SECTION 9

AUXILIARY AND EMERGENCY SYSTEM

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SECTION 9
AUXILIARY AND EMERGENCY SYSTEM

9.1 CHEMICAL AND VOLUME CONTROL SYSTEM

The chemical and volume control system (CVCS) is designed to:

1. Adjust the concentration of chemical neutron absorber in the reactor coolant for reactivity control
2. Maintain the proper water inventory in the reactor coolant system (RCS)
3. Provide the required seal water flow for the reactor coolant pump shaft seals
4. Provide high pressure flow to the emergency core cooling system (ECCS)
5. Maintain the proper concentration of corrosion inhibiting chemicals in the reactor coolant, and
6. Reduce the coolant inventory of corrosion products and fission products.

During the normal operation, this system also supplies:

1. Hydrogen to the volume control tank and, when desired, to the letdown line upstream of the volume control tank (VCT) via a hydrogen sparger
2. Nitrogen as required for purging the volume control tank
3. Hydrazine and lithium hydroxide as required via the chemical mixing tank to the charging pumps suction.

9.1.1 Design Bases

9.1.1.1 Redundancy of Reactivity Control

In addition to the reactivity control achieved by the rod cluster control assemblies (RCCA) as detailed in Section 7 and Section 3, reactivity control is provided by the CVCS which regulates the concentration of boric acid solution neutron absorber in the RCS. The system is designed to prevent, for various malfunctions, uncontrolled or inadvertent reactivity changes which might cause operation in excess of design limits.

9.1.1.2 Reactivity Shutdown Capability

Normal reactivity shutdown capability is provided by control rods with boric acid injection used to compensate for the long term xenon decay transient and for unit cooldown. Any time that the unit is at power, the total quantity of boric acid retained in the boric acid tanks and ready for injection always exceeds that quantity required for the normal cold shutdown. This quantity always exceeds the quantity of boric acid required to bring the reactor to hot shutdown and to compensate for subsequent xenon decay.

The boric acid solution is transferred from the boric acid tank by boric acid transfer pumps to the suction of the charging pumps which inject boric acid into the reactor coolant. Any charging pump and any boric acid transfer pump can be operated from diesel generator power on loss of offsite power.

Boric acid can be injected by one pump at a reactivity insertion rate which shuts the reactor down with no rods inserted in 90 minutes. In 90 additional minutes, enough boric acid can be injected to compensate for xenon decay, although xenon decay below the full power equilibrium operating level will not begin until approximately 25 hr after shutdown. If two boric acid transfer pumps are available, these time periods are reduced. Additional boric acid is employed if it is desired to bring the reactor to cold shutdown conditions.

On the basis of the above, the injection of boric acid is shown to afford backup shutdown reactivity capability, independent of control rod clusters which normally serve this function in the short term situation. Shutdown for long term and reduced temperature conditions can be accomplished with boric acid injection using redundant components.

The reactivity control systems provided are capable of making and holding the core subcritical for any cold shutdown, hot shutdown, or hot operating condition. The maximum excess reactivity expected for the core that must be controlled by the control rods and soluble neutron absorber (boron) occurs for the cold, clean condition at the beginning of life of the initial core. The RCC assemblies are divided into control and shutdown groups.

The control group, used in combination with chemical shim, provides control of the reactivity changes of the core throughout the life of the core at power conditions. This group of RCC assemblies is used to compensate for short term reactivity changes at power that might be 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 and fission product buildup and decay.

Manually initiated boric acid addition is used to maintain the shutdown margin for the long term conditions of xenon decay and station cooldown. After initiation, the boration sequence is automatically carried through as described in Section 9.1.2.2.

9.1.1.3 Reactor Coolant Purification Capability

The CVCS is capable of reducing the concentration of ionic isotopes in the purification stream. This is accomplished by passing the letdown flow through the mixed bed demineralizers which remove ionic isotopes except those of cesium, molybdenum and yttrium with a minimum decontamination factor of 10. Through occasional use of the cation bed demineralizer, the concentration of cesium-137 can be maintained below 1.0 μCi per cc, assuming 1 percent defective fuel. The cation bed demineralizer is capable of passing the normal letdown flow. Each mixed bed demineralizer is capable of processing the maximum letdown flow rate. If the normally operating mixed bed demineralizer resin has become exhausted, the second demineralizer can be placed in service. Each demineralizer is designed, however, to operate for approximately one core cycle with 1 percent defective fuel. Filters are provided at various locations to ensure removal of particulates and resin fines.

9.1.1.4 Codes and Classifications

System integrity is ensured by equipment design and construction conformance to applicable codes listed in Table 9.1-1, and by the use of austenitic stainless steel or other corrosion resistant materials in contact with both reactor coolant and boric acid solutions.

The code requirements for the regenerative heat exchanger and the excess letdown heat exchanger are indicated in Table 9.1-1. Pipe classifications conform to the requirements of Section 6.2.

The ASME Code, Section III, Class C, designation is based on:

1. Both heat exchangers can be double isolated from the reactor coolant system

2. Both heat exchangers are protected by a missile barrier (see Section 5.2.6 for a discussion of the postulated credible missiles which these heat exchangers are protected against)

3. Both heat exchangers are located inside the containment. Accordingly, the designation of "Class C" for these heat exchangers is justifiable and does not lead to any public hazard.

9.1.1.5 Chemical Additions

The CVCS provides a means for adding chemicals to the RCS which control the pH of the coolant during initial startup and subsequent operation, scavenge oxygen from the coolant during startup, and control the oxygen level of the reactor coolant due to radiolysis during all operations subsequent to startup.

9.1.1.6 Regulation of Reactor Coolant Inventory

The CVCS maintains the proper coolant inventory in the RCS for all normal modes of operation including startup from cold shutdown, full power operation, and plant cooldown. This system also has sufficient makeup capacity to maintain the minimum required inventory in the event of minor RCS leaks.

The CVCS flow rate is based on the requirement that it permit the RCS to be heated to or cooled from hot standby condition at the design rate and maintain proper coolant level.

9.1.2 System Design and Operation

The CVCS is shown in Figures 9.1-1 and 9.1-2 and the system performance requirements are given in Table 9.1-2.

9.1.2.1 System Description

Seal water inleakage to the reactor coolant system via the reactor coolant pump controlled leakage seal 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.

During unit operation, reactor coolant flows through the letdown line from the pump discharge side of one of the reactor coolant loop cold legs and, after processing, is returned through the charging line to the pump discharge side of the cold leg of another loop. An excess letdown line is also provided for removing coolant from the RCS in case letdown through the normal letdown path is unavailable or additional letdown is desired.

Each of the connections to the RCS has an isolation valve located close to the loop piping. The charging line has a check valve located downstream of the charging line isolation valves. 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 a letdown orifice which reduces the coolant pressure. The cooled low pressure water leaves the containment and enters the auxiliary building where it undergoes a second temperature reduction in the tube side of the nonregenerative heat exchanger followed by a second pressure reduction by the low pressure letdown valve. The coolant then passes through one of the mixed bed demineralizers where ionic impurities are removed. The cation bed demineralizer, located downstream of the mixed bed demineralizer, is used intermittently to control cesium activity in the coolant and also to remove excess lithium which is formed from the decay of B^{10}.

The coolant then flows through the reactor coolant filter and enters the volume control tank through a spray nozzle. Hydrogen is manually added to the vapor space in the volume control tank, which is predominantly hydrogen and water vapor, or directly to the letdown line upstream of the VCT via a hydrogen sparger. The hydrogen is supplied to the reactor coolant for the purpose of maintaining a low free oxygen concentration. Fission gases are removed from the system by diverting the letdown stream to the degasifier in the boron recovery system and also by venting the volume control tank to the gaseous waste system via the Boron Recovery System Degasifiers. Reactor coolant letdown flow is directed to the boron recovery system on a high level signal from the volume control tank. The capabilities also exist for continuous degasification of the reactor coolant letdown. When operating in this mode all or a portion of the letdown flow is diverted to the degasifiers of the boron recovery system with subsequent return to the VCT. From the volume control tank, the coolant flows to the charging pumps which raise the pressure above that in the RCS. The charging flow then enters the containment, passes through the tube side of the regenerative heat exchanger and returns to the RCS.

A portion of the high pressure charging flow is not returned through the regenerative heat exchanger, but instead is diverted through one of the seal injection filters and injected into the reactor coolant pumps. The injection flow (approximately 8 gpm per pump) enters each pump in the thermal barrier region where the flow splits with a portion (approximately 3 gpm) passing through the pump radial bearing, the controlled leakage seal package and returning to the charging pump suction. The remaining portion (approximately 5 gpm) passes through the thermal barrier heat exchanger and into the reactor coolant system where it constitutes a portion of the RCS makeup.

Boric acid is dissolved in the batching tank to a concentration of approximately 4 percent by weight. The batching tank is jacketed to permit heating of the solution with low pressure steam. The boric acid solution is transferred directly to the Evaporator Bottoms Tank by gravity prior to being transferred to the Boric Acid Hold Tank. Boric acid solution is metered from the discharge of an operating boric acid transfer pump for blending with primary grade water as makeup for normal leakage, or for increasing the reactor coolant boron concentration during normal load follow operation.

Excess reactor coolant inventory results from unit startup, load follow operations, fuel burnup, and unit shutdown. The excess coolant flows from the RCS through the letdown line and is directed to the boron recovery system where the liquid is processed to recover the boric acid and primary grade water.

During unit cooldown, prior to removing the reactor vessel head, when the residual heat removal system 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 nonregenerative heat exchanger, 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.

9.1.2.2 Reactor Coolant Makeup Control Modes

Makeup for unit leakage is regulated by the reactor makeup control which is normally set to blend water from one of the primary water storage tanks with concentrated boric acid solution to match the reactor coolant boron concentration. Makeup is added automatically to the volume control tank if the tank level falls below a preset point; the operator can also operate the makeup system manually.

The makeup system also provides concentrated boric acid or primary grade water to either increase or decrease the boric acid concentration in the reactor coolant system. To maintain the reactor coolant liquid inventory, for a given power reactor coolant at existing reactor coolant boric acid concentration is letdown to the boron recovery system.

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

1. The primary water storage 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, which supplies borated water for emergency makeup.

4. The chemical mixing tank, which is used to inject small quantities of solution when additions of chemicals for oxygen scavenging or pH control are necessary.

The reactor makeup control is designed to operate from the main control room by manually preselecting makeup composition to the charging pump suction header or the volume control tank in order to maintain the desired operating fluid inventory in the volume control tank and to adjust the reactor coolant boron concentration for reactivity control. The operator can stop the makeup operation at any time in any operating mode by either actuating the makeup stop switch or remotely closing the makeup stop valves.

Automatic Makeup

The "Automatic Makeup" mode of operation of the reactor makeup control provides dilute boric acid solution, preset 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. It operates on demand signals from the volume control tank level controller.

Under normal station operating conditions, the mode selector switch is set in the "AUTO" position and the boric acid and total makeup water flow controllers are set to give the same concentration of borated water as contained in the RCS. The mode selector switch must be in the correct position and the control energized by prior manipulation of the "Start" switch. A preset low level signal from the volume control tank level controller causes the automatic makeup control action to increase the speed on the normally running boric acid transfer pump to fast, open the makeup stop valve to the charging pump suction header, and open the boric acid flow control valve and the primary grade water flow control valve. The blend system is set up to provide a constant total (boric acid plus primary grade water) makeup flow rate. The total makeup flow rate is automatically controlled to a preset value by an orifice flowmeter downstream of the blender. The boric acid flow rate is controlled to an operator-selected set point by a magnetic flowmeter controller in the boric acid line. The total makeup flow controller controls the primary makeup water flow control valve to provide the difference between the boric acid flow and preset total flow rate thus controlling the makeup 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 point, the speed of the boric acid transfer pump is decreased to slow, the primary grade water and boric acid flow control valves close and the makeup stop valve closes.

Dilution

The "Dilute" mode of operation permits the addition of a preselected quantity of primary grade water at a preselected flow rate to the RCS. The operator sets the mode selector switch to "Dilute", the total makeup water flow controller setpoint to the desired flow rate, the total makeup water batch integrator to the desired quantity and actuates the makeup start. The start signal causes the makeup control action to open the primary grade water flow control valve, and open the makeup stop valve to the volume control tank inlet. The primary grade water is injected through the volume control tank spray nozzle and through the tank to the charging pump suction header. Excessive rise of the volume control tank water level is prevented by automatic actuation of a three-way diversion valve (by the tank level controller), which diverts the reactor coolant letdown flow to the boron recovery system. When the preset quantity of primary grade water has been added, the batch integrator causes the primary grade water flow control valve and the makeup stop valve to the volume control tank inlet to close.

Alternate Dilution

The "Alternate Dilute" mode is similar to the dilute mode except a portion of the dilution water 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 total makeup water flow controller setpoint to the desired flow rate, the total makeup water batch integrator to the desired quantity and actuates the makeup start. The start signal causes the makeup control action to open the makeup stop valve to the volume control tank, the makeup stop valve to the charging pump suction header and the primary grade water flow control valve. Primary 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 of reactor coolant to follow the initial Xenon transient and simultaneously to 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 diverts the reactor coolant letdown flow to the boron recovery system. When the preset quantity of primary grade water has been added, the batch integrator causes the primary grade water flow control valve and the makeup stop valves to close.

Boration

The "Borate" mode of operation permits the addition of a preselected quantity of concentrated boric acid solution at a preselected flow rate to the RCS. The operator sets the mode selector switch to "Borate", the concentrated boric acid flow controller setpoint to the desired flow rate, the concentrated boric acid batch integrator to the desired quantity and actuates the makeup start. Actuating the start switch opens the makeup stop valve to the charging pump suction header and the boric acid flow control valve and increases the speed of the normally operating boric acid transfer pump to fast. The concentrated boric acid is added to the charging pump suction header. The total quantity added in most cases will be small and will have only a minor effect on the volume control tank level. When the preset quantity of concentrated boric acid solution has been added, the batch integrator causes the boric acid transfer pump to return to slow speed and the concentrated boric acid flow control valve and the makeup stop valve to close.

All modes of operation must be manually selected by the operator and system start initiated. In the case of automatic makeup, initiation of system start sets the system up for automatic actuation in response to a "low" level in the volume control tank.

There is no other automatic actuation of the system (i.e., actuation requiring no operator action) except when derived from a volume control tank low-low level signal. In this event, the suction of the charging pump(s) is automatically aligned to take suction from the refueling water storage tank which contains boron at a minimum concentration defined within the plant technical specifications and Licensing Requirements Manual.

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 9.1-2. One set of values is given for the addition of boric acid from a boric acid tank with one transfer pump and one charging pump operating. The other set assumes 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.

Alarm Functions

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

1. Deviation of total makeup flow rate from control setpoint.

2. Deviation of concentrated boric acid flow rate from control setpoint.

3. High level in the volume control tank - this alarm indicates that the level in the tank is approaching high level and a resulting 100 percent diversion of the letdown stream to the boron recovery system.

4. Low level in the volume control tank - this is alarmed when the reactor makeup control selector is not set for the automatic makeup control mode and the level drops below the point where automatic makeup should be initiated. This alarm indicates that the level in the tank is approaching low-low level and results in realignment of charging pump(s) suction to the refueling water storage tank.

5. High temperature on inlet to the demineralizers - If letdown temperature increases to the alarm setpoint, the letdown flow is diverted around the demineralizers.

9.1.2.3 Charging Flow Control

Three single speed horizontal centrifugal charging pumps are used to supply charging flow to the RCS and perform the safety injection function as discussed in Section 6.3.

A flow transmitter on the charging line upstream of the regenerative heat exchanger transmits a signal to an indicator-controller in the control room. The charging flow controller regulates a throttling valve in the charging line to maintain a preset charging flow. An RCS pressurizer water level error signal resets the charging flow setpoint to take corrective action. The charging flow controller is provided with adjustable maximum and minimum limits which are preset and fixed on initial calibration of the station. The maximum limit of the charging flow comparator actuates an alarm indicating that a maximum flow condition exists at the charging flow control valve and that a second charging pump should be brought on the line. The minimum limit of the charging flow comparator also initiates an alarm and provides an electronic minimum flow stop on the charging flow control valve to preclude flashing downstream of the letdown orifices. The number of charging pumps required to be operable/functional for each reactor operating mode is stated in the Technical Specifications and Licensing Requirements Manual.

The response of the charging line throttling valve to changes in the flow control signal is maintained slow to reduce charging flow fluctuations due to short-term pressurizer level transients. If the pressurizer level increases, the error signal causes the control valve to throttle closed; conversely if the levels decrease, the valve is throttled open.

A pressure switch in the charging pump discharge header actuates an alarm if the discharge header pressure falls to a preset low level.

9.1.2.4 Components

A summary of principal component data is given in Table 9.1-3.

Regenerative Heat Exchanger

The regenerative heat exchanger is designed to recover heat from the letdown flow by reheating the charging flow which reduces thermal shock on the charging penetrations into the reactor coolant loop piping.

The letdown stream flows through the shell of the regenerative heat exchanger and the charging stream flows through the tubes.

The unit is of all welded austenitic stainless steel construction.

Letdown Orifices

The three letdown orifices control flow of the letdown stream and reduce the pressure to a value compatible with the nonregenerative heat exchanger design. Two of the orifices are designed to provide normal letdown flow. Each of these orifices is rated at 60 gpm at the normal operating pressure of 2,235 psig in the RCS. One of the two orifices is normally in use while the second serves as a standby. Either of these two letdown orifices can be used in conjunction with the third orifice, which has a lower capacity (45 gpm), to attain maximum purification flow at normal RCS operating pressure. All three orifices may be used in parallel in order to increase letdown flow when the RCS pressure is below normal. This arrangement provides a full standby capacity for control of letdown flow. A normally open trip valve is provided at each letdown orifice. The valve is actuated by two solenoid operated valves that deenergize upon a CIA signal or by interlocks with low pressurizer level or closure of regenerative heat exchanger inlet isolation valves. Each orifice is an austenitic stainless steel pipe containing a bored corrosion and erosion resistant insert. The orifices are placed in and taken out of service by remote manual operation of their respective isolation valves.

Nonregenerative Heat Exchanger

The nonregenerative heat exchanger cools the letdown stream to the operating temperature (<140°F) 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. All surfaces in contact with the reactor coolant are austenitic stainless steel and the shell is carbon steel.

Mixed Bed Demineralizers

Two flushable mixed bed demineralizers maintain reactor coolant purity. A Li^7 cation resin and a hydroxyl form anion resin are initially charged into one of the demineralizers. Both forms of resin remove fission and corrosion products. The resin bed is designed to reduce the concentration of ionic isotopes in the purification stream except for cesium, yttrium and molybdenum, by a minimum factor of 10.

The other demineralizer is normally charged with mixed bed resin. This demineralizer may be charged with cation, anion, or Li^7 cation resin with hydroxyl form anion resin in order to provide flexibility in the support of normal plant operation, shutdown cleanup operations following an activity release, or for reactor coolant deboration purposes.

Each demineralizer is sized to accommodate the maximum letdown flow. One demineralizer serves as a standby unit for use should the operating demineralizer become exhausted during operation. Each demineralizer has sufficient capacity to enable operation for approximately one core cycle with one percent defective fuel cladding.

The demineralizer vessels are made of austenitic stainless steel and are provided with suitable connections to facilitate resin replacement when required.

<u>Cation Bed Demineralizer</u>

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 Li^7. The demineralizer also has sufficient capacity to maintain the Cesium-137 concentration in the coolant below 1.0 uCi per cc with 1 percent defective fuel. If this demineralizer is not available, other CVCS demineralizers can be used to maintain the Cesium-137 concentration in the coolant below 1.0 uCi per cc with 1 percent defective fuel.

Alternately, the demineralizer may be charged with mixed bed, anion, or Li^7 cation resin with hydroxyl form anion resin in order to provide flexibility in the support of normal plant operation, shutdown cleanup operations following an activity release, or for reactor coolant deboration purposes.

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

<u>Deborating Demineralizers</u>

Two deborating demineralizers are provided. The resin in one of the demineralizers is normally a hydroxyl form anion resin. This demineralizer will normally be used toward the end of core life. The capacity is consistent with removing boron to compensate for core burnup beyond about 90 percent of the core cycle.

Alternately, the demineralizers may be charged with mixed bed, cation, anion, or Li^7 cation resin with hydroxyl form anion resin in order to provide flexibility in the support of normal plant operation, shutdown cleanup operations following an activity release, or for reactor coolant deboration purposes. The demineralizer vessels are made of austenitic stainless steel and are equipped with connections for resin replacement when required.

<u>Resin Fill Tank</u>

The resin fill tank is mobile and 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 demineralizer water and resin slurry can be sluiced into the demineralizer by opening the dump valve. The tank is made of austenitic stainless steel.

Reactor Coolant Filter

The filter collects resin fines and particulates from the letdown stream. For filter performance characteristics see Table 9.1-3. The vessel is made of austenitic stainless steel and is provided with connections for draining and venting. The design flow capacity of the filter ensures that the maximum purification flow rate is accommodated.

Volume Control Tank

The volume control tank is an operating surge volume compensating in part for reactor coolant releases from the RCS. When the tank level controller reaches the top of its control band, the remainder of the fluid is diverted to the boron recovery system. The volume control tank also acts as a head tank for the charging pumps. The leakage from the reactor coolant pumps controlled leakage seals is directed to the tank. Overpressure of hydrogen gas is maintained in the volume control tank to control the hydrogen concentration in the reactor coolant in accordance with the BVPS Chemistry Manual.

A spray nozzle is located inside the tank on the inlet line 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 Gaseous Waste System permits removal of gaseous fission products which are stripped from the reactor coolant and collected in this tank. The tank is constructed of austenitic stainless steel.

Letdown Line Hydrogen Sparger

A hydrogen sparger is provided to ensure proper hydrogen concentration of the charging fluid leaving the VCT. The sparger is installed within an expanded portion of the letdown line. It is in the form of a bayonet with a solid bottom surface and a sintered stainless steel top surface. The sparger is fed with hydrogen (recycled or new) which forms small bubbles that dissolve in the hydrogen depleted degassed letdown fluid.

Charging Pumps

Three charging pumps are provided to inject coolant into the RCS. The pumps are of the single speed horizontal centrifugal type, and all parts in contact with the reactor coolant are fabricated of austenitic stainless steel or other material of adequate corrosion resistance. Pump leakage is piped to the drain header. There is a minimum flow recirculation line on the charging pump discharge header to protect the charging pumps against a closed discharge valve condition.

Chemical Mixing Tank

The primary use of the chemical mixing tank is in the preparation of LiOH solutions for pH control 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 hydrazine concentration in the reactor coolant by 10 ppm. This capacity is also more than sufficient to prepare a solution of LiOH for the RCS.

The chemical mixing tank is made of austenitic stainless steel.

Excess Letdown Heat Exchanger

The excess letdown heat exchanger can be employed either when normal letdown is temporarily out of service to maintain the station in operation or it can be used to supplement maximum letdown during the final stages of heatup.

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. Surfaces in contact with reactor coolant are austenitic stainless steel, and the shell is carbon steel. Tube joints are seal welded.

Seal Water Heat Exchanger

The seal water heat exchanger removes heat from three sources: reactor coolant pump seal water return flow; bypass flow from the charging pumps and reactor coolant discharged 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 seal welded to the tubesheet. Surfaces in contact with reactor coolant are austenitic stainless steel, and the shell is carbon steel.

The normal load on the seal water heat exchanger consists of the nominal reactor coolant pump seal water return flow and the bypass flow from a single charging pump. The heat exchanger is capable of cooling the additional flow from either a second charging pump bypass stream or the reactor coolant discharged from the excess letdown heat exchanger. With this additional flow, the discharge flow from the seal water heat exchanger is approximately the same temperature as that normally maintained in the volume control tank.

Seal Water Return Filter

The filter collects particulates from the reactor coolant pump seal water return and from the excess letdown heat exchanger flow. For filter performance characteristics see Table 9.1-3. The vessel is constructed of austenitic stainless steel and is provided with connections for draining and venting. Disposable synthetic filter elements are used.

Seal Water Injection Filters

These filters collect particulates from the reactor coolant pump seal water inlet. For filter performance characteristics see Table 9.1-3. Two filters are provided in parallel, each sized for the maximum anticipated injection flow rate. The vessels are constructed of austenitic stainless steel and are provided with connections for draining and venting.

Boric Acid Filter

The boric acid filter collects particulates from the boric acid solution being pumped to the charging pump suction line or boric acid blender. For filter performance characteristics see Table 9.1-3. The filter is designed to pass the design flow of two boric acid transfer pumps operating simultaneously. The vessel is constructed of austenitic stainless steel, and the filter elements are disposable cartridges. Provisions are available for venting and draining the filter.

Boric Acid Tanks

The boric acid tanks' combined capacity of nominal 4 weight percent boric acid solution, either recovered from the boron recovery system or mixed in the batching tank, is more than adequate to meet the design basis of having 11,336 gallons available for cold shutdown, even if the most reactive control rod is not inserted. The boric acid hold tank in the Boron Recovery System supplements the capacity of the boric acid tanks.

The concentration of boric acid solution in storage is maintained between 4 and 4.4 percent by weight. Periodic manual sampling and corrective action, if necessary, 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 absorber concentration. The tanks are constructed of austenitic stainless steel. Two electric immersion heaters are provided for each boric acid tank.

Batching Tank

The batching tank is used for mixing a makeup supply of boric acid solution for transfer to the boric acid tanks. The boric acid solution is transferred directly to the Evaporator Bottoms Tank by gravity prior to being transferred to the Boric Acid Hold Tank. The tank may also be used for solution storage.

A local sampling point can be used for verifying the solution concentration prior to transferring it out of the tank. The tank is provided with an agitator to improve mixing during batching operations and a steam jacket for heating the boric acid solution. The primary grade water supply line to the batching tank contains an electric water heater that can preheat the water used for preparing the boric acid solution when steam is not available. The tank is constructed of austenitic stainless steel.

Boric Acid Transfer Pumps

Two horizontal centrifugal, two speed pumps with mechanical seals are supplied. Normally one pump is aligned with one boric acid tank and runs continuously at slow speed to provide recirculation for the boric acid tank. The second pump aligned with the second boric acid tank is then considered as a standby pump, with service being transferred as operation requires. This second pump also intermittently circulates fluid through the second tank. Manual or automatic initiation of the reactor coolant makeup system will activate the running pump to the higher speed to provide normal makeup of boric acid solution as required. Emergency boration, the supplying of 4 weight percent boric acid solution directly to the suction of the charging pump, can be accomplished by manually starting either or both pumps in either fast or slow speed. The transfer pumps also transfer boric acid solution from the batching tank to the boric acid tanks.

Boric Acid Blender

The boric acid blender promotes thorough mixing of boric acid solution and primary grade water for the reactor coolant makeup circuit. The blender consists of a conventional pipe tee fitted with a perforated tube insert. The material is austenitic stainless steel.

Valves

Valves that perform a modulating function are equipped with two sets of packing. Globe valves are normally installed with flow under the seats; however, in isolated instances they are installed with flow over the seats when such an arrangement reduces the possibility of leakage. Whenever possible, the valves are installed in such a way that the high pressure side is away from the packing when the valve is closed.

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 a relief valve on the charging line.

Piping

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

Electrical Heat Tracing

Redundant electrical heat tracing powered from separate emergency buses, is installed under the insulation on all piping, valves, line-mounted instrumentation and components normally containing concentrated boric acid solution. The heat tracing is designed to prevent boric acid precipitation due to cooling, by compensating for heat loss.

Exceptions are:

1. Lines which may transport concentrated boric acid, but are subsequently flushed with reactor coolant or other liquid of low boric acid concentration during normal operation.

2. The boric acid tanks, which are provided with immersion heaters.

3. The batching tank, which is provided with a steam jacket.

4. Pumps are located in heated enclosures, the enclosures being fabricated of insulating material. Duplicate tracing on sections of the CVCS normally containing boric acid solution provides backup if the operating tracing malfunctions.

9.1.3 System Design Evaluation

9.1.3.1 Availability, Reliability, and Redundancy

A high degree of functional reliability is ensured in this system by providing standby components where performance is vital to safety and by ensuring fail-safe response to the most probable mode of failure.

The CVCS has three high pressure charging pumps, each of which is capable of supplying the required reactor coolant pump seal and makeup flow. These pumps also serve as high-head safety injection pumps. Other than the charging pumps, the CVCS is not required to function during a LOCA, nor is it required to take action to prevent an emergency condition. During a LOCA, this system is isolated at the containment boundary except for the charging pumps and the piping in the safety injection flow path.

The generation of a safety injection signal automatically closes the motor-operated valves in the outlet line of the volume control tank and in the normal charging line, thus isolating the CVCS from the safety injection path. The letdown line is isolated by valves which automatically close as a result of a safety injection signal. Two centrifugal charging pumps are automatically started and commence pumping into the RCS immediately on alignment of flow paths as described in Section 6.3.

9.1.3.2 Leakage Provisions

CVCS components, valves, and piping designated for radioactive service are designed to limit leakage to the atmosphere. The components designated for radioactive service are provided with welded connections except where flanged connections are provided to permit removal for maintenance.

The volume control tank in the CVCS provides an inferential measurement of leakage from the CVCS as well as the RCS. Low level in the volume control tank actuates makeup at the prevailing reactor coolant boron concentration. The amount of leakage can be inferred from the amount of makeup added by the reactor makeup control system.

9.1.3.3 Incident Control

The letdown line and the reactor coolant pumps seal water return line penetrates the containment. The letdown line contains air-operated valves inside the containment and one air-operated valve outside the containment which are automatically closed by a containment isolation phase A signal.

The reactor coolant pumps seal water return line contains one motor-operated isolation valve inside and one outside the containment which are automatically closed by a phase A signal.

The seal water injection lines to the reactor coolant pumps are inflow lines penetrating the containment. Each line contains two check valves in series inside the containment and a motor-operated valve outside the containment, to provide isolation of the containment should a break occur in these lines. The motor-operated valves are closed by the operator following the injection phase of the accident.

The charging line is also an inflow line penetrating the containment. The line contains a check valve inside the containment and a motor-operated valve outside the containment which is closed by a safety injection signal.

The RCS fill line, an inflow line penetrating the containment, contains a check valve inside and a remote manual normally closed, administratively controlled isolation valve outside the containment to ensure proper containment isolation.

9.1.3.4 Malfunction Analysis

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 ensure 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 Section 14.2.

If a rupture takes place between a reactor coolant loop and the first isolation valve or check valve, an uncontrolled loss of reactor coolant occurs. The first isolation or check valve is always located as close as possible to the reactor coolant loop. The analysis of LOCA is discussed in Section 14.3.

When the reactor is subcritical, 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 BF3 counters and count rate indicators. Any appreciable increase in the neutron source multiplication including that caused by the maximum possible boron dilution rate is slow enough to give ample time to start a corrective action (boron dilution stop or boron injection) to prevent the core from becoming critical. The boron dilution accident is discussed in Section 14.

At least two separate and independent flow paths are available for reactor coolant boration, i.e., the charging line, or the reactor coolant pump labyrinth seals via the seal water injection lines. The total flow rate through the reactor coolant pump labyrinth seals is 15 gpm or about one-third of the normal charging flow rate. A longer boration period is the only consequence of using the seal water paths to borate the RCS instead of using the charging line. The malfunction or failure of one component does 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 refueling water storage tank outlet to the suction of the charging pumps.

On loss of seal injection water to the reactor coolant pump seals, seal water flow may be re-established by manually starting a standby charging pump. If the seal water injection flow is not re-established, the station can be operated indefinitely if component cooling water is available. The thermal barrier cooler has sufficient capacity to cool the reactor coolant flow which would pass through the thermal barrier cooler and seal leakoff from the pump volute.

9.1.4 Tests and Inspections

Periodic testing, calibration, and checking will be conducted on various instrument channels to ensure proper instrument response and operation of alarm functions. The testing of engineered safety feature valves is discussed in Section 6.

The CVCS is in continuous operation whenever the station is operating. Therefore, the availability of the necessary valves, pumps and heat exchangers is known.

9.1.5 Instrumentation Application

Process control instrumentation is provided to acquire data concerning key parameters about the CVCS.

The instrumentation furnishes input signals for monitoring and/or alarming purposes. Indications and/or alarms are provided for the following parameters:

1. Temperature
2. Pressure
3. Flow
4. Water level

The instrumentation also supplies input signals for control purposes. Some specific control functions are:

1. Letdown flow is diverted to the volume control tank upon high temperature indication upstream of the mixed bed demineralizers.
2. Pressure downstream of the letdown heat exchanger is controlled to prevent flashing of the letdown liquid.
3. Charging flow rate is controlled during charging pump operation.
4. Water level is controlled in the volume control tank.
5. Temperature of the boric acid solution in the batching tank is maintained.
6. Reactor makeup is controlled.

9.2 BORON RECOVERY SYSTEM

The boron recovery system collects radioactive reactor coolant and separates it into stripped liquid and gas. Within the system the stripped liquid is also evaporated to reclaim boric acid and primary grade water for future use or disposal.

Tank capacity is provided for storage prior to processing and to permit sampling and analysis of evaporator distillate and bottoms prior to reuse or disposal.

Monitoring devices to measure pressure, temperature, flow, radiation and liquid level are provided to ensure both the maintenance and effectiveness of control setpoints and the safe operation of the boron recovery system.

9.2.1 Design Bases

Although the station is expected to be operated primarily as a base load unit, the boron recovery system is designed to provide station capability for a flexible weekly load schedule, acceptance of unscheduled load demands, and two back-to-back cold shutdowns.

As a basis of design for the boron recovery system, a weekly load schedule has been postulated as: 2 days at 30 percent power; followed by 5 days each with 6 hr at 30 percent power; an increase to 100 percent power in 1 hr; 12 hr at full power; a decrease to 30 percent power in 1 hr; and 4 hr at 30 percent power.

Discharge of reactor coolant to the boron recovery system is dependent on the magnitude and frequency of the change of boron concentration in the RCS. The change is effected by bleed and feed via the chemical and volume control system. The boron recovery system is designed to enable the station, in the event of an unscheduled load demand, to return to full power at 5 percent of full load per minute ramp rate through most of each fuel cycle.

The two coolant recovery tanks in the boron recovery system are sized to hold the net volume produced by 30 days of maximum reactor coolant letdown to the boron recovery system when operating on the postulated weekly load schedule, plus borated water effluents from two back-to-back cold shutdowns at the end of the 30 day period, less 80 percent of the combined design capacity of the boron recovery evaporators for 30 days.

Back-to-back cold shutdowns consist of: a cold shutdown, 6 hr at cold shutdown, returning to operating temperature, remaining at operating temperature for 1 hr, a cold shutdown, remaining 6 hr at cold shutdown, and returning to operating temperature and full power operation.

Since excess reactivity in the core decreases during the fuel cycle, the full power equilibrium xenon boron concentration is reduced from about 1,030 ppm to 10 ppm over the fuel cycle. Load-following requires adjustment of boron concentration on either side of the full power equilibrium concentration. Load-following is defined herein as the changes in control rod insertion and reactor coolant boron concentration to permit operation of the reactor at a preselected series of power levels, while retaining the capability of responding to an unscheduled load demand up to full power at a ramp rate of 5 percent of full load per minute, over most of the fuel cycle. The maximum letdown volume generated by load-following occurs toward the end of each fuel cycle due to the required dilution of reactor coolant boron concentration. The letdown volume from boration or coolant operating temperature changes is negligible by comparison.

Three limitations on the amount of volume generated per unit time are:

1. The extent to which control rods are used.

2. The letdown flow rate limit imposed by the chemical and volume control system.

3. The extent of the fuel cycle during which the postulated load schedule may be followed.

Based on the above conditions, the maximum letdown volume generated by load-following and two back-to-back cold shutdowns, less evaporator capability, is approximately 375,000 gallon (approximately equivalent to two coolant recovery tanks). This maximum occurs at about 90 percent of the fuel cycle, or 140 ppm full power equilibrium xenon boron concentration, at which point the station load-following capability is reactor control limited.

The boron recovery system is capable of continuously separating dissolved gases from the reactor coolant letdown at its maximum flow rate. Provision has also been made to strip the dissolved gases from the normal letdown, returning the solution to the volume control tank. This reduces the dissolved radioactive gases in the coolant to a minimum. Additional allowance is made for collection and subsequent stripping of reactor coolant from leakage and drains.

The boron evaporators are capable of processing the average letdown rate until nearly the end of the fuel cycle. They are designed to produce a distillate with a boron content below 10 ppm, and concentrated bottoms ranging from 7,000 ppm boron, or 4 percent by weight boric acid to 20,000 ppm boron, or 12 percent by weight boric acid.

The two coolant recovery tanks are located indoors within individual cubicles which are missile protected up to the elevation required to contain the liquid in the coolant recovery tank, and are heated by space heaters. The primary water storage tanks are protected from freezing by forced pump flow through the primary water storage tank heaters and heat tracing of the loop seal.

Design data for boron recovery system components are given in Table 9.2-1. The boron recovery system is shown in Figure 9.2-1.

9.2.2 Description

Reactor coolant letdown containing dissolved hydrogen and fission product gases is normally directed to the "A" degasifier, by action of a three-way diversion valve, in response to manual diversion or level control on the volume control tank in the chemical and volume control system (Section 9.1). Liquid and non-aerated gases collected by the non-aerated vent and drain system (Section 9.7) are directed to the "B" degasifier. If the "A" degasifier is out of service or if letdown is greater than 60 GPM, the "B" degasifier may be aligned for processing the diverted stream. Dissolved gases are separated from the liquid in the degasifiers. Each degasifier has a feed rate of about 75 gpm. The total feed rate of 150 gpm is the maximum expected flow for the degasifier subsystem. Separation of dissolved gases at all reactor coolant letdown flow rates is thus ensured.

Normally the degasifiers operate without steam supplied to the steam heaters and with the degasifier circulation pump in operation. When the water leaves the degasifiers, it is routed to a coolant recovery tank. During station operation, the degasifiers have the capability to continuously degasify the reactor coolant; however, continuous degasification is not currently used. During station shutdown, the RCS is degasified by purging the VCT several times.

Noncondensable gases from both degasifiers are routed to the gas waste charcoal beds and compressed to the surge tank in the gaseous waste disposal system (Section 11.2.3). The surge tank discharges to the gaseous waste disposal system. Gas may also be returned to the volume control tank in order to return hydrogen and decayed fission product gases to the reactor coolant system but is not normally processed via this flow path.

Stripped liquid may be pumped from the degasifier to one of the cesium removal ion exchangers, if required, for removal of soluble fission and corrosion products. Following ion exchange, the liquid is filtered and stored in one of two coolant recovery tanks for further processing by the boron recovery evaporators.

The liquid in the coolant recovery tanks is pumped at constant flow rate to the boron recovery evaporators by one of two pumps. Either pump is capable of supplying one evaporator or both evaporators in parallel. Operation of the evaporators is automatic on selector control from the main control room. Manual override controls are provided for each function.

Evaporator distillate is collected in one of two test tanks for sampling. Noncondensable gases are discharged from the distillate accumulator to the vent and drain system, which combines these gases with the discharge from other vents and directs them to the gaseous waste disposal system (Section 11.2.3).

Once it has been determined that the distillate contains less than 10 ppm boron, it is pumped from the test tanks to the primary water storage tanks. Recirculation of either test tank is provided for sampling. Should test tank contents require demineralization, they may be routed through an ion exchanger prior to storage in the primary water storage tanks. Test tank liquid may also be discharged to one of the high-level waste drain tanks of the liquid waste disposal system (Section 11.2.4) if unsuitable for reuse as primary grade water, or unreclaimable by ion exchange.

The two primary water storage tanks are supplied with pumps and piping similar to the test tank arrangement to allow mixing of either tank for sampling. The pumps provide a constant pressure supply to the primary grade water header. Demineralized water is provided for startup and makeup purposes. The discharge of the PG water pumps can be aligned to supply BVPS-2.

The Primary Grade Water Tanks, located outside enclosed areas, are shown on Figure 9.2-1. Prevention of potential radioactive releases from these tanks is ensured by administrative controls during any filling evolution. Furthermore, limits on the allowable tank activity level in accordance with the Offsite Dose Calculation Manual ensure that any release concentrations will remain below regulatory limits.

Overfill protection for the primary grade water storage tanks is also provided by motor operated valves in the tank fill lines and by overflow lines directed to the fuel building sump. Motor operated valves are closed on high level automatically. Local manual control switches are also provided.

A heating system is available for circulation and/or heating the storage tank water, as necessary. Pumps located in the primary grade water pump room circulate water from the tanks through the heater and back to the tanks.

When the concentration of boric acid in the evaporator is at the desired value, it is pumped batchwise into the evaporator bottoms hold tank through a cooler and filters. The evaporator bottoms cooler is designed to reduce the bottoms temperature for storage.

The concentration of boric acid in the evaporator bottoms hold tank is adjusted with primary grade water to the desired reuse concentration, which is about 4 percent. The bottoms hold tank pump then returns the reclaimed boric acid to the chemical and volume control system or to the boric acid hold tank. The boric acid hold tank provides the surge volume.

When packaging of evaporator bottoms is desired for waste shipment, the cooled liquid is transferred to the evaporator bottoms hold tank. The concentrated liquid is then pumped to the solid waste disposal system (Section 11.2.5) for immobilization in a waste shipment container.

The control of each process within the boron recovery system is essentially automatic once the system setpoints are adjusted. The system setpoints are adjusted based on the setpoints established by subsystem tests performed prior to initial system startup. Operation of the evaporators is automatic upon cycle initiation from the main control board. Batch processing and proper sampling of all liquid ensures control of system effluent streams.

All process piping in the boron recovery system is fabricated of Type 304 stainless steel. All piping joints are welded except where flanged connections are required for maintenance. All valves handling radioactive gas are of the bellows seal type. Valves handling process fluids are stainless steel.

9.2.3 Evaluation

Where continuous processing capability is necessary, as in the degasifier subsystem, duplicate components or trains of equipment are provided. Individual equipment isolation valves are not provided where a duplicate subsystem exists, but are included in single systems where shutdown of the entire system for maintenance is a substantial handicap.

Valve setup error is minimized by the inclusion of remote operation of principal valves and interlocking of controls for proper valve orientation.

A failure analysis of major boron recovery system components is presented in Table 9.2-2.

9.2.4 Tests and Inspections

A program of tests and inspections ensures that the design bases capability of the system is maintained throughout station lifetime.

Prior to station startup, each unit of equipment or subsystem that may be independently operated is made to perform at design flow rates, temperatures, and pressures to establish control setpoints. These control setpoints are used to verify the subsequent proper operation of the subsystems and to ensure that overall system capability is not reduced by any component of the system.

The boron recovery system is frequently operated during the fuel cycle. This frequency of operation, with administrative control, ensures the proper performance of boron recovery system components.

Standby pumps are used at appropriate opportunities to ensure their operability. Preventive checks are performed to ensure that alternate systems or equipment will function upon failure of the operating unit.

9.3 RESIDUAL HEAT REMOVAL SYSTEM

9.3.1 Design Bases

The residual heat removal system (RHRS) shown in Figure 9.3-1 is designed to remove residual and sensible heat from the core and reduce the temperature of the RCS during the second phase of reactor 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 (Section 10). The RHRS is used for normal cooldown and is not considered essential to attain or maintain a safe shutdown.

The RHRS is designed to be placed in operation when the reactor coolant temperature has been reduced to approximately 350°F and the reactor coolant pressure is between 400 and 450 psig. These conditions are assumed to occur approximately 4 hrs after reactor shutdown. For a rated thermal power of 2900 MWt, the system is capable of reducing the temperature of the RCS from 350°F to 140°F within approximately 30 hrs after the RHRS is placed in operation. Actual cooldown times will vary depending on the river water temperature (32 - 90°F range) and equipment availability in the river water, component cooling water and RHR systems.

The system design precludes any significant reduction in the overall design reactor shutdown margin when the system is brought into operation for residual heat removal by equalizing the boron concentration and the temperature with the RCS.

System components whose design pressure and temperature are less than the RCS design limits are provided with redundant isolation means and a relief valve.

Any leakage from the RHRS goes either to the containment or to the component cooling system which is a closed system. Any migration of radioactivity would be detected by the containment particulate and gas monitors (Section 11) if the leakage was to the containment. If the leakage was to the component cooling water system, the component cooling water monitor would alarm in the event the radiation level reached a preset level above the normal background.

All piping and components of the RHRS are designed to the applicable codes and standards listed in Table 9.3-1. Since the system contains reactor coolant when it is in operation, austenitic stainless steel piping is used. The RHRS is a Seismic Category I system.

9.3.2 System Design and Operation

9.3.2.1 System Description

Two pumps and two heat exchangers perform the decay heat transfer functions for the reactor. After the RCS temperature has been reduced to approximately 350°F and the reactor coolant pressure is between 400 and 450 psig, further system cooling is initiated by aligning the residual heat removal pumps to take suction from the reactor outlet line (1A hot leg) and to discharge through the residual heat exchangers into the reactor inlet lines (1B and 1C cold legs). If only one RHR pump and one RHR heat exchanger are available, reduction of reactor coolant temperature is accomplished over a longer period of time.

During reactor cooldown, reactor coolant flows from the RCS. (1A hot leg) to the residual heat removal pumps, through the tube side of the RHR heat exchangers and back to the RCS via the 1B and 1C cold legs. The inlet line to the RHRS is located in the hot leg of reactor coolant loop number one (1A) between the RCS hot leg stop valve and the reactor core. The return line from the RHRS to the RCS connects to the cold legs of the 1B and 1C loops via the accumulator injection lines in the safety injection system. The heat loads are transferred by the RHR heat exchangers to the component cooling water in the component cooling system (Section 9.4).

The cooldown rate of the reactor coolant is controlled by regulating the reactor coolant flow through the tube side of the RHR heat exchangers. An automatically operated control valve in a bypass line around both RHR heat exchangers is used to maintain a constant reactor coolant flow through the RHRS while controlling reactor coolant temperature.

Coincident with the residual heat removal operation, a portion of the reactor coolant flow may be diverted to the CVCS for cleanup. The line used for this purpose connects to the RHRS on the common discharge header of the RHR heat exchangers.

The entire RHRS is located inside the containment with the exception of the line leading to the refueling water storage tank which penetrates the containment.

After refueling, the water level in the reactor cavity is lowered to the vessel flange by opening a valve at the residual heat removal pump discharge and then pumping the water into the refueling water storage tank. If desired, the refueling water can be diverted through the fuel pool purification system for cleanup prior to being returned to the refueling water storage tank.

The line from the RHR system to the RWST may also be used to return water from the RCS to the RWST during tests that inject large quantities of water into the RCS. Returning the water to the RWST will reclaim water instead of transferring the water to the coolant recovery tanks for processing. The line may be used by approved procedures under administrative control when the RCS is vented to atmosphere. Examples of the procedures that may use the flow path are SI full flow tests or SI accumulator discharge tests.

The RHRS is not an engineered safety features system.

9.3.2.2 Components

RHRS component original design data are listed in Table 9.3-2.

<u>Residual Heat Removal Heat Exchangers</u>

The RHR heat exchangers are of the shell and U-tube type with the tubes seal welded to the tubesheet. 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.

<u>Residual Heat Removal Pumps</u>

The two 50 percent capacity residual heat removal pumps are vertical centrifugal units with mechanical seals to limit reactor coolant leakage. All pump parts in contact with reactor coolant are austenitic stainless steel or of adequate corrosion resistant material.

RHRS Valves

The valves used in the RHRS are constructed of austenitic stainless steel or other adequately corrosion resistant materials such as Haynes alloy 25 and 17-4 PH stainless steel.

Manual stop valves are provided to isolate equipment for maintenance. Throttle valves are provided for remote, automatic, or manual control of residual heat exchanger tube side flow. Check valves prevent reverse flow through the residual heat removal pumps.

Isolation of the RHRS is achieved with two remotely operated stop valves in series in the inlet line from the reactor coolant loop number one (1A) hot leg to the suction side of the residual heat removal pumps, and by a check valve (located in the safety injection system) in series with a remotely operated stop valve in each line from the residual heat removal pumps discharge to the reactor coolant cold legs of loops two (1B) and three (1C). Overpressure in the RHRS is relieved through a relief valve to the pressurizer relief tank in the RCS.

Valves that perform a modulating function are equipped with two sets of packing and an intermediate leakoff connection that discharges to the vent and drain system (Section 9.7).

Manually operated valves have backseats to facilitate repacking and to limit stem leakage when the valves are open. Leakoff connections are provided where required by valve size and fluid conditions.

Residual Heat Removal Piping

All RHRS piping is austenitic stainless steel. Welded construction is used extensively throughout the system to minimize the possibility of leakage from pipes, valves, and fittings.

9.3.3 System Design Evaluation

9.3.3.1 Availability and Reliability

For RCS cooldown, the RHRS is provided with two pumps and two residual heat exchangers. If one of the two pumps and/or one of the two heat exchangers is not operative, safe operation of the unit is not affected; however, the time for cooldown is extended. Both RHR pumps are powered from a 4160 V emergency supply and can be operated on the emergency diesel generator on a loss of normal power.

9.3.3.2 Incident Control

The suction side of the RHRS is connected to the reactor coolant hot leg of the 1A loop and the discharge side to the cold legs of the 1B and 1C loops through the safety injection system. On the suction side, the connection is through two electric motor-operated valves in series. Each of the two isolation valves on the inlet line is interlocked with a separate pressure signal to prevent its being opened whenever the RCS pressure is greater than approximately 425 psig. The valves will also be automatically closed whenever the system pressure increases above approximately 600 psig. The isolation valve closest to the RHR system will incorporate a design which will, in addition to the defeat of the opening function described above, include a second (diverse) defeat of the opening function when the pressurizer vapor space temperature is above a preset value. On the discharge side, each connection is made through an electric motor-operated valve in series with a check valve. Each of the motor-operated discharge valves is

interlocked with a separate pressure signal to prevent it from being opened whenever the RCS pressure is greater than approximately 425 psig and to automatically close it when system pressure increases above approximately 600 psig. The interlocks and automatic closing action are derived from separate process control channels for each valve.

The purpose of the automatic valve closure interlock, which isolates the RHR System from the RCS, is to ensure RCS integrity and protect the RHR System from overpressurization by the RCS only during operating modes when RCS pressure is known to exceed RHR design pressure. RHR isolation valve automatic closure interlocks are manually defeated during normal RHR System operation. During RHR System operation, the pressurizer power operated relief valves (with a reduced setpoint) provide overpressure protection for the Reactor Coolant System and an RHR relief valve provides overpressure protection for the RHR System.

Defeating the RHR automatic valve closure interlock improves the plant shutdown safety posture by reducing the potential for inadvertent RHR isolation valve closure and the subsequent loss of RHR cooling capability. With the interlock defeated, remote control of RHR isolation valves from the control room will continue to be available. Thus, the RHR isolation valves may be promptly closed from the control room in the event of a RHR System leak. The automatic valve closure interlock is manually returned to operation during normal RHR System shutdown.

The fluid operating pressure is normally higher on the tube side of the RHR heat exchanger than on the shell side, ranging over an approximate range of 450 to 100 psig, so that in case of leakage, reactor coolant would leak into the component cooling water in the shell side. Abnormally high temperature or radiation level of the component cooling water would be indicated in the main control room. After the leakage condition is corrected, radioactivity in the component cooling water is reduced by bleed and feed.

The residual heat removal pumps are driven by open drip proof type motors with thermalastic epoxy insulation capable of operation in high humidity conditions. They are equipped with splash barriers to protect the motors in the event of a pipeline break in the area, which could possibly spray and wet the motors.

The inlet line from the RCS to the RHRS is between the reactor core and the reactor coolant hot leg loop isolation valve. Thus, if the reactor coolant hot leg or loop isolation valve is closed, the inlet from the RCS to the RHRS is not blocked.

9.3.3.3 Malfunction Analysis

A failure analysis of residual heat removal pumps, heat exchangers, and valves is presented in Table 9.3-3.

9.3.4 Tests and Inspections

The residual heat removal pump flow instrument channels are calibrated during each refueling operation.

The active components of the RHRS are in intermittent use during normal plant operation and no additional periodic tests are required. Periodic visual inspections and preventive maintenance are conducted following normal industrial practice.

9.3.5 Instrumentation Application

9.3.5.1 Temperature

1. Each RHR pump motor has bearing thermocouples for input to the computer and provides for high temperature alarm.

2. Temperature indicators are located both upstream and downstream of the residual heat exchangers. Both monitors input to a recorder in the control room.

9.3.5.2 Pressure

Pressure indication is provided upstream of the heat exchangers with high pressure alarm given in the control room.

9.3.5.3 Flow

Flow instrumentation on the discharge line controls a valve on the heat exchanger bypass line. When the flow through the heat exchangers is changed, the bypass flow is automatically corrected to maintain a preset discharge flow rate. This RHRS flow can be read in the main control room.

9.4 COMPONENT COOLING WATER SYSTEM

The primary component cooling water system consists of the component cooling water, the chilled water, and the neutron shield tank cooling water subsystems. These subsystems are used individually or in combination with each other to provide cooling water for removal of heat from components in the primary plant. The component cooling water subsystem is shown in Figure 9.4-1.

9.4.1 Design Bases

9.4.1.1 Component Cooling Water Subsystem

The component cooling water subsystem, serving the reactor coolant system and its auxiliary systems, is designed to supply water to cool the following components:

1. Reactor coolant pump thermal barriers, bearing oil coolers, and motor stators
2. Excess letdown cooler
3. Nonregenerative heat exchanger
4. Reactor coolant system (RCS) and steam generator blowdown sample coolers
5. Seal water heat exchanger
6. Residual heat removal (RHR) pumps seal coolers (during cooldown)
7. RHR exchangers (during cooldown)
8. Boron recovery system equipment
9. Containment penetration cooling coils
10. Fuel pool heat exchangers
11. CRDM shroud cooling coils
12. Liquid waste disposal system equipment
13. Gaseous waste disposal system equipment
14. Neutron shield tank cooler
15. Refueling water refrigeration units.

The maximum heat load occurs during the initial stages of RHR during unit cooldown. The component cooling water subsystem will reduce the temperature of the reactor coolant to 140 degrees F after a reactor shutdown (i.e., cooldown from 350 degrees F to 140 degrees F assuming 90 degree F river water temperature and no component cooling water heat exchanger tubes plugged). With tube plugging in component cooling water heat exchangers (maximum allowed based on water velocity/erosion considerations), the cooldown time is extended. During cooldown, the reactor coolant flow through the residual heat removal heat exchangers is throttled as necessary to maintain a maximum of 120 degrees F component cooling water out of the component cooling water heat exchangers.

Each primary plant component cooling water heat exchanger is capable of removing half of the normal cooldown heat removal load occurring 4 hours after a shutdown of the station.

The primary plant component cooling water heat exchangers and pumps, and the component cooling surge tank are designed as Seismic Category I components (Appendix B). The valves and interconnecting piping between the above components are also designed as Seismic Category I. The RHR heat exchangers, RHR pump seal coolers, fuel pool heat exchangers, and the component cooling piping connecting these components with the component cooling pumps and heat exchangers are designed as Seismic Category I.

The component cooling water subsystem normally supplies water to some safety-related items (RHR heat exchangers and fuel pool heat exchangers). However, the component cooling water subsystem is not used for accident purposes and is not considered part of the engineered safety features.

9.4.1.2 Chilled Water Subsystem

The chilled water subsystem is designed to perform the following functions:

1. Cool the water in the refueling water storage tank to a temperature of 45°F

2. Supply cooling water for containment air recirculation coolers

3. Supply cooling water to condition the air in various equipment areas such as service building, safeguards area, auxiliary building, pipe tunnel, turbine building basement, and those areas in the station which are normally occupied.

9.4.1.3 Neutron Shield Tank Cooling Water Subsystems

Subsystem is designed to circulate and cool the water in the neutron shield tank.

The water in the neutron shield tank, which is heated by neutron, gamma and thermal radiation, is normally maintained at a temperature below 135°F - the specified ambient temperature for proper operation of the neutron detectors.

A single cooler is sufficient for this application. There are no moving parts which would require redundancy maintenance.

9.4.2 Piping and Valves (Check and Manually-Operated Gate, Butterfly, Ball and Globe)

Piping is carbon steel, except for the shield tank cooling water subsystem which is austenitic stainless steel. Stainless steel is used to help maintain low levels of activated corrosion products. Welded construction is used throughout, with the exception of flanged connections for removable equipment. Carbon steel gate valves with high chrome or stellite or equivalent[(1)] hardfaced material seats are used in up to 8 inch sizes. Ball valves with stainless steel bodies are used in certain applications. The piping system conforms to the requirements of the American National Standards Institute (ANSI) Code for Pressure Piping, B-31.1.

9.4.3 System Description

9.4.3.1 Component Cooling Water Subsystem

Normal Operation

During operation, component cooling water is pumped through the shell sides of the primary plant component cooling water heat exchangers, where it is cooled by river water, and then through parallel circuits to cool the various components.

Three equally sized primary plant component cooling water pumps and three equally sized primary plant component cooling water heat exchangers are provided.

The component cooling water subsystem is a closed system with a surge tank at the pump suctions. The tank is the high point of the system and provides the required net positive suction head for proper operation of the pumps.

The subsystem is provided with two bypasses:

1. To recirculate water from the primary plant component cooling water pump discharge header back to the suction header.

2. To pass some of the flow around the primary plant component cooling water heat exchangers, to facilitate obtaining component cooling water at the desired quantity and temperature.

The following components cooled by the subsystem are located inside the containment:

1. The excess letdown cooler; reactor coolant pumps thermal barriers, oil coolers and motor stator air cooling coils; neutron shield tank cooler.

2. CRDM shroud cooling coils.

3. RHR heat exchangers, RHR pumps seal coolers.

4. Containment penetration cooling coils (outer).

The fuel pool heat exchangers are located in the fuel building. Pumps, tanks, heat exchangers, and the remainder of the equipment cooled by the subsystem are installed in the auxiliary building. One 24 inch main supply and one 24 inch main return line are used to transmit component cooling water to and from the containment. These 24 inch mains are connected directly by individual 18 inch lines to the RHR exchangers, which are at the extremities of the two piping loops. Smaller sized branches connected to the mains serve the remainder of the components being cooled inside containment. High point vents and low point drains are provided as required by piping configuration.

Each cooling water outlet line from a component contains a valve for controlling flow. This valve is either a manually-operated globe or butterfly type or an automatic air-operated type, positioned by pressure or temperature control signals originating in the systems being cooled.

The system is provided with valves for isolating the containment structure in accordance with the requirements of the containment isolation system (Section 5.3). The containment isolation phase A (CIA) signal isolates those portions of the component cooling system that are not needed for orderly unit shutdown. Items that receive the CIA signal include heat exchangers and chillers in the liquid waste and gaseous waste disposal equipment.

The component cooling water system is monitored from the main control room by indicators which display the following data:

1. Pump discharge pressure

2. Radioactivity, temperature, and flow in the supply mains immediately downstream from the heat exchangers

3. Level in the component cooling water surge tank.

Local indicators for pressure, temperature, level, and flow are provided on a general basis. Other important system parameters, such as selected temperatures, levels, pressures, and flows, are also monitored in the control room. Refer to Table 9.4-5 for a listing of parameters that are continuously monitored.

All equipment which might be overpressured by a combination of closed component cooling water inlet and outlet valves and heat input from the isolated equipment have thermal relief valves.

Components that have a single barrier between the component cooling water system and the reactor coolant system are:

1. Reactor coolant pump thermal barriers

2. Sample coolers: SS-E-4 reactor coolant cold leg; SS-E-5 reactor coolant hot leg; SS-E-6 pressurizer vapor space; and SS-E-7 pressurizer liquid space

3. Residual heat removal pump seal coolers

4. Residual heat removal heat exchangers

5. Excess letdown cooler

6. Nonregenerative heat exchanger

7. Seal water heat exchanger.

Table 9.4-6 shows the design pressure and temperature of the barriers confining primary coolant and the operating pressures and temperatures of the reactor coolant in the components listed above. The pressure and temperature design requirements of the barriers for the above components are greater than the reactor coolant operating pressures and temperatures during normal modes of plant operation.

The component cooling surge tank level is controlled at the 50 percent level. Tank capacity is sufficient to accommodate minor system surges and thermal swell. Makeup from the boron recovery system (Section 9.2) is admitted through an air-operated valve controlled automatically or by the control room operator. Letdown of the tank level is a manual operation performed by an operator from a station near the tank.

A chemical addition tank is connected to the primary plant component cooling pump discharge piping to facilitate the addition of corrosion inhibitor or for pH control. To add chemicals to the system, the tank is isolated, drained down, and filled with the necessary chemicals. The isolation valves are then opened and the discharge pressure of the pump forces water into the tank and injects the mixtures into the system at the pump suctions.

The chemical addition tank may be aligned to have continuous flow or may be isolated as necessary. Sampling is performed at the sampling station in the auxiliary building. Several local sample points are also provided.

Component cooling water subsystem component original design data are given in Table 9.4-1.

Loss of Component Cooling Water

Upon loss of the component cooling water, the reactor would be brought to safe shutdown, or hot standby condition, with all reactor coolant pumps tripped. The reactor coolant circulation will be accomplished via natural circulation through the steam generators. Residual heat removal and control of the primary system temperature, pressure, and volume can be achieved in this shutdown mode for an indefinite period during which time restoration of the component cooling water flow can be accomplished.

During any accident condition which initiates safety injection, cooldown is accomplished by the safety injection and recirculating spray systems. The component cooling system is not in use and its loss would not impair the plant's cooldown capability.

The main consequences of the loss of component cooling water are the loss of the reactor coolant pumps, and reduction of the residual heat removal capability needed to accomplish a cold shutdown. Charging pumps, emergency diesel generators and control room air conditioning are cooled by the river water system and are not affected. The high head charging pumps will be available to borate the reactor coolant system, if required, by adding boric acid from the boric acid tanks and the refueling water storage tank through the regenerative heat exchanger or the reactor coolant pump seal water flow paths.

The loss of cooling water to the reactor coolant pumps would not go undetected. Component cooling water is provided to each reactor coolant pump from the main header through individual lines to each reactor coolant pump. These individual lines then branch to each pump's upper and lower bearing coolers, stator coolers, and thermal barrier.

Temperature resistance bulbs are located on the outlet piping from upper and lower bearing coolers, stator coolers, and thermal barrier and would provide local and remote (control room) indication of loss of flow by sensing abnormal temperature. In addition, flow elements (restricting orifices) are located in these lines to sense pressure differential and will annunciate in the control room upon a low flow condition. Thermocouples are installed in the upper and lower thrust bearings and upper and lower radial guide bearings. An alarm will annunciate in the control room on high bearing temperature.

In the event of loss of component cooling water to one reactor coolant pump, there is adequate flow and temperature sensor redundance to provide detection and annunciation of the loss in the control room. For example, there are three low flow sensors in the component cooling lines for each reactor coolant pump as well as four thermal resistance bulbs for each reactor coolant pump. There are four thermocouples present for monitoring bearing oil temperatures for each pump. Partial seizure is not analyzed, but protection is provided for the loss of flow in one loop or the locked rotor, as discussed in Section 14.

None of these sensors will of themselves initiate action due to loss of component cooling water. Operator action is required. All pumps will perform as discussed above. For complete loss of component cooling water, plant shutdown will be initiated as discussed previously.

For the containment penetration cooling coils, the temperature of the adjoining concrete cannot withstand an extended period (12 hours) at a temperature higher than 200°F. Hose connections are provided from the fire protection system to provide an alternate source of cooling water to the coils. The 12-hour period available is ample time to connect all hoses.

Since the operating pressure and temperature of the component cooling water system does not meet the definition of a high energy piping system (less than 275 psig and 200°F), breaks and cracks of the system piping are not postulated. An analysis of the failure of the component cooling water system header (24"-CC-5-151-Q3) in the Auxiliary Building was, however, performed to determine the effect on safety-related equipment in that area. This analysis concluded that if the entire contents of the component cooling system were emptied due to a pipe rupture, no safety-related equipment would be affected since the maximum water level that could occur is less than the levels that would cause the equipment to fail.

Leakage Provisions

The principal methods of leak detection are chemical analysis, radioactivity analysis, abnormal surge tank level decreases or increases, and by indication of water makeup to the system.

The component cooling water system is not normally expected to contain radioactive water. Provisions are made, however, to preclude the possible spread of radioactive contamination in the event that a primary-to-component cooling water leak should occur. These precautions include isolation of each heat exchanger by manual shutoff of the inlet and outlet component cooling water valves, treating any leakage and water samples from these heat exchangers as radioactive, and installation of the heat exchangers within the auxiliary building. Any leakage is then returned to the liquid waste disposal system (Section 11.2.4) via the auxiliary building or containment sump pumps.

The heat exchangers that may develop a reactor coolant-to-primary component cooling water leak that are required to attain and maintain a safe shutdown are:

1. Non-regenerative heat exchanger
2. Reactor coolant pump thermal barriers
3. Residual heat removal exchangers.

Shutoff valves are manually operated from the control room and would be isolated as soon as the source of the leak were identified. During normal operation, the residual heat removal exchangers would not be in operation and the only sources of leakage would be from the reactor coolant pump thermal barrier and the non-regenerative heat exchanger. Any leakage from a reactor coolant pump thermal barrier would result in an excess flow alarm downstream from the offending thermal barrier and automatic isolation would occur.

Reactor coolant may enter via the non-regenerative heat exchanger or residual heat removal exchangers. Flow indicators and pressure indicators that can be monitored from the control room will allow the operator to determine the source of leakage and manually operate shutoff valves from the control room to isolate the leaking component when any component is in operation.

Isolation of any leakage would not result in any exposure to the operator, since all isolation valves can be manually operated from the control room.

Welded construction is used extensively throughout the system to minimize the possibility of leakage from pipes, valves, and fittings.

Small amounts of leakage inside the containment are not considered to be objectionable. Contamination could result from side-to-side leakage in a heat exchanger in the chemical and volume control, residual heat removal, or sampling systems, or from a leak in the thermal barrier of a reactor coolant pump. Temperature, level, and flow indicators in the main control room may be used to detect leakage at certain points. Elsewhere, leaks can be located by inspection or isolation.

9.4.3.2 Chilled Water Subsystem

The chilled water subsystem supplied with water from the chillers located in the turbine room basement provides cooling to the containment air recirculation cooling coils and the containment air compressor aftercoolers. An alternate source of water is provided from the river water system. The chilled water subsystem can be isolated and river water introduced by motor operated valves actuated from the control room.

The chilled water subsystem also cools the water in the refueling water storage tank before startup. Chilled water at 45°F or lower is circulated through the shell sides of the refueling water storage tank coolers. For this duty, the chilled water unit operates at continuously reducing capacity as the temperature of the refueling water decreases from the initial temperature (80 to 90°F) to a temperature consistent with the requirements specified in the Technical Specifications and the Licensing Requirements Manual.

Chilled water subsystem component design data are given in Table 9.4-2.

9.4.3.3 Neutron Shield Tank Cooling Subsystem

One neutron shield tank cooler, a neutron shield expansion tank, a corrosion control tank, and all necessary piping and valves constitute the subsystem.

The heated water in the neutron shield tank rises to the top of the tank and into the line connected to the neutron shield tank cooler. The cool water from the component cooling water subsystem circulates through the neutron shield tank cooler, cooling the heated neutron shield tank water. The neutron shield tank water circulates through the cooler by natural convection circulation. One full duty neutron shield tank cooler is provided to perform the required cooling. The neutron shield expansion tank accommodates thermal expansion of the neutron shield water. A level sensor on this surge tank sends a signal to the main control room to indicate low water level. A valve is actuated from the main control room to replenish the neutron shield tank subsystem from the component cooling water subsystem. The corrosion control tank is used for the addition of an inhibitor when the reactor is not operating; this is a manual operation.

Neutron shield tank cooling subsystem component design data are given in Table 9.4-3.

9.4.4 Availability and Reliability

The component cooling water system uses machinery and equipment of conventional and proven design. All components are specified to provide maximum safety and reliability.

Low flow, low pressure, high temperature, or high radioactivity level alarms alert the operator to malfunctions. If the malfunction causing low flow, low pressure, or high temperature is not corrected, components and systems served by the component cooling system may be inadequately cooled, requiring the operator to shut down the affected components and systems to prevent damage. In the event of high radioactivity, the component that is causing the increased levels is isolated, shutdown and repaired.

If, during normal operation, the performance of the component cooling water subsystem decreases to the point requiring unit shutdown, the system may be operated in the unit cooldown mode. If performance is inadequate for the normal cooldown mode, decay heat can be dissipated by evaporation of water from the steam generators.

During unit shutdown, the performance of the component cooling water subsystem may be inadequate for unit cooldown and cooling of the spent fuel pool for short periods of time. Various loss-of-cooling conditions for the spent fuel pool are analyzed in Section 9.5. The fuel pool cooling assumes component cooling water may be at elevated temperatures, or unavailable, for periods of time. Excessive temperatures do not result from temporary inadequacy of the component cooling water system. In addition, heat is removed by evaporation of water from the spent fuel pool with intermittent make-up. No hazardous conditions can develop.

One primary plant component cooling water pump and heat exchanger provide the required flow and cooling capability to maintain the supplied loads within normal operating parameters under most operating conditions. A second cooling water pump is aligned to start automatically on low pressure at the common discharge of the heat exchangers. The additional heat exchangers and pump are placed in service as necessary due to fluctuations in the river water temperature and to meet the various component cooling water load conditions during operation or cooldown.

Air-operated trip valves are installed in the outlet cooling water lines from the reactor coolant pumps' thermal barriers. A check valve is installed in each inlet cooling water line to the thermal barriers. In the event that a leak occurs in the thermal barrier cooling coil, an alarm annunciates in the main control room and the high pressure reactor coolant is safely contained by automatic closure of the appropriate isolation valve. The air-operated stop valves in the cooling water lines leaving the reactor containment, and in the reactor containment air recirculation cooler lines leaving the reactor containment close on containment isolation phase B signal (Section 5.3).

The neutron shield tank cooling water subsystem has no moving parts. Malfunction can occur only by loss of water through leakage, loss of natural convection circulation through blockage, or low heat transfer due to fouling. Malfunction is indicated by a low expansion tank level alarm or by high shield water temperature indication. If no operator action is taken, neutron shielding is decreased, but no hazardous conditions develop. Operation of the neutron shield tank water cooling subsystem is not normally required during unit cooldown or unit shutdown.

Chilled Water Subsystem

The chilled water subsystem has a design maximum temperature of 45°F during operation. There are no specific chilled water high temperature or low flow limits identified. Duplicate pumps or valves may individually malfunction causing a temporary increase in the chilled water temperature. This results in a reduction in cooling capacity in components served and may necessitate a reduction in some of the normal chilled water loads.

If the chilled water subsystem is malfunctioning to such an extent that the containment recirculation air coolers fail to maintain the containment ambient air temperature below the maximum permissible value, the alternate source of water from the river water system can be used to maintain the required temperature. If, after introducing river water, the containment ambient air temperature remains above the maximum permissible value, then unit operation must be curtailed. Also, malfunction of the chilled water system during cooling of the refueling water storage tank could delay unit startup because the tank must be cooled as specified in the Technical Specifications and the Licensing Requirements Manual.

The main piping loops in the component cooling water subsystem, fuel pool heat exchangers, RHR exchangers and residual heat removal pump seal coolers are Seismic Category I (Appendix B) and are designed accordingly. Also, these loops are analyzed and designed to meet associated thermal stress requirements.

Incident Control

The containment isolation valve arrangement for the component cooling water system is delineated in Section 5.3.

Malfunction Analysis

A failure analysis of equipment and components is presented in Table 9.4-4.

9.4.5 Tests and Inspections

During normal operation, the component cooling system is in continuous service, and integrated performance tests are not required. Component cooling water pumps and valves are tested periodically in accordance with Technical Specification requirements. Visual inspections are conducted following installation of spare parts or piping modifications, to confirm normal operation of the system. Routine pre-startup inspections are performed along with periodic observation and monitoring of system parameters during operation.

REFERENCES FOR SECTION 9.4

1. The term "equivalent" is described in UFSAR Section 1.8.2, "Equivalent Materials."

9.5 FUEL POOL COOLING AND PURIFICATION SYSTEM

9.5.1 Design Bases

The fuel pool cooling and purification system shown in Figure 9.5-1 is designed to remove the heat generated by the stored spent fuel assemblies, to permit unrestricted access to the working area both in and around the spent fuel storage pool, and to maintain optical clarity of the water in the spent fuel storage pool and the refueling cavity. The system is designed for continuous service.

The spent fuel pool heat exchangers with associated pumps are capable of removing the decay heat produced by 22 refueling batches plus a full off load while maintaining the pool temperature at 170°F or less. Purification facilities are provided to permit unrestricted access to the working area and to maintain optical clarity of the spent fuel pool and refueling cavity.

As detailed in Section 9.12.1, fuel in the spent fuel pool is stored in a vertical array using fuel assembly center-to-center spacing, installed Boral and administrative controls to ensure k_{eff} is 0.95 or less even if unborated water is used to fill the pool.

9.5.2 System Description

The spent fuel storage pool has a volume of approximately 51,000 cu ft. The pool is fully lined with stainless steel to prevent leakage. The design of the spent fuel pool is such that a dropped cask cannot result in a loss of water from the spent fuel area, nor in damage to spent fuel in the storage racks. The spent fuel pool is divided into two areas: the spent fuel area and the cask area. Because of the crane arrangement, dry fuel storage canisters and transfer cask or spent fuel shipping casks are only handled over the cask area and cannot be handled over the spent fuel area where the spent fuel storage racks are located. A stainless steel lined concrete wall is provided between the spent fuel area and the cask area. A gate is provided in the wall for the transfer of the spent fuel from storage racks to the spent fuel cask. The gate may be installed when the spent fuel cask is being moved into or out of the fuel pool.

The fuel basin is shielded from missiles emanating from the turbine building by the containment structure and by the auxiliary building. Loss of water as a result of tornado-generated winds sufficient to expose the spent fuel is not considered credible; Stone & Webster calculations drawn from the conclusions reported in APED 5696[1] show that the water level would not be reduced by more than 6 ft. The walls and floor of the fuel basin can withstand the impact of the tornado-generated missile described in Section 2.7.2, without taking credit for the energy-dissipation effect as the postulated missile enters the water above the fuel.

The components of the fuel pool cooling and purification system have design pressures of 150 psig or greater and design temperatures of 200°F or greater. All wetted parts of equipment and piping in contact with the fuel pool water, which is borated to refueling water concentration, are constructed of corrosion-resistant material.

The design of all the components in the fuel pool cooling and purification system complies with the following codes and regulations:

1. Spent fuel pool heat exchangers:

   a. Tubes - American Society of Mechanical Engineers (ASME) Boiler and Pressure Vessel Code, Section III.

   b. Shell - ASME Code, Section VIII.

2. Spent fuel pool ion exchanger - ASME Code, Section VIII.

3. Spent fuel pool cooling and purification system piping valves and fittings - American National Standards Institute B31.1, Code for Pressure Piping and Special Nuclear Cases.

Spent fuel pool cooling system equipment, piping, and associated component cooling water piping are designed for seismic conditions. The spent fuel pool cooling piping conforms to piping Class Q3, as discussed in Section 6.2.

All components of this system are designed for maximum expected flow conditions and ambient conditions inside the fuel building. All system equipment is located above maximum probable flood level.

The fuel pool water flows from the suction connection of the pool via two paralleled pumps, through the tube side of the two heat exchangers, and returns to the pool. The fuel pool heat exchangers are cooled by component cooling water flowing through the shell side. The component cooling flow is manually regulated locally to adjust the temperature of the water returning to the pool.

To ensure an adequate supply of cooling water to the fuel pool under all conceivable conditions, an emergency supply of cooling water from the engine driven fire pump is provided in addition to the normal water addition piping.

The loss of a fuel pool heat exchanger is considered extremely remote, due to the low pressure and inherent reliability of the heat exchanger equipment. For additional safety, each heat exchanger can be isolated from the rest of the system. In the event of failure of either of the fuel pool pumps, the failed pump can also be isolated. Should either pump fail, blind flanges are provided in the inlet piping of the heat exchangers for connecting a temporary pump which would take suction from the spent fuel pool via a temporary line.

Failure of any spent fuel pool cooling pump to operate can be detected by any one of the following annunciations in the main control room:

1. Low pump discharge pressure.

2. Control switch for pump in operating status and pump not running.

Loss of any heat exchanger can be detected by any one of the following annunciations or indications in the main control room:

1. Component cooling water supply isolation valve status; opened, closed.

2. Temperature indication and high temperature alarm for spent fuel pool and reactor cavity.

Local temperature indication on the inlet and outlet of the tube side of each heat exchanger is also provided to monitor heat exchanger performance.

The fuel pool pump discharge line penetrates the spent fuel pool at two locations:

1. In the spent fuel area.

2. In the cask area.

This ensures an adequate supply of cooling water to both areas under all conditions.

The Spent Fuel Pool Purification System uses two 400 gpm purification pumps, two 400 gpm fuel pool filters, a skimming assembly, and one mixed bed demineralizer for maintaining optical clarity inside the pool and for removal of dissolved and suspended solids.

Purification of the spent fuel pool water is accomplished by drawing water from the spent fuel pool using two parallel fuel pool purification pumps. These pumps can be lined up to discharge either through the fuel pool filters only (two are provided, each rated at 400 gpm flow), or through a single mixed bed fuel pool ion exchanger and then through the filters. This water is then returned to the spent fuel pool.

Approximately one and one half volumes of the spent fuel pool water can be processed daily through the spent fuel pool filters. Approximately one half volume of the spent fuel pool water can be processed daily through the mixed bed demineralizer at a water temperature of 140°F or below.

Local sample connections are provided to determine the decontamination factor of the ion exchanger and efficiency of the filters.

A flow indicator is provided to advise that purification flow is in progress to the ion exchanger. A differential pressure indicator across the filters and ion exchanger warns of fouling and indicates that replacement of the ion exchange resin and/or a filter cartridge is required.

A cross connect on the fuel pool cooling and purification system allows refueling water storage tank water to be processed through the spent fuel pool filters and ion exchanger.

Provisions for makeup of spent fuel pool water are provided by connections from:

1. Primary grade water supply.
2. Refueling water storage tank cooling systems.
3. Engine driven fire pump (via fire hose racks).

Connections are provided from the fuel pool ion exchanger to the solid waste disposal system for resin discharge.

The spent fuel pool is provided with a temperature indicator, a high temperature alarm to warn of insufficient cooling, and high and low level water level alarms. The high and low level alarms are sounded in the control room.

The activity levels in the spent fuel pool are determined by analysis of grab samples taken periodically from the fuel pool. The allowable operational levels of activity in the fuel pool are not based on a specific value of radioactive concentration per se, but on the ambient external dose rate and/or airborne activity concentration in the access areas above the pool. The allowable dose rates for personnel working in the fuel building are described in Section 11.5.

Ambient radioactivity concentrations in the access areas above the pool are monitored by radiation detection equipment described in Section 11. An area monitor with a range of 0.1 mr per hr to 1,000 mr per hr located on the fuel pool bridge monitors the external radiation levels. In addition, two detectors (one redundant) are located in the inlet plenum of the fuel building exhaust ventilation duct continuously monitor the gross ambient airborne activity concentrations in the fuel building exhaust air. The sensitivity of the detectors is 1 x 10^{-6} μCi/cc with a four decade range.

The water surface of the spent fuel pool and refueling cavity is maintained free of floating dust and other material by skimmers.

Water flows through the skimmer, into a purification pump, a filter, and back to the pool through the return nozzle located at the opposite end of the spent fuel pool from the skimmer. The skimmer system is not expected to be in continuous use, but is operated as needed to maintain the cleanliness of the spent fuel pool.

9.5.3 Evaluation

Under normal operating conditions, the fuel pool cooling and purification system is capable of maintaining acceptable low radiation level and high optical clarity. With a discharge of 157 fuel assemblies (full core offload) to the fuel pool, temperature peaks 136 hours after reactor shutdown at 156°F with a heat load of 40.75 x 10^6 Btu/hr. These results are based upon the following conservative assumptions:

1. Component cooling water supply at the maximum allowed value based on refueling start time.
2. Both fuel pool heat exchangers and pumps operating.
3. Spent fuel equal to 22 refueling batches is stored in the pool from previous refuelings.
4. Refueling begins at a minimum of 100 hours after reactor shutdown with fuel assemblies discharged to the pool at the average rate of six assemblies per hour.
5. Core power level of 2918 MWth.

With a partial loss of cooling circulation (one cooling pump inoperative), the cooling system is capable of maintaining the spent fuel storage pool water below 170°F. With the above spent fuel storage and a complete loss of cooling circulation, the spent fuel pool water temperature would rise to boiling in approximately 2.75 hours.

Under abnormal operating conditions with a full core discharged to the pool and one refueling load decayed 36 days and the previous off load decayed 400 days, the pool temperature peaks 136 hours after reactor shutdown at 169°F with a heat load of 49.13×10^6 Btu/hr. If a complete loss of cooling occurs under the abnormal condition, the spent fuel storage pool water will boil in approximately 2.33 hours.

Administrative controls ensure acceptable fuel pool temperature by controlling offloading of fuel based on component cooling water temperature and decay time (refer to License Amendment No. 247 of the Unit 1 Technical Specifications).

9.5.3.1 Operational Situations-Spent Fuel Pool Water Temperature Increase

Upon loss of normal station power, the spent fuel pool cooling pumps can be powered by the emergency diesel generators.

Failure of a cooling system pump to operate will be indicated in the control room by low discharge pressure alarms and annunciation of abnormal pump operating status. The following alarms and/or indications will alert the operator of loss of component cooling water supply to the heat exchangers:

1. High spent fuel pool water temperature (main control room).

2. Low pool water level alarms (main control room).

3. Component cooling water supply valve position indication (main control room).

Blind flanges are provided on the inlet and outlet of component cooling water lines connected to the spent fuel pool heat exchangers. These flanges can be used to connect temporary lines from the fire protection system in order to provide make-up water in case of a loss of component cooling water supply.

Concrete stresses within the fuel pool walls have been reviewed at a steady state condition of 200°F on the inside surface of the concrete, linear thermal gradient through the wall, and ambient temperature conditions on the exterior concrete surfaces. The rate of change of the spent fuel pool water temperature, with loss of cooling conditions as stated in Section 9.5, will not cause localized temperature gradients that will affect critical concrete stresses.

The water temperature inside the spent fuel pool rises above the ambient temperature and produces thermal stress which subjects the wall face inside the pool to compression. These thermal stresses in the walls are calculated on the basis of a linear thermal gradient. In order to evaluate the effect of thermal loading, the d section and a reduced value for Young's modulus of concrete. Under service conditions the large thickness of the wall (based on biological shielding requirements) limits deflection. The reinforcement which is designed to control racking conforms to paragraph 1508 (b) of American Concrete Institute (ACI) 318-63. A combination of the above factors, together with a well-distributed reinforcement detail in the tension side, limits cracking.

Provisions for makeup of spent fuel pool water are provided by connections from: primary grade water supply; refueling water storage tank cooling systems (Figure 6.4-1A); or an engine driven fire pump (via fire hose racks).

Failure of any spent fuel pool cooling line penetrating the pool will not permit the water level to drop below approximately 10 feet above the top of the active fuel. All lines that penetrate the spent fuel pool are provided with isolation valves located as close as possible to the penetration.

Piping design is such that it is not possible to siphon the spent fuel pool water level down as the result of a failed pipe or component to a water level below approximately 10 feet above the top of the active fuel. This level provides adequate shielding and cooling of the spent fuel while system repair is completed.

In the event that a failed or leaking fuel assembly is found during refueling, it can be placed in one of the sealed failed-fuel cans (if available) which are filled with water. The assembly and its container are stored in the spent fuel pool. Decay heat is transported to the container wall by natural convection, through the wall by conduction, and to the spent fuel pool water by natural convection.

9.5.3.2 Radiological Situations Spent Fuel Pool Water Temperature Increase

An increase in fuel pool water temperature could conceivably reduce the iodine nuclide retention capability of the pool water and nullify the effectiveness of the purification mixed bed demineralizer resin. The radiological effects from the increase of water temperature from 125°F to 200°F due to cooling system malfunction are based on the following parameters:

1. Nuclide release to the fuel based on an escape rate coefficient equal to 10^{-5} of the escape rate coefficient used for failed fuel nuclide release in the core during normal operation.

2. Fuel pool purification demineralizer flow rate of 150 gpm.

3. Six-month retention of full core inventory.

4. Fuel pool water volume of 5.1×10^4 cu ft.

5. Fuel building volume = 1.5×10^5 cu ft.

6. Fuel building exhaust rate = 2000 cfm.

7. 0.25 percent of fuel failed.

For fuel pool water temperature ranging from 125°F to 140°F, it is assumed that the water provides an effective iodine decontamination factor of 100 (i.e., 99 percent of the iodine released from the fuel is retained by the pool water). The mixed bed demineralizer in the fuel pool purification system provides a decontamination factor of 10 for iodine removal. For noble gas, the fuel pool water, and mixed bed demineralizer are assumed to have no removal credit. The average dose rate level in the fuel building from activity assumed to be released from the fuel pool is based on 100 hr of occupancy during the 6-month period that the spent fuel is assumed to be in the pool. The calculated average thyroid dose rate level is 1.4×10^{-4} mrem/hr and whole body dose rate is 7.6×10^{-4} mrem per hr.

For fuel pool water temperatures ranging from 140°F to 200°F, it is assumed that the water provides an effective iodine decontamination factor of 10 (i.e., 90 percent of the iodine release from the fuel is retained by the pool water). It is further assumed that the purification system mixed bed demineralizer is no longer functional at these relatively elevated temperatures. The expected dose rate levels in the fuel building with the elevated water temperatures are 4.6×10^{-1} mrem per hr to the thyroid and 7.6×10^{-4} mrem per hr whole body.

The above analyses assume the temporary corrective action to provide fuel pool cooling to be of long term duration resulting in sustained water temperature greater than 140°F. In reality, it is expected that the cooling system would be repaired and functional within a period of one week resulting in the pool water temperature returning to below 140°F. For the realistic case, the average occupational thyroid dose rate equals 1.2×10^{-1} mrem per hr and the dose rate to the whole body would remain unchanged. In summary, sufficient cooling capacity and makeup of water for the spent fuel pool is provided for normal and abnormal conditions.

9.5.3.3 Tornado-Generated Missiles

As noted in Section 9.5.2, the walls and floor of the concrete structure forming the spent fuel storage pool can withstand the impact of the tornado-generated missile described in Section 2.7.2. The stored fuel in the spent fuel pool is protected from tornado-borne missiles by 6 ft thick heavily reinforced concrete walls extending 31.33 ft above grade and approximately 24 ft of water covering the fuel racks.

The probability of a tornado occurrence at BVPS-1 is very slight, having a recurrence interval of between 2,100 years and 15,200 years, depending on the assumptions made, as discussed in Section 2.2.2.5.

The sides of the fuel pool are designed to withstand safely the effects of horizontal missiles. To endanger either the fuel elements or the liner, the potential missile would have to be acted upon by the maximum wind speed for a distance exceeding 300 ft, accelerated horizontally, ejected at a 45 degree angle by an elastic impact, maintain an orientation that will impart maximum velocity to the missile, and then be hurled to the direct location of the fuel pool and impact the pool at the critical angle required to possess enough energy to endanger the fuel. An analysis using this approach has been followed similar to that presented by Miller and Williams which shows the probability of such an event not to exceed 10^{-7}.[1]

In the highly improbable event that a missile would be generated that would enter the pool at the critical angle required to impact the fuel rack, the structural framework of the racks would be required to absorb the energy. The design of a typical fuel rack has been analyzed to have a strength in excess of 5,000 ft-lb, and typical assembly in excess of 1000 ft-lb before cladding damage will occur. This exceeds the impact energy of all credible missiles that could enter the pool. During closure operations of the spent fuel transfer cask, the intermediate wall of the Spent Fuel Pool separating the cask loading pit and the spent fuel storage area may be subject to damage from a tornado generated missile striking the transfer cask when the transfer cask is setting on the cask chair. The probability of a missile striking the cask sitting on the cask chair during cask closure activities has been evaluated as low. Although determined to be not credible, the event of a tornado generated missile striking the transfer cask with a loaded dry shielded canister on the cask pit chair during closure operations was postulated. The analyses concluded that some local damage to the top of the intermediate wall separating the cask pit from the spent fuel storage area may occur. However, the wall remains intact and the adjacent spent fuel is not affected.

An analysis of credible missiles has been made by D. F. Paddleford in "Effect of Tornado Missiles on Stored Spent Fuel,"[2] with the result that:

1. Postulated missiles that in fact could breach the spent fuel pool elevated walls would not damage the fuel assemblies.

2. Missiles, even if it is assumed that they somehow enter the pool, would have to enter and traverse the water with their minimum projected area oriented perfectly normal to the water surface to be considered a potential problem.

3. Only a certain selected few missiles having specific shapes (concrete slab less than 3 inches and exceeding 6 ft in length by 1 ft wide, and steel rods of 1/2 inch diameter exceeding 10 ft in length) unlikely to be available at the site would cause marginal considerations.

The analysis of a typical siding panel of the fuel building, assuming a perfectly normal entrance to the pool, on its smallest cross section, undeformed, was analyzed with a resultant loading on the racks of less than 5 percent of their design strength.

The fuel building is designed and detailed so that under the generated wind forces, the siding is designed to have failed at a maximum wind velocity of 170 mph (60 psf loading on windward side). See Figure Responses to BVPS Unit 2, 50-412, Amendment 3, Question 3.3, for figures and sketches related to this discussion.

The metal siding consists of an exterior 22 gage box rib type Galbestos metal sheet, 20 gage inner L2 liner panel sheet subgirts, and 1-1/2 inch insulation, manufactured by H. H. Robertson Company of Ambridge, Pennsylvania. The metal siding is supported on steel girts spaced at 10 ft-6 inch intervals on the lower 42 ft of the structure and 9 ft-5 inch intervals on the upper portion of the structure. The metal siding is constructed in maximum two span lengths and is fastened to structural steel girts.

Under positive wind pressure, the exterior Galbestos sheet, which is the strong element, fails by buckling when the maximum stress reaches the yield point of the material. The siding is constructed with two spans continuous over a center support. Under increasing load, it hinges at the center support and reverts to two simple spans. Continued deflection of the siding results in tearing of the inner liner sheet at the end fasteners and bending of the siding over the center support. The failure load from tests conducted on the siding compares to the calculated failure load and verifies the assumptions made.

Panel failure under negative wind load occurs first at the panel liner joints, where special pressure release fasteners are used, since facia load is transferred to the structural girt system through the subgirts attached to the liner lips. The fastener at the center of the liner is merely a stitch screw used to minimize flat plate oil canning.

The steel roof deck is fastened to steel members by special pressure release fasteners. The roof deck is expected to fail by failure of the fasteners, thus relieving the members from any overloading.

Based on calculations and verified by test, the building wall panels can be expected to fail under a positive wind load pressure of approximately 42 lb per sq ft for the lower portion of the building. Similarly, based on girts spacing of 9 ft-5 inches the upper portion of the building is expected to fail at about 53 lb per sq ft.

Factory Mutual tests of approved fasteners used for the panels have shown that the panels can be expected to fail under a negative wind load pressure of approximately 35 lb per sq ft negative pressure for the liner end joints. The panels can be expected to fail under a negative wind load pressure of approximately 37 lb/sq ft at the liner intermediate girts.

The siding panels being blown in toward the pool are assumed to fail. This mechanism of failure assumes the siding wraps around the middle of the three girts supporting the sheets. Any such panels that may be postulated to fall in the pool will have relatively large surface areas and have relatively light unit weights. These missiles would be slowed down upon entering the pool and have small impact velocities after traveling through the 20 ft of water before impacting the fuel racks. No damage to the fuel elements will occur because of these missiles entering the pool.

The ability of the bridge, trolley, and hoist spanning the spent fuel storage pool to withstand the tornado without failing in a manner that could damage the stored fuel has been analyzed.

The uplift from the tornado is ignored when considering the crane, since the crane is constructed with materials whose density and structural shapes will not present an area over which an uplift can be developed.

Sections 2.7.2 and 2.7.2.1 discuss the formulation of an equivalent wind loading pressure of 330 psf for maximum tornado wind speeds. Using a drag factor for rectangular plate of 1.3, the 330 psf wind loading and the conservatively calculated area of the crane of 200 sq ft results in a horizontal wind load of 85 kips. The crane is designed to remain in place for horizontal earthquake loads of 1.4g. The estimated weight of the crane is 62 kips. Earthquake restraints provide resistance to horizontal motion due to external loads of 1.4 x 62K or 86.8 kips.

Similarly, a horizontal tornado load on the front face of the crane and the related overturning moment are restrained by the vertical seismic wheel restraints. A 10 kip load at each wheel restraint is required to withstand tornado winds overturning moment. The wheel restraints are designed for 25 kips each to withstand hypothetical earthquake loads. Therefore, the crane will remain in place during the tornado, and no damage to the stored fuel will be sustained.

9.5.4 Tests and Inspections

During normal operation, the fuel pool cooling and purification system is in continuous use. Periodic sampling is necessary to check both the radioactivity and the boron concentration of the spent fuel pool water, the condition of the ion exchanger, and the condition of the filters. Periodic visual inspection and preventive maintenance of all components are conducted in accordance with normal nuclear power station practice.

REFERENCES FOR SECTION 9.5

1. Miller and Williams "Tornado Protection for Spent Fuel Pool," APED 5696, General Electric Corporation (November 1968.)

2. D. F. Paddleford, "Effect of Tornado Missiles on Stored Spent Fuel," WCAP-7572 (September 1970.)

3. Holtec Report HI-92791, Rev. 6, "Spent Fuel Pool Modification For Increased Storage Capacity, Beaver Valley Power Station Unit 1," forwarded to the NRC with Technical Specification Change Request No. 202, dated November 2, 1992, as supplemented by calculation 8700-DMC-3664, Rev. 0.

4. Non-Linear Analysis of the Cask Pit Wall for Impact From Indirect Tornado Missile Strike, calculation 8700-DSC-0348, Rev. 0.

9.6 SAMPLING SYSTEM

9.6.1 Design Bases

The sampling system is designed to provide liquid and gaseous samples from the various station fluid systems for laboratory analysis or for continuous monitoring. Two sample rooms are provided, one for possible radioactive fluids which is located in the auxiliary building, and one for nonradioactive fluids, located in the turbine building.

The nonradioactive sampling system does not require any special provisions and is designed in accordance with normal power plant practices for turbine plant sampling. The special features and precautions in the other sections apply only to the possibly contaminated system.

9.6.1.1 Special Features

Safety features are provided to protect laboratory personnel and prevent the spread of contamination from the auxiliary building sampling room when samples are being drawn. The sampling system is isolated at the containment boundary on a containment isolation phase A signal. Sampling system discharges are designed to limit flows under normal conditions and anticipated malfunctions or failure to preclude any fission product release in excess of 10 CFR 20 and the guidelines of Appendix I to 10 CFR 50.

Samples of process fluids and gases associated with both the primary system and the secondary system are either taken periodically or continuously monitored. Reactor coolant samples may be obtained during reactor operation from an operating loop and/or an isolated loop at a point in the hotleg of each loop chosen to give a representative sample of the loop. A reactor coolant sample can also be obtained during cooldown when the system pressure is low through the use of the sampling points in the residual heat removal system.

9.6.1.2 Sample Temperatures

Two general types of samples are obtained: high temperature samples (>150°F) such as the reactor coolant system samples; and low temperature samples (≤150°F) such as the chemical and volume control system demineralizer effluents. The various samples taken are listed in Table 9.6-1. The high temperature samples are reduced to approximately 115°F by first passing them through sample coolers. Component cooling water is used as the cooling medium for the possibly contaminated system. This arrangement minimizes the release of radioactive gases in the sampling panel.

9.6.1.3 Codes and Standards

The piping for the sampling system is designed to ANSI B31.1, Code for Pressure Piping, including nuclear code cases where applicable. Heat exchangers are designed to ASME Boiler and Pressure Vessel Code, Section VIII, and applicable nuclear code cases.

9.6.2 Description

All the sampling lines coming from within the containment contain high temperature samples with the exception of the pressurizer relief tank sample and the safety injection accumulator samples. Sampling lines coming from within the containment have air-operated or solenoid valves which can be remotely operated from a control panel in the auxiliary building sampling room, so that the desired sampling point can be selected. The primary coolant sampling lines and the safety injection accumulator sampling lines each join into their respective common headers downstream of the selection valve prior to penetrating the containment. The primary coolant samples flow through delay coils prior to penetrating the containment. These delay coils permit sufficient decay of Nitrogen-16 so that these samples can be handled in the sampling room. The pressurizer vapor space sample passes through capillary tubing which limits the flow of the steam.

All sampling lines penetrating the containment, with the exception of the steam generator blowdown sampling lines, have two remotely operated valves located in each line, one just inside and one just outside the containment. The steam generator blowdown sampling lines have just one trip valve each, located outside the containment. All trip valves close on a containment isolation phase A signal. In addition to closing on a containment isolation Phase A signal, the blowdown line trip valves will also close on a signal from Auxiliary Feed Pumps Start. All sampling line trip valves inside and outside the containment can be opened and closed from the main control board. The high temperature samples pass through sample coolers, sample capsules (with the exception of the steam generator blowdown sample line), and valves, in that order, located in the auxiliary building sampling room.

The sample coolers cool the high temperature samples to a temperature low enough for safe handling (approximately 115°F). Sample flows leaving the cooler are manually throttled and can be directed to a purge line or to the sampling panel. After sufficient purging, a pressurized sample can be obtained by filling a sample capsule, sealing the capsule and removing it from the sample line for analysis. An unpressurized sample can be obtained by diverting a portion of the flow to the sampling panel.

Low temperature samples from the containment and the auxiliary building flow to the auxiliary building primary sampling panel. Sampling lines from sampling points in the turbine building flow to the turbine building sampling panel. The high temperature samples flow through sample coolers and are then manually throttled. In general, these samples can either be directed to a purge line or to the sampling panel.

The purge flows of the various containment and auxiliary building samples are discharged to the volume control tank, or the vent and drain system. The samples in the auxiliary building sampling room discharge into a hooded sampling sink, which drains to the vent and drain system. The discharge flow from the steam generator blowdown sample panel can be diverted to either the river water system discharge header during normal operation or the liquid waste disposal system via the north sump in the auxiliary building in the event of a steam generator tube rupture.

In addition to the above facilities for periodic sampling, there are facilities in the sampling system for radiation monitoring of the steam generator blowdown samples, and continuous chemical monitoring of the condensate and feedwater systems. The potentially leaking steam generator can be determined when alarms are received from the radiation monitors in the condenser air ejector effluent or steam generator blowdown tank discharge line, indicating a possible primary-to-secondary leak in the steam generators. Chemical monitoring of the condensate and feedwater systems is required for detecting tube leaks in the main condenser.

Typical of the analyses performed on samples are boron concentration, fission product radioactivity level, dissolved gas content, and corrosion product concentration. Analytical results are used in regulating boron concentration, evaluating fuel element integrity, evaluating mixed-bed demineralizer performance, and regulating additions of corrosion controlling chemicals. The sampling system is designed to be operated manually, on an intermittent basis for conditions ranging from full-power operation to cold shutdown.

Anticipated sample size is in the order of 500-1000 cc, depending on the sample and the number and type of analyses to be performed.

Demineralized water is supplied to flush and clean the sampling sinks and sample containers.

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 sampling panel. Each purge line contains a flow indicator so that the purge flow rate can be regulated.

9.6.3 Evaluation

Except for the containment isolation valves, which close on a containment isolation phase A signal, the remainder of the sampling system is not required to function during an emergency, nor is it required to take action to prevent an emergency condition. It is, therefore, designed to perform in accordance with standard industrial practice.

If a critical sampling line becomes inoperable due to some malfunction, there is at least one alternate path which can be used to obtain a similar periodic sample or for continuous monitoring. For example, if the condensate pump discharge sample line becomes inoperative, condensate can be monitored continuously for conductivity with the condensate hotwell sampling line.

Radiological protection of personnel from sample system lines is provided through the use of biological shielding. Potentially radioactive sampling lines are run to the sample station in the auxiliary building (El. 735 ft-6 inch) through pipe trenches under El. 722 ft-6 inch and vertically through a pipe chase. These trenches and chase are encased by a minimum of 2 feet thick concrete, or concrete block (minimum density 135 lb per cu feet). Sample lines are run between containment and the auxiliary building in the underground trench, which is also shielded with concrete. Therefore, these lines pose no hazard to personnel. In the sample room all sample lines are run behind a sample panel wall which is also a minimum 2 feet thick concrete or concrete block biological shielding. The sample panel and wall protects personnel drawing samples or otherwise working in the area by minimizing direct exposure of personnel to potentially radioactive sample lines.

The sample panels and sinks are fitted with enclosed, ventilated hoods. The building exhaust system provides a sweep gas effect for the sample area, exhausting through the building exhaust stack. Any radioactive gases released from samples to the hood are swept away from personnel to the building exhaust. Radiation monitoring is provided in the building exhaust system as described in Section 11.3.3.3.5.

All high temperature samples are cooled to approximately 115°F to guard against personnel handling samples of excessive temperatures. High pressure sample lines from containment are provided with containment isolation valves inside and outside containment in addition to air-operated or solenoid selector valves upstream of the isolation valves inside containment. The air-operated selector valves are remotely operated from a panel at the sample station and are designed to fail closed. Other high pressure lines originating outside of containment are also fitted with similar remotely operated selector valves. Whenever the sample lines provided with remotely operated selector valves are not in use, the selector valve on any particular line is closed, and the line may be depressurized by draining it to the appropriate drain or waste gas header. This procedure minimizes the length of time the subject lines are pressurized. In this manner, should a tubing or fitting rupture occur due to accidental overpressure or oversurge, an operator in attendance at the sampling station can, with proper instrumentation monitoring, determine rupture occurrence and immediately isolate the affected line. Other sample lines outside containment may be manually isolated at the sample point.

Line rupture itself poses little danger to operating personnel or the public. Any fluid released is contained within shielded, enclosed areas. Released liquid is collected in sumps via a floor drain system and subsequently pumped to liquid waste tanks for processing and disposal. Gases are collected by the gaseous waste disposal system via sweep gas over the sumps. Also, post rupture uncontrolled flow rates are inherently small due to the small size of the sample lines. (0.245 inch ID for 3/8 inch tubing used for pressures less than 350 psig and 0.12 inch ID for 1/4 inch tubing used for pressures greater than 350 psig). Proper monitoring of instrumentation alerts the operator to the rupture occurrence and appropriate corrective measures can be instituted immediately.

Rupture due to oversurge is highly unlikely. Bypass valves are initially opened to allow flow to the appropriate drain or vent header for purging purposes. When the actual sample is to be taken, the operator can regulate the flow out of the sample spigot or through the sample bottle by the use of hand control valves and/or throttling globe valves.

9.6.4 Tests and Inspections

Most components are used regularly during power operation, shutdown, and/or cooldown, thus continuously ensuring the availability and performance of the sampling systems. The continuous monitors are periodically tested, calibrated and checked to ensure proper instrument response and operation of alarm functions. The frequency of each of these tests, calibrations, and inspections is given in the Technical Specifications.

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9.7 VENT AND DRAIN SYSTEM

9.7.1 Design Bases

The vent and drain system collects potentially radioactive fluids and gases from various systems and discharges them either to the gaseous waste disposal system (Section 11.2.3), the liquid waste disposal system (Section 11.2.4), or the boron recovery system (Section 9.2).

The drains are separated into those which contain air (aerated drains) and are sent to the liquid waste disposal system, and those which contain nonaerated reactor coolant fluid and are sent to the boron recovery system for processing and recovery.

The vents are separated into those which contain air (aerated vents) and those in which hydrogen and radioactive gases are predominant (gaseous vents). Both aerated and gaseous vents are separately sent to the gaseous waste disposal system.

There are no seismic requirements invoked for drainage systems which serve floor drains located below elevation 730 ft-0 inch. The simultaneous occurrence of an earthquake and a probable maximum flood is not considered to be a design basis event.

The only safety-related equipment required for safe shutdown located below the elevation of the PMF in the auxiliary building (see Figure 2.7-17) are the charging pumps. These pumps are individually sealed from water entering the auxiliary building by concrete walls, having all construction joints waterstopped and all penetrations below the elevation of the PMF sealed against the ingress of water.

9.7.2 Description

High level aerated drains (Section 11.2.4.1) are sent to the liquid waste disposal system for processing and disposal. Low level aerated drains are sent to the liquid waste disposal system for disposal.

Sumps are located in the:

1. Reactor containment
2. Incore instrumentation tunnel
3. Safeguards area
4. Tunnel between the reactor containment and auxiliary building
5. Auxiliary building
6. Fuel building
7. Solid waste area
8. Boric acid holdtank area
9. Decontamination building

10. Primary water storage tank area.

The fuel pool tell-tale drains receiver is pumped into the decontamination building sump. The collected aerated drainage is transferred by sump pumps to either a high or low contamination level drain tank in the liquid waste disposal system, depending on its activity level. If the contamination level of the drains is unknown, it is transferred to one of the high level waste drain tanks, checked, and if so indicated, transferred to a low level tank.

The containment sump collects open drains from the containment. The auxiliary building sumps collect floor drains, drains from equipment, ion exchanger drains, and filter drains. The other sumps collect floor drains in their respective areas. Drainage from ion exchangers, sample drains, laboratory, and spent resin flushing are also collected in the sumps.

Drainage from systems containing reactor coolant or from systems into which reactor coolant might leak (nonaerated drains) are collected in the primary drain transfer tanks for eventual return to the reactor coolant system. Primary drain transfer tank No. 1, located in the containment, receives drains directly from the reactor coolant pump No. 2 and 3 seal leakoff, the reactor flange leak detection line and, by administrative control, from the safety injection accumulators. All RCS valve stem leakoffs within the containment are routed to either the pressurizer relief tank or the No. 1 primary drain transfer tank. The reactor coolant loops can be drained directly or via the excess letdown heat exchanger (Section 9.1) to either primary drain transfer tank No. 1 or the reactor containment sump. The reactor vessel cannot be drained to the primary drain transfer tanks. Primary drain transfer tank No. 2, located in the auxiliary building, receives drains from valve stem leakoffs outside the containment and, by administrative control, from the volume control tank, and the sample system liquid header. The contents of the primary drain transfer tanks are pumped to the boron recovery system.

Large volumes of noncondensable gases are piped directly to the Degasifier Vent Chillers directly from the volume control tank vent lines, and pressurizer relief tank vent, bypassing the degasifiers. All the above mentioned lines tie into the 2 inch line immediately downstream of restricting orifice, but before the vent chillers. The volume control vent line will include a pressure control valve in order to prevent pressure surges within the system.

Aerated vents from the boron recovery system and liquid waste disposal system, the refueling water storage tank, the boric acid tanks, the component cooling surge tank, sumps and the spent resin dewatering tank are discharged via the gaseous waste disposal system.

The tanks, sumps, and pumps in the nonaerated and aerated drain systems are listed on Table 9.7-1.

Piping for the vent and drain system is designed in accordance with ANSI B31.1, Code for Pressure Piping. Containment isolation valves are provided in all vent and drain lines from the containment. For details see Section 5.3.

The equipment and floor drainage systems serving each seismic Category I structure or space are described below in the order listed in Table B.1-1. Also discussed are the precautions taken to prevent flooding by ingress of river water through the drainage system or other openings during a probable maximum flood.

Containment Equipment and Floor Drainage

Equipment drain lines in the containment convey condensate from control rod drive mechanism shroud cooling coils and air recirculation system cooling coils to the drainage trench supplying the containment sump. Floor drain lines in the containment convey liquid from solid floors at upper levels to the same drainage trench.

Flooding of the containment cannot occur by ingress of river flood water, since the drainage system is self-contained within the containment structure and is not susceptible to flood water.

Cable Vault and Cable Tunnel

There are no equipment or floor drains in the cable vault or cable tunnel. There are no sources of water located above these structures that could cause flooding. The elevation of the cable vault is above the probable maximum flood level and not susceptible to flooding.

Pipe Tunnel to Containment from Auxiliary Building(El. 722'6")

Floor drainage in the pipe tunnel is conveyed to the tunnel sump. The floor drainage system is designed to accommodate minor piping leaks or normal maintenance procedures. There are no precautions necessary to prevent flooding in the event of a major pipe rupture in any of the pressure piping systems located in the pipe tunnel. For the same reason, the pipe tunnel is not protected from ingress of river water during a probable maximum flood.

Main Steam Valve Area (El. 751'0")

Floor drainage in the main steam valve area is conveyed to the sump in the pipe tunnel. The floor drainage system is designed to accommodate minor piping leaks or normal maintenance procedures.

A decay heat line break within the Main Steam Valve Area is considered to be the limiting break. The break will not result in flooding as most of the escaping steam will be vented from the cubicle as superheated vapor.

The feedwater line is analyzed for a circumferential break outside the cubicle wall and at the containment end of the containment feedwater penetration. The floor penetrations above the auxiliary feedwater pumps and the quench spray pumps have been fitted with dams capable of handling the maximum possible level of water. These dams will prevent water from pouring onto electric motors. As stated above, floor drainage is conveyed to the sump in the pipe tunnel.

The auxiliary feedwater pipe, which branches from the main feedwater line to the first check valve in the valve cubicle area, is not considered as the consequences will be less serious than those described above.

The elevation of the main steam valve area is higher than the probable maximum flood and not subject to flooding.

Pump Room Below Main Steam Valve Area (El. 735'6")

Floor and equipment drainage in this pump room is conveyed to the sump in the pipe tunnel. The floor and equipment drainage system is designed to accommodate minor piping leaks, normal pump bedplate drainage, or normal maintenance procedures.

A major pipe break in the main steam valve area above will not cause flooding of the pump room, as discussed above. Minor flooding from any lines in the area is handled by drains in the pump room and the adjacent ventilation room. The elevation of the pump room is higher than the probable maximum flood level.

Safeguards Area (El. 747'0") and Safeguards Area Pipe Tunnel (El. 732'6")

There are no equipment or floor drains in the safeguards area. Major flooding of the safeguards area is prevented in the event of a pressure line rupture by valve closure isolating the affected piping section. The elevation of the safeguards area is higher than the probable maximum flood level.

Safeguards and Main Steam Valve Area Ventilation Rooms (El. 751’0")

Floor and equipment drainage in these ventilation rooms is conveyed to the sump in the pipe tunnel. The floor and equipment drainage system is designed to accommodate minor piping leaks, equipment condensate drainage, or normal maintenance procedures. There are no precautions necessary to prevent flooding in the event of a major pipe rupture in the main steam valve area for the reasons discussed above. The elevation of the ventilation rooms is higher than the probable maximum flood.

Primary Auxiliary Building

Floor and equipment drainage in the primary auxiliary building is conveyed to sumps in the lowest level of the building. The drainage system is designed to accommodate minor piping leaks, equipment drainage, and normal maintenance procedures. With the exception of the charging pump cubicle flood walls, and 14 inch high cubicle hatch curbs, there are no precautions necessary to prevent flooding in the event of a major pipe rupture in any of the pressure piping systems in the auxiliary building because open access doors will permit excess water to flow out of the affected area. With the exception of the lowest level of the auxiliary building, which is not protected against the probable maximum flood (PMF), all other floors are higher than the level of this flood. The only safety-related equipment required for safe shutdown on the lowest level of the primary auxiliary building are the charging pumps. These pumps are individually protected against the probable maximum flood and should not be affected by water accumulation due to pipe break or flood.

Fuel Building

Floor drainage in the fuel building is conveyed to a sump. The drainage system is designed to accommodate minor piping leaks, equipment drainage and normal maintenance procedures. There are no additional precautions necessary to prevent flooding in the event of a major pipe rupture in any of the pressure piping systems in the fuel building. A crack in the hot water lines around the fuel building at El. 767 was considered.

The elevation of the fuel building is higher than the level of the probable maximum flood.

Main Control Room

There are no equipment or floor drains in the main control room. There are no sources of water located above, or within, the main control room that could cause flooding. The elevation of the main control room is above that of the probable maximum flood.

Switchgear, Relay Room and Cable Tray Area in Service Building

There are floor, equipment, and roof drain lines running within the cable tray area. Since these lines are normally dry and nonpressurized (i.e., continuously pitched-gravity), they are not considered as potential sources of flooding.

A 1 inch domestic water line to two battery area safety showers exists within these areas. The common supply line to these showers is equipped with a water flow switch, physically located in the piping system upstream of this area, that annunciates in the control room.

In the event of a break in the domestic water line, water would accumulate only at El. 713 ft-6 inches, since the cable tray mezzanine (El. 725 ft-6 inches) is level and any water at that elevation would flow to El. 713 ft-6 inches via doorways and stairwells at the east and west entrances.

There is a recessed cable space at El. 713 ft-6 inches which has a volume of 875 cu ft. (based on 75 percent net free volume). For flooding purposes, a 75 percent net floor area of 13,550 sq ft. A survey of the area indicates, at minimum, a 2 inch water buildup could be tolerated without any adverse effects to any electrical equipment at El. 713 ft-6 inches. A complete rupture of the 1 inch domestic water line would result in a flow of 66 gpm which would be simultaneously annunciated in the control room. The recessed cable space would be filled in 99 minutes, after which water would accumulate at the rate of 1 inch in 128 minutes at the 713 ft-6 inch elevation, i.e., 3.78 hours from onset of the flood. At 5.92 hours (2 inches above El. 713 ft-6 inches), electrical equipment is still not threatened. The operator would have approximately 3 1/2 hours to isolate the water supply, and safely terminate the flood.

All cables for 4 kV service, 480 V service, control, and instrumentation for both primary and secondary plant use are of the same high quality construction. Each type of cable has been specified for wet and dry locations and will operate satisfactorily if submerged as proven by factory testing. The 4 kV power cable was submerged for a period of 24 hours before testing at the supplier's factory.

The flow switch annunciates on increasing flow at 0.90 gpm. Assuming an "unalarmed pipe crack" discharging at 0.90 gpm would result in a water accumulation of 1 inch in 11 1/2 days at the 713 ft-6 inch elevation. Plant personnel would pass through this area during this time span and detect the leak.

Curbs for minor leakages have been added beneath wall penetrations located below PMF elevations in the cable tray mezzanine area at El. 725 ft-6 inches and in the switchgear area, relay room, and process racks area west of relay room at El. 713 ft-6 inches.

Provisions have been made for using a portable emergency powered electric pump to remove water from the curbed areas. A level instrument is provided in the cable tray mezzanine, switchgear, and process racks curbed areas with a common alarm in the control room.

Battery Rooms

There are no equipment or floor drains in the battery rooms. There are no sources of water located above or within these rooms that could cause flooding. The battery rooms are structurally protected against ingress of water from the probable maximum flood.

Air Conditioning Equipment Room for Control Room

Floor drainage in the control room air conditioning equipment room is conveyed to the turbine building floor drainage system. An automatically primed deep seal running trap in the turbine building serves as an air seal to prevent inleakage of turbine room air to the control room equipment room through the drain line.

The sources of water into the air conditioning equipment room consist of redundant river water headers to the various equipment components. Four floor drains are provided with a 6 inch floor drain return header to remove any leakage that may occur as a result of a pipe rupture.

A level instrument with control room alarm has been added to indicate water accumulation on the floor resulting from a leak in one of the river water lines, assuming a closed valve in the normal drain line. 6 inch curbs will prevent water from flooding adjacent emergency electrical equipment. A 1/2 D x 1/2 T split in one of the 3 inch river water lines results in a leak of about 40 gpm and a control room air conditioning equipment room floor water accumulation of 1 inch every 16 minutes. With the level alarm set at 2 inches, this will allow sufficient time for an operator to stop the leak and open the drain line. Provisions have also been made for the use of an emergency powered portable electric pump to remove water from this room in the event of a mechanical failure of the drain valve which prevents it from being opened.

Since the elevation of the equipment room is below the elevation of the probable maximum flood, a block valve and check valve are provided in the drain line. The block valve will be closed to prevent river water from entering the equipment room during the probable maximum flood.

Diesel Generator Building

Floor drainage in the diesel generator building is conveyed to the yard storm sewer. The drainage system is designed to accommodate minor piping leaks, equipment drainage, and normal maintenance procedures. There are no precautions necessary to prevent flooding in the event of a pipe rupture since there are no "high energy" water lines within the diesel generator building. The river water lines in the diesel generator building have a maximum diameter of 6 inches. A passive failure of the EDG river water lines is limited to 50 gpm. The leak rate for this moderate energy piping is based on maximum pump seal leakage, flange failures, or similar system pressure-boundary violations postulated as a post-DBA passive failure rather than an initiating event. This leak rate can be adequately handled by the installed drainage system. The elevation of the diesel generator building is higher than the probable maximum flood.

River Water Pumps and Engine-Driven Fire Pump-Intake Structure

Each intake structure pump cubicle has an emergency-powered sump pump which is controlled by a float switch. The external flood protection doors leading into each intake structure pump cubicle are normally open with their associated security/fire doors normally closed. The interconnecting flood protection doors that are located between the pump cubicles are normally closed with their seals depressurized, along with their associated security/fire doors normally closed. These door seals will be pressurized in the event of an external flood or for seal testing purposes. This arrangement is designed to protect the interconnecting cubicles from the consequences of a major pipe rubber expansion joint failure.

Waste Gas Storage Area

There are no equipment or floor drains in the waste gas storage area. There are no sources of water located above or connected to the waste gas storage area that could cause flooding. The waste gas storage area is structurally protected against the ingress of water from the probable maximum flood.

Coolant Recovery Tank Structure

There are no equipment or floor drains in the coolant recovery tank structure. There are no sources of water located above or connected to the structure that could cause flooding. The elevation of the coolant recovery tank structure is higher than the level of the probable maximum flood.

Refueling Water Tank Enclosure and Mat

There are no equipment or floor drains in the refueling water tank enclosure. There are no sources of water located above or connected to the enclosure that could cause flooding. The elevation of the refueling water tank enclosure is higher than the level of the probable maximum flood.

Indication of Flooding Conditions

All areas previously listed are not subject to flooding conditions from the probable maximum flood because of elevation or structural provisions. Where small penetration leakage may occur during the probable maximum flood, provisions have been made to collect, indicate, and remove such leakage. Areas that are served by sumps are provided with high sump water level alarms which annunciate in the main control room. Such an alarm would be an indication of possible flooding.

9.7.3 Evaluation

The vent and drain system is sized to handle the maximum amount of liquids and gases expected during station operation.

Sumps located in critical areas are provided with a double pump arrangement. Operation of the pumps is alternated to obtain equal wear. One pump is on automatic service and the other on standby. When the water level in the sump reaches a selected level, one of the pumps starts. If the water level reaches a higher level, the second pump is started. The pump(s) stops

automatically upon emptying the sump. Each sump is also provided with a high level alarm. A containment isolation phase A (CIA) signal closes the containment isolation valves.

The primary drain transfer pumps are full-sized and independently controlled. Two pumps are provided for each tank. One pump is in automatic service and the other on standby. When the water level in the tank reaches a selected high level one of the pumps is started. If the water level reaches a selected higher level, the other pump also starts. The pump(s) stops automatically upon emptying the tank. Each tank is also provided with a high level alarm. A CIA signal closes the containment isolation valves.

During normal unit operation, proper operation of the vent and drain system precludes radiological hazard to the public or operating personnel from the drainage or venting of nuclear plant equipment or system components. All drainage liquid or vent gas which is potentially radioactive is collected and either returned to the reactor coolant system or processed for safe disposal.

All liquid drains originating inside containment are collected in either the pressurizer relief tank, primary drains transfer tank No. 1, or containment sump. This arrangement initially retains in containment all leakage originating from components within containment; thereby, segregating any radiation hazard to operating personnel and the public resulting from the leakage. Subsequent to collection, the leakage is either returned to the reactor coolant system or the liquid waste disposal system via an auxiliary building sump or the liquid waste disposal system via the safeguards building sump (reactor containment tunnel sump) discharge line. Gaseous buildup in the pressurizer relief tank can be vented to the gaseous waste disposal system.

Drainage liquid pumped from the containment sump to an auxiliary building sump still poses no danger to personnel. These sumps are remotely located, fitted with hoods, and subject to sweep gas ventilation.

Sweep gas ventilation affords a positive means of preventing gaseous outleakage from aerated tanks or confined areas. Application of exhaust ventilation to a tank or confined area draws out any gases contained therein and replaces them with outside air drawn in through a vent, "sweeping" this clean air through the ventilated area, and out again to the gaseous waste disposal system. The sweep gas prevents any gases entrained in the liquid from diffusing into the auxiliary building atmosphere. Leakage from certain components in the auxiliary building which are known to pose a radiation hazard is directed to the primary drains transfer tank No. 2 and is subsequently directed to the reactor coolant system or coolant recovery tanks after passing through the Boron Recovery System. Unit tankage which would be normally vented to the atmosphere, but contains potentially radioactive water is subject to sweep gas ventilation, preventing any escape of entrained gases to the atmosphere. The sweep gas is routed to the gaseous waste disposal system.

Other areas in which radioactive leakage or drainage may be generated, are designed with a drainage system with collection sumps. Any drainage collected is pumped to the liquid waste disposal system.

All radioactive and potentially radioactive leakage or drainage is either returned to the reactor coolant system through closed systems or collected in sumps and pumped to the liquid waste disposal system. In either case, the liquid is kept sufficiently segregated from operating personnel and the public so as to pose no radiation hazard. Redundant sump pumps are provided in critical areas to assure reliability of performance and function.

Because leakage and drainage to the auxiliary building sumps from the reactor coolant system and its subsystems have the potential of carrying entrained radioactive gases and hydrogen, these sumps are covered and ventilated with sweep gas which exhausts to the gaseous waste disposal system. In this manner, any entrained gases which may be released from the liquid in the sumps will enter the gaseous waste disposal system and not the auxiliary building atmosphere. Other area sumps do not have this provision since drainage to these sumps originates from systems which are not anticipated to contain entrained radioactive gases or hydrogen.

9.7.4 Tests and Inspections

Formal testing of this system is unnecessary since it is in normal day-to-day operation. Inspection and preventive maintenance are performed in accordance with normal station maintenance procedures.

9.8 COMPRESSED AIR SYSTEMS

9.8.1 Design Bases

The compressed air systems are designed to provide adequate compressed air capacity, of suitable quality and pressure, as required for normal station service and instrumentation. The flow diagrams of the systems are shown on Figure 9.8-1.

Operation of either the station air system or the containment air system is not required for station safety. However, since the systems are necessary for operation, redundancy of compressors and compressor power sources is provided.

No part of any safety-related equipment requires the supply of compressed air for shutdown. Safety-related air-operated valves fail in the correct position for assuring required safety functions. Valves that must respond during a shutdown or after a shutdown to maintain that state are either hand or motor operated.

Air from the station air system, which is outside the containment, can be supplied to components within the containment for containment instrument air and service air requirements. The supply lines are normally open for the containment instrument air supply and normally closed for the service air requirements during plant operation.

Design pressures are dictated by the expected uses of instrument and service air. Design temperatures are those resulting from extreme ambient conditions. Compressed air systems are moderate energy systems as defined in NUREG 0800, Section 3.6, BTP ASB 3-1, Appendix A. The instrument air dew point is to be maintained below the lowest indoor temperature expected at the station location.

9.8.2 Description

Two completely separate compressed air systems are provided: one supplies containment instrument air, instrument air external to the containment, and service air to the entire unit. The second supplies instrument air to the intake structure.

Two 100 percent design capacity electric motor-driven compressors are provided for the station compressed air system. A 100 percent design capacity diesel-driven compressor is provided to supply the instrument air system in case of the loss of both of the electric motor-driven compressors or when demand on the system exceeds the capacity of the electric motor-driven compressors. One 100 percent design capacity compressor is provided for the intake structure instrument air system. Each compressor is furnished with an intake filter and aftercooler. All compressors are of the nonlubricating type to prevent oil contamination of instrument air.

The station air compressors of the station compressed air system are located in the turbine building. Each station air compressor is of the rotary screw type. The standby electric motor-driven station air compressor is arranged to start automatically when the station air receiver outlet header pressure falls to 100 psig. The standby diesel-driven instrument air compressor is arranged to start automatically on low instrument air header pressure and is capable of starting and supplying the instrument air system regardless of the availability of offsite power. Service air is supplied from this header to valved service air hose connections throughout the unit. Instrument air is normally supplied from the service air header through a connection upstream of an automatically operated shutoff valve, and passes through one of the two drying units located downstream of the station air receivers, to the instrument air receivers. The air-operated shutoff valve will automatically trip closed in the event of a service air header leak to prevent further reduction in supply pressure to the instrument air receivers. When the normal electric-motor driven compressors are unavailable or when demand on the system exceeds the capacity of the electric motor-driven compressors, instrument air is supplied from the diesel-driven instrument air compressor through its drying unit and instrument air receiver IA-TK-2 to the instrument air headers.

There are two normal instrument air drying trains preceding instrument air receiver IA-TK-1. Each train consists of a dual tower regenerative desiccant dryer plus the necessary prefilter, afterfilter, and regeneration controls. The equipment is designed for completely automatic, continuous operation. The compressed air flows through one of the two desiccant filled drying towers where the air is dried by adsorbing the water vapor onto the desiccant. While one tower is on stream drying the compressed air, the other is off stream being regenerated. This is done by expanding a portion of the dried air to atmospheric pressure and passing it over the wet desiccant to desorb the water from the desiccant and carry it out of the dryer.

The standby instrument air drying train, associated with the diesel-driven instrument air compressor, preceding instrument air receiver IA-TK-2 consists of a 100 percent design capacity dual tower regenerative desiccant dryer plus the necessary prefilter, afterfilter, and regeneration controls. The equipment is designed for completely automatic, continuous operation. Saturated compressed air enters the dryer and passes over the desiccant where moisture in the air is adsorbed onto the desiccant beads. A portion of the dried air is redirected to the off-line tower to remove the adsorbed moisture during the regeneration process.

The intake structure instrument air compressor is located on the 705 ft elevation of the intake structure outside of the pump cubicles. The compressor is an oil free type compressor. The rated capacity of the compressor shall satisfy the system air flow demand. Moisture is removed from the compressed air by an air cooled aftercooler, with moisture separator. The instrument air is then dried by a twin tower, heatless, desiccant filled, flow regenerative air dryer with a prefilter and afterfilter. The dryer will provide 11 acfm of air at 100 psig and -40°F pressure dewpoint, from air with a relative humidity of 100 percent and a temperature of 100°F.

In addition to the two compressed air systems, adapters are provided on air supply lines to pneumatic valves necessary for safe shutdown. The adapters are provided so that bottled gas can be used as a backup emergency pneumatic source for manual operation of these valves.

9.8.3 Evaluation

The controls of each electric motor-driven air compressor are arranged so that one compressor of the station air system normally operates to supply system demand, while another compressor of the system serves as a standby unit. Each air compressor motor receives power from a different section of the station service bus. Therefore, if one power supply fails, the other compressor of the system is not affected.

The controls of the diesel-driven instrument air compressor are arranged so that the compressor serves as a standby or supplemental unit to the two electric-motor driven station air compressors; however, it supplies only the instrument air system. The diesel-driven instrument air compressor is self-contained and requires no external power supply or cooling water supply to start or operate. The diesel-driven instrument air compressor train auxiliaries and dryer are capable of operating from the nonsafety, onsite diesel generator power supply.

The station service air line penetrating the reactor containment is provided with a manual shutoff valve located outside the containment and a check valve located inside the containment. The station instrument air line penetrating the reactor containment, which supplies containment instrument air, is provided with a safety-related air operated valve located outside the containment. The design of the valve and control circuit ensures that the air operated valve fails closed on loss of air and power and closes upon receipt of a Containment Isolation Phase "B" (CIB) signal. This station instrument air line includes a check valve located inside containment. The air systems are not required for safe shutdown of the plant.

9.8.4 Tests and Inspections

During operation, periodic tests of the station air system are performed simulating low compressed air pressure to ensure proper starting of the standby electric motor-driven air compressor when required. Periodic tests of the diesel-driven instrument air compressor are performed to ensure proper starting and loading when required.

9.9 RIVER WATER SYSTEM

9.9.1 Design Bases

The river water system is designed to achieve the following prime objectives:

1. Normal operation and Unit 1 cool down requirements
2. Post Design Basis Accident (Section 14.3) cooling requirements.

9.9.1.1 Normal and Cool Down Operation

The river water system provides a continuous supply of cooling water to cool the following components during normal station operation:

1. At least one primary plant component cooling water heat exchanger and consequently all the components cooled by the component cooling water system (Section 9.4)
2. At least one charging pump lube oil cooler
3. One control room air conditioning condenser or one control room river water cooling coil (Section 9.13)
4. A backup to the Filter Water System for motor bearing cooling water and pump bearing lubrication water to the river water and raw water pumps.

When the unit is being cooled down, the normal heat removal requirements of the primary plant component cooling water heat exchangers are reduced as the individual components of the component cooling system are removed from service. The component cooling water is then used to remove the residual heat from the reactor through the use of the residual heat removal exchangers (Section 9.3). When the residual heat removal system is put into operation, the heat removal requirements of the river water system are increased above the normal levels then decrease as the rate of decay heat generation decreases.

9.9.1.2 Post Design Basis Accident Operation

In the event of a DBA, the river water system is designed to supply sufficient cooling water to the following components:

1. At least two recirculation spray heat exchangers (Section 6.4)
2. At least one charging pump lube oil cooler (Section 9.1)
3. At least one control room river water cooling coil (Section 9.13)
4. At least one emergency diesel generator cooling system heat exchanger (Section 8.5).

The river water system is designed to handle the above loads at a maximum river water temperature of 90°F at either the minimum river water level or the Probable Maximum Flood of El. 730.0. The minimum possible river elevation of 648.6 ft is applicable for Unit 1 shutdown requirements. The minimum operability level of 654 ft is applicable for normal operation and post DBA cooling requirements.

The river water system is designed as a Seismic Category I system (Appendix B) and is tornado and missile protected (Section 2.7) until the discharge from the various heat exchangers enters the turbine building. The closed fire doors connecting the adjacent compartments minimize the amount of water entering an adjacent compartment. Whatever water enters the compartments by passing around the fire doors will be removed by sump pumps.

The system is engineered and designed so that all components, pumps, and heat exchangers can be individually isolated thus providing for continual operation during equipment repair and maintenance.

9.9.2 Description

The river water system is shown on Figure 9.9-1.

The cooling requirements are achieved by the provision of three river water pumps. Each is able to deliver approximately 9,000 gpm and is designed to supply the quantity of water needed for the essential safety-related cooling requirements for all unit operating conditions.

The river water pumps are a-c motor driven vertical wet pit-type units. They are mounted above and take suction from three separate sections of the screenwell. The pump motors are in cubicles which are protected from flooding by the Probable Maximum Flood. The intakes of the pumps are at El. 640 ft 7 inches which is below the low river water level. The three pump motors are powered from the emergency buses. One pump receives power from one of the emergency buses, the second pump from the other emergency bus. The third pump is not normally connected to either bus, but can be manually connected to either. It cannot be connected to both emergency buses at the same time.

The intake structure is subdivided so that each of the three 100 percent capacity (with respect to their safety function) river water pumps are separated in missile protected and Seismic Category I cubicles. There are two doors in each cubicle - one opening toward the forward portion (north side) of the intake structure and the other providing access between each end compartment and its adjacent interior compartment. The two interior compartments have no door between them. The intake for each of these pumps is independent with separate screen and suction arrangements.

The Seismic Category I intake structure is a structure common to both BVPS-1 and BVPS-2. The River Water pumps (BVPS-1) and the Service Water pumps (BVPS-2) are housed in this structure, are considered essential systems, and are so designed. Both the River Water and Service Water systems are operated completely independent of each other and are designed to meet the single failure criterion. A cross-connect is provided between one of the two river water and one of the two service water discharge headers. This cross-connect is usually inoperable and is isolated from the two headers by two isolation valves. Catastrophic failure of one River Water or Service Water pump can disable the other pump located in the same bay. However, since three 100 percent pumps are provided for the River Water and the Service Water system and there is a cross-connect that can be used, there is no credible way that failure of one system can disable the other. The possibility of other essential and non-essential equipment failure damaging the essential River Water and Service Water piping and pumps is discussed in Section 5.2.6, and the current high-energy pipe heat study and such a failure mode is not considered credible. Refer to Section 1.7 for a discussion of other systems that will be shared by BVPS-1 and BVPS-2.

The pumps are valved to discharge into the two 24 inch headers which transport underground, by physically separate routes, river water from the screenwell to the primary auxiliary building basement. In addition, a Seismic Category I valve pit houses motor-operated valves and piping connections from the alternate intake structure which tie into the main river water headers. For a discussion of the alternate intake structure refer to Section 9.16.

A pipeline is provided to flush stagnant river water piping and perform river water pump flow testing with the recirculation spray coolers isolated. The 16 inch pipeline is connected upstream of the recirculation spray coolers to the 24 inch river water supply header (on both sides of the A header to B header cross-connect valve), and to the 24 inch river water header downstream of the coolers. During normal station operation the cooler bypass flow paths associated with this pipeline are isolated.

During normal station operation, river water is supplied to the components listed in Section 9.9.1.1. On a containment isolation phase B signal, which occurs following a DBA, the flow is diverted from the primary plant component cooling water heat exchangers to the recirculation spray coolers. River water continues to be supplied to the charging pump lube oil coolers and to one of the two 100 percent design capacity control room air conditioning condensers or one of the control room river water cooling coils. The inlet motor-operated valves to the diesel generator cooling system heat exchangers are opened on a safety injection signal, which is the same signal that starts the emergency diesel generators. If a DBA occurs, a safety injection signal would be initiated prior to a containment isolation phase B signal.

Each recirculation spray heat exchanger has a motor-operated valve at its inlet and outlet so that in the main control room it can be isolated from the main header and from the other recirculation spray heat exchangers. The charging pump lube oil coolers, the control room air conditioning condensers, and the control room river water cooling coils have manual valves under administrative control at their inlets and outlets so that they can also be isolated.

On a loss-of-normal station power, river water is supplied to the emergency diesel generator cooling system heat exchangers. River water continues to be supplied to the components listed in Section 9.9.1.1.

The operation of the various valves following a DBA or loss-of-normal station power is summarized in Table 9.9-1.

One of the river water headers can supply water to the suction of the steam generator auxiliary feedpumps (Section 10.3.5). This header is also connected to the discharge of the engine-driven fire pump, providing an alternate water supply to the steam generator auxiliary feedpumps. An alternate source of water is also provided for the containment air recirculation cooling coils by piping from each river water header through motor-operated valves to the cooling coils.

The River Water System additionally has the capability of supplying system components from the "A" header, and returning the water via the "B" header, or vice versa. The return water is then directed through the Auxiliary River Water System (Section 9.16) to the Alternate Intake Structure and ultimately to the river. This allows replacement of the expansion joints, if necessary, in the River Water 30" return header when the plant is in Mode 5.

Corrosion protection for the materials of the river water system is provided in three ways:

1. Chemical injection is used to prevent the growth of algae.

2. Piping around the recirculation spray heat exchanger which does not normally experience flow will have the water suitably inhibited.

3. A check on fluid conditions will be made by taking samples at suitable intervals.

Further, a high flow rate through the recirculation spray heat exchangers has been chosen to minimize the possibility of pitting corrosion occurring in the stainless steel tubes due to deposition of silt.

Chemical injection into the two 24-inch reactor plant river water headers is used to minimize the collection of silt in the system and control macro invertebrate growth. The main 30-inch turbine plant river water header is also equipped for chemical injection to minimize the collection of silt.

Component design data for the river water system is given in Table 9.9-2.

The minimum flow requirements for the various components following a DBA and for other modes of operation of the river water system are given in Table 9.9-3.

9.9.3 Evaluation

Figure 9.9-4 presents a composite diagram of the plant systems used to attain and maintain a normal shutdown condition. Figure 6.1-1 presents a composite diagram of all Engineered Safety Features Systems which are used to attain and maintain a safe shutdown condition following a DBA.

Station Auxiliary Systems heat rates and integrated heat rejected were calculated for both a normal shutdown and a DBA. Station auxiliary heat loads were based on information in Sections 9.4 and 9.9. Minimum safeguards criteria were assumed following the DBA. For the normal shutdown, the component cooling system heat rejected was assumed to remain constant. The actual system heat rejected would decrease as the unit shutdown continued and various pieces of equipment are removed from service.

During the normal shutdown the following pieces of auxiliary equipment were considered as part of the Station Auxiliary Systems heat load:

1. Charging pump lube oil cooler
2. Control room air conditioning condenser or control room river water cooling coil
3. River water and raw water pump cooling water and lube oil cooler
4. Component cooling water systems

Following a DBA, the following pieces of auxiliary equipment are considered as part of the station auxiliary heat load:

1. Charging pump lube oil cooler
2. Control room air conditioning condenser or river water cooling coil
3. Emergency Diesel Generator Cooling System Heat Exchanger.

Recirculation spray heat exchangers are not included since their heat load is included in the DBA heat loads.

For safe shutdown, the ultimate heat sink (Ohio River) must supply only the river water system. The river water pumps are vertical turbine pumps. The lip of the river water pump suction bell is nominally at 641 ft 3 inches with a minimum submergence required of 4 ft, which equates to a river El. 645 ft 3 inches, well below the minimum possible river El. 648.6 ft. At the extreme low water level of El. 648.6, submergence is 7.35 ft which provides margin above the 4 ft minimum submergence.

At the minimum possible river elevation (648.6 ft), the river flow assumes open channel flow characteristics at the rate of 800 cfs (360,000 gpm). River water system flow requirement to maintain safe shutdown is approximately 17 cfs (7.5 thousand gpm) or 2.1 percent of flow available.

At minimum flow conditions and at minimum river level, the Ohio River can fully meet the shutdown cooling water requirements of BVPS-1. Further, assuming that BVPS-2 requires a maximum of 7500 gpm of cooling water, less than 5 percent of available river flow would be required for both nuclear power stations.

The Ohio River, the Ultimate Heat Sink for the BVPS-1, meets the criteria of Safety Guide 27 as discussed in Section 1.3.3.27.

The minimum River Water System flow shown for a DBA in Table 9.9-3 is calculated on the basis of a minimum operability level of El. 654 and a maximum river water temperature of 90°F.

The River Water System bounding design criteria which has been determined for an extreme low river water level is:

1. A design basis accident occurs at Beaver Valley Power Station Unit No. 1 with the Ohio River at elevation of 654 ft mean sea level at the Intake Structure, with an extremely low river water flow rate of 800 cfs.

2. A coincident shutdown of Beaver Valley Power Station Unit No. 2 from full power operation.

3. Ohio River water temperature of 90°F.

4. A single failure in either a) an onsite system or b) in an offsite manmade structure. The limiting design basis offsite single failure is the loss of one lock or tainter gate in the downstream New Cumberland Dam. This is a passive failure as defined in Section 1.3.1 which culminates in an extreme low river water level of 648.6 ft mean sea level in a time frame as shown in Attachment 2.3C.

The cooling requirements to satisfy the above scenario bounds the cooling requirements for the postulated scenario which involves the failure of one tainter gate in the downstream dam, a subsequent normal shutdown of both units prior to reaching 650 ft river level, and a single failure in an onsite system.

A maximum solid blockage limit of 22 inches of silt has been established for each Intake Structure bay to ensure that sufficient ultimate heat sink cooling water remains available given the above design basis criteria. Silt, in this application, is defined as any obstruction which completely blocks flow from the Ohio River to the pump suction in an Intake Structure bay. This silt limit also requires that flow taken out of a single Intake Structure bay by station pumps be limited to a maximum of 7500 gpm whenever the Ohio River Water level is less than 650 ft mean sea level to ensure sufficient water level remains in the bay for pump NPSH/submergence requirements. 7500 gpm is sufficient to meet Beaver Valley Power Station Unit No. 1 post-DBA cooling requirements in the long term. There is no limit for flow out of an Intake Structure bay with river water level greater than 650 ft mean sea level.

The entire system is designed as a Seismic Category I system (refer to Appendix B) and is tornado and missile protected up to the entrance into the turbine building. If this piping in the turbine building were lost, the river water system would continue to operate. The turbine building basement would be filled with river water up to El. 708 ft. Water would then spill out the tube withdrawal opening in the north wall of the building and flow into the north yard and downhill to the river.

River water to all equipment is supplied via one 24-inch river water header. A completely redundant 24-inch river water header is provided to ensure a water supply in the event of a pipe rupture in the operating header. Duplication in the river water system of items such as pumps, the screenwell structure cubicles, the main pipe headers from the screenwell, and essential valves assures that the system will meet the single failure criterion (Section 1.3.1). All components, pumps and heat exchangers can be individually isolated, thus providing for continual operation during equipment repair and maintenance. Because of the duplication in the system, any component can be serviced while the system maintains sufficient cooling capacity to keep the unit in a safe condition. The alternate intake structure (Section 9.16) also ensures a continuous source of cooling capability in addition to the above system. Any single failure of the system can be detected by various instrumentation, such as temperature sensors, flow meters, etc., located throughout the system. Any failure can be isolated by isolating that portion of the failed system or by isolating the entire header and operating on the redundant header.

Only one of the three river water pumps is required to provide the cooling for the minimum number of components required following a DBA. These components are listed in Table 9.9-3. Use of one pump is based on the required heat duty of the recirculation spray coolers following a DBA, as discussed in Section 6. Assuming no outside source of a-c power, the emergency diesel generators (Section 8.5) are capable of accepting essential post accident load in sufficient time so that power is supplied to the river water pumps before any hazardous condition can develop. Motor-operated valves have been specified with no special fast opening requirements since there will be a delay in the operation of the recirculation spray pumps which is greater than the normal valve operating time.

Duplicate valves are used for all essential functions so that the failure of a single valve to operate correctly will not prevent the river water system from supplying the minimum flow requirements for its various modes of operation. For example, if one of the normally closed automatic motor-operated valves on the river water headers to the recirculation coolers should fail to open, the redundant parallel valve opening would supply water to the coolers. By appropriate valve operation, all the normally operating components are capable of being supplied through either redundant river water header.

All piping and equipment movements due to thermal or seismic effects have been analyzed and the piping and equipment designed to ensure that no damaging forces are exerted on piping or equipment nozzles.

Because the seismic movements of the intake structure and other buildings may differ from the seismic movements of the earth in which the pipe is buried, the most critical points are where the pipes pass into the structures. The pipes are provided with sleeves or enclosures which extend for a short distance out from the structures. The enclosures decouple the pipe from the soil and permit the pipe to accommodate differential movements between the soil and the structures. Flexible joints are used within the enclosures, as necessary, to limit stresses in pipe and to reduce length of enclosure. The radial clearance between pipe and enclosure is ample to accommodate both settlement and seismic motion.

Analysis of the river water system indicates that the system will function properly in the entire range from the extreme low water level in the Ohio River of 648.6 ft to the Probable Maximum Flood level of 730.0 ft.

9.9.4 Tests and Inspections

The major portion of the river water system is in continual use and requires no periodic testing.

The river water sides of the recirculation spray heat exchangers can be flow-tested in pairs, either when the unit is shut down for maintenance or when only one river water pump is shut down for unit operation. The purpose of the test is to assure adequate flow of river water to the coolers as specified in UFSAR Tables 9.9-3 and 14.3-5. The test is performed on an 18 month frequency (±25%).[2] By suitable valve operation, the output of one river water pump can be diverted to the recirculation spray coolers through one of the 24 inch river water lines, the other line being used to supply the primary plant component cooling water heat exchangers. Adequate flow in the river water lines supplying the coolers is verified using control room instrumentation.

The emergency diesel generators are tested on a regular basis. The cooling system must operate while the diesels are running so the regular test of the diesels serves as a flow test of the cooling system. Satisfactory operation of the diesel generators confirms the adequate functioning of the river water diesel generator cooling system.

The system is hydrostatically tested prior to unit startup and the equipment is accessible for periodic visual inspections during operation.

The motor-operated valves in the lines to and from the recirculation spray coolers are tested periodically to ensure satisfactory operation. They are normally set in the safe open position with the header valves in the 24 inch river water lines closed.

The recirculation spray heat exchanger header valves are checked during unit startup and periodically afterward to verify proper operation.

Provisions for preoperational testing of the sliding flood doors on the intake structure cubicles are as follows:

To verify the proper functioning of the cubicle flood doors, a test rig (plate) is provided. The test rig consists of 1/4 inch thick steel plate, with adequate stiffness, which is bolted to temporary angle frame attached to the concrete. The plate is sealed with a neoprene rubber or equivalent[(1)] gasket material. When installed, the test rig provides a void space on the cubicle side of the flood door. This void space is then filled with water through a test connection located near the bottom of the test rig. A vent connection is provided at the top of the test rig for venting of air when filling the void. When the void is completely filled and vented, a standpipe is attached to the vent connection. The standpipe (hose) is led above the cubicle roof (El. 730 ft) and filled with water to provide a static head pressure on the flood door equivalent to a PMF. The outside of the flood door is then observed for leakage.

Provisions for periodic testing of the flood doors will be as follows:

1. Inspection and cleaning around the frame.

2. Testing of seals with door in closed position to ensure that the seal is capable of maintaining desired air pressure. This will be done once every operating cycle by an air test. The seals will be inflated to 50 psig then isolated from the fill vessel. The seal pressure will be checked over a period of 100 hours (duration of PMF) for pressure loss. If after 100 hours the pressure does not fall below 40 psig the seal will be considered to be in good condition. A loss of pressure from 50 to 40 psig within 100 hours satisfies the recommendations of the manufacturer for maximum acceptable air loss. If the seal does not maintain at least 40 psig at the end of 100 hours or if it shows marked wear or surface peeling and cracking, the seal will be replaced.

3. Visual inspection of seals to determine when and if the seals require replacing.

The following information concerning the air supply system is provided as backup to the above discussion (item 2) of seal testing. The flood seal pressurization system provided for each seal is shown schematically in Figure 9.9-7.

The air vessels provided, one for each door seal, have sufficient capacity to inflate and maintain the flood door seals at the required pressure for the duration of the Probable Maximum Flood. The air vessels will be recharged when, due to leakage or as a result of testing during normal plant operation, the air pressure falls to 100 psig. Considering the design of the sealing system and the design margins incorporated into the system, it is not anticipated that any bottle replacement would be necessary during the Probable Maximum Flood.
Relief protection, set at 85 psig, is provided to protect against over-pressurization of the seals, whose maximum allowable pressure is 100 psig.

REFERENCES FOR SECTION 9.9

1. The term "equivalent" is described in UFSAR Section 1.8.2, "Equivalent Materials."

2. Commitment for river water flow rate testing of recirculation spray heat exchangers, BVPS Unit 1 License Amendment 252.

9.10 FIRE PROTECTION

The fire protection program is based on the NRC requirements and guidelines, Nuclear Electric Insurance Limited (NEIL) Property Loss Prevention Standards and related industry standards. With regard to NRC criteria, the fire protection program meets the requirements of 10 CFR 50.48(c), which endorses, with exceptions, the National Fire Protection Association's (NFPA) 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants – 2001 Edition. BVPS-1 has further used the guidance of NEI 04-02, "Guidance for Implementing a Risk-Informed, Performance-Based Fire Protection Program under 10 CFR 50.48(c)" as endorsed by Regulatory Guide 1.205, "Risk-Informed, Performance-Based Fire Protection for Existing Light-Water Nuclear Power Plants."

Adoption of NFPA 805, "Performance-Based Standard for Fire Protection for Light Water Reactor Electric Generating Plants", 2001 Edition in accordance with 10 CFR 50.48(c) serves as the method of satisfying 10 CFR 50.48(a) and General Design Criterion 3. Prior to adoption of NFPA 805, General Design Criterion 3, "Fire Protection" of Appendix A, "General Design Criteria for Nuclear Power Plants," to 10 CFR Part 50, "Licensing of Production and Utilization Facilities," was followed in the design of safety and non-safety related structures, systems, and components, as required by 10 CFR 50.48(a).

NFPA 805 does not supersede the requirements of GDC 3, 10 CFR 50.48(a), or 10 CFR 50.48(f). Those regulatory requirements continue to apply. However, under NFPA 805, the means by which GDC 3 or 10 CFR 50.48(a) requirements are met may be different than under 10 CFR 50.48(b). Specifically, whereas GDC 3 refers to SSCs important to safety, NFPA 805 identifies fire protection systems and features required to meet the Chapter 1 performance criteria through the methodology in Chapter 4 of NFPA 805. Also, under NFPA 805, the 10 CFR 50.48(a)(2)(iii) requirement to limit fire damage to SSCs important to safety so that the capability to safely shut down the plant is satisfied by meeting the performance criteria in Section 1.5.1 of NFPA 805.

A safety evaluation was issued on January 22, 2018 by the NRC, that transitioned the existing fire protection program to a risk-informed, performance-based program based on NFPA 805, in accordance with 10 CFR 50.48(c).

9.10.1 Design Basis Summary

9.10.1.1 Defense-in-Depth

The fire protection program is focused on protecting the safety of the public, the environment, and plant personnel from a plant fire and its potential effect on safe reactor operations. The fire protection program is based on the concept of defense-in-depth. Defense-in-depth shall be achieved when an adequate balance of each of the following elements is provided:

1. Preventing fires from starting,

2. Rapidly detecting fires and controlling and extinguishing promptly those fires that do occur, thereby limiting fire damage,

3. Providing an adequate level of fire protection for structures, systems, and components important to safety, so that a fire that is not promptly extinguished will not prevent essential plant safety functions from being performed.

9.10.1.2 NFPA 805 Performance Criteria

The design basis for the fire protection program is based on the following nuclear safety and radiological release performance criteria contained in Section 1.5 of NFPA 805:

1. Nuclear Safety Performance Criteria. Fire protection features shall be capable of providing reasonable assurance that, in the event of a fire, the plant is not placed in an unrecoverable condition. To demonstrate this, the following performance criteria shall be met.

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

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

    c. Decay Heat Removal. Decay heat removal shall be capable of removing sufficient heat from the reactor core or spent fuel such that fuel is maintained in a safe and stable condition.

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

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

2. Radioactive Release Performance Criteria. Radiation release to any unrestricted area due to the direct effects of fire suppression activities (but not involving fuel damage) shall be as low as reasonably achievable and shall not exceed applicable 10 CFR, Part 20, limits.

Chapter 2 of NFPA 805 establishes the process for demonstrating compliance with NFPA 805.

Chapter 3 of NFPA 805 contains the fundamental elements of the fire protection program and specifies the minimum design requirements for fire protection systems and features.

Chapter 4 of NFPA 805 establishes the methodology to determine the fire protection systems and features required to achieve the nuclear safety performance criteria outlined above. The methodology shall be permitted to be either deterministic or performance-based. Deterministic requirements shall be “deemed to satisfy” the performance criteria, defense-in-depth, and safety

margin and require no further engineering analysis. Once a determination has been made that a fire protection system or feature is required to achieve the nuclear safety performance criteria of Section 1.5, its design and qualification shall meet the applicable requirement of Chapter 3.

9.10.1.3 Codes of Record

The NFPA codes, standards and guidelines used for the design and installation of plant fire protection systems, as well as specific applications and evaluations of codes are described in BVPS Fire Safety Analysis (FSA), Volume 3, Enclosure 2, "NFPA 805 Codes of Record".

9.10.2 Systems Description

9.10.2.1 Required Systems

Nuclear Safety Capability Systems, Equipment, and Cables

Section 2.4.2 of NFPA 805 defines the methodology for performing the nuclear safety capability assessment. The equipment required for the nuclear safety capability assessment is identified in FSA Volume 3, Enclosure 4, "Nuclear Safety Capability Assessment Equipment List". Cables are identified in Enclosure 4, Reference 5.6, "Beaver Valley Unit 1 NFPA 805 Safe Shutdown, Non-Power Operation, and Fire PRA Cable Selection."

Fire Protection Systems and Features

Chapter 3 of NFPA 805 contains the fundamental elements of the fire protection program and specifies the minimum design requirements for fire protection systems and features. Compliance with Chapter 3 is summarized in FSA Volume 1, Section 14.0, "Fundamental Fire Protection Program and Design Elements". Section 14.2, "NFPA 805 Chapter 3 Compliance Discussion Locations" identifies documents that provide supporting details.

Chapter 4 of NFPA 805 establishes the methodology and criteria to determine the fire protection systems and features required to achieve the nuclear safety performance criteria of Section 1.5 of NFPA 805. These fire protection systems and features shall meet the applicable requirements of NFPA 805 Chapter 3. These fire protection systems and features are documented in the FSA Volume 3, Enclosure 10, Table 4-3.

Radioactive Release

The NEI 04-02 guidance defines the analysis of the radioactive release performance criteria as evaluation of the direct effects of fire suppression activities. The BVPS evaluation did not credit any installed SSC's and relies strictly on the fire brigade's pre-fire plans and the fire brigade's mitigation instructions for guidance in monitoring and controlling potentially contaminated releases. FSA Volume 1, Section 10.0, "Radioactive Release" describes the evaluation methodology, summarizes the results and provides references to more detailed aspects of the evaluation.

9.10.2.2 Definition of "Power Block" Structures

Where used in NFPA 805 Chapter 3 the terms "Power Block" and "Plant" refer to structures that have equipment required for nuclear plant operations. For the purposes of establishing the structures included in the fire protection program in accordance with 10 CFR 50.48(c) and

NFPA 805, the plant structures listed in FSA Volume 3, Enclosure 6 are considered to be part of the “power block”.

9.10.3 Safety Evaluation

The FSA documents the achievement of the nuclear safety and radioactive release performance criteria of NFPA 805 as required by 10 CFR 50.48(c). This document fulfills the requirements of Section 2.7.1.2 “Fire Protection Program Design Basis Document” of NFPA 805. The document contains the following:

1. Identification of significant fire hazards in the fire area. This is based on NFPA 805 approach to analyze the plant from an ignition source and fuel package perspective.

2. Summary of the Nuclear Safety Capability Assessment (at power and non-power) compliance strategies.

    a. Deterministic compliance strategies

    b. Performance-based compliance strategies (including defense-in-depth and safety margin)

3. Summary of the Non-Power Operations Modes compliance strategies.

4. Summary of the Radioactive Release compliance strategies.

5. Summary of the Fire Probabilistic Risk Assessments.

6. Key analysis assumptions to be included in the NFPA 805 monitoring program.

9.10.4 Fire Protection Program Documentation, Configuration Control and Quality Assurance

In accordance with Chapter 3 of NFPA 805 a fire protection plan documented in Procedure 1/2-ADM-1900 defines the management policy and program direction and defines the responsibilities of those individuals responsible for the plan’s implementation.

Procedure 1/2-ADM-1900:

1. Designates the senior management position with immediate authority and responsibility for the fire protection program.

2. Designates a position responsible for the daily administration and coordination of the fire protection program and its implementation.

3. Defines the fire protection interfaces with other organizations and assigns responsibilities for the coordination of activities. In addition, Procedure 1/2-ADM-1900 identifies the various plant positions having the authority for implementing the various areas of the fire protection program.

4. Identifies the appropriate authority having jurisdiction for the various areas of the fire protection program.

5. Identifies the procedures established for the implementation of the fire protection program, including the post-transition change process and the fire protection monitoring program.

BVPS administrative procedures identify the qualifications required for various fire protection program personnel.

The Augmented Quality Assurance Program, Appendix C, Fire Protection identifies the quality requirements of Chapter 2 of NFPA 805.

Detailed compliance with the programmatic requirements of Chapters 2 and 3 of NFPA 805 are contained in lower tier procedures and documents identified by Procedure 1/2-ADM-1900.

REFERENCES FOR SECTION 9.10

1. Deleted by Revision 32.
2. Deleted by Revision 32.
3. 10 CFR 50.48 - Fire Protection for Operating Nuclear Power Plants.
4. Deleted by Revision 32.
5. Deleted by Revision 32.
6. Deleted by Revision 32.
7. Deleted by Revision 32.
8. Deleted by Revision 32.
9. Deleted by Revision 32.
10. Deleted by Revision 32.
11. Amendment No. 301 to the Beaver Valley Power Station Unit No. 1 Operating License DPR-66.

9.11 WATER SUPPLY AND TREATMENT SYSTEMS

9.11.1 Design Basis

The water supply and treatment systems are designed to achieve the following objectives:

1. Clarify and filter Ohio River water.
2. Demineralize a portion of the filtered water.
3. Produce demineralized water containing less than 0.04 ppm dissolved solids and less than 0.01 ppm silica.
4. Provide sufficient storage of filtered and demineralized water.
5. Neutralize wastes to produce an effluent having a pH of 6.0 to 9.0 before discharge to the cooling tower blowdown stream.

The system is designed for automatic or manual control.

As operation of the water supply and treatment system is not necessary for safety. Redundancy of all equipment has not been provided.

9.11.2 Description

Primary Treatment Water System

River water, supplied by the raw water pumps to this system, is pumped by the water treating supply pumps to the clarifier.

From the clarifier the water flows by gravity to a bank of gravity filters. Flow from the filters then goes to the softener pump suction tanks for supply to the makeup demineralizer water system.

Makeup Demineralizer Water System

Water for the makeup demineralizer water system is pumped from the demineralizer pump suction tank to a vendor provided skid for processing. The water then flows to the turbine plant demineralized water storage tanks, the primary plant demineralized water storage tank or the two primary water storage tanks. The turbine plant demineralized water storage tanks supply makeup to the main condenser as described in Section 10.3.5. The primary plant demineralized water storage tank supplies water to the steam generator auxiliary feedpumps, as described in Section 10.3.5. The primary water storage tanks are described in Section 9.2.

The makeup demineralizer water system is designed to produce reactor-grade demineralized water. The demineralized water contains a total dissolved-solids content of less than 0.04 ppm and a silica concentration of less than 0.01 ppm.

Station Domestic Water System

Water for the station domestic water distribution system is provided from the Borough of Midland, Pennsylvania water system. There are no interconnections between the station domestic water system and systems having the potential for containing radioactive material.

Waste Neutralization

Wastes produced during the regeneration of the demineralizer system are discharged to the wastes sump. The wastes are monitored for pH and adjusted to achieve a pH in the range of 6.0 to 9.0 before discharge to the cooling tower blowdown.

9.11.3 Evaluation

The makeup demineralizer water system, along with the demineralized and primary water storage tanks, is sufficient to supply station makeup.

The Borough of Midland water supply to the station is sufficient to meet expected domestic water requirements.

Earthquake and tornado protection is provided only for the primary plant demineralized water tank, since this tank is required to bring the reactor to a safe shutdown. Refer to Section 10.3.5 for details.

The water treatment systems are not required for safe shutdown of the reactor.

9.11.4 Tests and Inspections

The water supply and treatment systems are in continual use and thus, do not require periodic testing to ensure operability.

All system equipment is tested for leakage and proper automatic control action prior to system operation. The conductivity of the demineralizer effluent is continuously monitored during operation.

9.12 FUEL HANDLING SYSTEM

The fuel handling system provides a safe, effective means of transporting and handling fuel from the time it reaches the station in an unirradiated condition until it leaves the station after post-irradiation cooling.

The system is designed to minimize the possibility of mishandling or of malfunctions that could cause fuel damage and potential fission product release.

The fuel handling system consists basically of:

1. The refueling cavity and fuel transfer canal which are flooded only during station shutdown for refueling

2. The spent fuel pool, which is kept full of water and is always accessible to operating personnel

3. The transfer tube, which connects the above two areas, fuel transfer system, RCC changing fixture, manipulator crane, new fuel elevator and other equipment used to manipulate the fuel assemblies.

9.12.1 Design Bases

9.12.1.1 Prevention of Fuel Storage Criticality

The new and spent fuel storage racks are administratively controlled so that assemblies are not inserted in other than the prescribed locations within the racks. New fuel assemblies are stored dry in a steel and concrete structure within the fuel building. The new fuel storage racks consist of a stainless steel support structure into which 70 stainless steel fuel guide assemblies are bolted in 14 parallel rows of five fuel guide assemblies each. New fuel assemblies are stored vertically, with a minimum center to center spacing of 21 inches. This will maintain the fuel in a subcritical condition with the effective multiplication factor, K less than 0.95, for both the full density (water at 68°F and 1 gm/cm^3) and low density (0.045 gm/cm^3) optimum moderation conditions. The spent fuel storage pool can accommodate 1622 assemblies (plus 2 assemblies in the failed fuel cannisters and any assemblies in the shipping cask area). Borated water is used to fill the spent fuel pool at a concentration to match that used in the refueling cavity and fuel transfer canal during refueling operations. The spent fuel is stored in a vertical array to maintain $k_{eff} \leq 0.95$ under normal storage conditions even if unborated water is used to fill the pool. The following features assure subcriticality:

1. Sufficient center-to-center spacing.

2. Installed neutron absorbing material (Boral) in rack walls (spent fuel racks only).

3. Administrative controls of storage configurations.

The continued presence of neutron absorbing material is ensured by a surveillance program. The program exposes neutron absorbing material samples to the same environment as the installed material. Samples are periodically retrieved for evaluation of material characteristics.

Limitations on the placement of fuel in the spent fuel pool are discussed in Sections 3.3.2.7 and 9.12.2.2.

9.12.1.2 Fuel Storage Decay Heat

The fuel pool cooling system which removes the spent fuel decay heat is described in detail in Section 9.5.

9.12.1.3 Fuel Building Radiation Shielding

The shielding provided in the fuel building and the resulting maximum doses outside of this building are discussed in Section 11.3.

9.12.1.4 Protection Against Radioactivity Release from the Fuel Building

Protection against radioactivity release from the fuel building is provided by the supplementary leak collection and release system described in Section 6.6. This system maintains the fuel building at subatmospheric pressure and ensures that any significant gaseous activity is filtered and released from the elevated release point. However, the DBA dose consequence analyses do not credit the SLCRS filters.

Because of the high radiation levels of the spent fuel when it first arrives in the fuel building and is placed in the spent fuel storage racks, a delay period is provided for decay of the fuel before there is any further movement of it in the fuel building.

9.12.1.5 Radiation Monitoring of the Fuel Building

The radiation monitors which protect the fuel building are described in Section 11.3.

9.12.2 System Design and Operation

9.12.2.1 System Description

The fuel handling system is shown in Figures 5.1-5 and 9.12-1.

The fuel handling system consists of the equipment needed for the refueling operation on the reactor core. Basically this equipment is comprised of cranes, handling equipment, and a fuel transfer system. The structures associated with the fuel handling equipment are the refueling cavity, the fuel transfer canal, the spent fuel pool, and the new fuel storage area.

New fuel assemblies received are removed one at a time from the shipping container and stored in the new fuel storage racks in the fuel storage area.

New fuel is delivered to the reactor by placing a fuel assembly into the new fuel elevator, lowering it into the spent fuel pool, and taking it through the fuel transfer system.

The reactor is refueled with fuel handling equipment designed to handle the spent fuel under water from the time it leaves the reactor vessel until it is placed in a cask for on-site dry storage or shipment from the site. Underwater transfer of spent fuel provides an effective, safe, and transparent radiation shield, as well as a reliable cooling medium for removal of decay heat. Boric acid is added to the water to ensure subcritical conditions.

The associated fuel handling structures may be generally divided into three areas:

1. The refueling cavity and fuel transfer canal which are flooded only during unit shutdown for refueling
2. The spent fuel pool which is kept full of water is always accessible to operating personnel
3. The new fuel storage area which is separate and protected for dry storage.

The fuel transfer canal and the spent fuel pool are connected by a fuel transfer tube. This tube is fitted with a blind flange on the canal end and a gate valve on the spent fuel pool end. The blind flange is in place except during refueling to ensure containment integrity. Fuel is carried through the tube on an underwater transfer car.

Fuel is moved between the reactor vessel and the fuel transfer canal by the manipulator crane. A RCC changing fixture is located on the fuel transfer canal wall for transferring control elements from one fuel assembly to another.

The upender at either end of the fuel transfer tube is used to pivot a fuel assembly to the horizontal position for passage through the transfer tube. After the transfer car transports the fuel assembly through the transfer tube, the upender at that end of the tube pivots the assembly to a vertical position so that it can be lifted out of the fuel container.

In the spent fuel pool, fuel assemblies are moved about by the fuel handling platform equipped with two hoists. The west hoist has a 10 ton block and either a 2 ton or 10 ton hook. The west hoist is rated for 2 tons regardless of which hook block is used. The 2 ton hook increases lift height for loading spent fuel assemblies into the dry fuel storage cask in the cask pit. The east hoist has either a 10 ton block and hook or a 2 ton block and hook. The east hoist is rated for 7,800 lbs maximum if 10 ton hook and block is installed. The 2 ton block and hook provides the ability to access the outer row of the spent fuel racks along the south side of the pool. When lifting spent fuel assemblies, the hoist uses a long-handled tool to ensure that sufficient radiation shielding is maintained. A shorter tool is used with a 10 ton hook and block to handle new fuel, but the new fuel elevator must be used to lower the assembly to a depth at which the west hoist, using the long-handled tool, can place the new assembly into the fuel transfer container in the upending device.

Decay heat, generated by the spent fuel assemblies in the spent fuel pool is removed by the spent fuel pool cooling system. After a sufficient decay period, the fuel may be removed from the racks and loaded into a shipping cask for removal from the site.

9.12.2.2 Fuel Handling Structures Refueling Cavity

The refueling cavity (Figures 5.1-5 and 9.12-1) is a Seismic Category I reinforced concrete structure lined with stainless steel. During refueling operations only, this cavity is filled with borated water to form a pool above the reactor vessel. The cavity is filled to a depth that limits the radiation at the top surface of the water to 50 mrem per hour during the brief periods while a spent fuel assembly is being handled and is at a maximum height within the cavity. This occurs while the assembly is moved by the manipulator crane into the transfer mechanism for removal from the containment structure.

The refueling cavity is separated into two (2) sections by a permanent five (5) foot high, stainless steel cofferdam across the mouth of the fuel transfer canal at elevation 738'10 1/4". When the pool's surface is above elevation 743'10 1/4", the pool is continuous throughout the cavity. For pool levels below elevation 743'10 1/4", the dam serves to create two (2) separate

pools of approximately 4,750 cubic ft. of water on the reactor's side of the dam, and 2,640 cubic ft. on the transfer canal side.[(1)]

To prevent leakage of refueling water into the lower portion of the containment, a reactor cavity water seal is provided. During the refueling operation, approximately twenty-seven (27) feet of water are on top of the reactor cavity water seal.

The seal consists of a stainless steel membrane (water seal plate) supported by a structural stainless steel frame. The frame members extend from the reactor vessel refueling seal ledge to the refueling cavity floor. The stainless steel membrane is seal welded around the circumference of the reactor vessel refueling seal ledge and the refueling floor embedment ring to prevent leakage and reduce the possibility of a complete seal failure when the refueling cavity is flooded. Analysis demonstrates that the support structure will adequately support itself, the sealing membrane and the water required for refueling.

The reactor cavity water seal is a Seismic Category 1 component and is designed to remain intact during a seismic event while flooded. It is also designed for the cyclic loading during heatup and cooldown cycles. Thermal growth does not affect the integrity of the support structure because the structure is not welded to either the seal ledge or to the embedment ring and is free to slide.

For access below the reactor cavity water seal, seven (7) access holes are provided. The access cover plates are installed and sealed using O-rings prior to flooding for refueling. These covers are removed to allow an airflow path for cooling the reactor vessel cavity during plant operation.

The amount of water in the refueling cavity subject to loss due to the seal's failure is approximately 33,100 cubic ft. Only water above and on the reactor's side of the cofferdam would be lost. Loss of refueling cavity water at a rate above the capability of the pumps and ion exchanger to clean up fluid from the sump and return it to the refueling cavity is considered an unlikely event.

In the unlikely event there is a total seal failure, no seal remaining at all, the water leaking through the voids would enter the instrumentation tunnel beneath the reactor and water would flow through the opening in the reactor cavity wall into the containment and there it would rise approximately 3 feet-8 inches above El. 692 ft-11 inches.

There are two reactor containment sump pumps at elevation 692 ft- 11 inches with a capacity of 25 gpm each. One pump starts automatically and the other has manual start and stop control. The two pumps and the 10 gpm sump pump located within the in-core instrumentation access tunnel are ineffective under these conditions. It would take 82.5 hours for the two 25 gpm pumps to remove the water. The units are able to operate in a submerged condition.

If the gate valve of the fuel transfer tube is open and remains open in conjunction with the seal leaking, then the volume of water in the spent fuel pool above elevation 743'10 1/4" (dam top) will flow into the containment via the opening in the reactor cavity wall and fuel transfer tube. The water level in the bottom of the containment will then rise to approximately 7 ft-9 inch maximum above El. 692 ft-11 inches.

The position of the gate valve in the fuel transfer tube is monitored at the transfer system control panel. A green interlock permissive light indicates that the gate valve is open.

All electrical equipment, open motors, switches and connectors are located above the critical flood level. For the postulated failure of the reactor cavity water seal, the lowest piece of safety-related RHR equipment required for refueling operations is located above the maximum possible water level of approximately El. 701 ft 0 inches.

The refueling cavity also provides storage space for the reactor upper internals, the rod cluster control assembly drive shafts, guide tube, miscellaneous refueling tools, and the reactor lower internals.

Fuel Transfer Canal

The fuel transfer canal (Figures 5.1-5 and 9.12-1) is an extension of the refueling cavity reaching to the containment end of the transfer tube which is about 3 ft from the containment wall. The fuel transfer canal is formed by two concrete shield walls that extend upward to the same elevation as the top of the refueling cavity. The walls and floor are lined with stainless steel. The floor of the canal is at a lower elevation than the refueling cavity to provide the greater depth required by the fuel transfer system upending device and the rod cluster control (RCC) assembly changing fixture which is located in an alcove on the side of the canal.

The outer end of the canal is a concrete shield wall that extends up to the same elevation as the top of the refueling cavity. The transfer tube penetrates this end wall. Radiation streaming through the tube is prevented by a shield ring between the containment liner and the end of the canal.

The inner end of the canal (towards the reactor) stops at the cofferdam. In the event of reactor cavity seal leakage, the cofferdam assures that water will remain at elevation 743'10 1/4" over the transfer canal so that fuel elements in transport can be safely stored until the pool can be refilled. The cofferdam is seismically designed as is a cover over the canal's floor drain.

Spent Fuel Pool

The spent fuel pool is designed to hold spent fuel assemblies and rod cluster control assemblies in underwater storage for a suitable decay period after their removal from the reactor. The structure is a seismic Category I heavy walled, reinforced concrete pool, located outside the containment structure, with the pool partially below ground grade. The interior of the pool is lined with a stainless steel plate. The pool is designed to accommodate a total of 1622 fuel assemblies.

Space is provided in the pool for the spent fuel storage racks, one dry storage transfer cask or one offsite shipping cask, the new fuel elevator, and fuel transfer and upending equipment. The spent fuel pool is divided into three areas:

1. The fuel transfer mechanism area.
2. The spent fuel storage area.
3. The spent fuel cask laydown area.

The fuel transfer mechanism area can be dewatered without emptying the entire pool. A concrete wall separates the fuel transfer mechanism area from the spent fuel storage area. The top of the wall is level with the top of the pool. By insertion of a gate dam at the open end of the transfer mechanism area, this area can be isolated from the spent fuel storage area.

A concrete wall also separates the spent fuel cask laydown area from the spent fuel storage area. The only penetration of the wall is a 24 inch slot to allow for underwater passage of spent fuel elements into the laydown area. The slot is entirely above the storage level of the spent fuel elements in the fuel pool.

The 13 stainless steel storage racks are of 2 separate designs, Region 1 and Region 2. Region 3 is the administratively controlled peripheral cells of Region 2. The racks are free standing on stainless steel bearing pads which distribute the rack load through the pool liner to the concrete floor. Fuel assemblies are placed in vertical cells within the racks, continuously grouped in parallel rows about 10.8 inches on center for Region 1 and 9.0 inches on center for Region 2. The rack is so arranged that the spacing between fuel assemblies cannot be less than that prescribed. The control rod clusters and burnable poison rods may be stored in the fuel assemblies.

Fuel storage in the spent fuel pool is segregated into three areas (Regions 1, 2 & 3). Region 1 will provide for storage of fuel with enrichment up to 5.0 weight percent. Criticality in Region 1 is prevented by spacing and the use of Boral in the storage location walls. Regions 2 and 3 provide for storage of fuel with burnup dependent enrichment limitations provided in the technical specifications. Criticality in Regions 2 and 3 is prevented by Boral and the technical specification administrative controls. The soluble boron in the pool water provides available negative reactivity to maintain k_{eff} less than or equal to 0.95 for postulated accidents that would affect an increase in reactivity. These limitations satisfy the design basis for preventing criticality outside the reactor where, including uncertainties, there is a 95% probability at a 95% confidence level that the k_{eff} of the fuel assembly array will be less than 0.95.

A movable platform with one 10 ton electric hoist with a 10 ton block and either a 2 ton or 10 ton hook on the west trolley and a 10 ton electric hoist with either a 2 or 10 ton hook and block on the east trolley runs over the spent fuel pool. The west hoist is rated for 2 tons regardless of which hook block is used. The east hoist is rated for 7,800 lbs maximum if 10 ton hook and block is installed. The fuel assemblies are moved with the long fuel handling tool suspended from one hoist. Within the spent fuel pool, the control rods may be handled separately from the fuel assemblies. Safety Guide 13, Regulatory Position C3, concerning measures to prevent cranes from passing over stored fuel when fuel handling is not in progress is discussed in UFSAR Section 1.3.3.13.

A 125-ton overhead crane and trolley is provided for moving the dry fuel storage transfer cask or spent fuel shipping cask into or out of the spent fuel cask laydown area in the fuel building. This crane passes only over the shipping cask loading area which can be isolated from the spent fuel storage area. This design conforms to the intent of Safety Guide 13, Regulatory Position C5, concerning dropping of heavy loads such as a fuel-cask (See UFSAR Section 1.3.3.13).

The fuel pool cooling and purification systems are described in Section 9.5.

<u>New Fuel Storage</u>

New fuel assemblies are stored dry in a steel and concrete structure within the fuel building. Both the elevation and type of construction of the new fuel storage space make it impossible to be flooded. The assemblies are stored vertically in racks in parallel rows, having a fuel assembly center-to-center distance of 21 inches. There is storage space for one-third (53 assemblies) of a core plus 17 spare assembly spaces. The steel rack construction prevents possible criticality by requiring that the spacing between fuel elements not be less than that prescribed. The assemblies which made up the remaining approximately two-thirds of the first core were stored dry in the spent fuel pool, until they were loaded into the core. A 10-ton hoist

is furnished for movement of the new fuel assemblies. The hoist is mounted on the motor driven platform that covers the fuel storage area.

Decontamination Facility

To permit both decontamination and shipping of large items without interruption, the decontamination facilities are located within a single-story steel frame structure adjacent to the fuel building. Both spent fuel casks and handling tools used in the fuel building may be cleaned and decontaminated within this structure. The rails of the 125-ton single failure proof fuel cask crane, which pass over the spent fuel cask laydown area within the fuel building, continue on into the decontamination building. This decontamination facility is described in detail in Section 9.15.

9.12.2.3 Refueling Equipment

Reactor Vessel Stud Tensioners

Stud tensioners are employed to secure or free the head closure joint at every refueling. The stud tensioner is a hydraulically operated device that uses oil as the working fluid. The device permits preloading and unloading of the reactor vessel closure studs at cold shutdown conditions. Stud tensioners minimize the time required for the tensioning or unloading operations. Three tensioners are provided, but tensioning or detensioning can be accomplished using two tensioners. The sequence using two or three tensioners maintains concentric loading or unloading patterns. The correction pass is performed using one or more tensioners depending on the location and number of studs to be adjusted. The studs are tensioned to their operational load in an optimized procedure. This allows for a shorter time to accomplish detensioning and tensionings which reduces exposure. This optimized procedure produces a more uniform tension, stays within the maximum stud load and tensioner pressures.

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. The lifting device is permanently attached to the reactor vessel head. Attached to the head lifting device are the monorail and hoists for the reactor vessel stud tensioners.

Reactor Internals Lifting Device

The reactor internals lifting device is a structural frame suspended from the overhead crane. The frame is lowered onto the guide tube support plate of the internals and is manually bolted, underwater, to the support plate by three bolts. Bushings on the frame engage guide studs in the vessel flange to provide guidance during removal and replacement of the internals package.

Manipulator Crane

The manipulator crane is a rectilinear bridge and trolley crane with a vertical mast extending down into the refueling cavity. The bridge spans the refueling cavity and runs on rails set into the operating floor near 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 down out of the mast to grip the fuel assembly. The gripper tube is long enough so that 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. Hardware modifications to the mast tube have been performed to allow for the use of the sipping process for identifying leaking fuel assemblies while the fuel is inside the mast tube.

All controls for the manipulator crane are mounted on a console on the trolley. The bridge and trolley positions are determined from an encoder system running on a gear track mounted for each axis. The position of the bridge and trolley can be read out on the system console. The drives for the bridge, trolley, and winch are variable speed and include speed variable control, based on location, and limited by a Programmable Logic Controller (PLC) to normal or reduced speed dependent on location. System interlocks are provided to prevent damage to fuel assemblies. The winch is also provided with limit switches plus a mechanical stop to prevent a fuel assembly from being raised above a safe shielding depth should the limit switch fail. In an emergency, the bridge, trolley, and winch can be operated manually using a handwheel on the motor shaft.

The Fuel Handling Equipment Interlock System is described below. In the description and analysis below, the interlocks have only been analyzed for failure in the permissive mode. The consequences of interlock failure in the interlocked mode are the same for all; the failed interlock would prevent normal operation of the equipment until repaired. There is no case where failure in the interlocked mode can result in a hazardous situation.

The manipulator crane design includes the following provisions to ensure safe handling of fuel assemblies:

1. Bridge, trolley, and hoist drives use multiple interlocks and error checking to prevent operation during a failure mode. A Failure Modes and Effects Analysis (FMEA) for the crane demonstrates that no single failure of an electrical component or device will result in uncontrolled motion of the equipment.

2. Bridge and trolley drive operation is prevented except when both gripper tube up position switches are actuated. The interlock is redundant and can withstand a single failure.

3. An interlock is supplied 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 back-up protection for this interlock, the air cylinder cannot apply sufficient force to disengage or operate the gripper in the loaded condition. This interlock is redundant and can withstand a single failure.

4. Two redundant contact switches open to prevent hoist motion when load is in excess of 110 percent of the fuel assembly weight. The interlock is redundant and can withstand a single failure.

5. An interlock of the hoist drive circuit in the up direction permits the hoist to be operated only when either the open or closed indicating switch on the gripper is actuated.

   The hoist-gripper position interlock consists of two separate circuits that work in parallel such that one circuit must be closed for the hoist to operate. If one or both interlocking circuits fail in the closed position, an audible and visual alarm on the console is actuated. The interlock, therefore, is not redundant, but can withstand a single failure, since both an interlocking circuit and the monitoring circuit must fail to cause a hazardous condition.

6. An interlock of the bridge and trolley drives prevents the bridge drive from traveling beyond the edge of the core unless the trolley is aligned with the fuel transfer canal centerline. The machine utilizes encoder inputs to a PLC to create and monitor safe travel zones. These boundary zones prevent damage to the machine by not allowing the mast to come into contact with pool walls, known obstructions or pool/canal floors. Trolley motion is limited to a small adjustment zone when the bridge is beyond the edge of the core and the bridge is in motion. Bridge and trolley drive interlocks prevent the bridge drive from traveling through the refueling canal if the trolley is not aligned with the refueling canal centerline. Bridge, trolley, and hoist travel limits are controlled by dual encoders that continuously error compare to prevent single failure. Failure of redundant encoders in combination with operator error could result in possible collision of the mast with the canal walls, vessel guide studs, or the upper internals in their stored position. The fuel assembly would be protected from damage by the outer mast. The mast is a pipe 16 inches in diameter and 1/2 inch wall thickness. The mast completely encloses the fuel and restrains the fuel from movement by guide bars, which limit fuel movement to 1/4 inch maximum. The fuel assembly is supported along its length by the guide bars.

7. Suitable restraints are provided between the bridge and trolley structures and their respective rails to prevent derailing due to the design basis earthquake. The manipulator crane is designed to prevent disengagement of a fuel assembly from the gripper under the design basis earthquake.

8. The main and auxiliary hoists are equipped with two independent braking systems. A solenoid release, spring set electric brake is mounted on the motor shaft. This brake operates in the normal manner to release upon application of current to the motor and set when current is interrupted. The second brake is a mechanically actuated load brake internal to the hoist gear box that sets if the load starts to overhaul the hoist. It is necessary to apply torque from the motor to raise or lower the load. In raising, the motor opens the brake by use of a cam. In lowering, the motor slips the brake allowing the load to lower. This brake actuates upon loss of torque from the motor for any reason and is not dependent on any electrical circuits. On the main hoist, the motor brake and the mechanical brake both meet a minimum of 150 percent of operating load.

Loss of redundancy in the interlocks on the refueling equipment can be determined only by a test conducted prior to each refueling. The test procedure is basically to set the interlock in its open or interlock mode and then to attempt to perform the related control operation.

Through the use of bypass circuits or simulated actions, it is possible to isolate the primary interlock circuit from its redundant backup and to individually check each.

The main hoist system is supplied with redundant paths of load support such that failure of any one component will not result in free fall of the fuel assembly. Two wire ropes are anchored to the winch drum to a load equalizing mechanism on the top of the gripper tube. Each cable system is designed to support 13,750 lb or 27,500 lb acting together.

The working load of fuel assembly plus gripper is approximately 2,500 lb.

The gripper itself has four fingers gripping the fuel assembly, any two of which will support the fuel assembly weight.

The gripper and hoist system are routinely load tested to 3,250 lbs.

Movable Platforms with Hoists

There is a motor driven movable platform with overhead truss and two independent 10-ton hoists in the fuel building.

One of the 10-ton hoists and the new fuel handling tool, manipulate fuel in the new fuel area up to the new fuel elevator. The 10-ton hoists and the long fuel handling tool, manipulate fuel in the spent fuel area between the new fuel elevator and the fuel transfer canal. Maneuverability over the entire area of the fuel building is provided.

The electric hoists can be moved the length of the platform's overhead truss. Both the vertical lift range of the spent fuel hoists and the length of the long fuel handling tool are designed to limit the maximum lift of a spent fuel assembly. A sufficient depth of water is thereby ensured for safe shielding above the fuel being handled. The maneuvered assembly passes within the pool just above the other stored fuel assemblies.

Both the new fuel handling tool and the long fuel tool are attached to, or detached from, the fuel elements by use of hand gripping levers located at the upper end of the tool. The new fuel handling tool does not become immersed in the pool, while the long fuel handling tool has sufficient length to keep the upper end of that tool above the pool surface at all times.

Fuel Transfer System

The fuel transfer system (Figure 9.12-1) includes a cable driven, underwater conveyor car that runs on tracks extending from the fuel transfer canal through the transfer tube and into the spent fuel pool and an upending frame at each end of the transfer tube. The mode of force used to propel the cart back and forth between the Reactor Containment and the Fuel Building is provided by two electric winches which are mounted in Containment outside the crane wall on the operating deck. One winch is used to pull the conveyor car into the Containment and one winch is used to pull the conveyor car into the Fuel Building. Cable tension is maintained at all times through a counter-torque system which is energized through the winch which is paying out cable. The upending frame in the transfer canal receives a fuel assembly in the vertical position from the manipulator crane. The fuel assembly is then lowered to a horizontal position for passage through the transfer tube and raised to a vertical position by the upending frame in the spent fuel pool. The west hoist on the movable platform takes the fuel assembly to a position in the spent fuel racks. An emergency cable is provided for manually pulling the transfer car back into containment. During unit operation, the conveyor car is stored in the fuel transfer canal in

the containment structure. A blind flange is bolted on the refueling canal end of the transfer tube to seal the reactor containment. Two seals are located around the periphery of the blind flange with leak-check provisions between them.

The electrical interlock functions on the fuel transfer system are as follows (none of these interlocks are designed to satisfy single failure criteria):

1. The gate valve must be open before the conveyor car can be moved. This is not considered a safety-related interlock because the worst that can happen is for the conveyor car to run into the valve disk. The conveyor car drive will stall before either component is damaged.

2. Both lifting frames must be in the horizontal position before the conveyor car can be moved.

3. The conveyor car must be against its travel limit stops before the lifting frames can be operated. This interlock is to make sure that the fuel container on the conveyor car is properly positioned before an attempt is made to raise it.

4. The manipulator crane must be over the core or the gripper must be at the top stop lifting position before the transfer system can be operated. This interlock is to prevent the operator from lowering the transfer system fuel container while the manipulator crane is in the process of inserting or removing a fuel assembly.

New-Fuel Elevator

The new-fuel elevator is located within the spent fuel pool and lowers new-fuel assemblies from the top to the bottom of the spent fuel pool. The new-fuel is transferred from the adjacent new-fuel storage space to the new-fuel elevator by the new fuel hoist on the movable platform using the new-fuel handling tool. Use of the elevator ensures that both the new-fuel handling tool and the hook and cable of the hoist do not become contaminated by immersion in the pool water. Transfer of the new-fuel assembly is accomplished with a hoist on the movable platform using the long fuel handling tool. The new-fuel elevator may be used for repair of irradiated fuel assemblies if a mechanical stop is added to prevent raising the irradiated fuel assemblies too close to the surface.

RCC Changing Fixture

A fixture is mounted on the fuel transfer canal wall for removing RCC elements from spent fuel assemblies and inserting them into new fuel assemblies. The fixture consists of two main components: a guide tube mounted to the wall for containing and guiding the RCC element, and a wheel-mounted carriage for holding two fuel assemblies and positioning them under the guide tube. The guide tube contains a pneumatic gripper on a winch that grips the RCC element 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 and releases the RCC element. The manipulator crane loads and removes the fuel assemblies into and out of the carriage.

Spent Fuel Handling Tool

This tool is used to handle new and spent fuel in the spent fuel pool. It is a manually actuated tool on the end of a long pole suspended from a hoist on the movable platform. An operator on the platform guides and operates the tool.

RCC Change Tool

The rod cluster control (RCC) change tool is a device used to remove an RCC from one fuel assembly and transfer it to another in the spent fuel pit. During use, this tool is supported from the spent fuel pit bridge hoist and is designed to operate from the bridge walkway.

The bottom of the tool is equipped with locating pins to orient the tool with respect to the fuel assembly nozzle.

The RCC tool is lowered onto a fuel assembly in the spent fuel racks. The gripper is then inserted into the RCC hub and activated to engage. The gripper and RCC are withdrawn from the fuel assembly and into a guide structure in the lower portion of the tool.

Once the RCC is fully withdrawn from the fuel, the tool is raised to permit movement to another fuel assembly. The tool is then lowered into the top nozzle of the other fuel assembly, the RCC is inserted into the fuel, and the gripper mechanism is disengaged.

New Fuel Assembly Handling Fixture

This short-handled tool is used to:

1. Handle new fuel on the operating deck of the fuel storage building.

2. Remove the new fuel from the shipping container.

3. Facilitate inspection and storage of the new fuel and loading of fuel into the new fuel elevator.

Refueling Water Storage Tank

The refueling water storage tank is a vertical flat-bottomed cylindrical tank with a dome head and is mounted on and secured to a reinforced concrete foundation. The tank has a 38 ft ID and 52 ft straight side height. The tank is fabricated of stainless steel.

The normal operating function of the refueling water storage tank is to contain the water required for filling both the refueling cavity and the fuel transfer canal. Normally, these two are filled when the reactor head is removed. In case of emergency there is sufficient water in the tank to refill the reactor vessel to above the coolant nozzles. The water in this tank is borated to approximately 2,400 ppm boron. Within the reactor vessel this boron concentration ensures reactor shutdown by at least 10 percent $\Delta k/k$ when all rod control cluster assemblies are inserted into the core and with the reactor cooled for refueling. A manhole is provided for internal inspection of the tank during the refueling period.

During normal operation the RWST is maintained within the volume range specified in Technical Specifications and the Licensing Requirements Manual. Two modes are provided for the return of reactor cavity water to the RWST, as follows:

1. Use of the Residual Heat Removal (RHR) pumps at a flow rate of approximately 2,500 gpm from the reactor cavity (Figure 9.3-1).

2. By use of RHR/fuel pool purification system piping (Figure 9.3-1 and 9.5-1) to pump the reactor cavity contents to the RWST via either the filters and demineralizer (150 gpm) or the filters only (400 gpm).

During normal system operation, two separate level alarms annunciate a liquid level one foot below normal. This is to alert the operator of any system leakages. When the RWST is filled after refueling operations, this high level alarm will annunciate and clear one foot below normal liquid level. Two separate high-high level alarms are also provided and will annunciate when the level exceeds 6 inches above normal liquid level.

The fission product activity level in the RWST immediately after refueling is approximately 1 x 10^{-2} microcuries per cc.

The tank is also used to supply water to certain engineered safety features (See Section 6), therefore, it is designed as a Category I component to withstand seismic loadings. An evaluation is made to ensure no loss of function following the hypothetical earthquake conditions. The connecting piping is designed to withstand seismic loading to ensure the functioning of the system.

9.12.3 Refueling Procedure

The refueling operation follows a detailed procedure which provides a safe, efficient refueling operation. The following significant points are assured by the refueling procedure:

1. The refueling water and the reactor coolant contains approximately 2,400 ppm boron. This concentration, together with the negative reactivity of control rods is sufficient to keep the core approximately 10 percent $\Delta k/k$ subcritical during the refueling operations. It is also sufficient to maintain the core subcritical if all of the rod cluster control assemblies were removed from the core.

2. The water level in the refueling cavity is high enough to keep the radiation levels within acceptable limits when the fuel assemblies are being removed from the core. This water also provides adequate cooling for the fuel assemblies during transfer operations.

The refueling operation is divided into four major phases. A general description of a typical refueling operation through the four phases is given below:

1. Phase I - Preparation

The reactor is shutdown and cooled to cold shutdown conditions with a final k_{eff} <0.9 (all rods in). Following a radiation survey, the containment vessel is entered. At this time, the coolant level in the reactor vessel is lowered to a point slightly below the vessel flange. Then the fuel transfer equipment and manipulator crane are checked for proper operation.

2. Phase II Reactor Disassembly

All cables, air ducts and vessel flange insulation are removed from the vessel head. The refueling cavity is then prepared for flooding by sealing off the reactor cavity; checking of the underwater lights, tools, and fuel transfer system; closing the refueling canal drain holes; and removing the blind flange from the fuel transfer tube. With the refueling cavity prepared for flooding, the reactor head is lifted slightly to check for levelness and to perform a rigging inspection. Additional inspections may be performed during this time using the reactor head as shielding. While monitoring the load cell, the reactor head is lifted to 128 inches. The full-length RCC CRDS are verified clear of the thermal sleeves. The reactor head is taken to its storage pedestal and the cavity is flooded. The control rod drive shafts are disconnected and, with the upper internals, are removed from the vessel. The fuel assemblies and rod cluster control assemblies are now free from obstructions and the core is ready for refueling.

3. Phase III - Fuel Handling

The refueling sequence is controlled by the Refueling Procedure. The sequence is based on the current Westinghouse fuel cycle loading plans, fuel handling specifications and the scheduled refueling operations.

The reactor fuel may be refueled in one of two ways. The core offload or the core shuffle. The core offload removes all of the fuel assemblies from the core, performs the insert changeouts, and then reloads the desired fuel assemblies into the core. The core shuffle, simultaneously offloads the spent fuel, performs fuel insert changeouts, and reloads the new fuel. These variations in the refueling sequence are necessary to support major maintenance on the reactor and reactor support systems, fuel handling, fuel insert changeouts and fuel inspections considerations, and to optimize the outage schedule.

Both the core shuffle and offload are performed with the equipment described as follows:

The manipulator crane transports fuel assemblies to and from the reactor, RCC change fixture and fuel transfer system upender. Fuel assemblies are lifted to a predetermined height sufficient to clear the reactor vessel and still leave sufficient water covering the fuel assembly. This limits any radiation hazard to the operating personnel. Limit switches are provided to ensure that the fuel assembly is in the fully up position before the manipulator crane can be moved.

The fuel transfer system transports fuel assemblies to and from the manipulator crane and motor-driven platform crane. The fuel assembly container of the fuel transfer car is pivoted to the vertical position by the upender. The manipulator crane is moved to line up the spent fuel assembly with the fuel assembly container. The manipulator crane loads the spent fuel assembly into the fuel assembly container of the transfer car. The container is pivoted to the horizontal position by the upender. The fuel transfer car with the fuel container is moved through the fuel transfer tube to the fuel storage area. At the fuel storage area the fuel assembly container is pivoted to the vertical position. The sequence is reversed to transport fuel assemblies to containment.

The motor-driven platform crane transports the fuel assemblies to and from the fuel transfer system and spent fuel racks. The spent fuel assembly is unloaded from the container by the spent fuel handling tool attached to the motor-driven platform crane. The spent fuel assembly is placed in a spent fuel storage rack.

The motor-driven platform crane also transports new fuel assemblies to and from the new fuel storage racks fuel transfer system and the spent fuel storage racks. This is accomplished with use of the new and spent fuel assembly handling tools and the new fuel elevator. New fuel assemblies are usually staged in the spent fuel pool prior to the core offload or core shuffle.

The motor-driven platform crane, with the appropriate handling tool is used to install and remove any of the fuel assembly inserts. This operation is performed either during the fuel shuffle or between the core offload and reload.

The RCC change fixture can be used to changeout RCCA inserts and inserts with RCCA type hubs. If the spent fuel assembly contains an RCC type insert, the fuel assembly may be placed in the RCC changing fixture by the manipulator crane. The RCC type insert is removed from the spent fuel assembly and put in a new fuel assembly or into another spent fuel assembly that has been previously placed in the changing fixture.

4. Phase IV - Reactor Assembly

Reactor assembly, following refueling, is essentially achieved by reversing the operations given in Phase II -Reactor disassembly.

9.12.4 Handling of Failed Fuel Assemblies

A fuel assembly which is suspected to be defective is stored in the spent fuel pool until shipment offsite is possible. A defective fuel assembly may also be stored at the ISFSI.

When a shipment is possible, the fuel element is transferred to a fuel cask. The cask complies with Federal Regulation 10CFR71, so that the defective fuel can be stored and shipped while sealed within it.

9.12.5 Onsite Spent Fuel Dry Storage (SFDS)

BVPS has implemented an on-site Independent Spent Fuel Storage Installation (ISFSI) facility that will be used for Spent Fuel Dry Storage (SFDS). The Spent Fuel Dry Storage (SFDS) operations at BVPS will be conducted under a general license in accordance with 10 CFR 72.

The ISFSI utilizes the AREVA/Transnuclear horizontal Spent Fuel Dry Storage system. The SFDS system includes the Transnuclear 37PTH or 32PTH1 dry shielded canister (DSC), the Horizontal Storage Module (HSM-H) and the OS200 Transfer Cask (TC).

The ISFSI site is located inside the site Protected Area (PA) southwest of the Fuel and Decontamination Buildings. The ISFSI is designed for the capacity of 60 horizontal storage units.

The welded DSC, constructed of steel, provides confinement and critically control for the storage and transfer of irradiated fuel. The concrete horizontal storage module (HSM-H) provides radiation shielding while allowing cooling of the DSC by natural convection during storage. The TC is used for transferring the DSC between the Fuel and Decontamination Buildings and the horizontal storage modules at the ISFSI. The TC also provide radiation shielding during transport operations.

9.12.5.1 Spent Fuel Dry Storage Operations

SFDS operations involve placing the DSC into the TC, placing the TC into the deep end of the cask pit section of the spent fuel pool and the placement of spent nuclear fuel from the spent fuel pool into the DSC. The loaded DSC is then transported in the TC to the onsite ISFSI storage facility. A haul path was evaluated to enable transport of spent nuclear fuel between the Decontamination Building and the ISFSI pad. The loaded DSC in the Transfer Cask is transported to the ISFSI pad and the DSC inserted into the horizontal storage module in its predetermined location on the ISFSI pad. Building structures and equipment have been evaluated for the imposed loads (References 3, 4, 5 and 6).

9.12.5.2 Operations Involving the Spent Fuel Cask Crane (1CR-15) and SFDS Heavy Load Handling

The spent fuel cask crane (1CR-15) has been replaced with a new crane meeting NUREG-0554 and NUREG-0612 requirements for designation as “single-failure-proof”, as discussed in Sections 9.12.2.2 and 16.2.6.4. The crane, crane control system, documentation and materials have been designed as necessary so that the hoist for the main hook meets applicable crane single-failure-proof guidelines. The crane has been evaluated and is capable of supporting a loaded TC during the Operating Basis Earthquake (OBE) and Safe Shutdown Earthquake (SSE) events. Heavy load handling special lifting devices are per ANSI-14.6, Transnuclear Calculation No. 13401-0206 (Reference 7) and Transnuclear Engineering Record No. 13401-ER-002 (Reference 8). Load handling during activities in support of spent fuel dry cask storage are administratively controlled.

9.12.6 System Design Evaluation

9.12.6.1 Incident Control

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.

Gamma radiation levels in the containment and fuel storage areas are continuously monitored. These monitors provide an audible alarm at the initiating detector indicating an unsafe condition. Continuous monitoring of reactor neutron flux provides immediate indication in the main control room of an abnormal core flux level.

Whenever fuel is being added to the reactor core, or is being relocated, continuous monitoring of neutron flux and verification of stable count rate will verify the subcriticality of the core.

During all phases of spent fuel transfer, the gamma dose rate at the surface of the water is 50 mr per hour or less. This is accomplished by maintaining a minimum of 8 ft-4 inches of water above the top of the fuel assembly during all handling operations.

The two cranes used to lift spent fuel assemblies are the manipulator crane and the movable platform. The manipulator crane contains positive stops which prevent the top of a fuel assembly from being raised to within 8 ft-4 inches of the normal water level in the refueling cavity. The hoists on the movable platform move spent fuel assemblies with a long-handled tool. Hoist travel and tool length likewise limit the maximum lift of a fuel assembly to within 8 ft-4 inches of the normal water level in the spent fuel pit.

The fuel pool elevator, located within the spent fuel pool, utilizes push button switches. An elevator operator must hold down the up push button switch to raise the fuel pool elevator. This ensures that the operator will be able to immediately stop the elevator upon any indication of trouble or increasing radiation levels in the fuel building. The new fuel elevator may be used for repair of irradiated fuel assemblies if a mechanical stop is added to prevent raising the irradiated fuel assemblies too close to the surface.

9.12.6.2 Malfunction Analysis

The analysis presented in Section 14.2.1 assumes damage to 137 fuel rods in an assembly as the basis for a fuel handling accident.

9.12.6.3 Seismic Considerations

The maximum design stress for the structures and for all parts involved in gripping, supporting, or hoisting the fuel assemblies is 1/5 ultimate strength of the material. This requirement applies to normal working load and emergency pullout loads, when specified, but not to earthquake loading. To resist design basis earthquake forces, the equipment is designed so that the stress in any load bearing part is less than 0.9 times the ultimate strength for a combination of normal working load plus design basis earthquake forces.

9.12.7 Tests and Inspections

Prior to initial fueling, preoperational checkouts of the fuel handling equipment were performed to ensure proper performance of the fuel handling equipment and to familiarize station operating personnel with operation of the equipment. A dummy fuel assembly was utilized for this purpose.

Upon completion of core loading and installation of the reactor vessel head, certain mechanical and electrical tests are performed prior to criticality. The electrical wiring for the rod drive circuits, the rod position indicators, the reactor trip circuits, the incore thermocouples, and the reactor vessel head water temperature thermocouples are checked at the time of installation. The checks are repeated on these electrical items before unit operation.

REFERENCES FOR SECTION 9.12

1. DLC response to NRC I. E. Bulletin 84-03, Reactor Cavity Water Seal.

2. Holtec Report HI-92791, Rev. 6, "Spent Fuel Pool Modification For Increased Storage Capacity, Beaver Valley Power Station Unit 1," forwarded to the NRC with Technical Specification Change Request No. 202, dated November 2, 1992, as supplemented by calculation 8700-DMC-3664, Rev. 0.

3. Transnuclear Engineering Evaluation No. 13401-EE-008, “Seismic Stability Assessment of the Wheelift Transporter”.

4. Transnuclear Engineering Evaluation No. 13401-EE-001, “Beaver Valley Power Station: Interference Study & Heavy Loads Summary for Pool-to-Pad Operational Evolutions”.

5. Transnuclear Calculation No. 13401-0202, “Stability of Loaded Transfer Cask in all Planned Configurations in the Decon Building”.

6. Calculation 8700-DSC-0355, “Transnuclear Equipment Loading Evaluation”.

7. Transnuclear Calculation No. 13401-0206, “OS200 Lifting Yoke Design Qualification for 8.0-inch Crane Hook”.

8. Transnuclear Engineering Record No. 13401-ER-002, “Beaver Valley Power Station: Cask Handling Crane Interface with ISFSI Equipment Design Inputs”.

9.13 VENTILATION SYSTEMS

9.13.1 Design Bases

The ventilation systems as shown in Figure 9.13-1, Figure 9.13-2, Figure 9.13-3, and Table 9.13-1 are designed to provide a suitable environment for personnel and equipment and prevent the spread of radioactive contamination. The ventilation exhaust air from areas subject to radioactive contamination are designed with provision for diverting the flow through the main filter banks in the supplementary leak collection and release system (Section 6.6).

Required flow rates for the ventilation systems are based on heat removal from warm areas and the prevention of accumulation of condensation or toxic vapors. Flow rates for ventilation systems serving areas subject to radioactive contamination are selected to maintain a slight negative pressure with a capture velocity so as to ensure only inleakage.

The common ventilation vent on top of the Auxiliary Building, discharging at El. 814.0, serves as a discharge for ventilation air. Ventilation discharge flow from areas subject to radioactive contamination are monitored for radioactivity prior to entering the common ventilation vent. See subsections 11.3.3.3.3 through 11.3.3.3.9, 11.3.4.4, and 11.3.4.5 for a description of the specific ventilation flows monitored for radioactivity.

9.13.2 Auxiliary Building

The top floor of the auxiliary building serves as a fan room containing the ventilation and filtration equipment for this building, the fuel building, the containment, and the containment contiguous areas.

As shown in Figure 9.13-1, shielded area exhaust systems "A" and "B", serving potentially contaminated areas in the auxiliary building, have a combined capacity of 62,000 cfm, exhausted by two fans of 31,000 cfm each. The two systems are designed on a once-through basis and each system is continuously monitored for radiation. Upon a high-high radiation alarm, automatic dampers divert either system's exhaust air stream through one of the main filter banks in the supplementary leak collection and release system and thence to the SLCRS Vent point. (With manual damper VS-D-7-8A normally closed during Mode 6, operator action is required to manually open the damper to accomplish diversion in the event of a high-high alarm.)

Exhaust systems for clean areas not subject to radioactive contamination discharge directly outdoors.

The maximum ambient air temperature limits of those areas containing safety related motors were considered as part of the BVPS Environmental Qualification Program (10 CFR 50.49) an, in general, are such that the ambient temperature plus the expectedmotor heat rise does not exceed the insulation rating of the motor windings. In areas where boric acid equipment or piping are located, the ambient air temperature is monitored to assure that boron precipitation will not occur.

Two general area supply air units of 27,000 cfm capacity each draw outside air through filters, cooling and heating coils and discharge through ductwork to supply makeup air to the exhausted areas in the auxiliary building. Where clean and potentially contaminated areas are adjacent, air is supplied to the clean area and exhausted from the potentially contaminated area to avoid the possibility of any spread of contamination. The supply units, in combination with the exhaust fans, are designed to create a slight negative pressure in the building so as to ensure inward leakage. Flow rates for air exhausted from shielded cubicles are designed to create an inward capture velocity of at least 50 fpm at the cubicle entrance.

9.13.3 Fuel Building

A recirculating air unit of 20,000 cfm capacity is provided for temperature and humidity control to inhibit the formation of condensation on surfaces in the fuel building. Fuel Building air is recirculated through filters, cooling coil, heating coil, supply fan, and distribution ductwork to the building.

The fuel building is also maintained under a negative pressure by the supplementary leak collection and release system as described in Section 6.6.

9.13.4 Main Control Area

As shown in Figure 9.13-2, Figure 9.13-3, and Table 9.13-1, the main control room, office and computer room, process instrument room, relay room and communications room are normally air-conditioned by two 70-ton 100 percent design capacity air-conditioning units located in the control room area mechanical equipment room. The system is zoned for temperature control of individual areas with auxiliary controls located within the main control room. Two river water cooling coils in the return air stream are also provided as an additional backup method of cooling the main control area if required. The river water cooling coils allow one of the air conditioning refrigeration units to be down for maintenance while still maintaining adequate redundancy capability for control room cooling under DBA conditions.

The control room air conditioning system is essential to maintain temperature and humidity for the proper operation of the computer and instrumentation and for the comfort and efficiency of the operating personnel during normal operation and accident conditions.

The control room air conditioning system includes two separate systems consisting of two 100 percent redundant air handling units, refrigeration condensing units, river water cooling coils, temperature control air compressors and controls, return air fans, and associated dampers. Each system's electrical components are supplied from separate emergency powered buses. The refrigeration condensing units are each supplied from two separate sources of cooling water. The operation of electrical components is monitored in the control room by means of red and green lights to indicate functioning conditions. The equipment is located in missile-protected areas and has been seismically analyzed and installed. Fan room flooding at river water elevations above El. 713 ft 6 inches is prevented by the use of backwater gate valves in the floor drainage system.

The normal unfiltered air intake plus in-leakage is 1250 cfm or less, total for both units, and the air conditioning unit flow rate is 33,500 cfm for the control room envelope.

Although the control boards are functionally and physically separate, BVPS-1 and BVPS-2 share a common control room. The control room areas of both units are open to each other and are, therefore, within the same pressure boundary. The emergency control room pressurization systems used during accidents are shared by both units.

Normal Operation

Although BVPS-1 and BVPS-2 control room areas are open to each other, in normal operation each area is served by separate HVAC systems. Thus two (2) 100% capacity HVAC systems serve the BVPS-1 area of the control room and two (2) additional 100% capacity HVAC systems serve the BVPS-2 area of the control room.

These HVAC systems maintain temperature and humidity conditions for the proper operation of control room equipment and instrumentation as well as for the comfort and efficiency of the operating personnel during both normal operation and accident conditions. During normal operation one (1) train of HVAC serving the BVPS-1 area is in operation with the second train in standby. A similar arrangement exists for the BVPS-2 area of the control room.

Accident Events

In the event of an accident, the control room envelope (CRE) as defined in the Control Room Envelope Habitability Program is pressurized by the control room emergency ventilation system (CREVS) while the normal HVAC systems continue to operate but in the 100% recirculation mode.

The CREVS fans move outside air through prefilters, charcoal filters and particle filters before discharging it into and pressurizing the control room. There are three (3) 100% CREVS systems.

The three (3) 100% capacity CREVS trains serve the CRE. Any one (1) of the three CREVS trains is capable of pressurizing the entire CRE. Two (2) CREVS trains are powered from train A and train B of BVPS-2, respectively. The third CREVS train is powered from either train A or train B of BVPS-1. All three (3) CREVS trains receive automatic start signals on a CIB signal from either unit, or a high control room radiation signal. The CREVS trains can also be manually started.

The two (2) CREVS trains powered from BVPS-2 are fully automatic meaning that both the fans and the dampers operate automatically on the start signal. Thus either of these CREVS trains can pressurize the CRE with no operator actions.

The CREVS train powered from BVPS-1 is not fully automatic. Its fan control switches are not maintained in the auto start position due to normally closed manual fan inlet and discharge dampers.

The BVPS-1 Technical Specifications requires any two (2) of the three (3) CREVS trains to be operable for Modes 1, 2, 3, 4 and during movement of irradiated fuel or movement over irradiated fuel.

If both fully automatic CREVS trains are OPERABLE no manual action is required to pressurize the control room even in the event of a single failure.

If either of the fully automatic CREVS trains is INOPERABLE or fails to operate, the third CREVS train provides single failure protection. In this case if the one remaining fully automatic CREVS train does not start (single failure) an operator can start the third system by manually opening the dampers and manually repositioning the fan start switch. Thus in this unusual situation, three (3) manual operator actions are required to activate CREVS and pressurize the control room. The manual inlet and discharge dampers located on the floor below the main control room must be opened and the control room operator must throw the fan start switch.

The radiological analysis is based on any one CREVS train being in operation within 30 minutes after a CIB signal. Operator's capability to satisfy the 30-minute requirement has been demonstrated.

Due to the physical layout of the suction ductwork serving the outside air pressurization fans, it is not considered necessary to provide any means of removing entrained water. The outdoor air intake ducts from which these fans take suction are designed for 100 percent outdoor air (33,500 CFM) to the control room air conditioning system. Following a DBA, the control room air conditioning system is operating at 100 percent recirculation, thereby resulting in only 1000 CFM being drawn up through the 48 inch by 75 inch outdoor air weather hood. This results in a velocity of approximately 40 feet per minute which is not sufficient to result in carryover of water droplets. Also, the suction taken off to this system is at right angles to the 48 inch diameter vertical intake duct.

To assure that the CREVS is capable of developing a positive pressure in the CRE, certain passageways and leakage paths from the CRE are required to be closed. The following precautions are used for controlling outleakage for the control room. Flexible sealing compounds are used between the annulus space between electrical cables and sleeves. Flexible boots are used for sealing the annulus space between pipes and sleeves. Redundant isolation dampers are provided in the control room area exhaust and supply air ducts. Floor drains have a primed deep seal water trap. All doors for the leakage barrier are provided with flexible seals. Walls and ceilings adjacent to shake spaces are provided with continuous water stops embedded in concrete. The ventilation duct dampers are tested and accepted as "bubbletight" at an air pressure of 10 psig through the butterfly type seat and packing. CRE unfiltered air inleakage testing is performed in accordance with Technical Specifications.

The majority of leakage is through door leakage paths. Data from Atomics International was used for determining leakage through and around hollow metal doors with weather stripping and flexible neoprene seals. The results presented in Figure A-4(1)C, Reactor Technology (NAA-SR-10100) by Atomics International were used. At a differential air pressure of 0.125 in water gage, the leakage is 12 cfm. There are thirteen doors used as leakage barriers in the Control Room areas. Therefore, a total leakage of 156 cfm is anticipated. A safety factor greater than two is provided to account for small leakages through penetrations and wear of door seals.

To assure that doors are not left open as potential leak paths, the doors either are self-closing or are controlled by administrative procedures which prohibit blocking open Control Room doors. To ensure that all these doors are closed following a DBA, the emergency procedures require that the plant attendants physically check that these doors are closed when returning to the main control area upon hearing the evacuation alarm.

As previously stated, the emergency air flow to the CRE is provided to ensure a positive outflow of air regardless of the magnitude of positive pressure. Periodic CRE unfiltered air inleakage tests will ensure that the CRE is operable.

Secondary control of the pressure boundary can be obtained (but is not mandatory) as follows:

> Automatic, adjustable fan speed controllers were added for the axial flow switchgear ventilation fans, VS-F-17 and VS-F-18. The fan speeds can be controlled to maintain a constant static air pressure in the ductwork downstream of VS-F-17. This constant static air pressure will therefore maintain constant static pressures in the normal switchgear and rod control M/G rooms to help maintain the Control Room process rack pressure boundary criteria.

In the event of a loss of power under either normal operating or accident conditions, all electrically powered motors and controls associated with the air conditioning and pressurizing systems are furnished with emergency power from the emergency diesel generators.

Refer to Section 8.5.3 for a discussion of the emergency battery room ventilation system and smoke detection system.

In the event of a fire in any of the individual zones in the main control area, separate manually actuated zone smoke dampers are provided to seal off the affected zone and permit the remaining zones to continue to function. Provision is also made for purging smoke from the fire area by positioning the proper outside air and exhaust air dampers.

Refer to Section 11.3.5 for a discussion of the Control Room Area Radiation Monitors (RM-RM-218A,B).

Self-contained breathing apparatus units and sufficient reserve air cylinders are available to support the minimum control room shift composition for at least five hours. This satisfies Regulatory Guide 1.78. Sufficient additional units are provided to support the members of the emergency squad stationed outside the control room for one hour, after which these personnel would move away from the area affected by the toxic release. Air cylinders brought from off-site locations may be used to extend capacity beyond five hours.

The main control area is thus designed to be continuously habitable under any foreseeable condition of operation.

9.13.5 Service Building

The chemistry laboratory hoods and potentially contaminated area shop exhaust systems discharge to the common monitored ventilation vent through prefilters and particulate filters. Suction hose facilities and exhaust hoods are provided in the potentially contaminated area shop to handle specific maintenance problems, and these also discharge to the common ventilation vent through the prefilters and particulate filters. All other ventilation exhaust systems such as locker room, cafeteria, and office air conditioning, discharge to the atmosphere locally.

9.13.6 Other Areas

Ventilation systems in the turbine building, screenwell house, and other nonradioactive areas all exhaust locally to the atmosphere.

9.13.7 Evaluation

Ventilation equipment is located in areas that are accessible for repair. Where level of radioactive contamination requires constant ventilation, redundant exhaust fans and dampers are provided. Redundant air conditioners, cooling coils, and temperature controls are provided for the main control area, thus verifying that the system meets the single failure criteria.

Individual radiation monitors are provided for each exhaust duct serving potentially contaminated areas in the auxiliary building. The use of these monitors aid in identifying any sources of contaminated air and, at a preset level, automatically diverting the contaminated air stream to the main filter banks in the supplementary leak collection and release system and thence to the SLCRS Vent point. Annunciators and indicators in the control room permit the operator to determine that the diverting dampers have been properly positioned.

The normal outdoor air supply of the main control area is secured by automatic operation following a DBA. The main control area is then maintained habitable through the use of three systems:

1. Emergency powered air conditioning units or river water cooling coils

2. Pressurization air

3. Separate fresh air intake fans with filters.

All of the air conditioning equipment for the main control area is designed as Seismic Category I (Appendix B) and is tornado-protected (Section 2.7.2).

The maximum control room temperature, if both air conditioning refrigeration units are inoperative, will not exceed 120°F by the use of the redundant river water cooling coils at a maximum river water temperature of 90°F. Thus control room electrical system performance will not be degraded if normal 75°F control room design air temperature is exceeded as electrical system malfunction does not occur at temperature levels below 120°F.

With the exception of the main control area equipment, supplementary leak collection and release system and containment refueling exhaust duct, the ventilation equipment discussed in this section is not designed as Seismic Category I.

9.13.8 Tests and Inspections

All systems are tested and inspected as separate components and as integrated systems. Anemometer or velometer readings are taken to ensure that all air systems are balanced to deliver the required air quantities at design conditions. All water coils are hydraulically tested for leakage prior to being placed in service.

Capacity and performance of centrifugal fans conform to the required conditions and are tested or rated in compliance with AMCA test codes and certified ratings program.

Air-conditioning unit coils are rated to conform to American Refrigeration Institute Standard 410 or American Society of Heating, Refrigeration and Air Conditioning Engineers, Inc. Standard 33.

Air distribution and dispersion devices comply with ADC test code 1062R1.

Materials and construction methods for air-conditioning and ventilation systems are inspected for compliance with National Fire Protection Association Standard 90A and 91.

9.14 FUEL OIL SYSTEMS

9.14.1 Design Bases

Separate fuel oil storage and supply systems are provided for the emergency diesel generators and engine-driven fire pump.

The safety-related portion of the emergency diesel generator fuel oil system, as shown in Figure 9.14-1, is composed of missile protected (Section 2.7.2) Seismic Category I (Appendix B) equipment. All lines are piping Class Q3. The design incorporates sufficient redundancy to prevent a malfunction or failure of an active or passive component from impairing the ability of the system to supply fuel oil to at least one of the emergency diesel generators.

The fuel oil supply to the diesel generators is further assured by tornado protected fuel oil lines and tanks in addition to cross-connected pump suction lines. The pumps take suction from two 20,000 gallon diesel generator fuel oil storage tanks and discharge to day tanks located within the diesel generator building.

The fuel oil supply system for the engine-driven fire pump is also missile protected and Seismic Category I.

9.14.2 Storage Facilities

Separate fuel storage facilities are provided for each system as indicated in Table 9.14-1.

9.14.3 Fuel Oil Suppliers

There are six major oil distributors within forty miles of the site. Tabulated in Table 9.14-2 are their normal diesel fuel stocks on hand and their delivery capabilities.

9.14.4 System Design

9.14.4.1 Emergency Diesel Generators

All fuel oil lines are adequately protected from corrosion by enclosure in a heated building, encasement in concrete, coating with bitumastic enamel and wrapping with coal tar saturated asbestos, or equivalent coating for piping buried directly in the ground.

Buried fuel oil storage tanks are protected on exterior with two coats of bitumastic applied after near white blast cleaning.

The day tanks and engine mounted tanks are vented at the diesel generator building roof in a missile protected enclosure.

The emergency diesel generator fuel oil storage tanks are buried underground and are kept filled to full capacity. Being buried, the oil temperature will be relatively constant. This constant temperature will reduce tank breathing and the subsequent accumulation of moisture by condensation. The fuel oil transfer pumps suction pipe is located above the tank bottom to prevent drawing water off the tank bottom, and maintains the usable fuel oil requirement specified in the Technical Specifications.

The day tanks and fuel oil pump relief valves return oil to the buried fuel oil tanks via an overflow line (one per tank).

In addition, to meet single failure criteria the fuel oil system for the two emergency diesel generators (Section 8.5) consists of the following features:

1. Two diesel generator fuel oil storage tanks, located underground, each with a storage capacity of 20,000 gallons. These tanks are built and stamped to ASME Section VIII Code. The capacity of both tanks is adequate for more than seven days full load operation of one diesel generator. Each tank has approximately four days capacity.

2. Fuel pump suction lines are cross-connected, as are the discharge lines, to provide fuel to either diesel generator from either tank.

3. There is a fuel suction strainer for each emergency diesel generator. The strainer is located on the suction line between the fuel oil pump and the buried fuel oil tank.

4. Two 100 percent capacity, electric motor driven, rotary, screw type pumps per diesel engine. These pumps transfer fuel from the diesel generator fuel oil storage tanks to the day tanks.

5. One day tank of approximately 500 gallon capacity per emergency diesel generator.

6. One engine mounted fuel oil tank of approximately 500 gallon capacity per emergency diesel generator. This fuel oil tank is fed by gravity from the day tank.

7. One engine-driven and one motor-driven positive displacement pump is mounted on each emergency diesel generator. These pumps take suction from the engine mounted fuel oil tank and discharge through filters to the engine fuel injectors. Pressure is regulated by relief valves and a by-pass system.

8. Two duplex type fuel filters per emergency diesel generator. The filters are located ahead of the fuel injectors.

9. One exhaust-silencer per emergency diesel generator exhausting at the building roof in a missile protected enclosure.

9.14.4.2 Engine Driven Fire Pump

The fuel storage tank for the engine driven fire pump is installed in the missile protected fire pump room in the intake structure. Fuel from the storage tank is supplied to the engine by an engine mounted pump. To prevent an accumulation of moisture in the diesel fuel oil, the oil is stored in a heated area in an elevated tank away from water sources. The engine driven fire pump and engine are FIA approved.

9.14.5 Evaluation

The emergency diesel generator fuel oil system is the most critical and is designed to remain operable under any one of the following conditions:

1. Maintenance outage or failure of any one diesel fuel pump.

2. Maintenance outage or failure of any one storage tank. The tanks are placed underground to minimize fire and other physical hazards.

3. Maintenance outage or failure of either of the redundant supply headers from the tanks.

4. Loss of outside power coincident with failure of one emergency diesel generator.

Seven days of onsite power generation is used as a conservative upper limit for design and safety evaluations of fuel storage capacity, even though it is highly improbable that the emergency diesel generators would be required to furnish station auxiliary power for this period. The onsite diesel fuel storage supply can be replenished within this time, as the station area is accessible by road, boat and air. In addition, relatively free access to the diesel generator fuel oil system will be permitted shortly after the DBA.

Fuel to the engine-driven fire pump is ensured by missile protecting both the fuel oil storage tank and the fuel oil supply lines between the tank and the engine-driven fire pump. The tank is sized to supply fuel oil for a period of 8 hours with pump operation at full capacity.

The fill lines for the emergency diesel generator fuel oil storage tanks are not designed as Seismic Category I nor are they missile protected.

9.14.6 Tests and Inspections

The fuel supply headers are hydrostatically tested during construction, and all active system components, pumps, valves, and controls are functionally tested during startup, and periodically thereafter. The fuel oil in storage is periodically checked for water, sediment, etc. to determine possible contamination or deterioration. The fuel oil inventory is also periodically checked. The frequency of checking is given in the Technical Specifications. To ensure the proper quality of new fuel oil, each delivery is tested before it is transferred from the holding tank to the emergency diesel generator storage tanks.

Periodically, samples will be drawn from the emergency diesel generator and the engine-driven fire pump storage tanks' bottom to check for water, viscosity and sediment. The presence and amount of water will determine the frequency of future checks.

9.15 DECONTAMINATION FACILITY

The decontamination facility is a steel frame and siding structure on the south side of the fuel building under the fuel cask trolley rails. This location makes it accessible for transporting major items in and out of the building. A rolling steel door in the south end of the building provides access for the fuel cask and equipment weighing more than 10 tons. The design data for the decontamination facility components is provided in Table 9.15-1.

Another entry at grade level gives access to the building and allows movement of equipment under the 10 ton bridge and trolley crane.

9.15.1 Design Bases

The facility is designed to provide an area in which equipment can be decontaminated without releasing activity to the environment in an uncontrolled manner. Decontamination procedures are specified to reduce surface contamination to a level such that the components can be maintained, inspected and handled in a safe manner.

9.15.2 Description

The decontamination structure is a steel frame and siding building abutting the west end of the fuel building's south wall. A 125 ton trolley runs through the high bay portion of the decontamination building, into the west end of the fuel building to the north, and on a high level runway out over the road to the south. A 10 ton bridge and trolley crane in the building is used to assist in dry cask storage operations and other work activities with minimum personnel exposure. A T-shaped rolling steel door encloses the high bay area from the outside when the 125 ton trolley is not in use. The fuel building and decontamination building are separated by essentially airtight doors.

Ventilation air is exhausted from the decontamination building through two roof exhaust fans and is used for normal ventilation of the decontamination building.

When a piece of equipment which is known to be contaminated must be cleaned, an enclosed air space is fabricated from which the air is exhausted to produce a slightly negative pressure differential with respect to the external area.

Personnel normally enter the decontamination building by passing through the fuel building from the auxiliary building. An emergency exit from the decontamination building is provided.

The interior surfaces of the building are covered with suitable materials to permit easy decontamination. Hose connections are provided for compressed air and primary grade water at each work area. A 1 inch connection from the fuel pool purification system is provided over the fuel pool and is used to hose down the spent fuel cask.

An ultrasonic cleaning tank is provided for immersion cleaning of components. There is also decontamination of equipment by washdown combined with scrubbing and soaking with or without agitation. The flexibility provided by various decontamination methods results in the best decontamination for a specific job while minimizing personnel exposure and limiting the release of radioactive material to the environment. Working personnel are monitored to ensure that established exposure limits are not exceeded.

A decontamination system is provided. A fluid waste treating tank is used to receive make-up water and a measured amount of chemicals from the chemical addition tank, to receive recycle and sump return and to supply the system via the tank's pump. A filter is provided for system cleanup.

9.15.3 Design Evaluation

The facility provides a contained area with all discharges controlled to prevent the inadvertent release of activity to the environment.

In the event of leakage from piping or equipment, all areas of the building are provided with drain connections which are routed to the sump.

9.15.4 Tests and Inspections

Periodic calibrations and checks using the integral check source are conducted on the radiation detection equipment in the ventilation system serving the decontamination facility to ensure proper operation.

Periodic swipe surveys of surfaces in the decontamination facility are taken to determine loose surface contamination concentrations.

9.16 AUXILIARY RIVER WATER SYSTEM

9.16.1 Design Basis

In response to the design basis event consisting of an explosion in conjunction with a gasoline barge impact, an alternate system is required to provide heat sink requirements when the Seismic Category I intake structure is disabled by the postulated event. In accordance with Regulatory Guide 1.27 - Ultimate Heat Sink, an alternate system should as a minimum be capable of providing its design function during site related historic events. Therefore, the alternate, designated as the Auxiliary River Water System (ARWS), is designed to the following criteria:

1. Historic earthquake 0.03g (surface motion).

2. Redundant pumps and motor-operated valves are provided to accommodate a single active failure.

3. Flood protected to El. 705 ft (Standard Project Flood).

4. Low river level capability to El. 654 ft.

5. No tornado protection.

6. Located to preclude damage from gasoline barge impact/ explosion which may disable the Seismic Category I intake structure.

7. Loss of offsite power is not considered coincident with requirement to start up ARWS; however, capability is provided for onsite emergency power in the long term.

8. A DBA is not considered coincident with requirements for ARWS.

9. Electrical and control requirements to meet IEEE 308-1971 standards.

10. Piping and valves to meet B31.1.0. Pumps to meet applicable ANSI and Hydraulic Institute Standards. Any pressure vessels to meet ASME Code VIII.

The ARWS is designed to accommodate unit shutdown from 100% reactor power and subsequent cooldown of the reactor coolant system to less than 200°F, after the postulated loss of the intake structure.

The ARWS is designed to duplicate the cooling capacity of the River Water System (RWS) specified in Section 9.9.1.1.

9.16.2 System Description

As shown in Figure 9.9-1, the ARWS consists of two 100% capacity pumps discharging to a 24 inch line and connected to the redundant 24 inch Seismic Category I river water supply lines via motor-operated valves (MOV's) located in separate Seismic Category I valve pits. From the point of connection to the river water lines the cooling water flows to the systems and equipment described in Section 9.9. The auxiliary intake structure location is shown on Figure 1.2-1.

The river water inlet to each ARWS pump is provided with an automatic self-cleaning traveling water screen system. Filtered water requirements for pump seals and screen washing are provided by an automatic self-cleaning water strainer.

The two 100% capacity pumps and two motor-operated valves are provided to accommodate the active failure criteria requirements for the system. The check valves in the Seismic Category I river water lines, located in the valve pit, automatically isolate the river water system from the disabled intake structure during the design basis event. This feature maintains the integrity of the River Water System to continue unit shutdown cooling water requirements when supplied from the ARWS.

The ARWS will be manually started from the unit control room whenever the RWS is unable to supply unit required cooling water. The barge impact and/or explosion design basis event will result in immediate loss of pressure and flow to both redundant river water supply lines. Loss of River Water System is indicated in the main control room by a zero reading on the river water pump motor amperage gage, breaker trip annunciation, or abnormal primary and secondary cooling system water temperatures. Upon detecting loss of the River Water System the operator will initiate unit shutdown and start both ARWS pumps. Starting of the ARWS pumps will also open respective header MOV's to supply cooling water to each redundant header. The MOV's can also be operated from the unit control room via individual control switches. Auxiliary river water system cooling effectiveness can be verified by adequate header pressures and various primary plant cooling water temperatures.

The ARWS is also capable of operating during loss of offsite power from redundant unit emergency buses as shown on Figure 8.1-1. When the RWS pumps are unavailable, an ARWS pump can be manually started and powered (in the event of loss of offsite power) from the emergency bus after the diesel loading sequence is complete.

In addition, the ARWS can also serve as a discharge for River Water when the River Water System (Section 9.9) is supplying the system components from the "A" header and returning the water via the "B" header, or vice versa, through the ARWS to the Alternate Intake Structure as described in Section 9.9.2.

9.16.3 Evaluation

The postulated gasoline barge impact with the intake structure and coincident explosion disabling the Reactor Plant River Water System is a low probability event and is outside those typically postulated by the NRC for reactor sites. Nonetheless, the ARWS provides defense in depth in assuring shutdown cooling capability. The requirement to operate the ARWS is not coincident with a postulated Design Basis Accident, but is coincident with the postulated gasoline barge impact event. The ARWS is a manually operated non-safety system provided with redundant pumps and valves on a header to accommodate a single active failure.

The ARWS is designed to provide cooling water to shut down the unit from 100% power and to subsequently cool down the reactor coolant to less than 200°F indefinitely, after the postulated loss of the Category I intake structures due to the new design basis event.

Redundancy of the ARWS pump and motor-operated valves permit acceptance of an active failure without impairing designed cooling water requirements. Furthermore, low pressure alarms and pressure indicators in the river water lines at the primary component cooling heat exchanger, pump motor amperes, and primary plant cooling system temperatures provide the operator with information to evaluate performance of ARWS and RWS.

The ARWS is also capable of operation during loss of offsite power, as power requirements can be provided from the redundant emergency buses when the RWS pumps are unavailable. The Licensing Requirements Manual establishes the functionality requirements of the ARWS.

9.16.4 Tests and Inspections

The major portion of the RWS is in continuous use and requires no periodic testing. However, the ARWS from the auxiliary river intake structure to the RWS connection in the valve pit can be tested periodically during unit operation, shutdown, or refueling periods.

During normal unit operation, cooling water is supplied from one river water pump via one 24 inch river water supply header. A standby river water pump is lined up to supply cooling water to the other redundant 24 inch header. The Auxiliary River Water System can be tested by starting the ARWS pump and opening the motor-operated valve associated with the standby river water header. The other ARWS pump can also be started to supply water to the safe river water header. Evaluation of the ARWS operation can be conducted by monitoring ARWS and RWS header pressures, motor amperes, and primary plant system temperatures.

The system is hydrostatically tested prior to acceptance and all active components are accessible for periodic visual inspection during unit operation.

9.16.5 Instrumentation Application

Operation and monitoring of the ARWS is accomplished in the unit control room. The following control, instrumentation, and annunciation functions are provided:

1. Control switches with status lights for each ARWS pump and each motor-operated valve

2. Ammeter for each ARWS pump motor

3. Pressure indicator, one for each river water supply header, located in the auxiliary intake structure

4. Annunciator, river water supply header low pressure, sensor located in the auxiliary intake structure

5. Annunciator, auxiliary intake structure local trouble (common alarm)

    a. Pump house temperature high/low

    b. ARWS pump seal water pressure low

    c. ARWS pump header pressure low

    d. ARWS pump bearing temperature high

    e. Strainer differential pressure high

    f. Heat trace trouble

9.17 COMMUNICATIONS SYSTEMS

Communications are essential for effective activation and implementation of the Emergency Preparedness Plan. BVPS-1 has several independent systems for outside communication to Federal, state, county authorities, to corporate management, and to offsite support groups. These systems include the Bell System, the PAX system, the Emergency Telecommunications System, and the industrial radio system. Onsite, the plant alarm system and the station paging system provide communication/notification for station personnel.

These multiple systems and redundancies ensure the performance of vital functions in transmitting and receiving information throughout the course of the emergency. The various types of communications systems mentioned above are described in Section 7.6 of the Emergency Preparedness Plan, Volume 1. In addition to these systems, the plant security force is equipped with a dedicated use communications system and operators have available an alternate safe shutdown communication system.

A discussion on security communications is contained in the Security Plan and a description of the alternate safe shutdown communications system is given below.

The design of the BVPS-1 precludes the necessity of having work stations on the plant site where it may be necessary for plant personnel to communicate with the control room or the emergency shutdown panel during and/or following transients and/or accidents in order to mitigate the consequences of the event and to attain a safe cold plant shutdown. However, in the instance of failure of both automatic and manual modes of reactor coolant or boron addition, it may be necessary to use local control of bypass valves. The area involved is the valve operation aisle just east of the charging pump pipe chase at El. 722 ft-6 inches in the auxiliary building. In the event of fire in the main control room cable spreading area, process instrument room or in the communication equipment and relay room, an alternate safe shutdown communications system provides independent communications for the operators to perform alternate safe shutdown of the plant even if the normal PAX system fails. This system uses the Emergency Response Facility (ERF) switchboard for all circuits. Emergency backup power is supplied to this system from the ERF diesel generator.

Sound levels in the valve operating aisle during all accident or transient conditions will be similar to those during normal operation and will not exceed 90 decibels (audio scale) (93 dB on flat response scale).

The types of communication systems available are listed in Table 9.17-1.
In addition, walkie-talkies available for the VHF band radio have been tested and proved to be effective from the valve operation aisle.

Communication through the page party system loud speakers to an operator anywhere in the valve operating aisle with good articulation is possible with background sound levels of at least 90 decibels (audio scale). Still higher background sound levels would be permissible with power articulation, since it is only necessary to alert the operator. Using the page party handset, communication is possible in either direction with background levels of up to 115 decibels (audio scale).

Satisfactory communication is possible using the UHF radio walkie-talkie with background sound pressure levels of 87 decibels (flat response scale) by holding the unit 2 inches from the ear or mouth. By holding the unit closer, communication capability would be improved accordingly.

Systems will be tested to ensure communication between stations and to ensure that it can be carried on under the design noise environmental conditions.

9.18 HEAVY LOAD HANDLING

The Beaver Valley position on the seven (7) guidelines of NUREG-0612 section 5.1.1 on "Control of Heavy Loads" is summarized as follows:

1. Guideline 1 Requirement for Safe Load Paths: Beaver Valley has an administrative procedure that defines requirements for use and control of Safe Load Path drawings. This administrative procedure contains most of the approved Safe Load Path drawings but also allows these drawings to be contained in other approved site procedures subject to the review and revision requirements of this administrative procedure. This procedure specifies that all NUREG-0612 heavy load lifts are to be performed using approved heavy load Safe Load Path drawings. This procedure also specifies that issue of new Safe Load Path drawings and changes to existing drawings are to be performed per the site approved 10 CFR 50.59 change process and specifies additional NUREG-0612 requirements that must be considered in making changes to Safe Load Path drawings.

2. Guideline 2 Requirement for Load Handling Procedures: Beaver Valley has an administrative procedure that defines requirements for performance of NUREG-0612 heavy load lifts. This procedure includes a checklist that must be used for performance of all NUREG-0612 heavy load lifts that ensures the requirements of Guideline 2 are met.

3. Guideline 3 on Crane Operator Training: Beaver Valley requires all crane operators to be trained and qualified, and the training program ensures that these personnel are trained and qualified to the requirements of ANSI/ASME B30.2 and other applicable crane standards.

4. Guideline 4 on Special Lifting Devices: The design of the Beaver Valley Unit 1 Special Lifting Devices have been evaluated and determined to be in compliance with the requirements of Guideline 4. In order to ensure continued compliance, the procedures for use of these devices requires performance of a visual examination by qualified personnel of critical welds and parts prior to the initial use each refueling outage. In addition, NDE of all major load carrying welds is performed each ten year in-service inspection interval.

5. Guideline 5 for Slings and Lifting Devices Not Specially Designed: The Beaver Valley administrative procedure for Heavy Loads and the administrative procedure for the Beaver Valley Rigging and Lifting Program both require that all other rigging equipment be designed, inspected and maintained to the applicable ANSI/ASME standards including ANSI/ASME B30.9 "Slings." In addition, the Beaver Valley administrative procedure for heavy loads requires that all slings to be used on NUREG-0612 heavy load lifts be rated for at least twice the normal required capacity to account for the potential effects of dynamic loading. This method has been determined to be conservative for accounting for the sum of both the static and dynamic loading, based on the potential acceleration from the maximum crane speed of all the site cranes capable of handling NUREG-0612 heavy loads.

6. Guideline 6 on Crane Inspection, Testing and Maintenance: The Beaver Valley administrative procedure has a checklist required to be used for performance of all NUREG-0612 Heavy Load Lifts which includes steps to verify completion of the crane periodic inspection (within previous 12 months or prior to use) and the daily "prior to use" visual inspection. In addition, periodic inspection procedures have been generated and scheduled to ensure completion of the required periodic inspections to the requirements of the applicable ANSI/ASME standards including ANSI/ASME B30.2.

7. Guideline 7 on Crane Design: All Beaver Valley cranes that handle NUREG-0612 heavy loads have been evaluated and determined to meet the requirements of Guideline 7 on crane design.

Beaver Valley followed the Nuclear Energy Institute Document, NEI 08-05, "Industry Initiative on Control of Heavy Loads, Section on Load Drop Analyses" concerning the evaluation of reactor vessel head heavy load lifts. In response to this initiative, a postulated reactor vessel closure head drop analysis was performed. For the load drop scenario, it is postulated that the closure head assembly falls and impacts flat and concentrically with the reactor vessel flange.

Using a dynamic finite element model, a closure head assembly drop and impact with the reactor vessel flange was simulated. The responses of the reactor vessel, reactor vessel support components and main loop piping were evaluated. The stresses and strains caused by the impact were evaluated to demonstrate acceptability based on maintaining the structural integrity of the critical components such that core cooling was not compromised and the core remains covered.

The analysis qualified the postulated drop of the closure head assembly 32 feet through air onto the reactor vessel flange. The analysis followed the methodology and assumptions for conducting reactor vessel head drop analyses as provided in NEI 08-05 and endorsed by NRC Regulatory Issue Summary (RIS) 2008-28, "Endorsement of Nuclear Energy Institute Guidance for Reactor Vessel Head Heavy Load Lifts."

The acceptance criteria used for the coolant retaining components, reactor vessel, piping and elbows, is taken from Appendix F, Section F-1341.2 of the ASME Code. The acceptance criteria for reactor vessel support steel were from NEI 08-05 guidance. The acceptance criteria for the concrete support bearing strength in the concrete under the reactor vessel supports was based on NEI 08-05 and ACI 349-97, Section 10.15. The acceptance criteria used from NEI 08-05 is consistent with the criteria endorsed in RIS 2008-28.

The analysis showed that the reactor vessel, main loop piping and the steel and concrete support structure for the reactor vessel are capable of meeting the acceptance criteria under a 32-foot drop of the reactor vessel closure head assembly, through air, onto the reactor vessel flange.

BVPS UFSAR UNIT 1

TABLES FOR SECTION 9

Table 9.1-1
CHEMICAL AND VOLUME CONTROL SYSTEM CODE REQUIREMENTS

Regenerative heat exchanger	ASME III(1), Class C
Nonregenerative heat exchanger	ASME III, Class C, Tube Side ASME VIII(2), Shell Side
Mixed bed demineralizers	ASME III, Class C(5)
Reactor coolant filter	ASME III, Class C
Volume control tank	ASME III, Class C(6)
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
Cation bed demineralizer	ASME III, Class C(5)
Boric acid tanks	ASME VIII
Batching tank	ASME VIII
Seal water injection filters	ASME III, Class C
Boric acid filter	ASME III, Class C
Seal water filter	ASME III, Class C
Resin fill tank	ASME VIII
Piping and valves	ANSI B31.I(3) and ANSI B16.5(4)

(1) ASME III - American Society of Mechanical Engineers, Boilers and Pressure Vessel Code, Section III, Nuclear Vessels.

(2) ASME VIII - American Society of Mechanical Engineers, Boilers and Pressure Vessel Code, Section VIII, Unfired Pressure Vessels.

(3) ANSI B31.1 - Code for Pressure Piping, American Standards Association, and special nuclear cases where applicable.

(4) ANSI B16.5 - Code for Steel Pipe Flanges and Flanged Fittings, American Standards Association.

(5) Mixed bed and Cation bed demineralizer vessels, purchased and constructed ASME III Class C, do not have a safety function. The components are exempted from ASME XI.

(6) Maintenance block valves in the pressure relief path are administratively controlled and locked in the open position during system operation.

Table 9.1-2

CHEMICAL AND VOLUME CONTROL SYSTEM PERFORMANCE REQUIREMENTS

Unit design life, yr	40**
Nominal seal water supply flow rate (three reactor coolant pumps operating), gpm	24
Nominal seal water return flow rate (three reactor coolant pumps operating), gpm	9
Normal letdown flow rate, gpm	60
Maximum letdown flow rate, gpm	120*
Normal charging pump flow rate (one pump including 60 gpm bypass flow), gpm	129
Normal charging line flow, gpm	45
Maximum rate of dilution, ppm per minute	17.5
Maximum rate of boration with one boric acid transfer pump and one charging pump (from initial RCS concentration of 10 ppm), ppm per minute	8
Cooldown rate equivalent to above rate of boration, F per minute	2.3
Rate of boration using two charging pumps and refueling water (from initial RCS concentration of 10 ppm), ppm per minute	6.2
Cooldown rate equivalent to above rate of boration, F per minute	1.7
Normal cooldown rate, F per minute (approx.)	0.8
Temperature of reactor coolant entering system at full power, F (approx.)	528.5 to 543.1
Temperature of coolant return to RCS at full power, F (approx.)	489
Amount of 4 percent boric acid solution required to meet cold shutdown requirements shortly after full power operation, gallons	11,300
Regenerative heat exchanger tube side inlet piping safety relief valve setpoint, psig	2,735

* Letdown may be increased to 180 gpm in Modes 4 and 5 to assist in RCS cleanup.

** The renewed facility operating license has extended the operating term of the plant to 60 years, but did not affect other parameters in this table.

Table 9.1-3

CHEMICAL AND VOLUME CONTROL SYSTEM
PRINCIPAL COMPONENT DATA SUMMARY

Heat Exchangers	Quantity	Heat Transfer Btu Per Hr	Design Flow Lb Per Hr Shell/Tube	Design Inlet Temperature, F Shell/Tube	Design Outlet Temperaure, F Shell/Tube	Design Pressure psig Shell/Tube	Design Temperature, F Shell/Tube
Regenerative	1	8.34×10^6	29,826/22,370	543.5/130	283/489.3	2,485/2,735	650/650
Nonregen-erative	1	15.72×10^6	494,000/59,700	100*/380	132/115	150/160	250/400
Seal Water	1	1.45×10^6	62,000/111,600	100*/138	123.4/125	150/200	250/250
Excess Letdown	1	3.23×10^6	83,500/7,500	100*/543.5	139/135.5	150/2,485	250/650

Pumps	Quantity	Type	Capacity gpm	Head	Design Pressure psig	Design Temperature F
Charging	3	Centrifugal	150	5,800 (min.)	2,800	300
Boric Acid Transfer	2	Centrifugal-2 Speed	75/37.5	235/59	150	250

Tanks	Quantity	Type	Volume	Design Pressure psig	Design Temperature F
Volume Control	1	Vertical	300 ft^3	75 Int/15 Ext.	250

*Temperature may exceed 104°F during periods of maximum river water temperature (90°F).

Table 9.1-3 (CONT'D)

Tanks	Quantity	Type	Volume	Design Pressure psig	Design Temperature F
Boric Acid	2	Vertical	7,500 gal	Atmos.	250
Chemical Mixing	1	Vertical	5.0 gal	150	200
Batching	1	Jacketed	400 gal	Atmos.	300

Filters	Quantity	Filtration Capability	Flow gpm	Design Pressure psig	Design Temperature F
Reactor Coolant	1	*20μ (Absolute)	150	200	200
Seal Water Return	1	*98% of 25μ	325	200	250 200 (epoxy coated glass fiber filter)
Seal Water Injection	2	*20μ (Absolute)	80	2,735	200
Boric Acid	1	*20μ (Absolute)	150	200	200

Demineralizers	Quantity	Type	Design Flow, gpm	Design Pressure psig	Design Temperature F
Mixed bed	2	Flushable	120	200	250
Cation bed	1	Flushable	60	200	250
Deborating	2	Flushable	120	175	200

*Maximum values given; filters designed to remove smaller particulates may be used based on operating conditions.

Table 9.1-3 (CONT'D)

	Quantiy	Design Flow gpm	Design Differential Pressure, psi	Design Pressure psig	Design Temperature F
Letdown orifices	3	1 at 45 2 at 60	1,900	2,485	650

Table 9.2-1

BORON RECOVERY SYSTEM COMPONENT DESIGN DATA

Degasifiers BR-EV-2A & 2B

Number	2
Capacity, each, gpm	75
Design Pressure,	0 psia to 100 psig
Design Temperature, °F	338/F.V. at 100
Material(1)	ASTM Type 304L SS
Design Code	ASME VIII

Coolant Recovery Tanks BR-TK-4A & 4B

Number	2
Capacity, each, gallons	195,000
Design Pressure, psig	0.5 & full of liquid
Design Temperature, °F	200
Material(1)	ASTM Type 304L SS
Design Code	API-650

Boron Evaporators BR-EV-1A & 1B

Number	2
Capacity, each, gpm	15
Design Pressure,	0 psia to 100 psig
Design Temperature, °F	338/F.V. at 100
Material(1)	ASTM Type 304L SS
Design Code	ASME VIII

Primary Water Tanks BR-TK-6A & 6B

Number	2
Capacity, each, gal	75,000
Design Pressure	Full of liquid,
Design Temperature, °F	120
Material(1)	ASTM Type 304L SS
Design Code	API-650

TABLE 9.2-1 (CONT'D)

BORON RECOVERY SYSTEM COMPONENT DESIGN DATA

Evaporator Bottoms Hold Tanks BR-TK-5

Number	1
Capacity, gallons	2,500 (full)
Design Pressure, psig	48
Design Temperature, °F	200
Material[(1)]	ASTM Type 304L SS
Design Code	ASME VIII

Evaporator Distillate Accumulators BR-TK-1A & 1B

Number	2
Capacity, each, gallons	230 (full)
Design Pressure, psig	100
Design Temperature, °F	300
Material[(1)]	Stainless Steel Type 304 SS
Design Code	ASME VIII

Test Tanks BR-TK-2A & 2B

Number	2
Capacity, each, gallons	14,500 (full)
Design Pressure, psig	0.5 & full of liquid
Design Temperature, °F	200
Material[(1)]	ASTM Type 304L SS
Design Code	API-650

Boric Acid Hold Tank BR-TK-7

Number	1
Capacity, gallons	35,000 (full)
Design Pressure, psig	0.5 & full of liquid
Design Temperature, °F	200
Material[(1)]	Type 304L SS
Design Code	API-650

TABLE 9.2-1 (CONT'D)

BORON RECOVERY SYSTEM COMPONENT DESIGN DATA

Degasifier Recovery Heat Exchangers BR-E-12A1, 12A2, & 12B1, 12B2

Number	2 Batteries of 2 Shells Each	
Total Duty, each battery, Btu/hr	3,000,000	
	Shell	Tube
Design Pressure, psig	150	200
Design Temperature, °F	350	350
Material(1)	Type 304 SS	Type 304 SS
Fluid	Gasified Borated Water	Degasified Borated Water
Design Code	ASME VIII	ASME VIII

Degasifier Steam Heaters BR-E-6A & 6B

Number	2	
Total Duty, each, Btu/hr	4,460,000	
	Shell	Tube
Design Pressure, psig	150	150
Design Temperature, °F	360	360
Material(1)	Carbon Steel	Type 304 SS
Fluid	Steam	Gasified Borated Water
Design Code	ASME VIII	ASME VIII

Degasifier Trim Coolers BR-E-10A & 10B

Number	2	
Total Duty, each, Btu/hr	710,000	
	Shell	Tube
Design Pressure, psig	150	200
Design Temperature, °F	350	350
Material(1)	Carbon Steel	Type 304 SS
Fluid	Component Cooling Water	Degasified Borated Water
Design Code	ASME VIII	ASME VIII

TABLE 9.2-1 (CONT'D)

BORON RECOVERY SYSTEM COMPONENT DESIGN DATA

Degasifier Vent Condensers BR-E-11A & 11B

Number	2	
Total, each, Btu/hr	905,000	
	Shell	Tube
Design Pressure, psig	150	150
Design Temperature, °F	350	350
Material[(1)]	Type 304 SS	Type 304 SS
Fluid	Steam and Noncon-densables (Radioactive)	Component Cooling Water
Design Code	ASME VIII	ASME VIII

Degasifier Vent Chillers BR-E-8A & 8B

Number	2	
Total Duty, each, Btu/hr	12,340	
	Shell	Tube
Design Pressure, psig	150	150
Design Temperature, °F	350	350
Material[(1)]	Type 304 SS	Type 304 SS
Fluid	Steam and Noncondens- ables (Radioactive)	Component Cooling Water
Design Code	ASME VIII	ASME VIII

Evaporator Reboilers BR-E-5A & 5B

Number	2	
Duty, each, Btu/hr	9,150,000	
	Shell	Tube
Design Pressure, psig	150	150
Design Temperature, °F	360	360
Material[(1)]	Carbon Steel	Type 316 SS
Fluid	Steam	0-12% Boric Acid
Design Code	ASME VIII	ASME VIII

TABLE 9.2-1 (CONT'D)

BORON RECOVERY SYSTEM COMPONENT DESIGN DATA

Evaporator Distillate Coolers BR-E-2A & 2B

Number	2	
Duty, each, Btu/hr	900,000	
	Shell	Tube
Design Pressure, psig	100	150
Design Temperature, °F	340	340
Material[(1)]	Type 304 SS	Type 304 SS
Fluid	Component Cooling Water	Water (up to 60 ppm Boric Acid)
Design Code	ASME VIII	ASME VIII

Evaporator Bottoms Cooler BR-E-3

Number	1	
Duty, Btu/hr	884,000	
	Shell	Tube
Design Pressure, psig	150	150
Design Temperature, °F	300	300
Material[(1)]	Carbon Steel	Type 304 SS
Fluid	Component Cooling Water	4-12% Boric Acid
Design Code	ASME VIII	ASME VIII

Primary Water Storage Tank Heaters BR-E-13A & 13B

Number	2	
Duty, each, Btu/hr	540,000	
	Shell	Tube
Design Pressure, psig	250	150
Design Temperature, °F	450	400
Material[(1)]	Carbon Steel	Type 304 SS
Fluid	Steam	Water (up to 60 ppm Boric Acid)
Design Code	ASME VIII	ASME VIII

TABLE 9.2-1 (CONT'D)

BORON RECOVERY SYSTEM COMPONENT DESIGN DATA

Degasifier Circulation Pumps BR-P-7A & 7B

Number	2
Type	Horizontal Centrifugal
Motor Horsepower, hp	20
Seal Type	Mechanical
Capacity, each, gpm	75
Head at Rated Capacity, ft	310
Design Pressure, psig	200 at 350°F
Material(1)	
Pump Casing	ASTM A-351 Gr. CF8M
Shaft	SAE 4140
Impeller	ASTM A-351 Gr. CF8M

Evaporator Feed Pumps BR-P-2A & 2B

Number	2 (one required)
Type	Horizontal Centrifugal
Motor Horsepower, hp	10
Seal Type	Mechanical
Capacity, each, gpm	150
Head at Rated Capacity, ft	122
Design Pressure, psig	270 at 350°F
Materials(1)	
Pump Casing	ASTM A-351 Gr. CF8M
Shaft	SAE 4140
Impeller	ASTM A-351 Gr. CF8M

Evaporator Circulation Pumps BR-P-6A & 6B

Number	2
Type	Horizontal Centrifugal
Motor Horsepower, hp	15
Seal Type	Double Mechanical
Capacity, each, gpm	2,200
Head at Rated Capacity, ft	19
Design Pressure, psig	150 at 250°F
Materials(1)	
Pump Casing	Type 316 SS
Shaft	Type 303 SS
Impeller	Type 316 SS

TABLE 9.2-1 (CONT'D)

BORON RECOVERY SYSTEM COMPONENT DESIGN DATA

<u>Evaporator Bottoms Pumps BR-P-1A & 1B</u>

Number	2 (one required)
Type	Horizontal Centrifugal
Motor Horsepower, hp	10
Seal Type	Mechanical
Capacity, each, gpm	44
Head at Rated Capacity, ft	190
Design Pressure, psig	100 at 300°F
Materials[(1)]	
Pump Casing	Type 316 SS
Shaft	Steel
Impeller	Type 316 SS

<u>Evaporator Bottoms Coolant Recirculation Pump BR-P-8</u>

Number	1
Type	Horizontal Centrifugal
Motor Horsepower, hp	5
Seal Type	Mechanical
Capacity, each, gpm	150
Head at Rated Capacity, ft	49
Design Pressure, psig	270 at 300°F
Materials[(1)]	
Pump Casing	Ductile Iron A-395
Shaft	SAE 4140
Impeller	A-276/Type 316 SS

<u>Evaporator Bottoms Hold Tank Pump BR-P-9</u>

Number	1
Type	Horizontal Centrifugal
Motor Horsepower, hp	1
Seal Type	Canned Pump
Capacity, each, gpm	20
Head at Rated Capacity, ft	50
Design Pressure, psig	100 at 200°F
Materials[(1)]	
Pump Casing	Type 316 SS
Shaft	Type 316 SS
Impeller	Type 316 SS

TABLE 9.2-1 (CONT'D)

BORON RECOVERY SYSTEM COMPONENT DESIGN DATA

<u>Evaporator Distillate Pumps BR-P-3A & 3B</u>

Number	2
Type	Horizontal Centrifugal
Motor Horsepower, hp	3/4
Seal Type	Mechanical
Capacity, each, gpm	20
Head at Rated Capacity, ft	32
Design Pressure, psig	150 at 350°F
Materials(1)	
Pump Casing	A-351 Gr. CF8M
Shaft	SAE 4140
Impeller	A-351 Gr. CF8M

<u>Test Tank Pumps BR-P-5A & 5B</u>

Number	2
Type	Horizontal Centrifugal
Motor Horsepower, hp	3
Seal Type	Mechanical
Capacity, each, gpm	50
Head at Rated Capacity, ft	58
Design Pressure, psig	270 at 350°F
Materials(1)	
Pump Casing	A-351 Gr. CF8M
Shaft	SAE 4140
Impeller	A-351 Gr. CF8M

<u>Primary Water Supply Pumps BR-P-10A & 10B</u>

Number	2
Type	Horizontal Centrifugal
Motor Horsepower, hp	30
Seal Type	Mechanical
Capacity, each, gpm	200
Head at Rated Capacity, ft	310
Design Pressure, psig	270 at 300°F
Materials(1)	
Pump Casing	A-351 Gr. CF8M
Shaft	SAE 4140
Impeller	A-351 Gr. CF8M

BORON RECOVERY SYSTEM COMPONENT DESIGN DATA

Boric Acid Hold Tank Pumps BR-P-11A & 11B

Number	2 (one required)
Type	Horizontal Centrifugal
Motor Horsepower, hp	3.5
Seal Type	Canned pump
Capacity, each, gpm	30
Head at Rated Capacity, ft	70
Design Pressure, psig	100 at 200°F
Materials(1)	
Pump Casing	Type 316 SS
Shaft	Type 316 SS
Impeller	Type 316 SS

Primary Grade Water Storage Tank Heater Pumps BR-P-14A & 14B

Number	2
Type	In-line Horizontal Centrifugal
Motor Horsepower, hp	3
Seal Type	Mechanical
Capacity, each, gpm	200
Head at Rated Capacity, ft	26
Design Pressure, psig	150
Materials	
Pump	316SS
Shaft	SAE 4140 Steel
Impeller	316SS

Coolant Recovery Filters BR-FL-1A & 1B

Number	2
Retention Size, microns	5
Filter Element, type	Wound fiber
Capacity Normal, each, gpm	150
Capacity Maximum, each, gpm	150
Housing Material(1)	Type 304 SS
Design Pressure, psig	150
Design Temperature, °F	250
Design Code	ASME VIII

Evaporator Bottoms Filters BR-FL-3A & 3B

Number	2
Retention Size, microns	25
Filter Element, type	Wound fiber
Capacity Normal, each, gpm	20
Capacity Maximum, each, gpm	50
Housing Material(1)	Type 304 SS
Design Pressure, psig	150
Design Temperature, °F	250
Design Code	ASME VIII

BORON RECOVERY SYSTEM COMPONENT DESIGN DATA

Cleanup Filter BR-FL-2

Number	1
Retention Size, microns	5
Filter Element, type	Wound fiber
Capacity Normal, each, gpm	150
Capacity Maximum, each, gpm	150
Housing Material(1)	Type 304 SS
Design Pressure, psig	200
Design Temperature, °F	250
Design Code	ASME VIII

Cesium Removal Ion Exchangers BR-I-1A & 1B

Number	2
Design Flow, gpm/sq ft	21
Resin Type	Cation or Mixed Bed
Resin Active Volume, each, ft^3	35
Design Pressure, psig	175
Design Temperature, °F	200
Material(1)	ASTM Type 304L SS
Design Code	ASME VIII

Cleanup Ion Exchangers BR-I-2A & 2B

Number	2
Design Flow, gpm/sq ft	21
Resin Type	Cation-Anion, Mixed Bed
Resin Active Volume, each, ft^3	35
Design Pressure, psig	175
Design Temperature, °F	200
Material(1)	ASTM Type 304L SS
Design Code	ASME VIII

Evaporator Overhead Condensers BR-E-7A & 7B

Number	2	
Duty, each, Btu/hr	7,800,000	
	Shell	Tube
Design Pressure, psig	100	150
Design Temperature, °F	350	350
Material(1)	Type 304 SS	Type 304 SS
Fluid	Steam (Radioactive, Max 60 ppm Boric acid)	Component Cooling Water
Design Code	ASME VIII	ASME VIII

TABLE 9.2-1 (CONT'D)

BORON RECOVERY SYSTEM COMPONENT DESIGN DATA

Evaporator Bottoms Hold Tank Vent Condenser BR-E-14

Number	1	
Duty, Btu/hr	1,500	
	Shell	Tube
Design Pressure, psig	150	200
Design Temperature, °F	250	250
Material[1]	Carbon Steel	Type 304 SS
Fluid	Component Cooling Water	Saturated Air
Design Code	None	None

Primary Water Makeup Deaerator Degasifier Tower BR-E-15[2]

Number	1
Duty, gpm	10
Quality, ppm	$0.1, 0_2$ 1.0,CO
Fluid	Demineralized water
Material[1]	Type 316 SS
Design Code	ASME VIII

NOTE:

1. Materials listed in this table may be replaced with materials of equivalent design characteristics. The term "equivalent" is described in UFSAR Section 1.8.2, "Equivalent Materials."

2. Equipment is "retired in place."

3. Component cooling inlet temperature range is approximately 80°-104°F depending on river temperature and plant operation.

Table 9.2-2

BORON RECOVERY SYSTEM MALFUNCTION ANALYSIS

	Component	Malfunction	Comments and Consequences
1.	Primary coolant containing dissolved gases	Leak	Reactor coolant fluid admitted to the boron RECOVERY system passes through the degasifier recovery exchangers and the degasifier steam heater before being stripped of radioactive gases and hydrogen in the degasifier. Once flow is admitted to the degasifier recovery exchangers, a flow exists through the degasifier steam heater to the degasifier unobstructed by any stop or regulating valves. The maximum pressure to which the components can be subjected by incoming letdown is governed by the degasifier relief valve setting. The design pressure of each component meets or exceeds this limit. Although the function of automatic control devices makes it very unlikely, the shell side of the degasifier steam heaters may be subject to auxiliary steam overpressure causing internal leakage. Since the steam pressure will be greater than the pressure of the reactor coolant flow through the heat exchanger, no outleakage of reactor coolant or entrained gases is possible. With the normal system arrangement of the degasifiers having steam to the degasifier steam heater isolated, the condensate and vent flowpaths are also isolated. Isolating the three flowpaths should ensure reactor coolant would be contained if the degasifier steam heater experiences a tube leak. Similarly, the degassed fluid flowing through the degasifier recovery exchangers is of a higher pressure than the incoming reactor coolant flow. Any leakage, therefore, will be inleakage of stripped liquid to the reactor coolant flow entering the system, precluding contamination of degassed liquid by liquid containing dissolved gases. The degasifier vent condenser and vent chiller provide a free flow path for stripped gases from the degasifer to the gaseous waste disposal system. Hence, these components can also be subjected to overpressure to the limit of the degasifier relief valve. The design of these components is significantly higher than the pressure at which the degasifier is relieved, thereby posing no danger to the components from overpressure. Stripped gases are collected from the degasifier vent chillers in the waste gas surge tank via the overhead gas compressors. The consequences of the release of the waste gas surge tank contents has been analyzed and are acceptable as reviewed against the accident dose requirements of 10 CFR 100. The details of this analysis are found in Section 11. In view of the foregoing, only minor leaks of nondegasified liquid are anticipated. The extent of contamination by any radioactive liquid is minimized since the equipment cubicles are fitted with floor curbs at the access way to contain liquid within the respective cubicle. Leakage is collected in the auxiliary building sumps of the vent and drain system (Section 9.7) and safely disposed of via the liquid waste disposal system (Section 11.2.4). Any radioactive gas released to the auxiliary building atmosphere is exhausted by the auxiliary building ventilation system via the auxiliary building ventilation vent stack. When a radiation level is detected in excess of its setpoint, a radiation monitor sampling the exhaust stream diverts the auxiliary building exhaust to the main filter banks of the supplementary leak collection and release system (Section 6.6).

Table 9.2-2 (CONT'D)

BORON RECOVERY SYSTEM MALFUNCTION ANALYSIS

	Component	Malfunction	Comments and Consequences
2.	Coolant recovery tank	Leak	Two coolant recovery tanks are provided in the system, and they are located in individual cubicles adjacent to the waste handling building. Each cubicle is designed to accommodate a tank leak of any magnitude, including tank rupture. The walls of the enclosure to the height required to contain the liquid from the rupture of a full tank are missile, tornado, and seismically designed. The personnel entranceways into these cubicles do not breach the required integrity of the enclosure walls. Therefore, a leak of any magnitude poses no radiation hazard to operating personnel or the public.
3.	Degasifier primary and equipment and controls associated with a degasifier train	Fail to function	Two degasifier trains are provided to guarantee performance at 50 percent of maximum reactor coolant letdown. Redundant provision of equipment and components negates loss of the entire degasification process due to any single failure or outage. Only load-follow capacity and startup dilution, which require maximum letdown rates, are affected by a 50 percent reduction in degasifier capacity.
4.	Degassed primary coolant	Leak or Overpressure	The two evaporators are fed by one of two feed pumps, one of which is a spare. The feed to the evaporators is flow control. Evaporation is accomplished by forced circulation through a steam heated reboiler. When aligned, the steam pressure is higher than 15 psig operating pressure of the evaporator system, so that any leakage in the exchanger will be into the evaporator rather than into the steam system. Similarly the condenser is cooled by cooling water higher in pressure than the operating pressure of the evaporator so that any leakage will be into the system. The evaporator system is protected by a relief valve set at 100 psig. All other components of this system are designed for the same pressure or higher. In the event that level in the evaporators is lost due to a severe leak, the operator will shut down the evaporator by push button, closing the feed valve. The evaporator room is provided with a curb to retain leaks if they do occur and to minimize any contamination. Leakage is collected in an auxiliary building sump and disposed of to the liquid waste disposal system. All radioactive gas present in the coolant letdown to the boron recovery system has been stripped from the coolant prior to evaporation. Any trace remaining released to the auxiliary building in the event of a leak is exhausted by the building ventilation system. There are high level alarms on the evaporator and the vaporator distillate accumulator to warn the operator of any malfunction that would indicate liquid building up in the system.

Table 9.2-2 (CONT'D)

BORON RECOVERY SYSTEM MALFUNCTION ANALYSIS

	Component	Malfunction	Comments and Consequences
5.	Primary water	Leak or Overpressure	The primary water contains little radioactivity. The test tanks are located inside the auxiliary building and the primary water storage tanks are located outside. Should a leak occur, the water would be collected in a sump and returned to the waste treatment area. The test tanks and storage tanks are protected from overpressure by open overflows to sumps.
6.	Evaporator Bottoms (4% Boric Acid)	Leak or Overpressure	The evaporator bottoms hold tank is located in the auxiliary building. Protection from radiation is provided by a shielded enclosure. A curb is provided at the entrance to the enclosure containing the tank to retain leaks if they should occur. In case the tank overflows or leaks develop the liquid is collected in a sump and pumped to waste treatment. The boric acid hold tank is protected from tornado by concrete walls. A sump is provided within the area to collect leaks should they occur. A pump will transfer the liquid to waste treatment.

Table 9.3-1

RESIDUAL HEAT REMOVAL SYSTEM CODE REQUIREMENTS

Residual heat exchangers -	Tube Side Shell Side	ASME III, Class C ASME VIII
Residual heat removal piping		ANSI B31.1
Residual heat removal valves		ANSI B31.1 ASME III, 68 ASA B16.5

Table 9.3-2

RESIDUAL HEAT REMOVAL SYSTEM COMPONENT DESIGN DATA

Residual Heat Removal Pumps

Quantity	2
Type	Vertical Centrifugal
Capacity (each), gpm	4,000
Material	Austenitic stainless steel and equivalent corrosion resistant materials
Design pressure, psig	600
Design temperature, F	400

Residual Heat Exchangers

Quantity	2
Type	Shell and U-tube
Design heat transfer rate (each)	
Btu/hr	33×10^6
Shell (component cooling water)	
Design temperature, F	200
Design pressure, psig	150
Design flow rate, lb/hr	5.12×10^6
Material	Carbon steel
Tube (Reactor Coolant)	
Design temperature, F	400
Design pressure, psig	600
Design flow rate, lb/hr	2×10^6
Material	Austenitic stainless steel

Table 9.3-3

RESIDUAL HEAT REMOVAL LOOP MALFUNCTION ANALYSIS

	Component	Malfunction	Comments and Consequences
1.	Residual heat removal pump	Rupture of casing	The casing and shell are designed for 600 psig and 400 F. The pump is protected from overpressurization by a relief valve in the system piping discharging to the pressurizer relief tank. The pump can be inspected and is located in the containment structure with protection against missiles. Rupture is not considered credible.
2.	Residual heat removal pump	Pump fails to start	One operating pump furnishes half of the flow required to meet design cooldown rate. Failure of a pump only increases the time necessary for reactor cooldown
3.	Residual heat removal pump	Stop valve in discharge line closed or check valve sticks closed	Pre-startup and operation checks confirm position of valves. If one discharge line is blocked, the second discharge line still provides flow.
4.	Remote operated valve inside con-tainment in pump suction line	Valve fails to open	Valve position indication light indicates that the valve has not opened. The operator can enter the containment and, utilizing the valve handwheel, open the valve manually or the unit can be cooled and maintained at an intermediate hot shutdown temperature by feed and bleed of the steam generator and steam dump.

TABLE 9.3-3 (CONT'D)

RESIDUAL HEAT REMOVAL LOOP MALFUNCTION ANALYSIS

	Component	Malfunction	Comments and Consequences
5.	Remote operated valve inside con-tainment in pump discharge line.	Valve fails to open	Two valves in parallel. If one valve fails open, flow passes through other valve.
6.	Residual heat exchanger	Tube or shell rupture	Rupture is considered very unlikely because of low operating pressure as compared to design pressure. In any event the faulty heat exchanger can be isolated and the remaining heat exchanger used for cooldown. With only one heat exchanger the time for cooldown is extended.
7.	Valve in bypass line around residual heat exchangers.	Valve sticks open	If valve sticks open part of flow will not pass through RHR heat exchanger. This increases time of cooldown.

Table 9.4-1

COMPONENT COOLING WATER SUBSYSTEM COMPONENT DESIGN DATA

Primary Plant Component Cooling Water Pumps

Number	3
Type	Horizontal centrifugal double suction
Bhp, normal/max	363/412
Seal	Single mechanical
Capacity, gpm	4,700
Head at rated capacity, ft	250
Design pressure, psig	275
Design temperature, °F	275
Materials(3)	
Pump casing	Ductile iron
Shaft	SAE-4340 Steel
Impeller	Bronze

Primary Plant Component Cooling Water Heat Exchangers

Number	3	
Duty, each, Btu/hr	33.0×10^6	
	Shell	Tube
Design pressure, psig	150	75 and full vacuum
Design temperature, °F	200	150
Materials(3)	Carbon steel	A-268, UNS S44660, ferritic stainless steel(2)
Fluids	Component cooling water	River water
Design code	ASME VIII	ASME VIII

Component Cooling Surge Tank

Number	1
Type	Cylindrical, vertical
Capacity, gallon	1,556
Design pressure, psig	50
Design temperature, °F	300
Material(3)	Carbon steel
Design code	ASME VIII

TABLE 9.4-1 (CONT'D)

COMPONENT COOLING WATER SUBSYSTEM COMPONENT DESIGN DATA

Chemical Addition Tank

Number	1
Type	Cylindrical, vertical
Capacity, gal	14
Design pressure, psig	150
Design temperature, °F	150
Material[(3)]	Carbon steel
Design code	ASME VIII

NOTES:

1. River water temperature range is 32 to 90°F.

2. As identified in ASME B&PV Code Case 1922.

3. Materials listed in this table may be replaced with materials of equivalent design characteristics. The term "equivalent" is described in UFSAR Section 1.8.2, "Equivalent Materials."

Table 9.4-2

CHILLED WATER SUBSYSTEM COMPONENT DESIGN DATA*

Rotary Screw Water Chillers	
Number	3
Type	Open type rotary screw water chilling packages
Capacity, tons	600
Temperature of water entering cooler, F	55
Temperature of water leaving cooler, F	45
Water flow through cooler, gpm max	1,080
Chilled Water Circulating Pumps	
Number	3
Type	Horizontal centrifugal pumps with horizontally- split casings
Capacity, gpm	1,080
Bhp, normal/max	67.8/70.4
Head at rated capacity, ft	215
Design pressure, psig	250
Design temperature, F	40-55

*Components of this subsystem also are used to meet building service requirements.

Table 9.4-3

NEUTRON SHIELD TANK COOLING WATER
SUBSYSTEM COMPONENT DESIGN DATA

<u>Neutron Shield Tank Cooler</u>

Number	1	
Duty, Btu/hr	80,000	
	<u>Shell</u>	<u>Tube</u>
Design pressure, psig	100	100
Design temperature, °F	300	300
Operating pressure, psig	25	15
Design operating temperature, in/out, °F	100*/105	125/115
Materials[(1)]	Type 316 SS	Type 316 SS
Fluids	Component cooling water	Shield tank water
Design code	ASME Section VIII	ASME Section VIII

* At maximum river water temperature of 90°F, the component cooling inlet temperature may exceed 104°F.

<u>Neutron Shield Tank Expansion Tank</u>

Number	1
Type	Cylindrical, vertical
Capacity, gallon	1,444
Design pressure, psig	25
Design temperature, °F	150
Material[(1)]	Type 304 SS
Design code	ASME VIII

<u>Corrosion Control Tank</u>

Number	1
Type	Cylindrical, vertical
Capacity, gallon	105
Design pressure, psig	120
Design temperature, °F	150
Material[(1)]	Type 304 SS
Design code	ASME VIII

<u>NOTE</u>:

1. Materials listed in this table may be replaced with materials of equivalent design characteristics. The term "equivalent" is described in UFSAR Section 1.8.2, "Equivalent Materials."

Table 9.4-4

CONSEQUENCES OF COMPONENT MALFUNCTIONS

	Components	Malfunctions	Comments and Consequences
1.	Primary plant component cooling water pumps	Pump casing ruptures	These pumps are designed as Seismic Category I. They are missile protected and may be inspected at any time. Rupture by missiles is not considered credible. Any pump can be isolated by suction and discharge isolation valves. Full system capability can be retained by using the standby pump.
2.	Primary plant component cooling water pumps	Original pump fails to start	Standby pump can be used to achieve full system capability.
3.	Primary plant component cooling water pumps	Standby pump fails to start	The appropriate second standby pump breaker can be racked on to the bus, and the pump can be started remotely from the main control room.
4.	Primary plant cooling water heat exchangers	Tube or shell ruptures	Each heat exchanger can be isolated in case of rupture. The remaining two heat exchangers are capable of performing to 100 percent system capability. Because of the system low operating pressure and temperature, Seismic Category I design, and missile protection arrangements, such rupture is considered unlikely.
5.	System valves	Improper position	Prevented by pre-startup and operational checks. When the system is in service such condition would be observed by operating personnel monitoring system parameters.

TABLE 9.4-4 (CONT'D)

CONSEQUENCES OF COMPONENT MALFUNCTIONS

	Components	Malfunctions	Comments and Consequences
6.	Component cooling water surge tank	Automatic level control fails to operate correctly	Handjack on makeup valve to tank used to manually position valve to achieve correct level.

Table 9.4-5

COMPONENT COOLING SYSTEM
PARAMETERS MONITORED IN CONTROL ROOM

Parameter	Component - Location	Type of Monitor
Pressure	Component cooling water pump bypass	Indicate on control board
Temperature	Neutron shield tank cooling system - upstream of component cooling water heat exchanger	Indicate, alarm, computer
Temperature	Neutron shield tank cooling system - downstream of component cooling water heat exchanger	Computer
Level	Neutron shield tank cooling system - surge tank	Indicate, alarm
Level	Component cooling water - surge tank	Indicate, alarm
Flow	Component cooling water - heat exchanger discharge, 8 in., 14 in., 24 in.	Indicate, alarm
Temperature	Component cooling water - heat exchanger common discharge	Indicate, alarm
Pressure	Component cooling water - pump common discharge	Alarm
Temperature	Containment air recirculation cooling coils - downstream lines	Indicate
Flow	Containment air recirculation cooling coils - downstream lines	Indicate
Temperature	CRDM shroud cooling coils - upstream lines	Computer
Temperature	CRDM shroud cooling coils - downstream lines	Computer
Temperature	Downstream of upper bearing lube oil cooler, each reactor coolant pump	Computer

TABLE 9.4-5 (CONT'D)

COMPONENT COOLING SYSTEM
PARAMETERS MONITORED IN CONTROL ROOM

Parameter	Component - Location	Type of Monitor
Flow	Downstream of upper bearing lube oil cooler, each reactor coolant pump	Indicate, alarm
Temperature	Downstream of stator cooler, each reactor coolant pump	Computer
Flow	Downstream of stator cooler, each reactor coolant pump	Indicate, alarm
Temperature	Downstream of lower bearing lube oil cooler, each reactor coolant pump	Computer
Flow	Downstream of lower bearing lube oil cooler, each reactor coolant pump	Indicate, alarm
Temperature	Combined stator and lube oil cooler flow, each reactor coolant pump	Indicate, alarm
Temperature	Downstream of thermal barrier, each reactor coolant pump	Indicate, alarm, computer
Flow	Downstream of thermal barrier, each reactor coolant pump	Indicate, alarm
Temperature	Downstream residual heat removal pump seal cooler	Computer
Temperature	Downstream neutron shield tank cooler	Computer
Temperature	Downstream excess letdown cooler	Computer

Table 9.4-6

COMPONENT COOLING WATER SYSTEM - REACTOR COOLANT SYSTEM
SINGLE BARRIER DATA

Component	Design Pressure (psig)	Design Temperature (F)	Operating Pressure (psig)	Operating Temperature (F)
1. Reactor coolant pump thermal barriers	2,485	650	2,250	560
2. Sample Coolers:				
a) SS-E-4	2,735	600	2,235	543
b) SS-E-5	2,735	700	2,235	611
c) SS-E-6	2,735	700	2,235	652
d) SS-E-7	2,735	700	2,235	652
3. Residual heat removal pump seal coolers	5,140	800	450	350
4. Residual heat removal heat exchangers	600	400	450	350
5. Excess letdown cooler	2,485	650	2,235	543
6. Non-regenerative heat exchanger	600	400	220	290
7. Seal water heat exchanger	150	250	100	136

Table 9.5-1

FUEL POOL COOLING AND PURIFICATION SYSTEM
COMPONENT DESIGN DATA

Fuel Pool Pumps

Number	2
Type	Horizontal centrifugal
Capacity, gpm (each)	750
Head at rated capacity, ft	74
Discharge pressure, psig	45
Design casing pressure, psig	240
Design temperature, °F	200
Pump speed, rpm	1,770
Motor horsepower	20
Materials,[(1)] wetted surfaces	Austenitic stainless steel

Pool Purification Pumps

Number	2
Type	Horizontal centrifugal
Capacity, gpm (each)	50-400
Head at rated capacity, ft	158-143
Discharge pressure, psig	80-74
Design casing pressure, psig	240
Design temperature, °F	200
Pump speed, rpm	1,750
Motor horsepower	30
Materials,[(1)] wetted surfaces	Austenitic stainless steel

Fuel Pool Heat Exchangers

Number	2
Design duty, Btu/hr (each)	11.4×10^6

	Shell	Tube
Fluid	Component cooling water	Fuel pool water
Flow, lb/hr	550,000	375,000
Operating temperature in/out, °F	100/121	140/110
Operating pressure, inlet, psig	45	45
Pressure drop, psi	10	11
Design pressure, psig	150	150
Design temperature, °F	200	200
Materials[(1)]	SA515-70	SA213, Type 304 SS

TABLE 9.5-1 (CONT'D)

FUEL POOL COOLING AND PURIFICATION SYSTEM COMPONENT DESIGN DATA

Fuel Pool Ion Exchanger

Number	1
Resin volume, cu ft	5
Resin Type	See Note 2 below
Operating flow, rated, gpm	150
Operating pressure, psig	80
Design pressure, psig	150
Operating temperature, °F	140
Design temperature, °F	200
Materials,[1] normally wetted surfaces	Austenitic stainless steel

Pool Filters

Number	2
Type elements	Cotton-wound
Size range, microns	1 to 125
Operating temperature, °F	140
Design temperature, °F	250
Operating pressure, psig	80
Design pressure, psig	150
Pressure drop, clean, psi	5
Rated flow, gpm	400
Materials,[1] wetted surfaces	Austenitic stainless steel

NOTE:

1. Materials listed in this table may be replaced with materials of equivalent design characteristics. The term "equivalent" is described in UFSAR Section 1.8.2, "Equivalent Materials."

2. The resin type is normally mixed bed resin; however, cation resin, anion resin, or charcoal may be used to maintain system purity.

Table 9.6-1

SAMPLING SYSTEM SAMPLES

High Temperature Samples

1. Pressurizer vapor

2. Pressurizer liquid

3. Residual heat removal liquid taken downstream of the residual heat removal pumps

4. Residual heat removal liquid taken downstream of the residual heat exchangers

5. Coolant taken from each of the reactor coolant loops

6. Steam generator blowdown liquid taken from each of the blowdown lines

* 7. Main steam taken from each of the main steam lines

* 8. Steam generator feedwater at the suction and discharge of the feedpumps

* 9. Heater drain pumps suction and discharge

Low Temperature Samples

1. Safety injection system accumulators (three)

2. Reactor coolant to chemical and volume control system demineralizers

3. Reactor coolant from chemical and volume control system demineralizers, upstream of reactor coolant filter

4. Reactor coolant effluent from chemical and volume control system cation demineralizer

5. Reactor coolant effluent from chemical and volume control system deborating demineralizers effluent (two)

6. Reactor coolant effluent from chemical and volume control system mixed bed demineralizers (two)

7. Volume control tank reactor coolant

8. Volume control tank gas space

TABLE 9.6-1 (CONT'D)

SAMPLING SYSTEM SAMPLES

9. Pressurizer relief tank gas space

*10. Component cooling water

11. Stripped reactor coolant from coolant recovery tanks

12. Degasifiers effluent stripped liquid (two)

13. Reactor coolant from each charging pump discharge

*14. Condenser hotwell

*15. Condensate pump discharge header

*16. Chilled water

*NOTE: Samples denoted by asterisks are nonradioactive and run to sample panel/sink in the turbine building. Remainder of samples are possibly contaminated and run to sample panel/sink in the auxiliary building.

Table 9.7-1

LIST OF DRAIN TANKS AND SUMPS IN
VENT AND DRAIN SYSTEM

Tank or Sump	Source of Liquid Waste	Pumps	Vented to Gaseous Waste Disposal via "Sweep Gas"
	Nonaerated Drains		
Primary Drain Transfer Tank No. 1, DG-TK-1	Valve stem leakoff within containment via pressurizer relief tank. Reactor coolant loops. Seal leakoff etc.	DG-P-1A and 1B	No
Primary Drain Transfer	Valve steam leakoff outside containment. Volume control tank drain. Sample system header and charging pump drains.	DG-P-2A and 2B	No
	Aerated Drains		
Tunnel Sump (Reactor Containment)	Tunnel piping leaks and equipment drains. Valve stem leakage from safety injection system. Chem. laboratory sinks. Thermal relief valves from chilled water system.	DA-P-6A and 6B (2)	No
Reactor Containment Sump	Safety injection accumulator relief valves. Tell-tale drain lines from refueling cavity. Drains from neutron shield tank cooling water subsystem. Drains going to DG-TK-1 can alternatively be routed to this sump.	DA-P-4A and 4B (2)	No

TABLE 9.7-1 (CONT'D)

LIST OF DRAIN TANKS AND SUMPS IN
VENT AND DRAIN SYSTEM

Tank or Sump	Source of Liquid Waste	Pumps	Vented to Gaseous Waste Disposal via "Sweep Gas"
	Aerated Drains (Cont'd)		
Incore Instrumentation Room Sump	Floor Drains	DG-P-5[1]	No
Fuel Building Sump	Floor drains from handling of spent fuel. Overflow from BR-TK-6A and B.	DG-P-2A and 2B[4]	No
Fuel Pool Tell-Tale Drains Receiver, DA-TK-1	Fuel pool water from leak test of spent fuel pool.	DA-P-10[3]	No
Safeguards Area Sump	Low Level leakage from safety injection suction. Leakage from containment piping. Valve steam leakage from safety injection system.	DA-P-1A and 1B[2]	
Decontamination Sump	Water from washing of contaminated items.	DA-P-7A and 7B[4]	No
Primary Water Storage Tank Sump	Primary water leakage around BR-TK-6A and B and pumps. Overflow from local steam condensate receiver.	DA-P-9[4]	
Boric Acid Hold Tank Sump	Leakage of 4% boric acid evaporator bottoms from pumps, piping etc.	DA-P-11[4]	No

TABLE 9.7-1 (CONT'D)

LIST OF DRAIN TANKS AND SUMPS IN
VENT AND DRAIN SYSTEM

Tank or Sump	Source of Liquid Waste	Pumps	Vented to Gaseous Waste Disposal via "Sweep Gas"
	Aerated Drains (Cont'd)		
Auxiliary Building - North Sump	Drain from main steam condensate system. Drain from boric acid tank. Drains and overflows from tanks, pumps, and filters in liquid waste disposal area, with a few drains from boron recovery area. Tank and filter drains from charging and volume control system. Drain from sinks in the sample rooms. Steam generator blowdown. Building floor drains.	DA-P-3A and 3B (4)	Yes
Auxiliary Building - South Sump	Drains and overflows from tanks, pumps, and filters in boron recovery area with a few drains from liquid waste disposal area. Drains from filters in fuel pool cooling and purification system. Building floor drains.	DA-P-3C and 3D (4)	Yes
Auxiliary Building - Southwest Sump	Relief valve discharge and tank drain from primary drain transfer system No. 2. Local spillage.	DA-P-8A and 8B (4)	Yes

TABLE 9.7-1 (CONT'D)

LIST OF DRAIN TANKS AND SUMPS IN
VENT AND DRAIN SYSTEM

Tank or Sump	Source of Liquid Waste	Pumps	Vented to Gaseous Waste Disposal via "Sweep Gas"
	Aerated Drains (Cont'd)		
East Waste Area Sump	Drains from tanks, pumps and filters in the solid waste disposal area. Floor drains in solid waste disposal area.	SW-P-2A and 2B(4)	Yes

(1) This pump transfers drains from incore instrumentation room sump to reactor containment sump.
(2) These pumps discharge to auxiliary building sump, north.
(3) Pumped to nearest floor drain.
(4) Pumped to either high or low level waste drain tanks depending on radioactivity.
(5) Pumped to east waste area sump.

Table 9.9-1

OPERATION OF RIVER WATER VALVES

Accident	Initial Valve Action River Water Valves
Design Basis Accident Loss-of-Coolant Coincident with Loss-of-Normal Station Power	1. Open valves to recirculation spray coolers
	2. Open valves to emergency diesel generator cooling system heat exchangers
	3. Close valves to primary plant component cooling heat exchangers
Loss-of-Normal Station Power	1. Open valves to emergency diesel generator cooling system heat exchangers
	2. All other valves remain in their normal operating position

Table 9.9-2

RIVER WATER SYSTEM COMPONENT
DESIGN DATA

River Water Pumps

Number	3
Type	Vertical turbine
Motor hp	500
Seals	Mechanical
Capacity, gpm	9,000
Head at rated capacity, ft	165
Design pressure, psig	120
Design temperature, °F	180
Materials[1]	
Discharge head	Carbon steel
Shaft	Type 416 SS
Impeller	Type 316 SS
Column	Carbon steel

NOTE:

1. Materials listed in this table may be replaced with materials of equivalent design characteristics. The term "equivalent" is described in UFSAR Section 1.8.2, "Equivalent Materials."

Table 9.9-3

RIVER WATER SYSTEM FLOW CONDITIONS
(ONE-PUMP BASIS)

System	Minimum Flow Required (gpm)	Number of Exchangers: Minimum Required	Number of Exchangers: Furnished
Design Basis Accident Recirculation Spray Heat Exchangers (3,900 gpm each)	7800	2	4
Diesel Generator Cooling System Heat Exchangers	350	1	2
Charging Pump Lube Oil Coolers	20	1	3
Control Room River Water Cooling Coil	100	1	2
Total	8,270		

Table 9.13-1

MAIN CONTROL AREA AIR CONDITIONS SYSTEMS
COMPONENT DESIGN DATA

Main Control Area Air Conditioning Units

Number	2
Capacity, cfm	33,500
Static Pressure, W.G.	5.12
Motor, Hp	60
Refrigeration Capacity, Mbh	810
Filter type	Roll plus high efficiency

Main Control Area Freon Condensing Units

Number	2
Motor, Hp	75
Refrigeration Capacity Mbh	840

Return Air Fans

Number	2
Capacity, cfm	33,200
Static Pressure, W.G.	4.375
Motor, Hp	40

Emergency Outside Air Supply Fans

Number	2
Capacity, cfm	1000
Static Pressure, W.G.	8.0
Motor, Hp	5
Filter type	Prefilter Charcoal and Particulate

Table 9.13-2

MAIN CONTROL ROOM CHARCOAL FILTER DATA

Flow	800 to 1000 cfm
Residence Time	0.301 secs. to 0.241 sec.
Amount of Charcoal	153 lbs. (approximately)
Charcoal Bed Thickness	2 in.
Type of Charcoal	ANSI N509, 1975 Nuclear Grade Carbon
Impregnants	Triethylene DiAmine and potassium iodide

Table 9.14-1

FUEL OIL SYSTEMS STORAGE CAPACITY

System	Total Storage Capacity, Gallons	No. of Tanks	Location	Criteria
Emergency diesel generators	40,000 (20,000 each)	2	Underground	7-day supply for one emergency diesel generator at full load
Emergency diesel generators	8,000	I	Underground	Receive and hold fuel oil while tests are performed
Engine-driven fire pump	450	1	Above ground inside missile protected building	8-hour supply at full load
ERFS diesel generator (Non-IE)	approx 29,100 available as configured	1	Underground	7-day supply with required loads (not safety related)

Table 9.14-2

MAJOR OIL DISTRIBUTORS WITHIN 40 MILES OF SITE[1]

Supplier	Distance from site, miles	Normal supply on hand, gallons	Tank truck capacity, gallons	Time req'd to make initial delivery, hours
Ashland	8	50,000 to 1,000,000	8,000	1
Texaco	18	945,000	6,700	1 1/4
Gulf	40	6,000,000	6,500	4
Universal	25	50,000	6,400	1 1/2
Mobil	4	1,300,000	6,700	3/4
Phillips	30	20,000	6,700	2

(1) As of 1973.

Table 9.15-1

DECONTAMINATION FACILITY COMPONENT DESIGN DATA

Fluid Waste Treating Tank

Number	1
Capacity, gallon	1400
Design pressure, psig	40
Design temperature, °F	120
Operating pressure	Atmospheric
Operating temperature, °F	70 to 120
Material[(1)]	Type 304 SS
Design Code	ASME VIII

Chemical Addition Tank

Number	1
Capacity, gallon	14
Design pressure	Full of water
Design temperature, °F	120
Operating pressure	Atmospheric
Operating temperature, °F	45 to 100
Material[(1)]	Type 304 SS
Design Code	None

Fluid Waste Treating Tank Filter

Number	1
Retention size, microns	5
Filter element, type	Wound fiber
Capacity, normal, gpm	30
Capacity, maximum, gpm	50
Housing material[(1)]	SS 304
Design pressure, psig	150
Design temperature, °F	250
Design code	ASME VIII

TABLE 9.15-1 (CONT'D)

DECONTAMINATION FACILITY COMPONENT DESIGN DATA

Fluid Waste Treating Tank Pump

Number	1
Type	Horizontal Centrifugal
Motor horsepower, hp	5
Seal type	Mechanical
Capacity, each, gpm	30
Heat at rated capacity, ft	50
Materials[(1)]	
Pump casing	AISI Type 316 SS
Shaft	AISI Type 316 SS
Impeller	AISI Type 316 SS

NOTE:

1. Materials listed in this table may be replaced with materials of equivalent design characteristics. The term "equivalent" is described in UFSAR Section 1.8.2, "Equivalent Materials."

Table 9.16-1

AUXILIARY RIVER WATER PUMP
DESIGN DATA

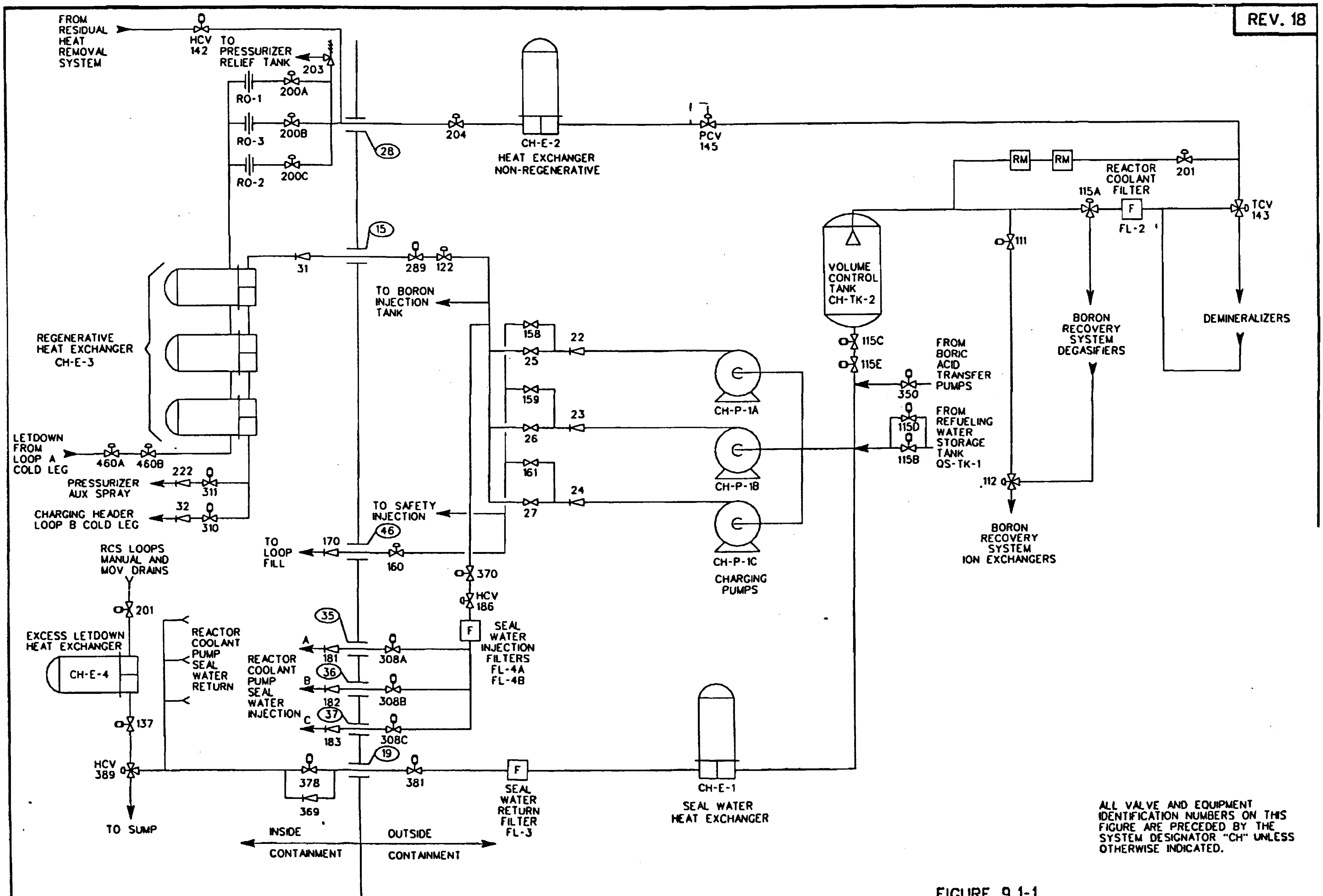
Number	2
Type	Vertical Turbine
Motor horsepower, hp	500
Seal Type	Mechanical
Capacity, each, gpm	9,000
Heat at rated capacity, ft	195
Design pressure, psig	120

Table 9.17-1

COMMUNICATIONS SYSTEMS AVAILABLE
FOR LOCAL CONTROL OF BYPASS VALVES

	Main Control Room	Emergency Shutdown Panel	Valve Operation Aisle	Back-Up Indicating Panel
1 page - 5 party line	X	X	X	X
PAX Tel.	X	X	X (1 each near stairway 1A and 2A)	X
Bell Tel.	X			
Two party jack system	X	X	X	
VHF high band radio handsets	X	X		
Alternate Safe Shutdown Communication System			X	X

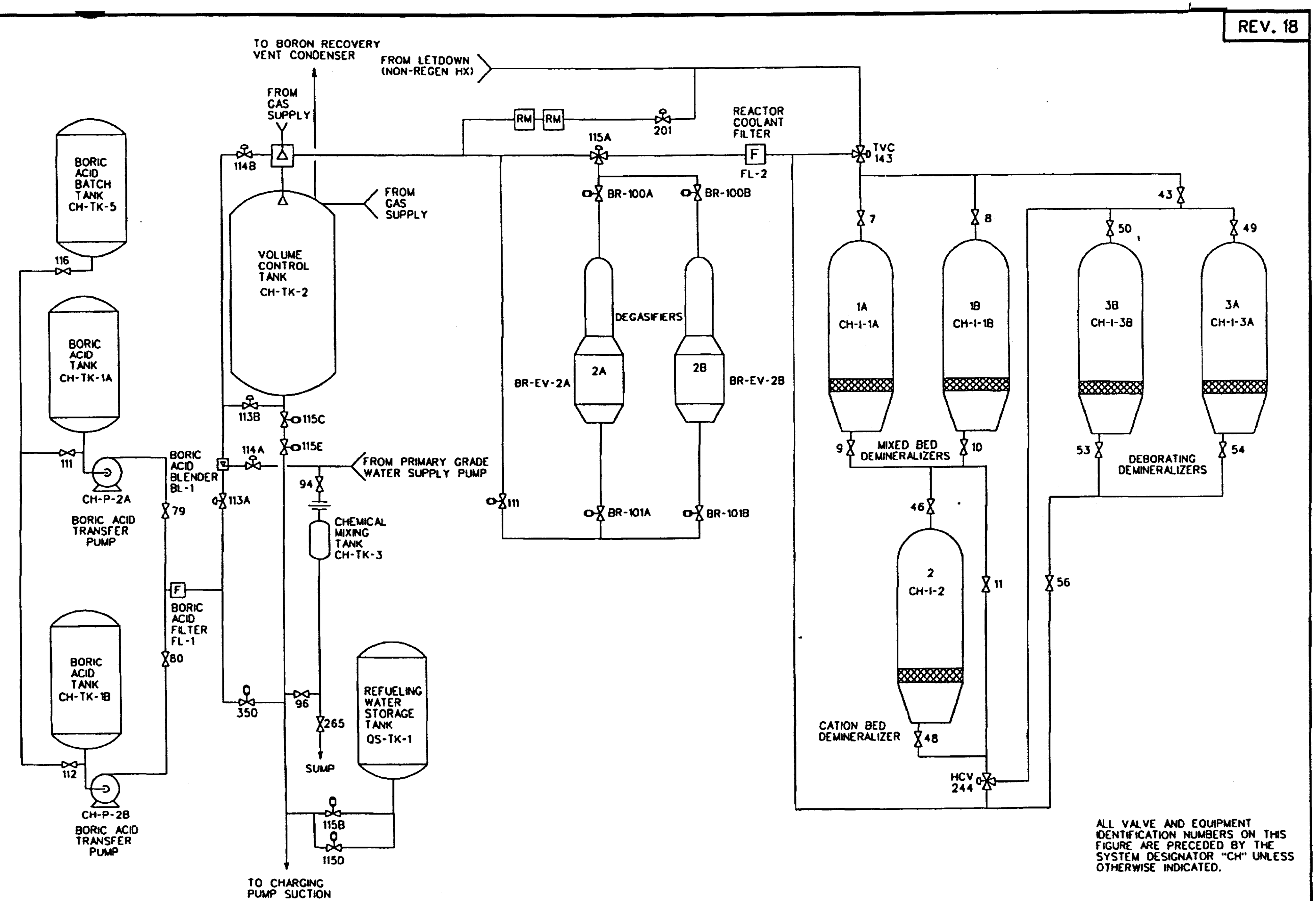
REV. 18



ALL VALVE AND EQUIPMENT IDENTIFICATION NUMBERS ON THIS FIGURE ARE PRECEDED BY THE SYSTEM DESIGNATOR "CH" UNLESS OTHERWISE INDICATED.

FIGURE 9.1-1
CHEMICAL AND VOLUME CONTROL SYSTEM

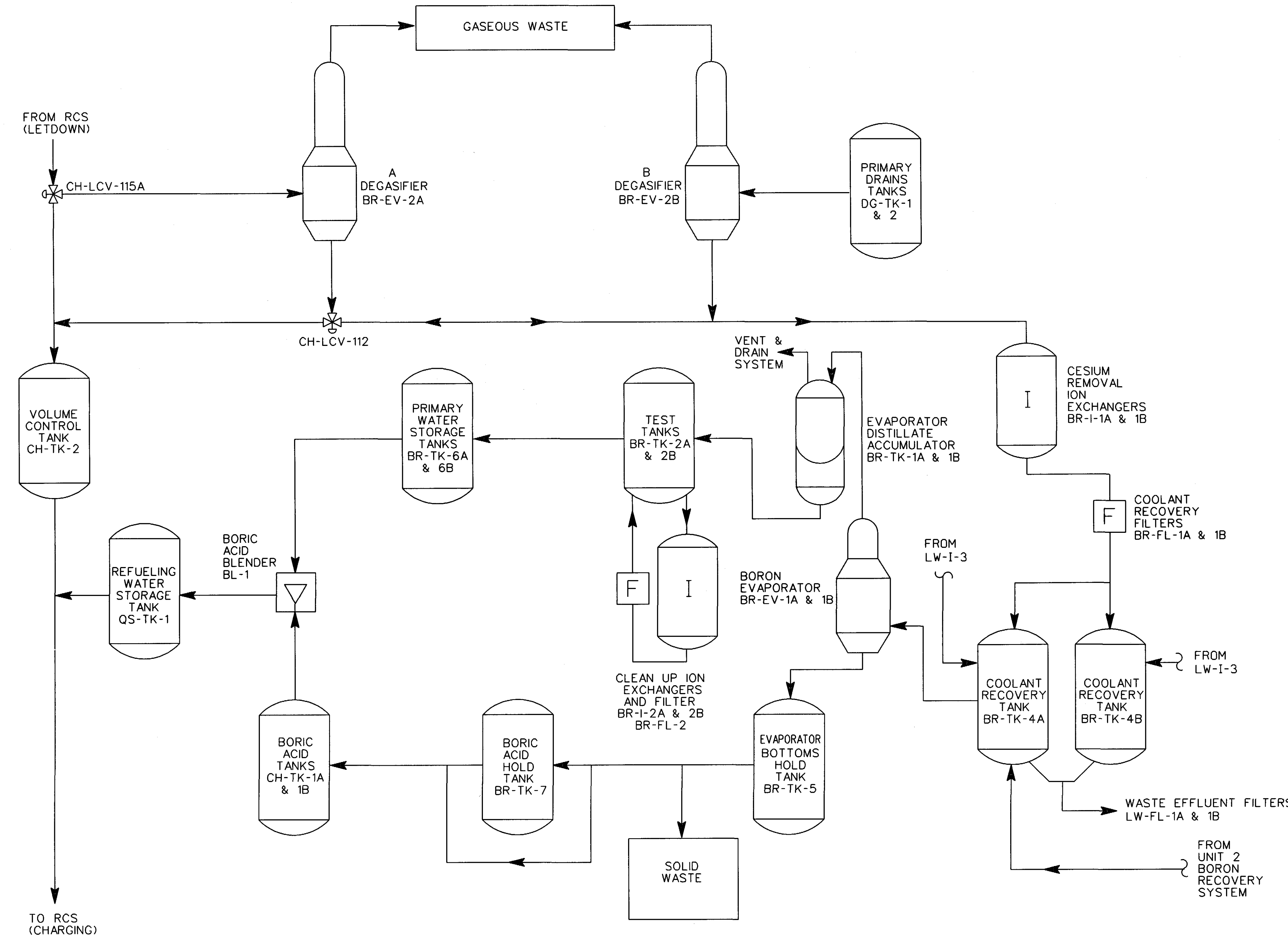
REFERENCE: STATION DRAWING RM-39A
BEAVER VALLEY POWER STATION UNIT NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT



ALL VALVE AND EQUIPMENT IDENTIFICATION NUMBERS ON THIS FIGURE ARE PRECEDED BY THE SYSTEM DESIGNATOR "CH" UNLESS OTHERWISE INDICATED.

FIGURE 9.1-2
CHARGING AND VOLUME CONTROL SYSTEM

REFERENCE: STATION DRAWING RM-39B
BEAVER VALLEY POWER STATION UNIT NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT

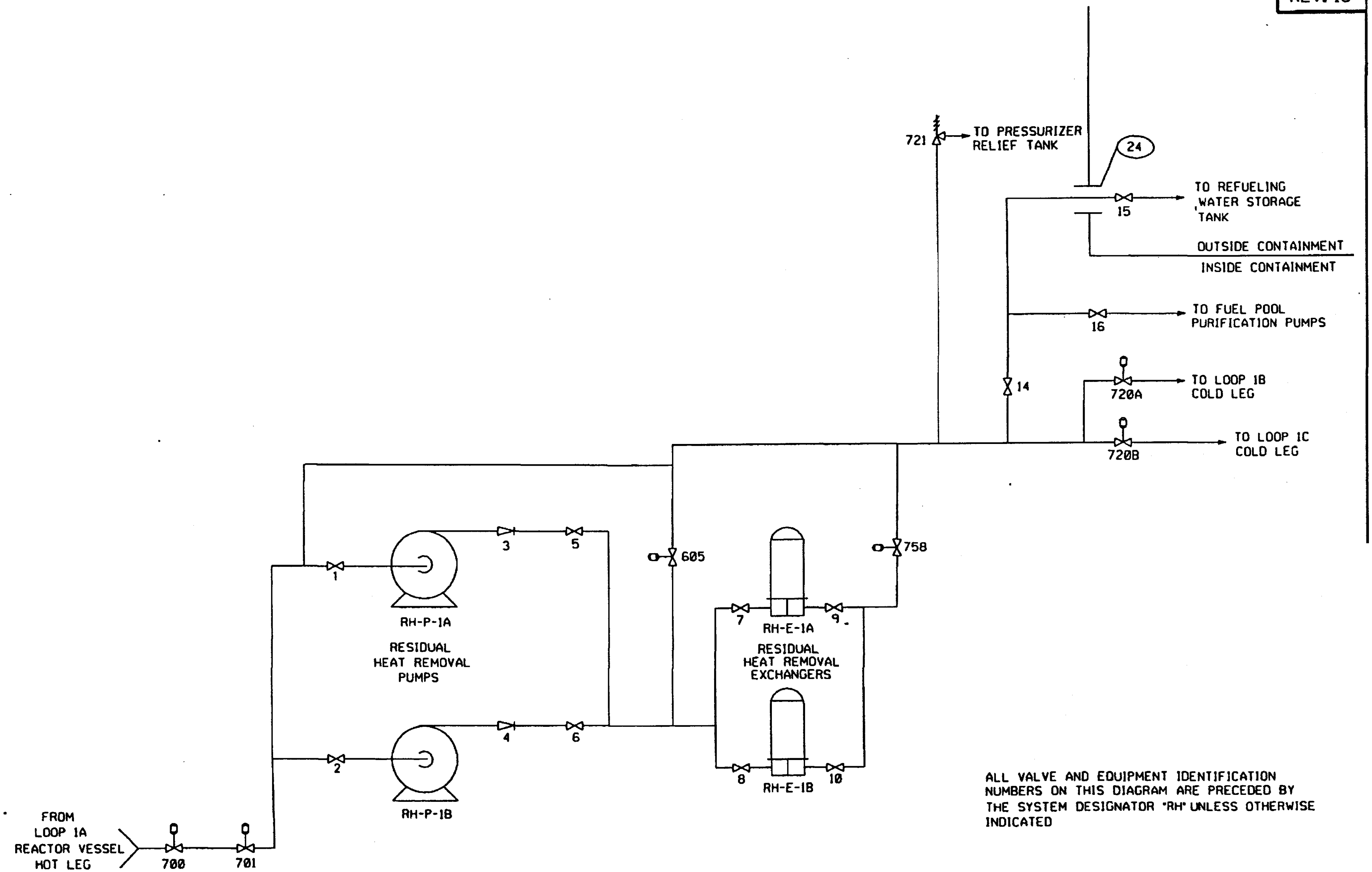


ALL VALVE AND EQUIPMENT IDENTIFICATION NUMBERS ON THIS FIGURE ARE PRECEDED BY THE SYSTEM DESIGNATOR "BR" UNLESS OTHERWISE INDICATED.

FIGURE 9.2-1
BORON RECOVERY SYSTEM

REFERENCE: STATION DRAWING RM-29A
BEAVER VALLEY POWER STATION UNIT NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT

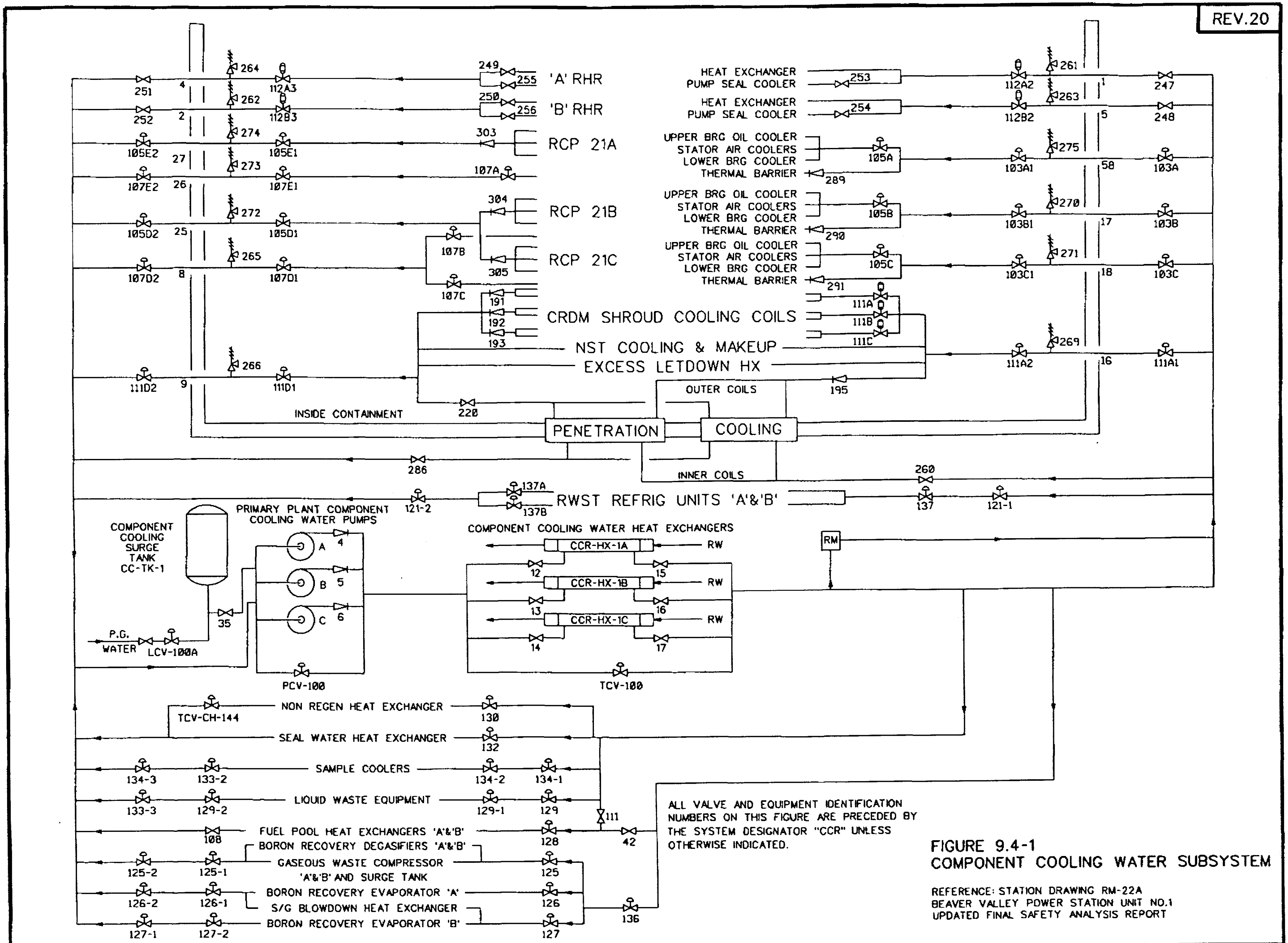
REV. 18



ALL VALVE AND EQUIPMENT IDENTIFICATION NUMBERS ON THIS DIAGRAM ARE PRECEDED BY THE SYSTEM DESIGNATOR "RH" UNLESS OTHERWISE INDICATED

FIGURE 9.3-1
RESIDUAL HEAT REMOVAL SYSTEM

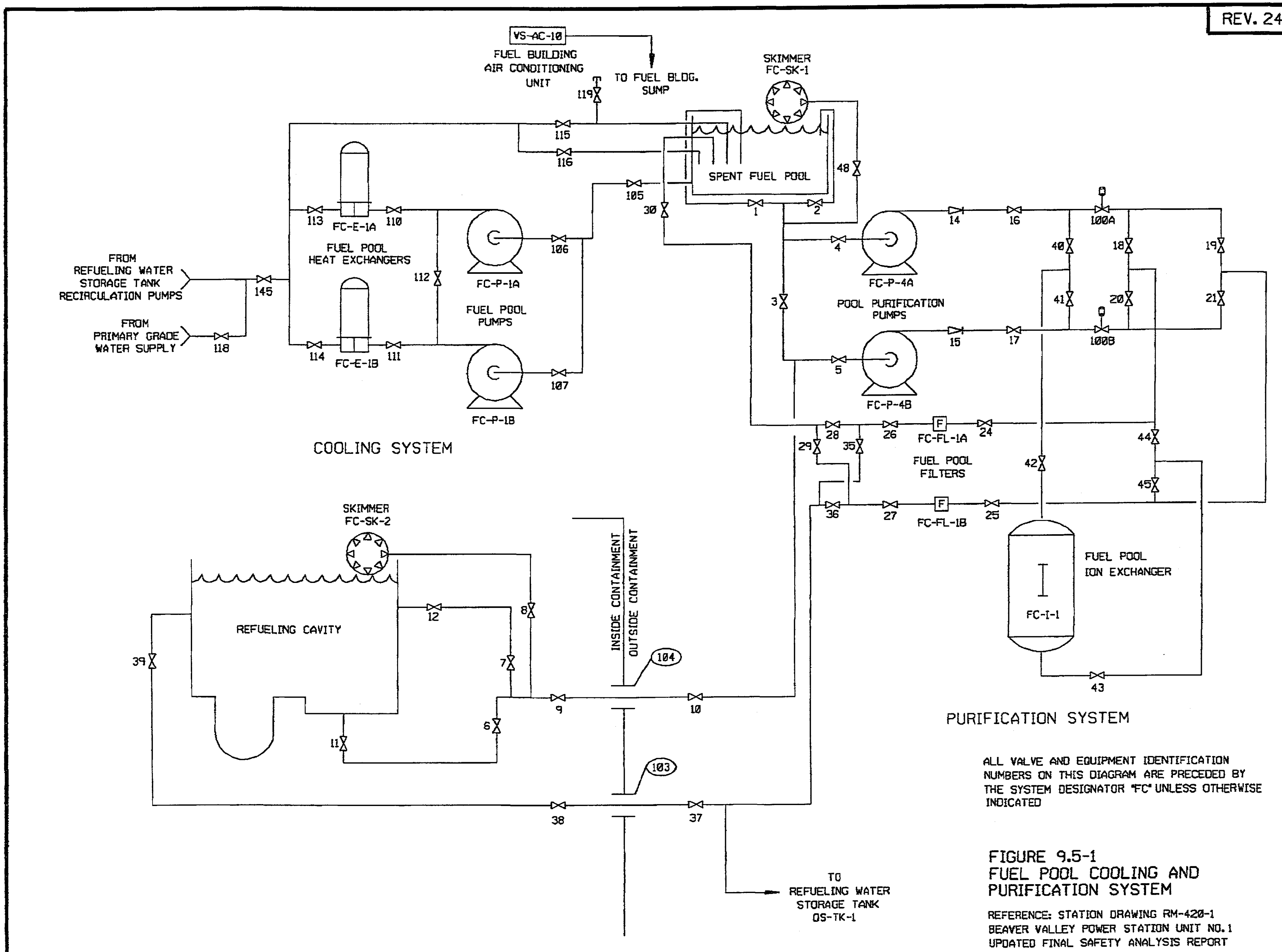
REFERENCE: STATION DRAWING RM-510
BEAVER VALLEY POWER STATION UNIT NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT



ALL VALVE AND EQUIPMENT IDENTIFICATION NUMBERS ON THIS FIGURE ARE PRECEDED BY THE SYSTEM DESIGNATOR "CCR" UNLESS OTHERWISE INDICATED.

FIGURE 9.4-1
COMPONENT COOLING WATER SUBSYSTEM

REFERENCE: STATION DRAWING RM-22A
BEAVER VALLEY POWER STATION UNIT NO.1
UPDATED FINAL SAFETY ANALYSIS REPORT



ALL VALVE AND EQUIPMENT IDENTIFICATION NUMBERS ON THIS DIAGRAM ARE PRECEDED BY THE SYSTEM DESIGNATOR "FC" UNLESS OTHERWISE INDICATED

FIGURE 9.5-1
FUEL POOL COOLING AND PURIFICATION SYSTEM

REFERENCE: STATION DRAWING RM-420-1
BEAVER VALLEY POWER STATION UNIT NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT

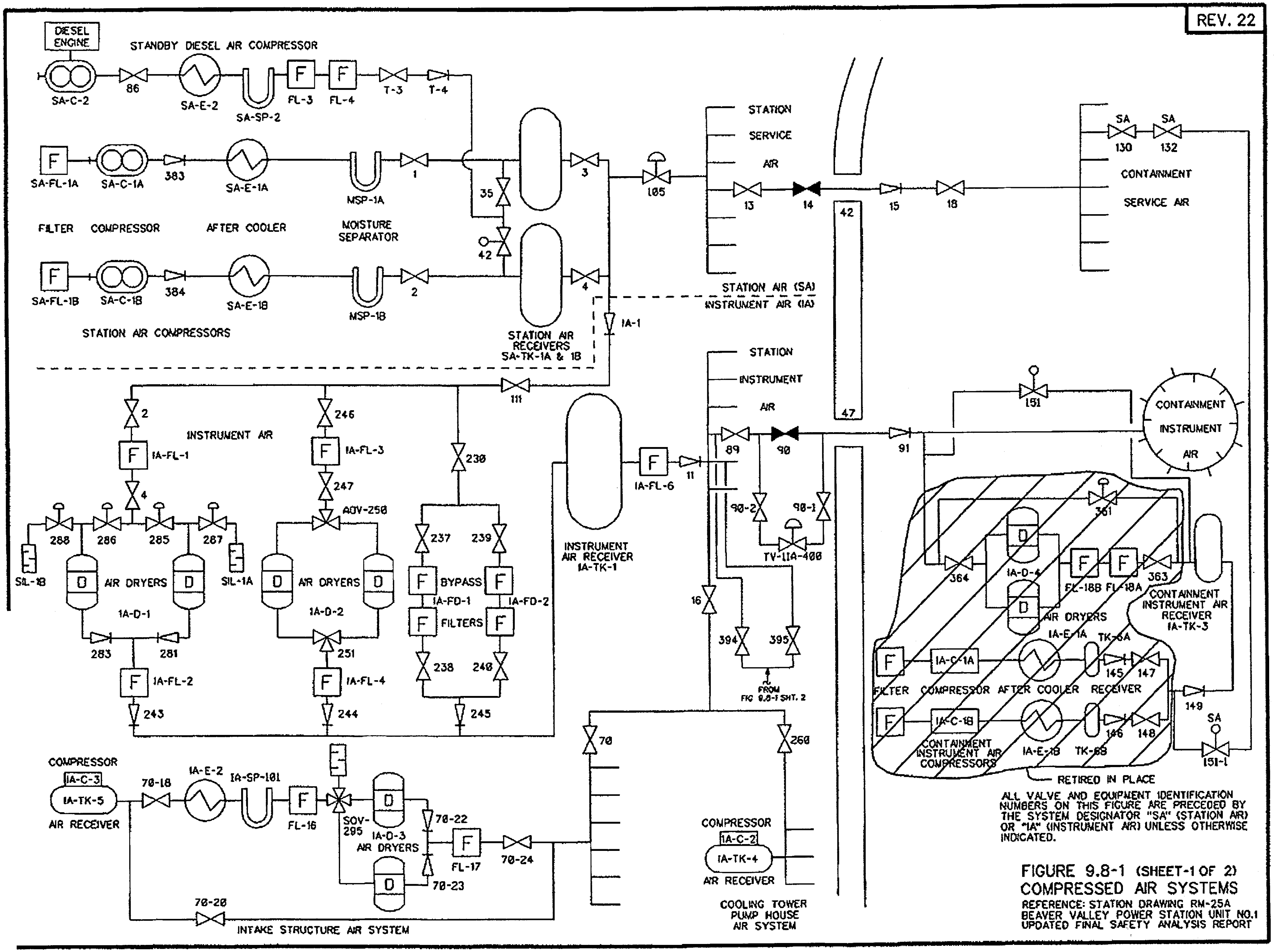
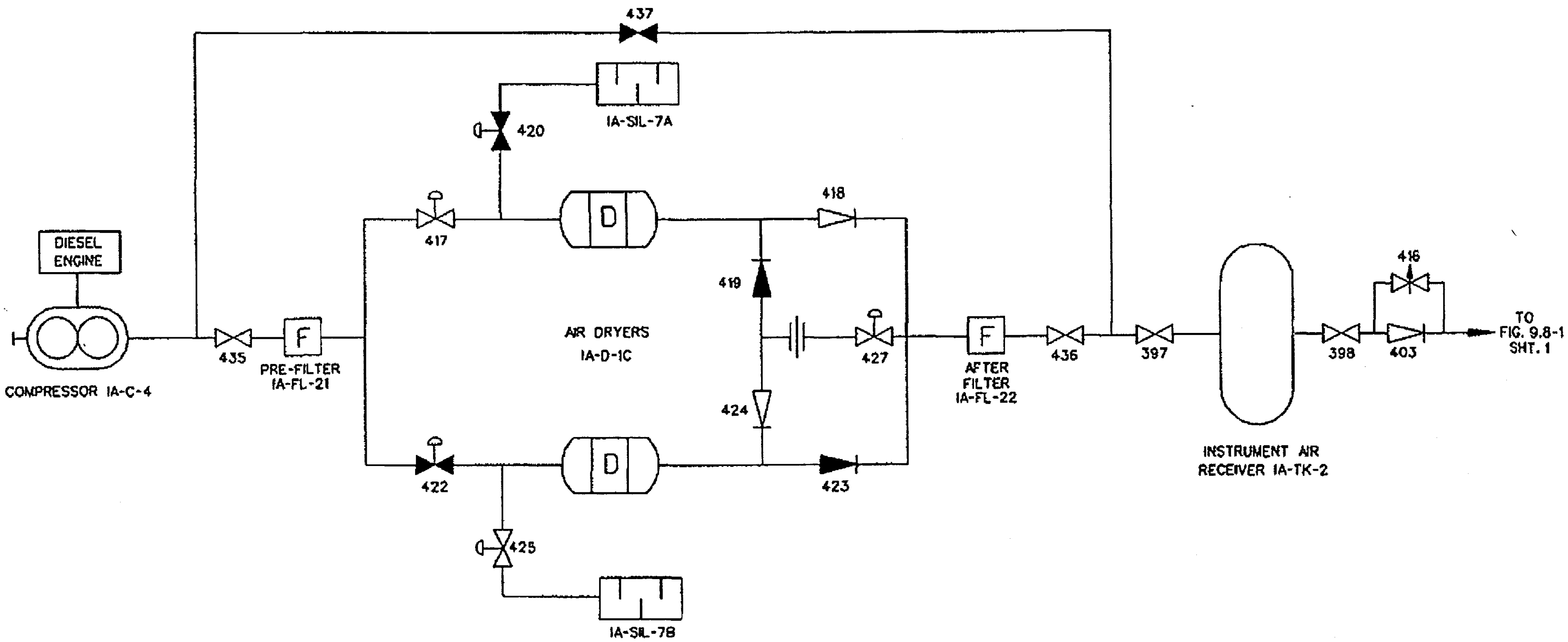


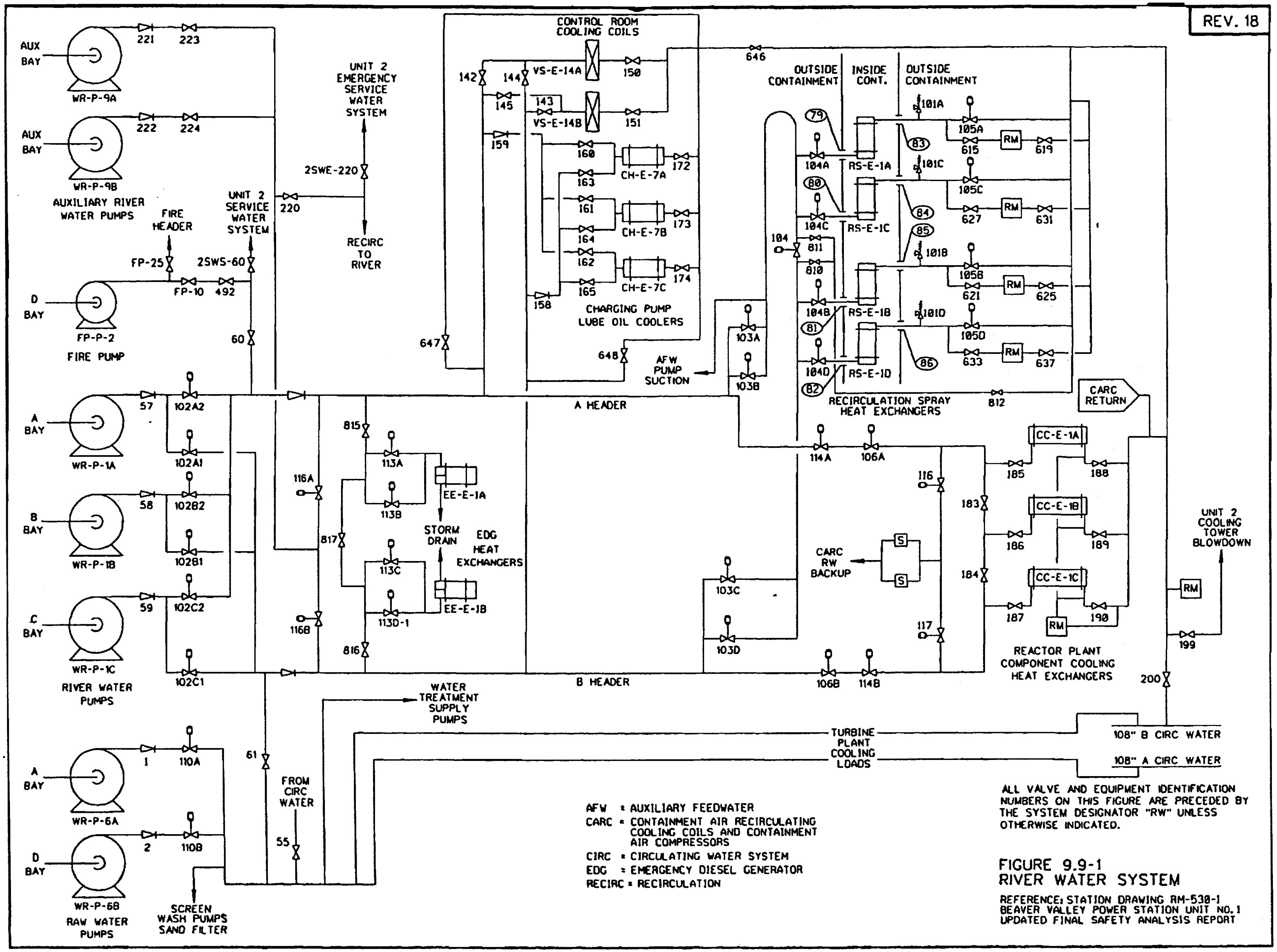
FIGURE 9.8-1 (SHEET-1 OF 2)
COMPRESSED AIR SYSTEMS
REFERENCE: STATION DRAWING RM-25A
BEAVER VALLEY POWER STATION UNIT NO.1
UPDATED FINAL SAFETY ANALYSIS REPORT

REV. 22



ALL VALVE AND EQUIPMENT IDENTIFICATION NUMBERS ON THIS FIGURE ARE PRECEDED BY THE SYSTEM DESIGNATOR "SA" (STATION AIR) OR "IA" (INSTRUMENT AIR) UNLESS OTHERWISE INDICATED.

FIGURE 9.8-1 (SHEET-2 OF 2)
COMPRESSED AIR SYSTEMS
REFERENCE: STATION DRAWING RM-25A
BEAVER VALLEY POWER STATION UNIT NO.1
UPDATED FINAL SAFETY ANALYSIS REPORT



AFW = AUXILIARY FEEDWATER
CARC = CONTAINMENT AIR RECIRCULATING COOLING COILS AND CONTAINMENT AIR COMPRESSORS
CIRC = CIRCULATING WATER SYSTEM
EDG = EMERGENCY DIESEL GENERATOR
RECIRC = RECIRCULATION

ALL VALVE AND EQUIPMENT IDENTIFICATION NUMBERS ON THIS FIGURE ARE PRECEDED BY THE SYSTEM DESIGNATOR "RW" UNLESS OTHERWISE INDICATED.

FIGURE 9.9-1
RIVER WATER SYSTEM

REFERENCE: STATION DRAWING RM-530-1
BEAVER VALLEY POWER STATION UNIT NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT

REV. 20

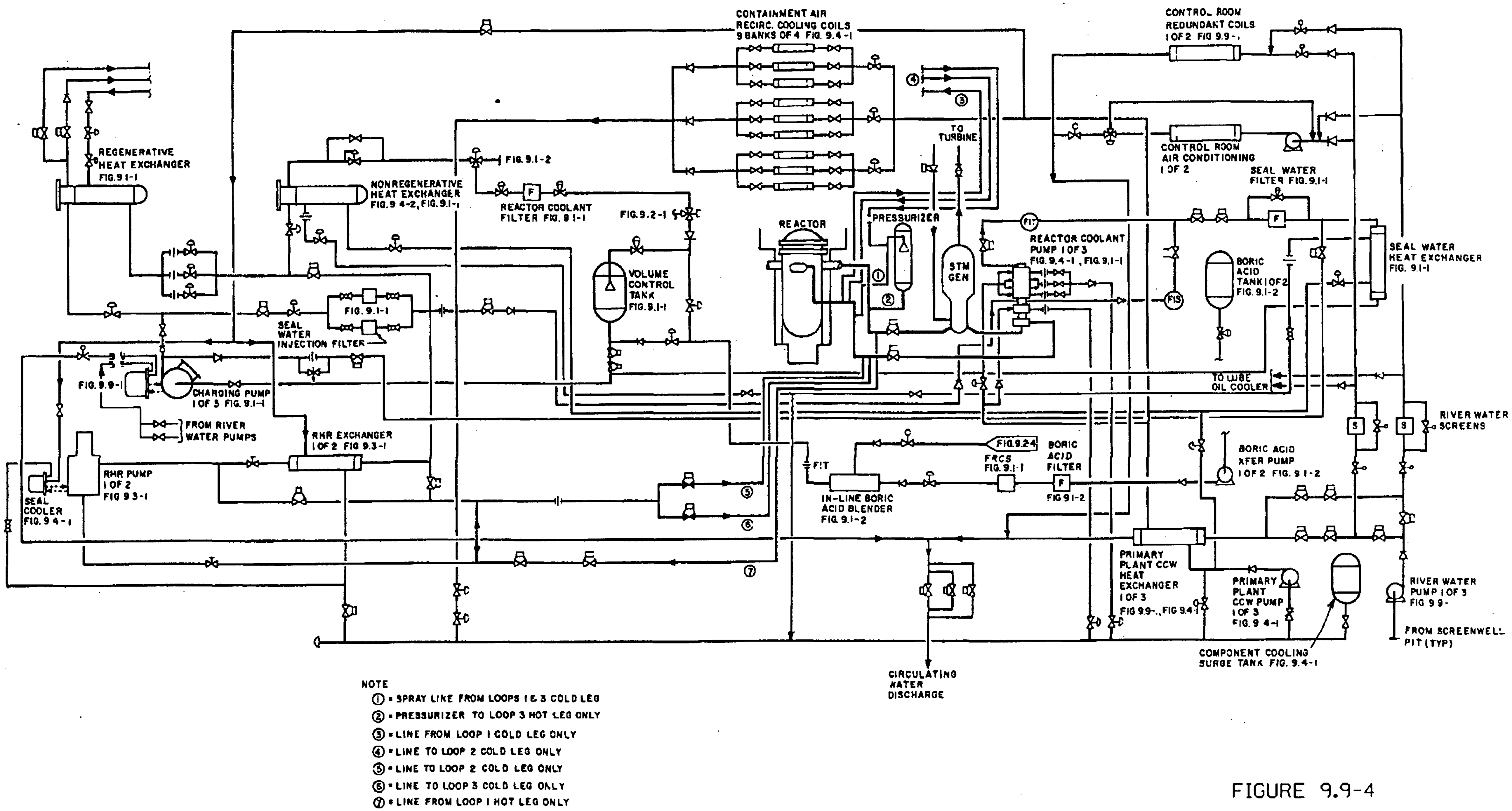
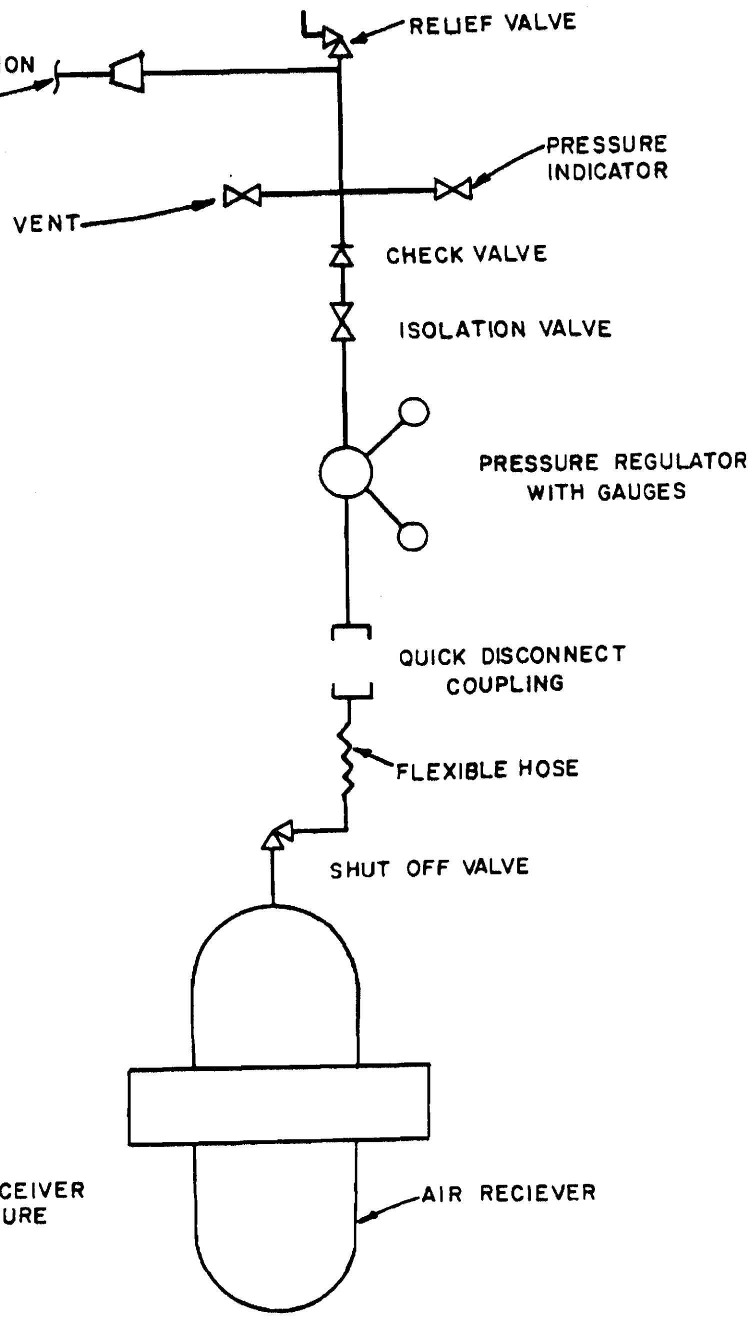


FIGURE 9.9-4

EQUIPMENT AND PIPING REQUIRED FOR NORMAL SHUTDOWN

BEAVER VALLEY POWER STATION NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT



NOTE:
ONE SYSTEM WITH AIR RECEIVER FOR EACH INTAKE STRUCTURE FLOOD DOOR

FIGURE 9.9-7
FLOOD SEAL PRESSURIZATION SYSTEM
BEAVER VALLEY POWER STATION UNIT NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT

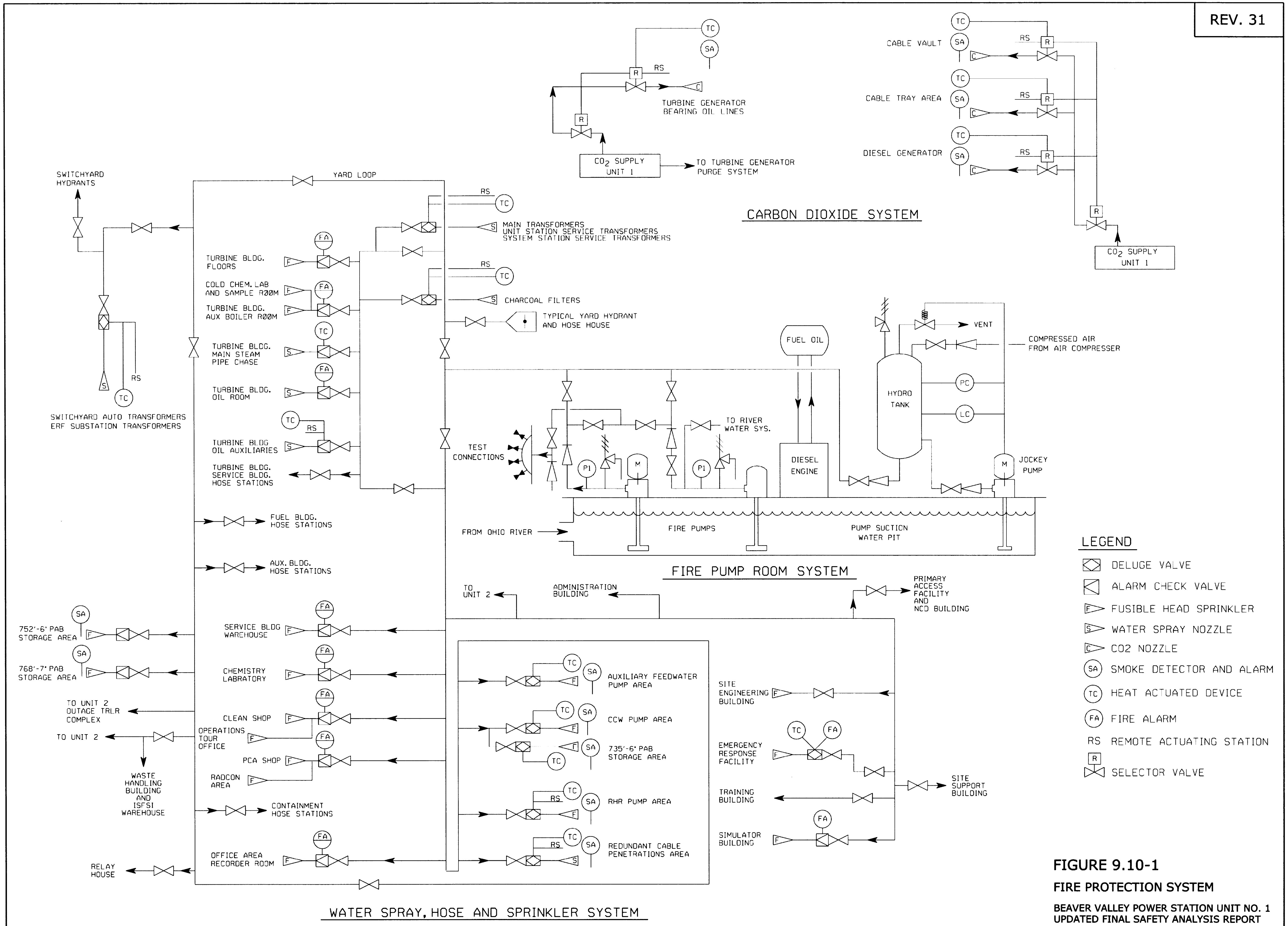
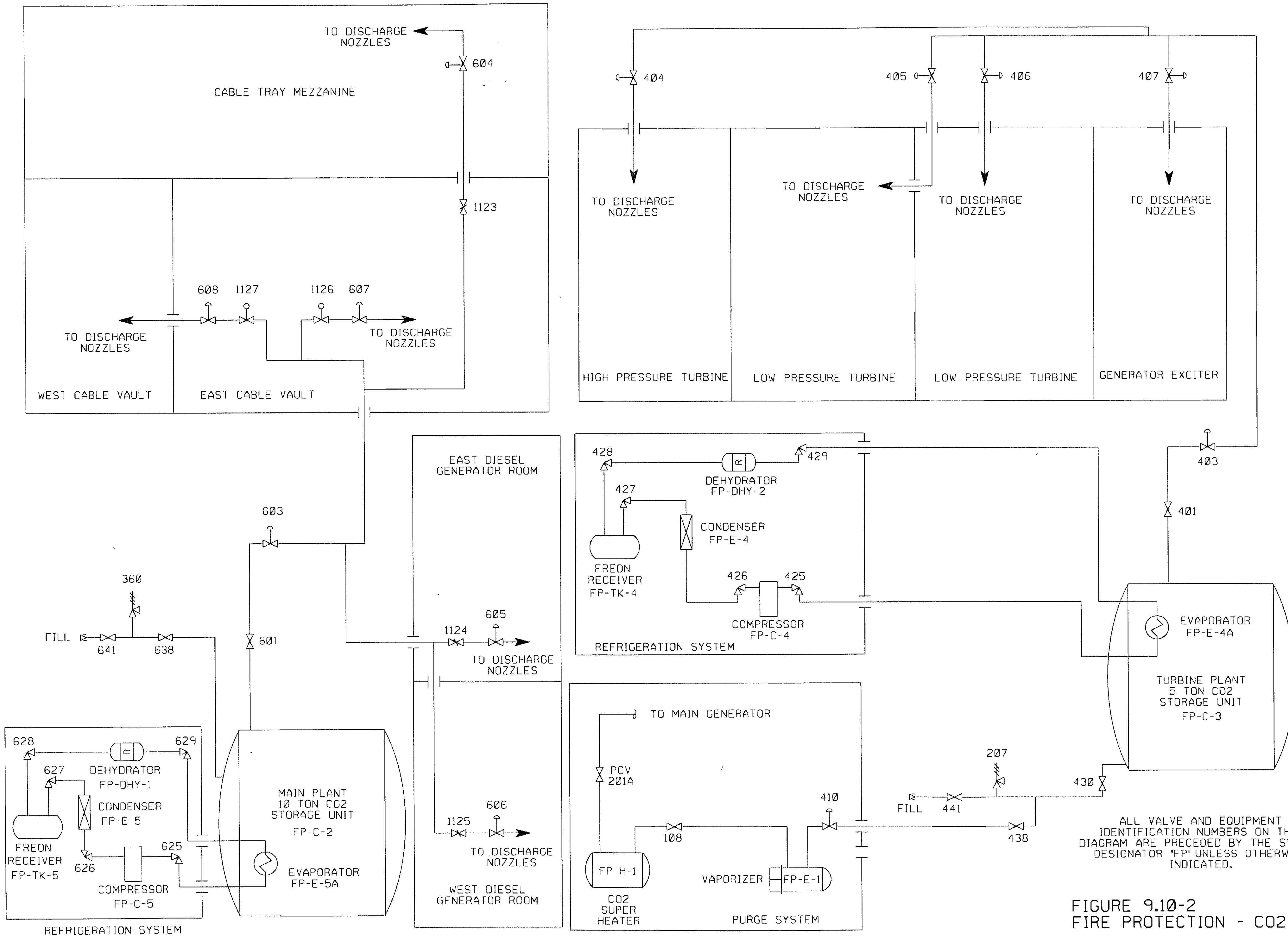


FIGURE 9.10-1

FIRE PROTECTION SYSTEM

BEAVER VALLEY POWER STATION UNIT NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT

K:\U1\UFSAR\FIGURES 9.1 THRU\FIG 9.10-1.dgn

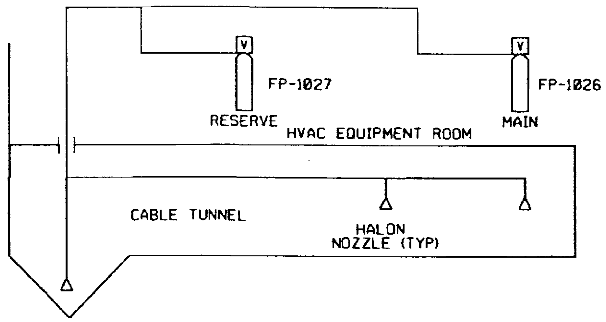


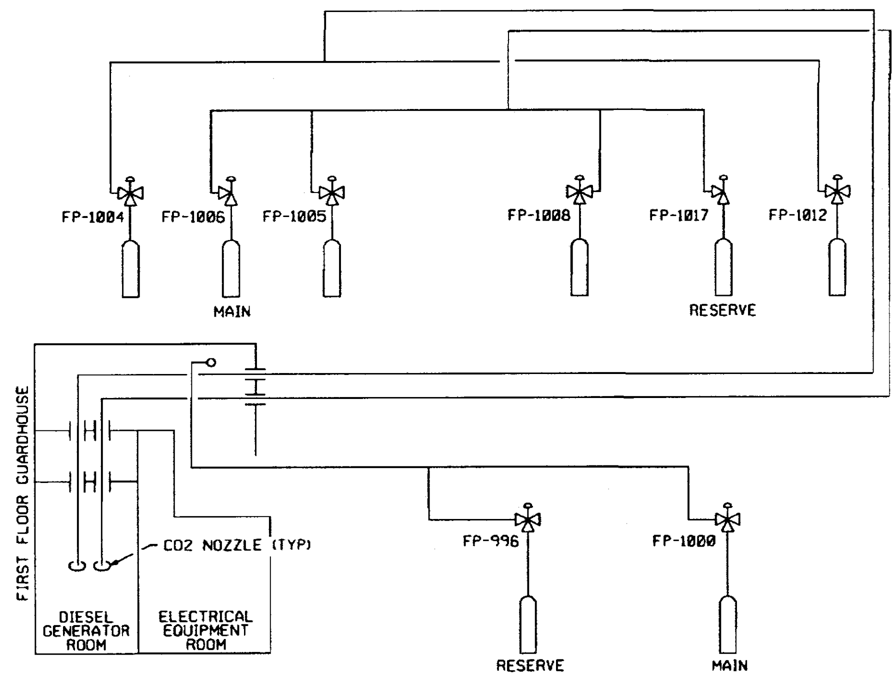
ALL VALVE AND EQUIPMENT IDENTIFICATION NUMBERS ON THIS DIAGRAM ARE PRECEDED BY THE SYSTEM DESIGNATOR "FP" UNLESS OTHERWISE INDICATED.

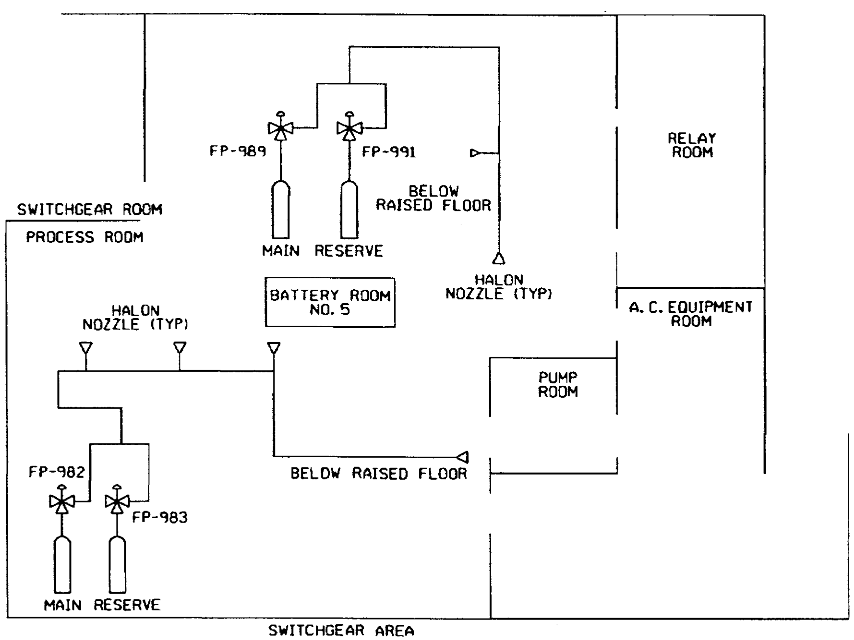
FIGURE 9.10-2
FIRE PROTECTION - CO2

REFERENCE: STATION DRAWING RM-533-3
BEAVER VALLEY POWER STATION NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT

REV. 23







ALL VALVE AND EQUIPMENT IDENTIFICATION NUMBERS ON THIS DIAGRAM ARE PRECEDED BY THE SYSTEM DESIGNATOR "FP" UNLESS OTHERWISE INDICATED.

FIGURE 9.10-3
FIRE PROTECTION-HALON AND CO2

REFERENCE: STATION DRAWING RM-533-4
BEAVER VALLEY POWER STATION UNIT NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT

REV. 12 (1/94)

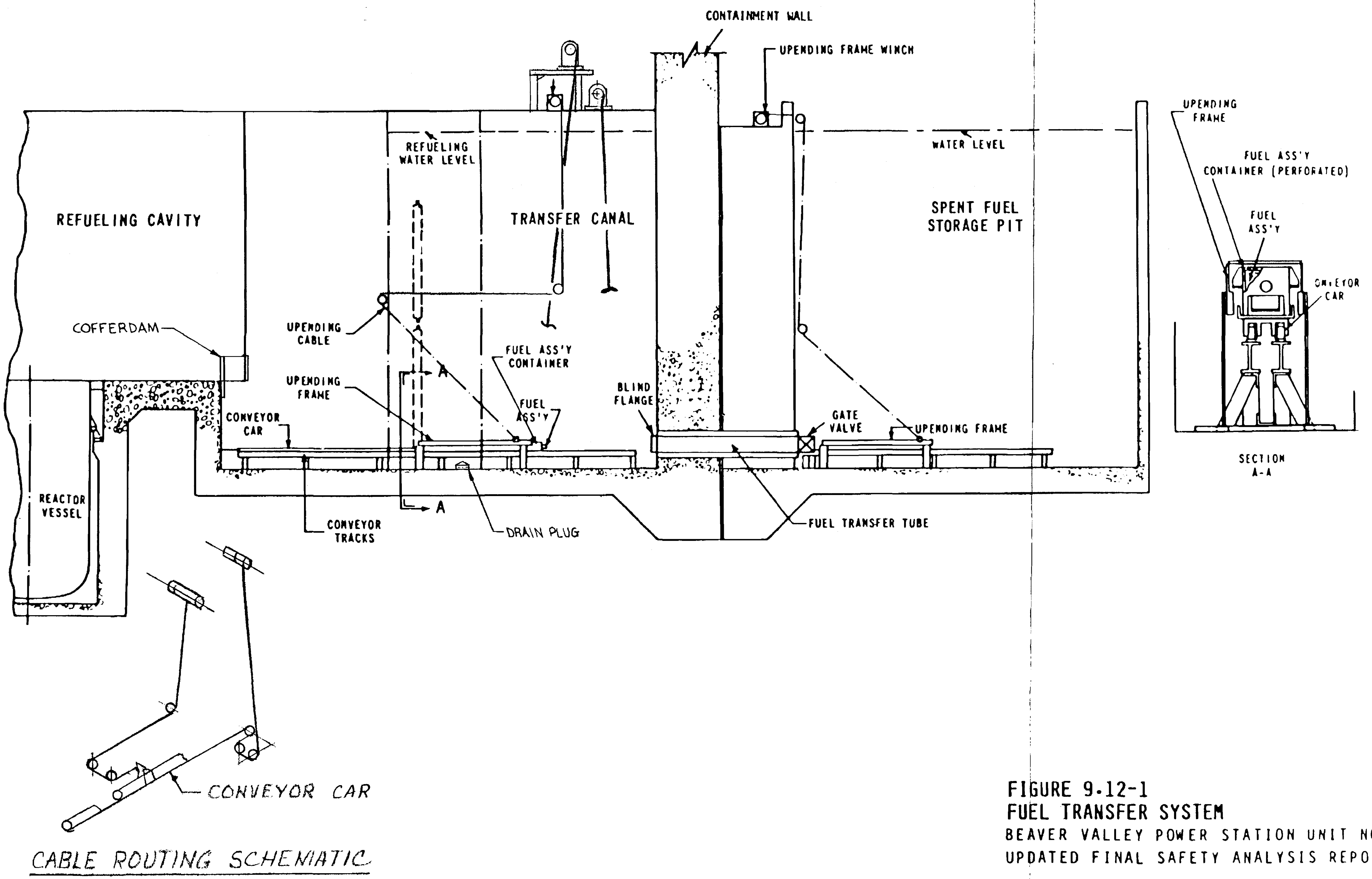


FIGURE 9.12-1
FUEL TRANSFER SYSTEM
BEAVER VALLEY POWER STATION UNIT NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT

REV13 1-95

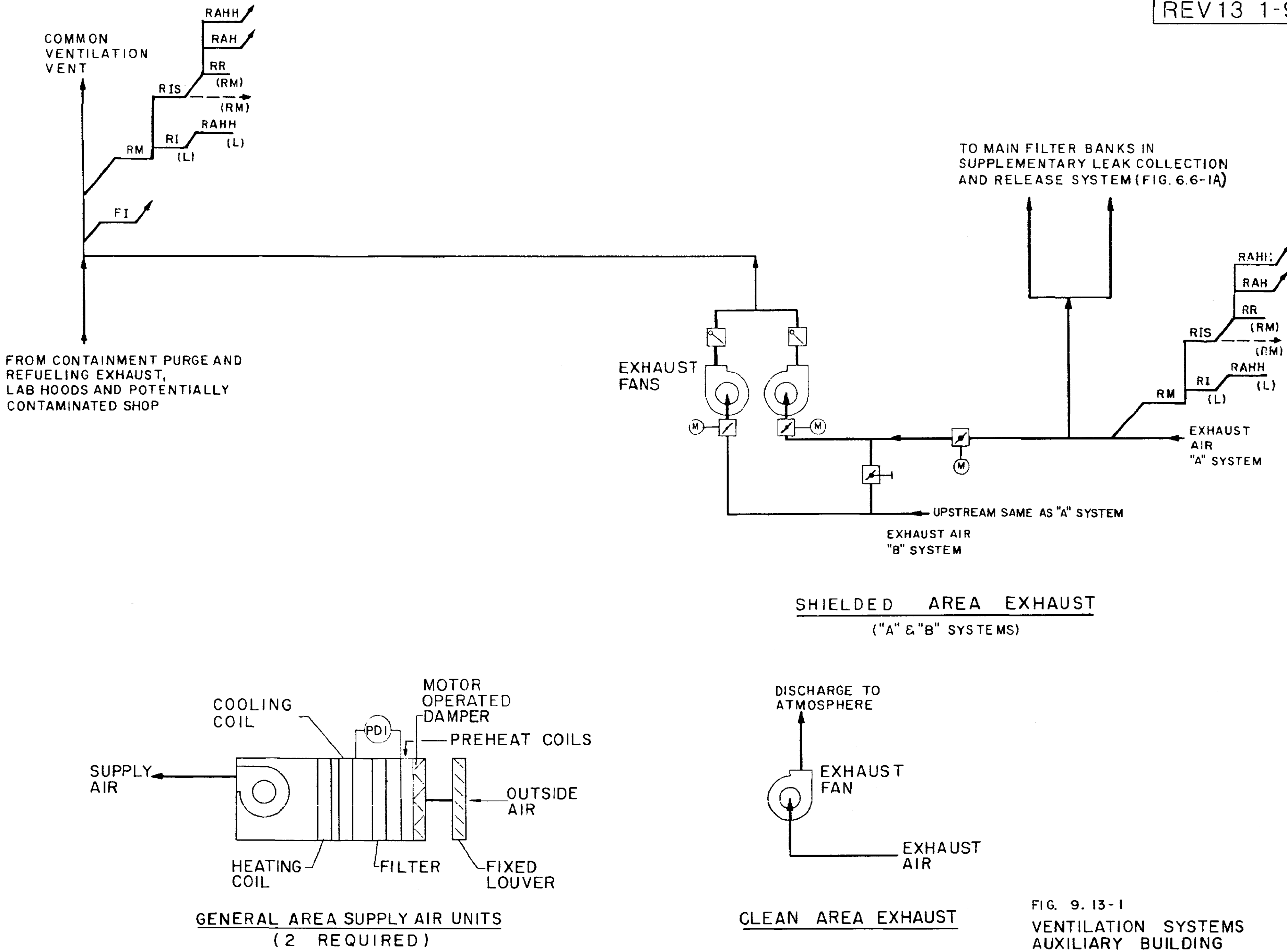
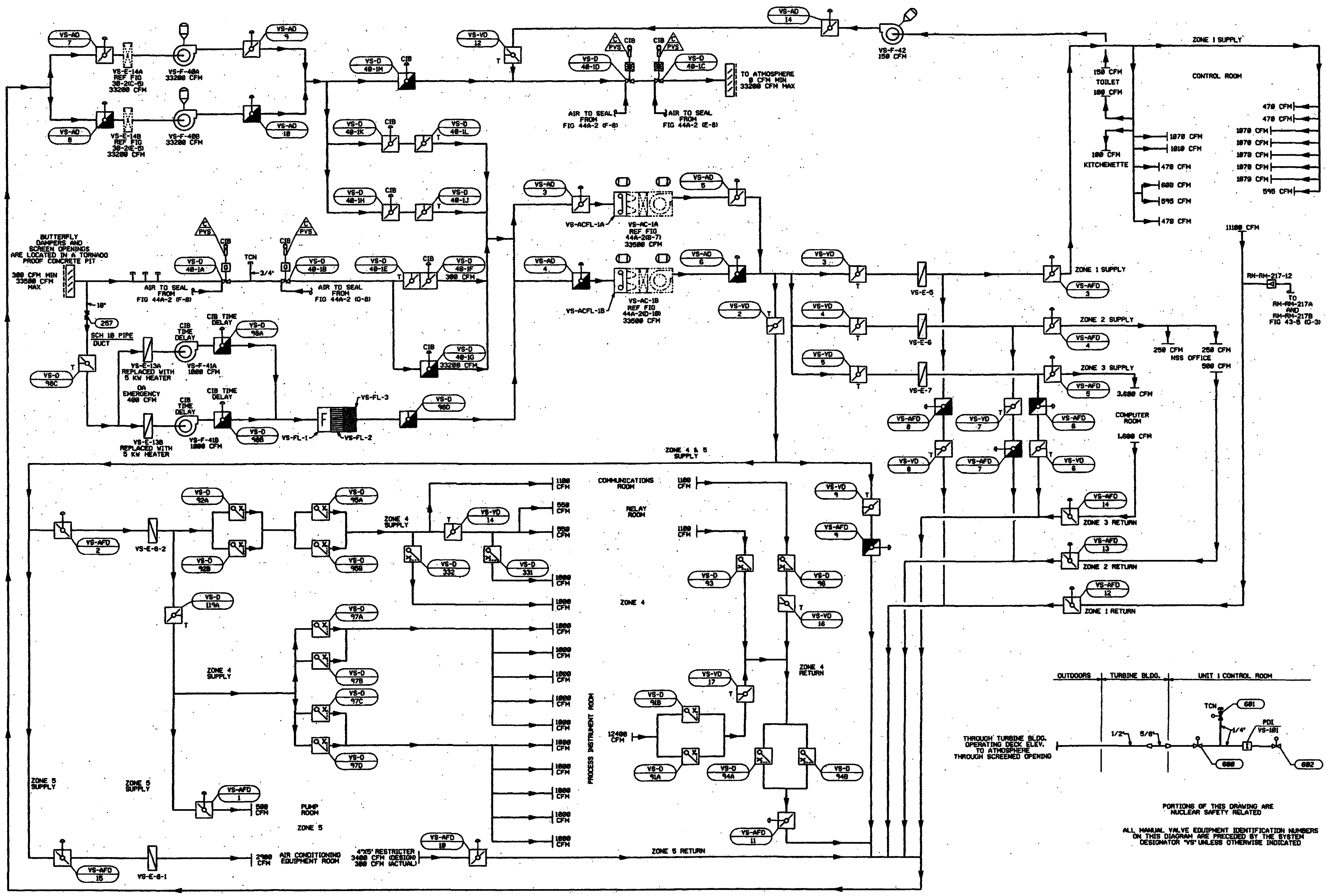


FIG. 9.13-1
VENTILATION SYSTEMS
AUXILIARY BUILDING
BEAVER VALLEY POWER STATION
FINAL SAFETY ANALYSIS REPORT

REV 28



PORTIONS OF THIS DRAWING ARE NUCLEAR SAFETY RELATED

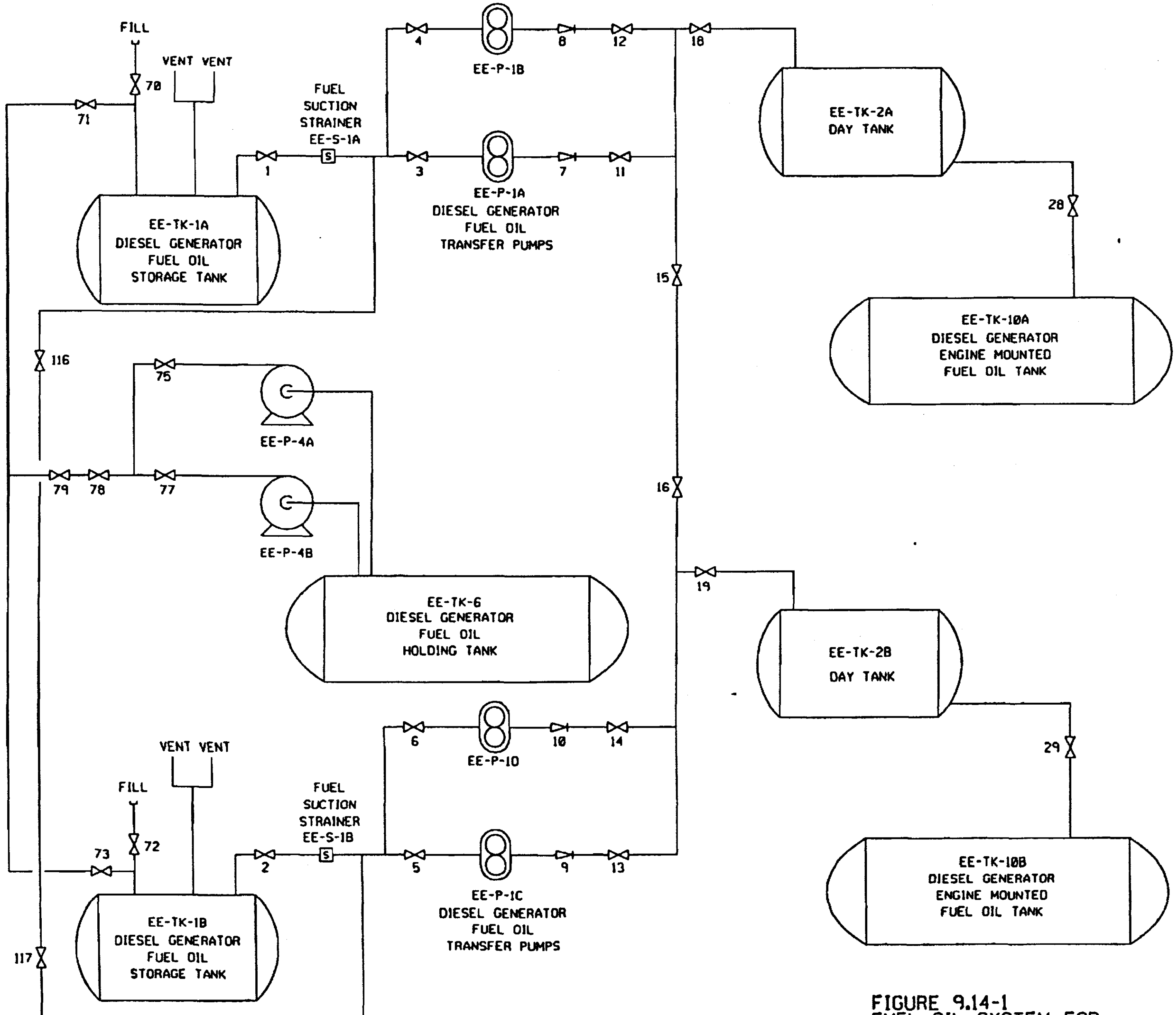
ALL MANUAL VALVE EQUIPMENT IDENTIFICATION NUMBERS ON THIS DIAGRAM ARE PRECEDED BY THE SYSTEM DESIGNATOR "VS" UNLESS OTHERWISE INDICATED

FIGURE 9.13-2
CONTROL ROOM AREA-AIR CONDITIONING

BEAVER VALLEY POWER STATION UNIT NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT

M:\U1\RM-0544A-04.DGN

REV. 18



ALL VALVE AND EQUIPMENT IDENTIFICATION NUMBERS ON THIS DIAGRAM ARE PRECEDED BY THE SYSTEM DESIGNATOR 'FO' UNLESS OTHERWISE INDICATED.

FIGURE 9.14-1
FUEL OIL SYSTEM FOR EMERGENCY DIESEL GENERATORS

REFERENCE: STATION DRAWING RM-436-2
BEAVER VALLEY POWER STATION UNIT NO. 1
UPDATED FINAL SAFETY ANALYSIS REPORT