9.0 AUXILIARY AND EMERGENCY SYSTEMS

9.0.1 Summary

The Auxiliary and Emergency Systems are supporting systems required for safe operation and servicing of the Reactor (Chapter 3) and the Reactor Coolant System (Chapter 4).

In some cases, operation of several systems is required to protect the Reactor and the Reactor Coolant System by controlling system conditions within specified operating limits. Other systems are called upon to operate under emergency conditions.

The systems included in this chapter are:

- Chemical and Volume Control System

This system provides for boron injection, chemical additions for corrosion control, reactor coolant clean-up and degasification, reactor coolant make-up, reprocessing of water letdown from the Reactor Coolant System, and reactor coolant pump seal water injection.

- Residual Heat Removal System

This system removes residual and sensible heat from the core and reduces the temperature of the Reactor Coolant System during the second phase of plant cool-down.

- Spent Fuel Pool Cooling System

This system removes the heat generated by spent fuel elements stored in the spent fuel pool, and maintains spent fuel pool water in clean and clear conditions.

- Component Cooling System

This system removes residual and sensible heat from the Reactor Coolant System via the Residual Heat Removal System during plant cooldown, cools the spent fuel pool water and the letdown flow to the Chemical and Volume Control System during power operation, and provides cooling to dissipate waste heat from various primary and secondary plant components.

- Sampling Systems

These systems provide the equipment necessary to obtain liquid and gaseous samples from the reactor plant systems.

- Fuel Handling System

This system provides for handling fuel assemblies, control rod assemblies, and some core structural components.

- Facility Service Systems

These systems include Fire Protection, Service Water, Auxiliary Building and Control Room Ventilation, and Compressed Air Systems.

9.1 APPLICATION OF PLANT DESIGN CRITERIA

The criteria, which apply to auxiliary and emergency systems, are presented in Section 1.4 of Chapter 1. Those specific attributes, which apply to one of the auxiliary or emergency systems, are listed and discussed in the appropriate system design basis section.

The applicable portions of the Missile Protection Criteria as stated in Sub-Chapter 1.4 apply to Class I equipment in this chapter.

As described in Chapter 7 and justified in Chapter 14, the reactor protection systems are designed to limit reactivity transients to DNBR above minimum allowable values due to any single malfunction in the reactivity control system. Each of the auxiliary cooling systems which serve an emergency function provide sufficient capability in the emergency operational mode to accommodate any single failure of an active component and still function in a manner to avoid risk to the health and safety of the public.

Adequate core cooling and containment heat removal can be maintained in case of gross leakage from either the component cooling system, essential service water system, or the residual heat removal system. Support of these Engineered Safety Feature (ESF) functions is discussed in Chapter 6.1.

The performance capability of the Residual Heat Removal System under partial components operation, and under accident conditions, is detailed in Chapter 6.

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

9.2 CHEMICAL AND VOLUME CONTROL SYSTEM

The Chemical and Volume Control System, is used to:

a. adjust the concentration of boric acid, i.e., the chemical neutron absorber for reactivity control,

b. maintain the proper water inventory in the Reactor Coolant System,

c. provide the required seal water flow for the reactor coolant pump shaft seals,

d. process reactor coolant effluent for reuse of boric acid and reactor makeup water,

e. maintain the proper concentration of corrosion inhibiting chemicals in the reactor coolant,

f. maintain the reactor coolant activities to within design limits, and

g. provide borated water for safety injection.

The system is also used to fill and hydrostatically test the Reactor Coolant System.

During normal operation, this system also has provisions for supplying the following chemicals:

a. Regenerant chemicals to the evaporator condensate demineralizers. This equipment is not normally used.

b. Hydrogen to the volume control tank.

c. Nitrogen as required for purging the volume control tank.

d. Hydrazine and lithium hydroxide as required via the chemical mixing tank to the charging pumps' suctions.

9.2.1 Design Bases

In addition to the reactivity control achieved by the rod cluster control (RCC) assemblies as detailed in Chapters 3 and 7, reactivity control is provided by the Chemical and Volume Control System (CVCS) which regulates the concentration of boric acid solution neutron absorber in the Reactor Coolant System.

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 plant cool down. Any time that the plant is at power, the quantity of boric acid retained in the boric acid tanks and ready for injection always exceeds that quantity required to bring the reactor to hot shutdown and to compensate for subsequent xenon decay.

The system is designed to allow for concurrent mixing and subsequent injection of boric acid solution. Thus the CVCS provides extended reactivity hold-down capability.

The reactivity control systems provided are capable of making and holding the core sub-critical for hot standby or hot operating conditions, including those resulting from power changes.

The chemical shim control serves to provide hot shutdown for the reactor as backup to the RCC assemblies.

The sizing of CVCS components and redundancy of its components and flow paths determine the CVCS reactor shutdown capability.

The boric acid solution is transferred from the boric acid tanks by boric acid transfer pumps to the suction of the charging pumps, which inject boric acid into the Reactor Coolant System. Any charging pump and any boric acid transfer pump can be operated from diesel generator power on loss of primary power.

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.

9.2.1.1 Codes and Classifications

Pressure retaining components (or compartments of components) of the CVCS which are exposed to reactor coolant comply with the following codes:

a. System pressure vessels - ASME Boiler and Pressure Vessel Code, Section III, Class C

b. System valves, fittings and piping - USAS B31.1.

Repair and replacement for pressure retaining components within the code boundary, and their supports, are conducted in accordance with ASME Section XI.

System integrity is assured by conformance to applicable codes listed in Table 9.2-1, and by the use of austenitic stainless steel or other corrosion resistant materials in contact with both reactor coolant and boric acid solutions.

The regenerative and excess letdown heat exchangers are designed as ASME Boiler and Pressure Vessel Code Section III, Class C. This designation is based on the following considerations: a) both heat exchangers can be double isolated from the Reactor Coolant System, b) both exchangers are located inside the containment, and c) both exchangers are protected by a missile barrier.

9.2.2 System Design and Operation

The Chemical and Volume Control System shown in Figures 9.2-1 through 9.2-6 provides a means for the injection of soluble neutron absorber in the form of boric acid solution, chemical additions for corrosion control and reactor coolant cleanup and degasification. This system also provides a means to add makeup water to the Reactor Coolant System, reprocess water letdown from the Reactor Coolant System, provide seal water injection to the reactor coolant pump seals, and fill and hydrostatically test the Reactor Coolant System.

A cross-tie on the discharge of the CVCS charging pumps, from one unit to the other, provides emergency flexibility. The cross-tie lines contain manual valves which are closed during normal operation.

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

System discharge from overpressure protective devices is directed to closed systems. Effluents removed from such closed systems are monitored and discharged under controlled conditions.

System design enables post-operational hydrostatic testing to test pressures required by the codes listed in Table 9.2-1.

9.2.2.1 System Description

During plant operation, reactor coolant flows through the letdown line from one of the reactor coolant loop cold legs on the suction side of the reactor coolant pump and is returned through the charging line on the discharge side of the reactor coolant pump of the same loop. An alternate charging connection is provided on the cold leg of a different loop. Current operating practice includes simultaneous use of both the normal and alternate charging connections. This practice has been adopted to address thermal stress concerns in piping connected to the reactor coolant system. An excess letdown line is also provided as an alternate in case the normal letdown circuit is inoperative.

Each of the CVCS connections to the Reactor Coolant System has an isolating valve. In addition, a check valve is located downstream of each charging line isolating valve. Reactor coolant entering the Chemical and Volume Control System 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 reactor containment and enters the auxiliary building where it undergoes a second temperature reduction in the tube side of the letdown heat exchanger followed by a second pressure reduction by the low pressure letdown valve. After passing through one of the mixed bed demineralizers, where ionic impurities are removed, coolant flows through the reactor coolant filter and enters the volume control tank through a spray nozzle.

Hydrogen is manually supplied to the vapor space in the volume control tank (VCT) which is predominantly hydrogen and water vapor. A self-regulating valve, or (Unit 2 only) manual bypass valve, is used to control hydrogen supply as determined by VCT pressure. The hydrogen within the tank is, in turn, the supply source to the reactor coolant. Fission gases are removed from the system by venting the volume control tank to the Waste Disposal System prior to a cold or refueling shutdown.

To enter the Reactor Coolant System the coolant flows from the volume control tank to the charging pumps which raise the pressure above that in the Reactor Coolant System. The coolant then enters the containment, passes through the tube side of the regenerative heat exchangers, and returns to the Reactor Coolant System. A portion of the high pressure charging flow is filtered and injected into the reactor coolant pumps between the pump impeller and the shaft seal so that the seals are not exposed to particulate matter in the reactor coolant. Part of the flow cools the lower radial bearing and enters the Reactor Coolant System through a labyrinth seal on the pumps shaft. The remainder, which is the shaft seal leakage flow, is filtered, cooled in the seal water heat exchanger and returned to the suction of the charging pumps.

Coolant injected through the reactor coolant pump labyrinth seals returns to the volume control tank by the normal letdown flow path through the regenerative heat exchanger. When the normal letdown route is not in service, labyrinth seal injection flow returns to the suction of the charging pumps through the excess letdown and seal water heat exchangers.

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

Boric acid is dissolved in hot water in the batching tank to a concentration of approximately four weight percent. The batching tank is jacketed to permit heating of the batching tank solution with low-pressure steam. One of four boric acid transfer pumps is used to transfer the batch to the boric acid tanks. The batching tank and the boric acid tanks are shared by Units 1 and 2. Small quantities of boric acid solution are metered from the discharge of an operating boric acid transfer pump for blending with the water supplied to makeup for normal leakage, or for increasing the reactor coolant boron concentration during normal operation. Electric immersion heaters maintain the temperature of the solution in the boric acid tanks high enough to prevent precipitation.

During plant startup, normal operation, load reductions and shutdowns, liquid effluents containing boric acid flow from the Reactor Coolant System through the letdown line and are collected in the holdup tanks (shared by Units 1 and 2) or the Volume Control Tank. As liquid enters the holdup tanks, the nitrogen cover gas is displaced to the gas decay tanks in the Waste Disposal System through the waste vent header. If the gas becomes highly oxygenated, a provision exists for bypassing the waste vent header and discharging the gas to the monitored plant vent, provided tank gas samples are found acceptable for release. The concentration of boric acid in the letdown fluid to the holdup tanks varies throughout core life from the refueling concentration to essentially zero at the end of the core cycle. A recirculation pump is provided to transfer liquid from one holdup tank to another. Contents of the CVCS holdup tanks may be discharged to the environment via the waste disposal system (Chapter 11).

Liquid effluent in the holdup tanks is generally processed as a batch operation through a recycle processing train for boric acid recovery. This liquid is pumped by the boric acid evaporator feed pumps through the evaporator feed ion exchangers (shared by Units 1 and 2) which remove cations (primarily lithium and long-lived cesium) and anions. The liquid then flows through the ion exchanger filter, and into the gas stripper section of the boric acid evaporator where dissolved gases are removed from the liquid. These gases are vented to the Waste Disposal System.

The liquid effluent from the gas stripper section enters the boric acid evaporator. The vapor produced in the boric acid evaporator leaves the evaporator condenser and is pumped through a condensate cooler where the distillate is cooled to the operating temperature of the evaporator condensate demineralizers. The evaporator condensate demineralizers are not normally used. When the evaporator condensate demineralizers are used then the non-volatile evaporator carry over is removed by one of the two evaporator condensate demineralizers. Evaporator condensate flows through the condensate filter and accumulates in one of the two CVCS monitor tanks (shared by Units 1 and 2). The other two monitor tanks are used for radioactive waste disposal. The dilute boric acid solution originally in the boric acid evaporators is concentrated to approximately four weight percent boric acid.

Subsequent handling of the condensate is dependent on the results of sample analysis of the monitor tank contents. Discharge from the monitor tanks may be pumped by the monitor tank drain pumps to the primary water storage tank, recycled through the evaporator condensate demineralizers, returned to the holdup tanks for reprocessing in the evaporator train or, if the sample analysis of the monitor tanks contents indicates sufficiently low levels, the contents may be discharged to the environment via the waste disposal system (Chapter 11).

Boric acid evaporator bottoms are sampled and then pumped to either the concentrates holding tank, the CVCS holdup tanks or the boric acid tanks as determined by the Shift Manager.

The concentrated solution can also be pumped from the evaporator to the waste disposal system waste evaporator for additional processing for disposal.

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

During plant cooldown when the residual heat removal loop is operating and the letdown orifices are not in service, a flow path is provided to remove fission products, corrosion and other impurities. A portion of the flow leaving the residual heat exchangers passes through the letdown 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 Reactor Coolant System.

9.2.2.2 Expected Operating Conditions

Tables 9.2-2 and 9.2-3 list the system performance requirements and data for individual system components, respectively. The component data contained in Table 9.2-3 is original equipment design and sizing data. The CCW system has been designed and analyzed to;

a. Operate in the range of 60°F to 105°F, except during periods of cooldown and post-LOCA operation, and

b. Operate at temperatures ≤120°F during cooldown and post-LOCA operation.

9.2.2.3 Reactor Coolant Activity Concentration

The parameters used in the calculation of the reactor coolant fission product inventory, including the expected coolant cleanup flow rate and the demineralizer effectiveness, are presented with the results of the calculations in Chapter 14. In these calculations the defective fuel rods are assumed to be present at locations uniformly distributed throughout the core. The fission product escape rate coefficients are therefore based upon an average fuel temperature.

Tritium is produced in the reactor from ternary fission in the fuel, irradiation of boron in the burnable poison rods and irradiation of boron, lithium and deuterium in the coolant. A discussion of tritium control is given in Section 9.2.3.

9.2.2.4 Reactor Makeup Control Modes

The reactor makeup control is designed to operate from the control room by manually pre-selecting 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 remotely closing the makeup stop valves.

Makeup water to the Reactor Coolant System is provided by Chemical and Volume Control System from the following sources:

a. The primary water storage tank, which provides water for dilution when the reactor coolant boron concentration is to be reduced;

b. The boric acid tanks, which supply concentrated boric acid solution when reactor coolant boron concentration is to be increased;

c. The refueling water storage tank which supplies borated water for alternate and emergency makeup; or

d. The chemical mixing tank, which is used to inject small quantities of hydrazine or pH control chemical when necessary.

Makeup for normal plant leakage is regulated by the reactor makeup control, which is set by the operator to blend water from the primary water storage tank with concentrated boric acid to match the reactor coolant boron concentration. Makeup is added automatically if the volume control tank level falls below a preset point.

9.2.2.5 Makeup

The "automatic makeup" mode of operation of the reactor primary water makeup control provides boric acid solution preset to match the boron concentration in the Reactor Coolant System. The automatic makeup compensates for minor leakage of reactor coolant without causing significant changes in the coolant boron concentration.

Under normal plant operating conditions, the mode selector switch and makeup stop valves are set in "Automatic Makeup Position." The following actions are initiated automatically when a low level signal is received from the volume control tank:

1. the operating boric acid transfer pump is switched to high speed or a boric acid transfer pump is manually switched on from an off position to high speed,
2. the primary water makeup pump starts,
3. the boric acid modulating valve and the primary water modulating valve are actuated, and
4. the valve to suction of charging pumps is opened.

The flow controllers then blend the makeup stream according to the preset concentration. This blending causes the volume control tank level to rise. At a preset high level point, the makeup is stopped, the primary water makeup pump stops, the primary water makeup control valve closes, the boric acid transfer pump is returned to low speed operation or is manually switched to an off position, the concentrated boric acid control valve closes and the makeup stop valve to the charging pump suction closes.

The system is also supplied with the capability for a “Manual Makeup” mode of operation which permits the operator to select the flow path, and the rate and quantity of both boric acid and primary water at the same time.

9.2.2.6 Dilution

The "dilute" mode of operation permits the addition of a pre-selected quantity of primary water makeup at a pre-selected flow rate to the Reactor Coolant System. The operator sets the makeup stop valves to the volume control tank and to the charging pump suction in the closed position, the mode selector switch to "dilute," the primary water makeup flow controller set point to the desired flow rate, sets the reactor primary water makeup batch integrator to the desired quantity and initiates system start. This opens the primary water makeup control valve to the volume control tank and starts a primary water makeup pump which will deliver primary makeup water to the volume control tank. From there the water goes to the charging pump suction header. Excessive rise of the volume control tank water level is prevented by automatic actuation (by the tank level controller) of a three-way diversion valve which routes the reactor coolant letdown flow to the holdup tanks. When the preset quantity of primary water makeup has been added, the batch integrator causes the primary water makeup pump to stop and the primary water makeup control valve to close.

The system is also supplied with the capability for an “Alternate Dilution” mode of operation which permits the operator to deliver dilution flow to the volume control tank and/or directly to the suction of the charging pump to allow quicker dilution than the dilute mode.

9.2.2.7 Boration

The "borate" mode of operation permits the addition of a pre-selected quantity of concentrated boric acid solution at a pre-selected flow rate to the Reactor Coolant System. The operator sets the makeup stop valves to the volume control tank and to the charging pump suction in the closed position, the mode selection switch to "borate," the concentrated boric acid flow controller set point to the desired flow rate, the concentrated boric acid batch integrator to the desired quantity, and initiates system start. The following actions are initiated automatically when a signal is received from the batch integrator of the reactor makeup control:

1. The boric acid transfer pump is switched to high speed,
2. the boric acid modulating valve is actuated, and
3. the valve to suction of charging pump is opened.

This will cause delivery of a four weight percent boric acid solution to the charging pump suction header. The total quantity added in most cases will be so small that it will have only a minor effect on the volume control tank level. When the preset quantity of concentrated boric acid solution is added, the batch integrator causes the boric acid transfer pump to return to low speed operation and closes the boric acid modulating valve and the makeup stop valve to the suction of the charging pumps. In the event of a volume control tank low-low level signal, the suction of the charging pumps is automatically aligned to take suction from the refueling water storage tank.

The amount of boric acid in the boric acid tank is sufficient to maintain the reactor subcritical by the necessary shutdown margin at hot conditions following a reactor trip from all credible operating conditions. The flow rate of boric acid from the boric acid tank is sufficient to follow the highest burnout rate of xenon following reactor startup from peak xenon conditions.

Sufficient volume of boric acid is available in the RWST to borate the reactor to cold shutdown conditions and maintain the reactor subcritical by the necessary shutdown margin following a reactor trip from all credible operating conditions. Plant operating procedures ensure the necessary shutdown margin is maintained at all times during the cooldown process. Additionally, the flowrate of boric acid from the RWST is sufficient to compensate for the maximum xenon burnout following a reactor startup from peak xenon conditions, which bounds the flow rate required to compensate for the xenon decay during a reactor shutdown from 100% rated thermal power at peak xenon conditions.

The system is also supplied with the capability for an “Emergency Boration” mode of operation to initiate rapid boration to restore shutdown margin in the event an abnormal condition results in an unexplained or uncontrolled reactivity increase.

9.2.2.8 Alarm Functions

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

a. Deviation of reactor primary water makeup flow rate from the control set point.

b. Deviation of concentrated boric acid flow rate from control set point.

c. Low level (makeup initiation point) in the volume control tank when the primary water makeup control selector is not set for the automatic makeup control mode.

d. Low level (between makeup initiation point and automatic alignment charging pump suction to refueling water storage tank) in the volume control tank to allow the operator to manually initiate makeup prior to refueling water automatic alignment.

9.2.2.9 Charging Pump Control

Positive Displacement Charging Pump[1]

The positive displacement charging pump has a variable speed drive and supplies charging flow to the Reactor Coolant System. The speed of this pump can be controlled manually, or

[1] The positive displacement charging pumps are not currently used for plant operations.

automatically by pressurizer level. During load changes the pressurizer level set point varies automatically with T_{avg}, compensating partially for the expansion or contraction of reactor coolant associated with T_{avg} changes. Charging pump speed will not change rapidly with pressurizer level control. If the pressurizer level increases, the speed of the pump decreases; conversely, if the level decreases, the speed increases. If the positive displacement charging pump reaches the high speed limit, it becomes necessary to place a centrifugal pump in operation to provide the higher flow capacity and to remove the positive displacement pump from service.

To ensure that the charging pump flow is always sufficient to meet both the seal water and minimum charging flow requirements, the pump has a variable control stop which prevents pump flow lower than the specified minimum. The control stop is variable to permit higher minimum flow limits to be set if mechanical seal leakage increases during plant life.

9.2.2.10 Centrifugal Charging Pumps

The centrifugal pumps are constant speed pumps with flow control accomplished by a modulating valve in the pump discharge line. When the positive displacement pump is in operation, this control valve is in the wide open position.

A flow transmitter on the charging line upstream of the regenerative heat exchanger transmits a signal to a controller which regulates a modulating valve in the charging line to maintain a preset charging flow. A pressurizer water level error signal resets the charging flow set point to take corrective action. The response of the charging line modulating valve to changes in the flow control signal is normally maintained slow to reduce charging flow fluctuations due to short term pressurizer level transients.

9.2.2.11 Components

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

9.2.2.12 Regenerative Heat Exchanger

The regenerative heat exchanger is designed to recover heat from the letdown flow by heating the charging flow, which minimizes reactivity effects due to insertion of cold water and reduces thermal shock on the charging penetrations into the reactor-coolant-loop piping.

The design also considers the limit of difference in temperature, which occurs during periods when letdown flow exceeds charging flow by a greater margin than at normal letdown conditions.

The letdown stream flows through the shell of the regenerative heat exchanger and the charging stream flows through the tubes. The unit is made of austenitic stainless steel, and is of all welded construction. It is a multi-shell U tube type heat exchanger using three shells.

9.2.2.13 Letdown Orifices

One of the three letdown orifices controls flow of the letdown stream during normal operation and reduces the pressure to a value compatible with the letdown heat exchanger design. Two of the letdown orifices are designed to pass normal letdown flow. The third orifice is designed to be used in conjunction with one normal letdown flow orifice to attain maximum purification flow at normal reactor coolant system operating pressure. The orifices are placed in and taken out of service by remote manual operation of their respective isolation valves. The standby orifice may be used in parallel with the normally operating orifice in order to increase letdown flow when the reactor coolant system pressure is below normal. This arrangement provides a full standby capacity for control of letdown flow. Each orifice is an austenitic pipe containing a bored corrosion and erosion resistant insert.

9.2.2.14 Letdown Heat Exchanger

The letdown heat exchanger cools the letdown stream to the operating temperature of the mixed bed demineralizers. Reactor coolant flows through the tube side of the exchanger while component cooling water flows through the shell. The letdown stream outlet temperature is automatically controlled by a temperature control valve in the component cooling water outlet stream. The unit is a multiple-tube-pass heat exchanger. All surfaces in contact with the reactor coolant are austenitic stainless steel, and the shell is carbon steel.

9.2.2.15 Mixed Bed Demineralizers

Two flushable mixed bed demineralizers assist in maintaining reactor coolant purity. A Li^7 cation resin and a hydroxyl form anion resin are charged into one demineralizer. A hydrogen cation resin and hydroxyl form anion resin are charged into the other demineralizer. This demineralizer is used to remove Li^7 whereas the other will not. 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, assuming one percent of fuel containing clad defects.

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

The demineralizer vessels are provided with suitable connections to facilitate resin replacement when required. The vessels are equipped with a resin retention screen. Each demineralizer has sufficient capacity for approximately one core cycle with one percent defective fuel rods.

9.2.2.16 Cation Bed Demineralizer

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

The demineralizer vessel is provided with suitable connections to facilitate resin replacement when required. The vessel is equipped with resin retention screens. The cation bed demineralizer has sufficient capacity for approximately one core cycle with one percent defective fuel rods.

9.2.2.17 Reactor Coolant Filter

The filter collects resin fines and particulates from the letdown stream. The vessel is provided with connections for draining and venting. The nominal flow capacity of the filter is equal to the maximum purification flow rate. Disposable filter elements are used.

9.2.2.18 Volume Control Tank

The volume control tank is an operating surge volume compensating in part for reactor coolant releases from the Reactor Coolant System as a result of level changes. The volume control tank also acts as a head tank for the charging pumps and reservoir for the leakage from the reactor coolant pump controlled leakage seal. Overpressure of hydrogen gas is maintained in the volume control tank to control the hydrogen concentration in the reactor coolant at 25 to 50 cc per kg of water (STP).

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 Waste Disposal System permits removal of gaseous fission products which are stripped from the reactor coolant and collected in the tank.

9.2.2.19 Charging Pumps

Three charging pumps are provided for injecting coolant into the Reactor Coolant System. Two are centrifugal pumps and the third is a positive displacement pump equipped with variable speed drive. All parts in contact with the reactor coolant are fabricated of austenitic stainless steel or other material of adequate corrosion resistance. The centrifugal pump packing glands and positive displacement pump stuffing box are provided with leakoffs to collect reactor coolant before it can leak to the outside atmosphere. Pump leakage is piped to the drain header disposal. The pump design prevents lubricating oil from contaminating the charging flow. The integral discharge valves on the positive displacement pump act as check valves.

The positive displacement pump is designed to provide the full charging flow and the reactor coolant pump seal water supply during normal seal leakage and normal letdown.[2] The centrifugal pumps have a higher flow capacity and are currently used in normal plant operation. Each pump was designed to provide charging and seal injection flows with normal letdown flow (75 gpm) or maximum letdown flow (120 gpm), provided that the RCS cold leg backpressure is at normal operating conditions, and provided that the charging pump minimum flow path is isolated during maximum letdown flow.

The positive displacement charging pump is designed to be used to hydrotest the Reactor Coolant System.

Either the positive displacement charging pump or a centrifugal charging pump can take suction from the volume control tank and discharge to the normal charging and reactor coolant pump seal water injection paths. When the positive displacement pump is not used, one of the

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[2] The positive displacement charging pumps are not currently used for plant operations.

centrifugal charging pumps is operated.[2] The flow paths remain the same but flow control is accomplished by a modulating valve on the discharge side of the centrifugal pumps. For periods when maximum letdown or purification flow is required, a centrifugal pump is operated to provide the necessary flow. The centrifugal charging pumps also serve as high head safety injection pumps in the Emergency Core Cooling System (Chapter 6).

9.2.2.20 Chemical Mixing Tank

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

The capacity of the chemical mixing tank is determined by the quantity of 35% hydrazine solution necessary to increase the hydrazine concentration in the reactor coolant by 10 ppm. This capacity is more than sufficient to permit the preparation of the appropriate quantity of pH control chemical solution for the Reactor Coolant System.

9.2.2.21 Excess Letdown Heat Exchanger

The excess letdown heat exchanger is designed to cool the amount of reactor coolant letdown equal to the nominal injection rate through the reactor coolant pump labyrinth seal, when the normal letdown path is not usable. The letdown stream flows through the tube side and component cooling water is circulated through the shell side. All surfaces in contact with reactor coolant are austenitic stainless steel and the shell is carbon steel. All tube joints are welded.

9.2.2.22 Seal Water Heat Exchanger

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

The unit is designed to cool the excess letdown flow, the pump seal water flow and the centrifugal charging pump by-pass flow to the temperature normally maintained in the volume control tank.

9.2.2.23 Seal Water Filter

This filter collects particulates from the reactor coolant pump seal water return and from the excess letdown heat exchanger flow. The filter is designed to pass the sum of the excess letdown flow and the maximum design leakage from the reactor coolant pump seals. The vessel is provided with connections for draining and venting. Disposable filter elements are used.

9.2.2.24 Seal Water Injection Filters

The filter collects particulates from the reactor coolant pump seal water inlet. Two filters are provided in parallel, each sized for the maximum design pump seal flow rate. The vessel is provided with connections for draining and venting. Disposable filter elements are used.

9.2.2.25 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. The filter is designed to pass the design flow of two boric acid transfer pumps operating simultaneously. The filter elements are disposable cartridges. Provisions are included for venting and draining the filter.

9.2.2.26 Boric Acid Tanks

Three boric acid tanks are shared by Units 1 and 2. The total boric acid tankage stores sufficient boric acid solution, recovered from the recycle processing train or mixed in the batching tank, for simultaneous hot shutdown shortly after full power operation is achieved. One tank provides sufficient boric acid solution for hot shutdown even if the most reactive RCC assembly is not inserted. One tank supplies boric acid for each reactor coolant makeup system during normal and emergency operations, while the third tank serves as a spare.

The concentration of boric acid solution in storage is maintained between 3.5 and 4.0% by weight. Periodic manual sampling and corrective action, if necessary, insures that these limits are maintained. As a consequence, measured amounts of boric acid solution can be delivered to the reactor coolant to control the chemical poison concentration. The combination overflow and breather vent connection has a water loop seal to minimize vapor discharge during storage of the solution.

9.2.2.27 Batching Tank

The batching tank (shared by both units) is sized to hold one week's makeup supply, per unit, of boric acid solution for transfer to the boric acid tanks. The basis for makeup is an arbitrary reactor coolant leakage of 1/2 gpm at beginning of core life. The tank may also be used for solution storage.

A local sampling point is provided for verifying the solution concentration prior to transferring it to the boric acid tank. The tank is provided with an agitator to improve mixing during batching operations. The tank is provided with a steam jacket for heating the boric acid solution to 120°F.

9.2.2.28 Boric Acid Tank Heaters

Each of two electric immersion heaters in each boric acid tank maintains the temperature of the boric acid solution from 105°F to 120°F with ambient air temperature of 40°F, thus ensuring a temperature in excess of the solubility limit. The solubility limit for 4.0 weight percent boric acid is reached at a temperature of 58°F. This temperature is sufficiently low that the normally expected ambient temperatures within the auxiliary building will maintain boric acid solubility. Heaters remain in place for manual or automatic operation in the event auxiliary building ambient temperature falls below the Technical Specification requirement. The heaters are sheathed in austenitic stainless steel.

9.2.2.29 Boric Acid Transfer Pumps

Two horizontal, centrifugal, two speed pumps with mechanical seals are available per unit. Although not required, one pump may be aligned to run continuously at low speed to provide recirculation of the boric acid system and the boric acid tank. The second pump can be aligned with the shared boric acid tank and is considered as a standby pump, with service being transferred as operation requires. This second pump also intermittently circulates fluid through the shared tank. Automatic initiation of the reactor coolant makeup system will align the running pump for high speed operation to provide normal makeup of boric acid solution as required. Manual operation of the boric acid transfer pumps (i.e., starting an inactive pump) can also be used to provide reactor coolant makeup as necessary. For emergency boration, supplying of boric acid solution to the suction of the charging pump can be accomplished by manually choosing either fast or slow speed and actuating either or both pumps. The transfer pumps also function to transfer boric acid solution from the batching tank to the boric acid tanks.

The design capacity of each pump is equal to the normal letdown flow with the capacity of both pumps being equivalent to the normal design capacity of one centrifugal charging pump. The design discharge pressure is sufficient to overcome any pressures which may exist in the suction manifold of the charging pumps (volume control tank relief valve setting). In addition to the automatic actuation by the makeup control system, and manual actuation from the main control board, these pumps may also be controlled locally at a local control center.

All parts in contact with the solution are of austenitic stainless steel. Connections are provided to enable the use of these pumps to flush the equipment and piping with primary water.

9.2.2.30 Boric Acid Blender

The boric acid blender promotes thorough mixing of boric acid solution and primary water makeup for the reactor coolant makeup circuit. The blender consists of a conventional pipe fitted with a perforated tube insert. The blender decreases the pipe length required to homogenize the mixture for taking a representative local sample.

9.2.2.31 Holdup Tanks

Two pairs of holdup tanks plus a single tank are provided to receive and hold for processing the letdown fluid from the Reactor Coolant System. Each pair of tanks has a crosstie between the liquid spaces and between the gas spaces of each tank, making each pair, in effect, a single tank. The single tank is half the capacity of each pair of tanks. The system is so arranged that normally one pair of tanks serves one unit, a second serves the other unit and the single tank is a spare to provide additional storage when necessary. One pair of tanks (or the single tank) at a time is processed by the boric acid evaporator. The total liquid capacity of the tanks is greater than three Reactor Coolant System volumes. The tanks are constructed of austenitic stainless steel.

9.2.2.32 Boric Acid Reserve Tank

The boric acid reserve tank provides a reserve supply of boric acid solution to augment the boric acid tanks. The tank augments the boric acid tanks as both a source and receiver of boric acid solution. This quantity of boric acid is not required by the Technical Specifications but provides additional operating margin for the boric acid makeup system and increases system reliability. The contents of the reserve tank may be mixed with one of the boric acid evaporator feed pumps. The boric acid evaporator feed pump supplies boric acid from the reserve tank to the boric acid tanks, the boric acid evaporator feed ion exchangers, ion exchange filter, boric acid evaporator, concentrates filter and concentrates holding tank transfer pumps. The contents of the reserve tank can also be processed via this path and returned to the reserve tank. Units 1 and 2 share the reserve tank. The tank is constructed of austenitic stainless steel.

9.2.2.33 Holdup Tank Recirculation Pump

The recirculation pump is used to mix the contents of a pair of holdup tanks for sampling or to transfer the contents to another pair of holdup tanks. The pump may also be used to fill the spent fuel pit transfer canal from the holdup tanks. The wetted surface of this pump is constructed of austenitic stainless steel.

9.2.2.34 Boric Acid Evaporator Feed Pumps

The three feed pumps (shared by both units) supply feed to the boric acid evaporator trains from the holdup tanks. The capacity of each pump is equal to the boric acid evaporator capacity. The non-operating pump is a standby and is available for operation in the event the operating pump malfunctions. These canned centrifugal pumps are constructed of austenitic stainless steel.

9.2.2.35 Evaporator Feed Ion Exchangers

Four flushable evaporator feed ion exchangers (shared by both units) remove cations (primarily cesium and lithium) and anions from the holdup tank effluent. Two of the demineralizers are of the mixed bed type and the other two are of the cation bed type. One of each type are in series in each processing train.

The design flow rate is equal to the boric acid evaporator processing rate. The demineralizer vessels are constructed of austenitic stainless steel and are provided with suitable connections to facilitate resin replacement when required. The vessels are equipped with resin retention screens.

9.2.2.36 Ion Exchanger Filters

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

9.2.2.37 Boric Acid Evaporators

A boric acid evaporator is provided which will process 30 gpm of dilute radioactive boric acid and produce distillate and concentrated boric acid stripped of the radioactive gases. The other boric acid evaporator and associated equipment has been converted to a radioactive waste evaporator as described in Chapter 11. Radioactive gas stripping is achieved by passing heated feed through packed towers employing stripping steam which removes nitrogen, hydrogen and fission gases from the feed and is designed to reduce the influent gas concentration by a factor of 10^5.

9.2.2.38 Evaporator Condensate Demineralizers

A demineralizer removes low-level contaminants from the evaporator condensate. The resin may be anion, cation or mixed bed, and is selected based on the condensate chemistry profile. The other demineralizer (anion) has been converted to a radioactive waste disposal function as described in Chapter 11. Facilities are provided for regeneration of anion resin. When regeneration is no longer feasible, the resin is flushed to the spent resin storage tank. Each demineralizer is sized for a flow rate equal to the evaporator flow rate. The demineralizer vessel is made of all-welded austenitic stainless steel, and is equipped with a resin retention screen. This equipment is not normally used.

9.2.2.39 Condensate Filter

The filter collects resin fines and particulates from the boric acid evaporator condensate stream. The vessel is made of austenitic stainless steel, and is provided with connections for draining and venting. Disposable filter elements are used. The design flow capacity of the filter is equal to the total installed boric acid evaporator flow rate.

9.2.2.40 Monitor Tanks

Two shared monitor tanks permit continuous operation of the evaporator train. When one tank is filled, the contents are analyzed and either reprocessed, discharged to the Waste Disposal System or pumped to the primary water storage tank. The other two monitor tanks have been converted to a radioactive waste disposal function as described in Chapter 11.

Each of the tanks has sufficient capacity to hold the condensate produced during 12 hours of operation from an evaporator at full output with only two lab analyses per day.

The tanks are fitted with a nylon, rubber-coated membrane to prevent absorption of oxygen by the water stored in the tank. The portion of the tank above the membrane is vented to the auxiliary building atmosphere.

9.2.2.41 Monitor Tank Pumps

Two shared monitor tank pumps discharge water from the monitor tanks. Each pump is sized to empty a monitor tank in approximately 3 hours. The pumps are constructed of austenitic stainless steel.

9.2.2.42 Deborating Demineralizers

When required, two anion demineralizers remove boric acid from the Reactor Coolant System fluid. The demineralizers are provided for use near the end of a core cycle, but can be used at any time when boron concentration is low. Hydroxyl based ion-exchange resin is used to reduce Reactor Coolant System boron concentration by releasing a hydroxyl ion when a borate ion is absorbed. Facilities are provided for regeneration. When regeneration is no longer feasible, the resin is flushed to the spent resin storage tank.

Each demineralizer is sized to remove the quantity of boric acid that must be removed from the Reactor Coolant System to maintain full power operation near the end of core life.

9.2.2.43 Concentrates Filter

The filter removes particulates from the evaporator concentrates. Design flow capacity of the filter can accommodate the total installed boric acid evaporator capacity. The vessel is provided with connections for draining and venting. Disposable filter elements are used.

9.2.2.44 Concentrates Holding Tank

The shared concentrates holding tank is sized to hold approximately the production of concentrates from one batch from both evaporators. The tank is supplied with an electrical heater which prevents boric acid precipitation.

9.2.2.45 Concentrates Holding Tank Transfer Pumps

Two shared holding tank transfer pumps discharge boric acid solution from the concentrates holding tank to the boric acid tanks. The canned centrifugal pumps are sized to approximately match the capacity of the boric acid evaporator concentrates pumps or to pump out the contents of the tank in approximately 1 hour. The wetted surfaces are constructed of austenitic stainless steel.

9.2.2.46 Electrical Heat Tracing

The boric acid concentration for Units 1 and 2 has been reduced to 4% and no longer requires heat trace. The minimum operating temperature for Units 1 and 2 boric acid piping is 63°F. The piping, valves, line-mounted instrumentation, and components, which normally contain boric acid solution, are monitored to ensure ambient temperatures are above 63°F.

Electrical heat tracing is provided for sections of piping, valves and equipment for freeze protection purposes where needed.

Alternate methods of flushing or heating are provided for:

a. 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.

b. The boric acid tanks which are provided with immersion heaters.

c. The batching tank which is provided with a steam heated jacket.

d. The concentrates holding tank, which is provided with an immersion heater.

9.2.2.47 Valves

Isolation valves are provided for all connections to the Reactor Coolant System. Lines entering the reactor containment also have check valves inside the containment to prevent reverse flow from the containment.

Relief valves are provided for lines and components that might be pressurized above design pressure by improper operation or component malfunction. Pressure relief for the tube side of the regenerative heat exchanger is provided by a locked or sealed open valve and a spring loaded check valve bypassing the charging line isolation valves.

9.2.2.48 Piping

Chemical and Volume Control System 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 and hydrostatic testing.

9.2.3 System Design Evaluation

9.2.3.1 Availability and Reliability

A high degree of functional reliability is assured in the Chemical and Volume Control System by providing standby components where performance is vital to safety and by assuring fail-safe response to the most probable mode of failure.

The Chemical and Volume Control System has three high pressure charging pumps, which are capable of supplying the required reactor coolant pump seal and makeup flow.

Aside from those components that are also part of the Emergency Core Cooling System (Chapter 6), the Chemical and Volume Control System is not required to function during a loss-of-coolant accident.

The generation of a safety injection signal, occasioned by a loss-of-coolant accident, automatically closes the motor-operated valves in the outlet line of the volume control tank and in the normal charging line thus isolating the Chemical and Volume Control System from the safety injection path. The letdown line and reactor coolant pump seal water return line are isolated at the containment boundary by valves which automatically close as a result of high containment pressure caused by a loss-of-coolant accident. The centrifugal charging pumps are also automatically started and commence pumping into the Reactor Coolant System immediately on alignment of flow paths.

9.2.3.2 Control of Tritium

Tritium is produced in the reactor coolant because of irradiation of boron, lithium, and deuterium in the coolant. Also as a design basis, 30% of the tritium produced in the fuel rods and in the burnable poison rods (initial cycle only) is assumed to be released to the coolant. Recent operating experience with zircaloy cores indicates that the amount of tritium released to the coolant is substantially less than the design basis (about 1% instead of 30%).

The Chemical and Volume Control System is used to control the concentration of tritium in the Reactor Coolant System. Essentially all of the tritium is in chemical combination with oxygen as a form of water. Therefore, any leakage of coolant to the containment atmosphere carries tritium in the same proportion as it exists in the coolant. Thus, the level of tritium in the containment atmosphere, when it is sealed from outside air ventilation, is a function of tritium level in the reactor coolant, the relative humidity of the air in the containment and the presence of leakage other than reactor coolant as a source of moisture in the containment air.

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

a. Possible plant personnel hazard during access to the containment must be limited. Leakage of reactor coolant during operation with a closed containment causes an accumulation of tritium in the containment atmosphere.

b. Undue public hazard due to release of tritium to the plant environment must be avoided.

Both of these criteria are met in this plant.

The tritium concentration recommended as an upper limit in the reactor coolant is 2.5 µci/cc (at 580°F coolant temperature). This value was chosen to assure that the tritium concentration in the atmosphere of the containment will be low enough to permit access without protective equipment. The concentration of tritium in the reactor coolant is maintained at an acceptable level by discharging part of the condensate from the boric acid recovery process to the Waste Disposal System (Chapter 11). The design basis for the monitor tanks is to process for release four RCS volumes per year for tritium control.

9.2.3.3 Leakage Provisions

Any undecayed tritium in the reactor coolant will eventually be released via the Waste Disposal System to the plant discharge stream. In the plant discharge stream, the tritium (and other liquid radwastes) is mixed with the plant circulating water flow.

Quality control of the material and installation of the Chemical and Volume Control System valves and piping which are designated for radioactive service is provided, in order to essentially eliminate leakage to the atmosphere.

The components designated for radioactive service are provided with welded connections to prevent leakage to the atmosphere. However, flanged connections are provided on each charging pump suction and discharge, on each boric acid pump suction and discharge, on the relief valves inlet and outlet, on three-way valves and on the flow meters to permit removal for maintenance.

The positive displacement charging pump stuffing boxes are provided with leakoffs to collect reactor coolant before it can leak to the atmosphere.

Diaphragm or ball valves are provided where the operating pressure is 200 psi or below and the operating temperature is 200°F or below. Leakage to the atmosphere is essentially zero for these valves.

The CVCS is included in our plant preventive maintenance program. The system is inspected, the leakage measured and repaired as necessary. This program is performed on this system at least once per refueling cycle.

9.2.3.4 Incident Control

The letdown line and the reactor coolant pumps seal water return line penetrate the reactor containment. The letdown line contains four air-operated valves inside the reactor containment (three in parallel and one in series with the parallel valves) and one air-operated valve outside the reactor containment which are automatically closed by the containment isolation signal.

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

The four seal water injection lines to the reactor coolant pumps and the charging line are inflow lines penetrating the reactor containment. Each line contains a check valve inside the reactor containment to provide isolation of the reactor containment should a break occur in these lines outside the reactor containment.

9.2.3.5 Malfunction Analysis

To evaluate system safety, failures or malfunctions were assumed concurrent with a loss-of-coolant accident, and the consequences analyzed, see Table 9.2-4 and Chapter 14.

If a rupture takes place between the reactor coolant loop and the first isolation valve or check valve, an uncontrolled loss of reactor coolant occurs. The analysis of the loss-of-coolant accident is discussed in Chapter 14.

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

When the reactor is 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 intermediate and source range detectors except after the P-6 setpoint is reached after which the source range detectors are de-energized. Any appreciable increase in the neutron source multiplication, including that caused by the maximum physical boron dilution rate of approximately 680 ppm per hour, is slow enough to give ample time to start a corrective action (boron dilution stop and/or emergency boron injection) to prevent the core from becoming critical. The maximum dilution rate is based on the abnormal condition of two centrifugal charging pumps delivering unborated makeup water to the Reactor Coolant System at a particular time during refueling when the boron concentration is at the maximum value and the water volume in the system is at a minimum.

At least two separate and independent flow paths are available for reactor coolant boration; i.e., the charging line, or the reactor coolant pumps labyrinth seals. The malfunction or failure of one component does not result in the inability to borate the Reactor Coolant System. An alternate flow path is also 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.

Boration during operating to compensate for power changes will be indicated to the operator from a combination of two sources:

a. the control rod movement and

b. the flow indicator in the boric acid transfer pump discharge line.

When the emergency boration path is used, three indications to the operator are available. The primary indication is a flow indicator in the emergency boration line. The charging line flow indicator will indicate boric acid flow since the charging pump suction is aligned to the boric acid transfer pump for this mode of operation. The change in boric acid tank level is another indication of boric acid injection.

On loss of seal injection water to the reactor coolant pump seals, seal water flow may be reestablished by manually rerouting the flow or starting a standby charging pump. During operation without seal injection flow, the thermal barrier cooler serves to remove heat from the reactor coolant flow that passes through the thermal barrier cooler, thereby controlling the No. 1 seal leak-off temperature. In the event seal water injection flow cannot be reestablished prior to the reactor coolant pump No. 1 seal leak-off flow temperatures exceeding the alarm setpoint, the plant will be tripped and the reactor coolant pump operation stopped. Process controls will be utilized to maintain adequate seal cooling after the affected RCP(s) are secured.

It can be concluded that proper consideration has been given to station safety in the design of the system.

9.2.3.6 Galvanic Corrosion

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

For example, the galvanic corrosion of Inconel versus 304 stainless steel resulting from high temperature tests (575°F) in lithiated, boric acid solution was found to be less than 20.9 mg/dm^2 for the test period of 9 days. Further galvanic corrosion would be trivial since the cell currents at the conclusion of the tests were approaching polarization. Zircaloy versus 304 stainless steel was shown to polarize at 180°F in lithiated, boric acid solution in less than 8 days with a total galvanic attack of 3.0 gm/dm^2. Stellite versus 304 stainless steel was polarized in 7 days at 575°F in lithiated boric acid solution. The total galvanic corrosion for this couple was 0.97 mg/dm^2.

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

9.2.3.7 Tests and Inspections

Those portions of the CVCS associated with the ECCS will be subject to the same type of inspections required for those systems as outlined in Chapter 6. Special tests and inspections for the remainder of the CVCS are not required because the system is in daily operation. Routine maintenance can be performed on system components during refueling.

9.2.4 References for Section 9.2

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

9.0 AUXILIARY AND EMERGENCY SYSTEMS

9.3 RESIDUAL HEAT REMOVAL SYSTEM

9.3.1 Design Bases

The Residual Heat Removal System is designed to remove residual and sensible heat from the core and reduce the temperature of the Reactor Coolant System during the second phase of plant cooldown. During the first phase of cooldown, the temperature of the Reactor Coolant System is reduced by transferring heat from the Reactor Coolant System to the Steam and Power Conversion System (Chapter 10).

The Residual Heat Removal System is normally placed in operation approximately four hours after reactor shutdown when the pressure and temperature of the Reactor Coolant System are approximately 400 psig and less than 350°F, respectively. The design residual heat load is based on the residual heat fraction of full core MW (thermal) power level that exists at 20 hours following reactor shutdown from an extended power run near full power.

An original design basis for the plant is that a normal plant cooldown to 140°F (dual trains of RHR) be achieved within 20 hours of reactor shutdown. An updated analysis was performed regarding cooldown times assuming that the lake water temperature was at the maximum projected value. For dual train cooldown, the analysis predicted that the RHR system will reduce the reactor coolant temperature to less than 140ºF within 24.4 hours of reactor shutdown, and for single train cooldown to less than 200ºF in 36 hours of reactor shutdown. This analysis was performed with the ESW at 88.9ºF. However, certain other parameters were assumed, during this analysis, to be at nominal values and include limited RHR and CCW heat exchanger tube plugging, and operator action to increase RHR flow for the single train cooldown case.

The single train cooldown evaluation is performed to demonstrate the plant design capability to achieve Mode 5 within 36 hours, as may be required to comply with various plant Technical Specification Actions. The single train cooldown evaluation is not part of the Chapter 14 accident analyses.

The licensing basis cooldown requirements are contained in Technical Specification Section 3.0.3, Limiting Condition for Operation (LCO) Applicability.

The design parameters of the system are shown in Table 9.3-2.

As a secondary function, the Residual Heat Removal System is used to transfer refueling water between the refueling water storage tank and the refueling cavity at the beginning and end of refueling operations.

In addition, portions of the system are utilized as parts of the Emergency Core Cooling System and the Containment Spray Systems. These functions and the associated analyses are discussed in Chapters 6 and 14.

The system design precludes any significant reduction in the overall design reactor shutdown margin when cooling water is introduced into the core for decay heat removal or during the emergency core cooling recirculation mode of operation.

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

All system active components which are relied upon to perform the system functions are redundant and the system design includes provision for hydrostatic testing of system components to applicable code test pressures.

Codes and Classifications

All piping and components of the Residual Heat Removal System are designed to the applicable codes and standards listed in Table 9.3-1. Since the loop contains reactor coolant when it is in operation, austenitic stainless steel piping is employed.

9.3.2 System Design and Operation

System Description and Operation

The Residual Heat Removal System (shown in Figure 9.3-1) consists of two residual heat exchangers, two residual heat removal pumps and associated piping, valves, and instrumentation. The instrumentation is discussed in Chapter 7.

During system operation, coolant flows from the Reactor Coolant System to the residual heat removal pumps, through the tube side of the residual heat exchangers and back to the Reactor Coolant System. The inlet line to the Residual Heat Removal System loop begins at the hot leg of one reactor coolant loop and the return line is connected to the cold legs of two separate reactor coolant loops. There are three separate return lines that connect the outlet of the heat exchangers to the reactor coolant loops. A 12-inch line was designed to be the normal return line for RHR cooling. The east and west 8-inch ECCS injection lines can also be aligned for RHR cooling return. The heat loads are transferred by the residual heat exchangers to the component cooling water.

When the plant is in Cold Shutdown with no RCP's running or prior to stopping all RCP's, a portion of RHR flow may be aligned for hot leg injection for temperature control in hot leg # 3.

The cooldown rate of the reactor coolant is controlled by regulating the flow through the tube side of the residual heat exchangers. A bypass line, which serves both residual heat exchangers, is used to regulate the temperature of the return flow to the reactor coolant system as well as to maintain a constant flow through the RHR system. Once cooldown is achieved, the RHR system continues to provide long-term decay heat removal cooling to the RCS. System operation is the same as for cooldown except that the CCW flow to the RHR heat exchanger may be reduced to accommodate the reduced RHR heat load when RCS temperature is reduced below 200°F.

During normal plant operation, the two motor operated cross-tie valves in the bypass line are closed to prevent deadheading of the weaker of the two RHR pumps during the initial high-head stage of safety injection. The cross-tie isolation valves, downstream of the RHR heat exchangers, are normally open to allow one RHR pump to inject into all four RCS cold legs. Check valves downstream of the RHR pump minimum flow branch connection prevent deadheading of the weaker of the two RHR pumps. During recirculation, the cross-tie isolation valves are closed to provide a second isolation to the RWST and to separate the trains to protect against a passive failure.

Coincident with plant cooldown, a portion of the reactor coolant flow is diverted downstream of the residual heat exchanger to the chemical and volume control system for volume control and cleanup.

Remotely-operated, double valving is provided to isolate the residual heat removal suction line from the reactor coolant system. The suction line valves are interlocked to prevent inadvertent opening whenever the RCS pressure exceeds design pressure of the RHR system. The residual heat removal discharge lines are isolated from the reactor coolant system by two check valves in series for each line and a remotely operated valve common to both lines. During power operation, the remotely-operated valves in the suction from the RCS and the 12-inch diameter cooling return line to the RCS are normally closed. Additionally, motive power is removed from the valves by opening the supply circuit breaker to further prevent inadvertent operation of these valves. The valves in the injection lines have power available during power operation, as these lines are required to be available for ECCS injection.

Independent reduced inventory (mid-loop) monitoring systems are installed to ensure adequate core cooling when the RCS is at reduced inventory especially during mid loop operation. These systems provide control room indication of RCS level and temperature as well as RHR system parameters to prevent possible vortexing and/or air entrainment which could affect the decay heat removal capability of the RHR system.

Components

Residual Heat Removal System component design data are listed in Table 9.3-2.

Residual Heat Removal Exchangers

Two identical heat exchangers are installed in the system. Each heat exchanger is designed to provide one-half of the capacity to meet design normal cooldown requirements. The installation of two heat exchangers assures that the heat removal capacity of the Residual Heat Removal System is only partially lost if one heat exchanger fails or becomes inoperative. Two heat exchangers also allows maintenance of one while the other is in operation.

The residual heat exchangers are of the shell and U-tube type. Reactor coolant circulates through the tubes, while component cooling water circulates through the shell. The tubes are welded to the tube sheet to prevent leakage of reactor coolant.

Residual Heat Removal Pumps

Two identical pumps are installed in the Residual Heat Removal System. Each pump is sized to deliver sufficient reactor coolant flow through the residual heat exchangers to meet the plant cooldown requirements. The use of two pumps, installed in parallel, assures that pumping capacity is only partially lost should one pump become inoperative. This also allows maintenance work to be done on one pump while the other pump is in operation. In addition to the residual heat removal duty, the pumps are used for transfer of refueling water before and after a refueling operation. The two residual heat removal pumps are vertical, in-line centrifugal units with mechanical seals to prevent reactor coolant leakage to the atmosphere. All pump parts in contact with reactor coolant are austenitic stainless steel or equivalent corrosion resistant material.

The valves used in the Residual Heat Removal System are constructed of austenitic stainless steel or equivalent corrosion resistant material.

Manual isolation valves are provided to isolate equipment for maintenance. Regulating valves are provided for remote manual control of the residual heat exchanger tube side flow, and for remote manual control of bypass flow. Check valves prevent reverse flow through the residual heat removal pumps. Isolation of the suction side of the Residual Heat Removal System is achieved by closing two motor-operated gate valves in series in the line from the Reactor Coolant System. Isolation of the discharge side is accomplished by two check valves in series in each line from the residual heat removal pump discharge to the Reactor Coolant System plus a motor-operated gate valve common to both discharge lines. Overpressure protection in the Residual Heat Removal System is provided by relief valves discharging to the pressurizer relief tank in the Reactor Coolant System whenever RCS pressure exceeds design pressure of the RHR system.

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

Residual Heat Removal Piping

All Residual Heat Removal System piping is austenitic stainless steel. All piping joints and connections are welded except where flanged connections are required to facilitate maintenance.

9.3.3 System Design Evaluation

Availability and Reliability

For Reactor Coolant System cooldown, the unit is provided with two residual heat removal pumps and two residual heat exchangers. If one of the two pumps or one of the two heat exchangers or one pump and one heat exchanger is not operable, safe cooldown of the plant is not compromised; however, the time for cooldown is extended.

To assure reliability, the two residual heat removal pumps are connected to two separate buses so that each pump will receive power from a different source.

An emergency power source is provided to supply essential electrical equipment if a total loss of power should occur while the system is in service. Each pump is connected to a separate emergency power supply.

Incident Control

The Residual Heat Removal System is connected to a reactor coolant loop hot leg on the suction side and to two of the reactor coolant loop cold legs on the discharge side. On the suction side isolation is effected through two motor-operated gate valves in series, both of which are interlocked through separate channels of the Reactor Coolant System pressure signals. These interlocks prevent inadvertent opening of the suction valves whenever the Reactor Coolant System pressure exceeds the design pressure of the Residual Heat Removal System. Only one RHR pump will be operated when the RCS is open to atmosphere to prevent damaging both pumps in the unlikely event that suction should be lost. On the discharge side, isolation is made through a motor-operated valve and two check valves in series. All of which are closed whenever the reactor is in an operating condition.

Should a large tube side to shell side leak develop in a residual heat exchanger, the water level in a component cooling surge tank would rise, and the operator would be alerted by a high water alarm. If the leaking residual heat exchanger could not be isolated from the Component Cooling System before the inflow completely filled the surge tank, the overflow-vent line would discharge the excess water to the drain header if the amount is small. Larger flows, which might pressurize the surge tank, would be discharged through the surge tank safety valve to the Waste Disposal System. Since the Residual Heat Removal System is required for long-term post-accident removal of decay heat from the reactor core and containment, independent piping systems are provided for the redundant active components so that excessive leakage resulting from the deterioration of, or failure in, some passive element in the system can be identified and isolated without complete system loss-of-function.

Massive failure of piping is not considered credible because long term operation of the system occurs only at low pressures and temperatures and the system is protected from environmental conditions by Seismic Class I structures.

The following section describes how core cooling will be restored in the event of loss of the RHR system during shutdown cooling when 1) the Reactor Coolant System is filled, and 2) the Reactor Coolant System is not filled. CNP generally follows the guidance of Westinghouse Owners Group (WOG) Abnormal Response Guideline, ARG-1, "Loss of RHR While Operating at Mid-Loop Conditions.

<u>Case 1: The Reactor Coolant System is Filled</u>

The operator would be alerted to the loss of RHR flow by monitoring instrumentation in the Control Room. If the cause of the flow loss were a RHR pump trip, the operator would attempt to restart the pump or start the second pump. If RHR pump flow cannot be established, at least 3 hours will be available to the operator to establish an alternate means of core cooling. This is the time it would take to heat the available RCS volume from 350 °F to 445 °F, which is the saturation temperature at 400 psi, assuming the maximum 24-hour decay heat load.

If a secondary heat sink is available, heat removal is accomplished via the steam generators. The operator can employ steam dump to either the main condenser or to the atmosphere, with make-up to the steam generators from the auxiliary feedwater system.

To establish this alternate means of heat removal, a short period of time is required to open the steam dump valves and to start up the auxiliary feedwater system.

The equipment and systems actuated can include (depending on whether steam dump is to the atmosphere or the main condenser) pressurizer heaters to maintain subcooling margin, the atmospheric discharge power relief valves, the main condenser steam dump valves, the auxiliary feedwater system, and other steam plant systems (needed only if steam dump is to the main condenser). The auxiliary feedwater system is a safety-related system.

If heat transfer to the secondary system is unavailable the operator can establish feed and bleed cooling utilizing either a centrifugal charging pump or a safety injection pump to feed cool water into the RCS and bleed out through the Pressurizer Power Operated Relief Valve (PORV) to the Pressurizer Relief Tank (PRT). Required flow is determined by the decay heat load and controlled by the operator. This method of decay heat removal is utilized until either the RHR can be restored or the coupling to the secondary system can be established.

Case 2: The Reactor Coolant System is not Filled

The operator would be alerted to the loss of RHR flow by monitoring instrumentation in the Control Room (See Case 1). To restore core cooling, several options would be available to the operator. If shut down cooling is lost due to a loss of a RHR pump and not due to a loss of inventory, shutdown cooling can be restored by starting the second RHR pump. If shutdown cooling is lost due to a loss of inventory, water level can be restored above the minimum for RHR operation and RHR shutdown cooling decay heat removal re-established, provided that RCS is intact so that the system can be refilled. Depending upon RCS level and temperature, refilling is accomplished by normal charging flow, gravity feed from the RWST or by injection from RWST using the centrifugal charging or safety injection pumps. Once the system is filled, the methods listed above in Case 1 would be available. The flow paths (or combination of flow paths) are detailed in plant procedures and generally follow the ARG-1 Guidelines.

If the RCS is not intact such as the RPV head being off, alternate heat removal methods such as filling the Refuel Cavity and operating the Spent Fuel Pit Cooling and Cleanup System can be employed while efforts continue to reestablish RHR shutdown cooling and/or RCS integrity to allow use of the secondary system cooling, as described in Case 1.

9.3.4 Malfunction Analysis

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

9.3.5 Tests and Inspections

The residual heat removal pump flow instrumentation is calibrated on a periodic basis. Periodic visual inspections and preventative maintenance are also conducted. Certain system components are tested in accordance with the applicable edition of the ASME Operation and Maintenance (OM) Code. (Refer to Chapter 6).

9.3.6 Safety Limits and Conditions

9.3.6.1 Limiting Conditions for Maintenance

a. Administrative controls at the Plant have been established to permit the removal of RHR system equipment from service only to perform absolutely required maintenance when the RHR system is operating in the decay heat removal mode. If the equipment has to be removed from service, consideration must be given to alternate decay heat removal methods.

b. Administrative controls at the plant have been established requiring that during the condition when the reactor coolant system is depressurized and vented with air in the steam generator tubes, and the reactor vessel head in place (with or without bolting), both RHR trains must be available with either both emergency diesel generators or one diesel generator and the alternate reserve source available.

9.3.6.2 Operational Requirements

a. A requirement to have only one RHR pump in operation whenever the reactor coolant system is drained to half-loop and vented, has been incorporated into applicable operating procedures. The second pump will be in manual standby. This requirement will reduce total system flow, which in turn reduces the possibility of vortex formation and air entrainment at the suction line.

b. Only one RHR pump will be operated when the RCS is open to the atmosphere to prevent damaging both pumps in the unlikely event that the suction valve from the RCS should close.

Unit 1: The motor operated valves in the RHR bypass line are normally closed during power operation. Closing these RHR cross-tie valves makes the mini-flow circuits for each RHR pump independent thereby removing the potential for deadheading the weaker pump. However, since a safety evaluation has determined that the RHR cross-tie valves cannot be closed at the same time the SI cross-tie valves are closed, the RHR cross-tie valves would be opened when the SI cross-tie valves are closed for testing or maintenance. Administrative controls have been established for protecting the RHR pumps in this situation when deadheading is possible.

Unit 2: The manual valves in the RHR bypass line are normally closed during power operation. Closing these RHR cross-tie valves makes the mini-flow circuits for each RHR pump independent thereby removing the potential for deadheading the weaker pump.

c. As an alternate decay heat removal method, the capability exits for single phase natural circulation cooling when the RCS is below 350°F and secondary side inventory can be maintained as a heat sink. A secondary side steam generator level above the top of the tubes is conservatively maintained for this capability. When the RCS is below 200°F, two steam generators with sufficient secondary side inventory are capable of removing the expected heat load.

9.4 SPENT FUEL POOL COOLING SYSTEM

9.4.1 Design Bases

The Spent Fuel Pool Cooling System shown in Figure 9.4-1 is designed to remove from the spent fuel pool the heat generated by stored spent fuel elements. The system serves the spent fuel pool which is shared between the two units.

The system design allows for the need to totally unload a reactor vessel (193 fuel assemblies) for maintenance or inspection at a time when as many as 3420 spent fuel elements are already residing in the spent fuel storage pool.

The system design incorporates two separate cooling trains sharing a common return line into the spent fuel pool. System piping is arranged so that failure of any pipeline does not drain the spent fuel pool below the top of the stored fuel elements.

The Spent Fuel Pool Cooling System has two cooling trains capable of maintaining pool temperature at or below 142.3°F when one complete core is unloaded and stored in the pool in addition to 3420 spent fuel assemblies already stored. The normal refueling practice at D.C. Cook is to perform a full core off-load.

The system design will keep the maximum bulk pool water temperature at or below 180°F for a full core off-load following a normal discharge of 88 assemblies several months prior with one cooling train operational. The design basis normal off-load scenario assumes the previous discharge occurs as short as approximately 5 months prior and a single failure of one spent fuel pool cooling train. The minimum time to boil in the event that both loops of the cooling system become inoperable is 5.8 hours, assuming a worst case maximum heat load and a bulk pool temperature of 142.3°F prior to the loss of cooling. In addition, the cooling system has been analyzed to maintain pool temperature at or below 142.3°F for the abnormal case of a full core off-load following a discharge of 88 assemblies as short as 30 days prior. The design basis abnormal off-load scenario assumes the previous discharge occurs as short as 30 days prior and credits both trains of spent fuel pool cooling. Any spent fuel pool off-loading scenario, including a full core off-load of two units, which meets the 180°F peak bulk pool temperature with one train of cooling and 5.8 hours to boil criteria is acceptable.

The acceptability of a peak pool temperature of 180°F with respect to the pool concrete integrity was established utilizing American Concrete Institute (ACI) 349-97, “Code Requirements for Nuclear Safety Related Concrete Structure”.

Codes and Classifications

All piping and components of the system are designed to the applicable codes and standards listed in Table 9.4-1.

9.4.2 System Design and Operation

System Description

Each of the two cooling loops in the Spent Fuel Pool Cooling System (see Figure 9.4-1) consists of a pump, heat exchanger, strainer, piping, associated valves and instrumentation. The pump draws water from the pool, circulates it through the heat exchanger and returns it to the pool. Component cooling water cools the heat exchanger.

Of the two trains of cooling in the spent fuel pool cooling system, during normal cooling of the pool, one train is in operation with the second train serving as back-up. The second train is also used as an alternate cooling system in case there is a loss of cooling event to the spent fuel pool. There is also a tie line, a 3 inch diameter pipe that connects the two independent cooling trains. This tie line will allow cross-tying the two independent trains of the cooling system if a pump and a heat exchanger in different trains malfunction. Although operating at reduced capacity, use of the cross-tie can extend the time to boil of the spent fuel pool during this loss of spent pool cooling event. The use of the cross connection will connect the south pump to the north heat exchanger or vice versa. The alignment of the flow between the north train and the south train is accomplished by opening the crosstie valves and throttling the heat exchanger valves as required.

The clarity and purity of the spent fuel pool water is maintained by passing up to 150 gpm of the cooling flow through a filter and demineralizer. Skimmers are provided to prevent dust and debris from accumulating on the surface of the water.

The refueling water purification pump and filter can be used separately or in conjunction with the spent fuel pool demineralizer to regain refueling water clarity after a crud burst in either unit. This can prevent loss of time during refueling due to poor visibility. The system is also used to maintain water quality in the Refueling Water Storage Tanks of both units.

The spent fuel pool pump suction lines penetrate the spent fuel pool wall above the fuel assemblies stored in the pool to prevent loss of water as a result of a suction line rupture. The pool is initially filled with water at the same boron concentration as in the refueling water storage tank.

There is sufficient capacity in the spent fuel pool to store up to 3420 spent fuel assemblies above and beyond the space required for the complete unloading of one unit (193 fuel assemblies). If any of this extra storage capacity is being utilized, it is by "cold" spent fuel assemblies. These are assemblies that have been removed from the reactor (e.g. during previous refuelings) and have been stored sufficiently long to reduce decay heat production to a relatively low level.

With the maximum heat loading 3420 spent fuel assemblies plus one complete core and two cooling trains operating, the temperature is analyzed to remain at or below 142.3°F. With the maximum heat loading of 3420 spent fuel assemblies plus one complete core and one cooling train operating, the temperature is analyzed to remain at or below 180°F.

If all cooling is lost and 3420 spent fuel assemblies are stored in the pool, the time required for the spent fuel pool to boil (approximately 211°F) with one complete core added, is approximately 5.8 hours assuming an initial bulk pool temperature of 142.3°F.

A failure consideration applicable to both units is a remote occurrence. However, should both cores require removal when up to 3420 fuel assemblies are already in the spent fuel pool, one of the cores is placed in the spent fuel pool and the other is left in its reactor vessel. The core added to the spent fuel pool brings the inventory up to 3613 assemblies, which can be safely handled. The other core is left in place in its reactor vessel, with the residual heat removal system in service, until there is space available for it in the spent fuel pool.

The spent fuel pool is located outside the reactor containment. During refueling the water in the pool can be isolated from that in the refueling canal by a gate valve so that there is only a very small amount of interchange of water as fuel assemblies are transferred.

Components

Spent Fuel Pool Cooling System component design data are listed in Table 9.4-2. The component data contained in Table 9.4-2 is original equipment design and sizing data. The CCW system has been designed and analyzed to:

a. Operate in the range of 60ºF to 105ºF, except during periods of cooldown and post-LOCA operation, and

b. Operate at temperatures ≤120°F during cooldown and post-LOCA operation.

Spent Fuel Pool Heat Exchangers

The two spent fuel pool heat exchangers are of the shell and U-tube type with the tubes welded to the tube sheet. Component cooling water circulates through the shell, and spent fuel pool water circulates through the tubes. The tubes are austenitic stainless steel and the shell is carbon steel.

Spent Fuel Pool Pumps

The two spent fuel pool pumps circulate water in the spent fuel pool cooling loops. All wetted surfaces of the pump are austenitic stainless steel, or equivalent corrosion resistant material. The pumps are operated manually from a local station.

Spent Fuel Pool Filter

The spent fuel pool filter removes particulate matter larger than 5 microns from the spent fuel pool water. The filter element is disposable. The vessel shell is austenitic stainless steel.

Spent Fuel Pool Strainer

A stainless steel strainer is located at the inlet of each fuel pool cooling suction line for removal of relatively large particles which might otherwise clog the spent fuel pool demineralizer or damage other components in the system.

Spent Fuel Pool Demineralizer

The demineralizer is sized to pass up to 150 gpm of the cooling flow to provide adequate purification of the fuel pool water for unrestricted access to the working area and to maintain water clarity.

Spent Fuel Pool Skimmer

A spent fuel pool skimmer pump, strainer, filter, and two skimmers are provided for surface skimming of the spent fuel pool water. This subsystem maintains the needed clarity for visual observations of the pool water.

Refueling Water Purification Pump

The shared refueling water purification pump provides for circulation of refueling water from either the refueling canal or the refueling water storage tank for purification. Its wetted surfaces are austenitic stainless steel.

Refueling Water Purification Filter

The refueling water purification filter removes particulate matter larger than 5 microns from the refueling water. The filter element is disposable.

Spent Fuel Pool Cooling System Valves

Manual stop valves are used to isolate equipment and manual throttle valves provide flow control. Valves in contact with spent fuel pool water are austenitic stainless steel or equivalent corrosion resistant material.

Spent Fuel Pool Cooling System Piping

All piping in contact with spent fuel pool water is austenitic stainless steel. The piping is welded except where flanged connections are used at the pumps, heat exchangers, and filters to facilitate maintenance.

9.4.3 Design Evaluation

Availability and Reliability

The availability of two cooling trains allows prolonged outages of either cooling loop.

Leakage Provisions

Whenever a leaking fuel assembly is transferred from the fuel transfer canal to the spent fuel storage pool, a small quantity of fission products may enter the spent fuel cooling water. A purification loop is provided for removing these fission products and other contaminants from the water.

Incident Control

The most serious failure of this system would be complete loss of water in the storage pool. To protect against this possibility, the spent fuel pool cooling connections enter near the water level so that the pool cannot be gravity-drained.

Malfunction Analysis

Failure analyses of system pumps, heat exchangers and valves are presented in Table 9.4-3.

9.4.4 Tests and Inspections

The active components of the system are in continuous use during normal plant operation. The spent fuel pit pumps are tested in accordance with the requirements of the applicable edition of the ASME Operation and Maintenance (OM) Code. Additionally, periodic visual inspections and preventive maintenance are conducted following normal industry practice.

9.5 COMPONENT COOLING SYSTEM

The Component Cooling System, shown in Figure 9.5-1, is duplicated for each unit. The only shared piece of equipment is the maintenance spare Component Cooling pump installed in the Unit 1 area. The miscellaneous service train can be fed from either safeguards train.

9.5.1 Design Bases

The system is designed to: a) remove residual and sensible heat from the Reactor Coolant System, via the Residual Heat Removal System, during plant shutdown; b) cool the spent fuel pool water and the letdown flow to the Chemical and Volume Control System during power operation; c) provide cooling to dissipate waste heat from various primary plant components, and d) provide cooling for safeguards equipment.

The system design provides radiation monitors for the detection of radioactivity entering the system from the Reactor Coolant System and its associated auxiliary systems, and includes provisions for isolation of system components.

All piping and components of the Component Cooling System have been designed to the applicable codes and standards listed in Table 9.5-1. Component cooling water contains a corrosion inhibitor to protect the carbon steel piping and equipment.

9.5.2 System Design and Operation

The Component Cooling Water (CCW) System provides cooling for the following heat sources:

Safeguards Train

a. Residual Heat Removal Heat Exchanger

b. Centrifugal Charging Pump Gear and Lube Oil Heat Exchangers

c. Safety Injection Pump Seal and Lube Oil Heat Exchangers

d. Residual Heat Removal Pump Seal Heat Exchangers

e. Containment Spray Pump Seal Heat Exchangers

Miscellaneous Services Train

a. Sample Heat Exchangers

b. Reciprocating Charging Pump Bearing and Fluid Drive Heat Exchangers

c. Spent Fuel Pit Heat Exchanger

d. Waste Gas Compressor and Seal Water Heat Exchangers

e. Reactor Coolant Pump Seal Water Heat Exchanger

f. Letdown Heat Exchanger

g. Boric Acid Evaporator Heat Exchangers

h. Steam & Feedwater Containment Penetration Heat Exchangers

i. Excess Letdown Heat Exchanger

j. Reactor Support Coolers

k. Reactor Coolant Pump Thermal Barrier Heat Exchanger

l. Reactor Coolant Pump Motor Upper Bearing Oil Cooler

m. Reactor Coolant Pump Motor Lower Bearing Oil Cooler

n. 15 GPM Waste Evaporator Heat Exchangers

o. Containment Air Recirculation Fan Motor Coolers

The CCW system has been designed and analyzed to:

a. Operate in the range of 60°F to 105°F, except during periods of cooldown and post-accident conditions, and

b. Operate at temperatures ≤120°F during cooldown and post-accident conditions.

Table 9.5-2 contains system and component flow information for 4 different operating conditions.

a. Normal Operation

This is defined as power operation and shutdown conditions that are controlled by the plant normal operating procedures. The values in Table 9.5-2 are nominal flow to components, required to support normal operation of the plant within the licensing basis. The Safeguards Train flows in Table 9.5-2 provide the required cooling at the CCW design basis temperature of 120°F. Because CCW temperatures and cooling requirements are lower during normal operating conditions, lower flow is acceptable. Plant procedures may consider uncertainty as appropriate.

b. Cooldown

This is defined as RCS temperature reduction using plant normal operating procedures in Mode 4 and 5. The values in Table 9.5-2 are the nominal flows to components, required to support cooldown of the plant within the licensing basis. Plant procedures may consider uncertainty as appropriate.

c. LOCA Injection

This is defined as post-LOCA injection for accident response and mitigation, using plant emergency operating procedures. The values in Table 9.5-2 are the minimum flows to components, required to meet the accident mitigation strategy in the accident analyses supporting the licensing basis. Plant procedures may consider uncertainty as appropriate.

d. LOCA Recirculation

This is defined as post-LOCA cold leg and hot leg recirculation for accident response and mitigation, using plant emergency operating procedures. The values in Table 9.5-2 are the minimum flows to components, required to meet the accident mitigation strategy in the accident analyses supporting the licensing basis. Plant procedures may consider uncertainty as appropriate.

The CCW system is arranged in three flow circuits, two parallel safeguards equipment trains, and one miscellaneous services train which can be served by either of the safeguards trains.

Since the heat is transferred from the component cooling water to the service water, the component cooling loop serves as an intermediate system between the reactor coolant and the service water system and insures that any leakage of radioactive fluid from the components being cooled is contained within the plant. The surge tank accommodates expansion and contraction, and ensures a continuous component cooling water supply. Because this tank is normally vented to the auxiliary building atmosphere, a radiation monitor is provided at the outlet of each component cooling heat exchanger to detect any inleakage of radioactive fluid. These monitors actuate an alarm and close the surge tank vent valve when the radiation level reaches a preset level above the normal background.

The Component Cooling System consists of two component cooling pumps, two component cooling heat exchangers, one surge tank and associated piping and valves to serve each unit.

One pump and heat exchanger, with associated equipment, forms a 100% train. Another use of the CCW pumps is to provide a CCW supply to the other unit in support of NFPA 805 safe and stable conditions. An additional pump is provided as an installed maintenance spare for either unit and is located in a crosstie header between the Unit 1 and 2 systems. The piping and valve arrangement is such that the maintenance spare can supply water to any one of the four trains, after the electrical controls have been transferred to it from the affected train.

One pump and one heat exchanger are required for the removal of residual and sensible heat from the reactor coolant system via the residual heat removal system during the cooldown of one unit. Full power operation of one unit, including cooling of a spent fuel pit heat exchanger, likewise requires one pump and one heat exchanger. Therefore, the remaining train serves as a standby and can be placed in service, if required, to increase system capability. Provision is made to add makeup to the system through lines connected to the surge tank.

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

a. Temperature recorder and alarm in the outlet lines for each of the component cooling heat exchangers.

b. A pressure and flow indicator in the supply line to each of the component cooling heat exchangers.

c. A radiation monitor at the outlet of each of the component cooling heat exchangers.

d. Flow indicators and/or alarms located in the discharge lines of the major heat exchangers served by the system.

e. Temperature indicators and/or temperature test points located in the discharge lines of the major heat exchangers served by the system.

In the event of a loss of coolant accident, one pump and one heat exchanger are capable of fulfilling system requirements. Following a LOCA, both trains receive an automatic start signal. Cooling water for the component cooling heat exchangers is supplied from the Essential Service Water System (Chapter 9) insuring a continuous source of cooling medium.

9.5.3 Components

Component Cooling System component design data are listed in Table 9.5-3.

Component Cooling Heat Exchangers

The component cooling heat exchangers are of the shell and tube type. Service water circulates through the tubes while component cooling water circulates through the shell side. The shell side is of carbon steel and the tubes are of arsenical copper.

Component Cooling Pumps

The component cooling water pumps which circulate water through the component cooling water loops are horizontal, centrifugal units and motor driven. The motors receive electric power from normal and emergency sources.

Component Cooling Surge Tank

The component cooling water surge tank accommodates changes in component cooling water volume and is constructed of carbon steel. In addition to piping connections at each pump's suction, the tank is provided with a means of adding a chemical corrosion inhibitor to the component cooling loop. The tank is internally divided (baffled) to form, in effect, two compartments. This arrangement provides redundancy for a passive failure during recirculation phase following a LOCA.

Valves

The valves used in the component cooling loop are constructed of carbon steel with the internals upgraded to stainless steel as needed during repairs. Isolation valves serving the stainless steel piping to the Reactor Coolant Pump Thermal Barrier are manufactured from stainless steel. Certain small valves (2” and under) are of a threaded bronze construction in the low pressure portions of the CCW system. Relief valves are provided for lines and components that could be pressurized beyond their design pressure by improper operation or malfunction.

The relief valves on the component cooling water lines downstream from each reactor coolant pump thermal barrier are designed to relieve excessive pressure that may be caused by over heating. The relief valve set pressure equals the design pressure of the particular segment of piping between the upstream check valve and downstream motor-operated discharge valves.

The relief valves on the cooling water lines downstream of the sample, excess letdown, seal water, spent fuel pit and residual heat exchangers are sized to relieve the volumetric expansion occurring if the exchanger shell side is isolated and high temperature liquid flows through the tube side. The set pressure is less than or equal to the design pressure of the shell side of the heat exchangers.

The relief valve on the component cooling surge tank is sized to relieve the maximum flow rate of water that would enter the surge tank following a rupture of a reactor coolant pump thermal barrier cooling coil. The set pressure assures that the design pressure of the component cooling system is not exceeded. The discharge of this valve is directed to the waste holdup tank.

The component cooling water surge tank vent-overflow line, which is open to the auxiliary building atmosphere, is equipped with an air-operated valve that will close automatically if radiation is detected in the system. A vacuum breaker valve is also provided to prevent collapsing this tank in the event of a large loss of water in the system.

Piping

The component cooling loop piping is carbon steel with flanged joints and connections at components, which might require removal for maintenance. Certain small (2" and under) portions of the piping system may be threaded. All other joints are welded. One exception to the carbon steel is that portion of the piping between the double check valves and the motor-operated discharge isolation valves for the reactor coolant pump thermal barrier cooling, which is stainless steel. Additionally, some selected portions at heat exchangers may be constructed of copper tubing/pipe.

9.5.4 System Evaluation

Availability and Reliability

The component cooling pumps, heat exchangers, and associated valves, piping and instrumentation are located outside of the containment and are therefore available for maintenance and inspection during power operation. Replacement of a pump, or maintenance on a heat exchanger is practical while redundant units are in service. Sufficient cooling capability is provided to fulfill all system requirements under normal and accident conditions. Adequate safety margins are included in the size and number of components to preclude the possibility of a component malfunction adversely affecting operation of safeguards equipment.

Incident Control

If outleakage occurs anywhere in the Component Cooling System, including a non-seismic I component served by the Miscellaneous Service Train, detection is accomplished by falling level in the surge tank. The surge tank is equipped with a low-level alarm that annunciates in the control room. Level alarms from the sumps to which this water will drain, also serve as leak indicators.

The leaking portion of the system is then shut down and isolated and the backup train is put in operation. To minimize the possibility of leakage from piping, valves, and equipment, welded construction is used wherever possible.

For leakage into the Component Cooling System, a high level alarm is provided in the Control Room for the surge tank.

The component cooling water could become contaminated with radioactive water due to a leak in any heat exchanger tube in the chemical and volume control, residual heat removal, sampling or the spent fuel pool cooling system or from a leak in a cooling coil for the thermal barrier cooler on a reactor coolant pump or from various pump seal water heat exchangers. The detection of this contamination is by a radiation monitor located at the outlet of each of the component cooling water heat exchangers.

Component cooling water flow at a reduced rate is automatically established to the residual heat removal heat exchanger at the safety injection signal. Since the thermal demand on this heat exchanger is minimal at this time, full design component cooling water flow is not required. When at least the RHR pumps and the CTS pumps suction has been transferred to the recirculation sump, and subsequently the component cooling water trains have been separated, full design flow will be established to the available RHR heat exchangers.

The component cooling water lines to and from the reactor support coolers and the excess letdown heat exchanger have valves outside the containment wall, which are automatically closed on the Phase A isolation signal.

If normal seal water supply is unavailable to the reactor coolant pumps, the cooling water to the RCP thermal barriers should be available to assure that there will be no mechanical damage to the pump. Therefore, isolation valves for the component cooling water for this service are not automatically closed until a Phase B (containment spray) containment isolation signal is received. The cooling water supply line to the reactor coolant pumps contains two remote-operated valves in series outside the containment wall. The return lines from the thermal barriers and RCP motor bearings each have two remote-operated valves in series outside the containment wall. These redundant valves assure the ability to isolate this circuit if a leak is detected. Leak detection is accomplished by flow alarms and indicators in the supply and return lines of this circuit.

Except for the normally closed makeup line and equipment vent and drain lines, there are no direct connections between the component cooling water and other systems. The equipment vent and drain lines outside the containment have manual valves which are normally closed unless the equipment is being vented or drained for maintenance or repair operations.

Malfunctions Analysis

A failure analysis of pumps, heat exchangers and valves is presented in Table 9.5-4.

9.5.5 Minimum Operating Conditions

Minimum operating conditions are given in the technical specifications.

9.5.6 Tests and Inspections

Pumps and certain valves in the Component Cooling System are tested in accordance with the applicable edition of the ASME Operation and Maintenance (OM) Code. Containment isolation valves will be tested periodically in accordance with procedures established in Chapter 5. Periodic visual inspection and preventative maintenance are conducted following normal industry practice.

9.0 AUXILIARY AND EMERGENCY SYSTEMS

9.6 SAMPLING SYSTEMS

9.6.1 Design Bases

Two separate systems provide means to obtain samples from various systems in each of the two units for chemical and radiochemical analysis. One system, the "Normal Sampling System" (NSS), provides for sampling during normal operation. The second system, the "Post-Accident Sampling (PAS) System", provides for sampling following a loss-of-coolant accident. Adequate safety features are provided to protect laboratory personnel and to prevent the spread of contamination from the sampling areas when samples are being drawn. The sampling systems' discharges are designed to limit flows to preclude any fission product release exceeding the 10 CFR 20 exposure limits under normal and anticipated malfunctions or failures. Upon a Phase A containment isolation signal, both systems are isolated at the containment boundary. The Post-Accident Sampling System has provisions for opening containment isolation valves under administrative control so post-accident samples can be taken to assess containment atmospheric and reactor coolant conditions. Reactor coolant samples can be obtained by the Normal Sampling System during reactor operation and during cooldown when the residual heat removal loop is in operation.

Sample Temperatures

High pressure, high temperature samples as well as the residual heat removal loop samples are cooled sufficiently to minimize generation of radioactive aerosols.

Codes & Standards

The sampling system piping up to and including the outermost containment isolation valve is designed to the 1967 Edition of the USAS B31.1, Code for Pressure Piping. Repair and replacement for pressure retaining components within the code boundary, and their supports, are conducted in accordance with ASME Section XI.

9.6.2 System Design

9.6.2.1 Normal Sampling System

The Normal Sampling System (NSS), shown in Figure 9.6-1, is designed to provide representative samples for laboratory analyses used to guide the operation of various primary and secondary systems throughout the plant during normal operation.

The Normal Sampling System also may be used as a vent path for purging non-condensable gases from the PZR (Pressurizer) steam space.

Provisions have been made for drawing samples in the Nuclear Sampling Room from each unit's:

a. Pressurizer steam space

b. Pressurizer liquid space

c. Two reactor coolant hot legs (loops 1 & 3)

d. Each of the four accumulators

e. Two residual heat removal lines

f. The Chemical and Volume Control System letdown line at the demineralizer inlet header

g. The Chemical and Volume Control System letdown line at the demineralizer outlet header

h. The volume control tank gas space

i. Each of the four steam generator blowdown lines

Typical of the analyses performed on samples are boron concentration, fission product radioactivity level, dissolved gas content, and corrosion product concentration. In addition, local sample points are provided at various locations outside the reactor containment for occasional sampling of other systems. These are not considered part of the sampling system. Analytical results are used to regulate boron concentration, evaluate fuel element integrity, evaluate mixed bed demineralizer performance, regulate additions of corrosion controlling chemicals and monitor primary and secondary water purity. Except for the steam generator blowdown sampling, the NSS is designed to be operated manually and intermittently for conditions from full power operation to cold shutdown.

Gamma spectrometric analyses of the liquid primary coolant samples are performed, where possible, without further preparation of the sample. In instances involving separation techniques, the time delay is dependent upon the particular component of interest. For normal routine analyses of liquid samples, including non-radioactive species, completion can usually be accomplished within 4 hours.

For gaseous components of primary coolant, liquid samples are collected in sampling vessels and degassed in the laboratory according to detailed plant procedures. Gas samples are then counted utilizing gamma spectrometry. In most cases, this can be accomplished within 1-2 hours after sampling.

The NSS incorporates means of purging a sample line for a sufficient period of time to ensure collection of a representative sample. Local flow, temperature and pressure measuring devices have been included in the nuclear sampling room to monitor these parameters.

Liquid samples are cooled and depressurized. Temperatures are maintained high enough after cooling to prevent solids from precipitating out. In addition, sample runs are kept to the minimum practicable and all sample lines and coils are constructed of materials compatible with coolant chemistry.

The reactor coolant sample points which are normally inaccessible and which require frequent sampling are permanently piped to a sampling room. The sample lines originating inside the reactor containment have remotely-operated isolation valves outside the containment. A delay coil located inside the containment provides for decay of short-lived radioactive isotopes present in the reactor coolant system samples. With the delay coil, it takes 2 1/2 to 3 1/2 minutes for a sample increment to reach the sampling room. The samples are cooled as they flow through the sample heat exchangers and the pressure is reduced by pressure-reducing needle valves. The sample flow is directed to the volume control tank through a purge line until sufficient volume has passed to obtain a representative sample. A portion of the flow is then diverted to the sample sink where the sample is collected. Reactor coolant gas samples and pressurizer steam samples are collected in sample vessels.

Liquid samples originating upstream and downstream of the Chemical and Volume Control System mixed bed demineralizer pass through a common sample line at the sample sink. Provisions have been made to also route these samples to a sampling vessel. The sample from the volume control tank gas space of the Chemical and Volume Control System is also collected in a sample vessel.

The volume control tank gas sample is purged through the vent header to the Waste Disposal System. The samples are collected in the sample vessel or other container and are transported to the laboratory for analysis. Because the pressurizer steam space sample, the reactor coolant dissolved gas sample, and the volume control tank gas space sample may contain some radioactive gases, their respective sample vessel stations are located in a small, well-ventilated cabinet within the sampling room to collect any gas which may be released during sample collection. The cabinet is vented to the auxiliary building ventilation system.

A continuous sample is drawn from each steam generator blowdown line at a point inside the containment. The samples are combined and automatically monitored for radiation and discharged to the blowdown tank drain line. A loss of flow alarm is provided on each sample line. In the event of a high radiation level, the steam generator blowdown and the sample isolation valves close automatically. Individual steam generator samples may then be drawn under administrative control to determine the radiation source.

The sample sink is located in a fume hood vented to the auxiliary building ventilation system. The sink drains to the Waste Disposal System. The work area around the sink has space for portable radiation monitoring equipment in addition to that for sample collection and storage. The sink perimeter has a raised lip to contain any spilled liquid. The sink and work area are stainless steel.

With the exception of the steam generator blowdown sample lines, the sample lines and a deionized water line for flushing, penetrate the sides of the hood. Local instrumentation is provided to permit manual control of sampling operations and to assure that the samples are at suitable temperatures and pressures before diverting flow to the sample sink. All purging flow rates are displayed by flow indicators.

Samples may be drawn for reactor coolant from a loop drain during core off-load conditions or for steam generator blowdown from a blowdown radiation monitoring loop drain during operational conditions when the normal sampling means is unavailable. The use of these drain taps to draw samples from system dead legs will be considered valid when proper equipment control and ALARA principles are followed

9.6.2.2 Post-Accident Sampling System

The Post-Accident Sampling (PAS) System shown in Figure 9.6-2, is designed to provide representative samples from designated plant fluid streams for laboratory analysis following a loss-of-coolant accident. This system provides for contingency sampling for reactor coolant, containment atmosphere, and containment sump (via RHR when in recirculation mode) following an accident. The system is common to both Units 1 and 2. It provides dilute liquid and gas grab sampling capability. Hydrogen sampling contingency can be performed for containment air sampling.

Provisions have been made for drawing liquid samples in each unit from:

a. Loop 1 hot leg, reactor coolant system

b. Loop 3 hot leg, reactor coolant system

c. Residual Heat Removal System

Connections into existing liquid sample lines are routed to the Post-Accident Sampling Panel located in the Spray Additive Tank Room. The Reactor Coolant System (RCS) samples from loop 1 and 3 hot legs and the RHR system sample pass through two heat exchangers prior to flowing to the sampling panel. Liquid waste from the sampling panel is purged to the PAS Waste Collection Tank and then pumped to the appropriate unit's containment.

Provisions have been made for drawing gas samples in each unit from the containment air space. A connection into an existing radiation monitor sample line is routed through a heat exchanger to the sampling panel where containment air samples are collected for laboratory analysis.

The containment air sample is taken after recirculating the air to obtain a representative sample. Gaseous waste from the sampling panel is returned directly to the appropriate unit's containment.

9.6.3 System Evaluation

9.6.3.1 Normal Sampling System

Availability and Reliability

The system is not required to function during an emergency, nor to take action to prevent an emergency condition. Therefore it is designed to perform in accordance with standard practice of the chemical process industry.

Leakage Provisions

Leakage of radioactive reactor coolant from this system within the containment is collected in the containment sump. Leakage of radioactive material from this system outside the containment is collected, via miscellaneous drains in the Waste Hold-Up system. The sampling stations within the room are provided with off-gas vents to the auxiliary building ventilation system. Liquid leakage through the valves at the sampling hood is drained to the Waste Disposal System.

Incident Control

With the exception of the steam generator blowdown samples, which are automatically isolated upon a high radiation or containment isolation signal, the system is designed for operation on an intermittent basis under administrative control. Sample lines penetrating the containment are equipped with isolation valves, which close on receipt of a containment isolation signal. The isolation valve in the CVCS letdown line outside the reactor containment also closes on receipt of a containment isolation signal, thereby isolating the demineralizer inlet and outlet sample lines in the event of a loss-of-coolant-accident.

Following a loss-of-coolant-accident, reliance on the Normal Sampling System to assay the primary water chemistry is not necessary since samples and analysis can be obtained using the Post-Accident Sampling System.

9.6.3.2 Post-Accident Sampling System

Availability and Reliability

The system is designed to function after a LOCA. It permits collecting liquid and gas samples. These samples can be transported to the laboratory for analysis.

Leakage Provisions

Leakage of radioactive reactor coolant from this system within the containment is collected in the containment sump. Liquid leakage outside the containment drains via miscellaneous drains to the Waste Disposal System. The Post-Accident Sampling Panel has provisions for handling gas leakage. Gaseous leaks within the PAS Panel are routed to the auxiliary building ventilation system via a HEPA and charcoal filtration unit.

Incident Control

Sample lines penetrating the containment are equipped with isolation valves, which close on receipt of a containment isolation signal. Isolation valves that need to be opened in order to obtain post-accident samples can be opened by overriding the appropriate containment isolation signal under administrative control in accordance with plant procedures.

9.7 REACTOR COMPONENTS AND FUEL HANDLING SYSTEM

The Reactor Components and Fuel Handling System provides a safe, effective means of transporting and handling fuel from the time it reaches the plant in an unirradiated condition until it leaves the plant after post-irradiation cooling. Each unit has its own fuel handling equipment within its containment and an independent fuel transfer mechanism. Other fuel handling equipment used in and around the spent fuel pool is shared.

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

The Reactor Components and Fuel Handling Systems consist basically of:

a. The reactor and refueling cavities.

b. The transfer canal and the spent fuel pool, which are accessible to operating personnel.

c. The Fuel Transfer System, which consists of an underwater conveyor, RCC changing fixture, new and spent fuel handling crane, manipulator crane, transfer tube, and new fuel elevator.

d. Fuel racks.

e. Polar crane.

9.7.1 Design Bases

During reactor vessel head removal, and while loading and unloading fuel from the reactor, the boron concentration is maintained at not less than that required to shutdown the core to a K_{eff} = 0.95. Refueling water boron concentration is verified in accordance with technical specification surveillance requirements to ensure the proper shutdown margin.

The new fuel storage racks are designed so that it is impossible to insert assemblies in other than the storage cells in the racks, thereby maintaining separation. The poisoned high density spent fuel storage racks are designed such that no assembly can be placed any closer to another assembly than that required by the critical analysis to maintain the required K_{eff} of ≤ 0.95. The new fuel storage rack accommodates 144 fuel assemblies, over two-thirds of a core, and a spent fuel storage pit accommodates 3613 fuel assemblies, slightly more than eighteen and one-half cores, plus the required spent fuel shipping cask area. Borated water is used to fill the spent fuel storage pool and maintain it at a concentration to match that used in the refueling cavity and refueling canal during refueling operations. (The fuel is stored in a vertical array with sufficient center-to-center distance between assemblies to assure $K_{eff} \leq 0.95$ [even if unborated water is used to fill the pool.])

The new fuel storage vault (NFSV) rack analysis is based on maintaining $K_{eff} \leq 0.95$ under full water density conditions and ≤ 0.98 under low water density (Optimum moderation) conditions.

The design basis for preventing criticality outside the reactor is that, including uncertainties, there is a 95 percent probability at a 95 percent confidence level that the effective neutron multiplication factor, K_{eff}, of the NFSV when flooded with full density water will be ≤ 0.95 as recommended by ANSI 57.3-1983 and NRC guidance. Furthermore, the effective neutron multiplication factor, K_{eff}, of the NFSV under optimum moderation (aqueous foam) conditions will be less than 0.98 as recommended by NUREG-0800.

Fuel assemblies and enrichments up to 4.55 w/o ^{235}U can be safely stored in the NFSV. The maximum 95/95 K_{eff} determined for full water density flooding is 0.9495 and the maximum 95/95 K_{eff} determined for optimum moderation flooding is 0.8974. Based on these previously calculated K_{eff} values, the acceptance criteria are met for both full and optimum water density flooding of the new fuel storage racks. A maximum nominal enrichment of 4.95 weight percent U-235 for Westinghouse fuel types is acceptable provided that sufficient integral fuel burnable absorber specified in Tech Spec Figure 5.6-4 for Unit 1 and Figure 5.6-4 for Unit 2 is present in each fuel assembly stored in the new fuel storage racks.

An exemption from the requirements of 10 CFR 70.24, which requires a criticality monitoring system and emergency procedures for the handling and storage of unirradiated fuel, has been granted. The basis for the exemption is that inadvertent or accidental criticality will be precluded through compliance with the Cook Technical Specifications, the geometric spacing of fuel assemblies in the new fuel storage facility and spent fuel storage pool, and administrative controls imposed on fuel handling procedures.

Detailed information is available for use by refueling personnel. These instructions, safety limits and conditions and the design of the fuel handling equipment incorporating built-in interlocks and safety features, provide assurance that no incidents can occur during the refueling operations that could result in a hazard to public health and safety.

An area radiation monitor, centrally located on the north wall of the 650 ft elevation above the spent fuel pool is provided. In addition to the spent fuel pool area monitor, a temporary, portable radiation monitor capable of emitting an audible alarm is mounted on the bridge crane during fuel handling operations. In the containment there are two, independent, redundant, train-oriented radiation monitor sets, each of which is capable of automatically isolating seven containment purge and exhaust valves. That is, the train A device will be capable of isolating the seven inboard purge and exhaust valves and the train B device will be capable of isolating the seven outboard valves. In addition to this, a temporary portable radiation monitor capable of sounding an audible alarm is mounted on the manipulator crane during core alterations. Temperature and redundant water level instruments are provided to guard against the loss of cooling capability.

9.7.2 System Design and Operation

9.7.2.1 System Description

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

In the reactor cavity, fuel is removed from the reactor vessel, transferred through the water and placed in the fuel transfer system by a manipulator crane. In the spent fuel pool, fuel is removed from the transfer system and placed in the poisoned high density storage racks with a long manual tool suspended from an overhead hoist. After a sufficient decay period, the fuel is expected to be removed from storage and loaded into a cask for removal from the site or continued on-site dry storage, unless it is desired to retain them in the spent fuel pool. Up to 3420 fuel assemblies may be stored and still retain capacity to store up to an additional 193 fuel assemblies which corresponds to a complete unloading of one unit. New fuel assemblies are received and eventually transferred to the spent fuel pool or new fuel storage vault for temporary storage or to the reactor core. The new fuel storage vault is sized for storage of the fuel assemblies and other nuclear fuel components normally associated with the replacement of up to 144 assemblies for either or both units. New fuel is loaded into the reactor by either lowering it into the refueling canal from the new fuel storage vault and taking it through the transfer system, by transferring it from the spent fuel pool via the transfer system or by transferring it directly from the receipt canister via the transfer system.

The refueling cavity, refueling canal and spent fuel storage pool are reinforced concrete structures with seam-welded stainless steel plate liners. These Class I structures are designed to withstand the anticipated earthquake loadings and to prevent liner leakage even in the event the reinforced concrete develops cracks.

9.7.2.2 Refueling Operation

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 contain approximately 2,400 ppm boron, or a boron concentration sufficient to ensure that the Keff $\leq$ 0.95, whichever provides more margin to criticality.
2. The water level in the refueling canal is maintained 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.
3. The handling of heavy loads is controlled to reduce the possibility of damage to nuclear fuel and/or equipment that may be required to achieve safe shutdown and continued decay heat removal. This is more fully described in Section 12.2.
4. RVCH Drop Analysis:
5. Reactor Vessel Head drop calculation for each unit is maintained in accordance with applicable design processes and procedures. The calculation provides assurance that public health and safety is maintained. Limitations and assumptions have been incorporated into applicable load handling procedures.

While one unit is being refueled, there are no restrictions on the operation of the other unit. Refueling of one unit does not affect the safety aspects of the other unit.

9.7.2.3 Refueling Procedure

9.7.2.3.1 Preparation

- The following general tasks are required prior to refueling:
- The reactor has been subcritical for at least 120 hours.
- The peak spent fuel pool (SFP) temperature during a full core offload with only one SFP cooling train operating has been verified to be less than or equal to 180°F. For all other scenarios, including a full core offload with both SFP cooling trains operating, the peak SFP temperature must be verified to be less than or equal to 150°F.
- A radiation survey is made. A procedure is followed to checkout the functioning and operability of radiation monitors important to refueling operations. This includes radiation monitors, both in the containment and in the auxiliary building spent fuel ventilation system.
- The reactor missile shields and the control rod drive mechanism (CRDM) seismic restraint are removed.
- The bulkhead sections between the reactor cavity and the refueling cavity are removed.
- CRDM cables and cooling air ducts are disconnected and removed.
- Reactor vessel head insulation and instrument leads are removed.
- The reactor vessel head nuts are loosened with the hydraulic tensioner.
- The reactor vessel head studs are removed.
- The canal drain holes are plugged and the fuel transfer tube flange is removed.
- Checkout of the fuel transfer device and manipulator crane is started.
- Guide studs are installed in three stud holes and the remainder of the stud holes are plugged. Only two of the three studs are required to align the vessel head.
- Install the reactor vessel to cavity seal.
- Final preparation of underwater lights and tools is made. Checkout of manipulator crane and fuel transfer system is completed.

- The reactor vessel head is unseated and raised.
- The lift of the reactor vessel head is stopped at several specified heights to check that:
  - the reactor head is level
  - the head is not binding on the guide studs
  - the protective sleeves for the instrument port seal assemblies are not being lifted.
- At the appropriate reactor vessel head lift height, a check is made that the RCCA drive shafts are clear of the CRDM housings, and are not being lifted with the head. The reactor vessel head is lifted to clear and is taken to its storage pedestal.
- The reactor cavity and refueling canal are flooded with water to the level required for unlatching the RCCA drive shafts.
- The control rod drive shafts are unlatched.
- The reactor vessel internals lifting rig is lowered into position and latched to the support plate.
- The reactor cavity and refueling canal are flooded with water to the level required for refueling.
- The reactor vessel upper internals are lifted out of the vessel and placed in the underwater storage rack.
- The core is now ready for refueling.

9.7.2.3.2 Refueling

Refueling is performed with the manipulator crane, following the general tasks listed below.

- Spent fuel, which is to be discharged, is removed from the core and placed on the fuel transfer conveyor for removal to the spent fuel pool.
- Partially spent fuel is relocated within the core or moved to the spent fuel pool.
- New fuel assemblies and the removed, partially spent fuel assemblies to be used in the upcoming cycle of operation are transferred from the new fuel storage area,

new fuel receipt canister or the spent fuel pool into the refueling canal and are brought through the transfer system and loaded into the core.

- Whenever fuel is added to the reactor core, the subcriticality of the core is verified.

If a transfer of the rod cluster control (RCC) elements between fuel assemblies is required and the reactor core is not completely off-loaded to the spent fuel pool, the assemblies can be taken to the RCC change fixture to exchange the RCC elements from one assembly to another. Should a full core off-load be performed during the refueling, the RCC exchange can be performed in the spent fuel pool with a long handled tool. Such an exchange is required whenever a spent fuel assembly containing RCC elements is removed from the core and whenever a fuel assembly is placed in or taken out of a control position during refueling rearrangements. If the previous core design contained burnable poison rod (BPR) elements, then fuel assemblies with BPR elements are moved to the spent fuel pool where the BPR element is removed using the handling tool, and a thimble plugging device may be inserted to restrict the flow through the guide thimbles.

9.7.2.3.3 Reactor Reassembly

The following general tasks are required following refueling:

- The fuel transfer car is parked and the fuel transfer tube isolation valve is closed.
- The reactor vessel internals package is replaced in the vessel. The reactor vessel internals' lifting rig is removed to storage.
- The control rod drive shafts are relatched to RCC elements.
- The manipulator crane is parked.
- The old seal rings are removed from the reactor vessel head, the grooves cleaned and new rings installed.
- The reactor vessel head is picked up and positioned over the reactor vessel.
- The water level is lowered and the reactor vessel head is lowered.
- The refueling cavity and refueling canal are completely drained and the flange surface is manually cleaned.
- The reactor vessel head is seated.

- The guide studs and the stud hole plugs are removed.
- The head studs are replaced and the head nuts are retorqued.
- The canal drain holes are unplugged and the fuel transfer tube flange is replaced.
- Electrical leads and cooling air ducts are reconnected to the CRDM's.
- Vessel head insulation, CRDM seismic restraints, and instrumentation leads are replaced.
- Remove the reactor vessel to cavity seal.
- Control rod drives are checked.
- The reactor missile shield is picked up with the polar crane and replaced.
- Pre-operational tests are performed.

9.7.2.4 Major Structures Required for Refueling

9.7.2.4.1 Refueling Cavity

The refueling cavity is a reinforced concrete structure that forms a pool above the reactor when it is filled with borated water for refueling.

The cavity is filled so that at least 23 feet of water is maintained over the reactor pressure vessel flange. The radiation at the surface of the water is limited to a level as low as reasonably achievable during those periods when a fuel assembly is transferred over the reactor vessel flange.

The reactor vessel flange is sealed to the reactor cavity by a Preferred Engineering mechanical seal, which prevents leakage of refueling water from the refueling cavity. This seal is installed after reactor cooldown but prior to flooding the refueling cavity for refueling operations.

The floor and sides of the refueling cavity are lined with stainless steel. The refueling cavity has been designed to be within the stress and strain limitations of the ACI Code 318-63, using working stress design criteria for operating conditions, and ultimate strength design criteria for accident conditions. Analysis of the refueling cavity has been made using the AEP FRAME Program. The heat generation rates due to radiation in the primary concrete were calculated by using a point kernel analysis technique. In addition to the reactor core sources, the code considers the capture gamma and inelastic neutron scattering contributions outside the core, and within the concrete.

9.7.2.4.2 Refueling Canal

The refueling canal is a passageway extending from the refueling cavity to the inside surface of the reactor containment. The canal is formed by two concrete shielding walls which extend upward to the same elevation as the reactor cavity. The floor of the canal is at a lower elevation than the reactor cavity to provide the greater depth required for the fuel transfer tipping device and the control cluster changing fixture located in the canal. The transfer tube enters the reactor containment and protrudes through the end of the canal. The canal is a stainless steel lined reinforced concrete structure.

The refueling cavity is large enough to provide storage space for the reactor upper and lower internals, the control cluster drive shafts, and miscellaneous refueling tools.

The refueling cavity and refueling canal are modeled as one unit; as a grid of beams and columns. The static and thermal loads are introduced as input at the node points of the gridwork. Seismic loading is entered using the acceleration responses determined from previous analyses. All stresses in a loading combination are combined algebraically. The seismic stresses are considered to be reversible in sign, so as to give maximum calculated combined stresses. The refueling cavity/refueling canal area is further checked for seismic condition by means of the FRAME Program dynamic routines.

9.7.2.4.3 Spent Fuel Storage Pool

The spent fuel storage pool consists of two portions, the pool proper and the transfer canal, separated by a structural wall. Both the pool and the canal are constructed of reinforced concrete and lined with stainless steel plate. A leak detection system indicates any leakage from the pool.

The transfer canal forms the auxiliary building terminal of the fuel transfer tube. Fuel transported into the transfer canal from either refueling canal is removed from the conveyor car by the new and spent fuel handling crane, passed through a closeable gate in the wall separating the canal from the spent fuel pool, and placed into poisoned high density spent fuel racks for storage. New fuel is either transferred from the new fuel storage area to the conveyer car in the transfer canal, and then to the appropriate refueling cavity for insertion into the reactors, transferred from the new fuel storage area to the spent fuel racks for temporary storage, and then transferred via the conveyer car to the appropriate refueling cavity for insertion into the reactors, or transferred from the new fuel receipt canister to the appropriate refueling cavity for insertion into the reactors via the conveyer car.

The spent fuel pool proper is designed for the underwater storage of spent fuel assemblies, control rod clusters and burnable poisons, thimble plugs and sources (primary and secondary) after their removal from the reactor. Other materials (such as Boral samples, reactor vessel irradiated specimen and highly radioactive materials) are also stored in the spent fuel pool.

Space is provided in the pool for the spent fuel cask which is placed in the pool during loading operations. Control rod clusters, burnable poison rod assemblies, thimble plugs and sources are typically stored in fuel assemblies.

Spent fuel assemblies are handled by a long handled tool suspended from a hoist on the new and spent fuel handling crane, and manipulated by an operator standing on the movable crane bridge. Fuel handling tools are typically stored on hangers along the south wall of the spent fuel pool.

The spent fuel storage racks are erected on the pit floor. The fuel assemblies are placed in vertical cells in the racks, continuously grouped in parallel rows of 8.97 inch centers in both directions. The racks are so designed that it is impossible to place two assemblies any closer than if they were in two adjacent cells, thereby ensuring the necessary spacing between assemblies.

Following the cavity seal failure at the Connecticut Yankee Plant, the possibility of this event was evaluated for the Cook Plant. A review of drain paths was performed. In no instance was the potential for a rapid drain-down identified.

9.7.2.4.4 Poison Cell / Rack Module Design Concept

The layout of the spent fuel storage cells is shown in Figure 9.7-2. Twenty three free-standing poisoned rack modules positioned with a prescribed and geometrically controlled gap between them will contain a total of 3613 storage cells (plus 3 triangle cells located at the SW, NW, and NE corners of the pool). Out of these cells, the peripheral cells located in each rack module are flux-trap cells∗ due to the gap between modules, and the interior ones are of the non-flux trap type. The storage cells suitable for storing fresh fuel (up to 4.95 wt % U-235 nominal enrichment) are uniquely identified (see Figures 9.7-3 and 9.7-4). Consistent with the concept of two region storage, the placement of fuel with a given burnup in the allowable location is administratively controlled. No credit is taken for soluble boron in normal refueling and full

∗ A flux trap construction means that there is a water gap between adjacent storage cells such that the neutrons emanating from a fuel assembly are thermalized before reaching an adjacent fuel assembly.

core off-load storage conditions. The essential cell data for all storage cells is given in Table 9.7-2.

The new spent fuel storage racks are free-standing and self-supporting. The principal construction materials for the new racks are SA240-Type 304 stainless steel sheet and plate stock, and SA564-630 (precipitation hardened stainless steel) for the adjustable support spindles. The only non-stainless material utilized in the rack is the neutron absorber material, which is boron carbide and aluminum-composite sandwich available under the patented product name "Boral".

The new racks are designed and analyzed in accordance with Section III, Division 1, Subsection NF of the ASME Boiler and Pressure Vessel (B&PV) Code. The material procurement, analysis, and fabrication of the rack modules conform to 10 CFR 50 Appendix B requirements.

9.7.2.4.5 Mixed Zone Three Region Storage (MZTR)

The high density spent fuel storage racks in the spent fuel pool will provide storage locations for up to 3613 fuel assemblies and will be designed to maintain the stored fuel, having an initial nominal enrichment of up to 4.95 wt% U-235, in a safe, coolable, and subcritical configuration during normal discharge and full core off-load storage and postulated accident conditions.

All rack modules for the spent fuel pool are of the "free-standing" type such that the modules are not attached to the pool floor nor do they require any lateral braces or restraints. These rack modules were placed in the pool in their designated locations using a specifically designed lifting device, and the support legs were remotely leveled (using a telescopic removable handling tool) by an operator on the fuel handling bridge. The leveling operation was done when the support legs were lifted off the floor. Except for the crane, no additional lifting equipment was needed while leveling was performed.

All modules in the spent fuel pool are of "non-flux trap" construction. However, the module baseplates extend out by 7/8" (nominal), such that the nominal gap between the adjacent walls of two neighboring racks is 2" (nom.). Thus, although there is a single screen of neutron absorber panel between two fuel assemblies stored in the same rack, there are two poison panels with a water flux trap (2" wide) between them for fuel assemblies located in peripheral cells of two facing modules. Out of these flux trap locations, and peripheral cell locations (cells adjacent to pool walls) a certain number of storage cells are designated for storing fresh fuel. In addition, a certain number of interior cells in each rack can be designated for storing fresh fuel of 4.95% wt. U-235 (maximum nominal) enrichment. In this manner, a sufficient number of locations without any burnup restriction (Region I cells) are provided to enable unrestricted full core off-load of the Donald C. Cook Nuclear Plant reactor in the spent fuel pool.

Each rack module is supported by at least four legs, which were remotely adjustable. Thus, the racks can be made vertical and the top of the racks can easily be made co-planar with each other. The rack module support legs are engineered to accommodate variations of the pool floor. The support legs also provide an under rack plenum for natural circulation of water through the storage cells. The placement of the racks in the spent fuel pool has been designed to preclude any support legs from being located over existing obstructions on the pool floor.

The spent fuel pool racks are subjected to mandated seismic loadings per the UFSAR. The Design Basis Earthquake (DBE) and Operating Basis Earthquake (OBE) seismic response spectra were provided and synthetic time histories were generated. These acceleration time histories were applied as inertia loads.

Under these seismic events, the rack modules have four designated locations of potential impact:

1. Support leg to bearing pad
2. Storage cell to fuel assembly contact surfaces
3. Baseplate edges
4. Rack top corners

The support leg to pool slab bearing pad impact would occur whenever the rack support foot lifts off the pool floor during a seismic event. The "rattling" of the fuel assemblies in the storage cell is a natural phenomenon associated with seismic conditions. The baseplate and rack top corners impacts would occur if the rack modules tend to slide or tilt towards each other during the postulated DBE or OBE seismic events.

A bearing pad, made of austenitic stainless steel, is interposed between the support foot and the liner such that the loads transmitted to the slab by the rack module under steady state as well as seismic conditions are diffused into the pool slab, and allowable local concrete surface pressures are not exceeded.

9.7.2.4.6 Material Considerations

Safe storage of nuclear fuel in the spent fuel pool requires that the materials utilized in the fabrication of racks be of proven durability and be compatible with the pool water environment.

This section provides the necessary information on this subject. Material composition for the various materials are summarized in Tables 9.7-3 through 9.7-5.

The following structural materials are utilized in the fabrication of the spent fuel racks:

a. ASME SA240-304 for all sheet metal stock.

b. Internally threaded support legs: ASME SA240-304.

c. Externally threaded support spindle: ASME SA564-630 precipitation hardened stainless steel.

d. Weld material - per the following ASME specification: SFA 5.9 ER308.

In addition to the structural and non-structural stainless material, the racks employ Boral, a patented product of AAR Brooks & Perkins, as the thermal neutron absorber material. A brief description of Boral, and its fuel pool experience list follows. Boral is a thermal neutron absorbing material composed of boron carbide and 1100 alloy aluminum. Boron carbide is a compound having a high boron content in a physically stable and chemical inert form. The 1100 alloy aluminum is a light-weight metal with high tensile strength which is protected from corrosion by a highly resistant oxide film. The two materials, boron carbide and aluminum, are chemically compatible and ideally suited for long-term use in the radiation, thermal and chemical environment of a spent fuel pool.

9.7.2.4.7 Compatibility with Coolant

All materials used in the construction of the spent fuel pool racks have an established history of in-pool usage. Their physical, chemical and radiological compatibility with the pool environment is an established fact at this time. Boral has been used in both vented and unvented configurations in fuel pools with equal success. The spent fuel pool rack construction at Cook Nuclear Plant allows full venting of the Boral space. Austenitic stainless steel (304) is widely used in nuclear power plants.

9.7.2.4.8 Criticality Considerations

The high density spent fuel storage racks are designed to assure that the effective neutron multiplication factor (K_{eff}) is equal to or less than 0.95 with the racks fully loaded with fuel of the highest anticipated reactivity, and flooded with unborated water at the temperature within the operating range corresponding to the highest reactivity. The maximum calculated reactivity includes a margin for uncertainty in reactivity calculations including mechanical tolerances. All uncertainties are statistically combined, such that the final K_{eff} will be equal to or less than 0.95 with a 95% probability at a 95% confidence level.

To assure the true reactivity will always be less than the calculated reactivity, the following conservative assumptions were made:

- Moderator is assumed to be unborated water at a temperature within the operating range that results in the highest reactivity (determined to be 20°C).
- The effective multiplication factor of an infinite radial array of fuel assemblies was used except for the boundary storage cells where leakage is inherent.
- Neutron absorption in minor structural members is neglected, i.e., spacer grids are analytically replaced by water.
- The design basis fuel assembly is a 15 X 15 (Standard) Westinghouse containing UO2 at a maximum initial enrichment of 4.95 + 0.05 wt% U-235 by weight. For fuel assemblies with natural UO2 blankets, the enrichment is that of the central enriched zone. Calculations confirmed that this reference design fuel assembly was the most reactive of the assembly types expected to be stored in the racks. Three separate storage regions are provided in the spent fuel storage pool, with independent criteria defining the highest potential reactivity in each of the regions as follows:
- Region 1 is designed to accommodate new fuel with a maximum enrichment of 4.95 + 0.05 wt% U-235, or spent fuel regardless of the discharge fuel burnup.
- Region 2 is designed to accommodate (high burnup) fuel of 4.95% initial enrichment burned to at least 50,000 MWD/MtU (assembly average), or fuel of other enrichments with a burnup yielding an equivalent reactivity.
- Region 3 is designed to accommodate (intermediate burnup) fuel of 4.95% initial enrichment burned to at least 38,000 MWD/MtU (assembly average), or fuel of other enrichments with a burnup yielding an equivalent reactivity.

The water in the spent fuel storage pool normally contains soluble boron, which would result in large subcriticality margins under actual operating conditions. However, the NRC guidelines, based upon the accident condition in which all soluble poison is assumed to have been lost, specify that the limiting k_{eff} of 0.95 for normal storage be evaluated in the absence of soluble boron. The double contingency principle of ANSI N-16.1-1975 and of the April 1978 NRC letter∗ allows credit for soluble boron under other abnormal or accident conditions since only a single independent accident need be considered at one time. Consequences of abnormal and accident conditions have also been evaluated, where "abnormal" refers to conditions which may reasonably be expected to occur during the lifetime of the plant and "accident" refers to conditions which are not expected to occur but nevertheless must be protected against.

9.7.2.5 Summary of Criticality Analyses

9.7.2.5.1 Normal Operating Conditions

The design basis layout of storage cells for the three regions is shown in Figure 9.7-3. In this configuration, the fresh fuel cells (Region l) are located alternately along the rack periphery (where neutron leakage reduced reactivity) or along the boundary between two storage modules (where the water gap provides a flux-trap, which reduces reactivity). High burnup fuel in Region 2 affords a low-reactivity barrier between fresh fuel assemblies and Region 3 fuel of intermediate burnup.

Prior to approaching the reactor end-of-life, not all storage cells are needed for spent fuel. Therefore, an alternative configuration may be used in which the internal cells are loaded in a checkerboard pattern of fresh fuel (or fuel of any burnup) with empty cells, as indicated in Figure 9.7-4. This configuration is intended primarily to facilitate a full core unload when needed, prior to the time the racks are beginning to fill up.

To provide for continued placement of the new and intermediate burnup fuel in the spent fuel pool as the storage racks approach a full condition, the A1, C1, E1 and A5 storage modules may have either the Mixed Zone Three Region (MZTR) or the interim (checkboard) storage pattern in any combination, while the remaining 19 modules must meet the MZTR requirements.

Figure 9.7-5 defines the acceptable burnup domains and illustrates the limiting burnup for fuel of various initial enrichments for both Region 2 (upper curve) or Region 3 (lower curve), both of which assume that the fresh fuel (Region 1) is enriched to 4.95% U-235. Criticality analyses show that the most reactive configuration occurs along the boundary between modules with the reactivity of the edge configuration being slightly lower∗. The bounding criticality analyses are

∗ USNRC letter of April 14, 1978, to all Power Reactor Licensees - OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications, including modification letter dated January 18, 1979.

∗ The thick base-plate on the rack modules extend beyond the storage cells and provide assurance that the necessary water-gap between modules is maintained.

summarized in Table 9.7-6 for the design basis storage condition (which assumes the single accident condition of the loss of all soluble boron) and the Table 9.7-7 for the interim checkerboard loading arrangement. The calculated maximum reactivity of 0.940 (same for both the normal storage condition and the interim checkerboard arrangement) is within the regulatory limit of a k_{eff} of 0.95. This maximum reactivity includes calculational uncertainties and manufacturing tolerances (95% probability at the 95% confidence level), an allowance for uncertainty in depletion calculations and the evaluated effect of the axial distribution in burnup. Fresh fuel of less than 4.95% enrichment would result in lower reactivities. As cooling time increases in long-term storage, decay of Pu-241 results in a continuous decrease in reactivity, which provides an increasing subcriticality margin with time. No credit is taken for this decrease in reactivity other than to indicate conservatism in the calculations.

For convenience, the minimum (limiting) burnup data in Figure 9.7-5 for unrestricted storage may be described as a function of the initial enrichment, E, in weight percent U-235 by fitted polynomial expressions as follows:

9.7.2.5.2 For Region 2 Storage

Minimum Burnup in NWD/MTU = - 22,670 + 22,220E - 2,260E^2 + 149E^3

9.7.2.5.3 For Region 3 Storage

Minimum Burnup in MWD/MTU = - 26,745 + 18,746E - 1,631E^2 + 98.4E^3

9.7.2.5.4 Abnormal and Accident Conditions

Although credit for the soluble poison normally present in the spent fuel pool water is permitted under abnormal or accident conditions, most abnormal or accident conditions will not result in exceeding the limiting reactivity (K_{eff} of 0.95) even in the absence of soluble poison. The effects on reactivity of credible abnormal and accident conditions due to temperature increase, boiling, assembly dropped on top of a rack, lateral rack module movement and misplacement of a fuel assembly have been analyzed. Of these abnormal or accident conditions, only one has the potential for a more than negligible positive reactivity effect.

The inadvertent misplacement of a fresh fuel assembly has the potential for exceeding the limiting reactivity, should there be a concurrent and independent accident condition resulting in the loss of all soluble poison. Administrative procedures to assure the presence of soluble poison during fuel handling operations will preclude the possibility of the simultaneous occurrence of the two independent accident conditions. The largest reactivity increase would occur if a new fuel assembly with a maximum enrichment of 5.00% U^{235} were to be positioned in an empty location not in accordance with the interim checkerboard pattern arrangement with the remainder of the module properly loaded with fuel of the highest permissible reactivity. Under this accident condition, credit for the presence of soluble poison is permitted by NRC guidelines*, and calculations indicate that the maximum k_{eff} (0.9295) under this accident scenario is significantly less than 0.95 with 800 ppm of soluble boron credited.

9.7.2.5.5 Thermal Considerations

A flow path for natural convection cooling of spent fuel assemblies is provided by a large hole in the module base at each fuel storage cell position. Additional cooling between cells is provided by smaller holes in the module base between cells and holes near the top of the side plate diaphragms. Additional area for flow to the bottom of the module is provided by space at the edge of the pool.

In summary, the high density (poison) spent fuel module design provides a significant increase in storage over the original non-poison spent fuel module design. The module is designed to meet all technical requirements for structural integrity, criticality, and cooling.

9.7.2.5.6 Cask Decontamination Facilities

Once the spent fuel cask has been loaded, it would be removed from the spent fuel pool and placed on a pad just beyond the pool for decontamination prior to onsite storage or shipment offsite.

The pad has a stainless steel lined base and a curb is provided around it to prevent the water and solvents used during decontamination from spreading over the auxiliary building floor. Drains in the floor of the pad remove the decontaminants to the waste disposal system for processing.

9.7.2.5.7 New Fuel Storage

New fuel assemblies and new control rod clusters are stored in a separate area, adjacent to the spent fuel pool, whose location facilitates the unloading of new fuel assemblies from delivery trucks. This storage vault is designed to hold new assemblies in specially constructed racks. A total of 144 storage positions are provided. Prior to initial core loading for Unit 1 assemblies in

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* Double contingency principle of ANSI N16.1-1975, as specified in the April 14, 1978, NRC letter (Section 1.2) and implied in the proposed revision to Reg. Guide 1.13 (Section 1.4, Appendix A).

excess of the number, which could be accommodated in the new fuel storage area, were stored in the dry spent fuel pool. For Unit 2, temporary storage facilities were established adjacent to the new fuel storage area.

Use of the new fuel storage vault is not required. New fuel may be loaded directly into the new fuel elevator from the new fuel shipping canister(s) if desired, for temporary storage in the spent fuel pool or for direct transfer to the appropriate refueling cavity for insertion into the reactor.

9.7.2.6 Major Equipment Required for Refueling

9.7.2.6.1 Reactor Vessel Stud Tensioner

Stud tensioners are used to make up the head closure joint.

The stud tensioner is a hydraulically operated (oil is the working fluid) device provided to permit preloading and unloading of the reactor vessel closure studs at cold shutdown conditions. Stud tensioners were chosen in order to minimize the time required for the tensioning or unloading operations. Three tensioners are provided and they are applied simultaneously to three studs 120° apart. However, procedures exist that allow use of only two tensioners 180° apart, if necessary. The studs are tensioned to their operational load in a fashion so as to prevent high stresses in the flange region and unequal loadings in the studs. An overstroke alarm is provided on each tensioner to alert the operator that a tensioner is about to reach maximum stroke. Charts indicating the stud elongation and load for a given oil pressure are included in the transient operating instructions. In addition, measurements of the elongation of the studs are performed after tensioning.

9.7.2.6.2 Reactor Vessel Head Lifting Device

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

9.7.2.6.3 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 manually bolted to the support plate by three bolts. Bushings on the frame engage guide studs in the vessel flange to provide closure guidance during removal and replacement of the internals package.

9.7.2.6.4 Manipulator Crane

The manipulator crane is a rectilinear bridge and trolley crane with a vertical mast extending down into the refueling water. The bridge spans the refueling cavity and refueling canal and runs on rails set into the floor along the edge of the cavity and canal. 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 from 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 into the mast tube. The fuel, while inside the mast tube, is transported to its new position.

While being transported, the fuel assembly may be tested for fuel leaks using an in-mast fuel sipping system. This system detects an increase in radioactive gases found at the top of the manipulator crane mast when leaking fuel is present. Small O.D. stainless steel tubing and nozzles were installed on the outside of the manipulator crane mast to facilitate transport of air to the bottom of the mast. Covers were added to the mast roller assemblies to minimize air leakage and crossflows during lateral movements of the mast, trolley, and crane. A cover was added to the top of the mast that creates a sealed environment in the top inner portion of the mast such that a small amount of gases/air present may be siphoned to the in-mast fuel sipping system detector. The detector, electronics, and air pump console of the in-mast fuel sipping system are typically installed prior to and removed after their use. These systems typically do not remain installed during reactor operation, although this activity is not precluded. There is no interferences caused by either the permanently or temporarily installed portions of the in-mast fuel sipping system to the operation of any limit switches. The presence of air bubbles in the mast with a fuel assembly was evaluated from thermohydraulic and criticality standpoints and found to be within acceptable limits.

All controls for the manipulator crane are mounted on a console on the trolley. The bridge and trolley are positioned on a coordinate system. Indications of bridge and trolley position are provided on the console. The drives for the bridge, trolley and winch are variable speed, with speed zones controlled by the Programmable Logic Controller (PLC). Electrical interlocks and limit switches on the bridge and trolley drives protect the equipment. In an emergency, the bridge, trolley and winch can be operated manually using hand-wheels on the motor shafts.

The suspended weight on the gripper tool is monitored by an electrical load cell indicator on the control console. A load in excess of 150 lbs. of the weight of a fuel assembly in water stops the winch drive from moving in the up direction. The 150 lbs. limit may be increased up to 200 lbs. in the event there is evidence of spurious trips resulting from load variations due to directional changes and/or cable and hose reel tensions. The gripper is interlocked through a weight sensing device and also a mechanical spring lock so that it cannot be opened when supporting a fuel assembly. Boundary zone values are compared to position encoder values that limit the normal operating area for the crane. The purpose is to prevent collision of the mast with the vessel guide studs, upper internals and the canal walls.

In addition to the travel limit switches on the bridge and trolley drives, the following safety features are incorporated in the system:

a. Bridge and trolley motion is interlocked with hoist operation to prevent operation of the hoist during bridge or trolley motion. The purpose of this interlock is to prevent simultaneous motion of the fuel assembly in the horizontal and vertical axis.

b. Bridge and trolley main motor drive operation is possible only when both the GRIPPER TUBE UP encoder position indicates that the gripper tube is fully retracted into the mast.

c. In addition, a speed control zone limits the speed and distance an extended assembly can move on the bridge and trolley axis.

d. With the gripper engaged, a solenoid valve in the air line which is utilized to disengage the gripper will not be activated if the load cell readout registers a weight greater than 1200 pounds.

e. The hoist drive circuit in the up direction is opened when the EXCESSIVE SUSPENDED WEIGHT switch is actuated. This switch is actuated at a reading of 150 lbs. in excess of fuel assembly weight. The purpose is to limit the pull that the hoist will put on a fuel assembly.

f. The hoist drive circuit in the up direction is operated only when either the OPEN or CLOSED indicating switch on the gripper is actuated. This interlock will prevent the lifting of a fuel assembly with the gripper only partially engaged.

g. Bridge and trolley drives are interlocked in the direction of the transfer system so that the bridge is prevented from traveling beyond the core area unless the trolley is aligned with the refueling canal centerline. A boundary breach requires use of an Override mode to enable a return within the boundary at reduced speed.

h. Travel of the hoist is prevented when loss of weight (approximately 150 pounds) occurs in any position other than gripper tube down in the core and transfer area. The 150 pound limit may be increased up to 200 pounds in the event there is evidence of spurious trips resulting from load variations due to directional changes and / or cable and hose reel tensions.

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 during the design basis earthquake.

Each manipulator crane used for movement of fuel assemblies within the reactor pressure vessel shall be demonstrated to be operable within 100 hrs prior to the start of such operation by performing a load test of at least 3250 pounds, and by demonstrating an automatic cut off when the crane load exceeds 2850 pounds.

The operability requirements for the manipulator cranes ensure that:

1. manipulator cranes will be used for movement of control rods and fuel assemblies
2. each crane has sufficient load capacity to lift a control rod or fuel assembly and
3. the core internals and pressure vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations.

Each auxiliary hoist and associated load indicator used for movement of control rods within the reactor pressure vessel shall be demonstrated operable within 100 hours prior to the start of such operations by performing a load test of at least 700 pounds.

During movement of control rods or fuel assemblies within the reactor pressure vessel, the manipulator crane and auxiliary hoist shall be determined operable with:

A. The manipulator crane used for movement of fuel assemblies having:

1. A minimum capacity of 3250 pounds, and
2. An overload cut off limit ≤ 2850 pounds.

B. The auxiliary hoist used for movement of control rods having:

1. A minimum capacity of 700 pounds, and
2. A load indicator which shall be used to prevent lifting loads in excess of 600 pounds.

If the above requirements are not satisfied suspend use of any inoperable manipulator crane and/or auxiliary hoist from operations involving the movement of control rods or fuel assemblies within the reactor pressure vessel.

9.7.2.6.5 New and Spent Fuel Handling Crane

The new and spent fuel-handling crane is a bridge with monorails on the overhead structure. The bridge consists of two spans which span the width of the fuel storage area (consisting of the spent fuel pool and the new fuel storage area) and travels its entire length. Electric monorail hoists are provided to travel the width of the fuel storage area. This crane was procured and installed in accordance with Seismic Class I requirements.

The fuel assemblies are moved within the spent fuel pool by means of a long handled tool suspended from the hoist. The hoist travel and tool length are designed to limit the maximum lift of a fuel assembly to a safe shielding depth. The maximum lift of a spent fuel assembly is also limited to a height of 15 inches above the spent fuel racks by means of a programmed normal up limit in the controls. During dry cask operations, this limit will remain active whenever spent fuel assemblies are carried over the storage racks. The limit will be bypassed in the cask loading area in order to lift fuel assemblies to a programmed cask up limit that will clear the top of the transfer cask. When moving new fuel assemblies, a shorter tool may be used since there are no shielding requirements.

The New and Spent Fuel Handing crane also includes a mechanical (passive) mechanism to ensure that the fuel handling tools will not drop into the Spent Fuel Pool.

9.7.2.6.6 Auxiliary Building Cranes

The auxiliary building is equipped with two, fully motorized, single-failure-proof overhead bridge cranes* that run on the same set of runway rails which extend the length of the building. The east crane is equipped with a 150-ton capacity main hoist and 20-ton capacity auxiliary hoist that are mounted on separated trolleys and are completely independent in design and operation (i.e. the auxiliary hoist has its own spent fuel pool travel limit switches). The main hoist/trolley is positioned north of the auxiliary hoist/trolley, and both trolleys move along the top of the crane bridge. In addition, the east crane is equipped with a 2,500 lb. capacity fully electric hoist that runs beneath a monorail, which is cantilevered off the east side of the idler girder. The east crane is operated by a radio remote control of the operator console mounted on the crane walkway. The west crane is equipped with a 150-ton capacity hoist mounted on a trolley that moves along the top of the crane bridge. The west crane is operated by means of a radio remote control or a pendant control.

Upper and lower geared hoist limit switches are provided on both cranes in addition to an upper weight type limit switch. Crane power cutout switches for emergency use have been provided at three separate locations. These switches enable the crane to be stopped, if necessary, by personnel other than the crane operator. The auxiliary building cranes are utilized to handle various items including the radiation protection shields, plant equipment, fuel assemblies, and containers of low-level radioactive waste. The east auxiliary building crane will be utilized to handle the spent fuel cask.

With fuel assemblies in the storage pool, loads in excess of 2,500 pounds shall be prohibited from travel over stored fuel assemblies in the storage pool. Loads carried over the spent fuel pool racks and the heights at which they may be carried over spent fuel racks containing fuel shall be limited in such a way as to preclude impact energies over 55,800 in.-lbs., if the loads are dropped from the crane.

∗ NUREG 0554, "Single-Failure Proof Cranes for Nuclear Power Plant, " specifies that a protective system for load hangups be provided for single-failure proof cranes. However, the overload system is not required by the Technical Specifications or Technical Specification Bases. It is permissible to operate the east auxiliary building crane main hoist with the overload protection circuitry defeated since the interlocks that prevent crane travel with loads over the spent fuel pool will not be impacted and since it still would be possible to safely shut down both units in case of a load drop.

The restriction on movement of loads in excess of 2,500 lbs. over stored fuel assemblies in the storage pool ensures that, in the event of a dropped load,

1. the activity release will be limited to that contained in a single fuel assembly, and
2. any possible distortion of fuel in the storage racks will not result in a critical array.

The 2,500 lb. load restriction is based on the combined nominal weight of a fuel assembly, a control rod assembly, and an associated fuel handling tool. Release of activity from a single fuel assembly is consistent with the assumption for the analysis for a fuel handling accident.

The restriction on movements of loads in excess of the impact energy limit, which is based on the kinetic energy of a dropped fuel assembly and control rod assembly weighing an average of 1550 lbs. (dry weight) from a height of 36" above the fuel storage rack, is to bound other loads.

To assure the above loading restrictions:

1. Crane interlocks which prevent crane travel with loads in excess of 2,500 pounds over stored fuel assemblies shall be demonstrated to be operable within 7 days prior to crane use, and at least once per seven days thereafter during crane operation. A procedure for bypassing interlocks with loads less than 2,500 lbs. is in place for those occasions when fuel assemblies are being moved over the spent fuel racks. Separately, crane interlocks were bypassed during the movement of steam generator sections in the auxiliary building for the Unit 1 steam generator replacement project at which time administrative controls were in place to prevent loads from passing over the spent fuel pool.
2. The potential impact energy due to dropping the crane's load is limited in such a way to be less than 55,800 in-lbs. prior to moving such loads over spent fuel racks containing fuel. Prohibiting loads greater than 2,500 pounds or loads at heights that would exceed the kinetic energy impact limit allows flexibility in the movements of fuel and other relatively light loads, while providing reasonable assurance that the consequences of a fuel handling accident will not be exceeded.

The main hoist load block of either auxiliary building crane and the auxiliary hoist load block of the east crane may be moved over the spent fuel pool if no load is being carried.

For the Dry Cask Storage Project, a keyed bypass was installed on the East Auxiliary Building Crane. When in bypass mode, this crane can be moved over the southeast comer of the Spent

Fuel Pool as long as the crane maintains its single failure-proof design status and the lifts are performed in accordance with the Control of Heavy Loads Program requirements described in Section 12.2.1. The crane can also be used to move a new fuel assembly to the "new fuel" elevator and over the transfer canal.

9.7.2.6.7 Polar Cranes

Each of the polar cranes is a 250/35 ton overhead crane running on circumferential rails in the containment buildings. The capacity of the main hoist is 250 tons and the capacity of the auxiliary hoist is 35 tons. The crane is normally operated by means of a radio control system, but capability for operating by means of pendant control is also provided. Upper and lower geared hoist limit switches are provided in addition to an upper paddle type limit switch. Crane power cutout switches for emergency use have been provided at two separate locations in each unit. There, switches enable the cranes to be stopped, if necessary, by personnel other than the crane operator. The polar cranes are utilized to handle equipment such as the reactor vessel head, upper internals, lower internals, missile shields, bulkhead sections, and special tools.

9.7.2.6.8 New Fuel Elevator

The new fuel elevator lowers new fuel assemblies into the transfer canal alongside the spent fuel pool so that the assemblies can be positioned in the upender for transfer into the reactor side of the canal. Administrative controls prevent spent fuel from being raised above the recommended safe water level.

For the Dry Cask Storage Project, a keyed bypass was installed on the East Auxiliary Building Crane. When in bypass mode, the crane can be moved over the "new fuel" elevator and the transfer canal.

9.7.2.6.9 Fuel Transfer System

The Fuel Transfer System, shown in Figure 9.7-1, is an underwater winch-powered cable driven conveyor car that runs on tracks extending from the refueling canal through the transfer tube and into the transfer canal. The conveyor car receives a fuel assembly in the vertical position from the manipulator crane or the new and spent fuel-handling crane. The fuel assembly is lowered to a horizontal position for passage through the tube, and then again raised to a vertical position at the opposite end of conveyor car travel.

During plant operation, the conveyor car is stored in the transfer canal. A blind flange is bolted on the transfer tube to seal the reactor containment, after transfer operations.

9.7.2.6.10 Rod Cluster Control Changing Fixture

A fixture is mounted on the refueling canal wall for removing Rod Cluster Control (RCC) elements from fuel assemblies and inserting them into other 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 the fuel assemblies and positioning fuel assemblies 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, another 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.

9.7.3 Design Evaluation

Incident Control

Gamma radiation levels in the containment and fuel storage areas are continuously monitored. These monitors, described in Section 9.7.1, provide an audible alarm at the initiating detector, indicating an unsafe condition (See Chapter 14.2.1.1 and 14.2.1.2 for details). Continuous monitoring of reactor neutron flux provides immediate indication in the control room of an abnormal core flux level.

Direct communication between the control room and the refueling cavity manipulator crane is available whenever changes in core geometry are taking place.

This provision allows the control room operator to inform the manipulator crane operator of any impending unsafe condition detected from the main control board indicators during fuel movement.

Malfunction Analysis

The analysis presented in Chapter 14 evaluates environmental consequences of a fuel-handling incident.

9.7.4 Tests and Inspection

Prior to initial fueling, preoperational checks of the fuel handling equipment were performed to ensure proper performance of the fuel handling equipment and to familiarize plant operating personnel and contract personnel with operation of the equipment.

Prior to subsequent refueling operations, the equipment is inspected for operating condition. Certain components, such as the fuel transfer car and manipulator crane, are operated and interlocks checked to ensure reliable performance prior to moving irradiated fuel.

9.0 AUXILIARY AND EMERGENCY SYSTEMS

9.8 FACILITY SERVICE SYSTEMS

The Facility Service Systems consist of the Fire Protection Systems, the Service Water System, and the Compressed Air System.

9.8.1 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. Donald C. Cook Nuclear Power Plant (CNP) 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 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 October 24, 2013 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).

9.8.1.1 Design Basis Summary

9.8.1.1.1 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.

9.8.1.1.2 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.

9.8.1.1.3 Codes of Record

The codes, standards and guidelines used for the design and installation of plant fire protection systems are as follows: (for specific applications and evaluations of codes refer to Engineering Equivalency Evaluation 14.1.1, “NFPA Code Conformance Report”).

- NFPA 10 – 1984, “Standard for Portable Fire Extinguishers”
- NFPA 12 – 1968, 2005, “Standard on Carbon Dioxide Extinguishing Systems”
- NFPA 12A – 1977, “Standard on Halon 1301 Fire Extinguishing Systems”
- NFPA 13 – 1971, 1983, 1991, “Standard for Installation of Sprinkler Systems”
- NFPA 14 – 1971, 1978, 1986, 1990, “Standard for the Installation of Standpipe, Private Hydrant and Hose Systems”
- NFPA 15 – 1973, “Standard for Water Spray Fixed Systems for Fire Protection”
- NFPA 20 – 1990, “Standard for the Installation of Stationary Pumps for Fire Protection”
- NFPA 22 – 1987, “Standard for Water Tanks for Private Fire Protection”
- NFPA 24 – 1987, “Standard for the Installation of Private Fire Service Mains and their Appurtenances”
- NFPA 30 – 1987, “Flammable and Combustible Liquids Code”
- NFPA 50A – 1999, “Standard for Gaseous Hydrogen Systems at Consumer Site”
- NFPA 51B – 1971, “Standard for Fire Prevention During Welding, Cutting, and Other Hot Work”
- NFPA 72D – 1967, 1979, “Installation, Maintenance, and Use of Proprietary Protective Signaling Systems”
- NFPA 72E – 1974, 1978, 1982, 1984, “Automatic Fire Detectors”
- NFPA 80 – 1970, “Standard for Fire Doors and Fire Windows”
- NFPA 80A – 1996, “Recommended Practice for Protection of Buildings for Exterior Fire Exposures”
- NFPA 90A – 1978, “Standard for the Installation of Air-Conditioning and Ventilation Systems”
- NFPA 101 – Current Edition, “Life Safety Code”
- NFPA 220 – 1999, “Standard on Types of Building Construction”

- NFPA 241 – 2000, “Standard for Safeguarding Construction, Alterations, and Demolition Operations”
- NFPA 251 – 1999, “Standard Methods of Fire Tests of Fire Endurance of Building Construction and Materials”
- NFPA 256 – 1998, “Standard Methods of Fire Tests of Roof Coverings”
- NFPA 600 – 2000, “Standard on Industrial Fire Brigades”
- NFPA 701 – 1999, “Standard Methods of Fire Tests for Flame Propagation of Textiles and Films”

9.8.1.2 System Description

9.8.1.2.1 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 systems equipment and cables required for the nuclear safety capability assessment are contained in the “Nuclear Safety Capability Assessment” (NSCA).

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 documented in the “NFPA 805 Fire Protection Program Manual” (NFPPM).

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 “D. C. Cook Fire Safety Analysis” (FSA).

The Fire Protection System is shown in Figures 9.8-1 and 9.8-2.

Radioactive Release

Structures, systems, and components relied upon to meet the radioactive release criteria are documented in the “NFPA 805 Fire Protection Program Manual” (NFPPM).

9.8.1.2.2 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 Table 9.8-1 are considered to be part of the "power block" as identified in the NRC NFPA 805 Safety Evaluation, dated October 24, 2013.

9.8.1.3 Safety Evaluation

The "D. C. Cook Fire Safety Analysis" (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:

- 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.
- Summary of the Nuclear Safety Capability Assessment (at power and non-power) compliance strategies.
  - Deterministic compliance strategies
  - Performance-based compliance strategies (including defense-in-depth and safety margin)
- Summary of the Non-Power Operations Modes compliance strategies.
- Summary of the Radioactive Release compliance strategies.
- Summary of the Fire Probabilistic Risk Assessments.
- Key analysis assumptions to be included in the NFPA 805 monitoring program.

9.8.1.4 Fire Protection Documentation, Configuration Control and Quality Assurance

In accordance with Chapter 3 of NFPA 805 a fire protection plan documented in Procedure PMI-2270, "Fire Protection Program," defines the management policy and program direction and defines the responsibilities of those individuals responsible for the plan's implementation. Procedure PMI-2270:

- Designates the senior management position with immediate authority and responsibility for the fire protection program.
- Designates a position responsible for the daily administration and coordination of the fire protection program and its implementation.
- Defines the fire protection interfaces with other organizations and assigns responsibilities for the coordination of activities. In addition, Procedure PMI-2270, "Fire Protection Program," identifies the various plant positions having the authority for implementing the various areas of the fire protection program.
- Identifies the appropriate authority having jurisdiction for the various areas of the fire protection program.
- Identifies the procedures established for the implementation of the fire protection program, including the post-transition change process and the fire protection monitoring program.
- Identifies the qualifications required for various fire protection program personnel.
- Identifies the quality requirements of Chapter 2 of NFPA 805.

Detailed compliance with the programmatic requirements of Chapters 2 and 3 of NFPA 805 is contained in the "NFPA 805 Fire Protection Program Manual" (NFPPM).

9.8.2 Compressed Air System

The Compressed Air System is shown on Figure 9.8-3.

9.8.2.1 Design Bases

Parameters included in design:

1. The system must provide redundant compressed air supplies for control and instrument air requirements.
2. The system must provide adequate compressed air capacity for:
   a. General Plant Service
   b. Control
   c. Instrumentation
   d. Testing
   e. Containment Penetration and Weld Channel Pressurization System
   f. Respiratory protection in the containment structure itself, as per compressed gas association commodity Spec. G-7.1 - 1966, per OSHA Standards and Interpretations 1910.134.
3. The system must provide a continuous supply of compressed air to vital systems under both normal and abnormal conditions.

9.8.2.2 System Description

The Compressed Air System includes the combined service and control instrument air sub-systems, the air supply for the Containment Penetration and Weld Channel Pressurization System and air respiratory protection at strategic location. Either of the two full capacity plant air compressors, one located in the turbine building of each unit, is capable of supplying compressed air to the plant air receivers for general service air for both units. In addition, the plant air compressors supply air to the dry control-instrument air receivers through redundant pre- and after-filters and dryers.

The Containment Penetration and Weld Channel Pressurization System air receivers are supplied from these dry control-instrument air receivers.

In addition, a standby control air compressor capable of supplying the control and instrument air for its unit is installed as a backup for the normal control-instrument air supply, i.e., the plant air compressors. This standby compressor is also capable of supplying air to the Containment Penetration and Weld Channel Pressurization System for its unit. The standby control air compressor is designed to start automatically upon detection of low air pressure in the plant air header.

The four air compressors (two plant, two control) are of the oil-free type to eliminate oil contamination of the control instrument air. Control and instrument air is also filtered and dried to remove any particulate matter and/or moisture which could interfere with the operation of any instrumentation and control equipment.

The Control Air System includes sufficient capacity to supply the control and instrument air requirements with the equivalent of approximately 5 minutes of control air output after a loss of power incident. Additionally, certain vital control valves within the containment are each equipped with a local receiver tank with capacity to activate the valve. Also, the control air compressors can be supplied with electric power from both normal and emergency sources so that a supply of compressed air can be made available in any foreseeable circumstance.

The Compressed Air System includes normal accessory equipment such as dryers, filters, storage receivers, after-coolers, and safety valves in addition to the compressors. A descriptive summary of the major pieces of equipment in the system is included in Table 9.8-2.

In addition, a 650CFM backup plant air compressor is installed on the Unit # 1 4KV Switchgear Room roof. The backup plant air compressor is tied into the normal plant compressed air system at both the inlet and outlet of the Containment Test Pressurization Air Dryer and can supply any service air or control instrument air load. The compressor must be manually started and aligned. The 575 V, 150 Amp power supply is provided from 600V Bus 11CMC.

9.8.2.3 Design Evaluation

The Compressed Air System is designed to provide a reliable source of compressed air for all plant uses.

During normal operation, either one of the two plant air compressors is capable of supplying the entire demand of both plant and control-instrument air requirements for both units.

Low plant air header pressure will automatically start the second plant air compressor. The air compressor is then manually loaded and placed in automatic pressure control. A lower control air header pressure in either unit will automatically start that unit's control air compressor. A further degradation in the plant air header pressure will cause the four air-operated isolation valves located in the plant air ring header to close, thus completely isolating the control air systems of the two units.

This system arrangement allows either unit's plant air system to be removed from service should that become necessary while allowing the remainder of the plant air system as well as both unit's control air system to continue in operation. This isolation can be achieved by closing the two air-operated isolation valves, which serve the effected unit.

In this manner, each unit still retains a backup supply of compressed air from its own control air compressor.

A failure in the control air system of one unit will not affect the control air system of the other unit because check valves in the control air off-takes from the plant air header prevent back flow.

Each control air header has safety relief valves to protect against over-pressurization.

9.8.2.4 Tests and Inspections

The compressed air systems is in service during all modes of operation. Flow and pressure instrumentation for the plant air system, and the control air system permit monitoring the systems for excessive air consumption, or inadequate compressor capacity.

Tests and inspections include:

- Routine functional testing of the standby plant and control air compressors, to ensure their readiness.
- Routine compressor inspections /overhauls.
- Routine prefilter, dryer desiccant, and after filter inspection/replacement.
- Routine air quality monitoring for moisture, and breathing air quality.
-

9.8.3 Service Water Systems

The Service Water Systems are shared by both units.

9.8.3.1 Design Basis

The Service Water Systems supply cooling water to various heat exchangers in both the primary and secondary systems of each unit. Provisions are made to ensure a continuous flow of cooling water to those systems and components necessary for plant safety both during normal operation or under accident conditions. Sufficient redundancy of piping and components is provided to insure that cooling is maintained to vital services at all times.

9.8.3.2 Description

Service water is provided by two independent systems, the Non-Essential Service Water System shown in Figures 9.8-4, 9.8-5 and 9.8-6 and the Essential Service Water System shown in Figure 9.8-7. Each system consists of four operational pumps, each with a duplex automatic backwashing strainer in its discharge line, and associated piping and valves. The design parameters of these components are listed in Table 9.8-3.

Non-Essential Service Water System

The Non-Essential Service Water System (NESW) supplies cooling water to the following SSCs. Turbine oil coolers, air compressors, chilled water subsystem and miscellaneous services, none of which are required for plant safety related functions. The chilled water subsystem provides cooling to the Upper and Lower Containment Ventilation Units, Instrument Room Ventilation Units, and Reactor Coolant Pump Motor Air Coolers. The NESW also provides an alternate source of water to equipment served by the Miscellaneous Sealing and Cooling Water (MSCW) System when the MSCW System is being serviced and/or operational. Cooling requirements are given in Table 9.8-4, Table 9.8-4A, and Table 9.8-4B. The number of main pumps normally operated to provide service water to the two units is dependent upon system flow demands. Typically, 2 or 3 main pumps are in service with at least one pump held in standby. All main pumps are able to take suction from either the Unit 1 or Unit 2 Circulating Water Intake Tunnels or discharge tunnels. The system discharges into either the Unit 1 or Unit 2 Circulating Water Discharge Tunnels. Thus, Non-Essential Service Water supply to both units is assured, even if the tunnels of one unit are out of service.

Following a loss of all off-site power, the non-essential service water main pumps are automatically started in the proper sequence as soon as the emergency diesel generator power becomes available. Under those conditions, the main pumps are primarily used to supply cooling water to the control air compressors in order to restore control air service. All motor-operated valves on the main non-essential water system are operated from the station battery system with the exception of the motor operated strainer discharge motor operators for each of the NESW automatic strainers, which are powered from the 600 VAC auxiliary bus. Crossties between the main pumps permit any one main pump to supply the initial requirements for both units upon loss of off-site power (LOOP).

The automatic discharge strainers of the pumps are provided with automatic backwashing. Each strainer contains an internal arm that cleans a segment of the strainer during backwash cycles. When the strainer is in service and if it becomes dirty or clogged, a high differential pressure signal initiates rotation of the internal arm that directs a small portion of the NESW flow into reverse flow across a basket segment, and directs the reverse flow and debris to a drain, maintaining the strainer in service during the backwash cycle.

Containment Chilled Water Subsystem

The Containment Chilled Water Subsystem consists of a closed-loop chilled water system and an open-loop condenser cooling system. The system consists of three water-cooled chillers per unit. Cooling water will be circulated through the Upper and Lower Containment Ventilation Units, Instrument Room Ventilation Units, Reactor Coolant Pump Motor Air Coolers, and through part of the new cooling system creating a closed loop. NESW will flow from the intake tunnel through the new cooling system and into the discharge tunnel, creating an open loop. Heat will be transferred from the closed loop via a chiller (evaporator and condenser) and/or a heat exchanger to the open loop NESW.

Alternately, direct cooling of the containment air handling units via the NESW can still be implemented if the chilled water subsystem is unable to operate.

Essential Service Water System

The Essential Service Water (ESW) System supplies cooling water to the following components:

a. Component Cooling Heat Exchangers
b. Containment Spray Heat Exchangers
c. Emergency Diesel Generators
d. Auxiliary Feedwater System
e. Control Room Air Conditioners
f. Auxiliary Feedwater Pump Enclosure Coolers

During normal operations essential service water is supplied continuously to the Component Cooling Heat Exchangers and the Control Room Air Conditioners. The Containment Spray Heat Exchangers are normally supplied only when the system is in operation. The Emergency Diesel Generators are automatically supplied with Essential Service Water when diesel generator(s) are in operation; however, Essential Service Water may be manually aligned to a diesel generator to help in establishing ESW minimum flow requirements or to provide flow for prevention of microbiological growth in the ESW supply piping. In addition, the essential service water system serves as back-up water sources to the auxiliary feedwater pumps for use when the condensate storage tank, the normal supply for the auxiliary feed-water system, is either empty or otherwise lost as a source of supply.

The system consists of four essential service water pumps, four duplex strainers and associated piping and valves. System piping is arranged in two independent headers, each serving certain components in each unit as follows:

a. Each essential service water header supplies cooling water to one of the two Containment Spray Heat Exchangers associated with each unit.

b. The heat exchangers for the two diesel-generator sets on each unit are served by the respective essential service water header on that unit. Remotely-operated valves are provided to supply each diesel with ESW from the redundant header for beyond-design-basis events.

c. Each essential service water header supplies cooling water to one of the two Component Cooling Heat Exchangers associated with each unit.

d. In each unit one essential service water system provides the source of feedwater for the turbine-driven auxiliary feedwater pump and the other to both motor-driven auxiliary feed pumps.

e. Each essential service water header supplies cooling water to one of the two Control Room Air Conditioners associated with each unit.

f. Each essential service water header supplies cooling water to two of the Auxiliary Feedwater Pump Enclosure Coolers.

The two headers are arranged such that a rupture in either header will not jeopardize the safety functions of the system. Each header is served by two essential service water pumps. Two pumps are sufficient to supply all service water requirements for unit operation, shutdown, refueling or post-accident operation, including a LOCA on one unit and a simultaneous hot shutdown in the other. System conditions will dictate when a third pump is required such as when CCW system heat load is high due to RHR system operation. All pumps receive a start signal in the event of an accident.

The pump discharge strainer backwash valves are provided with a backup control air source to support strainer backwash should the normal control air source become unavailable.

Since the thermal load on the Component Cooling Water Heat Exchangers is reduced after a safety injection signal, the Essential Service Water flow to these heat exchangers is automatically reduced to insure adequate flow to the Containment Spray Heat Exchangers if needed. Flow is automatically supplied to the Containment Spray Heat Exchangers during the recirculation mode if a containment spray signal has been initiated. Upon receipt of a Phase B isolation signal, full ESW design flow is established to both Containment Spray Heat Exchangers. The header and valving arrangement insures adequate service water flow under all normal and emergency conditions.

Table 9.8-5 contains data for ESW cooling flows under normal operation, cooldown, LOCA Injection and LOCA Recirculation conditions.

Normal Operation & Cooldown

The values in Table 9.8-5 for Normal Operation and Cooldown represent nominal flow conditions required to support normal operation of the plant within the current licensing basis. Values for component flows used in the plant are adjusted to include uncertainty and will depend upon the actual operating condition.

LOCA Injection & LOCA Recirculation

The Table 9.8-5 data for LOCA injection and recirculation is the minimum ESW cooling flow required for accident response and mitigation, using plant emergency operating procedures. During LOCA Injection, there is minimal heat load on the CCW system. The CCW flow is provided to the RHR heat exchanger during injection to provide a minimum flow path for the CCW pump. There is no RHR heat load on CCW during LOCA injection.

The values in Table 9.8-5 are the minimum flows to systems, required to meet the accident mitigation strategy in the accident analyses supporting the current licensing basis. Values for component flows used in the plant are adjusted to include uncertainty.

The Essential Service Water Pumps take suction from a separate section of the screenhouse, which cannot be isolated from the lake. As described in Sub-Chapter 10.6, lake water is supplied to the screenhouse forebay by three 16 foot diameter pipes which terminate approximately 2250 feet from shore. It is inconceivable that damage from barge or ship accidents or even natural phenomena could totally isolate these three pipes; however, motor operated sluice gates which normally separate the discharge from the intake can be opened providing another access to the lake. Furthermore, the maximum demand for the ESW system is only slightly more than one percent of the total circulating water system during normal operation.

The pumps are designed to operate as Class I equipment, with the motor drives located above the maximum flood level. The pump motors can be supplied with power from normal or emergency sources, thereby insuring a continuous flow of service water under all conditions.

Small ESW System leaks in the Auxiliary Building drain to the various Auxiliary Building sumps. A Sump High Level alarm will alert the operator to an increasing water level in the corresponding sump. Visual inspection performed by the operator will determine the source of the leak.

Large leaks in the ESW System will initiate alarms associated with one or more of the following: Low Header Pressure, High Pipe Tunnel Sump level, or High Auxiliary Building Sump Level. Any one of these alarms will alert the operator to a probable ESW pipe rupture. In addition, flow indicators are located in the ESW supply headers and in the supply lines to each Component Cooling and Containment Spray Heat Exchanger as well as each Diesel Generator return header. The header supply valves are remotely operated to enable isolation of the supply header or pump that has failed.

Radiation alarms monitor the Essential Service Water discharge for potential inleakage of radioactive liquid. Such inleakage is unlikely but is postulated to occur due to tube leaks in the containment spray heat exchangers during their use.

9.8.3.3 Design Evaluation

Non-Essential Service Water System

The Non-Essential Service Water System is not required for the maintenance of plant safety related functions in the event of an accident. During normal operation, the system remains functional even if one Unit is out of service and its circulating water tunnels are dewatered.

Essential Service Water System

The Essential Service Water System is designed to prevent any failure in its system from curtailing normal plant operation or limiting the ability of the engineered safeguards to perform their functions in the event of an accident. Since the Essential Service Water System is required for long term heat removal, it is designed to withstand a passive failure on a long term basis. Sufficient pump capacity is included to provide design service water flow under all postulated conditions. The headers are arranged such that even loss of a complete header does not jeopardize plant safety related functions. Table 9.8-6 gives a malfunction analysis of a pump, valve and strainer.

9.8.3.4 Tests and Inspections

System components were hydrostatically tested prior to station startup and are accessible for periodic inspections or tests during operation. Electrical components, switchovers, and starting controls are tested periodically.

The essential service water pumps and certain valves are tested in accordance with the applicable edition of the ASME Operation and Maintenance (OM) Code. Periodic testing of the non-essential service water pumps is conducted in accordance with normal industry practice.

9.9 AUXILIARY BUILDING VENTILATION SYSTEM

9.9.1 General Description

The auxiliary building ventilation systems, shown in Figures 9.9-1 and 9.9-2, consist of:

a. Engineered Safety Features Ventilation System (one per plant unit).

b. Fuel Handling Area Ventilation System (one shared system).

c. General Ventilation Systems (one per plant unit with crosstie).

d. General Supply System (one per plant unit).

The auxiliary building is basically a five-level compartmented structure containing the auxiliary nuclear equipment for both units. All equipment handling radioactive fluids is located on the lower four levels of the auxiliary building. The fourth level also houses the two control rooms and the ventilation equipment.

The auxiliary building ventilation systems are designed to maintain temperatures in the various portions of the building within design limits for operation of equipment and for personnel access for inspection, maintenance and testing as required.

9.9.2 Design Bases

Outside ambient conditions used for design purposes are 91°F summer dry bulb, 75°F summer wet bulb and -7°F winter dry bulb. Ventilation is based on limiting temperatures to a maximum design calculated for each area. Heating is provided to maintain a 60°F minimum temperature.

General ventilation systems serving the auxiliary building are once-through systems. Supply air is introduced to the areas least likely to be contaminated, and exhausted directly from those with the greatest contamination potential. Additionally, the exhaust systems are of greater capacity than the supply systems, thus maintaining the area within the auxiliary building pressure boundary at a slightly negative pressure. The auxiliary building pressure boundary is the area within the auxiliary building, which is maintained at a negative pressure by the HVAC system, as required for radiological control.

All exhaust air from the auxiliary building is directed to the unit vents. There is a vent for each unit. Each vent has radiation detectors for continuous monitoring of the exhaust air during release to atmosphere.

High efficiency particulate air filter cells are designed to remove as little as 99 percent of particulates. Performance characteristics of the charcoal adsorbers provide for removal of as little as 94.05 percent of methyl iodide, which includes a minimum charcoal filter efficiency of 95 percent and a maximum bypass leakage of one percent. Supply and exhaust unit roughing filters have a NBS duct spot efficiency (Cottrell Precipitate) of 75%.

9.9.3 System Descriptions

9.9.3.1 Engineered Safety Features Ventilation

The enclosures for the engineered safety features equipment for both units are located in the lower three levels of the auxiliary building. (The containment spray heat exchanger and residual heat exchanger enclosures extend up into the fourth level with access into the enclosures from the third level only.) The enclosures for each unit's safety feature equipment are ventilated by two separate ventilation systems. The areas serviced by this system are: the containment spray pump enclosures, the residual heat removal pump enclosures, the safety injection pump enclosures, the residual heat exchanger enclosures, the containment spray heat exchanger enclosures and the reciprocating and centrifugal charging pump enclosures. Figure 9.9-2 shows a flow diagram of the engineered safety features ventilation system and is typical for the system serving either unit.

The exhaust ventilation system is composed of two 25,000 cfm fan/ filter exhaust units (1 standby) which draw air from the auxiliary building through the equipment enclosures via a common vent shaft and discharge it to the unit vent. Each fan/filter unit is composed of a 100% capacity bank of roll media roughing filters, high efficiency particulate air filters, charcoal filters and a 100% capacity exhaust fan. (There is a bypass on the charcoal filter bank.) This is a Class I ventilation system, therefore each fan/filter unit receives power from a separate engineered safeguards system bus which can be fed from the diesel bus and all components up to the connection to the unit vent are of Class I design.

Normally, one fan/filter unit operates continuously, directing the exhaust air through the roughing filter and high efficiency particulate air filter, bypassing the charcoal filter, and discharging it to the unit vent. This operation aids in the air distribution within the auxiliary building, isolates the atmosphere in the enclosures by inducing a draft through the entering portals and removes any heat generated within the enclosures.

In the event of a Phase B Isolation signal the charcoal filter bypasses are automatically closed and the air is directed through the charcoal filters in addition to the roughing and high efficiency particulate air filters. There are two independent air operated, fail-closed, dampers in the charcoal filter bypass. These dampers are arranged in parallel. The charcoal filters can be placed in service when gaseous contamination warrants their operation. The standby fan unit starts on any train related ESF system pump start signal, or upon receipt of a safety injection signal.

Make-up air for the Engineered Safety Features Ventilation System is normally provided by the Auxiliary Building general supply. Partial make-up air can be provided during a loss of off-site power by three 15,000 cfm fans blowing outdoor air into the component cooling pump area of the Auxiliary Building (third level). The fans are Class I design and are provided for use in emergency conditions.

Power for two of the 15,000 cfm fans can be provided by the Unit No. 1 diesel-generators, and for the third by Unit No. 2 diesel-generators.

These fans are dual purpose during an emergency, aiding in providing safe ambient temperature for the component cooling pump motors and providing partial make-up air for the engineered safety features ventilation system. The capacity of these fans is less than the engineered safety features ventilation system exhaust fans, thus ensuring a negative pressure within the auxiliary building pressure boundary during an emergency.

In addition to the engineered safety features ventilation system described above, the emergency diesel-generator rooms, essential service water pump enclosures, safety related battery rooms, and the Control Rooms are ventilated by systems powered by the emergency diesels. These systems include supply and/or exhaust fans sized to maintain design ambient temperatures within the various rooms and enclosures. The Auxiliary Feedwater Pump Enclosure coolers, which are designed to maintain design ambient temperatures, receive cooling water from redundant ESW headers, and are powered by the emergency diesels.

9.9.3.2 Fuel Handling Area Ventilation System

The fuel handling area is a shared facility and its ventilation system is therefore a shared facility consisting of an exhaust system and a supply system.

The fuel handling area exhaust system is composed of two 30,000 cfm fans (1 standby) which draw air through a common slot exhaust plenum along the north side of the spent fuel pool to direct it through a filter housing and discharge it to the unit No. 1 vent. The filter assembly is composed of roll media roughing filters, high efficiency particulate air filters and charcoal filters. There is a normally open bypass on the charcoal filters.

The Fuel Handling Area Supply Air System is made up of four supply units composed of fans, filters and steam coils. Two 11,000 cfm supply units are located in the western section of the Fuel Handling Area and two 2,500 cfm supply units are located in the eastern section of the Fuel Handling Area. Normally, all four supply units operate, drawing outside air through the steam coils and filters and discharging it into the fuel handling area. The air is drawn through the Fuel Handling Area into the exhaust plenum, and passed through the roughing and high efficiency particulate air filters by a continuously operating exhaust fan and discharged into the unit No. l vent. The combined capacity of the four supply units is less than that of a single exhaust fan, thus the Fuel Handling Area, as well as the entire space within the auxiliary building pressure boundary, are maintained at a slightly negative pressure.

In the event that the area radiation monitors in the Fuel Handling Area give a high radiation signal the charcoal filter bypass dampers are tripped closed thus passing the exhaust air through the charcoal filters prior to discharge to the vent. This system can also be manually aligned through the charcoal filters. Since the area radiation monitors in the Fuel Handling Area do not actuate on a high radiation signal fast enough in the event of a Fuel Handling Accident in the Auxiliary Building to preclude the discharge of radioactive gases to the Unit 1 vent, a charcoal filter in the Fuel Handling Area Ventilation System is manually placed in service prior to irradiated fuel movement. A charcoal filter is also manually placed in service prior to crane operation with non-fuel loads over the Spent Fuel Pool. The Fuel Handling Area Supply Units are also tripped on the high radiation signal, thus ensuring a negative pressure within the space.

Operation of this system is the same for both summer and winter conditions. During winter operation the heating capacity of the supply units is supplemented by steam unit heaters located throughout the Fuel Handling Area.

9.9.3.3 General Ventilation System

All areas except the fuel handling area and the safeguard equipment areas are exhausted in each unit by a ventilation system consisting of two 50% capacity fans with roughing and high efficiency particulate air filters. There is no standby capacity in these systems, however there is a normally closed tie-line between the Unit No. 1 and Unit No. 2 exhaust units.

Normally, all fans operate at their design speed and direct their air flow through the filters and then to the unit vent. This operation induces a draft of 50 to 150 fpm through the entrance portals of the various enclosures thus removing any heat, vapors or particulate matter generated within the enclosures.

The hot laboratory chemical hood and cabinet exhaust fans, sample room sink hood and sample rack exhaust fans also discharge into this system. In the event of a high radiation signal from the vent monitor, the gas decay tank discharge is automatically closed.

The hot laboratory is located in the access control area of the Auxiliary Building. The access control area includes a radiation control office, a radiation protection supervisor's office, a chemical foreman's office, and other miscellaneous rooms which have no internal contamination potential, and a hot laboratory, chemical counting room, and R. P. counting room and decontamination area which are in a potential contamination area. The clean, or non-contaminated rooms are air-conditioned by a conventional, partial recirculation system which also pressurizes these areas. The potentially contaminated areas are air-conditioned by a once-through system with 100% fresh air supply of conditioned air, which is exhausted to the auxiliary building general exhaust system.

The spray additive tank room houses the post-accident sampling system panel and is normally ventilated by the auxiliary building general exhaust system. When necessary, the spray additive tank room can be isolated from the auxiliary building general exhaust system and ventilated by the spray additive tank room filter unit and the spray additive tank room sample filter unit. The spray additive tank room filter unit consists of a roughing filter, HEPA filter, charcoal absorber, a second HEPA filter, and fan. This unit combines makeup air from outdoors with recirculated air to both pressurize the room and remove radiation contamination in order to maintain the room habitable for plant personnel. The spray additive tank room sample filter unit exhausts air from the post-accident sampling system panel and discharges into the auxiliary building general exhaust system to prevent contamination from the panel being discharged to the room. The spray additive tank room sample filter unit consists of a canister HEPA filter, canister charcoal filter, and fan.

9.9.3.4 General Supply System

Normal make-up air from the outdoors for the engineered safety features ventilation system and the auxiliary building exhaust system is provided by the auxiliary building general supply system. This system consists of four 35,000 cfm capacity fans, 2 in each unit, with steam heating coils and air filters. There is no standby capacity in this supply air system.

Normally all fans operate at their design speed and direct outdoor air through the air filters and steam coils and into the building. The air is distributed throughout the building by the suction of the various exhaust ventilating systems.

The steam coils are activated during cold weather to temper the incoming air. Sufficient heat is added to the air flow to maintain the general ambient temperature of the building at or above the 60°F design minimum. Steam and/or electric heaters located in various areas of the building are used to ensure a satisfactory minimum temperature.

All ventilation system equipment is located within the building. During operation of either unit, all of its auxiliary building ventilation systems will be activated to "normal" operation. During shutdown of either unit, its auxiliary building ventilation systems may operate in part or in total to suit maintenance, inspection, testing, refueling, etc. conditions. .

9.9.4 Design Evaluation

The Auxiliary Building Ventilation and Heating Systems capacity is adequate for the maintenance of proper temperatures in the building under operating or shutdown conditions in all types of weather.

Sufficient redundancy is included in the Engineered Safety Features Ventilation System to insure proper operation of these systems with one active component out of service. In addition, one Engineered Safety Features exhaust fan is capable of providing sufficient airflow for the proper operation of the Component Cooling Water pumps and the Engineered Safety Features systems without reliance on the Component Cooling Water pump area supply fans or the Auxiliary Building general supply fans.

The Fuel Handling Area Ventilation System has sufficient redundancy to ensure proper operation of this system with one exhaust fan out of service. Charcoal, roughing and high efficiency particulate air filters on the Fuel Handling Area Exhaust System provide protection against release of radioactivity from this area to the atmosphere.

The General Ventilation System and the General Supply System each consist of two 50% capacity segments per unit with a crosstie between the Unit No. 1 and Unit No. 2 exhaust systems, thus minimizing the possibility of losing the total system of the plant unit.

Under normal operating conditions the total exhaust flow exceeds the total fan supply. Therefore all areas within the auxiliary building pressure boundary are at a negative pressure with respect to atmosphere. All exhaust flows from within the boundary are directed to the vent of the respective unit and monitored before release to atmosphere. All supply air is pre-filtered. The fuel handling area exhaust system is directed to the Unit 1 vent.

All systems are located within the building and generally grouped for ease of access, control and monitoring.

9.9.4.1 Test and Inspections

The systems are inspected, tested and balanced upon installation. Particulate and charcoal filters were individually tested by the manufacturer after fabrication and again after installation. The engineered safeguard ventilation system and the fuel handling area ventilation system are tested on a regularly scheduled basis over the life of the plant. Replacement filters will be tested in the same manner. Filter banks can be tested for leakage and dioctylphthalate smoke test efficiency while in place, and defective cells identified for removal and replacement.

9.10 CONTROL ROOM VENTILATION SYSTEM

9.10.1 General Description

The control rooms for Unit No. 1 and Unit No. 2 are both physically located on El 633' 0" of the auxiliary building with normal access from the turbine building. Control room air conditioning equipment is in an equipment room directly above the control room. Both control room envelopes are enclosed in a missile and tornado proof structure. The ventilation equipment room, the plant process computer room, and the control room are considered the Control Room Pressure Boundary (CRPB)/envelope. The control room ventilation system is shown in Figure 9.10-1.

9.10.2 Design Bases

The control room air conditioning system is designed to maintain room temperature within limits required for operation, maintenance and testing of plant controls and uninterrupted safe occupancy during post-accident shutdown.

The control room air conditioning system is designed to maintain a temperature of 85°F maximum dry bulb and 25-80 percent relative humidity under normal operating conditions. The design is based on outside temperatures ranging from -7°F winter dry bulb to 91°F summer dry bulb and 75°F summer wet bulb. The system operates during normal or emergency conditions as required.

Conditioned air is supplied to the Control Room envelope by either of two full-capacity air-handling units (one standby). Each unit includes a roughing filter, medium efficiency filter, chilled-water coil, and a fan. Downstream of each air handler in the duct system is an electric blast coil heater and a humidifier. Each unit is provided with chilled water from an associated liquid-chiller. Each air-handler/liquid-chiller combination is independently capable of fulfilling design objectives. Condenser water for each liquid chiller is taken from a different header of the Essential service water system. All Chiller packages are non-safety related sets of components qualified to Seismic Class I requirements and their performance has been evaluated for an ESW pump discharge temperature up to 90 °F. For emergency cooling essential service water can be manually diverted directly through the seismic Class I air handling coil, thus bypassing the liquid chillers.

Continuous pressurization of the Control Room envelope is normally provided by the air conditioning system to prevent the entry of dust and dirt. Emergency filtration and pressurization is provided by a separate air-handler with roughing filters, high efficiency particulate air filters and charcoal adsorbers. This unit can also be used in the recirculation mode as a cleanup system. Performance characteristics of the high efficiency particulate air filter cells provide for removal of as little as 99 percent of particulates. Performance characteristics of the charcoal adsorbers provide for removal of as little as 94.05 percent of methyl iodide, which includes a minimum charcoal filter efficiency of 95 percent and a maximum bypass leakage of one percent. All air conditioning equipment, pressurization fans and auxiliary equipment can be powered from emergency buses. The control room air conditioning system is designed to maintain room temperature within limits required for operation, maintenance and testing of plant controls and uninterrupted safe occupancy during post-accident shutdown.

9.10.3 System Operation

Two fresh-air intakes are provided for each Control Room envelope. Both air conditioning units share one intake. A separate intake is provided for the pressurizer/cleanup filter unit. The air handling units' fresh-air intake is fitted with redundant bubble tight motor-operated isolation dampers for Control Room envelope isolation. The pressurization/cleanup air intake is fitted with two motor-operated isolation dampers installed in parallel, each with 100% design flow capacity. Normally, a fixed proportion of room air and outside air is supplied to the Control Room envelope through one of the air-handling units. Temperature is controlled by thermostats located in the control room. Each liquid chiller has an independent control system. Outdoor air supplied to the Control Room envelope through the air-handling unit maintains a positive pressure with respect to the surrounding environs to prevent entry of dust, etc.

A toilet facility is located in the Unit No. 2 control room.

The Control Room Pressurization/Cleanup Filter Unit does not normally operate. In the event of a fire signal from the cable enclosure below the Control Room, the air conditioner fresh-air intake isolation dampers are closed, and one of the two isolation dampers in the pressurization system intake is opened. The Control Room Pressurization/Cleanup Filter Unit Fan starts. These operations are all performed automatically.

The Air Conditioning System then functions as a 100 percent recirculation system. The Pressurization/Cleanup Filter Unit fan supplies 100% pressurization air separately through the high efficiency particulate air and the charcoal filters before discharging into the Control Room envelope. The controls for isolating the normal fresh-air intake and starting the Emergency Pressurization/Cleanup Filter Unit are located in both the Control Room and the air conditioning equipment room and can be manually actuated from either room.

A high radiation signal from the Unit 1 or Unit 2 Control Room radiation monitor or a Safety Injection signal, from either unit, automatically initiates closure of the isolation dampers in the Air Conditioning System. The Air Conditioning System then functions in the 100 percent recirculation mode. Upon receipt of these same signals, one of the two isolation dampers in the pressurization/cleanup system intake goes to a minimum position to allow sufficient outdoor air into the system to pressurize the Control Room envelope. The Control Room Pressurization/Cleanup Filter Unit fan automatically starts in the partial recirculation mode to remove radioactive particulates and iodines from within the Control Room envelope and from the outdoor ventilation air used for pressurization.

A manually actuated override control can be used to supply additional variable amounts of outside air (over and above the minimum makeup air required for pressurization in the cleanup mode) through the Emergency Pressurization/Cleanup Filter Unit to purge the Control Room atmosphere (outdoor conditions permitting).

9.10.4 Design Evaluation

The control room and the ventilation equipment room are both enclosed in a missile-and tornado-proof concrete structure. The ventilation equipment room is directly accessible from the control room. All other areas in the vicinity of the Control Room envelope such as cable spaces, auxiliary building, turbine building, etc. are ventilated by systems which are completely independent of the Control Room envelope ventilation system, thus fire or smoke generated in such other areas would not impair the integrity or accessibility of the Control Room envelope. Two independent, full capacity air conditioning systems serve each Control Room envelope. Two full capacity fans are provided for the Control Room Pressurization/Cleanup Filter Unit of each Control Room envelope. Redundant motor-operated isolation dampers are installed in series in the fresh-air intake for the control room air conditioning system. Parallel motor-operated isolation dampers are installed in the intake for the Pressurization/Cleanup Filter Unit System. This redundancy ensures proper room conditions with one active component out of service or inoperable.

9.10.5 Incident Control

A safety injection signal automatically closes the normal control room air intake dampers, thus preventing possibly contaminated air from entering the room. The control room pressurization/cleanup filter unit is automatically operated to remove any particulates or iodine which may leak into the Control Room envelope. In the event the control room liquid chillers are not available, plant operation may continue provided Essential Service Water (ESW) is diverted directly through the air handling units and ESW supply temperature is ≤ 65°F.

9.10.6 Tests and Inspection

The systems were inspected, tested and balanced upon installation. Periodic testing is performed to insure system operability.

High efficiency particulate air filters and charcoal filters are tested after fabrication by the manufacturer, again after installation and periodically over the life of the plant.

9.10.7 Malfunction Analysis

A failure analysis of the Control Room Ventilation System is presented in Table 9.10-1.

CHEMICAL AND VOLUME CONTROL SYSTEM CODE REQUIREMENTS[1]

Regenerative heat exchanger	ASME III [2], Class C
Letdown heat exchanger	ASME III, Class C, Tube Side, ASME VIII, Shell Side
Mixed bed demineralizers	ASME III, Class C
Reactor coolant filter	ASME III, Class C
Volume control tank	ASME III, Class C
Seal water heat exchanger	ASME III, Class C, Tube Side, ASME VIII, Shell Side
Excess letdown heat exchanger	ASME III, Class C, Tube Side, ASME VIII, Shell Side
Cation bed demineralizer	ASME III, Class C
Seal water injection filters	ASME III, Class C
Boric acid filter	ASME III, Class C
Evaporator condensate demineralizers	ASME III, Class C
Concentrates filter	ASME III, Class C
Evaporator feed ion exchangers	ASME III, Class C
Ion exchanger filter	ASME III, Class C
Condensate filter	ASME III, Class C
Piping and valves	USAS B31.1[3] , ASME III Appendix F [4]

[1] Repairs and replacements for pressure retaining components within the code boundary, and their supports, are conducted in accordance with ASME Section XI.
[2] ASME III - American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section III, Nuclear Vessels.
[3] USAS B31.1 - Code for Pressure Piping, USA Standards, and special nuclear cases where applicable
[4] The evaluation criteria of ASME III Appendix F (faulted conditions) is applicable to: 1) RCP seal leak –off return line penetration piping between inside and outside containment isolation valves (CPN 37) and 2) Piping from the RCP seal bypass line check valves to the normally closed QRV-150 valve in the common discharge header.

CHEMICAL AND VOLUME CONTROL SYSTEM DESIGN PARAMETERS

General	
Original plant design life, years	40 [1]
Seal water supply flow rate:	
Normal, gpm	32
Maximum, gpm	Note [2]
Seal water return flow rate:	
Normal, gpm	12
Maximum gpm	Note [(1)]
Letdown flow:	
Normal, gpm	75
Minimum, gpm	45
Maximum, gpm	120
Charging flow:	
Normal, gpm	132 [3]
Minimum, gpm	25
Maximum, gpm	150 [(3)]
Temperature of letdown reactor coolant entering system, °F	Unit 1: 518.9 to 543.5 Unit 2: 511.4 to 547.6
Centrifugal pump miniflow, gpm	60 (each)
Temperature of charging flow directed to Reactor Coolant System, °F	495
Temperature of effluent directed to holdup tanks, °F	127

(volumetric flow rates in gpm are based upon 130°F and 2350 psig)

[1] Licensed life is 60 years in accordance with Chapter 15 of the UFSAR.
[2] This quantity is calculated, see Technical Specification #3.4.6.2e.
[3] Flow measured at QFI-200 (common discharge before RCP seal injection)

Principal Component Data Summary

Regenerative Heat Exchanger	
Number	1 (per unit)
Heat transfer rate at design conditions, Btu/hr	$10.3x10^6$
Shell Side	
Design pressure, psig	2485
Design temperature, °F	650
Fluid	Borated reactor coolant
Material of construction	Austenitic stainless steel

	Normal (Design)	**Maximum Purification**	**Heatup**
Flow, lb/hr	37,050	59,280	59,280
Inlet temperature, °F	545	545	547
Outlet temperature, °F	290	287	366

Principal Component Data Summary

Regenerative Heat Exchanger (cont)	
Tube Side	
Design pressure, psig	2735
Design temperature, °F	650
Fluid	Borated reactor coolant
Material of construction	Austenitic stainless steel

	Normal (Design)	Maximum Purification	Heatup
Flow, lb/hr	27,170	49,400	29,640
Inlet temperature, °F	130	130	130
Outlet temperature, °F	495	461	521

Principal Component Data Summary

Letdown Orifice	
Design pressure, psig	2485
Design temperature, °F	650
Normal operating inlet pressure, psig	2235
Normal operating temperature, °F	290
Material of construction	Austenitic stainless steel

	45 gpm	**75 gpm**
Number	1 (per unit)	2 (per unit)
Design flow, lb/hr	22,230	37,050
Differential pressure at design flow, psig	1900	1900

Principal Component Data Summary

Letdown Heat Exchanger	
Number	1 (per unit)
Heat transfer rate at design conditions (heatup), Btu/hr	14.8 x 106
Shell Side	
Design pressure, psig	150
Design temperature, °F	250
Fluid	Component cooling water[1]
Material of construction	Carbon steel

	Normal	**Heatup (Design)**	**Maximum Purification**
Flow, lb/hr	203,000	492,000	510,926
Inlet temperature, °F	95	95	95
Outlet temperature, °F	125	125	125

[1] The plant has been evaluated for a CCW Hx outlet temperature range of 60°F to 105°F. It is acceptable for the CCW temperature to rise to 120°F during cooldown and post-LOCA conditions. See Section 9.2.2.

Principal Component Data Summary

Letdown Heat Exchanger	
Tube Side	
Design pressure, psig	600
Design temperature, °F	400
Fluid	Borated reactor coolant
Material of construction	Austenitic stainless steel

	Normal	**Heatup (Design)**	**Maximum Purification**
Flow, lb/hr	37,050	59,280	59,280
Inlet temperature, °F	290	380 (max.)	380 (max.)
Outlet temperature, °F	127	127	127

Principal Component Data Summary

Mixed Bed Demineralizers	
Number	2 (per unit)
Type	Flushable
Vessel design pressure:	
Internal, psig	200
External, psig	15
Vessel design temperature, °F	250
Resin volume, each, ft^3	30
Vessel volume, each, ft^3	43
Design flow rate, gpm	120
Minimum decontamination factor as measured by I-131 removal[2]	10
Normal operating temperature, °F	127
Normal operating pressure, psig	150
Resin type	Cation and anion
Material of construction	Austenitic stainless steel

[2] Assuming one per-cent of fuel containing clad defects.

Principal Component Data Summary

Reactor Coolant Filter General:	
Number	1 (per unit)
Type	Disposable Cartridge
Flow rate,	
Nominal, gpm	120
Maximum, gpm	150
Vessel:	
Design pressure, psi	200
Design temperature, °F	250
Material of construction	Austenitic stainless steel
Cartridge:	
Maximum Design ΔPressure, psi	75
Design Temperature °F	180
Absolute Retention Size, micron	≤ 6

Principal Component Data Summary

Volume Control Tank	
Number	1 (per unit)
Internal volume, ft^3	400
Design pressure:	
Internal, psig	75
External, psig	15
Design temperature, °F	250
Operating pressure range, psig	0-40
Spray nozzle flow (maximum), gpm	120
Material of construction	Austenitic stainless steel

Principal Component Data Summary

Centrifugal Charging Pumps	
Number	2 (per unit)
Type	Horizontal centrifugal
Design pressure, psig	2800
Design temperature, °F	300
Shutoff head, psi	2530
Normal suction temperature, °F	115
Design flow rate, gpm	150
Design head, ft.	5800
Available NPSH, ft.	30
Material	Austenitic stainless steel

Principal Component Data Summary

Positive Displacement Charging Pump	
Number	1 (per unit)
Type	Positive displacement with variable speed drive
Design head, ft.	5800
Design temperature, °F	250
Design pressure, psig	3200
Design flow rate*, gpm	98
Available net positive suction head, ft.	40
Suction temperature, °F	127
Discharge pressure at 130°F, psig	2500
Material of construction	Austenitic stainless steel
Hydrostatic test pressure, psig	3125

* At 130°F, 2500 psig

Principal Component Data Summary

Chemical Mixing Tank	
Number	1 (per unit)
Capacity, gal	5
Design pressure, psig	150
Design temperature, °F	200
Normal operating temperature	Ambient
Material of construction	Austenitic stainless steel
Boric Acid Tank	
Number	3 (shared)
Capacity (each), gal	11,000
Design pressure	Atmospheric
Design temperature, °F	250
Normal operating temperature, °F	110-120
Material of construction	Austenitic stainless steel
Boric Acid Tank Electric Immersion Heater	
Number (two per tank)	6
Heat transfer rate, each, kW	10
Material of construction	Austenitic stainless steel sheath

Principal Component Data Summary

Batching Tank and Batching Tank Heater Jacket	
Number	1 (shared)
Type	Cylindrical with jacketed base
Capacity, gal	800
Design pressure	Atmospheric
Design temperature, °F	300
Steam temperature, °F	250
Steam pressure, psig	15
Initial ambient temperature	32
Final fluid temperature, °F	120
Heatup time, hrs	3 (approximately)
Tank material of construction	Austenitic stainless steel
Jacket material of construction	Carbon steel
Batching Tank Agitator	
Number	1 (shared)
Fluid handled, boric acid, wt%	12
Service	Continuous
Operating temperature, °F	120
Operating pressure	Atmospheric
Material of construction	Austenitic stainless steel

Principal Component Data Summary

Excess Letdown Heat Exchanger	
Number	1 (per unit)
Heat transfer rate at design conditions, Btu/hr	4.61×10^6

	Shell Side	**Tube Side**
Design pressure, psig	150	2485
Design temperature, °F	250	650
Design flow rate, lb/hr	115,000	12,380
Inlet temperature, °F	95	545
Outlet temperature, °F	135	195
Fluid	Component cooling water[3]	Borated reactor coolant
Material of construction	Carbon steel	Austenitic stainless steel

[3] The plant has been evaluated for a CCW Hx outlet temperature range of 60°F to 105°F. It is acceptable for the CCW temperature to rise to 120°F during cooldown and post-LOCA conditions. See Section 9.2.2.

Principal Component Data Summary

Seal Water Heat Exchanger	
Number	1 (per unit)
Heat transfer rate at design conditions, Btu/hr	2.49 x10^6

	Shell Side	**Tube Side**
Design pressure, psig	150	150
Design temperature, °F	250	250
Design flow, lb/hr	99,500	160,600
Normal operating flow, lb/hr (includes miniflow)	99,500	36,000
Design operating inlet temperature, °F	95	143
Design operating outlet temperature, °F	120	127
Fluid	Component cooling water[4]	Borated reactor coolant
Material of construction	Carbon steel	Austenitic stainless steel

[4] The plant has been evaluated for a CCW Hx outlet temperature range of 60°F to 105°F. It is acceptable for the CCW temperature to rise to 120°F during cooldown and post-LOCA conditions. See Section 9.2.2.

Principal Component Data Summary

Seal Water Filter	
General:	
Number	1 (per unit)
Type	Disposal Cartridge
Flow Rates,	
Nominal, gpm	12
Maximum, gpm	325
Vessel:	
Design pressure, psi	200
Design Temperature, °F	250
Material of construction	Austenitic stainless steel
Cartridge:	
Maximum Design ΔPressure, psi	80
Design Temperature, °F	200
Nominal Retention Size, micron	25

Principal Component Data Summary

Boric Acid Filter General:	
Number	1 (per unit)
Type	Disposable Cartridge
Design Flow Rate, gpm	150
Vessel:	
Design pressure, psi	200
Design Temperature, °F	250
Material of construction	Austenitic stainless steel
Cartridge:	
Maximum Design, ΔPressure, psi	150
Design Temperature, °F	250
Nominal Retention Size, micron	20

Principal Component Data Summary

Boric Acid Transfer Pump	
Number	4 (shared)
Type	Two-speed horizontal centrifugal
Design flow rate, each, gpm	75 at high speed
Design pressure, psig	150
Design discharge head, ft.	235
Design temperature, °F	250
Temperature of pumped fluid, °F	120
NPSHA at 135°F and 87.4 gpm, ft	11.75
NPSHR at 135°F and 87.4 gpm, ft..	7.24
Material of construction	Austenitic stainless steel
Boric Acid Blender	
Number	1 (per unit)
Design pressure, psig	150
Design temperature, °F	250
Material of construction	Austenitic stainless steel

Principal Component Data Summary

Cation Bed Demineralizer	
Number	1 (per unit)
Type	Flushable
Vessel design pressure:	
Internal, psig	200
External, psig	15
Vessel design temperature, °F	250
Resin volume, ft^3	20
Vessel volume, ft^3	30
Normal operating temperature, °F	127
Normal operating pressure, psig	150
Design flow, gpm	72
Resin type	Cation
Material of construction	Austenitic stainless steel

Principal Component Data Summary

Chemical Mixing Tank Orifice	
Number	1 (per unit)
Design pressure, psig	150
Design temperature, °F	200
Design flow, gpm	2
Material of construction	Austenitic stainless steel
Boric Acid Tank Orifice	
Number	3 (shared)
Design pressure, psig	150
Design temperature, °F	200
Design flow, gpm	3
Material of construction	Austenitic stainless steel

Principal Component Data Summary

Deborating Demineralizers	
Number	2 (per unit)
Type	Fixed bed
Vessel design pressure, psig	
Internal External	200 15
Vessel design temperature, °F	250
Resin Volume, ft^3	43
Vessel volume, ft^3	56
Normal flow, gpm	120
Normal operating temperature, °F	127
Normal operating pressure, psig	150
Resin type	Anion
Material of construction	Austenitic stainless steel

Principal Component Data Summary

Seal Injection Filters	
General:	
Number	2 (per unit)
Type	Disposal Cartridge
Flow Rates,	
Nominal, gpm	32
Maximum, gpm	80
Vessel:	
Design pressure, psig	2735
Design temperature, °F	200
Material of construction	Austenitic stainless steel Cartridge:
Cartridge:	
Maximum Design ΔPressure, psi	75
Design Temperature, °F	180
Absolute Retention Size, micron	≤ 6
No. 1 Seal By-Pass Orifice	
Number	4 (per unit)
Design pressure, psig	2485
Design temperature, °F	250
Design flow, gpm	1.0
Differential pressure at design flow, psi	300

Principal Component Data Summary

Holdup Tanks	
Number	5 (shared)*
Type	Horizontal, cylindrical
Capacity, each tank, gal.	64,000
Design pressure, psig	15
Normal operating pressure, psig	3
Design temperature, °F	200
Normal operating Temperature, °F	130
Material of construction	Austenitic stainless steel

* Two pairs of tanks plus single tank.

Principal Component Data Summary

Boric Acid Reserve Tank	
Number	1 (shared)
Type	Horizontal, cylindrical
Capacity, gal.	64,000
Design pressure, psig	15
Normal operating pressure, psig	2
Design Temperature, °F	200
Normal Operating Temperature, °F	115
Material of construction	Austenitic stainless steel
Recirculation Pump	
Number	1 (shared)
Type	Centrifugal
Design flow, gpm	500
Available NPSH at 130°F, ft.	15
Design head, ft.	100
Design pressure, psig	150
Design temperature, °F	200
Normal operating temperature, °F	150
Material of construction	Austenitic stainless steel

Principal Component Data Summary

Boric Acid Evaporator Feed Pumps	
Number	3 (shared)
Type	Canned
Design flow, gpm	30
Design head (TDH), ft.	320
Design pressure, psig	150
Design temperature, °F	200
Normal fluid temperature, °F	115
Material of construction	Austenitic stainless steel
NPSH at 115°F, ft.	15
Boric Acid Evaporator Package	
Number	1 (other used for radwaste)
Design flow/unit; gas stripper feed, gpm	30
Evaporator condensate, gpm	30
Evaporator concentrates (batch flow), gpm	45
Decontamination factors (design):	
Gas stripper	Approx. 10^5 (for gas)
Evaporator	Approx. 10^6 (for liquid)
Concentration of concentrates, boric acid, wt%	4
Concentration of distillate	<10 ppm boron as H_3BO_3 Conductivity 2.0 umhos/cm
Material of construction	Austenitic stainless steel

Principal Component Data Summary

Evaporator Condensate Demineralizers	
Number	1 (other used for radwaste)
Type	Fixed bed
Design temperature, °F	250
Design pressure:	
Internal, psig	200
External, psig	15
Resin volume, each, ft^3	20
Vessel volume, each, ft^3	30
Design flow, gpm	72
Normal operating pressure, psig	50
Normal operating temperature, °F	130
Resin type (south)	Anion
Resin type (north)	As required
Material of construction	Austenitic stainless steel

Principal Component Data Summary

Monitor Tanks	
Number	2 (shared) (other 2 shared for radwaste)
Type	Diaphragm, Cylindrical
Volume, each, gal.	21,600
Design pressure	Atmospheric
Design temperature, °F	150
Normal operating temperature, °F	120
Material of construction	Stainless steel
Monitor Tank Pumps	
Number	2 (shared)
Type	Centrifugal
Design flow, gpm	150
Design head, ft.	200
Design pressure, psig	150
Design temperature, °F	200
Material of construction	Austenitic stainless steel
NPSH, ft	15

Principal Component Data Summary

Evaporator Feed Ion Exchangers	
Number	4 (shared)
Type	Flushable
Design temperature, °F	250
Design pressure:	
Internal, psig	200
External, psig	15
Resin volume, each, ft^3	20 (2 of 4 units), 27 (2 of 4 units)
Vessel volume, each, ft^3	30 (2 of 4 units)
Normal flow, gpm	30
Normal operating temperature, °F	130
Normal operating pressure, Psig	75
Resin type	Cation (2 of 4 units), Mixed Bed (2 of 4 units)
Material of construction	Austenitic stainless steel

Principal Component Data Summary

Concentrates Filter General:	
Number	2 (shared)
Type	Disposable Cartridge
Design Flow Rate, gpm	40
Vessel:	
Design pressure, psi	200
Design Temperature, °F	250
Material of construction	Austenitic stainless steel
Cartridge:	
Maximum Design, ΔPressure, psi	75
Design Temperature, °F	200
Nominal Retention Size, micron	25
or	
Absolute Retention Size, micron	0.1 to 25

Principal Component Data Summary

Concentrates Holding Tank	
Number	1 (shared)
Type	Cylindrical, heated
Volume, gal.	2,000
Design Pressure	Atmospheric
Design temperature, °F	250
Normal operating temperature, °F	150
Material of construction	Austenitic stainless steel
Concentrates Holding Tank Electric Heater	
Number	1 (shared)
Heat transfer rate, KW	6.0
Material of construction	Austenitic stainless steel
Concentrates Holding Tank Transfer Pump	
Number	2 (shared)
Type	Centrifugal can
Design flow rate, gpm	40
Design head, ft.	150
Design temperature, °F	250
Design pressure, psig	150
Available NPSH at 180°F, ft.	10
Material of construction	Austenitic stainless steel

Principal Component Data Summary

Ion Exchanger Filter	
General:	
Number	2 (shared)
Type	Disposable Cartridge
Design Flow Rate, gpm	35
Vessel:	
Design pressure, psig	200
Design temperature, °F	250
Material of construction	Austenitic stainless steel
Cartridge:	
Maximum Design ΔPressure, psi	75
Design Temperature, °F	200
Nominal Retention Size, micron	25
or	
Absolute Retention Size, micron	0.1 to 25

Principal Component Data Summary

Condensates Filter	
General:	
Number	2 (shared)
Type	Disposable Cartridge
Design Flow Rate, gpm	35
Vessel:	
Design pressure, psi	200
Design Temperature, °F	250
Material of construction	Austenitic stainless steel
Cartridge:	
Maximum Design ΔPressure, psi	80
Design Temperature, °F	200
Nominal Retention Size, micron	25

Principal Component Data Summary

Relief Valves	No.	Fluid Discharged	Fluid Inlet Temperature °F	Set Pressure psig	Back Pressure psig Constant	Back Pressure psig Buildup	Capacity gpm
Letdown line (HP)	1	Water-Steam Mixture	385 (max.)	600	3	50	98,000 lb/hr
Seal water return line	1	Water	150	150	3	50	225
Charging pump's discharge	1	Water	130	2735	15	75	100
Letdown line (LP)	1	Water	127	200	15	12	200
Volume control tank	1	Hydrogen, nitrogen or water	130	75	3	12	350
Holdup tanks	3	Nitrogen water	130	12	3	3	235
Boric Acid Reserve Tank	1	Nitrogen Water	115	12	3	3	187

FAILURE ANALYSIS OF THE CHEMICAL AND VOLUME CONTROL SYSTEM

Component		Failure	Comments and Consequences
a.	Letdown Line	Rupture in the line inside the reactor containment	The remote air-operated valve located near the main coolant loop is closed on low pressurizer level to prevent supplementary loss of coolant through the letdown line rupture. The containment isolation valves in the letdown line are automatically closed by the containment isolation signal initiated by the concurrent loss-of-coolant accident. The closure of these valves prevents any leakage of the reactor containment atmosphere outside the reactor containment.
b.	Normal and alternate Charging Line	See above	The check valves located near the main coolant loops prevent supplementary loss of coolant through the line rupture. The check valve located at the boundary of the reactor containment prevents any leakage of the reactor containment atmosphere outside the reactor containment.
c.	Seal Water Return Line	See above	The motor-operated isolation valves located inside and outside the containment are manually closed or are automatically closed by the containment isolation signal initiated by the concurrent loss-of-coolant accident. The closure of that valve prevents any leakage of the reactor containment atmosphere outside the reactor containment.
d.	Letdown Line	Rupture in the line outside the containment	Any break between containment and the letdown heat exchanger would potentially result in flashing hot letdown fluid and would be identified by lo flow in the letdown line and other system indications. The increase in letdown flow caused by a break downstream of the letdown flow indicator would be matched by an automatic increase in the charging flow and a HI Letdown Flow Alarm. An operational level in the pressurizer would, therefore, be maintained. Ultimately, the operator would be alerted by a Lo Lo level alarm in the volume control tanks. (Other indications would be an increased charging flow and falling volume control tank level. Also, the area monitors in the auxiliary building would detect any increase in activity). By observing the flow meter on the letdown line, the operator could detect the increase in flow, depending on the location of the break. The break could then be isolated by closing the redundant isolation valves in the letdown lines. Any spillage would be drained and collected in the Radioactive Waste Disposal System, while residual gases from any flashed coolant would be circulated through particulate filters before being discharged to the atmosphere through the plant vent.

RESIDUAL HEAT REMOVAL SYSTEM CODE REQUIREMENTS[1]

Residual Heat Exchangers (Tube Side)	ASME B&PV Code Section III, Class C
(Shell Side)	ASME B&PV Code Section VIII
Residual Heat Removal System Piping and Valves	USAS B31.1, 1967 Edition

[1] Repairs and replacements for pressure retaining components within the code boundary, and their supports, are conducted in accordance with ASME Section XI.

RESIDUAL HEAT REMOVAL SYSTEM DESIGN PARAMETERS

General		
Original plant design life, years		40 [1]
Component cooling water supply temperature design, °F		95
Reactor coolant temperature at startup of decay heat removal °F		350
Time to cool Reactor Coolant System from 350°F to 140°F, hrs (design basis)		20
Refueling water storage temperature (minimum), °F		70
Decay heat generation at 20 hours after shutdown, Btu/hr		$77x10^6$
H_3BO_3 concentration in refueling water storage tanks, ppm boron		2400 to 2600 (Modes 1, 2, 3 & 4) 〈 2400 (Modes 5 & 6)
COMPONENTS		
Residual Heat Exchangers		
Number		2 (per unit)
Design heat transfer, Btu/hr		$41.1x10^6$
	Shell	Tube
Design pressure, psig	150	600
Design temperature, °F	200	400
Design flow rate, lb/hr	$2.475x10^6$	$1.48x10^6$
Design outlet temperature, °F	111.6	112.3
Design inlet temperature, °F	95	140
Fluid	Component cooling water [2]	Reactor coolant (borated demineralized water)
Material of construction	Carbon steel	Austenitic stainless steel

[1] Licensed life is 60 years in accordance with Chapter 15 of the UFSAR.

[2] The plant has been evaluated for a CCW Hx outlet temperature range of 60°F to 105°F. It is acceptable for the CCW temperature to rise to 120°F during cooldown and post-LOCA conditions. See Section 6.2.2.

RESIDUAL HEAT REMOVAL SYSTEM DESIGN PARAMETERS

Residual Heat Removal Pumps	
Number	2 (per unit)
Type	Vertical in-line, single stage, centrifugal
Design pressure, psig	600
Design temperature, °F	400
Shutoff head, psig	170
Design flow rate, gpm	3,000
Design head, ft.	350
Temperature of pump fluid, °F	70 - 350
Design Speed, rpm	1780
Motor Rating, HP	400
Normal fluid	Reactor coolant
Fluid during LOCA recirculation phase	Radioactive borated water with H_2 and NaOH in solution
Material of construction	Austenitic stainless steel

Piping and Valves	
Residual heat removal loop (piping and valves in isolated loop):	
Design pressure, psig	600
Design temperature, °F	400
Residual heat removal loop isolation valves and piping:	
Design pressure, psig	2,485
Design temperature, °F	650

RESIDUAL HEAT REMOVAL SYSTEM MALFUNCTION ANALYSIS

	Component	Malfunction	Comments and Consequences
1.	Residual heat removal pumps	Rupture of a pump casing	The casing and shell are designed for 600 psig and 400°F. The pump is protected from overpressurization by two normally closed valves in the pump suction line and by a relief line, containing a relief valve, back to the pressurizer relief tank. The pump is inspectable and is located in the auxiliary building protected against credible missiles. Rupture is considered unlikely but in any event the pump can be isolated.
2.	Residual heat Removal pump	Pump fails to start	One operating pump furnishes removal pump half of the flow required to meet design cooldown rate. Failure of the other pump to start increases the time necessary for plant cooldown.
3.	Residual heat removal pump	Manual valve on pump suction is closed	This is prevented by pre startup and operational check. The valve is normally locked or sealed open.
4.	Residual Heat removal pump	Stop valve on discharge line closed or check valve sticks closed.	Stop valve is locked or sealed open. Prestartup and operational checks confirm position of valves.
5.	Remote operated valves inside containment in pump suction line	Valve fails to open	In the improbable event that one of the remote operated valves on the suction line to the residual heat removal pumps is inoperable, an attempt will be made to open it manually. If this is impossible, the plant will be cooled to about 280 °F with steam dump from the team generators, and kept at that temperature for several weeks until decay heat could be matched by the letdown heat exchangers and by feed and bleed. Feed and bleed through the CVCS will done intermittently to prevent heat transfer through the regenerative heat exchanger. The pressurizer level will be to minimum during the bleed operation and to maximum during the feed operation. It is estimated that plant cooldown be accomplished within a month.
6.	Remote operated valves inside containment on pump discharge line	Valve fails to open	Pump discharge pressure gage shows pump shut-off head indicating no flow. An alternate return line may be opened and utilized to direct flow to the RCS.
7.	Residual heat exchanger	Tube or shell rupture	Rupture is considered unlikely, but in any event the faulty heat exchanger may be isolated.
8.	Residual heat exchanger vent or drain valve	Left open	This is prevented by prestartup operational checks.

SPENT FUEL POOL COOLING SYSTEM CODE REQUIREMENTS[1]

Spent Fuel Pool Heat Exchanger (tube side)	*ASME B&PV Code Section III, Class C*
(shell side)	*ASME B&PV Code Section VIII*
Spent Fuel Pool Filter	*ASME B&PV Code Section III, Class C*
Spent Fuel Pool Piping and Valves	*USAS B31.1*

[1] Repairs and replacements for pressure retaining components within the code boundary, and their supports, are conducted in accordance with ASME Section XI.

SPENT FUEL POOL COOLING SYSTEM COMPONENT DESIGN DATA

System cooling capacity, Btu/hr	29.8×10^6
Spent fuel pool heat exchanger	
Number	2 (Shared)
Design heat transfer, Btu/hr	14.9×10^6

	Shell	Tube
Design pressure, psig	150	150
Design temperature, °F	200	200
Design flow rate, lb/hr	1.49×10^6	1.14×10^6
Design inlet temperature, °F	95	120
Design outlet temperature, °F	105	106.9
Fluid	Component Cooling Water[1]	Spent fuel pool (borated demineralized water)

Spent fuel pool pump	
Number	2 (shared)
Design pressure, psig	150
Design temperature, °F	200
Design flow rate, gpm	2300
Minimum developed head, ft.	125
Temperature of pumped fluid, °F,	80 - 180
Fluid	Spent fuel pool water(borated demin. water)
NPSH, ft. (available/required)	30/10
Material	Austenitic Stainless Steel

[1] The plant has been evaluated for a CCW Hx outlet temperature range of 60°F to 105°F. It is acceptable for the CCW temperature to rise to 120°F during cooldown and post-LOCA conditions. See Section 9.4.2.

SPENT FUEL POOL COOLING SYSTEM COMPONENT DESIGN DATA

Spent fuel pool skimmer pump	
Number	1 (Shared
Design pressure, psig	50
Design temperature, °F	200
Design flow rate, gpm	100
Minimum developed head, ft.	50
Temperature of pumped fluid, °F	75 - 150
Fluid	Spent fuel pool water
NPSH, ft. (available/required)	30/2
Material	Austenitic Stainless Steel
Refueling water purification pump	
Number	1
Design pressure, psig	600
Design temperature, °F	200
Design flow rate, gpm	Nom. 100, Max 150
Minimum developed head, ft.	130
Fluid	Refueling water
NPSH, Ft. (available/required)	@ 100gpm 30/5, @ 150 gpm 43/7
Material	Austenitic stainless steel
Spent fuel pool demineralizer	
Number	1 (Shared)
Type	Flushable
Vessel design pressure, psig	
Internal -	200
External -	15
Vessel design temperature, °F	250
Design flow rate, gpm Maximum	150
Normal flow, gpm	100, Max 150
Normal operating temperature, °F	120
Normal operating pressure, psig	Approx. 50
Resin type	anion and cation

SPENT FUEL POOL COOLING SYSTEM COMPONENT DESIGN DATA

Spent fuel pool filter	
Number	1 (Shared)
Type	Replaceable (Cellulose and/or glass resin)
Internal design pressure, psig	200
Design temperature, °F	250
Design flow rate, gpm	Nom. 100, Max. 150
Filtration requirement	98% retention of particles above 5 micron
Spent fuel pool skimmer filter	
Number	1 (Shared)
Type	Replaceable (Cellulose and/or glass resin)
Internal design pressure, psig	200
Design Temperature, oF	250
Design flow rate, gpm	150
Filtration requirement	98% retention of particles above 5 micron
Refueling water purification filter	
Number	1 (Shared)
Type	Replaceable (Cellulose and/or glass resin)
Internal design pressure, psig	200
Design temperature, °F	250
Design flow rate, gpm	Nom. 100, Max. 150
Particle size retained, minimum, micron	5
Spent fuel pool strainer	2 (Shared)
Design flow rate, gpm	2300
Fluid	Borated demineralized water

SPENT FUEL POOL COOLING SYSTEM COMPONENT DESIGN DATA

Spent fuel pool skimmer strainer	
Number	1 (Shared)
Type	Basket
Design flow rate, gpm	100
Design pressure, psig	50
Design temperature, °F	200
Spent fuel pool skimmers	
Number	2 (Shared)
Design flow rate per unit, gpm	50

SPENT FUEL POOL COOLING SYSTEM MALFUNCTION ANALYSIS

Component		Malfunction	Comments and Consequences
1.	Spent fuel pool pumps	Rupture of a pump casing	The casing is designed for 150 psig and 200 °F which exceeds maximum operating conditions. The pump is inspectable and is located in the auxiliary building protected against credible accidents. Rupture is considered unlikely; however, the pump can be isolated.
2.	Spent fuel pumps	Pumps stops running and cannot be restarted	The second cooling train is used.
3	Spent fuel pool pump	Manual valves on pump suction is closed	This is prevented by prestart-up and operational check, etc.
4.	Spent fuel pool pump	Suction strainer plugs	The second train is used
5.	Spent fuel pool heat exchanger	Tube or shell rupture	Rupture is considered unlikely because of low operating pressure; however, the faulty heat exchanger can be isolated
6.	Spent fuel pool skimmer pump	Pump stops running and cannot be restarted	Spent fuel assemblies continue to be cooled by spent fuel pool sump. Pool water may become slightly murky possibly decreasing visual observations until pump is restored to service. Fuel pool water is clarified to some extent by-passing spent fuel pool water through spent fuel pool demineralizer.

COMPONENT COOLING SYSTEM CODE REQUIREMENTS[1]

Component cooling heat exchangers	ASME B&PV Code Section VIII 1968 Edition[2]
Component cooling surge tank	ASME B&PV Code Section VIII 1968 Edition
Component cooling loop piping and valves	USAS B31.1 1967 Edition

[1] Repairs and replacements for pressure retaining components within the code boundary, and their supports, are conducted in accordance with ASME Section XI.

[2] The component cooling water heat exchanger was designed and fabricated in accordance with ASME B&PV Code, Section VIII, 1968 edition requirements. Installation was in accordance with USAS B31.1, 1967 Edition.

COMPONENT COOLING WATER SYSTEM FLOW REQUIREMENTS PER TRAIN (GPM) [1]

Service	Normal Operation	LOCA Injection	LOCA Recirculation	Cooldown [2]
Safeguards Train [3]				
RHR Heat Exchanger	- [4]	- [5]	5000 [6]	4950
CCP PP Hx	45	26	45	45
SI PP Hx [7]		20	24	-
RHR PP Hx	-	5	10	10
CTS PP Hx [8]	-	3	3	-
Subtotal	45	54	5082	5005
Miscellaneous Train				
BA Evaporator	1442 [9]	-	-	-
SFP Hx [10]	2980	-		-
Waste Gas Compressors [11]	42.5	-		42.5

1 The values in this table are for the operating point described. Plant procedures may consider uncertainty as appropriate. See Section 9.5.
2 Cooldown refers to the operation that takes the reactor plant from hot shutdown (350 °F) and pressure to cold (140 °F) conditions. Once below 200 °F, which will preclude boiling, the CCW flow to the RHR heat exchanger may be reduced to accommodate the reduced RHR heat load.
3 The flows shown reflect the use of one safeguard's train. The second safeguard train may be placed in service provided the necessary equipment is operable. Single train operation results in minimum safeguard's requirements and a minimum cooldown.
4 This path is normally valved off with no flow through the path.
5 The flow through this path satisfies the miniflow requirements for the CCW Pump; there is no RHR heat load on CCW during LOCA injection.
6 UFSAR Chapter 14.3.4, "Containment Peak Pressure Transient", input assumption #21, provides the explanation of this input to the containment accident analysis of record.
7 Flow to the SI PP Hx and CTS PP Hx is not specified during the Normal or Cooldown mode since the SI PP and CTS PP are not required to operate during these modes of operation. However, the flow paths to the SI PP Hx and CTS PP Hx are open to permit flow.
8 Flow to the SI PP Hx and CTS PP Hx is not specified during the Normal or Cooldown mode since the SI PP and CTS PP are not required to operate during these modes of operation. However, the flow paths to the SI PP Hx and CTS PP Hx are open to permit flow.
9 Maximum flow; may be significantly reduced as necessary to control process temperatures.
10 SFP Hx is assumed to be on the non-accident unit.
11 Each of two, Waste Gas Compressor cooling can be aligned to either unit.

COMPONENT COOLING WATER SYSTEM FLOW REQUIREMENTS PER TRAIN (GPM) [1]

Service	Normal Operation	LOCA Injection	LOCA Recirculation	Cooldown [2]
Sample Coolers (U1/U2)	139/169 [12]	-		139/169 [13]
Post Accident Sampling System (U1/U2) [14]	-	-	8.5/32.5	-
Letdown Hx [15]	984 [16]	-	-	984 [17]
Seal Water Heat Exchanger	199	[4]	[4]	199
Ctmt. Pen. Cooling	300	-	-	300
CEQ Fan Mtrs [18]	-	15	15-	-
RCP Motors	404	-	-	404
RCP Thermal Barrier Hxs	140	-		140
Reactor Support Clrs	40	-	-	40
Subtotal (U1/U2)	6670.5/6700.5	15/15	23.5/47.5	2248.5/2278.5
Totals (U1/U2)	6715.5/6745.5	69/69	5105.5/5129.5	7253.5/7283.5

12 Maximum flow; may be significantly reduced as necessary to control process temperatures.
13 Maximum flow; may be significantly reduced as necessary to control process temperatures.
14 The 8.5/32.5 gpm (U1/U2) flow is based on the use of 3 model QC-563 (8 gpm ea. Unit 2) and 1 model QC-501 (8.5 gpm) sample coolers (Unit 1 and Unit 2).
15 The Letdown Hx is assumed to be inservice. The excess letdown Hx is placed inservice if the letdown Hx is unavailable. The excess letdown Hx's design flow rate is 230 gpm.
16 Maximum flow; may be significantly reduced as necessary to control process temperatures.
17 Maximum flow; may be significantly reduced as necessary to control process temperatures.
18 For LOCA Injection and Recirculation only one CEQ fan is required. An analysis was performed which determined acceptable performance at a reduced flow of 15 gpm for 1 fan.

Component Cooling System Component Design Data

Component Cooling Pumps	
Quantity	5 (incl. 1 Maintenance spare)
Type	Horizontal, centrifugal
Rated capacity, gpm	9000
Rated head, TDH, ft	190
Rated motor horsepower, HP	500
Rated motor speed, rpm	1170
Casing material	Cast iron
Design pressure, psig	150
Design temperature, °F	200
Component Cooling Heat Exchangers	
Quantity	4
Type	Shell and Tube
Heat transferred, Btu/hr	$76x10^6$
Shell side	
Component cooling water outlet Temp., °F	95[1]
Component cooling water inlet Temp., °F	114
Component cooling water	
Design flow rate, lb/hr	$4.0 x10^6$
Maximum flow rate, lb/hr	$4.5 x10^6$
Design Temperature, °F	200
Design pressure, psig	150
Tube side	
Service water inlet temperature, °F	76[2]
Service water outlet temperature, °F	92
Service water flow rate, lb/hr	$4.75x10^6$
Design pressure, psig	150
Design temperature, °F	200
Tube material	Arsenical copper

[1] These data reflect the original design of the components. The CCW system has been designed and analyzed to operate in the range of 60°F to 105°F. It is acceptable for the CCW Hx outlet temperature to rise to 120°F during cooldown and post-LOCA conditions.

[2] These data reflect the original design of the components. The system has been evaluated for an ESW pump discharge temperature of 88.9°F.

COMPONENT COOLING SYSTEM MALFUNCTION ANALYSIS

Component	Malfunction	Comments and Consequences
1. Component cooling water pump	Rupture of a pump casing	Isolate pump and start redundant pump. Minimum safeguards requirements only one out of two pumps.
2. Component cooling water pump	Pump fails to start	One operating pump will supply sufficient flow. Redundancy is sufficient to provide ample flow for any condition.
3. Component cooling water pump	Manual valve on a pump suction line closed	This will be prevented by pre-startup and operational checks. Further, during normal operation, each pump will be checked on a periodic basis, which would show that a valve was closed.
4. Component cooling water pump	Stop valve on discharge line closed or check valve sticks closed	Stop valve will be checked open by pre-startup and operational checks. The stop valve and the check valve will be checked by periodic operation of the standby pump during normal operation.
5. Component cooling heat exchanger	Tube or shell rupture	Isolate and valve in standby train.
6. Component cooling heat exchanger vent or drain valve	Left open	This will be prevented by pre-startup and operational checks. On the in service heat exchangers such a situation would be readily assessed by makeup requirements to system. On the out-of-service heat exchangers such a situation would be assessed during periodic inspection of general area.
7. Thermal Barrier Heat Exchanger	Tube Leak or Rupture	See Section 9.5.4 Detection by CCW Radiation Monitor or Surge Tank level. Redundant containment isolation valves provide means to isolate if a leak is detected (isolation would require plant shutdown).

MODULE DATA

Module I.D.	Quantity	Array Cell Size	Total Cell Count for the Module Type
A*	5	13x14	910
B	4	12x14	672
C	4	13x12	624
D	2	12x12	288
E	4	13x11	572
F	2	12x11	264
G	1	12x10	120
H**	1	13x14 - (8x2)	166
Total	**23**		**3616**

* Three of the A modules have one triangle cell to accommodate pool corner curvature.
** Non-rectangular module

COMMON MODULE DATA

Storage cell inside dimension:	8.75" ± 0.04"
Storage cell height (above the baseplate):	168 ± 1/16"
Baseplate thickness:	0.75" (nominal)
Support leg height:	5.25" (nominal)
Support leg type:	Remotely adjustable legs
Number of support legs:	4 (minimum)
Remote lifting and handling provision:	Yes
Poison material:	Boral
Poison length:	144"
Poison width:	7.5"
Cell Pitch:	8.97" (nominal)

1100 ALLOY ALUMINUM PHYSICAL AND MECHANICAL PROPERTIES

Density	0.098 lb/cu. in. , 2.713 gm/cc
Melting Range	1190-1215 °F, 643-657 °C
Thermal Conductivity (77 °F)	128 BTU/hr/sq ft/°F/ft, 0.53 cal/sec/sq cm/°C/cm
Coef. of Thermal Expansion (68-212 °F)	$13.1x10^{-6}$/°F, $23.6x10^{-6}$/°C
Specific heat (221 °F)	0.22 BTU/lb/°F, 0.23 cal/gm/°C
Modulus of Elasticity	$10x10^{6}$ psi
Tensile Strength (75 °F)	13,000 psi annealed, 18,000 psi as rolled
Yield Strength (75 °F)	5,000 psi annealed , 17,000 psi as rolled
Elongation (75 °F)	35-45% annealed, 9-20% as rolled
Hardness (Brinell)	23 annealed, 32 as rolled
Annealing Temperature	650 °F, 343 °C

CHEMICAL COMPOSITION (BY WEIGHT) - ALUMINUM (1100 ALLOY)

99.00% min.	Aluminum
1.00% max.	Silicone and Iron
0.05-0.20% max.	Copper
.05% max.	Manganese
.10% max.	Zinc
.15% max.	others each

BORON CARBIDE CHEMICAL COMPOSITION, WEIGHT %	
Total boron	70.0 min.
B^{10} isotopic content in natural boron	18.0
Boric oxide	3.0 max.
Iron	2.0 max.
Total boron plus total carbon	94.0 min.

BORON CARBIDE PHYSICAL PROPERTIES	
Chemical formula	B_4C
Boron content (weight)	78.28%
Carbon content (weight)	21.72%
Crystal Structure	rombohedral
Density	2.51 gm./cc-0.0907 lb/cu. in.
Melting Point	2450°C (4442 °F)
Boiling Point	3500°C (6332 °F)
Microscopic thermal-neutron cross-section	600 barn

SUMMARY OF CRITICALITY SAFETY ANALYSES

NORMAL STORAGE CONFIGURATION

Design Basis burnups at 4.95% ±0.05% initial enrichment	0 in Region 1 50 in Region 2 38 in Region 3
Temperature for analysis	20°C (68°F)
Reference K_∞ (KENO-5a)	0.9160
Calculational bias, δk	0.0090
Axial burnup effect	0.0037
UNCERTAINTIES	
Bias statistics (95%/95%)	±0.0021
KENO-5a statistics (95%/95%)	±0.0012
Manufacturing tolerances	±0.0064
Water-gap	±0.0045
Fuel enrichment	±0.0034
Fuel density	±0.0035
Burnup (38 MWD/KgU)	±0.0019
Burnup (50 MWD/KgU)	±0.0047
Eccentricity in position	±0.0019
Statistical combination of uncertainties[1]	±0.0110
TOTAL	0.9287 ±0.0110
Maximum reactivity (k_∞)	0.940

[1] Square root of sum of squares.

SUMMARY OF CRITICALITY SAFETY ANALYSES

INTERIM CHECKERBOARD LOADING

Design Basis burnups at 4.95% ±0.05% initial enrichment	0 in Region 1, 50 in Region 2, Region 3 - CHECKERBOARD, (FRESH FUEL AND EMPTY)
Temperature for analysis	20°C (68 °F)
Reference K_∞ (KENO-5a)	0.9168
Calculational bias, δk	0.0090
Axial burnup effect	0.0037
UNCERTAINTIES (Assumed same as the reference case)	
Bias statistics (95%/95%)	±0.0021
KENO-5a statistics (95%/95%)	±0.0012
Manufacturing tolerances	±0.0064
Water-gap	±0.0045
Fuel enrichment	±0.0034
Fuel density	±0.0035
Burnup (38 MWD/KgU)	NA
Burnup (50 MWD/KgU)	±0.0047
Eccentricity	±0.0019
Statistical combination of uncertainties[1]	±0.0108
TOTAL	0.9295 ±0.0108
Maximum reactivity (k_∞)	0.940

[1] Square root of sum of squares.

Power Block Definition

Power Block Structure	Fire Area(s)
Auxiliary Building	AA1, AA3, AA5/6, AA7, AA8, AA9, AA10, AA11, AA12, AA13, AA14, AA15, AA23, AA24, AA25, AA26, AA27, AA29, AA30, AA31, AA34, AA35, AA36/42, AA37, AA38, AA39A, AA39B, AA40, AA41, AA43, AA44, AA45A, AA45B, AA54, AA55
Unit 1 Containment Building	AA56
Unit 2 Containment Building	AA58
Turbine Building	AA2, AA2C, AA16, AA17, AA18, AA19, AA20, AA21, AA22
Control Rooms, Cable Vaults, & HVAC Equipment Areas	AA46, AA47, AA48, AA50, AA51, AA52, AA57A, AA57B
Service & Office Build (Containment Cooling Area Only)	AA2
Fuel Handling Areas	AA3
Screenhouse, ESW Pump & Tunnel Areas and Water Intake & Discharge System	AA2, AA2C, AA32, AA33, YD
Fire Pump House	YD
Offsite power distribution equipment (i.e., unit auxiliary and reserve transformers), portions of the non-safety power distribution system (i.e., main generator step up transformer, 745-345 and 34.5 kv switchyard transformers), and the Supplemental Diesel Generator Area.	YD

COMPRESSED AIR SYSTEM DESCRIPTIVE INFORMATION[1]

PLANT AIR SYSTEM

PLANT AIR COMPRESSOR	
Number	2 (one for each unit)
Type	Centrifugal
Discharge Pressure, psig	100
Discharge Temperature (approximate)°F	266
Capacity, icfm (with inlet conditions of 14.3 psia and 110°F)	1,485
PLANT AIR COMPRESSOR AFTERCOOLER	
Number	1 per compressor
Type	Shell & Tube
Tube Side Flow, icfm (air)	1,500
Shell Side Flow, gpm (water)	23
Shell Side Design Pressure, psig	150
Tube Side Design Pressure, psig	150
Shell Material	Carbon Steel
Tube Material	Admiralty
Design Code	ASME B&PV Code Section VIII
PLANT AIR RECEIVER	
Number	2 (one for each unit)
Capacity, ft^3	200
Design Pressure, psig	125
Design Temperature, °F	300
Operating Pressure, psig	100
Operating Temperature, °F	105
Material	Carbon Steel
Design Code	ASME B&PV Code Section VIII

[1] The information in this Table reflects manufacturer equipment ratings and specifications for the compressed air systems components. These values do not necessarily reflect design basis values for the compressed air systems.

COMPRESSED AIR SYSTEM DESCRIPTIVE INFORMATION

CONTROL AIR SYSTEM

CONTROL AIR COMPRESSOR	
Number	2 (one for each unit)
Type	Reciprocating
Nominal Discharge Pressure, psig	100
Discharge Temperature, °F	320
Capacity, icfm	338, @29.92 in HgA inlet, 480 RPM, 100 psig discharge
CONTROL AIR COMPRESSOR AFTERCOOLER	
Number	1 per compressor
Type	Shell & Tube
Tube Side Flow, icfm(air)	338
Shell Side Flow, gpm (water)	5
Shell Side Design Pressure, psig	150
Tube Side Design Pressure, psig	150
Shell Material	Carbon Steel
Tube Material	Admiralty
Design Code	ASME B&PV Code Section VIII
CONTROL AIR RECEIVER (WET CONTROL AIR)	
Number	2 (one for each unit)
Capacity, ft^3	500
Design Pressure, psig	125
Design Temperature, °F	300
Operating Pressure, psig	100
Operating Temperature, °F	95
Material	Carbon Steel
Design Code	ASME B&PV Code Section VIII

COMPRESSED AIR SYSTEM DESCRIPTIVE INFORMATION

CONTROL AIR SYSTEM (CONT'D)

CONTROL AIR PREFILTER	
Number	4 (two for each unit in parallel strings)
Capacity, scfm	325
Inlet pressure, psig	100
Inlet temperature, °F (saturated)	95
Effluent Dewpoint °F(at design pressure)	-40
Retention Size, microns	5
Type	Adsorbent
CONTROL AIR DRIER	
Number	8 (four for each unit in two parallel strings)
Capacity, scfm	325
Dew Point at 100 psig, °F	-40
Type	Convection
CONTROL AIR AFTER FILTER	
Number	4 (two for each unit in parallel)
Capacity, scfm	840
Retention Size, microns	4
Type	Adsorbent
CONTROL AIR RECEIVER (DRY CONTROL AIR)	
Number	4 (two for each unit in parallel strings)
Capacity, ft^3	500
Design Pressure, psig	125
Design Temperature, °F	300
Operating Pressure, psig	100
Operating Temperature, °F	105
Material	Carbon Steel
Design Code	ASME B&PV Code Section VIII

COMPRESSED AIR SYSTEM DESCRIPTIVE INFORMATION

CONTROL AIR SYSTEM (CONT'D)

BACKUP PLANT AIR COMPRESSOR WITH INTEGRAL SKID MOUNTED AIRCOOLED AFTERCOOLER	
Number	1 (common for both units)
Type	Rotary Screw
Nominal Discharge Pressure	100
Discharge Temperature (approximate)^{o}F	108
Capacity, icfm (with inlet conditions of 14.5 psia and 80^{o}F)	668
BACKUP PLANT AIR COMPRESSOR AIR RECEIVER	
Number	1
Capacity, ft^3	100
Design Pressure, psig	150
Design temperature, ^{o}F	450
Operating Pressure, psig	100
Operating Temperature, ^{o}F	105
Material	Carbon Steel
Design Code	ASME B & PV Code section VIII

Service Water Systems Components Design Data

Non-Essential Service Water Pumps	
Quantity	4
Type	Horizontal Centrifugal
Rated TDH (ft.)	175
Rated Capacity - (GPM)	4,500
Rated Motor Horsepower (HP)	250
Rated Motor Speed	1800 (nominal)
Casing material	Cast Steel
Non-Essential Service Water Strainers	
Quantity	4
Type	Automatic – Self Cleaning
Essential Service Water Pumps	
Quantity	4
Type	Vertical
Rated TDH (ft.)	145
Rated Capacity - (GPM)	10,000[1]
Rated Motor Horsepower (HP)	450
Rated Motor Speed	900 (nominal)
Casing material	Cast iron or Cast steel
Essential Service Water Strainers	
Quantity	4
Type	Duplex-automatic backwashing

[1] Flow rates up to 12,200 gpm have been evaluated as acceptable.

Non-Essential Service Water Requirements [1]

Service	Quantity			Flow - GPM					Remarks
	Installed	Min Req'd	Norm Req'd	Min. Design	Normal Design	Maximum Expected	Station Black-Out	Cont. or Inter. Serv.	
Unit No. 1 Main Oil Coolers	2	1	1	1000	1000	1258	-	C	
Unit No. 2 Main Oil Coolers	4	2	2	1056	1056	1056	-	C	
Unit No. 1 FPT Oil Coolers	4	2	2	270	270	270	-	C	
Unit No. 2 FPT Oil Coolers	4	2	2	354	354	354	-	C	
Unit No. 1 Main Turbine and Feed Pump EHC Control Fluid Coolers	1	1	1	60	60	60	-	C	
Unit No. 2 Main Turbine and Feed Pump EHC Control Fluid Coolers	1	1	1	60	60	60	-	C	
Unit 1 Containment Chiller Condensers	3	2	2	2900	2900	2900	2900	C	

[1] Water requirements based on 76°F maximum lake temperature except as noted. The system has been evaluated for operation with an NESW cooling water temperature of 88.9°F

Non-Essential Service Water Requirements [1]

	Quantity			Flow - GPM					
Service	**Installed**	**Min Req'd**	**Norm Req'd**	**Min. Design**	**Normal Design**	**Maximum Expected**	**Station Black-Out**	**Cont. or Inter. Serv.**	**Remarks**
Unit 2 Containment Chiller Condensers	3	2	2	2900	2900	2900	2900	C	
Unit No. 2 Generator Seal Oil Coolers	2	2	2	160	160	160	-	C	Based on 95°F cooling water
Former Technical Support Center A/C Units	3 [2]	3	3	77	77	77	-	C	Shared System
Glycol Refrigeration Condensers	10	6	7	360	420	600	-	C	Shared System
Ice Storage Condensing Units	2	1	1	20	30	30		I	Shared System & Used During Ice Loading Only.

[2] Does not include the 4th and 5th air-cooled units.

Non-Essential Service Water Requirements [1]

	Quantity			Flow - GPM					
Service	**Installed**	**Min Req'd**	**Norm Req'd**	**Min. Design**	**Normal Design**	**Maximum Expected**	**Station Black-Out**	**Cont. or Inter. Serv.**	**Remarks**
Ice Machines	3	2	2	10	50	75		I	Shared System & Used During Ice Loading Only.
Air Cooler Stage 1	1	1	1	20	20	20		I	Shared System & Used During Ice Loading Only.
Air Cooling Condensing Units Stage 2 & 3	2	2	2	20	30	30		I	Shared System & Used During Ice Loading Only.
Mixed Borated Water Condensing Unit	1	1	1	20	30	30		I	Shared System & Used During Ice Loading Only.
Plant Air Compressors [3]	2	1	1	80	80	160	160	C	Shared System
Control Air Compressors [4]	2	0	0	0	0	10	10	I	Shared System

[3] Includes compressor oil cooler, aftercooler and 1st and 2nd stage intercoolers.

[4] Includes compressor jacket cooler and aftercooler.

Non-Essential Service Water Requirements [1]

	Quantity			Flow - GPM					
Service	**Installed**	**Min Req'd**	**Norm Req'd**	**Min. Design**	**Normal Design**	**Maximum Expected**	**Station Black-Out**	**Cont. or Inter. Serv.**	**Remarks**
Degassifier Vacuum Pump 1st Stage	1	1	1	25	25	25	-	C	Shared System
Degassifier Vacuum Pumps 2nd Stage	2	1	1	50	50	50	-	C	Shared System
Demineralizer Make-Up System	1	0	1	0	600	800	-	I	Shared System
Heating Boiler Blowdown Flash Tank	1	0	0	110	110	110	-	I	Shared System
Unit No. 1 Auxiliary Feed Pumps	3	3	3	6	6	6	-	I	Bearing cooling water for turbine and motor-driven pumps
Unit No. 2 Auxiliary Feed Pumps	3	3	3	6	6	6	-	I	Bearing cooling water for turbine and motor-driven pumps
Miscellaneous Sealing and Cooling Water System (MSCW)	-	-	-	-	-	300	-	C	
Totals				9,564[5]	10,294	11,347			

[5] Actual operational data with cold NESW temperatures indicate nominal minimal flow is approximately 8000 gpm

Chilled Water Subsystem Nominal Design Flow Rates (Actual Values May Vary)

Containment Ventilation: Unit No. 1 Upper Units	93.9
Containment Ventilation: Unit No. 1 Lower Units	912
Containment Ventilation: Unit No. 2 Upper Units	93.9
Containment Ventilation: Unit No. 2 Lower Units	912
Unit No. 1 Instr. Room Vent	9.2
Unit No. 2 Instr. Room Vent	9.2
Unit No. 1 RCP Motor Air Coolers	148
Unit No. 2 RCP Motor Air Coolers	148

Chilled water can also be directed to the Steam Generator Blowdown Heat Exchanger at a design flow rate of 65 gpm and to the Steam Generator Blowdown Sample Heat Exchanger at a design flow rate of 14.1 gpm

Nominal Design Flow Rates for Alternate Configuration of NESW Directly To Containment AHUs:

Containment Ventilation: Unit No. 1 Upper Units	320
Containment Ventilation: Unit No. 1 Lower Units	1760
Containment Ventilation: Unit No. 2 Upper Units	320
Containment Ventilation: Unit No. 2 Lower Units	1760
Unit No. 1 Instr. Room Vent	50
Unit No. 2 Instr. Room Vent	50
Unit No. 1 RCP Motor Air Coolers	440
Unit No. 2 RCP Motor Air Coolers	440
Unit No. 1 Steam Generator Blowdown Sample Heat changer	11
Unit No. 2 Steam Generator Blowdown Sample Heat Exchanger	11
Unit No. 1 Steam Generator Blowdown Heat Exchanger	160
Unit No. 2 Steam Generator Blowdown Heat Exchanger	160

With less than design temperature NESW, actual flow rates may be lower and still provide adequate cooling.

Essential Service Water System Flow Requirements per Train (GPM)

Service [1]	**Normal Operation**	**LOCA Injection**	**LOCA Recirculation**	**Cooldown**
CCW HX	8700	5000 [2]	5000 [2]	9100
CTS HX	-	-	2400 (U2) [3] 2100 (U1) [4]	-
EDG CLRS	-	540	540	-
AFW SYS [5]	-	450	450	-
AFP Enclosure CLRS [6]	102	102	102	102
CRAC[7]	85	85	85	85
Totals	8887	6177	8277 - 8577	9287

[1] The flows shown reflect the use of one ESW train in service corresponding to one CCW safeguard's train. The second ESW train may be placed in service provided the necessary equipment is operable or the second CCW safeguard train is operating. Single train operation results in minimum safeguard's requirements and a minimum cooldown rate.

[2] Per update Westinghouse analyses, WCAP-14285, Revision 1, May, 1995.

[3] Per update Westinghouse analyses, WCAP-15302, December 13, 1999.

[4] Per EC-0000048860.

[5] This flow path is aligned manually and required only as a backup to the normal condensate supply to the Auxiliary Feedwater System.

[6] Auxiliary Feedwater Pump Enclosure Coolers will be provided with a continuous supply of ESW in all modes of operation. Flow is nominal based on cooler rated heat capacity. Different flows are allowable based on engineering analysis, provided required heat removal is achieved.

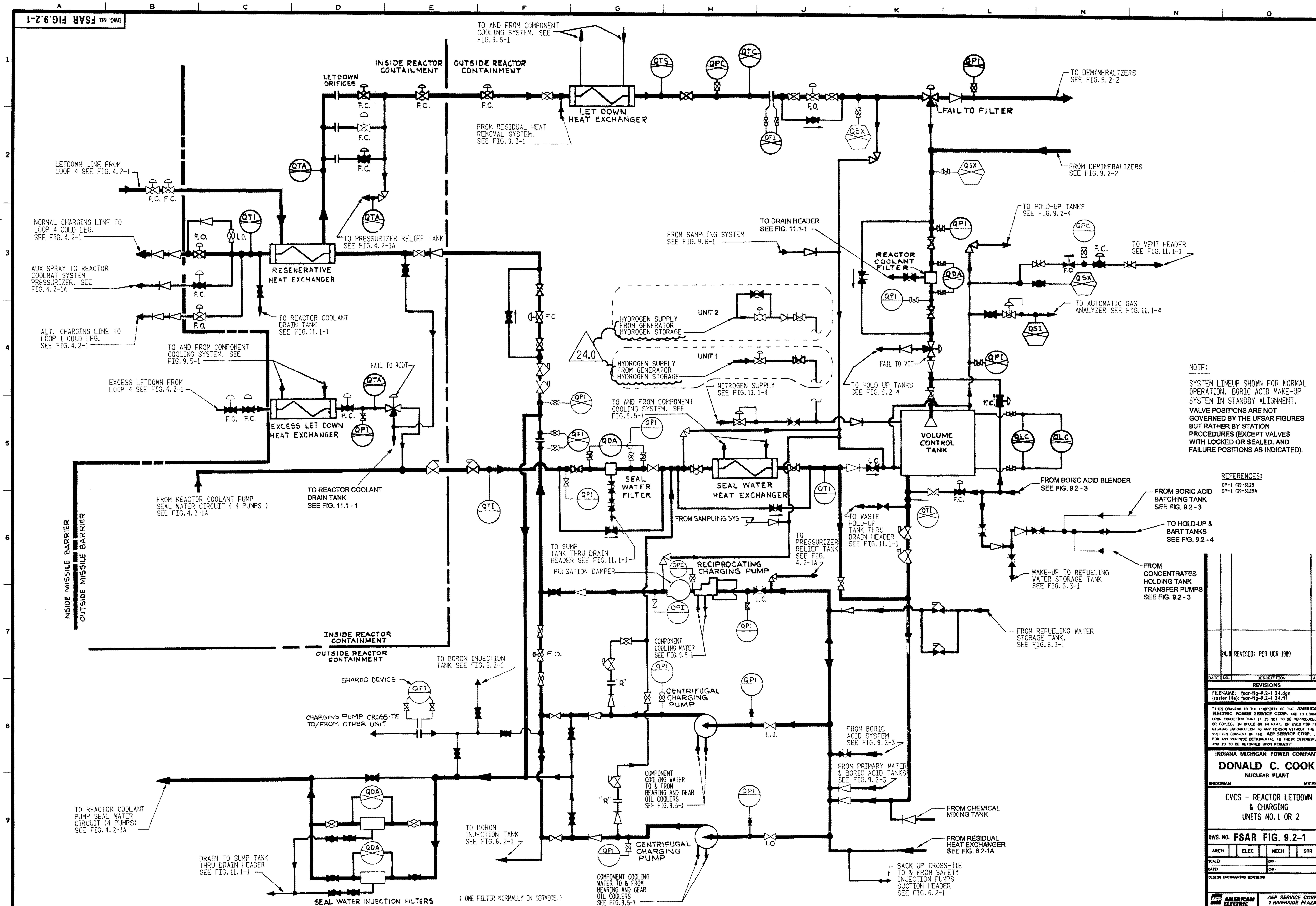
[7] Per FCN-51362-026 (U1-N), FCN-51362-016 (U1-S), FCN-51363-015 (U2-N), & FCN-51363-025 (U2-S)

ESSENTIAL SERVICE WATER SYSTEM MALFUNCTION ANALYSIS

COMPONENT		MALFUNCTION	COMMENTS AND CONSEQUENCES
1.	Essential service water pumps	Rupture of a pump casing	Isolate pump and start a redundant pump. Minimum requirements need only two out of four pumps.
2.	Essential service water pumps	Pump fails to start	One operating pump will supply sufficient flow for one operating Unit. Redundancy is sufficient to provide ample flow for any condition.
3.	Essential service water pump	Stop valve on discharge line closed or check valve sticks closed	The stop valve and the check valve will be checked by periodic operation of the off-duty pumps during normal operation.
4.	Essential service water pump strainer	Strainer casing rupture	Isolate and valve in spare train.
5.	Essential service water pump strainer vent or drain valve.	Left open	This will be prevented by prestartup and operational checks. On the out-of-service strainer, such a situation would be assessed during periodic checks.

CONTROL ROOM VENTILATION SYSTEM MALFUNCTION ANALYSIS

COMPONENT		MALFUNCTION	COMMENTS AND CONSEQUENCES
1.	Normal Intake Dampers	Failure to close	Two dampers in series ensure isolation of outside air normal intake.
2.	Pressurization/Cleanup Intake Dampers	Failure to open	Parallel dampers ensure outside air intake opens.
		Fully open or multiple dampers open	One damper partially open is correct alignment. A single failure may result in one damper fully open or 2 dampers partially open. Other failures are not considered credible.
3.	Pressurization/Cleanup Recirculation Damper	Failed to closed position	All air flowing through the filter is from the intake and air cleanup is limited to a single pass. Control Room dose consequences may increase. The position of the recirculation damper is administratively controlled so it does not need to change position in response to a radiological accident.


DWG. NO. FSAR FIG.9.2-1
TO AND FROM COMPONENT COOLING SYSTEM. SEE FIG.9.5-1
INSIDE REACTOR CONTAINMENT
OUTSIDE REACTOR CONTAINMENT
LETDOWN ORIFICES
LET DOWN HEAT EXCHANGER
FROM RESIDUAL HEAT REMOVAL SYSTEM. SEE FIG.9.3-1
TO DEMINERALIZERS SEE FIG.9.2-2
FAIL TO FILTER
FROM DEMINERALIZERS SEE FIG.9.2-2
LETDOWN LINE FROM LOOP 4 SEE FIG.4.2-1
NORMAL CHARGING LINE TO LOOP 4 COLD LEG. SEE FIG.4.2-1
AUX SPRAY TO REACTOR COOLNAT SYSTEM PRESSURIZER. SEE FIG.4.2-1A
ALT. CHARGING LINE TO LOOP 1 COLD LEG. SEE FIG.4.2-1
REGENERATIVE HEAT EXCHANGER
TO PRESSURIZER RELIEF TANK SEE FIG.4.2-1A
TO REACTOR COOLANT DRAIN TANK SEE FIG.11.1-1
TO AND FROM COMPONENT COOLING SYSTEM. SEE FIG.9.5-1
FAIL TO RCDT
EXCESS LETDOWN FROM LOOP 4 SEE FIG.4.2-1
EXCESS LET DOWN HEAT EXCHANGER
FROM REACTOR COOLANT PUMP SEAL WATER CIRCUIT (4 PUMPS) SEE FIG.4.2-1A
TO REACTOR COOLANT DRAIN TANK SEE FIG. 11.1-1
INSIDE MISSILE BARRIER
OUTSIDE MISSILE BARRIER
FROM SAMPLING SYSTEM SEE FIG.9.6-1
TO DRAIN HEADER SEE FIG. 11.1-1
REACTOR COOLANT FILTER
TO HOLD-UP TANKS SEE FIG.9.2-4
TO VENT HEADER SEE FIG.11.1-1
TO AUTOMATIC GAS ANALYZER SEE FIG.11.1-4
UNIT 2
HYDROGEN SUPPLY FROM GENERATOR HYDROGEN STORAGE
UNIT 1
HYDROGEN SUPPLY FROM GENERATOR HYDROGEN STORAGE
24.0
NITROGEN SUPPLY SEE FIG.11.1-4
FAIL TO VCT
TO HOLD-UP TANKS SEE FIG.9.2-4
TO AND FROM COMPONENT COOLING SYSTEM. SEE FIG.9.5-1
SEAL WATER FILTER
SEAL WATER HEAT EXCHANGER
VOLUME CONTROL TANK
NOTE:
SYSTEM LINEUP SHOWN FOR NORMAL OPERATION. BORIC ACID MAKE-UP SYSTEM IN STANDBY ALIGNMENT. VALVE POSITIONS ARE NOT GOVERNED BY THE UFSAR FIGURES BUT RATHER BY STATION PROCEDURES (EXCEPT VALVES WITH LOCKED OR SEALED, AND FAILURE POSITIONS AS INDICATED).
REFERENCES:
OP-1 (2)-5129
OP-1 (2)-5129A
FROM BORIC ACID BLENDER SEE FIG. 9.2-3
FROM BORIC ACID BATCHING TANK SEE FIG. 9.2-3
TO HOLD-UP & BART TANKS SEE FIG. 9.2-4
FROM CONCENTRATES HOLDING TANK TRANSFER PUMPS SEE FIG. 9.2-3
MAKE-UP TO REFUELING WATER STORAGE TANK SEE FIG.6.3-1
TO SUMP TANK THRU DRAIN HEADER SEE FIG.11.1-1
FROM SAMPLING SYS
TO WASTE HOLD-UP TANK THRU DRAIN HEADER SEE FIG.11.1-1
TO PRESSURIZER RELIEF TANK SEE FIG. 4.2-1A
RECIPROCATING CHARGING PUMP
PULSATION DAMPER
FROM REFUELING WATER STORAGE TANK. SEE FIG.6.3-1
INSIDE REACTOR CONTAINMENT
OUTSIDE REACTOR CONTAINMENT
TO BORON INJECTION TANK SEE FIG.6.2-1
SHARED DEVICE
CHARGING PUMP CROSS-TIE TO/FROM OTHER UNIT
COMPONENT COOLING WATER SEE FIG.9.5-1
CENTRIFUGAL CHARGING PUMP
FROM BORIC ACID SYSTEM SEE FIG.9.2-3
FROM PRIMARY WATER & BORIC ACID TANKS SEE FIG.9.2-3
COMPONENT COOLING WATER TO & FROM BEARING AND GEAR OIL COOLERS SEE FIG.9.5-1
FROM CHEMICAL MIXING TANK
TO REACTOR COOLANT PUMP SEAL WATER CIRCUIT (4 PUMPS) SEE FIG.4.2-1A
DRAIN TO SUMP TANK THRU DRAIN HEADER SEE FIG.11.1-1
SEAL WATER INJECTION FILTERS
(ONE FILTER NORMALLY IN SERVICE.)
TO BORON INJECTION TANK SEE FIG.6.2-1
CENTRIFUGAL CHARGING PUMP
COMPONENT COOLING WATER TO & FROM BEARING AND GEAR OIL COOLERS SEE FIG.9.5-1
FROM RESIDUAL HEAT EXCHANGER SEE FIG. 6.2-1A
BACK UP CROSS-TIE TO & FROM SAFETY INJECTION PUMPS SUCTION HEADER SEE FIG.6.2-1
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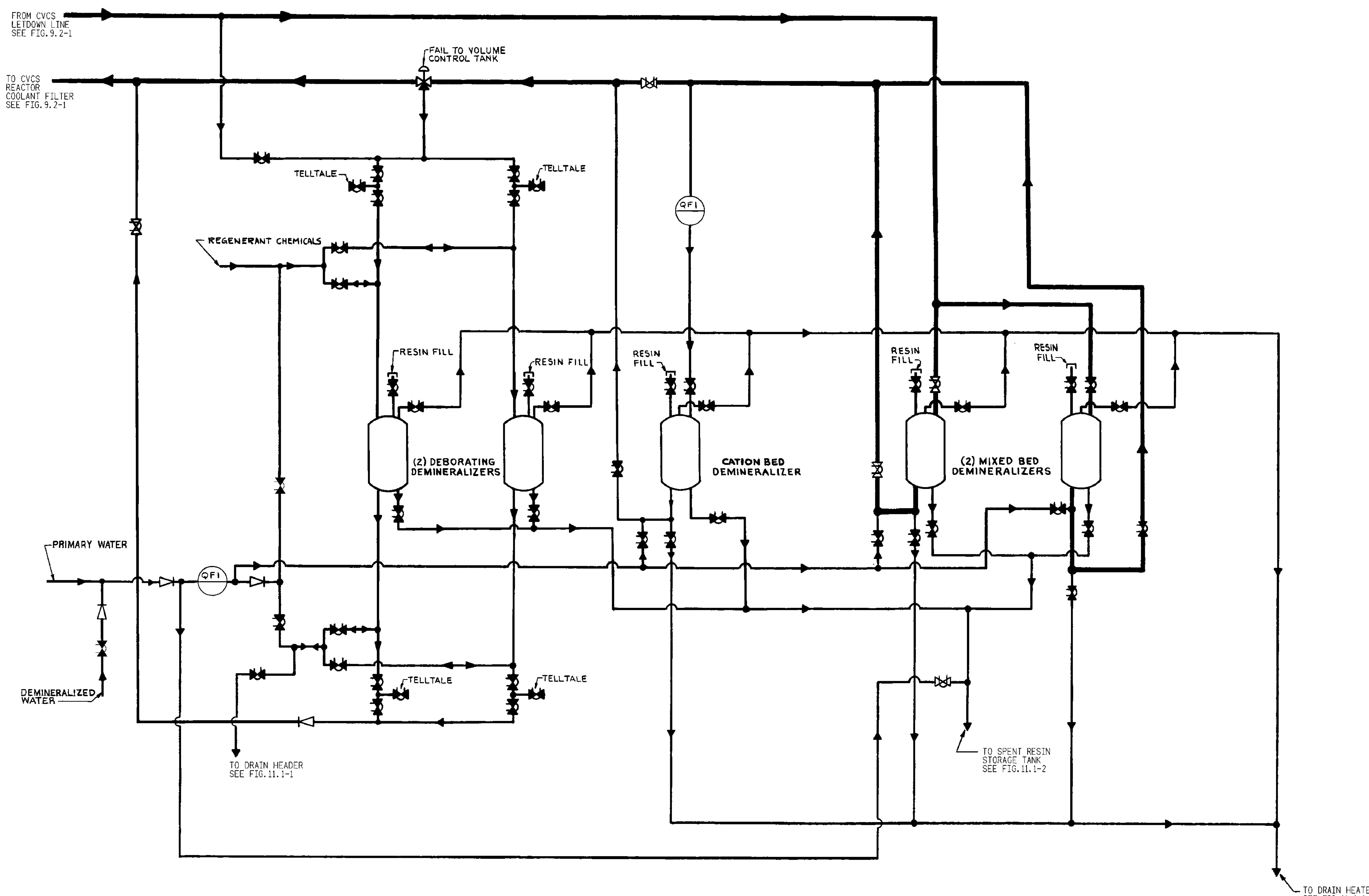
INDIANA MICHIGAN POWER COMPANY
DONALD C. COOK
NUCLEAR PLANT
BRIDGMAN MICHIGAN
CVCS - REACTOR LETDOWN & CHARGING UNITS NO.1 OR 2
DWG. NO. FSAR FIG. 9.2-1
ARCH | ELEC | MECH | STR
DESIGN ENGINEERING DIVISION
AEP AMERICAN ELECTRIC POWER
AEP SERVICE CORP. 1 RIVERSIDE PLAZA COLUMBUS, OH 43215



DWG. NO. FSAR FIG. 9.2-2

NOTE:

LINEUP SHOWN HAS ONE MIXED BED DEMINERALIZER IN SERVICE, OTHERS IN STANDBY. OTHER LINEUPS ARE ALLOWED IN ORDER TO CHANGE, SWITCH, FILL OR DRAIN DEMINERALIZERS.
VALVE POSITIONS ARE NOT GOVERNED BY THE UFSAR FIGURES BUT RATHER BY STATION PROCEDURES (EXCEPT VALVES WITH LOCKED OR SEALED, AND FAILURE POSITIONS AS INDICATED).

REFERENCES:

OP-1 (2)-5130

DATE	NO.	DESCRIPTION	APPD.
	163	REVISED PER UCR 98-UFSAR-0589, 99-UFSAR-1182, 99-UFSAR-1185	

REVISIONS

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INDIANA MICHIGAN POWER COMPANY

DONALD C. COOK

NUCLEAR PLANT

BRIDGMAN MICHIGAN

CVCS REACTOR COOLANT DEMINERALIZATION
UNITS NO. 1 OR 2

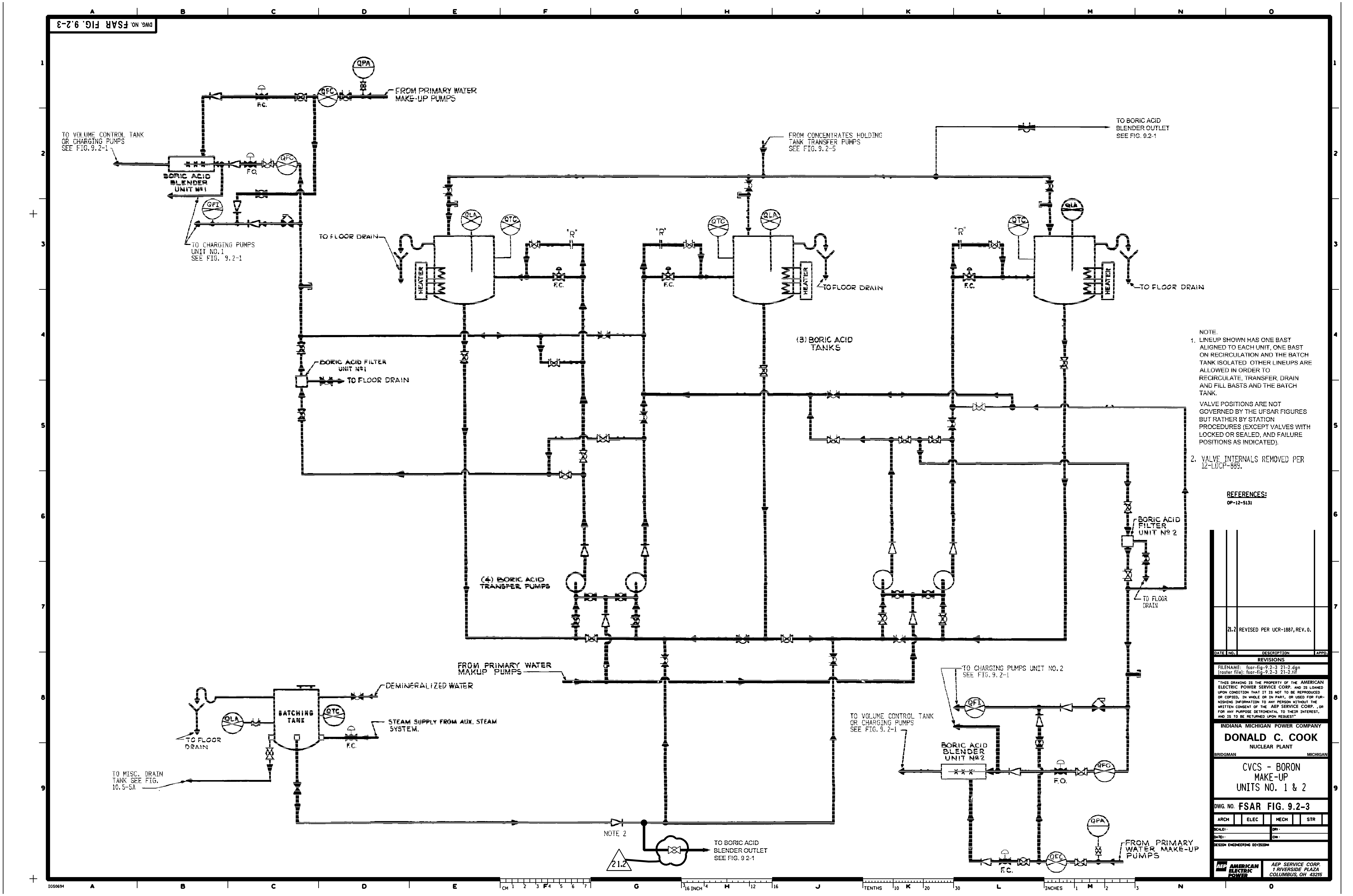
DWG. NO. **FSAR FIG. 9.2 - 2**

ARCH		ELEC		MECH		STR	

SCALE: DR:
DATE: CH:
DESIGN ENGINEERING DIVISION:

AMERICAN ELECTRIC POWER

AEP SERVICE CORP.
1 RIVERSIDE PLAZA
COLUMBUS, OH 43215



DWG. NO. FSAR FIG. 9.2-3
FROM PRIMARY WATER MAKE-UP PUMPS
QPA
QFC
F.C.
F.O.
TO VOLUME CONTROL TANK OR CHARGING PUMPS SEE FIG. 9.2-1
BORIC ACID BLENDER UNIT Nº1
QFI
TO CHARGING PUMPS UNIT NO.1 SEE FIG. 9.2-1
FROM CONCENTRATES HOLDING TANK TRANSFER PUMPS SEE FIG. 9.2-5
TO BORIC ACID BLENDER OUTLET SEE FIG. 9.2-1
TO FLOOR DRAIN
QLA
QTC
"R"
HEATER
(3) BORIC ACID TANKS
BORIC ACID FILTER UNIT Nº1
TO FLOOR DRAIN
NOTE:
1. LINEUP SHOWN HAS ONE BAST ALIGNED TO EACH UNIT, ONE BAST ON RECIRCULATION AND THE BATCH TANK ISOLATED. OTHER LINEUPS ARE ALLOWED IN ORDER TO RECIRCULATE, TRANSFER, DRAIN AND FILL BASTS AND THE BATCH TANK.
VALVE POSITIONS ARE NOT GOVERNED BY THE UFSAR FIGURES BUT RATHER BY STATION PROCEDURES (EXCEPT VALVES WITH LOCKED OR SEALED, AND FAILURE POSITIONS AS INDICATED).
2. VALVE INTERNALS REMOVED PER 12-LDCP-889.
REFERENCES:
OP-12-5131
BORIC ACID FILTER UNIT Nº 2
TO FLOOR DRAIN
(4) BORIC ACID TRANSFER PUMPS
FROM PRIMARY WATER MAKEUP PUMPS
DEMINERALIZED WATER
BATCHING TANK
STEAM SUPPLY FROM AUX. STEAM SYSTEM.
TO FLOOR DRAIN
TO MISC. DRAIN TANK SEE FIG. 10.5-5A
NOTE 2
21.2
TO BORIC ACID BLENDER OUTLET SEE FIG. 9.2-1
TO CHARGING PUMPS UNIT NO. 2 SEE FIG. 9.2-1
TO VOLUME CONTROL TANK OR CHARGING PUMPS SEE FIG. 9.2-1
BORIC ACID BLENDER UNIT Nº2
FROM PRIMARY WATER MAKE-UP PUMPS
21.2 REVISED PER UCR-1887, REV. 0.
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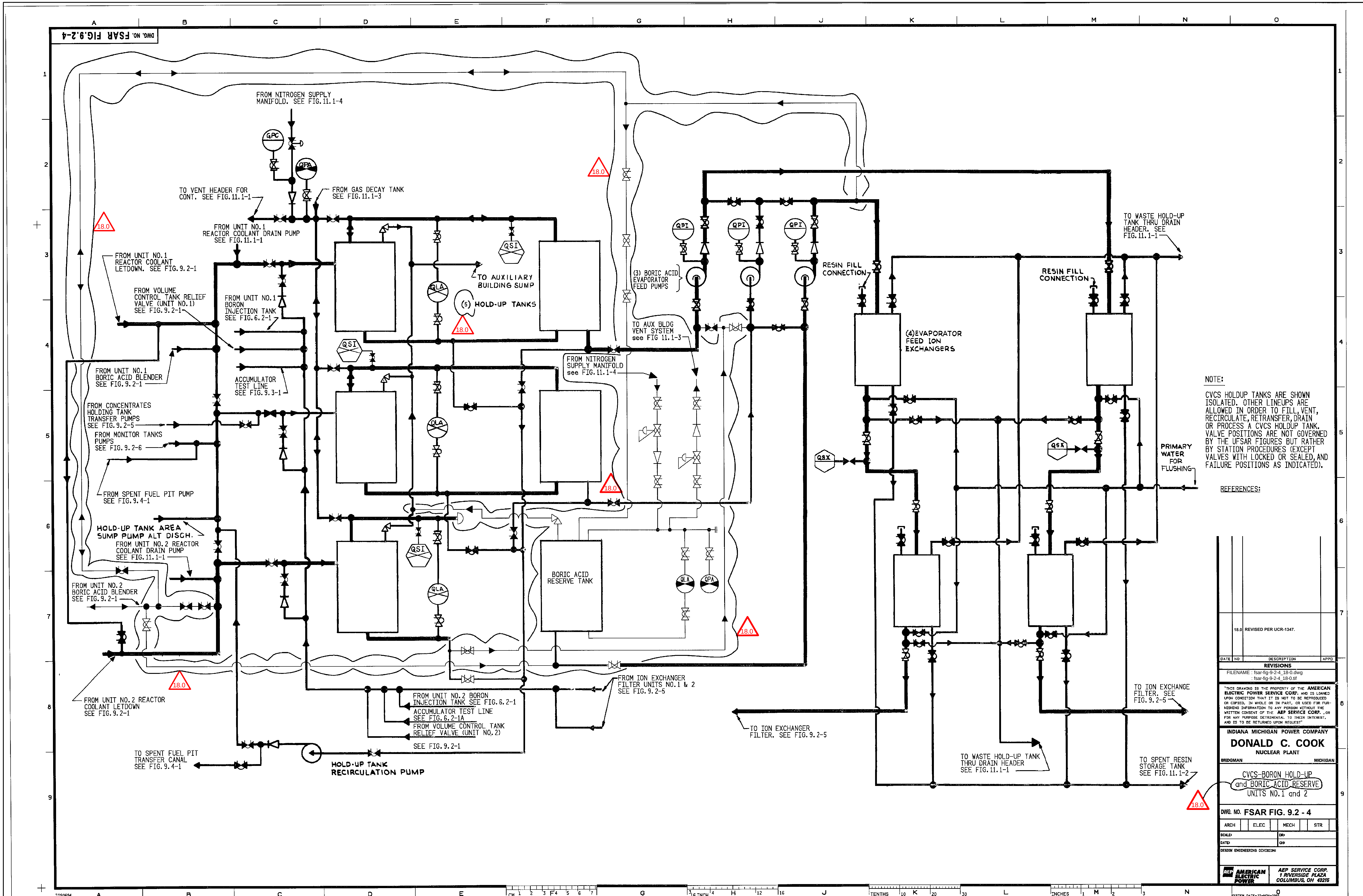
INDIANA MICHIGAN POWER COMPANY
DONALD C. COOK
NUCLEAR PLANT
BRIDGMAN MICHIGAN
CVCS - BORON MAKE-UP UNITS NO. 1 & 2
DWG. NO. FSAR FIG. 9.2-3
ARCH ELEC MECH STR
SCALE: DR:
DATE: CK:
DESIGN ENGINEERING DIVISION
AMERICAN ELECTRIC POWER
AEP SERVICE CORP. 1 RIVERSIDE PLAZA COLUMBUS, OH 43215


DWG. NO. FSAR FIG. 9.2-4
FROM NITROGEN SUPPLY MANIFOLD. SEE FIG. 11.1-4
GPC
QPA
TO VENT HEADER FOR CONT. SEE FIG. 11.1-1
FROM GAS DECAY TANK SEE FIG. 11.1-3
FROM UNIT NO. 1 REACTOR COOLANT DRAIN PUMP SEE FIG. 11.1-1
FROM UNIT NO. 1 REACTOR COOLANT LETDOWN. SEE FIG. 9.2-1
FROM VOLUME CONTROL TANK RELIEF VALVE (UNIT NO. 1) SEE FIG. 9.2-1
FROM UNIT NO. 1 BORON INJECTION TANK SEE FIG. 6.2-1
TO AUXILIARY BUILDING SUMP
(5) HOLD-UP TANKS
QSI
QLA
FROM UNIT NO. 1 BORIC ACID BLENDER SEE FIG. 9.2-1
ACCUMULATOR TEST LINE SEE FIG. 9.3-1
FROM CONCENTRATES HOLDING TANK TRANSFER PUMPS SEE FIG. 9.2-5
FROM MONITOR TANKS PUMPS SEE FIG. 9.2-6
FROM SPENT FUEL PIT PUMP SEE FIG. 9.4-1
HOLD-UP TANK AREA SUMP PUMP ALT DISCH.
FROM UNIT NO. 2 REACTOR COOLANT DRAIN PUMP SEE FIG. 11.1-1
FROM UNIT NO. 2 BORIC ACID BLENDER SEE FIG. 9.2-1
FROM UNIT NO. 2 REACTOR COOLANT LETDOWN SEE FIG. 9.2-1
TO SPENT FUEL PIT TRANSFER CANAL SEE FIG. 9.4-1
HOLD-UP TANK RECIRCULATION PUMP
FROM UNIT NO. 2 BORON INJECTION TANK SEE FIG. 6.2-1
ACCUMULATOR TEST LINE SEE FIG. 6.2-1A
FROM VOLUME CONTROL TANK RELIEF VALVE (UNIT NO. 2) SEE FIG. 9.2-1
BORIC ACID RESERVE TANK
FROM ION EXCHANGER FILTER UNITS NO. 1 & 2 SEE FIG. 9.2-5
FROM NITROGEN SUPPLY MANIFOLD see FIG. 11.1-4
TO AUX BLDG VENT SYSTEM see FIG 11.1-3
(3) BORIC ACID EVAPORATOR FEED PUMPS
QPI
RESIN FILL CONNECTION
(4) EVAPORATOR FEED ION EXCHANGERS
QSX
TO ION EXCHANGER FILTER. SEE FIG. 9.2-5
TO WASTE HOLD-UP TANK THRU DRAIN HEADER. SEE FIG. 11.1-1
PRIMARY WATER FOR FLUSHING
TO ION EXCHANGE FILTER. SEE FIG. 9.2-5
TO WASTE HOLD-UP TANK THRU DRAIN HEADER SEE FIG. 11.1-1
TO SPENT RESIN STORAGE TANK SEE FIG. 11.1-2
18.0
NOTE:
CVCS HOLDUP TANKS ARE SHOWN ISOLATED. OTHER LINEUPS ARE ALLOWED IN ORDER TO FILL, VENT, RECIRCULATE, RETRANSFER, DRAIN OR PROCESS A CVCS HOLDUP TANK. VALVE POSITIONS ARE NOT GOVERNED BY THE UFSAR FIGURES BUT RATHER BY STATION PROCEDURES (EXCEPT VALVES WITH LOCKED OR SEALED, AND FAILURE POSITIONS AS INDICATED).
REFERENCES:
18.0 REVISED PER UCR-1347.
DATE NO DESCRIPTION APPD
REVISIONS
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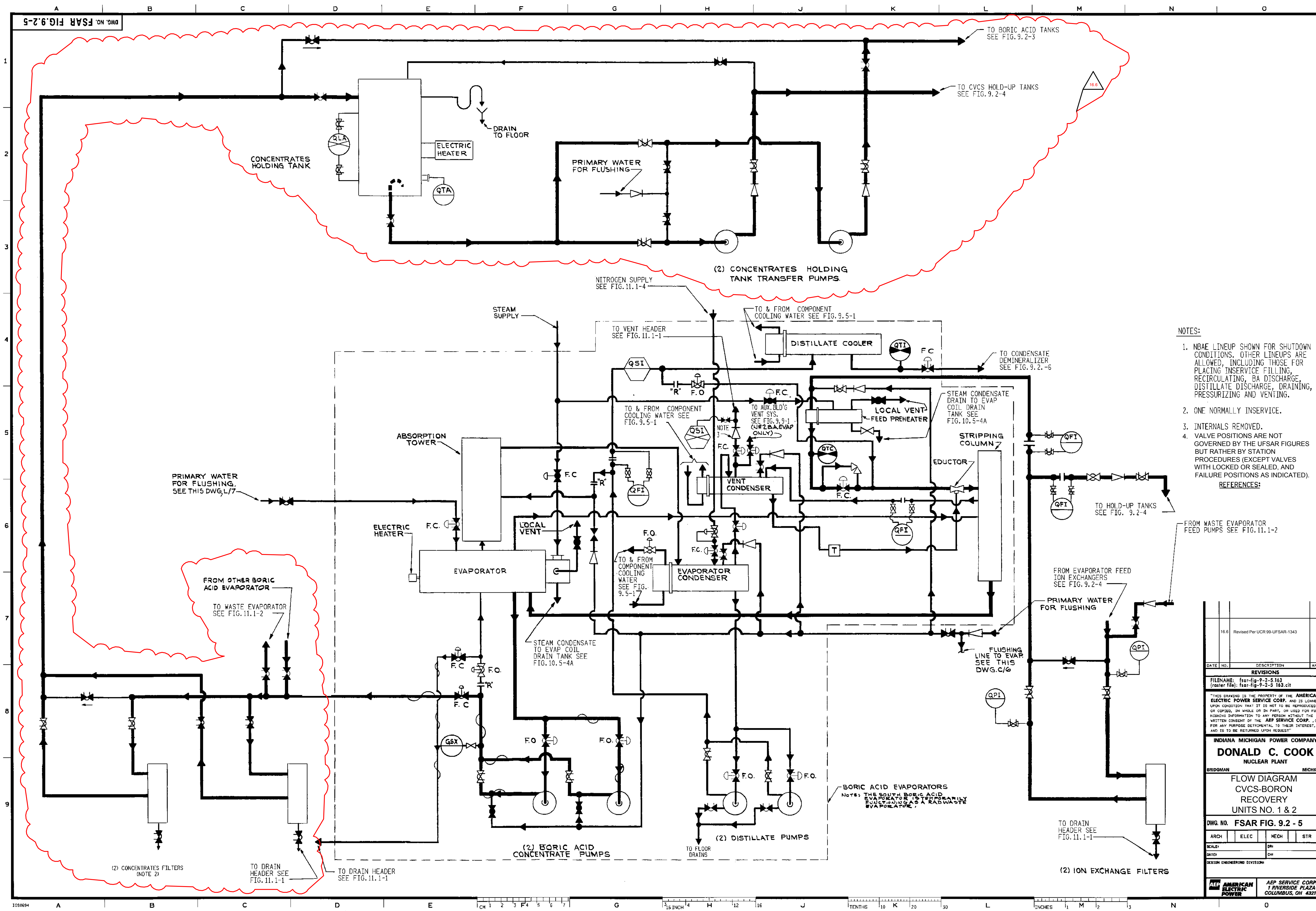
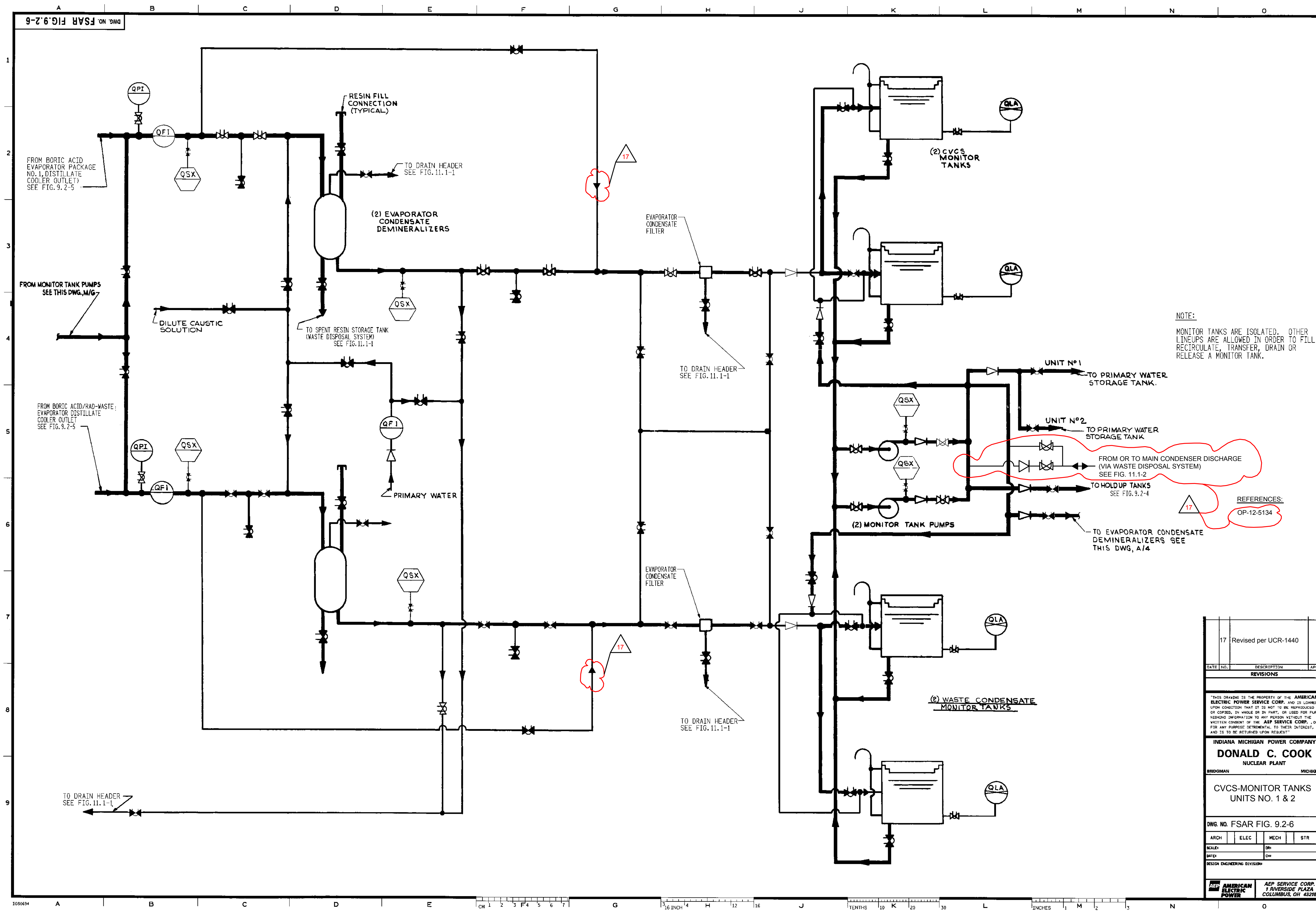
INDIANA MICHIGAN POWER COMPANY
DONALD C. COOK
NUCLEAR PLANT
BRIDGMAN MICHIGAN
CVCS-BORON HOLD-UP and BORIC ACID RESERVE UNITS NO. 1 and 2
DWG. NO. FSAR FIG. 9.2 - 4
ARCH ELEC MECH STR
SCALE: DR:
DATE: CH:
DESIGN ENGINEERING DIVISION:
AEP AMERICAN ELECTRIC POWER
AEP SERVICE CORP. 1 RIVERSIDE PLAZA COLUMBUS, OH 43215
SYSTEM DATE-22-NOV-2002
SYSTEM TIME-13 00



DWG. NO. FSAR FIG. 9.2-5
TO BORIC ACID TANKS SEE FIG. 9.2-3
TO CVCS HOLD-UP TANKS SEE FIG. 9.2-4
DRAIN TO FLOOR
ELECTRIC HEATER
CONCENTRATES HOLDING TANK
QLA
QTA
PRIMARY WATER FOR FLUSHING
(2) CONCENTRATES HOLDING TANK TRANSFER PUMPS.
NITROGEN SUPPLY SEE FIG. 11.1-4
STEAM SUPPLY
TO VENT HEADER SEE FIG. 11.1-1
TO & FROM COMPONENT COOLING WATER SEE FIG. 9.5-1
DISTILLATE COOLER
QTI
F.C
TO CONDENSATE DEMINERALIZER SEE FIG. 9.2.-6
QSI
"R"
F.O
F.C.
TO & FROM COMPONENT COOLING WATER SEE FIG. 9.5-1
TO AUX. BLD'G VENT SYS. SEE FIG. 9.9-1 (N°2 BA EVAP ONLY)
NOTE 3
LOCAL VENT
FEED PREHEATER
STEAM CONDENSATE DRAIN TO EVAP COIL DRAIN TANK SEE FIG. 10.5-4A
ABSORPTION TOWER
QSI
QTC
STRIPPING COLUMN
EDUCTOR
QFI
PRIMARY WATER FOR FLUSHING. SEE THIS DWG. L/7
VENT CONDENSER
TO HOLD-UP TANKS SEE FIG. 9.2-4
ELECTRIC HEATER
LOCAL VENT
FROM WASTE EVAPORATOR FEED PUMPS SEE FIG. 11.1-2
EVAPORATOR
TO & FROM COMPONENT COOLING WATER SEE FIG. 9.5-1
EVAPORATOR CONDENSER
T
FROM EVAPORATOR FEED ION EXCHANGERS SEE FIG. 9.2-4
PRIMARY WATER FOR FLUSHING
FROM OTHER BORIC ACID EVAPORATOR
TO WASTE EVAPORATOR SEE FIG. 11.1-2
STEAM CONDENSATE TO EVAP COIL DRAIN TANK SEE FIG. 10.5-4A
FLUSHING LINE TO EVAP. SEE THIS DWG. C/6
QPI
QSX
BORIC ACID EVAPORATORS
NOTE: THE SOUTH BORIC ACID EVAPORATOR IS TEMPORARILY FUNCTIONING AS A RADWASTE EVAPORATOR.
(2) BORIC ACID CONCENTRATE PUMPS
(2) DISTILLATE PUMPS
TO FLOOR DRAINS
(2) CONCENTRATES FILTERS (NOTE 2)
TO DRAIN HEADER SEE FIG. 11.1-1
TO DRAIN HEADER SEE FIG. 11.1-1
TO DRAIN HEADER SEE FIG. 11.1-1
(2) ION EXCHANGE FILTERS
NOTES:
1. NBAE LINEUP SHOWN FOR SHUTDOWN CONDITIONS. OTHER LINEUPS ARE ALLOWED, INCLUDING THOSE FOR PLACING INSERVICE FILLING, RECIRCULATING, BA DISCHARGE, DISTILLATE DISCHARGE, DRAINING, PRESSURIZING AND VENTING.
2. ONE NORMALLY INSERVICE.
3. INTERNALS REMOVED.
4. VALVE POSITIONS ARE NOT GOVERNED BY THE UFSAR FIGURES BUT RATHER BY STATION PROCEDURES (EXCEPT VALVES WITH LOCKED OR SEALED, AND FAILURE POSITIONS AS INDICATED).
REFERENCES:
16.6 Revised Per UCR 99-UFSAR-1343
DATE NO. DESCRIPTION APPD.
REVISIONS
FILENAME: fsar-fig-9-2-5.163 (raster file): fsar-fig-9-2-5 163.cit
INDIANA MICHIGAN POWER COMPANY
DONALD C. COOK
NUCLEAR PLANT
BRIDGMAN MICHIGAN
FLOW DIAGRAM
CVCS-BORON
RECOVERY
UNITS NO. 1 & 2
DWG. NO. FSAR FIG. 9.2 - 5
ARCH ELEC MECH STR
AEP SERVICE CORP. 1 RIVERSIDE PLAZA COLUMBUS, OH 43215
AMERICAN ELECTRIC POWER



DWG. NO. FSAR FIG. 9.2-6
FROM BORIC ACID EVAPORATOR PACKAGE NO. 1, DISTILLATE COOLER OUTLET) SEE FIG. 9.2-5
RESIN FILL CONNECTION (TYPICAL)
TO DRAIN HEADER SEE FIG. 11.1-1
(2) EVAPORATOR CONDENSATE DEMINERALIZERS
EVAPORATOR CONDENSATE FILTER
(2) CVCS MONITOR TANKS
FROM MONITOR TANK PUMPS SEE THIS DWG. M/G
DILUTE CAUSTIC SOLUTION
TO SPENT RESIN STORAGE TANK (WASTE DISPOSAL SYSTEM) SEE FIG. 11.1-1
NOTE:
MONITOR TANKS ARE ISOLATED. OTHER LINEUPS ARE ALLOWED IN ORDER TO FILL, RECIRCULATE, TRANSFER, DRAIN OR RELEASE A MONITOR TANK.
UNIT N° 1
TO PRIMARY WATER STORAGE TANK.
UNIT N° 2
TO PRIMARY WATER STORAGE TANK
FROM BORIC ACID/RAD-WASTE EVAPORATOR DISTILLATE COOLER OUTLET SEE FIG. 9.2-5
FROM OR TO MAIN CONDENSER DISCHARGE (VIA WASTE DISPOSAL SYSTEM) SEE FIG. 11.1-2
TO HOLDUP TANKS SEE FIG. 9.2-4
REFERENCES:
OP-12-5134
PRIMARY WATER
(2) MONITOR TANK PUMPS
TO EVAPORATOR CONDENSATE DEMINERALIZERS SEE THIS DWG. A/4
(2) WASTE CONDENSATE MONITOR TANKS
17 Revised per UCR-1440
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INDIANA MICHIGAN POWER COMPANY
DONALD C. COOK
NUCLEAR PLANT
BRIDGMAN MICHIGAN
CVCS-MONITOR TANKS UNITS NO. 1 & 2
DWG. NO. FSAR FIG. 9.2-6
ARCH ELEC MECH STR
DESIGN ENGINEERING DIVISION
AMERICAN ELECTRIC POWER
AEP SERVICE CORP. 1 RIVERSIDE PLAZA COLUMBUS, OH 43215

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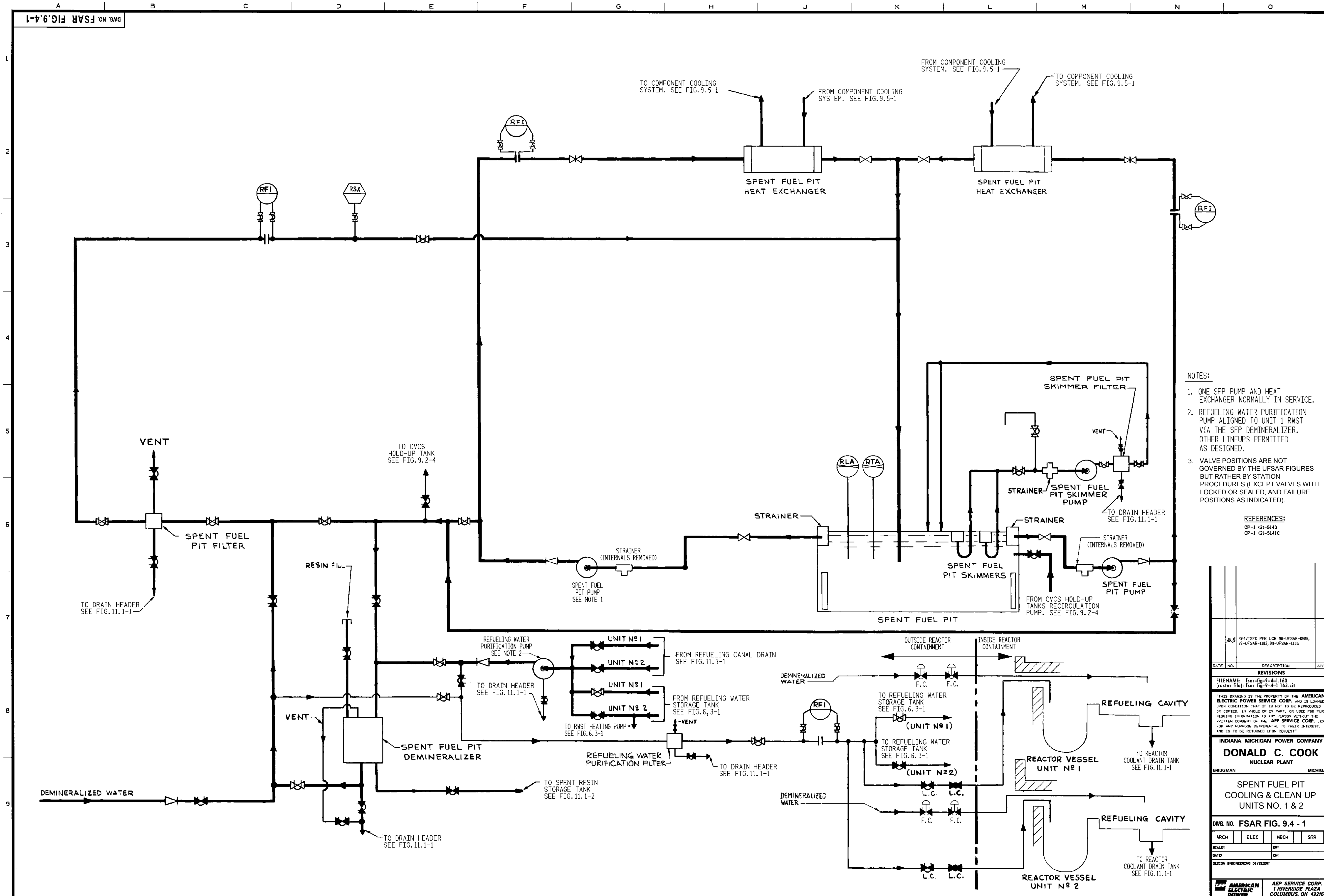
DWG. NO. FSAR FIG.9.4-1
TO COMPONENT COOLING SYSTEM. SEE FIG.9.5-1
FROM COMPONENT COOLING SYSTEM. SEE FIG.9.5-1
FROM COMPONENT COOLING SYSTEM. SEE FIG.9.5-1
TO COMPONENT COOLING SYSTEM. SEE FIG.9.5-1
RFI
RSX
SPENT FUEL PIT HEAT EXCHANGER
SPENT FUEL PIT HEAT EXCHANGER
VENT
TO CVCS HOLD-UP TANK SEE FIG.9.2-4
SPENT FUEL PIT FILTER
TO DRAIN HEADER SEE FIG.11.1-1
RESIN FILL
STRAINER (INTERNALS REMOVED)
SPENT FUEL PIT PUMP SEE NOTE 1
STRAINER
RLA
RTA
SPENT FUEL PIT SKIMMER FILTER
VENT
STRAINER
SPENT FUEL PIT SKIMMER PUMP
TO DRAIN HEADER SEE FIG.11.1-1
STRAINER
STRAINER (INTERNALS REMOVED)
SPENT FUEL PIT SKIMMERS
SPENT FUEL PIT PUMP
FROM CVCS HOLD-UP TANKS RECIRCULATION PUMP. SEE FIG.9.2-4
SPENT FUEL PIT
NOTES:
1. ONE SFP PUMP AND HEAT EXCHANGER NORMALLY IN SERVICE.
2. REFUELING WATER PURIFICATION PUMP ALIGNED TO UNIT 1 RWST VIA THE SFP DEMINERALIZER. OTHER LINEUPS PERMITTED AS DESIGNED.
3. VALVE POSITIONS ARE NOT GOVERNED BY THE UFSAR FIGURES BUT RATHER BY STATION PROCEDURES (EXCEPT VALVES WITH LOCKED OR SEALED, AND FAILURE POSITIONS AS INDICATED).
REFERENCES:
OP-1 (2)-5143
OP-1 (2)-5141C
REFUELING WATER PURIFICATION PUMP SEE NOTE 2
UNIT Nº 1
UNIT Nº 2
UNIT Nº 1
UNIT Nº 2
FROM REFUELING CANAL DRAIN SEE FIG.11.1-1
FROM REFUELING WATER STORAGE TANK SEE FIG.6.3-1
TO DRAIN HEADER SEE FIG.11.1-1
TO RWST HEATING PUMP SEE FIG.6.3-1
VENT
VENT
SPENT FUEL PIT DEMINERALIZER
REFUELING WATER PURIFICATION FILTER
TO DRAIN HEADER SEE FIG.11.1-1
DEMINERALIZED WATER
TO SPENT RESIN STORAGE TANK SEE FIG.11.1-2
TO DRAIN HEADER SEE FIG.11.1-1
OUTSIDE REACTOR CONTAINMENT
INSIDE REACTOR CONTAINMENT
DEMINERALIZED WATER
F.C.
F.C.
TO REFUELING WATER STORAGE TANK SEE FIG.6.3-1
(UNIT Nº 1)
TO REFUELING WATER STORAGE TANK SEE FIG.6.3-1
(UNIT Nº 2)
L.C.
L.C.
DEMINERALIZED WATER
F.C.
F.C.
L.C.
L.C.
REFUELING CAVITY
TO REACTOR COOLANT DRAIN TANK SEE FIG.11.1-1
REACTOR VESSEL UNIT Nº 1
REFUELING CAVITY
TO REACTOR COOLANT DRAIN TANK SEE FIG.11.1-1
REACTOR VESSEL UNIT Nº 2
REVISED PER UCR 98-UFSAR-0586, 99-UFSAR-1182, 99-UFSAR-1185
DATE NO. DESCRIPTION APPD.
REVISIONS
FILENAME: fsar-fig-9-4-1.163
(raster file): fsar-fig-9-4-1 163.cit
INDIANA MICHIGAN POWER COMPANY
DONALD C. COOK
NUCLEAR PLANT
BRIDGMAN MICHIGAN
SPENT FUEL PIT COOLING & CLEAN-UP UNITS NO. 1 & 2
DWG. NO. FSAR FIG. 9.4 - 1
ARCH ELEC MECH STR
DESIGN ENGINEERING DIVISION
AMERICAN ELECTRIC POWER
AEP SERVICE CORP. 1 RIVERSIDE PLAZA COLUMBUS, OH 43215

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DWG. NO. FSAR FIG. 9.6-1
TO NUCLEAR SAMPLING ROOM
FROM STEAM GENERATOR BLOWDOWN SEE FIG. 10.2-1, 10.2-1B
DSR
F.C.
Unit 1 only
MSX
FROM MAIN STEAM LEAD SEE FIG. 10.2-1, 10.2-1B
FROM OTHER MAIN STEAM LEADS
TO STEAM GEN. BLOWDOWN TREATMENT PUMP SEE FIG. 11.5-1
SEE THIS DWG C/2
RADIATION MONITOR
DRA
TO FLR DRAIN
SHIELDING
TO SAMPLING SINK SEE SK "M-4"
FROM COMPONENT COOLING SEE FIG. 9.5-1
DPI
DFA
DTI
SAMPLE COOLING COIL TYPE (TYP)
TO COMPONENT COOLING SEE FIG. 9.5-1
SAMPLING RACK "SRD"
FROM DEMINERALIZED WATER FOR FLUSHING
TO BLDG. EXHAUST VENTILATION SYSTEM
SAMPLING SINK HOOD
(Typical for all sample Connections except Steam Generator Blowdown).
TO UNIT NO 2 SINK
UNIT NO 1 SINK
UNIT NO 2 SINK (SIMILAR TO UNIT NO 1)
DEMIN. WATER
TO DRAIN HEADER FOR CONT. SEE THIS DWG. ℄ L/9
SKETCH "M-4"
ENCLOSED SAMPLE SINK
REFERENCES:
OP-1 (2)-5141
OP-1 (2)-5141A
OP-1 (2)-5141B
OP-1 (2)-5141D
NOTE:
SAMPLING SYSTEM SHOWN IN STANDBY ALIGNMENT.
VALVE POSITIONS ARE NOT GOVERNED BY THE UFSAR FIGURES BUT RATHER BY STATION PROCEDURES (EXCEPT VALVES WITH LOCKED OR SEALED, AND FAILURE POSITIONS AS INDICATED).
ACCUMULATOR SAMPLES SEE FIG. 6.2-1A
ISX
FROM LETDOWN LINE AT DEMIN. INLET SEE FIG. 9.2-1
FROM LETDOWN LINE AT DEMIN. OUTLET SEE FIG. 9.2-1
QSX
19.3
TO SAMPLING SINK. SEE SK. "M-4"
NSX
FROM PRESSURIZER STEAM SPACE SEE FIG. 4.2-1
TO POST ACCIDENT SAMPLING SYSTEM SEE FIG. 9.6-2
FROM VOLUME CONTROL TANK GAS SPACE SEE FIG. 9.2-1
FROM PRESSURIZER LIQUID SPACE SEE FIG. 4.2-1
TO BLDG. EXHAUST VENT SYSTEM.
TO VOLUME CONTROL TANK SEE FIG. 9.2-1
TO VENT HDR. SEE FIG. 11.1-1
FROM UNIT NO 2
NFI
QFI
IFI
SEE THIS DWG C/9
CAP
TO CVCS HOLD-UP TANKS
DEMINERALIZED WATER FOR FLUSHING
FROM UNIT NO. 2
"POST ACCIDENT UNDILUTED REACTOR COOLANT SAMPLE"
SKETCH "O-7"
INSIDE REACTOR CONTAINMENT
OUTSIDE REACTOR CONTAINMENT
OUTSIDE SAMPLING ROOM
INSIDE SAMPLING ROOM
NTI
ITI
NPI
IPI
PRESSURIZER STEAM SAMPLE VESSEL
HOT LEG SAMPLE VESSEL
PRESSURIZER LIQUID SAMPLE VESSEL
VOLUME CONTROL TANK SAMPLE VESSEL
FROM HOT LEG LOOP 3 SEE FIG. 4.2-1
DECAY COIL
FROM HOT LEG LOOP 1 SEE FIG. 4.2-1
FROM RESIDUAL HEAT REMOVAL LOOP SEE FIG. 9.3-1
TO POST ACCIDENT SAMPLING SEE FIG. 9.6-2
TO POST ACCIDENT UNDILUTED REACTOR COOLANT SAMPLE SEE SK "O-7"
SAMPLE HEAT EXCHANGER COIL TYPE (TYPICAL)
TO COMPONENT COOLING WATER SYS SEE FIG. 9.5-1
FROM COMPONENT COOLING WATER SYS SEE FIG. 9.5-1
SAMPLING RACK "B"
SAMPLING RACK "C"
FROM NUCLEAR SAMPLING SINK SEE THIS DWG. ℄ N/4.
FROM UNIT NO 2
FROM "POST ACCIDENT SAMPLING SEE FIG. 9.6-2
TO DRAIN HEADER SEE FIG. 11.1-1
19.3 REVISED PER UCR-1746.
DATE NO. DESCRIPTION APPD.
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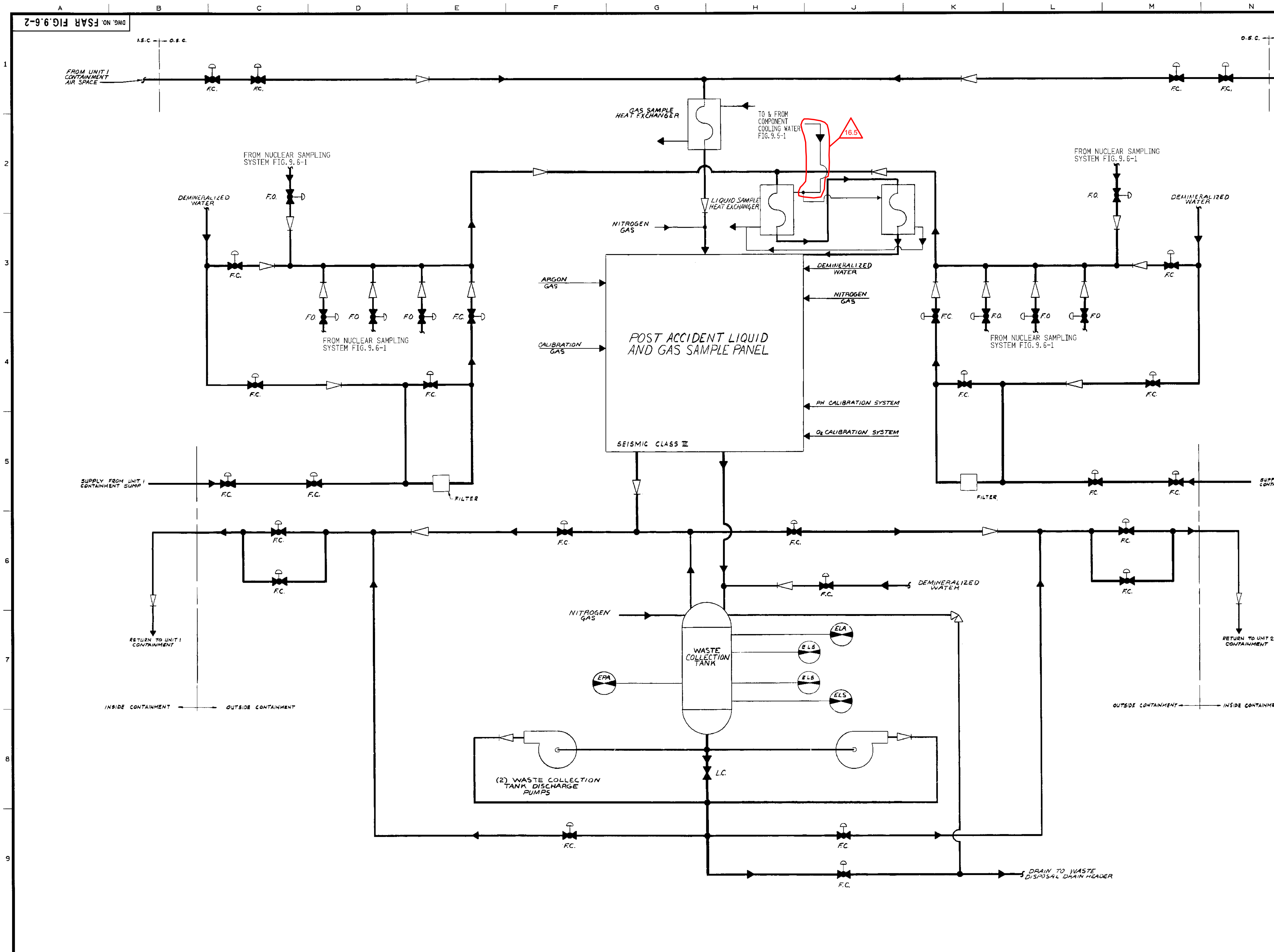
INDIANA MICHIGAN POWER COMPANY
DONALD C. COOK
NUCLEAR PLANT
BRIDGMAN MICHIGAN
SAMPLING
UNIT NO.1 OR 2
DWG. NO. FSAR FIG. 9.6-1
ARCH ELEC MECH STR
SCALE: DR:
DATE: CH:
DESIGN ENGINEERING DIVISION:
AEP AMERICAN ELECTRIC POWER
AEP SERVICE CORP. 1 RIVERSIDE PLAZA COLUMBUS, OH 43215



DWG. NO. FSAR FIG. 9.6-2

REFERENCES:
OP-1 (2)-5141D

NOTE:
VALVE POSITIONS ARE NOT GOVERNED BY THE UFSAR FIGURES BUT RATHER BY STATION PROCEDURES (EXCEPT VALVES WITH LOCKED OR SEALED, AND FAILURE POSITIONS AS INDICATED).

DATE	NO.	DESCRIPTION	APPD.
	16.5	Revised Per UCR 98-UFSAR-0793	

REVISIONS

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(raster file): fsar-fig-9-6-2 163.cit

INDIANA MICHIGAN POWER COMPANY

DONALD C. COOK
NUCLEAR PLANT

BRIDGMAN MICHIGAN

POST-ACCIDENT SAMPLING
UNITS NO. 1 AND 2

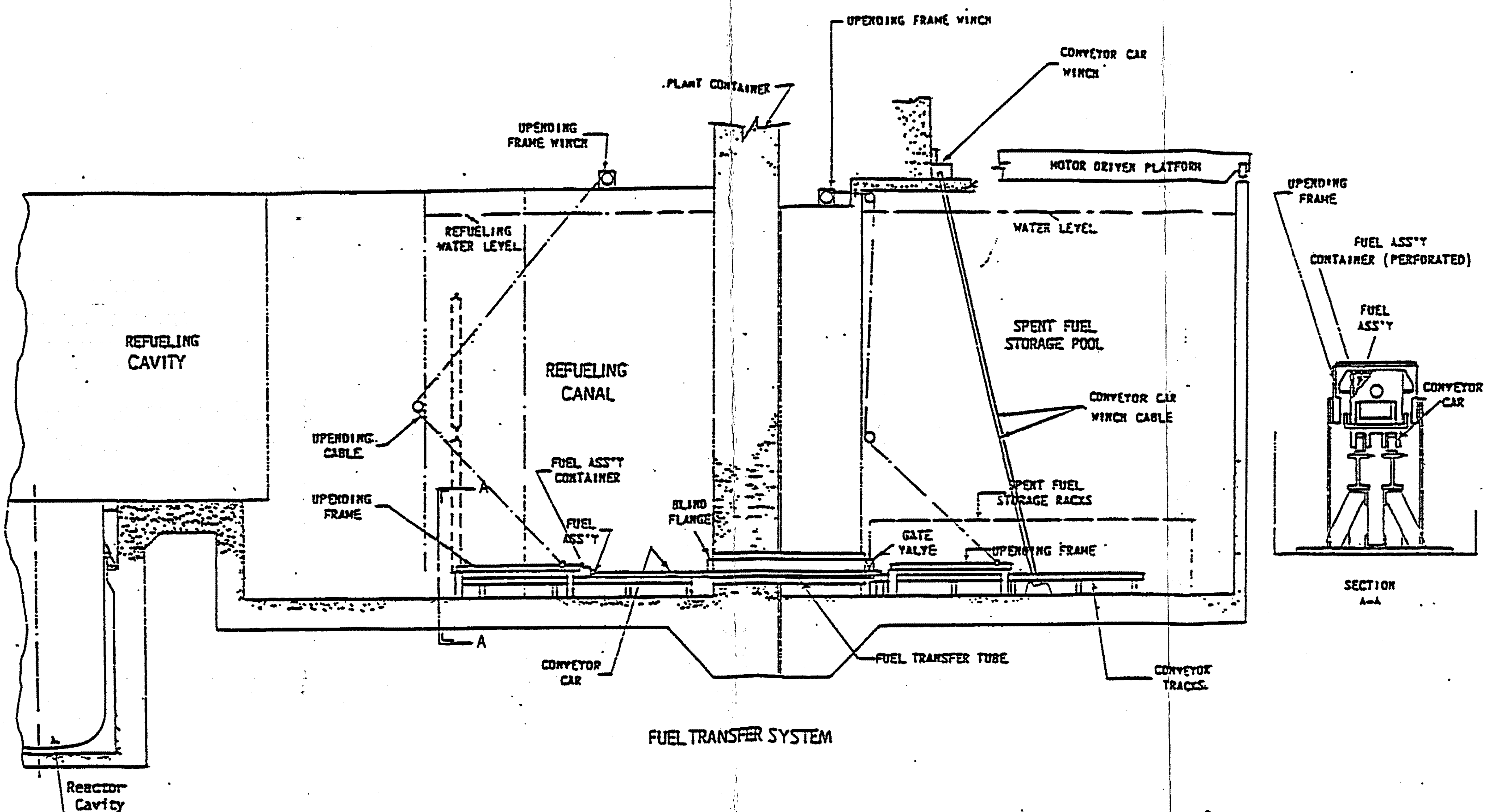
DWG. NO. FSAR FIG. 9.6 - 2

ARCH	ELEC	MECH	STR

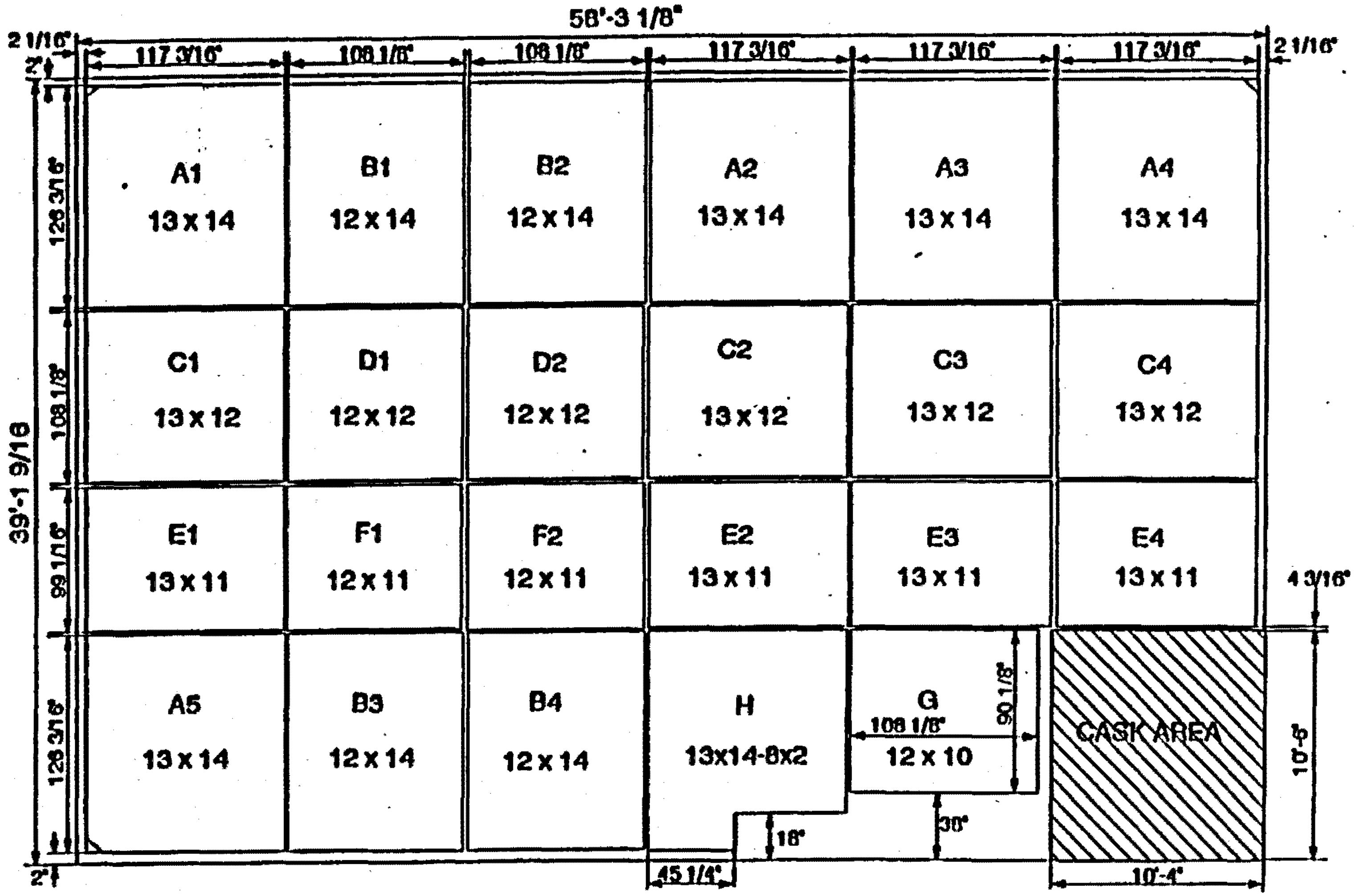
SCALE: DR:
DATE: CH:

DESIGN ENGINEERING DIVISION

AEP AMERICAN ELECTRIC POWER — AEP SERVICE CORP. 1 RIVERSIDE PLAZA COLUMBUS, OH 43215



TYPICAL FUEL TRANSFER SYSTEM
FIG. 9.7-1

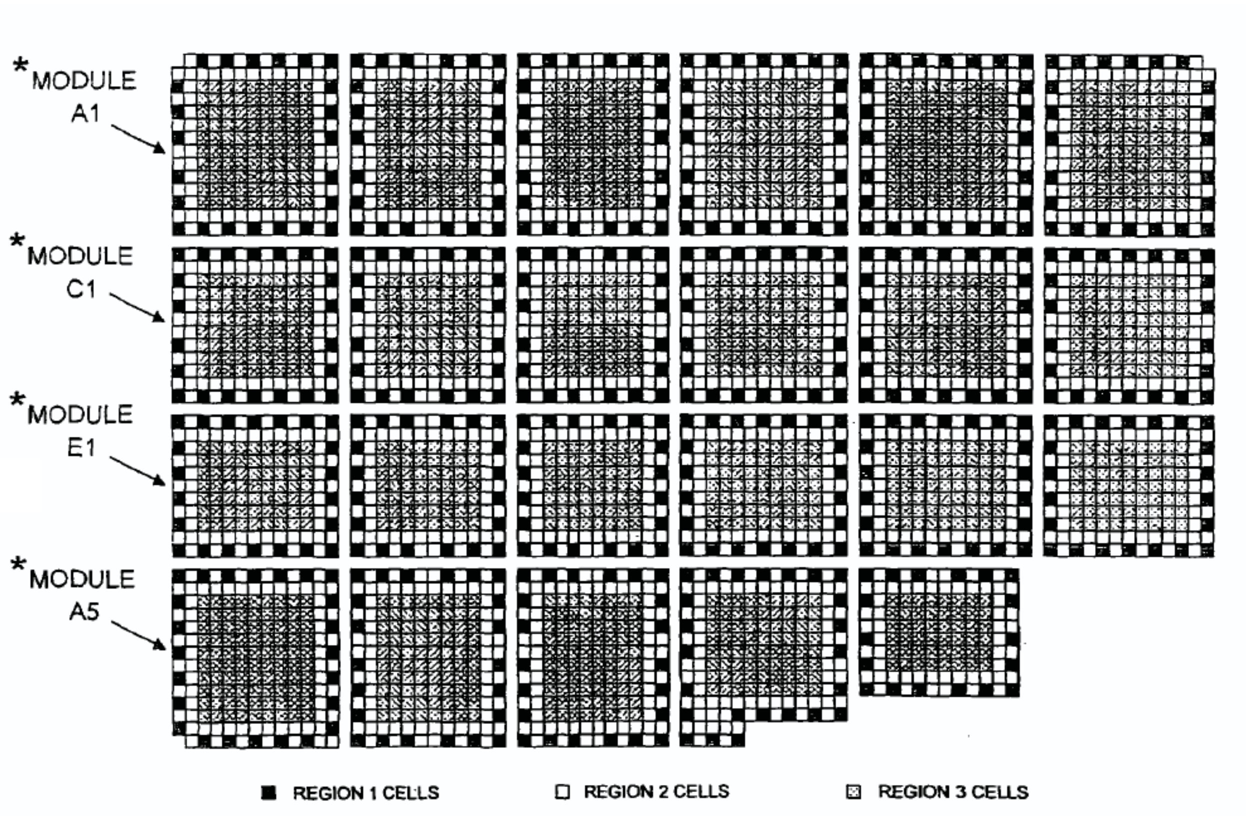


Typical Rack-to-Rack Gap: 2"

Total Storage: 3616 cells (include 3 triangular corner cells)

FIGURE 9.7-2: COOK SPENT FUEL POOL LAYOUT

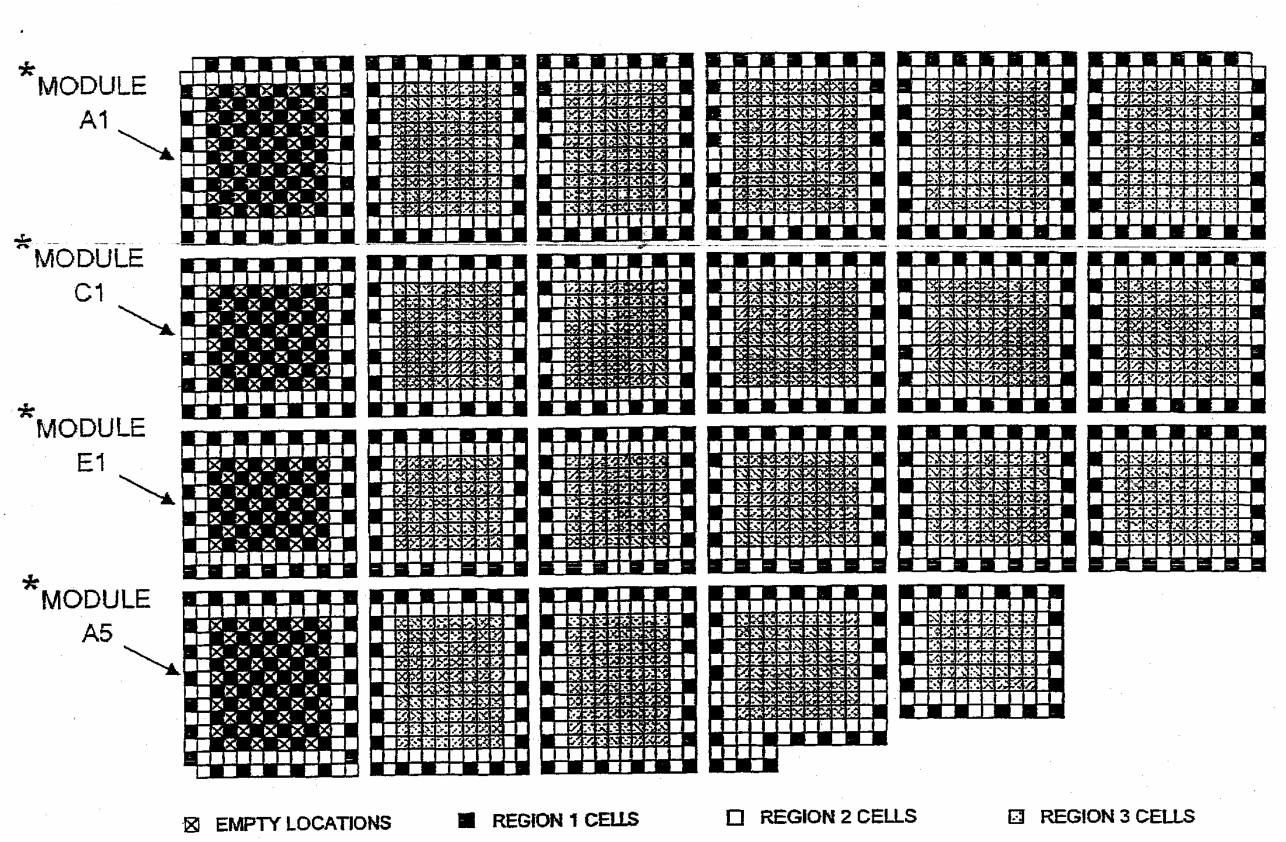
July 1994



* The storage pattern for any of these individual modules may be as shown in this figure or Figure 9.7-4.

Revision: **23**	Change Description: **UCR-1961**

AMERICAN ELECTRIC POWER COOK NUCLEAR PLANT NUCLEAR GENERATION GROUP BRIDGMAN, MICHIGAN	Title: **NORMAL STORAGE PATTERN (Mixed Three Zone)**	
	UFSAR Figure: **9.7-3**	Sheet 1 of 1



* The storage pattern for any of these individual modules may be as shown in this figure or Figure 9.7-3.

Revision: **23**	Change Description: **UCR-1961**

AMERICAN ELECTRIC POWER **COOK NUCLEAR PLANT** **NUCLEAR GENERATION GROUP** **BRIDGMAN, MICHIGAN**	Title: **INTERIM STORAGE PATTERN (Checkerboard)**	
	UFSAR Figure: **9.7-4**	Sheet 1 of 1

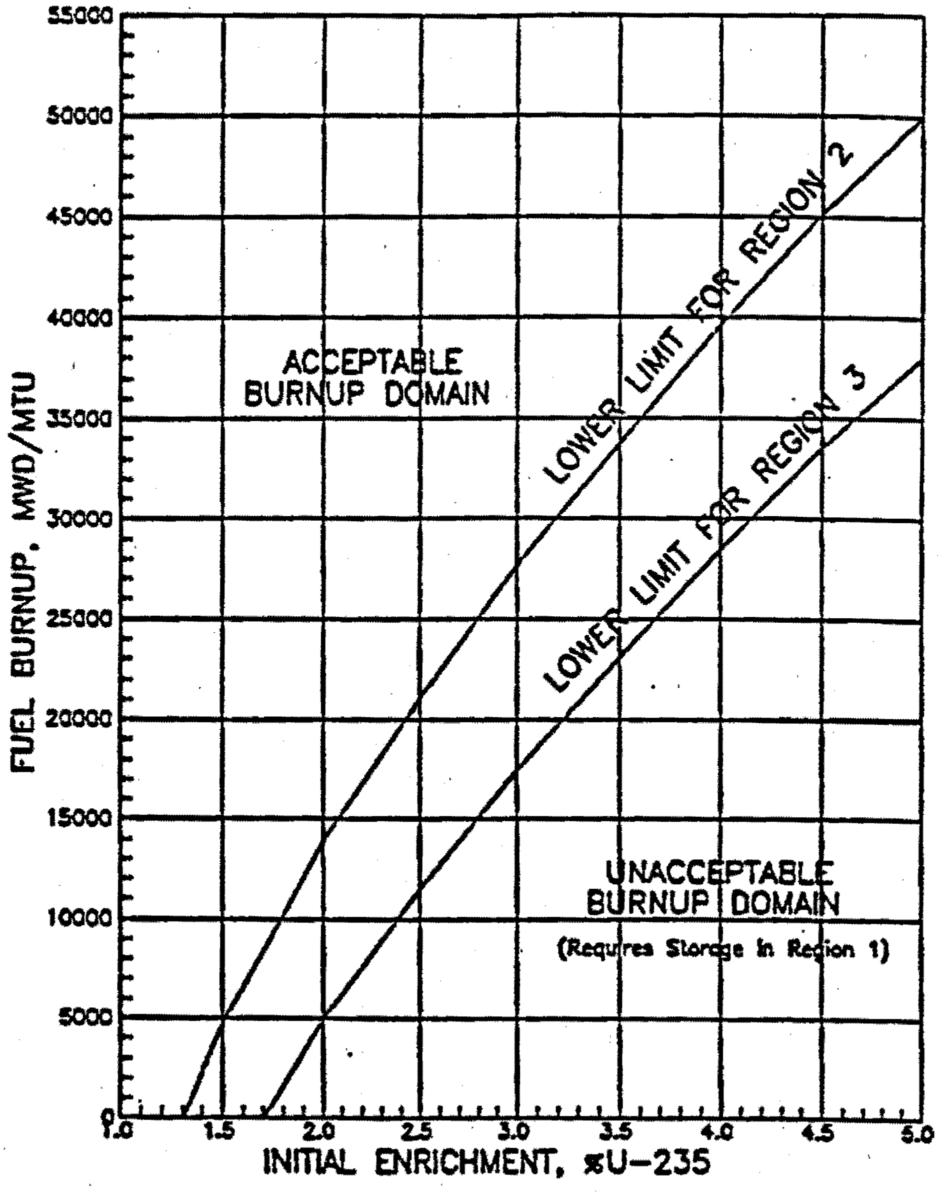


FIGURE 9.7-5: ACCEPTABLE BURNUP DOMAIN IN REGIONS 2 & 3

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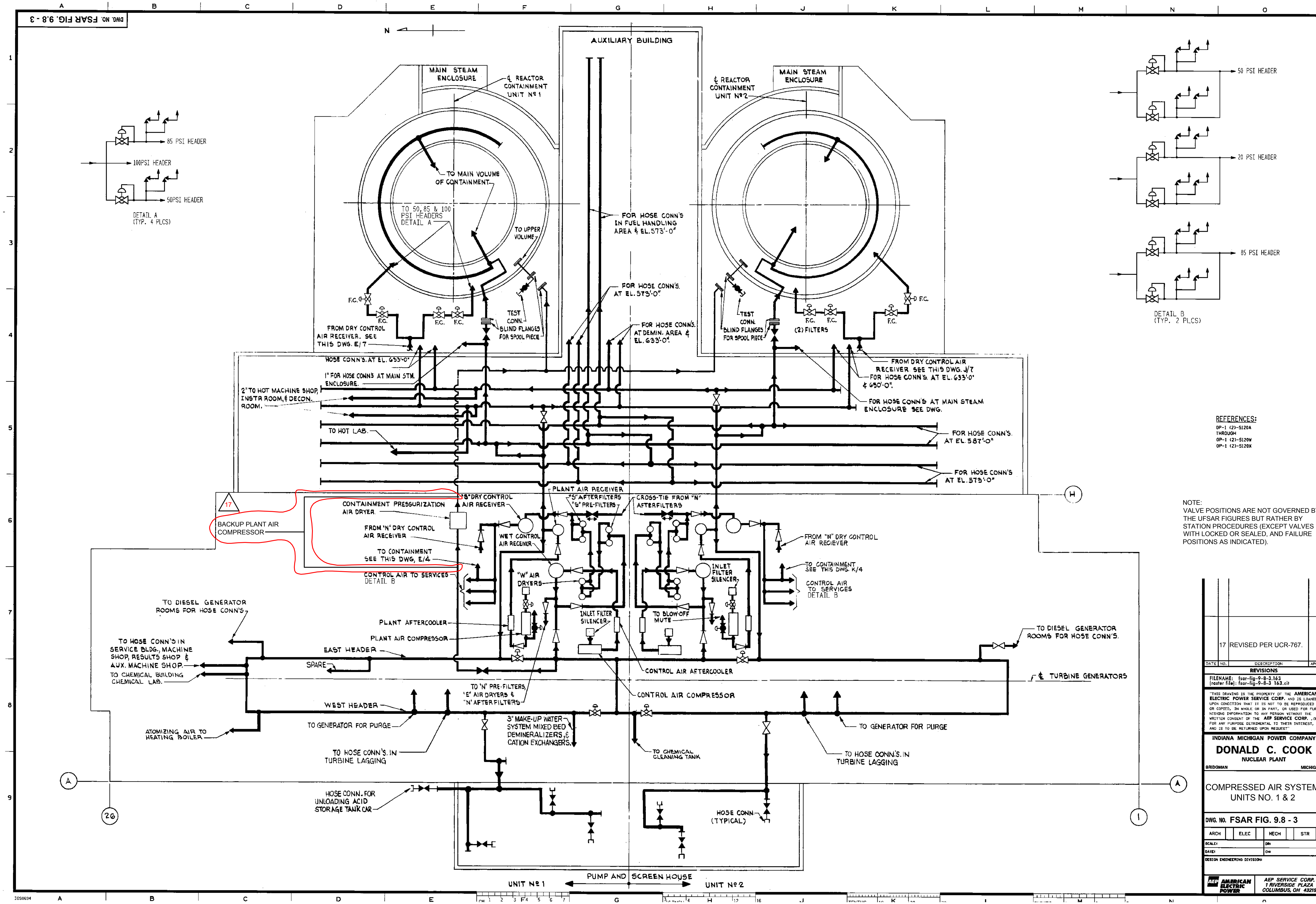
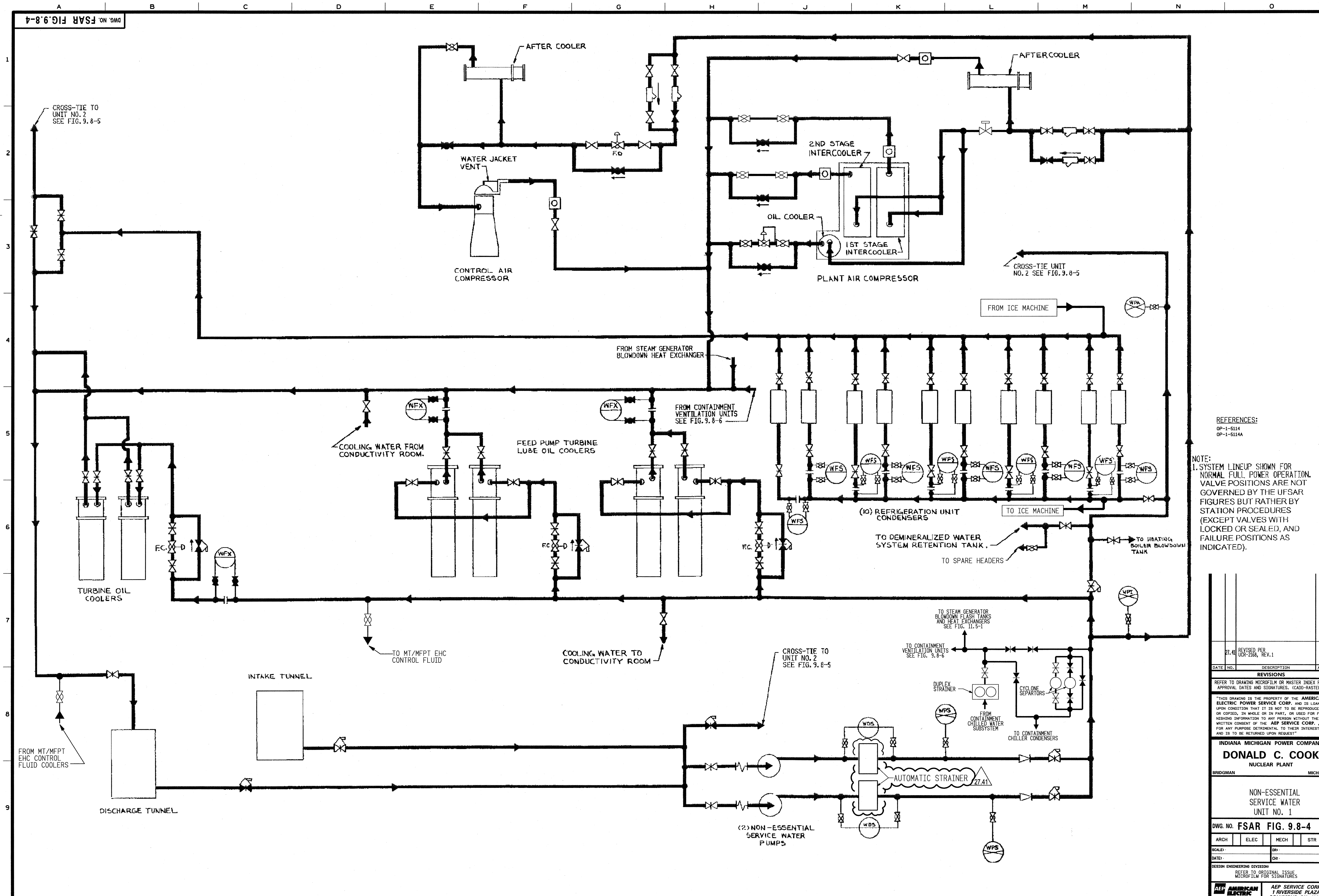


DWG. NO. FSAR FIG. 9.8 - 3
AUXILIARY BUILDING
MAIN STEAM ENCLOSURE
℄ REACTOR CONTAINMENT UNIT Nº 1
℄ REACTOR CONTAINMENT UNIT Nº 2
85 PSI HEADER
100PSI HEADER
50PSI HEADER
DETAIL A (TYP. 4 PLCS)
50 PSI HEADER
20 PSI HEADER
85 PSI HEADER
DETAIL B (TYP. 2 PLCS)
TO MAIN VOLUME OF CONTAINMENT
TO 50, 85 & 100 PSI HEADERS DETAIL A
TO UPPER VOLUME
FOR HOSE CONN'S IN FUEL HANDLING AREA ℄ EL.573'-0"
FOR HOSE CONN'S. AT EL.573'-0"
FOR HOSE CONN'S AT DEMIN. AREA & EL. 633'-0"
TEST CONN.
BLIND FLANGES FOR SPOOL PIECE
F.C.
(2) FILTERS
FROM DRY CONTROL AIR RECEIVER. SEE THIS DWG. E/7
HOSE CONN'S. AT EL. 633'-0"
1" FOR HOSE CONN'S AT MAIN STM. ENCLOSURE.
2" TO HOT MACHINE SHOP, INSTR ROOM, & DECON. ROOM.
TO HOT LAB.
FROM DRY CONTROL AIR RECEIVER SEE THIS DWG. J/7
FOR HOSE CONN'S AT EL. 633'-0" & 650'-0".
FOR HOSE CONN'S AT MAIN STEAM ENCLOSURE SEE DWG.
FOR HOSE CONN'S. AT EL. 587'-0"
FOR HOSE CONN'S AT EL. 573'-0"
REFERENCES:
OP-1 (2)-5120A THROUGH OP-1 (2)-5120W OP-1 (2)-5120X
NOTE:
VALVE POSITIONS ARE NOT GOVERNED BY THE UFSAR FIGURES BUT RATHER BY STATION PROCEDURES (EXCEPT VALVES WITH LOCKED OR SEALED, AND FAILURE POSITIONS AS INDICATED).
17
BACKUP PLANT AIR COMPRESSOR
CONTAINMENT PRESSURIZATION AIR DRYER
"S" DRY CONTROL AIR RECEIVER
PLANT AIR RECEIVER
"S" AFTERFILTERS
"S" PRE-FILTERS
CROSS-TIE FROM "N" AFTERFILTERS
FROM "N" DRY CONTROL AIR RECEIVER
WET CONTROL AIR RECEIVER
TO CONTAINMENT SEE THIS DWG, E/4
CONTROL AIR TO SERVICES DETAIL B
"N" AIR DRYERS
INLET FILTER SILENCER
TO BLOW-OFF MUTE
FROM "N" DRY CONTROL AIR RECIEVER
TO CONTAINMENT SEE THIS DWG. K/4
CONTROL AIR TO SERVICES DETAIL B
TO DIESEL GENERATOR ROOMS FOR HOSE CONN'S.
PLANT AFTERCOOLER
PLANT AIR COMPRESSOR
EAST HEADER
SPARE
TO HOSE CONN'S IN SERVICE BLDG., MACHINE SHOP, RESULTS SHOP & AUX. MACHINE SHOP.
TO CHEMICAL BUILDING CHEMICAL LAB.
CONTROL AIR AFTERCOOLER
CONTROL AIR COMPRESSOR
℄ TURBINE GENERATORS
TO 'N' PRE-FILTERS, 'E' AIR DRYERS & 'N' AFTERFILTERS
WEST HEADER
TO GENERATOR FOR PURGE
3" MAKE-UP WATER SYSTEM MIXED BED DEMINERALIZERS & CATION EXCHANGERS.
ATOMIZING AIR TO HEATING BOILER
TO HOSE CONN'S. IN TURBINE LAGGING
TO CHEMICAL CLEANING TANK
HOSE CONN. FOR UNLOADING ACID STORAGE TANK CAR
HOSE CONN. (TYPICAL)
PUMP AND SCREEN HOUSE
UNIT Nº 1
UNIT Nº 2
17 REVISED PER UCR-767.
DATE NO. DESCRIPTION APPD.
REVISIONS
FILENAME: fsar-fig-9-8-3.163 (raster file): fsar-fig-9-8-3 163.cit
INDIANA MICHIGAN POWER COMPANY
DONALD C. COOK NUCLEAR PLANT
BRIDGMAN MICHIGAN
COMPRESSED AIR SYSTEM UNITS NO. 1 & 2
DWG. NO. FSAR FIG. 9.8 - 3
ARCH ELEC MECH STR
DESIGN ENGINEERING DIVISION
AMERICAN ELECTRIC POWER
AEP SERVICE CORP. 1 RIVERSIDE PLAZA COLUMBUS, OH 43215



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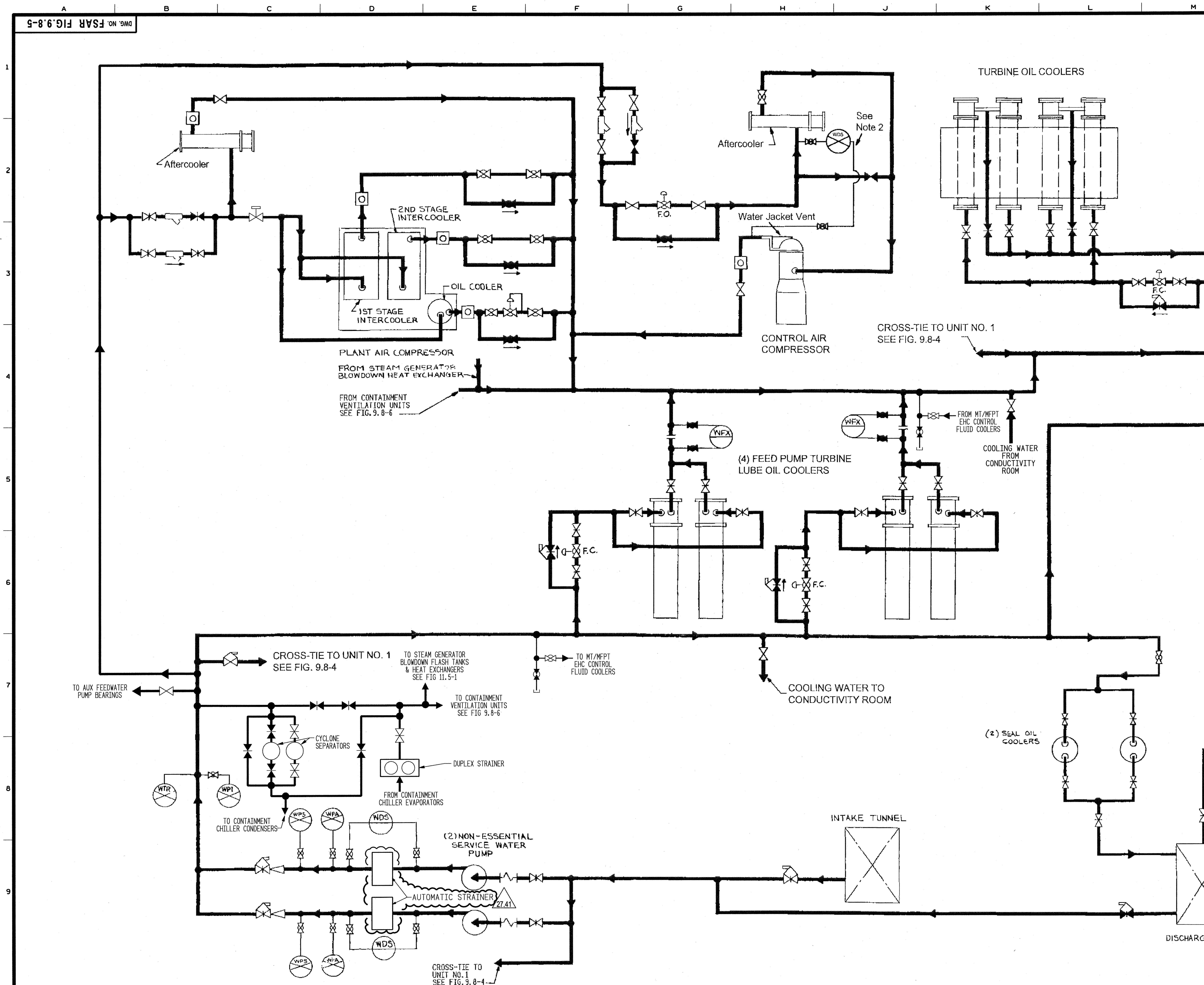

NOTE A:
1. SYSTEM LINEUP SHOWN FOR NORMAL FULL POWER OPERATION. VALVE POSITIONS ARE NOT GOVERNED BY THE UFSAR FIGURES BUT RATHER BY STATION PROCEDURES (EXCEPT VAVLES WITH LOCKED OR SEALED, AND FAILURE POSITIONS AS INDICATED).

2. Unit 2 only.

REFERENCES:
OP-2-5114
OP-2-5114A

DATE	NO.	DESCRIPTION	APPD.
	27.41	REVISED PER UCR-2168, REV.1	

REVISIONS

REFER TO DRAWING MICROFILM OR MASTER INDEX FOR APPROVAL DATES AND SIGNATURES. (CADD-RASTER)

INDIANA MICHIGAN POWER COMPANY

DONALD C. COOK

NUCLEAR PLANT

BRIDGMAN MICHIGAN

NON ESSENTIAL
SERVICE WATER
UNIT NO.2

DWG. NO. **FSAR FIG. 9.8-5**

ARCH		ELEC		MECH		STR	

SCALE: · DR: ·
DATE: · CH: ·
DESIGN ENGINEERING DIVISION:

AEP AMERICAN ELECTRIC POWER
AEP SERVICE CORP.
1 RIVERSIDE PLAZA
COLUMBUS, OH 43215

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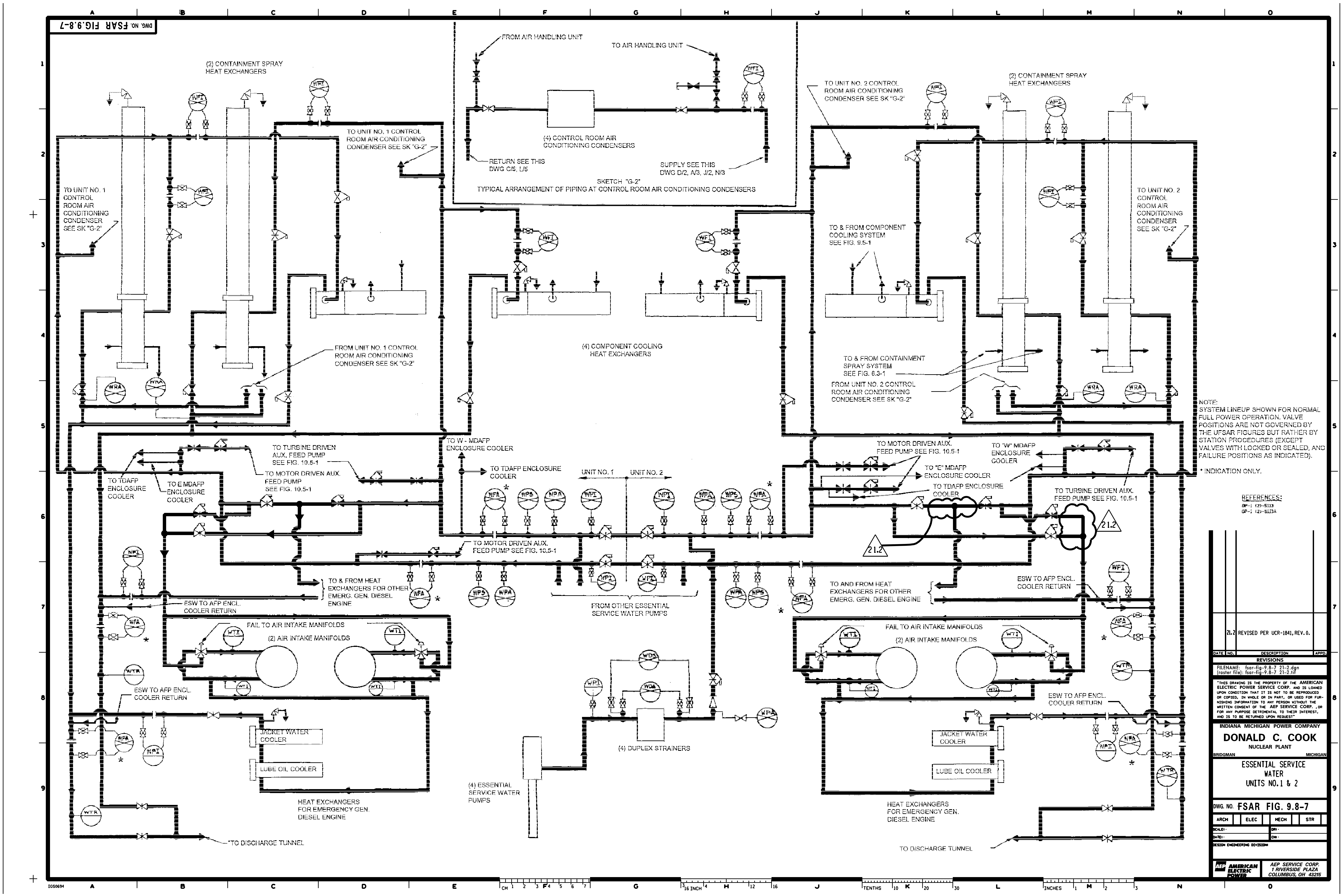

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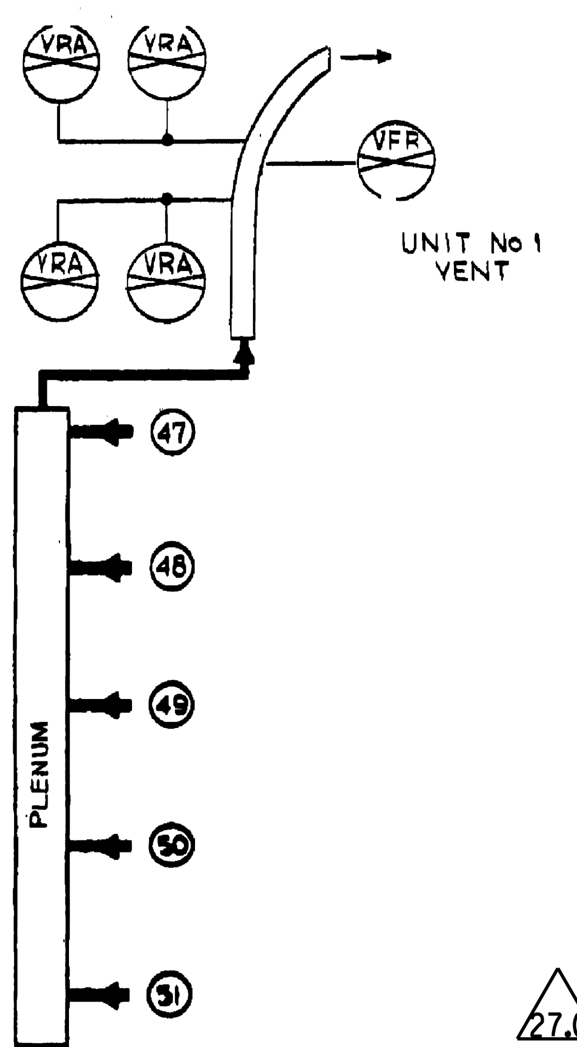
DWG. NO. FASR FIG 9.9-1

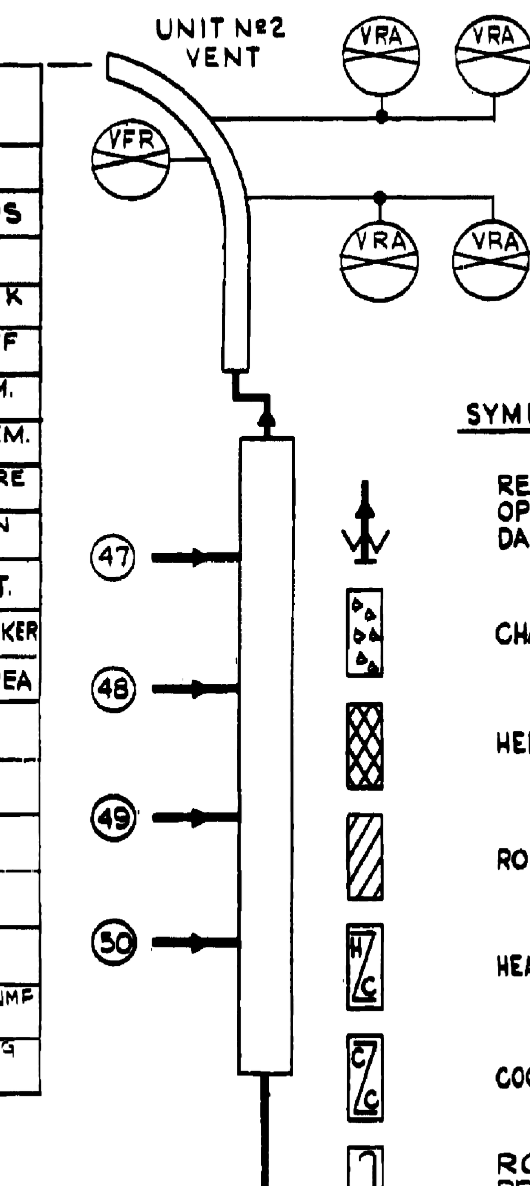
ITEM Nº	COORD.	Nº OF CONNS.	LOCATION (AREA OR ENCLOSURE)
1	E6	1	DECONTAMINATION AREA
2	G4	2	LET DOWN HEAT EXCHANGER
3	G5	1	HOT TOOL DECONTAMINATION FACILITY EXHAUST
4	D5 & K5	2	FAN EQUIPMENT ROOM
5	D5 & K5	2	CATION BED DEMIN.
6	D5, F5, H5 & K5	4	MIXED BED DEMIN.
7	E5 & J5	5	DEBORATING DEMIN. & SPENT FUEL PIT DEMIN.
8	F5 & G5	4	EVAP. FEED ION EXCHANGER
9	C7 & J7	2	NON-ESSENTIAL SERVICE WATER VALVE AREA
10	C7 & J7	2	BORIC ACID INJECTION TANK
11	E6	1	WASTE GAS COMPRESSORS
12	F6 & G6	2	VOLUME CONTROL TANK
13	F6 & G6	2	SEAL WATER HEAT EXCHANGER
14	E6	1	CONCENTRATE HOLDING TANK
15	D7	1	PIPE TUNNEL
16	H6, J6, K6	4	MONITOR TANKS
17	B7 & C6	2	HOT LABORATORY HOOD EXHAUST
18	C7	1	HOT LABORATORY CABINET EXHAUST
19			
20	A8 & B8	6	DRUM STORAGE & DRUMMING AREA

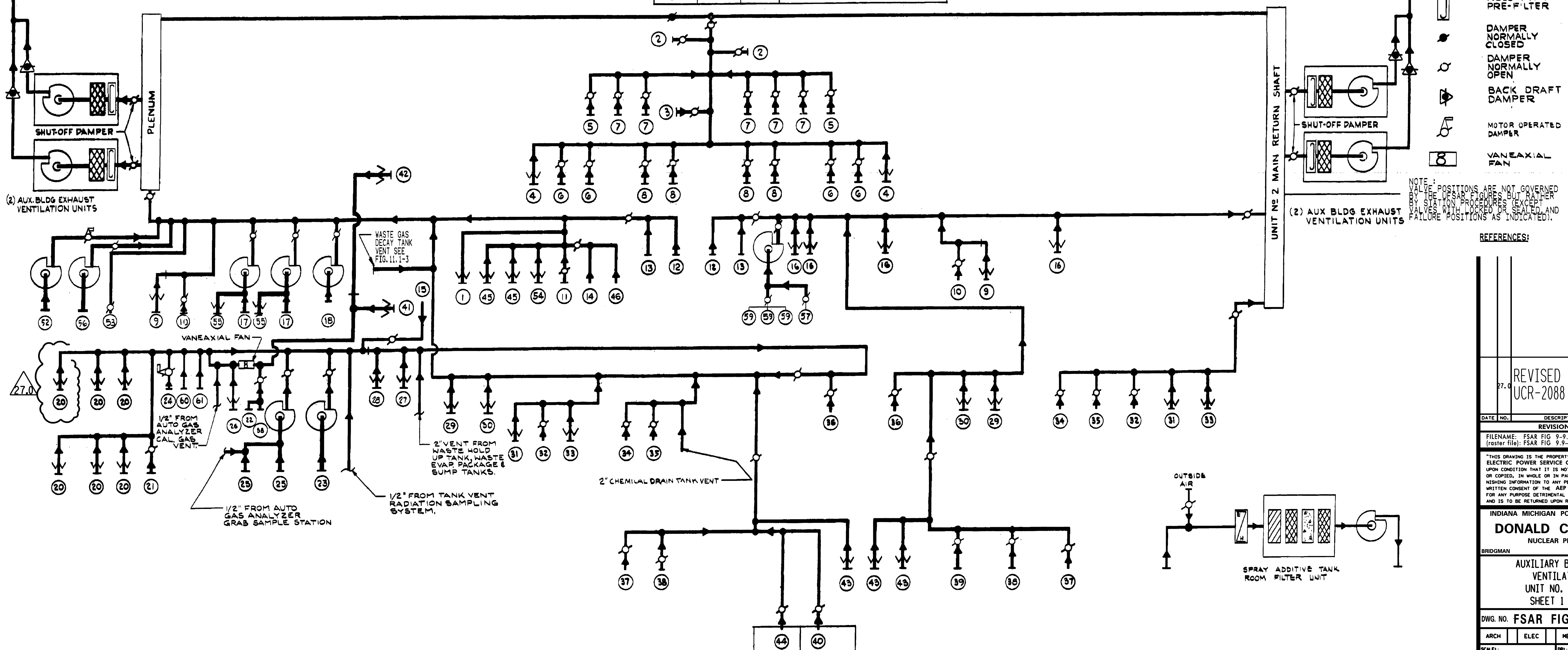
27.0

ITEM Nº	COORD.	Nº OF CONNS.	LOCATION (AREA OR ENCLOSURE)
21	B8	1	SPENT RESIN STORAGE TANK
22	C8	1	WASTE EVAP. PACKAGE (2 GPM)
23	C8	1	SAMPLING ROOM SAMPLE SINK HOOD
24	B8	1	SPRAY ADDITIVE TANK
25	C8	2	SAMPLE ROOM RACK 'C'
26	C8	1	WASTE EVAP. FEED FILTER
27	D8	1	WASTE HOLD-UP TANK
28	D8	1	REFUELING WATER FILTER
29	E8 & K8	2	PIPEWAY
30	E8 & J8	2	CONCENTRATE & SEAL WTR. INJECT FILTER
31	E8 & L8	2	VALVE OPERATING GALLERY
32	E8 & L8	2	BORIC ACID EVAPORATOR
33	F8 & L8	2	ELEC. PENETRATION
34	F8 & K8	2	PIPE TUNNEL
35	F8 & K8	2	GAS DECAY TANK AREA
36	J8 & H8	2	REACTOR COOLANT FILTER
37	F9 & K9	2	PIPE TUNNEL
38	G9 & K9	2	REACTOR COOLANT DRAIN TANK PUMP
39	J9	1	SUMP TANK & PUMP
40	H9	1	AUX. BLDG. SUMP
41	D7	1	SPENT FUEL PIT PUMPS
42	D6	1	NEW FUEL STORAGE ROOM

ITEM Nº	COORD.	Nº OF CONNS.	LOCATION (AREA OR ENCLOSURE)
43	H9 & J9	3	HOLD-UP TANKS
44	H9	1	BORIC ACID EVAP. FEED PUMPS
45	E6	2	HOT MACHINE SHOP
46	F6	1	CONCENTRATE HOLDING TANK TRANSFER PUMPS
47	N2 & A2	2	FROM CONTAINMENT PRESS RELIEF SYSTEM. SEE DWG. 5947A
48	N3 & A2	2	FROM INSTR. ROOM PURGE SYSTEM. SEE DWG. 5947A
49	N3 & A3	2	FROM CONTAINMENT PURGE SYSTEM. SEE DWG. 5947A
50	N3 & A3	2	FROM ENGINEERED SAFETY FEATURE VENT. SYSTEM. SEE DWG. 12-5948A
51	A/4	1	FROM FUEL HANDLING VENTILATION SYSTEM. SEE DWG. 12-5948A
52	A/7	1	HOT LABORATORY HOOD EXHAUST.
53	B/7	1	ACCESS CONTROL AREA TOILETS & LOCKER
54	E/7	1	PERSONNEL DECONTAMINATION AREA
55	B/7	2	HOT LABORATORY HOOD BY PASS
56	A/7	1	ATOMIC ABSORPTION HOOD VENT
57	G/7 & H/7	2	VENT FROM LAUNDRY ROOM
58	C/8	1	15 GPM WASTE EVAPORATOR
59	C/7	3	LAUNDRY ROOM DRYER
60	B/7	1	VENT FROM PA SAMPLING VACUUM PUMP DISCHARGE
61	B/7	1	EXHAUST FROM PA LIQUID & GAS SAMPLING PANEL CLEAN-UP UNIT 12-HV-SATSPFN







SYMBOLS

- REGISTER WITH OPPOSED BLADE DAMPER
- CHARCOAL FILTER
- HEPA FILTER
- ROUGHING FILTER
- HEATING COIL
- COOLING COIL
- ROLL TYPE PRE-FILTER
- DAMPER NORMALLY CLOSED
- DAMPER NORMALLY OPEN
- BACK DRAFT DAMPER
- MOTOR OPERATED DAMPER
- VANEAXIAL FAN

NOTE: VALVE POSITIONS ARE NOT GOVERNED BY THE UFSAR FIGURES BUT RATHER BY STATION PROCEDURES (EXCEPT VALVES WITH LOCKED OR SEALED AND FAILURE POSITIONS AS INDICATED).

REFERENCES:

DATE	NO.	DESCRIPTION	APPD.
	27.0	REVISED FOR UCR-2088 REV 1	

REVISIONS

FILENAME: FSAR FIG 9-9.1.27
(raster file): FSAR FIG 9.9-1_27.TIF

INDIANA MICHIGAN POWER COMPANY

DONALD C. COOK

NUCLEAR PLANT

BRIDGMAN MICHIGAN

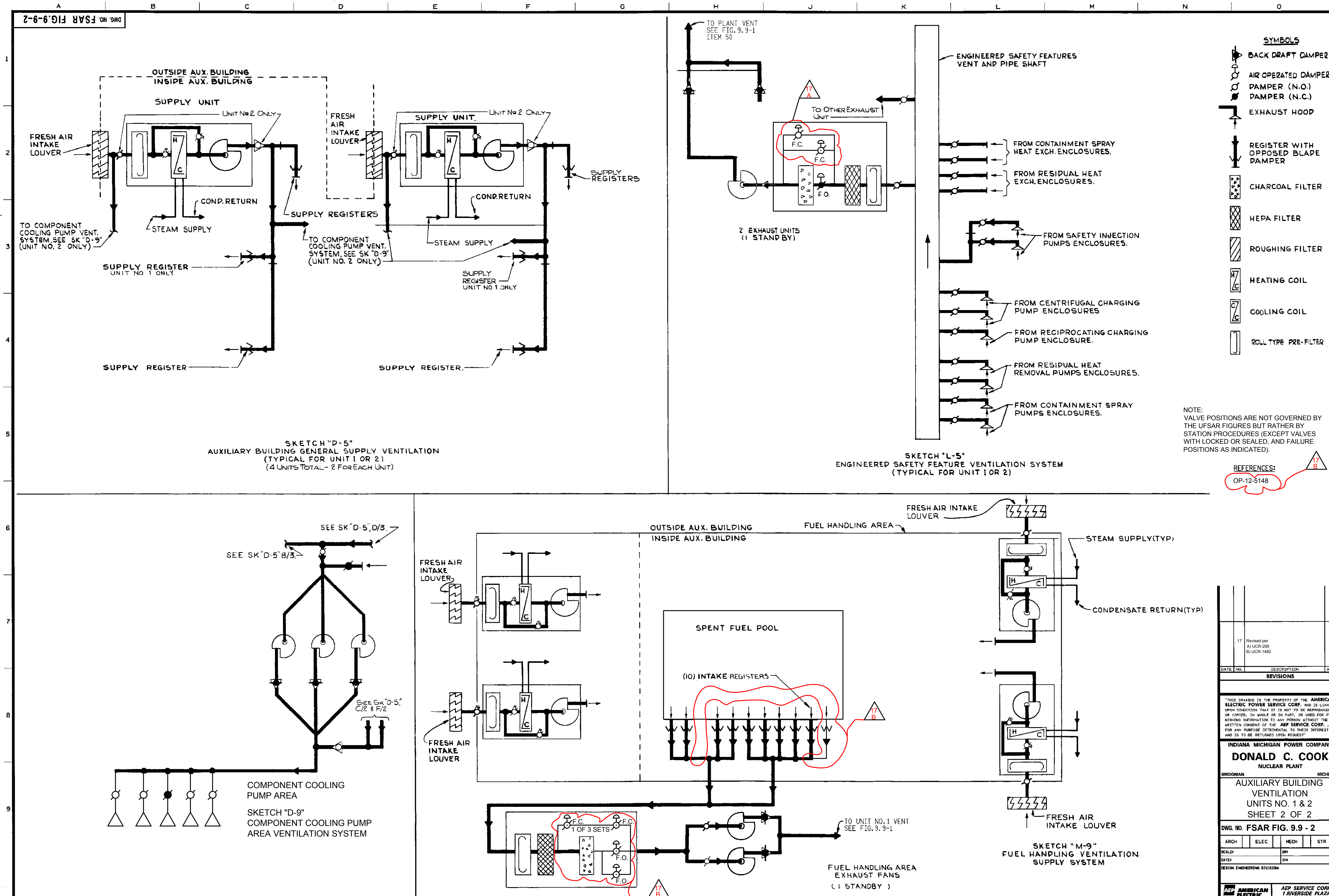
AUXILIARY BUILDING
VENTILATION
UNIT NO. 1 & 2
SHEET 1 OF 2

DWG. NO. FSAR FIG. 9.9-1

ARCH		ELEC		MECH		STR	

SCALE: DR:
DATE: CH:
DESIGN ENGINEERING DIVISION:

AEP AMERICAN ELECTRIC POWER — AEP SERVICE CORP. 1 RIVERSIDE PLAZA COLUMBUS, OH 43215



DWG. NO. FSAR FIG. 9.9-2
OUTSIDE AUX. BUILDING
INSIDE AUX. BUILDING
SUPPLY UNIT
UNIT No 2 ONLY
FRESH AIR INTAKE LOUVER
COND. RETURN
STEAM SUPPLY
SUPPLY REGISTERS
TO COMPONENT COOLING PUMP VENT. SYSTEM, SEE SK "D-9" (UNIT NO. 2 ONLY)
SUPPLY REGISTER UNIT NO 1 ONLY
SUPPLY REGISTER
SKETCH "D-5"
AUXILIARY BUILDING GENERAL SUPPLY VENTILATION
(TYPICAL FOR UNIT 1 OR 2)
(4 UNITS TOTAL - 2 FOR EACH UNIT)
TO PLANT VENT SEE FIG. 9.9-1 ITEM 50
ENGINEERED SAFETY FEATURES VENT AND PIPE SHAFT
TO OTHER EXHAUST UNIT
F.C.
F.O.
2 EXHAUST UNITS (1 STAND BY)
FROM CONTAINMENT SPRAY HEAT EXCH. ENCLOSURES.
FROM RESIDUAL HEAT EXCH. ENCLOSURES.
FROM SAFETY INJECTION PUMPS ENCLOSURES.
FROM CENTRIFUGAL CHARGING PUMP ENCLOSURES
FROM RECIPROCATING CHARGING PUMP ENCLOSURE.
FROM RESIDUAL HEAT REMOVAL PUMPS ENCLOSURES.
FROM CONTAINMENT SPRAY PUMPS ENCLOSURES.
SKETCH "L-5"
ENGINEERED SAFETY FEATURE VENTILATION SYSTEM
(TYPICAL FOR UNIT 1 OR 2)
SYMBOLS
BACK DRAFT DAMPER
AIR OPERATED DAMPER
DAMPER (N.O.)
DAMPER (N.C.)
EXHAUST HOOD
REGISTER WITH OPPOSED BLADE DAMPER
CHARCOAL FILTER
HEPA FILTER
ROUGHING FILTER
HEATING COIL
COOLING COIL
ROLL TYPE PRE-FILTER
NOTE:
VALVE POSITIONS ARE NOT GOVERNED BY THE UFSAR FIGURES BUT RATHER BY STATION PROCEDURES (EXCEPT VALVES WITH LOCKED OR SEALED, AND FAILURE POSITIONS AS INDICATED).
REFERENCES:
OP-12-5148
17 B
SEE SK "D-5" D/3.
SEE SK "D-5" B/3.
SEE SK "D-5" C/2 & F/2
COMPONENT COOLING PUMP AREA
SKETCH "D-9"
COMPONENT COOLING PUMP AREA VENTILATION SYSTEM
FRESH AIR INTAKE LOUVER
OUTSIDE AUX. BUILDING
INSIDE AUX. BUILDING
FUEL HANDLING AREA
STEAM SUPPLY (TYP)
CONDENSATE RETURN (TYP)
SPENT FUEL POOL
(10) INTAKE REGISTERS
1 OF 3 SETS
TO UNIT NO. 1 VENT SEE FIG. 9.9-1
FUEL HANDLING AREA EXHAUST FANS
(1 STANDBY)
SKETCH "M-9"
FUEL HANDLING VENTILATION SUPPLY SYSTEM
17 Revised per A) UCR-295 B) UCR-1482
REVISIONS
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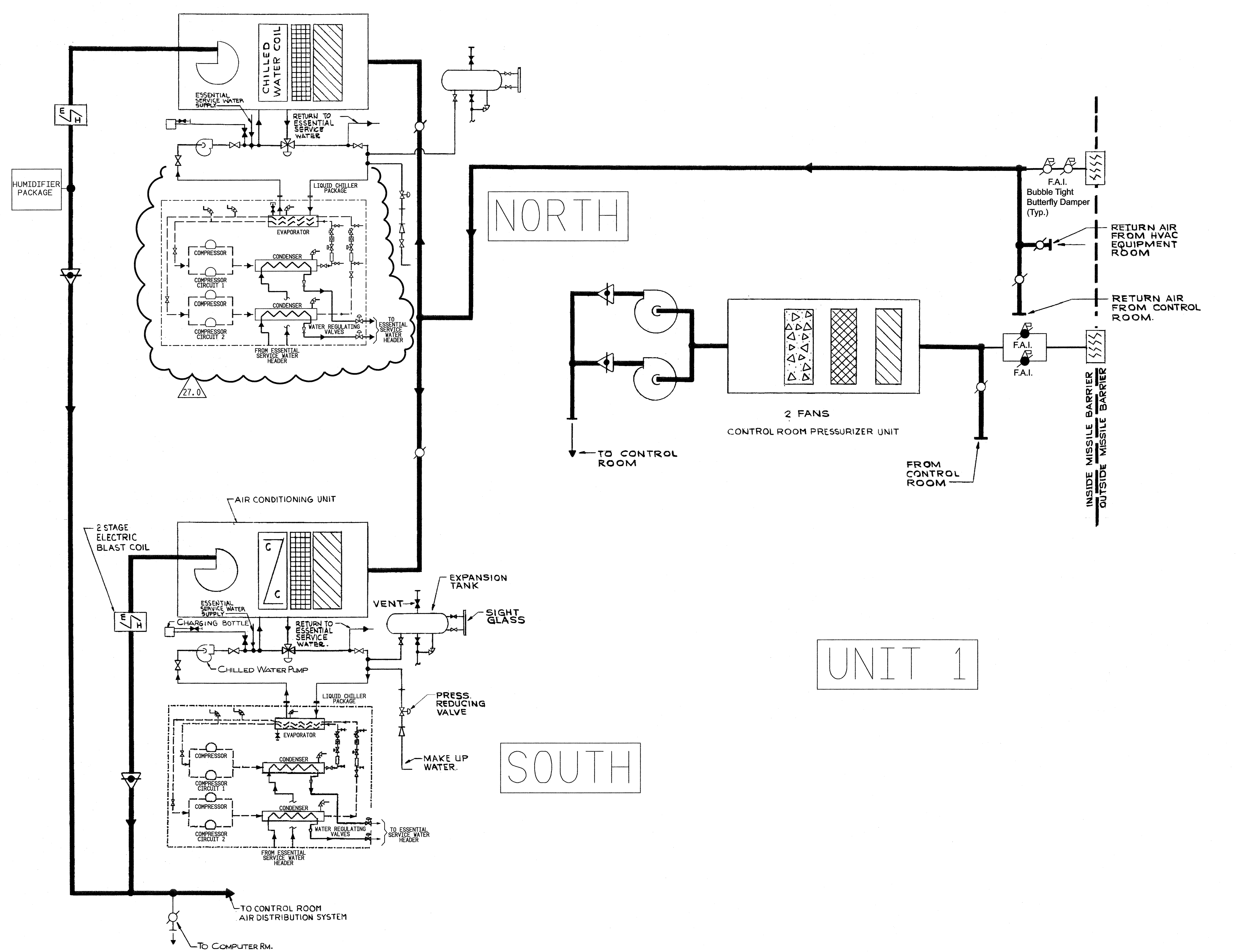
INDIANA MICHIGAN POWER COMPANY
DONALD C. COOK
NUCLEAR PLANT
BRIDGMAN MICHIGAN
AUXILIARY BUILDING VENTILATION
UNITS NO. 1 & 2
SHEET 2 OF 2
DWG. NO. FSAR FIG. 9.9-2
AMERICAN ELECTRIC POWER
AEP SERVICE CORP. 1 RIVERSIDE PLAZA COLUMBUS, OH 43215

DWG. NO. FSAR FIG. 9.10-1



SYMBOLS

- MANUAL VOLUME DAMPER
- NORMALLY OPEN DAMPER
- NORMALLY CLOSED DAMPER
- BACK DRAFT DAMPER
- ROUGHING OR PREFILTER
- ABSOLUTE FILTER
- CHARCOAL FILTER
- CHILLED WATER COIL
- CONTROL OFF REFRIGERANT LINE TEMPERATURE OR PRESSURE
- ELECTRIC BLAST COIL HEATER
- MEDIUM EFFICIENT FILTER

NOTE:
VALVE POSITIONS ARE NOT GOVERNED BY THE UFSAR FIGURES BUT RATHER BY STATION PROCEDURES (EXCEPT VALVES WITH LOCKED OR SEALED, AND FAILURE POSITIONS AS INDICATED).

REFERENCES:
OP-1 (2)-5149

DATE	NO.	DESCRIPTION	APPD.
	27.0	REVISED PER UCR-2057, REV.0	

REVISIONS

REFER TO DRAWING MICROFILM OR MASTER INDEX FOR APPROVAL DATES AND SIGNATURES. (CADD-RASTER)

INDIANA MICHIGAN POWER COMPANY
DONALD C. COOK
NUCLEAR PLANT
BRIDGMAN MICHIGAN

CONTROL ROOM
VENTILATION
UNIT NO.1

DWG. NO. FSAR FIG. 9.10-1

ARCH | ELEC | MECH | STR

SCALE: DR:
DATE: CH:
DESIGN ENGINEERING DIVISION:

AMERICAN ELECTRIC POWER
AEP SERVICE CORP.
1 RIVERSIDE PLAZA
COLUMBUS, OH 43215

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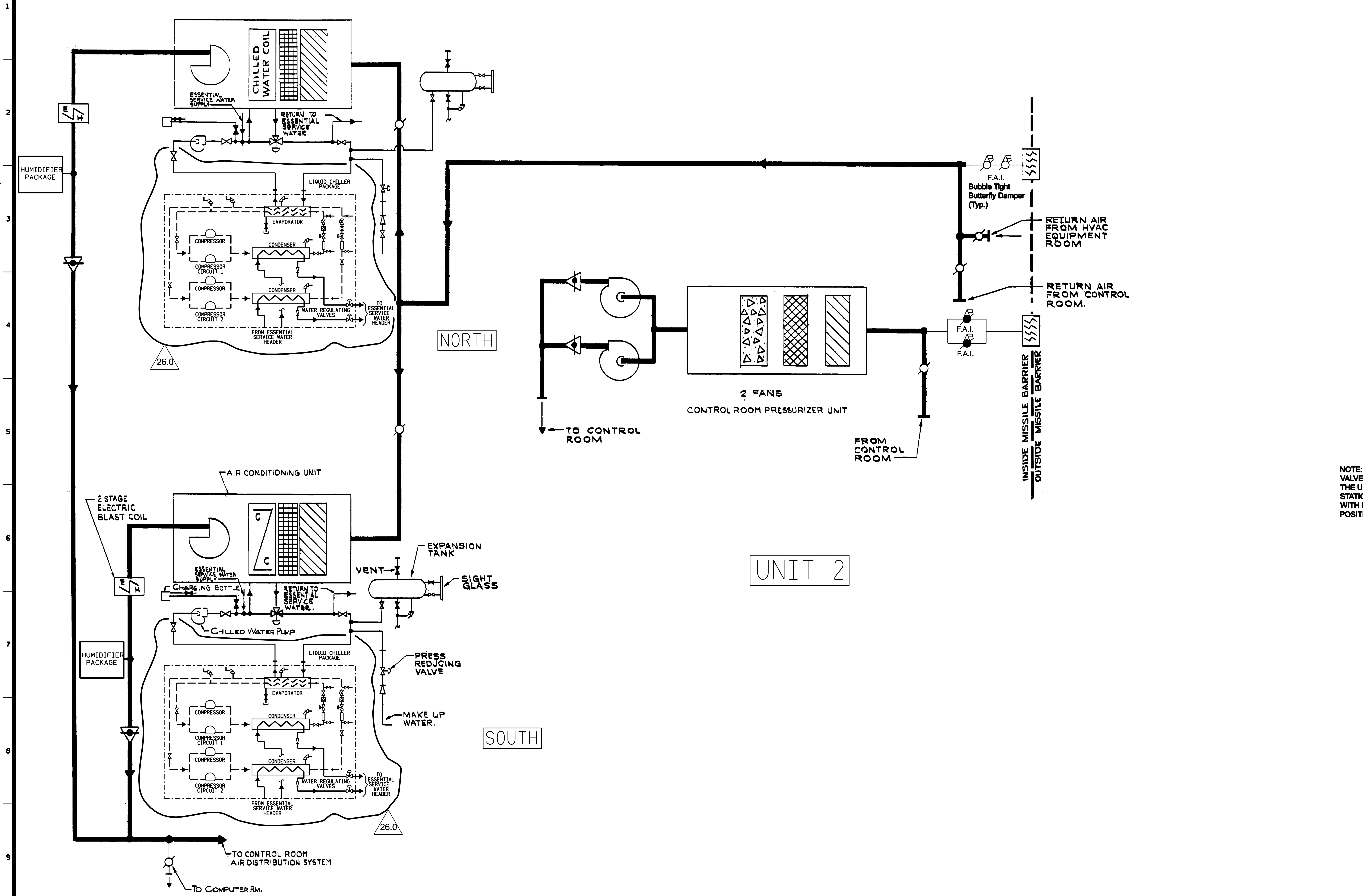

SYMBOLS

MANUAL VOLUME DAMPER

NORMALLY OPEN DAMPER

NORMALLY CLOSED DAMPER

BACK DRAFT DAMPER

ROUGHING OR PREFILTER

ABSOLUTE FILTER

CHARCOAL FILTER

CHILLED WATER COIL

CONTROL OFF REFRIGERANT LINE TEMPERATURE OR PRESSURE

ELECTRIC BLAST COIL HEATER

MEDIUM EFFICIENT FILTER

NOTE:
VALVE POSITIONS ARE NOT GOVERNED BY THE UFSAR FIGURES BUT RATHER BY STATION PROCEDURES (EXCEPT VALVES WITH LOCKED OR SEALED, AND FAILURE POSITIONS AS INDICATED).

REFERENCES:
OP-1 (2)-5149

DATE	NO.	DESCRIPTION	APPD.
	26.0	Revised per UCR-2002, UCR-2083, UCR-2068 & UCR-2069	

REVISIONS

REFER TO DRAWING MICROFILM OR MASTER INDEX FOR APPROVAL DATES AND SIGNATURES. (CADD-RASTER)

INDIANA MICHIGAN POWER COMPANY

DONALD C. COOK
NUCLEAR PLANT

BRIDGMAN MICHIGAN

CONTROL ROOM
VENTILATION
UNIT NO. 2

DWG. NO. FSAR FIG. 9.10 - 2

ARCH | ELEC | MECH | STR

SCALE: DR:
DATE: CH:
DESIGN ENGINEERING DIVISION

AEP AMERICAN ELECTRIC POWER — AEP SERVICE CORP. 1 RIVERSIDE PLAZA COLUMBUS, OH 43215