there are two 50 percent capacity pumps available and alternate cooling provisions can easily be made with a RHR heat exchanger. In the event the entire core has to be unloaded and stored, and one pump is out of service, two safety Class I sources of water (the RWST and a 6" service water supply line) are available to provide the necessary cooling until the failed pump is placed into service. In an event of a loss of both SFP pumps and/or SFP heat exchanger, alternate cooling is provided by evaporative cooling process.

9.3.3.7 Incident Control

9.3.3.7.1 Component Cooling System

Although the Component Cooling System is a closed and missile protected system inside containment, containment isolation valves are available and may be closed, if needed. Each of the cooling water supply lines to the reactor coolant pumps contains a check valve inside and two remotely operated valves outside the containment wall. The cooling water supply line to the excess letdown heat exchanger contains a check valve inside and a manual valve backed up by a remotely operated valve outside the containment wall. The equipment vent and drain lines outside the containment have manual valves which are normally closed unless the equipment is being vented or drained for maintenance or repair operations.

Following a LOCA, one component cooling pump and one component cooling heat exchanger accommodate the heat removal loads during recirculation. If either a component cooling pump or component cooling heat exchanger fails, the standby pump and the standby heat exchanger provide 100 percent backup. Valves on the component cooling return lines from the safety injection, containment spray, and residual heat removal pumps are normally open. Each of the component cooling inlet lines to the residual heat exchangers has a normally closed remotely operated valve. If one of the valves fails to open at initiation of long-term recirculation, the valve, which does open supplies a heat exchanger with sufficient cooling capacity to remove the heat load.

If a break of a component cooling line occurs inside the containment, adequate valving is available outside the containment on the component cooling supply and return lines to isolate the leak (see Figure 9.3-3). None of the components inside the containment require component cooling water during recirculation. If a break of a component cooling line occurs outside the containment, the leak could either be isolated and repaired, or the system could be shutdown for repairs depending on the position in the system at which the break occurred. Access is available to required components. During this period, no heat removal from the containment by the RHR System is required since the Containment fan coil units capability using service water exceeds decay heat generation.

Once the leak is isolated or the break has been repaired, makeup water is supplied from the plant makeup water system or emergency makeup is provided by the Service Water System.

9.3.3.8 Residual Heat Removal System

The RHR System is connected to the hot legs of both reactor coolant loops on the suction side and to the cold leg of one reactor coolant loop on the discharge side. On the suction side, the connections are through two electric motor-operated gate valves in series. Once closed, each valve is interlocked to prevent opening unless the RCS pressure is below the RHR System design pressure. On the discharge side, the connection is through a check valve in series with an electric motor-operated gate valve. All of these are closed during normal operation.

Control Room annunciation and pressure relief valves on the common portion of the RHR suction line are provided to protect the reactor pressure vessel from non-ductile failure by a low temperature overpressurization transient and the RHR System from an overpressurization event. Valve RHR-33 is the original relief valve and provides overpressurization protection for the RHR System. The low temperature overpressure protection (LTOP) valve, RHR33-1 was added to ensure reactor vessel integrity at low operating temperatures (Reference 1, Reference 2, and Reference 3).

The KPS LTOP system was designed to mitigate the consequences of low temperature, overpressurization events that could result in exceeding allowable reactor pressure vessel fracture toughness limits. Low temperature overpressure events are pressure transient events, which occur at operating temperatures below the enable temperature (i.e. low temperature) as defined in ASME Boiler and Pressure Vessel Code, Case N-514, “Low Temperature Overpressure Protection.” ASME Boiler and Pressure Vessel Code, Case N-514 defines the enabling temperature as follows:

“The greater of 200°F or the limiting PRV adjusted nil-ductility reference temperature at the ¼T through-wall location plus 50°F.”

Based on the Appendix G pressure-temperature limits calculated through 33[1] effective full power years (EFPY), the KPS LTOP system is to be operable at or below the RCS temperature of 200°F.

Overpressurization transients can result from either a mass input or an energy input transient, and typically occur when the RCS is in a water solid condition. The LTOP design assumed as the most limiting mass input transient, an inadvertent start of one SI pump, with two RHR & RXCPs operating. The most limiting energy input (i.e. thermal expansion) transient assumed an initial starting of a RXCP with an assumed secondary to primary temperature difference of 100°F, with two RHR pumps operating.

The set pressure of RHR33-1 is 500 psig.

1. Although the curves were developed for 33 EFPY, they are limited to 28 EFPY (corresponding to the end of cycle 28) by Reference 4.

The LTOP System is considered operable when all four valves on the RHR suction piping (valves RHR-1A, 1B, 2A, & 2B) are open, and RHR33-1 is able to relieve RCS overpressure events without violating the maximum allowable pressure temperature limits calculated in accordance with 10 CFR 50, Appendix G. If one train of RHR suction is isolated, the LTOP System is considered operable if the suction valves and valve breakers in the other train are open and tagged open. If both trains of RHR suctions are isolated or valve RHR33-1 is inoperable, the LTOP system is considered operable if an alternate vent path is provided. This alternate vent path must have the same or greater pressure relieving capacity as the LTOP valve.

9.3.3.9 Spent Fuel Pool Cooling System

The most serious failure possible with this system is a complete loss of water in the spent fuel pool. Several provisions have been made to protect against this possibility.

The possibility of a line failure causing complete drainage is precluded by the fact that the suction lines do not extend more than 2 feet below normal operating level. This leaves a margin of 22 feet above the top of the fuel assemblies. The return lines enter the pool above the top of the fuel assemblies and have check valves installed immediately adjacent to their penetration into the spent fuel pool wall, thus preventing siphoning of water from the pool in the highly unlikely event of a line failure.

9.3.3.10 Malfunction Analysis

A failure analysis of pumps, heat exchangers and valves is presented in Table 9.3-5.

9.3.4 Test and Inspection Capability

The following components of the Component Cooling System have safeguards functions:

- Component Cooling Pumps
- Residual Heat Removal Pump Seals
- Safety Injection Pump Seals
- Containment Spray Pumps
- Residual Heat Removal Heat Exchanger

The pre-operational tests for the Component Cooling System include the following:

The Component Cooling System will be operated as required to check the automatic start feature of each Component Cooling Pump, the isolation capability of the Component Cooling Pumps and Heat Exchangers to verify the ability to isolate these components for repair or replacement while the system is still in operation. The isolation capability of each individual serviced component shall be checked to verify the ability to isolate any component for repair or replacement while the balance of the system is still in operation. The low flow alarm and the high flow isolation functions of the Reactor Coolant Pump cooling water flow controllers shall be checked to demonstrate proper setpoint and operation. The low flow alarms shall be checked for the Safety Injection, Containment Spray and Residual Heat Removal Pump gland coolers. Throttle valve positions, individual component flows and Component Cooling Pump flow capacities, shall be determined for normal plant operation, plant shutdown and plant cooldown.

The RHR pumps flow instrument channels can be calibrated during shutdown.

The active components of the Auxiliary Coolant Systems are in either continuous or intermittent use during normal plant operation. Thus, no additional periodic tests are required. Periodic visual inspections and preventive maintenance can be conducted as necessary without interruption of cooling system operation.

9.3 References

1. Letter from S. A. Varga (NRC) to D. C. Hintz (WPS), “Transmitting the NRC SER for Removal of Automatic Closure Feature on the Suction Side RHR Isolation Valves,” Letter No. K-85-21, January 16, 1985
2. Letter from S. A. Varga (NRC) to C. W. Geisler (WPSC), “Transmitting the DRAFT NRC SER and Technical Evaluation Report, for Proposed Amendment to the Operating License Relating to the Reactor Vessel Overpressure Mitigating System,” Letter No. K-83-156, August 2, 1983
3. Letter from C. R. Steinhardt (WPSC) to Document Control Desk (NRC), “Submitting Amended Response to GL 90-06 and Proposed Amendment No. 101C to the Technical Specifications, Relating to PORV, Block Valve Reliability and LTOP,” Letter No. NRC-93-082, May 5, 1993
4. Letter from William O. Long (NRC) to M. L. Marchi (WPSC), “Transmitting the NRC SER for Amendment No. 144 to the Operating License, Revising the Pressure/Temperature Limits and LTOP Requirements,” Letter No. K-99-039, April 1, 1999
5. Holtec International Corporation document HI-992245, “Bulk Temperature Analysis For The Kewaunee Spent Fuel Pools and Transfer Canal,” Revision Original
6. Calculation C11168, “SFPCS Heat Removal Capacity,” November 2, 2000
7. WCAP-16040-P, “Power Uprate Project, Kewaunee Nuclear Power Plant, NSSS and BOP Licensing Report,” February 2003 (Proprietary)

Table 9.3-1
Component Cooling System Component Data

Component Cooling Pumps	
Quantity	2
Type	Horizontal centrifugal
Design flow rate (each), gpm	3650
Total developed head, ft H_2O	220
Motor horsepower	250
Casing material	cast steel
Design pressure, psig	150
Design temperature, °F	200
Component Cooling Heat Exchangers	
Quantity	2
Type	Fixed tube sheet, horizontal
Design heat transfer, Btu/hr	28.9E+6
Shell side (component cooling water):	
Operating design inlet temperature, °F	110.8
Operating design outlet temperature, °F	95
Design flow rate, lb/hr	1.83E+6
Design temperature, °F	200
Design pressure, psig	150
Material	Carbon steel
Tube side (service water):	
Operating design inlet temperature, °F	66
Operating design outlet temperature, °F	89
Design flow rate, lb/hr	1.26E+6
Design pressure, psig	150
Design temperature, °F	200
Material	Admiralty
Component Cooling Surge Tank	
Quantity	1
Volume, gal	2000
Normal water volume, gal	1000
Design pressure (internal), psig	100
Design pressure (external), psig	Vacuum breaker provided
Design temperature, °F	200
Material	Carbon steel
Component Cooling System Piping and Valves	
Design pressure, psig	150
Design temperature, °F	200

Table 9.3-1

Component Cooling System Component Data

Reactor Coolant Pump thermal Barrier Piping	
Design pressure, psig	2485
Design temperature, °F	650

Table 9.3-2

Residual Heat Removal System Component Data

Reactor coolant temperature at startup of RHR, °F	< 400
Time to cool Reactor Coolant System from 350°F to 140°F, hr (all equipment operational at uprated power of 1772 MWt)	≤ 18
Refueling water storage temperature, °F	40 (see Chapter 4 and 5)
Decay heat generation at 20 hr after shutdown condition, Btu/hr (at uprated power of 1772 MWt)	37.4E+6
H_3BO_3 concentration in RWSTs, ppm boron	≥ 2400
RHR pumps:	
Quantity	2
Type	Vertical centrifugal
Design point flow rate (each), gpm	2000
Design point - Total developed head, ft H_2O	280
Motor horsepower	200
Material	SS
Design pressure, psig	600
Design temperature, °F	400
RHR pump room sump pumps:	
Quantity	2
Type	Canned
Typical operating capacity, gpm	60
Typical operating head, ft	55
Motor rating, hp	3.35
Material (wetted surface)	Austenitic stainless steel
Residual heat exchangers:	
Quantity	2
Type	Shell and U-tube, vertical
Design heat transfer, Btu/hr	26.OE+6
Shell side (component cooling water):	
Operating design inlet temperature, °F	95
Operating design outlet temperature, °F	116.1
Design flow rate, lb/hr	1.25E+6
Design pressure, psig	150
Design temperature, °F	350
Material	Carbon steel

Table 9.3-2

Residual Heat Removal System Component Data

Tube side (reactor coolant):	
Operating design inlet temperature, °F	160
Operating design outlet temperature, °F	133.5
Design flow rate, lb/hr	1.0E+6
Design pressure, psig	600
Design temperature, °F	400
Material	Stainless steel

Table 9.3-3
Spent Fuel Pool Cooling System Component Data

Spent Fuel Pool heat exchanger:	
Quantity	1
Type	Shell and U-tube, horizontal
Design heat transfer, Btu/hr	8.5E+6
Shell side (service water):	
Operating design inlet temperature, °F (see Note [a])	66
Operating design outlet temperature, °F	96.9
Design flow rate, lb/hr	275,000
Design pressure, psig	150
Design temperature, °F	150
Material	Carbon steel
Tube side (Spent Fuel Pool water):	
Operating design inlet temperature, °F	120
Operating design outlet temperature, °F	100
Design flow rate, lb/hr	425,000
Design pressure, psig	150
Design temperature, °F	150
Material	Austenitic stainless steel
Spent Fuel Pool pump data:	
Quantity	2
Type	Horizontal, centrifugal
Design flow rate, gpm	450
Total developed head, ft H_2O	200
Motor horsepower	40
Design pressure, psig	300
Design temperature, °F	150
Material	Austenitic stainless steel
Spent Fuel Storage Pool:	
Normal water level, ft.	39
Boron concentration, ppm	2100
Spent Fuel Pool filter:	
Quantity	2
Type	Replaceable cartridge
Internal design pressure of housing, psig	150
Design temperature, °F	150
Design flow rate, gpm	450

Table 9.3-3
Spent Fuel Pool Cooling System Component Data

Spent Fuel Pool demineralizer:	
Quantity	1
Type	Flushable
Design pressure, psig	200
Design temperature, °F	250
Design flow rate, gpm	60
Resin volume, cu ft	20
Vessel volume, cu ft	27
Refueling water purification pump:	
Quantity	1
Type	Vertical, centrifugal
Design flow rate, gpm	85
Total developed head, ft H_2O	270
Design pressure, psig	150
Design temperature, °F	120
Spent Fuel Pool cooling loop piping and valves:	
Design pressure, psig	125
Design temperature, °F (See Note[b])	120
Refueling water purification loop piping and valves:	
Design pressure, psig	150
Design temperature, °F	120

a. The design service water temperature is 66°F; however, the Spent Fuel Pool Cooling System has been analyzed up to 80°F.

b. The design temperature of the spent fuel pool cooling loop piping and valves is 120°F; however, these components have been analyzed to 400°F. A limit of 150°F is chosen to be consistent with other design temperatures in the table.

Table 9.3-4
Auxiliary Coolant System Code Requirements

Component cooling heat exchangers	ASME VIII*
Component cooling surge tank	ASME VIII
Component cooling system piping and valves	USAS B31.1**
Residual heat exchangers	ASME III***, Class C, tube side; ASME VIII, shell side
Residual heat removal piping and valves	USAS B31.1
Spent fuel pool filter	ASME III, Class C
Spent fuel pool heat exchanger	ASME III, Class C, tube side; ASME VIII, shell side
Spent fuel pool demineralizer	ASME III, Class C
Spent fuel pool system piping and valves	USAS B31.1

* ASME VIII - American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section VIII.

** USAS B31.1 - Code for Pressure Piping, and special nuclear cases where applicable.

*** ASME III - American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section III, Nuclear Vessels.

Table 9.3-5
Auxiliary Coolant System Failure Analyses

Component	Failure	Comments and Consequences
1. Component cooling water pumps	Rupture of pump casing	The casing and shell are designed for 150 psi and 200°F, which exceed maximum operating conditions. The pumps are inspectable and protected against credible missiles. Rupture is not considered credible.
2. Component cooling water pumps	Pump fails to start	One operating pump will supply sufficient flow for emergency cooling.
3. Component cooling heat exchanger	Tube or shell rupture	Rupture is considered improbable because of low operating pressures. Each heat exchanger can be isolated and one heat exchanger can carry the total emergency heat load.
4. RHR pump	Rupture of a pump casing	The casing and shell are designed for 600 psi and 400°F. The pump is protected from overpressurization by two normally closed valves in the pump suction line and by an open relief line, containing a relief valve, back to the pressurizer relief tank. The pump is inspectable and is located in the Auxiliary Building, protected against credible missiles. Rupture is considered unlikely, but in any event the pump can be isolated.
5. RHR pump	Pump fails to start	One operating pump furnishes half of the flow required to meet design cooldown rate. This increases the time necessary for plant cooldown.
6. Remote operated valves inside containment in the residual heat removal pump suction lines	Valve fails to open in one of the suction lines	One suction path is adequate for plant cooldown; however, the time required to cool down may increase slightly.
7. Remote operated valves inside containment on the RHR pump discharge line	Valve fails to open	Pump discharge pressure gauge shows pump shutoff head indicating no flow. The low head safety injection lines may be opened and utilized to direct flow directly into the reactor vessel. A reactor coolant pump must be operated.

Table 9.3-5
Auxiliary Coolant System Failure Analyses

Component	Failure	Comments and Consequences
8. Residual heat exchanger	Tube or shell rupture	Rupture is considered unlikely, but, in any event, the faulty heat exchanger may be isolated.
9. Spent fuel pool pump	Rupture of a pump casing	The casing and shell are designed for 300 psig and 150°F, which exceeds maximum operating conditions. The pump is inspectable and is located in the Auxiliary Building, protected against credible accidents. Rupture is considered unlikely; however, pump can be isolated.
10. Spent fuel pool pump	Pump stops running and cannot be started	The system will operate at less than full capacity until stopped pump is repaired or replaced. The system can adequately remove heat from the pool following removal of one-third of the core from the reactor. In the event the entire core has to be unloaded and stored, and one pump is out of service, two safety Class I sources of water (the RWST and a 6" service water supply line) are available to provide the necessary cooling until the failed pump is placed into service.
11. Spent fuel pool heat exchanger	Tube or shell Rupture	Rupture is considered unlikely; however, faulty heat exchanger can be isolated. Tube plugging is a short-term operation and can be accomplished before a significant increase in pool temperature occurs. Alternate cooling provisions are available from a RHR heat exchanger through existing piping and valving arrangements. This alternate cooling would only be required when the core is unloaded and stored in the spent fuel pool (i.e., no fuel in the reactor); thus, it does not reduce safety features equipment availability.

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Figure 9.3-1 Residual Heat Removal System Flow Diagram

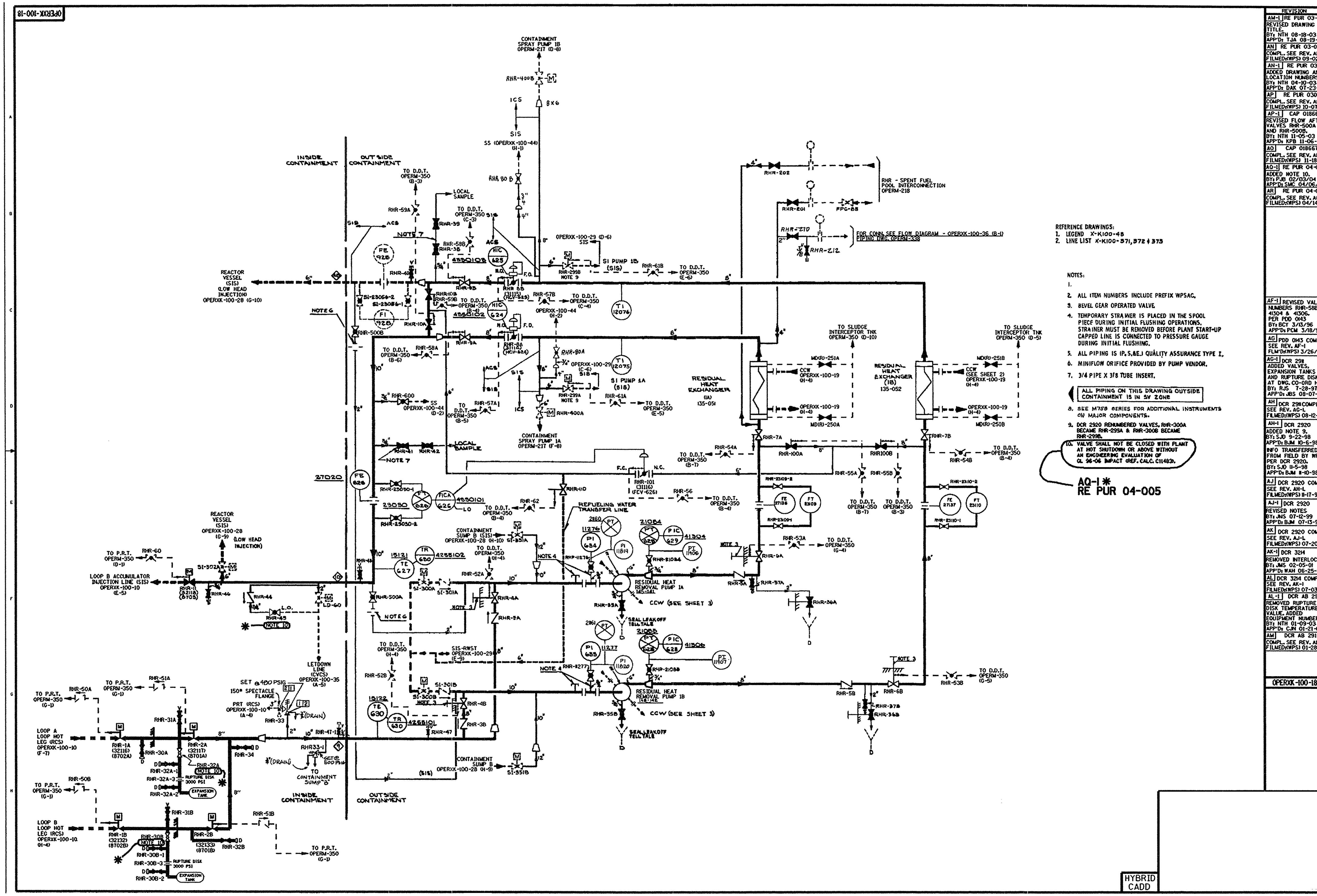


Figure 9.3-2 Component Cooling System Flow Diagram

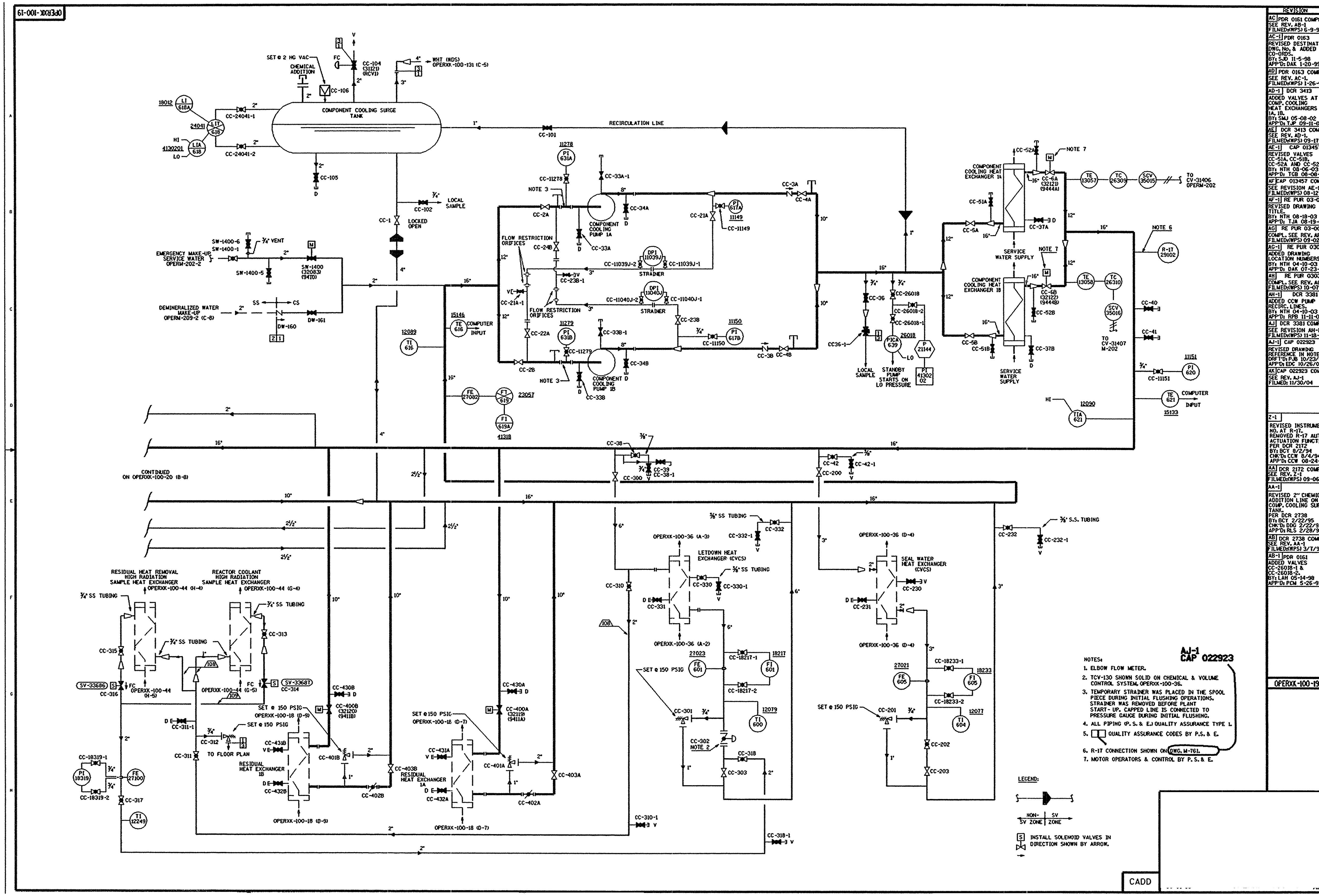


Figure 9.3-3 Component Cooling System Flow Diagram

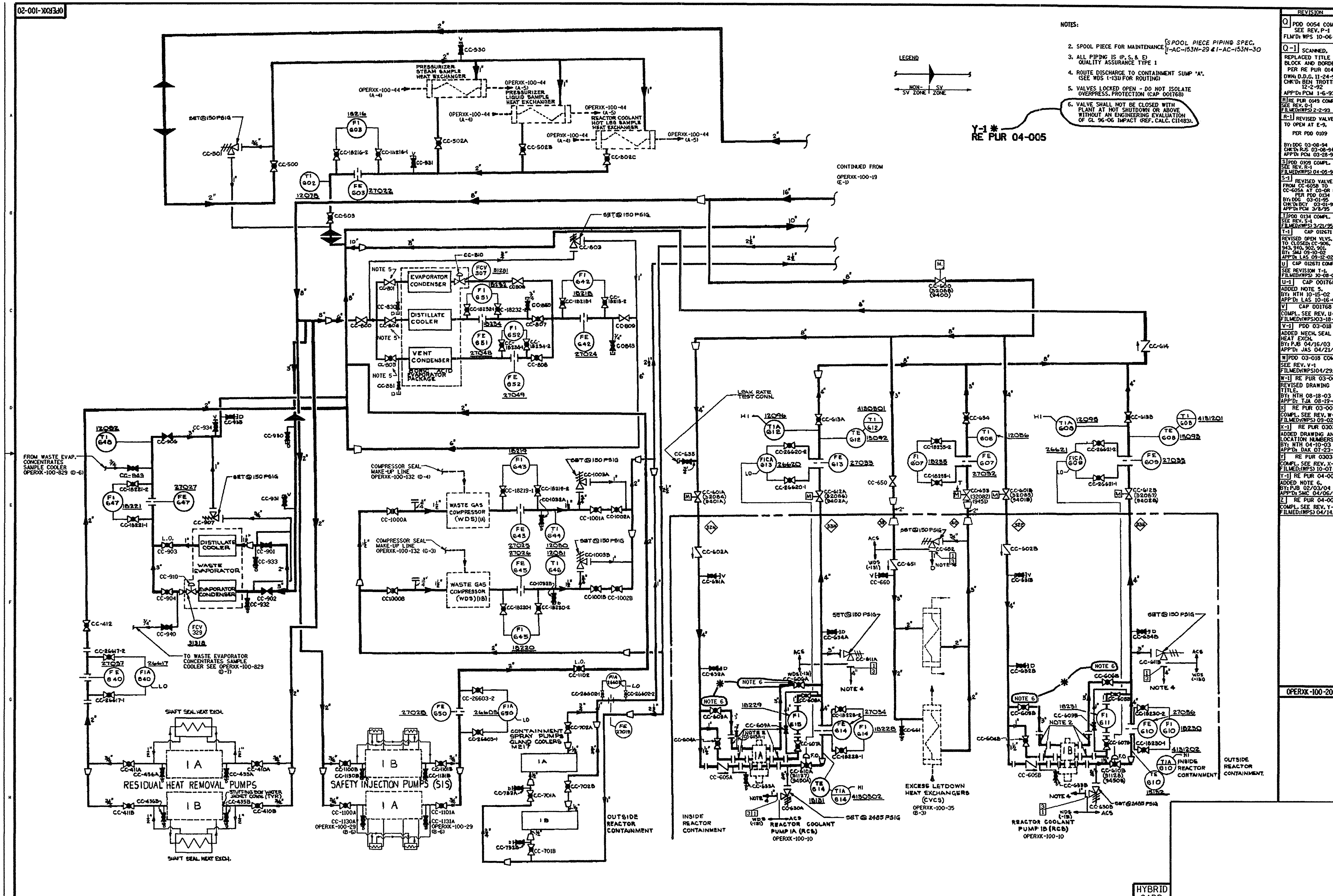
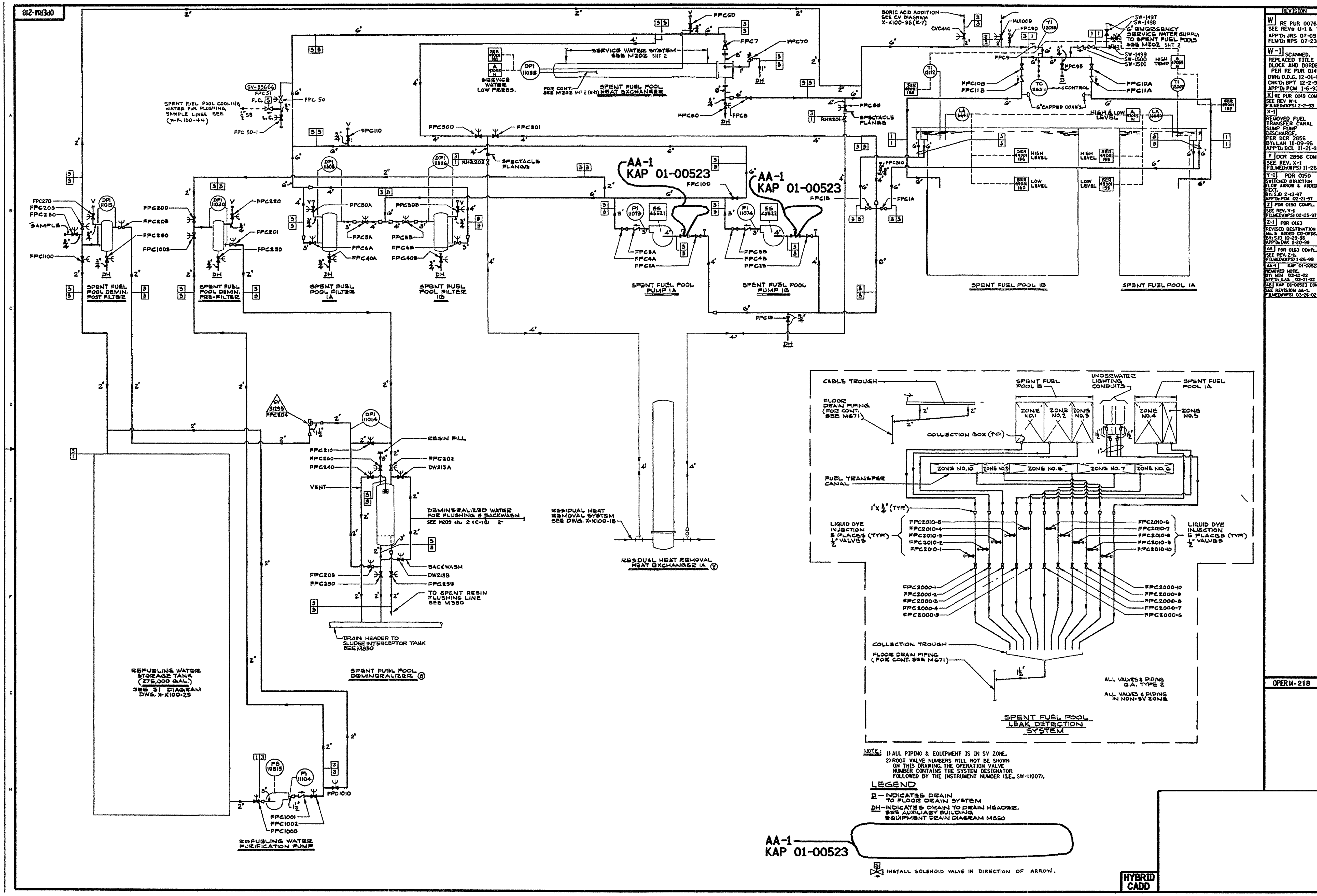


Figure 9.3-4 Spent Fuel Pool Cooling and Cleanup Systems-Flow Diagram



9.4 SAMPLING SYSTEM

9.4.1 Design Basis

9.4.1.1 Performance Requirements

This system provides fluid samples for laboratory analysis to evaluate reactor coolant, and other reactor auxiliary systems chemistry during normal operation. It has no emergency function. This system is normally isolated at the containment boundary.

Sampling System discharge flows are limited under normal and anticipated fault conditions (malfunctions or failure) to preclude any fission product releases beyond the limits of 10 CFR 20.

9.4.1.2 Design Characteristics

The system is capable of obtaining reactor coolant samples during reactor operation and during cooldown when the system pressure is low and the Residual Heat Removal System is in operation. Access is not required to the containment.

Sampling of other process coolants, such as tanks in the Waste Disposal System, is accomplished locally. Equipment for sampling secondary and non-radioactive fluids is separated from the equipment provided for reactor coolant samples. Steam generator blowdown samples can be taken either in the sample room or at the secondary system analytical and sampling panel (see Figure 10.2-9). Leakage and drainage resulting from the sampling operations are collected and drained to tanks in the Waste Disposal System.

Two types of samples are obtained by the system: high temperature, high-pressure RCS samples which originate inside the reactor containment; and low temperature, low-pressure samples from the CVCS and Auxiliary Coolant System.

9.4.1.3 High Pressure - High Temperature Samples

A sample connection is provided from each of the following:

- The pressurizer steam space
- The pressurizer liquid space
- One hot leg of the RCS
- Steam generator liquid.

9.4.1.4 Low Pressure - Low Temperature Samples

A sample connection is provided from each of the following:

- The mixed bed demineralizer inlet header

- The mixed bed demineralizer outlet header
- The RHR System, just downstream of the heat exchangers
- The volume control tank gas space.

9.4.1.5 Expected Operating Temperatures

The high-pressure, high temperature samples and the RHR System samples leaving the sample heat exchangers are held to a temperature of 130°F.

9.4.1.6 Codes and Standards

System component code requirements are given in Table 9.4-1.

9.4.2 System Design and Operation

The Sampling System, shown in Figure 9.4-1, provides the representative samples for laboratory analysis. Analysis results provide guidance in the operation of the RCS, Auxiliary Coolant System, and CVCS. Analyses show both chemical and radiochemical conditions. Typical information obtained includes reactor coolant boron and chloride concentrations, fission product radioactivity level, hydrogen, oxygen, and fission gas content, activated corrosion product concentration, and chemical additive concentration.

The information is used in regulating boron concentration adjustments, evaluating fuel element integrity and mixed bed demineralizer performance, and regulating additions of corrosion-controlling chemicals to the systems. The Sampling System is designed to be operated manually on an intermittent basis. Samples can be withdrawn under conditions ranging from full power to cold shutdown.

Sampling System equipment is located inside the Auxiliary Building with most of it in the sampling room. The delay coil and sample lines with remotely operated valves are located inside the Reactor Containment Vessel.

Reactor coolant hot-leg liquid, pressurizer liquid, and pressurizer steam samples originating inside the Reactor Containment Vessel flow through separate sample lines to the sampling room. Each of these connections to the RCS has a remote operated root valve located close to the sample source. The samples pass through the reactor containment, to the Auxiliary Building, and into the sampling room, where they are cooled (pressurizer steam samples condensed and cooled) in the sample heat exchangers. Before leaving the containment, the reactor coolant hot leg samples pass through a delay coil to assure sufficient elapsed time for N^{16} decay. The sample stream pressure is reduced by a manual-throttling valve located downstream of each sample pressure vessel. The sample stream is purged to the volume control tank in the CVCS until sufficient purge volume has passed to permit collection of a representative sample. After sufficient purging, the sample pressure vessel is isolated pending laboratory analysis of the contents.

Alternately liquid sample may be collected by bypassing the sample pressure vessels. After sufficient purge volume has passed to permit collection of a representative sample, a portion of the sample flow is diverted to the sample sink where the sample is collected.

The reactor coolant sample originating from the RHR System of the Auxiliary Coolant System has a remote-operated, normally-closed isolation valve located close to the sample source. The sample line from this source is connected into the sample line coming from the hot leg at a point upstream of the sample heat exchanger. Samples from this source can be collected either in the sample pressure vessel or at the sample sink, as with hot-leg samples.

Liquid samples originating at the CVCS letdown line at the mixed bed demineralizer inlet and outlet pass directly through the purge line to the volume control tank. Samples are obtained by diverting a portion of the flow to the sample sink. If the pressure is low in the letdown line, the purge flow is directed to the drain header in the Waste Disposal System. The sample line from the gas space of the volume control tank, delivers gas samples to the sampling room. Purge flow for these samples is discharged to the vent header in the Waste Disposal System.

The sample sink, which is contained inside a fume hood, contains a drain line to the Waste Disposal System.

Local instrumentation is provided to permit manual control of sampling operations and to ensure that the samples are at suitable temperatures and pressures before diverting flow to the sample sink.

9.4.2.1 Components

A summary of principal component data is given in Table 9.4-2.

9.4.2.2 Sample Heat Exchangers

There are three sample heat exchangers; one for the pressurizer steam sample, one for the pressurizer liquid sample, and one for the reactor coolant loop hot leg and the RHR loop samples. These shell-and-tube sample heat exchangers reduce the temperature of samples from the pressurizer steam space, the pressurizer liquid space, and the reactor coolant to 130°F before samples reach the sample vessels and sample sink. The tube side of the heat exchangers is austenitic stainless steel; the shell side is carbon steel.

The inlet and outlet tube sides have socket-weld joints for connections to the high-pressure sample lines. Connections to the component cooling water lines are socket-weld joints. The samples, flow through the tube side and component-cooling water from the Auxiliary Coolant System circulates through the shell side.

9.4.2.3 Delay Coil

The reactor coolant sample line contains a delay coil, consisting of coiled tubing, which has sufficient length to provide at a minimum a forty seconds sample transit time within the containment and an additional twenty seconds transit time from the reactor containment to the sampling hood. This allows for decay of short-lived isotopes to a level that permits normal access to the sampling room.

9.4.2.4 Sample Pressure Vessels

The high-pressure sample trains and the RHR System sample train can be directed through a pressure vessel to obtain liquid or gas samples at RCS design temperature and pressure. Isolation valves are furnished with the vessel. The vessels, valves and couplings are austenitic stainless steel.

9.4.2.5 Sample Sink

The sample sink is located in a hooded enclosure, which is equipped with an exhaust ventilator. The work area around the sink and the enclosure is large enough for sample collection and storage for radiation monitoring equipment. The sink perimeter has a raised edge to contain any spilled liquid. In addition to the incoming sample lines, the enclosure is penetrated by a demineralized water line, which discharges into the sink. The sink and work areas are stainless steel.

9.4.2.6 Piping and Fittings

All liquid and gas sample lines are austenitic stainless steel tubing and are designed for high-pressure service. Socket welded joints[1] are used throughout the Sampling System. Lines are located as to protect them from accidental damage during routine operation and maintenance.

9.4.2.7 Valves

Remotely operated stop valves are used to isolate all sample points and to route sample fluid flow inside the reactor containment. Manual stop valves are provided for component isolation and flow path control at all normally accessible Sampling System locations. Manual throttle valves are provided to adjust the sample flow rate as indicated on Figure 9.4-1.

Check valves in the purge line prevent gross reverse flow of gas from the volume control tank into the sample sink.

All valves in the system are constructed of austenitic stainless steel or equivalent corrosion resistant material.

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1. Socket welded joints are specified for pipe-to-pipe connections. Piping to tubing connections may be made with fittings that allow welded connections at the pipe end and swaged connections at the tube end, provided that the swaged fittings meet or exceed the maximum working pressure of the tubing.

An isolation valve is provided, outside the reactor containment on all sample lines leaving the containment, which trips closed upon actuation of the containment isolation signal.

9.4.3 System Evaluation

9.4.3.1 Leakage Provisions

Leakage of reactor coolant from this system within the containment is evaporated to the containment atmosphere and removed by the cooling coils of the Containment Air Cooling System. Leakage of sampling fluid, from the most likely places outside the containment, is collected by placing the entire sampling station under a hood provided with a connection to the plant vent. Liquid leakage from the valves in the hood is drained to the Waste Disposal System.

9.4.3.2 Incident Control

The system operates on an intermittent basis and under administrative manual control.

Qualifications, criteria, and organization for chemistry capability and experience at this plant are given in the Dominion Nuclear Facility Quality Assurance Program Description. Analysis of reactor coolant and other samples for boron content will be done in the plant chemistry laboratory by qualified chemistry personnel.

Minimum normal frequencies for sampling and analysis are listed in the Technical Specifications.

9.4.3.3 Malfunction Analysis

To evaluate system safety, the failures or malfunctions are assumed concurrent with a LOCA, and the consequences analyzed. The results are presented in Table 9.4-3.

From this evaluation it is concluded that proper consideration has been given to plant safety in the design of the system.

9.4.3.4 High Radiation Sampling System

Sampling provisions following a major fuel damaging accident are provided by the High Radiation Sampling System (HRSS). The HRSS has been installed to specifically address the requirements of TMI Action Item II.B.3 of NUREG-0737 and provides the capability to obtain and analyze reactor coolant and containment atmospheric samples following an accident which results in major fuel damage, without exceeding the allowable exposure limits for emergency response personnel. The system is isolated at the containment boundary, and discharge flows from the system are limited under normal and anticipated fault conditions (malfunction or failure) to preclude any fission product release beyond the limits of 10 CFR 20 (See NRC SERs in Reference 1 and Reference 2).

Based on an analysis performed by Westinghouse (Reference 3) and acceptance by the NRC (Reference 4), the requirements contained in NUREG 0737 for post-accident sampling have been changed. These changes allowed for the elimination of the requirement for post-accident sampling as long as the following recommended actions were performed:

1. Contingency plans for obtaining and analyzing highly radioactive sample of reactor coolant, containment sump and containment atmosphere post-accident shall be maintained.
2. The capability for classifying fuel damage events at the alert level threshold shall be maintained.
3. The capability to monitor radioactive iodines that have been released to off-site environments shall be maintained.

9.4 References

1. NRC Safety Evaluation Report, S. A. Varga (NRC) to C. W. Giesler (WPS), Letter No. K-83-161, August 2, 1983
2. NRC Safety Evaluation Report, S. A. Varga (NRC) to D. C. Hintz (WPS), Letter No. K-85-126, June 20, 1985
3. WCAP-14986, Revision 2, "Post-Accident Sampling System Requirements: A Technical Basis," July 2000
4. NRC Safety Evaluation Report, John G. Lamb (NRC) to Mark Reddemann (NMC), Letter No. K-02-006, January 16, 2002, Issuance of Amendment No. 160, which eliminated the requirements for the Post-Accident Sampling System

Table 9.4-1
Sampling System Code Requirements

Sample heat exchanger	ASME III*, Class C, tube side; ASME VIII**, shell side
Piping and valves	USAS B31.1 (1967)***

* ASME III - American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section III, Nuclear Vessels.

* ASME VIII - American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section VIII, Unfired Pressure Vessels.

*** SAS B31.1 - Code for Pressure Piping and special nuclear cases where applicable.

Table 9.4-2
Sampling System Components

Sample Heat Exchanger

General	
Number	3
Type	Shell and coiled-tube
Design heat transfer rate (duty for 652.7°F sat. steam to 127°F liquid), each, Btu/hr	2.14E+5
Shell	
Design pressure, psig	150
Design temperature, °F	350
Component cooling water flow, gpm	40
Operating cooling water temperature, in (max.), °F	105
Operating cooling water temperature, out (max.), °F	130
Material	Carbon steel
Tubes	
Design pressure, psig	2485
Design temperature, °F	680
Sample flow, normal, each, lb/hr	~209
Operating sample temperature, in (max.), °F	653
Operating sample temperature, out (max.), °F	127
Material	Austenitic stainless steel
Sample Pressure Vessels	
Number, total	1
Volume, ml	97
Design pressure, psig	2485
Design temperature, °F	680
Manual Throttle Valves	
Normal Operating temperature, °F	120-130
Design pressure, psig	2485
Body design temperature, °F	680
Piping	
Design pressure, psig	2485
Design temperature, °F	680

Table 9.4-3
Malfunction Analysis of Sampling System

Sample Trains	Malfunction	Comments and Consequences
Pressurizer steam space sample, pressurizer liquid space sample, hot-leg sample	Remotely operated sampling valve inside reactor containment fails to close	Remotely operated valve outside the reactor containment closes on containment isolation signal
Any of the above sample trains	Sample line break outside containment	Same as above

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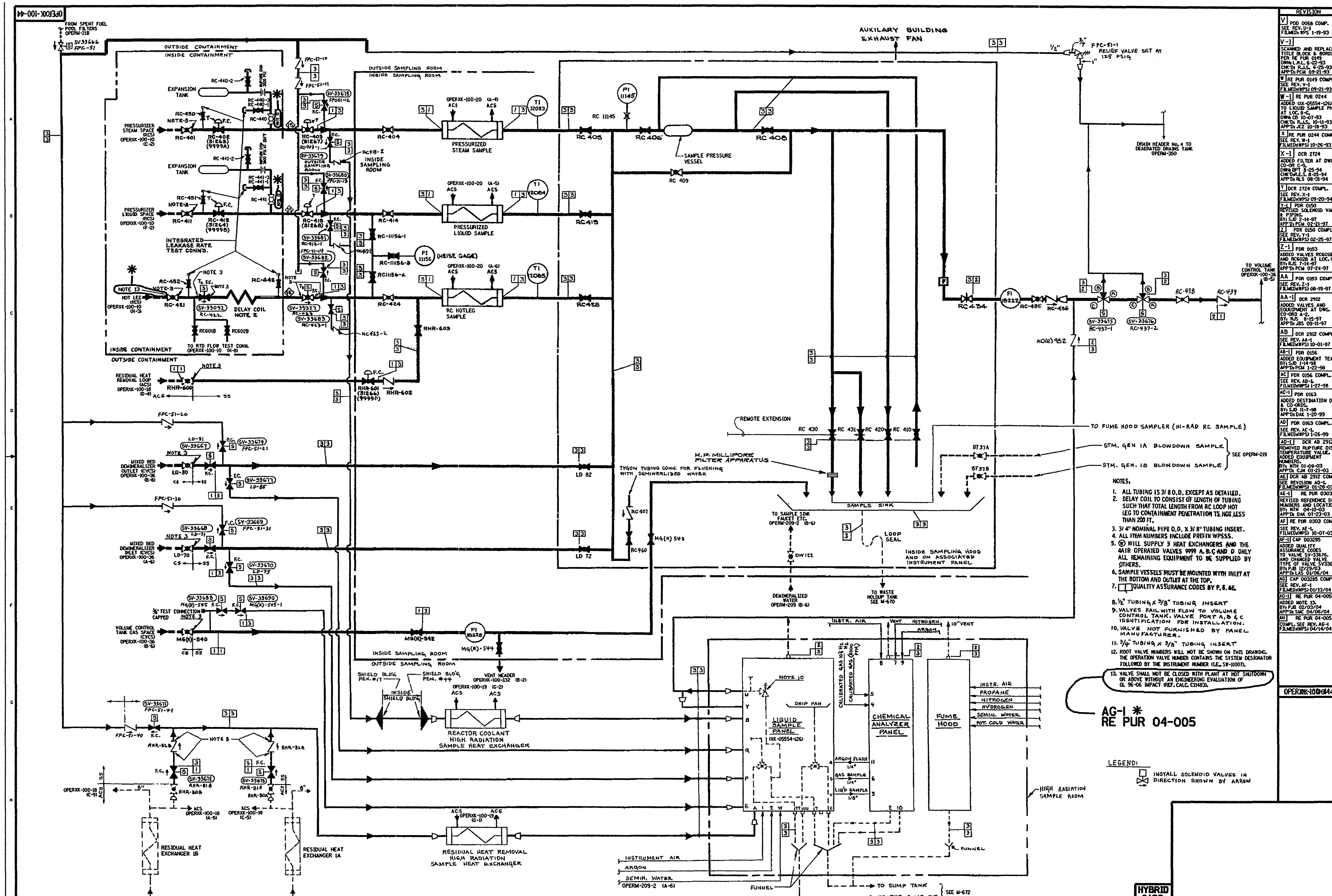
Figure 9.4-1 Sampling System



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9.5 FUEL HANDLING SYSTEM

The Fuel Handling System provides a safe and effective means of transporting and handling fuel from the time it reaches the plant in an unirradiated condition until it leaves the plant after postirradiation cooling. The system is designed to minimize the possibility of mishandling that could cause fuel damage and potential fission-product release.

The Fuel Handling System consists basically of:

- The reactor refueling cavity, which is flooded only during plant shutdown for refueling;
- The spent fuel pool, which is kept full of water during and after the first refueling and is accessible to operating personnel; and
- The Fuel Transfer System, consisting of an underwater conveyor that transports fuel assemblies between the reactor refueling cavity and the spent fuel pool.

9.5.1 Design Basis

9.5.1.1 Prevention of Fuel Storage Criticality

Criterion: Criticality in the new and spent fuel storage pools shall be prevented by physical systems or processes. Such means, as geometrically safe configurations shall be emphasized over procedural controls (GDC 66).

The new and spent fuel storage racks, which have accommodations as defined in Table 9.5-1, are designed so that it is impossible to insert assemblies in other than the prescribed locations. In addition, the spent fuel pool has areas set aside for accepting the spent-fuel shipping casks. Cask loading is done underwater. Borated water is used to fill the spent fuel storage pool at a concentration to match that used in the refueling cavity during refueling operations. The fuel in the spent fuel pool and new fuel storage pit is stored vertically in an array with sufficient center-to-center distance or solid neutron absorber between assemblies to assure K(eff) $\leq$ 0.95, even if unborated water were to fill the space between the assemblies.

Detailed instructions and procedures are available for use by refueling personnel. These instructions and procedures, combined with the design of the fuel handling facilities and the fuel handling equipment incorporating built-in interlocks and safety features, provide assurance that no incident could occur during the refueling operations that would result in a hazard to public health and safety. Kewaunee complies with 10 CFR 50.68.

9.5.1.2 Fuel and Waste Storage Decay Heat

Criterion: Reliable decay heat removal systems shall be designed to prevent damage to the fuel in storage facilities and to waste storage tanks that could result in radioactivity release which would result in undue risk to the health and safety of the public (GDC 67).

The refueling water provides a reliable and adequate cooling medium for spent fuel transfer. Heat removal from the spent fuel pool is provided by the Spent Fuel Pool Cooling System, which is discussed in Section 9.3.

9.5.1.3 Fuel and Waste Storage Radiation Shielding

Criterion: Adequate shielding for radiation protection shall be provided in the design of spent fuel and waste storage facilities (GDC 68).

Adequate shielding for radiation protection is provided during reactor refueling, by conducting all spent fuel transfer and storage operations underwater. This permits visual control of the operation at all times while maintaining personnel exposure during refueling operations ALARA. Spent Fuel Pool water level is indicated by a level transmitter, which causes an audible alarm in the Control Room on high or low level. Water removed from the spent fuel storage pool must be pumped out since there are no gravity drains.

Gamma radiation is continuously monitored in the Auxiliary Building. A high radiation level signal is alarmed locally and is annunciated in the Control Room.

9.5.1.4 Protection Against Radioactivity Release From Spent Fuel and Waste Storage

Criterion: Provisions shall be made in the design of fuel and waste storage facilities such that no undue risk to the health and safety of the public could result from an accidental release of radioactivity (GDC 69).

All fuel storage facilities are contained and equipment designed so that accidental releases of radioactivity directly to the atmosphere are monitored and do not exceed the guidelines of 10 CFR 50.67.

The reactor refueling cavity, fuel transfer canal, and spent fuel storage pool are reinforced concrete structures with seamwelded stainless steel plate liners. These structures are designed as Class I structures to withstand the anticipated earthquake loadings so that the liner prevents leakage even in the event the reinforced concrete develops cracks. All operating areas in the fuel storage facilities are ventilated. The exhausts of the ventilation system in the Waste Storage and Drumming Areas are monitored for radioactivity and are discharged via the vent through the top of the Auxiliary Building.

9.5.2 System Design and Operation

The reactor is refueled with equipment designed to handle the spent fuel underwater from the time it leaves the reactor vessel until it is placed in a cask for shipment from the site. Boric acid is present in the water to ensure sub-critical conditions during refueling.

The Fuel Handling System, shown in Figure 9.5-1, may be generally divided into two areas: the reactor refueling cavity, flooded only during plant shutdown for refueling, and the spent fuel pool. These two areas are connected by the Fuel Transfer System consisting of an underwater conveyor that carries the fuel through an opening in the Reactor Containment Vessel.

The refueling cavity is flooded with borated water from the RWST. In the refueling cavity, fuel is removed from the reactor vessel, transferred through the water and placed in the Fuel Transfer System by a manipulator crane. In the fuel transfer canal the fuel is removed from the Fuel Transfer System with a long manual tool suspended from an overhead hoist and placed in storage racks in the spent fuel pool.

New fuel assemblies are received and stored in accordance with approved plant procedures and Technical Specifications. New fuel is delivered to the reactor by taking it through the transfer system. The new fuel storage pit is sized for storage of the fuel assemblies and control rods normally associated with the replacement of one-third of a core.

Fuel handling data are given in Table 9.5-1.

9.5.2.1 Major Structures Required For Fuel Handling

9.5.2.1.1 Reactor Refueling Cavity

The reactor refueling cavity is a reinforced concrete structure that forms a pool above the reactor when it is filled with borated water for refueling. The cavity is filled to a depth that limits the measured radiation levels at the surface of the water to 50 mr/hr during those brief periods when a fuel assembly is transferred over the reactor vessel flange.

The reactor vessel flange is sealed at the bottom of the reactor-refueling cavity by a segmented seal system, which prevents leakage of refueling water from the refueling cavity. This seal is installed after reactor cooldown but prior to flooding the cavity for refueling operations.

Also, following cooldown, prior to flooding the refueling cavity, seals or nozzle dams can be installed in the hot and cold leg nozzles of the steam generators. With nozzle dams in place, work can continue in the steam generator primary channel head while refueling operations occur.

The refueling cavity is large enough to provide storage space for the reactor upper internals, the spare RCC drive shafts, and miscellaneous refueling tools. Space is also allowed for the storage of the lower internals if required.

The floor and sides of the refueling cavity are lined with stainless steel.

A portion of the floor of the cavity is at a lower elevation than the reactor vessel flange to provide the greater depth required for the fuel transfer system tipping device, the temporary storage of the upper and lower internals, and the RCCA changing fixture located in the cavity. The fuel transfer tube enters the reactor containment and protrudes through the end of the cavity.

9.5.2.2 Refueling Water Storage Tank

The normal function of the RWST is to supply borated water to the refueling cavity for refueling operations, and the capacity of the tank is sized on this basis. In addition, the tank provides borated water for the supply to the SI System. This is described in Chapter 6.

The water in the tank is borated to a concentration which assures reactor shutdown by at least 5 percent $\Delta K/K$ when all RCCAs are inserted and when the reactor is cooled down for refueling.

The tank design parameters are given in Chapter 6.

9.5.2.3 Spent Fuel Pool

The spent fuel pool is designed for the underwater storage of spent fuel assemblies, RCCAs, burnable poison rod assemblies, thimble plugs, etc. after their removal from the reactor. The pool is sized to accommodate a total of 1205 assemblies (Reference 6) and a fuel-shipping cask.

The spent fuel storage pool is divided into three storage compartments and a fuel transfer canal as shown in Figure 9.5-2. The storage compartments and the fuel transfer canal are connected by fuel transfer slots except for the opening between the north pool and the canal pool, these slots can be closed off with pneumatically sealed gates. The slot between the north pool and the canal pool must remain open whenever fuel is stored in the canal pool to provide for proper cooling. The elevations of the slot bottoms is above the elevation of the top of the stored spent fuel. The pneumatically sealed gate inflation system is supplied by the instrumentation air system whose air compressors are connected to the emergency diesel generators. The instrument air supply is backed up by a nitrogen cylinder supply.

The original spent fuel racks have been replaced with high-density spent fuel racks, permitting a larger number of spent fuel assemblies to be stored in the pool. These high-density racks located in the north and south pools, are double-walled stainless steel cans with a boron carbide plate sandwiched in between. The center-to-center spacing has been reduced from 21 inches to 10 inches minimum, thereby increasing the maximum combined storage capability of the north and south pools from 168 assemblies to 990 assemblies (see NRC SER in Reference 4).

An additional storage pool was created at the north end of the fuel transfer canal. The canal pool has the capacity to store 215 fuel assemblies. The racks in the canal pool are designed to store old (cool) spent fuel assemblies. The racks are constructed from stainless steel materials and use Boral as the neutron absorber (Reference 6).

Spent fuel assemblies are handled by a long-handled tool suspended from an overhead monorail electric hoist and manipulated by an operator standing on a movable bridge over the pool.

The spent fuel pool structure, with its 3-foot to 6-foot thick walls, is designed as a Class I structure that fully meets the seismic and tornado design criteria given in Appendix B. The fuel pool structure is also designed to withstand the hydraulic pressure of the contained water, as well as other credible static and dynamic loadings.

In addition, the spent fuel storage structure has been designed to minimize the loss of water due to a dropped cask accident. The bottom of the pool is a reinforced concrete slab 5 feet 11 inches thick that conforms to the ACI Standard Code Requirements for Reinforced Concrete (ACI-318-63) and attains a minimum compressive strength of 4000 psi. All reinforcing bars are made from intermediate grade, new billet steel and conform to ASTM A 615 Grade 60 specifications.

All inside surfaces of the spent fuel pool are lined with type 304 stainless steel plate with a minimum thickness of 3/16 inch. A leakage detection and collection system is also provided.

9.5.2.4 Spent Fuel Storage Racks (North and South Pool)

The spent fuel storage racks are ruggedly constructed of fusion-welded, type 304 stainless steel structural members. Individual storage compartments are provided in which the top of the fuel is below the top flange of the fuel storage compartments. Storage racks in 9 x 10 arrays are arranged in parallel rows. The fuel assemblies are spaced on 10-inch centers (minimum) for high-density storage. The individual arrays are fastened together by bars, which run horizontal near the top and bottom of the rack assembly forming a unified lattice arrangement. Each can is welded to the lattice forming the rack assembly as a single rigid structure. This framework is capable of absorbing considerable impact energy without permanent deformation.

9.5.2.5 Spent Fuel Storage Racks (Canal Pool)

The racks installed in the canal pool are constructed of type 304 stainless steel structural members. Individual storage compartments (cells) are provided in which the top of the stored fuel is below the top edge of the cells. The racks consist of 4 free standing rack modules. One of the modules contains an array that is 5 cells wide by 10 cells long. The other three modules contain 5 cell by 11 cell arrays. The cell walls are welded to the walls of adjacent cells along the vertical seams. The cells are also welded to the base plate of each module. The base plates are supported by adjustable pedestals that rest on stainless steel bearing pads that are located on the floor of the fuel transfer canal.

9.5.2.6 New Fuel Storage

New fuel assemblies and new inserts are stored in a separate new fuel storage pit, adjacent to the spent fuel storage pool, as shown in Figure 9.5-2. This separate storage pit is designed to hold 44 new fuel assemblies in specially constructed racks, and is utilized primarily for the storage of the α replacement core.

9.5.2.7 Major Equipment Required For Fuel Handling

9.5.2.7.1 Reactor Vessel Stud Tensioner

The stud tensioner is hydraulically operated (oil as the working fluid) device provided to permit pre-loading and unloading of the reactor vessel closure studs at cold shutdown conditions. Stud tensioners were chosen in order to minimize the time required for the tensioning or unloading operations. Three tensioners are provided and they are normally applied simultaneously to three studs 120 degrees apart. One hydraulic pumping unit operates the tensioners, which are hydraulically connected in parallel. The studs are tensioned to their operational load in two steps to prevent high stresses in the flange region and unequal loadings in the studs. Relief valves are provided on each tensioner to prevent over tensioning of the studs due to excessive pressure.

9.5.2.8 Reactor Vessel Head Lifting Device

The reactor vessel head lifting device consists of a welded and bolted structural steel frame with suitable rigging to enable the crane operator to lift the head and store it during refueling operations.

9.5.2.9 Reactor Internals Lifting Device

The reactor internals lifting device is a structural frame used in moving the reactor internals. The device is lowered onto the upper or lower internals and is manually bolted to the support plate by three bolts. Bushings on the fixture engage guide studs mounted on the vessel flange to provide close guidance during removal and replacement of the internals package.

9.5.2.10 Manipulator Crane

The manipulator crane is a rectilinear bridge and trolley crane with a vertical mast extending down into the refueling water. The bridge spans the refueling cavity and runs on rails set into the floor along the edge of the refueling cavity. The bridge and trolley motions are used to position the vertical mast over a fuel assembly in the core. A long tube with a pneumatic gripper on the end is lowered out of the mast to grip the fuel assembly. The gripper tube is long enough so the upper end is still contained in the mast when the gripper end contacts the fuel. A winch mounted on the trolley raises the gripper tube and fuel assembly up into the mast tube. The fuel is transported while inside the mast tube to its new position. The manipulator can lift only one fuel assembly at a time.

All controls for the manipulator crane are mounted on a removable console, which is located on the trolley. The bridge is positioned on a coordinate system laid out on one rail. A closed-circuit television system mounted on the console and bridge indicates the position of the bridge. The trolley is positioned with the aid of a scale on the bridge structure. The scale is read directly by the operator at the console. The drives for the bridge, trolley, and winch are variable speed and include inching controls. Electrical interlocks and limit switches on the bridge and trolley drives protect the equipment. A limit switch in the weight indicator parallels the gripper engage switch. The potential exists to lift a fuel assembly, which could be held by a disengaged gripper. Before the operator sees the indication on the remote digital readout, the fuel assembly could drop. The parallel limit switch provides one additional interlock to back up on operator's action. In an emergency, the bridge, trolley, and winch can be operated manually using a handwheel on the motor shaft.

Safety features are incorporated in the system as follows:

1. travel limit switches on the bridge and trolley drives;
2. bridge, trolley, and winch drives which are mutually interlocked to prevent simultaneous operation of any two drives;
3. a position safety switch, the GRIPPER TUBE UP position switch, which prevents bridge and trolley main motor drive operation except when it is actuated or in inching mode;
4. an interlock, which prevents the opening of a solenoid valve in the air line to the gripper except when zero suspended weight is indicated by a force gage. As backup protection for this interlock, the mechanical weight-actuated lock in the gripper prevents operation of the gripper under load even if air pressure is applied to the operating cylinder;
5. an overload interlock, which opens the hoist drive circuit in the up direction when the loading is in excess of a preset limit;
6. an interlock on the hoist drive circuit in the up direction, which permits the hoist to be operated only when either the OPEN or CLOSED indicating switch on the gripper is actuated; and
7. an interlock of the bridge and trolley drives, which prevents the bridge drive from traveling beyond the edge of the core unless the trolley is aligned with the refueling cavity centerline. The trolley drive is locked out when the bridge is beyond the edge of the core.

Suitable restraints are provided between the bridge and trolley structures and their respective rails to prevent derailing. The manipulator crane is designed to prevent disengagement of a fuel assembly from the gripper in the event of a Design Basis Earthquake.

9.5.2.11 Spent Fuel Pool Bridge

The spent fuel pool bridge is a gantry crane spanning the spent fuel pool, which carries electric monorail hoists on an overhead structure. The fuel assemblies are moved within the spent fuel pool by means of a long-handled tool suspended from the hoist. The hoist travel and tool length are designed to limit the maximum lift of a fuel assembly to a safe shielding depth.

9.5.2.12 Auxiliary Building Crane

The Auxiliary Building crane which is used, for handling spent fuel shipping casks is designed to minimize the possibility of dropping such a cask. The fuel handling crane has been designed, fabricated, and qualified in accordance with the Electric Overhead Crane Institute Standard No. 61, American National Standard Institute Standard B-30.2.0, 1967 Edition (and Pioneer Service and Engineering Company Standard Specification for Powerhouse Overhead Electrical Traveling Cranes). The rated load on the crane main hook and cable is 125 tons. All crane structural members have been designed to withstand impact loads per applicable specifications. A seismic evaluation has been performed for the loaded condition. Numerous safety features have been incorporated in the design of the crane. Among these are the following:

1. design of the hoist cables incorporates a design factor of five, based on the rated load and efficiency of lifting tackle,
2. all parts subjected to dynamic forces, such as gears, shafts, drums, blocks and other integral parts, have a design factor of five;
3. two separate magnetic holding brakes are provided as well as an eddy-current control brake. Each magnetic brake provides a braking force in excess of 150 percent of rated load. The eddy-current brake assures that smooth lowering and hoisting speeds can be maintained regardless of the load on the hook and that the load can be safely lowered even if both magnetic brakes fail;
4. the crane is capable of raising, lowering and transporting occasional loads of 125 percent of rated load without damage or distortion to any crane part;
5. the crane is provided with a type-302 stainless steel cable with a high-strength steel core. The cable is made up of 6 strands consisting of 37 wires to each strand;
6. during acceptance inspection, the hook was subjected to a 200 percent overload test followed by magnetic particle inspection;
7. motor controllers with at least five uniformly proportional steps for control in each direction are provided for all crane motions. Built-in time delays are provided between steps. This will provide a smooth uniform acceleration, which eliminates sudden jerking of the cask;
8. a fail-safe remote radio control system is provided for the crane. A selector switch determines whether the radio controls or master controls are used. All of the safety features built into the control system apply when either the radio transmitter or master control is used. The radio has a system for transmitting and receiving signals so it is not credible to duplicate this signal by other means. The signals used are different for every crane, so the transmitter for other cranes will not actuate the receiver panel for this crane.

9.5.2.13 Fuel Transfer System

The Fuel Transfer System shown in Figure 9.5-1 is an out-of-water electric gear motor driven transfer car. The gear motor unit drives a vertical line shaft, which extends down into the canal. A transfer of motion occurs in a series of bevel gears and an outboard sprocket, which drives the transfer car. The gear motor is equipped with overload protection devices to prevent damage to fuel assemblies and conveyor hardware. The travel of the transfer car is limited by an out-of-water limit switch system. The transfer car runs on tracks extending from the refueling cavity through the transfer tube and into the fuel transfer canal. The conveyor car receives a fuel assembly in the vertical position from the manipulator crane. The fuel assembly is lowered to a horizontal position for passage through the tube and then is raised to a vertical position in the fuel transfer canal.

During plant operation the conveyor car is stored in the fuel transfer canal. The gate valve is closed and a blind flange is bolted on the transfer tube to seal the reactor containment.

9.5.2.14 Rod Cluster Control Changing Fixture

A fixture is mounted on the reactor refueling cavity wall for removing RCCAs from spent fuel assemblies and inserting them into new fuel assemblies. The fixture consists of a guide tube mounted to the wall for containing and guiding the RCCAs, and a wheel-mounted carriage for holding and positioning fuel assemblies under the guide tube. The guide tube contains a pneumatic gripper on a winch that grips the RCCA and lifts it out of the fuel assembly. By repositioning the carriage, a new fuel assembly is brought under the guide tube and the gripper lowers the RCCA and releases it. The manipulator crane loads and removes the fuel assemblies into and out of the carriage.

9.5.3 System Evaluation

9.5.3.1 Incident Protection

9.5.3.1.1 Fuel Handling

Underwater transfer of spent fuel provides essential ease and corresponding safety in handling operations. Water is an effective, economic, and transparent radiation shield and a reliable cooling medium for removal of decay heat.

Basic provisions to ensure the safety of refueling operations are:

1. Gamma radiation levels in the Control Room and fuel storage areas are continuously monitored as described in Section 11.2.3. These monitors provide an audible alarm at the initiating detector indicating an unsafe condition. Continuous monitoring of reactor neutron flux provides immediate indication and alarm in the Control Room of an abnormal core flux level.

2. Violation of containment integrity is not permitted when the reactor vessel head is removed unless adequate shutdown margin is maintained.
3. Whenever new fuel is added to the reactor core, a reciprocal curve of source range neutron count-rate is plotted to verify the sub-criticality of the core.

Direct communication between the control room operator and the refueling cavity manipulator crane operation is available whenever changes in core geometry are taking place. This provision allows the control room operator to inform the manipulator crane operator of any impending unsafe condition detected from the main control board indicators during fuel movement.

An analysis is presented in Section 14.2.1 concerning damage to one complete assembly, assumed as a conservative limit for evaluating environmental consequences of a fuel handling accident. This analysis takes no credit for removal of fission products by the Spent Fuel Pool Ventilation System.

9.5.3.2 Tornado Missile

From Section 2.7.2 and a review of literature on tornado considerations (Reference 1, Reference 2 and Reference 3) in the design of spent fuel storage facilities, it may be concluded that:

1. the probability of a tornado occurrence is in the order of 1 in 20,000;
2. massive loss of water due to tornado-induced wind forces or tornado-generated missile cannot happen;
3. both the water cushion that remains above the stored spent fuel and the spent fuel storage racks afford a high degree of protection to stored fuel from tornado missiles.

The spent fuel pool structure and fuel storage racks provide adequate protection for stored spent fuel against the damaging effects of all credible tornado missiles. The structure precludes missile entry from all directions except for the open tops of the pools.

Results of an evaluation of a spectrum of tornado-driven missiles indicates that a 4-inch diameter or smaller pipe, wooden debris, or metal panel siding ripped off the building are the most probable sources of missiles that can impact the fuel pool. The design criteria for these tornado missiles are discussed in Appendix B. Wooden debris may enter the fuel pool, but such debris would not damage the fuel because of the buoyant force exerted by the approximately 25 feet of water that is expected to remain over the stored fuel. Even considering the highly improbable removal of pool water due to tornado action, investigations (Reference 5) have concluded that adequate protection from a wide spectrum of tornado missiles is provided by a minimum of 20 feet of water above the spent fuel with any resultant site boundary dose due to damage to the spent fuel an order of magnitude below 10 CFR 100 guidelines. In the unlikely event a metal siding panel passed into a fuel pool, the panel would not be expected to damage the fuel because the smallest projected edge would exceed the largest dimension of any opening in a fuel storage compartment. It is also expected that such a metal panel would, at most, cause only minor damage to a fuel storage rack.

The missile evaluation also indicated it was possible for a building girt, used to support and fasten the metal siding, to become a missile. As a result, special fasteners for the siding panels have been used to eliminate girts as potential tornado missiles. This is accomplished by employing blowout pressure relief panels whose fasteners are designed to fail before failure of the girts to which the panels are attached. Fasteners for all other panels attached to girts are designed so that if a girt fails, it will remain attached to the panel. Such panels are unlikely to enter the fuel storage area because of their geometry.

In the unlikely event that a 4-inch pipe becomes a missile, the 25 feet of water covering the fuel would create hydrodynamic forces that would tend to minimize the possibility of an end-on hit. Even with an end-on hit the fuel racks would afford significant protection to the stored fuel.

9.5.3.3 Cask Handling

The north spent fuel pool will be used for loading the spent fuel shipping cask; interlocks on the Auxiliary Building crane, prevent the transport of heavy loads such as the shipping cask over the spent fuel pool.

The spent fuel cask handling overhead crane has been provided with an interlock system which precludes the trolley passing over spent fuel storage areas. Redundant limit switches are furnished to assure that the exclusion area is not inadvertently traversed by the malfunction of a limit switch.

An override feature is provided to administratively allow free movement of the trolley when spent fuel is not stored in the pool. The override is achieved by the use of a key lock switch. The key will be under the control of the shift manager.

The protection provided to minimize the effects of a dropped cask accident is presented in Table 9.5-2.

9.5.3.4 Malfunction Analysis

An analysis is presented in Section 14.2.1 concerning damage to one complete fuel assembly, assumed as a conservative limit for evaluating environmental consequences of a fuel-handling incident. Analyses are also presented for evaluating the environmental consequences of turbine and tornado missiles in the spent fuel pool.

9.5.4 Test and Inspection Capability

Upon completion of core loading and installation of the reactor vessel head, certain mechanical and electrical tests were performed prior to initial criticality. The electrical wiring for the rod drive circuits, the rod position indicators, the reactor trip circuits, the in-core thermocouples and the reactor vessel head water temperature thermocouples were tested at the time of installation. The tests were repeated on these electrical items before initial plant operation.

Prior to cask lifting operations, an operation test of the Auxiliary Building crane was conducted with no load on the hook. This test verified that all controls are operating correctly. Following these tests, a load test was conducted which raised a simulated fuel cask approximately one foot off its carrier (reference Construction Test No. CT-C.04). The prime purpose of this test was to verify that there is no load movement after a fixed period of time.

9.5 References

1. Doan, P. L., "Tornado Considerations for Nuclear Power Plant Structures, Nuclear Safety 11, 4:296-308," July-August 1970
2. Miller, D. R., and W. A. Williams, "Tornado Protection for Spent Fuel Pit, General Electric Report APED 5696," November 1968
3. Doan, P. L., "Tornado and Tornado Effect Considerations for Nuclear Plant Structures Including Spent Fuel Storage Pool, Trans. Amer. Nucl. Soc., 12:167-8," June 1969
4. NRC Safety Evaluation Report, A. Schwencer (NRC) to E. W. James (WPS), letter, December 1, 1978
5. WCAP-7572, "Effect of Tornado Missiles on Stored Spent Fuel," September 1970
6. Lamb, J. B. (NRC) to M. E. Reddemann (NMC) transmitting the NRC SER for Amendment No. 150 to the Operating License, allowing 215 spent fuel assemblies to be stored in the new north canal pool, Letter No. K-01-009, January 23, 2001

Table 9.5-1
Fuel Handling Data

New Fuel Storage Pit:	
Core storage capacity	1/3+
Equivalent fuel assemblies	44
Center-to-center spacing of assemblies, in.	21
Maximum k_{eff} with unborated water	0.9
Spent Fuel Storage Pool (North and South Pools Combined):	
Core storage capacity	8.18
Equivalent fuel assemblies	990
Number of space accommodations for spent fuel shipping casks	1
Minimum Center-to-center spacing of assemblies, in.	10
Maximum k_{eff} with unborated water	<0.95
Spent Fuel Storage Pool (Canal Pool):	
Storage capacity (fuel assemblies)	215
Maximum K_{eff} with unborated water (See Note 1)	< 0.95
Nominal rack cell lattice spacing, in.	8.3
Miscellaneous Details:	
Width of refueling canal, ft	4
Wall thickness for spent fuel storage pool, ft	4 to 6
Weight of fuel assembly with RCCA (dry), lb	1390
Capacity of RWST, gal	276,500
Minimum contents of RWST, gal	272,500
Quantity of water required for refueling, gal	275,000

Note 1: Burnup and fuel assembly age restrictions apply.

Table 9.5-2

Design Conformance with Safety Guide 13

Regulatory Position		Kewaunee Design Feature	
Section 1.	The spent fuel storage facility (including its structures and equipment except as noted in Section 6 below) should be designed to Category I seismic requirements.	Item 1.	The Kewaunee Spent Fuel Storage Structure is designed as a Class I seismic structure and fully meets the seismic criteria given in Appendix B of the USAR.
Section 2.	The facility should be designed to prevent cyclonic winds and missiles generated by these winds from causing significant loss of watertight integrity of the fuel storage pool and to prevent missiles generated by cyclonic winds from contacting fuel within the pool.	Item 2.	The Kewaunee Spent Fuel Storage Facility, as a Design Class I structure, fully meets the tornado protection criteria given in Appendix B of the USAR. Thus, the full storage pool affords adequate protection against loss of watertight integrity from tornado missiles. In addition, the tornado missile evaluation presented in Section 9.5.3 of the USAR concludes that the spent fuel pool structure and fuel storage racks provide adequate protection for stored spent fuel against the damaging effects of all credible tornado missiles.
Section 3.	Interlocks should be provided to prevent cranes from passing over stored fuel (or near stored fuel, in a manner that could result in tipping the load over on stored fuel in the event of crane failure) when fuel handling is not in progress. During fuel handling operations, the interlocks may be bypassed and administrative control used to prevent the crane from carrying loads that are not necessary for fuel or other prohibited areas. The facility should be designed to minimize the need for bypassing such interlocks.	Item 3.	The spent fuel storage pool is divided into three storage compartments and a fuel transfer canal as shown in Figure 9.5-2 of the USAR. The south pool is reserved for the storage of spent fuel. The north pool is currently reserved for loading the spent fuel cask and interim storage of spent fuel. The canal pool is reserved for old (cool) spent fuel. The Auxiliary Building Crane design has been modified to provide interlocks that prevent moving any heavy loads over the north and south pools and the canal pool. Administrative controls will be such that no heavy loads will be transported over, or placed in any of the pools, when fuel is stored in that pool.

Table 9.5-2

Design Conformance with Safety Guide 13

Regulatory Position		Kewaunee Design Feature	
Section 4.	A controlled leakage building should be provided enclosing the fuel pool. The building should be equipped with an appropriate ventilation and filtration system to limit the potential release of radioactive materials. The building need not be designed to withstand extremely high winds but leakage should be suitably controlled during refueling operations. The design of the ventilation and filtration system should be based on the assumption that the cladding of all of the fuel rods in one fuel bundle might be breached. The inventory of radioactive materials available for leakage from the building should be based on the assumptions given in the Regulatory Guide, entitled “Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident for Boiling and Pressurized Water Reactors.”	Item 4.	The fuel storage pool is housed in a steel frame/metal siding portion of the Auxiliary Building. The spent fuel pool is ventilated by an air sweep system, which maintains a continuous air sweep across the pool when the system is operating. Exhaust air from the system passes through HEPA filters before being discharged to atmosphere through the monitored Auxiliary Building Vent. This system design was modified to provide a charcoal filtration capability that is used whenever fuel damage could occur in the spent fuel pool, as specified in Technical Specifications.

Table 9.5-2
Design Conformance with Safety Guide 13

Regulatory Position		Kewaunee Design Feature	
Section 5.	The spent fuel storage facility should have the following provisions with respect to the handling of heavy loads, including the refueling cask: a. Cranes capable of carrying heavy loads should be prevented, preferably by design rather than by interlocks, from moving into the vicinity of the pool, or b. The fuel pool should be designed to withstand, without leakage, which could uncover the fuel, the impact of the heaviest load to be carried by the crane from the maximum height to which it can be lifted. If this latter approach is followed, design provisions should be made to prevent this crane, when carrying heavy loads, from moving in the vicinity of stored fuel.	Item 5.	As pointed out under Section 3 above, the Auxiliary Building Crane Design was modified to provide interlocks to prevent heavy loads passing over stored spent fuel. In addition, the spent fuel storage structure has been designed to minimize the loss of water due to a dropped cask accident. Further, because fuel shipping cask handling is restricted to the small north pool, a dropped cask accident that might damage the bottom of the small pool will not affect the watertight integrity of the liner in the large pool. This conclusion is predicated on the cask drop accident assumptions and analytical results discussed in Section 9.5.3.
Section 6.	Drains, permanently connected system and other features that by maloperation or failure could cause loss of coolant that would uncover fuel should not be installed or included in the design. Systems for maintaining water quality and quantity should be designed so that any maloperation or failure in such systems (including failures resulting from the Design Basis Earthquake) will not cause fuel to be uncovered. These systems need not otherwise meet Category I seismic requirements.	Item 6.	No drains have been provided for the spent fuel storage pool. Since the pump suction connections extend no more than 2 ft below normal water level, there is also no possibility of inadvertently draining pool water below that level. To ensure adequate cooling of the stored fuel assemblies, pool water return lines extend above the top of the fuel assemblies. To ensure against inadvertently draining of the pool by a siphon effect, each return line has a check valve to prevent reverse flow.

Table 9.5-2

Design Conformance with Safety Guide 13

Regulatory Position		Kewaunee Design Feature	
Section 7.	Reliable and frequently tested monitoring equipment should be provided that will alarm both locally and in a continuously manned location if the water level in the fuel storage pool falls below a predetermined level or if high local radiation levels are experienced. The high radiation level instrumentation should also actuate the filtration system.	Item 7.	Level measuring instrumentation and radiation monitoring equipment are provided which alarm both locally and in the Control Room. As noted above, the air filtration system will be in operation whenever spent fuel is handled.
Section 8.	A seismic Category I makeup system should be provided to add coolant to the pool. Appropriate redundancy or a backup system for filling the pool from a reliable source such as a lake, river, or on-site seismic Category I water storage facility should be provided. If a backup system is used, it need not be a permanently installed system. The capacity of the makeup systems should be such that water can be supplied at rate determined by consideration of the leakage rate that would be expected as the result of damage to the fuel storage pool from the dropping of loads, from earthquakes, or from missiles originating in high winds.	Item 8.	A fuel pool-service water intertie line, which will allow rapid refilling of the pool, is provided. The service water system is a Class I system and is contained in a Class I structure. The intertie line and discharge lines to the pool are protected from potential missiles or tornado forces by the concrete structures in which they are housed or embedded. The water delivery is therefore designed and capable of delivering water to the pools during or immediately following a tornado.

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Figure 9.5-1 Fuel Handling System

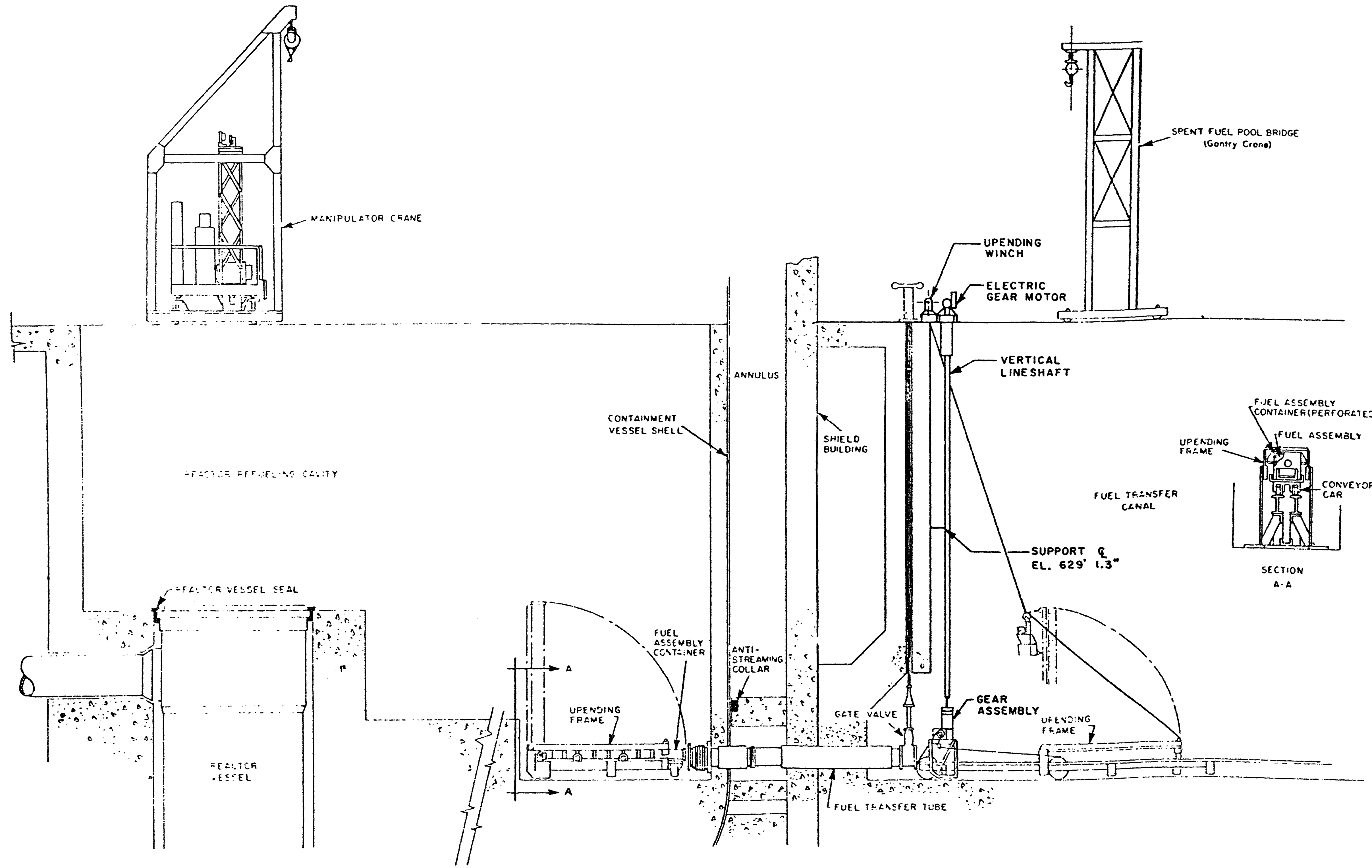
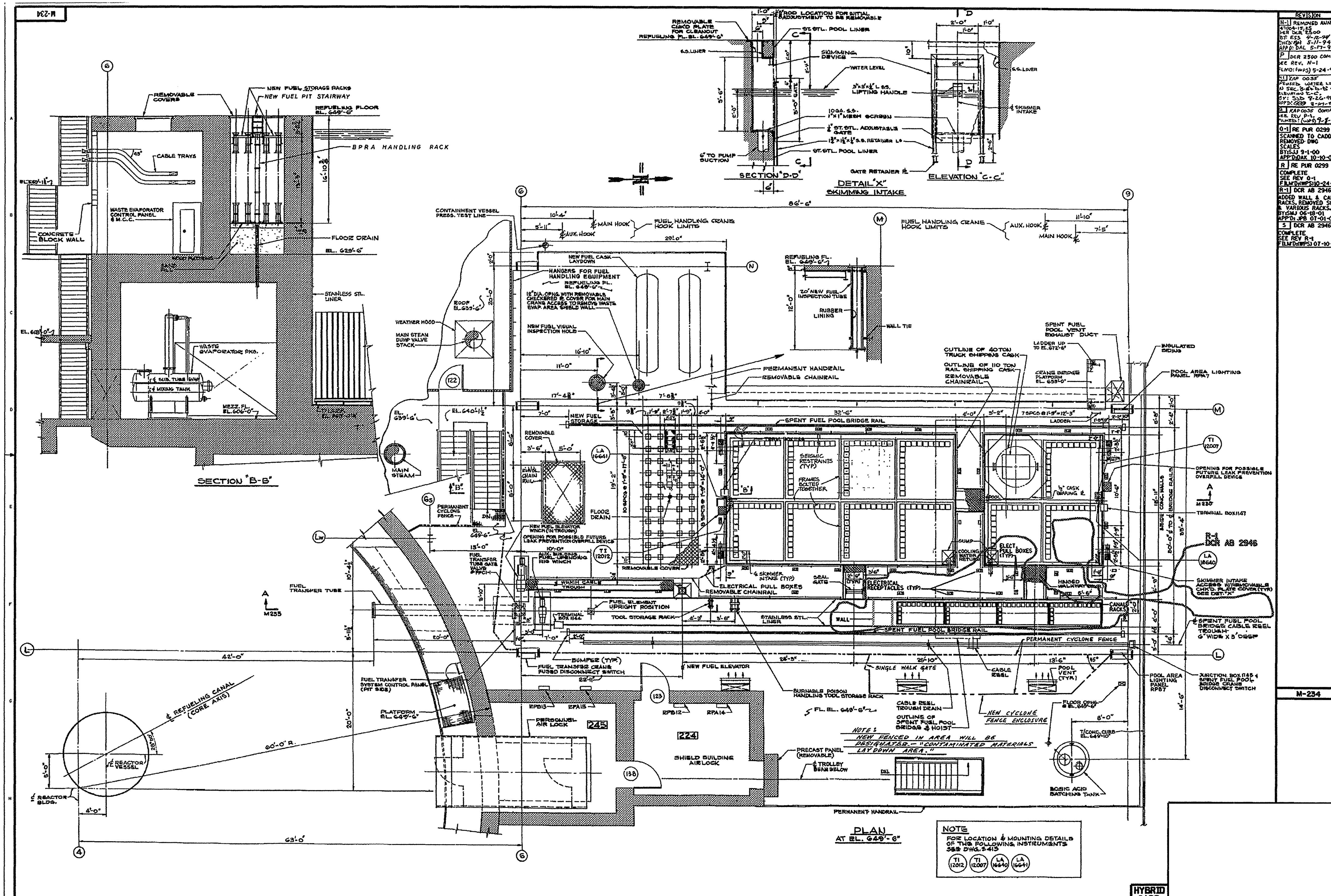


Figure 9.5-2 Spent Fuel Pool and New Fuel Storage Plan



9.6 FACILITY SERVICES

9.6.1 Fire Protection System

The KPS Fire Plan defines and guides the implementation of KPS fire protection practices in order to ensure adequate preventive, corrective and mitigative measures. The plan was prepared in response to and in accordance with the guidance provided in Generic Letter 86-10. Refer to the KPS Fire Plan for licensing and design requirements.

9.6.2 Service Water System

9.6.2.1 Design Basis

The Service Water System has been designed to provide redundant cooling water supplies to: the Containment Fan Coil Units, diesel generators, air compressors, component cooling heat exchangers, safety injection pump stuffing boxes and/or coolers, and control room air conditioners.

This system also supplies water for the deluge systems for ventilation system charcoal filters as described in Section 9.6.5. The design includes provisions for isolation of nonessential components following an accident. Lake Michigan is the source of service water.

Some non-Class I and non-redundant coolers and equipment are supplied from the service water headers. These include the circulating water pump seals and bearing coolers, traveling screen wash nozzles, service water pump strainer backwash, fire protection jockey pump, station and instrument air compressors and aftercoolers, spent fuel pool heat exchanger, spent fuel pool emergency makeup, and numerous hose connections located inside containment and in other Class I areas.

Also, train specific service water isolation to the non-safety features equipment in the Turbine Building will occur upon a SI sequence concurrent with a low service water header pressure. This ensures that safety-related equipment will receive sufficient flow under all conditions.

Additionally, there are 2 non-safeguard cooling coils inside containment that use service water to cool exhaust from the CRDM shroud during normal plant operations. A SIS isolates these cooling coils from the containment fan coil flow path.

The Service Water System is capable of sufficient heat removal based on highest expected temperatures of cooling water, maximum loading and leakage allowances. The system is monitored and operated from the Control Room. Isolation valves are incorporated in all service water lines penetrating the containment.

Electrical power requirements for the Service Water System are supplied by any of the power sources described in Section 8.2.3.

9.6.2.2 System Design and Operation

The Service Water System flow diagram is shown on Figure 9.6-1, Figure 9.6-2 and Figure 9.6-3. The Service Water System supplies cooling water to equipment in the steam plant, to the Containment fan coil units and to the reactor auxiliary systems.

Service water is normally withdrawn from the Circulating Water System which takes water from the lake through a deep-water, multiple inlet, submerged conduit. The Circulating Water System is described in Section 10.2. The water passes through the submerged intake to a forebay in the screenhouse structure; through four traveling water screens and to a screen well, from which service water pumps take suction. The screen structure design includes provisions for de-watering to permit inspection and maintenance of associated equipment.

Design precludes the interruption of water flow to more than two of the four service water pumps while one-half of the screen structure is de-watered.

Provisions for intaking service water are included in the sizing of the circulating water intake, which is the normal source of service water. In addition, two alternate sources of service water are provided. One alternate source consists of two auxiliary intakes on the circulating water intake pipe downstream of the main intakes. Each of these intakes is capable of providing the required amount of service water. The design of the auxiliary intakes are such that they will not be damaged by frazil ice. An alternate source of service water is provided by an interconnecting pipe between the circulating water discharge structure and the screenhouse forebay. This interconnecting provides a redundant source of service water in the extremely unlikely event the main intake line becomes blocked. The redundant path is always available since the valve CW-500 in the interconnecting line (Figure 10.2-7) is locked open.

Supply of service water for essential services is redundant and can be maintained in case of failure of one of the supply headers.

Four electric motor-driven vertical, wetpit, single-stage, double suction, water lubricated, 1800 rpm, centrifugal service water pumps are provided. During normal or accident operation, each service water pump has a capacity of 6400 gpm. Two of the service water pumps serving one header receive power from one of two 4160-V safety-feature buses and two from the other safety feature bus.

Under the conditions of a concurrent LOCA and loss of off-site power, either pair of pumps are capable of supplying the required cooling capacity to the containment fan-coil units, diesel generators and other components. The two service water supply headers run through the Screenhouse Tunnel to the Class I portion of the Turbine Building. Isolation valves are provided in the supply lines to the non-safety features equipment located in the Turbine Building. Isolation of the turbine building may occur remotely from the Control Room or automatically upon a SI sequence concurrent with a low service water header pressure. This automatic isolation actuates on a train basis to maintain cooling as long as possible to the non-safety features equipment. The return paths are separated to the fullest extent possible to minimize the possibility of flooding problems should a return line fail. Service water returns to the Circulating Water System discharge.

The Reactor Containment Vessel fan coils are supplied in pairs from the Service Water System. Each coil inlet line is provided with a manual shutoff valve and drain valve. Similarly, each discharge line from the coils is provided with a manual shutoff valve and drain valve. This arrangement allows each coil to be isolated individually for leak testing or to be drained and maintained open to the atmosphere during the integrated leak test. The fan coil units are made up of carbon steel headers with copper tubing and fins.

Each train has a flow control orifice in the 10 inch service water discharge headers. The orifices were sized to provide the required flow and tested in the SI line-up. The orifices are stainless steel and mounted between flanges so that if system flow characteristics change, they can be easily re-sized.

Each fan coil discharge line is monitored for radioactivity following a LOCA by routing a small bypass flow through a common radiation monitor. Upon indication of radioactivity in the effluent, the cooler discharge line would be monitored individually to locate the defective fan coil by isolating all but one of the units on the line and observing the radiation monitor. However, since the fan coils and service water lines are completely closed inside containment, no leakage is expected from these units. The Service Water System pressure at the inlet to the containment is maintained above the peak containment pressure calculated for a LOCA.

Automatically operated valves are provided to isolate sections of the Service Water System should malfunctions occur following the Design Basis Accident. The operator thus has the capability of keeping at least one-half of the system operational from the control room. In the event of a loss of normal auxiliary power, each of the two diesel generators is sized to supply two service water pumps in addition to the other vital engineered safety feature loads.

The Service Water System is in operation at all times during plant operation and shutdown and, therefore, is in a high state of readiness for any abnormal or emergency plant condition.

The service water lines to the Auxiliary Building primarily supply cooling water to the two component cooling heat exchangers. These heat exchangers are utilized to remove core residual heat through the RHR System. The RHR System is employed during normal shutdown operations. It also is placed in service following a LOCA for cooling of the recirculated flow from the reactor containment sump.

The capacity of two of the four service water pumps is sufficient to supply the cooling water requirement for safely controlling a LOCA.

The diesel generators employ jacket cooling and shell-and-tube heat exchangers. Adequate heat absorption capacity is provided to operate the diesel generators until the Service Water System starts.

The Service Water System is capable of supplying water to the suction of the AFW pumps. Should the supply of water from the condensate storage tank fail for any reason, the auxiliary feedwater pumps will trip on low suction or discharge pressure. Water from the service water header can then be supplied by the opening of normally closed motor operated valves from the Control Room. The AFW pumps can then be restarted.

Isolation valves are provided to isolate nonessential services in the event of a system malfunction. The operator may isolate the entire Turbine Building service water supply through remote actuation of these valves.

Flow metering devices are installed at various locations to provide means of determining equipment performance. Flow indication is provided outside containment for service water to each fan coil unit. A leak in any one of the headers or its associated fan coil unit will be detected by increasing containment sump level. Independent and redundant containment sump level instruments are qualified for all postulated plant conditions. During normal operation, Containment Sump A level indication, alarms and pump run times are used to calculate the leak rate. For large leaks, the source is identified by isolating the outlet of each fan coil unit and comparing the flow rates through the headers. For small leaks, the service water headers and fan coil units are accessible during operation for inspection.

For post accident conditions, analysis has shown the maximum containment wide range level is reached when Safety Injection is switched from the RWST to containment sump recirculation. Subsequent cooling of Containment contents results in a very small decrease in the liquid volume. The decrease in liquid volume due to heat removal more than compensates for the volume added by condensed steam. By trending Containment sump level versus time, leaks are detected. For large leaks, comparison of the flow through the headers, with the outlet valves isolated will identify the leak source. For small leaks, isolating the supply to each header and observing the effect on containment level will identify the leak. Manual or remotely operated valves are available to isolate the leak while maintaining sufficient equipment for containment heat removal.

9.6.2.3 Malfunction Analysis

The Service Water System is designed to prevent a component failure from curtailing normal station operation. As shown on Figure 9.6-1, Figure 9.6-2, Figure 9.6-3, and Figure 9.6-4, the service water header can be isolated to provide two independent systems with no cross-connections. Upon manual or automatic initiation of SI, both service water header isolation valves are closed separating the service water system into two independent headers. In addition, the four containment fan coil service water return motor valves are opened and the two containment fan coil service water flow control valves are closed to provide service water flow through all containment fan coil units. The turbine building service water header valves are arranged such that they fail as-is on loss of instrument air and only one valve may be open at any time. Sufficient compressed air is stored within a Class I accumulator for each Turbine Building header valve to permit a valve closure following a loss of instrument air.

In the event of a major malfunction in either half of the system, it is possible to isolate the portion of the system affected and maintain service to all essential services of the plant. In addition to the header isolation valves, each component also has individual isolation valves to permit removing any piece of equipment from the system. The pumping capacity is sufficient during the incident to supply all safety features equipment. Additionally, supply to the non-safety features equipment in the Turbine Building will be maintained except under the concurrent actuation of the SI sequence and a low service water header pressure on the supplying header.

In the extremely unlikely event that the main intake line becomes blocked, the circulating water pumps could lower the screenhouse forebay water level to below the service water pump inlet housing. Protection against this occurrence is afforded by tripping the circulating water pumps upon a redundant coincident forebay extreme low water level signal.

9.6.3 Auxiliary Building Ventilation Systems

9.6.3.1 Design Basis

The Auxiliary Building Ventilation System is designed to provide maximum safety and convenience for operating personnel, with equipment arranged so that potentially contaminated areas are separated from clean areas. Ventilating air is recirculated in clean areas only.

The Auxiliary Building Ventilation System was designed to maintain a minimum general area temperature of 60°F in the building with an outside temperature of -20°F, and a maximum air temperature not to exceed 10°F above the outside air temperature.

To insure that the building is under a slight negative pressure, total air exhaust quantity always exceeds supply air quantity (by typically a minimum of 10 percent during normal operations).

Exhaust fans are capable of discharging the required amount of air through the final filters regardless of whether the filters are clean or dirty. The maximum pressure drop across the filter bank is 3-inch water gauge when dirty.

The ventilation equipment complies with accepted industry standards for power plant equipment and with all applicable state and local codes and regulations. Redundant equipment is provided for those systems where public health and safety may be endangered or where safety features equipment operation may be impaired, in case of malfunctions. Particulate filters used are of high-efficiency type capable of removing at least 99.97 percent of 0.3-micron diameter smoke particles. These filters meet all requirements of AEC Health and Safety Bulletin 212, dated 25 of June 1965.

The ventilating equipment is accessible for periodic testing and inspection during normal operation.

9.6.3.2 System Descriptions

The Auxiliary Building has separate normal ventilation systems to serve the auxiliary equipment areas, the Spent Fuel Pool area, the non-radioactive area, and the Control Room area, as shown in Figure 9.6-5. The path of ventilating air is from clean or low activity area toward areas of progressively higher activity.

Ventilation air is drawn from outside, through two makeup air units. The system is balanced to maintain the Auxiliary Building at a pressure slightly negative with respect to atmospheric and adjacent Turbine Building pressures. Exhaust air is discharged through HEPA filters to remove all particulates.

Special provisions are included to exhaust air through activated charcoal beds and high-efficiency filters from areas subject to possible radioactive contamination, through the use of the Auxiliary Building Special Ventilation System. This system is actuated by either a Safety Injection Signal, or a high radiation signal from the Auxiliary Building Vent monitor. This system is described in Section 9.6.5.

The Spent Fuel Pool is ventilated by an air sweep system, which maintains a continuous air sweep across the pool when the system is operating. In normal operation, exhaust air from the system passes through HEPA filters before being discharged to atmosphere through the monitored Auxiliary Building Vent. Charcoal filters are provided, which are bypassed during normal operation. Administrative procedures assure that the bypass is closed during fuel handling operations. Also, the monitor in the Auxiliary Building vent will close bypass dampers if they are open, in event of high radiation.

The Turbine Building is provided with a recirculated air ventilation system, with provisions for makeup as required from a fresh-air louver. Exhaust air is discharged directly to atmosphere through exhaust fans mounted on the roof and on the east wall of the mezzanine floor. 100 percent makeup air is supplied.

The Administration Building is equipped with a multizone-type air conditioning system, arranged to receive outside air for makeup, to recirculate conditioned air, and to exhaust air to the Turbine Building and/or directly to atmosphere.

9.6.4 Control Room Air Conditioning System

9.6.4.1 Design Basis

The Control Room Air Conditioning System is designed to provide a reliable means of cooling and filtering air supplied to the Control and Relay Room under both normal and post-accident conditions.

The Control Room Air Conditioning System is normally in operation, providing cooled and filtered air to the Control Room and Relay Room. There is normally a 20 percent fresh air makeup to the Control Room from the Auxiliary Building air conditioning unit air intake. The air passes through roughing filters, cooling coils, and fans in one of the two 100 percent air conditioning units, and is then distributed to the Control and Relay Room. Heating coils, supplied from the Auxiliary Building hot water converter, provide for comfort heating. A humidifier is located in the common supply air duct. Service water can be aligned directly to the cooling coils in the air handler in the event that both chilled water units are not available.

Following a LOCA, the Control Room air conditioning will begin a 100 percent recirculation mode of operation. During this period, approximately 20 percent of the recirculation flow will pass through one of the two Control Room post-accident recirculation filters and fans. Each filter assembly consists of a pre-filter, HEPA filter, charcoal filter and deluge protection for the charcoal filters. The filters will assure that any minute amounts of in-leakage either prior to or following system actuation will be removed, and allow the operators to continuously occupy the Control Room.

The Control Room Air Conditioning System has a designed capacity of 600,000 Btu/hr to remove 537,400 Btu/hr from the areas served and to maintain the Control Room temperature at 75°F ± 10°F dry bulb under summertime design conditions (95ºF ambient temperature), and 70°F ± 10°F dry bulb under wintertime design conditions. Relative humidity is maintained at about 30-40 percent for personnel comfort. Alternate cooling mode (service water aligned directly to the air-handling unit cooling coils) will maintain the Control Room/relay room environment for equipment operation. The maximum temperature when in alternate cooling mode is 100ºF with 80ºF service water and 95ºF ambient air temperature. An evaluation was completed to assure necessary equipment would remain available at the higher room temperature. The alternate cooling mode is the safety-related means of cooling.

Since the Control Room Air Conditioning System is normally operating at all times; the air conditioning unit availability can be readily verified by transferring from one redundant unit to the other under normal conditions. The availability of service water alternate cooling can be checked without the inadvertent introduction of service water to the chilled water loop by cycling valves in a test mode. Periodically, 100 percent recirculation mode of operation can be demonstrated without affecting normal operation. Charcoal and HEPA filters meet the same design standards as those given for the Shield Building Ventilation System, as discussed in Section 5.5, and the Auxiliary Building Special Ventilation System as described in Section 9.6.5.

9.6.4.2 System Description

The Control Room Air Conditioning System provides cooling and heating of recirculated and fresh air used to ventilate the Control Room for either personnel comfort or equipment cooling.

The Control Room Air Conditioning is shown on the flow diagram, Figure 9.6-6.

The system consists of two 100 percent capacity air conditioning units, which filter and cool air supplied to the Control and Relay Rooms. Under normal conditions, approximately 20 percent of this air is exhausted to the Turbine or Auxiliary Buildings, and the remainder is recirculated to the Control Room Air Conditioning unit where it is mixed with fresh air from the Auxiliary Building Air Conditioning Unit air intake. During wintertime conditions the fresh air is pre-heated by a hot water coil.

Each air conditioning unit is equipped with a cooling coil. The cooling coil is supplied with chilled water from a water chiller package consisting of compressors, evaporator, condensers, expansion tank, and chiller pump. The condenser is cooled by the Service Water System. The chiller pump supplies the chilled water to the air conditioning unit-cooling coil. A water chiller package is provided for each air conditioning unit. The redundant trains of associated components are supplied with cooling water and power from separate sources. Additionally, each air-handling unit cooling coil can be supplied with service water directly if both water chillers are unavailable. Direct service water cooling (referred to as alternate cooling) is manually initiated from the Control Room on Mechanical Vertical Panel "A." A reheat coil and humidifier are located in the common air duct. The reheat coil is supplied with hot water from a hot water converter in the building heating system.

The heating of the Control Room is considered non-critical since it is provided for comfort only. Normal electrical heat loads in the Control Room provide sufficient heat to maintain temperatures sufficiently high for satisfactory instrument performance.

For post-accident recirculation, two redundant fan and filter units are provided to filter the Control Room air. Each unit is rated at approximately 20 percent of the total airflow. These units operate in parallel, and can be used to filter fresh air, drawn from the outside, if the operator desires. The filter units consist of particulate, absolute and charcoal filters, and each filter assembly is designed for 2500-cfm airflow.

9.6.4.3 Actuation and System Operation

One Control Room Air Conditioning Unit is normally operating. If that unit or its associated chiller package should fail to operate, the second system is automatically placed into service by a trip signal from the operating unit. Each unit is also manually controlled from the Control Room for both normal and alternate modes.

The 100 percent recirculation mode of operation is initiated by either a Safety Injection Signal or high radiation as detected by the radiation monitor at the outlet of the air conditioning unit. At this time, all outside air is stopped by closure of the dampers in the fresh air supply, and both post-accident recirculating fans are started. A portion of the total flow through the air conditioning unit equipment is filtered through the PAC filter assemblies.

The operator can add fresh air to the Control Room under post-accident conditions by first verifying that conditions at the intake plenum will not cause contamination of the Control Room atmosphere, then manually opening the selected outside air damper and the damper at the suction of the post-accident recirculation fans. Excess air is exhausted to the Turbine Room through a backdraft damper. The operator can confirm the incoming fresh/recirculation air mix is not contaminating the Control Room atmosphere by observing a radiation monitor channel in the Control Room. Damper control switches are spring return to the normal position so that when the damper control switches are released the dampers automatically return to the recirculation configuration.

Although the possibility of a fire in the Control Room is extremely remote, provisions have been made to prevent recirculation of smoke-filled air to the Control Room. As required, thermally actuated fire dampers have been installed where the ventilation system passes through firewalls. A smoke detector mounted in the return air duct will automatically close a damper in the return air duct, thereby causing all air to be exhausted into the Turbine Building and making the Control Room a 100 percent makeup system. Thus, in the unlikely event of a Control Room fire, continued occupancy by the operators is possible. In addition, breathing masks and fresh-air supply units are available for operator use in case of such an emergency.

In event of a high energy steam or feedwater line rupture in the plant, dampers will automatically close off all the fresh air supply to the system and will close off the relief vent to the turbine room, as well as other ducts leading to the normal Auxiliary Building Ventilation system.

9.6.5 Auxiliary Building Special Ventilation System (Zone SV)

9.6.5.1 Design Basis

The Auxiliary Building Special Ventilation System is designed to reliably collect any potential Containment System leakage that might bypass the Shield Building annulus and to cause it to pass through charcoal filters before reaching the environment. To ensure confinement of such bypass leakage and its removal by the Auxiliary Building Ventilation System, the areas within the

Auxiliary Building where there is the potential for such leakage are designated as Zone SV. This special volume is about 0.9E+6 ft^3. Figure 1.2-2, 1.2-4, 1.2-6, 1.2-7 and 1.2-10 indicate the areas included in Zone SV. Some of the major systems within this zone are:

- Residual Heat Removal System
- Safety Injection System
- Containment Vessel Internal Spray System
- Reactor Coolant Letdown System and Volume Control Tank
- Shield Building Ventilation System
- Component Cooling System
- Chemical and Volume Control System
- Waste Gas Handling System
- Auxiliary Building Special Ventilation System

A detailed analysis of leakage paths is given in Appendix H.2. It demonstrates that the leakage through the Containment System does not impose a severe burden on the Auxiliary Building Special Ventilation System and that such leakage will be very small as compared to the potential leakage to the Shield Building annulus. Some of the data in Table 5.2-3 is regrouped and presented in Table H.2-1, H.2-2 and H.2-3.

9.6.5.2 Design Leak Rate

The leakage specification for the Zone SV stipulates there be no ex-filtration of air from the Zone SV boundary with only one of the redundant trains of the Auxiliary Building Special Ventilation System in operation. Initial acceptance tests demonstrated the integrity of the Zone SV Perimeter, as well as the capacity of the Ventilation System. Because of the steep head-versus-capacity curve for the system fans, the flow is relatively unaffected by the vacuum attained, therefore, vacuum has little significance in demonstrating the functional performance of Zone SV and the associated Auxiliary Building Special Ventilation System.

The significant parameter in demonstrating the performance of the Zone SV is the face velocity across openings in the perimeter. The industrial ventilation practice as given by the American Conference of Government Industrial Hygienists recommends a minimum face velocity of 50 ft/min for exhaust hoods. Margin has been provided for the Kewaunee design by sizing the Auxiliary Building Special Ventilation System to produce an adequate negative pressure, so that with a door open, airflow through the opening will have a minimum face velocity of 80 ft/min.

An estimate can be made as to the allowable area since the leakage velocity is inversely proportional to the total area of the perimeter openings. Based on a requirement of 50 ft/min and the fan capability of 9000 cfm, (fan capacity with maximum allowable pressure drop across the filters), a nominal opening of approximately 200 square feet would appear acceptable.

9.6.5.3 System Description

The entire perimeter of the Auxiliary Building Special Ventilation Zone is defined as a medium-leakage type barrier. The walls, ceilings, and floors are constructed of poured concrete or masonry with sealed joints where required. Doors penetrating the perimeter of the Auxiliary Building Special Ventilation Zone will be fitted and weather-stripped as required to minimize leakage.

The Auxiliary Building Special Ventilation System is provided with redundant exhaust intakes, connecting redundant filter assemblies to the exhaust ductwork through electric heaters, roughing, absolute, charcoal and absolute filters, and then to redundant exhaust fans, as shown in Figure 9.6-7.

Components in the Special Ventilation System are of QA Type I construction. Electrical components are connected to the emergency power system provided by the diesel generators. Components are designed to withstand a total gamma radiation of 5×10^7 RADS and an operating temperature of 200°F.

The initiating signal for the Auxiliary Building Special Ventilation System is a Safety Injection Signal, described in Chapter 7, or a signal from the detection of high radiation in the Auxiliary Building Vent. When the Auxiliary Building Special Ventilation System is actuated, the normal supply and exhaust ducts from Zone SV are closed automatically, and the normal supply and exhaust fans for the Auxiliary Building are tripped.

To provide continued cooling for safety features components located within Zone SV under these conditions, small fan-coil units have been located throughout the Auxiliary Building to recirculate and cool the air in Zone SV. These units receive cooling water from the Class I Service Water System.

9.6.5.4 Doors

Zone SV is bounded by a system of single doors for low traffic usage and interlocked doors for high traffic usage. Single doors are administratively controlled to ensure that the doors are not inadvertently left open. The airlock passage doors are interlocked so that only one door may be open at a time. When the first door is opened, the second airlock door is automatically locked and cannot be opened until the open door is closed. These interlocks fail in the locked position when de-energized by a power failure. Provision is made for overriding the interlock in an emergency through use of a key, which is located between each set of zone SV doors to prevent trapping of personnel. All Zone SV doors are provided with heavy-duty closers without “hold open” devices.

The interlocks, administrative controls, heavy-duty closers and airflow into the Zone SV due to pressure differential, provide assurance that these doors will be closed for proper functioning of the Auxiliary Building Special Ventilation System.

9.6.5.5 Fans

The Zone SV exhaust fans are vane-axial, direct-connected fans, each capable of maintaining 100 percent design conditions, are of standard construction. Accessories include inlet bell, horizontal mounting legs and vibration isolation base. They will maintain at least the minimum air flow requirements when filter resistance is at the maximum.

9.6.5.6 Filter Assemblies

The filter assemblies are composite units consisting of roughing filters, electric heating elements, HEPA filter section, impregnated charcoal bed filter section, and a HEPA filter after section. Each section is designed as follows:

1. The roughing filter is designed as a standard particulate-removal air filter.
2. The heating coil is capable of sufficiently increasing the incoming air temperature to ensure a 70 percent relative humidity entering the charcoal bed assuming an initial relative humidity of 80 percent entering the heaters (see KPS Environmental Qualification Plan). The heaters are verified by testing to be capable of lowering the humidity from 80 to 70 percent. An interlock with the Zone SV exhaust fans assures that the electric heater cannot operate unless the fan starts.
3. The high-efficiency particulate filters are designed to have 99.97 percent DOP removal efficiency on a 0.3-micron smoke particle through complete filter (medium, frame, and gasket) when operated at rated capacity and at 20 percent of rated capacity. Filter design is water and fire resistant and meets all requirements of AEC Health and Safety Bulletin 212-1965.
4. The iodine filter is an impregnated activated charcoal bed, designed to remove ≥ 90 percent radioactive methyl iodide when exposed to an atmosphere at 66°C, 95 percent relative humidity, based on a filter depth of 2 inches and a residence time of 0.25 seconds. The ignition temperature for the charcoal used is greater than 640°F.
5. Charcoal is contained in Type 304 stainless steel trays, with three trays per module. Each module is approximately the same face size as the HEPA filters. Trays are removable and arranged for a leak-tight fit to the support frame.
6. Each charcoal filter bank is equipped with a Fire Protection System, consisting of sprinkler nozzles, distribution piping to the exterior of the housing and an electrical solenoid deluge valve. The deluge valve is actuated by a heat detector near the charcoal. An air test connection is provided to permit both the spray nozzles and solenoid valves to be tested.

Filter design, evaluation, reliability, and testing is the same as the Shield Building Ventilation System described in Section 5.5.4, 5.5.5, and 5.5.6.

The charcoal filters were sized based on ventilation flow requirements. This gave filters that are similar (including fission product retention capability) to those being used in the Shield Building Ventilation System described in Section 5.5.4. Since the iodine handling capability of these charcoal filters is so much greater than any possible iodine loading in this system, no further calculations are required.

The exhaust air is discharged through the Auxiliary Building Vent, which extends through the roof of the Auxiliary Building. The Auxiliary Building Special Ventilation System flow capacity is sufficient to provide a measurable total negative pressure in Zone SV under the credible environmental and operating conditions.

9.6.5.7 Reliability and Testing

The following inspections and tests were performed to provide assurance that the functional intent of the system was achieved during the manufacture of the components and the construction of the system:

1. All ducting and filter assemblies were given an air pressure and leak rate test.
2. Each filter assembly received a filter performance test. Each HEPA and charcoal filter bank was tested in-place to verify performance.
3. Dimensional tolerances on filter assemblies and frame assemblies were checked to assure that suitable gasket compression was uniformly achieved on the filter sealing faces.
4. Charcoal filters initially used Barnebey Cheney Type 727 Activated charcoal. This material had been qualified by previous AEC testing. Each batch was tested by the manufacturer to assure that it had a removal capability equivalent to that of the activated charcoal used in the qualification testing.

   NOTE**:** *The charcoal currently in use conforms to Regulatory Guide 1.52, Revision 1, except as modified by ASTM D3803-89.*
5. Each charcoal bed filter was assembled at the manufacturer's shop and given a Flow Resistance Test and a Leak Test.
6. High-efficiency particulate absolute filters were randomly tested to demonstrate the filter's ability to withstand a pressure differential of 10 inches of water without loss of filtering efficiency.
7. HEPA filters of identical design to those in the filter assemblies were subjected to a rough handling test (¾ inch amplitude at 200 cycles/min.) following which the filter demonstrated no loss of filtering efficiency.

9.6.5.8 Initial Testing

A pre-operational acceptance test was performed to demonstrate the capability of the Auxiliary Building Special Ventilation System to accomplish the following functions:

1. To start both trains of redundant equipment upon initiation by a simulated normal actuation signal with coincident isolation of the normal ventilation supply and exhaust ducts for the Zone SV.
2. With each train operated alone, produce ¼ inch water vacuum in the Special Ventilation Zone with measurable vacuum at all points within the perimeter of the Zone.
3. To demonstrate by means of smoke tests for each redundant train, that there is no ex-filtration of air from various openings in the Zone SV boundary.

9.6.5.9 Incident Control

The Auxiliary Building Special Ventilation System, shown schematically on Figure 9.6-6, provides a sub-atmospheric pressure in, ventilation for, and fission-product removal from the Auxiliary Building Zone SV. The single system of trunk and branch ducting has 100 percent redundant roughing-particulate-charcoal filters, exhaust fans, and discharge ducting to the Shield Building vent stack. The system is automatically activated by either a Safety Injection Signal or by a high-radiation signal from the Auxiliary Building Normal Ventilation System exhaust monitors.

9.6 References

1. NRC SER, J. G. Giitter (NRC) to D. C. Hintz (WPS) Letter No. K-88-112, May 12, 1988
2. NRC Safety Evaluation Report, S. A. Varga (NRC) to E. R. Mathews (WPS) Letter No. K-81-210, December 22, 1981
3. NRC Safety Evaluation Report, S. A. Varga (NRC) to E. R. Mathews (WPS) Letter No. K-81-19, February 13, 1981
4. NRC Safety Evaluation Report Supplement, Memorandum for A. Schwencer (NRC) from G. Lainas (NRC) Letter, November 20, 1979

Figure 9.6-1 Service Water System-Flow Diagram Sheet 1

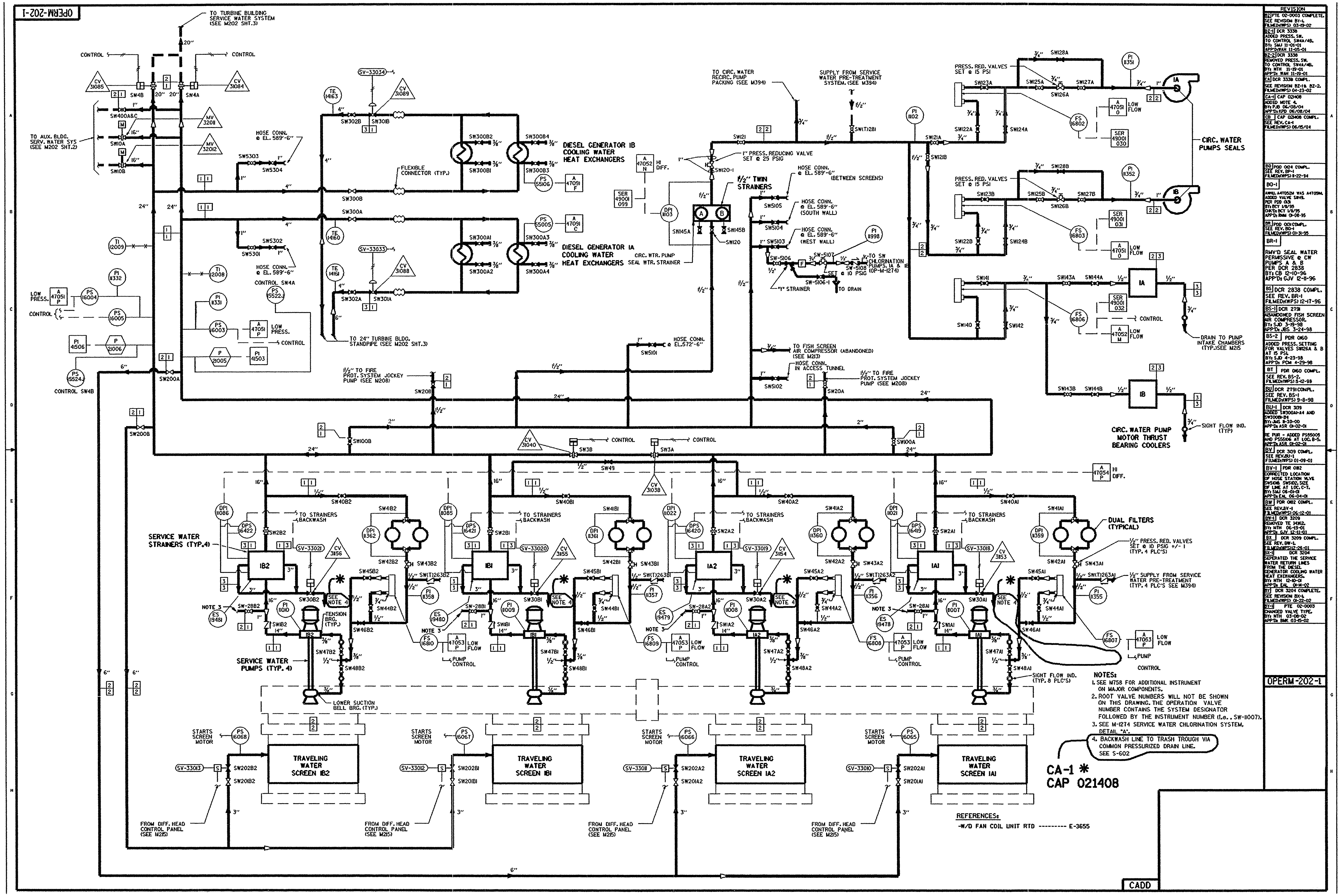


Figure 9.6-2 Service Water System-Flow Diagram Sheet 2

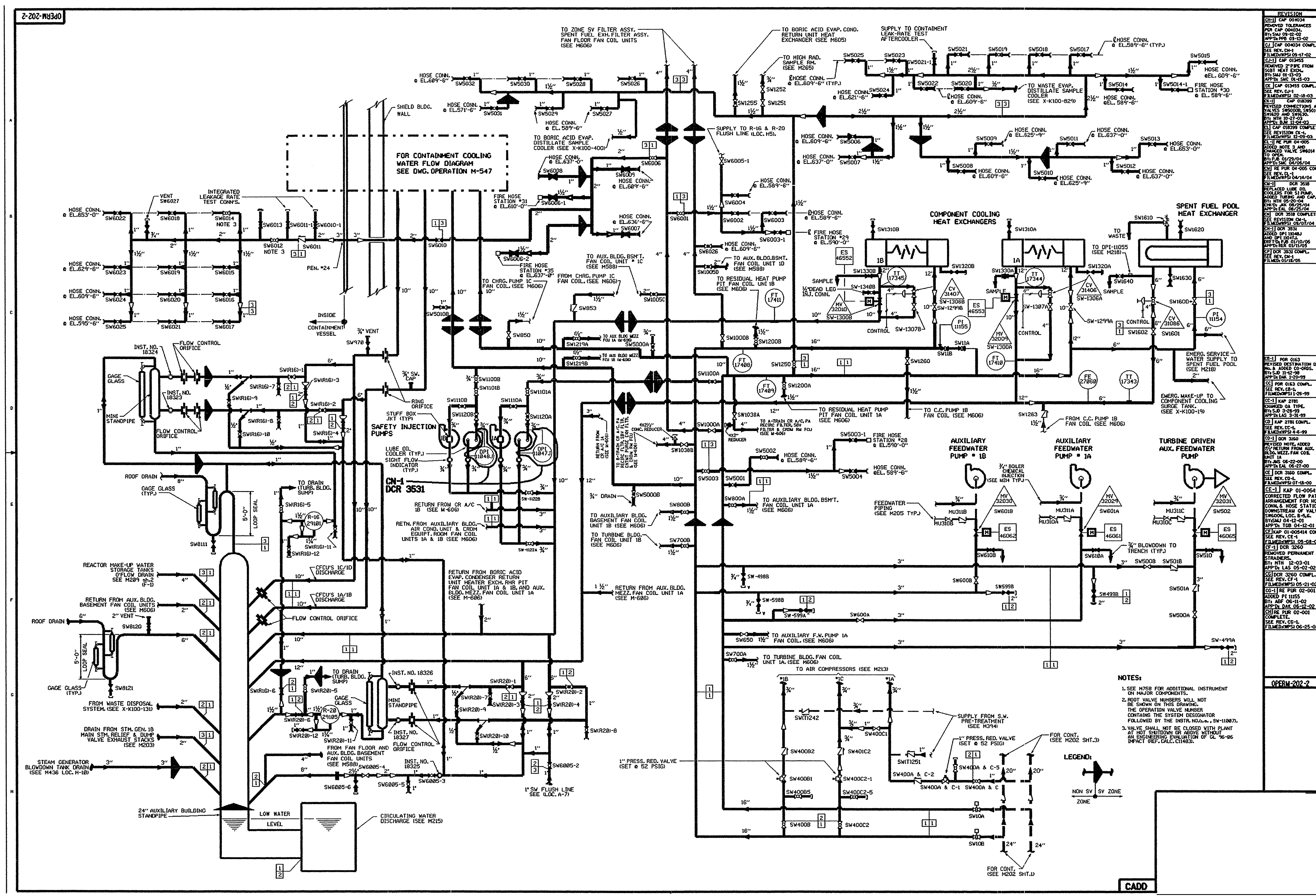


Figure 9.6-3 Service Water System-Flow Diagram Sheet 3

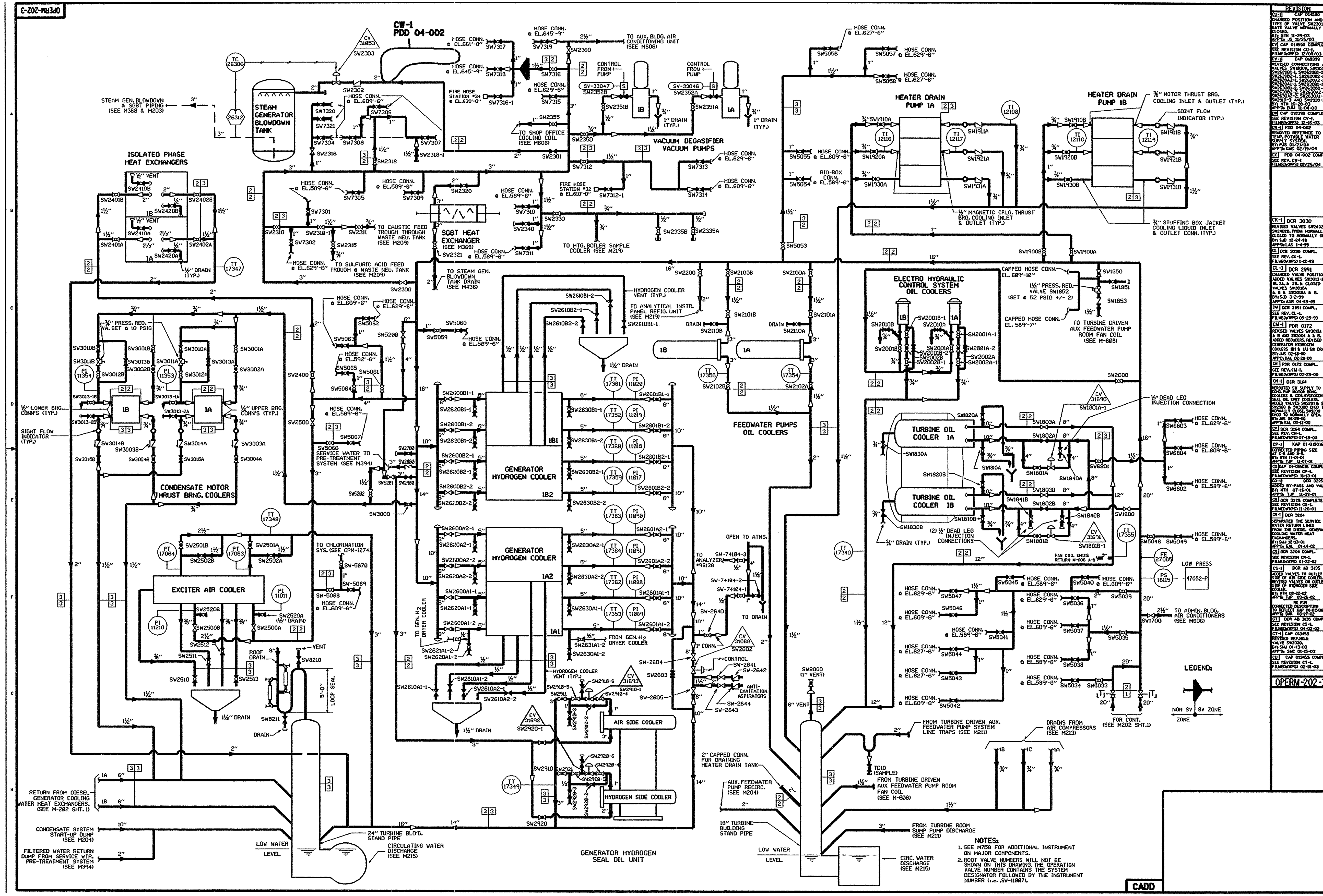


Figure 9.6-4 Service Water System-Flow Diagram Sheet 4

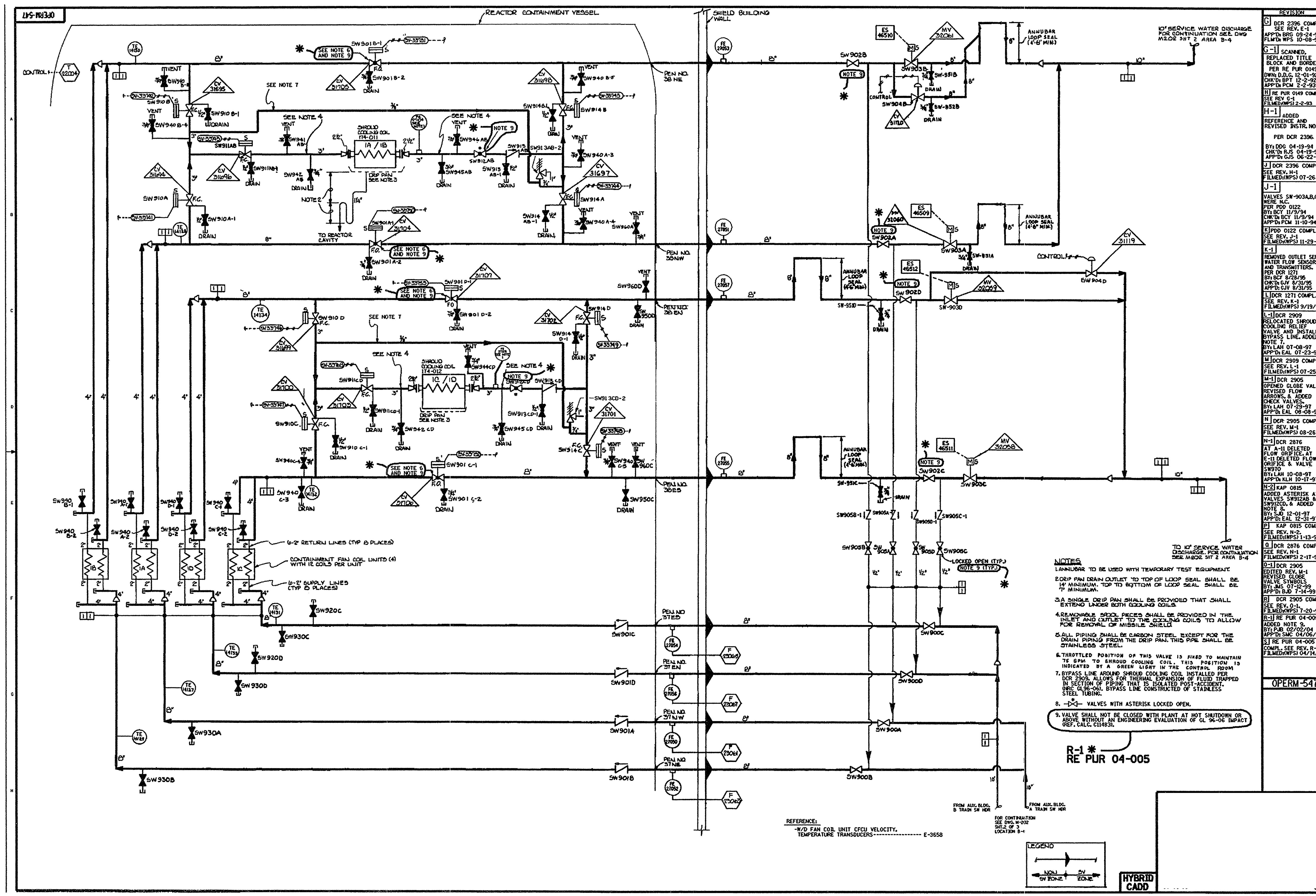


Figure 9.6-5 Auxiliary Building Ventilation System-Flow Diagram

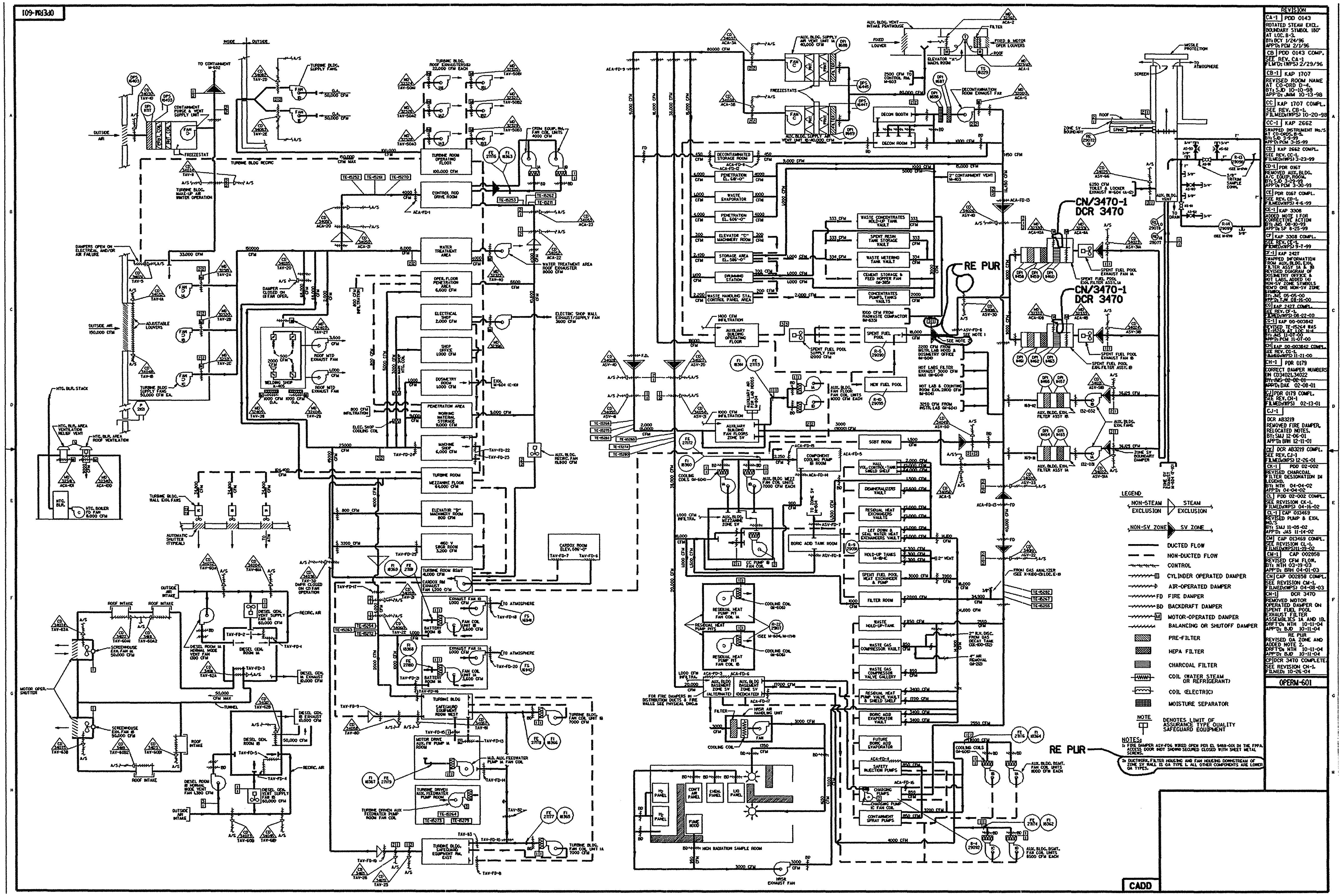


Figure 9.6-6 Control Room Air Conditioning System-Flow Diagram

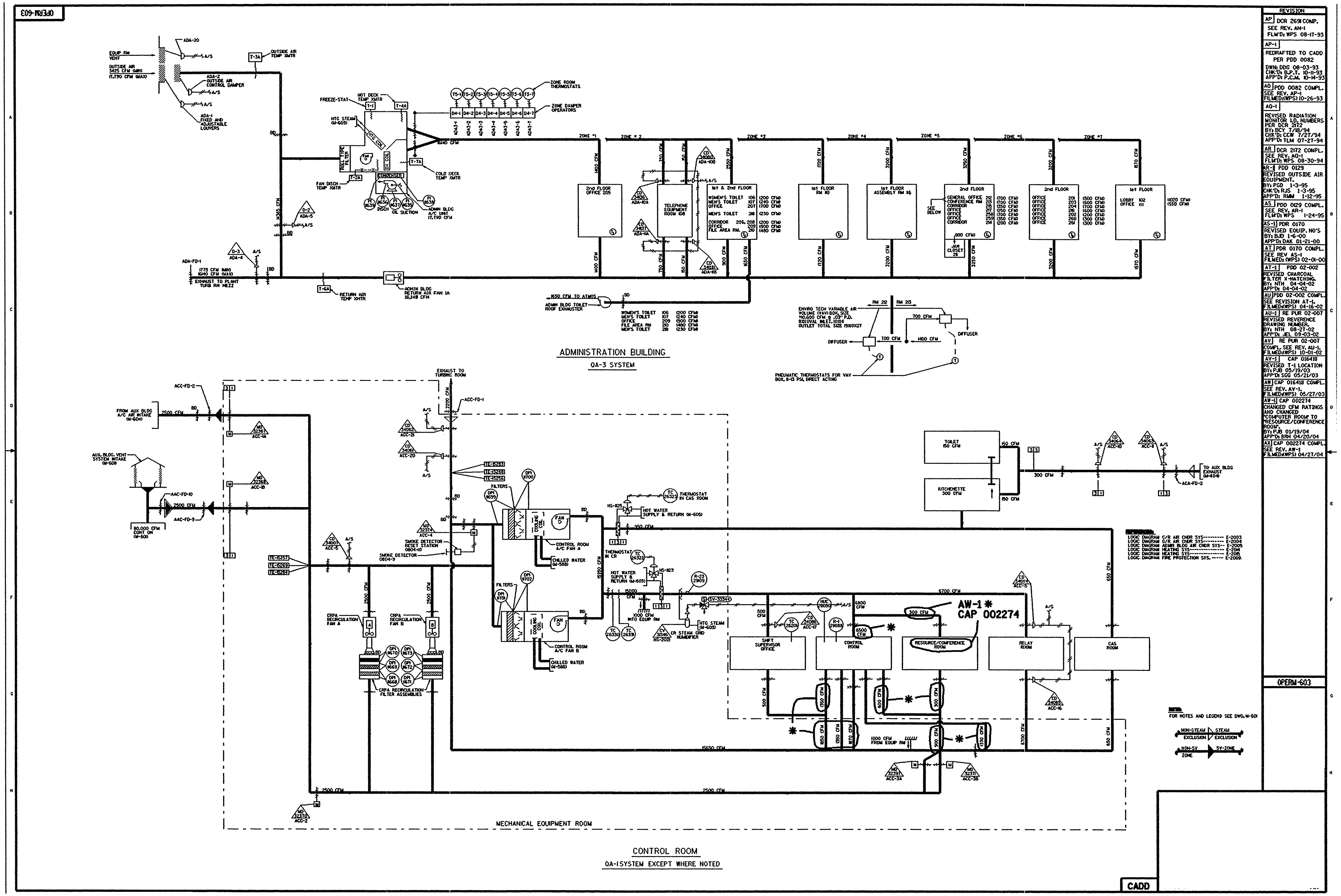
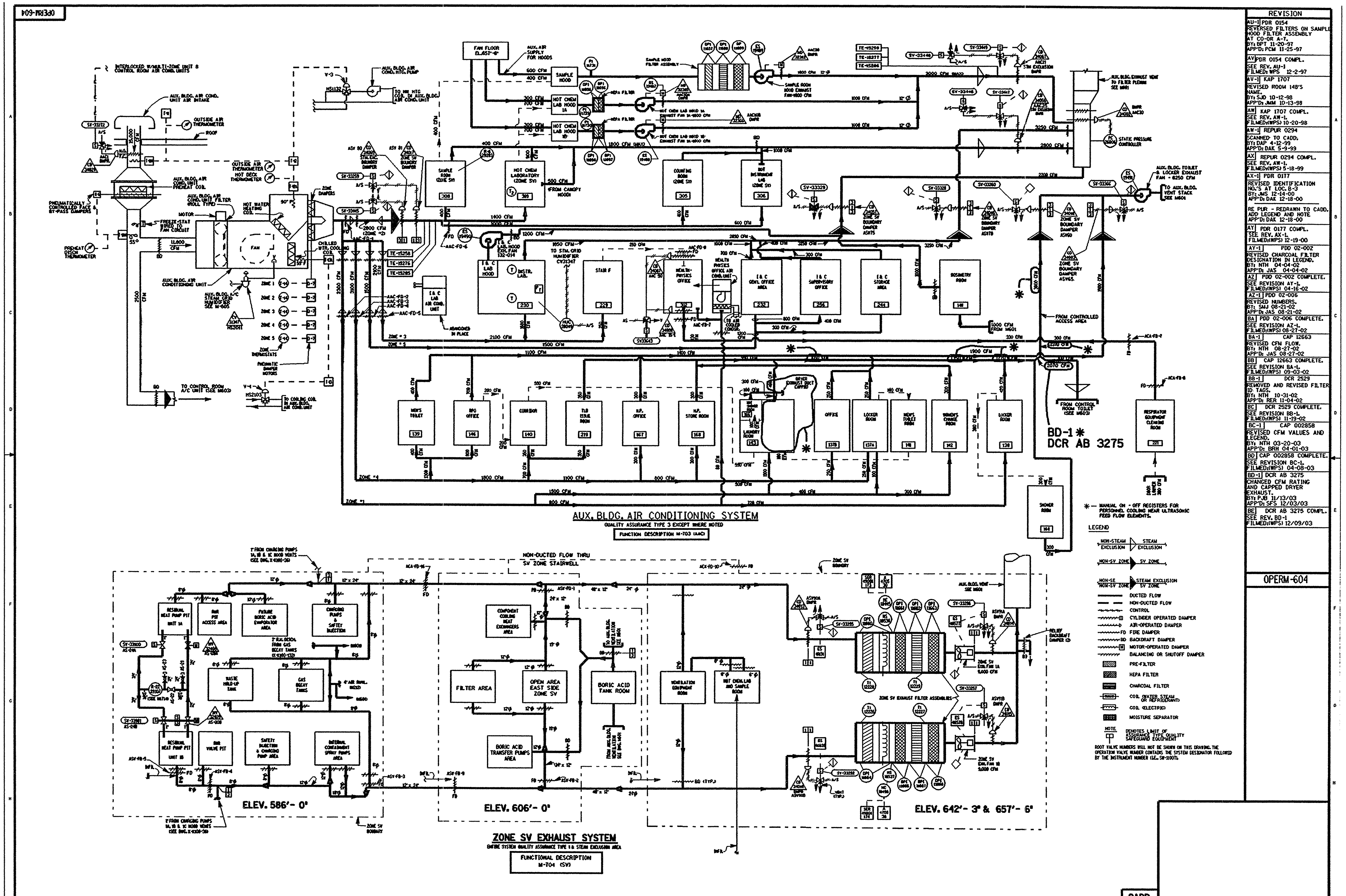


Figure 9.6-7 Auxiliary Building Special Ventilation System-Flow Diagram



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9.7 EQUIPMENT AND SYSTEM DECONTAMINATION

9.7.1 Design Basis

Activity outside the core could result from fission products defective fuel elements, fission products from tramp uranium left on the cladding in small quantities during fabrication, products of n - υ or n - ρ reactions on the water or impurities in the water, and activated corrosion products. Fission products in the reactor coolant associated with normal plant operation and tramp uranium are generally removed with the coolant or in subsequent flushing of the system to be decontaminated. The products of water activation are not long lived and may be removed by natural decay during reactor cooldown and subsequent flushing procedures. Activated corrosion products are the primary source of the remaining activity.

The corrosion products contain radioisotopes from the reactor coolant, which have been absorbed on or have diffused into the oxide film. The oxide film, essentially magnetite (Fe_3O_4) with oxides of other metals including Cr and Ni can be removed by chemical means presently used in industry.

Water from the RCS and the spent fuel pool is the primary potential source of contamination outside of the corrosion film of the RCS. The contamination could be spread by various means when access is required. Contact while working on primary system components could result in contamination of the equipment, tools and clothing of the personnel involved in the maintenance. Also, leakage from the system during operation or spillage during maintenance could contribute to the contamination of the equipment, tools, and clothing.

9.7.2 Methods of Decontamination

Surface contaminants, which are found on equipment in the primary system and the spent fuel pool are in contact with the water and are removed by conventional techniques of flushing and scrubbing as required. Tools are decontaminated by flushing and scrubbing since the contaminants are generally on the surface only of non-porous materials. Personnel and their clothing are decontaminated according to the standard health physics requirements.

Those areas of the plant, which are susceptible to spillage of radioactive fluids, are painted with a sealant to facilitate decontamination that may be required. Generally, washing and flushing of the surfaces is sufficient to remove any radioactivity present.

The corrosion films generally are tightly adhering surface contaminants and must be removed by chemical processes. The removal of these films is generally done with the aid of commercial vendors who provide both services and formulations. Since decontamination experience with reactors is continually being gained, specific procedures may change for each decontamination case.

Portable components may be cleaned with a combination of chemical and ultrasonic methods if required.

9.7.3 Decontamination Facilities

Decontamination facilities on site consist of a decontamination room located in the Auxiliary Building near the personnel access airlock, a decontamination sink in the hot instrument laboratory, and a cask decontamination pad located adjacent to the Spent Fuel Storage Pool. Fuel handling tools and other tools can be cleaned and decontaminated in the cask decontamination pad area.

In the cask decontamination pad, the outside surfaces of the shipping casks are decontaminated, as required, by using water detergent solutions and manual scrubbing to the extent required. When the outside of the casks are decontaminated, the casks may be removed with the Auxiliary Building crane.

In the equipment cleaning areas, located in the decontamination room and in the hot instrument laboratory, equipment and tools can be decontaminated by using water, detergent solutions, and manual scrubbing to the extent required.

For personnel, a decontamination shower and wash sink are located adjacent to the Radiation Protection Office.

9.8 SYSTEM EVALUATION

The Shield Building Ventilation System in the event of a loss-of-coolant accident; will produce a vacuum in the Shield Building annulus and will cause all leakage from the Containment Vessel to be mixed in the annulus volume, and recirculated through a filter system before its deferred release to the environment through the exhaust fan that maintains vacuum.

The free-standing Reactor Containment Vessel is designed to accommodate the maximum internal pressure that would result from the Design Basis Accident. For initial conditions typical of normal operation, 120°F and 15 psia, an instantaneous double-ended break with minimum safety features effectiveness results in a peak pressure transient as presented in Section 14.3.4, Containment Integrity Evaluation.

The Reactor Containment Vessel has been successfully strength-tested at 51.8 psi and leak-tested at 46.0 psi to meet acceptance specifications previous to installation of penetrations.

Section 14.3.5, Off-Site Dose Consequences is based on a conservatively chosen reference set of assumptions regarding the sequence of events relating to activity release and attainment of vacuum in the Shield Building annulus, the effectiveness of filtering, and the leak rate of the Containment Vessel as a function of time.

The initial leak rate in this analysis is 0.5 weight percent of air per 24 hours at the peak containment pressure of 46 psig. The resulting doses are less than the guidelines presented in 10 CFR 50.67.

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9.9 INTERNAL STRUCTURES

NOTE: In 2001, Kewaunee Nuclear Power Plant replaced the original Model 51 steam generators with Model 54F steam generators. The relevant changes that the replacement steam generators produce on the primary system include a decrease in the resistance to primary flow (and therefore an increase in the primary flow rate) and an increase in the primary volume (more tubes have been added to the replacement steam generators). The resistance reduction results from a 6 percent increase in tube flow area within the steam generator. The associated primary flow rate increase is calculated to be 6 percent. The primary volume increase is 3 percent within the steam generator and 1.2 percent considering the total primary system volume. Since the subcompartment pressurization rate and peak pressures resulting from a leak in the primary piping or surge line could be impacted by the flow resistance in the lines and the total primary side volume, a change to these values would be possible if the LOCA analysis were to be redone. The impact is minimal based on the magnitude of the resistance and volume changes and the fact that the key driving factors (pipe diameter and pressure difference) remain unchanged. Therefore, the analysis for subcompartment pressurization for the original steam generators, as described in the following sections, remains a bounding analysis. Instead of re-performing this analysis, KNPP used the leak-before-break provision of General Design Criterion 4 and KNPP's design basis (Reference 14, Reference 15 and Reference 19).

Subcompartment pressurization for the breaks of smaller primary lines remain bounded by the existing primary loop LOCA analysis results. The next largest diameter branch connection off of the primary loop piping is the 12 inch Safety Injection line. Since flow rate (and resulting pressure buildup) in subcompartments is proportional to the pipe flow area (note that the pressure difference would remain the same), and since the 12 inch diameter pipe has a flow area of about 1/7 of the 31 inch pipe, the pressure buildup inside containment for a pipe break of a 12 inch pipe is significantly less than that for the 31 inch pipe. The minor changes caused by the changeout of the steam generators have no impact on this conclusion and the existing LOCA analysis easily bounds the results for the smaller diameter pipes.

9.9.1 Description

The Reactor Containment Vessel contains the reactor cavity, which houses the reactor pressure vessel, two reactor coolant system compartments, a refueling cavity, which is located between the primary coolant compartments and above the reactor cavity, and miscellaneous equipment rooms and work areas (see Figure 5.1-1).

9.9.2 Design Basis

The reactor cavity, the reactor coolant system compartments, and all equipment rooms housing Class I components or systems as defined in Appendix B, are designed as Class I structures. The design criteria for structures as described in Section B.6 of Appendix B is applicable with the exception of environmental loads such as snow, wind and tornado, which do not affect containment internal structures.

The design has been performed in accordance with the applicable portions of the following design codes:

- American Concrete Institute Code 318 "Building Code Requirements for Reinforced Concrete," 1963 Edition
- American Institute of Steel Construction "Specification for the Design, Fabrication and Erection of Structural Steel Buildings," 1963 Edition
- American Welding Society Code D 1.0 "Standards for Arc and Gas Welding in Building Construction"
- International Conference of Building Officials "Uniform Building Code," 1967 Edition
- Atomic Energy Commission publication TID 7024 "Nuclear Reactors and Earthquakes"

The design pressure differentials across walls and slabs of the enclosed compartments in the internal structure have been calculated for the hypothetical double-ended pipe rupture at any location within the Reactor Coolant System compartments or within the piping penetration of the reactor cavity shield up to the last field weld on the Reactor Coolant System piping. The pressure differentials used for this design are:

- Reactor Cavity (Nozzles) - 475 psia
- Reactor Vessel Gap - 100 psia
- Reactor Steam Generator and Pump Vaults - 25 psia

In addition to the peak pressure differentials, the Reactor Coolant System compartment walls are designed for simultaneous action of a single jet impingement load and the Operational Basis Earthquake.

The supports for all NSSS equipment including the reactor pressure vessel, the steam generators, the primary coolant pumps and the pressurizer are designed to remain within the elastic range following the rupture of any pipe, except the reactor coolant loop piping and the pressurizer surge line, combined with the Design Basis Earthquake. Elimination of postulated breaks in the reactor coolant loop piping is justified by application of "leak-before-break" technology in accordance with the limited 7 scope rule change of General Design Criteria 4 (51 FR 12502, dated 4/11/86). The analyses performed, which demonstrate leak-before-break in the reactor coolant loop, are reported in Reference 12 and Reference 13. NRC approval is provided in Reference 14 and Reference 15. Elimination of postulated breaks in the pressurizer surge line is justified by a leak-before-break analysis using methodology described in the

Standard Review Plan (52 FR 32626, dated 8/28/87). The analysis, which demonstrates leak-before-break for the pressurizer surge line, is documented in Reference 18. NRC approval is provided in Reference 19.

The internal reinforced concrete and steel structures are designed using the working stress method, and are based on the allowable stresses as set forth in Table 9.9-1.

The design was reviewed to assure that any resulting deflections or distortions will not prevent the proper functioning of the structure or piece of equipment, will not endanger adjacent structures or components, will not allow the uncontrolled release of radioactive material, and will not prevent the safe shutdown and isolation of the reactor. Earthquake stresses were compared with stresses implied by the damping values used, to assure that the analysis was consistent.

The reactor cavity is designed to contain safety injection water up to the level of the reactor nozzles, and reduce the likelihood of secondary missiles during a pipe rupture.

Pipe whip restraints are provided, as required, to prevent consequential damage to the Reactor Coolant System, Engineered Safety Features systems, Containment Isolation Systems, and the Reactor Containment Vessel as the result of a double-ended rupture of high-pressure piping within containment (except the reactor coolant loop and the pressurizer surge line).

The design of the internal structures has been examined to assure that credible missiles from high-pressure equipment will not cause a loss of function within the structure, the Reactor Coolant System, the Containment Isolation Systems or the Reactor Containment Vessel. The principal barriers against missiles are the reinforced concrete biological shield, the shield walls around the Reactor Coolant System compartments, and miscellaneous local barriers. A removable steel plate is located above the reactor vessel head to block any missile that could be generated by a failure of the RCCA drive mechanism.

9.9.2.1 Reactor Coolant System Compartments and Refueling Cavity

There are two Reactor Coolant System Compartments each housing one loop of the primary nuclear steam supply components, namely; pressurizer, steam generator, and reactor coolant pump. The compartments are vertical vaults of reinforced concrete construction with flat walls having an irregular geometric wall configuration.

9.9.2.2 Loads

The vault compartment structure and refueling cavity are designed to carry the vertical and horizontal live and dead loads of adjacent floors and structures that are attached to it as well as its own live and dead loads and horizontal loads imposed on the compartment walls from the equipment components contained within and external to the compartments.

Also included in the Design are:

- Jet Load

  Jet impingement loads on compartment surfaces that are postulated to occur in the event of a pipe rupture.

- Internal Pressure

  Compartment internal pressure buildup that would occur during a LOCA blowdown. These loads are defined in Section 9.9.2.

- Pipe Rupture Reaction Load

  These are reaction forces on equipment and structures caused by a postulated pipe rupture, except ruptures in the reactor coolant loop piping and the pressurizer surge line.

- Thermal Load

  These are stress loads from thermal gradients through compartment walls that are caused by a LOCA.

- Residual Construction Loads

  These are internal residual construction stress loads resulting from the shrinkage of the various levels of construction pours of concrete.

- Refueling Cavity Pool

  Vertical pool water load and lateral hydraulic water pressure loads from water in the refueling cavity pool. These loads are treated as live loads since they will only occur during reactor shut-down.

- Seismic Loads

  Seismic loads are determined from the John A. Blume analysis (Reference 3) of the Reactor-Auxiliary-Turbine buildings using data for the appropriate mass points under consideration.

9.9.2.3 Method of Analysis

The general structural model consists of a composite of the compartment walls, the adjoining biological shield, the attached floor systems and structures, and the refueling cavity pool.

The analytical model is broken down into two district interacting models (1) laminar horizontal frames analyzed by STRESS programs and (2) svertical structural elements analyzed to carry unbalanced load reactions to the base by the box girder action of the compartment walls.

This latter model could more aptly be described as a modified folded plate structure.

The local effects of forces are analyzed by considering loading conditions on simplified but conservative models limited to the immediate and close by adjoining areas of loading.

The above described analysis results were later confirmed by an independent finite element method of analysis that indicated conservative margins in the results of the original analysis on the order of 30 percent and greater.

9.9.2.4 Design

Concrete structures are designed in accordance with the working strength design methods of the ACI-318 Code.

Concrete is specified to attain a minimum strength of 4000 psi in 28 days. Reinforcement is ASTM-A615, Grade 60. No bar sizes greater than No. 11 are used. No welding of reinforcement was permitted.

In those portions of concrete structures subject to dynamic loadings and having heavy tensile reinforcement, particular attention was given to enhance the ductility of the concrete by the addition of stirrup systems (Reference 10).

9.9.2.5 Design Criteria

The design stress limits for the various design condition categories are shown on Table 9.9-2. This criteria is in accordance with Section 9.9 and B.6.

Criteria and stress limits are shown for reinforced concrete and structural steel. Structural steel criteria and stress limits applies to those members that are a functioning integral part of the compartment structure system, i.e.; embedded items, structural wall-tie members, etc.

Pipe rupture reaction forces and resulting jet impingement loads will act on limited extents of the structures. Because of the instantaneous nature of these loads they can produce very high local stress concentrations on parts of the structures. Accordingly, these loads require a somewhat different treatment than categories 1, 2, & 3. Table 9.9-2 has been expanded to include category-4, “Pipe Rupture Reaction Loads and Jet Loads”.

This category will include the local effects of pipe rupture reaction loads and jet forces. The criteria and increased stress limits presented in this category take into account the self-equilibrating nature of these localized stresses. This is analogous to the techniques employed in the design of pressure vessels. The design stress limits indicated are consistent with the design margins used throughout the plant.

Because of the time related effects of the reaction forces induced by a pipe rupture when considered simultaneously with internal compartment pressure buildup from a LOCA blowdown, two limiting loading conditions were established for category-4, as shown in Table 9.9-2.

9.9.2.6 Design Verification

Subsequent to the design process, an analysis was performed which used as-built data to verify the adequacy of the design for the reactor coolant system compartments.

9.9.2.7 Analysis Model

A five-volume model was used to calculate the peak differentials for the steam generator vault. The volumes represented were the dome, the base compartment, the foundation void and the two steam generator vaults. Time dependent equations of conservation of mass, energy and momentum, and the equation of state were used in the calculation. Flow inertia effects between the volumes were calculated. This model calculated critical flow conditions for application under high pressure differentials. The model assumed 100 percent entrainment of the water, which emerges from the break. The time intervals used in the calculations up to the time of the peak differential were one millisecond steps.

The pressures in the pipe annulus and reactor annulus were established through a steady state analysis of the peak mass and energy flow in the region. This study utilized two-phase critical flow relations. The pressure in the compartment below the reactor was established by addition of a sixth volume, representing this compartment, to the model used to calculate the peak differentials of the steam generator vaults. A steam and energy flow was input to this new volume to calculate the resultant peak differential. The time interval in use when the peak pressure was calculated was four milliseconds, although the time steps of one millisecond and two milliseconds were used earlier in the transient. The lengthening of the time step is due to the low rate of change of the blowdown after initial flow rise through the break. The shorter time steps are used to follow the rapidly changing portion of the blowdown.

9.9.2.8 Analysis Assumptions

The steam generator vaults of the Kewaunee plant, shown schematically in Figure 9.9-1, have been analyzed to determine the peak pressure differentials. The peak differentials have been calculated for a double-ended rupture of a reactor coolant pipe. One steam generator vault has a volume of 21,740 ft^3 and vent areas of 360 ft and 111.3 ft; the other vault has a volume of 24,500 ft and vent areas of 394 ft and 106 ft. Volumes and vent areas used in the analysis are shown in Figure 9.9-1.

The reactor cavity region has been analyzed for the effects of a longitudinal split inside the pipe sleeve of area equivalent to the cross-sectional area of a reactor coolant pipe. A circumferential failure of the pipe at this location would result in a much smaller flow discharge area because the vessel, pipe, and sleeve arrangement is such that no significant relative movement can take place. The compartments analyzed are the pipe sleeve, reactor vessel annulus, and the compartment below the reactor vessel. The volume of the pipe annulus containing the break is 117.3 ft and the vent area is 23.5 ft. The volume of the reactor vessel annulus is 223 ft and the vent area is 54.2 ft. The volume of the compartment under the reactor vessel is 4709 ft and the vent area is 33.2 ft. The areas reported above are net as-built vent areas. Volumes and vent areas used in the analysis are shown in Figure 9.9-2.

9.9.2.9 Analysis Results

The maximum pressure achieved in the smaller steam generator vault at the time of the peak differential pressure is 28.7 psig. The maximum differential between the steam generator vault and the dome is 27.4 psi and the maximum differential between the steam generator vault and the base compartment is 23.9 psi.

The maximum pressure calculated in the pipe annulus is 350 psig and this has also been selected as peak differential pressure. The maximum pressure in the reactor annulus is 84 psig, and again the peak differential has been selected as 84 psi. Finally, the peak pressure in the cavity below the vessel is 25 psig and the peak differential pressure is 24.3 psi.

The jet force loads from rupture of various pipes used in compartment structure designs are very conservative, and are as follows:

Break Location	Jet Force
• Primary loop hot leg	1800 kips
• Primary loop cold leg	1600 kips
• Crossover (pump suction leg)	2250 kips
• Steam line	813 kips
• Feedwater line	277 kips

The above jet forces are calculated by the method described in Section 10A.12.

The pressure differential values discussed above are from the re-analysis of compartment pressures. A comparison with the values assumed for design purposes are shown below:

Location	Re-analysis	Design Values
• Steam generator vault – base compartment	23.9 psi	25 psi
• Steam generator vault – bulk containment volume	27.4 psi	25 psi
• Reactor vessel gap – bulk containment volume	84.3 psi	85.3 psi
• Pipe annulus – bulk containment volume	350 psi	460.3 psi

The compartment structures surrounding the steam generator vaults have considerable margin in their design because of the design method used initially. The design was first done using a simplified analysis and allowance was made for uncertainties. A subsequent finite element analysis showed the structural design could withstand nearly twice the originally predicted differential pressure loading.

9.9.2.10 Analysis Conclusions

9.9.2.10.1 Steam Generator Vault

Compartment Structural Design

The steam generator vault design depends primarily on steel reinforcement to carry the loads induced by compartment differential pressures, jet impingement forces and steam supply equipment reaction forces resulting from pipe rupture. Loads from compartment pressure differentials represent only a minor part of the total design load. The wall thickness was determined from biological shielding requirements and the resulting thickness exceeds that required by the loading configuration. The horizontal and vertical steel reinforcing bars carry the loads associated with the hoop stresses and longitudinal stresses, respectively. Steel reinforcing bars formed into stirrups carry the shear load. Figure 9.9-3 shows the orientation of the stirrups.

The simplified analysis initially used in the design of the compartments was a quasi-folded plate method. This method exaggerates the loading on the transverse and longitudinal frames examined during the analysis because some actual loads are accounted for twice, and this repetition leads to a structure that is very conservatively designed. The finite element analysis subsequently performed, demonstrated the degree of conservatism of the initial analysis.

9.9.2.10.2 Structural Analysis of Steam Generator Compartment Model

The finite element computer code Structural Analysis Program (SAP) was used for the analysis. The structure for the compartment around Steam Generator 1A was modeled. The compartment configuration is shown in plan view in Figure 9.9-4 and in section on Figure 9.9-5. The model consists of vertical steam generator and reactor coolant pump shield walls, mezzanine floor, missile shield slabs, knuckle wall, and steam generator support steel beams.

The structural model consists of four different types of elements connected by 2151 nodal points. The number and type of elements are:

- 967 three dimensional brick elements
- 40 three dimensional beam elements
- 42 elastic shell elements
- 111 boundary elements

9.9.2.11 Loads

Structural integrity was analyzed for the 24 loading cases listed in Table 9.9-3. The loads are defined in the following discussion.

The maximum pressure differential load was determined from the compartment pressure analysis described above. The maximum pressure differential was 27.4 psi at 0.17 sec. after the break.

The seismic loads are determined from the analysis provided by Reference 3. The Operational Basis Earthquake (OBE) has been defined as having ground acceleration of 0.06g. The Design Basis Earthquake (DBE) has been defined as having ground acceleration of 0.12g. For this analysis an equivalent static load method was used.

Pipe rupture forces (namely jet impingement and pipe reaction forces) were developed from a detailed analysis of the steam generator supports and calculations done expressly for this analysis.

9.9.2.12 Load Combinations

Loading combinations for design and stress investigation of the compartment structures are defined in Table 9.9-2 (see notes in Table 9.9-2).

- Load Combination 1: Normal Operation

  Stresses are very low during normal operations since the operating loads are extremely low when compared to accident induced loads. The massive self supporting walls and missile slabs are designed to restrain equipment and piping of Reactor Coolant System from lateral movements during hypothetical accidents.

  Similarly, compartment structures are also designed to provide biological shielding from radiation, and this requires wall thickness in excess of those required for structural integrity.

- Load Combination 2: Design Basis Accident

  This load combination includes maximum calculated internal differential pressure plus Operating Basis Earthquake. The wall acceleration used was 0.12g.

- Load Combination 3: Safe Shut-Down

  Load Combination 3 is similar to Load Combination 2 above except that the earthquake loading is the Design Basis Earthquake. The wall acceleration used was 0.24g.

- Load Combination 4: Local Effects Consideration

  This combination includes operating load and DBE in addition to an appropriate superposition of the time dependent pressure differential loads, pipe rupture reaction loads and jet impingement loads.

The highest loading results from a double-ended crossover pipe break. The peak load occurs at the time when the pressure differential load is 0.766 of its maximum value.

Load Combination 4 has also been modified to include hypothetical pressure differential loads of 50 psi and 75 psi.

9.9.2.13 Discussion of Results of Finite Element Analysis

Concrete walls are modeled with three-dimensional brick elements. The computer output consists of stresses at the element faces with the general convention described in Figure 9.9-6 and, as indicated, shear stresses are part of the unique solution for each loading case.

The most highly stressed elements of each load combination were selected for further detailed examination. The stresses on the brick elements were converted to axial forces, shear forces and moments, at each face. Then the stresses in the steel reinforcing bars and stirrups and concrete were calculated. These stresses were compared with Code allowable values as given in Table 9.9-2.

Because most sections are subjected to axial loads, the shearing capacity of the concrete was computed by:

$$v_c = 1.75\sqrt{f'_c\left(1.0 + 0.004\,N/A_g\right)}$$

where N = axial force (if N is tension, negative sign reduces concrete shearing capacity), for Load Combinations 2 and 3. For Load Combination 4 the shearing capacity of the concrete was computed by:

$$v_c = 2\left(1.0 + 0.002\,N/A_g\right)\sqrt{f'_c}$$

The stresses for the critical sections are summarized in Table 9.9-4. For Load Combination 4.a) the calculated stresses are below the stress limits. The assumed allowable concrete stress design limit of 3000 psi is well below the actual measured 28-day compression strength of 6000 psi.

Further, in regards to the reinforcing bars specified by code, minimum yield strength is 60,000 psi but it has a tested average strength of 70,000 psi.

The margin of the compartment structural capability is indicated by the results for Load Combination 4. c) where, even for a hypothetical pressure differential load of 75 psi, the critical concrete stress is below both the specified and actual measured strength and the reinforcing steel stresses are below the specified and measured yield limits. Essentially no permanent deformation would be expected to occur.

Initially only the compartment walls of Unit 2 of Prairie Island, which are similar to Kewaunee compartments, were analyzed by a finite element analysis. In this case reinforcing was provided by 40 percent more than the analysis required.

9.9.2.14 Structural Analysis of Reactor Vessel Cavity Compartment

The load carrying ability of the reactor vessel cavity compartment is provided primarily by the 7.33-ft thick steel reinforced concrete walls. Steel supports for the reactor vessel are also encased in the concrete structure.

The dead load consists of the weight of the walls, equipment and water in the refueling pool. Thermal loads result from the neutron attenuation-induced thermal gradient in the wall. The long term design differential pressure for the cavity is 100 psi, however, a short-term differential pressure of 185 psi and an upper bound pressure differential of 480 psi have been considered for analysis purposes.

Seismic loads are determined from Reference 3. The highest structural stress and design limits for different load combinations are given in Table 9.9-5.

9.9.2.15 Nozzle Cavity

The applicable design loads for the nozzle cavity are as follows:

1. Differential pressure load due to pipe rupture in the nozzle region. The most critical stresses and the associated stress limits for the design differential pressure of 475 psi and an upper bound differential pressure of 1400 psi are given in Table 9.9-6.
2. Reaction loads imposed by the pipe restraints on the nozzle cavity walls for a pipe rupture in the nozzle region. Under normal operating conditions, the restraints impose negligible loads on the cavity walls. The normal operating temperature in the concrete is kept below the ASME B&P.V. Code, Section III, Division 2, allowables of 150°F overall and 200°F local. The rupture loads imposed by the restraints will result in the indentation of the restraint tips into the steel liner and a confined local bearing load on the concrete cavity wall.

9.9.2.16 Structural Supports for Primary System Components

9.9.2.16.1 Loads

The structural support system for the steam generators and reactor coolant pumps is designed for the following loads:

- Operating Loads

  Consists of dead loads, live loads and loads caused by thermal expansion.

- Seismic Loads

  Seismic loads are determined from J.A. Blume Response Acceleration Spectra Curves (Reference 4) for the appropriate mass points under consideration.

- Pipe Rupture Reaction Loads

  These are reaction forces on equipment and structures caused by a postulated pipe rupture, except ruptures in the reactor coolant loop piping and the pressurizer surge line.

- Operating Loads On Ventilated Support Structures

  Operating loads consist of:

  a. Reactor vessel vertical dead loads,

  b. Lateral radial friction loads caused by radial thermal expansion of the reactor vessel,

  c. Internal loads within the ventilated support structures caused by local temperature gradients.

9.9.2.17 Method of Analysis

The steam generator analytical structural model consists of three related structural support systems: Pin-ended support columns, a lower lateral bumper support system, and an upper lateral suppressor dampened support system (see Figure 9.9-8). The three support systems are analyzed statically as a deformation related single system. Appropriate dynamic load amplification factors are applied to the dynamic loads of pipe rupture reactions.

The reactor coolant pump analytical structural model consists of pin-ended support columns and pin-ended horizontal tie members located at the tops of the columns (see Figure 5.9-11). This model is analyzed statically with a similar treatment of dynamic load amplification factors as described above.

The reactor vessel is the structural rigid center of the two loop system. The analytical structural model consists of six vertical support columns rigidly tied together at their tops and anchored into the massive concrete biological shield structure (see Figure 9.9-7).

The ventilated support structures as described in Section 4.2.2 are modeled as a deformation related group of structures designed to support the reactor vessel vertically and to restrain its horizontal movement. The entire analytical model is treated by statical methods with appropriate application of dynamic load amplification factors.

An independent time-history dynamic analysis of the complete assembly of the separate structural support systems modeled together with the complete nuclear steam supply equipment components and primary loop piping was made by the Westinghouse Electric Corp. The results of these two independent analyses were found to be a consistent agreement. Also, an independent time-history dynamic analysis of the steam generator, its support system, and the steam piping was made by Pioneer Service & Engineering Co. and the results of this analysis were also found to be in consistent agreement with the original static analysis.

9.9.2.18 Design

9.9.2.18.1 Reactor Vessel (Figure 9.9-7)

The reactor vessel is supported on six vertical steel H-Columns embedded in the biological shield concrete. The tops of these columns are furnished with ventilated support structures to provide for a suitable temperature gradient between the heated parts of the reactor vessel coming in contact with the supports and the supporting steel columns and surrounding concrete below. These ventilated support pads are discussed in Section 4.2.2.

Fitted key slot blocks that are furnished with the reactor and bolted to the ventilated support pads provide for the free radial thermal expansion of the reactor vessel. Machined keys that are integral with the reactor vessel nozzles and support lugs are shimmed for sliding fits in the key slots and restrain the reactor vessel from movement in any horizontal direction. Thus the center point of the Reactor Vessel is rigid and has a “zero” lateral movement.

The tops of the steel H-Columns are connected together by means of a structural tee horizontal bracing system that is welded to a continuous outer steel band. This entire bracing system is embedded in concrete to provide a rigid anchorage. Stud anchors are welded to the flanges and web along the column length to assure the composite action of concrete in carrying the vessel loads, thereby providing additional load carrying margins in the supporting structural system.

9.9.2.18.2 Steam Generators (Figure 9.9-8)

Vertical support consists of four steel H-Columns hinged at each end to provide for unrestrained movement in the direction of thermal expansion. The column ends are securely bolted into the steam generator support lugs and anchored by embedded bolts at the base to provide for uplift forces that are postulated to occur during lateral loading of the support system.

Two lateral levels of supports are provided for the lateral seismic and pipe rupture loads.

The lower lateral support provides restraint in three of the six directional degrees of freedom by means of close clearance bumper-pedestals that are mounted on the compartment walls and on horizontal girders that span between compartment walls. These bumper-pedestals were fitted and shimmed for close clearance during pre-operational heatup.

The upper lateral support system consists of a horizontal ring girder that is fitted and shimmed to the contour of the steam generator shell with clearance allowance for the thermal expansion of the girder-shell system. The upper lateral support girder provides restraint in four directions by means of three attached bumper-pedestal plates that come in close clearance contact with wall mounted bumper plates in the hot position of the generator. The fourth side of the girder is provided with a hydraulic suppressor whose longitudinal axis is in alignment with the direction of thermal expansion through the centerline of the steam generator. The suppressor is attached to the girder and the compartment wall by means of pivoted linkage brackets to allow for vertical thermal displacement of the girder. The hydraulic suppressor provides for freedom of movement in the direction of thermal expansion during heatup and cooldown and provides dampened restraint to the sudden loads of seismic action and pipe rupture.

The reactions of jet forces in the main steam line at the top of the steam generator are restrained by means of two cable anchors that are fitted with yokes welded to the pipe bends.

Based on independent flow stability computations in conjunction with information submitted by WPSC, the NRC has concluded that the KNPP primary loop piping complies with the revised General Design Criterion 4 (GDC-4) of Appendix A to 10 CFR Part 50. Thus, the probability or likelihood of large pipe breaks occurring in the primary coolant system loops is sufficiently low such that the dynamic effects associated with postulated pipe breaks need not be a design basis. The NRC's Safety Evaluation Report provided in Reference 14 approves this application of fracture mechanics "leak-before-break" (LBB) technology. This, along with the elimination of arbitrary intermediate breaks in accordance with NRC Branch Technical Position MEB 3-1, Rev. 2, provided the basis for the removal of three of the four snubbers on the upper lateral support of each of the Steam Generators. Reference 15 is the NRC's Safety Evaluation for KNPP's Steam Generator snubber reduction.

9.9.2.18.3 Primary Piping (Figure 5.9-9)

The reactions of jet forces in primary loop piping caused by pipe rupture, which was postulated in the original design prior to elimination of the dynamic effects of primary loop rupture (see page 5.9-2), are restrained by means of heavy structural steel weldment brackets rigidly anchored at various points throughout the internal concrete structures. Since the NRC approved the use of the "leak-before-break" analysis at KNPP (Reference 14), the need for the RCS crossover leg whip restraints has been eliminated. Analysis by Westinghouse, as part of the Steam Generator Replacement and the Power Uprate Projects, demonstrate the allowable stresses and displacements of the RCS piping, steam generators, primary equipment supports and primary equipment nozzle loads remain within acceptance criteria.

In order to limit the asymmetric LOCA loads for the postulated reactor coolant pipe break in the nozzle region, restraints have been provided for the hot and cold leg pipes in the penetrations of the reactor shield wall. The bearing pressure on the pipe exerted by the restraints is limited to 15KSI for the postulated accident conditions. Gaps between pipes and restraints are maintained as required by the original design. The lateral movement of pipe at the restraint is limited to 1½ inches, excluding the deformation of the pipe but including the gap. A factor of safety is provided against premature buckling of compression members. The structural steel supporting the restraints was designed for load combinations and allowable stresses as specified in the FSAR using the latest editions of applicable codes at the time of original design.

9.9.2.18.4 Pressurizer (Figure 5.9-10)

The pressurizer is supported by the base concrete slab of the pressurizer vault and anchored thereto by means of 24 high strength bolts equally spaced on the circumference of the pressurizer support skirt.

9.9.2.18.5 Reactor Coolant Pumps (Figure 5.9-11)

Vertical support consists of three steel H-columns hinged at each end for vertical support and uplift hold-down while providing unrestrained freedom of movement laterally in the direction of thermal expansion during heatup and cooldown.

The connection between the top of the support columns and the pump support brackets is by means of a single high strength steel threaded rod. Between the support bracket and the top of the column are three high-strength tie bars with slotted holes at each end to accommodate the thermal expansion of the pump loop. The tie bars are anchored to the compartment walls and will prevent whipping of the pump in the event of a pipe rupture and/or a seismic event.

The structures are of heavy welded construction. The materials are specified to be ASTM-A588 high strength, low-alloy steel except for the coolant pump horizontal linkage bars, which are ASTM-A517. The connecting bolts to the steam generators are high strength precipitation hardening stainless steel. The connecting bolts to the pumps are high strength Maraging alloy steels. Anchor bolts are specified to meet ASTM-A540 requirements.

The design and fabrication was specified in accordance with the AISC Specifications for the “Design, Fabrication, and Erection of Structural Steel for Buildings” and selected provisions of the ASME Boiler and Pressure Vessel Code. Welding is in accordance with Section IX of the ASME Code.

In general, the final design sizing of members and welds as determined by the specified design criteria have been increased approximately 40 percent above the minimum design requirements to provide adequate margins in the design.

All anchor bolts carrying dynamic loads of equipment and that are subject to the forces of pipe rupture reactions are designed to be positively anchored by means of embedded anchor plates.

All plate material was specified to be ultrasonically tested. Steel plates, structural shapes and anchor bolts were specified to meet Charpy impact test requirements of 15 ft-lb average and 12 ft-lb minimum at 40°F. Anchor bolts are magnetic particle inspected.

All welding procedures were pre-qualified. All welders were qualified for the project work regardless of previous qualifications. All welds were either ultrasonically inspected or magnetic particle inspected in progressive steps. All weld thicknesses over 1½ inches were stress-relieved.

All work was specified to at least meet the minimum equivalent quality assurance requirements of Appendix IX of the ASME Boiler and Pressure Vessel Code.

9.9.2.19 Design Criteria

Because of the impulse nature of pipe rupture dynamic loads and the short duration of these loads, on the order of 10 milliseconds or less, there will be high strain-rate loadings present in the response of the structures.

This factor of the dynamic material behavior has been taken into account in establishing the design criteria since the results of tests (Reference 11) have shown that the stresses associated with the initiation of yielding have been found to increase with increased strain-rates (Reference 11).

Table 9.9-7 indicates the design stress limits that were used for the various design condition categories. The stress limits that are indicated for concrete apply to the anchorage design of the support embedments.

9.9.2.20 Reactor Cavity Biological Shield

The biological shield is a massive reinforced concrete structure that surrounds the reactor vessel and all its nozzles and immediate piping. The shield structure is supported in the vicinity of the containment vessel bottom head. It supports the reactor vessel by composite action with the embedded support columns. The top of the shield forms the floor of the refueling cavity pool. Its interconnection with the steam generator compartments and floor systems integrates its design function into the total interior structure complex.

Neutron absorption by the concrete biological shield structure at the level of the reactor core will result in a temperature of 118°F on the inside surface of the concrete facing the reactor vessel and an internal concrete temperature which will peak at about 190°F two (2) feet inside the concrete; these temperatures are not deleterious to the cement.

The thermally induced stresses in the concrete shield structure caused by neutron absorption were structurally analyzed using the "IBM M003 FLEXIBILITY ANALYSIS" program.

The concrete and reinforcement design of the concrete shield structure is based on the "Working Stress Design Method" of the ACI 318-63 code.

To provide generous margins in the design of the six (6) structural steel columns that support the reactor vessel and that are embedded in the concrete of this region, the structure was analyzed as an independent vertical support system and was assumed to carry the total reactor vertical load without relying on the surrounding concrete. The embedded structural steel members were designed in accordance with the requirements of the AISC, "Specifications for the Design, Fabrication and Erection of Structural Steel for Buildings".

The design criteria used is stated in Table 9.9-1 except that stress caused by thermal effects is included in all categories of loading.

9.9.2.21 Loads

The shield structure is designed to carry the vertical and horizontal live and dead loads of adjacent floors and structures that are attached to it as well as its own dead load.

Also included in the design are:

- Internal Pressure

  Pressure forces within the confines of the structure that are postulated to build-up in the event of a LOCA. These pressures are identified in Section 9.9.2.

- Neutron Absorption

  Stress load from neutron absorption. This is discussed above.

- Safety Injection Water

  Water pressure loads from the pooling of safety injection water in the reactor cavity up to the level of the nozzles.

- Reactor Vessel

  Reactor vessel dead and live loads.

- Refueling Cavity Pool

  Refueling cavity pool water load. This load is treated as a live load and will occur only during reactor shutdown.

- Residual Construction Loads

  These are internal residual construction stress loads resulting from shrinkage of the various levels of construction pours of concrete.

- Seismic Loads

  Seismic loads are determined from the John A. Blume analysis (Reference 3) of the Reactor-Auxiliary-Turbine buildings using data for the appropriate mass points under consideration.

- Pipe Rupture Reaction Load

  These are reaction forces on equipment and structures caused by a postulated pipe rupture.

9.9.2.22 Method of Analysis

The biological shield was analyzed in the integrated structural model complex as previously described.

9.9.2.23 Design

Design is same as outlined in the Reactor Coolant System Compartments and Refueling Cavity.

9.9.2.24 Design Criteria

The design stress limits for the various design condition categories are as set forth in Table 9.9-8. Because of the massive nature of that portion of the shield structure below the vessel supports and its general involvement in structural action with the compartment structures, the design criteria established generally followed those criteria set for the compartment structures.

The shield structure above the vessel supports will respond to applied loads of a more local nature, therefore, the criteria established in Table 9.9-2 for the local effects of pipe rupture were used.

9.9.2.25 Floor Systems

Floor systems consist of the following types:

- Monolithic reinforced slab and beam construction,
- Composite concrete slabs and steel framing,
- Steel grating on steel framing.

The floors form an integral part of the knuckle support system as well as the interior concrete complex as described in preceding sections.

9.9.2.26 Loads

The floors are designed to carry the following loads:

- Dead Loads
- Floor Live Loads
- Equipment Loads
- Construction Loads
- Seismic Loads
- Loads Imposed on the Floor System caused by a postulated pipe rupture.

9.9.2.27 Method of Analysis

To avoid impractical and unwieldy analytical models the floor systems were modeled with any one or combination of the following models as was appropriate:

- Individual floor support system
- Floor and knuckle support system
- Floor and interior structure system

In most instances, floors were modeled as individual floor support systems since steel framing structures are secondary structures and therefore do not contribute to the structural support of the primary structures under the conditions of DBA, OBE, and DBE. These supporting steel structures as well as the composite floor structures are analyzed and designed to include the seismic effect of the structure dead loads and supported equipment.

9.9.2.28 Design and Design Criteria

Concrete floor systems are designed in accordance with the Working Stress Design Methods of the ACI-318 Code.

Structural steel is designed in accordance with the AISC Specifications for the "Design, Fabrication and Erection of Structural Steel for Buildings".

Figure 5.9-12 shows the typical arrangement of the reinforcing steel patterns for reinforced concrete beam floors.

Figure 5.9-13 shows the typical arrangement of the reinforcing patterns for the steam generator and reactor coolant pump enclosure walls.

Design criteria is in accordance with Table B.6-2.

9.9.2.29 Stresses in Interior Structures

Table 9.9-9 summarizes the allowable and the maximum actual stresses in the four major categories of structures, which have been treated in detail in preceding sections.

The results of these analyses demonstrate that the critical stresses in all structures meet the criteria for allowable design stresses.

Provisions for reinforcing bar bond and anchorage in tension zones were in accordance with the requirements of the ACI Standard - Building Code Requirements for Reinforced Concrete ACI-318-63.

All splices and dowels were staggered wherever feasible and splices were located at points of minimum tension stresses. Regardless of the actual stress levels, splice laps and dowel lengths applicable for maximum stresses were used in accordance with Section 805(b) of the ACI Code where:

$$L = \frac{fs\ As}{\Sigma O\,(.75u)}$$

fs = 24,000 psi grade 60 reinforcing steel

fs = 20,000 psi grade 40 reinforcing steel

$$u\,(top\,bars) = \frac{3.4\sqrt{f'_c}}{D}$$

$$u\,(others) = \frac{4.8\sqrt{f'_c}}{D}$$

Anchorage of tension bars were in accordance with Section 1301(b) of the ACI Code where:

$$L = \frac{fs\ As}{\Sigma O\,(u)}$$

Regardless of the actual stress levels, the embedment length required for maximum allowable stress was used.

In the Shield Building wall, dome roof and internal Reactor Building concrete, the embedment and splice lengths “L” were increased an additional 10 percent for conservatism.

Coefficients of 1.5 and 1.67 times the allowable ACI Code working stress design values were used for DBE and tornado conditions respectively.

Further conservatism is realized because many design conditions were governed by construction loading and also the fact that the f'c = 4000 psi used in design is in reality 4800 psi or larger.

9.9.3 Materials

The specifications and working drawings for materials and their installation are of such scope and detail that the quality of internal structures is assured and the desired degree of integrity is obtained.

9.9.3.1 Concrete

The basic specifications for concrete materials and construction practices are as delineated in Section 5.2.2, Materials (Concrete). No high density concrete is used for structures. A small quantity of pre-cast high density shielding concrete may be provided at selected areas, but will not serve a structural function.

9.9.3.2 Reinforcement

Standards and specifications for reinforcement steel, testing, and construction methods are as follows:

- ACI 315 Manual of Standard Practice for Detailing Reinforced Concrete Structures
- ACI 318 Building Code Requirements for Reinforced Concrete
- CRSI 63 Recommended Practice for Placing Reinforcing Bars
- CRSI 65 Recommended Practice for Placing Bar Supports, Specifications and Nomenclature
- AWS D1.0 Code for MC and Gas Welding in Building Construction
- AWS D12.0 Recommended Practices for Welding Reinforcing Steel, Metal Inserts and Connections in Reinforced Concrete Construction
- ASME Boiler and Pressure Vessel Code, Welding Qualifications, Sections IX
- ASTM A615 Standard Specifications for Deformed Billet Steel Bars for Concrete Reinforcing, Grade 40 bars and Grade 60 bars
- ASTM A185 Specifications for Welded Steel Wire Fabric for Concrete Reinforcement

9.9.3.3 Structural Steel

Standards and specifications for structural steel, testing and construction and methods are as follows:

- “Specification for the Design, Fabrication and Erection of Structural Steel for Buildings,” published by the American Institute of Steel Construction
- “Code of Standard Practice for Steel Buildings and Bridges,” published by the American Institute of Steel Construction
- “Code for Welding in Building Construction,” published by the American Welding Society (AWS D1.0 and D3.0)
- “Boiler and Pressure Vessel Code,” Section VIII “Unfired Pressure Vessels,” and Section IX “Welding Qualifications,” published by the American Society of Mechanical Engineers

ASTM:

A36	Specifications for Structural Steel
A502	Specifications for Steel Structural Rivets
A325	Specifications for High-Strength Bolts for Structural Steel Joints, Including Suitable Nuts and Plain Hardened Washers
A354	Specifications for Quenched and Tempered Alloy Steel Bolts and Studs with Suitable Nuts
A490	Specifications for Quenched and Tempered Alloy Steel Bolts for Structural Steel Joints
A588	Specifications for High-Strength Low Alloy Structural Steel with 50,000 psi Minimum Yield Point to 4 in. thick
A240	Specifications for Chromium and Chromium-Nickel Stainless Steel Plate Sheet and Strip for Fusion-Welded Unfired Pressure Vessels - Type 304
A298	Specifications for Corrosion-Resisting Chromium and Chromium-Nickel Steel Covered Welding Electrodes
A276	Specifications for Hot-Rolled and Cold-Finished Stainless and Heat Resisting Steel Bars
A306	Specifications for Carbon Steel Bars Subject to Mechanical Property Requirements, Grade 70
A370	Methods and Definitions for Mechanical Testing of Steel Products

A593	Specifications for Charpy V-Notch Testing Requirements for Steel Plates for Pressure Vessels
E109	Method for Dry Powder Magnetic Particle Inspection
E164	Method for Ultrasonic Contact Inspection of Weldments

ASME:

Appendix VI	Section VIII	Methods for Magnetic Particle Examination
Appendix U	Section VIII	Ultrasonic Examination of Welds
N-532	Section III	Requirements for Postweld Heat Treatment

9.9.4 Construction

9.9.4.1 Concrete in Ellipsoidal Bottom

The placement of internal reinforced concrete in the bottom and in the knuckle region of the ellipsoidal bottom of the Reactor Containment Vessel followed the concrete fill and chemical grouting operations of the concrete fill support under the Vessel (Section 5.2.1) after the lapse of sufficient time for the concrete fill and chemical grouting to develop strength to carry the loads to be imposed by the internal concrete. The internal concrete was placed in a manner to minimize shrinkage effects of concrete. Also prior to placement of internal concrete at the knuckle, the temporary steel stiffeners for the steel ellipsoidal bottom of the vessel were removed.

9.9.4.2 Reactor Cavity Structure

The reactor cavity structure was designed and constructed as a composite structural component by using shim lags on embedded members, to integrate the steel structural supporting system for the Reactor Pressure Vessel with the reinforced concrete mass required for shielding.

The reactor cavity is fabricated from stainless steel liner plate in the areas of the nuclear core and carbon steel liner plate above and below the core area. During operation of the reactor, the liner plate in the core area will function to minimize the circulation of concrete dust by the Ventilation System, with its attendant effects on the buildup of radiation sources within the accessible areas of containment. In addition, during the placement of concrete the liner plate functioned as a form.

9.9.4.3 Steam Generator and Reactor Coolant Pump Vaults

The vault structures were designed and constructed as two systems which are complementary rather than composite:

1. The structural steel system within the vault consists of supporting legs and pinned brackets, upper and lower lateral supports, stops, eye bar and pin retainers. All of these members are secured by anchor bolts rather than by embedment into the concrete walls of the vault. The structural steel systems, basically acts to transmit the load to the reinforced concrete internal structure.
2. The reinforced concrete walls of the vault resist the radial and lateral loads transmitted by the structural steel system through tension in anchor bolts and bearing against load distribution plates. The loads are dissipated through the reactions of the mass of the walls. The massive reinforced concrete base dissipates the vertical loading of the structural steel (supporting legs) system.

9.9.4.4 Reactor Refueling Cavity

The Reactor Refueling Cavity is a complementary system consisting of formed reinforced concrete and stainless steel liner plate. The reinforced concrete wall was placed in two lifts except at the fuel transfer tube opening, where three lifts were required. The floor of the Reactor Refueling Cavity for the most part is the top lift of the reactor cavity concrete. The 3/16" stainless steel liner plate is field “seal” welded to stainless steel angles that are embedded and anchored in the reinforced concrete walls and floor.

9.9.4.5 Work Areas

The work areas consist of three types of construction, which are as follows:

- Top surface of mass (reinforced) concrete - i.e., basement floor
- Reinforced concrete construction, columns, beams and slabs - i.e., mezzanine floor
- Structural steel columns and beams and reinforced concrete structural slab - i.e., refueling and operating floor.

Chapter 9 References

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Table 9.9-1
Allowable Stresses - Internal Structures

Reinforced Concrete - Structural Steel

Loading Conditions	Reinforced Concrete	Structural Steel
1. Dead loads plus live loads plus DBA load	ACI 318-63 Allowable Values	AISC Allowable Values
2. Dead loads plus live loads plus DBA load plus OBE load (0.06 g)	ACI 318-63 Allowable Values	AISC Allowable Values
3. Dead loads plus live loads plus DBA load plus DBE load (0.12 g)	1½ Times ACI 318.63 Allowable Values	1 1/2 Times AISC Allowable Values
4. Dead loads plus live loads plus pipe rupture (Note 1)	$f_c = 0.75\ f'_c$ $f_s = 0.90$ Y.S.	$f_s = 0.90$ Y.S.
5. Missile loads	$f_c = 1.25\ f'_c$ f_s = Stress @ 0.50 Ultimate Strain	$f_s = 0.90$ Y.S.

f'_c = Minimum 28-day compressive strength of concrete
f_c = Compressive stress in concrete
f_s = Tensile stress in steel
Y.S. = Specified minimum yield strength or yield point of steel
ACI Allowable Values are working stress design allowable stresses

Note 1: Criterion for pipe rupture is based on maximum usable in pipe as per Westinghouse Electric Corporation “Support Design Criteria - Reactor Coolant System Components”.

Table 9.9-2
Design Stress Limits For Reactor
Coolant System Compartments and Refueling Cavity

Design Condition Categories	Design Stress Limits for Reinforced Concrete		Design Stress Limits for Structure Steel	
	Criteria	Stress Limits	Criteria	Stress Limits
1. Normal Operation Operating Loads[(1)]	ACI-318 (WSD)	Code Allowable	AISC (Part 1)	Code Allowable
2. Design Basis Accident Operating Load + DBA (Maximum Internal Differential Pressure Load)[(2)] + OBE	ACI-318 (WSD)	Code Allowable	AISC (Part 1)	Code Allowable
3. Safe Shutdown[(4)] Operating Load + DBA (Maximum Internal Differential Pressure Load[(2)] + DBE	ACI-318 (WSD)	1.5 x (Code Allowable)	AISC (Part 1)	1.5 x (Code Allowable)
4. Pipe Rupture Reaction Loads and Jet Loads (Local Effect)[(5)]				
a. Operating Load + DBA (Maxi mum Pipe Rupture Reaction Load[(2)] + DBE	ACI-318 (WSD)	0.75 f'_c (concrete) 0.9 Y.S. (reinforcement)	AISC (Part 1)	0.9 Y.S.
b. Operating Load + DBA (Pipe Rupture Reaction Load and Internal Differential Pressure Load)[(2)(3)] + DBE	ACI-318 (WSD)	0.75 f'_c (concrete) 0.9 Y.S. (reinforcement)	AISC (Part 1)	0.9 Y.S.

Table 9.9-3
Loads and Load Combinations for Finite Element Analysis

Loading Case Number	Loads and Load Combinations
1	Pipe Rupture; Hot Leg, Horizontal Run-A
2	Pipe Rupture; Hot Leg, Horizontal Run-S
3	Pipe Rupture; Cross-over Leg, Vertical Run, S.G. Side-A
4	Pipe Rupture; Cross-over Leg, Vertical Run, S.G. Side-B
5	Pipe Rupture; Cross-over Leg, Horizontal Run-A
6	Pipe Rupture; Cross-over Leg, Horizontal Run-B
7	Pipe Rupture; Cold Leg - A
8	Pipe Rupture; Cold Leg - B
9	Pipe Rupture; Cross-over Leg, Run-A (Split)
10	Pipe Rupture; Cross-over Leg, Run-B (Split)
11	Load Combination: 0.766 DDP* + (Case 1) + DBE**
12	Load Combination: 0.766 DDP + (Case 2) + DBE
13	Load Combination: 0.766 DDP + (Case 3) + DBE
14	Load Combination: 0.766 DDP + (Case 4) + DBE
15	Load Combination: 0.766 DDP + (Case 5) + DBE
16	Load Combination: 0.766 DDP + (Case 6) + DBE
17	Load Combination: 0.766 DDP + (Case 7) + DBE
18	Load Combination: 0.766 DDP + (Case 8) + DBE
19	Load Combination: 0.766 DDP + (Case 9) + DBE
20	Load Combination: 0.766 DDP + (Case 10) + DBE
21	Design Differential pressure = 27.4 psi
22	DDP + DBE
23	DDP + (0.5) x DBE
24	DDP + (1.0) x DBE

* Design Differential Pressure (DDP)
** Design Basis Earthquake (DBE)

Table 9.9-4
Steam Generator Vault Maximum Stress and Stress Limits

Load Combination	Design Stress Limits for Reinforced Concrete: Design Stress (psi)	Design Stress Limits for Reinforced Concrete: Stress Limit (psi)	Design Stress Limits For Reinforcing Steel: Design Stress (psi)	Design Stress Limits For Reinforcing Steel: Stress Limit (psi)
1. Normal Operation Operating Loads	Not Critical	Not Critical	Not Critical	Not Critical
2. Design Basis Accident				
Operating Load + DBA (Maximum Internal Differential Pressure Load) + OBE	639	1800	19,590	24,000
3. Safe Shut-Down				
Operating Load + DBA (Maximum Internal Differential Pressure Load) + DBE	644	2700	21,000	36,000

Table 9.9-4
Steam Generator Vault Maximum Stress and Stress Limits

Load Combination	Design Stress Limits for Reinforced Concrete: Design Stress (psi)	Design Stress Limits for Reinforced Concrete: Stress Limit (psi)	Design Stress Limits For Reinforcing Steel: Design Stress (psi)	Design Stress Limits For Reinforcing Steel: Stress Limit (psi)
4. Pipe Rupture Reaction Loads and Jet Loads (Local effect)				
a. Operating Load + DBA (Pipe Rupture Reaction Load & Internal Differential Pressure Load) (27.4 psi) + DBE	1267	3000	49,475	54,000
b. Operating Load + DBA (Pipe Rupture Reaction Load & Internal Differential Pressure Load) (50 psi) + DBE	1250	3000	51,550	54,000
	1320	4000	57,480	60,000
c. Operating Load + DBA (Pipe Rupture Reaction Load & Internal Differential Pressure Load) (75 psi) + DBE				

Table 9.9-5
Reactor Vessel Cavity
Maximum Stress And Stress Limits

	Design Stress Limits for Reinforced Concrete		Design Stress Limits for Reinforcing Steel	
	Design Stress (psi)	Stress Limit (psi)	Design Stress (psi)	Stress Limit (psi)
1. Normal Operating Loads (Dead Loads + Thermal)	389	1800	6150	24,000
2. Dead Loads + Thermal + OBE + Differential Pressure (100 psi)	479	1800	12,351	24,000
3. Dead Loads + Thermal + DBE + Maximum Differential Pressure (185 psi)	569	2700	22,800	36,000
4. Dead Loads + Thermal + DBE + Maximum Hypothetical Differential Pressure (480 psi)	569	4000	59,284	60,000

Table 9.9-6
Nozzle Cavity Maximum Stress and Stress Limits

	Design Stress (psi)	Stress Limit (psi)
1. Differential Pressure = 475 psi	10,133	20,000
2. Differential Pressure = 1400 psi	29,870	45,000

Table 9.9-7
Design Stress Limits for Structural Equipment Supports

Design Condition Categories	Design Stress Limits (1) Steel	Concrete
1. Normal Conditions:		
Operating Loads	0.6 Fy	0.45 f'c
2. Upset Condition:		
Operating Loads + OBE	0.6 Fy	0.45 f'c
3. Emergency Condition:		
Operating Loads + DBE	0.9 Fy	0.675 f'c
4. Faulted Condition:		
a. Operating Loads + DBA (2)	0.9 Fy	0.675 f'c
(1) Duration: > 0.01 Sec	0.95 Fy or 0.8 F_u	0.75 f'c
(2) Duration: ≤ 0.01 Sec	Fy or 0.85 Fu	0.75 f'c
b. Operating Loads + DBA + DBE	1.1 Fy or 0.9 F_u	0.85 f'c
(1) Duration: > 0.01 Sec		
(2) Duration: ≤ 0.01 Sec		

NOTES:

1. Fy = Minimum Yield Stress

   F_u = Ultimate Tensile Stress

   f'_c = Minimum 28-Day Compressive Strength of Concrete

2. DBA Load Includes:

   a. Pipe Rupture Reaction Loads

   b. LOCA Temperature Transients Through Steel

Table 9.9-8
Design Stress Limits for Biological Shield Structures

Structural System and Design Condition Categories	Design Stress Limits for Reinforced Concrete		Design Stress Limits for Structure Steel	
	Criteria	Stress Limits	Criteria	Stress Limits
1. Shield Structure Below Reactor				
Vessel Supports				
a. Normal Operation: Operating Loads(1)	ACI-318 (WSD)	Code Allowable	AISC (Part 1)	Code Allowable
b. Design Basis Accident: Operating Load + DBA + OBE	ACI-318 (WSD)	Code Allowable	AISC (Part 1)	Code Allowable
c. Safe Shutdown: Operating Load + DBA + DBE	ACI-318 (WSD)	1.5 x (Code Allowable)	AISC (Part 1)	1.5 x (Code Allowable)

Table 9.9-8
Design Stress Limits for Biological Shield Structures

Structural System and Design Condition Categories	Design Stress Limits for Reinforced Concrete		Design Stress Limits for Structure Steel	
	Criteria	Stress Limits	Criteria	Stress Limits
2. Shield Structure Above Reactor				
Vessel Supports				
a. Normal Operation: Operating Loads[1]	ACI-318 (WSD)	Code Allowable	AISC (Part 1)	Code Allowable
b. Design Basis Accident: Operating Load + DBA + OBE	ACI-318 (WSD)	0.75 f'_c (concrete) 0.9 Y.S. (reinforcement)	AISC (Part 1)	0.9 Y.S.
c. Safe Shutdown: Operating Load + DBA + DBE	ACI-318 (WSD)	0.75 f'_c (concrete) 0.9 Y.S. (reinforcement)	AISC (Part 1)	0.9 Y.S.

NOTES:

1. Operating load includes: (a) dead loads, (b) live loads, (c) neutron absorption, (d) reactor vessel operating weight, and (e) residual construction loads.
2. DBA load includes: (a) internal pressure, (b) pipe rupture reaction loads, and (c) safety injection water load.

Table 9.9-9
Allowable and Critical Stresses in Reactor Building Interior Structures

		Design Stress Limits (psi)		Actual Maximum Stresses (psi)	
Structural Stress and Design Condition Categories	Loading Criteria	f'_c	f_s	f'_c	f_s
1. Equipment Compartments					
a. Normal Operation	Operating Loads	1800	24000	200	5900
b. Design Basis Accident	Operating Loads + DBA + OBE	1800	24000	610	24000
c. Safe Shutdown	Operating Loads + DBA + DBE	2700	36000	2200	3200
d. Pipe Rupture Reaction Loads and Jet Loads (Local Effect)	1. Operating Loads + DBA (Maximum Pipe Rupture Reaction Load) + DBE	3000	54000	2260	48430
	2. Operating Loads + DBA (Pipe Rupture Reaction Load and Internal Differential Pressure Load) + DBE	3000	54000	2890	53200
2. Structural Equipment Supports					
a. Normal Condition	Operating Loads	1500	30000*	200	5850*
b. Upset Condition	Operating Loads + OBE	1800	30000*	500	19700*
c. Emergency Condition	Operating Loads + DBE	2700	45000*	2190	24700*
d. Faulted Condition	Operating Loads + DBA:				
	1. Duration: >0.01 Sec	2700	45000*	(1)	(1)
	2. Duration: ≤ 0.01 Sec	3000	47500*	(1)	(1)
	Operating Loads + DBA + DBE:	3000	56000*		
	1. Duration: > 0.01 Sec	3400	56000*	2630	56000*
	2. Duration: ≤ 0.01 Sec				
3. Biological Shield					
Structure Below Reactor Vessel Supports					
a. Normal Operation	Operating Loads	1800	24000	420	20900
b. Design Basis Accident	Operating Loads + DBA + OBE	1800	24000	500	21500
c. Safe Shutdown	Operating Loads + DBA + DBE	2700	36000	500	21500
Structure Above Reactor Vessel Supports					
a. Normal Operation	Operating Loads	1800	24000	60	20900
b. Design Basis Accident	Operating Loads + DBA + OBE	3400	54000	70	25400
c. Safe Shutdown	Operating Loads + DBA + DBE	3400	54000	70	25400
4. Floor Systems					
Monolithic Concrete Beam and Slabs					
a. Normal Operation	Operating Loads	1800	24000	1800	22600
b. Design Basis Accident	Operating Loads + DBA + OBE	1800	24000	1800	23900
c. Safe Shutdown	Operating Loads + DBA + DBE	2700	36000	1900	25300
Composite Floors (Concrete Slab on Steel Framing) (All Design Condition Categories)	(1) Operating Loads + OBE	1800	24000	1570	22400
	(2) Operating Loads + DBE	2700	36000	1660	23800

Table 9.9-9
Allowable and Critical Stresses in Reactor Building Interior Structures

Structural Stress and Design Condition Categories	Loading Criteria	Design Stress Limits (psi) f'_c	Design Stress Limits (psi) f_s	Actual Maximum Stresses (psi) f'_c	Actual Maximum Stresses (psi) f_s
Steel Grating on Steel Framing (All Design Condition Categories)	(1) Operating Loads + OBE	NA	22000*	NA	25000*
	(2) Operating Loads + DBE	NA	33000*	NA	25000*

NOTES:
f_s (without asterisk) = Tensile Stress in Reinforcing Steel
f_s (*) = Tensile Stress in Structural Steel
(l) Stresses not calculated

Table 9.9-10
Maximum Stresses in the Containment Shell at the External Embedment Line

Calculated Stress (psi)

	Vertical and Horizontal Seismic Loads		Vessel Dead Loads Plus Internal Accident Pressure[3]	Total		Stress (psi)[2]	
Type of Stress	OBE	DBE	(41.4 psi)	OBE	DBE	OBE	DBE
Membrane:							
Meridian	423	846	7878	8301	8724	17500	20300
Circumferential	238	476	3594	3832	4070	17500	20300
Surface:							
Meridian	259	518	22438[1]	22697	22956	26250	30450
Circumferential	78	156	6052	6130	6208	26250	30450
Shear:							
Tangential	293	586	NA	293	586	NA	NA
Transverse	5.3	10.6	NA	5.3	10.6	NA	NA

(1) Accident Pressure (41.4 psi) ONLY
(2) See USAR Table 5.2-1
(3) Includes stresses caused by axial temperature gradient in embedment region.

Table 9.9-11
Steel and Concrete Stresses in Containment Vessel Bottom at Mezzanine Floor Line

Calculated Stress (psi)

	Horizontal Seismic Loads Only		Vessel Dead Loads Plus Internal Accident Pressure[3]	Total		Allowable Stress (psi)[2]	
Type of Stress	OBE	DBE	41.4 psi)	OBE	DBE	DBE	DBE
Average Vessel Plate Membrane Hoop Stress in 20 in. Wide Band	6330	12660	14330	20660	26990	26250	30450
Approximate Bearing Stress of 20 in. Wide Hoop Band Bearing Against Internal Concrete	15	30	NA	15	30	1500[2]	2250[2]

(1) See USAR Table 5.2-1.
(2) Maximum Allowable Bearing Stress on Concrete (see ACI 318-63)
(3) Includes stresses caused by axial temperature gradient in embedment region.

Table 9.9-12
Combined Stresses

	Top of Cold Spot	Bottom of Cold Spot	5.5 Feet Below Cold Spot	Allowable Stress Intensity
With Operating Basis Earthquake				
Membrane Stresses Only – Including Local Membrane (No Temperature Stresses)				
Meridional	10.00 ksi	12.61 ksi	14.99 ksi	
Circumferential	17.45	17.43	9.24	
Max. Stress Intensity	17.50	18.99*	16.00	17.50 ksi
Membrane and Bending Stresses (No Temperature Stresses)				
Meridional	11.27 ksi	13.00 ksi	22.19 ksi	
Circumferential	21.19	18.37	11.40	
Max. Stress Intensity	21.23	19.892	22.79	26.25 ksi
All Stresses, Including Temperature				
Meridional	16.50 ksi	14.22 ksi	23.60 ksi	
Circumferential	27.85	24.72	11.82	
Max. Stress Intensity	27.86	25.34	24.09	52.50 ksi
With Design Basis Earthquake				
Membrane Stresses Only - Including Local Membrane (No Temperature Stresses)				
Meridional	10.12 ksi	12.87 ksi	15.14 ksi	
Circumferential	17.45	17.43	9.24	
Max. Stress Intensity	17.50	19.04	16.14	20.30 ksi
Membrane and Bending Stresses (No Temperature Stresses)				
Meridional	11.39 ksi	13.26 ksi	22.34 ksi	
Circumferential	21.19	18.37	11.40	
Max. Stress Intensity	21.23	19.87	22.94	30.45 ksi
All Stresses, Including Temperature				

Table 9.9-12
Combined Stresses

	Top of Cold Spot	Bottom of Cold Spot	5.5 Feet Below Cold Spot	Allowable Stress Intensity
Meridional	16.62 ksi	14.48 ksi	23.75 ksi	
Circumferential	27.85	24.72	11.82	
Max. Stress Intensity	27.86	25.35	24.23	52.50 ksi

* The maximum membrane stress intensity is 20530 psi and occurs at 6 in. below the bottom of this "cold spot". Because this point is not within 2.5 $\sqrt{Rt}$, of another point where the membrane stress intensity exceeds 1.1 S_m or 19250 psi, and since the height of the area stressed above 1.1 S_m does not exceed 0.5 $\sqrt{Rt}$, this area is classified as a local stress region with an allowable membrane stress intensity of 1.5 S_m or 26250 psi in accordance with Paragraph N412(j) of Section III of the ASME Boiler and Pressure Vessel Code.

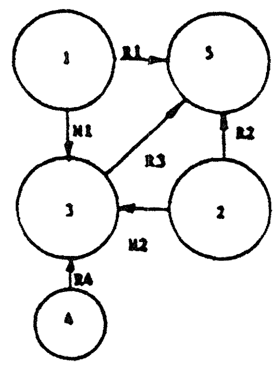
Table 9.9-13

Assumptions and Results of Calculation of Wall Temperatures and Total Stresses

Wall density	144 lbm/ft^3
Conductivity	0.73 Btu/hr ft°F
Heat capacity	0.2 Btu/lbm °F
Containment temperatures	output of the revised CONTEMPT code with condensate heat transfer rate 25% higher than that of Tagami's
Annulus air temperatures	output of the Shield Building Computer Code with best estimated heat transfer rates
Compression stress for concrete:	
Allowable	1.8 ksi
Evaluated*	1.51 ksi
Tension stress for steel:	
Allowable	24 ksi
Evaluated*	22.42 ksi

* Including all stresses due to thermal, dead load, live load, design earthquake, membrane shear, and out-of-plumb effects.

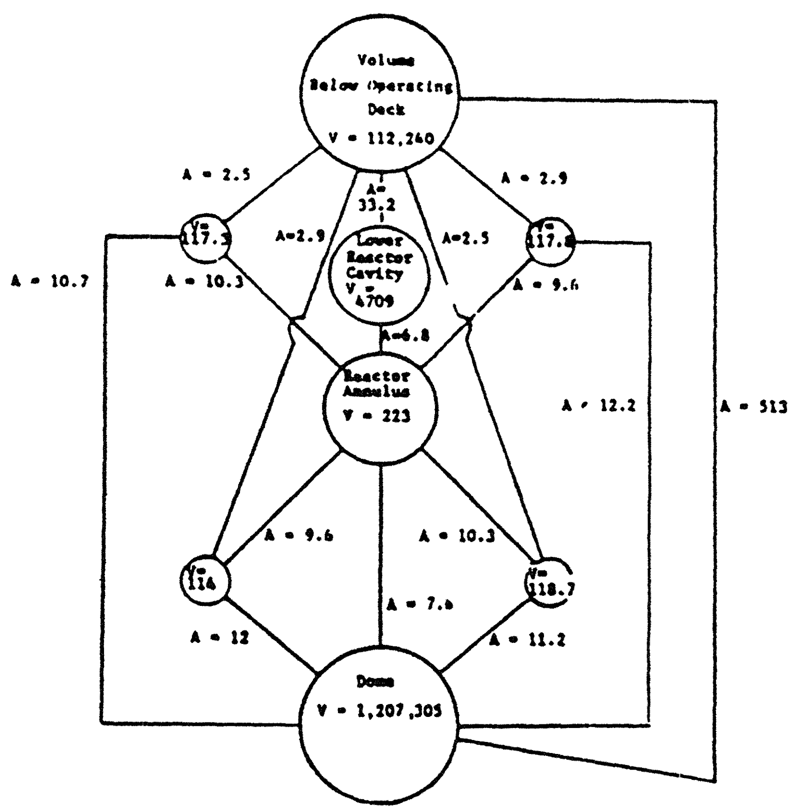
Figure 9.9-1 Steam Generator Vault Compartment Differential Pressure Analysis



Compartment	Volume (ft^3)
1 (Steam generator vault)	21,740
2 (Steam generator vault)	24,500
3 (Base compartment)	66,000
4 (Foundation void)	23,935
5 (Bulk containment volume)	1,184,000

Flow path	Minimum flow area (ft^2)
R1	360
R2	394
R3	174
R4	12.6
M1	111.3
M2	106.0

Figure 9.9-2 Volume and Flow Path Schematic of Reactor Cavity Region

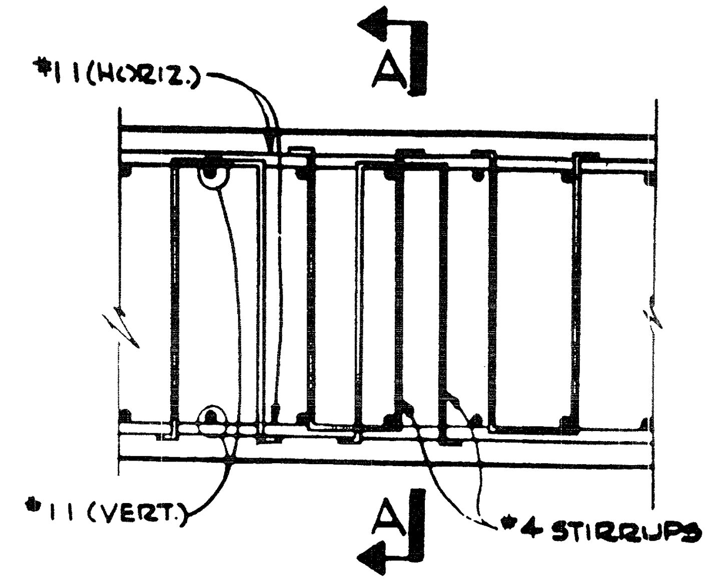


Unnamed Compartments are Pipe Annuli

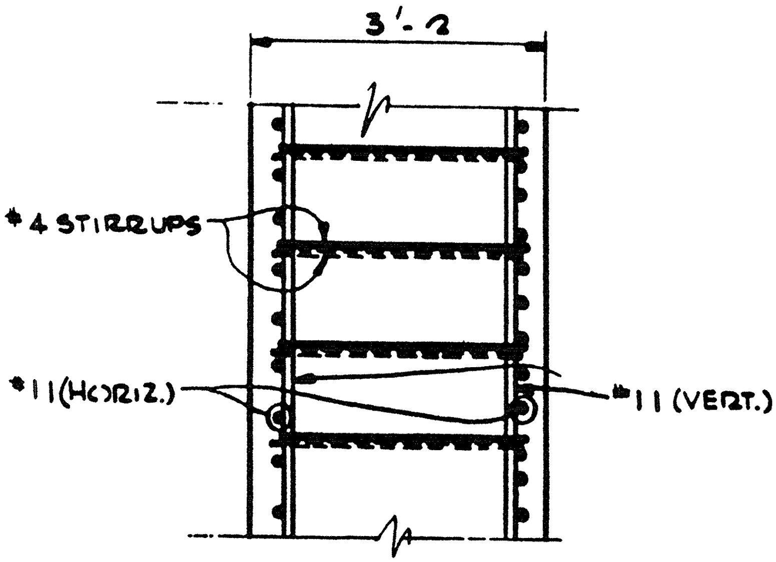
V = Volume ft^3

A = Area ft^2

Figure 9.9-3 Plan View and Section of Wall Segment Showing Stirrup



TYP. STIRRUP DETAIL IN WALLS



SECTION A-A

Figure 9.9-4 Plan View or Structural Model of Steam Generator Compartment (Height 68 Ft.)

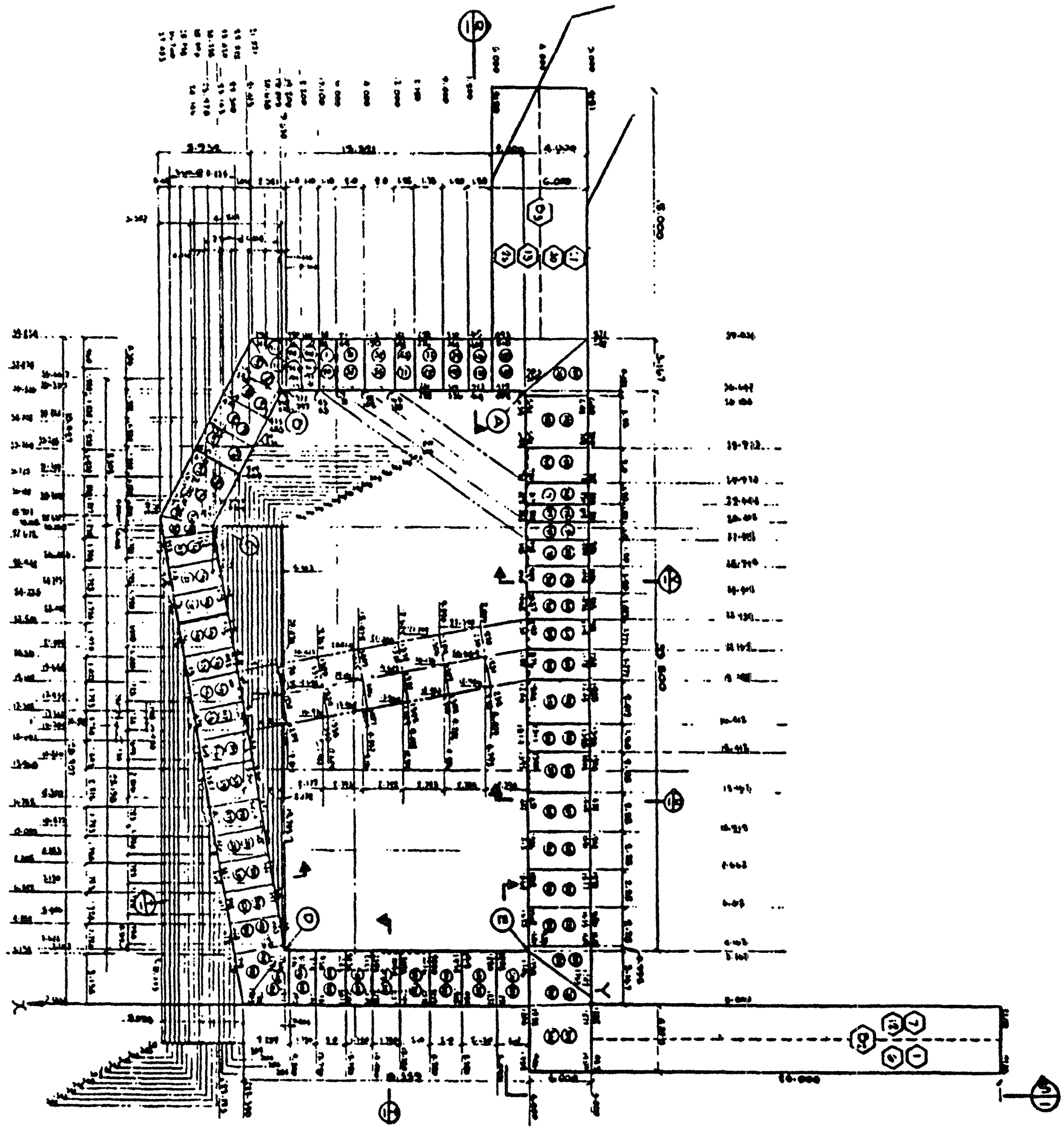


Figure 9.9-5 Steam Generator and Reactor Cavity Compartment Configuration - Section

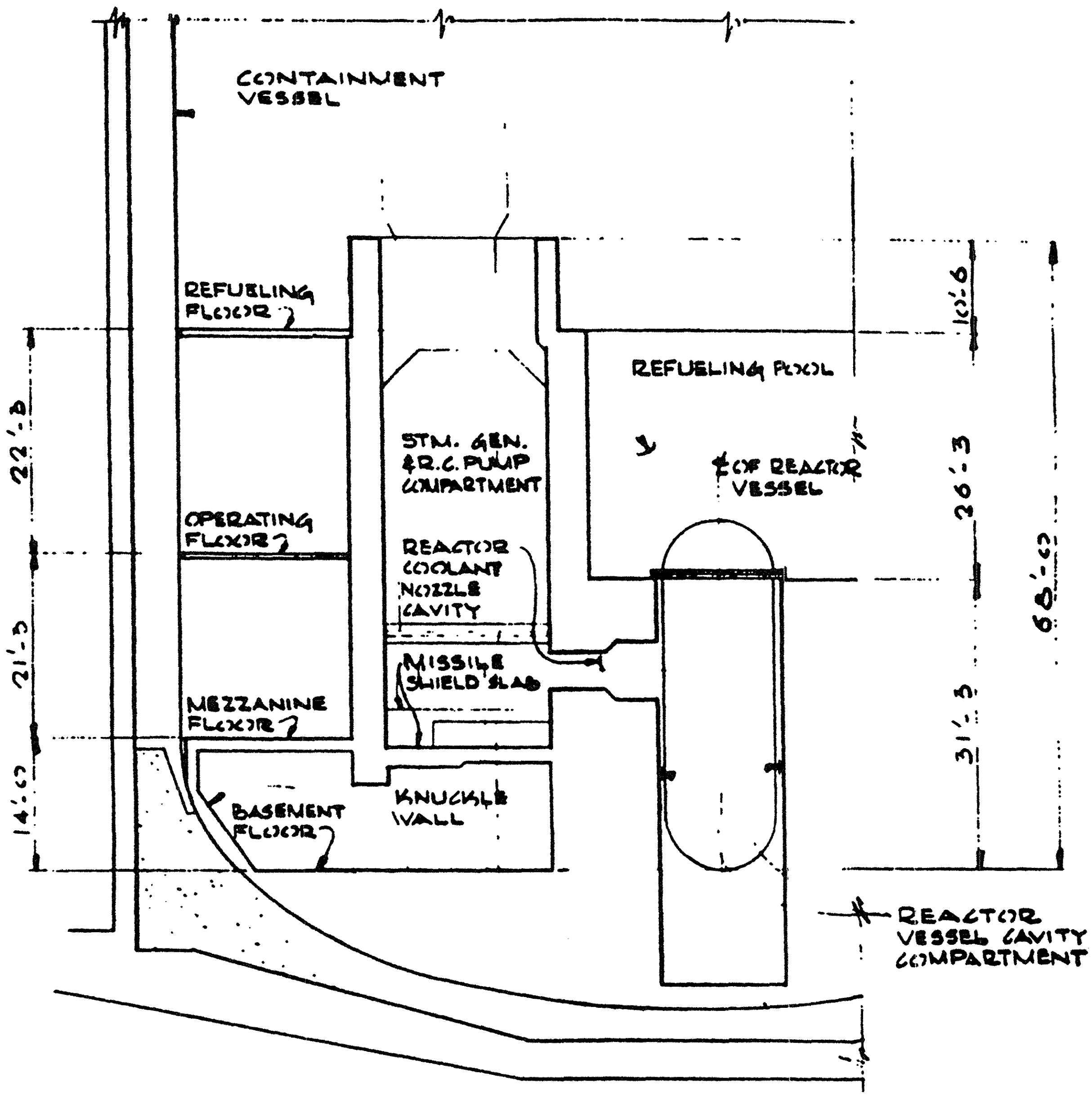
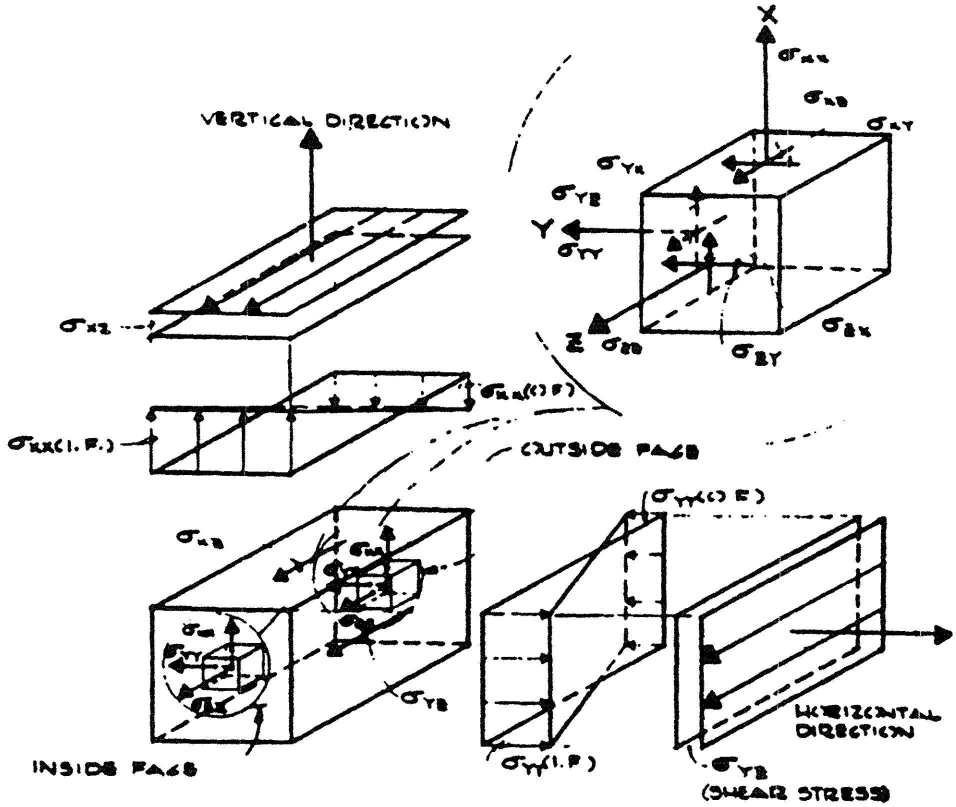
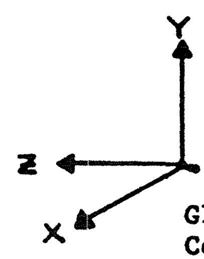


Figure 9.9-6 Stress Output for Three-Dimensional Brick Element in Local Coordinate System



NOTES:

1. Inside Face: Facing Steam Generator and Reactor Coolant Pump.
2. Outside Face: Opposite of Inside Face of Wall.
3. σ_{xx}, σ_{yy}, σ_{zz}: Positive for Tension and Negative for Compression.



Global Coordinate System

Figure 9.9-7 Reactor Vessel Supports

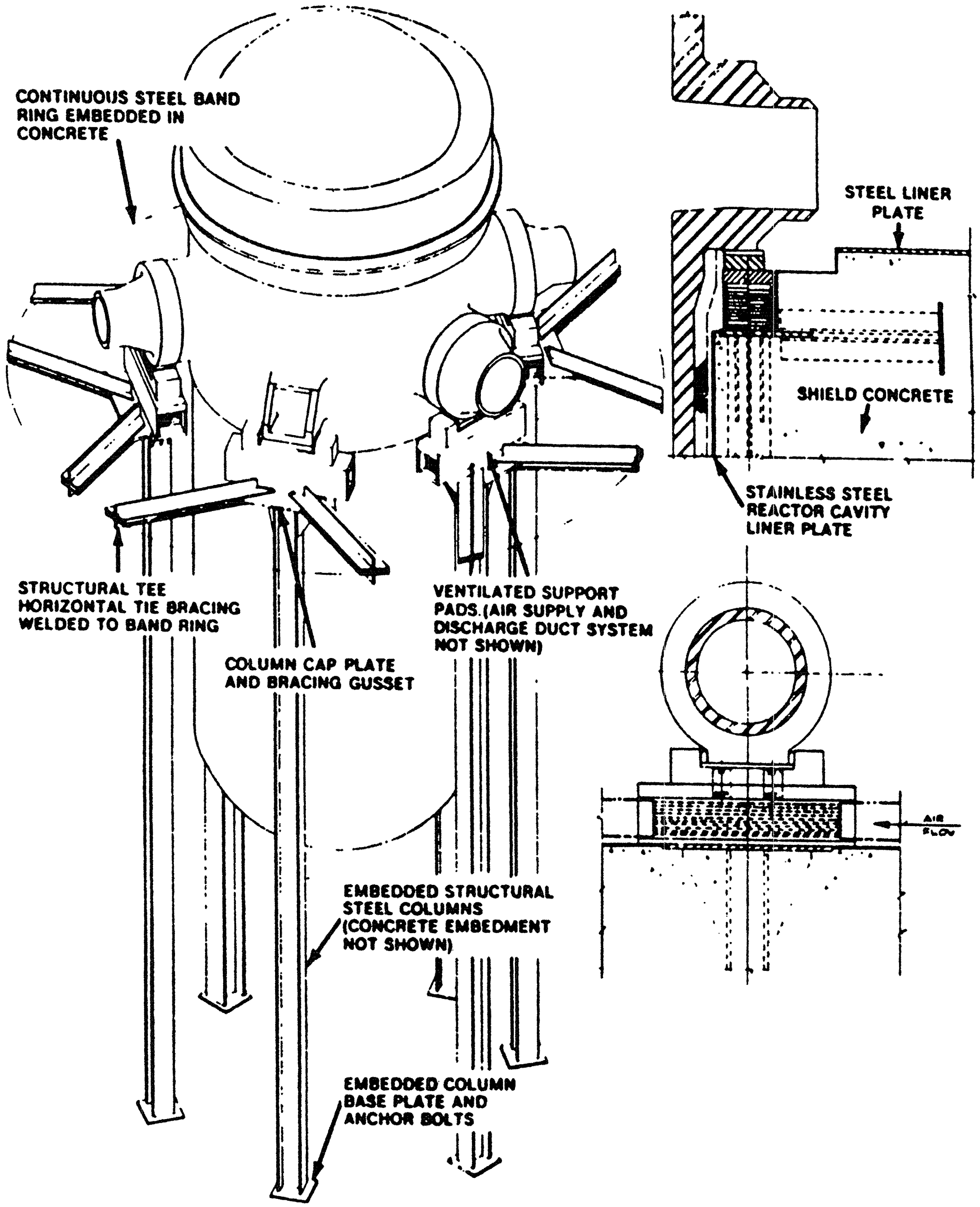


Figure 9.9-8 Steam Generator Supports