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CHAPTER 9.0. AUXILIARY SYSTEMS

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9.0. AUXILIARY SYSTEMS

9.1. Fuel Storage and Handling

9.1.1. New Fuel Storage

9.1.1.1. Design Bases

1. Storage space shall be provided for one refueling batch for each reactor or a total of at least 138 fuel assemblies. (12 additional space cells are provided.)

2. New fuel shall be stored in an array such that k_{eff} will be less than 0.90 even if flooded with unborated water.

3. The new fuel storage facility shall be capable of withstanding loads imposed by the dead load of the fuel assemblies, impact and handling, and the safe shutdown earthquake (SSE) and 1/2 SSE. Any resulting damage shall not be such as to increase k_{eff} above 0.90. The facility shall not be required to withstand loads that would be imposed by dropping heavy objects onto it, but the movement of such objects over it shall be administratively prohibited.

9.1.1.2. Facilities Description

The location of the new fuel storage area is shown in Figures 9.1-1, 9.1-4, and 9.1-5. Figure 9.1-2 shows the typical design of the new fuel storage racks.

The racks are multiple cell structures which can be fastened together in any number to form an assembly that can be firmly bolted to anchors in the floor and walls of the new fuel vault. These racks have been designed to withstand the Safe Shutdown Earthquake loads as well as dead loads of the fuel assemblies. The racks are designed in accordance with AEC Design Criteria 61 and 62.

The racks are constructed so that it is impossible to insert fuel assemblies except in prescribed locations which have a minimum center-to-center spacing of 21 inches in both directions. The spacing is sufficient to assure $k_{eff} < 0.90$ even if immersed in unborated water.

Storage rack analysis uses an infinite array of assemblies. Therefore, the sharing of the new fuel storage facilities presents no problem from a criticality standpoint. 1

9.1.1.3. Safety Evaluation

The new fuel storage racks shall be designed to meet the requirements of a Category I structure and Safety Guide 13. The analysis shall be performed using the applicable response spectra curves for the plant. Results of the analysis will show that the stored assemblies will not impact against an adjacent supporting structure or violate the stored configuration.

9.1.2. Spent Fuel Storage

9.1.2.1. Design Bases

1. Spent fuel storage space shall be provided by separate storage pits. Each storage pit shall provide for one refueling batch from a reactor plus a complete core unloading of the reactor, for a total of one and one-third cores or 288 fuel assemblies. (14 additional space cells are provided.) The combined storage capacity of the two separate pits shall be 548 fuel assemblies. 11

2. The fuel shall be stored in an array such that k_{eff} will be less than 0.90 even if the water in the storage pit contains no boron and if the fuel is unirradiated.

3. The depth of shielding water over the spent fuel in storage shall be such that the gamma dose rate at the surface is 2.5 mR/h or less. There shall be a minimum of 9'6" of water above the top of a fuel assembly during all handling operations.

4. The spent fuel storage facility shall be capable of withstanding loads imposed by the dead load of the fuel assemblies, impact and handling, and SSE. Damage to spent fuel pit and storage racks shall not be sufficient to cause a loss of water below the top of the racks or to increase k_{eff} above 0.90.

5. The facility shall be designed so that the building cranes cannot travel above stored spent fuel.

6. The spent fuel shipping cask loading area shall be separated from the spent fuel storage area so that in the event a cask is dropped, the fuel is not damaged nor is a loss of water below the top of the storage racks incurred.

9.1.2.2. Facilities Description

The location of the spent fuel storage facility is shown in Figures 9.1-1 9.1-4 and 9.1-5. Figure 9.1-2 shows the design of the spent fuel storage racks.

There are two spent fuel pits, each of which is a reinforced concrete structure which rests on the rock formation which underlies the Bellefonte site. The pits are designed to withstand safe shutdown earthquake forces, as is the auxiliary building in which the pit is located. The pits are lined with stainless steel plate to ensure watertightness.

Each spent fuel pit structure includes a fuel storage area, a spent fuel cask loading area, and a transfer canal which connects the refueling canal in the reactor building via the fuel transfer tube. The following discussion shows how the facilities comply with the design bases given in paragraph 9.1.2.1:

1. Each fuel storage area contains 9 spent fuel storage racks having a capacity of 32 fuel assemblies each, for a total of 288 assemblies. The racks include provisions for storage of spent control rods and burnable poison rods.

2. The spent fuel storage racks provide a minimum spacing of 21 inches between fuel assemblies. This spacing is such that k_{eff} is below 0.90 if the racks are filled with fuel assemblies having the highest anticipated enrichment even when flooded with unborated water. The racks are designed so that fuel can be inserted only in the designated spaces.

3. The normal depth of water in the spent fuel storage area is about 40 ft. The depth of water above the racks is about 26 ft. With the racks full of spent fuel, the gamma dose rate at the water surface is less than 2.5 mR/h.

   Fuel in transit in the spent fuel pit must clear the sill between the transfer canal and the storage area and the sill between the storage area and the cask loading area. These sills are about 16' 2" above the pit floor. The top of a fuel assembly (height = 13' 10") which clears a sill by 12 inches would be about 10 ft. below the surface of the water. Hence, the water level could be 6 inches below normal level and still meet the criterion of maintaining at least 9' 6" of water above the top of the active fuel in a fuel assembly. A low level alarm is annunciated locally when the water level drops to 6 inches below normal level. The hoist on the spent fuel pit bridge has a mechanical stop which prohibits lifting a fuel assembly higher than a level at which its top if 102 inches below the alarm level. (The active fuel in the fuel assembly is 9' 6" below alarm water level.)

4. The spent fuel storage racks shown in Figure 9.1-2 are designed in accordance with AEC Design Criteria 61 and 62.

   The racks are multiple cell structures which can be fastened together in any number to form an assembly that can be firmly bolted to anchors in the floor of the spent fuel pit. The spent fuel racks, including supports, are made of aluminum and are constructed so that it is impossible to insert assemblies except in prescribed locations having a minimum center-to-center spacing of 21 inches in both directions. The spacing is sufficient to assure k_{eff} <0.90 even if immersed in unborated water.

   The racks and held down hardware have been designed to withstand SSE and 1/2 SSE loads as well as the dead load of the fuel assemblies.

   The fuel hoist which withdraws the fuel assembly from the fuel storage racks is designed with an overload limit which stops the hoist automatically in the up direction any time the load is 250 pounds ± 20 pounds 1
   greater than the weight of the fuel assembly plus the mast weight. (<u>Note</u>: This limit is to protect the fuel assembly — not the storage racks.)

5. The design of the spent fuel handling facility is such that neither of the two building cranes can travel over the spent fuel storage areas. As shown in Figure 9.1-1 the crane that handles the spent shipping cask is prevented by rail stops from carrying its load beyond the cask loading areas. The crane which serves the rest of the auxiliary building travels between and not over the fuel pits.

6. Each cask loading area is separated from the respective fuel storage area by a reinforced concrete wall. A slot in the wall permits movement of fuel from the storage area to the loading area. The sill of the slot is above the top of the fuel in the storage racks. A gate

which closes the slot is in place whenever the cask is being loaded into or raised from the cask loading area. With this arrangement, a cask drop accident which damaged the loading area to the extent that the water was lost would not cause a significant loss of water from the storage area.

9.1.2.3. Safety Evaluation

The spent fuel storage racks shall be designed to meet the requirements of a Category I structure and Safety Guide 13. The analysis shall be performed using the applicable response spectra curves for the plant. Results of the analysis will show that the stored assemblies will not impact against an adjacent structure or violate the stored configuration.

9.1.3. Spent Fuel Cooling System and Cleanup System

9.1.3.1. Design Bases

The spent fuel cooling system will be designed, in accordance with AEC General Criterion 61, to remove decay heat from the fuel stored in both spent fuel storage pools and to maintain water clarity. The system will maintain the temperature of each spent fuel storage pool water at less
than 150F, with a heat load based on removing the decay heat generated 11
from a one-third batch of core fuel assemlies. The batch which has been irradiated for 1044 days will be removed from the reactor and placed in the pool in 170 hours.

The system will have sufficient additional cooling capacity to store a newly refueled core that has to be removed from the reactor and placed in the storage pool in addition to the fuel assemblies previously placed in the pool. With a total of 1-1/3 cores in storage, the system will be capable of maintaining the spent fuel pool temperature below 150F while removing the total decay heat load of 33.3 x 10^6 Btu/h from the following combination of stored fuel assemblies:

1. One-third core irradiated for 1044 days and cooled for 70 days. 11

2. One-third core irradiated for 634 days and cooled for 10 days.

3. One-third core irradiated for 342 days and cooled for 10 days.

4. One-third core irradiated for 50 days and cooled for 10 days.

The combination of one pump and one cooler will maintain the water in one 11
spent fuel pool below 200F with the maximum amount of fuel (1-1/3 cores) stored in the pool. In addition to its primary function, the system will provide for the removal of fission and corrosion products from the spent fuel pool water, the fuel transfer canal water, and the contents of the borated water storage tank (BWST) in order to maintain water clarity for fuel operations. The system provides for filling the fuel transfer canal and the cask loading area. The system is also provided with interconnections to the decay heat removal system as a supplementary cooling source.

9.1.3.2. System Description

9.1.3.2.1. General

The spent fuel cooling system is shown schematically in detail on the Process and Instrumentation Drawing, Figure 9.1-3. The system performance data is shown in Table 9.1-1 and major individual system component data is shown in Table 9.1-2. All components required for the safe shutdown of the plant are Seismic Category I per AEC Safety Guide 29. The following is a brief functional description of system components on a per unit basis.

1. Spent Fuel Cooler - One spent cooler is provided to maintain the spent fuel storage pool water below 150F with a one-third batch of irradiated core in the storage pool as explained in 9.1.3.1. A spent fuel cooler common to both Units 1 and 2 is provided to maintain the spent fuel storage pool water temperature below 150F with one and one-third batches of irradiated core in a storage pool as explained in 9.1.3.1. The cooling system for the 2-pool arrangement is designed to handle the heat load imposed by a total of 1-1/3 cores. One and one-third core in each of the two pools is not considered to be a credible design base. |11 |11

2. Spent Fuel Coolant Pump - The spent fuel coolant pump takes suction from the spent fuel storage pool and recirculates the fluid back to the storage pool after passing through the spent fuel cooler. During refueling operations the spent fuel pump is also used to fill the refueling canal with borated water from the borated water storage tank. A spare spent fuel coolant pump provides coolant flow to the spare spent fuel cooler as mentioned in paragraph 1. |11

3. Spent Fuel Coolant Demineralizer - The demineralizer will maintain the quality of storage pool water with less than 4 ppm dissolved solids (excluding boric acid).

4. Spent Fuel Filters - The spent fuel coolant filters will remove particulate matter from the water of the spent fuel storage pool during circulation.

5. Borated Water Recirculation Pump - This pump will remove water from the BWST for demineralizing and filtering. In addition the pump may be used to transfer water from the refueling canal to the purification loop for demineralization and filtration. It may also be used to empty the refueling canal if the spent fuel coolant pump is not available for use.

6. Skimmer - The skimmer system will remove foreign matter from the top of the pool water to maintain the clarity required for visual observation of the spent fuel pool. The system consists of a strainer, pump, and a filter and associated valving, piping, and instrumentation. The skimmer system will operate independently from the spent fuel coolant demineralizer and filter to maintain surface clarity of the pool water. |11

7. Pipes and Valves - All piping in contact with the spent fuel pool water will be stainless steel. The piping will be welded except where flanged connections are used at the heat exchangers, the pumps, and the filters. The equipment is isolated by manual stop valves, and manual throttle valves are used for flow control. Valves in contact with the pool water are constructed of stainless steel.

9.1.3.2.2. Mode of Operation

1. General - The spent fuel cooling system will normally operate to fulfill main functions. The first is to maintain the pool water at temperatures less than 150F with one-third of a core of stored fuel at expected operating histories and with one pump and cooler operating. The second function is to purify the spent fuel pool coolant so that it will be of acceptable clarity during fuel handling operations. 11

   Prior to transferring the spent fuel from the reactor core to the spent fuel storage pool, the pool is initially filled with demineralized water and then borated. The selected fuel assemblies are then transferred to the storage pool.

2. Spent Fuel Storage Pool Recirculation Mode - The first function of the system is to recirculate the storage pool water from the pool through the pump and cooler and back to the pool. The spent fuel pump will take suction from the spent fuel pool, deliver the water through the tube side of the cooler and return it to the pool.

3. Spent Fuel Cooling System Purification Mode - The spent fuel cooling system provides a purification loop to maintain water clarity in the spent fuel storage pool, fuel transfer canal, and the borated water storage tank (BWST) following refueling.

   The purification loop consists of one demineralizer and two 3-micron filters. Manually operated stop valves are provided at the inlet and outlet to the demineralizer. A bypass equipped with a manually operated stop value is also provided for bypassing the demineralizer. Sluicing connections are also provided to the waste disposal system to permit disposal of spent demineralizer resins. Resin fill connections are also provided to replenish resins.

   The discharge from the demineralizer is passed through either of two full capacity filters arranged in parallel. Each filter is provided with a manually operated stop valve at its inlet and outlet to permit isolation for replacement of cartridges and maintenance. The discharge from the filters is normally returned to the pool via the spent fuel storage pool return line.

   The fuel pool skimmer loop consists of fuel pool surface skimmers, skimmer pump suction strainer, skimmer pump and skimmer particles and other materials. The storage pool has overflow weirs to provide skimming of pool surface for removal of debris that may tend to float and reduce the ability to visually observe fuel handling operations in the pool. The undesired particles are removed by pumping through the skimmer filter, and the effluent is returned back to the pool. A strainer is located at the inlet of the skimmer pump for removal of relatively large particles.

During refueling, it is necessary that the refueling canal water purity be maintained. This is done by recirculating canal water from the bottom of the canal by use of the borated water recirculation pump, which discharges to the spent fuel system demineralizer and filters from where it is directed to the suction of the decay heat removal pumps. The water then passes into the reactor vessel and flows back into the refueling canal. A check valve is provided in the line to the suction of the decay heat pumps to prevent BWST water from flooding the spent fuel pool in the event valves are improperly operated.

If leaking fuel assembly is transferred from the fuel transfer canal to the spent fuel pool, a small quantity of fission product activity may enter the spent fuel pool cooling water even though the assembly's cladding temperature is lowered and leakage may reasonably be expected to be minimized. The purification loop will remove these fission products and other contaminants from the water. Special provisions will be made for the storage of defective fuel assemblies.

A larger, but still small, quantity of contaminants can be expected to appear in the borated water storage tank (BWST) after refueling since the water has been intimately mixed with the reactor coolant in the refueling canal. Connections are provided to permit post-refueling purification of this water directly from the BWST. To accomplish this, the borated water recirculation pump draws water from the BWST and directs it through the demineralizer and filters of the spent fuel cooling system and returns it to the BWST. Suitable manually-operated stop valves are provided for controlling water flow to the demineralizer, or for isolating these connections when not in use.

4. Filling and Draining Refueling Canal - During the refueling operation, the refueling canal is filled with borated water from BWST using the spent fuel coolant pumps. The canal can also be filled with borated water using the decay heat pumps. This should be considered as a backup operation due to possible spreading of radioactive crud and clouding of the water.

   When the refueling operation is completed, the spent fuel coolant pumps are used to transfer the bulk of the water back to the BWST. The decay heat pumps can also be used to drain the canal down to the level of the reactor vessel flange, then the spent fuel pumps must be used. Remaining fluids are drained to the reactor building normal sump.

5. Special or Infrequent Operation - A special operation of the cooling system involves the removal of decay heat from 1-1/3 core stored in one of the pools. This unlikely situation is assumed to occur after the removal of the entire core assembly from a unit shortly following

the removal of the normal 1/3 core reload. The maximum heat load consists of 1-1/3 cores stored in the spent fuel storage pool irradiated as given in 9.1.3.1. Both the manual and standby pumps and heat exchangers will be in operation to handle the heat load to maintain the pool temperature at or below the maximum designed temperature. In the event of a loss of spent fuel cooling, cooling will be provided by the use of the corresponding decay heat removal system. Suitable connections are provided from the spent fuel cooling system to the suction side of the decay heat pumps passing through the decay heat coolers and back to the spent fuel cooling system. One cooler may be 11
shut down for servicing without endangering the system, the standby cooler taking its place. No special provisions are taken for shutdown, since any part of the system may be shut down without serious consequences.

During operation of the spent fuel cooling system, a source of demineralized water is available as makeup for the spent fuel storage pool. A connection from the chemical addition batch controller to the spent fuel pump suction provides chemicals for maintaining spent fuel storage pool boron concentration. A pool boron concentration similar to that of the reactor coolant concentration is required since, during spent fuel transfer, the pool water mixes with the reactor coolant. Repair of equipment located in the fuel transfer canal during refueling operations is also considered as an infrequent 11
operation. To permit access to the fuel transfer canal, canal drainage is required. This is accomplished by utilizing a portable submersible pump to transfer water to the BWST. A flow path from the BWST to the transfer canal is available for refilling the canal.

9.1.3.3. Safety Evaluation

9.1.3.3.1. Reliability Considerations

The spent fuel cooling system will provide adequate capacity and component redundancy to ensure that stored spent fuel is cooled even when unexpectedly large amounts of fuel are in storage. Ample time will be available to ensure that protective action can be taken even in the unlikely event of multiple component failures or complete cooling loss. The system will be arranged so that piping or component failures cannot cause an uncontrolled loss of water from the pool. The system will not perform an emergency function and will not be directly connected to the RC system.

9.1.3.3.2. Codes and Standards

Each component of this system will be designed to the applicable code or standard as noted in Table 9.1-2.

9.1.3.3.3. System Isolation

The major penetration of the reactor building will be through the fuel
transfer tubes. The fuel transfer tubes will be isolated inside the re-
actor building by a blind flange connection in the refueling canal. 11
Isolation is provided for fill and drain lines.

9.1.3.3.4. Leakage Considerations

The design of the spent fuel cooling system will consider potentially radioactive service by limiting or restraining leakage of possibly radioactive fluid. The system will be constructed entirely by welding except for sections requiring bolted flanges for maintenance. Principal valves will have stem leakoff provisions.

9.1.3.3.5 Failure Considerations

The failure of a single active component in this system will not permit
uncovering of the stored spent fuel because of pool water boil-off under
normal operating conditions -- since the system design will incorporate
a manual full-capacity spent fuel coolant pump for each pool 11
and a common pump to back up either units' spent fuel cooling pumps. There
will be a full capacity spent fuel cooler for each spent fuel pool and a
common standby for both. If there is a complete loss of cooling, consider-
able time will be required to raise the temperature of the pool water to
boiling; during this time, cooling can be restored through interconnections
with the decay heat removal system.

9.1.3.4. Inspection and Testing Requirements

Active and passive components of the spent fuel cooling system will be examined periodically to determine the operating condition. Periodic visual inspections and preventive maintenance will be conducted according to sound maintenance practice.

9.1.3.5. Radiological Considerations

The fuel handling and storage area housing the spent fuel pool will be monitored for gaseous activity, and gases will be exhausted to the environment through the unit vent. The design will provide for air-testing the flanged end of the fuel transfer tubes for leaktightness after use. Further radiological evaluation of the system is presented in Chapters 11 and 12.

9.1.3.6. Instrumentation Application

Instrumentation application in the spent fuel cooling system is as follows:

1. The following process variables are measured and locally indicated, and a signal is transmitted that will actuate alarms in the control room.

a. Spent fuel coolant filter differential pressure.

b. Spent fuel purification strainer differential pressure.

c. Spent fuel pump suction strainer differential pressure.

d. Spent fuel skimmer filter differential pressure.

e. Spent fuel skimmer strainer differential pressure.

2. The spent fuel purification demineralizer differential pressure is measured, and a signal is transmitted that will actuate alarms in the control room.

3. The following process variables are measured and locally indicated:

   a. Spent fuel storage pool purification flow.

   b. Spent fuel coolant pump discharge pressure.

   c. Borated water recirculation pump discharge pressure.

   d. Spent fuel storage pool temperature.

   e. Spent fuel skimmer pump discharge pressure.

4. The spent fuel storage pool level is measured, and a signal is transmitted that will actuate alarms and provide indication in the control room.

5. The spent fuel coolant flow is measured and locally indicated, and a signal is transmitted to the plant computer for indication and alarm.

6. The following process variables are measured, and a signal is transmitted to the plant computer for indication and alarm.

   a. Spent fuel storage pool temperature.

   b. Spent fuel coolant demineralizer inlet temperature.

9.1.4. Fuel Handling System

The fuel handling system provides a safe, effective means of transporting and handling fuel from the time it reaches the station in an unirradiated condition until it leaves the station after postirradiation cooling. Those components within the system which transport, position, raise, lower or house the fuel assembly during transport shall be designed to meet the requirements of a Catoegory I structure and Safety Guide 13.

Also the system will be designed and constructed to minimize the possibility of mishandling or maloperations that could damage fuel assemblies or cause a release of fission products. The requirements of the AEC General Design Criterion 62 will be fulfilled by the design of the fuel handling equipment and storage pool.

The reactor will be refueled by using equipment designed to handle the spent fuel assemblies under water from the time they leave the reactor vessel until they are placed in a cask for shipment from the site. Underwater transfer of spent fuel assemblies will provide an effective, transparent radiation shield as well as a reliable cooling medium for the removal of decay heat. The use of borated warer will ensure subcritical conditions during refueling.

9.1.4.1. Design Bases

The fuel handling system consists of equipment and structures utilized for handling new and spent fuel assemblies in a safe manner.

The followng design bases apply to the fuel handling system:

1. Fuel handling devices have provisions to avoid dropping or jamming of fuel assemblies during transfer operation.

2. Fuel lifting and handling devices are capable of supporting maximum loads under safe shutdown earthquake conditions.

3. The fuel transfer system, where it penetrates the containment, has provisions to preserve the integrity of the containment pressure boundary.

4. Cranes and hoists used to lift spent fuel have a limited maximum lift height so that the minimum required depth of water shielding is maintained.

9.1.4.2. System Description

The fuel handling equipment consists of the equipment needed for refueling the reactor.

Basically this equipment is comprised of:

1. Cranes and handling equipment.

2. A fuel transfer system.

3. Fuel storage racks.

The following structures are associated with the fuel handling equipment:

1. Reactor cavity.

2. Refueling canal.

3. Spent fuel storage pit, including transfer canal and cask loading area.

4. New fuel storage area.

5. Decontamination facilities

6. Cask conveyance handling area.

New fuel is received in containers which hold 2 assemblies. A shipment of new fuel is received in the rail car and truck room in the auxiliary building. Containers are unloaded using the auxiliary building crane. The assemblies are unpacked and inspected, and are installed in the new fuel storage vault. The new fuel assemblies are handled with a new fuel handling tool and special lifting slings suspended from the auxiliary hook of the building crane.

New fuel to be transferred to the reactor is removed from the storage vault using the new fuel handling tool on the auxiliary hook. The new fuel assemblies are transferred from the new fuel storage vault,one at a time, to the new fuel elevator located in the cask loading area. The assemblies are placed in the new fuel elevator which has been raised to the up position, to receive a new fuel assembly. The new fuel handing tool is disengaged from the assembly, raised to the up position and moved from over the new fuel elevator. The new fuel elevator containing the new fuel assembly is lowered to the bottom position. The new fuel assemblies are removed from the container using the fuel storage handling bridge and are placed either in the fuel transfer system or in the spent fuel storage racks.

Figure 9.1-1 shows the fuel handling and storage facilities schematically. Although the figure does not conform to the actual plant layout as shown in Figures 9.1-4 and 9.1-5, it shows the essential features of the system.

A refueling outage commences with shutdown of the reactor. Following shutdown and entry of the refueling crew into the reactor building, refueling is initiated with removal of the missile shield. The annular space between the reactor vessel flange and the bottom of the refueling canal is sealed off by a seal clamped to a canal shield plate flange and the reactor vessel flange. Removal of the reactor closure head, control rod drives, and their service structure is initiated. Two stud tensioners are used to minimize the time required to remove and replace the head. The stud tensioners are hydraulically operated devices that permit preloading and unloading of the reactor closure studs. Each tightens or loosens one stud at a time but both can be used simultaneously at stud locations 180 degrees apart. During preloading, the studs are tensioned to their operational load in two steps in a predetermined sequence. The posttensioning elongation of the studs is verified by micrometer measurements. Relief valves are provided on each stud tensioner to prevent overtensioning of the studs due to excessive pressure.

Following removal of the studs from the tapped holes in the reactor vessel, the studs and nuts are supported in the closure head by specially designed spacers. Removal of the studs with the reactor closure head minimizes handling time and reduces the possibility of thread damage.

The reactor closure head assembly is maneuvered by a handling fixture supported from the reactor building crane. It is lifted out of the canal onto a head storage stand. The stand is designed to protect the gasket surface of the closure head. The lift is guided by two closure head alignment studs installed in two of the stud holes. These studs also provide proper alignment of the reactor closure head with the reactor vessel and internals when the closure head is replaced after refueling. At the head storage location, special stud and nut handling fixtures can be used to remove the studs and nuts from the reactor closure head for inspections and cleaning. A stud storage rack is provided. After these operations, the refueling canal is filled with borated water from the borated water storage tank. The reactor building crane and an internals handling adapter are used to remove the plenum assembly from the reactor and store it under water on a stand that is provided on the floor of the refueling canal.

Refueling operations are carried out from two refueling bridges spanning the refueling canal. The main bridge is used to shuttle spent fuel assemblies from the core to the transfer station and new fuel assemblies from the transfer station to the core. During this operation, the auxiliary bridge is used to relocate partially spent fuel assemblies in the core as specified by the fuel management program. |11

Fuel assemblies are handled by pneumatically operated fuel grapple attached to a telescoping mast mounted on a trolley that moves laterally on each bridge. Control and orifice rod assemblies are handled by a grapple attached to a second mast on the main bridge in the reactor building.

The main (2-mast) bridge moves a spent fuel assembly from the core under water to the transfer station from which the fuel assembly is lowered into a fuel transfer carriage fuel basket. A grapple attached to the second mast is used to transfer a control rod assembly to a new fuel assembly. The main refueling bridge carries this new fuel assembly with the control rod assembly to the reactor and places it in the core while the spent fuel assembly is being transferred to the spent fuel storage pool. |11

The fuel handling and control rod handling mechanisms are designed so that the fuel and rod assemblies are withdrawn into the mast tube for protection before transfer. Interlocks are provided to prevent operation of the bridges or trolleys until the assemblies have been hoisted to the upper limit in the mast tube. Mandatory slow zones are provided for the hoisting mechanisms during insertion of fuel and rod control assemblies. The slow zones are in effect during insertion into and removal from the reactor core, fuel transfer basket, or fuel storage racks. The controls are appropriately interlocked to prevent simulatneous movement of the bridge, trolley, or hoists. The grapple mechanisms are interlocked with the hoists to prevent vertical movement unless the graples are either fully operated or fully closed. The fuel grapple is designed so that when loaded with the fuel assembly, it cannot be opened as a result of operator error or electrical or hydraulic failure. Hard stops are provided to prevent raising an assembly above the minimum shielding depth in the event of an uplimit failure.

New fuel is moved into the reactor building, and spent fuel is moved from the reactor building by means of the fuel transfer systems, of which there are two per reactor unit.

Each transfer system consists of a fuel transfer tube, two rotating frames, one carriage and fuel basket, and control consoles in the reactor building and in the auxiliary building. The transfer tube has a gate valve closure in the auxiliary building and a manually installed and removed gasketed cover plate in the reactor building. See Figure 9.1-1 showing the two transfer systems per reactor unit.

New fuel is moved from the spent fuel storage area into the transfer canal and is placed in the fuel basket, which is in a vertical position in the rotating frame. The frame is rotated to the horizontal, and the fuel in the basket is moved through the transfer tube on the transfer carriage. In the reactor building, the basket is rotated to the vertical by the rotating frame. The fuel assembly is picked up by the fuel handling mechanism on the main refueling bridge and is transferred through the refueling canal to the space above the reactor. |11

The fuel transfer mechanisms are underwater, air motor-driven carriages that run on tracks extending from the transfer canal through the transfer tubes and into the refueling canal in the reactor building. The rotating fuel basket mounted on each fuel transfer carriage receives fuel assemblies in a vertical position. The fuel basket is rotated to a horizontal position for passage through the transfer tube and then rotated back to a vertical position in the refueling canal for vertical removal of the fuel assembly. The fuel transfer mechanisms are designed to permit rotation of the carriage fuel basket to be initiated from the building in which it is being loaded or unloaded. Each transfer tube contains tracks for the fuel transfer carriage, a gate valve on the auxiliary building side, and flange closure on the reactor building side. The closure plate has two gaskets inside the bolt circle diameter. A leak test channel is welded to the back side of the transfer tube flange and to the transfer tube to provide a test annulus concentric with the flange-to-tube weld. A test port drilled in the transfer tube flange allows the annulus between the seals and the leak test channel to be pressurized with plant air to leak test the seals and the tube-to-flange weld.

All electrical equipment is located above water for greater reliability and ease of maintenance. The hydraulic system that actuates the rotating fuel basket does not use storage pool water for operation to minimize contamination. Demineralized water is used.

The simplified movement of a transfer carriage through the horizontal fuel transfer tube minimizes the danger of jamming or derailing. All operating mechanisms for the transfer carriage system are located in the auxiliary building for ease of maintenance and accessibility for inspection before the start of refueling operations.

All spent fuel assembly transfer operations are conducted under water. The water level in the refueling canal provides adequate shielding over the top of the active fuel in the spent fuel assemblies during movement from the core into storage. The depth of the water over the fuel assemblies and the thickness of the concrete walls of the transfer canal is sufficient to limit the maximum radiation levels in the working area to a safe working level.

The fuel storage handling bridge (one per reactor unit) travels on rails. It covers an area which includes the transfer canal, the spent fuel storage area, and the cask loading area. Fuel assemblies are handled with a hydraulically operated fuel grappel attached to a telescoping mast. The mast is mounted on a trolley that moves laterally on the fuel storage handling bridge. A mechanical stop is provided to prevent raising the top of a fuel assembely closer than 9 feet 6 inches from the normal pool surface. As with all of the bridges, the bridge, trolley, and hoist functions can be accomplished manually in the event of a power failure to return equipment and components being handled to a safe shutdown position.

During the refueling period, the water level in both the refueling canal and the spent fuel storage pool is the same since the fuel transfer tube valve will be continuously open. This eliminates the need for an interlock between the fuel transfer carriage and fuel transfer tube valve operations except to verify full-open valve position.

Water in the fuel transfer canal, the refueling canal and the reactor vessel has a minimum boron concentration of 1800 ppm. Although not required for safe storage of spent fuel assemblies, the spent fuel storage pool water is borated so that water in the refueling canal and the reactor vessel is not diluted during fuel transfer operation. The boron concentration of the reactor vessel water is sufficient to maintain core shutdown if all of the control rod assemblies were removed from the core even though only a few control rods will be removed at any one time during the fuel shuffling and replacement.

During operation of the reactor, the carriages will be stored in the spent fuel storage pool. A bolted and double o-ring gasketed closure plate on the reactor building flange of the fuel transfer tube provides containment isolation.

The refueling canal in the reactor building and the transfer canal, fuel storage area, and cask loading area in the auxiliary building are lined with stainless steel for leaktightness and ease of decontamination. The fuel transfer tubes are appropriately attached to these liners to maintain leak integrity. The spent fuel storage pool is designed to preclude accidental drainage.

After refueling is completed, the water in the reactor building refueling canal is drained and pumped to the BWST.

Use of the fuel handling equipment follows a preplanned operational procedure, which permits safe handling of spent fuel. Interlocks, bridge travel and load limiting devices, and other protective measures prevent accidents that could cause fuel assembly damage or potential fission product release. The system does not impose limits on reactor operation.

Before each refueling operation, the fuel handling equipment is visually inspected to determine its condition. Repairs and servicing indicated by this inspection are performed before the refueling operation. Equipment such as bridges and hoists will be operated to ensure reliable performance for the refueling operation.

Suspected defective fuel will be removed from the core and tested for leakage. Leakage testing will be performed in the deep end of the reactor building refueling canal in a container that can be sampled for fission products. Leaking fuel assemblies that are removed from the core and verified for leakage will be transferred to the spent fuel pool via the fuel transfer tube. The failed fuel, placed in a failed fuel container, is stored in the spent fuel storage rack. Following a suitable decay period, the fuel will be transferred to a spent fuel shipping cask for shipment to a reprocessing plant. The cask is designed in compliance with 10 CFR 71.

9.1.4.3. Spent Fuel Shipping Cask Handling

Spent fuel is shipped in a cask weighing approximately 100 tons and having a capacity of 10 PWR fuel assemblies.

The cask will be shipped by rail to the plant site from the fuel reprocessing facility. Rail facilities are provided directly to the cask receiving area in the auxiliary building.

In the cask receiving area, the personnel exclusion frame which surrounds the cask is removed using the auxiliary building crane. Tiedowns and any other necessary shipping equipment are removed and, if necessary, this equipment may be stored in the receiving area.

The cask is raised to the vertical position using the auxiliary building overhead crane. Next, the cask is moved to the cask decontamination facility for cleaning. If temporary cask storage is required, the cask is moved to the cask storage area. The cask is kept within 6 inches above the floor during its transfer. After the cask has been properly positioned in the cask storage area and the tiedowns are attached, the crane may be disengaged from the cask.

The spent fuel assemblies to be shipped to the fuel reprocessing facility are stored in the spent fuel storage pool which is adjacent to and separated from the cask loading area by a removable gate. When the spent fuel assemblies are ready for shipment, the cask is transferred to the cask loading area. Handrails are removed and replaced as required. The cask head is removed, after which the gate separating the cask loading area and the spent fuel storage pool is removed. Space is provided for temporary storage of this equipment. The spent fuel is moved from the storage racks to the shipping cask with the fuel storage handling bridge. During this operation the fuel assemblies are always under water.

After the fuel assemblies have been loaded into the cask, the cask head is replaced. The loaded cask is lifted from the cask loading pit with the overhead crane. As it emerges from the water, it is rinsed with demineralized water. It is then transferred to the adjacent cask decontamination facility.

The cask decontamination facility is located in a pit to prevent the spread of radioactive contamination as a result of splashing. The cask is positioned on a circular rotating platform located on the floor of the pit, and the pit cover is installed. As the cask is slowly rotated, jets of high-pressure water wash the entire surface of the cask. The jets are mounted on an elevator and move upward or downward over the length of the cask. The pit is equipped with a personnel elevator platform to allow all areas of the surface of the cask to be checked for radioactive contamination. If excessive contamination is detected, the cleaning operation is repeated.

When the decontamination of the cask has been completed, the cask is again transferred to the cask storage area. As described above, tiedowns are attached to the cask and the crane hook is disengaged while preshipment tests are conducted.

When the loaded cask is ready for shipment, it is transferred to the cask receiving area using the overhead crane. In the cask receiving area the cask is lowered to the rail car, connected to the support frame, and lowered to the horizontal position. All tiedowns are replaced and the cask is secured to the support frame and rail car. Cask cooling equipment remains on the rail car during the cask handling operation. This equipment is placed in operation when the cask is in place on the rail car. Temporary cooling equipment is connected to the cast at other locations as required. After the exclusion frame is attached to the rail car, the cask is ready for shipment to the fuel reprocessing facility.

9.1.4.4. Component Description

Main Fuel Handling Bridge (One Per Unit)

The main fuel handling bridge, located in the reactor building, operates between the reactor core and the fuel transfer stations (or fuel storage racks and the failed fuel detection can in the deep portion of the refueling canal near the transfer station). One mast assembly is provided for fuel assembly handling, and one mast assembly is provided for control component (control rods, orifice rods, axial power shaping rods, and burnable poison rods) handling. Provision is made for installing a closed-circuit television system between the fuel handling mast and control component handing mast on the trolley of the main bridge.

Auxiliary Fuel Handling Bridge (One Per Unit)

The auxiliary fuel handling bridge, located in the reactor building, operates over the reactor core area to move partially spent fuel assemblies from one location to another in the core as required by the fuel management program. A fuel handling mast assembly identical to the main fuel handling bridge is provided. Bridge and trolley construction, positioning, mast rotation, and hoist functions are identical to the main fuel handling bridge. Space is provided for installation of a closed-circuit television system.

Fuel shuffle operations are performed during the travel-unload-load-return cycle of the main fuel handling bridge.

Fuel Storage Handling Bridge (One Per Unit)

The fuel storage handling bridge, located in the auxiliary building, operates over the transfer canal fuel storage pool and cask loading area. It is used to move fuel assemblies from the new fuel transfer container to the fuel storage racks, from the storage racks to the transfer system basket, from the transfer system basket to the storage racks, and from the storage racks to the spent fuel shipping cask.

A control component handling mast installed on the fuel storage handling bridge makes this bridge identical to the main fuel handling bridge (except for bridge span).

Closed-circuit television may be installed on the fuel storage handling bridge.

Bridges - General

The bridge and tolley have 5-speed step control. The fuel and control rod handling masts have 2-speed control.

Fuel handling mast speed is 5 feet per minute in the core, storage racks, and transfer basket, and 20 feet per minute at other elevations.

Bridge and trolley trucks are supplied with flame-hardened flat wheels, with guide rollers on one bridge truck and one trolley truck. Antiderailing and overturning devices are provided for the bridges and trolley.

Controls are interlocked so only the bridge, trolley, one of the hoists, or the TV mast can be operated at one time. The trolley or bridge can be moved only when the fuel grapple, control rod grapple, or TV mast is in the "up" position. Indicating lights, bypass switches, and bypass indicating lights are provided.

Bridge positioning is accomplished by the operator observing a precalibrated selsyn-operated dial with hour and minute hands position readout. Two readouts are provided, one for the fuel handling mechanism and one for the control rod handling mechanism. The trolley is positioned by visual reference to a pointer moving along a fixed scale with precision markings.

Vertical position indication is provided by operator observation of a numbered tape which is attached to the fuel handling inner mast. Two tapes are used to give the position of the control rod handling mast. One tape is attached to the handling mast and another to the control rod grapple. Tapes are automatically retracted by a spring motor driven reel.

Manual release of the fuel grapple and control rod grapple can be accomplished with the use of proper procedures.

A load monitor measures and displays loads on the grapples. The monitor provides the limiting circuits for slack cable (low load), overload, opening the grapples, and a test circuit for the orifice rod assembly.

The fuel handling mast is designed for manual rotation through an angle of 180 degrees. The manual rotating mechanism is located on the upper part of the mast and can be actuated by the operator while standing on the bridge trolley.

The fuel handling mast is designed to permit a manually actuated lift of 6000 pounds (maximum). The control rod handling mast is designed to be self-limiting at 375 pounds (maximum push or pull), although greater loads than this can be applied manually.

Bridges, trolley, and hoist functions may be operated manually in event of a power failure to return equipment and components being handled to a safe shutdown position.

Rails for the fuel handling bridges are 40-pound ASCE; the guide rail is machined on the top, bottom, and sides of the rail head. Rails are furnished with alignment plates on approximately 1-foot centers. The alignment plates have four leveling screws.

The guide rail is level and straight within 0.030 inch. The other rail is at the same elevation. level within 0.030 inch.

Bridge stops are provided. Alignment plates are attached to embedments in the floor.

Closed-Circuit Television System (Normally operated from the Main Handling Bridge)

A self-contained closed-circuit television system for underwater viewing is installed on the main fuel handling bridge. It can also be installed on the auxiliary or the fuel storage bridge. The system includes a radiation-resistant camera in a stainless steel waterproof container, high resolution monitor, telescoping stainless steel mast, motorized mast rotation (approximately 300 degrees), control box with pendant switches, all necessary cable and cable reels, remote optical focus, integral lighting, interlock of telescoping mast with bridge and trolley (with override switch), and other items necessary for submerged operation.

The camera can be tilted 37 degrees from the vertical centerline to permit reading the control component identification numbers

Fuel Transfer System (Two Per Unit)

The complete fuel transfer system consists of two roating frames, one carriage and fuel basket, a fuel trasnfer tube with a gate valve in the auxiliary building, and a manually installed and removed gasketed cover plate in the reactor building, and control consoles are located in the reactor building and in the auxiliary.

Two transfer basket rotating frames are provided for each transfer system, one for the auxiliary building and one for the reactor building. As the carriage and fuel basket approaches a rotating frame, the fuel basket engages the rotating frame, and after reaching its travel limit, a hydraulic cylinder rotates the fuel basket to a vertical attitude. The fuel handling bridge then removes from or inserts a fuel assembly into the fuel basket. The fuel assembly may contain a control or orifice rod.

A self-propelled carriage with a moving underwater air motor and trailing air hoses is used to transport the fuel basket which pivots at the center of the carriage.

If a carriage should become hung up in the fuel transfer tube or an air line should be lost, an emergency pullout cable can be attached to the fuel storage building crane to withdraw the carriage into the fuel storage building.

The air motor supply and exhaust hose reels and the emergency pullout cable reel are fitted with spring motors which automatically pay out and retract hose and cable as the carriage moves.

Control panels in the reactor buildings control only the rotating frames in the reactor buildings. Control panels in the fuel storage building control the rotating frames in the fuel storage building and the horizontal travel of the carriages. Carriages and rotating frames are interlocked to prevent inadvertent movement of carriages when baskets are in the vertical position and rotation of the fuel baskets when the carriages are not in the home positions.

Fuel Transfer Tube and Gate Valves (Two Each Per Unit)

The fuel transfer tube serves as a passageway between the reactor building and the auxiliary building during refueling. The portion of the fuel transfer tube inside the reactor building is an integral part of the reactor containment during reactor operation.

The tube is welded to the refueling canal wall liner and bellows expansion sleeves are field welded to the tube and the primary containment liner and the fuel transfer canal liner. The transfer tube outside diameter is 30 inches.

A gasketed cover plate is bolted to the transfer tube flanged end in the reactor building. A test connection is provided to permit the cover plate gaskets and the flange-to-tube weld to be leak tested after the cover plate is replaced at completion of refueling operation.

A tube support in the fuel storage building supports the transfer tube and a gate valve which is bolted to the flanged end of the tube. The support is designed to accommodate transfer tube thermal expansion and seismic loads.

The manually operated gate valve is locked closed during reactor operation. A drain connection is provided in the tube to monitor the gate valve if it should leak during the reactor operating period.

The fuel transfer carriage assembly moves through the transfer tube on rails welded to the inside of the transfer tube. The transfer carriage drive sprocket engages a roller chain welded to the transfer tube to provide means for driving the fuel transfer carriage.

An electrical interlock prevents the fuel transfer carriage from being moved until the gate valve is fully open.

New Fuel Transfer Container (One Per Plant)

The new fuel transfer container is used to lower fuel assemblies from the auxiliary building working floor level to the cask loading area where fuel assemblies are removed with the fuel storage handling bridge and placed in the fuel storage racks or the transfer carriage basket.

The container has compartments for four fuel assemblies. Suitable rigging to the auxiliary building crane permits loading fuel assemblies into the container at the auxiliary building working floor. The container is lowered to rest on a prepositioned index plate on the pool floor, and fuel assemblies are removed using the fuel storage handling bridge.

New Fuel Handling Tool (One Per Plant)

The new fuel handling tool is used to remove and transfer new fuel assemblies from the shipping containers to the new fuel storage racks or the new fuel transfer container for insertion into the auxiliary building spent fuel pools.

It is suspended from the building hoist by a 2-bridled sling for lowering over the upper end fitting of the vertical fuel assemblies. When the tool is seated on the fuel assembly end fitting, the grapper fingers are in proper alignment with the established lifting areas on the end fitting and can be manually moved inward beneath the lifting bars. Projections on the gripper fingers prevent them from being disengaged until the fuel assembly load has been relaxed.

The tool can handle a fuel assembly with or without an inserted rod assembly and is normally used and stored in the auxiliary building. This tool can be attached to a fuel assembly, new or irradiated, or to a rod assembly handling container, under water if necessary, to supplement the bridge-mounted grapple. If used for this purpose, the tool is guided into postiion by a long-handled hook or guide tool and a long-handled wrench tool is used to actuate the grippers.

Rod Assembly Handling Tool (One Per Unit)

The rod assembly handling tool is a manually operated grapple used primarily for handling new control and orifice rod assemblies. It can also be used for handling axial power shaping rod assemblies or burnable poison rod assemblies. It consists of a stainless steel tube with lifting handle at one end and a removable grapple head at the other end. Vertical movement of a center rod moves three balls into an internal groove in the rod assembly coupling for engagement of the tool to the rod assembly. Stepped slots in the center rod actuating sleeve postiion the center rod in the "open," "check," or "locked" position. Rotating the knurled sleeve 120 degrees moves the rod down approximately 1 inch to disengage the orifice (or burnable poison) rod assembly from the fuel assembly and lock the tool onto the rod assembly for removal. Rotating the sleeve back 60 degrees will return the rod approximately 1/2 inch to the "check" position. When in this position, a pull on the tool will verify that the rod assembly is firmly latched onto the fuel assembly after rod assembly has been reinstalled into another fuel assembly. The tool, approximately 18 inches long, is required for removing and transferring new rod assemblies from their shipping containers to new fuel assemblies or into the rod assembly handling container for transfer into the storage pool. It is suspended from the building hoist by a single leg sling.

If an irradiated rod assembly must be removed from a spent fuel assembly in the fuel storage pool, the grapple head must be unscrewed from the short handle and screwed into an alternate long handle. The grapple head can be guided onto a submerged rod assembly with the aid of a long-handled hook or guide tool. A rod assembly cannot be reinserted into a fuel assembly under water with this tool.

This tool is normally used and stored in the auxiliary building.

Rod Assembly Handling Container (One Per Plant)

The rod assembly handling container has the same external size envelope as a fuel assembly and consists of a modified fuel assembly lower end fitting and upper end fitting connected by four corner angles with sixteen guide tubes positioned between the end fittings to provide guidance for the rod assembly rods. It will accept an orifice rod, control rod, burnable poison rod, or axial power shaping rod assembly for temporary storage or for handling and transfer between the reactor buildings and the auxiliary building.

This container can be handled by either the fuel handling bridge fuel handling mechanism grapple or by the new fuel handling tool.

This container could be located in a space in the fuel handling rack in the reactor building canal to facilitate rod assembly shuffling but it would normally be located in the auxiliary building, either in a space in a new fuel storage rack or a spent fuel storage rack for use in transferring a new rod assembly into the reactor building.

An irradiated rod assembly can be inserted into or removed from the rod assembly handling container by the main fuel handling bridge control rod handling mechanism grapple. A new rod assembly can be inserted or withdrawn by the control rod grapple or by the rod assembly handling tool.

Long-Handled Tools

The long-handled tools consist of a segmented handle to which can be attached various types and shapes of tool heads. The square aluminum handle consists of a 20-foot-long main section with lifting bail for attachment to a hoist hook and a 15-foot-long adapter section connected to the main section by two ball-lock pins. The removable stainless steel tool heads supplied with the handle include a hook tool, a guide tool, and several sizes of socket or wrench tools. These heads screw into the end of the adapter section and are locked onto the handle by a jam nut and a slide bar.

An extension section of correct length can be added between the main and adapter handle section to produce a longer tool to permit reaching the top of the core or the bottom of the fuel transfer canal or the fuel storage pool. Two handles will be supplied for use in the reactor building and one for use in the fuel storage building.

The long-handled tools are intended to be used manually to perform miscellaneous guiding and light wrenching operatings associated with control rod and fuel assembly handling.

Borescope (One Per Plant)

This sytem is designed for internal observation of core area and includes interchangeable objectives and viewing heads to alter magnification and field of view and to change the borescope optically to serve as a periscope-type instrument for more distant viewing. All parts are corrosion resistant. The unit is designed for mounting on the fuel handling bridge trolleys and has built-in illumination.

The system includes a 35mm camera for attachment to the eyepiece for photography. The camera will receive either Polaroid film or conventional film.

Internals Storage Stand

The internals storage stand is a stainless steel weldment designed to support the plenum assembly for temporary storage during each refueling period and to support the core support and plenum assembly for temporary storage during reactor vessel internal surface inspection.

Head Storage Stand

The head storage stand is a combination carbon steel and stainless steel weldment designed to support the reactor vessel closure head assembly for temporary storage and maintenance during each refueling period.

Internals Indexing Fixture

The internals indexing fixture is a stainless steel weldment designed to radially orient the plenum assembly and the core support assembly for insertion into and removal from the reactor vessel.

Stud Tensioners

The stud tensioner is a hydraulically operated jacking device which elongates and preloads the reactor closure head studs at cold shutdown conditions to permit tightening or loosening of the stud nuts without the use of torque wrenches or bolt heaters.

Stud Handling Tool

The stud handling tool is an assembly consisting of an air motor, motor reducer assembly, motor and spring support assembly, drive screw, stud hook and transfer yoke, and stud balance springs. This tool removes the 650 lb closure studs from the reactor vessel flange tapped holes after the stud nuts have been loosened and re-inserts the studs into the tapped holes after the closure head assembly has been set onto the reactor vessel.

Stud Tensioner Hoist

The stud tensioner hoist is a commercial 2-ton hand-operated chain hoist with a geared trolley. Two of these hoists are mounted on a circular monorail attached to the closure head service structure for supporting and positioning the stud tensioners over each of the closure head studs.

Stud Hoist

The stud hoist is identical to the stud tensioner hoist. Two additional 2-ton hoists are mounted on the circular closure head service structure monorail for supporting the stud handling tools which remove the studs from the reactor vessel flange tapped holes.

Head and Internals Handling Fixture

The head and internals handling fixture is a carbon steel weldment designed for handling the closure head assembly (reactor vessel closure head, closure head service structure, and control rod drives) and the reactor vessel internals (internals indexing fixture and internals handling adapter which are used for insertion and removal of the plenum assembly and the core support assembly).

Internals Handling Adapter

The internals handling adapter is an assembly designed for handling the two reactor vessel internals assemblies -- the core support assembly during initial insertion into the reactor vessel, the plenum assembly for refueling and the combined plenum and core support assembly for periodic reactor vessel inspection. This adapter is used in conjunction with the head and internals handling fixture, the internals handling extension and the internals indexing fixture for internals handling.

Internals Handling Extension

The internals handling extension is a carbon steel assembly designed to connect the head and internals handling fixture to the main hook of the reactor building crane.

Failed Fuel Container

The failed fuel container is an aluminum receptacle designed to contain loose pieces of grossly failed fuel assembly. It is stored in the spent fuel storage pool.

Stud Nut Wrench

The stud nut wrench is a hand wrench provided for turning the stud nuts on the closure studs. These wrenches should normally not be needed since the nuts should turn freely by hand but one is supplied to help free a jammed nut from a stud.

Stud Tensioner Sling

The stud tensioner sling is a 4 legged bridle sling used for moving each stud tensioner from its storage location in the reactor building into the transfer canal and for transferring the load of the tensioner from the reactor building crane to the stud tensioner hoist.

Stud Impact Wrench

The stud impact wrench is a commercial air operated wrench with a rated torque capacity of 4000 ft-lb. Although this wrench is furnished for the express purpose of dislodging any closure head stud which may become stuck in a reactor vessel flange tapped hole, it is not anticipated that it will be needed. A low torque wrench has been purposely chosen to prevent thread damage to the tapped hole if a stud removal should require torque beyond the capacity of the stud handling tool air motor.

Stud Impact Wrench Adapter

The stud impact wrench adapter is required to couple the square impact wrench shaft to the round closure head stud upper end.

Stud Handling Adapter

The stud handling adapter is required to couple the stud handling tool to the closure head stud or alignment stud upper end.

Stud Nut Handling Fixture

The stud nut handling fixture is a threaded plug which can be attached to the stud nuts for handling when removing the nuts from the studs for inspection.

Stud and Nut Handling Sling

The stud and nut handling sling is a single leg cable sling for handling a closure stud, stud nut or stud, nut and washer set assembly using the reactor building crane. The sling can be attached to the stud nut handling fixture for nut handling or to the stud handling tool transfer yoke for stud handling.

Stud Hole Chase Tool

The stud hole chase tool is a thread tap supplied for chasing the threads of the reactor vessel stud tapped holes to move foreign matter. This tap would only be used in those tapped holes where some difficult was experienced when removing a stud, or for holes into which refueling water may have leaked during the refueling process.

Table 9.1-1. Spent Fuel Cooling System Performance Data

System cooling capacity, Btu/h	
Normal	
1/3 core irradiated 1044 days	14.0×10^6
Maximum	
1-1/3 core irradiated	43.86×10^6 11
System design pressure, psig	200
System design temperature, F	250

Table 9.1-2. Spent Fuel Cooling System Equipment Data

Spent fuel coolers per unit (a common standby cooler is shared by both units)		
Quantity	1	
Type	Tube and shell	
Material, shell/tube	CS/SS	
Design pressure, shell/tube, psig	200/200	
Design temperature, shell/tube, F	250/250	
Heat transferred normal/maximum 10^6, Btu/h	14.04/43.86	11
Flow rate (shell/tube) 10^5, lb/h (shell flow rate is 5.5 x 10^5 when spare is operating)	8.25/8.25	
ASME Code/Class (tube)	III/3	
ASME Code (shell)	VIII	
Spent fuel coolant pumps per unit (a common standby pump is shared by both units)		
Quantity	1	
Type	Horizontal centrifugal	
Material	Stainless steel	
Flow rate, gpm	1650	
Rated head, ft	231	
Design pressure, psig	200	
Design temperature, F	250	
ASME Code/Class	III/3	
Motor horsepower, hp	150	9
Spent fuel coolant demineralizers per unit		
Quantity	1	
Flow rate, gpm	200	11
Resin volume, ft^3	60	
Vessel material	Stainless steel	
Design pressure, psig	200	
Design temperature, F	250	
ASME Code/Class	III/3	

Table 9.1-2. (Cont'd)

Spent fuel coolant filters per unit		
Quantity	2	
Micron size, nominal	3	
Flow rate, gpm	200	
Type	Disposable element	
Vessel material	Stainless steel	
Design pressure, psig	200	
Design temperature, F	250	
ASME Code/Class	III/3	
Borated water recriculation pumps per unit		
Quantity	1	
Type	Centrifugal	
Material	Stainless steel	
Flow, gpm	200	
Head, ft	231	
Design pressure, psig	200	
Design temperature, F	250	
ASME Code/Class	III/3	
Motor horsepower, hp	25	9
Spent fuel storage pool volume, ft^3	43,240	
Spent fuel skimmer inlet strainers per unit		
Quantity	1	
Micron size, nominal	300	
Flow rate, gpm	100	
Type	Disposable element	
Vessel material	Stainless steel	
Design pressure, psig	150	
Design temperature, F	200	
ASME Code/Class	III/3	

Table 9.1-2. (Cont'd)

Spent fuel skimmer pumps per unit

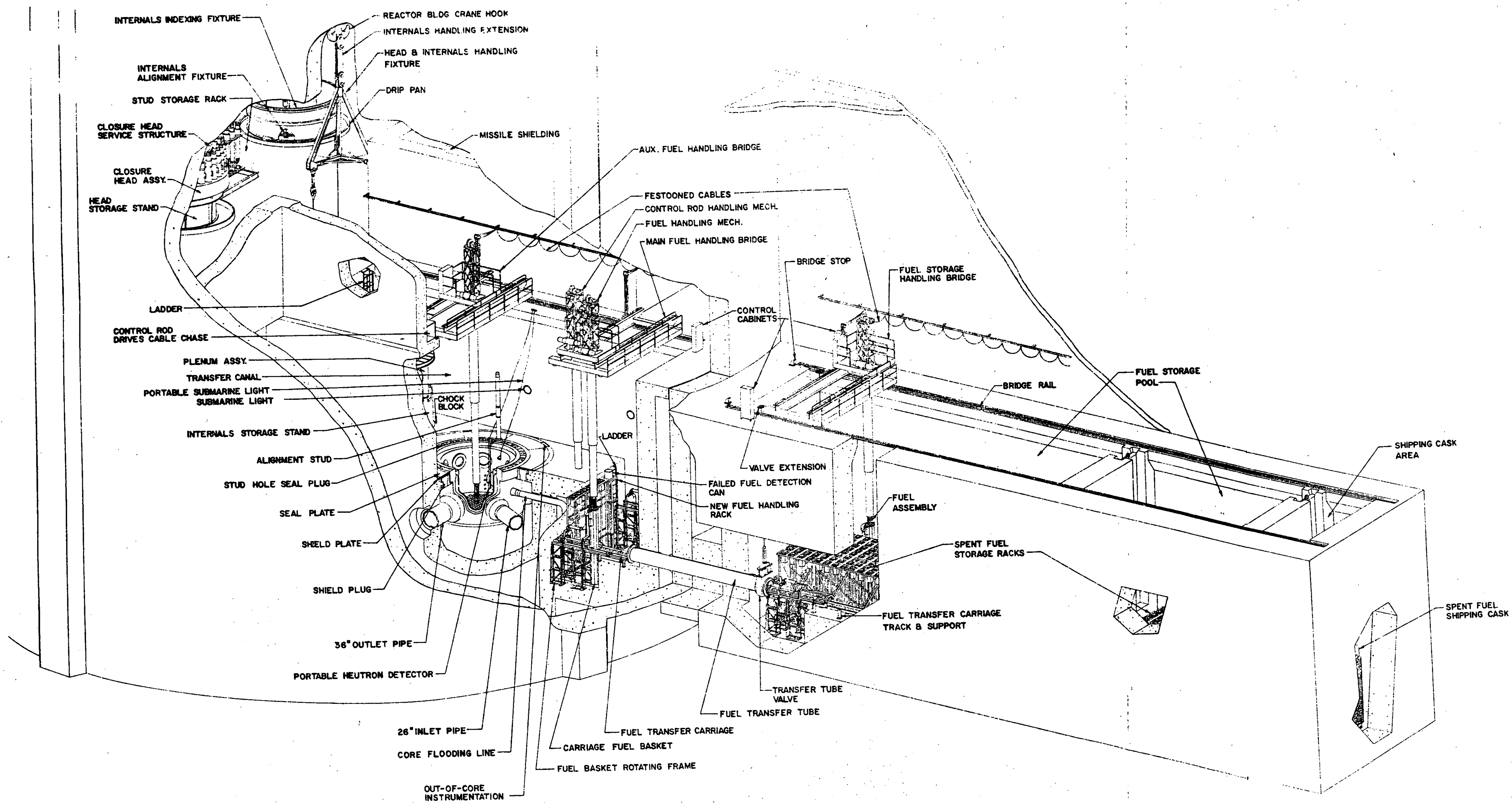
Quantity	1
Type	Centrifugal
Material	Stainless steel
Flow, gpm	100
Head, ft	150
Design pressure, psig	150
Design temperature, F	200
ASME Code/class	III/3
Motor horsepower, hp	10 9

Skimmer filters per unit

Quanity	1
Micron size, normal	25
Flow rate, gpm	100
Type	Disposable element
Vessel material	Stainless steel
Design pressure, psig	150
Design temperature, F	200
ASME Code/class	III/3

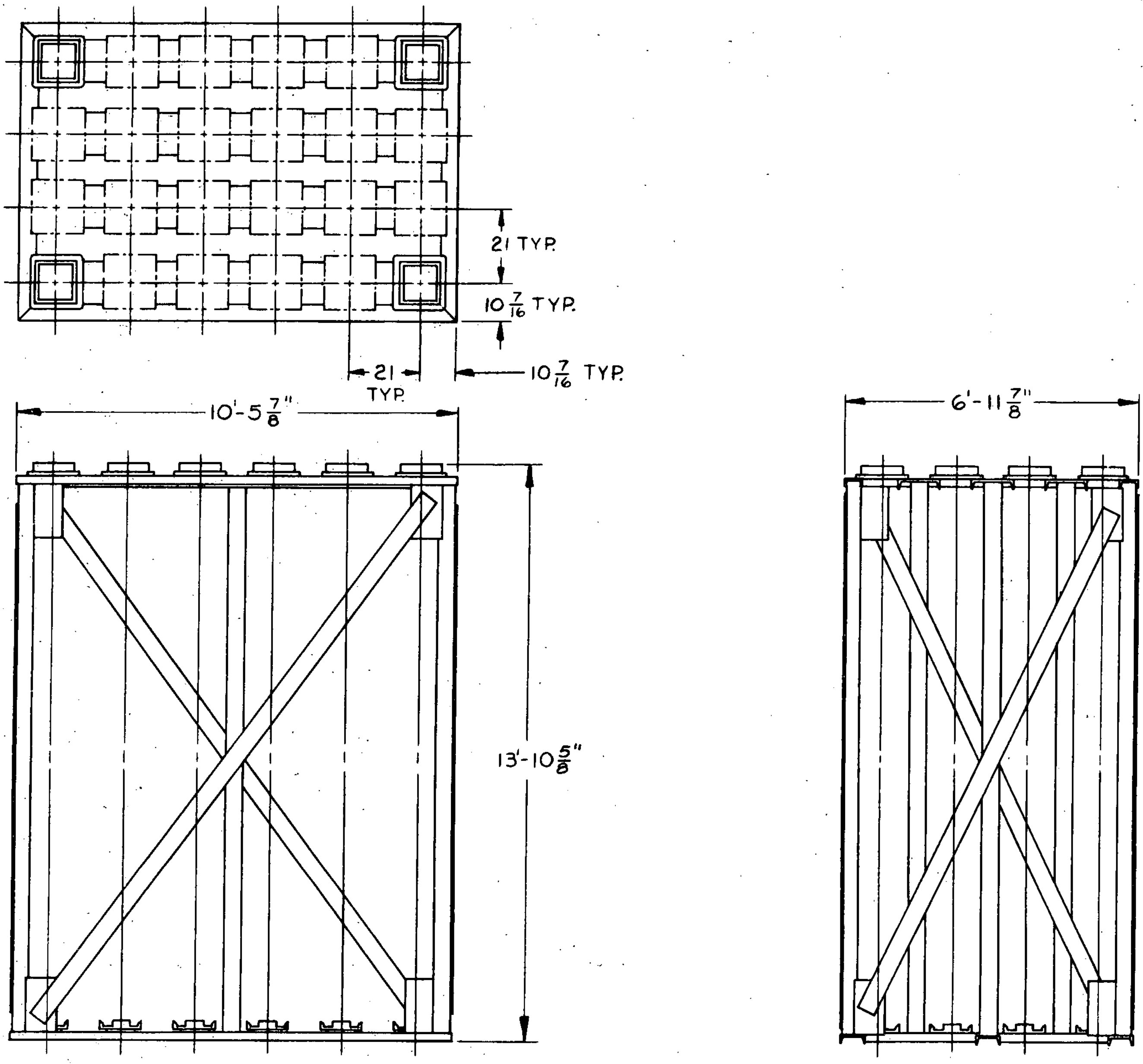
Spent fuel pump suction strainer, per unit

Quantity	1
Flow rate, gpm	1650 - 3300
Type	Disposable element
Vessel material	Stainless steel
Design pressure, psig	200 11
Design temperature, F	250
ASME Code/class	III/3



TYPICAL REFUELING SYSTEM

FIGURE 9.1-1



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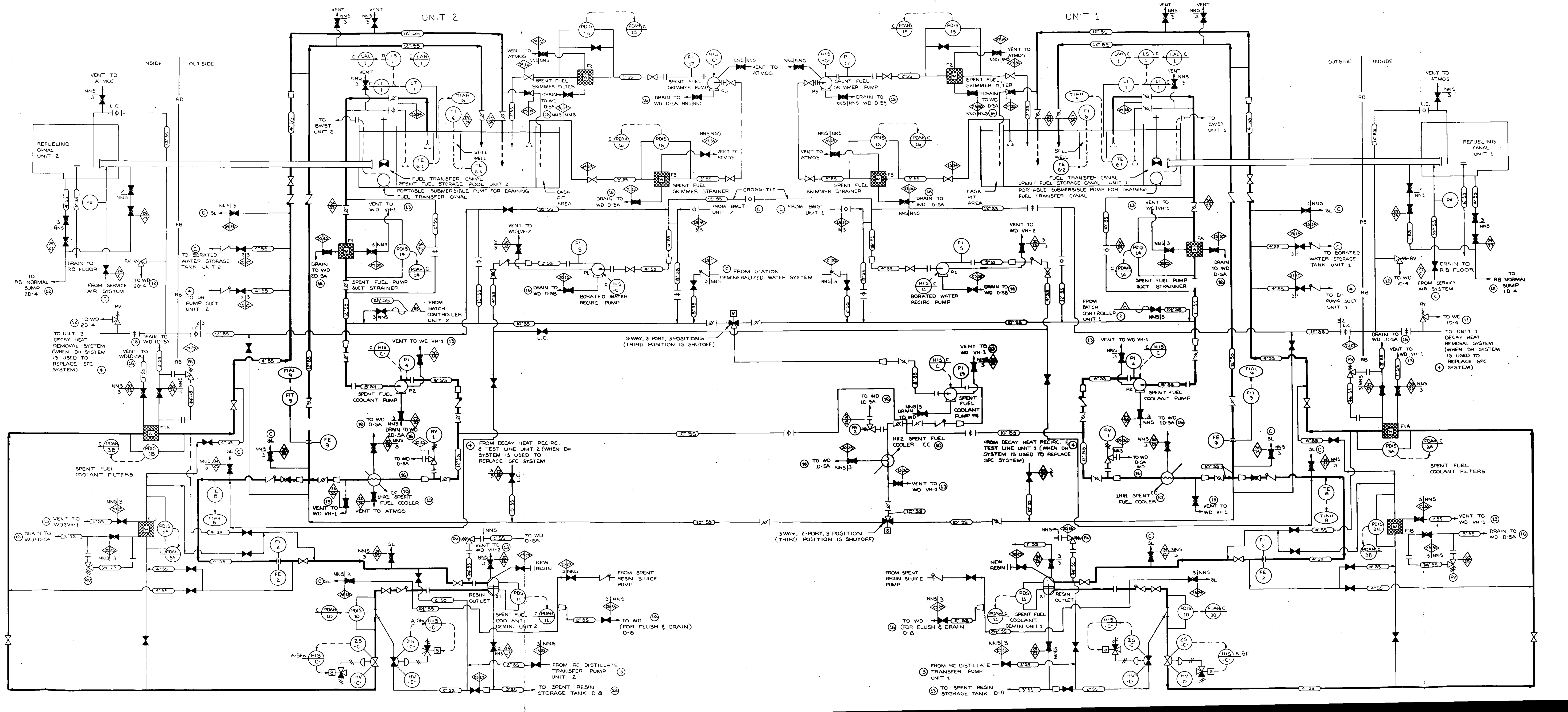
FUEL STORAGE RACK

FIGURE 9.1-2



RESERVATIONS:

1. PIPE SIZES CONTINGENT UPON FINAL PIPING CONFIGURATION.
2. SCHEDULES OF PIPING TO BE ESTABLISHED BY CUSTOMER.
3. FOR COMPLETE FILLING AND DRAINING OF SYSTEM, ADDITIONAL VENTS AND DRAINS MAY BE NECESSARY BECAUSE OF PIPING LAYOUT. FLUSH CONNECTIONS FOR INITIAL CLEANING OF SYSTEM TO BE DETERMINED BY CUSTOMER.
4. CUSTOMER TO INVESTIGATE NEED FOR AND PROVIDE IF REQUIRED PROTECTION OF ISOLATED EQUIPMENT AGAINST PRESSURE BUILDUP DUE TO AMBIENT TEMPERATURE CHANGE.

NOTES:

1. FINAL LAYOUT OF PIPING IN THE SPENT FUEL POOL WILL DEPEND ON CONFIGURATION OF SPENT FUEL POOL.
2. INSTRUMENTATION STRING NUMBERS 1-19 USED ON THIS DRAWING.

DESIGN CONDITIONS:

- 17 — 450 PSIG & 350°F
- 20 — 200 PSIG & 300°F
- 23 — 150 PSIG & 200°F
- 25 — 200 PSIG & 250°F
- 26 — 100 PSIG & 200°F
- 25 — 75 PSIG & 150°F
- 26 — BLDG DESIGN & 200°F
- 26 — 50 PSIG & 200°F
- 34 — ATMOS & 200°F
- C — CUSTOMER'S SYSTEM

REFERENCE DRAWINGS

- (3) CA & BR SYSTEM
- (4) DH SYSTEM
- (10) CC SYSTEM
- (12) WD SYSTEM
- (13) WD SYSTEM
- (16) WD SYSTEM
- (C) CUSTOMER'S SYSTEM

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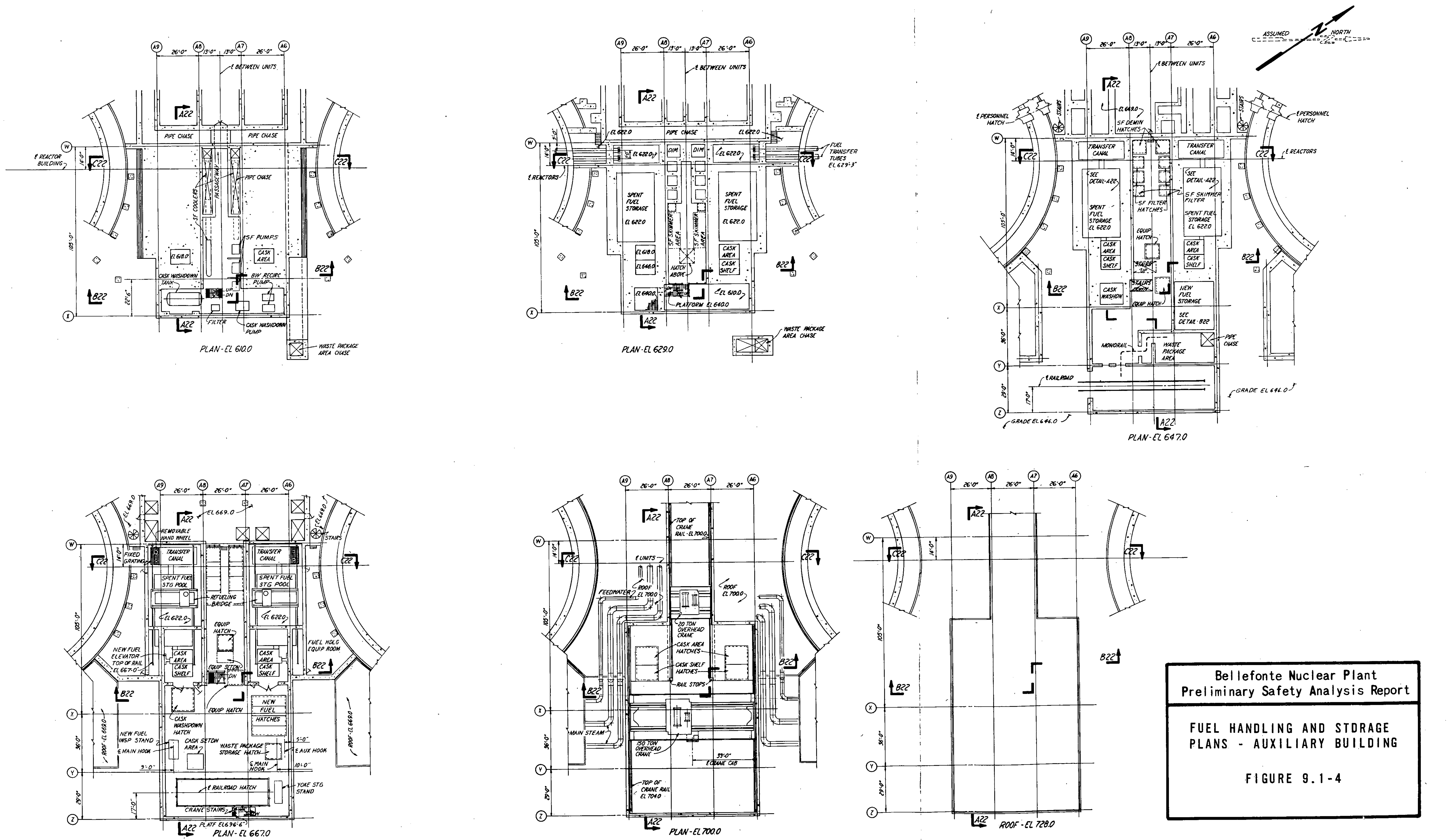
SPENT FUEL COOLING SYSTEM

FIGURE 9.1-3

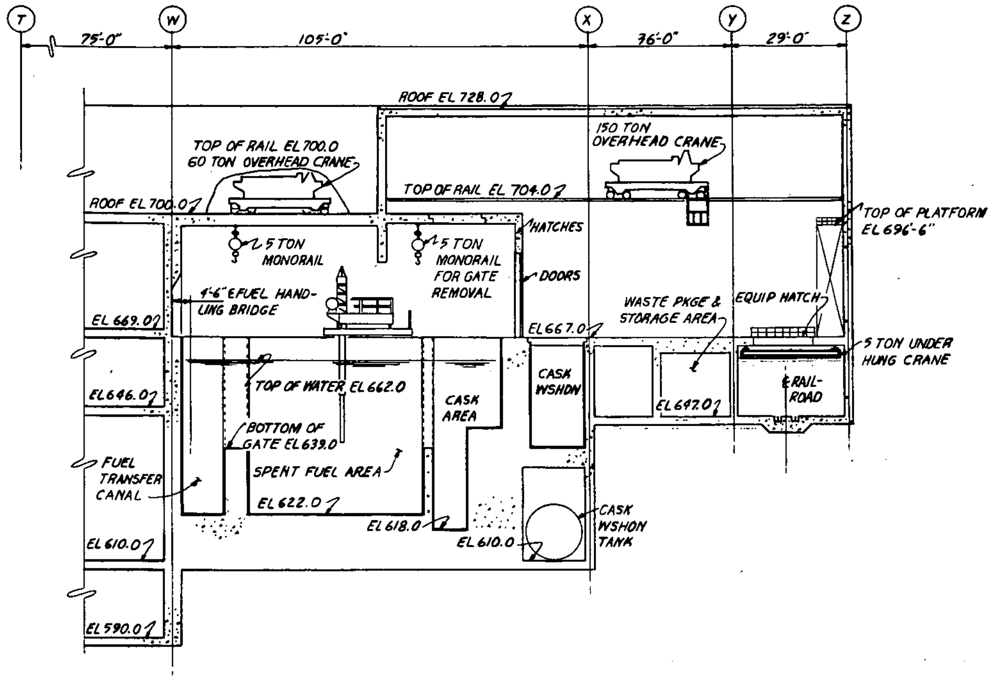
REVISED PER AMEND. 11, MAY 15, 1974



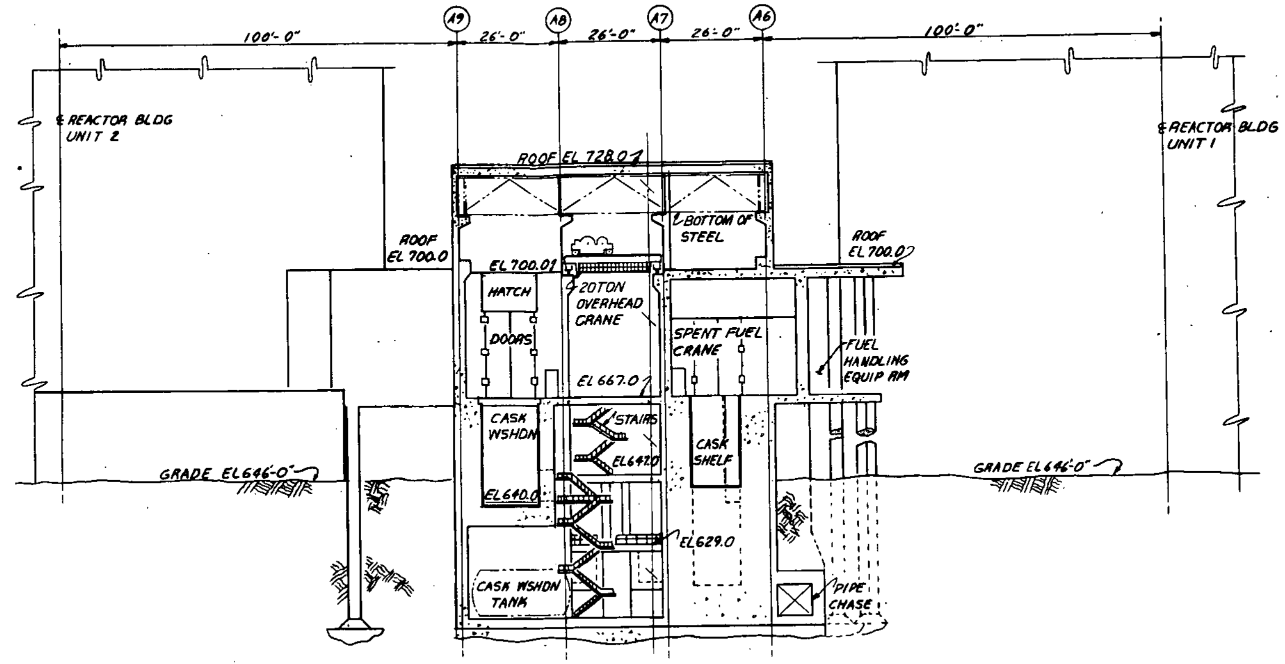
Bellefonte Nuclear Plant
Preliminary Safety Analysis Report

FUEL HANDLING AND STORAGE PLANS - AUXILIARY BUILDING

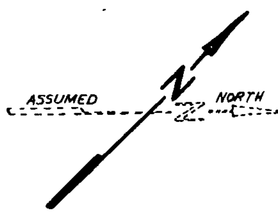
FIGURE 9.1-4

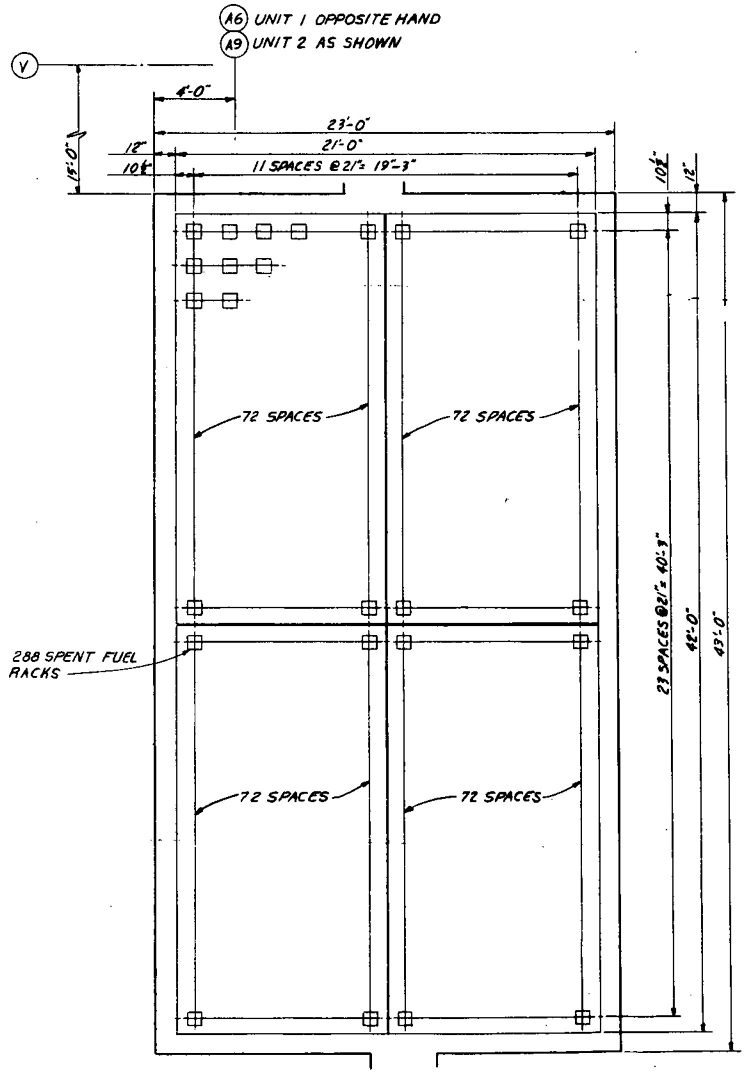


SECTION A-A

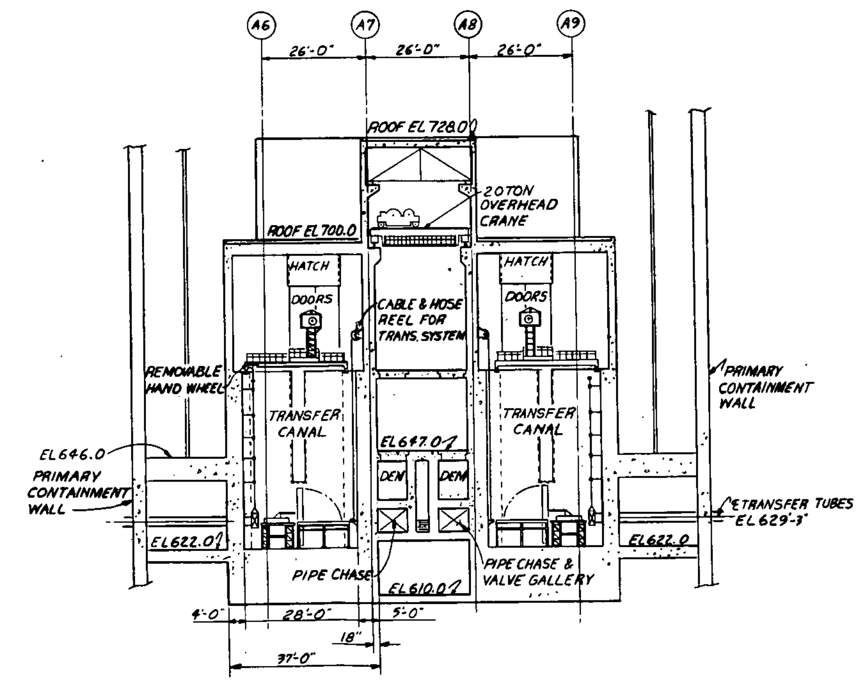


SECTION B22-B22

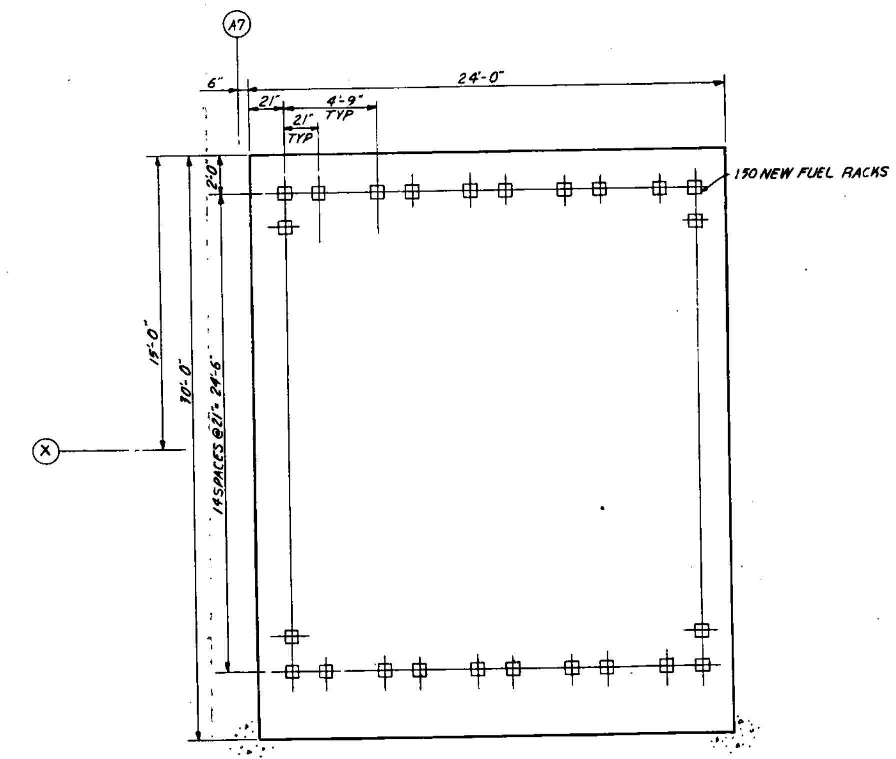




DETAIL-A
SPENT FUEL PIT
STORAGE RACK LAYOUT
SCALE ¼"=1'-0"



SECTION C22-C22



DETAIL-B
NEW FUEL STORAGE
RACK LAYOUT
SCALE ⅜"=1'-0"

SCALE 1"=20'-0" EXCEPT AS NOTED

Bellefonte Nuclear Plant
Preliminary Safety Analysis Report

FUEL HANDLING AND STORAGE
SECTIONS - AUXILIARY BUILDING
FIGURE 9.1-5

TVA DWG. NO. 47W274-22 R0

9.2. WATER SYSTEMS

9.2.1. Essential Raw Cooling Water System

9.2.1.1. Design Bases

The Essential Raw Cooling Water System is designed in accordance with AEC General Criterion 44, to provide a heat sink for removal of heat from plant essential equipment. This heat is then rejected to the ultimate heat sink as described in section 9.2.5. The system is designed to the standards of an engineered safety features system (this is not an ESF System) to ensure a continuous flow of cooling water to those systems and components necessary for plant safety either during normal plant operation or under accident conditions. Cooling requirements are based on a maximum river temperature of 95 F. The system is designed to deliver, with a single failure, adequate cooling water for all credible cases including the worst combination of design events (i.e., a LOCA in one unit, shutdown of the other unit, abrupt loss of downstream dam and all offsite power, and a simultaneous Safe Shutdown Earthquake).

Following a LOCA, the ERCW System will be required to remove the heat from reactor building sump water recirculation via the decay heat removal coolers via the component coolers, and also provide cooling water to the reactor building coolers and control building air conditioners.

Sufficient redundancy of piping and components is provided to accomodate maintenance requirements, the single failure criteria, and to ensure that cooling is maintained to vital heat loads at all times.

The ERCW System is designed to operate without interruption except upon the loss of offsite electric power. If this occurs, the ERCW pumps will automatically restart and operate on the onsite auxiliary electric power. (See Figure 9.2-3, 9.2-4, and 9.2-5.)

The ERCW system is designed to ANS Safety Class 3 and Category I seismic requirements, except that containment penetrations and piping inside the reactor building are designed to ANS Safety Class 2, Category I Seismic, and the portion of yard discharge piping, valves, and fittings going from the Unit 2 diesel 12
generator building to the cooling towers is designed to ANSI B31.1, Nonseismic. The system shall be impervious to the effects of tornadoes, hurricanes, floods, rain, snow, or ice as defined in Chapter 3 of this document.

9.2.1:2. System Description

9.2.1.2.1. General

The Essential Raw Cooling Water System (see Figure 9.2-1 and 9.2-2) consists of eight ERCW pumps, associated piping, strainers, and valves. Only 6 pumps (3 on each header loop) are needed to provide full redundancy for 2 unit operation. The other 2 pumps are maintenance spares. The pumps and strainers are located at the intake pumping station, and associated piping and valves are located at the intake pumping station, yard, diesel generator building, auxiliary building, and the reactor building. All 12
equipment, piping, and structures are seismically designed except the

discharge piping to the cooling towers. Cooling water is suppled to the various heat exchangers by the essential BAW Cooling Water pumps at all times. 12

The intake pumping station is designed to provide separated equipment and piping such that a failure of one train of redundant equipment does not affect the other train. Essential equipment controls (ERCW pumps, etc.) are located in the main control room. The eight ERCW pump motors are mounted on the deck of the intake pumping station above probable maximum flood, and the pumps' suction is below minimum lake level (El 571.0, due to loss of downstream dam) to ensure continuous pump operation and ERCW supply at all lake levels. For design data for these pumps see Table 9.2-1.

The ERCW pumps furnish water to the Essential Raw Cooling Water System which is designed to supply cooling or makeup water to the following components:

1. Component cooling water coolers
2. Control rod drive cooling water coolers
3. Emergency diesel generator heat exchangers
4. Reactor building coolers
5. Auxiliary building ventilation coolers
6. Control building water chillers
7. Air compressors (including the aftercoolers)
8. Emergency makeup to steam generator via the Auxiliary Feedwater System
9. Emergency makeup for the Component Cooling Water System
10. Emergency makeup for the spent fuel pools

The only loads on the system during normal operation are the component cooling water coolers, control rod drive cooling water coolers, the air-conditioning and ventilation systems, the air coolers, the air compressors, and the reactor building coolers.

The essential raw cooling water pumps take suction from the intake station sump which is supplied by the intake channel which is in turn supplied from the river channel. Water is supplied to the auxiliary building from the intake pumping station through two dependent sectionalized supply header loops. A supply header loop consists of two supply headers, with a valve to isolate flow to a unit basis when desired. Supply headers 1A and 2A with isolation valve comprise the "A" header loop. Supply headers

1B and 2B likewise comprise the "B" header loop. Four essential raw cooling water pumps (includes one maintenance spare) are assigned to each header loop. Physical separation is sufficient so that a rupture in either header loop will not affect continuous operation of the other header loop nor jeopardize the safety functions of the system. Isolation valves are provided on each header inside the auxiliary building at the penetration of the outside walls, such that line break will be isolated to reclaim and maintain the unimpaired section of the ruptured loop and two pumps (or one if one pump is out for maintenance) in operation. Leak detection is provided for each header section, with annunciation and alarm in the control room. The affected header section will be isolated by remote manual valve operation from the control room. The ERCW supply and discharge headers are designed with physical separation to support the separate trains of engineered safeguard equipment.

Cooling water discharges from the various heat exchangers served by the Essential Raw Cooling Water System will normally go to the condenser circulating water cooling tower basins. Each discharge header (A and B) is connected to both cooling towers (Unit 1 and Unit 2) and shutoff valves are provided such that the discharge can be directed to either cooling tower. Normally, header B will discharge to the Unit 1 cooling tower and header A will discharge to the Unit 2 cooling tower. During normal operation, when the cooling towers are in operation, the discharged ERCW serves as makeup for cooling tower evaporative and drift losses. When the cooling towers are not in operation, ERCW discharge is directed to the river through the cooling towers' blowdown line. However, an emergency route to the yard drainage pond is provided for the ERCW discharge in the event that the cooling towers or blowdown lines are out of service.

The ERCW System piping is arranged in two independent header loops each serving certain components in each unit as follows:

1. Each header loop supplies cooling water to one of the two component cooling coolers associated with each unit. |11

2. Each header loop supplies cooling water to one of the two control rod drive cooling water coolers associated with each unit.

3. Each of the emergency diesel generator heat exchangers is served by both header loops. Upon the start of a diesel generator, the motor-operated shutoff valves automatically open to both ERCW supply headers. When the ERCW flow is established, one of the shutoff valves is closed by remote manual actuation from the control room. One header comprises the normal source and the other is held in standby.

4. Each header loop provides cooling water to the control room and electrical room air-conditioning system, the auxiliary building ventilation coolers, and the reactor building coolers.

5. Each header loop supplies cooling water to one of the two air compressors associated with each unit.

6. Each header loop provides a backup source of feedwater for the auxiliary feed pumps associated with each unit.

7. Each header loop provides emergency makeup water (backup) to one of the component cooling surge tanks associated with each unit.

8. Each header loop provides emergency makeup water (backup) to the spent fuel pool associated with each unit.

The operation of three pumps on one header loop is sufficient to supply all minimum cooling requirements for the 2-unit plant for unit shutdown, refueling, or postaccident operation. (See Tables 9.2-2, 9.2-3, and 9.2-4 for minimum and maximum flow requirements.) However, additional pumps are normally started for unit shutdown or refuleing. Six pumps operate during the hypothetical combined modes of one unit LOCA and the other unit in a cooldown with loss of offsite power and four diesels in operation. Each emergency diesel generator provides auxiliary power for two ERCW pumps. In an accident, the ESF signal automatically starts a minimum of six ERCW pumps (eight if available), thus providing full redundancy.

The excess pumps will be shut off by remote manual actuation from the control room. Such an arrangement assures adequate cooling water flow under both normal and emergency conditions. As previously stated, three ERCW pumps will supply adequate cooling water for the 2-unit plant during any postulated combination of modes of units' operation and adverse environmental occurrences. Only one engineered safeguard train of equipment is required to operate in each unit to accomplish safeguard functions in one unit and extended cooldown or refueling in the other unit; therefore, the ERCW System minimum requirement shall provide cooling to one component cooling cooler in each unit to support the above requirements.

9.2.1.2.2. Component Design

All components piping of the Essential Raw Cooling Water System are ANS Safety Class 3 and Seismic Category I except where piping penetrates the reactor primary containment and yard discharge piping to the cooling towers. The containment penetrations, including containment isolation valves, and piping inside the reactor building, are ANS Safety Class 2 and Seismic Category I. Essential raw cooling water pumps, piping, valves, and strainers are designed to ASME Section III, Code Class 3 standards, except yard discharge piping from the Unit 2 diesel generator building to the cooling towers will be designed to ANSI B31.13 Standards, Nonseismic. 12

9.2.1.3. Safety Evaluation

9.2.1.3.1. General

The Essential Raw Cooling Water System is designed to prevent a single failure from limiting the ability of the safety systems to perform their functions in the event of natural disasters or plant accidents, Sufficient pump capacity is included to provide design cooling water flow under all conditions. The system provides redundant components which are powered from redundant sources of emergency power.

The system is designed to furnish a continuous supply of cooling water under normal conditions, as well as under the following extreme circumstances:

1. Tornado or other violent weather conditions which disrupt normal auxiliary power. Pumps are designed for tornadic winds and are dispersed to minimize common impact damage.

2. Earthquake involving seismic loading of structures and equipment as well as abrupt loss of main river dams above and below the plant site.

   The intake pumping station structure is designed to seismic Category I. Equipment including traveling water screens, the ERCW pumps, strainers, and piping is also seismically designed to Category I. The ERCW strainers are designed for continuous backwash to prevent plugging.

   The system provides for the simultaneous occurrence of Safe Shutdown Earthquake, a loss-of-coolant accident in one unit and shutdown of the other unit, and the loss of upstream and downstream dams either individually or concurrently.

3. Probable maximum flood. This extremely low probability event does not affect the continuous operation of the ERCW system. The pump motors are protected above the probable maximum flood elevation including maximum wave runup and will continue to operate during this event. The probable maximum flood design also assumes coincident or subsequent loss of the upstream and downstream dams. The intake pumping station is designed to function during and after these events. Settling basin capacity is provided to ensure minimum water quantity requirements.

4. Downstream dam failure. This event does not affect the ERCW system operation. The intake pumping station is designed to function after this event and will always have a minimum pool elevation of ample depth and supply to maintain flow requirements to the ERCW system.

   Cooling water is supplied from the river to the various heat exchangers by the essential raw cooling water pumps at all times. Tables 9.2-2, 9.2-3, and 9.2-4 show the essential raw cooling water flow requirements for various combined modes of operation for the 2-unit plant. When offsite power is lost, the four emergency diesel generators are used to provide auxiliary power for operation of the ERCW pumps. In the case of an emergency diesel generator or power train failure, the number of ERCW pumps operating will still meet the minimum flow requirements for any postulated conditions as indicated in Table 9.2-2.

9.2.1.3.2. System Reliability

For analysis of the ERCW system, the following events are postulated to have occurred concurrent with a LOCA in one unit and the other unit in a cooldown mode:

1. Loss of offsite power

2. Loss of downstream dam

3. Loss of one emergency power train <u>or</u> loss of one ERCW supply header loop

Emergency diesel generators will supply auxiliary power for the pumps and motor-operated valves in case of loss of offsite power. The pumps automatically restart on auxiliary power. Loss of the downstream dam does not interrupt operation of the ERCW system, since the intake pumping station is designed to handle this event. A minimum river level will be available to provide water to the ERCW pumps' suction. The loss of one emergency power train <u>or</u> the loss of one supply header loop means that cooling water must be supplied with the three (four if maintenance spare is available) remaining ERCW pumps and supply header loop. This is sufficient to provide cooling water. Three pumps and one header loop can supply the minimum cooling water requirements for all combined modes of operation (normal and accident) for the 2-unit plant. See ERCW system flow requirements, Table 9.2-2. Minimum flow requirements are established by operating only one component cooling cooler per unit. One cooler has the capacity to handle the unit in LOCA, in an extended cooldown, or during refueling. Three ERCW pumps, operating on one loop header, provide sufficient cooling water for the minimum flow requirements of the two units, as shown in Table 9.2-2.

9.2.1.3.3. System Isolation

In order to preclude leakage of radioactivity from primary containment, all lines that penetrate the containment are provided with two isolation barriers in accordance with Criterion 57 of the General Design Criteria (GDC).

One barrier is the closed piping system inside the containment; a motor-operated valve with remote manual actuation provides the second containment isolation barrier.

For each unit, a radiation detector monitors a sample from the three reactor building cooler discharge lines to detect inleakage of radioactivity during accident conditions. Valving allows sampling each line individually to determine the source of the inleakage. Each of the two ERCW discharge headers (A and B) has a radiation monitor to detect leakage of radioactive fluids into the ERCW system. ERCW activity released from the plant is monitored with the detector that takes a continuous sample of cooling tower blowdown at a point downstream of all plant radioactive liquid discharge. Isolation valves are provided to remove leaking equipment from service and contain any radioactive leak.

9.2.1.3.4. Corrosion and Organic Fouling

A corrosion allowance is provided in the system design by increasing the pipe and equipment wall thicknesses to accomodate 40 years plant life in accordance with the applicable codes.

Control of organic fouling is provided by the use of strainers, periodic injection of a slimicide (acrolein) for control of Asiatic clams, and provisions for backflushing the supply headers. The strainers (auto backwash type) are located in the intake pumping station on the discharge side of each ERCW pump and are capable of removing particles and organic matter larger than 1/8-inch diameter.

In order to retard growth of Asiatic clams, provisions are made for periodic and seasonal acrolein injection sufficient to develop a residual of 0.2 ppm for 50 minutes daily during a 3-week period twice a year. The Asiatic clam season exists during these 3-week periods. In order to maintain acrolein control in dead ended piping, all of the normal nonoperating heat exchangers are operated periodically, except supply lines to the auxiliary feedwater pumps. A small bypass line with a shutoff valve is provided for each supply line dead end to the auxiliary feedwater pumps, and piped to the ERCW discharge header. These bypass lines are used during acrolein feed to maintain a residual of acrolein in these feedlines.

9.2.1.4. Tests and Inspections

All system components are hydrostatically tested prior to station startup and are accessible for periodic inspections during operation. All components, switchovers, starting controls, and the integral systems are tested periodically.

9.2.1.5. Instrumentation Application

The instrumentation in the ERCW system provides measurements which are used for indication, alarm, and interlock as follows:

1. The total ERCW flow is measured and a signal transmitted that actuates alarms and provides flow indication in the control room.

2. The ERCW flow for each component cooling cooler is measured and a signal transmitted that actuates alarms and provides flow indication in the control room.

3. ERCW flow to and from each reactor building cooler (located inside primary containment) is measured and a signal transmitted that provides flow indication and actuates differential flow alarms in the control room.

4. The ERCW flow for each control rod drive cooler is measured and a signal transmitted that actuates alarms and provides low indication in the control room. A signal is transmitted to the plant computer for alarm of low flow.

5. Indication in control room on status of ERCW pumps.

6. Indication in control room of status of power-operated valves.

7. Radiation monitors are located to detect potential radioactive inleakage to the ERCW system with indication and alarms in the control room.

8. ERCW flow and pressure are measured for various equipment and locally indicated.

9. ERCW outlet temperatures for various equipment are measured and signals are provided to the plant computer for temperature indications.

10. Each ERCW plant discharge header is provided pressure indication with high-pressure alarm in the control room. The alarm will signal that the discharge line is restricted or that both discharge header valves are closed. Manually actuated, motor-operated valves are used to divert the ERCW discharge to the holding pond. The valves operate automatically on the ESF signal. The discharge header valve to the holding pond will open on the ESF signal, and the discharge header valve to the cooling towers close.

9.2.1.6. Reference Drawings

Figures 3.8-36 through 3.8-40 - Water Supply Intake Pumping Station Equipment Plans.

9.2.2. Component Cooling Water System

9.2.2.1. Design Bases

The component cooling water system (CC system) is designed to provide cooling water for various system components and heat exchangers during both normal and emergency conditions.

The CC system acts as an intermediate heat sink for the removal of heat from the following components. This heat is then rejected to the essential raw cooling water system.

1. Letdown coolers.
2. Reactor coolant drain tank cooler.
3. Reactor coolant pump seal area coolers.
4. Reactor coolant pump motor and lube oil coolers.
5. Seal return coolers.
6. Pressurizer sample cooler.
7. Steam generator sample coolers.
8. Evaporator coolers.
9. Waste gas compressor coolers.
10. Decay heat removal coolers.
11. Spent fuel coolers.

12. Control rod drive mechanism (contained on a separate cooling system).

13. Seal coolers of the decay heat removal pumps.

The CC system is also used to remove heat from the decay heat coolers following a loss-of-coolant accident (LOCA).

9.2.2.2. System Description

9.2.2.2.1. General

The schematic flow diagram for the component cooling water system is shown in Figure 9.2-6 and 9.2-7. Design data is provided in Table 9.2-5. The system described herein is for a single unit. Component list is for a single unit.

The component cooling water system is operated during all stages of normal unit operation. During emergency operation of the unit (LOCA), the component cooling water system will be required to remove the heat from reactor building sump water recirculation via the decay heat removal coolers.

The component cooling water system is a closed cooling system consisting of two separate cooling loops with a physical separation barrier located between the loops. A separate closed system, consisting of two separate cooling loops in the auxiliary building with a physical separation barrier between them, is provided for the control rod drive mechanism cooling coils. This system has two full-capacity pumps, two full-capacity coolers, one surge tank, and necessary piping, valves, instrumentation, and controls.

The main component cooling water system loop consists of three full-capacity pumps, two full-capacity coolers, two surge tanks, two wire wound filters, and necessary piping, valves, instrumentation, and controls and is divided into two carbon steel sub-loops "A" and "B". For most of normal system operation, the "A" sub-loop serves all the components requiring cooling and contains one component cooling water cooler, two component cooling water circulation pumps (one operating and one on normal standby), one surge tank, a filter recirculation line, and one decay heat removal cooler. A low pressure interlock (from the pressure switch on the inlet line to | 11
components inside the reactor building) to the normal standby pump is provided to automatically maintain cooling flow.

The "B" sub-loop is on emergency standby to serve the second decay heat removal cooler and contains one component cooling water cooler, one component cooling water circulation pump, a filter circulation line, and one surge tank. Two component cooling water circulation pumps are provided in the "A" sub-loop so that one pump will be available for emergency cooling operation with one pump in the "A" sub-loop down for maintenance. During cooldown of the unit from 6-20 hours and later stages of operation requiring the two decay heat removal coolers, both the "A" and "B" sub-loops are in operation to maintain cooling. The filter recirculation lines are supplied with a wire wound, back flushable filter for removal of contaminants, mainly iron oxide. For emergency operation all non-essential components are automatically isolated and both sub-loops are used solely for cooling of the essential heat loads.

The separate control rod drive (CRD) cooling loops are made entirely of stainless steel except for the Cu-Ni tubes in the coolers, which reduces the amount of ferrous particles in the component cooling water. This in turn reduces the possibility of flow blockage due to attraction of the ferrous particles by the magnetic field produced by the stator of the control rod drives. For normal operation of a CRD cooling loop one CRD cooling water cooler and nne CRD cooling water circulation pump are in operation with one pump and cooler on standby. A low flow interlock is provided to the standby pump to automatically maintain cooling flow to the control rod drive mechanisms on loss of a pump. The CRD cooling loop serves no emergency function.

9.2.2.2.2. Component Design

All components and piping of the component cooling water system are ANS Safety Class 3 and meet Safety Guide 29, with the exception of the valving and piping for reactor building containment penetrations which are ANS Safety Class 2 and meet Safety Guide 29. 11

9.2.2.3. Modes of Operation

1. Normal Operation -- The heat loads that the component cooling water system is required to handle will vary as a function of the stage of unit operation. Normal operation heat loads will be received from various combinations of the following: letdown coolers, reactor coolant drain tank cooler, reactor coolant pump seal area coolers, reactor coolant pump motor and lube oil coolers, seal return coolers, pressurizer sample cooler, steam generator sample coolers, waste gas compressor cooler, evaporator coolers, spent fuel coolers, decay heat removal coolers, and the seal coolers of the decay heat pump. The control rod drive mechanisms are cooled by a separate cooling system.

   Normal component cooling water system operation will require the use of one component cooling water circulation pump and one component cooling water cooler in the "A" loop. During stages of operation when there is a substantial decay heat load, two component cooling water circulation pumps and two component cooling water coolers are used. To more equally divide the loads between the two sub-loops, and in turn reduce the sizing requirements on the component cooling water coolers and circulation pumps, the system has been split to separate the heat loads inside the reactor building and those outside the reactor building. Normally, one component cooling water cooler and circulation pump serve both inside and outside reactor building loads. With the decay heat removal coolers on line, the loads are realigned so that one decay heat removal cooler and inside reactor building loads are served by one component cooling water cooler and circulation pump. On the other sub-loop the second decay heat removal cooler and outside reactor building loads are served by one component cooling water cooler and circulation pump.

Each unit's component cooling water system is designed to normally handle one spent fuel cooler load. However, remote operated valving is provided on the spent fuel coolers to allow for a heat burden equal to two spent fuel cooler loads to be directed to one unit's component cooling water system if the heat and flow loads on the other unit's system so require. Remote operated valving is also supplied to allow evaporator heat loads to be directed to the second unit component cooling water system if required. These remote operated valving arrangements have sufficient interlocks to prevent inadvertent mixing between units' cooling water systems and in turn prevent the possible draining of one unit's cooling water system into the other. This ensures the integrity of the component cooling water system for post-accident operation. The system piping and valving is designed to allow the decay heat removal coolers to be operated independently from each other or from other operating coolers. Thus, each decay heat removal cooler has the capability of being served by one component cooling water cooler and one circulation pump during its operation.

As stated previously, the heat and flow loads that the component cooling water system is required to handle will vary with the stage of unit operation. The following subsections describe the required operation of the component cooling water system during the various stages of unit operation.

a. Heatup -- During the heatup stage of normal unit operations, the unit's component cooling water system is required to remove heat from one letdown cooler, the reactor coolant pump coolers, the pressurizer and/or steam generator sample coolers, the seal return cooler, and one spent fuel cooler. For this stage of operation, the component cooling water system is operated with one component cooling water circulation pump and one component cooling water cooler. Control rod drive mechanism cooling is also required and can be accomplished through the use of one CRD cooling water circulation pump and one CRD cooling water cooler.

b. Power Operation -- In the normal power operation of a unit, the unit's component cooling water system must remove heat from both letdown coolers, the reactor coolant drain tank cooler, the reactor coolant pump coolers, the two reactor coolant bleed evaporators, the waste disposal evaoprators, the waste gas compressor, the pressurizer and/or steam generator sample coolers, one seal return cooler, and one spent fuel cooler. The component cooling water system requires the use of one component cooling water circulation pump and one component cooling water cooler to handle the above heat and flow loads. Control rod drive mechanism cooling is required with the use of one CRD cooling water circulation pump and one CRD cooling water cooler.

c. Cooldown (Before and After 6 Hours) -- In the cooldown stage during the six-hours before the decay heat removal system is needed, heat removal is required from one letdown cooler, the reactor coolant pump coolers, the two reactor coolant bleed evaporators, the waste disposal evaporators, the waste gas compressor, the pressurized and/or steam generator sample cooler, the seal return cooler, and the spent fuel coolers. The heat loads from the bleed and waste disposal evaporators, the waste gas compressor, and one spent fuel cooler are handled by the second unit's component cooling water system. The component cooling water system for the unit in the cooldown stage handles the flow and heat loads imposed by all other above listed sources and uses one component cooling water circulation pump and one component cooling water cooler.

In the cooldown stage after the six hour period, the unit's component cooling water system is required to remove heat from one letdown cooler, the reactor coolant pump coolers, the seal return cooler, one spent fuel cooler, two decay heat removal coolers, and the decay heat pump seals. Due to the high flow requirements, both sub-loops of the main component cooling loop are used. Prior to bringing the decay heat removal coolers on line for 6 - 20 hour cooldown stage, the system is realigned as described previously. One decay heat removal cooler and inside reactor building loads are served by one sub-loop, and the other decay heat removal cooler and outside reactor building loads are served by the second separate sub-loop. The circulation pumps will be operating at the upper end of the design flow range but still within the capacity of the pumps and coolers.

Control rod drive mechanism cooling water is circulated in the separate CRD cooling loop with the use of one CRD cooling water circulation pump and one CRD cooling water cooler. However, the heat load from the control rod drive mechanisms is neglibible during cooldown.

d. Shutdown -- At shutdown of a unit, heat removal is required from the two reactor coolant bleed evaporators, the waste disposal evaporators, the waste gas compressor, the spent fuel coolers, two decay heat removal coolers, and decay heat pump seals. The evaporators, the waste gas compressor and one spent fuel cooler heat and flow loads are handled by the second unit's component cooling water system. The component cooling water system for the unit at shutdown removes heat from the other above listed sources. Heat removal for the component cooling water system in the shutdown stage is accomplished through the use of both sub-loops of the main component cooling loop with the line-up as described in the cooldown after 6 hour stage. One sub-loop serves one decay heat removal cooler and one spent fuel cooler, and the other sub-loop serves the second decay heat removal cooler with the decay heat pump seal coolers on either sub-loop. The circulation pumps will operate within the design flow range of these pumps.

Control rod drive mechanism cooling is not required during this stage of operation.

e. Refueling -- During the refueling operation, heat removal is required from the evaporator, the waste gas compressor, two spent fuel coolers, two decay heat removal coolers, and the decay heat pump seal coolers. The evaporators, the waste gas compressor, and one spent fuel cooler are handled by the second unit's component cooling water system. The unit being refueled handles the other above listed loads using both sub-loops of the main component cooling loop. One sub-loop serves one decay heat removal cooler, one decay heat pump seal cooler, and the spent fuel cooler. The other sub-loop serves the second decay heat removal cooler and the second decay heat pump seal coolers. The circulation pumps will operate within the design flow range for these pumps.

   Control rod drive mechanism cooling is not required during this stage of operation.

f. Cold Iron (>150 hours after 0 power) -- In the cold iron stage of operation the unit has been shutdown for an extended time period. The heat removal requirements are from the evaporators, the waste gas compressor, the spent fuel coolers, one decay heat removal cooler, and the decay heat pump seal coolers. The evaporators' loads, the waste gas compressor, and one spent fuel cooler are handled by the second unit's component cooling water system. The other above listed loads are handled by the component cooling water system in cold iron. One sub-loop of the main component cooling loop has the capacity to meet the loads imposed during cold iron. The decay heat load imposed during this stage of operation is small enough that one decay heat removal cooler can maintain the required cooling. The circulation pump will operate within the design flow range of these pumps.

   Control rod drive mechanism cooling is not required during this stage of operation.

2. Emergency Operations -- In emergency conditions following a LOCA, the component cooling water system is required to remove the emergency heat load from reactor building sump water recirculated through the decay heat removal coolers. Upon receipt of an ESF signal, the component cooling water system's reactor building isolation valves and the flow stoppage valves for non-essential components outside the reactor building are actuated. In addition to closing the isolation valves the ESF signal opens the control valves downstream of the decay heat removal coolers and actuates the component cooling water circulation pumps to initiate cooling flow to the decay heat removal coolers. It should be noted that in the "A" sub-loop containing the standby pump, the ESF signal to the standby pump is blocked out to prevent overflowing of the decay heat removal coolers due to two circulation pumps in operation on the sub-loop. Approximately 30 minutes after the receipt of the ESF signal, the reactor building sump water is directed through the decay heat removal coolers.

During emergency operation, cooling flow to the seal coolers of the low pressure injection pumps may be required. These pump seals require cooling only if pump temperature exceeds 250F. It is expected that the decay heat pump will exceed this temperature and require seal cooling. The flow and heat loads required for the cooling of these pump seals is negligible compared to the main emergency operation loads, but are considered.

9.2.2.4. Safety Evaluation

1. General -- Emergency use of the decay heat removal coolers requires that each cooler is served by one sub-loop of the main component cooling loop, and that each sub-loop serves only essential loads. This requirement ensures that there is always adequate cooling for emergency operation. Thus, all non-essential coolers must have cooling flow stopped to them prior to reactor building sump recirculation. For components inside the reactor building, flow stoppage is accomplished by the automatic closing of the reactor building isolation valves. This arrangement provides sufficient redundancy to ensure isolation of the inside reactor building components. Flow stoppage for non-essential components outside the reactor building is provided by a remote operated, ESF actuated valve in the inlet and outlet lines to these components. This arrangement provides sufficient redundancy to ensure flow stoppage to non-essential components outside the reactor building. Necessary connections are provided for the evaporators, the waste gas compressor and spent fuel coolers to allow flow and heat loads from these sources to be diverted to the second unit's component cooling water system if required.

   Of primary importance to component cooling water system emergency operation is an adequate supply of essential raw cooling water to the component cooling water coolers. This ensures that the decay heat removal-component cooling water systems' emergency heat removal capability is maintained.

2. System Isolation -- Both the main component cooling water system loop and the separate CRD cooling loop penetrate the reactor building. Building isolation for the CC system lines which direct flow into the reactor building is provided by two ESF actuated, one a motor operated valve inside the building, and one an air operated valve outside the building. Building isolation for the CRD system lines which direct flow into the reactor building is provided by an ESF actuated, motor operator valve outside the reactor building and a check valve inside the building.

   To provide reactor building isolation for the two lines which direct component cooling water flow out of the reactor building, each line is supplied with an ESF actuated, motor operated valve inside and immediately outside the reactor building. This arrangement also conforms with Criterion 57 of the 55 GDC.

3. System Reliability -- Each CC surge tank is provided with a high and two low level alarms to warn the operator of loss of water from the CC system and inleakage of water to the CC system. Loss of water from the CC system could lower the surge tank level enough to result in loss of sufficient NPSH for the circulation pumps. Inleakage to the CC system could result in overpressurization of the CC system. Therefore, an overflow line containing an in-line relief valve is provided on the surge tank which allows any excessive inleakage to the CC system to be directed to the waste disposal system and thus prevent overpressurization. Also, the high level alarm, in conjunction with the high radiation alarm on each of the sub-loops, provide a means of detecting tube failures in the various coolers served by the CC system.

   In addition to the surge tank overflow line, system over-pressurization protection is provided by the relief valves on the shell-side of each of the coolers. Coolers that have high pressure tubeside flows (i.e., letdown and the decay heat removal coolers) have relief valves sized to handle the flow resulting from a tube rupture. All other coolers in the system are provided with relief valves to provide overpressurization protection due to thermal expansion of isolated shellside water with the tubeside flowing.

9.2.2.5. Tests and Inspections

Fluid samples are periodically taken from the CC system for analysis. The ESF components and equipment will be periodically tested for reliability.

The component cooling water system will be examined periodically to determine its operating condition. Periodic visual inspection and preventive maintenance will be conducted according to sound maintenance practices.

9.2.2.6. Instrumentation Application

The instrumentation in the component cooling water system provides measurements which are used for indication, alarm, and interlock as follows:

1. The total component cooling water flow is measured and a signal transmitted that will actuate high and low level alarms and provide indication in the control room.

2. The component cooling water flow for the non-essential components inside the reactor building is measured and a signal transmitted that will actuate alarms and provide indication in the control room. Low flow interlock signals are also provided to the controlled bleedoff valves from the RC pump seals in the makeup and purification system.

3. The component cooling water pressure to the nonessential components inside the reactor building is measured and a signal is transmitted to a low pressure interlock to the normal standby pump to automatically provide cooling flow.

4. The component cooling water flow for non-essential components outside the reactor building is measured and a signal transmitted that will provide indication in the control room.

5. The control rod drive cooling water flow is measured, locally indicated, and a signal transmitted that will actuate alarms in the control room. A signal is transmitted to the plant computer for alarm of low flow. A low flow interlock signal is also provided to the control rod drive cooling water pumps.

6. The following process variables are measured and locally indicated;

   a. Component cooling water pump discharge pressure. 11

   b. Control rod drive cooling water pump discharge pressure.

7. The component cooling water filter differential pressure is measured, locally indicated and a signal transmitted that will actuate alarms in the control room.

8. The following process variables are measured and a signal is transmitted that will actuate an alarm in the control room:

   a. Decay heat cooler component cooling water flow.

   b. Spent fuel cooler component cooling water cooler flow.

9. The following process variables are measured and a signal is provided to the plant computer for indication:

   a. Letdown cooler component cooling water outlet temperature.

   b. RC pump seal cooler component cooling water outlet temperature.

   c. RC pump motor and lube oil coolers component cooling water outlet temperature.

   d. RC drain tank cooler component cooling water outlet temperature.

   e. Seal return coolers component cooling water outlet temperature.

   f. Sample coolers component cooling water outlet temperature.

   g. Waste gas compressor cooling water outlet temperature.

h. Auxiliary waste disposal evaporator cooler component cooling water outlet temperature.

i. Waste disposal evaporator cooler component cooling water outlet temperature.

j. RC bleed evaporator coolers component cooling water outlet temperature.

k. Decay heat cooler component cooling water outlet temperature.

l. Spent fuel cooler component cooling water outlet temperature.

m. Component cooling water cooler inlet temperature.

n. Control rod drive mechanism inlet temperature.

9. The following process variables are measured and a signal is provided to the plant computer for indication and alarm:

   a. Control rod drive mechanism outlet temperature.

   b. Component cooling water cooler outlet temperature.

10. The following process variables are redundantly measured and redundant signals are transmitted that will actuate alarms and provide indication in the control room:

    a. Control rod drive cooling water surge tank level.

    b. Component cooling water surge tank levels.

9.2.2.7. Radiological Considerations

No part of the component cooling water system requires shielding to provide radiation protection for personnel.

The component cooling water is a closed system which provides an additional barrier to fission product release from the plant during normal operation, during anticipated operational occurrence and during accident conditions. The only possible source of radioactivity in this system is leakage across heat exchangers from systems cooled by the component cooling water. Activity in the component cooling water resulting from such leakage during normal operation is always at a low level. The actual level is monitored under all conditions, by radiation detectors, one located downstream from each heat exchanger. Activity in the component cooling water system can be released from the plant in one of three ways: (1) leakage across the component cooling water heat exchangers to the ERCW and subsequent release in the cooling tower blowdown or by evaporative losses, (2) leakage, which is collected by the waste disposal system, from other parts of the system and (3) overflow from the component cooling water surge tank which is also collected by the waste disposal system. None of these paths of component cooling water activity released from the plant can be a significant part of expected plant activities released from either gaseous or nongaseous sources.

Component	Failure	Comment	
RB-CC isolation valve	Fails to close	Three other isolation valves will ensure stoppage of CC flow to nonessential components inside the RB.	11
Outside RB component isolation valve	Fails to close	Redundant valve in series will ensure stoppage of CC flow to nonessential components outside the RB.	
CC pump	Second pump fails to start or operating pump stops	Redundant CC pump will provide flow to one DH cooler and one CC cooler to maintain sufficient emergency cooling. The spare pump in the "A" sub-loop can be brought on line to provide cooling for the second DH-CC cooler combination.	
DH cooler CC flow control valve	Fails to open	Sufficient heat removal can be provided by redundant loop to ensure adequate emergency cooling capability.	
Piping in one emergency loop	Breaks	Sufficient heat removal can be provided by redundant loop to ensure adequate emergency cooling capability.	
CC cooler	Tube break	Sufficient heat removal can be provided by redundant loop to ensure adequate emergency cooling capability.	

9.2.2.8. Prevention of Corrosion

The majority of the component cooling water system is constructed of carbon steel except for the separate stainless steel control rod drive (CRD) cooling water loop. Therefore, the cooling water quality must be maintained within certain limits to keep corrosion under control. Chemical treatment of the component cooling water system (both main and separated CRD loops) is accomplished by adding specified chemicals to the surge tanks. The following is the proper water quality for the component cooling water as verified by periodic samples and analyses:

Component cooling water

pH @ 77F	8.8-9.2
Dissolved solids, ppm max. (including additives)	25[(a)]
Chloride, ppm max. (as Cl^-)	1.0
Oxygen, ppm max.	0.1
Morpholine, ppm	2-6[(b)]
Hydrazine, ppm	15-25

CRD cooling water coolers

Quantity	2
Type	Shell & tube
Material, shell/tube	SS/Cu-Ni
Cooling capacity, 10^6 Btu/h	1.215
Cooling water flow, 10^5 lb/h	1.085
Raw water flow, 10^5 lb/h	1.085
Design press., shell/tube, psig	200/150
Design temp, shell/tube, F	200/200
ASME Code, Section/Class	III/3
Seismic Category	I

(a) Cation conductivity may be used in lieu of dissolved solids specification. The cation conductivity specification is 25 μmho/cm max.

(b) The morpholine range is equivalent to the pH range.

9.2.3. Demineralized Makeup Water System

9.2.3.1. Design Bases

The system is designed to supply demineralized water for makeup to the condensate system; condensate demineralizer regeneration; and to the demineralized water system for miscellaneous services in the reactor, auxiliary, control, service, and turbine buildings.

The system also supplies treated water, which has not been demineralized, to the condenser circulating water pumps for bearing lubrication.

Space is provided in the turbine building for future treatment facilities which will increase the capacity by 100% if additional makeup is required.

9.2.3.2. System Description

The demineralized water makeup system consists of three subsystems: a 730,000-gallon-per-day net filtration plant, a 2-train demineralizer system having a net capacity of 216,000 gallons per train per day, and the demineralized water storage and distribution system. The system supplies treated and demineralized water to both power generating units.

Flow diagrams are shown in Figures 9.2-8, 9.2-9 and 9.2-10.

Water supply to the filtration plant is from the raw service water system. Prior to entering the clarifier, the raw water is chlorinated. After the water has entered the clarifier, coagulants are added by the dry chemical feeders which flocculate suspended solids.

Clarified water then flows through the filters to the clearwell tanks. The filters normally operate simultaneously, but the system can be operated while one filter is being backwashed. The filter effluent turbidity is continually monitored and recorded to ensure high water quality.

The water-treatment plant principal piping system is carbon steel, and chemical feed lines are polyvinyl chloride. All water-treatment plant equipment is located in the turbine building.

The second subsystem is a 2-train demineralizer consisting of cation, anion, and mixed-bed units; a common 2-stage vacuum deaerator; and shared regeneration facilities. The resins remove dissolved ions from the water by the ion-exchange process.

The demineralizer supply pumps take suction from the clearwells and pump water through the cation unit into the 2-stage vacuum deaerator where carbon dioxide is reduced to 10 ppm. The demineralized water transfer pumps take suction from the vacuum deaerator and pump through the anion and mixed bed units to the demineralized water storage and distribution system. During periods of low demand, the mixed-bed effluent is automatically recycled. The demineralizer trains can operate individually or simultaneously.

Equipment specifications require that the demineralizer effluent quality not exceed the following levels:

Silica (ionized soluble) - 0.01 ppm
Electrolyte (ionized soluble solids) - 0.04 ppm as $CaCo_3$
Conductivity - 0.10 micromho

Regenerant wastes are collected in a waste surge tank and then pumped through a neutralizer to the neutralized waste sump. The neutralizer is designed to produce an effluent with a pH of between 6.0 and 8.0.

From the neutralized waste sump, the waste water is pumped to the cooling tower blowdown stream. All demineralizer equipment is located in the turbine building, except the acid storage tank and acid supply pump which are located in the yard.

Backwash water from the filters and demineralizers and blowdown from the clarifier flow by gravity to the water-treatment plant waste sump located below the turbine building floor. The waste is then pumped to one of two settling basins located in the yard. The suspended solids settle out in the basins, and the decanted water is pumped back to the water-treatment plant inlet. The sludge is periodically removed and buried.

The makeup demineralizers and waste neutralization equipment process and drainage piping is plastic-lined steel.

The third subsystem is the demineralized water storage and distribution system. The demineralized water system consists of a 15,000-gallon demineralized water storage tank, a 15,000-gallon cask decontamination supply tank, a main piping loop, and distribution headers. The loop supplies water for various services in the reactor buildings, auxiliary building, office building, control building, service building, and turbine building. These services include emergency showers, eyewash stations, water for cask washdown, refueling canal wall wetting, spent fuel pit makeup, and makeup water for equipment associated with nuclear service systems and other miscellaneous services.

The main loop supplies all demineralized water in the plant except that furnished to the condensate system and the washdown water for the cask washdown room, which is supplied by the cask decontamination supply tank. The demineralized water system principal piping is aluminum. Insulating flanges are used where dissimilar metals are joined.

9.2.3.3. Safety Evaluation

The demineralized water makeup system is not required for maintenance of plant safety in the event of an accident and is not a part of the engineered safety systems; therefore, the reactor containment isolation valves and the piping connecting the valves are the only portion of this system which have a Nuclear Safety Class designation in accordance with ANS Safety Class 2.

All pipe and supports in the control building, auxiliary building, and reactor buildings are designed for seismic loading to prevent damage to adjacent safety-related equipment necessary for the safe shutdown of the plant in case of a nuclear accident.

9.2.3.4. Test and Inspection

Prior to startup, all piping and equipment is hydrostatically tested. After startup, visual inspection of the system components and instrumentation monitoring is adequate to verify system operability.

9.2.3.5. Instrumentation Applications

The water-treatment plant is provided with instrumentation to monitor gravity filter and clearwell levels, high-pressure differential across the filters which initiates back wash, and turbidity of the filter effluent. The turbidity is recorded and high turbidity annunciation is provided locally and in the main control room. Each cation, anion, and mixed-bed tank is provided with flow recorders which indirectly give an indication of resin exhaustion. The demineralizer beds are also provided with conductivity recorders, high conductivity alarms, and high differential pressure alarms. During periods of low demand, the demineralizer system effluent is automatically recycled. The recycle is controlled by a flow switch in the demineralizer system effluent line. The flow switch opens a flow control valve in the recycle line during periods of low demineralized water demand and closes for high demand.

The demineralized water storage tank level is maintained by a flow control valve in the demineralized water supply line. The valve opens when the tank level falls below a predetermined set point, and after the tank fills, the valve closes. The cask decontamination tank fills by gravity from the demineralized water storage tank. High and low level switches annunciate both tank levels in the control room, allowing the operator to continually check tank levels and monitor control valve performance.

9.2.4. Potable and Sanitary Water Systems

9.2.4.1. Potable Water System

9.2.4.1.1. Design Bases

The purpose of this system is to furnish potable water to all facilities on this project.

The initial quantity of potable water required will be approximately 40,000 gallons to partially fill four 10,000-gallon storage tanks on the turbine building roof, and the system pipes. The average daily requirement for normal operation will be approximately 12,000 gallons based on the maximum number of personnel on the site; approximately 335 employees using water at the rate of 35 gpd per capita.

9.2.4.1.2. System Description

Potable water is supplied to this plant from the local municipal water system. It is classified as an approved system by the Alabama Department of Public Health.

The flow diagram for the potable water system is shown in Figure 9.2-11. Static pressure at the plant site is reduced at several points in the system. The potable water supply flows by gravity from the four storage tanks on the turbine building roof.

The yard distribution system conveys potable water to the various buildings and to other points of usage. Where 4-inch lines are required, cast-iron mechanical joint valves, pipe, and fittings are used. Three-inch and smaller lines and fittings are polyvinyl chloride used with iron body valves. Concrete backing is poured where lines change direction or dead end.

The radiochemical laboratory, the hot instrument shop and titration room sinks, plumbing fixtures, water coolers, water heaters, emergency showers, eye and face wash fixtures, hospital sinks, and a footbath are supplied from the four storage tanks. Most fixtures are supplied from a return line from the storage tanks so as to not deplete the chlorine residual in the tanks. Other fixtures which are located at relatively high elevations and remote locations are supplied directly by the supply line to the storage tanks. Potable water lines are sized so as not to exceed a velocity of 7 or 8 fps in the pipes.

Potable water is supplied to the chlorinator and chemical feeders for the makeup water-treatment plant located in the turbine building.

No potable water lines are located in the reactor building.

Inside the buildings, hard-drawn copper tubing and solder joint fittings are used for all water lines except those in the diesel generator building where galvanized steel pipe and galvanized malleable iron fittings are used.

Potable water lines extending into the auxiliary building to serve the battery rooms have shutoff valves in the control building.

Several usage points are supplied water from valved connections for use in preventing possible backup of undesirable constituents into the plant water system. Battery room sinks in the control building are supplied potable water from a valved connection in the turbine building. The radiochemical laboratory, the hot instrument shop, and the titration room sinks in the auxiliary building are also supplied from this header with a valved connection in the control building.

9.2.4.1.3. Safety Evaluation

The potable water system has no effect on plant safety.

9.2.4.1.4. Tests and Inspections

All potable water systems are tested by pressurizing them to 150 psig and inspecting them for leaks. Where piping is in walls, the tests are made prior to erection of masonry walls. Fixtures are accessible for inspection during normal operation. Periodic inspections are made of the water level control switches.

9.2.4.1.5. Instrumentation Applications

Water level in the four potable water storage tanks is controlled by level switches which actuate a flow control valve located in the tank supply line. The water levels in the tanks are controlled by level switches. The level switches on one tank actuate an alarm in the control room on high and low level in all tanks.

9.2.4.2. Sanitary Water System

9.2.4.2.1. Design Bases

The quantity of sanitary water handled, treated, and disposed of is approximately 12,000 gpd, the estimated usage of potable water.

9.2.4.2.2. System Description

Sanitary water is collected in the control building, service building, turbine building, office building, and auxiliary building, and is conveyed from the buildings by the yard sewerage system to the sewage treatment facilities. The sewage effluent is pumped into the condenser circulating water system which is a recirculating system.

The reactor building has no sanitary water. Although potable water is supplied to the battery room in the auxiliary building, it is collected in the floor drainage system in the control building.

Sanitary water from the radiochemical laboratory, titration room, and the hot instrument shop in the auxiliary building is handled separately from the sanitary water system described here. See section 9.3.3, Equipment and Floor Drainage System.

All embedded lines and fittings are extra-heavy cast-iron soil pipe, bell and spigot with neoprene gaskets. Exposed lines are galvanized steel pipe and the fittings are the black cast-iron drainage type.

Vent lines are galvanized steel pipe and fittings are galvanized malleable iron.

Sanitary water from the turbine building, auxiliary building, and control building flows by gravity to the yard collection system.

Sanitary water from portions of the office and service buildings, which are below grade, is conveyed to the yard collection system by duplex sewage ejectors.

The duplex mechanically operated ejectors have cast-iron pots of 100-gallon capacity each. Plant service air, to operate the ejectors, is supplied at 100 psi and is reduced to the ejector manufacturer's recommendation.

The sanitary water from these buildings is then conveyed through the yard system to the septic tanks. The yard system consists of vitrified clay pipe and precast manholes. Two septic tanks are sized for the total daily flow from the plant.

The effluent of the septic tanks flows by gravity to two dosing tanks which work alternately. An automatic sewage siphon in each dosing tank discharges the septic effluent to each of two sand filters at a rate of flow to deliver 0.9 gpm for each 10 square feet of filter area. The effluent from each sand filter flows to a common chlorine contact tank which retains the sanitary water for not less than a 2-hour period. The effluent is then pumped into the supplementary makeup water system to the cooling towers and is recycled.

9.2.4.2.3. Safety Evaluation

Containment is provided on the sanitary water drain between the auxiliary building and the control buildings by means of a running trap inside of the auxiliary building. Potable water drips in the trap to keep it full at all times. A needle valve is used to control the flow of water. All plumbing fixtures and water coolers, except water closets and urinals, have chromium-plated traps. The water closets and urinals have vitreous china traps which are an integral part of the fixture.

These traps prevent fumes, odors, or gases from coming back through the fixtures.

The only safety function of the sanitary water system is that of maintaining sanitary conditions in the plant. This safety function is adequately fulfilled by the installed equipment, and the system design and construction meets the requirements of the Alabama Department of Public Health.

9.2.4.2.4. Tests and Inspections

All embedded lines and fittings, extra-heavy cast-iron pipe, and fittings are tested to 5-psi air pressure with soaped joints while still exposed.

The chlorine residual of the effluent from the sand filter is checked and recorded daily. These reports are submitted at the end of each month to the Alabama Department of Public Health, Division of Stream Pollution Control.

9.2.4.2.5. Instrumentation Applications

There are three air pressure gauges at each pressure-reducing station preceding the duplex sewage ejector unit. Each pressure-reducing station has a bypass and a pressure gauge preceding each pressure reducing valve. These gauges will read 100 psi.

There is a pressure gauge downstream from each pressure-reducing valve.

9.2.5. Ultimate Heat Sink

9.2.5.1. Design Bases

To provide the sink safety functions, the ultimate heat sink is designed to: (1) provide sufficient cooling for simultaneous safe shutdown of either unit coincident with the dissipation of heat from a loss-of-coolant accident (LOCA) in the other reactor, and (2) maintain the two units in the safe shutdown condition, dissipating decay heat from both units, for a minimum of 30 days without replenishment. These capabilities are required by Safety Guide 27.

The ultimate heat sink design is formulated to ensure performance of these sink safety functions with a single water source and intake structure for the most severe natural phenomena, and for other appliable site-related events. These include the Safe Shutdown Earthquake (SSE)[1] tornado, severe storm, and probable maximum flood (PMF).[2]

The ultimate heat sink will provide its safety function for reasonably severe site events related to the heat sink. River blockage or diversion[3] and aircraft crashes[4] are taken as being outside the domain of credible events.

The ultimate heat sink will provide its function despite single failures of either the upstream of downstream dams. The failure of the seismically qualified intake channel is taken as being outside the domain of credible events.

The occurrence of an SSE coincident with a LOCA and a loss of either the downstream or up stream dam is considered as a design basis event for Bellefonte Nuclear Plant. Although the ultimate heat sink can provide its safety function during and after a tornado or the PMF, the coupling of these events with a LOCA is not considered credible for BNP.

9.2.5.2. Sink Description

During normal operation, the Bellefonte Nuclear Plant will transfer the majority of the waste heat to the atmosphere via the condenser circulating water system and the cooling towers. These features are not part of the 1
ultimate heat sink. Some heat is transferred to the river via the essential raw cooling water system.

Should the functional capability of either the main steam condensers or the cooling towers be lost, the plant would be shut down and the Tennessee River-Guntersville Reservoir waters would serve as the ultimate heat sink. (The yard drainage pond is available as an emergency discharge route should normal discharge paths become unavailable.) The intake channel supplies the essential raw cooling water system with sufficient river water for emergency service and a return path to the river for adequate heat removal. The sink features[5] are constructed to ensure an adequate plant flow of cooling water for emergency operations as given in table 9.2-4 for the condition of LOCA plus shutdown.[6] This intake channel is designed to function during and after an SSE, severe storm, or tornado.

The original river channel bottom is at elevation 569' and extends 2000 feet from the river to the intake pumping station. The channel structure, with its upper wall slope of 1:5, is designed for normal and seismic loads. The channel empties into the intake pumping station sump whose bottom is at elevation 557'.

The capabilities of the upstream and downstream dams to withstand the SSE, and floods are described in sections 2.4.4 and 2.4.3, respectively.

9.2.5.3. Safety Evaluation

Streamflow data prior to Guntersville Dam closure shows a minimum natural flow of 3400 cfs, corresponding to a low water surface elevation of 571 feet at the intake site.[7]

Loss of Guntersville Dam, with the reservoir initially at its normal summer level of 595 feet and the upstream Nickajack gates completely closed, would result in a water surface drawdown to 571 feet at the intake channel in approximately 24 hours. This ignores all contributions to streamflow south of Nickajack Dam and north of Bellefonte Island.

Water released from Nickajack Dam will reach the intake site within 8 to 12 hours. The essential corrective action taken, in the unlikely event of downstream dam failure, is the opening of Nickajack Dam floodgates to ensure sufficient flow at the intake channel. Though this action would be initiated soon after such an occurrence, the river response times stated illustrate the time margin that is allowed. (A redundant telecommunication system serves necessary control and observation locations.)[8]

Sufficient redundancy exists at Nickajack Dam to ensure floodgate operability under all credible accident conditions.[9] Since the natural flow of 3400 cfs is sufficient to yield the design water surface elevation of 571 feet, no holding capacity is required at Nickajack Dam to provide the ultimate heat sink. Furthermore, since this is the minimum natural flow of record, the heat sink "30-day capability" is met.

For calculation of essential raw cooling water pump submergence requirements, the 571' water surface elevation was not taken as the level of the water in the ERCW sump (bottom elevation 557'). An assumed water surface slope of 0.1% over the 2000-foot channel length, and an additional loss of 0.5 foot, for the transition from forebay to sump were used resulting in a sump low water surface elevation of 568.5 feet. The ERCW pumps require approximately 8 feet of submergence.

Their intake location at elevation 559' provides adequate head under the worst possible water availability conditions. These pumps are sized so as to provide a minimum plant flow[6] for emergency service. This is the flow sufficient to mitigate the consequences of a LOCA in one unit, with simultaneous safe shutdown of the other unit, and possible subsequent long-term dual maintenance of the safe shutdown condition until appropriate corrective measures can be executed.

The effects of the failure of Nickajack is discussed in section 2.4.4.

REFERENCES

(Section 9.2.5)

(1) PSAR section 2.5.2.10.

(2) PSAR section 2.4.3.

(3) PSAR section 2.4.11.

(4) PSAR section 2.2.3.

(5) PSAR section 2.4.8.1.

(6) PSAR section 9.2.1 and Table 9.2-4.

(7) PSAR section 2.4.11.3.

(8) PSAR section 9.5.2.4.

(9) PSAR section 2.4.11.2.

9.2.6. Condensate Storage Facilities

The condensate storage facilities include the condensate storage tanks, related piping, and transfer pumps which are shown on Figure 10.4-1.

9.2.6.1. Design Bases

The condensate storage tanks maintain the hotwell level, supply condensate to the auxiliary feedwater pumps, and provide makeup to the auxiliary boilers (house boilers).

The size of each condensate storage tank is based on the volume needed to store a condensate and feedwater system fill plus 300,000 gallons for auxiliary feedwater.

The condensate storage tanks, transfer pumps, and related piping up to the auxiliary feedwater isolation valves will be nonnuclear safety (NNS).

9.2.6.2. System Description

The condensate storage facilities consist of one 700,000-gallon steel, epoxy-phenolic lined storage tank per unit. Through piping crossties and transfer pump, either one of the two tanks can be used on either or both units for flexibility during operation.

The condensate storage tanks are located outside the turbine building with supply piping to the hotwell and return piping from the condensate piping system downstream of the demineralizer. Makeup to the storage tanks is provided by the water treatment plant.

9.2.6.3. Safety Evaluation

Each condensate storage tank is equipped with a standpipe which provides a minimum of 300,000 gallons of condensate always available to the auxiliary feedwater pumps (for operation time see subsection 10.4.7.2, Auxiliary Feedwater). The exposed suction piping to the auxiliary feedwater pumps inside the auxiliary building will be in accordance with ASME B&PV Code, Section III, Class 3, 1971 edition requirements. Inside the auxiliary building the auxiliary feedwater suction piping will be embedded in seismically qualified concrete to the point where it leaves the floor and attaches to the auxiliary feedwater pumps. Isolation valves will be provided in the suction piping immediately after it turns up from the floor. 1

In the event of a passive failure that allows flow from the condensate storage tank into the auxiliary building, covered dropout hatches located in the floor at elevation 590.0 will prevent flooding of safety equipment. The passive sump located at elevation 579.0 is sized to contain the condensate storage tank volume.

The storage tanks are not seismically qualified, nor tornadoproof; any loss of the tanks would be compensated for by using essential raw cooling water for auxiliary feedwater supply (see subsection 10.4.7.2, Auxiliary Feedwater).

Since all condensate in the storage tanks comes from either the makeup water-treatment plant or the condensate system downstream of the condensate demineralizer, no concentration or radioactivity is expected to occur in the tanks, and no special provisions are provided to contain leakage from the tanks.

For freeze protection the condensate transfer pump and exposed piping outside the powerhouse are heat traced. Due to the quantity of water stored in the condensate storage tanks, no freeze protection is provided for these tanks.

9.2.6.4. Tests and Inspections

A visual inspection of the tank linings and a hydrostatic test will be the only inspections and tests required.

9.2.6.5. Instrumentation Applications

The condensate storage tanks will have float-type level indicators with a minimum level annunciation in the control room.

9.2.7. Raw Cooling Water System

9.2.7.1. Design Bases

The raw cooling water system is a nonessential shared system which serves both units and is capable of satisfying the cooling water demands for turbine building equipment during the full range of turbine operation. The system, consisting of pumps, strainers, associated piping, and valves, receives its water from the condenser circulating water intake conduits, and returns the water to the condenser circulating water discharge conduits. The condenser circulating water system is a closed cycle cooling system which utilizes cooling towers to provide the ultimate heat sink.

This system is not part of the Engineered Safety Features System (ESF), and none of its components are located near ESF equipment. Therefore, the system is not seismically designed, nor does it have a nuclear safety class, and is not available during loss of offsite power.

9.2.7.2. System Description

The system is provided with two 14,000-gpm raw cooling water pumps per unit (see Table 9.2-6 for pump design data) which are located in the condenser pit of the turbine building. These pumps are controlled locally or from the main control room. Normally, one pmup per unit is operating and one pump per unit is on standby. The standby pump is started on a low flow alarm automatically, when the discharge header flow drops to a preset level.

The raw cooling water pumps take suction from their respective unit's condenser circulating water intake conduit through individual pump strainers and discharge through two supply lines (one per unit) to loop headers which surround each unit in the turbine building. Isolation valves are provided in the interconnecting lines between the unit loop headers to allow for unit isolation, if required. Each loop header contains isolation and sectionalizing valves so segments of the system can be isolated for maintenance without removing the rest of the system from service. These valves also provide for backflushing sediment from the headers by reversing flow and increasing the velocity. Flush lines are piped to the year drainage system, which in turn flows to the yard drainage holding pond.

The loop headers provide cooling water directly to the following equipment (see Figure 9.2-12 for the flow diagram):

1. Main turbine lubricating oil heat exchangers.

2. Feedwater pump-turbine oil heat exchangers.

3. Electrical bus heat exchangers.

4. Condenser vacuum pump heat exchangers.

5. Hydrogen seal oil heat exchangers.

6. Main turbine control fluid heat exchangers.

7. Condensate booster pump heat exchangers.

8. Stator cooling water tank coolers.

9. Other miscellaneous pumps and equipment.

After heat has been transferred to the cooling water from this equipment, the water is discharged into unit return headers. These headers are then routed to their respective unit's condenser circulating water discharge conduit and thence to the cooling towers.

Cooling water for the generator hydrogen and generator stator heat exchangers is supplied by the circulating pumps (see Table 9.2-6). These pumps, one operating pump and one standby pump per unit, take suction from the loop headers and discharge into lines which supply the heat exhangers. The cooling water outlets discharge into headers which divide into two lines, each of which includes flow control valves. These valves, controlled by the temperature of the hydrogen heat exchangers, divert all or part of the cooling water back to the suction of the circulating pumps. Water not recycled through the heat exchangers is routed to the unit return headers and one to the condenser circulating water discharge conduits in the same manner as the other heat exchangers previously discussed.

9.2.7.3. Safety Evaluation

The raw cooling water system is not required for maintenance of plant safety in the event of a loss-of-coolant accident (LOCA), Safe Shutdown Earthquake, or probable maximum flood conditions.

9.2.7.4. Tests and Inspections

All system piping and components are hydrostatically tested prior to station startup and, with the exception of piping embedded in concrete, are accessible for periodic inspection after startup.

9.2.7.5. Instrumentation Applications

Temperature controls on most of the various heat exchangers operate as follows: A thermocouple is placed in the output of the process line leaving heat exchangers. The process line may contain oil, water, air, or gas. A control valve in the process line controls the flow of the fluid passing through the coolers depending on the exit temperature of the cooled fluid from the coolers. The cooling water flow is throttled with a manually throttled valve that is adjusted (and locked) to the flow rate. Temperature indicators and test wells are installed at the cooling water inlets and outlets of the various heat exchangers for operation and test when required. Pump discharge lines are monitored locally by pressure gauges and flowmeters to determine which pump or pumps are operating properly.

9.2.8. Borated Water Storage Tank

9.2.8.1. Design Bases

The borated water storage tank (BWST) fulfills two basic requirements:

1. It provides an adequate supply of borated water (boron concentration of approximately 1800 ppm) for use during refueling operations.

2. It provides an adequate supply of borated water (boron concentration of approximately 1800 ppm) for use by the high-pressure injection system (HPIS), the low-pressure injection system (LPIS), and the reactor building spray system (RBSS) in the event of a loss-of-coolant accident (LOCA).

The following criteria are used to fulfill the above requirements:

1. The size of the tank is sufficient to contain the largest of the following:

   a. The amount of water required to fill the refueling canal, transfer canal, and two fuel transfer tubes. (390,480 gallons) 7

   b. The amount of water required, in addition to that in the core flooding tanks, to establish the emergency cooling recirculation mode following a LOCA (i.e., the depth of water provided in the reactor building will be sufficient to provide free flow to the emergency sump and to provide adequate suction head for the DHR and RBS pumps). (570,000 gallons)

   c. The amount of water required to supply the HPIS, LPIS, and RBSS (with all pumps operating at runout condition) for a period of time (10 minutes or more) sufficient to allow the operator to properly assess the situation and establish the recirculation mode following a LOCA. The suction requirements are tabulated below: 7

Two HPIS pumps @ 700 gpm	=	1,400 gpm
Two LPIS pumps @ 6,500 gpm	=	13,000 gpm
Two RBSS pumps @ 2,400 gpm	=	4,800 gpm
Total	=	19,200 gpm

Figure 9.2-13 shows the BWST design parameters. The tank volume to the low level alarm was determined by the quantity of water required to fill the static containment volume to the top of the emergency sumps, plus the volume required to provide sufficient NPSH to the pumps. An additional volume was added to provide one minute of pumping time at the maximum rate indicated above (19,200 gpm) between the low and low-low level alarms. A further reserve volume provides an additional 4.9 minutes of pumping time. The tank volume thus provides nearly 30 minutes of pumping time to the low level alarm for all six pumps running at maximum capacity. Below the low level alarm, nearly six additional minutes of pumping time (capacity) are provided. At the time when the low level alarm point is reached (whether in the minimum 29.7 minutes or in an extended period), there will be a sufficient depth of water in the containment to ensure adequate NPSH for the pumps, and nearly six minutes will be provided at the maximum pumping rate for conversion to the recycle mode of operation. 7

The 570,000-gallon nominal tank capacity is equivalent to 76,208 ft^3. Of this, 25,294 ft^3 will fill the containment volume to the top of the emergency sumps (see response to question 6.65), and the remaining 50,914 ft^3 will provide a head of 32.49 ft for the pump suction. This head is more than adequate, as discussed in the response to question 9.37.

2. The tank is designed to accommodate both static and dynamic loading effects associated with a Safe Shutdown Earthquake (SSE) or tornado 13
winds.

3. The tank is designed for a lifetime of 40 years with essentially no maintenance except for normal inspection, cleaning, and flushing.

9.2.8.2. System Description

There is one borated water-storage tank (minimum capacity, 700,000 gallons) per nuclear unit for the Bellefonte Nuclear Plant, being located outside at approximately grade level at El 646.0 and in close proximity to its respective reactor building. The functional requirements for the tank are discussed in Chapter 6.

The tank is equipped with an atmospheric vent. The vent is designed to 13
pass a volume flow rate of air that is greater than the maximum withdrawal rate (20,000 gpm) from the tank. Necessary precautions have been taken in the design of the vent to assure that birds, animals, and/or other foreign objects including rain cannot enter the tank.

A sketch of the tank showing its capacity, water levels, and general speci-
fications is provided in Figure 9.2-13. For additional design information 13
about the BWST, see section 3.7, 3.8, and Figure 3.8-40c.

9.2.8.3. Safety Evaluation

The BWST is essential for the safety of the plant during the injection mode (approximately 30 minutes) of emergency cooling following a LOCA as well as other design bases events (DBE). The design and fabrication of the BWST is of sufficient integrity such as to preclude any significant probability of a failure of the BWST coincident with the occurrence of a LOCA, during the injection mode of a LOCA, or any other DBE requiring the BWST. The BWST and the reactor coolant system are both designed to withstand the effects of an SSE. The BWST is designed to withstand the effects of a tornado wind.

9.2.8.4. Test and Inspections

Periodic visual inspections will be made during the life of the plant to assure the structures integrity.

9.2.8.5. Instrument Application

Refer to Chapter 7.

BLANK

Table 9.2-1. Essential Raw Cooling Water Pumps Performance Data

Quantity	8
Type	Vertical, turbine
Rated capacity gpm	12,000
Design head, ft	240
Motor horsepower, hp	900
Material	CS
Design pressure, psig	200
Design temperature, F	150
ASME Code, Section/Class	III/3
Seismic Category	I

Table 9.2-2. Essential Raw Cooling Water System Minimum Flow Requirements, gpm (Based on 95 F Cooling Water)

Equipment	Normal operation 2 Units full power	1 Unit LOCA 1 Unit full power (1)	2 Units cooldown (1)	1 Unit LOCA 1 Unit cooldown (1)	1 Unit LOCA 1 Unit refueling (1)
Component Cooling Water Heat Exchangers	21,000	21,000	21,000	21,000	21,000
Emergency Diesel Generator Heat Exchangers	-	2,000	2,000	2,000	2,000
Control Building Water Chillers	800	800	800	800	800
Auxiliary Building Cooling Systems	524	624	400	562	647
Air Compressors	200	200	200	200	200
Spent Fuel Cooling Pit Makeup	-	100	100	100	100
Control Rod Drive Cooling Water Coolers	406	203	406	203	-
Reactor Building Coolers	2,000	6,100	2,000	6,100	6,100
Auxiliary Turbine Driven Feedwater Pump	-	-	1,200 (2)	1,200 (2)	-
Auxiliary Motor Driven Feedwater Pump	-	600 (2)	1,200 (2)	600 (2)	600 (2)
TOTAL	24,930	31,627	29,306	32,765	31,447
Minimum Number of Pumps Required	2	3	3	3	3
Number in Service	3	4	4	4	4
Number Available	4	4	4	4	4

(1) Minimum flow requirements are based on the assumption of only one power train operating in each unit, thus eliminating cooling water flow requirement for redundant equipment.

(2) The cooldown unit requires 1200 gpm only for the first six hours of the cooldown mode. The LOCA unit will only require the steam generators be filled with ERCW to maintain pressure in the steam generator to contain the primary leak.

Table 9.2-3. Essential Raw Cooling Water System Maximum Flow Requirements, gpm (Based on 95 F Cooling Water)

Equipment	Normal operation 2 Units full power	1 Unit LOCA 1 Unit full power (1)	2 Units cooldown (1)	1 Unit LOCA 1 Unit cooldown (1)	1 Unit LOCA 1 Unit refueling (1)
Component Cooling Water Heat Exchangers	21,000	31,500	42,000	42,000	42,000
Emergency Diesel Generator Heat Exchangers	-	4,000	4,000	4,000	4,000
Control Building Water Chillders	800	800	800	800	800
Auxiliary Building Cooling Systems	524	624	400	562	647
Air Compressors	200	200	200	200	200
Spent Fuel Cooling Pit Makeup	-	100	100	100	100
Control Rod Drive Cooling Water Coolers	406	203	406	203	-
Reactor Building Coolers	3,000	9,150	3,000	9,150	9,150
Auxiliary Turbine Driven Feedwater Pump	-	-	1,200 (2)	1,200 (2)	-
Auxiliary Motor Driven Feedwater Pump	-	600 (2)	1,200 (2)	600 (2)	600
TOTAL	25,930	47,177	53,306	58,815	57,497
Minimum Number of Pumps Required	2	4	5	5	5
Number in Service	3	5	6	6	6
Number Available (Assuming one pump on each loop out for maintenance)	6	6	6	6	6

(1) Maximum flow requirements are based on all equipment being available for operation.

(2) The coolant unit requires 1200 gpm only for the first six hours of the cooldown mode. The LOCA unit will only require the steam generator be filled with ERCW to maintain pressure in the steam generator to contain the primary leak.

Table 9.2-4. Essential Raw Cooling Water System Minimum Flow Requirements for Unit 1 Operation Prior To Unit 2 Operation, gpm (Based on 95 F Cooling Water)

Equipment	Heatup	Full power	Cooldown (1)	LOCA (1)	Refueling (1)
Component Cooling Water Heat Exchangers	10,500	10,500	10,500	10,500	10,500
Emergency Diesel Generator Heat Exchangers	-	-	1,000	1,000	-
Control Building Water Chillers	800	800	800	800	800
Auxiliary Building Cooling Systems	524	624	400	562	647
Air Compressors	100	100	100	100	100
Spent Fuel Cooling Pit Makeup	-	-	100	100	100
Control Rod Drive Cooling Water Coolers	203	203	203	-	-
Reactor Building Coolers	1,000	1,000	1,000	5,100	1,000
Auxiliary Turbine Driven Feedwater Pump	-	-	1,200(2)	-	-
Auxiliary Motor Driven Feedwater Pump	-	-	-	600(2)	-
TOTAL	13,127	13,227	15,303	18,762	13,147
Minimum Number of Pumps Required	1	1	1	2	1
Number in Service	2	2	2	2	2
Number Available	2	2	2	2	2

(1) Minimum flow requirements are based on the assumption of a loss of one power train.

(2) The cooldown unit requires 1200 gpm only for the first six hours of the cooldown mode. The LOCA unit will only require steam generators be filled with ERCW to maintain pressure in the steam generator to contain the primary leak.

Table 9.2-5. Component Performance Data (On a per Unit Basis)

Component Cooling Water Pumps

Quantity	3	
Type	Horizontal centrifugal	
Rated capacity, emergency/normal, gpm	7,500/10,500	
Design head, emergency/normal, ft	280/150	
Motor horsepower, hp	700	9
Material	SS	
Design pressure, psig	200	
Design temperature, F	200	
ASME Code, Section/Class	III/3	
Seismic Category	I	

Component Cooling Water Heat Exchangers

Quantity	2	
Type	shell/tube	
Material, shell/tube	CS/CuNi	
Cooling capacity, Btu/h	112.8×10^6	
Cooling water flow (raw water), gpm	10,500	11
Design pressure, shell/tube, psig	200/200	
Design temperature, shell/tube, F	200/200	
ASME Code, Section/Class	III/3	
Seismic Category	I	

CRD Cooling Water Circulation Pump

Quantity	2	
Type	Centrifugal	
Capacity, gpm	220	
Head, ft	230	
Design pressure, psig	200	
Design temperature, F	200	
Material	SS	
ASME Code, Section/Class	III/3	
Motor horsepower, hp	25	9
Seismic Category	I	

Table 9.2-5. (Cont'd)

CRD Cooling Water Heat Exchangers

Quantity	2	
Type	Shell/tube	
Material, shell/tube	SS/CuNi	
Cooling capacity, Btu/h	121.5×10^4	
Cooling water flow (raw water), gpm	215	
Cooling water flow, lb/h	1.085×10^5	
Design pressure, shell/tube, psig	200/200	11
Design temperature, shell/tube, F	200/200	
ASME Code, Section/Class	III/3	
Seismic Category	I	

Table 9.2-6. Raw Cooling Water System Pump Design Data

Raw Cooling Water Pumps

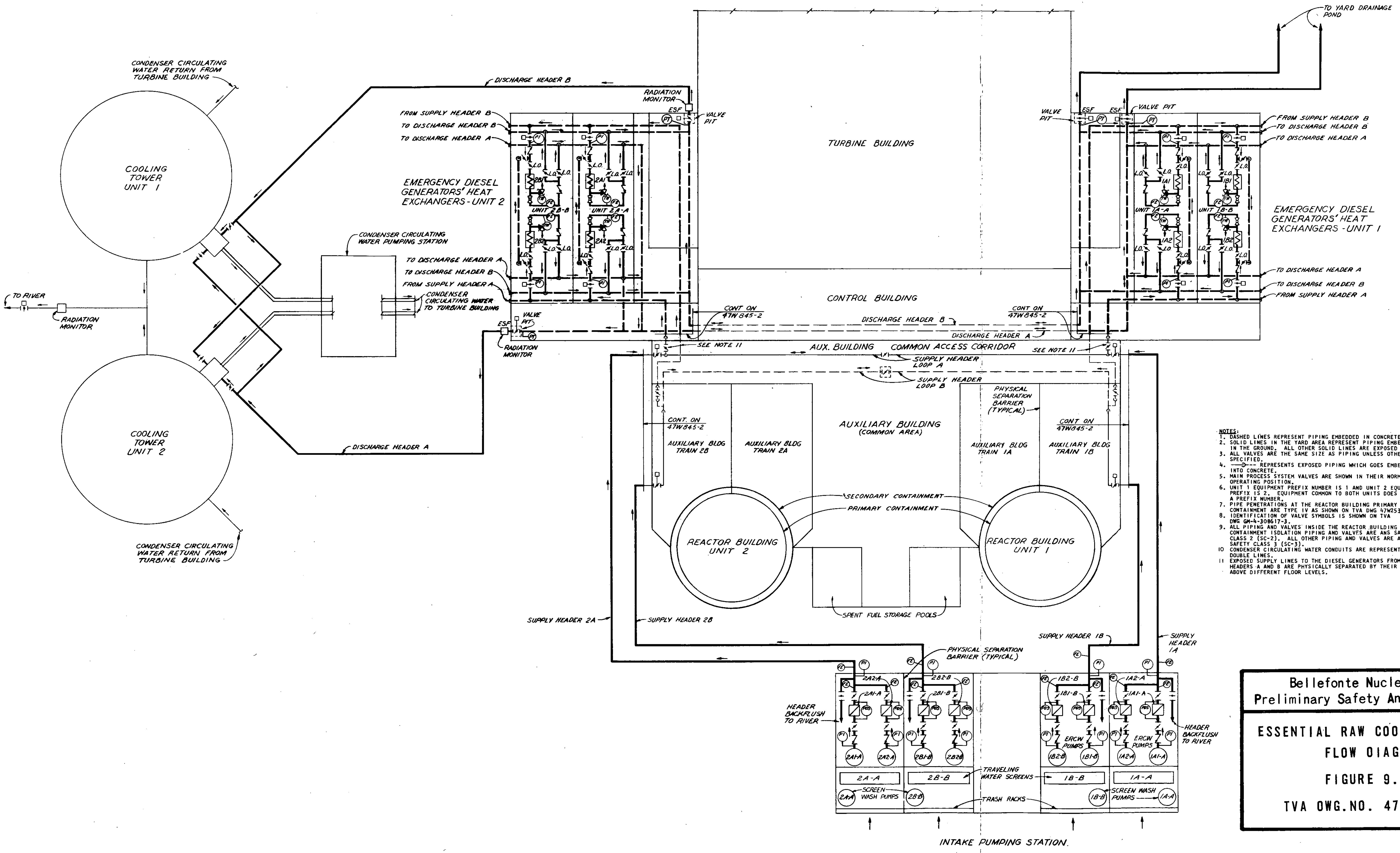
	Unit 1	Unit 2
Quantity	2	2
Type	Horizontal mixed flow	
Rated capacity (gpm)	14,000	14,000
Rated head (feet of water)	55	55
Motor horsepower	250	250

Raw Cooling Water Circulating Pumps

	Unit 1	Unit 2
Quantity	2	2
Type	Horizontal mixed flow	
Rated capacity (gpm)	6,000	6,000
Rated head (feet of water)	45	45
Motor horsepower	100	100

Table 9.2-7. Raw Cooling Water System Flow Requirements (Based on 95 F Cooling Water)

Equipment	Unit 1 RCW quantity GPM, Maximum	Unit 2 RCW quantity GPM, Maximum
Hydrogen coolers	2,980	2,980
Main lubricating oil coolers	2,630	2,630
Stator coolers	2,630	2,630
Seal oil coolers	266	266
Main control fluid coolers	328	328
Generator bus coolers	600	600
Backflush raw water headers	210	210
Total steady flow	9,644	9,644
Feedwater pump coolers	700	700
Condenser vacuum pumps	400	400
Condensate booster pumps	300	300
Miscellaneous loads (pump, etc.)	250	250
Total intermittent flow	1,650	1,650
Total maximum flow	11,294	11,294



NOTES:

1. DASHED LINES REPRESENT PIPING EMBEDDED IN CONCRETE.
2. SOLID LINES IN THE YARD AREA REPRESENT PIPING EMBEDDED IN THE GROUND. ALL OTHER SOLID LINES ARE EXPOSED PIPING.
3. ALL VALVES ARE THE SAME SIZE AS PIPING UNLESS OTHERWISE SPECIFIED.
4. ——⊃--- REPRESENTS EXPOSED PIPING WHICH GOES EMBEDDED INTO CONCRETE.
5. MAIN PROCESS SYSTEM VALVES ARE SHOWN IN THEIR NORMAL OPERATING POSITION.
6. UNIT 1 EQUIPMENT PREFIX NUMBER IS 1 AND UNIT 2 EQUIPMENT PREFIX IS 2. EQUIPMENT COMMON TO BOTH UNITS DOES NOT HAVE A PREFIX NUMBER.
7. PIPE PENETRATIONS AT THE REACTOR BUILDING PRIMARY CONTAINMENT ARE TYPE IV AS SHOWN ON TVA DWG 47W253-11.
8. IDENTIFICATION OF VALVE SYMBOLS IS SHOWN ON TVA DWG GM-4-30B617-3.
9. ALL PIPING AND VALVES INSIDE THE REACTOR BUILDING AND ALL CONTAINMENT ISOLATION PIPING AND VALVES ARE ANS SAFETY CLASS 2 (SC-2). ALL OTHER PIPING AND VALVES ARE ANS SAFETY CLASS 3 (SC-3).
10. CONDENSER CIRCULATING WATER CONDUITS ARE REPRESENTED BY DOUBLE LINES.
11. EXPOSED SUPPLY LINES TO THE DIESEL GENERATORS FROM LOOP HEADERS A AND B ARE PHYSICALLY SEPARATED BY THEIR LOCATION ABOVE DIFFERENT FLOOR LEVELS.

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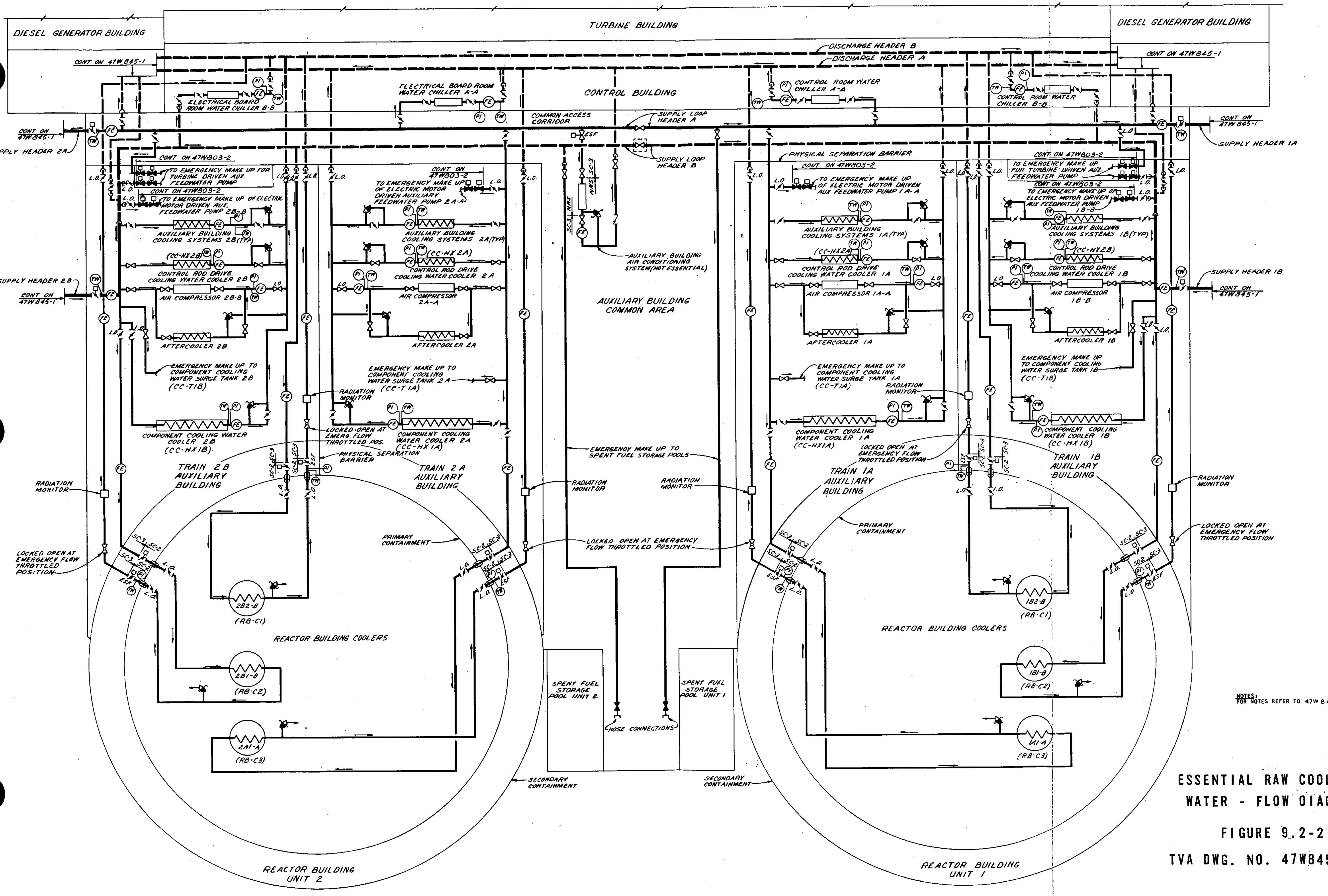
ESSENTIAL RAW COOLING WATER-FLOW DIAGRAM

FIGURE 9.2-1

TVA DWG.NO. 47W845-1 RO

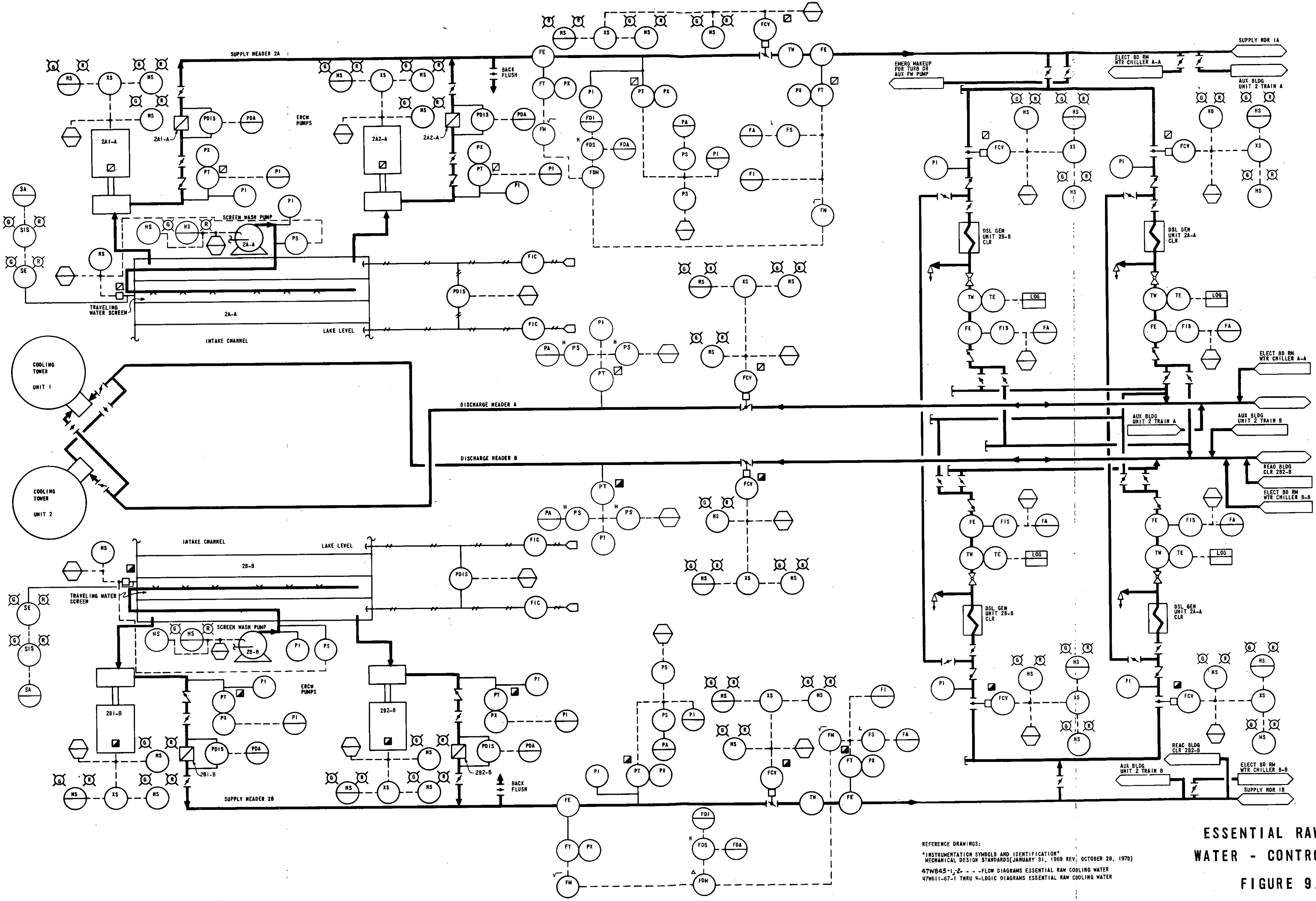


NOTES:
FOR NOTES REFER TO 47W845-1

ESSENTIAL RAW COOLING WATER - FLOW DIAGRAM

FIGURE 9.2-2

TVA DWG. NO. 47W845-2 RO



REFERENCE DRAWINGS:

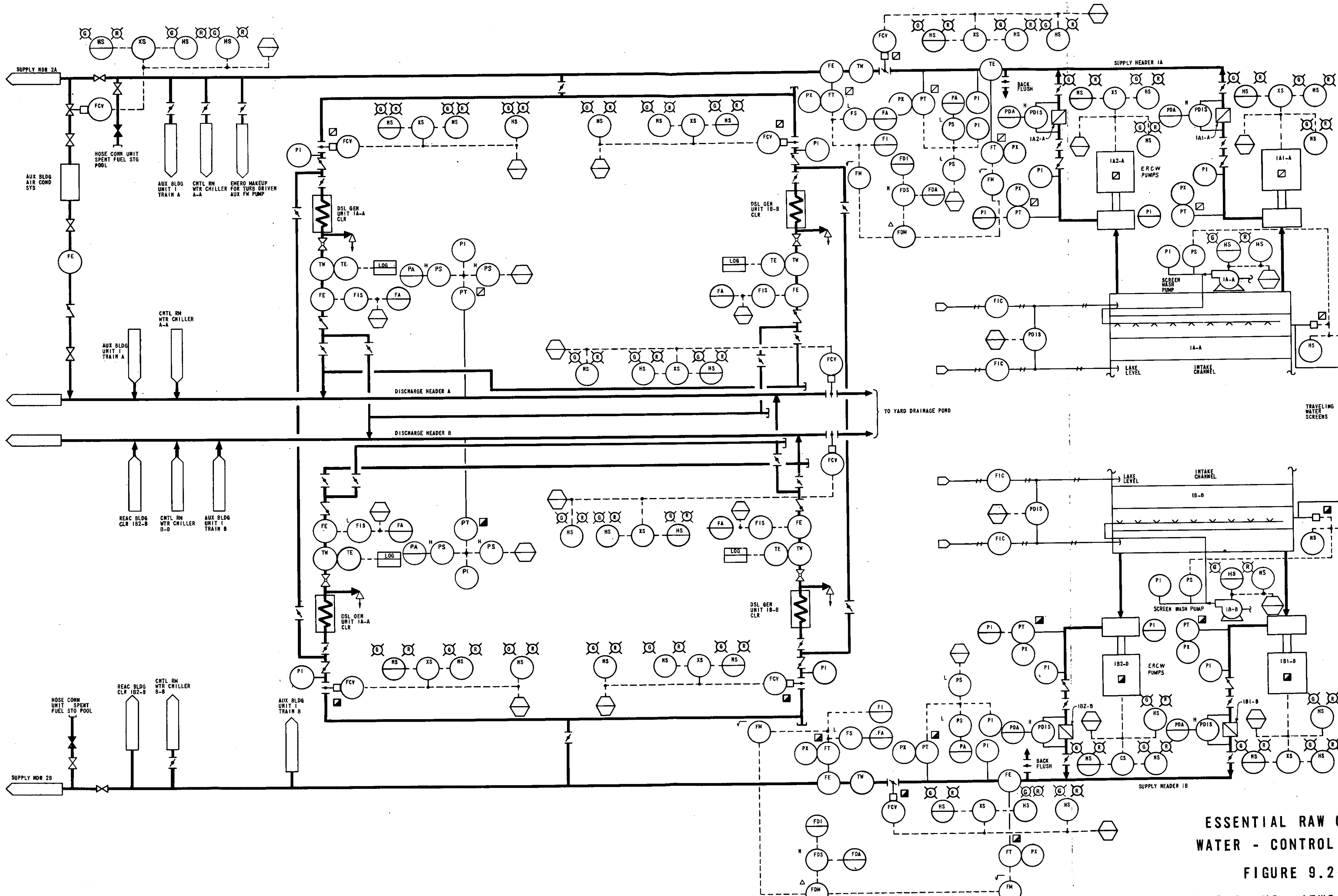
"INSTRUMENTATION SYMBOLB AND IDENTIFICATION"
MECHANICAL DESIGN STANDARDS(JANUARY 31, 1969 REV; OCTOBER 28, 1970)

47W845-1,-2- - - -FLOW DIAGRAMS ESSENTIAL RAW COOLING WATER
47W611-67-1 THRU 4-LOGIC DIAGRAMS ESSENTIAL RAW COOLING WATER

ESSENTIAL RAW COOLING WATER - CONTROL DIAGRAM

FIGURE 9.2-3

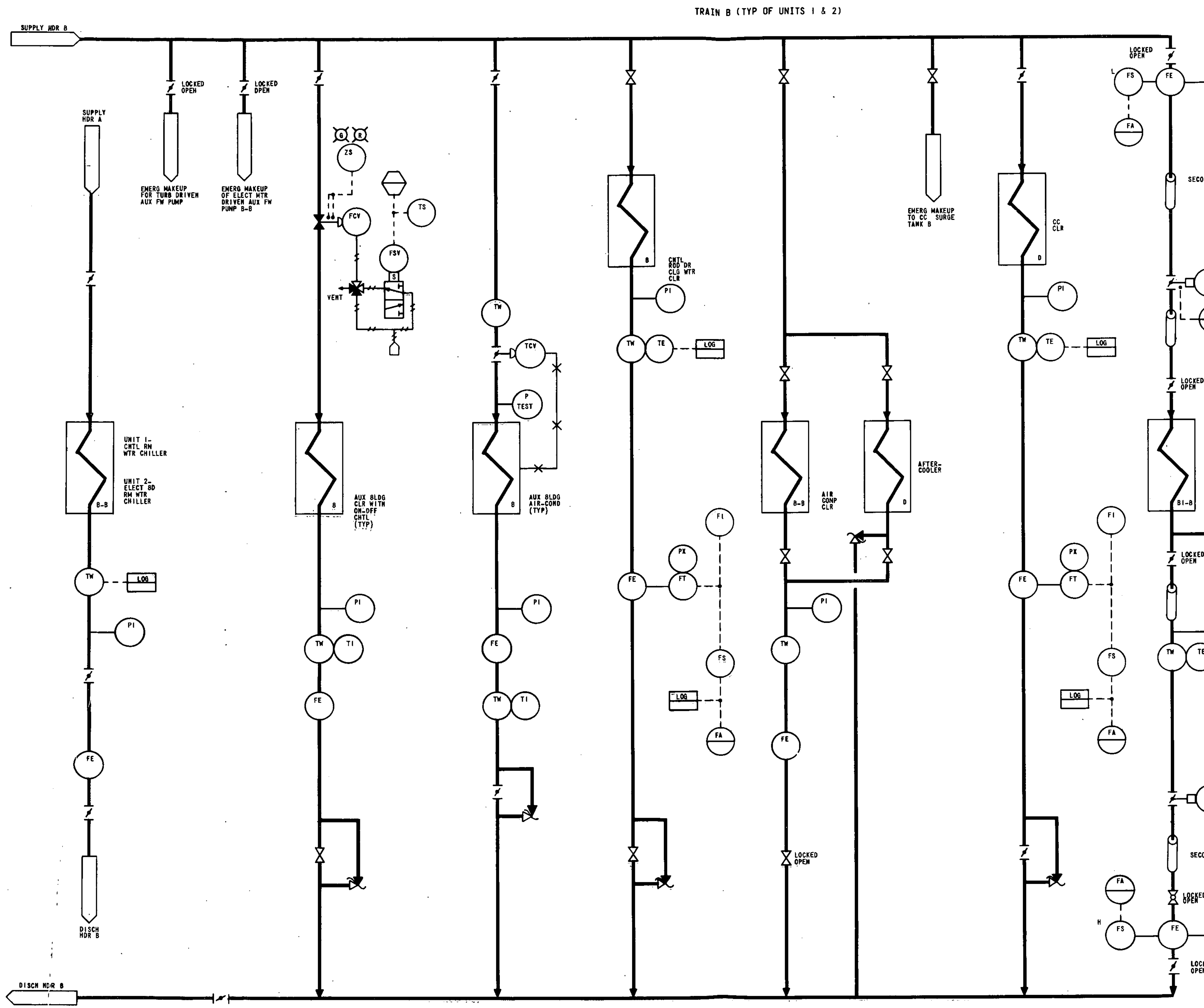
TVA DWG. NO. 47W610-87-1 R0

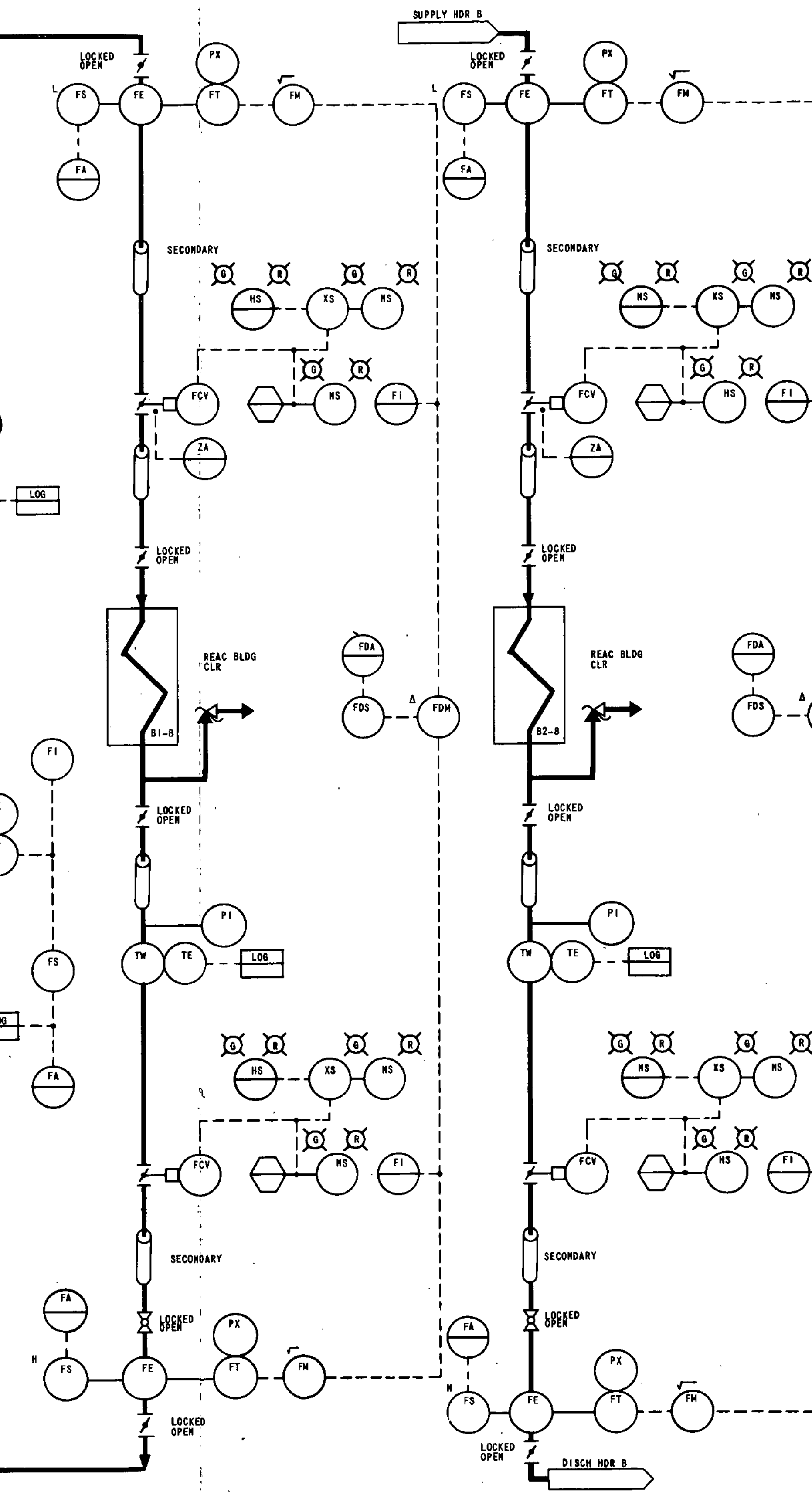


ESSENTIAL RAW COOLING WATER - CONTROL DIAGRAM

FIGURE 9.2-4

TVA DWG. NO. 47W610-87-2 RO





ESSENTIAL RAW COOLING WATER - CONTROL DIAGRAM

FIGURE 9.2-5

TVA DWG. NO. 47W810-67-4 R0

RESERVATIONS:

1. PIPE SIZES CONTINGENT UPON FINAL PIPING CONFIGURATION.
2. PIPE SCHEDULE NUMBERS TO BE ESTABLISHED BY CUSTOMER.
3. FOR COMPLETE FILLING AND DRAINING OF THE SYSTEM, ADDITIONAL VENTS AND DRAINS MAY BE NECESSARY. FLUSH CONNECTION LOCATIONS FOR INITIAL CLEANING TO BE DETERMINED BY CUSTOMER.
4. CUSTOMER TO INVESTIGATE NEED FOR AND PROVIDE IF REQUIRED PROTECTION OF ISOLATED EQUIPMENT AGAINST PRESSURE BUILDUP DUE TO AMBIENT TEMPERATURE CHANGE.
5. IN ACCORDANCE WITH NOTE 4, THE SYSTEM LAYOUT SHALL BE SUCH THAT THE SUM OF THE SURGE TANK ELEVATION HEAD AND THE PUMP SHUTOFF HEAD SHALL NOT EXCEED THE DESIGN PRESSURE OF THE SYSTEM.
6. INPUT TO LOW FLOW INTERLOCK TO CONTROLLED BLEEDOFF VALVES (INSTRUMENT STRINGS 12-2) IS DEPENDENT UPON FINAL SELECTION OF RC PUMPS. INPUT TO INSTRUMENT STRING 12-2 MAY BE CHANGED TO CC SYSTEM INLET TO RC PUMP SEAL COOLERS (NODE 94).

REFERENCE DRAWINGS:

NO.	SYSTEM
1	RC SYSTEM
2	MU SYSTEM
4	DH SYSTEM
8	SFC SYSTEM
9	CC SYSTEM
10	CC SYSTEM
12	WC SYSTEM
14	ESSENTIAL RAW COOLING WATER SYSTEM (CUSTOMER)
16	WC SYSTEM
17	WD SYSTEM
C	CUSTOMER SYSTEM

NOTES:

1. ITEMS WITH COMPONENT NUMBERS SUPPLIED BY B&W.
2. INSTRUMENTATION STRING AND VALVE NUMBERS 1-18, 20, 21-24, 28, 29, 30, 31 AND 43 USED ON THIS DRAWING.
3. SURGE TANK ELEVATIONS FOR THE SYSTEMS SHOWN ON THIS DRAWING TO BE SUCH THAT THE PUMPS NPSH REQUIREMENT WILL BE MET UNDER ALL NORMAL OPERATING CONDITIONS.
4. SURGE TANK ELEVATIONS ARE TO BE HIGHER THAN ANY OTHER PORTION OF THEIR RESPECTIVE SYSTEMS.
5. BUILDING ISOLATION VALVES ARE NOT NORMALLY USED FOR THROTTLING PURPOSES, EXCEPT FOR HV-22
6. TRAIN A&B ARE SEPARATED BY A PHYSICAL BARRIER EXCEPT IN THE SECONDARY CONTAINMENT.
7. STANDBY SIGNAL WILL BE TO THE DESIGNATED STANDBY PUMP ONLY.
8. ESF SIGNAL WILL BE TO THE DESIGNATED RUN PUMP ONLY.

DESIGN CONDITIONS

- 3: 2500 PSIG @ 670°F
- 20: 200 PSIG @ 300°F
- 20: BUILDING DESIGN PRESSURE @ 300°F
- 30: 50 PSIG @ 200°F
- 34: ATMOS PRESS @ 200°F
- 30: 200 PSIG @ 200°F
- 35: 200 PSIG @ 350°F

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COMPONENT COOLING WATER SYSTEM

FIGURE 9.2-6

REVISED PER AMEND. 11, MAY 15, 1974

RESERVATIONS:

1. PIPE SIZES CONTINGENT UPONG FINAL PIPING CONFIGURATION.
2. PIPE SCHEDULE NUMBERS TO BE ESTABLISHED BY CUSTOMER.
3. FOR COMPLETE FILLING AND DRAINING OF THE SYSTEM, ADDITIONAL VENTS AND DRAINS MAY BE NECESSARY. FLUSH CONNECTION LOCATIONS FOR INITIAL CLEANING TO BE DETERMINED BY CUSTOMER.
4. CUSTOMER TO INVESTIGATE NEED FOR AND PROVIDE IF REQUIRED PROTECTION OF ISOLATED EQUIPMENT AGAINST PRESSURE BUILDUP DUE TO AMBIENT TEMPERATURE CHANGE.

REFERENCE DRAWINGS:

NO.	SYSTEM
1	RC SYSTEM
2	MU SYSTEM
4	DH SYSTEM
8	SF SYSTEM
9	CC SYSTEM
10	CC SYSTEM
11	SP SYSTEM
12	WO SYSTEM
13	WD SYSTEM
15	CA & BR SYSTEM
16	WD SYSTEM
17	WD SYSTEM
C	CUSTOMER SYSTEM

NOTES:

1. ITEMS WITH COMPONENT NUMBERS SUPPLIED BY B&W EXCEPT VALVES HV-26 A&B, HV-29 A&B, HV-30 A&B, HV-31 A&B HV-32 A&B, HV-33 A&B, & HV-34 A&B.
2. INSTRUMENTATION AND VALVE NUMBERS 24 THRU 42 USED ON THIS DRAWING.
3. BUILDING ISOLATION VALVES ARE NOT NORMALLY USED FOR THROTTLING PURPOSES.

DESIGN CONDITIONS:

30 50 PSIG @ 200°F
34 ATMOS @ 200°F
36 200 PSIG @ 200°F

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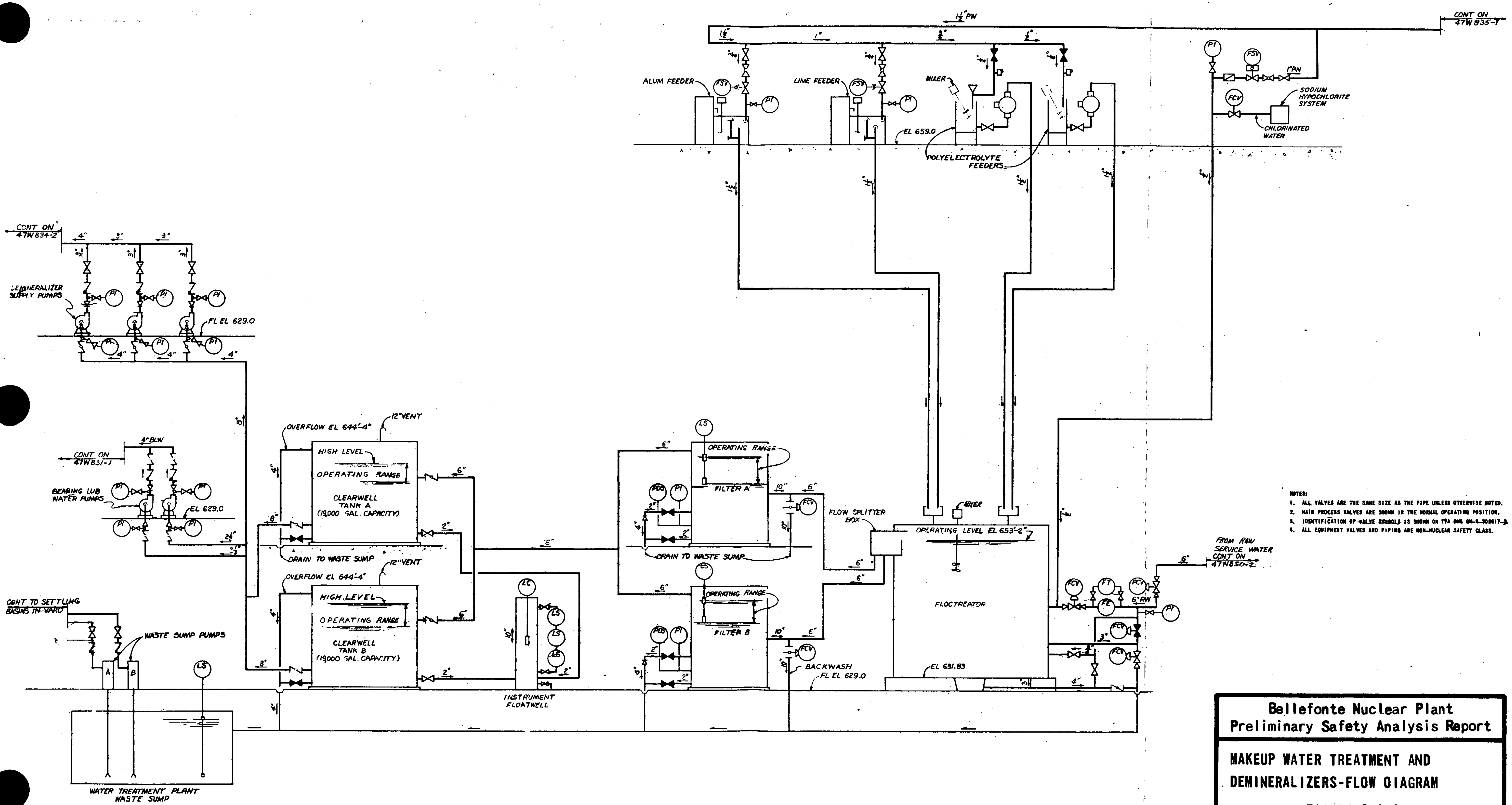
COMPONENT COOLING WATER SYSTEM

SHEET 2 OF 2

FIGURE 9.2-7



NOTES:

1. ALL VALVES ARE THE SAME SIZE AS THE PIPE UNLESS OTHERWISE NOTED.
2. MAIN PROCESS VALVES ARE SHOWN IN THE NORMAL OPERATING POSITION.
3. IDENTIFICATION OF VALVE SYMBOLS IS SHOWN ON TVA DWG [illegible]
4. ALL EQUIPMENT VALVES AND PIPING ARE NON-NUCLEAR SAFETY CLASS.

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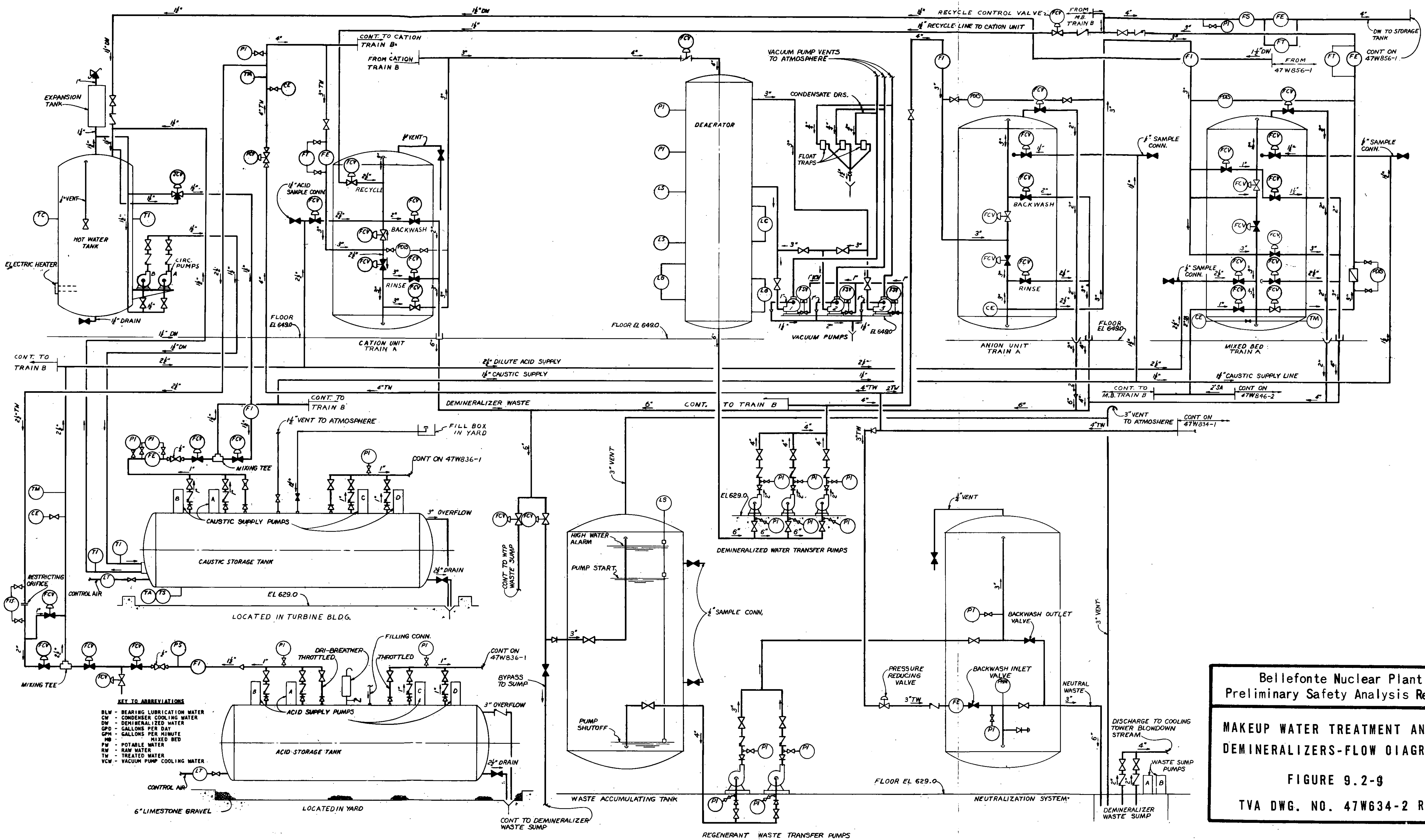
MAKEUP WATER TREATMENT AND DEMINERALIZERS-FLOW DIAGRAM

FIGURE 9.2-8

TVA DWG. NO. 47W834-1 R9S

REVISED PER AMEND. 12, AUG. 14, 1974



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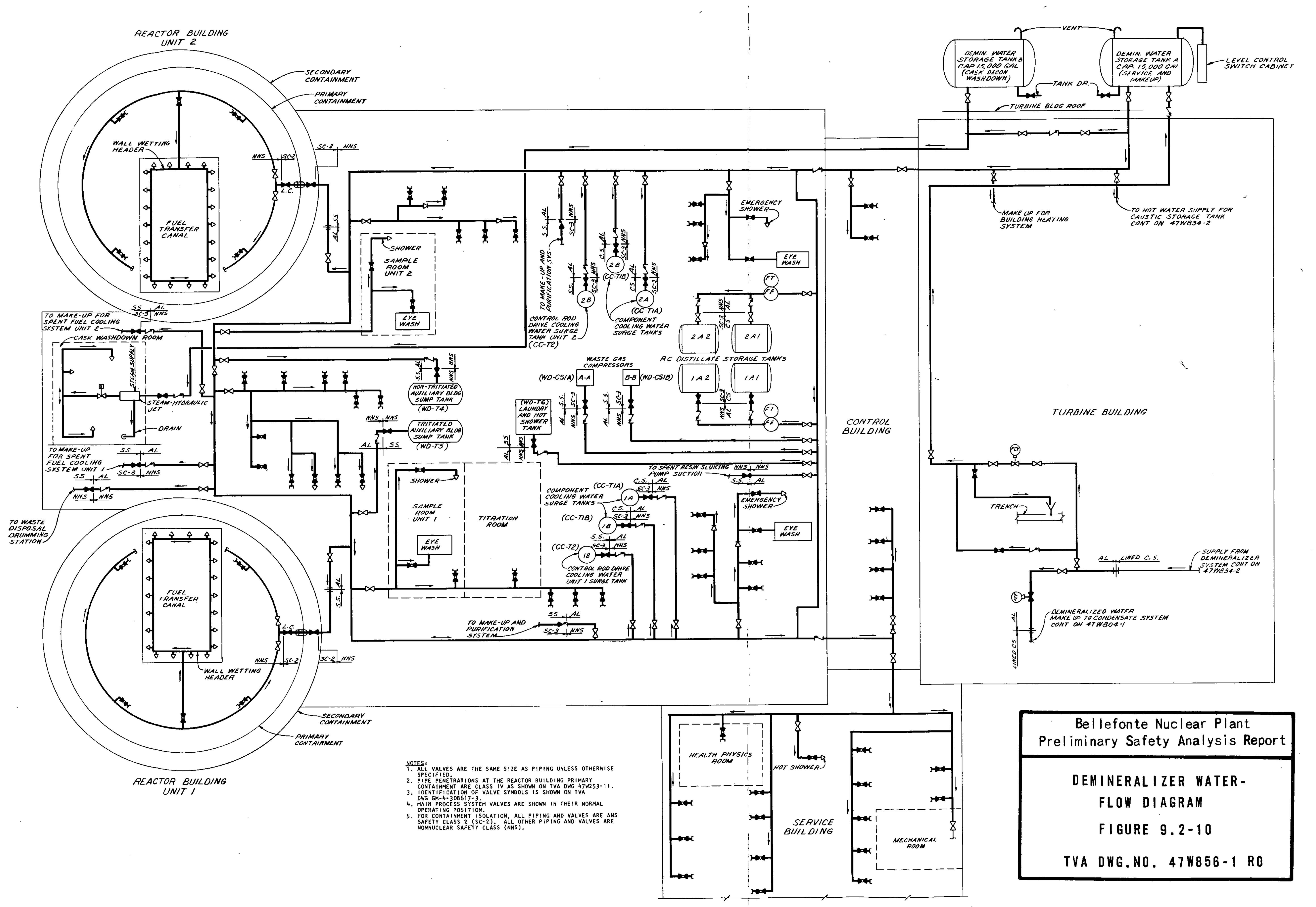
MAKEUP WATER TREATMENT AND DEMINERALIZERS-FLOW DIAGRAM

FIGURE 9.2-9

TVA DWG. NO. 47W634-2 R0



NOTES:

1. ALL VALVES ARE THE SAME SIZE AS PIPING UNLESS OTHERWISE SPECIFIED.
2. PIPE PENETRATIONS AT THE REACTOR BUILDING PRIMARY CONTAINMENT ARE CLASS IV AS SHOWN ON TVA DWG 47W253-11.
3. IDENTIFICATION OF VALVE SYMBOLS IS SHOWN ON TVA DWG GM-4-30B617-3.
4. MAIN PROCESS SYSTEM VALVES ARE SHOWN IN THEIR NORMAL OPERATING POSITION.
5. FOR CONTAINMENT ISOLATION, ALL PIPING AND VALVES ARE ANS SAFETY CLASS 2 (SC-2). ALL OTHER PIPING AND VALVES ARE NONNUCLEAR SAFETY CLASS (NNS).

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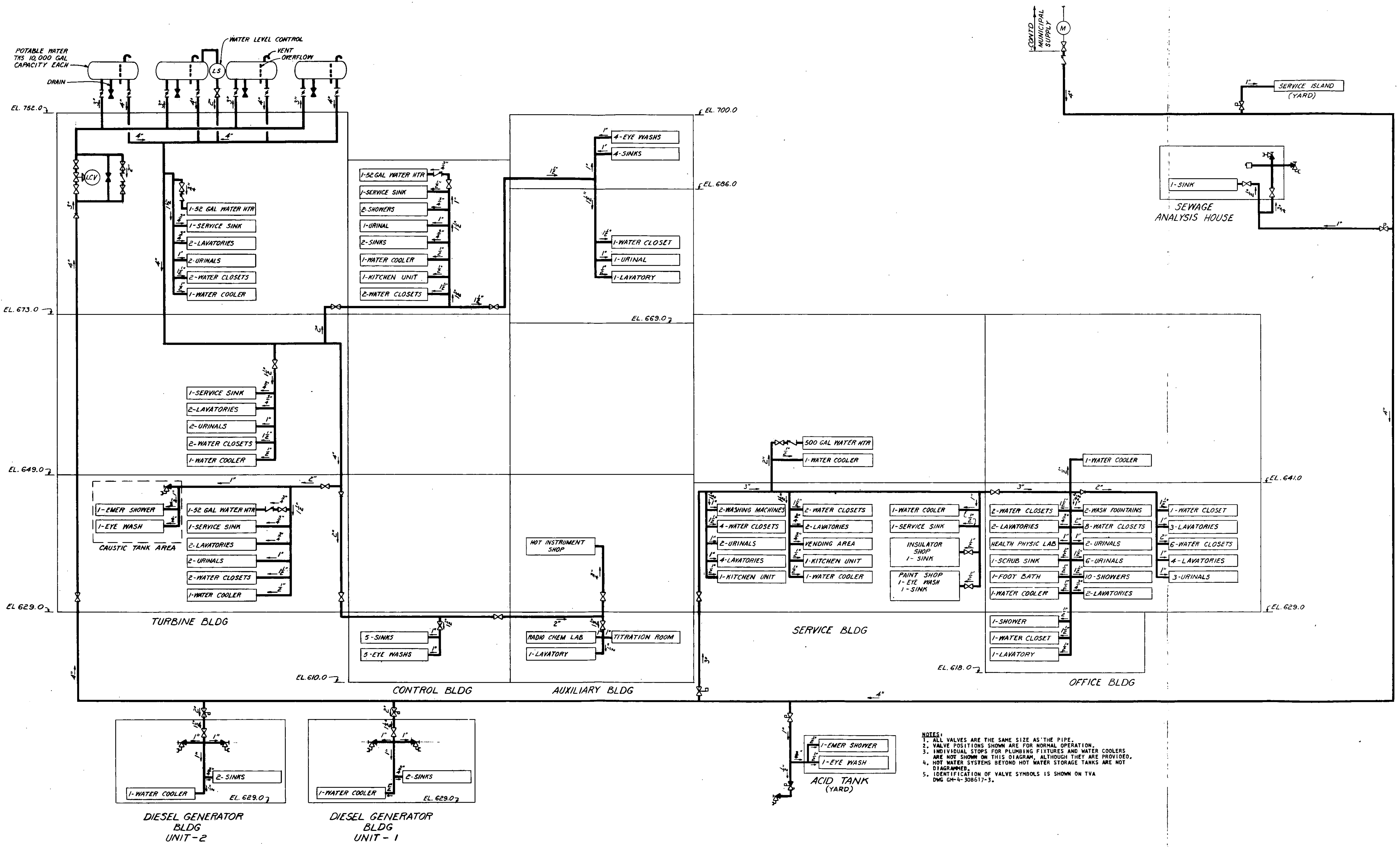
DEMINERALIZER WATER-FLOW DIAGRAM

FIGURE 9.2-10

TVA DWG.NO. 47W856-1 RO



NOTES:
1. ALL VALVES ARE THE SAME SIZE AS THE PIPE.
2. VALVE POSITIONS SHOWN ARE FOR NORMAL OPERATION.
3. INDIVIDUAL STOPS FOR PLUMBING FIXTURES AND WATER COOLERS ARE NOT SHOWN ON THIS DIAGRAM, ALTHOUGH THEY ARE PROVIDED.
4. HOT WATER SYSTEMS BEYOND HOT WATER STORAGE TANKS ARE NOT DIAGRAMMED.
5. IDENTIFICATION OF VALVE SYMBOLS IS SHOWN ON TVA DWG GM-4-308617-3.

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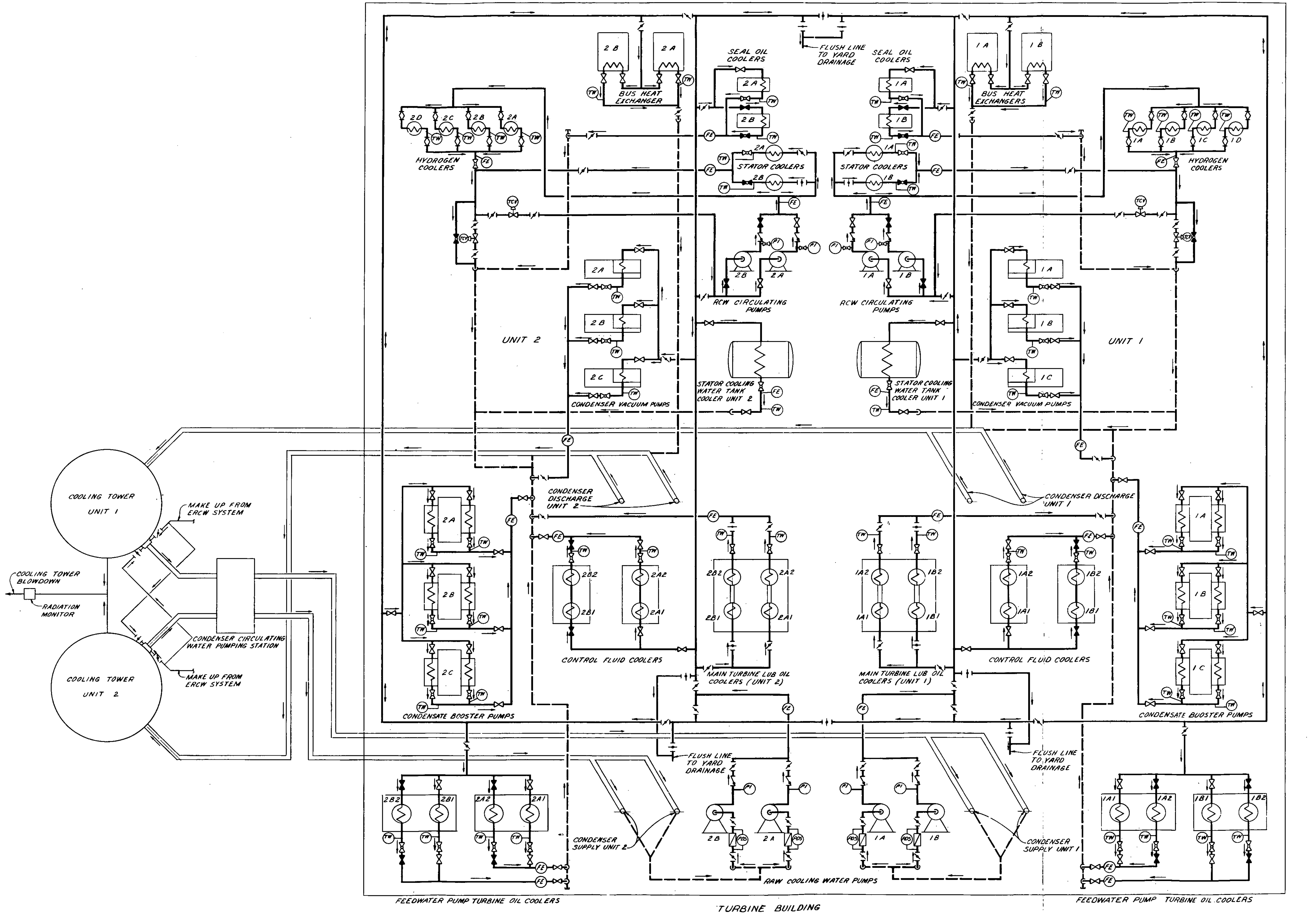
POTABLE WATER DISTRIBUTION SYSTEM - FLOW DIAGRAM

FIGURE 9.2-11



NOTES:
1. ALL VALVES ARE THE SAME SIZE AS PIPING UNLESS OTHERWISE SPECIFIED.
2. IDENTIFICATION OF VALVE SYMBOLS IS SHOWN ON TVA OWG GM-4-308617-3.
3. MAIN PROCESS SYSTEM VALVES ARE SHOWN IN THEIR NORMAL OPERATING POSITION.
4. UNIT 1 EQUIPMENT PREFIX NUMBER IS 1 AND UNIT 2 EQUIPMENT PREFIX IS 2. EQUIPMENT COMMON TO BOTH UNITS DOES NOT HAVE A PREFIX NUMBER.
5. ALL EQUIPMENT, PIPING, AND VALVES ARE NONNUCLEAR SAFETY CLASS.
6. DASHED LINES REPRESENT PIPING EMBEDDED IN CONCRETE.
7. SOLID LINES IN THE YARD AREA REPRESENT PIPING EMBEDDED IN THE GROUND. ALL OTHER SOLID LINES ARE EXPOSED PIPING.
8. ——⊃— — REPRESENTS EXPOSED PIPING WHICH GOES EMBEDDED INTO CONCRETE.
9. CONDENSER CIRCULATING WATER CONDUITS ARE REPRESENTED BY DOUBLE LINES.

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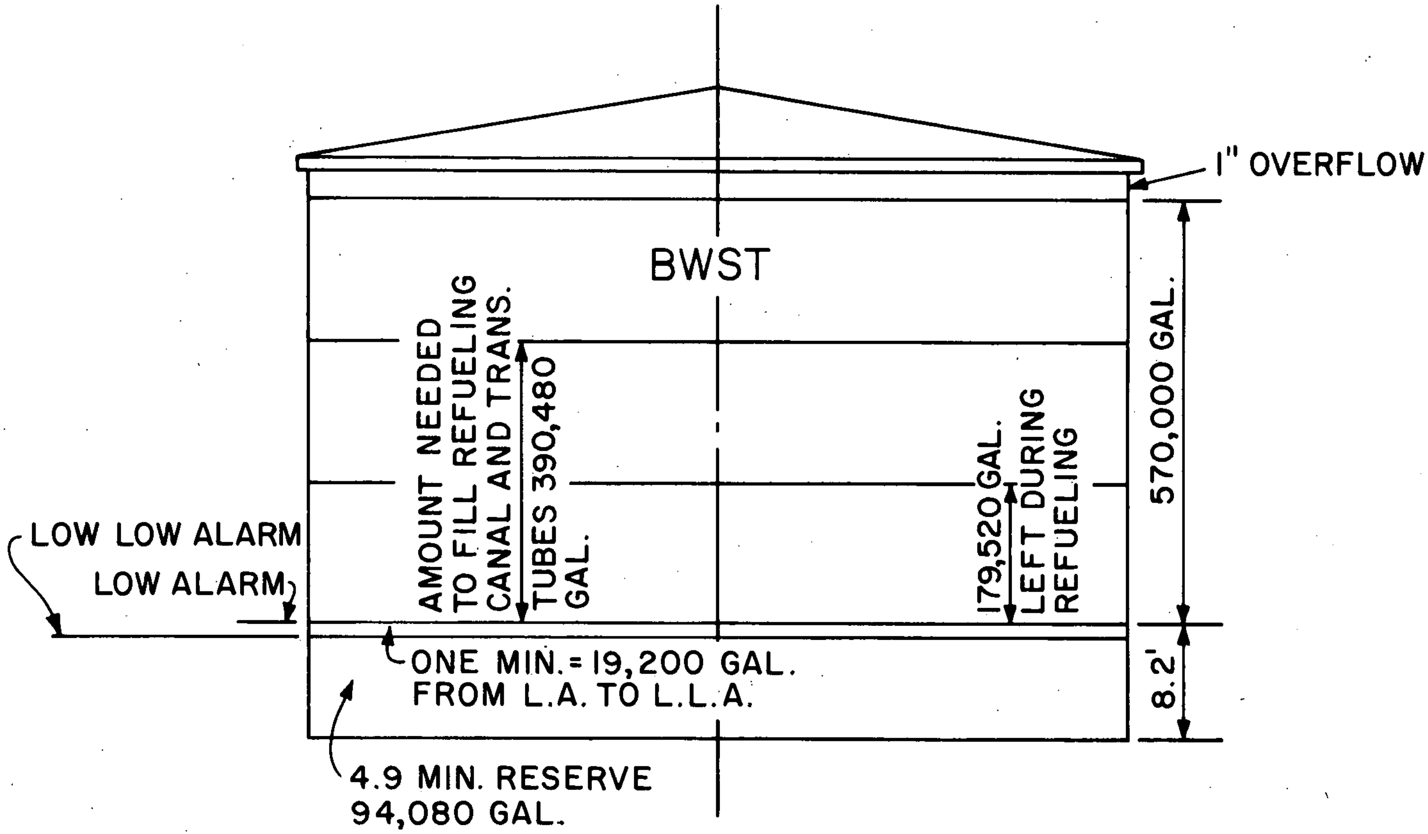
RAW COOLING WATER-
FLOW DIAGRAM

FIGURE 9.2-12

TVA DWG.NO. 47W844-1 R0

FIGURE 9.2-13



SPEC:

DIA. = 50 FT.
WALL HEIGHT = 48 FT.
NORMAL VOL. = 700,000 GAL.
USEABLE VOL. = 570,000 GAL.
MAX. FLOW RATE = 19,200 GPM
(THIS IS WITH 2DHR, 2RBS AND 2MU & HPI PUMPS AT RUNOUT)
TIME REQUIRED TO EMPTY TANK = 29.7 MIN. (APPROX.)
GAL./FT OF HEIGHT = 14,682 GAL./FT.

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BORATED WATER STORAGE TANK SECTION AND WATER CAPACITIES

FIGURE 9.2-13

REVISED BY AMEND 13, OCT. 1975

9.3. Process Auxiliaries

9.3.1. Compressed Air Systems

9.3.1.1. Design Bases

The compressed air system is capable of supplying normal and emergency plant operating requirements and is divided into three separate air systems.

1. Essential air system, see Figures 9.3-1 and 9.3-3.

2. Control air system, see Figures 9.3-1, 9.3-2, and 9.3-3.

3. Service air system, see Figures 9.3-1 and 9.3-2.

The major components of the compressed air system are described in Table 9.3-1.

The essential air system is designed to provide oil free control air, dried to a low dewpoint and free of foreign materials, to all essential pneumatically operated instruments, controls, and final operators.

The control air system is designed to provide oil free control air, dried to a low dewpoint and free of foreign materials, to all pneumatically operated instruments, controls, and final operators, such as control valves throughout the entire plant and yard. Wherever necessary, air for breathing apparatus can be supplied from control air or service air mains, both of which are oil free.

The service air system is designed to provide oil free air to hose connections throughout the plant, yard, and miscellaneous equipment. Since the air has passed through only an aftercooler, this is not considered to be a dry air system; however, moisture traps are provided to remove condensate.

The system is designed to provide air on a priority basis in the order listed above with provisions to assure that essential air is not lost because of a single failure in one of the two trains per unit.

9.3.1.2. System Design

9.3.1.2.1. General

Essential air, control air, and service air systems are supplied by four seismically qualified, motor-driven, nonlubricated, 2-stage, reciprocating compressors located in the auxiliary building at El 610.0. One compressor is located in each of four trains (Unit 1, Trains A and B; and Unit 2, Trains A and B). Each compressor is designed for continuous operation at 610 scfm at 100 psig. Air discharges from the aftercoolers at 110F maximum temperature. Each of the compressors is sized to handle the total plant essential and control air requirements under normal conditions with sufficient additional capacity to handle minimal service air requirements. Each compressor is equipped with an intercooler between stages and the final stage discharge is connected to a horizontal, 2-stage water-cooled aftercooler of shell-and-tube design with condensate separator, which in turn will be connected to an air receiver of 266-cubic-foot actual capacity.

Each receiver provides storage for essential air, control air, and service air. The essential and control air is filtered through one filter unit consisting of parallel banks of stone-type filters with a manual bypass around the filter unit for maintenance. Air from the stone filters discharges to a regeneration cycle will be determined from the consumption data and programmed to assure a dew-point of between -20 and -40F. A manual bypass is provided around the dryer for maintenance. The filter-dryer station handles both essential and control air.

The control air is supplied through a back pressure control valve which closes if the essential air supply to the particular train falls below 80 psig. This assures that essential air requirements take priority over control air requirements. Closure of the the back pressure valve is annunciated in the main control room. The actuating control must be manually reset before control air is restored.

The service air is supplied through a back pressure control valve which closes if the receiver pressure falls below 84 psig. This assures that both essential and control air take priority over service air requirements. Closure of the back pressure valve is annunciated in the main control room. The actuating control must be manually reset before service air is restored.

Motor-operated flow control valves are actuated to close on an engineered safety features actuation signal. These valves isolate the essential air header from the control air and service air supply headers and assure essential air for each individual train (see control diagram in Figure 9.3-3).

Control air from the filter-dryer station discharges into a large common header which runs the length of the auxiliary building. The header is sized large to provide additional storage capacity.

Control air is routed from the main header through smaller branch headers to various locations in the plant. Branch headers to the auxiliary building are provided with shutoff valves for isolation purposes and to facilitate adding future control air connections without requiring plant shutdown. Branch headers to the turbine building are provided with air-actuated isolation valves located inside the auxiliary builidng and remotely operated from the main control room.

The essential air system is seismically designed with train separation. Each essential air header will normally supply essential control air only to its respective train (i.e., train A essential air serves only train A components of safety systems). In case of loss of a compressor in a particular train, the air receiver is sized with sufficient volume to actuate air-operated containment isolation valves in that particular train. On the loss of offsite power the compressor will automatically restart on the auxiliary power supply. Control air from adjacent trains is available in event the compressor cannot be restarted.

The service air main headers are looped around each unit in the turbine, reactor, and auxiliary buildings and routed to other building in the powerhouse and yard. These headers also provide the system with more stored air capacity.

Moisture traps and air safety relief valves are provided where needed.

In order to get the cleanest possible air, the intake for the air compressors is ducted or piped from the air intake rooms located at El 686.0, for each trained area of the auxiliary building. Ambient air conditions for the air intake rooms normally are: 1

Dry bulb temperature, F	60-80
Relative humidity, %	20-90

The essential air system includes only that portion of the compressed air system that begins with the first check valve downstream of each compressor and extends to and includes the control valves in the compressed air system which 1 are actuated by an accident signal, and the appropriate accident signal actuated valves within the following systems:

Code No.	System[(a)]
1	Main, Reheat, and Auxiliary Steam System
2	Condensate, Demineralizer, and Distilled Water System
3	Main and Auxiliary Feedwater System
62	Makeup and Purification System
63	Core Flooding System
70	Component Cooling System
74	Decay Heat Removal System

[(a)]For code number and systems, see Appendix 7A, Figure 7A-2.

9.3.1.2.2. Component Design

All components and piping of the essential air system are ANS Safety Class 3 and Seismic Category I. The control air system and service air system are nonnuclear safety class; however, all pipe hangers for piping located in Category I structures are designed for seismic requirements.

9.3.1.3. Design Evaluation

During normal operating conditions the following safety features, environmental requirements, and redundancy are designed into the system:

1. The compressors are located at El 610.0 of the auxiliary building, one compressor in each of the trained areas. The air intake for each air compressor is in the intake rooms at El 686.0. Filters for the intake air are designed to remove all foreign materials.

2. Electric power to compressors is supplied from the emergency power system, one compressor per power train. Compressors automatically restart when emergency power is supplied from the diesel generators.

3. Safety relief valves on compressors, discharge headers, and receivers are set at 115 psig and designed to pass the specified cubic feet per minute air discharge to assure that the system cannot be overpressurized.

4. Aftercoolers are designed to ensure discharge air temperature at 110F maximum. Cooling water to compressors and aftercoolers is supplied from the essential raw cooling water system.

5. The compressed air system includes sufficient receiver capacity to supply essential air after a loss-of-power accident until restart of compressors by power supplied by diesel generators. This ensures continued operability 7
of the essential air system.

6. The service air system is automatically isolated when the pressure in the header supplying control and essential air drops to 84 psig. Also, the control air system is isolated when the pressure in the essential air header drops to 80 psig. This will conserve all the remaining air for the essential air system.

7. The control and service air systems have the necessary valving to isolate each train from the main header for maintenance or testing of components. Service air is provided to the reactor buildings for use only after reactor cold shutdown or in a refueling mode. Valves on either side of containment penetrations are locked closed at all other times.

8. Essential air system headers are routed on a train basis to all valves and instruments essential to plant safety.

9. Pipe hangers for the service air system piping and components located in Category I structures are designed for seismic requirements to assure safety of other essential equipment in the area.

10. Pipe hangers for the control air system piping and components located in Category I structures are designed for seismic requirements to assure safety of other essential equipment in the area.

11. All essential air piping and components essential to plant safety are designed to ANS Safety CLass 3 and Category I seismic requirements.

12. Availability of compressed air systems is guaranteed by automatic control of compressors. All four compressors operate on loading-unloading with staggered starting pressure set points for the compressors. If demand exceeds capacity of the first compressor, the receiver pressure drops to the starting point for the second compressor, etc. This arrangement provides four compressor units common to the control and service air headers while providing separation and isolation of essential air supply to each safeguard trained zone (1A, 1B, etc.).

13. Air actuated header isolation valves in the turbine building control and service air headers make it possible to maintain control and service air in the auxiliary and reactor buildings should a pipe rupture downstream of the valves.

9.3.1.4. Tests and Inspections

Preoperational tests are performed on the essential air system.

Testing of the essential air system is primarily limited to those portions which serve essential safety functions.

All system components are hydrostatically tested prior to station startup at 150 psig and are checked for leaks. All components are accessible for periodic inspections during operation. In normal operation, one or more of the compressors are idle, and may be operated in a no-load condition subject to immediate loading or left idle and tested in accordance with the technical specification.

Visual inspection of the system components, instrumentation, and trouble alarms is adequate to verify operability.

9.3.1.5. Instrumentation Applications

Pressure is the control variable in the operation of this system. Once pressure is built up, compressors only need to run intermittently. Their operation is controlled by pressure switches. Other pressure instruments provide control room indication, alarm actions, and computer-input information.

9.3.1.5.1. Indicators in The Main Control Room

Train A and train B air receiver pressures are indicated on respective unit main control room panels.

Control air header pressure indication on the common header is supplied to provide each unit operator with information on the air supply for nonessential instruments.

Service air header pressure indication on the common header is also supplied to each unit operator.

9.3.1.5.2. Alarms in The Main Control Room

The following parameters are alarmed to alert the unit operators of impending trouble or abnormal operation:

1. Air compressor discharge high temperature.

2. Air receiver low pressure.

3. Air dryer discharge high dewpoint.

4. Service air supply valve closure below 84 psig.

5. Control air supply valve closure below 80 psig.

6. Control air to auxiliary building low pressure.

7. Control air to turbine building low pressure.

8. Service air header low pressure.

9.3.1.5.3. Computer

The following parameters are monitored by each unit's computer:

1. Train A air receiver pressure.

2. Train B air receiver pressure.

3. Control air header pressure.

4. Service air header pressure.

The computer records data on a routine basis, records the time of any abnormal condition, alarms on low limits, and displays data or alarms on cathode ray tubes or computer printout.

9.3.1.5.4. Local Instrumentation

Pressure switches which monitor compressor discharge pressure are supplied for automatic control of compressor operation.

Local indicators are provided for aftercooler discharge temperature and for receiver pressure.

9.3.2. Process Sampling System

9.3.2.1. Design Bases

Remote sampling is provided to facilitate laboratory analysis of specimens from all critical locations in the primary, secondary, and auxiliary systems. The samples are analyzed for total solids, boron, lithium 7, pH, conductivity, dissolved oxygen, dissolved hydrogen, hydrazine, chlorides, fluorides, silica, iron, copper, etc. (Reference: Table 5.2-6, Water Chemistry Specifications.) Those samples associated directly with unit operations are obtained via remote facilities. All others are provided with local grab sample capability. The system is designed to provide samples needed for post-LOCA or flood-mode operation as well as normal operation.

The remote lines are routed to a hot sampling room and, where necessary, decay coils are provided for the decay of the short-lived, high-energy isotopes.

The piping and equipment in the sampling system is designed to the following codes, latest issue:

1. Pressurizer cooler and core flooding tank cooler — ASME Code, Section III, Class C.

2. Steam generator cooler — ASME Code, Section VIII.

3. Piping — ANSI B31.7, Code Class III for Nuclear Piping, including nuclear code cases where applicable.

4. Valves — ASME Code, Section III, Class 2.

9.3.2.2. System Description

Figure 9.3-4 gives sampling locations for various liquid samples and one gas sample. A selected group of these are taken to a remote, hot sampling room located in the auxiliary building. The remainder is taken at local grab stations located throughout the auxiliary building. A boron analyzer with recorder is provided for continuous analysis of reactor letdown fluid. Gas sampling, with the exception of the makeup tank gas space, is performed by a gas analyzer which is associated primarily with the waste disposal system.

Various analysis-type instruments are located in the radiochemical laboratory and counting room located in the auxiliary building.

The balance of plant sampling is brought to a common location (titration room) and is not described herein because it is not safety related.

For a description of sampling for the primary purpose of process and effluent radiological monitoring, see subsection 11.4.3.

9.3.2.2.1. Liquid Samples, Remote

Provisions are made to obtain liquid samples from a remote hot sampling room for each unit. These samples are listed below.

Liquid Samples, Remote

	Sample location	Approx sample size (Mℓ)	Frequency of sampling
1A.	Pressurizer water space (1)(2)	1000	Weekly
1B.	Pressurizer steam space (1)(2)	1000	Weekly
2.	Letdown line upstream of breakdown orifices	1000	Weekly
3.	Steam generator A secondary side (1)	1000	Daily
4.	Steam generator B secondary side (1)	1000	Daily
5.	Core flooding tanks	100	Weekly
6.	Letdown line upstream of prefilter (2)(3)	1000(4)	(4)
7.	Letdown line downstream of prefilter	1000	Weekly
8.	Letdown line downstream of demineralizer	1000	Weekly
9.	Makeup tank liquid space	100	Optional
10.	Makeup downstream of deborating demineralizer	100	During operation
11.	Decay heat cooler discharge	1000	Daily during system operation

Sample location	Approx sample size (Mℓ)	Frequency of sampling
12. Control rod drive cooling pump suction	1000	Weekly
13. Component cooling pumps discharge	1000	Weekly

Notes:

(1) Provided with cooling coils to reduce temperatures sufficiently to permit safe handling and minimize generation of radioactive aerosols.

(2) Provided with sample containers.

(3) Provided with decay coil.

(4) See table below for further information on reactor coolant system analysis

Reactor Coolant System Analysis

Analysis required	Sample point	Sample size, Mℓ	Frequency of sampling
Filterable crud	Letdown line	1000	Weekly
Boric acid	Letdown line	50	Daily
Lithium	Letdown line	25	Weekly
pH	Letdown line	25 hundred	1-2 days
Dissolved oxygen	Letdown line	1000	Optional
Chlorides	Letdown line	25	1-2 days
Fluorides	Letdown line	25 procedure	102 days
Total dissolved gas	Letdown line	100	Weekly
Hydrazine	Letdown line	25	Daily
Sodium	Letdown line	25	Optional
Tritium	Letdown line	1000	Weekly
Hydrogen	Downstream of demineralizer	1000	1-2 days

9.3.2.2.2. Liquid Samples, Grab

Provisions are made to obtain local grab samples of the following liquids (all samples are common to both units except 40, 41, and 42). These samples are listed below.

Liquid Samples, Grab

	Sample location	Approx sample size, Mℓ	Frequency of sampling
1.	LiOH mix tank	100	After mixing
2.	Boric acid mix tank	100	After mixing
3.	RC bleed transfer pump 1A1 and 1A2 recirculation line	500	After filling or before pumping out RC bleed hold-up tanks
4.	RC bleed transfer pump 2A1 and 2A2 recirculation line	500	After filling or before pumping out RC bleed holdup tanks
5.	RC distillate transfer pump 1A1 and 1A2 recirculation line	500	After filling or before pumping out RC distillate storage tanks
6.	RC distillate transfer pump 2A1 and 2A2 recirculation line	500	After filling or before pumping out RC distillate storage tanks
7.	Concentrated boric acid, tank 1A	100	Anytime in question, usually weekly
8.	Concentrated boric acid, tank 2A	100	Anytime in question, usually weekly
9.	Evaporator distillate test tank 1A	1000	Before pumping out
10.	Evaporator distillate test tank 2A	1000	Before pumping out
11.	Downstream of evaporator distillate demineralizer 1A	1000	During use
12.	Downstream of evaporator distillate demineralizer 2A	1000	During use
13.	Downstream of deborating demineralizer strainers	1000	During use
14.	RC bleed evaporator 1A concentrate	1000	As required for evaporator operation
15.	RC bleed evaporator 2A concentrate	1000	As required for evaporator operation
16.	RC bleed evaporator 1A distillate	1000	As required for evaporator operation
17.	RC bleed evaporator 2A distillate	1000	As required for evaporator operation
18.	RC bleed evaporator demineralizer A discharge	1000	While filling RC bleed evaporator
19.	RC bleed evaporator demineralizer B discharge	1000	While filling RC bleed evaporator

Sample location	Approx sample size, Mℓ	Frequency of sampling
20. Upstream of spent fuel coolant demineralizer	500	Only to determine demineralizer efficiency
21. Downstream of spent fuel coolant demineralizer	500	Daily during refueling
22. Downstream of spent fuel coolant filter	1000	Daily during refueling
23. Borated water recirculation pump discharge	500	During purification or pumping of BWST, fuel transfer canal, or spent-fuel storage pool
24. Spent fuel cooling pump discharge	1000	Weekly
25. Nontritiated waste holdup tank	1000	After addition or before pumping out
26. RC drain tank	500	During system operation
27. Nontritiated waste holdup tank	1000	Before addition or after pumping out
28. Waste evaporator distillate test tank A	1000	Before pumping out
29. Tritiated waste evaporator feed tank	1000	Before pumping out
30. Tritiated waste evaporator concentrate	1000	As required by evaporator operation
31. Tritiated waste evaporator distillate	1000	As required by evaporator operation
32. Nontritiated auxiliary building sump tank	500	Before pumping out
33. Laundry and hot shower drain tank	500	Before pumping out
34. Spent resin liquid sluicing pump discharge	500	As required by sluicing operations
35. Spent resin transfer pump discharge	500	As required during drumming operations
36. Tritiated waste holdup tank	1000	After addition or before pumping out
37. Waste evaporator distillate test tank B	1000	Before pumping out
38. Tritiated auxiliary building sump tank	500	Before pumping out
39. Chemical drain tank	1000	During use
40. Borated water storage tank	1000	Weekly and after each makeup

	Sample location	Approx sample size, Mℓ	Frequency of sampling	
41.	Sodium hydroxide storage tank	1000	Weekly and after each makeup	
42.	Downstream of auxiliary waste evaporator distillate test tank demineralizer[a]	1000	As required by evaporator operation	
43.	Auxiliary waste evaporator distillate test tank 1 recirculation line	1000	As required by evaporator operation	
44.	Auxiliary waste evaporator distillate tank 2 recirculation line	1000	As required by evaporator operation	11
45.	Auxiliary waste evaporator concentrates[a]	1000	As required by evaporator operation	
46.	Auxiliary waste evaporator distillate[a]	1000	As required by evaporator operation	
47.	Auxiliary waste evaporator feed[a]	1000	As required by evaporator operator	

[a] Refer to Chapter 11, Figure 11.2-3.

9.3.2.2.3. Gaseous Samples

Provisions are made to obtain sample Sr-14 from the makeup tank gas space for each unit. This sample is collected in a sample container which is located in the same sample hooded area provided for liquid samples.

Provisions are made to deliver the following gaseous samples to the hydrogen-oxygen analyzing station (or to a sample container) common to both units (see waste disposal system description, subsection 3.2.5. and paragraph 11.3.2, System Description of Gaseous Waste System).

Provisions are made to deliver gaseous samples from the inlets to the two hydrogen recombiners inside the reactor building to two hydrogen-oxygen analysis stations in the auxiliary building. Capability is provided in the main control room to monitor the results of these analyses.

Gaseous Samples, Remote

Sample location
1. Nontritiated vent header
2. Tritiated vent header
3. Nontritiated auxiliary building sump tank
4. Tritiated auxiliary building sump tank
5. Waste gas decay tank A
6. Waste gas decay tank B
7. Waste evaporator system, nontritiated
8. Waste evaporator system, tritiated
9. RC bleed holdup tank 1A
10. RC bleed holdup tank 2A
11. RC distillate storage tanks 1A1 and 1A2
12. RC distillate storage tanks 2A1 and 2A2
13. RC drain tank
14. Tritiated waste holdup tank
15. Nontritiated waste holdup tank
16. RC bleed evaporator 1A
17. RC bleed evaporator 2A

9.3.2.3. Safety Evaluation

During a loss-of-coolant accident, the pressurizer sample lines originating inside the containment are automatically isolated at the containment boundary.

Remote manual operation of the isolation valves in the sampling lines from the steam generators and from the core flooding tanks can be accomplished from either the main control room or the hot sampling room.

Adequate safety features are provided to protect laboratory personnel and prevent the spread of contamination from the sampling room.

Sampling system discharges are designed to limit flows under normal operation and anticipated malfunctions or failures to preclude any fission product release leading to exposures exceeding 10 CFR 20 limits. All remote samples are provided with recirculation lines to facilitate purging without excessive waste of water or excessive sink contamination.

Sinks are provided with a supply of demineralized water for washdown. All sampling hood vents are filtered using HEPA-type filters and are exhausted to the station vent. The sample sinks drain to the nontritiated waste holdup tank. Shielding is provided where required, and portable radiation monitoring equipment is provided in the vicinity of the sink and hood enclosures. All local grab samples are provided with adequate facilities to obtain representative samples without undue hazard to personnel. Local samples are grouped at convenient locations in the auxiliary building and each local sampling group has a hood connected to a ventilating system through a HEPA filter and a drain connected to the nontritiated waste holdup tank.

Since the sampling system does not perform any safety function, reactor operation need not be terminated when any part malfunctions; however, the operating requirements for water quality as listed in Table 5.2-6 will be observed.

9.3.2.4. Test and Inspections

All components of the sampling system will be tested for operability in the startup test program. Since the sampling system will be used in normal operation, no formal test program will be required beyond routine surveillance during operation. The two hydrogen-oxygen analysis stations for reactor building containment gaseous samples are not needed for normal operation but will be tested periodically to assure operability when they are required.

9.3.2.5. Instrumentation

In the hot sampling room the remote sampling lines have local indicators for pressure, temperature, and flow to aid the operators in taking samples.

Analysis instrumentation will be used to monitor selected liquid samples in the remote sampling room for conductivity, pH, and hydrazine.

The following analysis instrumentation is located in either the radiochemical laboratory or the counting room:

1. Multichannel pulse height analyzer system.

2. Liquid scintillation counter (for tritium analysis).

3. Low background proportional counting system (for alpha or beta).

4. Well counter scintillation system (for gross gamma).

5. Gas proportional system (for alpha or beta).

6. Gas chromotograph for automatic analysis of hydrogen and oxygen in gas samples.

9.3.3. Equipment and Floor Drainage System

9.3.3.1. Design Bases

The equipment and floor drainage system is designed primarily to collect equipment and floor drainage in such a manner that the segregation and safe disposal of radioactive and nonradioactive effluents will be assured during the various modes of operation of the plant. This will be accomplished by providing:

1. Separate drain collection headers for tritiated and nontritiated radioactive drains.

2. Separate open drain headers from each ESF zone in the auxiliary building to provide zonal separation.

3. A means of leak detection and emergency drainage in the equipment rooms of engineered safety features equipment.

4. Separate open drain headers for each section of the diesel generator buildings.

5. Equipment to segregate condensate demineralizer backwash and area floor drains from other turbine building drainage.

6. Piping to the radioactive waste disposal system for all reactor building equipment and floor drainage.

Secondary functions of the system will be to dispose of all roof drainage and to provide means of collecting and disposing of oil drainage.

9.3.3.2. System Description

9.3.3.2.1. General

The equipment and floor drainage system serves the diesel generator, turbine, service and office, control, reactor, and auxiliary buildings. The following type drains are used in the design of these buildings:

1. Roof drains (see Figure 9.3-5).

2. Floor drains (see Figures 9.3-5 and 9.3-6).

3. Open equipment drains (see Figures 9.3-5 and 9.3-6).

4. Oil drains (see Figures 9.3-5 and 9.3-6).

5. Closed equipment drains.

6. Leak detector drains (see Figures 9.3-5 and 9.3-6).

7. Emergency drains (see Figure 9.3-6).

Roof drains for all buildings are collected and discharged into the yard drainage system where they are routed to the holding pond and subsequently discharged into the river.

All drains routed to and collected in the turbine building sumps (oil sump and station sump) are of the following categories:

1. Potentially oily waste water.

2. Oil-free water but containing minute quantities of the following:

   a. Radioactivity

   b. Chemicals such as used in secondary feedwater treatment

   c. Miscellaneous crud from washing floors

Water drained as (1) is routed to the oil sump (adjacent to the station sump) where oil is allowed sufficient time to separate from the water. Afterwards, water which is collected at the bottom of the oil sump, is allowed to drain to the station sump. Subsequently, oil is accumulated until a sufficient amount is collected to be pumped into oil drums for disposal.

Water drained as (2) is routed directly to the station sump until a level alarm signals that the sump is ready to be pumped. At this time a sample is taken for analysis before pumping. The contents of the station sump is then pumped to the yard drainage pond provided the crud, dissolved solids, and radioactivity levels are below acceptable plant discharge levels. If the levels are unacceptable, provisions will be made to process this batch of liquid.

9.3.3.2.2. Diesel Generator Buildings

Each diesel generator building (one per plant unit) is divided into three areas — General, Train A, and Train B. Drains from the first floor of each area are collected and routed separately to the diesel generator building sump. Three-way valves allow the drainage to normally flow into a common header which is routed to the turbine building oil sump. However, if a probable maximum flood occurs, the 3-way valves are manipulated to shut off any possible backflow from the turbine building and divert the drainage into the diesel generator building sump where it is pumped to the turbine building station sump. Also, first floor emergency drains from the train A and train B areas to the diesel generator building sump are provided to handle water resulting from an essential raw cooling water line break. All second floor drains are collected on a diesel unit basis and routed directly to the turbine building oil sump or to the diesel generator building sump during probable maximum flood conditions.

9.3.3.2.3. Turbine Building

Turbine building drains fall into five categories — potentially radioactive floor drains, nonradioactive floor drains, nonradioactive open equipment drains, chemical drains, and oil drains.

Floor drains located in the condensate demineralizer area on floor El 609.0 are potentially radioactive (see Figure 9.3-5). These drains are collected in the demineralizer area sump tank (see Figure 10.4-5) and pumped to the nontritiated waste holdup tank in the auxiliary building (see Figure 9.3-6) and processed in the radwaste system.

Chemical drains located in the condensate demineralizer area (see Figure 10.4-5) are collected in the waste accumulator, neutralized, sent to the demineralizer area sump tank, and then pumped to the nontritiated waste holdup tank for processing in the radwaste system.

Open equipment drains and nonradioactive floor drains are collected and routed to the station sump which is located at the lowest level of the turbine building. The station sump is equipped with two vertical turbine-type pumps which transfer the drainage into the yard drainage system.

Oil drains are located in curbed areas where oil spillage is likely and routed to the oil sump. The oil sump is also located at the lowest level of the turbine building and is connected to the station sump by two lines. One line is near the top to allow passage of oil from the station sump to the oil sump and the other line is near the bottom to allow passage of water from the oil sump to the station sump. Each of these lines has a check valve and isolation valve to prevent contamination of water with oil and vice versa. The oil sump has a vertical turbine-type pump which pumps oil through a flexible connection into a 55-gallon drum when a sufficient amount of oil has been collected.

9.3.3.2.4. Service Building

The service building has floor drains and open equipment drains in the hot shower and laundry area. These drains are considered radioactive drains and routed to the hot shower and laundry drain tank in the auxiliary building.

Other service building drains are nonradioactive floor drains and oil drains. The floor drains are collected and transferred to the station sump in the turbine building. The oil drains, which are located in the machine shop and auxiliary boiler areas, are routed to the turbine building oil sump.

9.3.3.2.5. Control Building

All control building drains are nonradioactive. These drains are collected and transferred to the turbine building station sump. To prevent backflow from the station sump into areas located below probable maximum flood level, check valves and shutoff valves are provided in the lines that drain these areas.

9.3.3.2.6. Reactor Building

Reactor building drains consist of radioactive floor drains and radioactive closed equipment drains. The floor drains are collected in the reactor building normal sump and then normally transferred to the nontritiated waste holdup tank by the reactor building sump pumps. However, connections are provided in the discharge of the reactor building sump pumps to route the contents of the reactor building normal sump to the tritiated waste holdup tank. This is done only when the tritium concentration of the drainage becomes excessive.

The closed equipment drains are collected in either the reactor building normal sump (normally nontritiated) or the reactor coolant drain tank (normally tritiated). These drains are shown on liquid waste disposal system, Figure 11.2-1.

9.3.3.2.7. Auxiliary Building

9.3.3.2.7.1. General

The auxiliary building drainage system consists mainly of oil drains and the following radioactive drains:

1. Floor drains

2. Open equipment drains

3. Closed equipment drains

4. Leak detector drains

5. Emergency drains

The floor drains, open equipment drains, and closed equipment drains are divided into two classes — tritiated and nontritiated. (See liquid waste disposal system, Figure 11.2-2 for closed equipment drains and Figures 9.3-5 and 9.3-6 for floor drains and open equipment drains.) The leak detector drains and emergency drains are considered tritiated. As has already been stated, zonal separation of all open drains is provided to prevent backflow and cross ventilation between the ESF zones in case of an accident.

9.3.3.2.7.2. Tritiated Drains

The tritiated drains normally carry water with a tritium content of 10% or greater than the tritium content of the reactor coolant. These drains, with the exception of the emergency drains, are collected and sent directly to either the tritiated waste holdup tank (for drains above El 590.0) or the tritiated auxiliary building sump tank (for drains at El 590.0 or below). The emergency drains are collected in the passive sump and then drained into the tritiated auxiliary building sump tank. After reaching either of the tritiated tanks, the drainage is then transferred to the tritiated radwaste processing equipment.

9.3.3.2.7.3. Nontritiated Drains

Nontritiated drains carry water with a tritium content of less than 10% of the tritium content of the reactor coolant. Most of these drains are collected and routed directly to either the nontritiated waste holdup tank or the nontritiated auxiliary building sump tank. However, some special cases exist. Among these are the drains from the radiochemical laboratory, the cask washdown area, the decontamination area, and the titration room.

The radiochemical laboratory, in addition to the normal tritiated and nontritiated drains, has a chemical drain which is routed to the chemical drain tank. The contents of this tank are directed to the plant discharge line, drummed, or sent to the nontritiated radwaste processing equipment, depending on the tritium level and the chemical contents.

Cask washdown area drainage is routed directly to the cask washdown drain tank where it is sampled and tested. It is then discharged into the plant discharge line if the radioactivity level is low enough. This line is monitored on a continuous basis. If the level of radioactivity is too high, then the contents of the cask washdown drain tank are directed to the nontritiated radwaste processing equipment.

The decontamination area has three drains — a nontritiated floor drain, a chemical drain, and a laundry drain. The drains are capped and only one of them is used at any given time. When contaminated equipment is brought to this area, the nontritiated drain is used if only water is required to remove the contamination. If acid is required, the chemical drain is used, and the drainage is sent to the chemical drain tank. Finally, if soap and water are necessary to remove the contamination, the laundry drain is used and the drainage is directed to the hot shower and laundry drain tank. Like the chemical drain tank, the laundry and hot shower drain tank contents are put in the plant discharge line, drummed, or routed to the nontritiated radwaste processing equipment.

The titration room has sample lines from the turbine building which continuously supply condensate samples. These samples are rerouted back to the hotwell or to the turbine building station sump for disposal. However, floor drains are provided to collect any spillage or washdown water in the titration room. These drains are routed to the nontritiated waste holdup tank.

9.3.3.2.7.4. Oil Drains

Where there is a possibility of considerable oil leakage (transformers, etc.), curbed areas equipped with floor drains are provided. Drainage flows by gravity through valved lines and is collected in 55-gallon drums for further disposal.

9.3.3.3. Safety Evaluation

The equipment and floor drainage system is not an engineered safety features system. Therefore, most of the piping and components are non-nuclear safety class. However, the system is designed to prevent the spread of a nuclear accident and the subsequent release of radioactive contamination to the environment. Specific examples of the safety design features are as follows:

1. In the lines which drain the reactor building sump tanks to either the tritiated or nontritiated waste holdup tank, automatic isolation valves are provided on the inside and outside of the reactor building containment. These valves close on the ESF signal and prevent the release of radioactive contamination from the reactor building to the auxiliary building in case of an accident. The containment isolation valves and penetration piping are ANS Safety Class 2.

2. In areas where ESF equipment (decay heat coolers, decay heat pumps, reactor building spray pumps, high-pressure injection pumps, and diesel generators) is located, leak detectors and emergency drains are provided. The leak detectors inform plant personnel of major leaks in piping and equipment, and the emergency drains provide a way to dispose of the leakage. This ensures the operability of adjacent equipment in the event of an accident.

3. If an essential raw cooling water line break occurs in any generator area of the diesel generator buildings, emergency drains provide a means for rapid water disposal to the sump and prevent the loss of a diesel unit. Also, normal first floor drains are routed to the turbine building oil sump to reduce the possibility of oil fires in the diesel generator buildings.

4. The auxiliary building has zonal separation (train 1A area, train 2A area, radiochemical laboratory, titration room, etc.). In order to prevent an accident from spreading from one zone to another, zonal separation of the open drains is provided. This is accomplished by routing the collection header from each zone to either the sump tanks or holdup tanks and then providing a liquid seal.

5. A sump tank in the condensate demineralizer area is provided to collect demineralizer wastes and area floor drains. The contents of this sump tank may be radioactive; therefore, they are pumped to the nontritiated waste holdup tank in the auxiliary building.

6. The floor drains in the annulus between the primary and secondary containments of each reactor building are routed to a small sump located in the annulus. The sump drains to the nontritiated waste holdup tank in the auxiliary building. A water seal is provided at the holdup tank so that a negative pressure can be maintained in the annulus. A leak detector located in the sump alarms in the control room to notify plant personnel of any pipe rupture or leak.

7. Exposed floor drains, equipment drains, and roof drains in the reactor, auxiliary, diesel generator, and control buildings are seismically qualified. This requirement is to prevent the drains from falling on essential piping and equipment during seismic events.

9.3.3.4. Tests and Inspections

All drains are cleaned free of rust, dirt, scale, and welding scrap and blown out with compressed air before final closure. Also, the auxiliary building and reactor building drains are leak tested.

9.3.3.5. Instrumentaion Applications

The equipment and floor drainage system includes the following instrumentation:

1. Leak detectors for auxiliary building ESF equipment.

2. Level controls for the drainage sumps and tanks.

3. Pressure measurement in the discharge of all system pumps.

4. Leak detectors for the first floor of each diesel generator unit.

5. Leak detectors for the reactor building secondary containment.

The auxiliary building ESF equipment rooms' leak detectors consist of sump-type floor drains and level indicators. When the water builds up to a predetermined level in the floor drain, a level indicator will annunciate in the control room. The drains allow normal leakage to drain to the tritiated auxiliary building sump tank or the tritiated waste holdup tank. However, in the event of major line ruptures or equipment leaks, the drains will not handle the leakage, thus allowing a buildup of water in the room. When the water builds up to a predetermined level in the room, the emergency drain dropout panel ruptures and allows the water to drain to the emergency sump.

The level controls for the drainage sumps and tanks consist of low-level and high-level switches. The low-level switches ensure a minimum level to cover the suction of the pumps and the high-level switches alarm to warn of overflowing.

Pressure gauges are provided for pressure measurement and are read locally with the exception of the reactor vessel cavity sump pumps and the reactor building pit unwatering pump which are inaccessible since they are located in the reactor buildings.

Each diesel generator building leak detector (one per diesel generator unit is provided) consists of a small sump with a level switch. If a major leak occurs, such as an essential raw cooling water line break, the water level will build up until it overflows into the sump. As the sump level increases, the level switch annunciates in the control room.

Each reactor building secondary containment annulus is provided with a leak detector sump (one per unit) to detect the effects of a pipe break or leak and annunciate in the control room.

9.3.4. Makeup and Purification System

9.3.4.1. Design Bases

9.3.4.1.1. The makeup and purification system is designed to accommodate the following functions during normal reactor operations:

1. Provide preoperational fill and operational makeup to the reactor coolant system.

2. Provides injection water to the reactor coolant pump seals.

3. Provides the capability of removing corrosion and fission products from the reactor coolant system during purification operations.

4. Controls the boron concentration of the reactor coolant by the addition of boric acid.

5. In conjunction with pressurizer, the system will accommodate temporary changes in reactor coolant volume due to small temperature changes.

6. Maintain proper concentration of hydrogen and corrosion inhibiting chemicals in the reactor coolant.

7. Supply borated water to the core flooding tanks as makeup.

8. Provides makeup to the reactor coolant system for protection against small breaks in the reactor coolant pressure boundary.

9.3.4.1.2. The makeup and purification system is designed to control and maintain the reactor coolant inventory and also to control the boron concentration of the reactor coolant system through the processes of makeup and letdown of the reactor coolant.

9.3.4.1.3. Purification of the letdown fluid by removal of corrosion or fission products is also accomplished by the makeup and purification system. Two letdown coolers are provided to remove heat from the reactor coolant prior to entering the purification demineralizers and filters. The normal letdown flow rate is 50 gpm which permits recirculation of approximately one reactor coolant system volume through the purification train during a 24 hour period. The maximum letdown flow rate is set at 200 gpm. This flow rate permits changing boron concentration by bleeding coolant from the reactor coolant system during xenon peaking following a 50% power change. During this period, unborated reactor grade water is added to the reactor coolant system to dilute the boric acid concentration in the reactor coolant system; this is done to compensate for the negative reactivity addition resulting from the xenon peaking.

9.3.4.1.4. The makeup portion of the makeup and purification system is designed to provide normal makeup to the reactor coolant system via seal inleakage through the reactor coolant pump seals and via the normal makeup line. Makeup required with a small break in the reactor coolant pressure boundary is directed to the reactor coolant system via the normal makeup line.

9.3.4.1.5. Each of the letdown coolers, purification demineralizers, and purification filters are sized for one half the maximum letdown flow rate. The purification demineralizer prefilter is sized for the maximum letdown flow rate.

9.3.4.1.6. The letdown and makeup process also accommodates thermal expansion and contraction of the reactor coolant system during startup and shutdown transients.

9.3.4.1.7. The makeup and purification system also functions to provide emergency high pressure injection following a loss of coolant accident (LOCA). This function is explained in Chapter 6.

9.3.4.2. System Description and Evaluation

9.3.4.2.1. General

The makeup and purification system is shown schematically on the process and instrumentation drawing, Figure 9.3-7. Tables 9.3-2 and 9.3-3 list the system performance and individual component design characteristics.

1. Letdown Cooler — The letdown cooler reduces the temperature of the letdown flow from the temperature of the reactor coolant system to a temperature suitable for demineralization and injection to the reactor coolant pump seals. Heat from the letdown coolers is rejected to the component cooling water system.

2. Letdown Flow Control — The normal letdown flow rate at reactor operating pressures is controlled by a fixed block orifice. A parallel, normally closed, remotely operated valve can be opened to obtain flow rates up to the maximum letdown capability. This valve is also used to maintain the desired letdown rate at reduced reactor coolant pressures. In addition, there is a second parallel, normally closed valve which may be manually positioned for flow control.

3. Letdown Flow Radiation Monitoring — A fission product detector is placed in the letdown line to monitor the activity of the letdown fluid. It is located upstream of any purification equipment so that reactor coolant fission product activity can be accurately indicated.

4. Boron Meter — This device measures the boron concentration of reactor coolant.

5. Purification Demineralizer Prefilter — The purification demineralizer prefilter is designed to remove particulate matter from the letdown stream prior to entering the purification demineralizers. The filter prohibits any accumulation of radioactive crud in the demineralizer resin and in the downstream piping of the purification system and the waste disposal system.

6. Purification Demineralizers — The mixed-bed demineralizer can process one reactor coolant volume in 24 hours at the normal letdown rate. Since the reactor coolant may be contaminated with fission and corrosion products, the resins will remove certain radioactive impurities. (All given in Chapter 11).

7. Purification Filters — Two purification filters are installed in parallel to remove particulates from the effluent of the purification demineralizers. This will prevent solids from entering the makeup tank.

8. Makeup Pumps — The makeup pumps are designed to return the purified letdown fluid to the reactor coolant system and to supply seal water to the reactor coolant pumps. One pump is normally in service.

9. Reactor Coolant Pump Seal Return Coolers — The seal return coolers are 11
sized for one makeup pump recirculation flow plus reactor coolant pump seal leakoff flow. Heat from these coolers is rejected to the component cooling system. Two coolers are provided with one normally in operation.

10. Makeup Tank — The makeup tank serves as a receiver for letdown, seal return, chemical addition and system makeup and provides NPSH for the makeup pump. The tank also accommodates temporary changes in system coolant volume.

11. Seal Injection Filter — Filters are provided in the reactor coolant pump seal injection line to remove particulates which could enter the pump seals and result in increased seal wear.

9.3.4.2.2. Mode of Operation

The makeup and purification system is operated during all phases of the nuclear steam system (NSS) operating life, including startup, power operation, and shutdown. The system will also be operated during refueling by employing the purification equipment through interconnections to the decay heat removal system. During normal NSS operation, one makeup pump continuously supplies high pressure water from the makeup tank to the seals of each of the reactor coolant pumps and to the reactor coolant system through the makeup line. Makeup flow to the reactor coolant system is regulated by the makeup control valve which operates on signals sensing pressurizer level.

Reactor coolant pump seal injection flow is automatically controlled to the desired rate. A portion of the water supplied to the pump seals leaks off as controlled bleedoff and returns to the makeup tank after passing through the seal return cooler. The remainder of the water leaks into the reactor coolant system as reactor coolant system makeup.

Seal water inleakage to the reactor coolant system makes necessary a continuous letdown flow of reactor coolant to maintain the desired coolant inventory. Letdown flow is also required for removal of impurities and boric acid from the reactor coolant. The letdown flow is cooled by both of the letdown coolers, 11
reduced in pressure by the letdown orifice, and then passed through a prefilter and a purification demineralizer, to a three-way valve which directs the coolant to the makeup tank or to boron recovery system.

Normally, the three-way valve is positioned to direct the letdown flow to the makeup tank. If the boric acid concentration in the reactor coolant is to be reduced, the three-way valve is positioned to divert the letdown flow to the chemical addition and boron recovery system, where the boric acid is removed, or to the waste disposal system. Demineralized water is then added to the reactor coolant via a flow integrator to the makeup tank. During normal operation, the flow integrator, the control rod drive interlock, or the operator will terminate dilution. Through the use of boric acid control, the plant operator may accommodate xenon transients resulting from plant load following.

The makeup tank also receives chemicals for addition to the reactor coolant. A hydrogen overpressure is maintained in the tank to insure that a predetermined amount of dissolved hydrogen remains in the reactor coolant. Chemicals in solution are injected into the makeup tank, which serves as a final mixing location.

The MU system in Figure 9.3-7 shows three connections from the decay heat removal system; one connects just downstream of the letdown orifice, and the other two tie directly into the line which interconnects the MU pump inlets.

System control is accomplished remotely from the control room with the exception of periodic manual switching of the seal return coolers. The letdown flow rate is set for flow rates other than normal by remotely positioning the letdown flow control valve to pass the desired flow rate. The spare purification demineralizer can be place in service by remote positioning of the demineralizer isolation valves. A third demineralizer is provided to remove cesium in the reactor coolant prior to refueling.

Emergency operation of the makeup system is described in Chapter 6.

9.3.4.2.3. Normal System Function

1. Initial Fill — Prior to heatup and pressurization of the reactor coolant system, the reactor coolant system is filled with coolant at the refueling boron concentration by using the reactor coolant bleed transfer pumps or with the reactor coolant distillate transfer pumps and boric acid pumps. The fill line bypasses the makeup tank and makeup pumps whereby fluid is injected into the RC system through the normal makeup control valve. When the fill operation is completed, the auxiliary fill line is secured; makeup and inventory control is then continued by operation of a makeup pump.

2. Reactor Coolant Inventory Regulation — The makeup control valve is the primary device which controls the inventory of the RC system through most phases of RC system operation. In addition to the coolant added to the system through the makeup control valve, a large portion of normal makeup water is added through each RC pump seal as seal inleakage. The control valve is normally automatically controlled by pressurizer level which controls the makeup valve to regulate flow.

3. During power operation, the makeup and purification system supplies all inventory regulation of the RC system. A constant inventory of the RC system is maintained by the combination of makeup flows through both the makeup control valve and RC pump seal inleakages. Total makeup flow to the system equals letdown flow plus RC pump seal leakages.

   During plant shutdown, the temperature in the RC system is decreased thus causing contraction of the total reactor coolant inventory. The volume of the makeup tank is not sufficient to replace the contraction volume of the reactor coolant system during cooldown and requires additional makeup. A source of demineralized water is available for additional makeup, and boric acid is available to control the reactor coolant boron concentration.

   Reactor Coolant System Chemical Shim Reactivity Control

   a. General — One function of the makeup and purification system is to provide and maintain a specific boric acid concentration in the reactor coolant system.

      The boric acid concentration of the reactor coolant system is controlled by increasing or decreasing the boric acid concentration by the addition of boric acid or demineralized water, respectively. Concentrated boric acid is supplied to the MU system at two points on the line leading to the purification filters. The third source of boric acid is from the BWST supply which connects to the MU pump inlet cross tie line. Each line is independent of the others.

   b. Plant Startup and Heatup — During the initial filling of the reactor coolant system, borated water at a concentration which maintains the core 1% $\Delta k/k$ subcritical at the cold condition is added to the reactor coolant system. After the reactor coolant system is brought to operating temperature and pressure, the reactor is brought critical by rod withdrawal and subsequent boric acid concentration reduction.

   c. Power Operation — Following criticality, the neutron poisoning caused by the buildup of xenon and samarium is compensated for by reduction of the boric acid concentration in the reactor coolant system. During normal, steady state operation, the boric acid concentration of the reactor coolant system is adjusted to compensate for the following:

      (1) Core reactivity depletion due to fuel burnup and fission product accumulation.

      (2) Core reactivity transients due to xenon buildup and decay due to daily load changes.

Adjustment of boric acid concentration to account for the core reactivity depletion is made periodically; adjustments on the boric acid concentration to account for xenon buildup and decay is made following a plant load change.

When the thermal load on the reactor decreases due to a decrease in plant load, the integrated control system (ICS) inserts the control rod assemblies (CRA) to match the reduced requirement. The operator then determines the amount of boric acid reduction required to maintain the CRD at their new nominal position and at the same time compensate for the anticipated xenon buildup and peak. If the reduced power operation extends beyond the xenon-peaking period, it becomes necessary for the operator to increase the boric acid concentration in the reactor coolant to counteract the positive reactivity insertion due to xenon decay.

When the thermal load on the reactor increases due to increase in plant load, the ICS withdraws the control rod assemblies to match the increased requirement. For a considerable time span following the load increase, the increased flux produces a poison depletion and corresponding positive reactivity insertion; during this initial phase of load increase, it is necessary for the operator to add boric acid to the reactor coolant to maintain the control rods in the nominal position. Immediately following the poison depletion, a poison repletion will be experienced and the removal of boric acid is necessary to maintain the nominal rod position for the higher power level. Both of these actions will occur subsequently following a load increase.

d. Shutdown — During the period when the reactor coolant system is being cooled and depressurized, the subsequent contraction will be accompanied by an increase in boric acid concentration to provide the reactor coolant system with the proper boric acid concentration to keep the reactor 1% $\Delta k/k$ subcritical at cold condition.

4. Reactor Coolant Purification — One of the major functions of the makeup and purification system is to purify the reactor coolant system during all phases of reactor coolant system operation.

For those periods during which the pressure and temperature of the reactor coolant system are low and the decay heat removal pumps are providing reactor coolant system circulation, the reactor coolant may be purified through interconnections with the decay heat removal system and the purification equipment of the makeup and purification system. The makeup and purification system is supplied with water from the outlet of the decay heat removal coolers which is then routed through the purification path in the makeup and purification system back to the suction of the decay heat removal pumps.

During periods of heatup leading to power operation, the decay heat system is not in use and reactor coolant purification is performed through the normal letdown flow path. When the normal letdown flow path is used, the purification interconnections between the decay heat removal system and the makeup and purification system are secured.

During periods of large temperature transients such as heatup and cooldown, it is expected that the greatest amount of crud (suspended contaminants) will be released from the reactor coolant system surfaces; for this reason, the fluid which is to be purified is first passed through a purification demineralizer prefilter to prevent contaminants of high radiation levels from accumulating in the demineralizer resin beds. Letdown fluid continues to the mixed bed purification demineralizers located downstream of the prefilter; their function is to remove ionic corrosion products and certain fission products, specifically Mn-54, Mn-56, Co-58, Fe-59, Co-60 and Cr-51. The mixed-bed purification demineralizers are boric acid saturated and will remove reactor coolant system impurities other than boron. The effluent from the purification demineralizers is then filtered through the purification filters so that all solid impurities are removed.

The purification demineralizer prefilter, the purification demineralizers, and the purification filters provided purified reactor coolant quality water of reactor grade for injection to the reactor coolant system both as regular makeup and as reactor coolant pump seal injection water; the makeup and purification system is to be operated at all times with these three purification devices continuously in operation.

5. Chemical Adjustment for Corrosion Control — The makeup and purification system is the vehicle through which the corrosion inhibiting chemicals are added to the reactor coolant system. All of the chemicals which are added as solutions are delivered to the makeup and purification system upstream of the purification filters by the equipment in the chemical addition and boron recovery system. The two main functions which the chemicals perform are those of oxygen concentration control and pH control. Specific chemicals are added and adjusted during all phases of reactor operation including startup, normal operation, shutdown and refueling to enable the functions of oxygen and pH control.

   Hydrazine is added to the reactor coolant for oxygen scavenging when the temperature within the reactor coolant system is less than 400F and there is no neutron flux in the reactor.

   For those periods of time when the temperature is above 400F or there is flux in the reactor core, oxygen scavenging is effected by maintaining the correct partial pressure of hydrogen in the makeup tank so that the specified equilibrium concentration of hydrogen is maintained dissolved in the reactor coolant.

   The chemical media employed for pH control is Lithium ($_3Li^7$) added in the form of lithium hydroxide (LiOH). Lithium is also produced in the core region of the reactor due to irradiation of the boron dissolved in the reactor coolant. The concentration of boric acid in solution determines the initial pH of the reactor coolant.

6. Reactor Coolant Pump Seal Injection — The makeup and purification system provides a continuous and dependable supply of clean, high pressure water to the reactor coolant pump seals. Seal injection water is supplied at all times when the reactor coolant pumps are operating so that the mechanical seals are subjected to a nearly constant temperature environment. In line

filters are provided in the seal injection line to prevent particulate matter from entering the seal cavity. Individual flow control valves regulate flow to each pump seal.

The reactor coolant pump may be operated without seal injection water or
component cooling water; the pump must be secured immediately upon loss of
both cooling water supplies. A loss in seal inlet flow to any reactor cool- 7
ant pump, with a loss of component cooling water, will automatically close
the seal return valve for that particular pump. In this way flow reversal
through the RC pump is prevented.

7. Makeup for the Core Flooding Tanks — The core flooding tanks, lines, and check valves will be checked for operability during each scheduled shutdown. The water which is permitted to flow out of the core flooding tank during the test is replenished by use of the makeup lines coming from the discharge header of the makeup pumps.

8. Abnormal Makeup — The makeup and purification system provides makeup to reactor coolant system to replenish inventory lost due to a small rupture in the reactor coolant system pressure boundary. The makeup control valve senses a decrease in pressurizer level and positions itself to maintain level. A high flow alarm is associated with the increase in additional makeup. Following this alarm will be a low makeup tank level alarm. Additional makeup is provided by connections from three sources of concentrated boric acid in conjunction with demineralized water from the chemical addition and boron recovery system to the makeup pumps.

9.3.4.3. Safety Evaluation

9.3.4.3.1. Reliability Considerations

The makeup and purification system provides essential functions for the normal operation of the reactor steam system. Redundant components and alternate flow paths have been provided to improve system reliability.

In addition to the letdown orifice, the system has two full-capacity control valves in parallel with the orifice. One of these control valves is manually operated, and one is remotely operated from the console.

The unit has two makeup pumps, each capable of supplying the required reactor 11
coolant pump seal and makeup flow. One pump is required during normal reactor
plant operations. The makeup system is supplied with full-sized redundant
equipment. One of each of the two letdown coolers, normal purification de-
mineralizers, and purification filters will perform the required duty with
normal letdown flow. Both of these devices will permit twice the normal let-
down flow rate. 11

The chemical composition is controlled by sampling and by makeup as necessary as described in this section. Additional information is included in sections 9.3.4.3.4, 9.3.4.3.5, and 9.3.4.5.

9.3.4.3.2. Code and Standards

Each component of this system will be designed, fabricated, and inspected to the code or standard, as applicable, noted in Tables 9.3-2, 9.3-3, and Chapter 3.

9.3.4.3.3. System Isolation

The letdown line and the reactor coolant pump controlled bleedoff line are outflow lines which penetrate the reactor building. These lines contain electric motor-operated isolation valves inside the reactor building and pneumatic valves outside the reactor building which are automatically closed by an engineered safety features (ESF) signal. The injection line to the reactor coolant pump seals is an inflow line penetrating the reactor building. This line contains stop-check valves inside the reactor building and a remotely operated valve on the outside of the reactor building. The emergency high pressure injection lines are used for injecting coolant to the reactor vessel after a loss-of-coolant accident. After use of the lines for emergency injection is discontinued, the electric motor-operated isolation valves in each line outside the reactor building may be closed remotely by the control room operators for isolation.

The ESF signal for isolation of the letdown and pump seal return is initiated upon (1) an RC system pressure of 1600 psig or less, or (2) a reactor building pressure of 4 psig or greater. 11

9.3.4.3.4. Leakage and Failure Consideration

Design and installation of the components and piping in the makeup and purification system considers radioactive service. Except where flanged connections have been installed for ease of maintenance, the system is of all-welded construction. Principal valves have double packing with provisions for leakoff connections.

The effects of failures and malfunctions in the makeup and purification system concurrent with a loss-of-coolant accident are presented in Chapter 6.

Paragraph 9.3.4.3.3 describes system isolation. These analyses show that redundant safety features are provided where required. For pipe failures in the makeup and purification system, the consequences depend upon the location of the rupture. If the rupture were to occur between the reactor coolant loop and the first isolation valve or stop check valve, it would lead to an uncontrolled loss of coolant from the reactor coolant system.

A single active failure will not prevent boration when desired for reactivity control since several alternate paths are available for adding boron to the reactor coolant system. These are: (1) through the normal makeup line, or (2) through the reactor coolant pump seals. In addition, two redundant sources of boric acid and redundant pumping and delivery equipment are available to the makeup system. If pump suction is unavailable from the makeup tank, a source of borated water is available from the borated water storage tank during reactor power operation.

The MU system design incorporates pressure and flow measuring instruments at strategic locations. Leakage is detected through the interpretation of these signals, i.e., high differential pressures, low pressures, low flow rates, etc.

9.3.4.3.5. Operational Limits

Alarms or interlocks are provided to limit variables or conditions of operation that could cause system malfunctions. The variables or conditions of operation that are limited are as follows:

1. Makeup Tank Level — Low water level in the makeup tank is alarmed and interlocked to the three-way valve. Low-low water level will switch the three-way valve from the bleed position to its normal position.

2. Letdown Line Temperature — A high letdown temperature in the letdown line downstream of the letdown coolers is alarmed and interlocked to close the pneumatic letdown isolation valve, thus protecting the purification demineralizer resins.

3. Dilution Control — The dilution cycle is initiated by the operator. Several safeguards are incorporated into the design to prevent inadvertent excessive dilution of the reactor coolant.

   a. The dilution valves are interlocked so that the operator must preset the desired dilution batch size before initiating the dilution cycle. The dilution cycle will automatically terminate when the dilution flow has integrated to the preset batch size.

   b. Interlocks on the regulating control rod bank automatically terminate the dilution cycle if the regulating rod group (group 6) returns to the position predicted by the ICS boron feed and bleed control or group 5 is inserted into the core to less than 25% withdrawn position. This is designed to prevent inadvertent excessive boron dilution of the reactor coolant.

   c. The operator may manually terminate the dilution cycle at any time.

4. Small Break Detection — The makeup and purification system is provided with a high flow alarm in the normal makeup line that will alarm at a preset value. When the valve positions open to permit larger quantities of flow needed to compensate for flow lost through a small break in the reactor coolant system pressure boundary, the makeup flow control valve will position open to maintain pressurizer level. The makeup tank low level alarm will be actuated after a break has occurred; the time the alarm sounds depends on the size of the break. Operator action is required to establish additional makeup either by opening valves to permit suction from the borated water storage tank or by lining up makeup from the demineralized water storage tanks. The boron concentration of the reactor coolant is controlled by starting the boric acid addition pump when demineralized water is used as a source of makeup.

9.3.4.4. Tests and Inspections

The system has provisions for periodic flow testing to test system operational reliability. This provision is explained in Chapter 6.

Periodic visual inspections and preventive maintenance will be conducted according to sound maintenance practice. The purification demineralizers and filters are periodically switched to allow equal usage. Building isolation valve operability will be tested during scheduled plant shutdown. 11

9.3.4.5. Instrumentation Application in The Makeup and Purification System

Instrumentation application in the makeup and purification system.

The instrumentation in the makeup and purification system provides measurements which are used to indicate, record, alarm, interlock and control process variables such as level and flow as follows:

1. The following process variables are measured and a signal is transmitted that provides indication in the control room.

    a. Letdown flow.

    b. Trimbleed flow.

    c. Makeup pumps discharge header pressure.

    "Trimbleed" is the flow controlled by FV35 on the MU system diagram (Figure 9.3-7). This flow allows for an adjustment in the boric acid concentration without utilizing the three-way valve HV11.

2. The following process variables are measured and a signal is transmitted that will actuate alarms and provide indication in the control room.

    a. Letdown filter differential pressure.

    b. Boric acid flow.

    c. Makeup filter differential pressure.

    d. Makeup tank pressure.

    e. High pressure injection flow.

    f. Makeup flow.

3. The letdown temperature is measured and a signal is transmitted that will actuate alarms and provide indication in the control room. A signal for high temperature interlock is also provided to close the letdown isolation valve.

4. The following process variables are measured and locally indicated.

   a. Makeup pump discharge pressure.

   b. Makeup control valve bypass flow.

5. The RC pump seal bleed off flow is measured, locally indicated and a signal is transmitted that will actuate alarms in the control room.

6. The RC pump seal inlet flow is measured and a signal is transmitted that will actuate alarms and provide indication in the control room. An analog signal is also provided for positioning the RC pump seal inlet flow control valve.

7. The RC pump seal header pressure is measured and a signal is transmitted that provides indication in the control room. The pressure signal is compared to the wide range pressurizer pressure and a differential pressure signal is derived. The differential pressure signal is transmitted to provide indication in the control room. The differential pressure signal is also transmitted to the RC pump seal header pressure control valve to maintain correct RC pump seal header pressure.

8. The makeup tank level is measured by redundancy manually selectable transmitters. The selected signal is transmitted to the control room to be recorded and will actuate alarms. A low level interlock signal is also provided for the feed and bleed circuitry (batch controller).

9. The following process variables are measured and a signal is transmitted that will actuate alarms in the control room.

   a. Purification demineralizer differential pressure.

   b. Purification demineralizer outlet strainer differential pressure.

10. The following process control valves are manual/electric positioned from the control room.

    a. Letdown flow control valve.

    b. Trim bleed control valve.

    c. Boric acid flow control valve.

11. The makeup flow control valve is controlled by a signal from the pressurizer level controller. This is explained in greater detail in Chapter 7.

12. Signals from the following process variables are transmitted to the plant computer for indications and/or alarm.

    a. Letdown temperature.

    b. Letdown flow.

    c. Letdown pressure.

d. Makeup tank temperature.

e. Makeup tank level.

f. RC pump seal differential pressure control.

13. The makeup system feed and bleed controls (batch controller) is a device that measures the amount of boric acid or demineralized water added to the reactor coolant system. The feed and bleed controls (batch controller) will automatically terminate the addition when the quantity preselected by the operator is reached.

9.3.4.6. Radiological Considerations

Shielding of system components for personnel radiation protection is described in Chapter 12. Paths for leakage to the environment of radioactivity in the system fluids and quantities of radioactivity that follow those paths are described in Chapters 11 and 12.

9.3.5. Decay Heat Removal System

9.3.5.1. Design Bases

The decay heat removal (DH) system is designed, in accordance with AEC General Criterion 34, to remove decay heat from the reactor core and sensible heat from the RC system during the latter stages of cooldown. The system also provides auxiliary spray to the pressurizer for complete depressurization, maintains the reactor coolant temperature during refueling and provides an alternate means for filling and draining the refueling canal if the spent fuel cooling system is not available for some reason for this function. In the event of a loss-of-coolant accident, the system injects borated water into the reactor vessel and is used for long-term emergency cooling. The emergency functions (low-pressure injection) of this system are described in Chapter 6.

All components of the decay heat removal system have a category I seismic classification as required by AEC Safety Guide 29. The system design is in conformance with safety guide one as shown in section 6.3.2.1.4.

9.3.5.2. System Description

9.3.5.2.1. General

The DH system is designed to reduce the temperature of the RC system to one that permits refueling after the steam generators have reduced the coolant from the normal operating temperature to the DH system switchover temperature During this latter mode of cooldown, the DH system removes the decay heat generated by the core, removes sensible heat from the RC system, and provides cooled auxiliary spray to the pressurizer for final RC system depressurization. After cooldown, the DH system keeps the RC temperature low enough for refueling operations and provides an alternate means for filling and draining the fuel transfer canal.

The DH system consists of two DH removal pumps; two DH removal coolers discharging through isolation, check, and throttle valves; and interconnecting piping into the RC system. Since the system is designed to perform both normal and emergency functions, separate and redundant flow paths and equipment are provided to prevent the failure of a single component from reducing the system's performance capability below an adequate level.

The DH system is shown schematically in detail in Figure 9.3-8. Table 9.3-4 lists the major component design data; a brief description of each major component follows:

1. Decay Heat Removal Pumps — Two pumps, arranged in parallel, will be designed for continuous operation during the period required for the removal of decay heat for refueling. Both pumps will be available for emergency operation. The design flow will be that required for both pumps to cool the RC system from 305 to 140F in 14 hours starting 6 hours after shutdown. The steam generators will be used to cool the RC system from operating temperature to 305F.

2. Decay Heat Removal Coolers — The coolers will remove decay heat from the circulating reactor coolant during a routine shutdown. Both coolers will be operated to cool the circulated reactor coolant from 305 to 140F in 14 hours starting 6 hours after shutdown.

3. Borated Water Storage Tank — This tank will be located outside the reactor building and the auxiliary building. It will contain a minimum of 2100 ppm boron in solution and will be used as a source of water for filling the fuel transfer canal during refueling. The BWST will also supply borated water for emergency cooling to the reactor building spray system, the DH system, and the high-pressure injection system as described in section 9.2.8. |11

9.3.5.2.2. Modes of Operation

The decay heat removal (DH) system has two primary functions — normal and emergency. The emergency function is employed in the event of a postulated major reactor coolant system rupture resulting in rapid depressurization of the RC system. The emergency function is explained in Chapter 6. The normal function of the system is to provide plant heatup and cooldown with the associated depressurization of the RC system for maintenance or inspection of the RC system and scheduled refueling. During refueling, when the RC system is depressurized, the reactor coolant is recirculated through the DH system to remove residual heat generated by the remaining portion of the core. The DH system provides pressurizer auxiliary spray when a decrease in RC pressure is desired. During plant startup from a cold condition, the DH system maintains forced flow through the core until the RC pump(s) are started. A secondary function of the DH system is to fill and partially drain the refueling canal during refueling if the spent fuel cooling system is not for some reason, available for this function. The following is a brief description of each normal function of the DH system.

Normal System Functions

1. Plant Startup — During plant startup from the cold shutdown condition, the RC system is normally brought to pressure and temperature by the initial heat of the pressurizer heaters and then by the addition of heat dissipated

to the reactor coolant from the RC pump(s). The DH system maintains forced flow through the core until the RC pumps are started. The RC pumps are dependent on sufficient net positive suction head before the pumps can operate without cavitation. Following RC pump startup, the suction valves associated 8
with the DH system are closed and prevented from reopening following pressurization of the RC system by the interlocks described in section 6.3.2.16. The system is placed in low-pressure injection standby to take suction from the BWST and the reactor building sump.

2. Decay Heat Removal — Plant cooldown is accomplished during the DH removal mode by recirculating coolant from one of the reactor vessel outlet lines to the DH removal pumps, through the DH removal coolers, and returning it to the vessel through the core flooding nozzles. The DH letdown line from the RC system to the suction of the DH removal pumps is provided with two electric-motor-operated (EMO) stop valves in series in the reactor building. Both valves are interlocked with auto-closure on high pressure signal 11
from a pressurizer wide-range pressure transmitter.

3. Depressurization of The RC System — An auxiliary pressurizer spray line provides a means for depressurizing the RC system after the RC pumps are stopped during cooldown; this line routes a small flow from the DH removal cooler discharge line to the pressurizer spray line.

   The line tees off the crossconnect between the two decay heat loops allowing depressurization with either DH pump operating.

4. Refueling — During refueling, the DH system provides long-term cooling of the reactor coolant and the refueling canal water after shutdown to maintain the coolant and the refueling canal water at the refueling temperature. Heat removed by the DH coolers is dissipated to the component cooling system. During this period, the system interfaces with the spent fuel cooling (SFC) system for purification of the refueling canal water. The refueling canal water is purified by the SFC system purification loop. The effluent from this purification loop is discharged into the DH system through a connection to the DH letdown line. This flow path allows water to circulate from the refueling canal through the SFC purification equipment and back to the canal via the DH system.

5. Refueling Canal Filling and Draining — A secondary function of the DH system is to fill and partially drain the refueling canal during refueling. This function is normally fulfilled by the SFC system. The fuel transfer canal may be completely filled by the DH removal pumps taking suction from the BWST. The discharge from the pumps is routed into the refueling canal via the SFC system connection downstream of the DH cooler. Either DH pump string may be used for this operation.

   The connection between the SFC system and the DH letdown line within the reactor building is used to partially empty the refueling canal using the DH removal pump. The canal water is pumped to the BWST via the test line. When the water level reaches the reactor vessel flange, final drainage of the lower refueling canal is accomplished using the SFC pumps.

6. Backup Cooling for Spent Fuel Pool — The DH system provides secondary backup cooling for the spent fuel storage pool. 1

The DH system is capable of cooling the maximum number of irradiated cores (1-1/3) stored in the pool, as explained in section 9.1.2, when there is no previously irradiated fuel in the associated reactor. 11

Cross-connecting lines between the SFC systems provide this backup function. These cross-connects are sized to pass the SFC system design flow rates. Since the DH pump is rated at 5000 gpm and the SFC pumps are rated at 1650 gpm, throttling will be required to prevent excessive flows. This is accomplished by remote-manual throttling of the valve at the discharge of the DH coolers.

Emergency Function

Chapter 6 describes the emergency function of the DH system.

9.3.5.3. Safety Evaluation

The system is designed to perform both normal and emergency functions. Separate and redundant flow paths and equipment are provided to prevent a single component failure from reducing the system performance below a safe level. The loss of one pump or cooler will only slow the cooldown. During normal cooldown, one decay heat pump and one decay heat cooler make up the "normal operating system." A manually operated stop valve at the suction and discharge of each pump permits isolation of the pump for maintenance. A check valve at the discharge of each pump provides (1) protection against water leaking backward into the system if the pumps are stopped and (2) protection for the pumps against water hammer damage. 11

The discharge from each of the DH removal coolers is routed through an electric-motor-operated (EMO) stop valve outside the reactor building, through the reactor building (RB) penetration, and then to a core flooding line which discharges to the reactor through one of the core flooding nozzles. Check valves — one in each core flooding line and another in each decay heat injection line within the reactor building — provide isolation for the DH system. The EMO stop valve provides reactor building isolation and protection for the DH system from the higher reactor system pressure if the check valves inside the reactor building leak.

A redundant decay heat line is provided around the two EMO valves (letdown valves) to the suction of the DH removal pumps to allow a means of obtaining a flow path in the event that it is not possible to use the normal DH letdown line. The redundant line contains two closed, manually operated stop valves in series inside the reactor building and one manually operated stop valve outside the building. A crossover connection from the spent fuel cooling system is provided to this redundant DH letdown line to provide a flow path for backup cooling of the spent fuel storage pool using DH coolers.

Relief valves in both DH letdown lines (inside the reactor building downstream of the second valve from the RC system) provide overpressure protection of the DH system from component failures or operator errors during plant cooldown with the DH system in operation. The possible incidents that could cause overpressurization of the DH system include loss of power to the DH pumps or loss of DH cooling, all pressurizer heaters energized, makeup pump actuated with makeup control valve full open, core flooding tank isolation valve opens, and all high-pressure injection pumps actuated. These relief valves are sized to handle the flow rate that would give the fastest rate of pressure increase required to reach the DH system design pressure from among the faults stipulated above. Any relieved fluid is routed to the RC drain tank in the waste disposal system.

Two remote-manual, pneumatically operated valves are located at each DH removal cooler; each is a throttle valve. One, located in the outlet of the DH cooler, is used to control flow through the cooler and to limit DH pump runout; the other is located in the cooler bypass and, in conjunction with the cooler outlet throttle valve, is used to control the temperature of the DH fluid returned to the reactor vessel.

Relief valves are provided in the suction and discharge lines of the pumps for overpressure protection from ambient temperature changes or isolation valve leakage. Any relieved fluid is routed to the waste disposal system.

A small recirculation line around the DH removal pumps and coolers is provided to prevent the pumps from overheating in the event a valve fails to open, a stoppage occurs in the discharge piping, or the reactor pressure is too high to allow flow.

During reactor power operation, all equipment of the decay heat removal system is idle; and all isolation valves are closed. Under loss-of-coolant accident conditions, fission products may be recirculated in the coolant through the exterior piping system. Potential leaks have been evaluated to obtain the total radiation dose due to leakage from this system. The evaluation is discussed in Chapter 15.

9.3.5.4. Test and Inspections

The delivery capability of the DH removal pumps may be tested periodically by opening the valve in the line from the BWST, starting one pump, and opening the corresponding valve in the test line to circulate the flow back to the BWST. Pump discharge pressure and flow indication are required to determine whether the pump is operating normally.

Equipment performance may also be verified while the system is in the normal DH removal mode. At this time, pump and motor performance, cooler characteristics, and line pressure drops may be determined to indicate whether any equipment degradation has taken place.

9.3.5.5. Instrumentation Application

The instrumentation in the DH system provides measurements that are used to indicate and alarm as follows:

1. These process variables are measured and a signal is transmitted that actuates alarm and provides indication in the control room.

a. DH removal injection flow.

b. DH removal cooler outlet temperature.

c. DH removal cooler inlet temperature.

2. DH removal pumps discharge pressure is measured and indicated locally.

9.3.6. Chemical Addition and Boron Recovery System

9.3.6.1. Design Bases

The chemical addition and boron recovery system serves the nuclear steam system (NSS) during normal operations as follows:

1. Stores reactor coolant bleed and evaporator distillate.

2. Recovers boron from the reactor coolant for reuse.

3. Prepares, stores, and transfers lithium hydroxide for pH control and hydrazine for oxygen control.

4. Stores and transfer the quantities of concentrated boric acid needed to achieve cold shutdown of the reactor with a stuck rod of greatest worth at any time during core life.

5. Stores and has the capability to transfer the quantities of concentrated boric acid needed to achieve cold shutdown of the reactor with all rods stuck, during any time in the core life.

During a loss-of-coolant accident (LOCA), the chemical addition and boron recovery system is isolated from the reactor building by isolation valves in the makeup and purification system.

The chemical addition and boron recovery system accommodates the effluents resulting from reactor transients (including startup, shutdown, and load maneuvering) based on 335 days' power operation and 30 days' refueling per year. The reactor operation is based on an equilibrium fuel loading operated at a power factor of 0.8; i.e., 292 days of full power days/0.8 power factor = 365-day cycle, of which 30 days — included in the power factor — are for refueling. The system also accommodates the effluents transferred as a result of xenon control transient requirements based on a daily load swing of 100%-50%-100% power up to 90% of core life for the equilibrium fuel cycle.

By limiting and controlling load swing operations, a 30-day back-to-back refueling may be performed. This, however, cannot be interpreted as a design condition. 11

A minimum reactor coolant bleed holdup capacity of one reactor coolant volume per reactor is provided; and a minimum distillate water storage capacity of one reactor coolant volume per reactor is provided. The minimum bleed holdup tank and distillate water storage tank sizes are obtained from the thermal expansion and dilution volumes letdown during reactor startup with an equilibrium fuel loading.

9.3.6.2. System Description and Evaluation

9.3.6.2.1. General

The chemical addition and boron recovery system is shown schematically in Figures 9.3-9 and 9.3-10. The chemistry of the reactor coolant system to be maintained by the system is outlined in Table 5.2-6. Tables 9.3-5 and 9.3-6 list the individual component data.

9.3.6.2.2. Chemical Addition System

The chemical addition and boron recovery system provides concentrated boric acid and demineralized water as required for changing the boron concentration of the reactor coolant system. The boric acid supply system shall be capable of providing sufficient boric acid solution to increase the reactor coolant system's boron concentration at the end of life to the concentration necessary to achieve cold shutdown with a cooldown rate of 50F/h. The system shall also possess sufficient redundancy of components and storage facilities so that if a failure should occur, the system would still be capable of supplying the required boric acid. The boric acid supply system shall contain a sufficient amount of boric acid to achieve a cold shutdown with a stuck control rod of greatest worth.

The lithium hydroxide and hydrazine addition portion of this system adds lithi-
um hydroxide or hydrazine to the reactor coolant. The ^{7}Li concentration in the
reactor coolant is maintained at 0.2 to 2.0 ppm by the addition of lithium
hydroxide. Hydrazine is added during subcritical operation to maintain a hy- 11
drazine concentration between 0.1 and 1.0 ppm. Hydrogen overpressure in the
makeup tank is maintained to provide the hydrogen concentration in the reactor
coolant at 15 to 40 cc/kg H_2O.

9.3.6.2.3. Boron Recovery System

The boron recovery portion of the chemical addition and boron recovery system recovers boric acid from the reactor coolant for reuse. Boric acid is recovered from the evaporators used to process reactor coolant stored in the RC bleed holdup tanks. Concentrated boric acid is stored in the concentrated boric acid storage tank and the boric acid addition tank and is available for injection during cold shutdown of the RC system. Distillate from the evaporator is stored in the distillate storage tanks.

The RC bleed holdup tanks provide storage for the reactor coolant bleed during startup and normal operation. The distillate storage tanks provide storage for distillate recovered from the processing of reactor coolant bleed.

During normal operation, reactor coolant is bled from the RC system to change the boron concentration of the coolant. For approximately the first 90% of the fuel cycle, the reactor coolant is processed to recover the boron by bleeding reactor coolant to the RC bleed holdup tank and replacing the volume bled with distillate. During the last 10% of fuel cycle, the reactor coolant is routed through deborating demineralizers to remove the boron and returns the effluent to the RC system via the makeup and purification system.

The chemical addition and boron recovery system stores sufficient quantities of concentrated boric acid and demineralized water to control the boron concentration in the RC system during startup and normal operation and to obtain cold shutdown with a stuck rod of greatest worth at any time during core life.

9.3.6.2.4. Modes of Operation

One main function of the chemical addition and boron recovery system is to provide storage for reactor coolant bleed and evaporator distillate during all phases of reactor operation.

1. Heatup — During heatup, when the RC system is heating, the coolant expands and must be continuously bled until heatup is completed. The RC bleed holdup tanks provide coolant storage for this phase of operation. Before reactor criticality, the boron concentration in the RC system is decreased from the refueling concentration to the BOL concentration. The boron concentration is reduced by bleeding reactor coolant to the RC bleed holdup tanks and replacing the bleed with evaporator distillate or demineralized water. The evaporator distillate is stored in the distillate storage tanks.

2. Power Operation — During normal operation, the RC bleed holdup tanks provide storage for the bleed from load transients and fuel depletion until it is processed. The distillate storage tanks store evaporator distillate used for makeup to the makeup and purification system during bleed operation. Two RC bleed holdup tanks and four distillate storage tanks can be 11
shared by both units. The RC bleed holdup tanks are located side by side with inlet and discharge lines cross-tied so that either tank may be used for either unit. The distillate storage tanks are also located side by side and have cross-tied inlet and discharge lines.

3. Cooldown — During cooldown, water from the distillate storage tank is returned to the RC system to compensate for the decreased volume due to contraction of the coolant. Concentrated boric acid is also added to the coolant as necessary to achieve the proper boron concentration.

4. Processing of Stored Reactor Coolant Bleed — Transferring reactor coolant bleed to the RC bleed holdup tank provides a feed source for boric acid recovery. The fission products and lithium are removed by the RC bleed evaporator demineralizers prior to concentration. The purified reactor coolant (dilute boric acid solution) is concentrated by evaporators to a 5 wt% boric acid solution and is transferred to the concentrated boric acid storage tank. The distillate is pumped to the distillate test tanks for analysis. If the distillate meets specifications it is pumped to the distillate storage tanks for reuse. If it does not meet specifications, it is demineralized by the distillate demineralizers and rechecked; if the demineralized distillate does not meet specifications it is pumped back to the RC bleed holdup tanks for reprocessing.

5. Changing RC Boron Concentration — The chemical addition and boron recovery system provides concentrated boric acid and water as required for changing the boron concentration of the reactor coolant bleed.

a. Increasing Boron Concentration in Reactor Coolant — The boron concentration in the reactor coolant may be increased by the addition of boric acid stored at 5 to 7 wt%. To increase the boron concentration, the reactor coolant is diverted from the makeup and purification system to the RC bleed holdup tank. During this period, concentrated boric acid solution is pumped at a rate equal to the bleed rate into the makeup and purification system until the desired boron concentration in the RC system is obtained. The concentration can also be increased by pumping concentrated boric acid solution into the RC system via the makeup and purification system during reactor cooldown. During this period, the reactor coolant volume contracts, thus providing a void volume that can accept the concentrated boric acid volume without the need for bleeding the reactor coolant. Both methods can utilize 5 wt% concentrated acid obtained from concentration of the RC bleed or 7 wt% concentrated boric acid solution. The 7 wt% concentrated boric acid solution is prepared in a mix tank and transferred to the chemical addition tank for storage and subsequent use. This method offers the versatility of providing a quick and unlimited supply of concentrated boric acid solution. A redundant source of boric acid is available for addition to the coolant from a normally secured crossover from the borated water storage tank to the suction of the boric acid pumps.

b. Decreasing Boron Concentration in Reactor Coolant — The boron concentration in the RC system may be decreased by diverting reactor coolant from the makeup and purification system to an RC bleed holdup tank. During this period, water is pumped from a distillate storage tank at a rate equal to the bleed rate into the makeup and purification system until the desired boron concentration is reached. At the end of core life when the boron concentration is low (core life approximately 90%), it is not economical to recover the boric acid. At this stage the coolant is diverted to the deborating demineralizers for removal of the boric acid. The deborated effluent from the demineralizers is returned to the coolant via the makeup and purification system. This operation is controlled by the three-way valve described in section 9.2.

6. Storage and Transfer of Concentrated Boric Acid Solution for Reactor Shutdown — Sufficient concentrated boric acid solution will be in storage and can be transferred from the chemical addition and boron recovery system to borate the reactor coolant for refueling or cold shutdown. During cooldown, as the temperature of the coolant decreases, boric acid solution is transferred to the makeup and purification system to adjust the coolant to the refueling concentration. For cold shutdown, sufficient boric acid solution is available to borate the coolant to a 1% $\Delta k/k$ subcritical condition with a stuck control rod of greatest worth at any time during core life. The boric acid solution is available from the concentrated boric acid storage tank, the boric acid addition tank, and the borated water storage tank.

7. Preparation, Storage, and Transfer of Other Chemicals —

a. Lithium Hydroxide — LiOH is used for pH control in the reactor coolant. The largest additions are required after refueling and after new purification demineralizer resins are placed in operation. LiOH is also used to replace the lithium removed by the RC bleed evaporator demineralizers. The solution is made up to a 1 to 2 wt% lithium hydroxide solution and is transferred by positive displacement pumps to the reactor coolant via the makeup and purification system. The two pumps provided can pump to either reactor unit.

b. Hydrazine — Hydrazine is used as an oxygen scavenger in the reactor coolant. It is used during the initial fill of reactor coolant and after refueling when the coolant contains a large quantity of oxygen. Hydrazine is also added during cooldown when large quantities of makeup are added to compensate for contraction of the coolant. The hydrazine is stored in a drum with a nitrogen blanket to prevent oxygen pickup by the hydrazine. The solution is transferred by positive dis- 11
placement pumps to the reactor coolant via the makeup and purification system. The two hydrazine pumps can pump to either reactor unit.

c. Caustic Solution — Caustic is used to neutralize the waste evaporator feed solution in the waste disposal system and to regenerate the deborating demineralizers. In the waste disposal system, caustic is added to the evaporator feed tank to increase the solubility and reduce the volatility of the borates. At the end of the fuel cycle, after the deborating demineralizers have been used, the caustic is used to regenerate the demineralizer resins. The caustic is prepared as a 50 wt% solution and is transferred by a positive displacement pump to the demineralizers for regeneration of the resin.

9.3.6.3. Safety Evaluation

9.3.6.3.1. System Isolation

The boron recovery portion of the chemical addition and boron recovery system is not required during an emergency condition, nor is the chemical addition portion of the system used in an emergency or a LOCA. Both portions of this system are essentially isolated from the RC system by being connected to the makeup and purification system. Reactor coolant bleed is isolated at the makeup and purification system letdown line penetrating the reactor building. Chemical addition is isolated from the RC system by the motor-operated isolation valves on the outlet of the makeup tank, which are directed to close by signal from the engineered safety features (ESF) system during a LOCA.

9.3.6.3.2. Leakage Considerations

Leaking reactor coolant and chemicals added from the chemical addition system inside the reactor building are contained and collected in the building sump. Outside the reactor building, leakage from this system is directed to the floor drains and collected in the waste disposal system.

9.3.6.3.3. Codes and Standards

Each component of this system will be designed fabricated, and inspected to the applicable code or standard noted in Tables 9.3-5 and 9.3-6.

9.3.6.3.4. Reliability Considerations

The boron recovery portion of this system provides the essential functions for normal operation. The system's redundant components and alternate flow paths improve system reliability.

The boric acid supply system is capable of providing sufficient boric acid solution to increase the RC system boron concentration at the end of life to the concentration necessary to achieve cold shutdown with a cooldown rate of 50F/h. The system also has sufficient redundancy of components and storage facilities so that if a failure should occur, the system would maintain the capability of supplying the required boric acid. The boric acid supply system contains sufficient boric acid to achieve cold shutdown with a stuck control rod of greatest worth as defined in Chapter 4.

To ensure dissolving and maintaining the solubility of the boric acid, all components and piping, including tanks, pumps, and valves, have redundant heating devices. The equipment is to be maintained at a minimum temperature equal to the precipitation temperature for the boric acid concentration that is used plus a 10F margin. No heating devices are required where the boric acid is maintained at concentrations that would not normally precipitate. All components requiring electrical power are provided with separate and redundant power sources. Redundant flow paths and pumps are provided for the addition of boric acid to the makeup and purification system of either unit. Sample connections are provided to permit determination of the boric acid concentration of the fluids that are in storage before they are used.

9.3.6.4. Tests and Inspections

Fluid samples are periodically taken from the RC system and the auxiliary systems for analysis. The samples are subjected to chemical and radiochemical analyses (as approriate) to determine boron concentration, fission and corrosion product activity levels, dissolved gas concentration, chloride concentration, pH and conductivity levels, noncondensible gases in the pressurizer steam space, and gas compositions in various vessels. The analytical results are used to regulate boron control adjustments, to monitor fuel rod integrity, to evaluate ion exchanger and filter performance, to specify chemical additions to various systems, and to maintain the proper hydrogen overpressure in the makeup tank.

The system requires no functional testing. It will be examined periodically to determine its operating condition. Periodic visual inspections and preventive maintenance will be conducted according to sound maintenance practice.

9.3.6.5. Instrumentation Application

The instrumentation in the chemical addition and boron recovery system provides measurements that are used to indicate, alarm, and interlock as described below:

1. The following process variables are measured and a signal is transmitted that will actuate alarms in the control room:

   a. Deborating demineralizer outlet strainer differential pressure.

   b. Evaporator distillate demineralizer outlet strainer differential pressure.

   c. RC bleed evaporator demineralizer outlet strainer differential pressure.

2. The following process variables are measured, locally indicated, and a signal is transmitted that will actuate alarms in the control room:

   a. Deborating demineralizer outlet strainer differential pressure.

   b. Evaporator distillate demineralizer outlet strainer differential pressure.

   c. RC bleed evaporator demineralizer outlet strainer differential pressure.

3. The following process variables are measured and locally indicated:

   a. Distillate transfer pump discharge pressure.

   b. RC bleed transfer flow.

   c. RC bleed transfer pump discharge.

   d. Hydrazine pump discharge pressure.

   e. Hydrazine pump quantity.

   f. Lithium hydroxide pump discharge pressure.

   g. Lithium hydroxide pump quantity.

   h. Boric acid flow.

   i. Boric acid pump discharge pressure.

   j. Evaporator distillate test tank pump discharge pressure.

   k. Caustic pump discharge pressure.

4. The distillate transfer flow is measured and a signal is transmitted that provides indication in the control room.

5. The following process variables are measured and a signal is transmitted that will actuate alarms and provide indication in the control room:

   a. Distillate storage tank level.

   b. RC bleed tank level.

   c. Boric acid addition filter differential pressure.

   d. Concentrated boric acid tank level.

   e. Evaporator distillate test tank level.

6. The temperature of the concentrated boric acid storage tank is measured and a signal is transmitted that will actuate alarms and provide indication in the control room. A signal is also provided for heater control to maintain the fluid temperature within a predetermined range.

7. The boric acid mix tank temperature is measured, indicated locally, and a signal is transmitted that will actuate alarms and provide indication in the control room. A signal is also provided for heater control to maintain the fluid temperature within a predetermined range.

8. The boric acid mix tank level is measured, indicated locally and a signal is transmitted that will actuate alarms and provide indication in the control room. An interlock signal is also provided to de-energize the heaters on low level.

9. The boric acid addition tank temperature is measured and a signal is transmitted that will activate alarms and provide indication in the control room. A signal is also provided for heater control to maintain the fluid temperature within a predetermined range.

10. The boric acid addition tank level is measured, indicated locally, and a signal is transmitted that will actuate alarms and provide indication in the control room.

11. The distillate transfer flow control valve is manually/electrically positioned from the control room.

9.3.7. Failed Fuel Detection System

9.3.7.1. Design Bases

The failed fuel detection system provides a means for identifying out-of-reactor fuel assemblies containing failed fuel rods (rods that have not maintained clad integrity). The system utilizes the gamma radiation of fission product gases escaped from the fuel rods as the means of failure identification.

A closed loop of piping in the fuel transfer canal including a fuel assembly container, coolant circulating pump, radiation detection system, measurement instrumentation, controls, and data readout equipment is utilized. A control console is contained within one portable unit. Coolant fluid is circulated within the loop to accumulate fission products from the failed rods. Flow of this coolant through detection system allows for identification of higher-than-background radiation levels within the coolant, confirming the presence of fission product material.

This wet-sipping technique also provides for water or gas sample collection for off-line evaluation. The system provides for fuel assembly can and sample manifold flushes for decontamination purposes.

Criteria for design of this system are as follows:

1. The system shall not, under either normal or emergency conditions, endanger system operating or maintenance personnel due to operating radiation levels.

2. The maximum incremental radiation exposure that operating personnel could obtain from routine use of this system shall be 15 mRem/h.

3. The system shall permit evaluation of irradiated fuel assemblies for the entire fuel exposure range without damage to these fuel assemblies during examination.

4. The system shall be capable of functioning within the refueling canal, interfacing with available services, space, and handling facilities.

5. The system shall be capable of evaluating a complete reactor load of fuel assemblies on a continual basis.

9.3.7.2. System Description

Utilization of the system to detect failed fuel rods in an intact fuel assembly uses a closed piping loop with a container for the fuel assembly, heat exchanger, pump, detection system, measurement instrumentation, controls, and data readout equipment.

The cooling fluid within the loop (i.e., the gas or liquid coolant) is also the medium that transports the potentially leaking substances from the fuel assembly to the detection system. This cooling fluid is monitored by the measurement instrumentation for fission-product radioactivity and removal of water or gas samples from this cooling fluid piping is also possible for utilization of off-line counting.

Following placement of the suspect fuel assembly within the fuel assembly can (see Figure 9.3-11, the can is flushed with deionized water, once-through flow, with the discharge being directly into the refueling canal water. The system is then made into a closed loop and deionized water is circulated to permit the accumulation of fission products. An accumulation of radioactive fission products would be indicated by the detection system and measurement instrumentation.

Should a gaseous sample be desired for off-line counting, nitrogen would be bubbled through the fuel assembly to act as a collector and carrier of fission product gases. These gases would then be extracted from the recirculating water by means of a gas trap. The gases would then be passed through a moisture separator and a liquid-nitrogen-cooled sampling cylinder. Xenon contained in the gaseous stream would be liquified in this cold trap, but the carrier nitrogen would continue to the gas waste line, to be disposed of appropriately.

The expected radiation levels within the sample system are not sufficiently high to require the use of radiation shielding for personnel exposure reduction; necessary shielding to reduce background radiation levels for the detection system is provided.

The design of the fuel assembly can negates the requirement for operator determination of a gas-water interface, as the can outlet is above the top of the active fuel length.

A manifold flush mode is designed into the system to allow for removal of accumulated fission products following evaluation of a failed assembly. The manifold can thus be decontaminated by action initiated at the control console.

9.3.7.3. Safety Evaluation

Final system design, component specification and installation details are needed prior to undertaking the analyses and evaluation. A future amendment to this PSAR will contain the safety evaluation.

9.3.7.4. Tests and Inspections

For the same reason stated in 9.3.7.3 the information on tests and inspections will be provided in a future amendment.

9.3.7.5. Instrumentation Applications

The controls are completely enclosed in a single cabinet.

This unit contains the switches which activate the following functions:

1. AC power.
2. Flush manifold.
3. Prime manifold.
4. Flush can.
5. Recirculate.
6. Recirculate alarm.
7. Valve status (open or closed).

Valving is performed automatically by depressing the function switches. Stainless steel solenoid valves, with position indicators for interlocking, provide the means of doing this. The solenoid valves and the pump motor will be properly interlocked to prevent improper action when a function switch is depressed. "Prime Manifold" will be "Automatic" or "Manual." The status of valves will be shown by lights on a flow diagram inscribed on the control panel. If the "Recirculate" function ceases to operate after being activated, an audible alarm will be sounded. The alarm sounds even if recirculate is intentionally stopped.

The water recirculate and flush flow rates are measured with stainless steel rotating transmitters and are indicated remotely by meters mounted in the front panel of the instrument turret. The meters will be calibrated at 30 gpm full scale.

Table 9.3-1. Compressed Air System Component Performance Data

Air Compressors	
Quantity	4
Type	Nonlubricated reciprocating
Rated capacity, scfm	610
Motor horsepower	125
Material, HP/LP	Meehanite/Alum
Design pressure, psig	125
Design temperature, F	260
ASME Code, Section/Class	NA
Seismic category	I
Aftercoolers	
Quantity	4
Type	Shell/tube
Cooling water flow, gpm	20
Temperature of discharge air, F	110
Design pressure, psig	125
ASME Code, Section	VIII
Air Receiver	
Quantity	4
Capacity, ft^3	266
Design pressure, psig	125
ASME Code, Section/Class	III/3

Table 9.3-2. Makeup and Purification System Performance Data

Normal letdown flow, gpm	50
Maximum letdown flow, gpm	200
Total seal flow to each reactor coolant pump, gpm	8-15
Seal inleakage to reactor coolant system per reactor coolant pump, gpm	7-14
Temperature to seals, normal/maximum, F	120/200
Purification letdown fluid temperature, normal/max., F	120/150
Makeup tank normal operating pressure range, psig	15-35
Makeup tank water vol., nominal, ft^3	800

Table 9.3-3. Makeup and Purification System Component Data

Makeup Pump

Type	Horizontal, multistage, centrifugal, mechanical seal	
Rated capacity, gpm	306	
Rated head, ft	6500	
Motor horsepower, hp	900 nameplate	
Pump material	SS wetted parts	
Design pressure, psig	3500	11
Design temperature, F	200	

Letdown Cooler

Type	Shell and tube	
Heat transferred, 10^6 Btu/h	23.9	
Letdown flow, 10^4 lb/h	4.9830	11
Letdown cooler inlet/outlet temperature, F	570/120	
Material, shell/tube	CS/SS	
Design pressure (shell/tube), psig	200/2500	
Design temperature (shell/tube), F	350/600	
Cooling water flow (each), 10^5 lb/h	3.05	
Code (tube/shell)	ASME III-2/III-3	

Reactor Coolant Pump Seal Return Cooler

Type	Shell and tube	
Heat transferred, 10^6 Btu/h	2.67	
Flow rate, 10^5 lb/h	1.05	
Seal return temperature change, F	145 to 120	
Material (shell/tube)	CS/SS	
Design pressure (shell/tube), psig	200/150	
Design temperature (shell/tube), F	200/200	
Cooling water flow (each), 10^4 lb/h	9.25	
Code (tube/shell)	ASME III-3/III-3	11

Table 9.3-3. (Cont'd)

Makeup Tank

Volume, ft^3	1200
Design pressure, psig	100
Design temperature, F	200
Material	SS
Code	ASME III-2

Purification Demineralizer Prefilter

Flow rate, gpm	200
Material	SS
Design pressure, psig	150
Design temperature, F	200
Code	ASME III-2

Purification Demineralizer

Type	Mixed bed, boric acid saturated
Material	SS
Resin volume, ft^3	50
Flow, gpm	100
Vessel design pressure, psig	150
Vessel design temperature, F	200
Code	ASME III-2

Purification Filter

Flow rate, gpm	100
Material	SS
Design pressure, psig	150
Design temperature, F	200
Code	ASME III-2

Table 9.3-3. (Cont'd)

Seal Injection Filter

Flow rate, gpm	60
Material	SS
Design pressure, psig	3200
Design temperature, F	200
Code	ASME III-2

Table 9.3-4. Decay Heat Removal System Equipment Data

Decay Heat Removal Cooler

Quantity	2 full-capacity	
Type	Shell and tube	
Heat transferred (normal/max.), 10^6 Btu/h	41/146	11
Shell/tube flow, gpm	7500/5000	
Material, shell/tube	CS/SS	
Shell/tube design temperature, F	200/350	
Shell/tube design pressure, psig	200/675	
ASME Code/Class (shell)	III/3	
ASME Code/Class (tube)	III/2	
Seismic category, shell/tube	I/I	

Decay Heat Removal Pump

Quantity	2
Type	Centrifugal
Rated capacity/TDH, gpm/ft	5000/385
Rated runout capacity/TDH, gpm/ft	6500/250
Pump material	SS
Design pressure, psig	675
Design temperature, F	350
Horsepower requirement, hp	700
Seismic design category	I
ASME Code, Section/Class	III/2

Table 9.3-5. Chemical Addition System Equipment Data

Tanks	
Boric acid mix tank	
Quantity	1
Type	Vertical cylindrical
Volume (nominal), ft^3	320
Design pressure	Atmospheric
Design temperature, F	200
ASME Code/class	VIII/no code stamp
Material	SS
Boric acid addition tank	
Quantity	1
Type	Horizontal cylindrical
Volume (nominal), ft^3	2500
Design pressure, psig	4
Design temperature, F	200
ASME Code/class	III/3
Material	SS
Concentrated boric acid storage tanks	
Quantity	2
Type	Horizontal cylindrical
Volume (nominal), ft^3	4200
Design pressure, psig	4
Design temperature, F	200
ASME Code/class	III/3
Material	SS
Lithium hydroxide mix tank	
Quantity	1
Type	Vertical cylindrical
Volume, ft^3	9.4
Design pressure	Atmospheric
Design temperature, F	200
ASME Code/class	VIII/no code stamp
Material	SS
Caustic mix tank	
Quantity	1
Type	Vertical cylindrical
Volume, ft^3	27
Design pressure	Atmospheric
Design temperature, F	200
ASME Code/class	VIII/no code stamp
Material	SS

Table 9.3-5. (Cont'd)

Pumps		
Boric acid pump		
Quantity	4	11
Type	Centrifugal (horiz/vert)	
Capacity, gpm	25	
Head, ft	210	11
Design pressure, psig	150	
Design temperature, F	200	
ASME Code/class	III/3	
Material	SS	
Lithium hydroxide pump		
Quantity	2	
Type	Reciprocating piston	
Capacity, gph	10	
Head, ft	231	
Design pressure, psig	150	
Design temperature, F	200	
ASME Code/class	NA	
Material	SS	
Hydrazine pump		
Quantity	2	
Type	Reciprocating piston	
Capacity, gph	10	
Head, ft	231	
Design pressure, psig	150	
Design temperature, F	200	
ASME Code/class	NA	
Material	SS	
Caustic pump		
Quantity	1	
Type	Reciprocating piston	
Capacity, gph	120	
Head, ft	231	
Design pressure, psig	150	
Design temperature, F	200	
ASME Code/class	NA	
Material	SS	
Boric acid filter		
Quantity	2	
Type	Disposable element	
Flow rate, gpm	100	
Design pressure, psig	150	
Design temperature, F	200	
Shell material	SS	
Micron size (nominal), μ	25	
ASME Code/class	III/3	

Table 9.3-6. Boron Recovery System Equipment Data

Tanks	
Evaporator distillate test tank	
Quantity	2
Type	Vertical cylindrical (flat bottom)
Volume (nominal), ft^3	1900
Design pressure, psig	4
Design temperature, F	200
ASME Code/class	VIII/none
Material	SS
Distillate storage tanks	
Quantity	4
Type	Horizontal cylinder (dished ends)
Volume, ft^3	13,000
Design pressure, psig	150
Design temperature, F	200
ASME Code/class	III/3
Material	SS
Reactor coolant bleed holdup tank	
Quantity	2
Type	Horizontal cylinder (dished ends)
Volume, ft^3	16,000
Design pressure, psig	50
Design temperature, F	200
ASME Code/class	III/3
Material	SS
Reactor coolant bleed evaporator	
Quantity	2
Capacity, gpm	30
Design pressure, psig	150
Design temperature, F	200
ASME Code/class	III/3
Material	SS
Demineralizers	
Evaporator distillate demineralizer	
Quantity	2
Type	Nonregenerative (mixed bed)
Material	SS
Resin volume, ft^3	30
Flow rate, gpm	80
Vessel design pressure, psig	150
Vessel design temperature, F	200
ASME Code/class	III/3

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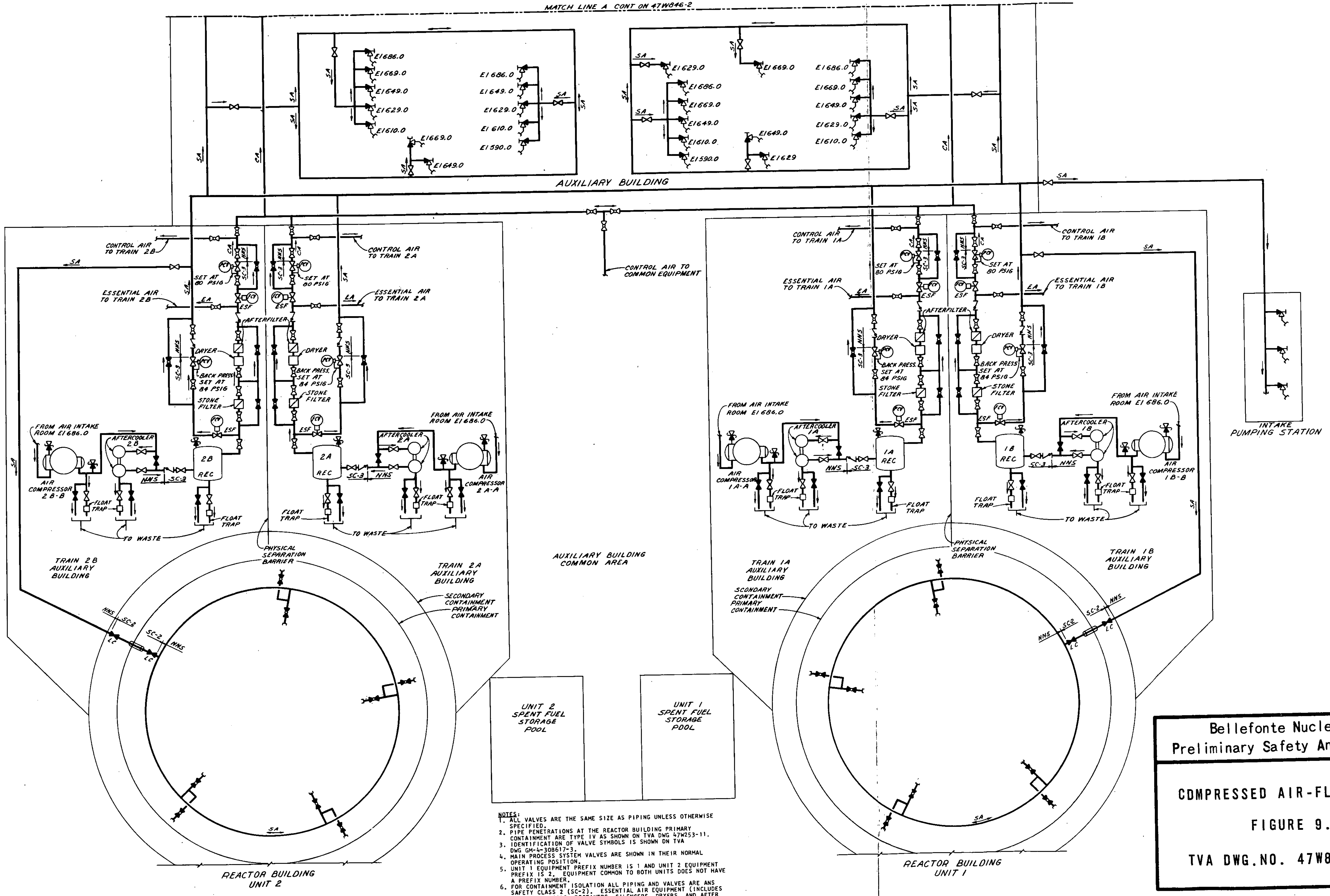
Table 9.3-6. (Cont'd)

Reactor coolant bleed evaporator demineralizer		
Quantity	2	
Type	Nonregenerative (boric acid-saturated, mixed bed)	
Material	SS	
Resin volume, ft^3	34	
Flow rate, gpm	35	11
Vessel design pressure, psig	250	
Vessel design temperature, F	200	
ASME Code/class	III/3	
Deborating demineralizer		
Quantity	6	
Type	Regenerative	
Material	SS	
Resin volume, ft^3	65	
Flow rate, gpm	100	
Vessel design pressure, psig	150	
Vessel design temperature, F	200	
ASME Code/class	III/3	
Pumps		
Reactor coolant distillate transfer pump		
Quantity	4	
Type	Centrifugal	
Capacity, gpm	200	
Head, ft	295	11
Design pressure, psig	150	
Design temperature, F	200	
ASME Code/class	III/3	
Material	SS	
Required NPSH, ft	25	
Reactor coolant bleed transfer pump		
Quantity	4	
Type	Centrifugal	
Capacity, gpm	50	
Head, ft	450	
Design pressure, psig	250	
Design temperature, F	200	
ASME Code/class	III/3	
Material	SS	
Required NPSH, ft	25	
Evaporator distillate test tank pump		
Quantity	2	
Type	Centrifugal	
Capacity, gpm	80	
Head, ft	230	

Table 9.3-6. (Cont'd)

Design pressure, psig	150
Design temperature, F	200
ASME Code/class	NA
Material	SS
Required NPSH, ft	25
RC bleed transfer pump flow indicators	
Quantity	2
Type	Rotometer
Capacity, gpm	0-10
Design pressure, psig	150
Design temperature, F	200
ASME Code/class	III/3
Material	SS



NOTES:

1. ALL VALVES ARE THE SAME SIZE AS PIPING UNLESS OTHERWISE SPECIFIED.
2. PIPE PENETRATIONS AT THE REACTOR BUILDING PRIMARY CONTAINMENT ARE TYPE IV AS SHOWN ON TVA DWG 47W253-11.
3. IDENTIFICATION OF VALVE SYMBOLS IS SHOWN ON TVA DWG GM-4-308617-3.
4. MAIN PROCESS SYSTEM VALVES ARE SHOWN IN THEIR NORMAL OPERATING POSITION.
5. UNIT 1 EQUIPMENT PREFIX NUMBER IS 1 AND UNIT 2 EQUIPMENT PREFIX IS 2. EQUIPMENT COMMON TO BOTH UNITS DOES NOT HAVE A PREFIX NUMBER.
6. FOR CONTAINMENT ISOLATION ALL PIPING AND VALVES ARE ANS SAFETY CLASS 2 (SC-2). ESSENTIAL AIR EQUIPMENT (INCLUDES AIR COMPRESSORS, RECEIVERS, SILENCERS, DRYERS, AND AFTER COOLERS). PIPING, AND VALVES ARE ANS SAFETY CLASS 3 (SC-3). ALL OTHER PIPING AND VALVES ARE NONNUCLEAR SAFETY CLASS (NNS).
7. SEE DWG 47W610-32-1 FOR INSTRUMATION

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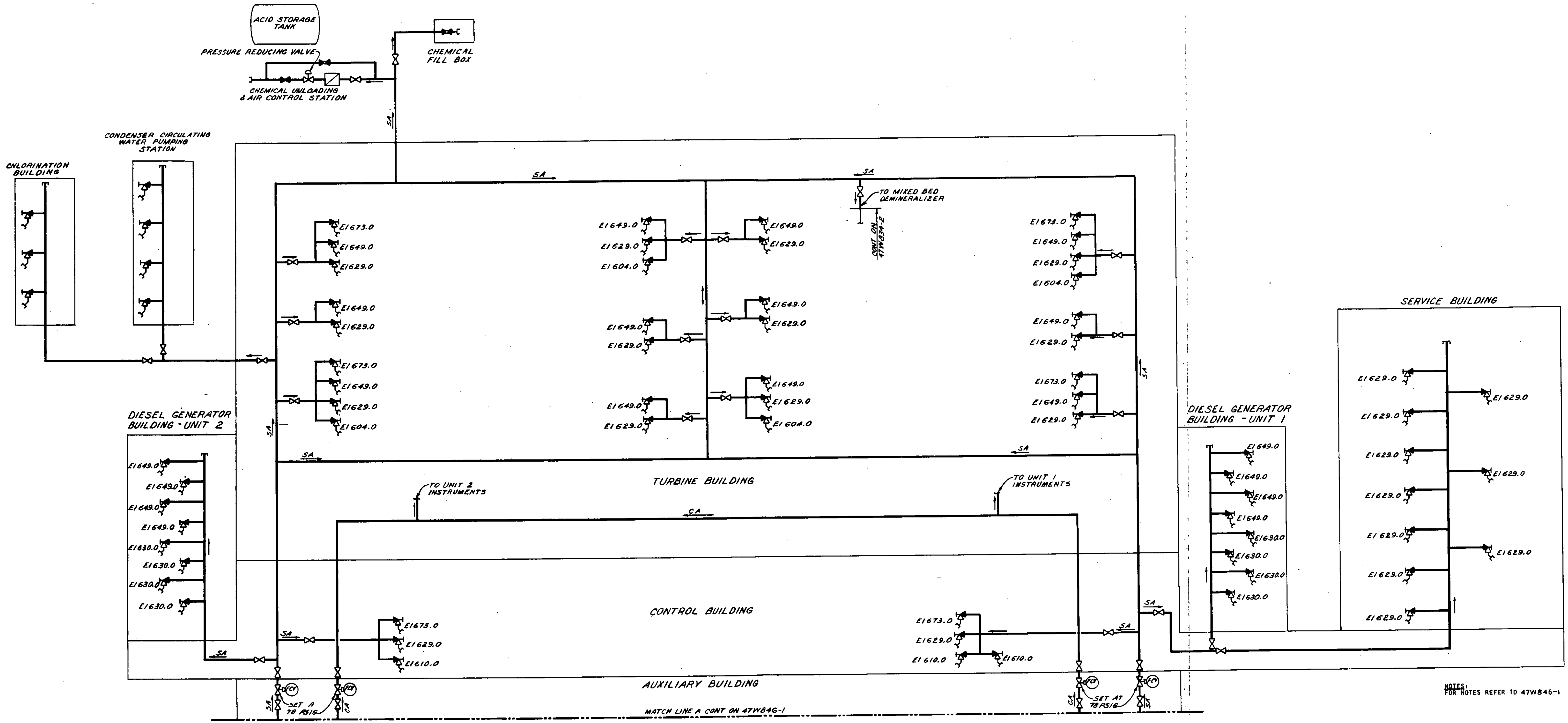
COMPRESSED AIR-FLOW DIAGRAM

FIGURE 9.3-1

TVA DWG.NO. 47W846-1 R0



NOTES:
FOR NOTES REFER TO 47W846-1

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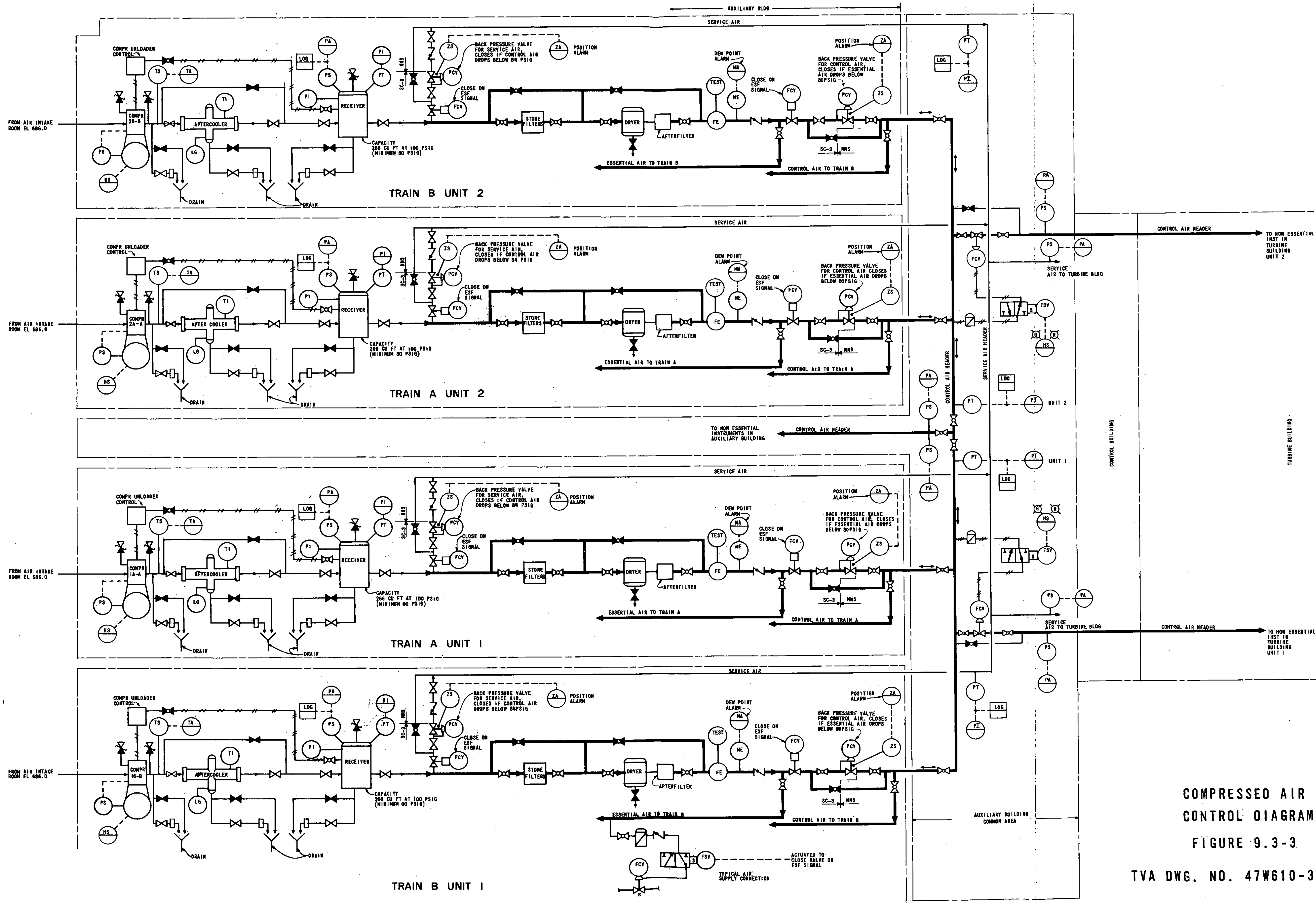
COMPRESSEO AIR - FLOW OIAGRAM

FIGURE 9.3-2

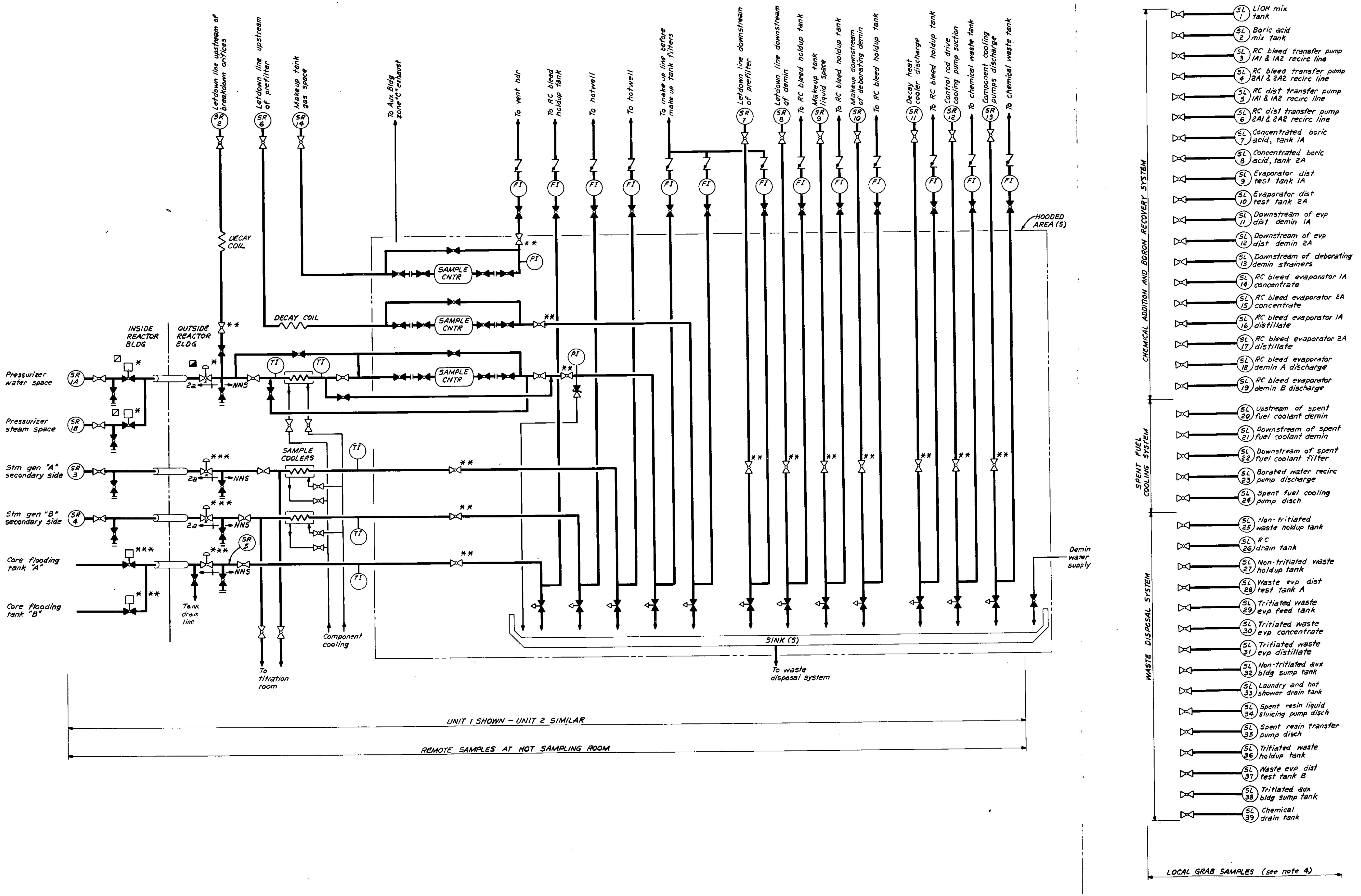
TVA OWG. NO. 47W846-2 RO

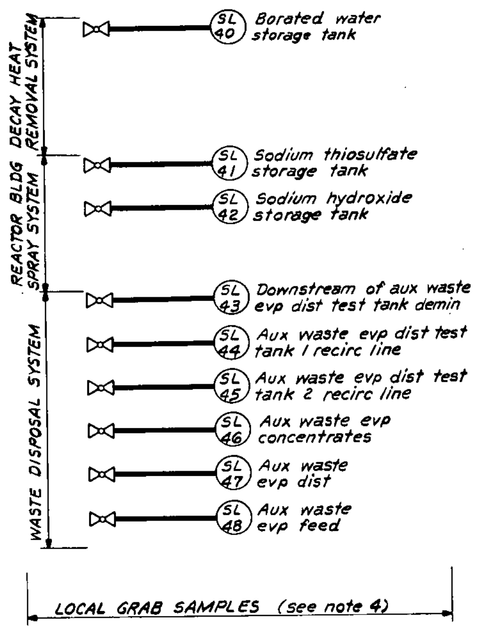


COMPRESSED AIR -
CONTROL DIAGRAM

FIGURE 9.3-3

TVA DWG. NO. 47W610-32-1 RO





NOTES:
1. * These isolating valves may be operated from remote sampling station but are subject to automatic closure by the engineered safe-guards systems.
2. ** These throttling valves to be set for proper flow and handle removed.
3. Piping and instrument symbols are shown in Appendix 7A.
4. Local grab samples SL-20, 21, 22, 23, 24, 26, 40, 41, and 42 are on unit basis; others are on common systems.
5. *** These valves may be operated from the main control room and from the hot sampling station.

LEGEND:

SR Sample remote

SL Sample local

Dist - Distillate

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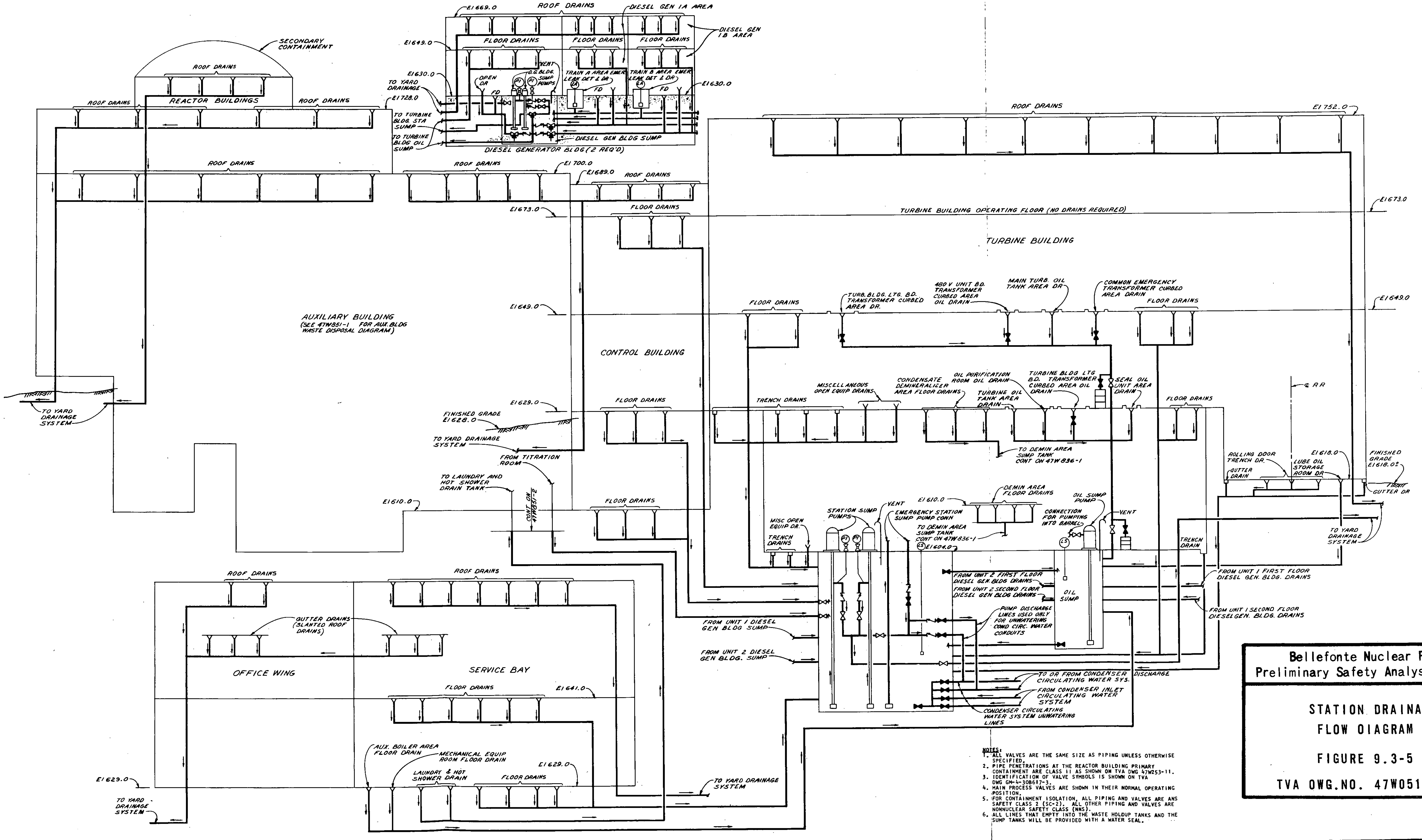
PROCESS SAMPLING-FLOW DIAGRAM

FIGURE 9.3-4

TVA DWG.NO. 47W861-1 R1



NOTES:
1. ALL VALVES ARE THE SAME SIZE AS PIPING UNLESS OTHERWISE SPECIFIED.
2. PIPE PENETRATIONS AT THE REACTOR BUILDING PRIMARY CONTAINMENT ARE CLASS II AS SHOWN ON TVA DWG 47W253-11.
3. IDENTIFICATION OF VALVE SYMBOLS IS SHOWN ON TVA DWG GM-4-30B617-3.
4. MAIN PROCESS VALVES ARE SHOWN IN THEIR NORMAL OPERATING POSITION.
5. FOR CONTAINMENT ISOLATION, ALL PIPING AND VALVES ARE ANS SAFETY CLASS 2 (SC-2). ALL OTHER PIPING AND VALVES ARE NONNUCLEAR SAFETY CLASS (NNS).
6. ALL LINES THAT EMPTY INTO THE WASTE HOLDUP TANKS AND THE SUMP TANKS WILL BE PROVIDED WITH A WATER SEAL.

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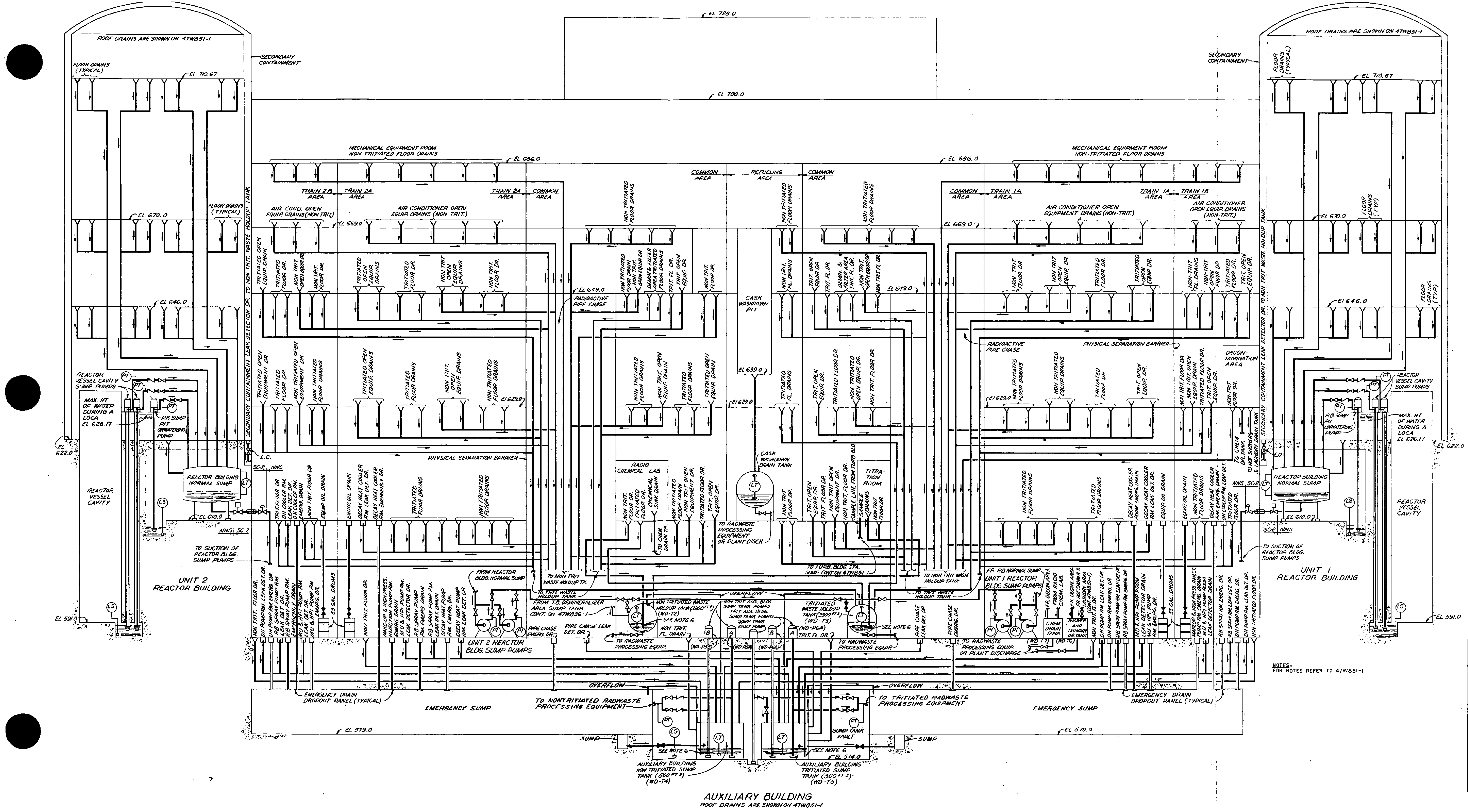
STATION DRAINAGE-
FLOW DIAGRAM

FIGURE 9.3-5

TVA DWG. NO. 47W051-1 R0



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STATION DRAINAGE-
FLOW DIAGRAM

FIGURE 9.3-6

TVA DWG.NO. 47W851-2 R0

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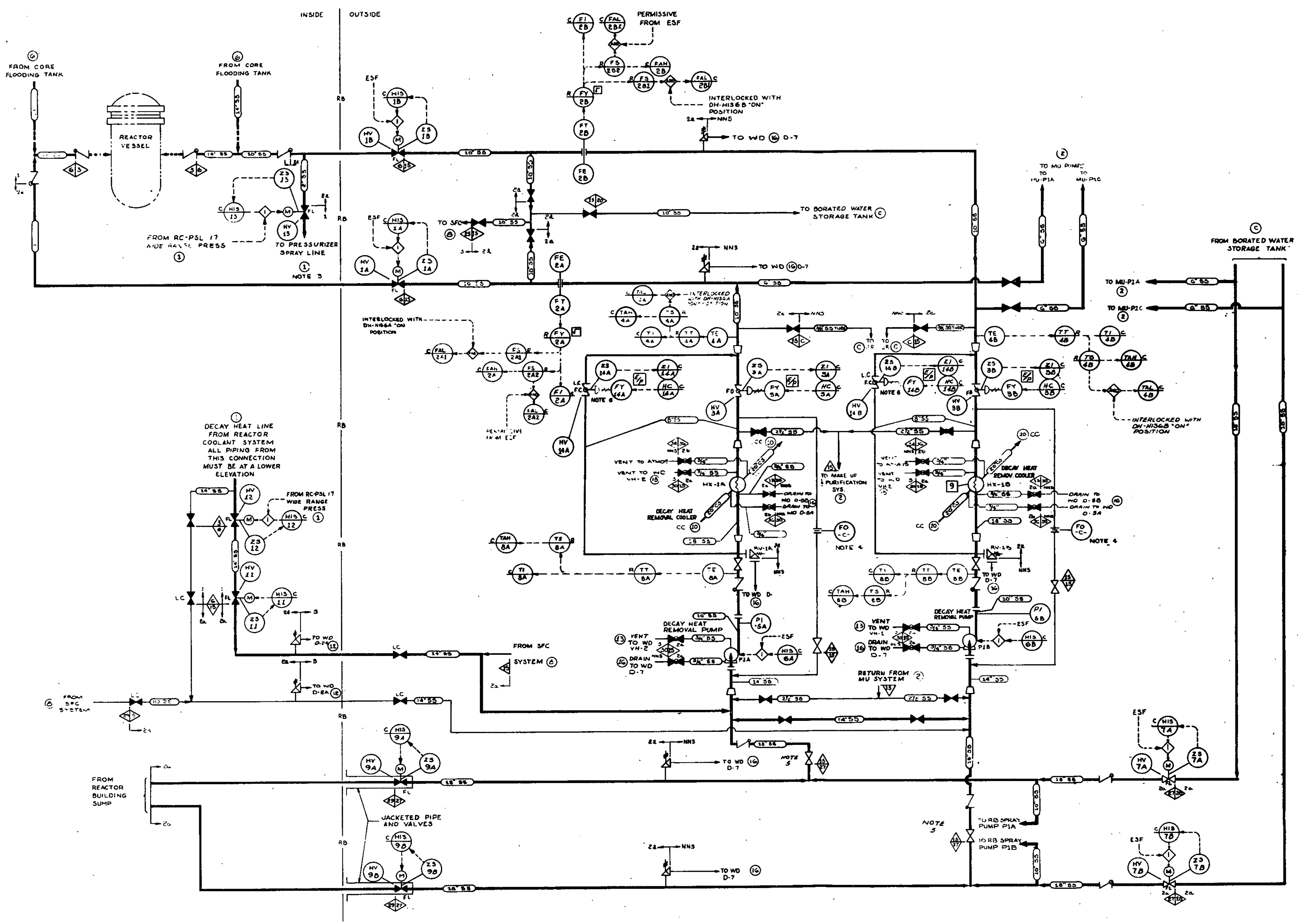
MAKEUP AND PURIFICATION SYSTEM

FIGURE 9.3-7

REVISED PER AMEND. 11, MAY 15, 1974



RESERVATIONS

1. PIPE SIZES CONTINGENT UPON FINAL PIPING CONFIGURATION.
2. PIPE SCHEDULE NUMBERS TO BE ESTABLISHED BY CUSTOMER.
3. CUSTOMER TO PROVIDE PROTECTION OF ISOLATED EQUIPMENT AGAINST PRESSURE BUILDUP DUE TO AMBIENT TEMPERATURE CHANGE.
4. FOR COMPLETE FILLING AND DRAINING OF THE SYSTEM ADDITIONAL VENTS AND DRAINS MAY BE NECESSARY DUE TO PIPING LAYOUT. FLUSH CONNECTIONS FOR INITIAL CLEANING OF THE SYSTEM TO BE DETERMINED BY THE CUSTOMER.
5. BUILDING ISOLATION VALVES ARE NOT NORMALLY INTENDED FOR THROTTLING SERVICE.

NOTES

1. ITEMS WITH COMPONENT NUMBERS SUPPLIED BY B&W.
2. INSTRUMENT AND VALVE NUMBERS 1-14 USED ON THIS DRAWING EXCEPT ED.
3. THIS SPRAY LINE SHOULD ORIGINATE FROM THE LP INJECTION LINE WHICH HAS THE HIGHER PRESSURE DROP. THE DRIVING HEAD FOR THE SPRAY FLOW IS THE PRESSURE DROP IN THE LP INJECTION LINE FROM THE SPRAY LINE CONNECTION TO THE REACTOR VESSEL. THE DESIRED FLOW RATE IS APPROXIMATELY [illegible] GPM.
4. MINIMUM DH PUMP RECIRCULATION FLOW RATE SHOULD BE [illegible] GPM; THE ORIFICE ΔP SHOULD BE 300 FT AND THE ORIFICE LEADS MUST BE SIZED TO ABSORB THE REMAINDER OF THE PUMP SHUTOFF TDH. WITH THE RECIRCULATING LINE OPEN THE DH PUMP MUST DELIVER 5000 GPM AT A REACTOR VESSEL PRESSURE OF 100 PSIG.
5. THIS VALVE (VALVES) MUST BE LOCKED CLOSED BEFORE OPENING HV11 AND HV12.
6. THIS VALVE (VALVES) MUST BE LOCKED CLOSED DURING NORMAL REACTOR POWER OPERATION. COOLER BYPASS UTILIZED DURING COOLDOWN ONLY.

- (1) RC SYSTEM
- (2) MU SYSTEM
- (6) CF SYSTEM
- (7) RBS SYSTEM
- (5) SFC SYSTEM
- (10) CC SYSTEM
- (12) WD SYSTEM
- (13) WD SYSTEM
- (15) CA & BR SYS
- (16) WD SYSTEM
- (C) CUSTOMER SYSTEM

- (3) 2500 PSIG @ [illegible]°F
- (4) 2500 PSIG @ [illegible]°F
- (6) 2500 PSIG @ 300°F
- (15) 875 PSIG @ 350°F
- (16) 600 PSIG @ 200°F
- (20) 200 PSIG @ 300°F
- (25) 200 PSIG @ 250°F
- (27) 75 PSIG @ 300°F
- (28) 75 PSIG @ 150°F
- (29) BUILDING DESIGN PRESS @
- (30) 50 PSIG @ 250°F
- (35) ATMOS PRESS @ 200°F
- (36) 200 PSIG @ 200°F
- (C) CUSTOMERS SYSTEM DESIGN

Bellefonte Nuclear Plant
Preliminary Safety Analysis Report

DECAY HEAT REMOVAL SYSTEM

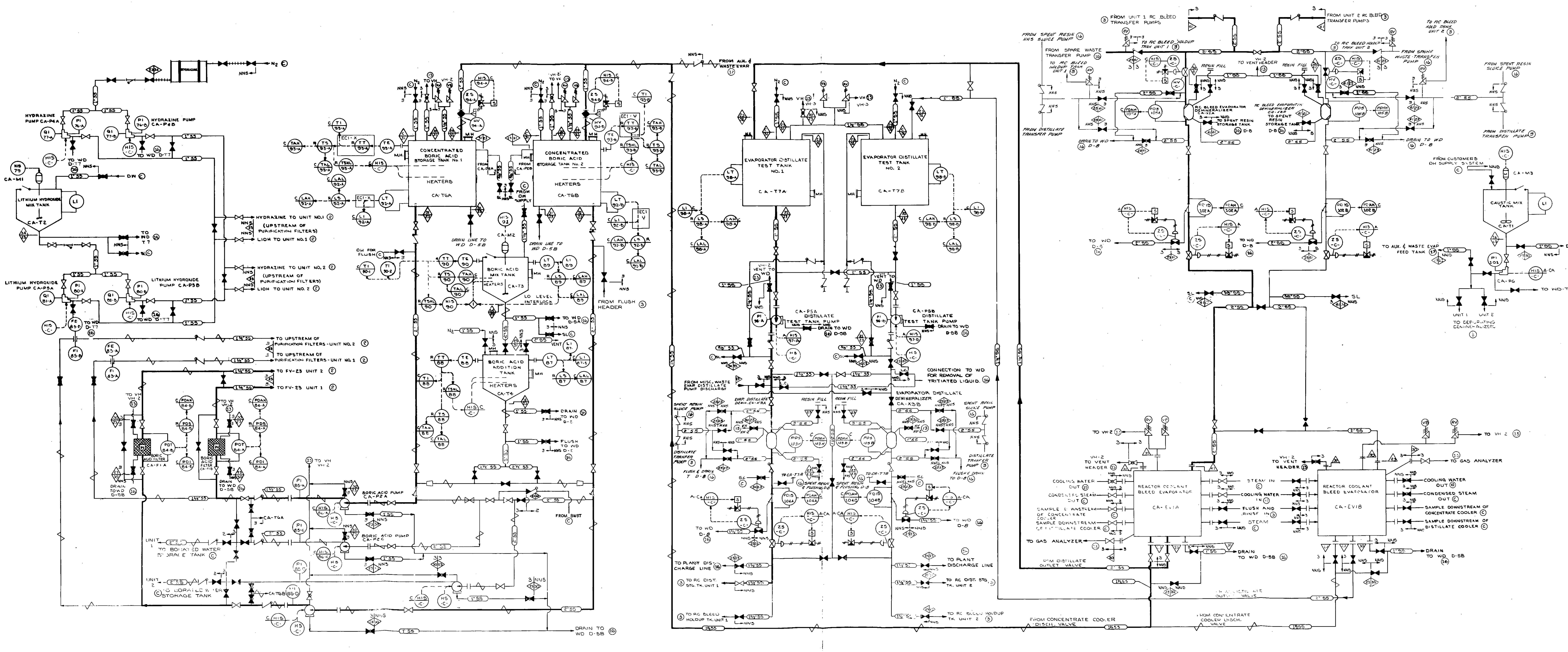
FIGURE 9.3-8

Bellefonte Nuclear Plant
Preliminary Safety Analysis Report

CHEMICAL ADDITION AND BORON RECOVERY SYSTEM

FIGURE 9.3-9

REVISED PER AMEND. 11, MAY 15, 1974



RESERVATIONS:

1. PIPE SIZES CONTINGENT UPON FINAL PIPING CONFIGURATION.
2. PIPE SCHEDULE NUMBERS TO BE ESTABLISHED BY CUSTOMER.
3. CUSTOMER TO PROVIDE PROTECTION OF ISOLATED EQUIPMENT AGAINST PRESSURE BUILDUP DUE TO AMBIENT TEMPERATURE CHANGE.
4. FOR COMPLETE FILLING AND DRAINING OF THE SYSTEM, ADDITIONAL VENTS AND DRAINS MAY BE NECESSARY. FLUSH CONNECTION LOCATION FOR INITIAL CLEANING OF THE SYSTEM TO BE DETERMINED BY THE CUSTOMER

NOTES:

1. ITEMS WITH COMPONENT NUMBERS SUPPLIED BY B&W
2. INSTRUMENT AND VALVE NUMBERS 76-105 USED ON THIS DRAWING.
3. ALL EQUIPMENT SHOWN ON THIS DRAWING SHARED BY BOTH UNITS.

2 MU SYS.
3 CA & BR SYS
4 DH SYS
10 CC SYS.
12 WD SYS
13 WD SYS
C CUSTOMER SYS.
17 WD SYS.

DESIGN CONDITIONS

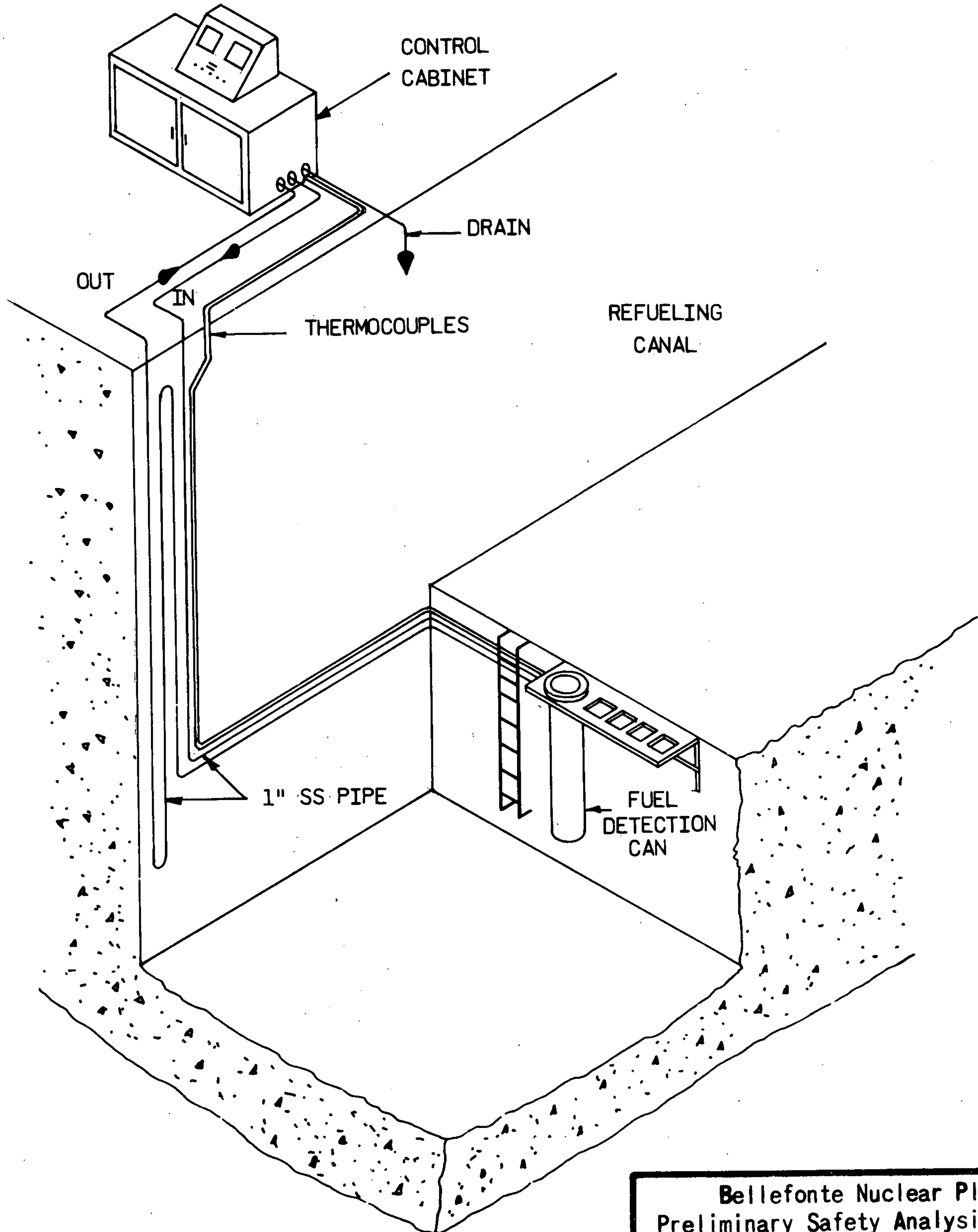
23 150 PSIG @ 200° F
30 50 PSIG @ 200° F
31 4 PSIG @ 200° F
34 ATMOS @ 200° F
61 250 PSIG @ 200° F

Bellefonte Nuclear Plant
Preliminary Safety Analysis Report

CHEMICAL ADDITION AND BORON RECOVERY SYSTEM

FIGURE 9.3-10

REVISED PER AMEND. 11, MAY 15, 1974



Bellefonte Nuclear Plant
Preliminary Safety Analysis Report

TYPICAL LOCATION OF FAILED
FUEL SIPPING SYSTEM

FIGURE 9.3-11

9.4. Air Conditioning, Heating, Cooling, and Ventilation

9.4.1. Control Building

9.4.1.1. Background

The control building will contain the main control room, or "nerve center", which is used to remotely operate the plant. In addition, it will contain offices, shops, service rooms, electrical board rooms, locker rooms, toilet rooms, and kitchen.

The building will be located between the area occupied by the reactor and auxiliary buildings and the area occupied by the turbine building. The control building will contain four floors, referred to by their elevations expressed in feet. For instance, the first floor is at EL 610, the second floor at EL 629, the third floor at El 649, and the fourth floor at El 673.

The control building air-conditioning systems shall be designed to maintain the temperature and humidity in the main control room and the various other rooms in the control building within acceptable limits for the operation of plant controls, for the maintenance and testing of the controls as required, and for the uninterrupted safe occupancy of the main control room during an accident and the period following.

The building pressurizing air supply system shall be designed to normally maintain the control building at a positive pressure relative to the outdoor pressure or to the adjoining buidling.

For the conditions during and following an accident, the main control room will be isolated and maintained at a slight positive pressure relative to the outdoor pressure.

All main control room equipment will operate normally within the rated tem-
perature range of 50F to 104F. At temperatures above 104F, failure rates for
this control room equipment will tend to rise somewhat and some instrumenta-
tion inaccuracies may arise. The full capacity air conditioning system re- 1
dundancy discussed above, however, reduces the probability of over-tempera-
ture operations to negligible values. Loss of ventilation problems are 7
discussed further in Section 3.11.4.

9.4.1.2. Design bases

The control building air-conditioned spaces shall be maintained at approximately 75F and 50% relative humidity for the protection of instruments and controls and for the comfort and safety of the operators. These conditions shall be continuously maintained during all modes of normal or accident plant operation and during outdoor design conditions of 97F DB and 79F WB in summer and 5F in winter.

During normal plant operation, a continuous stream of fresh air will be mechanically supplied to the building to replace that exhausted, to maintain the building at a slight positive pressure relative to the outdoor pressure to minimize the possibility of air inleakage.

Isolation of the main control room will automatically occur upon the actuation of an ESFAS from either unit or upon indication of high radioactivity, high 11
temperature, high chlorine or smoke concentrations in the outside air supply stream to the building. The air intake will be equipped with radiation monitors and chlorine and smoke detectors that will indicate and annunciate in the main control room. Upon isolation, the following conditions will exist:

1. The control room emergency air cleanup supply fans will operate to recirculate a portion of the control room air-conditioning system air through the cleanup trains composed of HEPA filters and charcoal absorbers.

2. The control room emergency presurizing air supply fans will operate to supply a reduced stream of outside air to the control room air-conditioning system to keep the main control room pressurized relative to the outdoors and relative to the other floors to minimize the inleakage of unprocessed or contaminated air. However, if the isolation is initiated by the chlorine detectors the entry of makeup air into the main control room will be stopped. Under such circumstances a portion of the recirculated control 9
room air will be processed through the air cleanup unit until a manual switch is operated to start the emergency ventilation mode used for the other accidents.

3. The building normal pressurizing air supply fan in use prior to the accident will continue to operate to supply outside air to the control building lower floors only.

4. The toilet rooms exhaust fan will be stopped and dampers closed to isolate the main control room floor.

5. The spreading room supply and exhaust fans and the battery room exhaust fans will continue to run.

Isolation of the main control room may also be manually initiated at any time by the control room operators.

The following building air-conditioning and ventilating system components shall each be provided with two 100% capacity units:

1. Control room and electrical board room air-conditioning systems, water chillers, air handling units, chilled water piping, and pumps.

2. Building pressurizing air supply fans.

3. Control room emergency air cleanup supply fans and filter assemblies.

4. Control room emergency pressurizing air supply fans.

Double isolation dampers will be provided in each exhaust duct connection from the main control room floor to the outdoors.

All essential air-conditioning and ventilating equipment and systems including isolation dampers will be designed to withstand the Safe Shutdown Earthquake.

The Bellefonte Nuclear Plant heating, ventilating and air conditioning systems are designed to maintain the proper temperatures within the plant in all weather situations. To assure this capability, outside weather conditions for weather extremes in the plant area are used to size this equipment. These design bases climatic conditions are as follows:

Winter median of annual extremes, F dry bulb	6
Winter design temperature, F dry bulb	10
Winter relative humidity, maximum, %	100
Coincident wind velocity	Light
Summer design conditions	
Dry bulb, F	97
Wet bulb, F	78

1

In December through February, 99 percent of hourly readings have been equal to or above the winter design temperature. During the months of June through September, 99 percent of hourly readings have been equal to or below the summer design dry bulb temperature, and 99 percent have been equal to or below the summer design wet bulb temperature.

9.4.1.3. System description

The control building heating, ventilating, air-conditioning, and air cleanup systems are shown on Figure 9.4-1, and will consist of the following systems:

1. Air-conditioning systems.
2. Building air pressurizing system.
3. Control room emergency air cleanup systems.
4. Control room emergency air pressurizing system.
5. Battery rooms ventilating system.

6. Miscellaneous ventilating systems.

9.4.1.3.1. The main control room and the electrical board rooms air-conditioning systems, water chillers, and chilled water pumps will be located in separate compartments of the mechanical equipment room at El 610. The electrical board rooms system air handling units will also be located in separate compartments at El 610. The main contol room system air handling units will be located in separate compartments of the mechanical equipment room at El 629.

Each of the above two air-conditioning systems will be provided with two 100% capacity water chilling units, air handling (fan-coil) units, chilled water pumps, and chilled water piping arranged in two parallel 100% capacity combinations. Each air-conditioning system shall also be provided with an assemblage of air supply and return ducts, dampers, heaters, grilles, and controls.

All air, fresh and recirculated, will be filtered by passing through filter banks having a National Bureau of Standards (NBS) dust spot efficiency rating of approximately 85%. Two 100% capacity filter banks shall be provided for each air-conditioning system, one per air handling unit, for a total of four for the building. This arrangement will provide for one filter bank per air-conditioning system to be in continuous service while the other bank is on standby and available for servicing.

Electric heaters will be mounted in air supply ducts serving rooms requiring heat, and they shall each be thermostatically controlled to maintain design room conditions.

9.4.1.3.2. The building air pressurizing system will be located in separate compartments within the mechanical equipment room at El 649.

All fresh air for the control building will be taken from the outdoors through either of two intake hoods located on the building roof at El 689 with one each located near the north and south ends of the building.

Each of these fresh air ducts will contain monitors to assess the quality of the incoming air. Redundant smoke and temperature sensors will be provided in the normal air intake ducting to measure steam and smoke concentrations that may be drawn in during normal operations. Redundant radiation monitors will also be provided to measure the amount of airborne radiation entering the normal air intake ducting and the emergency air intake ducting. Outputs from this instrumentation above specific thresholds will initiate an automatic transfer of the control building heating, ventilating, air-conditioning and air cleanup system from the normal to the emergency mode of operation. 1

The building air pressurizing system will contain two 100% capacity supply fans. During normal operation, the building will be pressurized relative to the outdoors with fresh air to minimize air inleakage. The quantity of air supply shall be automatically controlled as required to maintain approximately 1/8-inch water positive static pressure in the building relative to the outdoors.

Special doors, latches, and seals and/or the protective enclosure will be utilized at all exits from the El 673 floor level to assure a capability for maintaining a positive pressure within the main control room. All of these will be sufficiently strong to resist any differential pressures that may arise. Those leading to the turbine building will have adequate structures and door latches and/or will be protected by an appropriate enclosure to withstand loads that may arise in steam pipe break accidents and during a tornado. 1

9.4.1.3.3. The main control room emergency air cleanup system will be located in separate compartments in the mechanical equipment room near the north end of the El 649 floor.

This system shall be provided with two 100% capacity emergency air cleanup supply fans and two 100% capacity air cleanup filter assemblies arranged in two parallel 100% capacity fan-filter trains. Each air cleanup filter assembly shall consist of a bank of prefilters followed by a bank of HEPA filters which will be followed by a bank of 4 charcoal absorbers enclosed within a plenum. Each plenum shall be provided with static pressure differential indicators, thermometers, connections for inplace testing of filters and absorbers, and access doors for filter and absorber maintenance.

This system will automatically operate upon an accident signal or upon indi-
cation of high radioactivity, high temperature, high chlorine or smoke con- 4
centrations in the building fresh air supply. This system may also be manu-
ally started from the main control room at any time.

Upon main control room isolation, the system will operate to automatically start both of the emergency air cleanup supply fans and will stop the toilet room exhaust fan. Double isolation dampers in each duct connection to the outdoors will automatically close.

The system shall operate to route a portion of the control room air-conditioning system return air through both of the HEPA filter-charcoal adsorber trains. The cleaned air will thus be recirculated to the main control room by the air-conditioning system.

To prevent condensation of moisture on the charcoal at system start, an electric heater will operate to continuously maintain the charcoal temperature above 85F during system shutdown. Upon system start, the charcoal temperature will thus be higher than the possible entering air wet bulb temperature. These heaters will automatically shut down upon system start.

The system may be manually operated from the main control room at any time upon the indication of smoke or gaseous contamination on the control room floor.

<u>9.4.1.3.4.</u> The main control room emergency air pressurizing supply system will contain two 100% capacity supply fans located in separate compartments within the mechanical equipment room near the north end of the El 649 floor.

Fresh air will be supplied by either or both fans from a separate air intake
for each fan. These two air intakes will be located approximately at El 689 1
with one at the north end of the building and the other at the south end of
the building. These intakes will be approximately 320 feet apart.

Fresh air will be taken from either of both of two air intake hoods provided near each end of the control building roof.

During main control room isolation, the emergency air pressurizing supply fans will automatically run to supply a reduced flow of fresh air to the control room air-conditioning system return air, to a point just upstream of the emergency air cleanup filter trains. The mixture of return and fresh air will pass through the filter train, and the cleaned air will be supplied to the control room air-conditioning system. The control room will thus be slightly pressurized relative to the outdoors or adjoining spaces to minimize the inleakage of unprocessed air.

During control room isolation, the normal air pressurizing supply system fan will continue to supply fresh air to the electrical board rooms air-conditioning system.

9.4.1.3.5. The battery rooms and the battery board rooms ventilation system will consist of two 100% capacity supply fans and two 100% capacity exhaust fans located in the mechanical equipment room at El 610. Filtered fresh air will be mechanically supplied to the battery board rooms from which it will pass through the corridor to the battery rooms and be mechanically exhausted to the outdoors. Fire dampers, provided in each room's air supply duct, exhaust duct, and wall opening, will automatically close to isolate the room upon high temperature. The battery rooms and battery board room ventilation system shall be required to operate when the plant is under normal operating conditions but not under accident conditions. 7

9.4.1.3.6. The spreading room will be ventilated by two 100% capacity supply fans located in the mechanical equipment room near the north end of the El 649 floor, and by two 100% capacity exhaust fans located within the spreading room. The supply air is taken from the mechanical equipment rooms. The spreading room will be normally maintained at a slight negative pressure relative to the adjoining air-conditioned spaces.

The mechanical equipment rooms at El 610 and 629 will be ventilated at all times by routing a portion of the electric board rooms air-conditioning system return air through the rooms to the spreading room supply fan system.

During normal operation the kitchen, toilet, and locker rooms at El 673 will be ventilated by exhausting a portion of the main control room air-conditioning system return air through the rooms. The toilet and locker rooms roof ventilator exhaust fan will be located in the roof and will discharge to the outdoors.

9.4.1.4. Safety evaluation

9.4.1.4.1. The main control room air-conditioning system and the electric board rooms air-conditioning system are essential equipment and will each be served by two separate full-capacity air cooling assemblies. Each assembly shall consist of a water-coded condenser-compressor water chiller, an air handling (fan-coil) unit, chilled water pump and piping, and a steam humidifier.

The control building air-conditioning systems will be of engineered safety features (ESF) quality. Redundant full-capacity compressors, pumps, and air handling units will be served from separate trains of the emergency power system and essential raw cooling water system.

Redundant full-capacity equipment trains shall be separated by being installed in individual fire-resistant and missileproof compartments.

A filter bank will be provided in the inlet air plenum to each of the four air handling units. Each filter bank will contain filter cells rated at approximately 85% efficiency based on NBS dust spot method. Each filter bank will be provided with a static pressure differential indicating gauge.

9.4.1.4.2. The two building pressurizing air supply fans will be of ESF quality and will be connected to separate trains of emergency power.

9.4.1.4.3. The two main control room emergency air cleanup fans (one redundant) will be of ESF quality and shall be connected to separate trains of the emergency power system.

The main control room emergency air cleanup system was designed to be in general accordance with the guidelines set forth in Regulatory Guide 1.52, "De-
sign, Testing, and Maintenance Criteria for Atmospheric Cleanup System Air 9
Filtration and Adsoprtion Units of Light-Water-Cooled Nuclear Power Plants." A tabulation showing the degree of compliance with this regulatory guide is shown in Table 9.4-2.

Each of the main control room emergency air cleanup filter trains will consist
of a bank of prefilters to remove larger-sized particulates, followed by a 4
bank of HEPA filter cells rated at 99.9% efficiency based on 0.30-micron DOP test, mounted in series with a bank of carbon adsorber modules rated at 99.99% efficiency for removal of elemental iodine and 95.0% efficiency for removal of methyl iodide. Each HEPA filter cell will be rated for an initial resistance of 1.0-inch water gauge when clean and will be replaced with a new filter cell upon an increase in resistance to 2.0 inches. Each HEPA filter bank will be provided with a static pressure differential indicating gauge. An electric heater will maintain the charcoal temperature during system shutdown at a temperature above the possible entering air wet bulb temperature upon system startup.

9.4.1.4.4. The two main control room emergency pressurizing air supply fans are of ESF quality and shall be connected to separate trains of the emergency power system.

9.4.1.4.5. Two isolation dampers will be mounted in series in each of the control room floor toilet rooms exhaust duct to the outdoors, and in the control room floor pressure relief duct to the outdoors. These dampers will be designed to low-leakage, seismic Category I requirements. Each pair of dampers will be served from separate trains of the emergency power system and each damper will fall in the closed position.

9.4.1.4.6. The spreading room supply and exhaust fans have no safety-
related task to perform. Therefore, these fans will not be connected to emer- 11
gency power.

9.4.1.4.7. The battery board rooms supply and exhaust fans are not ESF equipment. The battery board room air supply filters will be rated at approximately 85% efficiency based on the NBS dust spot method.

9.4.1.4.8. The locker and toilet rooms exhaust fan will not be connected to emergency power.

9.4.1.4.9. An electric steam generator will be provided for air-conditioning system humidity control. Humidifiers, mounted in each air-conditioning system air handling unt, will automatically supply steam to the conditioned air stream in response to a humidistat for each system. The control room system humidistat will be a wall mounted in the main control room, and the electric board rooms system humidistat will be mounted in the system return air duct. The steam generators will not be ESF equipment and will not be connected to emergency diesel power.

9.4.1.4.10. Several features incorporated into the main control room
ventilation system help minimize the hazard from an accidental release of toxic 1
chlorine or acrolein gas. A listing of the beneficial features will include:

a. Air intake locations about 60 feet above the ground. Such a feature minimizes the concentration at the normal air intakes of heavier-than-air gases such as acrolein and chlorine. These gases will tend to settle close to the ground.

b. The isolation of normal air intakes upon detection of high chlorine con-
centrations. Emergency air pressurization is not started when there is 9
high chlorine concentration outdoors.

c. The air distribution system for the main control room that has redundant air cleanup equipment for processing recirculated air. This feature provides a capability for removing any acrolein or chlorine which could somehow enter the main control room.

These features and the fact that only a small amount of acrolein will be stored
onsite at a considerable distance from the control building air intakes reduce
the toxic gas hazard. Table 9.4-1 identifies the acrolein storage positions,
lists the quantity that may be stored at each location, and gives the distance 12
between the storage locations and both of the control room air intakes.

No chlorine is stored onsite. Chlorination requirements will be satisfied as needed through the use of a hyperchlorite system. This system does not produce free chlorine in the gaseous state.

Chlorine detectors capable of detecting 1 ppm of chlorine or more are located
in the control bay air intakes. These detectors initiate automatic isolation 7
and the capabilities b and c listed above. No detectors or alternate
means of detection are provided for acrolein. As seen in Table 9.4-1, only a small amount of acrolein is located on site; it is kept at a very large distance from the control room. Also, the evaporation rate of this chemical is moderate at normal temperatures and pressures due to high vapor pressure, but it is in liquid form at normal temperatures. Hence, the effects from accidental releases of acrolein are insignificant. No other toxic gases are located within the site boundary.

9.4.1.5. Test and Insprection Requirements

The control building air-conditioning systems will be in continuous operation and will be accessible for periodic inspection. Essential electrical components, switchovers, and starting controls shall be tested initially and periodically.

The building pressurizing air supply system will be in continuous operation, and the fans will be accessible for periodic inspection.

The control room emergency air cleanup system fans and filter assemblies, and the emergency pressurizing air supply fans will be tested peiodically. The HEPA filters will be tested in place initially and periodically with dioctylphthalate (DOP), and the charcoal adsorbers will be tested initially and periodically with Freon 112. Charcoal surveillance specimens will be periodically evaluated to assure iodine adsorptivity.

Air radiation monitors, thermostats, and smoke detectors shall be each calibrated and tested periodically using a calibrated check source to verify the instruments response and alarm functions.

The battery rooms and spreading room ventilating system fans shall be accessible for periodic inspection.

Environmental conditions will vary only slightly under normal operation, and simulated conditions should be such as to prove design conditions.

9.4.2. Auxiliary Building

9.4.2.1. Design bases

The auxiliary building air-conditioning, heating, cooling, ventilating, and air cleanup systems will be designed to maintain an acceptable building environment for the protection of plant equipment and controls; for the comfort and safety of operating personnel; to allow personnel access for the operation, inspection, maintenance, and testing of mechanical and electrical equipment; and to limit the release of radioactivity to the atmosphere.

The building environmental control systems will be designed to maintain building temperatures in the main occupied areas between 50F minimum and 104F maximum during outdoor temperatures ranging from 5F in winter and 97F in summer, during all modes of plant operation including postaccident conditions. Maximum ambient temperatures in engineered safety features (ESF) and other safety equipment rooms or cubicles will not be allowed to exceed 120F.

To control the release of airborne radioactive material, filtered ventilating air will be supplied to clean areas, then routed to areas of progressively greater contamination possibilities. Areas of the building, which have potential for radioactive contamination, will be maintained at a slight negative pressure relative to the outdoor pressure, and exhaust air will be processed by high efficiency particulate filters and charcoal adsorbers before being released to the atmosphere. All building exhausts will be discharged to the outdoors by means of an exhaust stack provided for each plant unit and extending approximately 25 feet above the reactor building parapet wall.

In order to reduce the quantity of building air exhaust and its attendant filtering equipment, potentially contaminated areas will be ventilated for contamination control. Air coolers will be provided for removal of heat from mechanical and electrical equipment in excess of that provided by ventilation.

Building environmental control system fans, air coolers, and air-conditioning 9
units, essential to the operation of safety-related equipment, will be
assigned to redundant equipment trains having separate emergency power and
raw cooling water sources.

For environmental control and isolation purposes, the auxiliary building
will be considered to be divided into separately controlled and isolated 11
zones as follows:

1. Mechanical equipment zone 1A (Unit 1 - Power train A).
2. Mechanical equipment zone 1B.
3. Mechanical equipment zone 2A.
4. Mechanical equipment zone 2B.
5. Essential electrical equipment zone 1A
6. Essential electrical equipment zone 1B.
7. Essential electrical equipment zone 2A.
8. Essential electrical equipment zone 2B.
9. Units 1 and 2 fuel-handling zones.
10. Common equipment zone.

All of the above temperature control and/or air cleanup systems except the
common equipment zone ventilation and air cleanup system have safety-related
functions to perform during accidents. Therefore, these are considered to be
engineered safety feature systems. Each of these systems is designed to the 11
requirements of seismic Category I.

9.4.2.2. System description

9.4.2.2.1. The auxiliary building ventilating, cooling, heating, air-conditioning, and air cleanup systems are shown on Figure 9.4-2. They will consist of the following subsystems:

1. Building air supply systems.
2. Building air exhaust and air cleanup systems.
3. Engineered safety feature (ESF) equipment and essential electrical equipment coolers.
4. Auxiliary control room air-conditioning system.
5. Miscellaneous ventilating and air-conditioning systems.

The building supply and exhaust fans, air supply heating and filtering assemblies, and exhaust air cleanup filter trains will be located in the fan rooms at El 686. Air cooling units will be provided for rooms or cubicles containing engineered safety feature equipment. Air-conditioning units will be provided for essential electrical equipment rooms.

9.4.2.2.2. Building Air Supply Systems — Outdoor air will be mechanically supplied to each building zone from five fan rooms located at El 686. An air intake room, located within each fan room 1A, 1B, 2A, and 2B, will contain air intake louver, air filter bank, heating coils, two 100% capacity supply fans for the mechanical equipment rooms, and two 100% capacity supply fans for the essential electrical equipment rooms. The common area supply fan room will contain air intake louvers, air filter banks, heating coils, two 100% capacity supply fans for the Unit 1 fuel-handling area, two 100% capacity supply fans for Unit 2 fuel-handling area, and four 50% capacity supply fans for the common equipment areas. Outdoor air will be taken from above the control building roof for air intake rooms 1A and 2A and the common area supply fan room. Air will enter the building north wall for air intake room 1B and through the building south wal for air intake room 2B. Outdoor air supply filters for each air intake will have a nominal efficiency of approximately 85% based on NBS atmospheric dust spot test.

During periods when the outdoor ambient air temperature is below 60F, thermostatically controlled hot water heating coils will be used to temper the incoming air.

9.4.2.2.3. Building Air Exhaust Systems — The mechanical equipment room, fuel-handling area rooms, common area equipment rooms, and essential electrical equipment rooms will be exhausted to the building exhaust stacks by fans located in the fan rooms at El 686. Except for the essential electrical equipment rooms, all exhaust air will normally be routed through air cleanup filter trains before being discharged to the stack.

Two 100% capacity exhaust fans and one air cleanup filter assembly for the mechanical equipment rooms, and two 100% capacity exhaust fans for the essential electrical equipment room will be provided in each fan room 1A, 1B, 2A, and 2B.

Four 50% capacity exhaust fans and two 50% capacity air cleanup filter assemblies will be provided in the common area exhaust fan room. Two 100% capacity exhaust fans and one air cleanup filter assembly for the Unit 1 fuel-handling area will be located in the Unit 1 purge exhaust room and two 100% capacity exhaust fans and one air cleanup filter assembly for the Unit 2 fuel-handling area will be located in the Unit 2 purge exhaust room.

Each air cleanup filter assembly will consist of a demister which will also serve as a prefilter, followed by a bank of HEPA filter cells which will be in series with a bank of charcoal adsorbers and a second bank of HEPA filters. 9

Each air cleanup filter assembly will be provided with static pressure differential indicators, thermometers, connections for inplace testing of filters and adsorbers, and access doors for filter and adsorber maintenance.

9.4.2.2.4. Engineered Safety Equipment and Essential Electrical Equipment Coolers — Cubicles or areas containing engineered safety equipment will be provided with air cooling units in addition to the normal ventilating systems. These coolers will be designed to limit the rooms maximum ambient temperature to 120F during equipment operation, and will automatically operate to provide cooling whenever the equipment is started. A thermostat will assure that the cooler will remain in operation until the low limit temperature setpoint is reached.

Packaged air-conditioning units will be provided for cooling the essential electrical equipment rooms in addition to the cooling normally provided by the ventilating system. These units will be sized to limit the room's maximum ambient temperature to 104F and each unit will be thermostatically controlled..

Engineered safety equipment coolers and essential electrical equipment room air-conditioning units will each operate from the same redundant power train and essential raw cooling water source assignments as the equipment they serve.

9.4.2.2.5. Auxiliary Control Room Air-Conditioning System

The auxiliary control room will be maintained at 75F and 50% relative humidity year around by either of two 100% capacity air-conditioning units each provided with pressurizing air supply fan. These units will be located in fan room at El 686, and outdoor air for pressurizing will be taken from the air intake rooms.

Air-conditioning units cooling water will be supplied from the emergency raw cooling water systems and the units will be fed from separate trains of the emergency power.

9.4.2.2.6. Miscellaneous Ventilating and Air-Conditioning Systems

The battery rooms will be adequately ventilated to prevent the possible accumulation of hydrogen gas. Two battery rooms, located in each fan room 1A, 1B, 2A, and 2B, will be ventilated together by either one of two 100% capacity exhaust fans discharging to the outdoors by means of the exhaust stack. Air will be allowed to endter each battery room through wall openings from the fan room. Fire dampers will be provided in each air supply and exhaust wall opening. Electric heater, located in each battery room, will be designed to maintain not less than 60F during cold weather.

The radiochemical laboratory, titration room, and instrument shop will each be maintained at 75F and 50% relative humidity by a common chilled water system and separate fan-coil units for each room. Two 100% capacity water chillers and chilled water circulating pumps will be provided and will be served from separate power trains and raw cooling water sources. Filtered outdoor air will be supplied to each room by the air handling units and none will be recirculated. Hoods, located in each room, will be exhausted to the building exhaust stack by a separate exhaust fan for each hood and each hood exhaust shall be provided with a HEPA filter. Water chillers, fans, and filters will be located in the El 686 fan rooms. These systems are not ESF equipment nor will they be connected to emergency power.

Main steam pipeway rooms, located on the east and west sides of each reactor building, will be ventilated by a roof ventilator exhaust fan for each room. Outdoor air will be induced to the lower part of each room, through wall openings for the east rooms, and through a duct from a hood above the roof for the west rooms. These fans will be manually operated and will not be connected to emergency power.

9.4.2.3. Design evaluation

The operation of the auxiliary building air-conditioning, heating, cooling, ventilating, and air cleanup systems will be required to assure the integrity of the capability to shut down the reactor and maintain it in a safe shutdown condition.

The environmental control systems will also incorporate features to assure reliable operation during normal plant operation. These features will include the installation of redundant principal system components assigned to separate emergency power and cooling water sources.

The fuel-handling areas will be maintained at a slightly negative pressure to minimize exfiltration of air to other portions of the building.

A radiation-monitoring system will be provided to monitor and record and annunciate high activity in the building exhausts.

Each air cleanup filter train will consist of a demister which will also serve as a prefilter, followed by a bank of HEPA filter cells rated at 99.9% efficiency based on 0.30-micron DOP test, mounted in series with a bank of carbon adsorber modules rated at 99.99% efficiency for removal of methyl iodide. A second bank of HEPA filters will be installed downstream of the carbon adsorbers. Each HEPA filter bank will be provided with a static pressure differential indicating gauge.

All ESF air cleanup systems in the auxiliary building air exhaust system are 9
designed to comply with the major positions of Regulatory Guide 1.52. These air cleanup systems are listed below:

1. Unit 1 mechanical equipment zone exhaust air cleanup units.
2. Unit 2 mechanical equipment zone exhaust air cleanup units.
3. Units 1 and 2 fuel handling zone air cleanup units.

Tables 9.4-3 and -4 present further details of each system's compliance with the guide.

9.4.2.4. Tests and Inspection

The auxiliary building environment control systems will be in continuous operation and will be accessible for periodic inspection. Essential electrical components, switchovers, and starting controls shall be tested initially and periodically. Fans will be accessible for periodic inspection. The mechanical equipment zones air cleanup system fans and filter assemblies will be tested periodically. The HEPA filters will be tested in place initally and periodically with DOP, and the charcoal adsorbers will be tested initially and periodically with Freon 112. Charcoal surveillance specimens will be evaluated to assure iodine adsorptivity.

Air radiation monitors, thermostats, and smoke detectors shall be calibrated and tested periodically using a calibrated check source to verify the instrument's response and alarm functions.

Environmental conditions will vary only slightly under normal operation, and simulated conditions will prove design conditions.

9.4.3. Radwaste Area

9.4.3.1. Design bases

The auxiliary building radwaste areas will be located within the auxiliary building common areas between units at El 610, 629, and 649. These areas will be ventilated for the protection of plant equipment and controls; for the comfort and safety of operating personnel; to allow personnel acces for the operation, inspection, maintenance, and testing of mechanical and electrical equipment; and to limit the release of radioactivity to the building or to the atmosphere.

The radwaste areas will be ventilated by the environmental control systems of the auxiliary building, refer to PSAR section 9.4.2. These systems will be designed to maintain building temperatures in the main occupied areas between 50F minimum and 104F maximum during outdoor temperatures ranging from 5F in winter and 97F in summer, during all modes of plant operation.

To control the release of airborne radioactive material, filtered ventilating air (heated when necessary) will be supplied to clean areas, then routed to areas of progressively greater contamination possibilities. Areas of the building which are subject to radioactive contaimination will be maintained at a slight negative pressure relative to the outdoor pressure. Exhaust air from these areas will be processed by high efficiency particulate filters and charcoal adsorbers before being released to the atmosphere. All building exhaust will be discharged to the outdoors through the exhaust stack provided for each plant unit that extends above the reactor building parapet wall.

Building environmental control system fans, air coolers, and air-conditioning units, essential to the ventilation of radwaste area equipment, will be assigned to redundant equipment rains having separate emergency power assignments.

9.4.3.2. System description

The radwaste area ventilating systems are shown on Figure 9.4-2. These systems will be a part of the auxiliary building common area equipment zone ventilating, cooling, heating, air-conditioning, and air cleanup systems described in subsection 9.4.2.

Filtered and heated (when required) outdoor air will be mechanically supplied to the access areas adjacent to the radwaste equipment rooms by four 50% capacity supply fans located in the common area supply fan room at El 686.

Radwaste equipment room or cubicle air will be exhausted by two of four 50% capacity exhaust fans located in the common area exhaust fan room at El 686 and exhausting to the outdoors via the reactor building exhaust stack. This air is normally routed through two 50% capacity air cleanup filter trains located at El 686 before being discharged to the atmosphere. In order to prolong filter life, a bypass duct, provided around each air cleanup train, will allow air to be exhausted without passing through the filters, during plant shutdown or at times when air contamination conditions will permit.

Each air cleanup filter train will consist of a bank of HEPA filter cells followed by a bank of charcoal adsorbers enclosed within a plenum or room.

Each air cleanup filter assembly will be provided with static pressure differential indicators, thermometers, connections for inplace testing of filters and adsorbers, and access doors for filter and adsorber maintenance.

9.4.3.3. Design evaluation

The operation of the radwaste area ventilating and air cleanup systems will be required to assure a safe outside environment during normal operations. These systems will not be required to assure the integrity of the reactor coolant pressure boundary, nor the capability to shut down the reactor and maintain it in a safe shutdown condition.

The environmental control systems will incorporate features to assure reliable operation which will include the installation of redundant principal system components assigned to separate emergency power and cooling water sources.

The radwaste areas will be maintained at a slightly negative pressure to minimize exfiltration of air to other portions of the building.

A radiation-monitoring system will be provided to monitor and record and annunciate high activity in the building exhausts.

Each air cleanup filter train will consist of a bank of HEPA filter cells rated at 99.97% efficiency based on 0.30-micron DOP test, mounted in series with a bank of carbon adsorber modules rated 99.99% efficiency for removal of elemental iodine and 95.0% efficiency for removal of methyl iodide.

Each HEPA filter bank will be provided with a static pressure differential indicating gauge.

A single failure in the radwaste air handling system could reduce the number of air changes per hour in the radwaste area for a brief period of time. Such a change will allow the airborne activity level to increase somewhat. However, no safety hazard should arise because:

(a) The loss of airflow in the ducting would be annunciated in the main control room. Such an indication would inform the control room operator of the failure and make him aware of the need to switch to the full capacity standby redundant radwaste area ventilation system. 1

(b) all exhaust air from the radwaste area will be processed through air cleanup trains containing a HEPA filter bank and a carbon adsorber bank. Such processing assures that no hazardous activity releases to the environs will take place.

9.4.3.4. Tests and inspection

The radwaste area environment control systems will be in continuous operation and will be accessible for periodic inspection. Essential electrical components, switchovers, and starting controls shall be tested initially and periodically. Fans will be accessible for periodic inspection. The air cleanup system fans and filter assemblies will be tested periodically. The HEPA filters will be tested in place initially and periodically with DOP, and the carbon adsorbers will be tested initially and periodically with Freon 112. Charcoal surveillance specimens will be periodically evaluated to assure iodine adsorptivity.

Air radiation monitors, thermostats, and smoke detectors shall be calibrated and tested periodically using a calibrated check source to verify the instrument's response and alarm functions.

Environmental conditions will vary only slightly under normal operation, and simulated conditions should prove design conditions.

9.4.4. Turbine Building

9.4.4.1. Design bases

The turbine building heating, cooling, ventilating, and air cleanup systems will be designed to maintain an acceptable building environment for the protection of plant equipment and controls; for the comfort and safety of operating personnel; to allow personnel access for the operation, inspection, maintenance, and testing of mechanical and electrical equipment; and to limit the release of radioactivity to the atmosphere.

The building environmental control systems will be designed to maintain building temperatures between 60F minimum and 110F maximum during outdoor temperature conditions ranging 5F to 97F, and during all modes of plant operation.

All building exhaust systems will discharge to the outdoors from an exhaust housing located on the turbine building roof. Radiation monitoring of each building exhaust will be monitored, and recorded and annuciated in the main control room.

9.4.4.2. System description

Outdoor air will be mechanically supplied to maintenance areas of the basement floors at El 649 and 629 and also to the main operating floor at El 673. These supply fans will be generally located within a fan room provided between the plant units and at approximate El 702. Fresh air will enter from near the east end of the building.

Basement air will be allowed to rise through floor gratings and openings to the main turbine room at El 673 and will be mechanically exhausted to the outdoors. These fans will be located near the roof of the turbine room, between the plant units, and will discharge to the outdoors from near the west end of the building. An exhaust housing, located on the roof, will facilitate the continuous monitoring for high levels of radioactivity. Exhaust duct systems will be provided where required for heat removal from transformers and electric boards.

During cold weather, when building air supply and exhaust fans may be stopped to conserve heat, a reduced quantity of heated outdoor air will be supplied to the building El 629 basement floor. Heating coils, mounted in the air supply duct, will be designed to heat the air to approximately 60F when 5F outdoors. These hot water heating coils will be served by the building heating system.

The demineralizer tank and equipment rooms will be exhausted to the outdoors by two 100% capacity centrifugal fans and air cleanup filter assemblies. The rooms will be kept at a slight negative pressure to assure that all air leakage will be inward. Basement air will enter each room through door grilles equipped with backflow or air check dampers and adjustable dampers for manually balancing airflow quantities. The filter assemblies will consist of HEPA filter cells provided with static pressure gauges.

An air cleanup capability will be provided for purification of the condenser off-gas. Both HEPA filters and carbon adsorbers will be utilized in this cleanup system.

The turbine building heating system will consist of a medium-temperature hot water system which will include unit heaters, heating coils, pumps, heat exchangers, tanks, nitrogen pressurizing equipment, instruments and controls, and a piping distribution assemblage. The system heat exchangers, tanks, and pumps will be located near the southeast corner of the El 629 floor.

During normal plant operation, steam supply to the building heating system heat exchangers will be taken from the turbogenerator cold reheat steam. During plant shutdown, heat exchanger steam will be taken from the plant auxiliary boiler.

The turbine building will be heated during cold weather by unit heaters strategically located throughout the building.

Heating coils provided in the auxiliary building fresh air supply systems will be served by the turbine building hot water heating system.

9.4.4.3. Safety evaluation

The operation of the turbine building ventilating or heating systems will not be required to assure the integrity of the reactor coolant pressure boundary, or the capability to shut down the reactor and maintain it in a safe condition.

The systems will be designed to assure their reliable operation during normal plant operation . The demineralizer area exhaust fans and air cleanup system will include redundant components and fail-safe positions for control actuators.

9.4.4.4. Tests and inspection

The turbine building environment control systems will be in continuous operation and will be accessible for periodic inspection. Essential electrical components, switchovers, and starting controls shall be tested initially and periodically. Fans will be accessible for periodic inspection.

Air radiation monitors and thermostats will be calibrated and tested periodically using a calibrated check source to verify the instrument's response and alarm functions.

Environmental conditions will vary only slightly under normal operation, and simulated conditions should be such as to prove design conditions.

Testing of the turbine building heating, ventilating, and air-conditioning systems will be limited in scope as the systems will be self-supporting and will have no interlocks with other equipment.

The air cleanup system fans and filter assemblies will be tested periodically. The HEPA filters will be tested in place initially and periodically with DOP and carbon adsorbers will be tested initially and periodically with Freon 112.

9.4.5. Diesel Generator Buildings

9.4.5.1. Design bases

The diesel generator building ventilation systems will be designed to maintain an acceptable building environment for the protection of the diesel generators, electrical boards and equipment, batteries, and for the safety of operating personnel.

Each diesel generator unit room will be separately heated and ventilated to limit the room ambient temperature to not more than 120F when the entering air is 97F and diesel generator is operating, and to not less than 50F when 5F outdoors and the diesel generator is not operating.

Battery areas will be ventilated at all times for hydrogen removal, and electrical board rooms will be ventilated to limit the room ambient temperature to 104F when the entering air is 97F.

9.4.5.2. System description

Fresh air will be allowed to enter and be routed through each diesel generator room and exhausted to the outdoors by one of two 100% capacity exhaust fans provided for each diesel generator unit. Each pair of exhaust fans will be connected to its respective diesel generator engineered safety power supply, and one fan will automatically start upon diesel generator start. Fire dampers, provided in each air supply and exhaust, will automatically close to isolate any diesel generator in case of fire.

Battery room and electrical board room provided for each diesel generator unit will each be ventilated by a separate exhaust fan which will discharge to the outdoors.

Electric unit heaters, thermostatically controlled, will be strategically located throughout the building to keep maintenance areas at not less than 50F with outdoor temperature of 5F.

9.4.5.3. Safety evaluation

Each diesel unit will be redundant and the ventilation system will be a part of the diesel unit. Ventilating systems will be designed to seismic Category I requirements. Fans will be connected to emergency power, and the main exhaust fans will be redundant for reliability.

9.4.5.4. Tests and inspections

The diesel generator building ventilating and heating systems will be accessible for periodic inspection. Essential electrical components, switchovers, and starting controls will be tested initially and periodically.

Environmental conditions will vary only slightly under normal operation, and simulated conditions should prove design conditions.

9.4.6. Service Building

9.4.6.1. Design bases

The service building heating, ventilating, air-conditioning, and air cleanup systems will be designed to maintain an acceptable building environment for the protection of plant equipment and controls, for the comfort and safety of operating personnel, and to limit the release of radioactivity to the atmosphere.

Building heating will generally be accomplished by electrical unit and duct-mounted heaters, thermostatically controlled.

The building environmental control systems will be designed to maintain building temperatures between 60F minimum and 104F maximum during outdoor temperature conditions ranging from 5F to 97F.

Air-conditioned spaces will be maintained at approximately 75F and 50% relative hnmidity the year around.

Laboratories, decontamination rooms, and areas of the building which are subject to radioactive contamination will be maintained at a slight negative pressure relative to the adjoining areas, and exhaust air will be processed by high efficiency particulate filters before being discharged to the atmosphere.

9.4.6.2. System discription

Supply and exhaust fans, packaged air-conditioning units, associated ducting, dampers, and controls will be provided as required to maintain individual space environmental design requirements. Outside air will be supplied and mixed with return air, and building exhausts will be discharged to the outdoors.

The waste baling area, decontamination cell, and health physics laboratory hood will each be mechanically exhausted through HEPA filters to the outdoors.

9.4.6.3. Safety evaluation

The building environmental control systems will incorporate features that will assure their reliable operation during normal plant operation.

Each air cleanup system HEPA filter cell shall be rated at 99.97% efficiency based on 0.30-micron DOP test. Each filter bank will be provided with a static pressure differential indicating gauge.

A radiation-monitoring system will be provided to monitor and record and annunciate high activity in the exhausts from the waste baling, decontamination, and health laboratory areas.

9.4.6.4. Tests and inspection

The HEPA filters will be tested in place initially and periodically with DOP.

Air radiation monitors and thermostats shall be calibrated and tested periodically using a calibrated check source to verify the instrument's response and alarm functions.

Environmental conditions will vary only slightly under normal operation, and simulated conditions should be such as to prove design conditions.

9.4.7. Reactor Building Purge Ventilation System

The reactor building ventilation system provides outside air to ventilate the primary containment and the secondary containment. During reactor operations, the portion of the system serving the primary containment is normally shut down and primary containment ventilator purge lines are closed. When personnel enter the primary containment during reactor operation, a portion of the system serving the primary containment may be operated to provide fresh air for breathing and contamination control. During refueling operations, the system provides fresh air for breathing and contamination control as well as for cooling.

During power operation, cooling of the reactor building is accomplished with the containment air cooling system discussed in subsection 6.2.2.

The secondary containment is purged by a portion of the reactor building purge ventilation system. This portion of the system operates during power operation and during refueling. It is isolated and shut down during containment isolation conditions. The secondary containment purge not only supplies outdoor air for cooling and breathing, it maintains the secondary containment annulus at a negative pressure during normal operations so that in the event of an accident, which required containment isolation, the secondary containment annulus will remain negative throughout the course of the accident.

9.4.7.1. Design bases

The reactor building purge ventilation system is designed to maintain the environment in the primary and secondary containment within acceptable limits for equipment operation and for personnel access during inspection, testing, and maintenance operations. The design bases include provisions for:

1. Supplying fresh air for breathing when primary or secondary containment is occupied.

2. Cleaning up airborn activity in the primary and secondary containment during normal operation.

3. Cooling within the primary containment during refueling.

4. Cooling within the secondary containment during normal operation.

5. Establishing and maintaining the secondary containment at a negative pressure -- a precondition for postaccident primary containment leakage cleanup.

9.4.7.2. System description

The reactor building purge ventilation system flow diagram is shown in Firgur 9.4-3. The portion of the system serving the primary containment contains an air supply subsystem and an air exhaust subsystem. The air supply subsystem brings outside air in through an auxiliary building air intake, passes it through dust filters, across a heating coil, and to a supply fan. From the supply fan, the air is routed into the containment and released at several locations within containment. The exhaust system draws air from several containment inlets including an arrangement of inlets that provide for an exhaust sweep across the surface of the refueling pool. The exhaust flow passes through an air cleanup assembly containing a prefilter bank, a HEPA filter bank, and a carbon absorber bank. After leaving the air cleanup assembly, the exhaust is routed to an exhaust fan and then to the reactor building unit vent. As the exhaust air rises in the reactor building vent, it is monitored for radiation by the unit vent exhaust monitor. Both the supply and the exhaust subsystem have two 50% capacity fans and filter banks.

A small volume of the primary containment is subdivided from the remainder of primary containment and used as an instrument room. This instrument room serves to provide a protective environment for most of the instrumentation inside the primary containment. During normal operation relief panels in the subdividing walls are provided to assure pressure equalization within the containment during a LOCA. Instrument room cooling during power operation is accomplished with a chilled water system. The chilled water system is located in the auxiliary building with the chilled water routed to the two instrument room air handling unit cooling coils. The supply and exhaust subsystems of the primary containment ventilation system contain a small single low-capacity fan for purging just the instrument room. Such a capability is provided because it is anticipated that the instrument room will have to be entered more frequently than the primary containment. The exhaust from the instrument room is processed through the filtration systems provided for the primary containment purge ventilation system.

The primary containment penetrations for the ventilation supply and exhaust subsystems are designed to primary containment requirements. These are discussed in detail in subsection 6.2.4., Containment Isolation Systems.

The primary containment contains three air cleanup units that recirculate the containment atmosphere through HEPA filters and carbon adsorbers. These units are used to clean the containment atmosphere prior to purging of the primary containment. They are used only for pre-access; not for post accident conditions.

The primary containment purge ventilation system is normally controlled from the main control room. However, local control capability is provided to facilitate maintenance and test operations. The controls are designed to have simultaneous starting and stopping of the matching supply and exhaust equipment. The controls are also designed to givn an automatic shutdown and isolationupon receipt of containment isolation signal.

The secondary containment building ventilation purge system is shown in Figure 9.4-3. This system contains an air supply and exhaust subsystem. The supply subsystem obtains outside air from an auxiliary building air intake and routes it through dust filters and heating coils, and releases it into a manifold extending completely around the lower volume of the annulus formed between the primary containment and the secondary containment building. The exhaust subsystem draws air from the inlet above the primary containment dome, through a cleanup system, and discharges it to the unit station vent after passing the unit station vent radiation monitor. the secondary containment ventilation purge subsystem contains two 100% capacity trains. Each train contains a supply fan, an exhaust fan, and a pressure control assembly that regulates pressure in the annulus. The dust filters in the air supply subsystem and the HEPA filters and carbon adsorbers in the exhaust subsystem are shared with the primary containment purge ventilation system.

The secondary containment portion of the reactor building purge ventilation system maintains the secondary containment at a negative pressure as a precondition for secondary containment operation after containment isolation. A secondary containment pressure control assembly modulates dampers regulating the supply air so as to maintain the secondary containment at a negative pressure of approximately 1 inch of water. The secondary containment functioning is described in subsection 6.2.1.

The secondary containment purge subsystem is normally controlled from the main control rooms. These controls provide capability for manual startup, shutdown, and automatic control of the secondary containment pressure. The secondary containment building purge ventilation system is automatically shut down and isolated upon receipt of a containment isolation signal. Local starting and stopping switches are also provided to facilitate maintenance of testing operations.

The secondary containment pressure control assembly contains redundant pressure differential sensors, control circuitry, and motor control modulating dampers in each of the two inlet ducts. Upon startup of the system, the differential control unit adjusts the air supply rate to maintain the secondary containment pressure at a negative pressure of about 1-inch water gauge. The control unit alarms and automatically transfers to the redundant units upon failure of the operating system to keep the secondary containment pressure within the required pressur limits.

9.4.7.3. Safety evaluation

The primary containment and the instrument room purge, the primary containment air cleanup system, and the secondary containment purge system are not engineered safety features, and credit for LOCA mitigation is not claimed. Containment isolation signals automatically shut down and isolate the purge systems.

In the event of a refueling accident inside the containment, radiation monitors located near the surface of the fuel pool will initiate containment isolation. The isolation valves on the primary containment purge exhaust system will be 4
completely closed before any radioactivity has passed through the purge exhaust system ducting to the exterior of the primary containment.

The primary containment exhaust is monitored by a fast response radiation detector which provides automatic primary containment isolation upon detection of set point radioactivity in the exhaust air stream. This monitor assures that primary containment will be isolated in the event of a LOCA condition when the primary containment is being purged during normal operation. This radiation monitor is discussed in chapter 11.

The negative pressure maintained in the secondary containment annulus by the secondary containment ventilation purge system is considered in the postaccident primary containment leakage. The portion of the system that maintains the negative pressure is not designed to engineered safety feature standards because the system creates just the preconditon, and functioning of the system after containment isolation is not required. The equipment that isolates the secondary containment ventilation purge system and the actuating instrumentation, however, is designed to engineered safety feature requirements. The set point for the secondary containment negative pressure is sufficiently low that pressure changes within secondary containment due to accident-induced effects within the primary containment do not allow the secondary containment pressure to reach atmospheric anytime in the postaccident condition. This capability assures that any leakage from the outside into the secondary containment will be collected along with any primary containment leakage, and processed through HEPA and carbon adsorbers before release into the atmosphere.

9.4.7.4. Inspection and testing requirements

Prior to power operation, tests will be conducted to assure that the reactor building purge ventilation system performs as designed. Preoperational tests and periodic tests thereafter will be conducted to confirm that the features of the system that are required to mitigate accidents perform as designed. Specifically, inplace cold-DOP tests will confirm HEPA filter efficiency, and Freon tests will confirm carbon adsorber efficiency. Radiation monitors that provide containment isolation upon high radiation will be calibrated and tested periodically. ehe automatic shutdown and isolation of the primary containment and the secondary containment ventilation purge upon primary containment isolation will be confirmed periodically. Verification of the automatic startup capability of the standy redundant secondary containment purge system, upon failure of the operating system, will also be demonstrated periodically.

TABLE 9.4-1.

Toxic Gas Storage and Proximity to the Control Building Air Intakes

Toxic gas storage location	Type of toxic gas	Normal quantities stored	Distance from control building normal air intake, ft	Distance from control building emergency air intake, ft
At the ERCW intake	Acrolein	2 to 4 370-pound cylinders	3640	3900

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Table 9.4-2. Regulatory Guide 1.52 — Section Applicability for the Main Control Room Air Cleanup Units

Reg. Guide section	Applicability to this system	Comment index	Reg. Guide section	Applicability to this system	Comment index	
C.1.a	Yes	Note 1	C.3.i	Yes	--	
C.1.b	Yes	--	C.3.j	Yes	Note 9	12
C.1.c	Yes	--	C.3.k	Yes	--	
C.1.d	Yes	--	C.3.1	Yes	--	
C.1.e	Yes	--	C.3.m	No	Note 5	
C.2.a	Yes	Note 2	C.3.n	Yes	--	12
C.2.b	Yes	--	C.4.a	Yes	--	
C.2.c	Yes	--	C.4.b	Yes	--	
C.2.d	Yes	Note 3	C.4.c	No	Note 10	12
C.2.e	Yes	--	C.4.d	Yes	Note 11	
C.2.f	Yes	--	C.4.e	Yes	--	
C.2.g	Yes	--	C.4.f	Yes	--	
C.2.h	Yes	--	C.4.g	Yes	--	
C.2.i	Yes	--	C.4.h	Yes	--	
C.2.j	Yes	Note 4	C.4.i	Yes	--	12
C.2.k	Yes	--	C.4.j	Yes	--	
C.2.1	No	Note 5	C.4.k	Yes	--	
C.2.m	Yes	Note 6	C.4.1	Yes	--	
C.3.a	No	Note 2	C.4.m	Yes	--	
C.3.b	No	Note 7	C.5.a	Yes	--	
C.3.c	Yes	--	C.5.b	Yes	--	
C.3.d	Yes	--	C.5.c	Yes	--	
C.3.e	Yes		C.6.a	Yes	--	
C.3.f	Yes	--	C.6.b	Yes	--	
C.3.g	Yes	--	C.6.c	Yes	--	
C.3.h	Yes	Note 8				12

Notes:

1. The design basis LOCA is the postulated DBA.
2. No demisters are provided since no significant source of water is present in the volume served. Relative humidity control will be handled by ESF air coolers. The cooling system reduces the air temperature to 55F. This factor, plus the fact that all water piping is seismic category I, provides adequate capability to maintain the relative humidity in the main control room below 70%.
3. No significant pressure surges are foreseen for this system; thus, no 12 special protective devices are needed.
4. The air cleanup units will have a totally enclosed steel housing, which will be sufficiently rigid to be moved as an intact unit.
5. The ducting upstream of the fans and filters is under negative pressure and therefore only inleakage is expected. Ducts downstream of filters handle only clean air. The emergency pressurization portion of the system that may handle contaminated air under positive pressure will comply with the position stated in the regulatory guide.

Table 9.4-2. (Cont'd)

6. The system under consideration is an ESF system.
7. Humidity control is maintained by ESF air cooling units.
8. The use of aluminum is not restricted.
9. An analysis has been performed to determine the temperature rise in the
carbon adsorbers due to radiolytic heating. It was assumed that all the
radioactive decay energy from iodine was deposited in the carbon. Fur-
ther, it was assumed that there was no heat loss to the framing or the
air passing through the carbon beds. The rate of energy release corre- 12
sponding to the maximum activity on the adsorbers was calculated to be
9.1×10^{-6} kW. In such an analysis this yields a temperature rise of
0.006F/hour. No radioactive heat loss has been assumed. Water sprays are
therefore not required for this system due to the very low temperature
rise in the carbon even with maximum decay heat as used in this case.
10. Compliance with this section is not a licensing requirement.
11. Five feet between mounting frames will be provided if space permits.

Table 9.4-3. Regulatory Guide 1.52 — Section Applicability for the Auxiliary Building Units 1 and 2 Mechanical Equipment Zone Exhaust Air Cleanup Units

Reg. Guide section	Applicability to this system	Comment index	Reg. Guide section	Applicability to this system	Comment index	
C.1.a	Yes	Note 1	C.3.i	Yes	--	
C.1.b	Yes	--	C.3.j	Yes	Note 8	12
C.1.c	Yes	--	C.3.k	Yes	--	
C.1.d	Yes	--	C.3.l	Yes	--	
C.1.e	Yes	--	C.3.m	No	Note 5	
C.2.a	Yes	Note 2	C.3.n	Yes	--	
C.2.b	Yes	--	C.4.a	Yes	--	
C.2.c	Yes	--	C.4.b	Yes	--	
C.2.d	Yes	Note 3	C.4.c	No	Note 9	12
C.2.e	Yes	--	C.4.d	Yes	Note 10	
C.2.f	Yes	--	C.4.e	Yes	--	
C.2.g	Yes	--	C.4.f	Yes	--	
C.2.h	Yes	--	C.4.g	Yes	--	
C.2.i	Yes	--	C.4.h	Yes	--	
C.2.j	Yes	Note 4	C.4.i	Yes	--	
C.2.k	Yes	--	C.4.j	Yes	--	
C.2.l	No	Note 5	C.4.k	Yes	--	
C.2.m	Yes	Note 6	C.4.l	Yes	--	
C.3.a	Yes	--	C.4.m	Yes	--	
C.3.b	Yes	--	C.5.a	Yes	--	
C.3.c	No	Note 2	C.5.b	Yes	--	
C.3.d	Yes	--	C.5.c	Yes	--	
C.3.e	Yes		C.6.a	Yes	--	
C.3.f	Yes	--	C.6.b	Yes	--	
C.3.g	Yes	--	C.6.c	Yes	--	
C.3.h	Yes	Note 7				12

Notes:

1. The design basis LOCA is the postulated DBA.
2. Demisters will serve also as prefilters; no other prefilters are provided.
3. No significant pressure surges to this system are foreseen. Thus, no special protective devices are required.
4. The air cleanup units will have a totally enclosed steel housing. The housing will be sufficiently rigid to be moved as an intact unit. 12
5. Ducting upstream of the fans and air cleanup units is normally under negative pressure with respect to the surroundings; so only inleakage is expected. Malfunctioning dampers upstream of the air cleanup units could cause a momentary positive pressure in the ducting that would induce outleakage. However, this outleakage would be into the same air volume being served, therefore, no advantage is envisioned in using low leakage ducting in this part of the system. Ducts downstream of the air cleanup units handle only clean air, so leakage has no safety significance.

Table 9.4-3. (Cont'd)

6. The system under consideration is an ESF system.
7. The use of aluminum and zinc is not restricted since the system is a secondary system.
8. An analysis was conducted to determine the temperature rise in the carbon adsorbers due to radiolytic heating. It was assumed that all the radioactive decay energy from iodine was deposited in the carbon. Further there was no heat loss to the framing of air passing through the carbon beds. In such an analysis, the rate of energy release corresponding to the maximum activity on the adsorbers was calculated to the 5.1×10^{-2} kW. Air flow through the carbon and energy deposition was assumed to be uniform in all beds. This yields a temperature rise for the carbon adsorber assembly of less than 0.48F/hour. Water sprays are therefore not required for this system due to the very low temperature rise in the carbon even with maximum decay heat as used in this case. 12
9. Compliance with this section is not a licensing requirement.
10. Five feet between mounting frames will be provided if space permits.

Table 9.4-4. Auxiliary Building Fuel Handling Area Exhaust Air Cleanup Units

Reg. Guide section	Applicability to this system	Comment index	Reg. Guide section	Applicability to this system	Comment index	
C.1.a	Yes	Note 1	C.3.i	Yes	--	
C.1.b	Yes	--	C.3.j	Yes	Note 10	12
C.1.c	Yes	--	C.3.k	Yes	--	
C.1.d	Yes	--	C.3.1	Yes	--	
C.1.e	Yes	--	C.3.m	No	Note 8	12
C.2.a	Yes	Notes 2,3	C.3.n	Yes	--	
C.2.b	No	Notes 4,5	C.4.a	Yes	--	
C.2.c	Yes	--	C.4.b	Yes	--	
C.2.d	Yes	Note 6	C.4.c	No	Note 11	12
C.2.e	Yes	--	C.4.d	Yes	Note 12	
C.2.f	Yes		C.4.e	Yes	--	
C.2.g	Yes	--	C.4.f	Yes	--	
C.2.h	Yes	--	C.4.g	Yes	--	
C.2.i	Yes	--	C.4.h	Yes	--	
C.2.j	Yes	Note 7	C.4.i	Yes	--	12
C.2.k	Yes	--	C.4.j	Yes	--	
C.2.1	Yes	Note 8	C.4.k	Yes	--	12
C.2.m	Yes	--	C.4.1	Yes	--	
C.3.a	Yes	--	C.4.m	Yes	--	
C.3.b	Yes	--	C.5.a	Yes	--	
C.3.c	No	Note 2	C.5.b	Yes	--	
C.3.d	Yes	--	C.5.c	Yes	--	
C.3.e	Yes	--	C.6.a	Yes	--	12
C.3.f	Yes	--	C.6.b	Yes	--	
C.3.g	Yes	--	C.6.c	Yes	--	
C.3.h	Yes	Note 9				12

Notes:

1. The postulated DBA is the design basis fuel handling accident.
2. Demisters will also serve as prefilters; no other prefilters are provided.
3. Two different needs affect the sizing of the auxiliary building fuel handling area exhaust air cleanup system. One of these is the fuel handling accident in the auxiliary building. The other is the ventilation needs to keep an acceptable air purity in the auxiliary building fuel handling area during normal fuel handling operations. In evaluating these needs, it was found that the ventilation capacity needed to maintain a safe working environment in the fuel handling area in the auxiliary building is substantially greater (more than twice as large) than that needed to mitigate the effects of a fuel handling accident. Therefore, the system was sized for the normal ventilation needs.

   Such a practice assures that ample capacity will be available to cope with a fuel handling accident. Since the fuel handling operations will take place only if the auxiliary building fuel handling area ventilation system is in operation, at least 200% of the purging capacity needed to clean up

Table 9.4-4. (Cont'd)

the auxiliary building fuel handling area atmosphere in the post-accident period will be in operation at the time an accident occurs. Adequate redundancy is, therefore, provided to perform the only accident mitigation operation to be conducted by this ventilation system.

4. The short duration of the air cleanup operation (about 2 hours) needed following the postulated DBA identified in note 1 makes this requirement unnecessary because the probability of a destructive event of sufficient capacity to damage both air cleanup units installed side-by-side is extremely small.

5. The absence of violence during the postulated DBA identified in note 1 is another reason why separation of these air cleanup units is unnecessary.

6. No pressure surges are envisioned during or after the fuel handling accident.

7. The air cleanup units will have a totally enclosed steel housing. These housings will be sufficiently rigid to be moved as an intact unit.

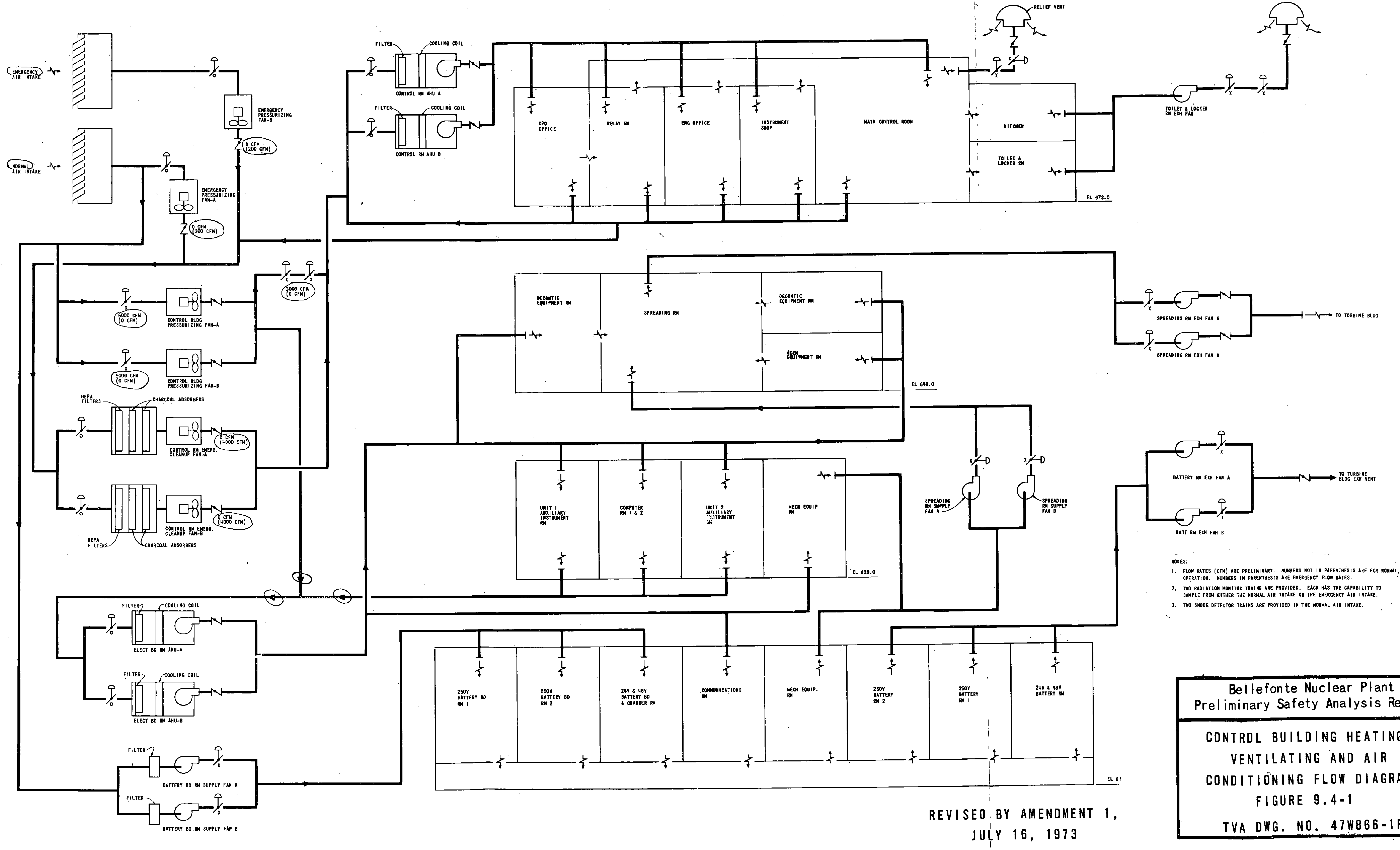
8. Low leakage ducting is not expected to provide safety enhancement for this system. Air in ducting upstream of the filters will be at a negative pressure with danger to the surroundings. Thus, only inleakage would occur. Ducting downstream of the filters handles only cleaned air; exfiltration would cause no danger.

9. Use of aluminum and zinc is not restricted.

10. An analysis was performed to determine the temperature rise in the carbon 12
adsorbers due to radiolytic heating. It was assumed that all the radioactive decay energy from iodine was deposited in the carbon. Further it was assumed that there was no heat loss to the framing or air passing through the carbon beds. The rate of energy release corresponding to the maximum activity on the adsorbers was calculated to be 6.5×10^{-4} kW. This yields a temperature rise of 0.44F/hour. No radioactive heat loss has been assumed. Water sprays are therefore not required for this system due to the very low temperature rise in the carbon even with maximum decay heat as used in this case.

11. Compliance with this section is not required by the AEC Division of Reactor Licensing.

12. Five feet between mounting frames will be provided if space permits.

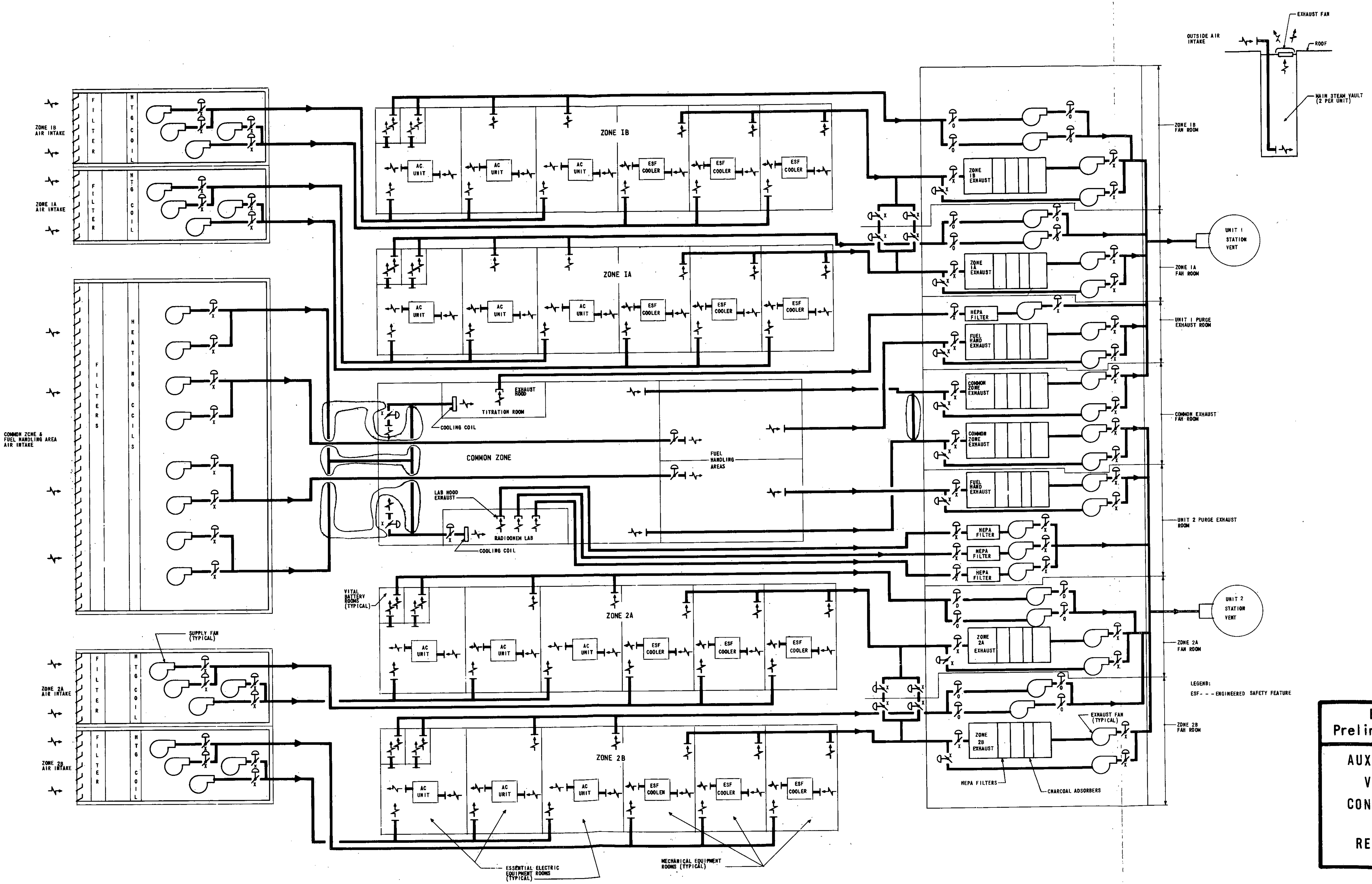


REVISED BY AMENDMENT 1,
JULY 16, 1973

Bellefonte Nuclear Plant
Preliminary Safety Analysis Report

CONTROL BUILDING HEATING
VENTILATING AND AIR
CONDITIONING FLOW DIAGRAM
FIGURE 9.4-1

TVA DWG. NO. 47W866-1R1

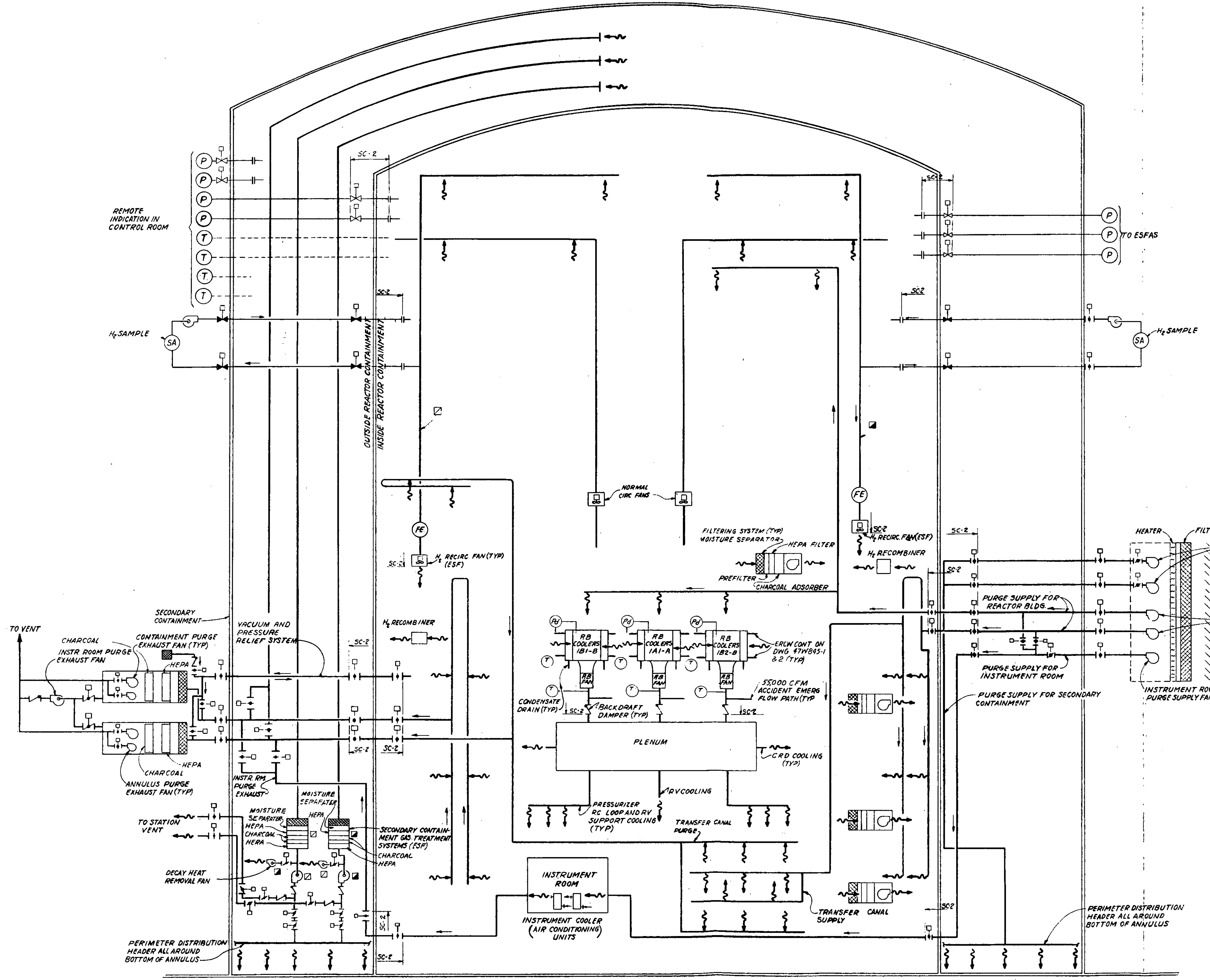


Bellefonte Nuclear Plant
Preliminary Safety Analysis Report

AUXILIARY BUILDING HEATING, VENTILATING AND AIR CONDITIONING-FLOW DIAGRAM

FIGURE 9.4-2

REVISED BY AMMENDMENT 1, JULY 16, 1973



NOTES:
1. PIPING AS SHOWN FOR UNITS 1 AND 2.
2. ALL EQUIPMENT SHALL BE PREFIXED BY THE UNIT NUMBER, I.E., 1-A1, 2-A1. EQUIPMENT COMMON TO BOTH UNITS DOES NOT HAVE A PREFIX NUMBER.
3. FOR PIPING SYMBOLS SEE FIGURE
4. SC-2 | NNS INDICATES LIMITS OF ANS SAFETY CODE CLASS.
5. ▨ DENOTES TRAIN A ◩ DENOTES TRAIN B

Bellefonte Nuclear Plant
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REACTOR BUILDING PURGE, COOLING, VENTILATION & H_2 REMOVAL

FIGURE 9.4-3

REVISEO BY AMMENDMENT 2 AUG. 6, 1973

9.5. Other Auxiliary Systems

9.5.1. Fire-Protection Systems (See Figures 9.5-1, 9.5-2, and 9.5-3)

9.5.1.1. Design Bases

The fire-protection system is designed to achieve the following objectives:

1. Provide fire protection in those areas of essential systems or control where a fire could prevent the operation of the reactor plant systems which are determined to be essential to public safety.

2. Provide fire extinguishment by fixed systems of water or carbon-dioxide type which are actuated automatically or manually in those areas where the danger of fire is of the greatest potential.

3. Provide manually operated fire extinguishing equipment including water connections for fire hose reels and cabinets and portable equipment of the wheeled and/or hand-carried type for use by personnel at all potential hazard areas throughout the property.

4. Provide fixed temperature detectors where automatic actuation of the carbon dioxide fire-protection system is specified and ionization type fire detectors in areas where products of combustion are a better indication of fire and it is desired to inspect the area before the carbon 12
dioxide fire-protection system is actuated. Fire detectors for the water fire-protection system are of the rate-of-rise and fixed temperature type.

5. Provide curbs around all oil facilities to prevent oil fires from spreading where a potential oil rupture exists.

6. Sound alarms in the affected area and after approximately a 20-second delay, discharge carbon dioxide into that area. There is sufficient delay time between the sounding of the alarm and the discharge of the carbon dioxide to allow personnel to leave the area.

7. All portions of the fire-protection systems necessary to protect safety related equipment in the auxiliary building, control building, diesel 12
generator buildings, and reactor buildings are designed for seismic requirements.

8. Pipe and pipe hangers for all portions of the fire-protection system located in seismic Category I structures are designed for seismic requirements to assure the integrity of other essential equipment in the area.

9.5.1.2. System Description

9.5.1.2.1. General

The high pressure fire-protection system furnishes water to all points throughout the plant area and to buildings where water for firefighting may be used. In other areas where water would create a hazard due to

the nature of the equipment or the type of fire (e.g., electrical or oil),
a low pressure carbon-dioxide fire-protection system is provided. The
carbon-dioxide system also furnishes carbon dioxide for generator purging.

A yard fire main is provided to loop the periphery of the plant buildings.
Sectionalizing valves are provided to permit isolation of potential faults
in the loop. Hydrants are appropriately located throughout the plant areas.
Water for firefighting in the yard, turbine building, and service and office
building is furnished by four disel engine-driven fire pumps that take their 12
suction from the condenser circulating water system and discharge into the
yard loop. These pumps are each rated at 2000 gpm at 350-foot head.

A second fire protection system is provided by four raw service water and 12
auxiliary building fire pumps located at the river pumping station. Each
pump delivers 600 gpm at 400-foot head to the yard loop through a seismically
qualified piping system and thence to the auxiliary building. The section of
the yard loop served by these pumps is seismically qualified and isolated
from the non-seismic part of the loop by motor-operated isolation valves.
These isolation valves are automatically closed when the auxiliary building
fire pumps are started to provide water for fire protection in the auxiliary
building. The pumps are normally operated to provide raw service water and
controlled by level switches on the raw water storage tank; however, when used
for fire protection, the pumps are manually actuated from pushbutton stations
located in the auxiliary building, adjacent to fire hose racks. The raw 12
service water and auxiliary building fire pumps and motor-operated isolation
valves in the yard loop are provided auxiliary power from the diesel generators.

Pressure is maintained on the entire system at all times by the raw service
water storage tanks. A valve automatically closes to isolate the two raw 12
service water storage tanks when the diesel engine-driven fire pumps start.
The valve is interlocked with the diesel fire pumps.

The service water system in the turbine building will supply raw water
to the water treatment plant. The condenser circulating water will be
chlorinated via the hyperchorite system located in the turbine building. 12
Condenser water requirements are given in Table 9.5-3.

Carbon dioxide is furnished by two separate units. Each unit has a storage
capacity of 17 tons. A unit is located in each of the two diesel-generator
buildings. Each carbon dioxide-unit supplies carbon dioxide to the
necessary rooms of its respective plant unit. Plant common rooms are
supplied from the Unit 1 carbon-dioxide unit with the exception of the
spreading room, which is furnished equally by both units. The Unit 1 carbon-dioxide
unit supplies all carbon dioxide used for fire protection in non-seismic
structures. A pneumatic discharge timer is included in this supply
line to maintain the integrity of the carbon-dioxide system in the event of
pipe rupture in the nonseismic areas.

The CO_2 storage unit is provided with a refrigeration unit that maintains
the temperature of the liquid CO_2 at its normal storage temperature of 0°F. 12
Should the refrigeration unit fail, automatic bleed is provided to maintain
the unit at a safe pressure level. As the CO_2 is bled off, expansion

occurs which in turn lowers the temperature level, thus maintaining a safe,
nonexplosive level of stored CO_2. In addition, relief valves and rupture
discs (with relief line to the roof) are provided and preset to a safe
pressure setting below tank design pressure. As an added protection, wall 12
blowout panels and vent openings are provided for the room wherein the units
are located, and all walls of the room are designed to withstand any momen- 12
tary pressure surge created until the blowout panels relieve, should
the storage tank accidently lose its total volume to the atmosphere.

An automatic water fog or spray system is provided for:

1. Main power transformers.
2. Unit and startup transformers
3. Hydrogen storage area
4. Acetylene and oxygen storage (service building)
5. Oil room and carpenters' shop (service building)

A manually actuated water fog or spray system (upon inspection after detection) is provided for:

1. Turbine oil reservoirs
2. Selected hazard areas in the turbine building

A CO_2 fire protection system is provided for the following areas:

1. Standby diesel generator rooms, fuel oil pump room and electrical board rooms (automatic electro-manual).
2. Paint shop and paint storage room (automatic electro-manual). 12
3. Spreading room, computer room, and auxiliary instrument rooms (manual-electric).
4. Other selected hazards in the turbine building (automatic electro-manual).

9.5.1.2.2. Component Design

All components and piping of the high-pressure fire-protection system for
the auxiliary building are designed for seismic requirements and in accor-
dance with the Power Piping Code, ANSI B31.1-1967, and the appropriate NFPA
standards. The raw service water and auxiliary building fire pumps provide
a minimum of 250 gpm per hose outlet on the auxiliary building roof for 12
fire protection (See Table 9.5-1 for pump design data.) The remaining
portion of the high-pressure fire-protection system, which includes the
turbine building, other nonessential structures, and the yard, is designed
in accordance with AWWA and the Power Piping Code, ANSI B31.1-1967, nonseismic,
and the appropriate NFPA Standards. The diesel engine-driven fire pumps are
designed to conform to the Hydraulic Institute, Centrifugal Pump Section.
The following ANSI Standards are also applicable, as noted:

1. ANSI B58.1, Part B, Vertical Turbine Pumps
2. ANSI B16.1, Flanges (CI)
3. ANSI B16.5, Flanges (Steel)
4. ANSI B31.1, Power Piping

Refer to Table 9.5-2 for pump design data.

All components and piping of the carbon dioxide fire protection system are designed in accordance with Power Piping Code, ANSI B31.1-1967, and the 12
appropriate NFPA Standards. Equipment, piping, and valves in the control building and diesel generator buildings are designed for seismic requirements.

9.5.1.3. Design Evaluation

The fire-protection systems provide a reliable source of water or other firefighting agent to all portions of the plant. The fire pumps' dis- 12
charge headers are provided with backpressure valves that are designed to maintain a preset pressure on the fire mains regardless of the demand.

Adequate drainage and physical separation barriers between trains of equipment prevent damage to one train of equipment while a fire is being fought in the adjacent zone. The use of solid streams of water on hot pipes or electrical equipment is prevented by the selection of proper hose nozzles for areas where these problems exist. Adequate alarms and control are employed in the carbon-dioxide system to prevent adverse effects to plant personnel or equipment.

A minimum of two fire detection devices are installed in any given area of the plant not normally occupied by operators or where rapid fire detection is essential. Reliability is maintained on the basis of redundancy of detectors and alarm wiring, or the two out of three logic input system where rate-of-rise, smoke, and temperature detectors are all used in a given area.

1
All detection devices are tested and approved by a recognized National Testing Agency such as Factory Mutual or Underwriters. Smoke detectors also have a system utilizing under voltage relays to detect failures in the detector system. Ionization type detectors are adjustable for sensitivity. All detectors which actuate the systems and the alarms in the control room, are installed in accordance with the applicable standards of the latest edition of the National Fire Protection Association Codes.

In addition, portable firefighting equipment is located throughout the plant for fighting small fires. Adequate chemical agents are provided for those areas where the use of water would be inappropriate or would constitute a hazard.

To minimize the potential for fires, the following design criteria is used:

1. Noncombustible construction is used throughout plant construction.
2. Fire doors are used to close off rooms containing flammables.
3. Equipment containing flammable fluids is curbed and drained to remove fluid due to spillage, leakage, or rupture. 12
4. Above grade tanks in the yard containing flammable fluids are located to reduce potential hazards and are diked to retain any leakage.
5. Fire retardant covering on power cables.
6. Fire barriers are used in cable penetrations between walls and floors.
7. Noncombustible fluid is used in power transformers.
8. Fire resistant fluid is used in the turbine hydraulic system.

9.5.1.4. Tests and Inspections

All fire pumps and system valves will be periodically tested and run to ensure proper performance when required. In addition, appropriate tests will be performed on all portable equipment to ensure it is properly charged and in good working condition.

9.5.2. Plant Communications System

9.5.2.1. Reference List of Figures

9.5.2.2. Design Bases

9.5.2.2.1. Interplant Systems

The design basis for interplant communications is to provide dependable communications for reliable operation (see Figure 9.5-7). The primary vehicles utilized to meet this requirement are insulated shield wire carrier (ISWC) and microwave radio (MW). Backup links for use if ISWC and MW fail simultaneously are the public telephone lines, power system operations VHF radio link, and the VHF radios assigned to public safety service (security) employees. See section 9.5.2.4. for a description and Figure 9.5-4 for a simplified diagram of the systems provided.

9.5.2.2.2. Intraplant Communications

The design basis for the intraplant communications is to provide sufficient equipment of various types so that the plant has adequate communications to start up, continue safe operation, or safely shut down (see Figure 9.5-7).

See section 9.5.2.3. for a description and Figure 9.5-5 for a simplified diagram of the systems provided.

9.5.2.3. General Description Intraplant Communications

The plant communications systems will be installed and maintained by TVA. The installation will include (1) a private automatic exchange (PAX) telephone system with code-call, (2) manual telephone system, (3) administrative telephone with intercom in offices, (4) paging system, (5) sound-powered telephone systems, (6) closed-circuit television systems, (7) an evacuation alarm system, and (8) operational intercom systems.

9.5.2.3.1. Telephone System

9.5.2.3.1.1. Private Automatic Exchange (PAX)

A 200-line PAX will be installed to provide primary 2-way communications throughout the Bellefonte Nuclear Plant. This PAX will be equipped with provisions for:

1. Regular 2-way telephone conversation.

2. Code-call and answer.

3. Fire alarm.

4. Paging call and answer over PAX instruments.

5. Executive right-of-way.

6. Single-digit access to electrical control room turret.

7. Single-digit access to system direct dialing circuits.

8. Revertive call switch.

The power for the PAX will be supplied from a 48-volt battery charger power board system which will consist of two 100-ampere chargers with dual a-c source voltage, supplied from train A and train B diesel-backed boards, backed by a 48-volt battery capable of supplying the load for 8 hours without the chargers.

9.5.2.3.1.2. Manual Telephone Switchboard

A 20-line,cordless, pushbutton switchborad will be installed on the electrical control area desk between Units 1 and 2 control rooms. This board will be equipped typically as follows:

4 keys - Automatic signaling unit to carrier

3 keys - 2-way trunks to carrier and PAX

2-keys - Touch tone to carrier

1 key - Executive right-of-way trunk

9 keys - Future

1 key - Fire-alarm trunk

Provision will be made for remote answering of the turret from the Unit 1 control desk, Unit 2 control desk, and the shift engineer's office.

9.5.2.3.1.2. Code-Call

Code-call will use the same end devices as the paging and evacuation systems, which will be installed throughout the plant area. The PAX is equipped with a special code-call switch to permit dialing of code numbers and answering by dialing a special answer number. The conversation is limited to 60 seconds to prevent tieup of the equipment. Code-call is on a par with paging and below evacuation, all clear, and fire-alarm in priority.

9.5.2.3.2. Administrative Telephone System

These multiple-line telephones will be installed as follows:

Power Stores offices - 6, 6 keys (5 lines, 1 hold)

Plant results supervisor - 1, 6 keys (5 lines, 1 hold)

Assistant plant results supervisor - 1, 6 keys (5 lines, 1 hold

Safety engineer - 1, 6 keys (4 lines, 1 hold, 1 spare)

Health physics officer - 2, 6 keys (4 lines, 1 hold, 1 spare)

Engineering offices - 7, 6 keys (5 lines, 1 hold)

Maintenance office No. 1 - 2, 6 keys (4 lines, 1 hold, 1 spare)

Operations office - 3, 6 keys (4 lines, 1 hold, 1 spare)

Assistant administrative officer - 1, 6 keys (4 lines, 1 hold, 1 spare)

Administrative officer - 1, 12 keys (6 lines, 1 hold, 5 spares)

Assistant superintendent - 1, 12 keys (5 lines, 1 hold, 1 exclusion, 5 spares)

Superintendent - 2, 12 keys (6 lines, 1 hold, 5 spares)

Receptionist - 2, 12 keys (6 lines, 1 hold, 5 spares)

Two of the lines on each instrument will be used for intercom. The power supply will be connected to the local 120-volt ac source.

9.5.2.3.3. Intercom System

9.5.2.3.3.1. Administrative Intercom System

This intercom system, an integral part of the administrative telephone system, will consist of two separate intercom lines and will be common between all the instruments listed in section 9.5.2.2. Power will be supplied from the administrative telephone system power supply.

9.5.2.3.3.2. DPO Engineer's Intercom

This intercom will consist of the following:

1 - 6-station master intercom in supervisor's office

1 - 6-station master intercom in DPO engineers' shop

1 - Staff intercom in DPO engineer's office

1 - Staff intercom on communications room test bench

Power will be supplied from the local 120-volt a-c source.

9.5.2.3.3.3. Power Stores Intercom and Paging System

This intercom system will consist of the following:

1 - 12-station master intercom in storekeeper's office

1 - 12-station master intercom in secretary's office

15 - Remote speakers in storeroom

15 - Intercom callback pushbuttons in storeroom

2 - Staff intercoms at stores issue counter

Power will be supplied from the local 120-volt a-c source.

9.5.2.3.3.4. Operational Intercom

The following equipment will be installed for this system:

1 - 12-station master intercom, Unit 1 control room desk

1 - 12-station master intercom, Unit 2 control room desk

1 - 12-station master intercom, electrical control room desk

1 - 12-station master intercom, shift engineer's office

Power will be supplied from the local 120-volt a-c source.

9.5.2.3.4. Paging System

The paging system will share the code-call and evacuation alarm system speakers which will be installed throughout the entire plant area. Paging handsets will be provided on both unit control desks, the electrical control desk, the shift engineer's desk, and the auxiliary control room desk. In addition to the paging handset locations, this equipment may be accessed from any PAX telephone. Paging is on a par with code-call and below evacuation, all clear, and fire alarm in priority, except when it is temporarily advanced to just below evacuation during emergencies.

9.5.2.3.5. Sound-Powered Telephone Systems

9.5.2.3.5.1. Plant Operation

The following separte systems will be provided for each of the two units--a total of 12 systems:

SP1 - Sound-powered jack system for turbine control, generator, and auxiliary power system

SP2 - Sound-powered jack system for feedwater, steam, and condensate system

SP3 - Sound-powered jack system for reactor control

SP4 - Sound-powered jack system for reactor coolant and auxiliary steam

SP5 - Sound-powered jack system for engineered safeguards and auxiliaries

SP6 - Sound-powered jack system for reactor refueling system

9.5.2.3.5.2. Backup Control Center System

The backup control center system will consist of two completely redundant systems. Each subsystem will be wired direct and independent of any other communications systems. Wiring routes will avoid the spreading room, unit control rooms, and auxiliary instrument room. Sound-powered equipment and circuits will be provided in the diesel generator building, 480-volt a-c shutdown board rooms, 6.9 kV a-c shutdown board room, auxiliary building, and auxiliary control room.

9.5.2.3.5.3. Health Physics System

A direct circuit will be provided between the health physics office and the Units 1 and 2 control rooms (physically on the electrical control area desk). Two pairs will be provided between the locations -- one for magneto signaling and one for talking over sound-powered handsets.

9.5.2.3.5.4. Diesel Building to Main Control Room

A direct circuit will be provided between the shielded waiting room in the diesel generator building and the main control building at the diesel generator control panel. A 4-conductor waterproof cable will be provided between these locations -- one twisted pair for magneto signaling and one twisted pair for talking over sound-powered handsets.

9.5.2.3.6. Closed-Circuit Television

9.5.2.3.6.1. Gatehouse Employee Entrance Surveillance

This system will include a camera, loudspeaker, and gate release mechanism in the gatehouse, and a television monitor, loudspeaker with amplifier, and gate release switch on the Unit 1 operator's desk.

9.5.2.3.6.2. Refueling Floor and Control Room

Two cameras will be provided for remotely viewing refueling operations. Permanent wiring for the cameras will be terminated in plug receptacles on the refueling floor at each unit. A camera will be installed in the spent fuel pit in a similar configuration. Another camera will be provided in the Units 1 and 2 control rooms. A 27-inch monitor will be provided in the visitor's lobby and a 14-monitor on the Unit 1 control desk. This system is to be used primarily by operations but is also useful in providing visitor information.

9.5.2.3.7. Evacuation Alarm System

See Figure 9.5-6.

This system will be designed to utilize a solid-state amplifier speaker end device which is capable of an output of 100 db at 10 feet and is shared with the all clear, code-call, fire alarm, and paging systems. The order of priority is fixed by relay logic as follows:

1. Evacuation alarm.

2. All clear.

3. Fire alarm.

4. Code-call and paging.*

The solid-state tone generators will be plug-in devices and may be easily replaced. A maximum of eight different tones will be available of which four will be used.

The output impedance of the tone generator will be 56 ohms and it will be capable of operating at 8 ohms. The input impedance of the amplifier-speaker unit will be 30,000 ohms so that the generator will be capable of driving a maximum of 3000 amplifier-speaker units in parallel. This application will require slightly over 300 units, which will result in a parallel impedance of about 100 ohms.

Design consideration will be given to increase reliability with the following features provided:

1. Redundant operating centers.

2. Three separate tone generator consoles.

3. Two physically separated, 24-volt dc fuse panels with approximately half of the amplifier-speaker units in each area of the plant fed from each fuse panel via alarm-type fuses.

*Paging and code-call are equal in priority except that paging can be temporarily advanced in priority to No. 2 so that the person actuating the evacuation signal may give oral instructions.

4. Power will be supplied to the fuse panels from two chargers, operating in parallel, each capable of providing the full load and backed up by a battery capable of supplying the load for 8 hours without the chargers.

5. d-c supervision of each individual audio pair.

6. Failure of paging selector switches results in plant-wide paging.

7. Manual bypass of evacuation alarm actuating devices.

9.5.2.4. General Description Interplant Systems

See Figure 9.5-4.

9.5.2.4.1. Microwave Radio

One microwave circuit operating in the 7 GHZ bank will be provided. This plant will be an additional repeater station on the power system control center (PSCC), Wilson circuit 8610. Twenty-four channels will be provided to the PSCC and twelve to Wilson Dam.

The equipment will be solid-state employing waveguide to 10-foot parabolic antennas which in turn are aligned with tower-mounted passive reflectors to direct the signal to the proper repeater station.

A 24-volt battery and charger-power board system is installed for the exclusive sue of these microwave circuits.

9.5.2.4.2. Insulated Shield Wire Carrier (ISWC) Circuits

One channel will be provided, routed on ISW to Widows Crrek, bypassed there to Raccoon Mountain on ISW, and cross-connected there to microwave radio for transmission to the power system control center.

The equipment will be solid-state, powered by the same 48-volt d-c battery charger-power board system described in section 9.5.2.3.1.1.

9.5.2.4.3. Commercial Telephone Service

Public telephone service will be provided for the following offices:

* Plant superintendent

*Assistant plant superintendent

Operations supervisor

*Administrative officer

*Receptionist

Shift engineer

Electrical operator

Power stores

DPO engineers

Health physics

Public safety - Two lines

Safety engineer

Waiting room

Treatment room

Pay station in office building

* Also provided with access to Chattanooga TVA PAX.

9.5.2.4.4. Power Operations VHF Radio

This system will consist of one base station with operating stations in the control room and the Division of Power Operations (DPO) engineer's office. It is intended for daily use by the DPO engineers but will not be used in the control room except in emergencies.

9.5.2.4.5. Public Safety Service (PSS) VHF Radio

This system will consist of VHF radio mobile equipment in PSS vehicles, portables in the possession of PSS officers, and fixed equipment in the gatehouse and in the visitor's reception area of the plant. This system is not intended for use by plant operating personnel, but it could be used in an emergency.

9.5.2.5. Evaluation

The following evaluation is intended to show redundancy and adequacy of the plant communications systems.

9.5.2.5.1. Interplant Systems

See Figures 9.5-4 and 9.5-7.

The ISWC circuit and the microwave circuit will enter the plant via different means, and the microwave circuit has redundant channels. There will be one ISWC circuit and a microwave circuit with 24 channels to the PSCC and 12 channels to Wilson. The radio frequency (RF) equipment on the microwave circuit will have two working sets of RF operating in the frequency diversity mode. The power for the microwave will be fed from its own 24-volt battery charger supplied for the telephone system. Each charger will be fed from two a-c sources and each battery will be capable of operating its system at least 8 hours without chargers.

The major electronic portions of both systems will be housed in the communications room which will be located in the control building. This building will be a seismic Category I structure and virtually fireproof.

The public telephone lines will also be routed through protectors in the communications room but will not be dependent on onsite power for operation.

The power system operations VHF radio will be located in or near the electrical control room and not in the communications room so that the complete loss of the communications room would not cause the loss of the link.

The public safety service (PSS) VHF radios will not be located either with the power system operations VHF radio or in the communications room. They will be located in the gatehouse, in the visitors' reception area of the plant, in PSS vehicles, and as portables in the possession of PSS officers. It is not probable that all units will be out of service simultaneously.

See Figure 9.5-7 for the availability of interplant communications during various postulated conditions.

9.5.2.5.2. Figures 9.5-5, 9.5-6, and 9.5-7

The automatic telephone equipment, which will be one of the primary systems, will be designed so that failures in individual switches or lines will not interrupt service. However, such failures will be annunciated and repairs will be made promptly. The main switching equipment for this system will be located in the communications room which will be in a seismic Category I building.

The administrative telephone system switching equipment will be located in the office building and will function without the PAX for those trunks which are not PAX trunks.

The paging speakers which are shared with the code-call and evacuation systems will be dispersed throughout the plant. Single or multiple open circuits or amplifier fialures in individual units will not prevent the remaining equipment from functioning. The failure of the PAX equipment will not impair the use of the paging equipment from the local paging stations at the unit operator's desk, the electrical control room operator's desk, the shift engineer's desk, and the auxiliary control room.

The various intercommunications systems will not be dependent on each other nor on the PAX or paging equipment for operation. The sound-powered telephone systems will be completely independent of power, each other, and all other systems provided. As long as a complete metallic path exists between instruments, communications will be maintained since the instruments supplied with these systems will be very rugged and will successfully withstand high shocks, negligence, and abuse. If permanently installed wires are rendered unusable for any reason, a temporary pair of wires will be used with the sound-powered instruments.

The evacuation alarm system will be designed for survivability with the following features:

1. Duplicate operating locations - One on the electrical control room desk and the other in the auxiliary control room. Manual bypass of the controls will be provided at each location.

2. Three tone generator consoles powered from separate sources:

   a. The operating consoles.

   b. A standby console which automatically will be inserted upon power failure on the operating console. It may also be manually switched at any time.

   c. A third console which may be manually substituted for either of the other consoles.

3. Plug-in Features:

   a. The tone generators will be solid-state plug-in devices, and spares will be readily available.

   b. The amplifier in the speaker unit will be solid-state, easily unplugged and replaced. Spares will be stocked.

4. The power leads to each speaker-amplifier will be fused and annunciated.

5. The signal leads to each speaker-amplifier will be supervised with d-c while idle. Any occurrence which causes a short of the signal leads will cause the fuse to blow and annunciate. The rest of the units will function normally with single or multiple open-circuited leads to individual speaker-amplifiers.

6. The power will be quite reliable since it will be supplied from two chargers, operating in parallel, and each capable of handling the full load. The chargers will be backed up by a battery capable of supplying the load for 8 hours without the chargers. The power will be distributed from two fuse panels located in widely separated bays in the communincations room. Each fuse panel will feed power by separte routes to approximately half of the end devices in each area of the plant.

See Figure 9.5-7 for the availability of intraplant communications during various postulated conditions.

9.5.2.6. Inspection and Tests

General

All systems will be carefully installed and checked for proper operation initially by construction forces. Routine maintenance will be performed by operating personnel on a regular basis and includes such items as checking for proper switch operation, oiling where necessary, checking for proper operating levels, visual inspection, etc.

The most comprehensive testing, however, will result from the heavy daily usage of the equipment and the subsequent reports of any troubles encountered by the users. Individual power failures in the equipment will be annunciated and repairs will be made promptly.

9.5.3. Lighting Systems

9.5.3.1. Design Bases

There shall be three basic lighting systems in the plant designated as follows: normal, standby, and emergency. These systems shall be designed in accordance with the recommendations of the Illuminating Engineering Society, National Electrical Code, TVA Electrical Design Memorandums, and good engineering practice to provide the best visual performance for the particular seeing task required for the plant operation while meeting safety and comfort requirements.

The normal system shall be designed to economically provide the amount and quality of illumination to meet normal plant operations and maintenance requirements.

The standby system, on loss of the normal lighting system, provides the minimum illumination level necessary for the safe shutdown of the reactor and the evacuation of personnel from the plant if the need should occur. It forms an integral part of the normal lighting requirements but is fed from an entirely independent source.

The emergency lighting system, fed from independent d-c voltage sources, shall provide immediately the minimum illumination level in areas vital to the safe shutdown of the reactor when the other lighting systems are unavailable.

9.5.3.2. Description of the Plant Lighting Systems

All plant lighting systems shall have the following features in common: adequate capacity and rating for the operation of all loads connected to the systems, independent wiring and power supply, overcurrent protection for conductor and equipment using nonadjustable circuit breakers, and copper conductor with 600-volt insulation run in metal raceways.

The insulated cable and lighting materials used inside the primary containment area shall be resistant to nuclear radiation and chemical environmental conditions in this area as well as excessive humidity and temperature extremes. Lighting materials containing aluminum and mercury are excluded from use inside the primary containment area.

Conduits, boxes, and fixtures located in the Category I structures shall be provided with supports or safety catches designed to meet the Safe Shutdown Earthquake (SSE) requirements of the plant.

The plant lighting system consists of three basic schemes, the first of which is the normal lighting. This system is for general lighting of the plant; the major power supply is through feeders from the 6.9-kV common boards to a split bus 6.9-kV lighting board with automatic transfer, to 3-phase, 120/208-volt a-c transformers feeding lighting boards distributed throughout the main plant. The 120/208-volt lighting boards are located in the turbine building, auxiliary building, service building, office building, etc. These lighting boards feed the normal lighting cabinets, designated by the prefix LC, located near lighting load centers. In the power boards and control rooms, alternate rows of fixtures or alternate fixtures are fed from different lighting boards to prevent total blackout in a particular area in case of failure of one of the other lighting boards or cabinets.

The second system is the standby lighting which forms a part of the normal lighting requirements and is energized at all times. This system is fed from 480-volt energized safety load centers to 3-phase, 120/208-volt a-c transformers to each standby lighting cabinet, designed by the prefix LS. These load centers have a normal and alternate AC power supply and are backed up by the diesel generators in event of a power failure. The cable feeders to the standby cabinets located in the Category I structures shall be routed in redundant raceways, and the fixtures are dispersed among the normal lighting fixtures in the vital and less critical areas of the plant.

The third lighting system is referred to as the emergency system. The feeder to this system shall be held electrically in the off position until a power failure occurs on the ac system; then the emergency lighting cabinets, designated by the prefix LD, will be automatically energized from the 125-volt d-c vital battery boards. This system is an engineered safety feature (ESF), and the cable feeders to the LD cabinets shall be routed on the redundant cable tray or in conduit. The fixtures shall be incandescent type and shall be dispersed among the normal and standby fixtures with adjacent emergency fixtures being fed from redundant power-trained LD cabinets in the control rooms.

In power board rooms and other areas of zones A and B separation, lighting circuits of any of the three systems located in either of these zones will not feed lighting fixtures located in the other zone.

9.5.3.3. Diesel Generator Building Lighting System

The diesel generator building lighting cabinets will be fed through 480-208/120-volt ac, 3-phase local lighting transformers, which in turn are fed from the diesel 480-volt motor control centers respectively. Each of these diesel motor control centers are fed from a 480-volt engineered safety load center board which has a normal and emergency supply. In the event of an ac power failure to the 6.9kV engineered safety boards that feed these load centers, the diesel generator will start in time to provide the 480-volt ac power requirements for the safe shutdown of the plant through the standby feeders to the 480-volt diesel control centers boards, thus supplying power again to the diesel generator building lighting transformers. Each diesel generator has its own lighting cabinet which supplies lighting power to its generator room.

The emergency lighting supply for each diesel generator room is obtained from the 125-volt dc feeders for the diesel generators. The power supply to the dc lighting system is held electrically in the off position until a power failure occurs on the a-c system, then the emergency lights are automatically energized from the d-c feeder.

9.5.3.4. Safety-Related Functions of the Lighting Systems

The normal ighting will be safe for the operation and evacuation of the plant to the extent of the design criteria for seismic mounting of conduits, boxes,fixtures, and the reliability of the power source feeding it, along with the quality of the materials and field installation.

The standby system provides low-level lighting in the vital areas and exit points for the safe shutdown of the reactor and evacuation of personnel.

The emergency lighting in the vital areas is adequate for safe shutdown of the reactor and the evacuation of personnel from hazardous areas during any design basis event.

9.5.3.5. Inspection and Testing Requirements

Following the complete installation of a lighting system, it shall be tested and inspected and all short circuits, grounding of potential conductors, other faults, etc., shall be eliminated and all damaged or non-operable fixtures replaced or repaired. The operation of the lighting system shall be observed during the initial and periodic testing of the normal and alternate feeder systems and emergency power to the various boards from which these lighting systems are fed. Maintenance and relamping of the normal and standby lighting systems shall be according to routine plant operating procedures.

The emergency lighting system shall be tested periodically by tripping the holding coil circuit fed from the LS standby cabinet, thus closing the feeder circuit to the LD emergency cabinet. All emergency lamps shall be inspected and replacements made where necessary. A written record of dates and results of these tests shall be maintained by plant personnel responsible for these tests.

9.5.4. Diesel Generator Fuel Oil System

9.5.4.1. Design Bases

From seven-day tanks to day tank, the deisel generator fuel oil system is de-
signed to ANS Safety Class 3 and Category I seismic requirements and shall be 8
impervious to the effects of tornadoes, hurricanes, floods, rain, snow, or ice
as defined in Chapter 3 of this document.

The system is capable of supplying No. 2 diesel fuel to all four diesel generator sets operating at full load for a period of not less than 7 days.

The design code requirements for this system are as follows:

1. Diesel generator building 7-day diesel oil supply tanks — ASME Boiler and Pressure Vessel Code, Section VIII.

2. Yard fuel oil storage tanks — American Petroleum Institute Standard 650.

3. Diesel generator building piping — ASME Boiler and Pressure Vessel Code. Section III, Class 3. 8

4. Diesel generator building safety-related valves, pumps, and associated equipment — ASME Boiler and Pressure Vessel Code, Section III, Class 3.

5. Yard piping, valves, and associated equipment — Power Piping Code, ANSI B31.1-1967.

The system is designed with the ability to meet single failure criteria.

9.5.4.2. Description

The diesel generator fuel oil system (Figure 9.5-8) provides four 7-day diesel oil supply tank assemblies, one for each diesel generator unit. The tanks serve as liners embedded in the Category I diesel generator building substructure, and will provide approximately 75,000 gallons of fuel for each generating unit.

Fuel transfer from the 7-day diesel oil supply tanks to each of the 550-gallon diesel generator day tanks is accomplished by two motor-driven day-tank fuel transfer pumps, one per diesel engine. Each transfer pump is capable of supplying fuel to either engine's day tank. A 200-gpm diesel building fuel transfer pump located in the fuel oil transfer pump room of each diesel generator building is provided to accomplish the following functions:

1. Transfer oil from any 7-day diesel oil supply tank assembly to any other.

2. Transfer oil from any 7-day diesel oil supply tank assembly to either of two 100,000-gallon yard fuel oil storage tanks.

3. Reject oil from the system through one of two reject connections in the yard.

A 200-gpm diesel fuel oil unloading pump located at the railcar unloading station is used to transfer fuel oil from rail cars to either of the 100,000-gallon yard fuel oil storage tanks or directly to the 7-day diesel oil supply tanks. (Also, the 7-day diesel oil supply tanks may be filled directly from a tank truck at fill connections located outside the diesel generator buildings.)

A 200-gpm yard diesel fuel transfer pump located adjacent to the yard fuel oil storage tanks is provided to accomplish the following functions:

1. Transfer oil from a tank truck to either of two yard fuel oil storage tanks.

2. Transfer oil from either yard fuel oil storage tank to the other.

3. Transfer oil from either yard fuel oil storage tank to any one of the four diesel generator building 7-day diesel oil supply tank assemblies.

4. Reject oil from either yard fuel oil storage tank through one of two reject connections in the yard.

Rejection of oil due to possible contamination from any source is considered to be an abnormal requirement. However, should this occur adequate means are provided to reject any volume of oil to the reject connections, and thence to yard rail tank cars or tank trucks. Ultimately, this oil will be removed from the plant area or used for other purposes.

Administrative control over inventory, normal testing per technical specifications, and use of fuel oil for the auxiliary boilers should preclude the rejection of any oil under normal operating circumstances.

The yard fuel oil storage tanks are diked to contain the oil in the event of a tank rupture. The oil piping in the yard is routed in concrete trenches to allow periodic inspection of the oil lines. The trenches drain to the yard drainage system which discharges to the yard drainage holding pond. Therefore, in the event of an oil line break, the oil will be contained in the holding pond and subsequently removed.

9.5.4.3. Evaluation

In the event any of the previously mentioned adverse environmental conditions occur, each diesel generating unit is assured of having at least 7 days of fuel supply because:

1. Each diesel oil supply tank assembly will store a 7-day supply of diesel fuel.

2. Each assembly is embedded in the concrete substructure of the seismic Category I diesel generator building. 8

3. Assemblies will be separated by 18 inches of concrete.

4. All piping, valves, and remaining equipment are Category I seismic.

The ability to meet the single failure criterion is shown by the following:

1. Each day tank diesel fuel transfer pump is capable of supplying fuel to the day tank of either diesel engine. Each day tank will contain two sets of level switches. One set is positioned at a higher level

than the other, thereby maintaining one pump as the "lead" pump and the other as "backup". A selector switch is provided to alternate the "lead" and "backup" pumps.

2. Redundant paths from each day tank to its corresponding diesel engine are provided by the d-c motor-driven pump and the engine-driven pump.

Should the yard fuel oil storage tanks or the transfer piping between these tanks and the diesel generator buildings be damaged or destroyed by an earthquake or tornado, and fuel oil be needed after 7 days, additional oil will be procured and delivered by tank truck for refilling the 7-day diesel oil supply tanks. This is accomplished by dispensing the fuel directly to the tanks through fill connections immediately outside the diesel generator buildings.

All fuel oil lines feeding the diesel generator building oil transfer pump and the 7-day oil supply tanks enter the top of the tanks and are designed and located such that an inadvertent line break for any cause will not siphon any of the contents of the 7-day storage tanks. A bottom drain is not provided.

A 0.125 inch corrosion allowance is provided in the design of the wall thickness for the diesel generator building 7-day diesel oil supply tanks. The safety-related fuel oil piping and fittings within the diesel generator buildings have more than ample corrosion allowance, since they are designed per the 8
ASME Boiler and Pressure Vessel Code, Section III, Class 3, and operated at pressures considerably lower than the maximum allowable for the size and schedule piping and fittings used.

9.5.5. Diesel Generator Cooling Water System

A closed circuit circulating water cooling system is furnished for each engine as shown in Figure 9.5-9. Each system includes a pump, heat exchanger, 1
expansion tank, and all accessories required for a cooling loop. The expansion tanks are equipped with a float valve to provide a means for supplying makeup water from the demineralized water system to the closed cooling loops. The engine cooling water (closed loop) is circulated through the shell side of the heat exchanger by a shaft-driven pump off the engine.

The diesel generator cooling water system (closed loop) components are designed in accordance with the applicable standards of ASTM, TEMA, ANSI, and ASME, Section VIII.

The heat sink for the engine cooling water loop is provided by the essential raw cooling water system. Essential raw cooling water flows through the tube side of the heat exchanger. (Refer to section 9.2.1. and flow diagram Figure 9.2-1).

The cooling water system is designed to satisfy the single failure criteria. Each diesel generator set is provided with two closed circuit engine cooling water loops, one for each engine. A failure in one cooling loop does not affect the other loop. The cooling water (heat sink) to each heat exchanger is provided by redundant headers from the essential raw cooling water system. A failure of one ERCW header does not jeopardize the essential raw cooling water supply to the heat exchanger from the other header.

9.5.6. Diesel Generator Starting System

Each diesel engine is equipped with an independent pneumatic starting system
mounted on a common base, complete with valves, piping, controls, etc., as 1
shown in Figure 9.5-10. Two full-capacity starting air motors are provided for each diesel engine.

The diesel generators are automatically started on the loss of off-site power or by the Engineered Safety Features Acutation System (ESFAS) signal as discussed in 8.3.1.

Two air accumulators are provided for each diesel engine. Each accumulator is sized for an air storage capacity sufficient to crank the engine five times without recharging. One accumulator serves as a standby for the other. Each set of accumulators is equipped with shutoff valves, pressure gauges, safety valves, and low-pressure alarm contacts for use on 125-volt d-c circuit.

Two electric-motor-driven air compressors are provided for each diesel generator power package. Each compressor is sized to recharge one set of accumulators in 30 minutes. The compressor motors are 480 volts, 3 phase, 60 Hz.

The starting system is designed to satisfy the single failure criteria. Redundant equipment is provided for each power package such as air compressors, accumulators, starting air motors, and accessories so that a single failure will not jeopardize the design starting capacity of the system. The starting
air supply system, as defined by IEEE Std 387, is designed to ASME Section III, 8
Class 3.

9.5.7. Diesel Generator Lubricating System

The engine lubricating oil system provides full pressure lubrication to the various moving parts of the engine and all surfaces requiring lubrication,
as shown in Figure 9.5-9. The system also supplies oil for cooling of the 1
piston and lubrication of the piston pin bearing surfaces.

Oil filters and oil coolers are provided in the lubricating oil system for filtering and cooling the oil. These filters are all metal self-cleaning type with automatic bypass. Lubricating oil is cooled via the lubricating oil cooler by the closed circuit cooling water system (refer to 9.5.5, Diesel Generator Cooling Water System). Lubricating oil is heated to ensure rapid starting when the engine is not operating.

The lubricating oil pumps are positive displacement type, mounted externally for accessibility, and shaft-driven from the diesel engines.

An individual lubricating oil system is provided for each diesel generator engine. A failure in the lubricating oil system of one diesel generator does
not effect the operation of any other diesel generator. The lubricating sys- 8
tem is internal to the diesel generators and meets the requirements of IEEE Std 387.

Table 9.5-1. Raw Service Water and Auxiliary Building Fire Pumps

Quantity	4
Type	Vertical turbine
Rated capacity, gpm	600
Rated head, feet of water	400
Motor horsepower	100

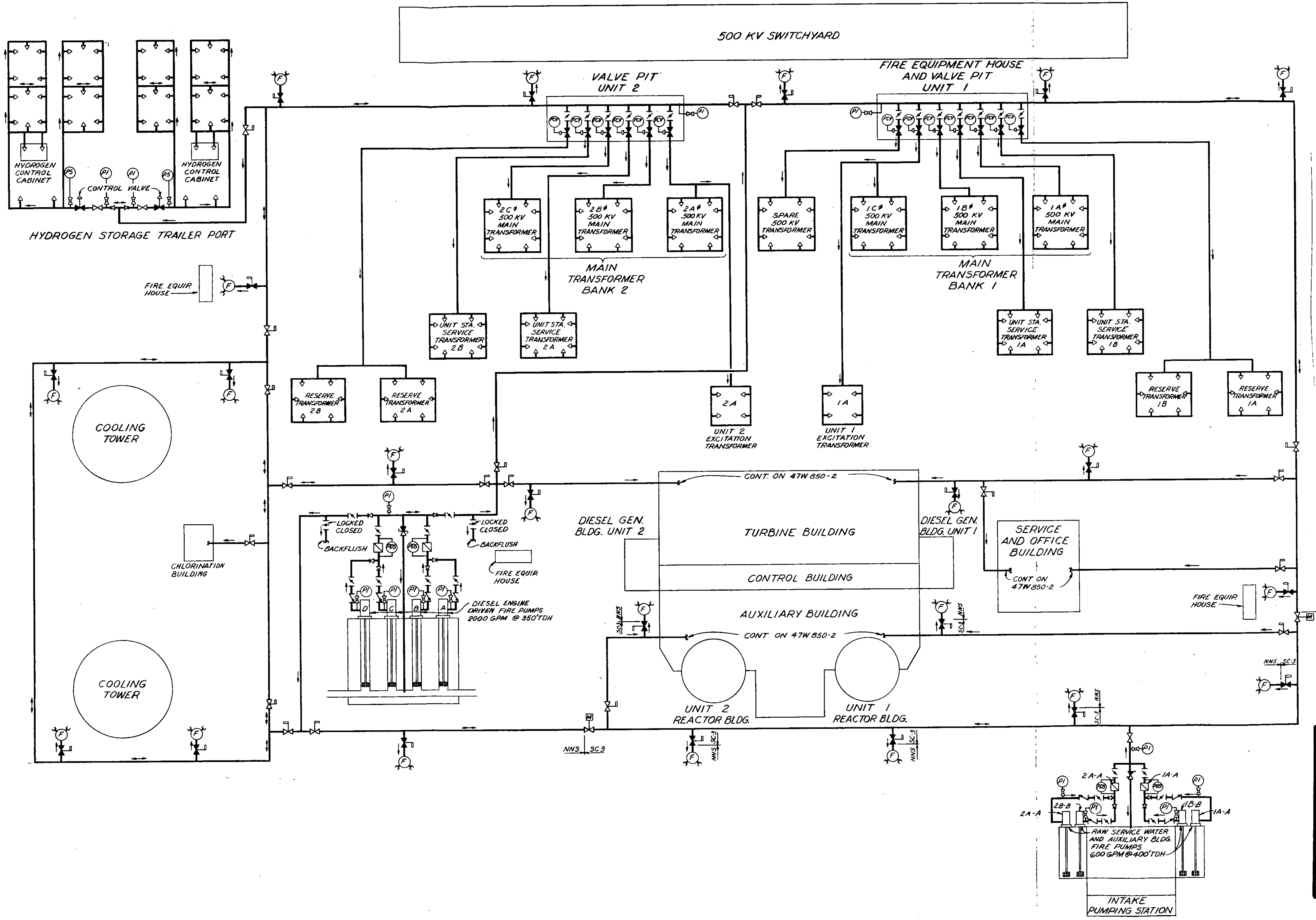
Table 9.5-2. Engine-Driven Diesel Fire Pumps

Quantity	4
Type	Vertical turbine
Rated capacity, gpm	2000
Rated head, feet of water	350
Motor horsepower	250

Table 9.5-3. Service Water Requirements

Service	Constant usage, gpm	Intermittent use, gpm	
Service connections and floor wash	50	75	
Air conditioning service bay and office wing[a]	500	--	12
Water-treatment plant makeup	500	--	
Yard sprinkling	--	200	
Total	1050	275	

(a) Estimated maximum quantities



NOTES:

1. ALL VALVES ARE THE SAME SIZE AS PIPING UNLESS OTHERWISE SPECIFIED.
2. IDENTIFICATION OF VALVE SYMBOLS IS SHOWN ON TVA DWG GM-4-308617-3.
3. MAIN PROCESS VALVES ARE SHOWN IN THEIR NORMAL OPERATING POSITION.
4. UNIT 1 EQUIPMENT PREFIX NUMBER IS 1 AND UNIT 2 EQUIPMENT PREFIX IS 2. EQUIPMENT COMMON TO BOTH UNITS DOES NOT HAVE A PREFIX NUMBER.
5. THE RAW SERVICE WATER AND AUXILIARY BUILDING FIRE PUMPS, STRAINERS, VALVES, AND PIPING WHICH FORM THE AUXILIARY BUILDING FIRE-PROTECTION LOOP ARE ANS SAFETY CLASS 3 (SC-3). ALL OTHER PIPING AND VALVES ARE NONNUCLEAR SAFETY CLASS (NNS)

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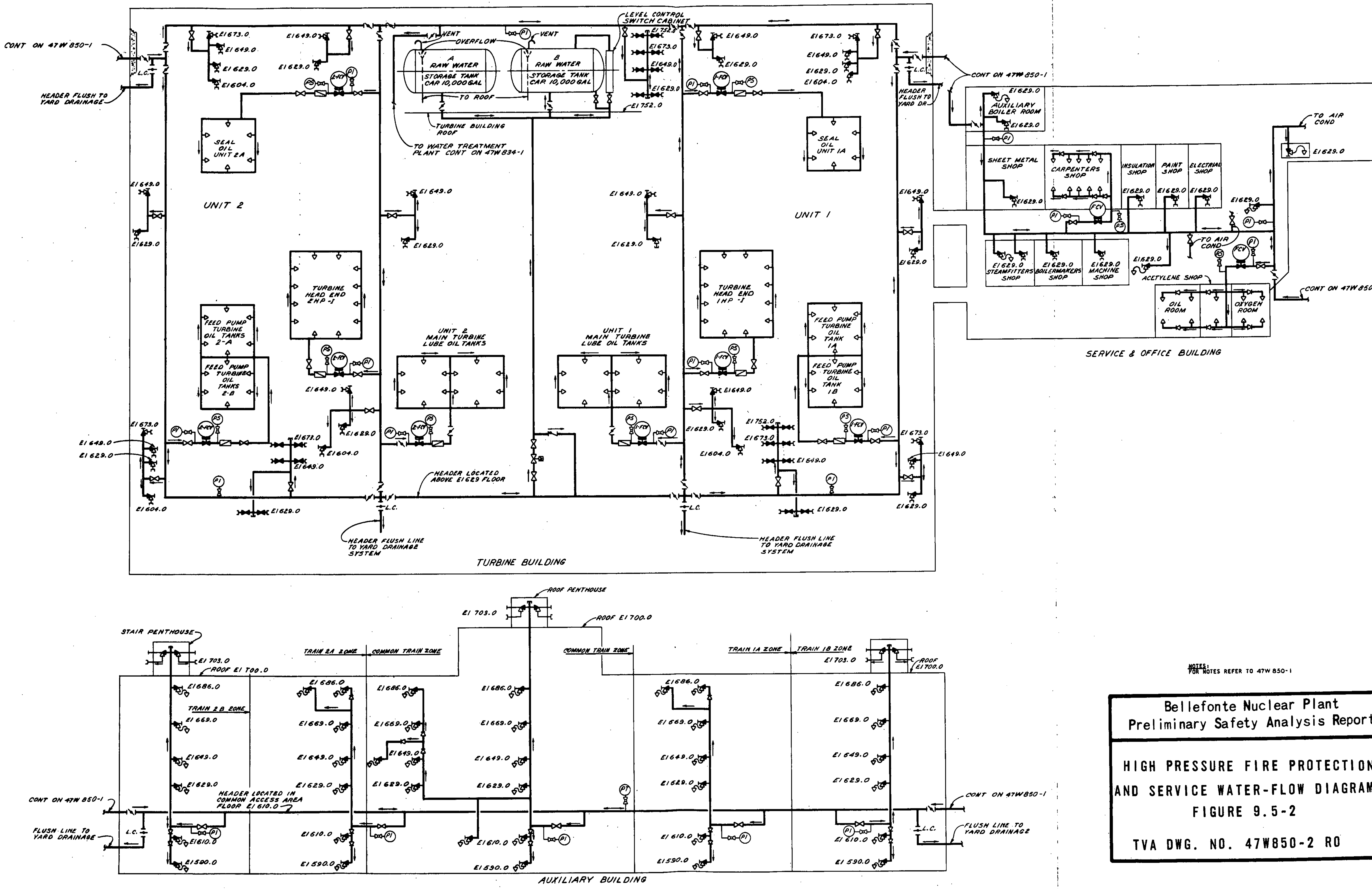
HIGH PRESSURE FIRE PROTECTION
&
SERVICE WATER-FLOW DIAGRAM

FIGURE 9.5-1

TVA DWG.NO. 47W858-1 R0



NOTES:
FOR NOTES REFER TO 47W 850-1

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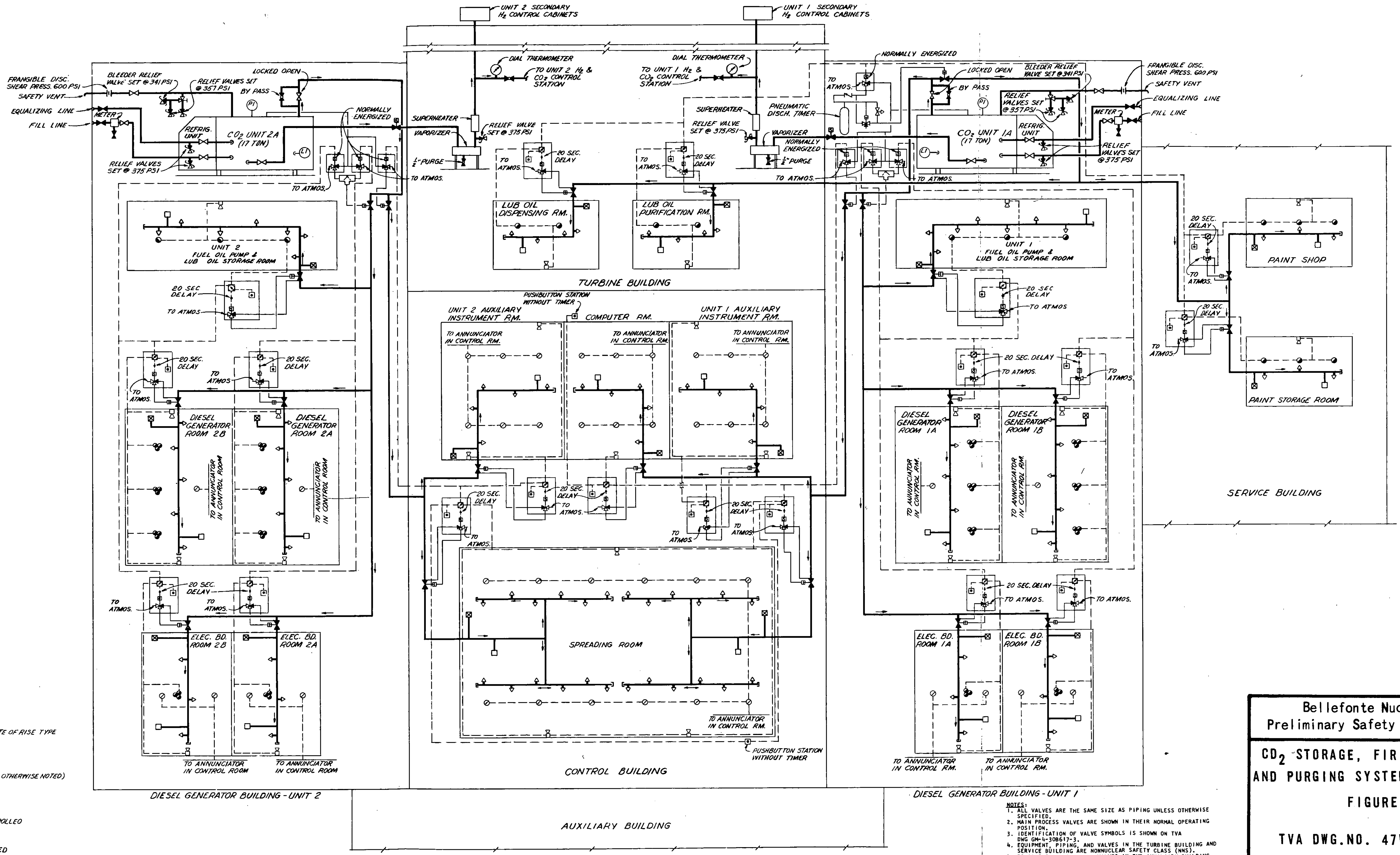
HIGH PRESSURE FIRE PROTECTION AND SERVICE WATER-FLOW DIAGRAM

FIGURE 9.5-2

TVA DWG. NO. 47W850-2 R0



LEGEND:

- ⊘ FIRE DETECTOR - IONIZATION TYPE
- ● THERMO SWITCH (FIRE DETECTOR) RATE OF RISE TYPE
- ⇾ CO_2 NOZZLE
- ☐◁ PRE DISCHARGE ALARM
- ⊡ PUSHBUTTON STA. WITH TIMER (UNLESS OTHERWISE NOTED)
- ⊘ TIMER
- THREE-WAY SOLENOID VALVE
- PISTON OPERATED VALVE - CO_2 CONTROLLED
- ⊠ DOOR CONTROL - CO_2 OPERATED
- ☐ DAMPER CONTROL - CO_2 OPERATED

NOTES:
1. ALL VALVES ARE THE SAME SIZE AS PIPING UNLESS OTHERWISE SPECIFIED.
2. MAIN PROCESS VALVES ARE SHOWN IN THEIR NORMAL OPERATING POSITION.
3. IDENTIFICATION OF VALVE SYMBOLS IS SHOWN ON TVA DWG GM-4-30B617-3.
4. EQUIPMENT, PIPING, AND VALVES IN THE TURBINE BUILDING AND SERVICE BUILDING ARE NONNUCLEAR SAFETY CLASS (NNS).
5. EQUIPMENT, PIPING, AND VALVES IN THE AUXILIARY BUILDING, CONTROL BUILDING, AND DIESEL GENERATOR BUILDING ARE ANS SAFETY CLASS 3 (SC-3) AND SEISMIC CLASS 1.

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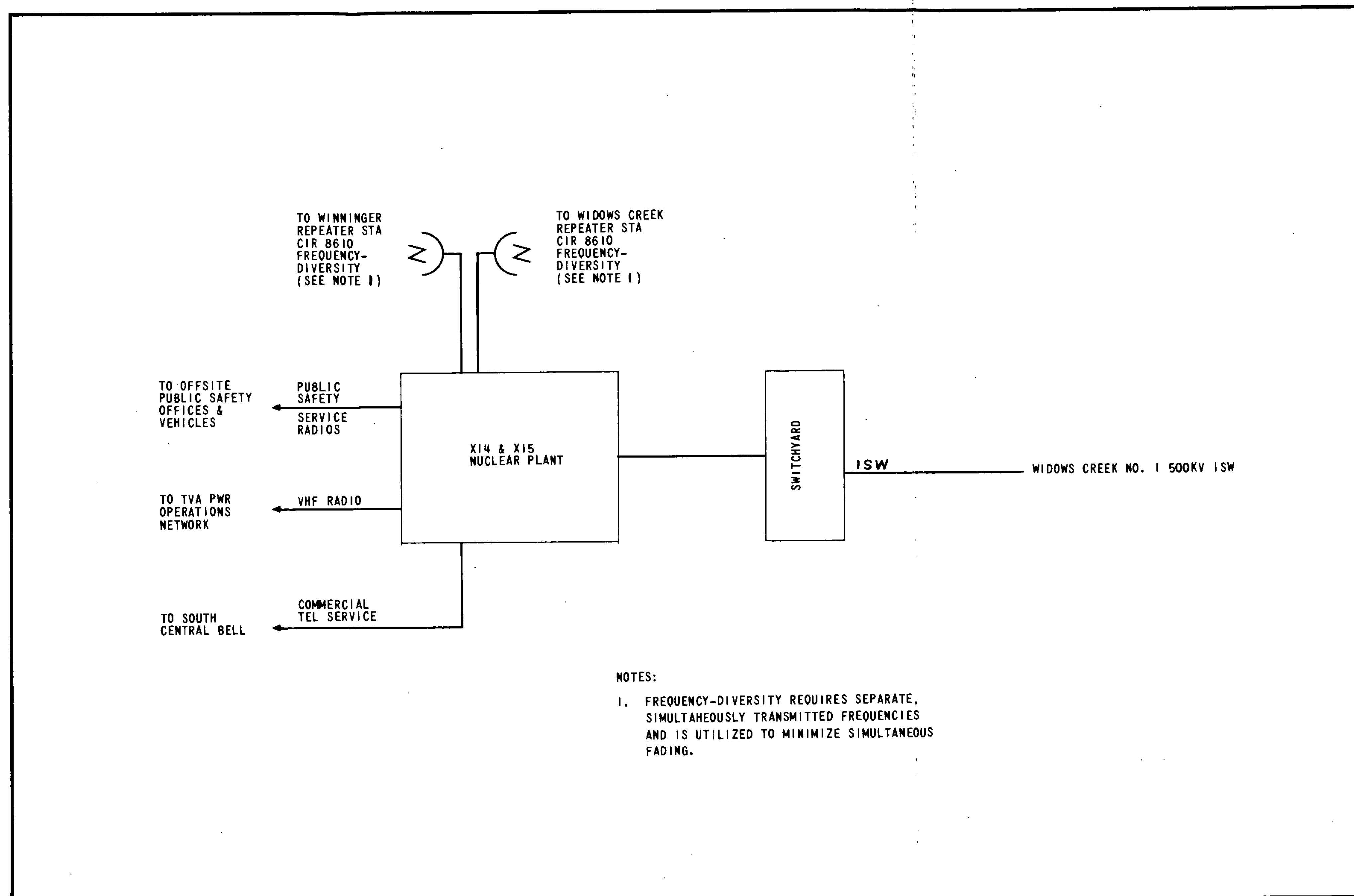
CO_2 STORAGE, FIRE PROTECTION AND PURGING SYSTEM-FLOW DIAGRAM

FIGURE 9.5-3

TVA DWG.NO. 47W843-1 R0



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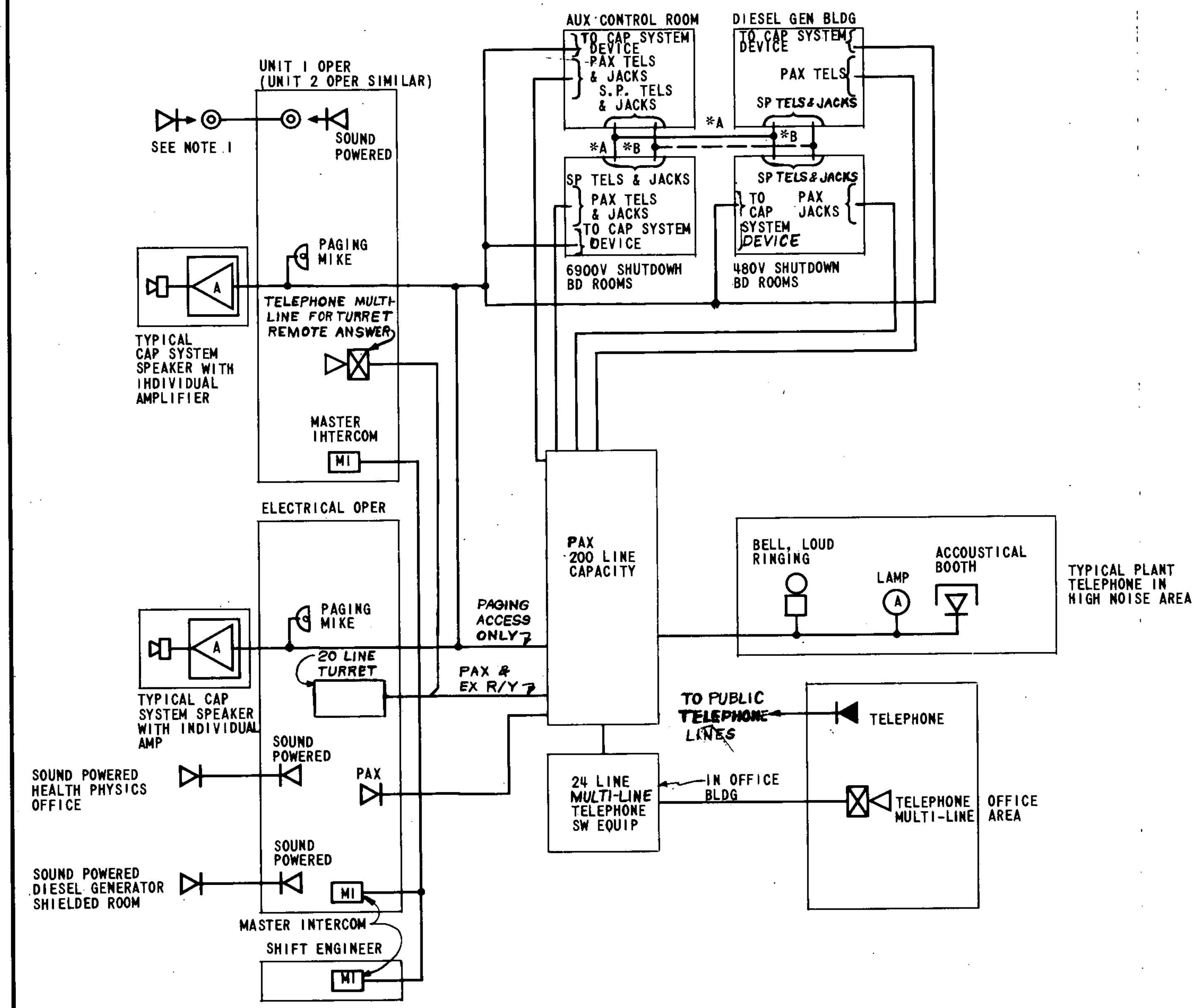
PLANT-TO-OFFSITE COMMUNICATIONS

FIGURE 9.5-4



NOTES:

1. SIX SYSTEMS OF JACKS ARE PROVIDED FOR USE WITH SOUND POWERED INSTRUMENTS:
   SP-1 TURBINE CONTROL, GENERATOR & AUX-POWER
   SP-2 FEEDWATER, STEAM & CONDENSATE SYSTEM
   SP-3 REACTOR CONTROL
   SP-4 REACTOR COOLANT SYSTEM & AUXILIARY STEAM
   SP-5 ENGINEERED SAFEGUARDS & AUXILIARY SYSTEMS
   SP-6 REFUELING SYSTEM
2. THE CAP SYSTEM REFERRED TO IS A COMBINED CODE-CALL, FIREALARM, EVACUATION, ALL CLEAR & PAGING SYSTEM. FIGURE C SHOWS THIS SYSTEM IN MORE DETAIL.

* - SYSTEM A & B ARE CCOMPLETELY REDUNDANT

FIGURE 9.5.2.B
X14 & X15 NUCLEAR PLANT
INTRAPLANT COMMUNICATIONS
(SIMPLIFIED)

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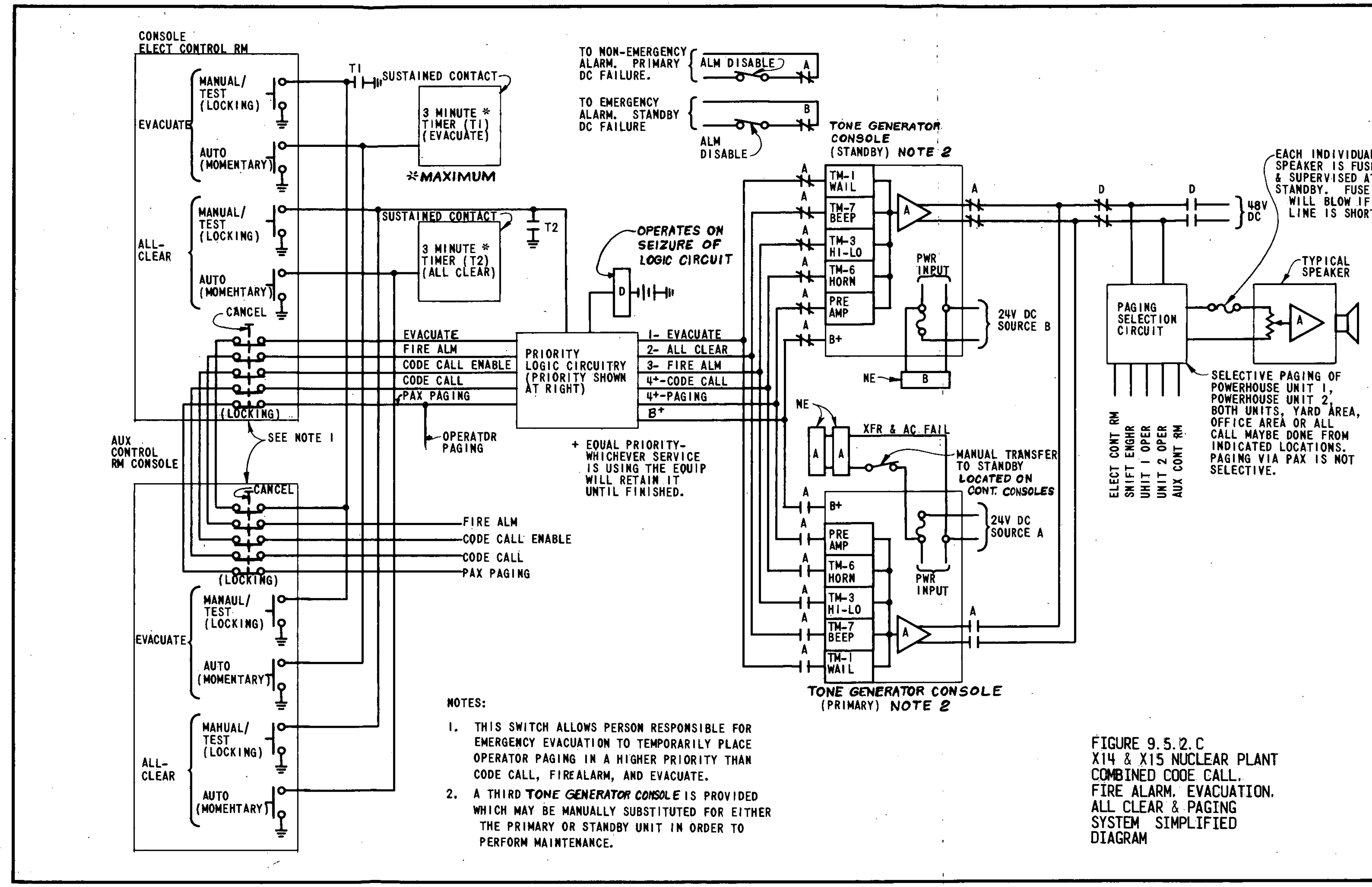
INTRAPLANT COMMUNICATIONS

FIGURE 9.5-5



NOTES:

1. THIS SWITCH ALLOWS PERSON RESPONSIBLE FOR EMERGENCY EVACUATION TO TEMPORARILY PLACE OPERATOR PAGING IN A HIGHER PRIORITY THAN CODE CALL, FIREALARM, AND EVACUATE.
2. A THIRD TONE GENERATOR CONSOLE IS PROVIDED WHICH MAY BE MANUALLY SUBSTITUTED FOR EITHER THE PRIMARY OR STANDBY UNIT IN ORDER TO PERFORM MAINTENANCE.

FIGURE 9.5.2.C
X14 & X15 NUCLEAR PLANT
COMBINED COOE CALL.
FIRE ALARM. EVACUATION.
ALL CLEAR & PAGING
SYSTEM SIMPLIFIED
DIAGRAM

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COMBINED COOE CALL, FIRE ALARM EVACUATION, ALL CLEAR ANO PAGING SYSTEM- SIMPLIFIEO DIAGRAM

FIGURE 9.5-6

POSTULATED CONDITIONS	INTERPLANT COMMUNICATIONS					INTRAPLANT COMMUNICATIONS							
	MICROWAVE RADIO (MW)	INSULATED SHIELD WIRE CARRIER	PUBLIC TELEPHONE LINES	POWER SYSTEM OPERATIONS RADIO	PUBLIC SAFETY SERVICE (SECURITY) VHF RADIO	PRIVATE AUTOMATIC TELEPHONE EXCHANGE (PAX)	MANUAL TELEPHONE TURRET	ADMINISTRATIVE KEY-OPERATED TELEPHONE SYSTEM	SOUND POWERED TELEPHONE SYSTEMS	OPERATIONAL INTERCOM SYSTEM	CLOSED-CIRCUIT TELEVISION (CCTV)	CODES, ALARMS & PAGING SYSTEM	
FIRE IN COMMUNICATIONS ROOM (TOTAL DESTRUCTION)				X	X			PARTIAL	X	X	X		
FIRE IN CABLE TUNNEL TO SWITCHYARD	X			X	X	X	X	X	X	X	X	X	
FIRE IN CONTROL ROOM	X	X	X		X	X		X	PARTIAL	PARTIAL		PARTIAL	
DBA	X	X	X	X	X	X	X		X			X	
SSE					PARTIAL (VEHICULAR & PORTABLE UNITS)				X				
LOSS OF OFFSITE POWER	X	X	X	X	X	X	X		X			X	
LOSS OF ALL AC POWER FOR UP TO 8 HOURS	X	X	X		X	X	X		X			X	
MAXIMUM POSSIBLE FLOOD				X	X				PARTIAL				
TORNADO (MICROWAVE ANTENNAS & REFLECTORS DESTROYED)		X	X	X	X	X LOSS OF MW TRUNKS	X LOSS OF MW TRUNKS	X	X	X	X	X	

NOTES:

1. X IN BLOCK INDICATES AVAILABILITY OF THE SERVICE DURING THE POSTULATED CONDITION.
2. PARTIAL IN BLOCK INDICATES THE LOSS OF THAT PORTION OF THE SYSTEM LOCATED WHERE THE ACCIDENT OCCURRED. THE SURVIVING EQUIPMENT WILL REMAIN FUNCTIONAL.

FIGURE 9.5.2.D
COMMUNICATIONS EQUIPMENT
AVAILABILITY TABLE
X14 & X15 NUCLEAR PLANT

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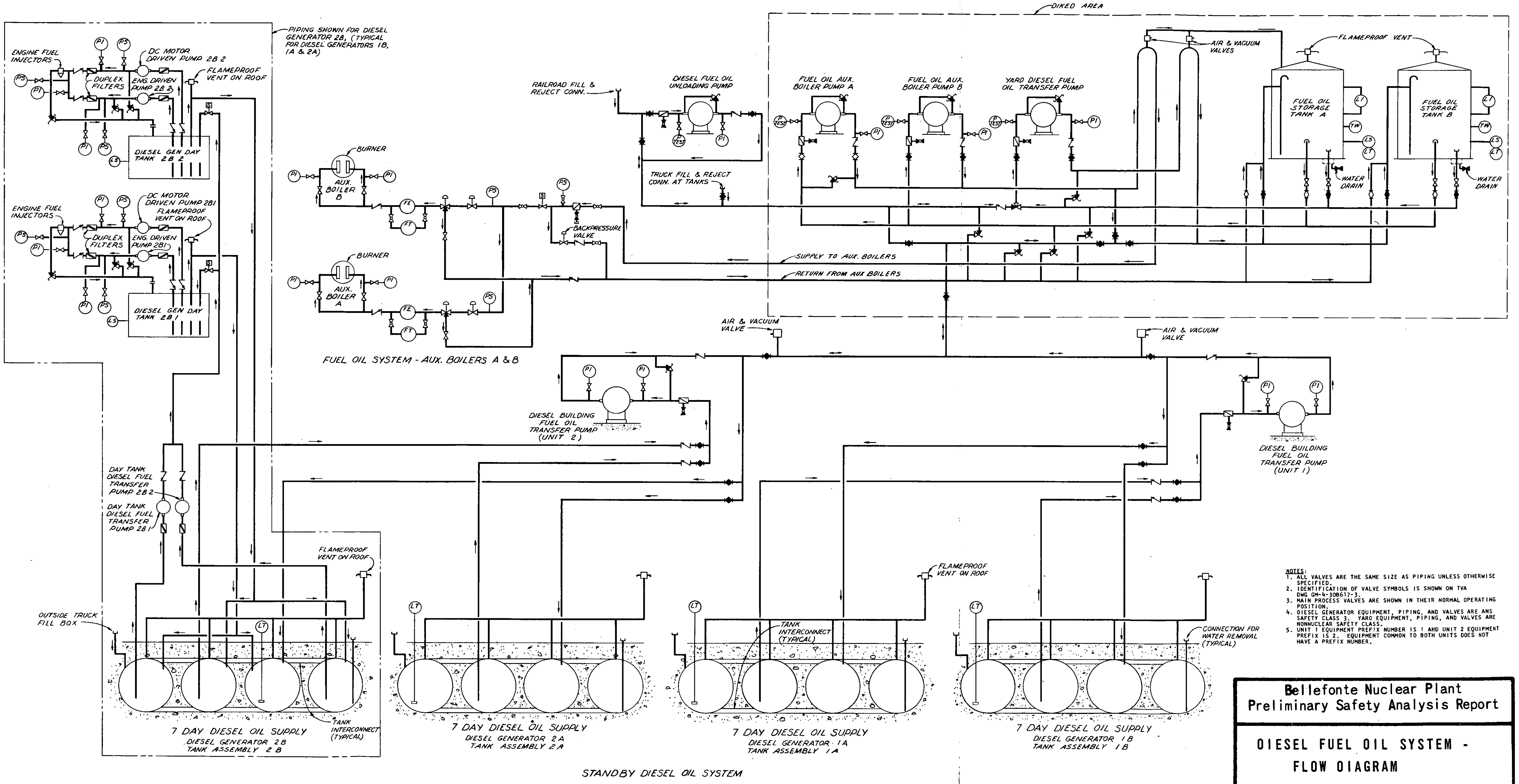
PLANT COMMUNICATIONS
EQUIPMENT AVAILABILITY TABLE

FIGURE 9.5-7



NOTES:
1. ALL VALVES ARE THE SAME SIZE AS PIPING UNLESS OTHERWISE SPECIFIED.
2. IDENTIFICATION OF VALVE SYMBOLS IS SHOWN ON TVA DWG GM-4-308617-3.
3. MAIN PROCESS VALVES ARE SHOWN IN THEIR NORMAL OPERATING POSITION.
4. DIESEL GENERATOR EQUIPMENT, PIPING, AND VALVES ARE ANS SAFETY CLASS 3. YARD EQUIPMENT, PIPING, AND VALVES ARE NONNUCLEAR SAFETY CLASS.
5. UNIT 1 EQUIPMENT PREFIX NUMBER IS 1 AND UNIT 2 EQUIPMENT PREFIX IS 2. EQUIPMENT COMMON TO BOTH UNITS DOES NOT HAVE A PREFIX NUMBER.

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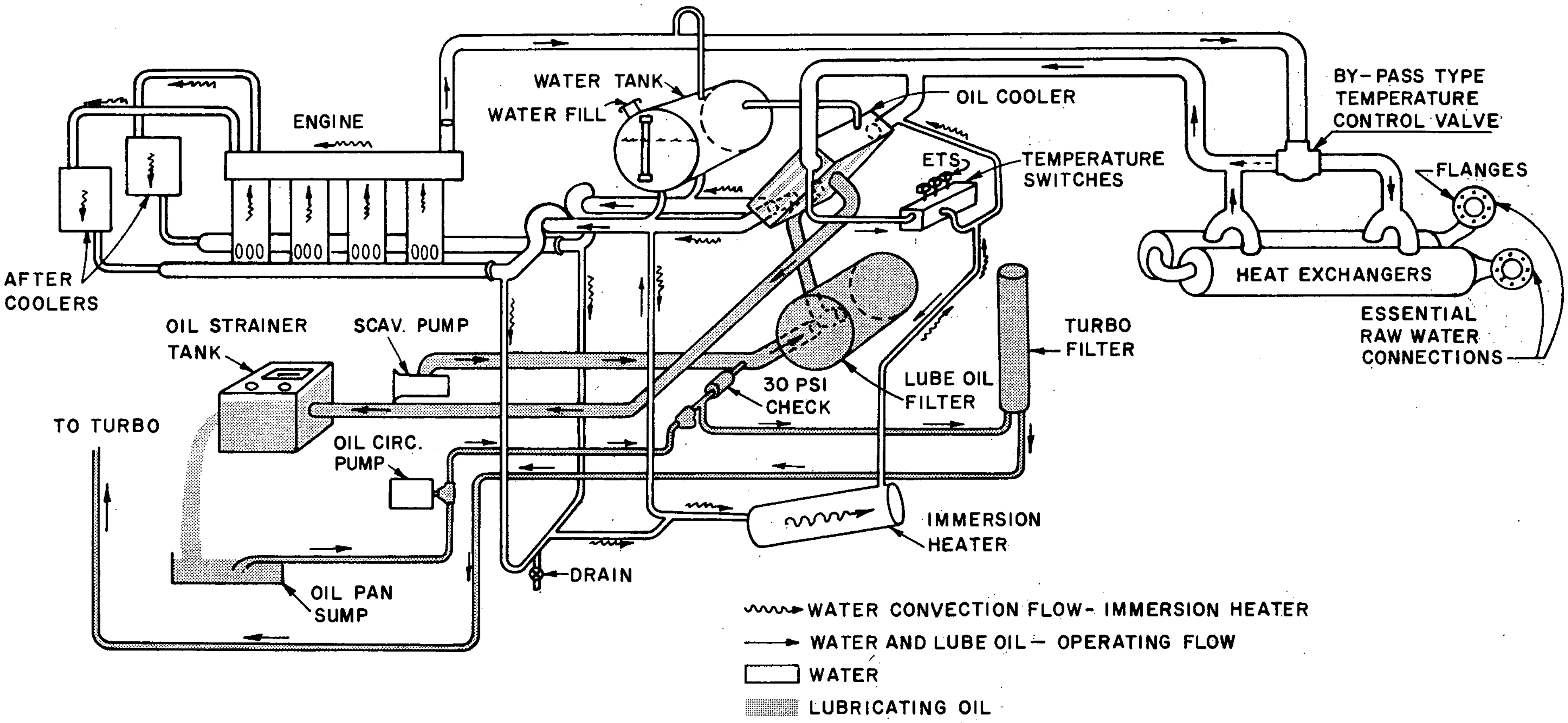
DIESEL FUEL OIL SYSTEM - FLOW DIAGRAM

FIGURE 9.5-8

TVA DWG.NO. 47W840-1 R0



LUBRICATING OIL SYSTEM
AND ENGINE COOLING WATER SYSTEM
WITH IMMERSION HEATER SYSTEM - TURBOCHARGED UNITS

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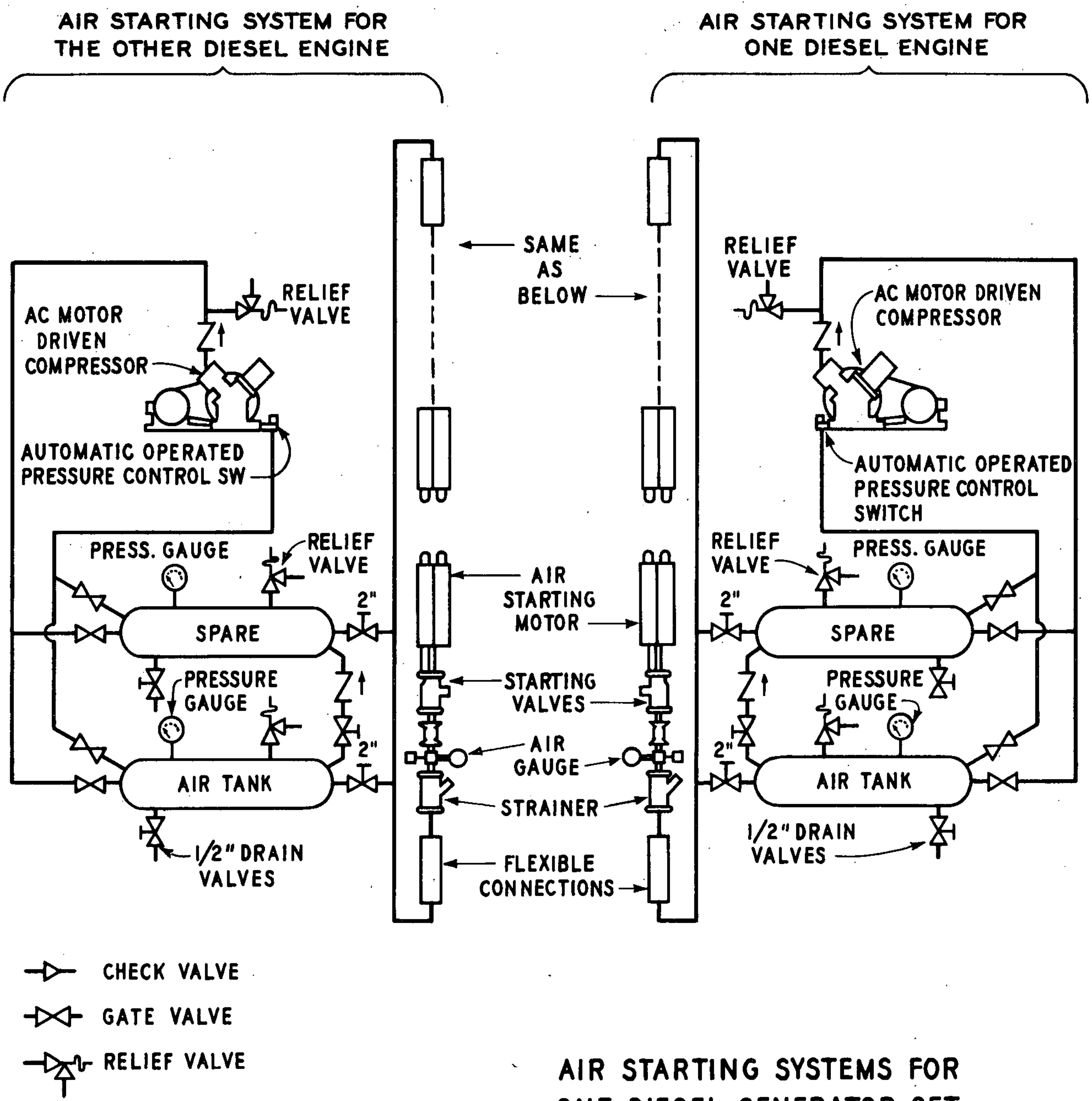
LUBRICATING DIL SYSTEM AND
ENGINE COOLING WATER SYSTEM
FIGURE 9.5-9

REVISED PER AMENB 11, MAY 15, 1974



AIR STARTING SYSTEMS FOR
ONE DIESEL GENERATOR SET

Bellefonte Nuclear Plant Preliminary Safety Analysis Report
AIR STARTING SYSTEMS FOR ONE DIESEL GENERATOR SET FIGURE 9.5-10 REVISED PER AMEND 11, MAY 15, 1974

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CHAPTER 10.0. STEAM AND POWER CONVERSION SYSTEM

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10.0. STEAM AND POWER CONVERSION SYSTEM

10.1. Summary Description

The steam and power conversion system is designed to produce electrical power from heat produced in the reactor and transmitted through the steam generators. The waste heat is rejected to the atmosphere through the plant's cooling water system.

The major components of the steam and power conversion system will be: turbine-generator, main condenser, vacuum pumps, turbine seal system, turbine bypass system, hotwell pumps, condensate booster pumps, main feed pumps, main feed pump turbine (MFPT), condenser-feedwater heater, feedwater heaters, heater drain pumps, and condensate storage system.

The superheated steam produced by the steam generators will be expanded through the high-pressure turbine and then exhausted to the moisture separator/reheaters. The moisture separator section will remove the moisture from the steam and the two-stage reheaters will superheat the steam before it enters the low-pressure turbines. The steam will then expand through the low-pressure turbines and exhaust into the main condenser where it will be condensed and deaerated and then returned to the cycle as condensate. The heat rejected in the main condenser will be removed by the circulating water system.

The first-stage reheater is supplied with steam from an extraction point; the condensed steam is cascaded to the No. 2 heater. The second-stage reheater is supplied with main steam; the condensed steam will cascade to the highest pressure (No. 1) heater.

Condensate from the hotwells is transmitted by the hotwell pumps through the gland steam condenser, the demineralizer, and into the suction of the condensate booster pumps. The condensate booster pumps pump the water through main feed pump condensers, five stages of feedwater heaters, and into the suction of main feedwater pumps, which forward the water through two stages of feedwater heaters and into the inlet of the steam generators. Heat for the feedwater heating cycle will be supplied by the moisture separator reheater drains and by steam from the turbine extraction points. The cycle arrangement is shown in Figure 10.1-1, General Steam Cycle Diagram. The important design and performance characteristics of the steam and power conversion system are summarized in Tables 10.1-1 and 10.1-2.

The safety related equipment described in this chapter includes the auxiliary feedwater system, the feedwater isolation valves and downstream piping, and the main steam isolation valves and upstream piping, including ASME Code safety valves and the modulating power operated relief (atmospheric dump) valves.

Table 10.1-1. Important Design and Performance Characteristics — Bellefonte Nuclear Plant Units 1 and 2

Vertical Steam Generators

Manufacturer	Babcock & Wilcox	
Type	Once-through	
Number	2 per unit	
Length (overall),	75'-5"	
Diameter (OD, maximum), in.	146.75	
Heating surface, ft^2	51,500	
Tubes (straight tubes)		
Dimensions	0.625 in. OD × 0.034 min wall,	
Material	Inconel (ASME-SB-163)	
Number	16,056	
Operating conditions at 100% load		
Steam flow rate (each), 10^6 lb/h	8.00	
Steam temperature, F	602	11
Steam pressure, psia	1060	
Steam quality (superheat), F	50	

Turbogenerator

Manufacturer	Brown Boveri Corporation	
Turbine rating (max), kW	1,332,000	
Turbine type	Horizontal, tandem-compound, single reheat, extraction, condensing, 1800-rpm, 1 HP and 2 LP turbines with four-flow exhaust and 52-inch last-stage buckets	
Generator type and max rating	1 direct connected, H_2 cooled rotor, water-cooled stator, 1,480,000 kVA, 0.9 pf, 75 psig H_2, 3 ph, 60 Hz, 4 pole, 24,000 V, 35,640 A, 0.58 scr	
Exciter type and rating	1 shaft-driven, brushless, 7600 A, 660 V, 1800 rpm	
Heat rate		
kW	1,263,215	11
Btu/kWh	9782	
Description	Guaranteed performance based on extraction for feedwater heating,	

Table 10.1-1. (Cont'd)

	including all losses in the unit, also exciter and rheostat losses, rated throttle steam conditions, and 2.0 in. of Hg abs exh press., with zero makeup
Moisture separator and reheaters	
Number	2 per unit
Type	Wire mesh separator, two-stage reheat
Size	99'-2" length, 13'-0" diameter
<u>Main Feedwater Pump Turbine</u>	
Number (1 turbine per pump)	2
Type	Multi-stage, dual inlet
Throttle pressure	LP steam, 125 psia; HP steam, 1032 psia
Throttle temperature	LP steam, 521; HP steam, 602F
Back pressure	9 in. Hg abs
Extraction points	None
Rated horsepower	14,000
Main feedwater pumps	
Number	2
Type	Single-stage, double suction, centrifugal, horizontal
Design point	19,200 gpm, 2600 ft head
Condensate booster pumps	
Number	3 11
Type	Single-stage, double suction, centrifugal, horizontal
Design point	8155 gpm, 1025 ft head
Motor design	2250 hp, 3550 rpm, 6600 V, 3 ph, 60 Hz, horizontal, constant speed
No. 3 heater drain pumps	
Number	3
Type	Single-stage, double suction, centrifugal, horizontal
Design point	4600 gpm, 1100 ft head

Table 10.1-1. (Cont'd)

Motor design	2000 hp, 3550 rpm, 6600 V, 3 ph, 60 Hz, horizontal, constant speed
No. 7 heater drain pumps	
Number	2
Type	Multi-stage, centrifugal, horizontal
Design point	2000 gpm, 1100 ft
Motor design	800 hp, 3550 rpm, 6600 V, 3 ph, 60 Hz, horizontal, constant speed
Condensate hotwell pumps	
Number	3
Type of pump	Multi-stage, single suction, vertical
Design point	8155 gpm, 315 ft head
Motor design	700 hp, 1180 rpm, 6600 V, 3 ph, 60 Hz, vertical, constant speed
Condenser	
Number	1
Type	Horizontal, single shell, single-pass, surface, deaerating, dual pressure
Surface, ft^2	860,000
Tube sheets	Copper bearing steel
Waterboxes	Divided, two inlet and two outlet bottom connections
Hotwell data	Deaerating type, storage capacity of hotwell at normal operating level, 42,300 gal
Air removal equipment	
Number	3
Type	Mechanical, vacuum
Design point	Suction press., in. Hg abs - 1.0, rated capacity, each - 10 scfm
Motor design	460 V, 3 ph, 60 Hz, horizontal, constant speed
Feedwater Heaters	
Number	14 (7 stages, divided into 2 streams) plus 2 feed pump turbine condensers
Type	Closed, horizontal, U-tube
Tubes	304 SS

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Table 10.1-1. (Cont'd)

Turbine Bypass Valves

Number of condenser dump valves	3 (modulating)	6 (ON/OFF)
Design pressure	1235 psig and 29 in. Hg vacuum	
Design temperature, F	630 and 79	630 and 79
Flow per valve, lb/h	855,000	992,000 11
Valve actuation	Compressed air	Compressed air
Main steam pressure at valve inlet (for above flow), psig	1035	1035
Minimum flow/valve at 50 psia inlet pressure, lb/h	22,500	--
Time to open (full stroke), s	3	0.5
Failure position	Closed	Closed

Power Operated Relief (Atmospheric Dump) Valves

Number of power operated relief valves	
On/off type	4
Modulating type	2
Design pressure	1235 psig and 29 in. Hg vacuum
Design temperature, F	630 and 79
Flow per valve	
On/off type, lb/h	850,000
Modulating type, lb/h	591,500
Valve actuation	Compressed air
Main steam pressure at valve inlet (for above flow), psig	1035
Time to open (full stroke)	
On/off type, s	0.5
Modulating type, s	3
Failure position	Closed

Safety Relief Valves

No. valve	Set press., psig	Accumulation		Blowdown	
		%	Press., psig	%	Press., psig
5	1225	3	1262	2-4	1176
6	1245	3	1282	2-4	1195

Table 10.1-2. General Steam Cycle Operating Conditions

p = Pressure, psia; t = Temperature, F; m = Flow rate, lb/h

Location[a]	Rated power	Stretch power	Location	Rated power	Stretch power	
Main Steam						
MS 1	p = 1060.0 t = 601.7 m = 16,068,943	1080 606.1 16799056	MS 7	p = 143.5 t = 355.0 m = 5,651,065	151.9 359.4 5882851	
MS 2	p = 1032.5 t = 601 m = 13,242,082	1048 600 13147374	MS 8	p = 133.1 t = 521.6 m = 5,003,921	142.9 524.1 5208845	
MS 3	p = 1022.2 t = 596.8 m = 362,026	1038 600.7 369492	MS 9	p = 143.5 t = 355.0 m = 5,651,065	151.9 359.4 5882851	
MS 4	p = 1022.2 t = 596.8 m = 362,025.5	1038 600.7 369492	MS 10	p = 133.1 t = 521.6 m = 4,929,960	142.9 524.1 5208845	11
MS 5 Normal	p = 1060.0 t = 601.7 m = 2,093,314	1080 606.1 2902363	MS 11	p = 1.081 t = 104.34 m = 4,477,170	0.884 97.7 4725936	
MS 6	p = 1060.0 t = 601.7 m = 0(b)	1080 606.1 0	MS 12	p = 1.669 t = 119.50 m = 4,138,599	1.439 114.2 4300560	
			MS 13	p = 129.5 t = 521.0 m = 295,557	135.8 522.9 273362	
Extraction						
EX RH1	p = 444.0 t = 459.2 m = 644,856	463.3 459.2 684256	EX H5	p = 23.75 t = 234.4 m = 358,553	25.05 237.4 364863	
EX EX H1N H1 Normal	p = 666.2 t = 539.7 m = 1,533,268	700.1 547.1 1605128	EX H6	p = 11.90 t = 199.1 m = 404,483	12.82 202.7 343147	

(a) Locations marked on Figure 10.1-1.

(b) Used for emergency and startup.

Table 10.1-2. (Cont'd)

Location	Rated power	Stretch power	Location	Rated power	Stretch power
EX	p = 666.2	700.1	EX	p = 0	6.097
H1	t = 539.7	547.1	H7A	t = 0	168.5
Overload	m = 1,533,268	1605128		m = 0	106595
EX	p = 666.2	700.1	EX	p = 0	5.382
H1N	t = 539.7	547.1	H7B	t = 0	163.2
Overload	m = 560,046	1297235		m = 0	76419
EX	p = 302.7	315.8	EX	p = 4.22	3.23
H2	t = 413.5	417.4	TA	t = 155.1	144.4
	m = 959,033	1031849		m = 147,778.5	136681
EX	p = 145.0	151.9	EX	p = 4.22	3.23
H3	t = 351.8	355.4	TB	t = 155.1	144.4
	m = 898,695	965709		m = 147,778.5	136681
EX	p = 52.14	55.06			
H4	t = 349.6	351.1			
	m = 481,241	506062			
Condensate					
C	p = 1.669	1.44	C	p = 300.0	532
1	t = 119.5	114.2	8	t = 151.8	161.6
	m = 2,899,509	2979248		m = 5,119,132	5349234
C	p = 100.0	120	C	p = 300.0	514
2	t = 119.5	114.2	9	t = 195.7	197.7
	m = 8,698,528	9028023		m = 5,119,132	5349234
C	p = 100	120	C	p = 300.0	502
3	t = 119.7	114.4	10	t = 231.4	232.4
	m = 8,698,528	9028023		m = 5,119,132	5349234
C	p = 100.0	120	C	p = 300.0	488
4	t = 119.7	114.4	11	t = 277.8	278.9
	m = 8,698,528	9028023		m = 5,119,132	5349234
C	p = 300	550	C	p = 300	476
5	t = 119.8	114.4	12	t = 348.8	350.4
	m = 4,349,264	4514012		m = 5,119,132	5349234
C	p = 300.0	550	C	p = 300.0	476
6	t = 152.2	143.2	13	t = 349.8	352.3
	m = 4,349,264	4514012		m = 8,034,471.5	8399519
C	p = 300.0	532			
7	t = 152.2	160.6			
	m = 4,349,264	4514012			

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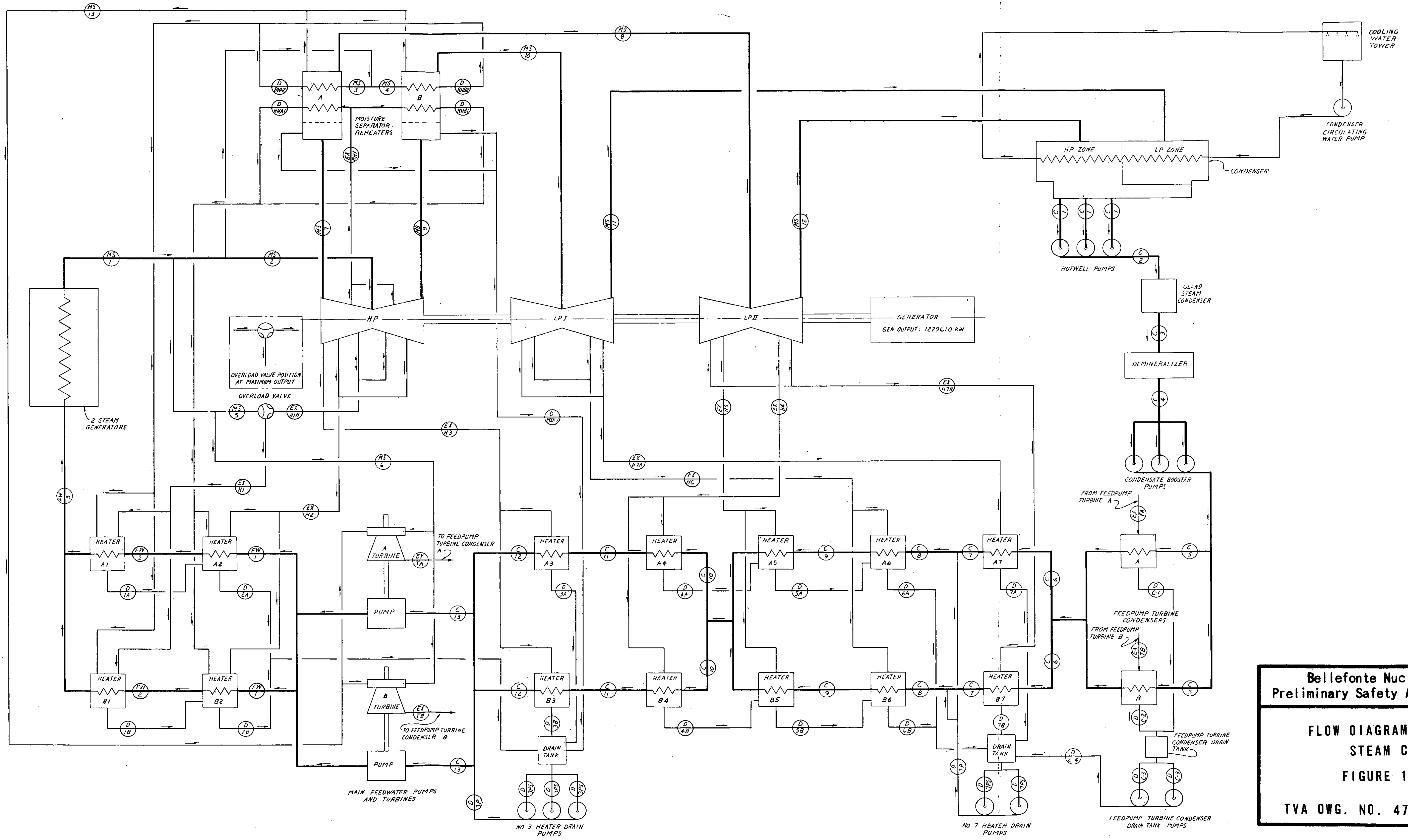
Table 10.1-2. (Cont'd)

Location	Rated power	Stretch power	Location	Rated power	Stretch power	
Feedwater						
FW 1	p = 1247.3	1250	FW 3	p = 1247.3	1214	
	t = 352.1	354.5		t = 489.0	492.4	
	m = 8,034,471.5	8399159		m = 16,068.943	16799056	
FW 2	p = 1247.3	1232.5				
	t = 410.5	414.5				
	m = 8,034,471.5	8399159				
Heater Drains and Vents						
D RHA1	p = 439.5	458.7	D 2A	p = 287.6	300.0	
	t = 454.0	458.3		t = 362.1	364.5	
	m = 322,428	342128		m = 1,930,604	2030108	
D RHA2	p = 1011.7	1027.6	D 2B	p = 287.6	300.0	
	t = 546.0	547.9		t = 362.1	364.5	11
	m = 362,025.5	369492		m = 1,930,604	2030108	
D RHB1	p = 439.5	458.7	D 3A	p = 137.7	144.2	
	t = 454.0	458.3		t = 351.8	355.4	
	m = 322,428	342128		m = 475,509	482854	
D RHB2	p = 1011.7	1027.6	D 3B	p = 137.7	144.2	
	t = 546.0	547.9		t = 351.8	355.4	
	m = 362,025.5	369492		m = 475,509	482854	
D 1A	p = 632.9	665.1	D 4A	p = 49.79	52.31	
	t = 420.5	424.6		t = 241.4	242.4	
	m = 1,128,659.5	1172056		m = 240,620.5	253031	
D 1B	p = 632.9	665.1	D 4B	p = 49.79	52.31	
	t = 420.5	424.6		t = 241.4	242.4	
	m = 1,128,659.5	1172056		m = 240,620.5	253031	
D 5A	p = 22.56	23.80	D C-3	p = 4.22	3.23	
	t = 205.7	207.7		t = 155.1	144.4	
	m = 419,897	435462		m = 147,778.5	136681	
D 5B	p = 22.56	23.80	D C-4	p = 11.31	5.5	
	t = 205.7	207.7		t = 155.1	144.6	
	m = 419,897	435462		m = 295,557	273362	
D 6A	p = 11.31	12.18	D 3PS	p = 137.7	144.2	
	t = 119.1	171.6		t = 351.4	355.4	
	m = 202,241.5	605236		m = 1,943,560	2013188	

Table 10.1-2. (Cont'd)

Location	Rated power	Stretch power	Location	Rated power	Stretch power	
D	p = 11.31	12.18	D	p = 300.0	476	
6B	t = 199.1	171.6	3P	t = 351.8	355.4	
	m = 202,241.5	605236		m = 5,830,679	6100571	
D	p = 0	5.79	D	p = 11.31	5.5	11
7A	t = 0	168.5	7PS	t = 194.3	165.8	
	m = 0	106595		m = 769,917	835222	
D	p = 0	5.11	D	p = 300	532	
7B	t = 0	163.2	7P	t = 194.3	165.8	
	m = 0	76419		m = 1,539,834	1670444	
D	p = 4.22	3.23	D	p = 142.1	148.9	
C-1	t = 155.1	144.4	MSR-1	t = 354.2	357.8	
	m = 147,778.5	136681		m = 1,070,780	1074651	
D	p = 4.22	3.23				
C-2	t = 155.1	144.4				
	m = 147,778.5	136681				



Bellefonte Nuclear Plant
Preliminary Safety Analysis Report

FLOW DIAGRAM GENERAL STEAM CYCLE

FIGURE 10.1-1

TVA DWG. NO. 47W800-1 R0

10.2. Turbine Generator

10.2.1. Design Bases

The purpose of the turbine generator is to use steam supplied by the PWR system in the conversion of thermal energy to electrical energy, and to provide extraction steam for feedwater heating. The turbine generator together with its associated systems and their control characteristics will be integrated with the features of the reactor and its associated systems to obtain an efficient and safe energy conversion and power generation unit.

The turbine receives steam from the two steam generators and converts the thermal energy to electric energy. The generator produces 1,234,950 kW when the turbine passes 14,053,000 lb/h of steam at throttle conditions of 1050 psia, 50F superheat, and at a back pressure of 2.0 inches Hg absolute, with 0% makeup under normal conditions. Under emergency conditions the turbine protection system provides the necessary protection for the turbine-generator equipment. This protection system is arranged to trip the turbine on low condenser 1
vacuum, excessive shaft vibration, abnormal thrust bearing wear, low bearing oil pressure, and indication of electrical distribution system malfunctions. In addition, two overspeed governors are provided to prevent overspeed.

The intended mode of operation for the units is that they will be utilized primarily as base loaded units.

The functional limitation on the turbine imposed by the nuclear steam system (NSS) is that the NSS accepts step load changes of ±10% and ramp load changes of ±5% per minute over the range of 15% to 90% and ±3% per minute over the range of 90% to full power.

10.2.2. Description

The turbine generator unit will consist of the following components: turbine, generator, exciter, controls, and required support subsystems. The turbine will be a tandem compound double-stage reheat unit with 52-inch last-stage blades, designed for inlet steam conditions of 50F superheat and 1050 psia with a throttle flow of 14,053,000 lb/h. It will consist of a double-flow, high-pressure turbine and two double-flow low-pressure turbines with extraction nozzles arranged for seven stages of feedwater heating. Exhaust steam from the HP 1
unit will pass through two moisture separator/reheaters before entering the low-pressure turbines. The separators will reduce the moisture content of the steam after which it will be superheated in the reheaters.

The generator will be a direct-connected, hydrogen-cooled, three-phase, 60-Hz, 24,000-volt, 1800-rpm synchronous generator rated at 1,480,000 kVA, 0.90 power factor (pf), with a short circuit ratio of 0.58, designed with conductor cooling of the armature winding. Hydrogen gas pressure is 75 psig, and conductor coolant is demineralized water. The excitation system is rated at 7600 amperes and 660 volts, using cabinet-mounted solid state electronic devices for the establishment of generator field current.

The turbine control system is functionally divided into two parts: the speed/load control system and the safety or turbine protection system. Speed/load of the turbine is controlled by a mechanical hydraulic control (MHC) system. 7
The safety system, or turbine protection system (TPS), and the MHC system execute their respective functions hydraulically, with certain subfunctions

being carried out pneumatically, electrically, or mechanically. A feature of both systems is the use of mineral oil or turbine lube oil as the hydraulic medium for initial functions, and the use of a fire-resistant control fluid as the final medium. This arrangement permits the use of the fire-resistant fluid near and around high-temperature portions of the turbine and confines the mineral oil to low-temperature and more readily protectable areas of the turbine generator, reducing the consequences of fire damage due to an accidental release of pressurized combustible fluids. The MHC system and the TPS are shown schematically in Figure 10.2-1; the legend for Figure 10.2-1 is listed in Figures 10.2-2 through 10.2-7.

The MHC speed/load control system is generally of conventional design, using hydraulic servo-valve-controlled power pistons as the final control elements for precise positioning of the steam flow control valves. The error or command signal to the servo-valves is generated at the speed governor, the governor output pressure signal being the equivalent algebraic sum of the turbine shaft speed and the reference or loading pressure generated by the operating device (item 1205, Figure 10.2-1). The governor output error/demand oil pressure signal is transmitted to control, overload, and intercept valve servo-valves, resulting in steam flow through the turbine that is proportional to the output pressure of the speed governor. (Steam flow versus error/demand oil pressure is linearized by appropriate nonlinear characterization of the power cylinder feedback signal to the servo-valves at the respective valve operator assemblies.)

The oil pressure error/demand signal can also be modulated by the initial pressure limiter (item 1223, Figure 10.2-1), a device whose function is to
reduce turbine load in the event of a precipitous decrease in inlet steam 7
pressure. This prevents opening of the control valves on loss of steam pressure, which normally would occur as the control system tries to hold the load at a constant value. The limiter remains inactive during pressure increases or slow pressure reductions.

In addition, the input reference oil pressure signal to the speed governor flows through a derivative element (item 1203, Figure 10.2-1). The ultimate function of this device is to bring about a rapid decrease of the error/demand oil pressure signal and, sequentially, a rapid closure of the control, overload, and intercept valves on the occurrence of a high acceleration rate of the turbine shaft (such as a potential shaft overspeed occasioned by a generator breaker trip at full load). The derivative device performs an anticipatory function, working through the control, overload, and intercept valves to prevent or ameliorate any possible overspeed.

The fluid pressure error/demand signal is also modulated by a pair of vacuum limiting devices (items 1221 and 1222, Figure 10.2-1). Their purpose is to bring about a reduction in turbine inlet steam flow, the steam flow reduction being proportional to an increase in exhaust backpressure. The limiting action begins on an increase in backpressure to 5.0 in. Hg abs and provides for full closure of all of the control valves at a backpressure of 9.0 in. Hg abs.

A pressure gradient relay is installed (item 1220, Figure 10.2-1) to limit the loading rate of the turbine. It limits the rate of pressure increase in the fluid error/demand system, so that a maximum sudden pressure rise is limited to a certain ratio, which is spring-adjustable. Above this rate, the pressure rises at a constant dp/dt, which is also adjustable.

The turbine protection system utilizes two emergency or overspeed governors as the ultimate protective devices to protect the turbogenerator against excessive overspeed. The two overspeed governors (items 2201 and 2203, Figure 10.2-1) are built into the turbine shaft; they consist of spring-loaded bolts having eccentric centers of gravity. At overspeed, the bolts will move to their extreme position, thereby unlatching the respective trip pawls. The unlatched pawls permit the oil pressure in the respective emergency oil circuits to go to zero, resulting in the fast closure of all main stop, overload stop, and reheat stop valves; they also initiate a closing signal to all turbine nonreturn valves. The overspeed governors are sequentially set to operate at 110 and 112% of rated speed. The overspeed bolts will reset automatically at a speed slightly above rated speed (approximately 103%).

Each overspeed governor can be on-line tested periodically to ensure that the mechanism is functionable and free to move. The testing operation temporarily isolates one emergency oil system and then introduces oil into the rotating shaft and to the overspeed bolt in such a manner that the center of mass of the bolt is so altered as to induce the required bolt movement and subsequent trip. The oil inflow pressure buildup is monitored by a pressure gage; if the bolt is free and in good condition, the oil pressure required to induce bolt movement is always the same for each simulated test. Monitoring of trip oil pressure thereby produces data for comparative observations of the condition of the overspeed governor. The use of two overspeed governors, coupled with the isolation hardware for the overspeed governor under test, leaves system protection to the other overspeed governor, so that overspeed protection is maintained, and the safety of the set is ensured even while testing. The tested emergency system is reset automatically through the test device (item 7
4203, Figure 10.2-1), which, before returning to its normal position, resets the trip pawl and ensures that safety system pressure can be restored before isolation of the tested system is negated.

The emergency oil system can also be tripped by either of two vacuum trip devices (items 2205 and 2206, Figure 10.2-1). They are designed to trip the turbine on a rise in backpressure of either condenser; the value at which the turbine is tripped is approximately 9 in. Hg abs.

The emergency oil system can also be tripped by energizing either of two solenoid valves (items 2501 and 2503, Figure 10.2-1).

The MHC system and the emergency trip system are interconnected by the safety oil system (item 2107, Figure 10.2-1). Loss of emergency oil releases the holding force on the emergency trip device (item 2213, Figure 10.2-1); the trip device then moves to discharge safety oil pressure to drain (and therefore to trip) the MHC operating device (item 1205) and, sequentially, to trip the control, overload, and intercept valves. Loss of emergency oil to relay 2211 releases emergency fluid 2108 to drain, thereby tripping closed the overload stop, the main stop, and the reheat stop valves. In addition, through relay 2225, loss of safety oil also dumps the emergency fluid to drain, serving as an additional means of tripping the stop valves.

The turbine can only be reset by running the operating device of the MHC system to its zero or "closed overtravel" position, thereby ensuring that all control and intercept valves are closed before any of the turbine stop valves can be opened.

Automatic starting, synchronizing, loading, and shutdown of the turbogenerator and its condensate and feedwater systems is accomplished by an on-line, closed-loop automatic control system, called Turbomatic (T-M) by its manufacturer, the Brown Boveri Corporation. T-M consists mainly of two principal systems: a silicon solid-state system of wired logic sequential control, and a starting and loading controller which controls the runup and loading of the set by means of a small digital computer. These two systems are so integrated as to permit bringing the turbogenerator, its auxiliaries, and the condensate and feedwater systems from a zero or cold shutdown condition up to a full-load condition automatically but excluding control of the feedwater pump and its drive turbine.

The sequential control issues control commands in program steps when definite conditions are fulfilled; these conditions vary from one program step to another. This sequential control, when switched on, determines whether the primary criteria of the first program step are fulfilled. If the primary criteria are fulfilled, the program step orders are automatically executed within a specified tolerance time. The execution of step orders is monitored by so-called secondary criteria. If all primary criteria are not fulfilled, the program stops automatically, and appropriate signals indicate to the operator the nature of the program interruption. If all the secondary criteria are fulfilled, the system automatically proceeds to the next program step, in which its primary criteria are checked, etc., and so on from step to step.

A number of program steps that logically belong together are combined into a program section. The program sections correspond to logical operating conditions of the plant that can be maintained for long periods of time; for example, 7

Startup of lube oil system and shaft turning gear.

Startup of condensing and feedheating plant.

Warming up and starting of turbine.

Excitation of alternator.

Synchronizing and loading of turbogenerator.

The wired logic of the T-M system is set up in accordance with the contents of the total program, which contains the number of program sections and steps that are needed, what primary and secondary criteria must be checked in the individual steps, what orders must be issued, and what tolerance periods apply. The wiring in the T-M system is set up in accordance with the contents of the program, and the contacts of the output relays of each step are connected to their respective switchgear controls.

When the turbine has been suitably prepared for startup and all other conditions set forth in the program are met (such as an adequate steam supply being available), the unit is then run up from turning gear speed to rated speed, automatically synchronized, and loaded. The turbine runup is effected by means of the operating device (item 1205, Figure 10.2-1), which controls the steam admission valves. The motor driving the operating device is controlled by pulses from the computer equipment furnished as part of the T-M. Analog inputs to the computer are the steam temperature to the turbine and significant representative turbine metal temperatures. With this temperature information and with specific design knowledge of the thermal stress characteristics of those metal parts of the turbine that comprise the steam path, the computer program has been written to permit a maximum rate of change of steam flow through the turbine without exceeding predetermined design margins. The computer can then determine the maximum permissible rate of change of speed during runup, or rate of change of load during a demand change, and will emit corresponding control pulses — positive or negative — to the operating device of the turbine.

The T-M has interface connection provisions for the control system of the NSS; a signal will be sent to the NSS in the event that the T-M is limiting load, and the T-M also can accept a signal from the NSS to the effect that the NSS is limiting load.

The T-M system is operated from a control panel in the unit control room. The
panel contains all the buttons required for starting, stopping, or reversing
the program, as well as indicating lights for the primary and secondary cri- 7
teria and for program steps and sections and load set equipment to set the
desired value of the load the T-M is to impose on the turbogenerator.

The turbine can be operated automatically through the T-M, or it can be operated manually under direct control of the operator. The T-M can also be used in the "operator guide" configuration, wherein all turbogenerator control is handled manually by the operator and the T-M follows the operation, indicating the sequence of orders that are followed by having no actual control of the turbine. In the event of a power supply failure to the T-M, it immediately releases control of the turbine to operator manual and also switches the T-M to "operator guide."

The safety devices of the turbogenerator (i.e., lube oil supervision, high vibration, generator trips, etc.) act independently of the T-M. These systems are connected directly to the tripping systems of the turbogenerator. If they cause a trip, the T-M is informed by certain reversal criteria (built into the program) and thus, the T-M activates the corresponding reversal of the program to an appropriate program step.

The T-M power supply is independent of the plant auxiliary systems. The power supply is from a separate battery system, which is equipped with its own charger. In normal service the system is supplied from the plant system through the battery charger, the battery being float charged. Should the a-c supply fail, the capacity of the battery is sufficient to keep the T-M system supplied for several hours. Should this supply fail, an alarm is sounded coincident with transfer of the T-M from "automatic" back to "operator guide." One spare charger will be installed with a changeover switch to be used for both turbogenerators.

Any influence of the turbine controls on the reactor coolant system is controlled by the integrated control system (ICS) which is described in section 7.7.1.2. Analyses of the most severe of these influences are given in Chapter 15. 7

10.2.3. Turbine Missiles

A plant arrangement having the turbines and safety-related components and structures positioned as shown in Figure 10.2-8 was selected to reduce the probability of unacceptable damage from a turbine disintegration to an acceptably small value. In such an arrangement, the essential safety-related components and structures are placed in locations having a very low probability of being struck by a turbine missile. The following components and structures were considered essential for safety:

1. Both reactor secondary containment buildings. 7

2. Both spent fuel pit rooms in auxiliary building.

3. Other parts of auxiliary building housing essential ESF equipment on upper floor.

4. The entire control building.

5. Both diesel buildings.

The only essential ESF installation not considered was the emergency raw cooling water system intake structure. This was omitted because the high degree of redundancy provided at this installation (just three of the eight pumps are needed to preserve safety) gives a capability for supplying sufficient cooling water after a turbine missile has struck any part of this installation.

10.2.4. Evaluation

The following operational transients can occur, caused by operation of turbine, generator, or distribution system protection equipment.

1. Turbine trip.

2. Generator trip.

3. Turbine fast momentary valving.

4. Turbine fast sustained valving.

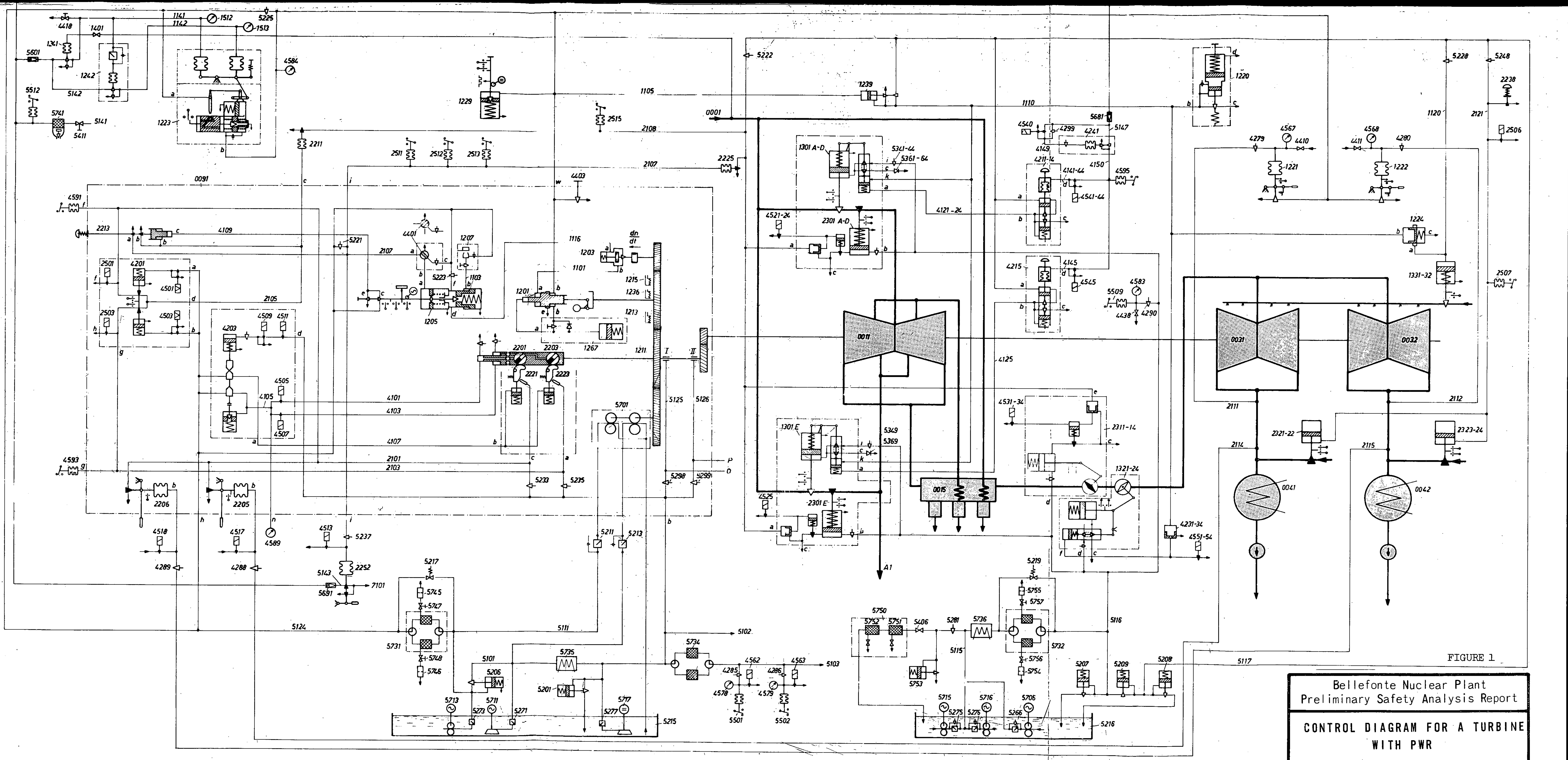
The analysis of the consequences of the severest of these events with respect to reactor safety are discussed in Chapter 15, Accident Analysis.

There can be any number of component or system operational abnormalities that can be postulated to produce a turbogenerator load transient. However, since the effects of such abnormalities can be no worse than a turbine or generator trip, these occurrences are not formally listed.

Any noble gas activity in the secondary system as well as the particulate activity present due to moisture carryover from the steam generators enters the high-pressure turbine.

The subsequent activity entering the low-pressure turbine is reduced due to the moisture separation that occurs between the exit of the high-pressure turbine and the entrance to the two low-pressure turbines.

Activity levels in the turbine are expected to be low and the shielding is provided by the piping, turbine casing, and other components. If any additional shielding is required in local areas, it will be provided so that unlimited access to the turbine area is possible. Details of the shielding design are discussed in section 12.1.



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CONTROL DIAGRAM FOR A TURBINE WITH PWR

FIGURE 10.2-1
TVA DWG. NO. TGR 203377
ADDED BY AMEND. 7, NOV. 30, 1973

Nr.	Plan	Object	Department
		The letters in the last column have the following meaning :	
		TGA,TGD,TGR,TGW = The devices are to be delivered from the corresponding departments	
		TGR-Lb = Device to be delivered from dept.TGR and is build-in to the front bearing pedestal	
		TGR-Rb = Device to be delivered from dept.TGR and is build-in to the control group	
		KEC5(TGR) = Device to be purchased by KEC5 upon instruction from dept. TGR	
		FL = Device to be delivered by customer or sub-contractor	
		Main elements of set	
0001	A6	Live steam supply	
0011	C8	HP-cylinder	
0015	D8	Reheater/waterseparator	
0031	C10	LP-cylinder 1	
0032	C12	LP-cylinder 2	
0041	D11	Condenser 1	
0042	D12	Condenser 2	
0091	B3	Front bearing pedestal	
		Control	
1101	C6	Speed set point oil circuit	
1103	C5	Pulsator circuit	
1105	A6	Speed governor oil circuit	
1110	A9	Control circuit for the control valves and intercept flaps, control fluid	
1116	B6	Turbine speed set point oil circuit,input to acceleration limiter	
1120	A12	Control circuit for the water injection valves for the LP-cylinder	
1141	A3	Output signal of live steam pressure transmitter 1241	
1142	A3	Output signal of delay relay 1242	
1201	C5	Speed governor	TGR-Lb
1203	B6	Acceleration limiter	TGR-Lb
1205	C4	Operating device	TGR-Lb
1207	B5	Pulsator	TGR-Lb
1211	C6	Gearing	TGR-Lb
1213	C6	Speed transmitter	TGR-Lb
1215	C6	Speed transmitter	TGR-Lb
1220	A10	Pressure gradient relay	TGR-Rb
1221	B11	Vacuum limiter to condenser 1	TGR-Rb
1222	B11	Vacuum limiter to condenser 2	TGR-Rb
1223	A3	Live steam pressure limiter	TGR-Rb
1224	B12	Control relay to water injection valves 1331-32	TGR-Rb

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LEGEND TO CONTROL DIAGRAM

FIGURE 10.2-2

TVA DWG. NO. TGR 410457

ADDED BY AMEND. 7, NOV. 30, 1973

Nr.	Plan	Object	Department
1229	A5	Load limiter	TGR-Rb
1236	C6	Speed transmitter	TGR-Lb
1239	A8	Circuit separating relay,control oil-control fluid, control circuit for the control valves	TGR-Rb
1241	A2	Live steam pressure transmitter to 1223	TGR-Rb
1242	A2	Delay relay to 1223	TGR-Rb
1267	C6	Accumulator to speed governor	TGR-Lb
1301A-D	B7	Control valves	TGR
1301E	D7	Bypass control valve	TGR
1321-24	D9	Intercept flaps	TGR
1331-32	C12	Water injection valves for LP-cylinders	TGR
1401	A2	Isolating valve to live steam pressure limiter 1223	TGR-Rb
1512	A3	Pressure gauge to 1141	TGR-Rb
1513	A3	Pressure gauge to 1142	TGR-Rb
		Safety devices	
2101	D4	Emergency circuit left	
2103	D4	Emergency circuit right	
2105	C3	Common emergency circuit	
2107	B6	Safety circuit, oil	
2108	A6	Safety circuit, control fluid	
2111-12	D10/C12	Vacuum impulse pipe to the vacuum limiters 1221/22	
2114-15	D10/D12	Vacuum impulse pipe to the vacuum trips 2205/06	
2121	A12	Control circuit to the vacuum breakers	
2201	C5	Overspeed trip, 110 %	TGR-Lb
2203	C5	Overspeed trip, 112 %	TGR-Lb
2205	D3	Vacuum trip to condenser 1	TGR-Lb
2206	D3	Vacuum trip to condenser 2	TGR-Lb
2211	B4	Relay between emergency circuit 2105 and safety circ.2108	TGR-Rb
2213	B2	Emergency trip	TGR-Lb
2221	C5	Reset device to 2201	TGR-Lb
2223	C5	Reset device to 2203	TGR-Lb
2225	B7	Relay between safety circuit 2107 and safety circuit 2108	TGR-Rb
2238	A12	Trip valve for the vacuum breakers	TGR-Rb
2252	E4	Safety relay for power assisted check valves in the extraction lines	TGR-Rb

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LEGENO TO CONTROL OIAGRAM

FIGURE 10.2-3

TVA OWG. NO. TGR 41045B
AODED BY AMEND. 7, NOV 30, 1973

Nr.	Plan	Object	Department
2301 A-D	B7	Main stop valves	TGR
2301 E	D7	Bypass stop valve	TGR
2311-14	D9	Reheat stop flaps	TGR
2321-24	D11/12	Vacuum breakers	TGR
2501	C2	Solenoid trip, left	TGR-Lb
2503	C2	Solenoid trip, right	TGR-Lb
2506	A12	Solenoid trip to the vacuum breakers	TGR-Rb
2507	C12	Pressure switch for indicating "vacuum broken"	TGR-Rb
2511	B4	Pressure switch in safety circuit 2107 for reactor signal	TGR-Rb
2512	B5	Pressure switch in safety circuit 2107 for reactor signal	TGR-Rb
2513	B5	Pressure switch in safety circuit 2107 for reactor signal	TGR-Rb
2515	A6	Pressure switch in the safety circuit 2108	TGR-Rb
		Test	
4101	D4	Test circuit left	
4103	D4	Test circuit right	
4105	D3	Common test circuit	
4107	D4	Reset circuit for the overspeed trips	
4109	B3	Reset circuit for the emergency trip	
4121-24	B8	Test circuit to control valves 1301 A-D	
4125	C9	Test circuit to bypass valve 1301 E	
4141-44	B9	Air circuit for testing the control valves 1301 A-D	
4145	B9	Air circuit for testing the bypass valve 1301 E	
4149	B9	Common air circuit	
4150	B9	Output from delay relay 4241	
4201	C2	Emergency circuit separating relay	TGR-Lb
4203	C3	Test device	TGR-Lb
4211-14	B9	Test relay for the control valves 1301 A-D	TGR
4215	B9	Test relay for the bypass valve 1301 E	TGR
4231-34	D10	Test relay for the intercept flaps 1321-24	TGR
4241	B9	Delay relay	TGR-Rb
4279	B10	Test orifice for the vacuum limiter 1221	TGR-Rb
4280	B11	Test orifice for the vacuum limiter 1222	TGR-Rb
4285	E7	Test orifice for the lube oil pressure switch 5501	TGR-Rb
4286	E7	Test orifice for the lube oil pressure switch 5502	TGR-Rb
4288	E3	Test orifice for the vacuum trip 2205	TGD
4289	E3	Test orifice for the vacuum trip 2206	TGD
4290	C10	Test orifice for the control fluid pressure switch 5509	TGR-Rb
4299	B9	Bypass orifice for the solenoid valve 4540	TGR-Rb
4401	B4	Test valve for adjusting the control valves without safety oil pressure	TGR-Lb
4403	B6	Test valve for adjusting the control valves	TGR-Lb
4410	B11	Test valve for the vacuum limiter 1221	TGR-Rb
4411	B11	Test valve for the vacuum limiter 1222	TGR-Rb
4418	A2	Test valve for the live steam pressure limiter 1223	TGR-Rb

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LEGEND TO CONTROL DIAGRAM

FOGIRE 10.2-4
TVA DWG. NO. TGR 410459
AOOEO BY AMENO.7, NOV 30,1973

Nr.	Plan	Object	Department
4438	C10	Test valve for the control fluid pressure switch 5509	TGR-Rb
4501	C3	Solenoid valve to emergency circuit separating device,left	TGR-Lb
4503	C3	Solenoid valve to emergency circuit separating device,right	TGR-Lb
4505	C3	Solenoid valve to select test circuit, left	TGR-Lb
4507	D3	Solenoid valve to select test circuit,right	TGR-Lb
4509	C3	Solenoid valve to test device 4203	TGR-Lb
4511	C3	Solenoid valve to test device 4203	TGR-Lb
4513	E4	Solenoid valve to test power assisted check valves in the extraction lines	TGR-Rb
4517	E3	Solenoid valve to test the vacuum trip 2205	TGR
4518	E2	Solenoid valve to test the vacuum trip 2206	TGR
4521-24	B7	Solenoid valve to test the main stop valves 2301A-D	TGR
4525	D7	Solenoid valve to test the bypass stop valve 2301 E	TGR
4531-34	C9	Solenoid valve to test the reheat stop flaps 2311-14	TGR
4540	B9	Solenoid valve to delay relay 4241	TGR-Rb
4541-44	B9	Solenoid valve to test the control valves 1301 A-D	TGR
4545	C9	Solenoid valve to test the bypass control valve 1301E	TGR
4551-54	E10	Solenoid valve to test the intercept flaps 1321-24	TGR
4562	E7	Solenoid valve to test the lube oil pressure switch 5501	TGR-Rb
4563	E7	Solenoid valve to test the lube oil pressure switch 5502	TGR-Rb
4567	A11	Pressure gauge to vacuum limiter 1221	TGR-Rb
4568	A11	Pressure gauge to vacuum limiter 1222	TGR-Rb
4578	F7	Pressure gauge to lube oil pressure switch 5501	TGR-Rb
4579	F7	Pressure gauge to lube oil pressure switch 5502	TGR-Rb
4583	C10	Pressure gauge to control fluid pressure switch 5509	TGR-Rb
4584	A3	Pressure gauge in the control circuit 1105	TGR-Rb
4589	E3	Pressure transmitter to circuit 4105	TGA-Rb
4591	B2	Pressure switch to emergency circuit, left	TGR-Rb
4593	D2	Pressure switch to emergency,right	TGR-Rb
4595	B10	Pressure switch to circuit 4150	TGR-Rb
		Air and oil supply	
5101	E5	Lube oil after pump	
5102	E7	Lube oil to journal bearings	
5103	E7	Lube oil to thrust bearing	
5111	E5	Control oil after pump	
5115	E8	Control fluid after pump	
5116	E9	Control fluid supply circuit, HP	
5117	E10	Control fluid supply circuit, IP	
5124	E4	Control oil supply circuit, IP	
5125	D6	Lube oil for gear bearing 1	
5126	D6	Lube oil for gear bearing 2	
5141	A2	Air supply circuit (min.40 psig - max.240 psig)	

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LEGENO TO CONTROL OIAGRAM

FIGURE 10.2-5

TVA DWG. NO. TGR 410460

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Nr.	Plan	Object	Department
5142	A2	Air circuit, 20 psig	
5143	E4	Air circuit, 15 psig	
5147	A10	Air circuit, 35 psig	
5201	F5	Constant pressure valve, lube oil	TGR
5206	F5	Constant pressure valve, IP-control oil	TGR
5207	E9	Constant pressure valve,HP-control fluid	TGR
5208	E10	Constant pressure valve,IP-control fluid	TGR-Rb
5209	E10	Reducing valve,HP-IP control fluid	TGR-Rb
5211	E6	Check valve with bypass,IP-circuit	TGR
5213	E6	Check valve with bypass, lube oil circuit	TGR
5215	F6	Lube oil tank	TGD
5216	F9	Control fluid tank	TGD
5217	E4	Spring loaded check valve,control oil	TGR-Rb
5219	E9	Spring loaded check valve,control fluid	TGR
5221	B4	Feed orifice,safety circuit, oil	TGR-Lb
5222	A7	Feed orifice,safety circuit, control fluid	TGR-Rb
5223	C5	Feed orifice,set point oil circuit 1116	TGR-Lb
5225	A3	Feed orifice,control circuit 1105	TGR-Rb
5228	A12	Feed orifice, control circuit 1120	TGR-Rb
5233	D5	Feed orifice, emergency circuit left	TGR-Lb
5235	D5	Feed orifice, emergency circuit right	TGR-Lb
5237	E4	Feed orifice, safety relay 2251	TGR-Rb
5248	A12	Feed orifice,control circuit to the vacuum breaker	TGR-Rb
5266	F9	Check valve to pump 5706	TGR
5271	F5	Check valve to pump 5711	TGR
5273	F5	Check valve to pump 5713	TGR
5275	F8	Check valve to pump 5715	TGR
5276	F8	Check valve to pump 5716	TGR
5277	F6	Check valve to pump 5717	TGR
5281	E8	Feed orifice,control fluid regeneration	TGR
5298/99	D6	Feed orifice to circuit 5125-26	TGR
5341-44	B8	Feed orifice to 1301 A-D	TGD
5349	D8	Feed orifice to 1301E	TGD
5361-64	B8	Check valve to 1301 A-D	TGR
5369	D8	Check valve to 1301 E	TGR
5406	E6	Isolating valve to control fluid regeneration	TGR
5411	B2	Isolating valve in air supply circuit	TGR-Rb

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LEGEND TO CONTROL DIAGRAM

FIGURE 10.2-6

TVA DWG. NO. TGR 410461

ADDED BY AMEND. 7, NOV 30,1973

Nr.	Plan	Object	Department
5501	F7	Pressure switch, lube oil circuit, 0,4 x Pn	TGR-Rb
5502	F7	Pressure switch, lube oil circuit, 0,6 x Pn	TGR-Rb
5509	C9	Pressure switch, control fluid	TGR-Rb
5512	A2	Pressure switch for air pressure supervision	TGR-Rb
5601	A2	Air pressure filter reducing valve,20 psig	TGR-Rb
5681	A9	Air pressure filter reducing valve, 35 psig	TGR-Rb
5691	E4	Air pressure filter reducing valve, 15 psig	TGR-Rb
5701	D6	Main pump for lube and control oil	TGR-Lb
5706	F9	Control fluid pump 1	TGR
5711	F5	Auxiliary pump for lube oil	TGR
5713	F5	Auxiliary pump for control oil	TGR
5715	F8	Control fluid pump 2	TGR
5716	F9	Control fluid pump 3	TGR
5717	F6	Emergency pump for lube oil	TGR
5731	E4	Control oil filter	TGR-Rb
5732	E9	Control fluid filter	TGR
5734	E6	Lube oil filter	TGD
5735	E5	Lube oil cooler	TGW
5736	E8	Control fluid cooler	TGW
5741	A2	Air filter	TGR-Rb
5745/46	E4	Flow control device to control oil filter 5731	TGR-Rb
5747/48	E4	Vent cock to control oil filter 5731	TGR-Rb
5750	E8	Regeneration unit for control fluid	KEC5(TGR)
5751	E8	Activ earth filter	KEC5(TGR)
5752	E8	Fine filter	KEC5(TGR)
5753	F8	Constant pressure valve,control fluid regeneration	TGR
5754/55	E9	Flow control device to control fluid filter 5732	TGR
5756/57	E9	Vent cock to control fluid filter 5732	TGR
7101	E4	Impulse to the check valves in the extraction lines	

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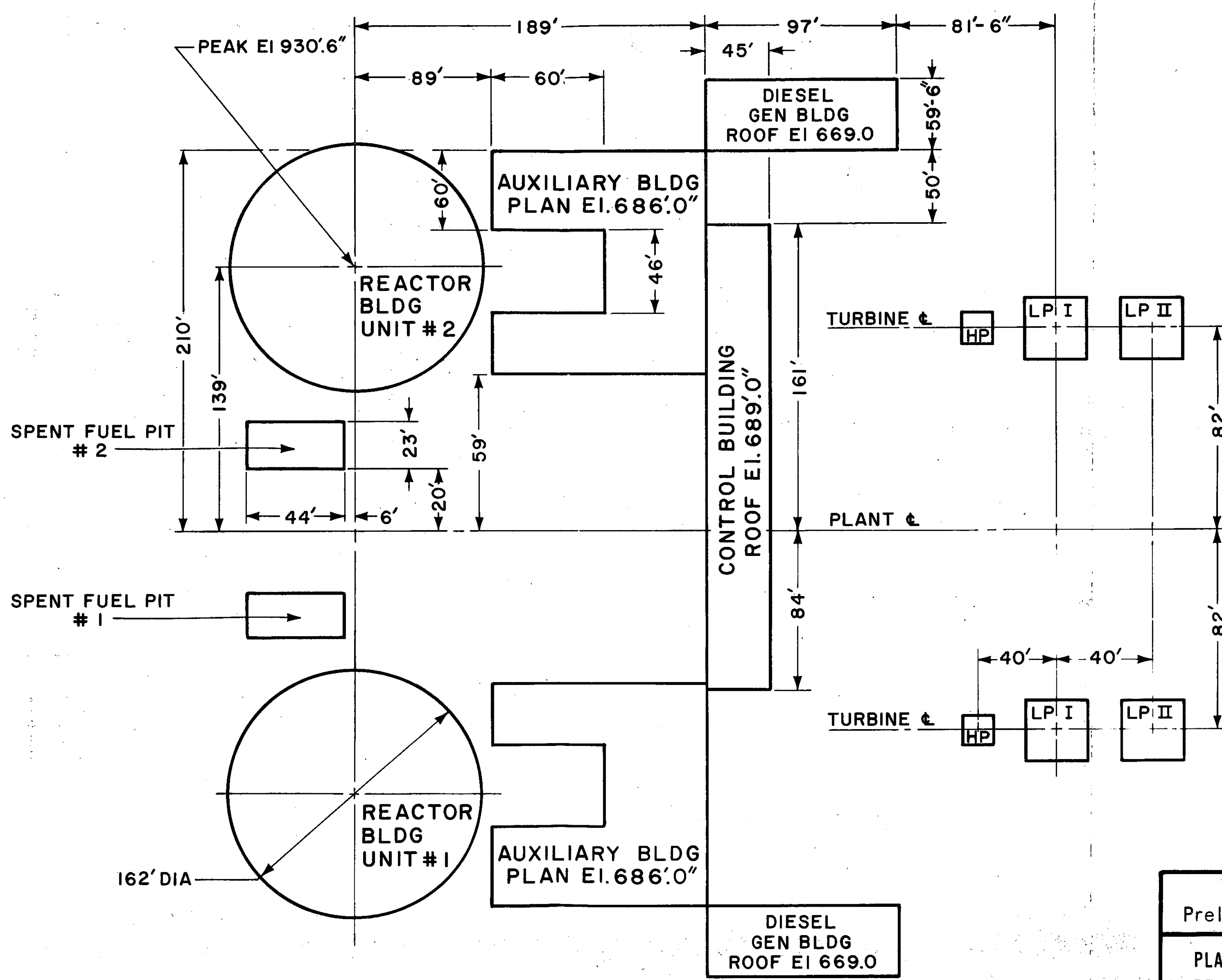
LEGENO TO CONTROL DIAGRAM

FIGURE 10.2-7

TVA DWG. NO. TGR 410462

AODEO BY AMEND. 7, NOV. 30,1973



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PLAN VIEW OF ESSENTIAL SAFETY RELATED ITEMS FOR TURBINE MISSILE ANALYSIS

FIGURE 1D.2-8

ADDED BY AMEND. 7, NOV. 30, 1973

10.3. Main Steam Supply System

10.3.1. Design Bases

The main steam supply system is designed to conduct steam from the steam generator outlets to the turbine-generator and to the turbine bypass system. It further supplies steam to the turbine seals, to the main feedwater pump turbines, and to the auxiliary feedwater pump turbine.

The main steam supply system from the steam generators to and including the main steam isolation and check valves is designed according to requirements of ANSI Safety Class 2 and the ASME Boiler and Pressure Vessel Code, Section III.

A failure or malfunction of the Class 2 or 3 portion of the main steam system must not:

1. Reduce flow capabilities of the auxiliary feedwater system.

2. Prevent safe shutdown and cooldown of the reactor.

3. Initiate a loss-of-coolant accident.

4. Cause failure of any other steam (or feedwater) line.

5. Result in the containment pressure exceeding design value.

6. Impair the containment integrity.

7. Allow uncontrolled blowdown of more than one steam generator.

The steam supply to the auxiliary feedwater pump turbine downstream of the check valve is designed according to requirements of ANSI Safety Class 3 and the ASME Boiler and Pressure Vessel Code, Section III. The remainder of the main steam supply system, all piping downstream of the main steam isolation and check valves, is designed to the requirements of the ANSI Power Piping Code B31.1.

The main steam system is designed for the following conditions:

Pressure	1235 psig
Temperature	630F
Flow, lb/h	14,800,000
Pressure drop from steam generator to turbine	25 psi

10.3.2. System Design Description

10.3.2.1. System Description

The main steam supply system for Unit 1 (identical for Unit 2) is shown on Figure 10.3-1.

The Safety Class 2 and 3 portion of the main steam supply system and all its parts downstream of the main steam isolation and check valves inside the main steam safety valve rooms, including pipe hangers, restraints, and anchors, are Category I. This piping is within a Category I building. The Safety Class 3 portion of the steam supply line for the auxiliary feedwater turbine is also Category I.

The main steam supply system includes self-actuating safety relief valves to provide emergency pressure relief for the steam generators and includes diaphragm-operated atmospheric dump valves for automatically or manually controlled steam discharge to the atmosphere, see section 10.4.4.2, Power Operated Relief Valve (Atmospheric Dump) System.

There are eleven self-actuating safety relief valves per steam generator upstream of the main steam isolation valves with a combined capacity of 9.9×10^6 lb/h or 117% of the maximum capacity of each steam generator. The steam generator safety valves provide emergency pressure relief for the steam generator and the settings are as follows:

Valve, quantity	Set pressure, psig	Accumulation		Blowdown	
		Percent	Pressure, psig	Percent	Pressure, psig
5	1225	3	1262	4	1176
6	1245	3	1282	4	1195

Steam is conducted from each steam generator through the primary and secondary containment and out through the main steam isolation and check valves by two 32-inch OD pipes. The main steam isolation and check valves are located outside and as close as possible to the secondary containment building inside the main steam safety valve rooms.

In the event of a main steam pipe rupture the air-operated main steam isolation valves are capable of closing and isolating a steam generator within 7.5 seconds of the detection of high containment pressure or low steam generator pressure. The main steam isolation valves are 32-inch globe valves of the Y-type with straight through flow utilizing pressurized air for opening and spring power for closing. Upon loss of control air the isolation valves return to the closed position.

The check valves prevent reverse flow of steam in the event of accidental pressure reduction in any steam generator or its associated piping. The check valves are counterweighted to reduce pressure loss in the valve and are supplied with air-operated side mounted closing cylinders.

From each of the two main steam safety valve rooms two 32-inch OD main steam lines are routed across the roof of the auxiliary building and the control building penetrating finally the turbine building wall. Inside the turbine building two main steam lines originating from the same steam generator combine to one 42-inch OD line before entering the main steam header. This header serves as cross connection of the two steam generators for the purpose of pressure equalization before the steam enters the main steam stop and control valves directly ahead of the high pressure turbine via four 30-inch OD lines. The header further distributes steam to the condenser steam dump valves and to the moisture separators and reheaters. For description of condenser dump valves, see section 10.4.4.1, Turbine Bypass System.

Dependability of steam supply for the auxiliary feedwater turbine is provided by two separate 8-inch steam lines each tying into the main steam line system of a different steam generator upstream of the main steam isolation valves.

The main steam supply system is not normally radioactive, but in the event of a primary-to-secondary leak in the steam generators radioactive material could be released to the secondary system. The condenser vacuum pump radiation monitoring system and the turbine building exhaust radiation monitoring system will detect radiation that results from a primary-to-secondary leakage.

10.3.2.2. Materials and Standards

The steam supply system is manufactured from carbon steel.

All ANSI Safety Class 2 and Class 3 portions of the system conform to the requirements of the ASME Boiler and Pressure Vessel Code, Section III.

For the ANSI Safety Class 2 portion of the Main Steam Piping System, the following material specifications and standards apply:

1. Main steam line straight pipe sections including the safety valve headers are manufactured from rolled and welded plate conforming to the ASTM Electric-Fusion-Welded Steel Pipe Standard A 155, Grade KCF 70, Class 1 and the ASTM Carbon Steel Plate Standard A 516, Grade 70.

2. Pipe fittings are manufactured from welded plate conforming to the ASTM Pipe Fitting Standard A 420, Grade WPL1, and the ASTM Carbon Steel Plate Standard A 516, Grade 70.

3. Valves in this portion of the main steam line system conform to NC-3500 of the ASME Boiler and Pressure Vessel Code, Section III.

The following material specifications and standards apply for the ANSI Safety Class 2 and Class 3 portions of the steam supply lines for the auxiliary feedwater turbine:

1. Straight pipe sections comply with the ASTM Seamless and Welded Steel Pipe Standard A 333, Grade 6.

2. Pipe fittings comply with the ASTM Pipe Fitting Standard A 420, Grade WPL-6.

3. Valves in the Safety Class 2 and Class 3 portions conform to Articles NC-3500 and ND-3500 of the ASME Boiler and Pressure Vessel Code, Section III respectively.

For the remainder of the main steam piping system downstream of the main steam isolation and check valves, the following material specifications and standards apply:

1. Main steam line straight pipe sections including the main steam header are manufactured from rolled and welded plate conforming to the ASTM Electric-Fusion-Welded Steel Pipe Standard A 155, Grade KC 70, Class 1 and the ASTM Carbon Steel Plate Standard A 515, Grade 70.

2. Pipe fittings are manufactured from welded plate conforming to the ASTM Pipe Fitting Standard A 234, Grade WPB and the ASTM Carbon Steel Plate Standard A 515, Grade 70.

3. Valves in this portion of the main steam piping system conform to the requirements of the ANSI Power Piping Code B31.1.

10.3.3. Safety Evaluation

Safety considerations with respect to the main steam supply system are directed toward maintaining the capability of controlled reactor cooldown following any kind of accident and minimizing release of radioactive material to the environment.

Although individual pieces of equipment may be damaged, the capability of carrying out the required function through operation of redundant or diverse equipment will be maintained. This capability is shown to be adequate by the analyses in Chapter 15.

The portion of the main steam supply system designed to ANS Safety Class 2 and Safety Class 3 is Category I seismically qualified. This part of the system is protected from missiles and pipe whip by restraints, physical separation, or barriers.

Redundant electrical power and air supplies assure reliable system operation and safe shutdown procedures.

The safety relief valves provide 117% relieving capacity to protect the system from overpressure. The capacity provided by the turbine bypass and power operated relief valves is not included in the safety relief valve capacity of 117%. The dump valves, since they have a set pressure slightly lower than the safety relief valves, prevent lifting of the safety valves.

The maximum capacity of any single relief or dump valve does not exceed a flow of 900,000 lb/h or 5.4% of the maximum capacity of both steam generators to limit steam release if any one valve is inadvertently stuck open.

Only one atmospheric dump valve is required for plant cooldown following any credible accident.

The safety relief valves and modulated power operated (atmospheric dump) valves are located inside the safety valve rooms and are vented to the outside by vent pipes. For protection of the auxiliary building where one safety valve room is located, the safety valve room has 4-foot-thick concrete walls without doors and extends upward to the roof line which is covered by a light roof to allow steam escape to the outside in case of an accident involving steam discharge into the safety valve room.

The auxiliary and control building roofs and walls in the vicinity of the main steam and feedwater lines have been strengthened to protect the equipment inside the buildings from effects of a pipe rupture.

If primary-to-secondary leakage occurs in a steam generator, the condenser vacuum pump and the turbine building exhaust radiation monitoring systems provide redundant detection of radioactivity in the secondary system.

The vacuum pump exhaust is passed through a HEPA filter and charcoal absorber train before being discharged through the turbine building exhaust vent. A bypass around the filter train is used only during filter maintenance and when the vacuum exhaust flow is in excess of 50 cfm as during unit startup.

10.3.4. Tests and Inspection

All Safety Class 2 and 3 pressure-retaining components of the main steam supply system and their supports are designed to comply with the ASME Boiler and Pressure Vessel Code, Section XI, Inservice Inspection of Nuclear Reactor Coolant Systems.

Performance tests of individual components in manufacturer's shops, integrated preoperational tests of the whole system, and periodic performance tests of the actuating circuitry and mechanical components will assure reliable performance.

Test procedures for initial tests and operation are discussed in Chapter 14.

10.3.5. Water Chemistry

In the once-through steam generator, blowdown is not used to control the concentration of contaminants in the secondary system water. Instead, the level of contaminants is controlled by water treatment methods. The treatment maintains the water quality within the limits of the criteria specified in Table 10.3-1. A combination of treatment methods is employed to maintain water quality, namely, demineralization, chemical addition and deaeration. A discussion of these methods and their purpose in the secondary water chemistry program is presented below:

Demineralization

Condensate from the condenser hotwell is treated via mixed bed demineralizers to remove soluble and suspended materials, including contaminants due to condenser inleakage. The demineralizers (shown on flow diagram, Figure 10.4-7) have mixed beds which consist of an ammonia type cation resin and a hydroxyl form anion resin. The condensate system contains five mixed bed demineralizers. Normally, four demineralizers are in service and one is in standby.

Demineralizers are also used to treat makeup water to the secondary system. Makeup water is supplied from the demineralized water system to the condenser hotwell, and meets the secondary chemistry requirements.

Chemical Addition and Deaeration

Chemicals are continuously fed to the condensate system for corrosion control. Corrosion is inhibited by controlling the pH and oxygen content of the secondary system water. A pH range of 9.3 to 9.5 is maintained continuously by regulating the input of ammonia to the condensate system. Oxygen is removed both by deaeration and by addition of hydrazine. Sufficient hydrazine is fed to the condensate system to provide a residual quantity at the inlet to the steam generator.

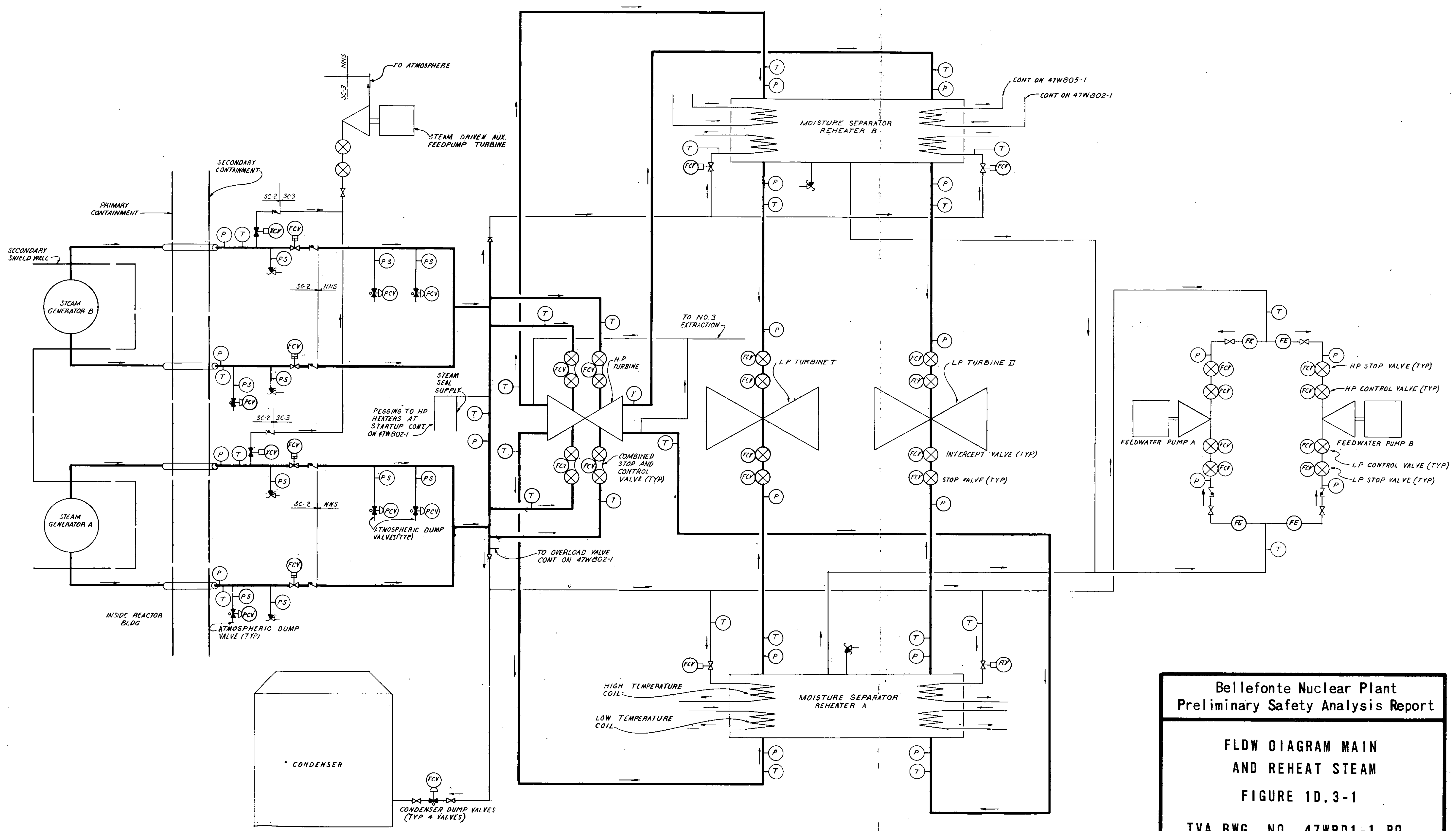
Utility fossil plant experience has demonstrated these water treatment methods to be effective. Plant secondary water quality criteria have been shown to be met when these treatment methods are employed.

Although stringent water quality criteria are in effect for the steam generator system, a significant accumulation of deposits may occur during extended use. In order to minimize degradation of component performance and generator life, it is necessary to clean the steam generators periodically with the aid of chemical cleaning agents.

The chemicals added to the secondary system water have no effect on the radioiodine partition coefficient, which for the once-through steam generator system is assumed to be 1.0. On the other hand, the radioiodine partition coefficient for the air ejector system is increased by the presence of ammonia. As a result of the lack of quantitative data on the distribution of iodine species and their respective concentrations an exact partition coefficient cannot be computed; however, from available data it is apparent that the coefficient will increase as a function of increasing pH for a specific temperature. It should be noted that for site boundary dose calculations involving the air ejector system this increase is not considered, and coefficients for plain water are used.

Table 10.3-1. Total Solids (Dissolved and Suspended) at 50 ppb Maximum

Specific conductivity at 25C, μ mho/cm	Less than 0.1
Dissolved oxygen as O_2 (max), ppb	7
Total silica as SiO_2 (max), ppb	20
Total iron as Fe (max), ppb	10
Total copper as Cu (max), ppb	2
pH at 77F (adjusted with ammonia)	9.3 - 9.5



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FLOW DIAGRAM MAIN AND REHEAT STEAM

FIGURE 1D.3-1

TVA DWG. NO. 47WBD1-1 RO

10.4. Other Features of Steam and Power Conversion System

10.4.1. Main Condenser

10.4.1.1. Design Bases

The design basis for the main condenser is to provide a heat removal rate of at least 7.846 × 10^9 Btu/h per unit for the steam system by condensing the steam from the turbine exhaust at a dual back pressure of 2.0/3.4 inches of mercury, absolute. 5

10.4.1.2. System Description

To provide sufficient capability to meet the functional requirements as stated in section 10.4.1.1, the main condenser has been designed with the following specifications:

Total surface area, ft^2	860,000	5
Circulating water quantity, gpm	410,000	
Number of shells	1	
Number of passes/shell	1	
Cleanliness, %	95	
Duty, 10^9 Btu/h	7.846	5
Hotwell storage/shell (normal), gallon	42,300	
Oxygen content of condensate, cc/liter	0.005	
Bypass system		
Flow, lb/h	8,510,000	5
Enthalpy, Btu/lb	1,252.3	

The condenser is a single shell, dual pressure, dual neck unit. Each pressure zone is connected to a low-pressure turbine by a rubber-belt type expansion joint in the neck. The required impingement baffles are provided to protect the tubes from incoming drains and steam dumps. Provisions have been made for mounting of three one-half capacity low-pressure extraction feedwater heaters in the neck of each pressure zone.

The main condenser system will produce back pressures of 2.0 inches of mercury absolute in the low-pressure zone and 3.4 inches of mercury absolute in the high-pressure zone, when operating at rated turbine output with 74.7F cooling water and 95% clean tubes. A continuous tube cleaning system is provided to keep the condenser operating at peak performance. 5

The condenser is designed to remove dissolved gases from the condensate, limiting oxygen content to 0.005 cc/liter at any load during normal operation.

The condenser can accept a bypass steam flow of approximately 50% of maximum guaranteed steam generator flow, without exceeding the turbine low vacuum 5

trip point, or the high exhaust hood temperature trip point, with yearly average circulating water temperature. This bypass steam dump to the condenser is in addition to the normal duty expected with a throttle flow of 50% of 5
maximum guaranteed steam generator flow.

The correct secondary cycle water inventory is maintained by the automatic bypass-makeup condensate system. The level controller(s), which are sensitive to the hotwell level, position the bypass valve or makeup valve (to or from condensate storage) as required to maintain the hotwell water level within preset limits.

10.4.1.3. Safety Evaluation

The condenser could become ineffective because of the loss of some or all of its cooling water and/or because of excessive air leakage. Either of the above conditions would cause the condenser pressure to increase and upon reaching 8 to 10 inches mercury absolute, the turbine would trip and consequently cause a reactor scram when unit load is above approximately 50%.

The residual heat would then be removed as steam through the turbine bypass valves until they were tripped closed because of high condenser back pressure or the loss of the circulating water pumps. After the turbine bypass valves are tripped closed, the residual heat would be removed as steam through the power operated relief valves to the atmosphere. The ASME Code safety valves 5
would not be utilized.

10.4.1.4. Inspection and Testing

The condenser will be tested for leaks by completely filling the shell with condensate. The waterboxes will be tested by operating the condenser circulating water pump(s) and closing the condenser discharge butterfly valves. Manways provide access to water boxes, tube sheets, lower steam inlet section, shell, and hotwell for purposes of inspection, repair, or tube plugging.

10.4.1.5. Instrumentation

Sufficient level controller, level switches, pressure switches, temperature switches, etc., will be provided to permit personnel to conveniently and safely operate this condenser system.

10.4.1.6. Performance Requirements

For the purpose of this report the performance requirements are the specifications listed in the paragraphs for design bases, 10.4.1.1, and system description, 10.4.1.2.

10.4.1.7. Inventory of Radioactive Contaminants

The inventory of radioactive contaminants in the main condensers is a function of the percentage of defective fuel rods, the escape rate coefficients, the steam generator primary-to-secondary leak rate, the secondary system demineralizer decontamination factors, steam generator, and condenser partition factors. Table 10.4-1 gives the calculated radioactivity inventory for a primary-to-secondary leak rate of 20 gpd and 0.25% failed fuel during power operation and shutdown. The factors and coefficients used are the same as those used in section 11.1. For a PWR, the inventories during power operation and during shutdown differ significantly only as a result of the time allowed for radioactive decay during shutdown.

10.4.1.8. Potential for Hydrogen Buildup

There are several possible mechanisms for hydrogen production in the secondary system. Hydrogen produced by these mechanisms carries over to the condenser. Among the mechanisms are radiolysis of steam generator steam side water, corrosion of piping and tubing, and primary-to-secondary leakage. Radiolysis of water produces negligible amounts of hydrogen (less than 0.001 scfm). The water chemistry of the secondary system is such that hydrogen evolution by corrosion is also negligible. Of the possible mechanisms, primary-to-secondary leakage is the only one capable of supplying measurable quantities of hydrogen. However, for large primary-to-secondary leakage rates (1.0 gpm/unit), the hydrogen volumetric rate would be less than 0.01 scfm. This rate is small when compared to the air ejector system capacity of 30 scfm. Consequently the potential for hydrogen buildup is negligible.

10.4.1.9. Air Inleakage

The condenser has been designed to accommodate an air inleakage rate of 30 scfm. The expected air inleakage rate is approximately 15 scfm.

10.4.1.10. Control Functions

Any influence of the condenser controls on the reactor coolant system is controlled by the integrated control system (ICS) which is described in section 7.7.1.2. Analyses of the most severe of these influences are given in Chapter 15.

10.4.2. Main Condenser Evacuation System

10.4.2.1. Design Basis

The design basis for the main condenser evacuation system is to create and maintain condenser back pressure at 1.0 in. Hg, absolute by removing noncondensable gas and air inleakage. The design air inleakage rate is 30 scfm.

10.4.2.2. System Description (Flow Diagram, Figure 10.4-1)

This system provides sufficient capacity to meet the functional requirements as stated in section 10.4.1.1. The main condenser evacuation system has three electrically driven mechanical vacuum pumps per unit and operates to the following specifications.

Mode of operation		Capacity
Full power:	Three pumps operating at a suction of 1-inch Hg absolute, scfm	30
Startup:	Three pumps initial operation at 15-inch Hg absolute, scfm	2400
Normal:	Two pumps operating at a suction of 1-inch Hg absolute, scfm	20

Two pumps operating in parallel will be adequate to remove the maximum expected air inleakage rate of 15 scfm at the rated back pressure. The third pump is arranged to start automatically as the condenser absolute pressure increases to 5 in. Hg. There are no crossties between unit one and unit two vacuum systems.

The discharge from all three vacuum pumps is normally routed through a HEPA filter charcoal adsorber train. The filtered offgas is monitored with a radiation detector before it is exhausted through one of the turbine building roof ventilators.

Dilution provided by the normal ventilator air flow limits radiation exposure rates on the turbine building roof to acceptable levels.

The offgas piping is arranged to permit bypass of the HEPA filter-charcoal adsorber train during startup and during replacement of filter units. The offgas is still monitored by the radiation detector during this bypass operation. Details of the radiological evaluation are in Chapter 11.

10.4.2.3. Safety Evaluation

One of the three vacuum pumps is considered as spare. This spare unit will automatically start when the condenser absolute pressure increases to 5 in. of mercury. Should the pressure continue to increase (because of inadequate air removal capability), the turbine would trip.

This turbine trip would occur at approximately 8.9 in. Hg absolute.

10.4.2.4. Inspection and Testing

The operating characterisitcs for each vacuum pump will be established throughout the operating range by factory tests.

A flow meter is provided in the discharge of each vacuum pump. Periodic readings of these flow meters will indicate whether or not the air inleakage to the condenser is under 20 scfm. These readings will also indicate the effectiveness of the operating vacuum pumps.

Radioactivity will be constantly monitored in the discharge of the vacuum pumps.

10.4.2.5. Instrumentation

The necessary pressure and temperature switches are provided to automatically start the standby vacuum pump or shutdown a malfunctioning (vacuum) pump.

A radiation monitor is provided on the condenser vent line for added environmental protection.

10.4.3. Turbine Gland Sealing System

10.4.3.1. Design Bases

The system will be designed to provide means of sealing the turbine shafts and valve stems. The purpose of sealing is to prevent inleakage of air to the turbine or outleakage of steam into the turbine room. Since very little radioactive contamination is expected in the main steam, the sealing steam will be taken from the main turbine cycle. During startup, sealing steam will be taken from the auxiliary boiler system.

10.4.3.2. System Description

The seals are labyrinth type. The system will consist of a supply header kept at slightly positive pressure, and an exhaust header kept at slightly negative pressure. Steam to the supply header will come from the steam seal regulator valves and leak-off from high-pressure glands. A dump valve to No. 6 extraction line will release the excess pressure in the supply header. Steam from the exhaust header will be piped to a gland steam condenser where most of the steam component is condensed and returned to the secondary cycle. The noncondensibles are forced by the exhauster through piping to the outside of the building.

10.4.3.3. Safety Evaluation

The outer glands are kept under negative pressure to prevent steam leakage into the turbine room. A 100% spare capacity exhauster will be provided to start automatically if the running exhauster fails, and an alarm will also sound in the control room. The exhaust from the exhauster fan is piped to outside of the building where it will readily disperse. If the second exhauster also fails, steam will leak into the turbine room.

10.4.3.4. Tests and Inspections

The regulating valves will be tested by the vendor before shipment. After installation, the system will be tested by the construction forces for proper operation. Pressure gauges installed on the supply and exhaust headers will help the operator monitor the system's proper operability.

10.4.3.5. Instrumentation

Sufficient pressure controllers, pressure transmitter, pressure indicating switches, etc., will be provided to permit personnel to conveniently and safely operate this steam seal system.

10.4.4. Steam Generator Dump System

10.4.4.1. Turbine Bypass System

10.4.4.1.1. Design Basis

The turbine bypass system function is to (1) maintain secondary system pressure within allowable code limits and (2) reduce the magnitude of nuclear system transients following large turbine load reduction by dumping throttle steam directly to the main condenser, thereby creating an artificial load on the reactor.

The turbine bypass system has the following functional requirements:

1. Permit a direct bypass flow to the main condenser of 50% of rated turbine flow.

2. Provide plant flexibility during operation by allowing turbine load changes in excess of the base NSS design without reactor trip.

3. Provide controlled cooldown of the NSS.

4. Assist in achieving stable startup of the plant by acting as an artificial load when turbine generator demand is less than 15% of rated load.

10.4.4.1.2. System Description (See Flow Diagram, Figure 10.3-1)

To provide the capability for meeting the functional requirements of section 10.4.4.1.1, the turbine bypass system for each unit has been designed with the following specifications:

Number of condenser dump valves	3	6	
Design pressure, psig/in. Hg vac	1235/29	1235/29	11
Design temperature, F	630 and 79	630 and 79	
Flow per valve, lb/h	855,000	992,000	
Valve actuation	Compr air	Compr air	
Main steam pressure at valve inlet (for above flow), psig	1035	1035	
Minimum flow per valve at 50 psia inlet pressure, lb/h	22,500	--	
Time to open (full stroke), s	3	0.5	11
Failure position	Closed	Closed	

The steam exits the steam generator via two steam leads through the containment and into the turbine building. All four steam leads from the two steam generators are cross-connected immediately upstream of the turbine stop valves. 11
Piping is run from this header to the nine turbine bypass valves and then to the condenser.

Normal control is provided by comparing main steam pressure to a programmed setpoint. Steam pressure of 50 psi greater than the setpoint initiates opening of the valves. Manual control (direct use of valve loading signal) is also provided for use during unit startup and shutdown.

During a large load rejection, the turbine governor valves and intercept
valves close which results in an increasing steam pressure. The steam pres-
sure continues to increase to the turbine bypass valve setpoint and may in-
crease to the power operated relief (atmospheric dump) valve setpoint. Dur-
ing abnormal operation, the pressure may increase to the safety valve setpoint 5
(see Chapter 15). The turbine bypass valves allow for 50% of the rated full
steam flow to go directly to the condensers; the power operated relief (at-
mospheric dump) valves permit an additional 27% of full steam flow to vent 11
directly to the atmosphere. The safety valves have a capability of allowing
117% steam relief to the atmosphere. Steam venting to the atmosphere occurs
following a large loss of load until the steam pressure decreases following
reactor runback to a point where the turbine bypass can handle all the excess
steam generated.

The turbine bypass valves are designed to meet the requirements of ASME Code for Nuclear Power Plant Components, Section III (1971), Class 2 and MC.

10.4.4.1.3. Safety Evaluation

Loss of the control air supply to the diaphragms of the bypass valves will prevent the valves from opening; or, if the valves are open, will trip them closed. In the event of loss of all control air, the steam generators will still be protected during all transients by the ASME code safety valves.

Inadvertent or accidental opening of any one valve (bypass, power operated relief, or code safety) during power operation will not subject the reactor coolant system to an uncontrolled depressurization and cooldown since the capacity of each valve is relatively small.

Failure of the turbine bypass system can result in discharge of steam to the atmosphere through the power operated relief or code safety valves. If tube leaks were present prior to the incident, some radioactivity accumulated in the steam generator shell-side water would be discharged through these valves. Total plant releases during any blowdown through these valves will be well within criteria established by 10 CFR 100. Total plant releases during a load rejection transient are discussed in Chapter 15.

10.4.4.1.4. Inspection and Testing

This equipment will be tested in accordance with the ASME Section III Code requirements. Periodic tests will be performed to assure that the system remains capable of its functional requirements. Since this system was designed in accordance with ANS B31.1 and does not require Class 2 components (even through the bypass valves are manufactured as Class 2 components), inservice inspection is not required.

10.4.4.1.5. Instrumentation

Sufficient instrumentation has been provided to permit this system to:

1. Satisfy all its functional requirements.

2. Protect the turbine (from high condenser pressure).

3. Be safe and convenient for plant personnel to operate.

10.4.4.2. Power Operated Relief Valve (Atmospheric Dump) System

10.4.4.2.1. Design Basis

The power operated relief (atmospheric dump) valve function is to (1) maintain secondary system pressure within allowable code limits without lifting the ASME Code safety valves, (2) reduce the magnitude of nuclear system transients following large turbine load reduction by dumping throttle steam directly to the atmosphere to supplement the turbine bypass system, thereby creating an additional artificial load on the reactor, and (3) allow load changes and pressure transients without activation of the safety valves. 5 5

The power operated relief valves have the following functional requirements:

1. Permit a direct flow to the atmosphere of 27% of rated turbine flow when main steam containment isolation valves are open. 11

2. Provide plant flexibility during operation by allowing turbine load changes in excess of the turbine bypass system capability without reactor trip.

3. Provide controlled cooldown of the NSS when the condenser is unavailable.

4. Provide relief capability to avoid lifting code safety valves.

5. Permit direct flow to the atmosphere of 591,500 pph per steam generator when main steam containment isolation valves are closed.

10.4.4.2.2. System Description (See Flow Diagram, Figure 10.3-1)

To provide the capability for meeting the functional requirements of section 10.4.4.2.1, the power operated relief valve system for each unit has been designed with the following specifications:

Number of power operated relief valves	
On/off type	4
Modulating type	2
Design pressure	1235 psig and 29 in. Hg vacuum
Design temperature, F	630 and 79

Flow per valve	
On/off type, lb/h	850,000
Modulating type, lb/h	591,500
Valve actuation	Compressed air
Main steam pressure at valve inlet (for above flow), psig	1035
Time to open (full stroke)	
On/off type, s	0.5
Modulating type, s	3
Failure position	Closed

One modulating valve is assigned to each steam generator and is located adjacent to the Code safety valves upstream of the main steam containment isolation valve. These valves are required to permit shutdown when the condenser is not available and when the steam generator is isolated. Each valve can be opened independently by remote manual control.

The four on/off valves are connected to main steam piping downstream of the main steam containment isolation valves. Normal control is provided by comparing main steam pressure to a programmed setpoint. Steam pressure of 75 psi greater than the setpoint initiates opening of the valves. Manual control (direct use of valve opening signal) is also provided for use during unit shutdown.

During a large load rejection, the turbine governor valves and intercept valves close which results in an increasing steam pressure. The steam pressure continues to increase to the turbine bypass valve setpoint and may increase to the power operated relief valve setpoint. During abnormal operation, the pressure may increase to the safety valve setpoint (see Chapter 15). The turbine bypass valves allow for 50% of the rated full steam flow to go directly to the condensers; the power operated relief valves permit an additional 27% of full |11
steam flow to vent directly to the atmosphere. The safety valves have a capability of allowing 117% steam relief to the atmosphere. Steam venting to the atmosphere occurs following a large loss of load until the steam pressure decreases following reactor runback to a point where the turbine bypass can handle all the excess steam generated.

The modulating valves are designed to meet the requirements of ASME Code for Nuclear Power Plant Components, Section III (1971), Class 2 and MC. On/off valves are Class 3.

10.4.4.2.3. Safety Evaluation

Loss of the control air supply to the diaphragms of the power operated relief valves will prevent the valves from opening; or, if the valves are open, will trip them closed. In the event of loss of all control air, the steam generators will still be protected during all transients by the ASME Code safety valves.

Inadvertent or accidental opening of any one valve (bypass, power operated relief, or code safety) during power operation will not subject the reactor coolant system to an uncontrolled depressurization and cooldown since the capacity of each valve is relatively small.

Failure of the turbine bypass system can result in discharge of steam to the atmosphere through the power operated relief or code safety valves. If tube leaks were present prior to the incident, some radioactivity accumulated in the steam generator shell-side water would be discharged through these valves. Total plant releases during any blowdown through these valves will be well within criteria established by 10 CFR 100. Total plant releases during a load rejection transient are discussed in Chapter 15.

10.4.4.2.4. Inspection and Testing

This equipment will be tested in accordance with the ASME Section III Code requirements. Periodic tests will be performed to assure that the system remains capable of its functional requirements. The on/off valve piping was designed in accordance with ANS B31.1 and does not require Class 3 components (even though the valves are manufactured as Class 3 components); inservice inspection is not required. Since the modulating valves are Class 2 components, they will be subjected to inservice inspection in accordance with ASME Boiler and Pressure Vessel Code, Section XI, Rules for Inservice Inspection of Class 2 Nuclear Power Plant Components, ICS-261(b).

10.4.4.2.5. Instrumentation

Sufficient instrumentation has been provided to permit this system to:

1. Satisfy all its functional requirements.
2. Protect the turbine (from high condenser pressure).
3. Be safe and convenient for plant personnel to operate.

10.4.5. Condenser Circulating Water System (CCW)

10.4.5.1. Design Bases (CCW Flow Diagram, Figure 10.4-2)

The condenser circulating water system is a closed-cycle cooling water system which is separated on a unit basis. Each subsystem uses one natural draft cooling tower for cycle heat rejection from the main condensers. Heat dissipation to the atmosphere will be approximately 8×10^9 Btu/h for each tower.

The CCW system also supplies water to the auxiliary cooling equipment and the diesel-driven, high-pressure, fire-protection pumps.

The system is designed to meet the thermal pollution requirements of the state of Alabama.

The condenser circulating water system is not a part of the engineered safety feature system (ESF). The towers are constructed with nonflammable materials and are located away from ESF structures.

Water losses are replaced by the essential raw cooling water system (ERCW) and the tower makeup water system (TMUW). Both systems take water from Guntersville Reservoir through the lake intake pumping station. Level controls regulate the amount of water supplied by the tower makeup water system.

Acrolein, an unsaturated aldehyde, is periodically fed into both the ERCW and the TMUW systems at the intake pumping station to control Asiatic clams.

The evaporation losses in the cooling tower increase the dissolved solids content ratio of the CCW. A blowdown system maintains solids at about twice the TMUW concentration ratio.

Periodic chlorination is used for control of slime and algae in the CCW system.

The water of Guntersville Reservoir has a scaling effect rather than a corrosive nature; therefore, the use of inhibitors is not anticipated.

10.4.5.2. System Description

The condenser circulating water system is designed for individual unit operation. Major items in the system are: condensers, cooling towers, circulating water pumping station, water pumps, valves, and conduits. Secondary items include the following subsystems: chlorination, blowdown, and makeup water. (See Table 10.4-2, Circulating Water System Components.)

Water flows from the tower basin through the suction conduit to the circulating pumps located in the CCW pumping station. Isolation valves are used for each pump. Pump discharge flows through the supply conduit to the condensers and is returned to the cooling tower through the discharge conduit. Isolation valves are used for each section of the condenser. (Equipment plans, Figures 10.4-3 and 10.4-4.)

Tower evaporation and drift losses are replaced by the essential raw water system discharge which enters at the suction conduit. Tower makeup water is used to maintain the tower basin water level and to replace tower blowdown. Tower blowdown is used to control the dissolved solid concentrations in the CCW at a ratio of two times the dissolved solid content of the makeup water. Blowdown is discharged to Guntersville Reservoir through a diffuser to provide good distribution.

Periodic chlorination of the CCW is used to control the growth of slime and algae. Periodic use of Acrolein as a biocide for clam control is used in the ERCW and TMUW. Acrolein is essentially scrubbed from the CCW in the tower spray complex. No adverse environmental effect is anticipated from the blowdown water or the tower evaporation.

A mechanical tube cleaning system is used to clean the condenser tubes during on-line service.

When the basin water is drained to a settling area for tower basin inspection, the unit RCW is nonfunctional and the ERCW is diverted to the operating CCW system.

10.4.5.3. Safety Evaluation

The condenser circulating water system is not part of the engineered safety feature system (ESF).

The towers are constructed with noncombustible materials and are located more than one tower height from all ESF structures.

The modes of operation to stop the CCW pumps for an accident are:

1. Stop the pumps normally from the control room.

2. Stop the pumps through the rackout breaker in the switchgear.

3. Stop the pumps through offsite power grid dispatcher.

Normal operation uses four CCW pumps per unit. Functional loss of a pump will not require reduced power generation for the unit but will cause some loss in overall efficiency.

Tower makeup pumps, strainers, and piping located at the intake pumping station are separated from the ESF equipment.

10.4.5.4. Test and Inspection

The system is not an ESF; therefore, no special tests are required. Equipment and systems are tested and inspected to normal power plant specifications. Routine visual inspection of the system, components, instrumentation, and trouble alarms is used to verify that the system is operative.

10.4.5.5. Instrumentation

Plant safety instrumentation includes radiation monitors for the low level, nontritiated, radioactive, liquid waste discharge and discharge flow monitors for the ERCW system.

Nonsafety related instrumentation is used to monitor the system functions either locally and/or in the control room. Alarms are used in conjunction with the monitored functions as required.

The tower makeup water system is controlled automatically based on the tower basin water level. All other nonsafety related equipment is actuated manually either locally or in the control room depending on the operational requirements.

10.4.6. Condensate Cleanup System

10.4.6.1. Design Bases Power Conversion Objective

The objective of the condensate demineralizer system is to remove dissolved and suspended impurities from the secondary system. This demineralizer system removes corrosion products which are carried over from the turbine, condenser, feedwater heaters, and piping, and protects the steam generator feedwater against condenser tube leaks, removes condensate impurities which might enter the system in the makeup water, and isolates radioisotopes which will be carried over from the primary-to-secondary cycle in the event of a steam generator leak.

The system will polish the full flow of condensate under normal operating conditions of 16,155 gpm per unit and up to a maximum flow of 18,056 gpm.

The system will polish condensate before startup and restarts with the steam generator isolated from the feedwater system to prevent a high concentration of crud in the generator as steam generation begins. Before feedwater is introduced into the steam generator, the demineralizer effluent quality will be as shown in Table 10.3-1.

The system is capable of removing dissolved and suspended solids which would be carried over to the condensate system in the event of a condenser tube leak. The system protects the steam generator and turbine from condenser leak contaminants during a simultaneous steam generator tube leak.

The demineralizer effluent and feedwater quality will meet the quality shown in Table 10.3-1 during power generation. Allowable excursions from these limits as a result of condenser tube leaks are shown in Table 10.4-3.

10.4.6.2. System Description

The condensate demineralizer system for each power generation unit consists of five mixed-bed demineralizer vessels. The system also includes shared, external regeneration facilities, consisting of a resin separation and cation regeneration vessel, an anion regeneration vessel, and two resin storage tanks. In an effort to reduce the volume of regenerative waste, a reclamation system has been included in the condensate cleanup system. This system consists of acid and caustic reclamation tanks, a high-conductivity waste neutralization vessel, two 18,000-gallon regenerative waste holdup tanks, and a centrifugal separator. Flow diagrams are shown in Figure 10.4-5.

The demineralizers are shielded from the turbine building and valve galleries. Each of the five demineralizers is isolated in an individual concrete cell. No valves or other equipment with moving parts are located inside the cells. Each pair of demineralizers has a common valve gallery which is shielded from all demineralizers, other valve galleries, and the turbine building. Regeneration equipment is located in individual cells with a common valve gallery. All reclamation equipment is individually shielded from all other equipment and the turbine building.

The demineralizers, which are in parallel, are supplied by the condenser hotwell pumps via an inlet header. The demineralizer system has an automatic bypass valve which opens on failure of control air, when pressure drop across the system exceeds 65 psi, when the effluent pressure drops below 25 psi, when the condensate temperature exceeds 140F, or when the effluent conductivity exceeds 1 μ mho/cm during normal operation. An outlet header collects the effluent from the demineralizers, and the effluent provides the suction for the condensate booster pumps. During normal operation of a unit, all five demineralizers are in service with a spare resin charge in the resin storage tank. The vessels will be fabricated in accordance with the ASME Code for Unfired Pressure Vessels, Section VIII, latest edition, and will be rubber lined.

10.4.6.3. Safety Evaluation

Radioisotopes are released to the secondary system when there is a steam generator tube leak. These radioisotopes concentrate in the demineralizer beds but have essentially no effect on the resin ion exchange capacity. Although the radioisotope concentrations have no effect on resin capacity, they have activity levels which require shielding of demineralizer equipment. (See Table 11.5-2.)

Gaseous waste is removed from the demineralizer area by inducing a negative pressure on the demineralizer vessel cells, valve galleries, and regeneration equipment cells. This waste is scrubbed by HEPA filters, then released. Releases are continually monitored. (See Chapter 11.) Liquid radwaste is sent to the nontritiated waste holdup tank in the radwaste building until evaporated.

Radioactive solid waste removed from backwash water by the centrifugal separator is drummed and sent to the radwaste drum storage area.

Chapter 11 describes the activity level and removal of radioactive material from the system.

10.4.6.4. Tests and Inspections

The condensate cleanup system is used extensively during pipe cleaning and also undergoes a preoperational test prior to startup. After startup and during shutdowns each vessel in the system can be separately isolated for testing or visually inspected.

The system is designed so that all demineralizer vessels, regeneration equipment, and most valves can be separately isolated from the system if testing or inspection is required, with no curtailment or interruption of power generation. Flow control valves on inlet and outlet of demineralizer vessels and system bypass valves can be tested and inspected during shutdowns if required.

10.4.6.5. Instrumentation

Instrumentation and controls are provided to perform the following functions:

1. Measure, indicate, and record water conductivity in the influent header, the effluent line of each demineralizer, and the effluent header.

   High conductivity downstream of a particular demineralizer indicates resin exhaustion, and high conductivity influent indicates condenser tube leakage. Conductivity elements in the effluent header give an indication of system efficiency and total condensate water quality.

   High system conductivity as measured in the effluent header is annunciated locally and in the main control room, and high conductivity at all other points is annunciated at the local control panel.

2. Measure, indicate, and record pressure drop across each demineralizer.

   High-pressure drop across an individual demineralizer indicates crud buildup which is alarmed on the local control panel.

3. Measure pressure differential between influent and effluent headers and open valve bypassing the demineralizer units on high-differential signal.

   High system differential pressure and opening of the bypass valve are annunciated in the main control room.

4. Measure, record, and control flow rates so that regardless of pressure drop across individual units, flow rates to each unit are equal.

5. Measure, indicate, and record the dissolved silica content in the effluent of each demineralizer unit.

   High level silica content is annunciated locally.

6. Annunciate high-pressure differential across each resin trap.

10.4.7. Condensate-Feedwater System

10.4.7.1. Condensate and Main Feedwater System

10.4.7.1.1. Design Bases

The condensate-feedwater system is designed to supply a sufficient supply of condensate from the main condenser hotwell to the steam generator secondary side inlet for heat removal purposes under most normal and abnormal conditions except when feedwater isolation is required.

The maximum steam generator feedwater demand rate is expected to be 16,805,000 pph during the "stretch" operating condition of the NSS (3777± MWt). The feedwater temperature for this operating mode is expected to be 492.6F.

10.4.7.1.2. System Description

Flow diagrams, Figures 10.4-6 through 10.4-8 and Figure 10.4-1.

In order to perform the design objectives, the condensate-feedwater system has been designed with the following specifications:

1. Hotwell pumps:

   Number — 3
   Type — Multistage, single suction, vertical, centrifugal
   Design point — 8155 gpm, 315 ft-hd
   Motor design — 700 hp, 1180 rpm, 6600 V, 3 ph, 60 Hz, vertical, constant speed | 11

2. Condensate booster pumps:

   Number — 3
   Type — Single stage, double suction, horizontal, centrifugal
   Design point — 8155 gpm,1025 ft-hd
   Motor design — 2250 hp, 3550 rpm, 6600 V, 3 ph, 60 Hz, horizontal, constant speed | 11

3. Main feedwater pumps:

   Number — 2
   Type — Single stage, double suction, centrifugal
   Design point — 19,200 gpm, 2600 ft-hd 11

4. Main feedwater pump turbine:

   Number — 2
   Type — Multistage, dual inlet
   Throttle pressure — LP steam, 130 psia; HP steam, 1032 psia 11
   Throttle temperature — LP steam, 521F; HP steam, 602F
   Back pressure — 9" Hg absolute
   Extraction points — None
   Rated horsepower — 14,000 hp 11

5. Main feedwater pump turbine condenser:

   Number — 2
   Tube material — 304 stainless steel
   Channel design pressure* — 150 psi
   Channel design temperature* — 272F

6. Gland steam condenser:

   Number — 1
   Tube material — 304 stainless steel
   Channel design pressure* — 150 psi
   Channel design temperature* — 272F

7. Feedwater heaters:

   Number — 14 (2 streams of 7 heaters)
   Type — Closed, horizontal, U-tube
   Tube material — 304 stainless steel

Heater No.	Channel design pressure,* psi	Channel design temperature,* F	
1	1750	520	
2	1750	520	
3	625	380	
4	625	380	11
5	625	300	
6	625	300	
7	625	300	

- - - - -

*Channel side design conditions only tabulated here. For shell-side design conditions, see section 10.4.9, Heater Drains and Vents.

As discussed in Chapter 15, the following situations require periods of feedwater isolation:

1. Steam generator tube rupture (to preclude release of radioactivity to the environment).

2. Steam line break accident (to prevent uncontrolled depressurization and cooldown of the reactor coolant system).

The condensate and feedwater system pumps take suction from the main condenser hotwells and deliver water to the steam generators at an elevated temperature and pressure. These systems are designed to provide 14,795,000 lb/h of 477.7F water to the steam generators under turbine generator guaranteed load conditions.

Feedwater heaters are designed in accordance with HEI Standards for Closed Feedwater Heaters and the ASME Boiler and Pressure Vessel Code, Section VIII. Pumps are designed in accordance with Hydraulic Institute Standards for Centrifugal Pumps. All piping and valves from the condenser hotwell to the feedwater isolation valve is designed in accordance with ANSI B31.1, 1967, while the remainder of the feedwater system (downstream of the feedwater isolation valve) is nuclear Class 2 and is designed in accordance with ANSI B31.7.

The systems' boundaries extend from the condenser hotwell to the inlet nozzle of the steam generator. The hotwell of the condenser has a water storage capacity equivalent to approximately 2-1/2 minutes of full-load operation. Condensate is taken from the main condenser hotwells by three vertical, centrifugal, motor-driven hotwell pumps. The head imparted by these pumps is sufficient to provide adequate NPSH to the condensate booster pumps during all modes of unit operation. The hotwell pumps and condensate booster pumps (which start simultaneously as load increases) when operating in series, are capable of delivering required flow with sufficient NPSH to the main feedwater pumps under all normal operating conditions.

Feed flow to the individual steam generators is controlled automatically above 15% load by adjustment of a feedwater regulator valve in the piping to each steam generator. The valve position is determined by a three-element controller that uses steam generator water level, steam flow, and feedwater flow as the control variables. The regulator valves are pneumatically operated and are designed to fail closed on loss of air.

During startup and operation below 15% load, additional control is available from manual operation of small bypass valves around the feedwater regulator valves.

The systems normally operate at full load with three hotwell, three condensate booster and two main feed pumps in service. However, with all feedwater heaters in service and all heater drains being pumped forward, 100% of guaranteed load can be maintained with only two hotwell and two condensate booster pumps running. Unit load can be continuously maintained at 70% guaranteed load with only one main feed pump in operation.

Heating of the condensate-feedwater is accomplished by passing it through a series of closed heat exchangers as described below:

1. Gland Steam Condenser — This exchanger condenses the steam leakoff from all turbine shaft seals and removes the noncondensables (the result of shaft inleakage of air) from this steam. An externally connected, weighted check valve is provided to ensure minimum required flow through the condenser at low condensate flow conditions and to minimize pressure drop through the condenser during high condensate flow conditions.

2. Main Feed Pump Turbine Condensers — Each main feed pump turbine is equipped with an individual surface type condenser. Control valves in the inlet and outlet condensate piping to these condensers provide the ability to isolate a condenser if its associated drive turbine is rendered inoperative and to force all condensate flow through the operating condenser, thus allowing maximum power operation of the remaining drive turbine.

3. Feedwater Heaters — Two parallel strings of heaters, each consisting of five intermediate pressure feedwater heaters, and two high-pressure feedwater heaters are provided. The heaters are numbered from 1 to 7 with the highest pressure heater designated as No. 1. Motor-operated isolation valves are provided at the inlet to each No. 7 heater and the outlet to each No. 5 heater, the inlet of each No. 4 heater, and the outlet of each No. 3 heater, and at the inlet of each No. 2 heater and outlet of each No. 1 heater. Each of these isolation groups (1,2; 3,4; 5,6, and 7) is equipped with a bypass which is included with an automatic isolation valve that opens on high-high level in the isolation group. High-high level in a heater shell will cause the isolation of the group of heaters in the stream in which the high-high level occurred (either the 5, 6, 7, heaters, 3 and 4 heaters, or 1 and 2 heaters in either the A or B stream).

Minimum flow bypasses are provided for equipment protection. The condensate system minimum flow bypass is located immediately upstream of the No. 7 heaters. The bypass control valve received its operating signal from the station flow nozzle located upstream of the gland steam condenser. The valve plug's position is modulated to maintain at least 3500 gpm flow through the flow nozzle. This flow is sufficient to protect the hotwell and condensate booster pumps and to provide adequate cooling water to the gland steam condenser at all times.

The feedwater system has a minimum flow bypass line located downstream of each feedwater pump to permit direct recirculation back to the main condensers for protection of the main feed pumps. The bypass control valve modulates in a manner similar to the condensate minimum flow valve.

Piping is provided around the main feed pumps to allow filling the steam generators without operating the main feed pumps.

A demineralizer for maintenance of proper feedwater quality is located downstream of the hotwell pumps. Additional components of the condensate and feedwater systems include an injection water system to provide sealing water to all system pumps, condensate storage tanks which provide capability of controlling feedwater inventory by regulating condenser hotwell level and which provide storage of the water required for operation of the auxiliary feedwater system, and facilities for injection of hydrazine for oxygen scavenging and volatile amines for feedwater pH control.

Complete isolation of feedwater from all steam generators is required only if a feedwater isolation signal or a safety injection signal is initiated.

Inservice inspection of all Class 2 components of the feedwater system will be performed in accordance with ASME Boiler and Pressure Vessel Code, Section XI, Rules for Inservice Inspection of Class 2 Nuclear Power Plant Components, SC-261(b).

10.4.7.1.3. Safety Evaluation

Feedwater flow to the steam generators must be interrupted upon initiation of a feedwater isolation signal. This isolation, accompanied by a reactor trip, is accomplished by closure of redundant valves in the piping to each steam generator.

For additional protection, a feedwater isolation signal also trips both main feed pumps (from either train A or train B signal). Trip of these pumps then initiates an orderly shutdown of the remainder of the condensate-feedwater system pumps. The shutdown sequence is as follows:

1. Trip of both main feed pumps will initiate closure of condensate isolation valves at the inlet and outlet of both main feed pump turbine condensers.

2. A logic interlock prevents operation of hotwell pumps unless both valves serving at least one main feed pump turbine condenser are fully open. Thus, all hotwell pumps will trip.

3. Trip of the hotwell pumps will cause the NPSH available to the condensate booster pumps to drop below that required for operation and a low NPSH trip of the booster pumps will occur.

High level in any one steam generator initiates rapid closure of the feedwater regulator valve associated with the steam generator in which the high level occurs. A return to satisfactory steam generator level removes the isolation signal and allows the regulator valve to resume normal operation.

With all heater drains being pumped forward and all heater banks in service, each hotwell pump and each condensate booster pump is capable of delivering 50% of the unit guaranteed flow while imparting sufficient head to the feedwater to meet all system demands. Thus, loss of any one of the three hotwell pumps and/or any one of the three condensate booster pumps simply results in flow being transferred to the remaining operational pumps with the reactor coolant system being unaffected.

If the unit is operating below 55% guaranteed load, loss of one main feed pump has no effect on the reactor coolant system, since one main feed pump is capable of delivering greater than 55%. If the unit is operating above 55% guaranteed load and loss of one main feed pump occurs, the following events will occur:

1. Isolation of the main feed pump turbine condenser associated with the tripped pump. Thus the total remaining condensate flow is passed through the active main feed pump turbine condenser allowing maximum power operation of the active feed pump turbine.

2. Acceleration of the active drive turbine to its "high speed stop" speed.

3. Unit load runback is initiated and unit load decreases to 55%, thus allowing the operator to return to normal one main feed pump operation as soon as possible.

Insufficient NPSH at the main feed pump suction can result in a decrease in steam generator level. Low NPSH at the main feed pump suction is annunciated in the main control room, thereby alerting the unit operator to the need for a load runback to avoid a reactor coolant system transient.

The effect of equipment malfunction in the reactor coolant system is discussed in Chapter 15.

10.4.7.1.4. Inspection and Testing

The operating characteristics for each system pump will be established throughout the operating range by factory tests. Each hotwell and condensate booster pump casing will be tested hydrostatically to 150% of its shutoff head plus maximum suction pressure. All parts of each turbine driven main feed pump subject to hydraulic pressure in service will be hydrostatically tested to not less than 150% of the maximum pressure to which these parts are subjected when the pump is operating at rated speed and zero flow, with maximum suction pressure from the hotwell and condensate booster pumps.

All parts and assemblies of parts of the feedwater heaters will be hydrostatically tested and tested otherwise as required by applicable sections of Heat Exchange Institute Standards for Closed Feedwater Heaters; and Section VIII for Unfired Pressure Vessels of the ASME Boiler Code. Heater tubes will be tested as required by ASTM A 249, latest edition, except paragraphs 10.1 and 10.2.1 will be applicable.

Hydrostatic and other testing of the parts and assemblies of parts of the main feed pump turbine condensers channels and tubes will be in accordance with applicable sections of the Heat Exchange Institute Standards for Closed Feedwater Heaters and Section VIII, Unfired Pressure Vessels, of the ASME Boiler Code.

Manways or removable heads are provided on all heat exchangers to provide access to the tube sheet for inspection, repair, or tube plugging.

All Class 2 components of the feedwater system will be subjected to inservice inspection in accordance with ASME Boiler and Pressure Vessel Code, Section XI, Rules for Inservice Inspection of Class 2 Nuclear Power Plant Components, ICS-261(b).

10.4.7.1.5. Instrumentation

Sufficient level controllers, flow controllers, level switches, pressure switches, limit switches, temperature switches, etc., will be provided to permit personnel to conveniently and safely operate the condensate-feedwater system.

10.4.7.2. Auxiliary Feedwater System

10.4.7.2.1. Design Bases

The auxiliary feedwater system is designed to supply adequate secondary side cooling, in the event of a loss of main feedwater for any reason, to remove system stored and residual core energy in accordance with General Design Criteria 34. It may also be used in the event the main control room must be evacuated.

The auxiliary feedwater pumps are sized to provide sufficient feedwater to remove decay heat at 30 seconds after trip, plus reactor coolant pump heat, to permit a smooth transition from power operation to decay heat removal. The system is capable of delivering its rated flow against the highest steam generator safety valve setpoint plus pressure accumulated in the steam generator and lines. A supply of clean condensate water and an unlimited, redundant, backup supply of emergency raw water are available.

The auxiliary feedwater system is a safety system designed to engineered safety feature (ESF) standards. It will remain functional following an SSE, a loss of power, or a single failure in itself or its supporting systems.

10.4.7.2.2. System Design

Design Description

The two reactor units have separate auxiliary feedwater systems, as shown in Figure 10.4-8. Only one system is discussed here; the other is identical.

The 1200-gpm turbine-driven pump will supply 600-gpm flow to each steam generator, or 1200 gpm to either; each 600-gpm electric-driven pump will supply one-half of the full flow requirement to either steam generator. Any combination of two pumps, or the turbine-driven pump alone in the case of loss of all a-c power, will supply the design requirement of feedwater. In the event of a steam or feedwater line break, the system will deliver the required flow to the unaffected, operable steam generator.

One electric pump is powered by train A emergency power, and the other by train B; the turbine-driven pump receives steam from either or both steam generators, with the supply line from the inoperable steam generator isolated to prevent a blowdown of the unaffected steam generator. The turbine exhausts to the atmosphere.

The primary source of water for all auxiliary feedwater pumps is the unit's condensate storage tank (which is actually a part of the condensate system).

A minimum of 300,000 gallons in this tank is reserved for the auxiliary feedwater system; this is accomplished by means of a standpipe in the tank, through which other systems are supplied. As an unlimited backup water supply, a separate essential raw cooling water (ERCW) system header feeds each electric pump; the turbine pump receives backup water from either or both ERCW headers. The ERCW supply is automatically (or remote-manually) initiated on a two-out-of-three low-pressure signal in the condensate suction line. Since the ERCW system supplies poor quality water, it will not be used except in emergencies when the condensate supply is unavailable. The ERCW is described in section 9.2.1.

The auxiliary feedwater system is designed to deliver 40-90F water for pressures ranging from the decay heat removal (DHR) system cut-in point (the equivalent pressure in the steam generator) to the highest setpoint of the steam generator safety valve (1245 psig) plus pressure accumulated in the steam generator and lines. System piping downstream of the pumps is designed for pressures up to the maximum discharge head of the pumps. Table 10.4-4 lists design conditions for the nodal points on Figure 10.4-8.

A separate, ESF-quality power subsystem and control air subsystem serves each electric-driven auxiliary feedwater pump and its associated valves. The turbine-driven pump and its associated valves are served by both electric and control air subsystems, with appropriate measures precluding any interaction between the two subsystems. Details are given in Chapter 7.

Material Compatibility, Codes, and Standards

Generally, components will be of carbon steel. The condensate storage tanks will be lined to prevent corrosion; other components will be protected by chemical additions to the water as necessary.

Applicable codes and standards are given in Table 10.4-5; ANS PWR Criteria, 1972 draft, safety classifications are shown on Figure 10.4-8.

System Reliability

In addition to using high quality components and materials, the auxiliary feedwater system design provides sufficient pump capacity and water supply, with appropriate redundancy, for all cases for which the system is required. Only one steam generator is required to be usable for any credible accident condition.

Redundant electrical power and air supplies assure reliable system initiation and operation. The electric-motor-driven pumps are powered by either offsite or onsite sources; the turbine-driven pump takes steam from both main steam lines upstream of the main steam isolation valves.

Essential parts of the system are seismically qualified and protected against missiles by the physical separation of all redundant components.

10.4.7.2.3. Design Evaluation

Analyses of accidents that require auxiliary feedwater are given in Chapter 15. The system is designed to be capable of delivering sufficient feedwater flow over the required pressure range for all design bases considerations given in section 10.4.7.2.1.

In the event of a total loss of plant a-c power, including emergency onsite power, the turbine-driven auxiliary feedwater pump will take water from the condensate tank, which contains enough water for about 1 day of decay heat removal. All valves necessary to control flows in this case are operated on station batteries and air reservoirs, with a-c-powered makeup unnecessary in the short run; manual valve manipulation will be used in the long run.

In the similar feedwater and steam line break accidents, a minimum of 1200 gpm of feedwater can be automatically supplied to the unaffected steam generator despite any single active failure in the auxiliary feedwater system or any supporting system.

All other transients impose less severe requirements.

10.4.7.2.4. Tests and Inspections

Performance tests of individual components in the manufacturer's shop, integrated preoperational tests of the whole system, and periodic performance tests of the actuation circuitry and mechanical components will assure reliable performance.

During plant operation, the design allows the system to be tested by pumping condensate storage water to the steam generators. ERCW water will not be fed to the steam generators during tests; but the ERCW system will be tested separately, and the valves between the two systems will be tested by operating them while no ERCW water is being supplied.

Testing requirements are given in Chapter 16.

10.4.7.2.5. Instrumentation

The auxiliary feedwater pumps will be automatically actuated on:

1. Low discharge pressure on both main feedwater pumps.

2. Low feedwater level in either steam generator.

3. Low steam generator outlet pressure.

They may also be manually controlled.

Upon a low main steam line pressure signal, the isolation valves in the auxiliary feedwater injection lines close. The isolation valves reopen when the steam generator they serve repressurizes. The valves serving a steam generator that does not repressurize, indicating a steam or feedwater line rupture which cannot be isolated, remain closed.

Control valves downstream of each pump are modulated automatically to maintain the proper feedwater level in the steam generators. These valves will fail open on loss of air or control power.

Initiation and control of the auxiliary feedwater system is by the essential controls and instrumentation system (ECI) described in Chapter 7. All necessary instrumentation and control features are available in both the main and auxiliary control rooms. (See Figure 10.4-9 for a control diagram schematic.)

10.4.8. Steam Generator Blowdown System

The necessity for steam generator blowdown is not anticipated since periodic chemical cleaning of the steam generators will be used to remove suspended solids.

10.4.9. Heater Drains and Vents

10.4.9.1. Design Bases

The heater drain system is designed to remove and dispose of all drains from the moisture separators, reheaters, feedwater heaters, main feed pump turbine condensers, and gland steam condensers during all modes of unit operation.

The vent system is designed to adequately vent all heat exchangers to assure complete removal of noncondensible gases during all modes of unit operation.

10.4.9.2. System Description (Flow Diagram, Figure 10.4-10)

In order to perform the design objectives, the heater drain and vents systems have been designed with the following specifications:

1. No. 3 heater drain pumps:

   Number — 3
   Type — Single stage, double suction, centrifugal, horizontal
   Design point — 4600 gpm, 1100-ft head
   Motor design — 2000 hp, 3550 rpm, 6600 V, 3 ph, 60 Hz, horizontal, constant speed 11

2. No. 7 heater drain pumps:

   Number — 2
   Type — Multi-stage, centrifugal, horizontal
   Design point — 2000 gpm, 1100-ft head
   Motor design — 800 hp, 3550 rpm, 6600 V, 3 ph, 60 Hz, horizontal, constant speed 11

3. Feedwater heaters:

   Shell side design conditions only given here. See section 10.4.7, Condensate-Feedwater System, for channel side design conditions.

   Number — 14 (2 streams of 7 heaters)

Heater No.	Shell design pressure, in. Hg vacuum	Shell design temperature, F	
			11
1	750/30	650	
2	350/30	650	
3	175/30	400	

Heater No.	Shell design pressure, in. Hg vacuum	Shell design temperature, F	
4	50/30	400	11
5	50/30	300	
6	50/30	300	
7	50/30	300	

4. Main feed pump turbine condensers:

   Shell-side design conditions only given here. See section 10.4.6, Condensate-Feedwater System, for channel side design conditions.

   Number — 2
   Shell design pressure — 20 psi and 30 in. Hg vacuum
   Shell design temperature, F — 160

5. No. 7 heater drain tank:

   Number — 1
   Design pressure — 1" hg abs - 50 psig/30 in. Hg vacuum 11
   Design temperature, F — 175

6. No. 3 heater drain tank:

   Number — 1
   Design pressure — 175 psig/30 in. Hg vacuum 11
   Design temperature — 388F

7. High-pressure reheater drain tanks

   Number — 4
   Design pressure — 1200 psig
   Design temperature — 567F

8. Low-pressure reheater drain tanks:

   Number — 4
   Design pressure — 500 psig
   Design temperature — 467F

9. Main feed pump turbine condenser drain tanks:

   Number — 1
   Design pressure — 1" hg abs - 16 psig
   Design temperature — 101F

10. Main feed pump turbine condenser drain pumps:

    Number — 2
    Type — Single stage, single suction, centrifugal
    Design point — 300 gpm, 55 ft-head
    Motor design — 7.5 hp, 1750 rpm, 3 ph, 60 Hz, constant speed

The heater drains and vents system is designed to return water condensed by feedwater heating back to the condensate-feedwater system and to adequately vent all heat exchangers to assure complete removal of noncondensable gases during all modes of unit operation.

The tube and shell side of all feedwater heaters are equipped with manually valved vent lines to the main condenser for ventilation during unit startup. Venting to the main condenser during normal operation is accomplished through continuous "free blowing" orifices, sized in accordance with recommendations of the Heat Exchange Institute Standards for Closed Feedwater Heaters, 1968. The venting scheme for the moisture separators, high-pressure reheaters, and low-pressure reheaters is similar to that employed for the feedwater heaters.

The heaters are numbered from 1 to 7 with the highest pressure designated as No. 1. During normal unit operation, the No. 1 heater drains, composed of the high-pressure reheater drains and the No. 1 extraction, cascade into the shell of the No. 2 heater. The No. 2 heater drains, the No. 1 drains, plus the No. 2 extraction and the low-pressure reheater drains, cascade into the No. 3 heater drain tank. The No. 3 heater drains (No. 3 extraction) and the moisture separator drains also flow into the No. 3 heater drain tank. Water from the No. 3 heater drain tank is then pumped forward into the condensate cycle (at the main feed pump suction) by the No. 3 heater drain pumps.

The first extraction from the low-pressure turbines is condensed in the No. 4 heaters. These drains are cascaded into the shell of the No. 5 heaters. No. 5 heater drains (No. 5 extraction plus No. 4 heater drains) cascade to the No. 6 heater, whose drains cascade in turn to the No. 7 heater drain tank. The condensed No. 7 extraction, main feed pump turbine condenser drains, and other miscellaneous drains are also routed to the No. 7 heater drain tank. This water is pumped forward into the condensate system (at a point between the No. 7 and No. 6 heaters) by the No. 7 heater drain pumps.

Proper level is maintained in the Nos. 1, 2, 4, 5, and 6 feedwater heaters by modulating level control valves that receive their control signal from level indicating controllers mounted on the heater shells. Should the level drop below the normal control range, a low level alarm is sounded. High level in the shells of the No. 2, 5, and 6 heaters results in annunciation of a high level alarm. The No. 4 heater is equipped with modulating bypass to No. 7 heater drain tank valves. Should the level in a No. 4 heater exceed the normal control level, the bypass valve begins to open. Indication is given in the control room when the bypass valve leaves its seat. If the level exceeds the control range of the bypass valve, high level annunciation occurs. High-high level in a No. 1 or a No. 2 heater results in isolation (of both feedwater and extraction steam) of the appropriate bank of No. 1 and No. 2 heaters. High-high level in No. 4 heater results in isolation of the appropriate bank of No. 3 and 4 heaters. High-high level in a No. 5 and 6 heater results in isolation of the appropriate bank of No. 5, 6, and 7 heaters. Additional level increase in a No. 5, 6, 7 heater will initiate breaking of condenser vacuum and will result in a turbine trip.

All No. 3 and No. 7 heaters are "dry" shelled heaters. Thus, particular care in piping design was taken to ensure that choking of drains as a result of steam entrainment will not occur. Indication of high level in a No. 3 or No. 7 heater will result in isolation of the appropriate bank of feedwater heaters.

Level in the No. 3 heater drain tank is maintained within the proper range by modulating level control valves at the discharge of the No. 3 heater drain pumps. Level in excess of normal control range initiates opening of modulating bypass to condenser valves. Indication that the bypass to condenser valve has left the fully closed position is given in the control room. Additional increase in level to a point above the range of the bypass valves annunciates a high level alarm. Further increase in level will result in a turbine trip. Low level in the drain tank results in a trip of all operating No. 3 heater drain pumps.

A level control scheme identical to that for the No. 3 heater drain tank is provided for the No. 7 heater drain tank.

The moisture separator drains are routed to the No. 3 heater drain tank, low-pressure reheater drains to the No. 2 heater shells, and the high-pressure reheater drains to the No. 1 heater shells. Since all moisture separators and reheaters are "dry" shelled vessels, particular care in design of drain piping was taken to prevent choking of drain flow due to steam entrainment.

Moisture separator-reheater system drain control is provided by maintaining proper level in drain pots connected to the moisture separator shell, and drain tanks associated with the HP reheaters and LP reheaters. This level is controlled in the drain tanks by modulating level control valves (one per tank) that receive their signals from tank mounted level indicating controllers.

Level in excess of the normal control range causes a modulating bypass to condenser valve (one per tank) to open. Indication that a bypass valve has left the fully closed position is given in the main control room. Increase in level to above the control range of the bypass to condenser valve results in a high level alarm being annunciated in the control room. Low level alarm is also annunciated if the level drops below the normal control range.

Air assisted nonreturn valves are provided in each moisture separator-reheater drain line downstream of the point where the bypass to condenser piping is connected so that, in the event of a turbine pressure transient due to load rejection, the water stored in the feedwater heaters cannot flash back to the moisture separator-reheater. The bypass to condenser valves will still be available for level control during a transient of this type.

Feedwater heaters are designed in accordance with applicable sections of Heat Exchange Institute Standards for Closed Feedwater Heaters, and Section VIII, Unfired Pressure Vessels, of the ASME Boiler Code. The No. 3 and No. 7 heater drain tanks are designed in accordance with Section VIII, Unfired Pressure Vessels of the ASME Boiler Code. All moisture separators, reheaters, and associated drain tanks are designed to Section VIII of the ASME Code and require an ASME certification stamp. All piping and valves in the heater drains and vents system are designed in accordance with ANSI Standard B31.1.

A single drain tank receives the drains from both main feed pump turbine condensers. Normal water level in the tank is maintained by a level control valve at the drain pump discharge that receives its control signal from a level indicating controller mounted to the drain tank. Level below the control range of the controller results in annunciation of a low level alarm. Level above the control range results in annunciation of a high level alarm. A further increase in level (to the high-high level) initiates opening of a bypass to condenser valve. Direct operator action is required to close bypass valve after it has opened.

All heater drains will be returned to the condenser (for cleanup in the polishing demineralizer) during startup and low load operation. The three No. 3 heater drain pumps will start sequentially as the unit load increases above this minimum load setpoint. The particular order in which the pumps start is determined by the position of a selector switch. Conditions that must be satisfied before any pump can start include:

1. Level in the No. 3 heater drain tank above a permissive level setpoint.

2. Sufficient lubricating oil pressure.

The pumps are sequentially tripped on decreasing load. In addition, the pumps may be tripped by low level in the No. 3 heater drain tank, low lube oil pressure, or a motor protection signal.

Minimum flow for pump protection is provided by an automatic recirculation control valve at the discharge of each pump.

The No. 7 heater drain pumps are controlled and protected in the same manner as the No. 3 heater drain pumps.

The main feed pump condenser drains are equipped with two 100% capacity pumps which take suction from a single drain tank. One pump is started manually while the second pump is put on standby by placing the selector switch in the auto position. Should the pressure in the discharge of the active pump drop below a pressure switch setpoint, the standby pump is automatically started. A trip of the main turbine will result in tripping of all No. 3 and No. 7 heater drain pumps.

All pumps conform to applicable paragraphs of the centrifugal pump section of the latest Standards of the Hydraulic Institute.

10.4.9.3. Safety Evaluation

With few exceptions, the operating mode of the heater drains system has no effect on the reactor coolant system and the ability of the condensate-feedwater system to deliver feedwater to the steam generators in sufficient quantity to meet all system demands. However, some transient conditions can exist that do require proper interfacing between the heater drains system and other secondary cycle systems to prevent a reactor trip. These conditions will be analyzed and proper action taken to prevent undesirable effects on the reactor coolant system.

10.4.9.4. Inspection and Testing

All pumps, heaters, and pressure vessels in the heater drains and vents system will be tested by the manufacturer in accordance with the codes under which they are manufactured. Since there are no Class 2 components in this system, no inservice inspection is required.

10.4.9.5. Instrumentation

Sufficient level controllers, level switches, pressure switches, temperature switches, limit switch, etc., will be provided to permit personnel to conveniently and safely operate this drain system.

Table 10.4-3. Allowable Time Periods for Correction of High Feedwater Cation Conductivity

Cation conductivity range, μ mho/cm		Allowable time to correct feedwater conductivity before shutdown is required
Above	Below	
0.5	1.0	24 hours
1.0	2.0	12 hours
2.0	--	0 hour (initiate normal procedure for plant shutdown)

Table 10.4-4. Auxiliary Feedwater System — Design Parameters

Node point,[a] AW-	Design temperature, F	Design pressure, psia	Design flow rate, gpm
1	125	65	2400
2	125	65	600
3	125	65	1200
4	125	65	600
5	150	150	600
6	150	150	600
7	150	150	600
8	150	150	600
9	125	1400	600
10	125	1400	600
11	125	1400	600
12	125	1400	1200
13	125	1400	1200
14	125	1400	1200
15	125	1400	1200
16	125	1400	1200
17	125	1400	1200

[a]Locations marked on Figure 10.4-8.

Table 10.4-5. Auxiliary Feedwater System Applicable Codes and Standards

Condensate storage tank

a. AWWA D 100

Condensate supply to pump suction, piping, and valves

a. ANSI B31.1.0

Auxiliary feedwater pumps, valves, and piping

a. Category I Seismic Requirements

b. ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components — Classes 2 and 3

Essential raw cooling water pumps, valves, and piping

See section 9.2.1

Table 10.4-3. Allowable Time Periods for Correction of High Feedwater Cation Conductivity

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0.5	1.0	24 hours
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1	125	65	2400
2	125	65	600
3	125	65	1200
4	125	65	600
5	150	150	600
6	150	150	600
7	150	150	600
8	150	150	600
9	125	1400	600
10	125	1400	600
11	125	1400	600
12	125	1400	1200
13	125	1400	1200
14	125	1400	1200
15	125	1400	1200
16	125	1400	1200
17	125	1400	1200

[a]Locations marked on Figure 10.4-8.

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Auxiliary feedwater pumps, valves, and piping

a. Category I Seismic Requirements

b. ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components — Classes 2 and 3

Essential raw cooling water pumps, valves, and piping

See section 9.2.1

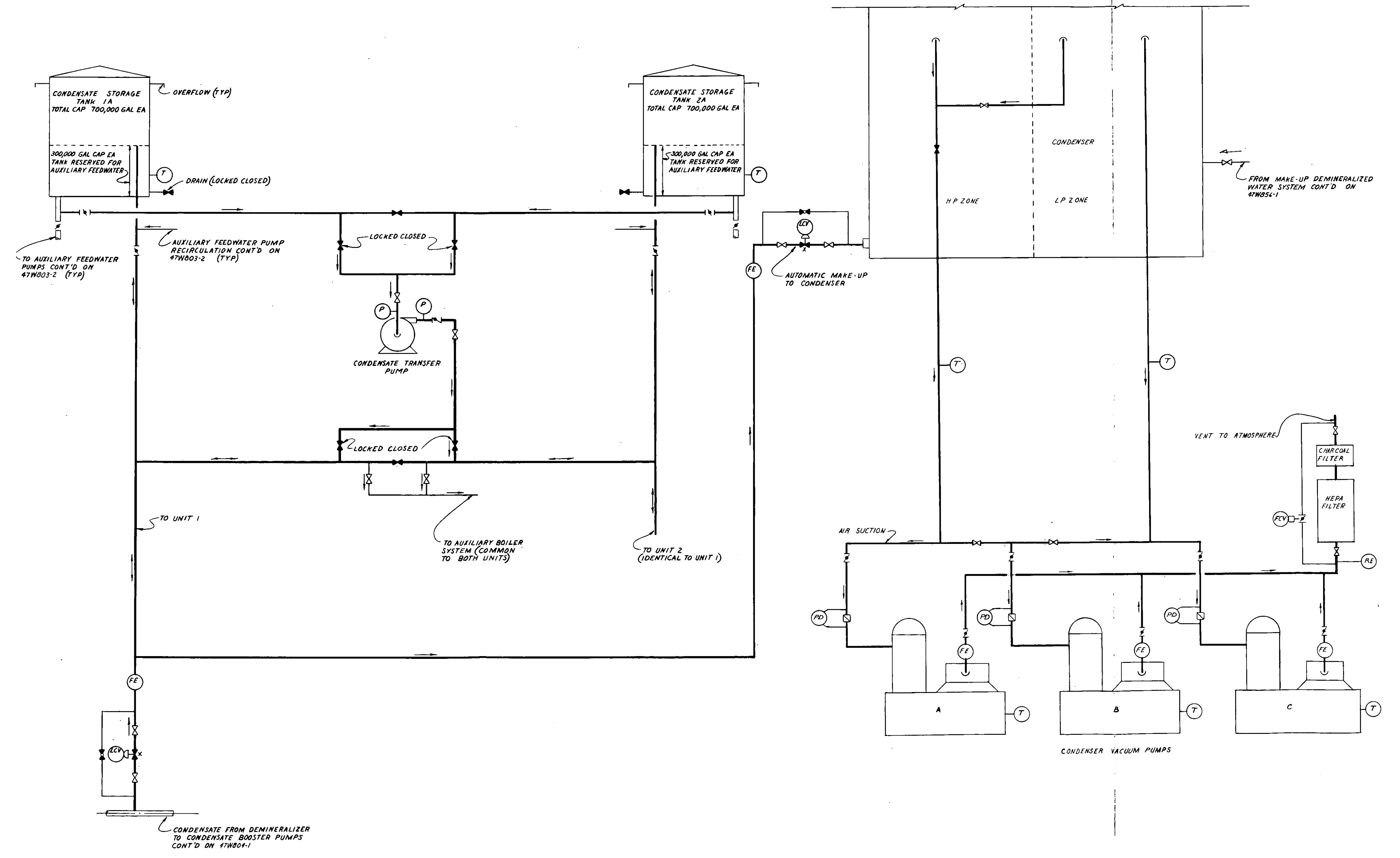
Table 10.4-1. Main Condenser Radioactive Contaminant Inventory During Power Operation and Shutdown

Isotope	Inventory, Ci/m^3	Decay constant, s^{-1}
^{84}Br	0.3930×10^{-8}	0.3850×10^{-3}
^{85}Br	0.1154×10^{-9}	0.3850×10^{-2}
$^{85}Kr^m$	0.0	0.4410×10^{-4}
^{85}Kr	0.0	0.2140×10^{-8}
^{87}Kr	0.0	0.1480×10^{-3}
^{88}Kr	0.0	0.6950×10^{-4}
^{88}Rb	0.8289×10^{-7}	0.6490×10^{-3}
^{89}Sr	0.7363×10^{-9}	0.1480×10^{-6}
^{90}Sr	0.2138×10^{-10}	0.7850×10^{-9}
^{91}Sr	0.5755×10^{-7}	0.1990×10^{-4}
^{92}Sr	0.1827×10^{-8}	0.7130×10^{-4}
^{90}Y	0.1282×10^{-10}	0.2980×10^{-5}
^{91}Y	0.4036×10^{-9}	0.1380×10^{-6}
^{99}Mo	0.3054×10^{-6}	0.2880×10^{-5}
^{106}Ru	0.2848×10^{-6}	0.2200×10^{-7}
$^{131}Xe^m$	0.0	0.6680×10^{-6}
$^{133}Xe^m$	0.0	0.3490×10^{-5}
^{133}Xe	0.0	0.1520×10^{-5}
$^{135}Xe^m$	0.0	0.7400×10^{-3}
^{135}Xe	0.0	0.2110×10^{-4}
^{129}I	0.7178×10^{-14}	0.2180×10^{-14}
^{131}I	0.5600×10^{-6}	0.9960×10^{-6}
^{132}I	0.2665×10^{-6}	0.8020×10^{-4}
^{133}I	0.7393×10^{-6}	0.9250×10^{-5}
^{134}I	0.6848×10^{-7}	0.2200×10^{-3}
^{135}I	0.3976×10^{-3}	0.2890×10^{-4}
^{134}Cs	0.2983×10^{-7}	0.1100×10^{-7}
^{136}Cs	0.1797×10^{-7}	0.6170×10^{-6}
^{137}Cs	0.6891×10^{-7}	0.8270×10^{-9}
^{138}Cs	0.3631×10^{-7}	0.3620×10^{-3}
$^{137}Ba^m$	0.3869×10^{-6}	0.4440×10^{-2}
^{139}Ba	0.1477×10^{-7}	0.1360×10^{-3}
^{140}Ba	0.9619×10^{-9}	0.6290×10^{-6}
^{140}La	0.3369×10^{-9}	0.4790×10^{-5}
^{144}Ce	0.7680×10^{-10}	0.2760×10^{-7}
^{51}Cr	0.2262×10^{-8}	0.2890×10^{-6}
^{54}Mn	0.2602×10^{-9}	0.2650×10^{-7}
^{59}Fe	0.2602×10^{-9}	0.1780×10^{-6}
^{58}Co	0.1357×10^{-7}	0.1130×10^{-6}
^{60}Co	0.7579×10^{-8}	0.4180×10^{-8}
^{95}Zr	0.1810×10^{-7}	0.1270×10^{-6}

Table 10.4-2. Condenser Circulating Water System Components

	Unit 1	Unit 2
Circulating water pumps		
Number	4	4
Type	Centrifugal, double suction	
Capacity, gpm at	116,900	116,900
Head, ft[a]	98	98
Isolation valves, motor-operated, butterfly	8	8
Cooling towers		
Number	1	1
Diameter, ft[a]	400+	400+
Height, ft[a]	500+	500+
Condensers		
Number	1	1
Isolation valves, motor-operated, butterfly	4	4
Circulating water conduits		
Suction	1	1
Supply	1	1
Discharge	1	1
Material		
Yard portion	Precast concrete, steel cylinder type	
Turbine building portion	Steel lined, concrete cast in place	
Plant bases		
Tower makeup water pumps		
Number	3; Shared	3; Shared
Type	Vertical, turbine	Vertical, turbine
Capacity, gpm at	21,000	21,000
Isolation valves, butterfly type	1 per pump (3)	1 per pump (3)
Makeup water strainers		
Number	3; Shared	3; Shared
Capacity, gpm at	21,000	21,000
Isolation valves, butterfly type	2 per strainer (6)	2 per strainer (6)

(a) Estimated.



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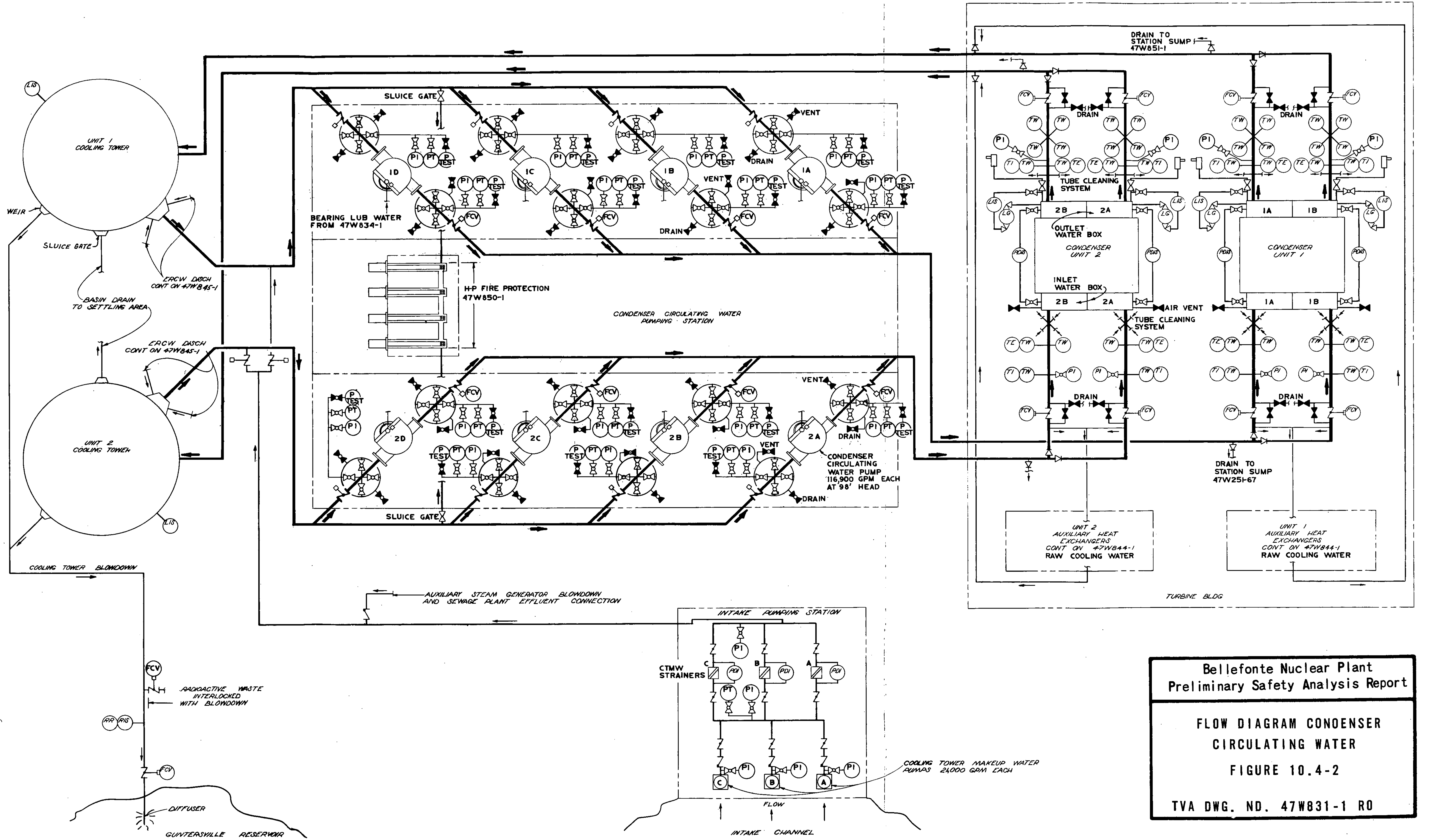
FLOW DIAGRAM CONDENSATE TRANSFER AND STORAGE, VACUUM REMOVAL

FIGURE 10.4-1

TVA DWG. NO. 47W004-2 R0



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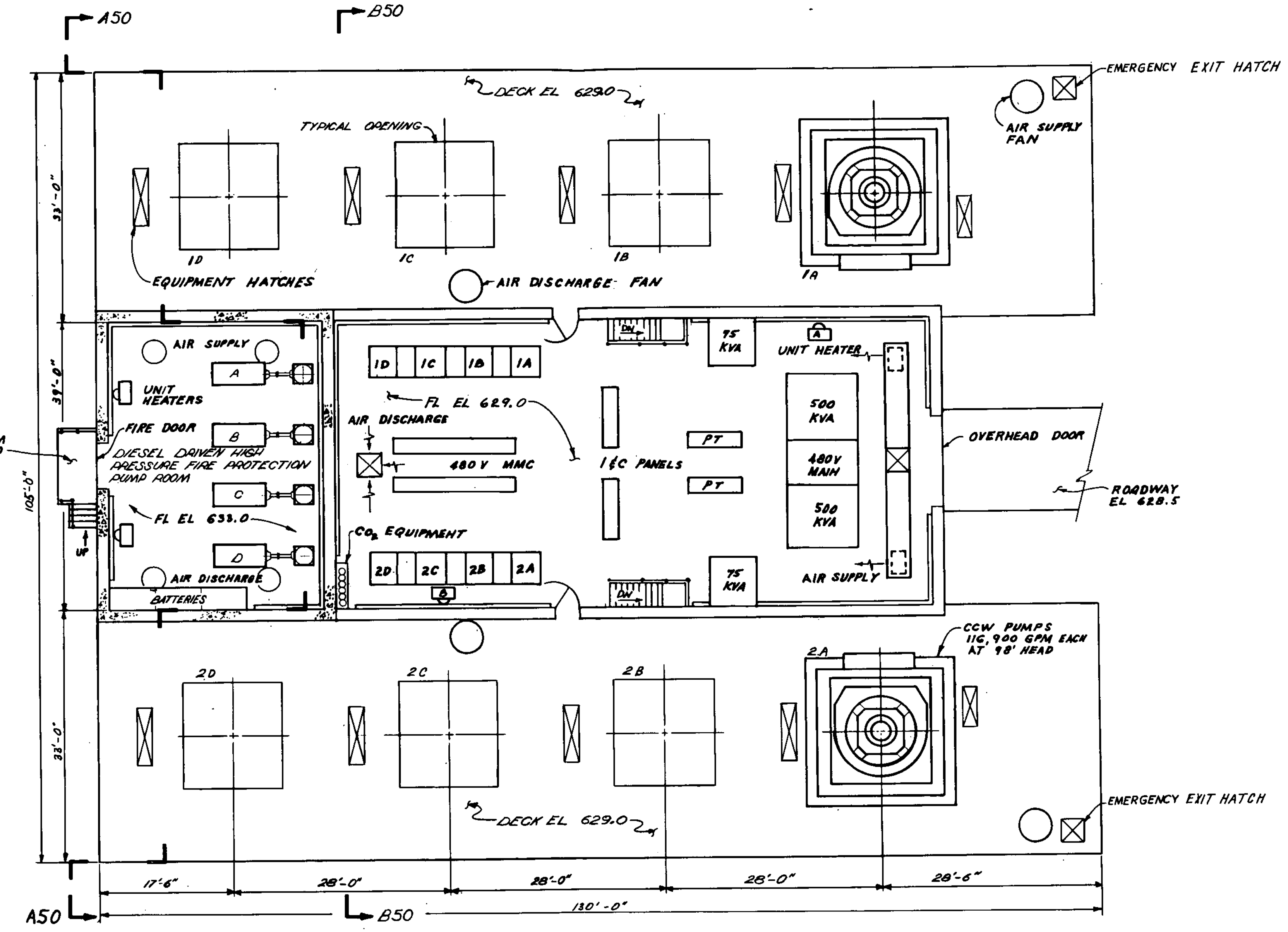
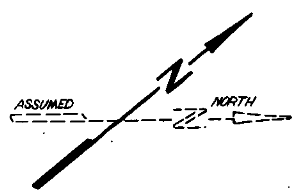
FLOW DIAGRAM CONDENSER CIRCULATING WATER

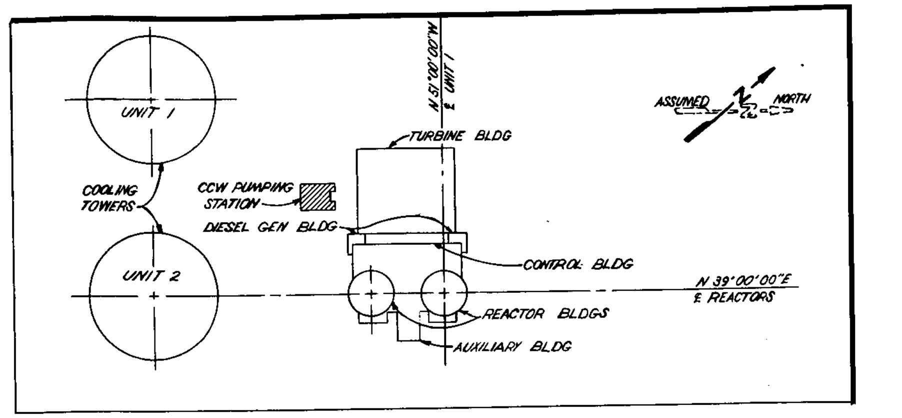
FIGURE 10.4-2

TVA DWG. NO. 47W831-1 R0

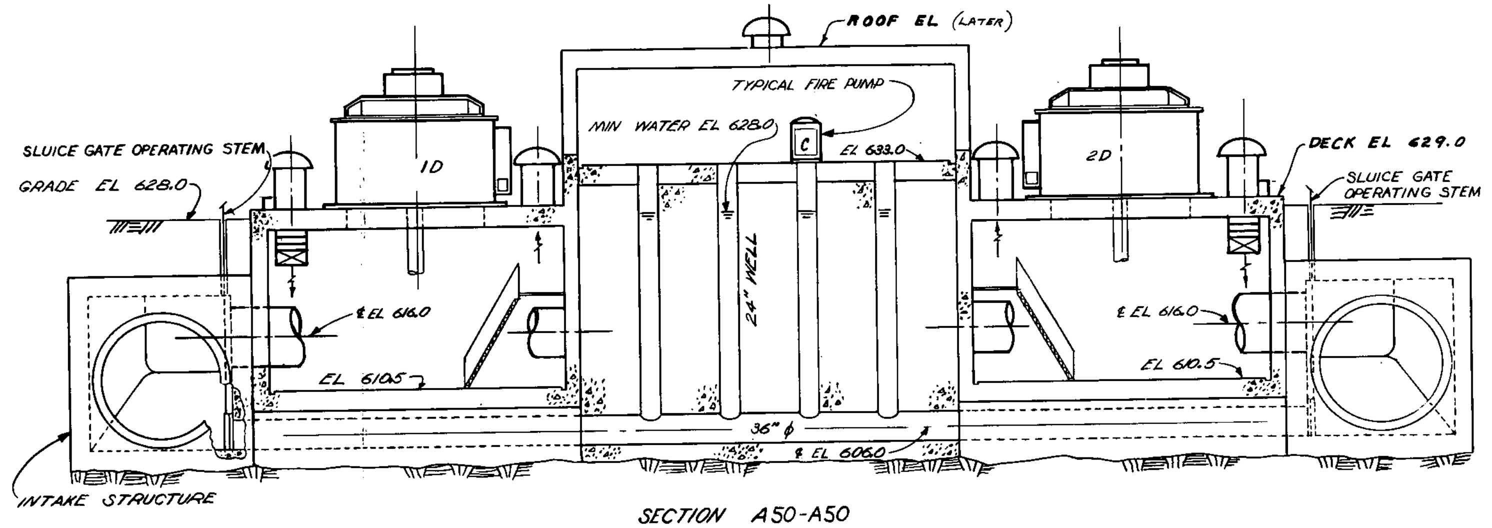


PLAN EL 629.0 & ABOVE

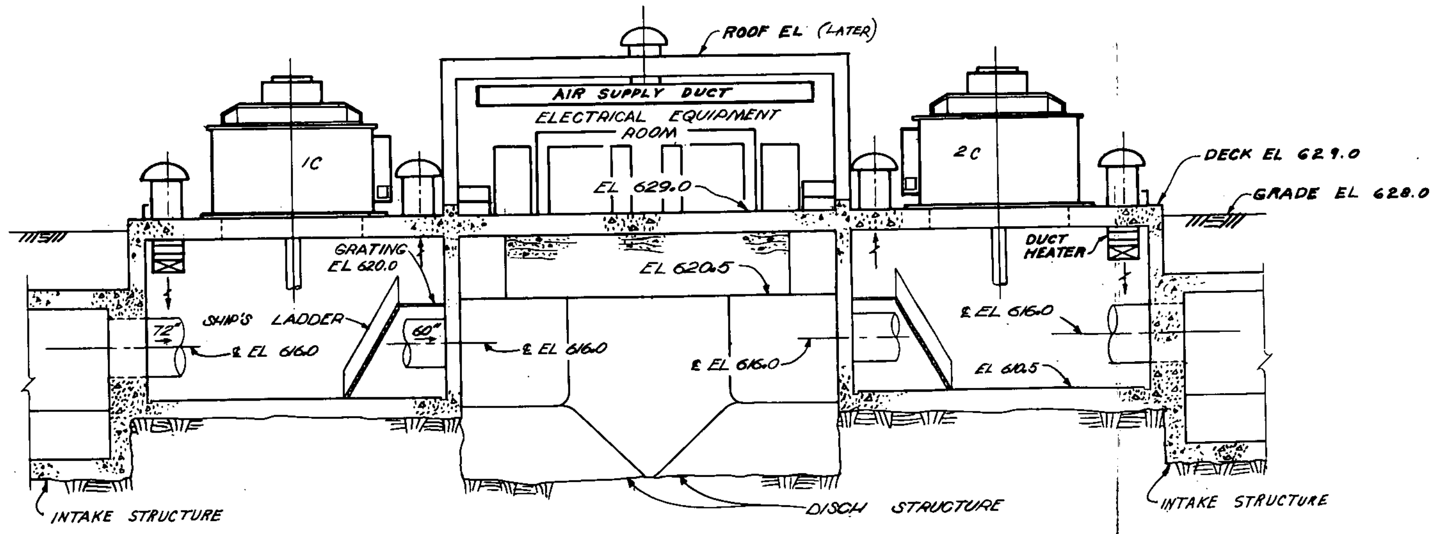




KEY PLAN
SCALE 1" = 300'



SECTION A50-A50



SECTION B50-B50

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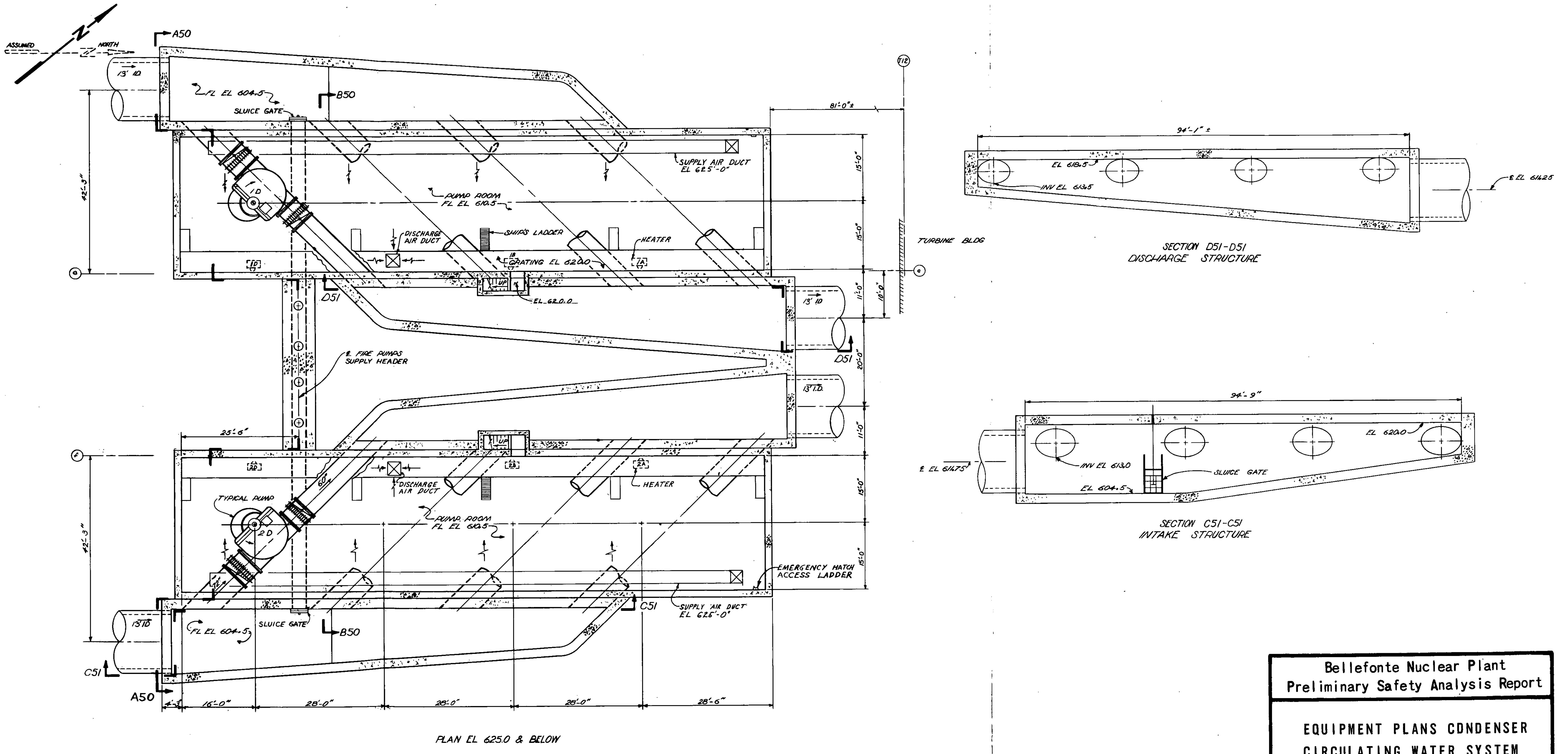
EQUIPMENT PLANS CONDENSER CIRCULATING WATER SYSTEM (SHEET 1)

FIGURE 10.4-3

TVA DWG. NO. 37W200-1 R0



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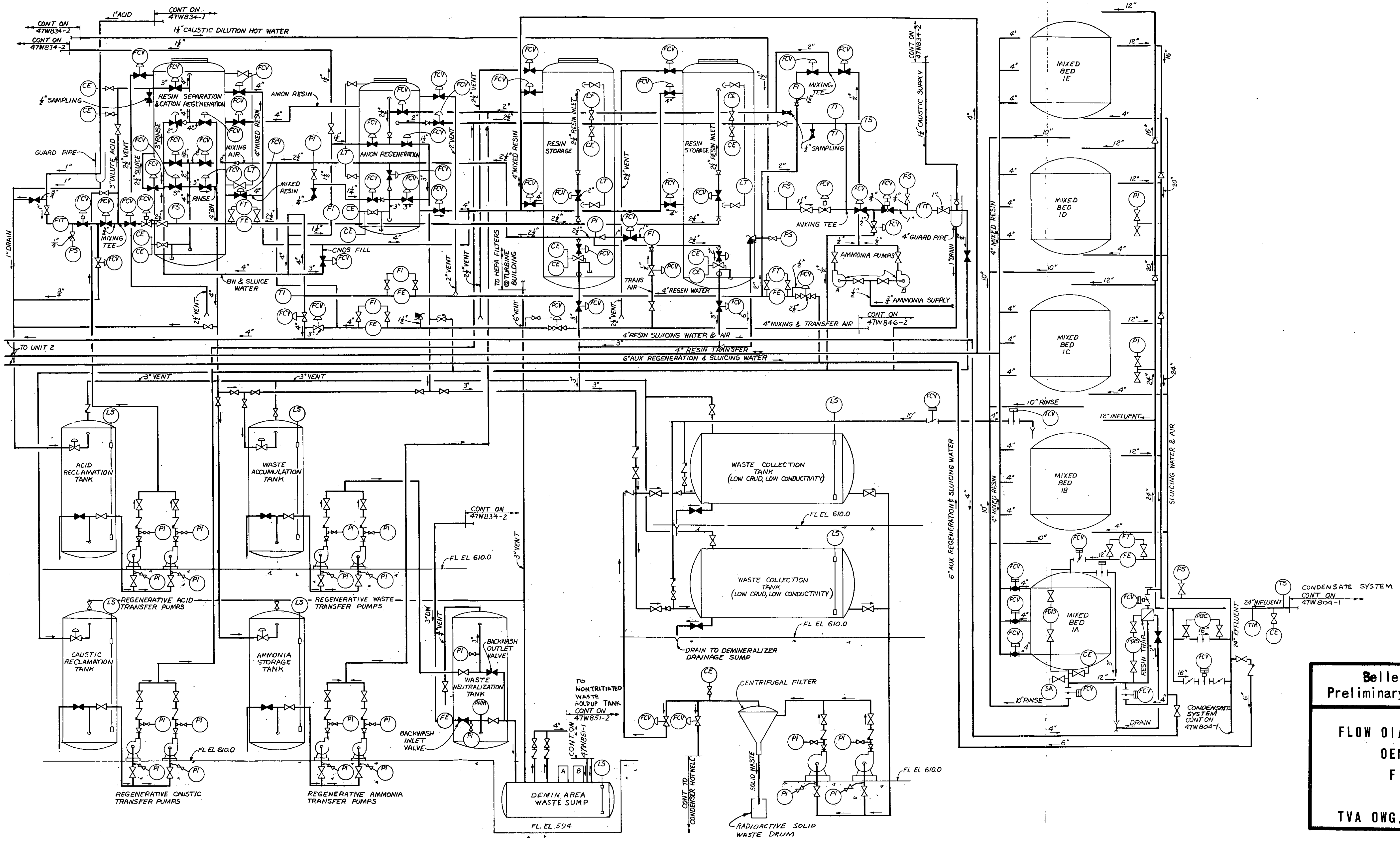
EQUIPMENT PLANS CONDENSER CIRCULATING WATER SYSTEM (SHEET 2)

FIGURE 18.4-4

TVA DWG. NO. 37W208-2 R0



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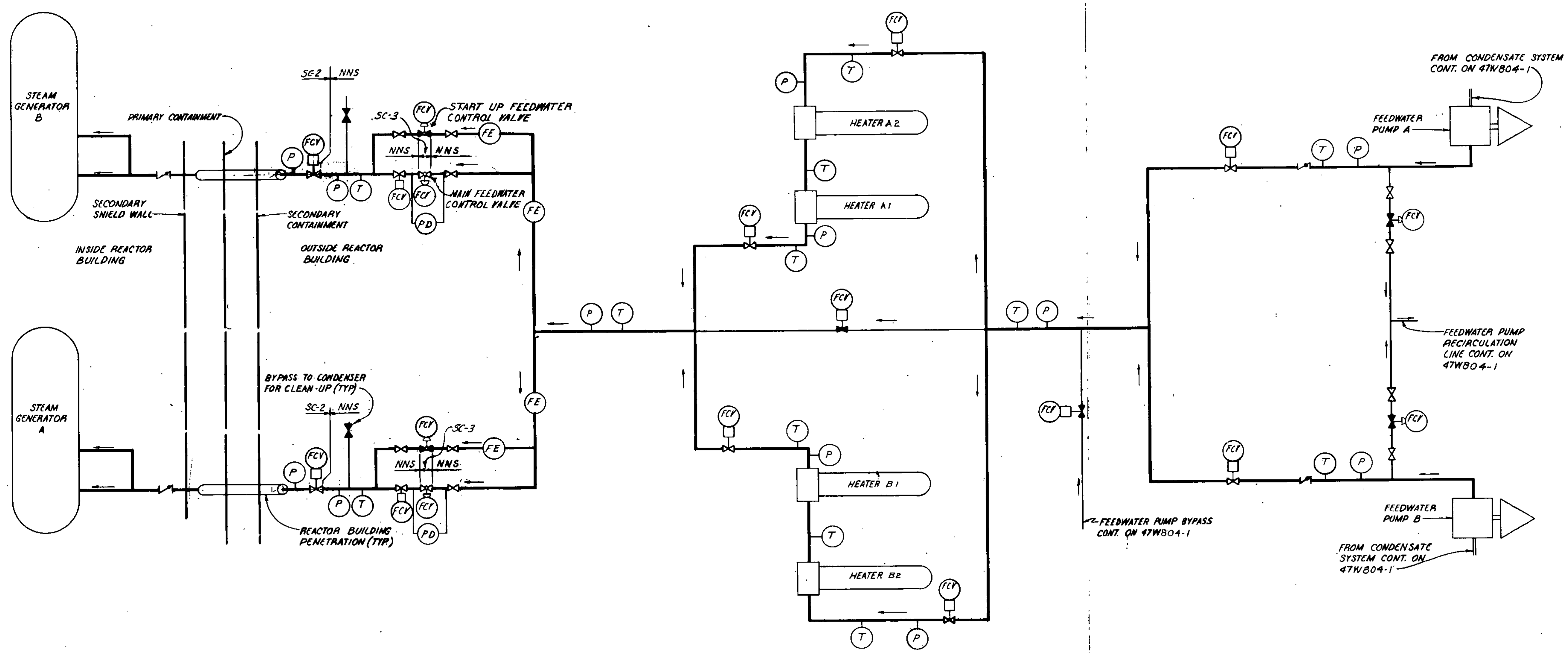
FLOW DIAGRAM CONDENSATE DEMINERALIZERS
FIGURE 10.4-5

TVA DWG. NO. 47W838-1 R0



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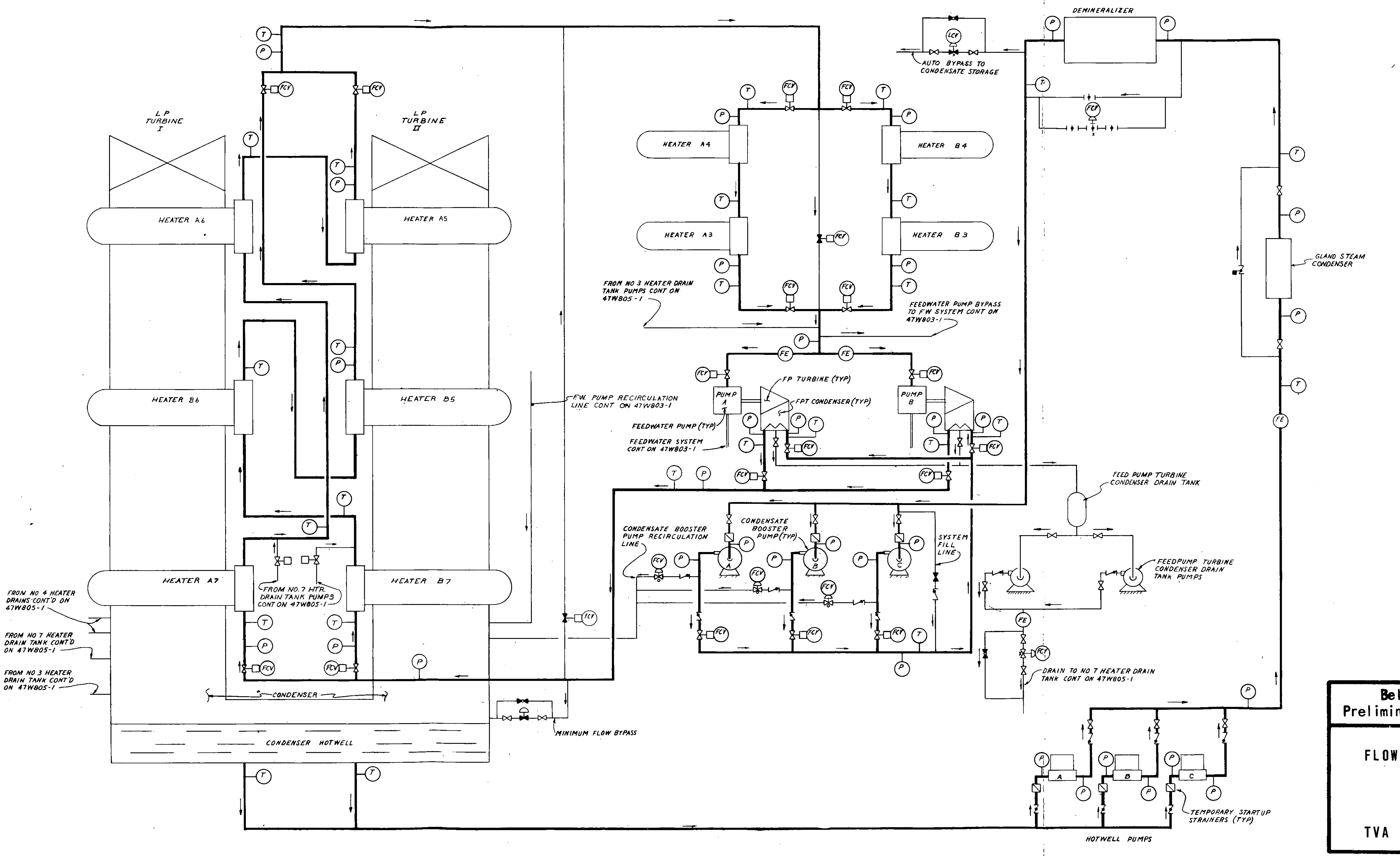
FLOW DIAGRAM FEEDWATER SYSTEM

FIGURE 10.4-8

TVA DWG. NO. 47W884-1 R0



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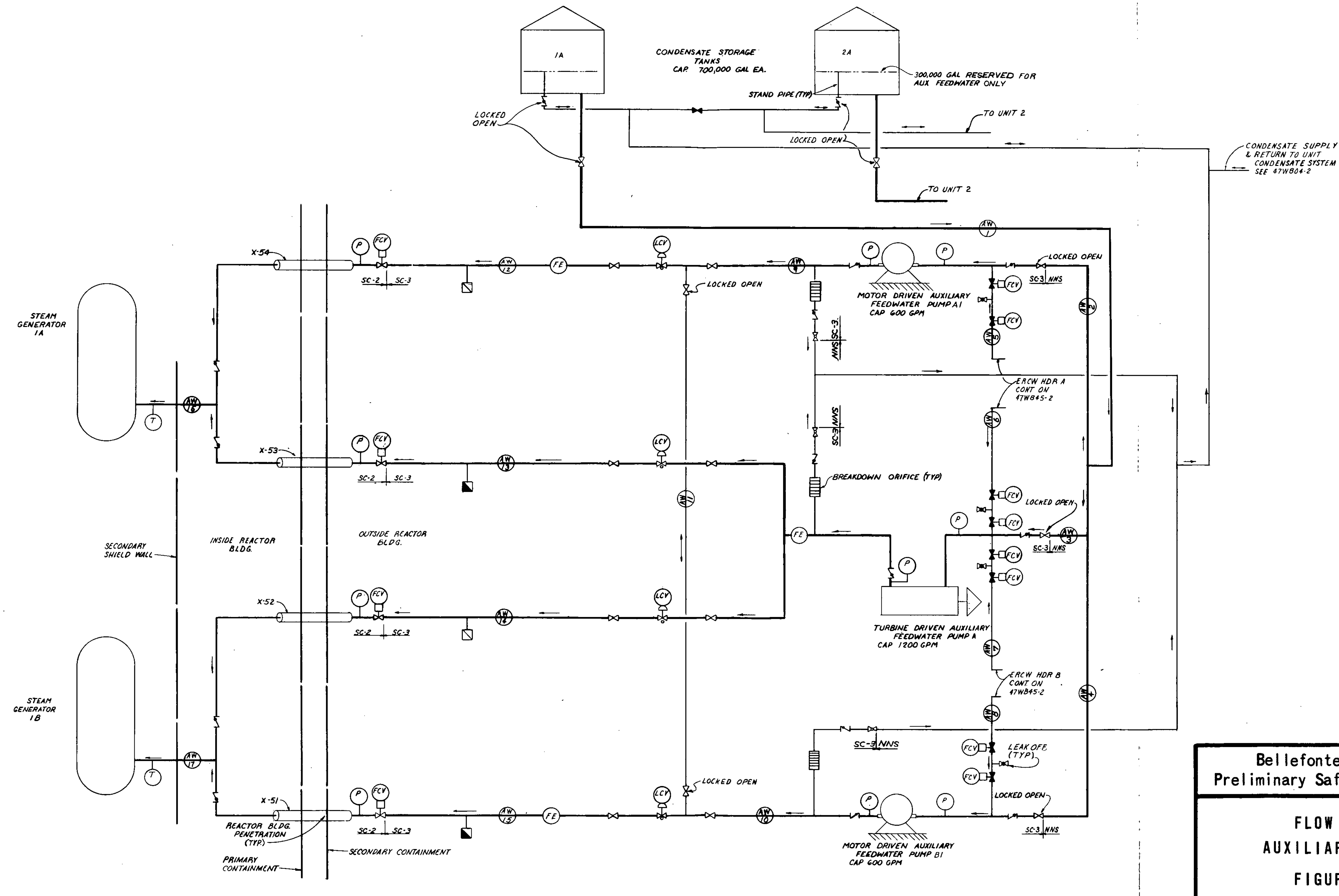
FLOW DIAGRAM CONDENSATE

FIGURE 10.4-7

TVA DWG. NO. 47W884-1 R0



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FLOW DIAGRAM
AUXILIARY FEEOWATER
FIGURE 10.4-8

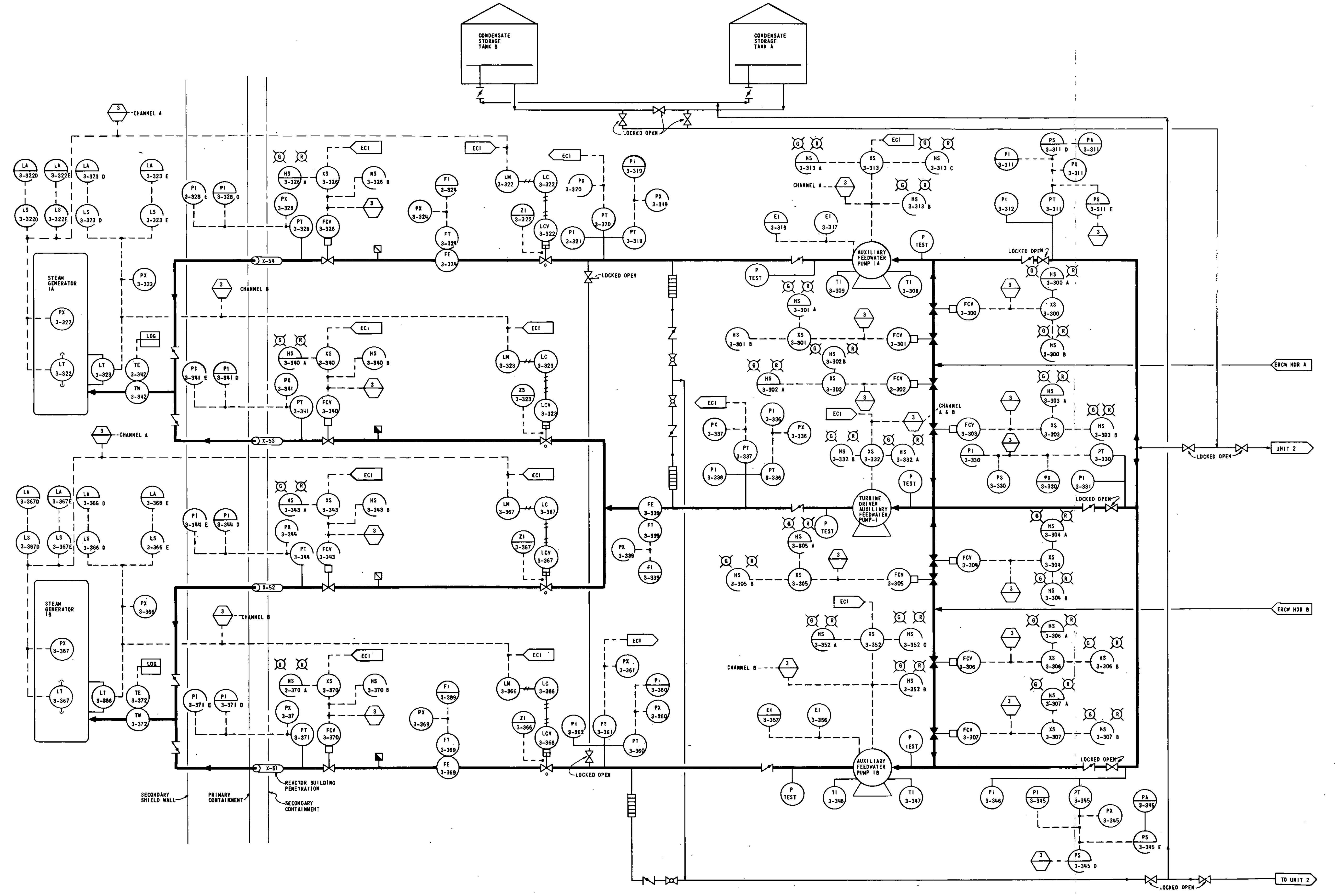
TVA OWG. NO. 47W803-2 RO

INSTRUMENT NUMBERS USED THIS DWG:

3-300	3-337
3-301	3-338
3-302	3-339
3-303	3-340
3-304	3-341
3-305	3-342
3-306	3-343
3-307	3-344
3-311	3-345
3-312	3-346
3-313	3-347
3-317	3-348
3-318	3-352
3-319	3-356
3-320	3-357
3-321	3-360
3-323	3-361
3-323	3-362
3-323	3-366
3-324	3-366
3-326	3-366
3-328	3-369
3-330	3-370
3-331	3-371
3-332	3-372
3-336	3-322
	3-367

INSTRUMENT NUMBERS OPEN THIS DWG:

3-308	3-349
3-309	3-350
3-310	3-351
3-314	3-353
3-315	3-354
3-316	3-355
	3-358
3-325	3-359
3-327	3-363
3-329	3-364
3-333	3-365
3-334	
3-335	3-368



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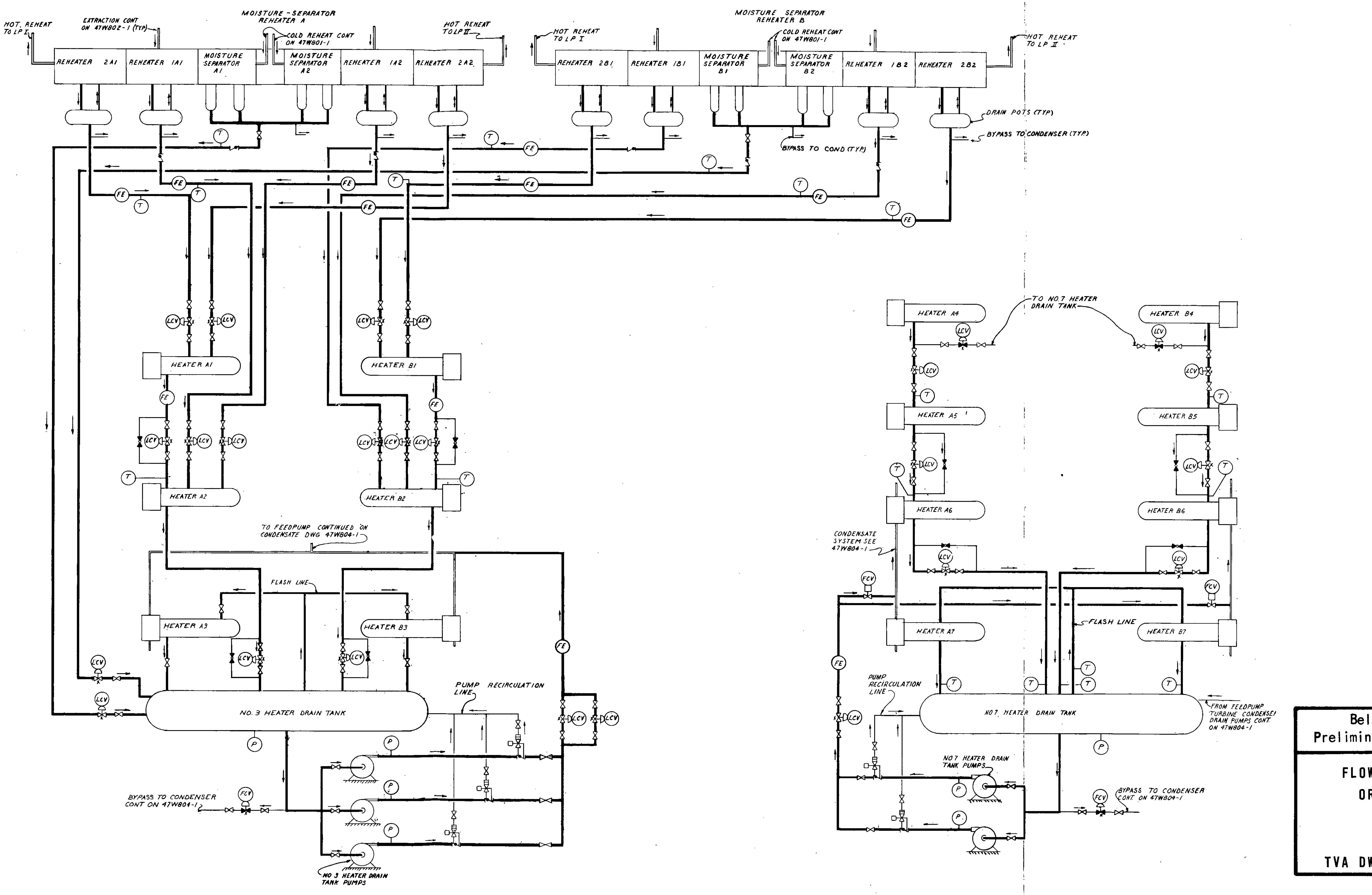
CONTROL DIAGRAM AUXILIARY FEEDWATER SYSTEM

FIGURE 10.4-9

TVA DWG. NO. 47W610-3-3 R0



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FLOW DIAGRAM HEATER, DRAINS, AND VENTS

FIGURE 10.4-1D

TVA DWG. NO. 47W805-1 R0

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11.0. RADIOACTIVE WASTE MANAGEMENT

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LIST OF TABLES (Cont'd)

LIST OF FIGURES

(Placed at end of sections)

FIGURE

11.0. RADIOACTIVE WASTE MANAGEMENT

This chapter provides information about the radioactive source calculations and the systems which affect the amount of radioactive material released from the plant or which mitigate the consequences or monitor the amount of such releases to ensure that the radioactivity levels are in accordance with 10 CFR 20 and the AEC's Numerical Guidelines on as low as practicable in 10 CFR 50. These criteria ensure protection of station personnel and the general public against excessive exposure due to the radiation from the release of radioactive liquids, gases, and solids.

All operations associated with radioactive materials are based on design criteria and procedures which ensure that the operations can be performed in accordance with 10 CFR 20 and the design objectives defined in proposed Appendix I of 10 CFR 50.

11.1. Source Terms

All radioactive wastes originate from either fission products produced in the fuel or activation products produced in the reactor coolant. Although essentially all fission product activity is contained within the fuel rods, minute quantities may enter the reactor coolant system. These fission products, together with the activation products produced in the reactor coolant, are transported throughout much of the plant in the reactor coolant and thus
are the source of radioactivity for other systems. This section describes 11
the methods used to calculate radioactivity levels in the core, reactor coolant system, secondary system, and auxiliary systems, and also discusses the leakage from these systems.

11.1.1. Activity in Core

The fission product activity in the fuel and the fuel rod gap is calculated by a digital computer code that solves the rate equations for fission product buildup in the fuel and leakage from the fuel to the fuel rod gap. The code considers 178 isotopes in 70 decay chains, with a maximum chain length of 5 isotopes. The activity can be calculated for up to 100 times; and for each time, the core power, thermal flux, and the fraction of power produced from any two fissile materials can be changed.

The general rate equation for the inventory of a radioactive nuclide, in the fuel N_f is:

$$\frac{dN_f}{dt} = RY + F\lambda' N'_f - \sigma\phi N_f - \lambda N_f - \alpha N_f$$

where

N_f = inventory of a radioactive nuclide in the fuel, atoms,

R = fission rate, fissions/s,

Y = independent fission yield of N_f,

$\lambda' N'_f$ = activity of N_f precursor in fuel, disintegrations/s,

F = fraction of precursor which decays to N_f,

$\sigma\phi$ = neutron capture rate N_f, s^{-1}

λ = decay constant of N_f, s^{-1},

α = escape rate coefficient of N_f, s^{-1}

The escape rate coefficient, α, represents the fraction of the activity in the fuel that is released, per unit time, from the fuel matrix. Values of these coefficients are derived from experimental data for most elements and are reported in the literature.(1) The experiments, involving purposely defected fuel elements in pressurized water loops, have been performed for a variety of fuel conditions. In selecting the coefficient values to be used in the activity calculations, a comparison of the experimental conditions and the design reactor operating conditions was made. Parameters considered in the comparison include: operating pressure and temperature of the primary coolant loop, type of coolant, size of fuel pellets, density of the fuel, enrichment of the fuel, fuel temperature, thermal output of the fuel (in kW/ft), fuel burnup, total length of the fuel, clad material and clad thickness. The values of the escape rate coefficients for elements of lesser importance were estimated based on the chemical similarity with elements for which the escape rate coefficient was experimentally determined.

The escape rate coefficients used in this chapter are given in Table 11.1-1.

The general rate equation for the inventory of a radioactive nuclide, N_g, in the fuel rod gap is:

$$\frac{dN_g}{dt} = \alpha N_f + F\lambda' N'_g - \lambda N_g$$

where

N_g = inventory of a radioactive nuclide in the fuel rod gap, atoms,

α = escape rate coefficient, s^{-1},

N_f = inventory of same radioactive nuclide in the fuel, atoms,

F = fraction of precursor which decays to N_g

$\lambda' N'_g$ = activity of N_g precursor in fuel rod gap, disintegrations/s

λ = decay constant of N_g, s^{-1}.

The fission product inventory in the fuel and the fuel rod gap was calculated by the code described above using the following assumptions:

1. Core full power operation at 3763 MWt during a 292-EFPD equilibrium cycle. One-third of the fuel has been irradiated for three full power equilibrium cycles, one-third irradiated for two full power equilibrium cycles, and the remaining one-third for one full power equilibrium cycle.

2. Operation with no defective fuel assemblies so the maximum fission product inventory in the core is calculated.

Tables 11.1-2 and 11.1-3 give the resulting fission product inventories in the core and in all the fuel rod gaps.

11.1.2. Activity in Reactor Coolant

11.1.2.1. Fission Product Activity

The fission product activity in the reactor coolant is calculated by a digital computer code that solves the rate equations for fission product buildup in the fuel, leakage from the fuel to the reactor coolant, and removal from the reactor coolant. The code considers 178 isotopes in 70 decay chains, with a maximum chain length of 5 isotopes. The activity can be calculated for up to 100 times, and for each time, the core power, thermal flux, fraction of power produced from any two fissile materials, fraction of defective fuel rods, and purification system parameters can be changed. The calculation assumes that the activity which leaks from the fuel pellets in defective fuel roads immediately reaches the reactor coolant.

The general rate equation for the inventory of a radioactive nuclide, N_c in the reactor coolant is:

$$\frac{dN_c}{dt} = \alpha N_f^* + F\lambda' N_c' - \lambda N_c - \beta N_c - \gamma N_c$$

where

N_c = inventory of a radioactive nuclide in the reactor coolant, atoms,

α = escape rate coefficient, s^{-1},

N_f^* = inventory of same radioactive nuclide in the fuel of defective fuel rods, atoms ($N_f^* = N_f \times$ fraction defective fuel rods),

N_f = inventory of same radioactive nuclide in the fuel, atoms,

F = fraction of precursor which decays to N_c,

$\lambda' N_c'$ = Activity of N_c precursor in reactor coolant disintegrations/s,

λ = decay constant of N_c, s^{-1},

β = removal rate coefficient of N_c by purification, s^{-1},

γ = removal rate coefficient of N_c by plate-out, s^{-1}.

The general rate equation for the inventory of a radioactive nuclide, N_p, in a coolant purification system demineralizer due to removal of N_p from the reactor coolant is

$$\frac{dN_p}{dt} = \beta N_c + F\lambda' N'_p - \lambda N_p - \eta N_p$$

where

N_p = inventory of a radioactive nuclide in a purification system deminerlaizer, atoms,

β = removal rate coefficient of N_c by purification, s^{-1},

N_c = inventory of same radioactive nuclide in the reactor coolant, atoms,

F = fraction of precursor which decays to N_p,

$\lambda' N'_p$ = activity of N_p precursor in a purification system demineralizer, disintegrations/s,

λ = decay constant of N_p, s^{-1},

η = removal rate coefficient of N_p from a purficiation system demineralizer, s^{-1}.

The fission product activity in the reactor coolant was calculated by the code described above using the following assumptions:

1. Core full power operation at 3763 MWt during 292-EFPD equilibrium cycle. One-third of the fuel has been irradiated for three full power equilibrium cycles, one-third irradiated for two full power cycles, and the remaining one-third for one full power equilibrium cycle.

2. Operation occurs with 0.25% failed fuel from the time of intial fuel insertion into the core, and the failed fuel is randomly distributed throughout the entire core.

3. Continuous reactor coolant purification through the purification demineralizer at an average rate of one reactor coolant system volume per day plus that required for load following operation. The removal efficiencies used were zero for the noble gases and tritium and 90% for all other elements except Cs, Y and Mo. Cs, Y and Mo were assumed to be removed in a separate deminerlizer which has a removal efficiency of 90% and which was in service only 20% of the time (a net removal efficiency of 18%). Tellurium decay is the major source of iodine activity, tellurium is known to rapidly plateout on reactor coolant system surfaces, the calculations assume that all the tellurium remains in the reactor coolant system (removal efficiency of zero) and serves as a source of iodine.

4. Bleed rates based on daily load reductions from 100% power to 50% power for the first 254 days of an equilibrium cycle, and base loaded full power operation thereafter.

5. All isotopes, except tritium, are removed with a 99.9% efficiency during the processing of the reactor coolant bleed for the first 254 days of a 292-EFPD equilibrium cycle. After that time, when the deborating demineralizers are used for bleed stream processing, and the removal of all isotopes from the bleed stream is assumed to cease.

6. The plateout coefficient (γ) for all nuclides was assumed to be zero.

Table 11.1-4 shows the reactor coolant activities for each nuclide at various times in the equilibrium cycle and also gives the average activities over the cycle. The maximum activity occurs after 292 days of full power operation.

11.1.2.2. Tritium Activity

Tritium is generated by several mechanisms, but only three are significant; (1) ternary fission in the fuel with subsequent transport to the reactor coolant, (2) neutron activation of boron in the reactor coolant passing through the core, and (3) neutron activation of lithium in the reactor coolant passing through the core.

1. Ternary fission

   Tritium is produced in the fuel by ternary fission. The rate equation for the tritium inventory in the reactor coolant resulting from ternary fission is

$$\frac{dT_1}{dt} = YfPD - (\lambda + k)T_1$$

where

T_1 = inventory of tritium from ternary fission in the reactor coolant system, atoms,

Y = tritium fission yield, 1.1×10^{-4} tritium atoms/fission[2], 11

f = fissions per second per unit thermal power,

P = core thermal power,

D = fraction of tritium escaping from the fuel and entering coolant,

λ = tritium decay constant, s^{-1},

k = loss rate from system, s^{-1}.

The fraction of tritium escaping from the fuel (D) includes both the fraction that diffuses through the Zircaloy cladding and the fraction that leaks from defective fuel rods. Based on work done at ORNL[3], 1% of the tritium produced in the fuel rods is assumed to enter the reactor coolant.

In order to calculate a maximum reactor coolant activity, the loss rate from the system (k) is assumed to be zero.

2. Boron Activation

Tritium is produced from the neutron activation of natural boron ($19.6\%\ ^{10}B$) dissolved in the reactor coolant primarily by the reaction $^{10}B\ (n, 2\alpha)\ ^{3}H$. The rate equation for the tritium inventory in the reactor coolant resulting from boron activation is

$$\frac{dT_2}{dt} = V_c \sigma_1 \phi_1 N_B - (\lambda + k) T_2$$

where

T_2 = inventory of tritium from boron activation, atoms,

V_c = water volume in the active core region, cc,

σ_1 = average reaction cross section for the $^{10}B\ (n, 2\alpha)\ ^{3}H$ reaction, cm^2,

ϕ_1 = total fast neutron flux, $n/cm^2/s$,

N_B = boron concentration as a function of time ($N_B = N_{B_o} - R_B t$), atoms/cc,

N_{B_o} = initial boron concentration, atoms/cc,

R_B = boron removal rate, atoms/cc/s,

λ = tritium decay constant, s^{-1},

k = loss rate from system (assumed to be zero), s^{-1}.

3. Lithium Activation

Tritium is produced from the lithium in the reactor coolant by the reactions:

$$^{6}Li(n,\alpha)^{3}H$$
$$^{7}Li(n,\alpha n)^{3}H$$

Lithium-7 ($99.9\%\ ^{7}Li$, $0.1\%\ ^{6}Li$) is added to the reactor coolant for pH control, and ^{7}Li is also produced by the activation of ^{10}B by the reaction $^{10}B(n,\alpha)^{7}Li$. The lithium concentration is assumed to be a constant 2 ppm since lithium is removed from the reactor coolant system during power operation to maintain the lithium concentration in the range 0.2 - 2.0 ppm. The rate equation for the tritium inventory in the reactor coolant resulting from lithium activation is

$$\frac{dT_3}{dt} = V_c \sigma_2 \phi_2 X_1 N_L + V_c \sigma_3 \phi_3 X_2 N_L - (\lambda + k) T_3$$

where

T_3 = inventory of tritium from lithium activation, atoms,

V_c = water volume in the active core region, cc

σ_2 = reaction cross-section for the $^6Li(n,\alpha)^3H$ reaction, cm^2,

ϕ_2 = thermal neutron flux, $m/cm^2/s$

σ_3 = average reaction cross-section for the $^7Li(n,\alpha n)^3H$ reaction, cm^2,

ϕ_3 = total fast neutron flux, n/cm^2,

N_L = lithium concentration, atoms/cc

X_1 = fraction of N_L which is ^6Li,

X_2 = fraction of N_L which is ^7Li,

λ = tritium decay constant, s^{-1},

k = loss rate from system (assumed to be zero), s^{-1}.

Even though only ^7Li is produced from boron activation, the calculations assume that the lithium in the reactor coolant remains at 99.9% ^7Li (the isotopic composition of the lithium added for pH control).

The calculated amount of tritium produced in/or entering the reactor coolant during an equilibrium cycle, using the above equations, is given in Table 11.1-5. This quantity of tritium becomes uniformly distributed in the reactor coolant system, the bleed holdup system, and the boron recovery system. During refueling operations, the tritium is further diluted when the refueling canal is filled from the borated water storage tank. The reactor coolant system tritium concentration at the end of each cycle and the refueling canal and borated water storage tank tritium concentration are given as a function of the number of cycles in Figure 11.1-1. The lower series of points on Figure 11.1-1 indicates the estimated concentration of tritium in the refueling volume (refueling canal and borated water storage tank). The upper series of points on Figure 11.1-1 indicates the estimated tritium concentration in the reactor coolant.

It is assumed that the tritium concentration in the primary coolant will be limited to 2.5 μCi/cc. When the tritium concentration in the primary coolant approaches 2.5 μCi/cc, a specific quantity of primary coolant is removed for offsite disposal and replaced with nontritiated demineralized water. The estimated quantities to be removed are shown on Figure 11.1-1 and will depend upon the actual tritium production in the primary system. The limiting tritium concentration may be changed should operating experience show that a change is warranted.

11.1.2.3. Corrosion Product Activity

Corrosion products in the reactor coolant become activated when passing through the core. The most important corrosion products are ^{51}Cr, ^{54}Mn, ^{55}Fe, ^{59}Fe, ^{58}Co, ^{60}Co, and ^{95}Zr. The corrosion product activity is dependent on many factors. The mass transport process is complex, and calculational methods to accurately predict corrosion product activity have not been successfully correlated with operational data. Without operational data for the specific reactor system, analytical predictions of the corrosion product activity levels are at best rough approximations. Therefore, in the absence of an acceptable calculation method, the best alternative is to extrapolate data from an operating reactor(4) to the power level of this plant.

The corrosion product activities in the reactor coolant are given in Table 11.1-6.

11.1.2.4. Nitrogen-16 Activity

Nitrogen-16 (^{16}N) is a concern only during reactor operation because of its short half life, 7.1 seconds. ^{16}N is produced by the reaction $^{16}O(n,p)^{16}N$ with an effective neutron energy threshold at 10.7 MeV. The equilibrium ^{16}N activity at any point in the reactor coolant system is calculated using a digital computer code which uses a calculational model that divides the core and reactor coolant system into various regions. The code input parameters for each region include the physical dimensions of the region, reactor coolant velocities and densities in each region, the 10.7 MeV neutron flux at the region inlet, and the appropriate radial, axial, and energy slopes in each core region. The 10.7 MeV neutron flux is approximated by the average value of the neutron flux over the energy range 10-12 MeV. 11

The code calculates the ^{16}N production in each core region by solving the rate equation

$$\frac{dN_j}{dt} = O_j \ (\phi\sigma)_j \ e^{(a_j H_j t/T_j)} - \lambda N_j$$

where

N_j = the concentration of ^{16}N produced in the j^{th} core region at time t, atoms/cc,

t = the continuous variable time, defining the instantaneous axial position of an element of reactor coolant in the j^{th} core region, s,

O_j = the concentration of ^{16}O in the j^{th} core region, atoms/cc,

$(\phi\sigma)_j$ = the mean ^{16}N activation rate per unit target atom at the inlet of the j^{th} core region (i.e., the mean value of the product of the ^{16}N activation cross-section and the average radial fast neutron flux at the region inlet integrated over the energy range 10.7 MeV to 20 MeV), atoms/s-atom,

a_j = the axial slope of the fast neutron flux above 10.7 MeV in the j^{th} core region, cm^{-1},

H_j = length of the j^{th} core region, cm,

T_j = the reactor coolant's total transit time from inlet to outlet of the j^{th} core region, s,

λ = ^{16}N decay constant, s^{-1}.

The j^{th} core region's contribution to the total specific activity of ^{16}N at specified points in the reactor coolant system is computed by calculating the specific activity of ^{16}N at the outlet of the j^{th} core region and applying corrections for radioactive decay and density variations encountered during transit to the specified point. The contribution from the j^{th} core region considers the fact that the reactor coolant is recirculated at a rapid rate and includes the activity contribution from the first and all subsequent passes. The total specific activity of ^{16}N at specified points is then computed by summing the activity contribution from each core region. 3

The ^{16}N activity at various points in the reactor coolant system is given in Table 11.1-7.

11.1.2.5. Argon-41 Activity

Argon-41 is produced by argon-40 activation. Argon-40 present in the argon which may be in the nitrogen cover gas of the distillate storage tank enters the reactor coolant dissolved in the makeup water. The equilibrium ^{41}Ar activity in the reactor coolant is calculated by assuming a constant production rate from Ar, and assuming decay is the only loss. The ^{40}Ar concentration in the reactor coolant is assumed to be constant at a value corresponding to the makeup water being saturated with atmospheric nitrogen (which is 1.185 volume percent ^{40}Ar).(5)

The equilibrium reactor coolant ^{41}Ar activity is 0.14 μCi/g, when the reactor is operating at full power.

11.1.3. Activity in the Secondary System

The Bellefonte Nuclear Plant incorporates once-through steam generators and uses condensate demineralizers to maintain the required feedwater quality. Concentration of radioactive materials in the secondary side of the steam generators as a result of a primary to secondary steam generator tube leak is minimized by this design. The condenser air ejector effectively removes gases from the secondary coolant and the condensate demineralizers remove most of the activity from the secondary feedwater. As a result, the concentration of activity in the secondary coolant is a small fraction of the concentration in the primary coolant. Table 11.1-8 gives the concentration of activity in the steam, the composite feedwater, and the portion of the feedwater that is processed through the condensate demineralizers. These activities are based on the following assumptions:

1. Average primary coolant activities for 0.25% failed fuel (see Table 11.1-14).

2. A primary to secondary steam generator leak rate of 110 pounds per day per unit.

3. A steam flow rate of 7.4×10^6 pounds per hour per steam generator.

4. 54% of the feedwater flow passes through the condensate demineralizers with a decontamination factor of 10 for each isotope.

5. All noble gases are stripped from the secondary coolant in the main condenser and exhausted by the condenser off-gas system.

6. The decay of isotopes in the secondary system is considered to be neglibible since the residence time is very short. This is a conservative assumption.

11.1.4. Activity in Auxiliary Systems

The auxiliary systems will handle fluids which will contain radioactivity. The initial source of radioactivity for these systems is the reactor coolant. The specific activity at various locations in the auxiliary system is calculated by starting with reactor coolant activity and considering the purification processes through which the fluid streams must pass in the various auxiliary systems. The main pieces of equipment that effect the specific activity of the fluid streams are the demineralizers, filters, and evaporators in the auxiliary systems.

Table 11.1-9 summarizes the activity sources in auxiliary systems. Decay was neglected in calculating these sources, and thus the activities of short half-life isotopes will actually be much lower than the values shown.

11.1.4.1. Liquid Activities

1. Reactor coolant system

   Since the reactor coolant activity forms the basis for all activities shown in Table 11.1-9, the activities in column A of the table are the maximum isotopic activities in the reactor coolant, taken from Table 11.1-4.

2. Reactor coolant bleed holdup tank

Reactor coolant will pass through the makeup and purification system demineralizer prior to reaching the bleed holdup tank. The reactor coolant bleed holdup tank specific activities, shown in column B of Table 11.1-9, are calculated by assuming makeup and purification demineralizer isotopic removal efficiencies as specified in section 11.1.2.1.

3. Concentrated boric acid storage tank

Liquid from the reactor coolant bleed holdup tank pases through the bleed evaporator demineralizer and the bleed evaporator prior to reaching the concentrated boric acid storage tank. The specific activities at the outlet of the bleed evaporator demineralizer, shown in column C of Table 11.1-9, are calculated assuming that the demineralizer provides a decontamination factor of 10 for all isotopes except xenon, kryton, and tritium. The concentrated boric acid storage tank specific activities, shown in column D of Table 11.1-9, are calculated by assuming that the noble gases have been removed by the evaporator and that the evaporator train has a concentration factor of 10^2 for all other isotopes.

4. Evaporator distillate test tank

Liquid from the reactor coolant bleed holdup tank passes through the bleed evaporator demineralizer and the bleed evaporator prior to reaching the evaporator distillate test tank. The specific activities at the outlet of the bleed evaporator demineralizer, shown in column C of Table 11.1-9, are calculated assuming that the demineralizer provides a decontamination factor of 10 for all isotopes except xenon, krypton, and tritium. The evaporator distillate test tank specific activities, shown in column E of Table 11.1-9, are calculated by assuming that the noble gases have been removed by the evaporator and that the evaporator train has a decontamination factor of 10^3 for all other isotopes.

5. Distillate storage tank

Liquid from the evaporator test tank passes through the evaporator distillate demineralizer prior to reaching the distillate demineralizer prior to reaching the distillate storage tank. The distillate storage tank specific activities, shown in column F of Table 11.1-9, are calculated by assuming an additional DF of 10 in the evaporator test tank activities.

6. Makeup tank (liquid phase)

The makeup tank liquid phase has the same specific activity as the reactor coolant bleed holdup tank, column B of Table 11.1-9.

7. Reactor coolant drain tank

The reactor coolant drain tank specific activity is assumed to be the same as the maximum reactor coolant specific activity for each nuclide listed in column A of Table 11.1-9.

8. Tritiated waste holdup tank

The tritiated waste holdup tank specific activity is assumed to be the same as the maximum reactor coolant specific activity for each nuclide listed in column A of Table 11.1-9.

9. Nontritiated waste holdup tank

The nontritiated waste holdup tank specific activity is arbitrarily assumed to be one-tenth of the maximum reactor coolant specific activity for each nuclide in column A of Table 11.1-9.

10. Waste evaporator feed tank

The waste evaporator feed tank specific activity is assumed to be the same as the tritiated waste holdup tank specific activity shown in column A of Table 11.1-9.

11. Waste evaporator concentrate

Liquid from the waste evaporator feed tank passes through the waste evaporator prior to being sent to the concentrated boric acid storage tank or the waste packaging station. The waste evaporator concentrate specific activities, shown in column G of Table 11.1-9, are calculated by assuming that the noble gases have been removed by the evaporator and that the evaporator train has a concentration factor of 10^2 for all other isotopes.

12. Waste evaporator distillate test tank

Liquid from the waste evaporator feed tank passes through the waste evaporator prior to being sent to the waste evaporator distillate test tank. The waste evaporator distillate activities, shown in column H of Table 11.1-9, are calculated by assuming that the noble gases have been removed by the evaporator and that the evaporator train has a decontamination factor of 10^3 for all other isotopes.

13. Borated water storage tank

During refueling, the refueling canal is flooded with water from the borated water storage tank. The specific activity in the borated water storage tank, column I of Table 11.1-9, is calculated by mixing the borated water storage tank with the 3100 ft^3 of reactor coolant which remains in the reactor vessel after the reactor coolant system is drained. The tritium concentration in the borated water storage tank is discussed in section 11.1.2.2.

11.1.4.2. Gaseous Activities

1. Makeup tank (gas phase)

   The gaseous activities in the makeup tank are calculated by assuming the gas phase is an ideal gas obeying Henry's law, and that the gas phase is in equilibrium with the liquid phase. The calculated makeup tank gas phase specific activities shown in column J of Table 11.1-9 are based on the liquid activities in column B.

2. Waste gas decay tank

   The maximum activity inventory in the waste gas decay tank occurs following reactor coolant degassification prior to refueling. The waste gas decay tank specific activities shown in column K of Table 11.1-9 assumes that one full tank contains all the xenon and krypton and 0.01% of the iodine in one reactor coolant system volume at the end of the equilibrium cycle.

11.1.4.3. Solid Activities

1. Makeup and purification demineralizer

   The specific activity in the makeup and purification demineralizer, shown in column L of Table 11.1-9, assumes that one 50-ft^3 demineralizer bed is used for reactor coolant purification during one entire core cycle. The demineralizer bed is assumed to have the following equivalent removal efficiencies: zero for noble gases and tritium; 18% for Cs, Y, and Mo; and 90% for all other elements.

2. Deborating demineralizer

   The specific activity in the deborating demineralizer, shown in column M of Table 11.1-9, assumes that one demineralizer is used for deboration for the last 10% of one cycle. Demineralizer efficiencies are 90% for iodine isotopes and zero for all other isotopes.

3. Reactor coolant bleed evaporator demineralizer

   The reactor coolant evaporator demineralizer specific activity, shown in column N of Table 11.1-9, assumes that one bleed evaporator demineralizer is used for one entire cycle. Demineralizer removal efficiencies are assumed to be the same as the purification demineralizer (see subsection 11.1.2.1).

4. Spent resin storage tank

   The spent resin storage tank specific activity, shown in column O of Table 11.1-9, is assumed to be the mean activity of the makeup and purification demineralizer, deborating demineralizer, and reactor coolant bleed evaporator demineralizer.

11.1.5. Activity Releases Due to Leakage

11.1.5.1. Estimate of Leakage (all phases within the plant)

In estimating releases from the waste disposal system it is assumed that 20 gpd of primary coolant per reactor unit leaks to the nontritiated waste disposal system and is processed in the auxiliary waste evaporator. In addition it is assumed that 1 gpd of coolant per reactor unit is released without processing, other than filtration, in the form of nontritiated wastes, including laundry and shower wastes, decontamination drains, and floor drains.

In estimating the release from the condensate demineralizer regeneration system, a primary-to-secondary steam generator leak of 20 gpd per reactor unit is assumed. In addition it is assumed that 100 pounds per day per reactor unit of feedwater leaks to the turbine building floor drain system.

It is assumed that 50 pounds per day of hot reactor coolant escapes to the containment. In addition, it is assumed that the radioactivity in 1 gpd, per reactor unit, of reactor coolant plus 19 gpd, per reactor unit, of liquid downstream of the purification demineralizers is released to the auxiliary building atmosphere.

11.1.5.2. Gaseous Releases Due to Leakage

Leakage to the containment is assumed to be released to the atmosphere on the following basis: (1) twelve containment purges per year, (2) operation of the containment auxiliary charcoal adsorber system for 8 hours prior to the purge, (3) a decontamination factor of 10 in the auxiliary charcoal adsorber for all radioisotopes except noble gases, and (4) an iodine decontamination factor of 100 in the charcoal filters in the purge exhaust system.

Leakage to the auxiliary building is released to the atmosphere through HEPA and charcoal adsorbers with a decontamination factor of 100 for iodine.

It is assumed that 9000 pounds of steam per hour per reactor unit goes to the gland seals. All noble gases in the steam are assumed to be released. A decontamination factor of 2000 for iodine across the gland seal condenser is assumed.

For estimating condenser offgas releases it is assumed that all of the noble gases leaked to the secondary system are released as condenser offgas. Iodine releases assume a steam generator internal partition factor of 1, a condenser air ejector partition factor of 2000 and a decontamination factor of 100 across the air ejector after-condenser and charcoal adsorber.

Estimates of releases resulting from feedwater leakage are based on a leakage rate of 100 pounds per day per reactor unit. For the portion of condensate that passes through the condensate demineralizers (54%), a decontamination factor of 10 is assumed. Except for noble gases, a decontamination factor of 10,000 is assumed from liquid to air.

Steam leakage of 100 pounds per day, per reactor unit, is assumed. It is further assumed that 90% of the iodine deposits in the system before reaching the leak points. Table 11.1-10 gives the estimated amounts of radioactivity in gaseous releases. Table 11.1-12 gives estimated sources of radioactivity in gaseous releases. The tritium in gaseous releases is assumed to be less than 300 curies per year for two reactor units. 11

11.1.5.3. Liquid Releases Due to Leakage

For estimating liquid releases a decontamination factor of 1000 is assumed for the auxiliary waste evaporator and it is assumed that the distillate is released. For estimating releases from the condensate demineralizer system it is assumed that the demineralizers remove 90% per pass of the nongaseous radioactivity from the condensate, that the individual demineralizers are regenerated at 30-day intervals, that the spent regenerants are decayed for 3 days, are processed by the auxiliary waste evaporator with a decontamination factor of 100, and the distillate is released. The estimated amounts of radioactivity in liquid releases are given in Table 11.1-11. Table 11.1-12 gives the estimated sources of radioactivity in liquid releases. Tritium releases in plant liquid effluents are assumed to be less than 300 curies per year for two reactor units.

11.1.5.4. Leakage Control

The primary sources of leakage are in connection with pump shaft seals and valve stem packing seals. These sources are reduced to a minimum by proper inspection, maintenance, and design. Abnormal leakage is detected by monitoring drain tanks and sumps.

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Table 11.1-1. Escape Rate Coefficients

Element	Escape Rate Coefficient, s^{-1}
Kr, X_e	6.5×10^{-8}
Br, I, Cs, Rb	1.3×10^{-8}
Mo, Nb, Ru	2.0×10^{-9}
Te, Se, Sb, Sn	1.0×10^{-9}
Sr, Ba	1.0×10^{-11}
Y, La, Ce	1.6×10^{-12}

Table 11.1-2. Total Core Fission Product Activity (In Curies) Versus Time In Equilibrium Fuel Cycle (In Full Power Days)

Isotope	4	10	30	60	90	120	150	180	210	240	254	292
Br-84	2.61(+7)	2.61(+7)	2.61(+7)	2.61(+7)	2.61(+7)	2.61(+7)	2.61(+7)	2.61(+7)	2.61(+7)	2.61(+7)	2.61(+7)	2.61(+7)
Br-85	3.34(+7)	3.34(+7)	3.34(+7)	3.34(+7)	3.34(+7)	3.34(+7)	3.34(+7)	3.34(+7)	3.34(+7)	3.34(+7)	3.34(+7)	3.34(+7)
Kr-85m	3.34(+7)	3.34(+7)	3.34(+7)	3.34(+7)	3.34(+7)	3.34(+7)	3.34(+7)	3.34(+7)	3.34(+7)	3.34(+7)	3.34(+7)	3.34(+7)
Kr-85	3.97(+5)	4.04(+5)	4.29(+5)	4.67(+5)	5.04(+5)	5.40(+5)	5.77(+5)	6.13(+5)	6.49(+5)	6.84(+5)	7.01(+5)	7.45(+5)
Kr-87	6.15(+7)	6.15(+7)	6.15(+7)	6.15(+7)	6.15(+7)	6.15(+7)	6.15(+7)	6.15(+7)	6.15(+7)	6.15(+7)	6.15(+7)	6.15(+7)
Kr-88	9.29(+7)	9.29(+7)	9.29(+7)	9.29(+7)	9.29(+7)	9.29(+7)	9.29(+7)	9.29(+7)	9.29(+7)	9.29(+7)	9.29(+7)	9.29(+7)
Rb-88	9.29(+7)	9.29(+7)	9.29(+7)	9.29(+7)	9.29(+7)	9.29(+7)	9.29(+7)	9.29(+7)	9.29(+7)	9.29(+7)	9.29(+7)	9.29(+7)
Sr-89	6.19(+7)	6.58(+7)	7.68(+7)	8.87(+7)	9.67(+7)	1.02(+8)	1.06(+8)	1.08(+8)	1.11(+8)	1.11(+8)	1.11(+8)	1.12(+8)
Sr-90	3.07(+6)	3.13(+6)	3.32(+6)	3.61(+6)	3.90(+6)	4.20(+6)	4.48(+6)	4.77(+6)	5.06(+6)	5.35(+6)	5.48(+6)	5.85(+6)
Sr-91	1.48(+8)	1.48(+8)	1.48(+8)	1.48(+8)	1.48(+8)	1.48(+8)	1.48(+8)	1.48(+8)	1.48(+8)	1.48(+8)	1.48(+8)	1.48(+8)
Sr-92	1.92(+8)	1.92(+8)	1.92(+8)	1.92(+8)	1.92(+8)	1.92(+8)	1.92(+8)	1.92(+8)	1.92(+8)	1.92(+8)	1.92(+8)	1.92(+8)
Y-90	3.05(+6)	3.09(+6)	3.28(+6)	3.58(+6)	3.87(+6)	4.16(+6)	4.45(+6)	4.74(+6)	5.02(+6)	5.31(+6)	5.45(+6)	5.81(+6)
Y-91	8.1(+7)	8.56(+7)	9.88(+7)	1.14(+8)	1.24(+8)	1.31(+8)	1.36(+8)	1.4(+8)	1.42(+8)	1.44(+8)	1.44(+8)	1.46(+8)
Mo-99	1.21(+8)	1.75(+8)	1.91(+8)	1.91(+8)	1.91(+8)	1.91(+8)	1.91(+8)	1.91(+8)	1.91(+8)	1.91(+8)	1.91(+8)	1.91(+8)
Ru-106	2.24(+7)	2.28(+7)	2.42(+7)	2.64(+7)	2.82(+7)	3.0(+7)	3.17(+7)	3.34(+7)	3.49(+7)	3.63(+3)	3.7(+7)	3.87(+7)
Xe-131m	3.53(+5)	3.78(+5)	5.91(+5)	7.54(+5)	7.92(+5)	7.99(+5)	8.0(+5)	8.0(+5)	8.0(+5)	8.0(+5)	8.0(+5)	8.0(+5)
Xe-133m	2.55(+6)	4.31(+6)	4.68(+6)	4.68(+6)	4.68(+6)	4.68(+6)	4.68(+6)	4.68(+6)	4.68(+6)	4.68(+6)	4.68(+6)	4.68(+6)
Xe-133	7.12(+7)	1.37(+8)	1.9(+8)	1.94(+8)	1.94(+8)	1.94(+8)	1.94(+8)	1.94(+8)	1.94(+8)	1.94(+8)	1.94(+8)	1.94(+8)
Xe-135m	4.66(+7)	4.66(+7)	4.66(+7)	4.66(+7)	4.66(+7)	4.66(+7)	4.66(+7)	4.66(+7)	4.66(+7)	4.66(+7)	4.66(+7)	4.66(+7)
Xe-135	2.08(+7)	2.08(+7)	2.08(+7)	2.08(+7)	2.08(+7)	2.08(+7)	2.08(+7)	2.08(+7)	2.08(+7)	2.08(+7)	2.08(+7)	2.08(+7)
I-129	1.1	1.12	1.18	1.28	1.38	1.49	1.6	1.7	1.81	1.91	1.96	2.10
I-131	4.17(+7)	6.56(+7)	9.5(+7)	1.01(+8)	1.01(+8)	1.01(+8)	1.01(+8)	1.01(+8)	1.01(+8)	1.01(+8)	1.01(+8)	1.01(+8)
I-132	8.56(+7)	1.31(+8)	1.48(+8)	1.48(+8)	1.48(+8)	1.48(+8)	1.48(+8)	1.48(+8)	1.48(+8)	1.48(+8)	1.48(+8)	1.48(+8)
I-133	1.88(+8)	1.96(+8)	1.96(+8)	1.96(+8)	1.96(+8)	1.96(+8)	1.96(+8)	1.96(+8)	1.96(+8)	1.96(+8)	1.96(+8)	1.96(+8)
I-134	2.48(+8)	2.48(+8)	2.48(+8)	2.48(+8)	2.48(+8)	2.48(+8)	2.48(+8)	2.48(+8)	2.48(+8)	2.48(+8)	2.48(+8)	2.48(+8)
I-135	1.9(+8)	1.9(+8)	1.9(+8)	1.9(+8)	1.9(+8)	1.9(+8)	1.9(+8)	1.9(+8)	1.9(+8)	1.9(+8)	1.9(+8)	1.9(+8)
Cs-134	1.85(+6)	1.88(+6)	2.00(+6)	2.18(+6)	2.39(+6)	2.60(+6)	2.83(+6)	3.07(+6)	3.33(+6)	3.60(+6)	3.72(+6)	4.08(+6)
Cs-136	5.62(+5)	7.63(+5)	1.11(+6)	1.26(+6)	1.29(+6)	1.29(+6)	1.29(+6)	1.29(+6)	1.29(+6)	1.29(+6)	1.29(+6)	1.29(+6)
Cs-137	3.82(+6)	3.89(+6)	4.14(+6)	4.5(+6)	4.86(+6)	5.22(+6)	5.58(+6)	5.94(+6)	6.3(+6)	6.65(+6)	6.82(+6)	7.27(+6)
Cs-138	1.96(+8)	1.96(+8)	1.96(+8)	1.96(+8)	1.96(+8)	1.96(+8)	1.96(+8)	1.96(+8)	1.96(+8)	1.96(+8)	1.96(+8)	1.96(+8)
Ba-137m	3.51(+6)	3.58(+6)	3.80(+6)	4.14(+6)	4.47(+6)	4.8(+6)	5.13(+6)	5.46(+6)	5.79(+6)	6.12(+6)	6.27(+6)	6.69(+6)
Ba-139	1.91(+8)	1.91(+8)	1.91(+8)	1.91(+8)	1.91(+8)	1.91(+8)	1.91(+8)	1.91(+8)	1.91(+8)	1.91(+8)	1.91(+8)	1.91(+8)
Ba-140	8.46(+7)	1.15(+8)	1.68(+8)	1.9(+8)	1.94(+8)	1.95(+8)	1.95(+8)	1.95(+8)	1.95(+8)	1.95(+8)	1.95(+8)	1.95(+8)
La-140	7.36(+7)	1.04(+8)	1.64(+8)	1.89(+8)	1.94(+8)	1.95(+8)	1.95(+8)	1.95(+8)	1.95(+8)	1.95(+8)	1.95(+8)	1.95(+8)
Ce-144	7.29(+7)	7.44(+7)	7.92(+7)	8.59(+7)	9.21(+7)	9.8(+7)	1.03(+8)	1.08(+8)	1.13(+8)	1.18(+8)	1.19(+8)	1.24(+8)

Table 11.1-3. Total Core Fuel Rod Gap Activity (In Curies) Versus Time In Equilibrium Cycle (In Full Power Days)

Isotope	4	10	30	60	90	120	150	180	210	240	254	292
Br-84	8.86(+2)	8.86(+2)	8.86(+2)	8.86(+2)	8.86(+2)	8.86(+2)	8.86(+2)	8.86(+2)	8.86(+2)	8.86(+2)	8.86(+2)	8.86(+2)
Br-85	1.13(+2)	1.13(+2)	1.13(+2)	1.13(+2)	1.13(+2)	1.13(+2)	1.13(+2)	1.13(+2)	1.13(+2)	1.13(+2)	1.13(+2)	1.13(+2)
Kr-85m	4.93(+4)	4.93(+4)	4.93(+4)	4.93(+4)	4.93(+4)	4.93(+4)	4.93(+4)	4.93(+4)	4.93(+4)	4.93(+4)	4.93(+4)	4.93(+4)
Kr-85	2.65(+5)	2.7(+5)	2.84(+5)	3.1(+5)	3.36(+5)	3.65(+5)	3.94(+5)	4.25(+5)	4.56(+5)	4.88(+5)	5.03(+5)	5.45(+5)
Kr-87	2.7(+4)	2.7(+4)	2.7(+4)	2.7(+4)	2.7(+4)	2.7(+4)	2.7(+4)	2.7(+4)	2.7(+4)	2.7(+4)	2.7(+4)	2.7(+4)
Kr-88	8.69(+4)	8.69(+4)	8.69(+4)	8.69(+4)	8.69(+4)	8.69(+4)	8.69(+4)	8.69(+4)	8.69(+4)	8.69(+4)	8.69(+4)	8.69(+4)
Rb-88	8.87(+4)	8.87(+4)	8.87(+4)	8.87(+4)	8.87(+4)	8.87(+4)	8.87(+4)	8.87(+4)	8.87(+4)	8.87(+4)	8.87(+4)	8.87(+4)
Sr-89	5.87(+3)	6.04(+3)	6.63(+3)	7.52(+3)	8.32(+3)	9.01(+3)	9.57(+3)	1.0(+4)	1.04(+4)	1.06(+4)	1.07(+4)	1.09(+4)
Sr-90	7.15(+2)	7.31(+2)	7.87(+2)	8.77(+2)	9.75(+2)	1.08(+3)	1.19(+3)	1.31(+3)	1.44(+3)	1.57(+3)	1.63(+3)	1.82(+3)
Sr-91	1.47(+3)	1.47(+3)	1.47(+3)	1.47(+3)	1.47(+3)	1.47(+3)	1.47(+3)	1.47(+3)	1.47(+3)	1.47(+3)	1.47(+3)	1.47(+3)
Sr-92	3.26(+2)	3.26(+2)	3.26(+2)	3.26(+2)	3.26(+2)	3.26(+2)	3.26(+2)	3.26(+2)	3.26(+2)	3.26(+2)	3.26(+2)	3.26(+2)
Y-90	7.1(+2)	7.23(+2)	7.78(+2)	8.67(+2)	9.64(+2)	1.07(+3)	1.18(+3)	1.3(+3)	1.42(+3)	1.56(+3)	1.62(+3)	1.8(+3)
Y-91	1.67(+3)	1.72(+3)	1.9(+3)	2.14(+3)	2.36(+3)	2.54(+3)	2.68(+5)	2.8(+3)	2.89(+3)	2.97(+3)	2.99(+3)	3.05(+3)
Mo-99	3.67(+4)	9.57(+4)	1.35(+5)	1.35(+5)	1.35(+5)	1.35(+5)	1.35(+5)	1.35(+5)	1.35(+5)	1.35(+5)	1.35(+5)	1.35(+5)
Ru-106	7.99(+5)	8.12(+5)	8.59(+5)	9.34(+5)	1.01(+6)	1.1(+6)	1.19(+6)	1.28(+6)	1.37(+6)	1.47(+6)	1.52(+6)	1.64(+6)
Xe-131m	2.67(+4)	2.93(+4)	4.49(+4)	6.8(+4)	7.72(+4)	7.97(+4)	8.03(+4)	8.05(+4)	8.05(+4)	8.05(+4)	8.05(+4)	8.05(+4)
Xe-133m	2.15(+4)	6.87(+4)	9.19(+4)	9.2(+4)	9.2(+4)	9.2(+4)	9.2(+4)	9.2(+4)	9.2(+4)	9.2(+4)	9.2(+4)	9.2(+4)
Xe-133	1.32(+6)	3.33(+6)	7.54(+6)	8.27(+6)	8.29(+6)	8.29(+6)	8.29(+6)	8.29(+6)	8.29(+6)	8.29(+6)	8.29(+6)	8.29(+6)
Xe-135m	2.98(+4)	2.98(+4)	2.98(+4)	2.98(+4)	2.98(+4)	2.98(+4)	2.98(+4)	2.98(+4)	2.98(+4)	2.98(+4)	2.98(+4)	2.98(+4)
Xe-135	1.54(+5)	1.55(+5)	1.55(+5)	1.55(+5)	1.55(+5)	1.55(+5)	1.55(+5)	1.55(+5)	1.55(+5)	1.55(+5)	1.55(+5)	1.55(+5)
I-129	2.64(-1)	2.7(-1)	2.89(-1)	3.21(-1)	3.55(-1)	3.91(-1)	4.3(-1)	4.7(-1)	5.13(-1)	5.58(-1)	5.8(-1)	6.41(-1)
I-131	2.91(+5)	4.65(+5)	1.01(+6)	1.26(+6)	1.29(+6)	1.29(+6)	1.29(+6)	1.29(+6)	1.29(+6)	1.29(+6)	1.29(+6)	1.29(+6)
I-132	2.7(+4)	5.9(+4)	8.25(+4)	8.32(+4)	8.32(+4)	8.32(+4)	8.32(+4)	8.32(+4)	8.32(+4)	8.32(+4)	8.32(+4)	8.32(+4)
I-133	2.25(+5)	2.71(+5)	2.71(+5)	2.71(+5)	2.71(+5)	2.71(+5)	2.71(+5)	2.71(+5)	2.71(+5)	2.71(+5)	2.71(+5)	2.71(+5)
I-134	1.55(+4)	1.55(+4)	1.55(+4)	1.55(+4)	1.55(+4)	1.55(+4)	1.55(+4)	1.55(+4)	1.55(+4)	1.55(+4)	1.55(+4)	1.55(+4)
I-135	8.58(+4)	8.58(+4)	8.58(+4)	8.58(+4)	8.58(+4)	8.58(+4)	8.58(+4)	8.58(+4)	8.58(+4)	8.58(+4)	8.58(+4)	8.58(+4)
Cs-134	2.42(+5)	2.49(+5)	2.72(+5)	3.09(+5)	3.50(+5)	3.92(+5)	4.37(+5)	4.85(+5)	5.36(+5)	5.89(+5)	6.15(+5)	6.89(+5)
Cs-136	8.29(+3)	9.83(+3)	1.69(+4)	2.35(+4)	2.57(+4)	2.63(+4)	2.64(+4)	2.65(+4)	2.65(+4)	2.65(+4)	2.65(+4)	2.65(+4)
Cs-137	9.27(+5)	9.46(+5)	1.01(+6)	1.12(+6)	1.24(+6)	1.36(+6)	1.49(+6)	1.63(+6)	1.77(+6)	1.93(+6)	2.0(+6)	2.21(+6)
Cs-138	2.37(+4)	2.37(+4)	2.37(+4)	2.37(+4)	2.37(+4)	2.37(+4)	2.37(+4)	2.37(+4)	2.37(+4)	2.37(+4)	2.37(+4)	2.37(+4)
Ba-137m	8.53(+5)	8.71(+5)	9.33(+5)	1.02(+6)	1.14(+6)	1.25(+6)	1.37(+6)	1.5(+6)	1.63(+6)	1.77(+6)	1.84(+6)	2.03(+6)
Ba-139	2.66(+3)	2.66(+3)	2.66(+3)	2.66(+3)	2.66(+3)	2.66(+3)	2.66(+3)	2.66(+3)	2.66(+3)	2.66(+3)	2.66(+3)	2.66(+3)
Ba-140	1.14(+3)	1.38(+3)	2.32(+3)	3.14(+3)	3.41(+3)	3.49(+3)	3.5(+3)	3.51(+3)	3.51(+3)	3.51(+3)	3.51(+3)	3.51(+3)
La-140	1.14(+3)	1.32(+3)	2.27(+3)	3.16(+3)	3.46(+3)	3.55(+3)	3.57(+3)	3.58(+3)	3.58(+3)	3.58(+3)	3.58(+3)	3.58(+3)
Ce-144	2.02(+3)	2.05(+3)	2.16(+3)	2.34(+3)	2.54(+3)	2.74(+3)	2.96(+3)	3.18(+3)	3.4(+3)	3.63(+3)	3.73(+3)	4.02(+3)

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Table 11.1-4. Fission Product Activity in Reactor Coolant Activity, µCi/g
Days Operation in Equilibrium Cycle

Isotope	4	30	60	90	120	150	180	210	240	254	292	Average
Br-84	8.51(-3)	8.51(-3)	8.51(-3)	8.51(-3)	8.51(-3)	8.51(-3)	8.51(-3)	8.48(-3)	8.44(-3)	8.37(-3)	8.62(-3)	0.841(-2)
Br-85	1.12(-3)	1.12(-3)	1.12(-3)	1.12(-3)	1.12(-3)	1.12(-3)	1.12(-3)	1.12(-3)	1.12(-3)	1.12(-3)	1.13(-3)	0.111(-2)
Kr-85m	0.468	0.467	0.465	0.462	0.458	0.453	0.446	0.436	0.414	0.386	0.492	0.444(+0)
Kr-85	2.10(-2)	3.95(-2)	3.95(-2)	3.81(-2)	3.56(-2)	3.23(-2)	2.81(-2)	2.29(-2)	1.62(-2)	1.12(-2)	0.461	0.571(-1)
Kr-87	0.266	0.266	0.266	0.265	0.264	0.264	0.262	0.260	0.256	0.250	0.270	0.260(+0)
Kr-88	0.840	0.839	0.836	0.833	0.828	0.823	0.814	0.802	0.775	0.739	0.868	0.809(+0)
Rb-88	0.839	0.837	0.835	0.832	0.827	0.822	0.813	0.799	0.771	0.732	0.872	0.807(+0)
Sr-89	7.53(-4)	8.52(-4)	9.30(-4)	9.82(-4)	1.02(-3)	1.03(-3)	1.02(-3)	9.68(-4)	8.66(-4)	7.39(-4)	1.60(-3)	0.964(-3)
Sr-90	2.01(-5)	2.18(-5)	2.37(-5)	2.56(-5)	2.75(-5)	2.92(-5)	3.00(-5)	3.00(-5)	2.81(-5)	2.46(-5)	5.66(-5)	0.281(-4)
Sr-91	8.10(-3)	8.10(-3)	8.10(-3)	8.11(-3)	8.11(-3)	8.08(-3)	7.96(-3)	7.74(-3)	7.30(-3)	6.70(-3)	9.52(-3)	0.788(-2)
Sr-92	2.69(-3)	2.69(-3)	2.69(-3)	2.69(-3)	2.69(-3)	2.68(-3)	2.67(-3)	2.64(-3)	2.58(-3)	2.49(-3)	2.85(-3)	0.264(-2)
Y-90	1.23(-5)	1.46(-5)	1.55(-5)	1.63(-5)	1.69(-5)	1.72(-5)	1.67(-5)	1.54(-5)	1.25(-5)	9.30(-6)	5.15(-5)	0.172(-4)
Y-91	3.79(-4)	5.14(-4)	5.42(-4)	5.50(-4)	5.42(-4)	5.18(-4)	4.73(-4)	4.07(-4)	3.09(-4)	2.23(-4)	1.64(-3)	0.528(-3)
Mo-99	0.225	0.486	0.476	0.462	0.446	0.425	0.395	0.353	0.287	0.224	0.779	0.412(+0)
Ru-106	2.76(-2)	3.01(-2)	3.26(-2)	3.50(-2)	3.73(-2)	3.91(-2)	3.97(-2)	3.91(-2)	3.62(-2)	3.15(-2)	7.08(-2)	0.373(-1)
Xe-131m	4.73(-2)	0.109	0.134	0.130	0.118	0.106	9.06(-2)	7.36(-2)	5.26(-2)	3.68(-2)	0.635	0.128(+0)
Xe-133m	0.165	0.519	0.500	0.477	0.450	0.419	0.381	0.331	0.259	0.195	0.885	0.414(+0)
Xe-133	6.10	30.3	29.5	27.5	25.4	23.0	20.2	16.9	12.5	8.94	81.0	0.248(+2)
Xe-135m	0.208	0.208	0.208	0.207	0.207	0.207	0.205	0.201	0.193	0.182	0.230	0.203(+0)
Xe-135	1.12	1.11	1.10	1.09	1.07	1.05	1.02	0.960	0.858	0.737	1.32	0.964(+0)
I-129	7.00(-9)	7.50(-9)	8.08(-9)	8.67(-9)	9.26(-9)	9.76(-9)	9.98(-9)	9.90(-9)	9.22(-9)	8.03(-9)	1.83(-8)	0.936(-8)
I-131	0.297	0.737	0.785	0.789	0.789	0.784	0.760	0.719	0.643	0.552	1.14	0.730(+0)
I-132	0.199	0.452	0.442	0.429	0.414	0.397	0.374	0.345	0.305	0.266	0.736	0.393(+0)
I-133	0.956	1.02	1.02	1.02	1.02	1.02	0.999	0.962	0.890	0.797	1.29	0.979(+0)
I-134	0.145	0.145	0.145	0.145	0.145	0.145	0.145	0.144	0.143	0.141	0.148	0.122(+0)
I-135	0.558	0.558	0.558	0.558	0.558	0.557	0.550	0.538	0.515	0.481	0.630	0.544(+0)
Cs-134	2.62(-2)	3.32(-2)	3.46(-2)	3.57(-2)	3.64(-2)	3.64(-2)	3.51(-2)	3.21(-2)	2.62(-2)	1.99(-2)	0.144	0.392(-1)
Cs-136	9.87(-3)	2.47(-2)	2.74(-2)	2.70(-2)	2.58(-2)	2.42(-2)	2.20(-2)	1.91(-2)	1.48(-2)	1.11(-2)	6.09(-2)	0.238(-1)
Cs-137	6.20(-2)	8.01(-2)	8.38(-2)	8.61(-2)	8.70(-2)	8.58(-2)	8.15(-2)	7.32(-2)	5.86(-2)	4.40(-2)	0.314	0.903(-1)
Cs-138	0.234	0.233	0.233	0.233	0.233	0.232	0.232	0.231	0.228	0.225	0.236	0.217(+0)
Ba-137m	5.69(-2)	7.34(-2)	7.68(-2)	7.89(-2)	7.97(-2)	7.87(-2)	7.47(-2)	6.71(-2)	5.36(-2)	4.03(-2)	0.288	0.252(-1)
Ba-139	2.38(-2)	2.38(-2)	2.38(-2)	2.38(-2)	2.38(-2)	2.37(-2)	2.36(-2)	2.35(-2)	2.31(-2)	2.25(-2)	2.47(-2)	0.234(-1)
Ba-140	6.49(-4)	1.19(-3)	1.33(-3)	1.36(-3)	1.36(-3)	1.36(-3)	1.31(-3)	1.24(-3)	1.11(-3)	9.49(-4)	1.99(-3)	0.126(-2)
La-140	2.02(-4)	4.06(-4)	4.61(-4)	4.73(-4)	4.75(-4)	4.72(-4)	4.49(-4)	4.13(-4)	3.50(-4)	2.80(-4)	8.30(-4)	0.435(-3)
Ce-144	7.46(-5)	8.14(-5)	8.84(-5)	9.49(-5)	1.01(-4)	1.06(-4)	1.07(-4)	1.06(-4)	9.76(-5)	8.46(-5)	1.90(-4)	0.101(-3)

Table 11.1-5. Tritium Production

Tritium source	Curies per equilibrium cycle
Ternary fission	158[a]
Boron Activation	648
Lithium Activation	60
TOTAL	866

[a] 100% of the total ternary fission produced tritium (D = 0.01)

Table 11.1-6. Corrosion Product Activity in Reactor Coolant

Isotope	Activity, μCi/g
^{51}Cr	5.0 (-3)
^{54}Mn	5.6 (-4)
^{55}Fe	1.9 (-2)
^{59}Fe	5.6 (-4)
^{58}Co	2.9 (-2)
^{60}Co	1.6 (-4)
^{95}Zr	3.8 (-2)

Table 11.1-7. Nitrogen-16 Activity in Reactor Coolant

Location	Activity, dps/g
Reactor Vessel Outlet	1.2(+7)
Steam Generator Inlet	1.0(+7)
Steam Generator Outlet	6.5(+6)
Reactor Vessel Inlet	5.6(+6)

Table 11.1-8. Secondary Coolant Activity Equilibrium Activity (μCi/g)

Isotope	Steam	Demineralized Feedwater	Feedwater	
^{84}Br	5.23(-09)	5.23(-10)	2.66(-09)	
^{85}Br	6.00(-10)	6.00(-11)	3.05(-10)	
^{85}Kr[a]	1.38(-07)	1.38(-08)	7.02(-08)	
^{85}Kr	1.77(-08)	1.77(-09)	9.00(-09)	
^{87}Kr	8.83(-08)	8.83(-09)	4.49(-08)	
^{88}Kr	2.51(-07)	2.51(-08)	1.28(-07)	
^{88}Rb	5.03(-07)	5.03(-08)	2.56(-07)	
^{89}Sr	6.10(-10)	6.10(-11)	3.10(-10)	
^{90}Sr	1.78(-11)	1.78(-12)	9.05(-12)	
^{91}Sr	4.98(-09)	4.98(-10)	2.53(-09)	
^{92}Sr	1.66(-09)	1.66(-10)	8.44(-10)	11
^{90}Y	1.09(-11)	1.09(-12)	5.54(-12)	
^{91}Y[a]	1.55(-11)	1.55(-12)	7.88(-12)	
^{91}Y	3.34(-10)	3.34(-11)	1.70(-10)	
^{92}Y	4.12(-12)	4.12(-13)	2.10(-12)	
^{99}Mo	2.60(-07)	2.60(-08)	1.32(-07)	
^{99}Tc[a]	3.31(-10)	3.31(-11)	1.68(-10)	
^{99}Tc	1.56(-19)	1.56(-20)	7.93(-20)	
^{106}Ru	2.36(-08)	2.36(-09)	1.20(-08)	
^{106}Rh	1.21(-08)	1.21(-09)	6.15(-09)	
^{129}I	5.92(-15)	5.92(-16)	3.01(-15)	
^{131}I	4.62(-07)	4.62(-08)	2.35(-07)	
^{132}I	2.48(-07)	2.48(-08)	1.26(-07)	
^{133}I	6.19(-07)	6.19(-08)	3.15(-07)	
^{134}I	7.64(-07)	7.64(-08)	3.89(-07)	
^{135}I	3.44(-07)	3.44(-08)	1.75(-07)	
^{131}Xe[a]	3.98(-08)	3.98(-09)	2.02(-08)	
^{133}Xe[a]	1.29(-07)	1.29(-08)	6.56(-08)	
^{133}Xe	7.70(-06)	7.70(-07)	3.92(-06)	
^{135}Xe[a]	6.36(-08)	6.36(-09)	3.23(-08	
^{135}Xe	3.00(-07)	3.00(-08)	1.53(-07)	
^{134}Cs	2.48(-08)	2.48(-09)	1.26(-08)	

Table 11.1-8. (Cont'd)

Isotope	Steam	Demineralized Feedwater	Feedwater
^{136}Cs	1.50(-08)	1.50(-09)	7.63(-09)
^{137}Cs	5.72(-08)	5.72(-09)	2.91(-08)
^{138}Cs	1.35(-07)	1.35(-08)	6.87(-08)
^{137}Ba (a)	2.23(-08)	2.23(-09)	1.13(-08)
^{139}Ba	1.47(-08)	1.47(-09)	7.48(-09)
^{140}Ba	7.97(-10)	7.97(-11)	4.05(-10)
^{140}La	2.75(-10)	2.75(-11)	1.40(-10)
^{144}Ce	6.36(-11)	6.36(-12)	3.23(-11)
^{144}Pr	1.87(-12)	1.87(-13)	9.51(-13)
^{51}Cr	3.16(-09)	3.16(-10)	1.61(-09)
^{54}Mn	3.53(-10)	3.54(-11)	1.80(-10)
^{55}Fe	1.20(-08)	1.20(-09)	6.10(-09)
^{59}Fe	3.54(-10)	3.54(-11)	1.80(-10)
^{58}Co	1.83(-08)	1.83(-09)	9.31(-09)
^{60}Co	1.01(-10)	1.01(-11)	5.14(-11)
^{95}Zr	2.40(-08)	2.40(-09)	1.22(-08)
^{95}Nb (a)	3.04(-14)	3.04(-15)	1.55(-14)
^{95}Nb	2.47(-13)	2.47(-14)	1.26(-13)

(a) Metastable Isotopes

Table 11.1-9. Activity Sources in Auxiliary Systems

Isotope	Liquid Source Activities, μCi/gm									Gaseous Activities, Curies/ft^3 @STP		Demineralizer Resin Activities, Curies/ft^3-cycle			
	A	B	C	D	E	F	G	H	I	J	K	L	M	N	O
Kr-85m	0.492	0.492	0.492	0	0	0	0	0	3.57(-2)	0.153	6.02(-3)	0	0	0	0
85	0.461	0.461	0.461	0	0	0	0	0	3.34(-2)	0.144	5.63(-3)	0	0	0	0
87	0.270	0.270	0.270	0	0	0	0	0	1.96(-2)	0.084	3.30(-3)	0	0	0	0
88	0.868	0.868	0.868	0	0	0	0	0	6.29(-2)	0.270	1.06(-2)	0	0	0	0
Rb-88	0.872	8.72(-2)	8.72(-3)	0.872	8.72(-6)	8.72(-7)	87.2	8.72(-4)	6.32(-2)	0	0	9.68(-2)	0	3.90(-3)	3.42(-2)
Sr-89	1.60(-3)	1.60(-4)	1.60(-5)	1.60(-3)	1.60(-8)	1.60(-7)	0.160	1.60(-6)	1.16(-4)	0	0	0.488	0	1.97(-2)	0.173
90	5.66(-5)	5.66(-6)	5.66(-7)	5.66(-5)	5.66(-10)	5.66(-11)	5.66(-3)	5.66(-8)	4.10(-6)	0	0	5.87(-2)	0	2.37(-3)	2.07(-2)
91	9.52(-3)	9.52(-4)	9.52 (-5)	9.52(-3)	9.52(-8)	9.52(-9)	0.952	9.52(-6)	6.90(-4)	0	0	3.11(-2)	0	1.25(-3)	1.10(-2)
92	2.85(-3)	2.85(-4)	2.85(-5)	2.85(-3)	2.85(-8)	2.85(-9)	0.285	2.85(-6)	2.07(-4)	0	0	2.88(-3)	0	1.16(-4)	1.02(-3)
Y-90	5.15(-5)	4.22(-5)	4.22(-6)	4.22(-4)	4.22(-9)	4.22(-10)	5.15(-3)	5.15(-8)	3.73(-6)	0	0	1.02(-4)	0	3.38(-5)	4.22(-5)
91	1.64(-3)	1.34(-3)	1.34(-4)	1.34(-2)	1.34(-7)	1.34(-8)	0.164	1.64(-6)	1.19(-4)	0	0	0.464	0	0.153	0.192
Mo-99	0.779	0.639	6.39(-2)	6.39	6.39(-5)	6.39(-6)	77.9	7.79(-4)	5.65(-2)	0	0	2.40	0	0.793	0.994
Xe-131m	0.635	0.635	0.635	0	0	0	0	0	4.60(-2)	0.099	7.76(-3)	0	0	0	0
133m	0.885	0.885	0.885	0	0	0	0	0	6.41(-2)	0.138	1.08(-2)	0	0	0	0
133	81.0	81.0	81.0	0	0	0	0	0	5.87	12.6	0.989	0	0	0	0
135m	0.230	0.230	0.230	0	0	0	0	0	1.67(-2)	0.036	2.82(-3)	0	0	0	0
135	1.32	1.32	1.32	0	0	0	0	0	9.57(-2)	0.206	1.61(-2)	0	0	0	0
I-131	1.14	0.114	1.14(-2)	1.14	1.14(-5)	1.14(-6)	1.14(+2)	1.14(-3)	8.26(-2)	0	1.39(-6)	60.3	0.180	2.43	21.3
132	0.736	7.36(-2)	7.36(-3)	0.736	7.36(-6)	7.36(-7)	73.6	7.36(-4)	5.33(-2)	0	8.99(-7)	0.393	1.32(-3)	1.58(-2)	0.139
133	1.29	0.129	1.29(-2)	1.29	1.29(-5)	1.29(-6)	1.29(+2)	1.29(-3)	9.35(-2)	0	1.58(-6)	8.51	2.50(-2)	0.343	3.02
134	0.148	1.48(-2)	1.48(-3)	0.148	1.48(-6)	1.48(-7)	14.8	1.48(-4)	1.07(-2)	0	1.81(-7)	4.99(-2)	1.43(-4)	2.01(-3)	1.77(-2)
135	0.630	6.30(-2)	6.30(-3)	0.630	6.30(-6)	6.30(-7)	63.0	6.30(-4)	4.57(-2)	0	7.70(-7)	1.48	4.26(-3)	5.96(-2)	0.524
Cs-134	0.144	0.118	1.18(-2)	1.18	1.18(-5)	1.18(-6)	14.4	1.44(-4)	1.04(-2)	0	0	0.270	0	8.90(-2)	0.111
136	6.09(-2)	4.99(-2)	4.99(-3)	0.499	4.99(-6)	4.99(-7)	6.09	6.09(-5)	4.41(-3)	0	0	0.677	0	0.224	0.280
137	0.314	0.257	2.57(-2)	2.57	2.57(-5)	2.57(-6)	31.4	3.14(-4)	2.28(-2)	0	0	41.3	0	13.7	17.1
138	0.236	0.194	1.94(-2)	1.94	1.94(-5)	1.94(-6)	23.6	2.36(-4)	1.71(-2)	0	0	9.85(-3)	0	3.27(-3)	4.07(-3)
Ba-139	2.47(-2)	2.47(-3)	2.47(-4)	2.47(-2)	2.47(-7)	2.47(-8)	2.47	2.47(-5)	1.79(-3)	0	0	1.30(-2)	0	5.24(-4)	4.61(-3)
140	1.99(-3)	1.99(-4)	1.99(-5)	1.99(-3)	1.99(-8)	1.99(-9)	0.199	1.99(-6)	1.44(-4)	0	0	0.165	0	6.64(-3)	5.82(-2)
La-140	8.30(-4)	8.30(-5)	8.30(-6)	8.30(-4)	8.30(-9)	8.30(-10)	8.30(-2)	8.30(-7)	6.02(-5)	0	0	7.64(-3)	0	3.08(-4)	2.70(-3)
Ce-144	1.90(-4)	1.90(-5)	1.90(-6)	1.90(-4)	1.90(-9)	1.90(-10)	1.90(-2)	1.90(-7)	1.38(-5)	0	0	0.150	0	6.07(-3)	5.30(-2)

Table 11.1-10. Estimated Amounts of Radioactivity In Gaseous Releases

Sources	Ci/yr Iodine-131	Ci/yr Noble Gases[a]
Containment purge	0.36×10^{-5}	0.30×10^{2}
Instrument room purge	0.38×10^{-6}	0.71×10^{-1}
Purification and makeup system gases vented to ABVS	0.88×10^{-3}	0.13×10^{4}
Waste gas decay tank venting	0.42×10^{-3}	0.17×10^{4}
Steam leakages	0.14×10^{-5}	0.45×10^{-3}
Turbine gland sealing system leakage	0.12×10^{-4}	0.84×10^{0}
Condenser offgases	0.12×10^{-4}	0.84×10^{3}
Feedwater Leakage	0.20×10^{-8}	0.13×10^{-8}

[a] Noble gases include: $^{85}Kr^{m}$, ^{85}Kr, ^{87}Kr, ^{88}Kr, $^{131}Xe^{m}$, $^{133}Xe^{m}$, ^{133}Xe, $^{135}Xe^{m}$, ^{135}Xe

Table 11.1-11. Estimated Amounts of Radioactivity in Liquid Releases For Two Reactor Units [(a)]

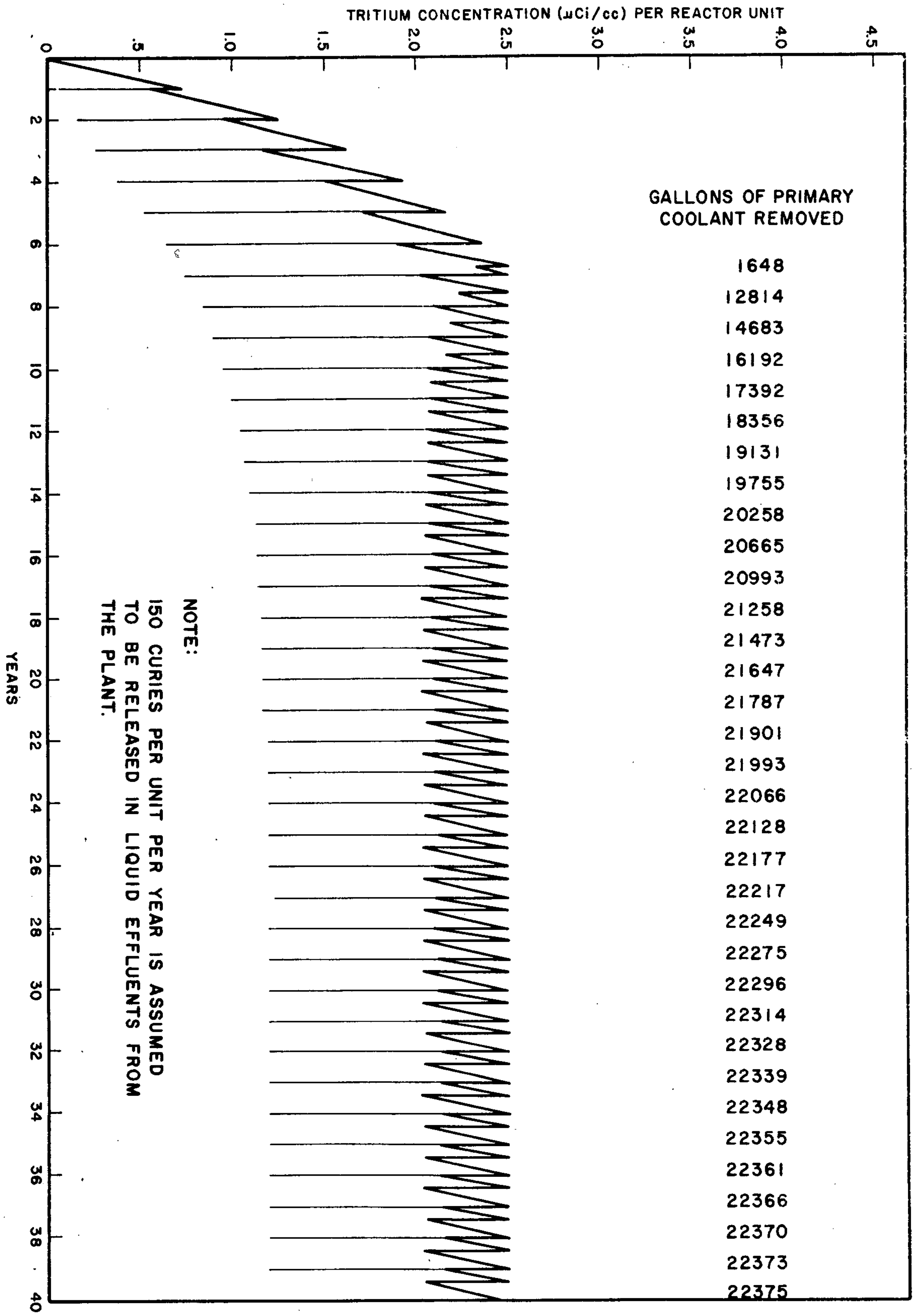
Origin of Release	Release, Ci/yr
Liquid waste disposal system	0.67(0) [(b)]
Processing condensate demineralizer spent regenerants	0.26(0)
Secondary system liquid leakage	0.200(-4)

(a) Total tritium released in liquid effluents is less than 300 curies per year for two reactor units

(b) 0.67×10^0

Table 11.1-12. Estimated Sources of Radioactivity Released to The Environment During One Year's Operation of Two Units

Isotope	Liquid Releases (Ci)	Gaseous Releases (Ci)	
^{84}Br	0.6922×10^{-7}	0.97×10^{-5}	
^{85}Br	0.6190×10^{-8}	0.20×10^{-7}	
$^{85}Kr^{m}$	0.0	$0.33 \times 10^{+2}$	
^{85}Kr	0.0	$0.156 \times 10^{+4}$	
^{87}Kr	0.0	$0.197 \times 10^{+2}$	
^{88}Kr	0.0	$0.60 \times 10^{+2}$	
^{88}Rb	0.4479×10^{-6}	0.57×10^{-4}	
^{89}Sr	0.8800×10^{-3}	0.12×10^{-3}	
^{90}Sr	0.9232×10^{-4}	0.82×10^{-9}	
^{91}Sr	0.2727×10^{-6}	0.13×10^{-6}	
^{92}Sr	0.2645×10^{-7}	0.42×10^{-7}	
^{90}Y	0.8194×10^{-5}	0.28×10^{-9}	
^{91}Y	0.5764×10^{-3}	0.16×10^{-7}	
^{99}Mo	0.3096×10^{-1}	0.12×10^{-4}	
^{106}Ru	0.9974×10^{-1}	0.62×10^{-5}	
$^{131}Xe^{m}$	0.0	$0.646 \times 10^{+2}$	
$^{133}Xe^{m}$	0.0	$0.326 \times 10^{+2}$	
^{133}Xe	0.0	$0.1971 \times 10^{+4}$	
$^{135}Xe^{m}$	0.0	$0.90 \times 10^{+1}$	
^{135}Xe	0.0	$0.762 \times 10^{+2}$	
^{129}I	0.8880×10^{-7}	0.44×10^{-8}	1
^{131}I	$0.1090 \times 10^{+0}$	0.13×10^{-2}	
^{132}I	0.3893×10^{-5}	0.49×10^{-3}	
^{133}I	0.3014×10^{-1}	0.12×10^{-2}	
^{134}I	0.1099×10^{-5}	0.19×10^{-3}	
^{135}I	0.3200×10^{-4}	0.64×10^{-3}	
^{134}Cs	$0.1094 \times 10^{+0}$	0.11×10^{-5}	
^{136}Cs	0.4859×10^{-2}	0.73×10^{-6}	
^{137}Cs	0.2768×10^{-0}	0.63×10^{-5}	
^{138}Cs	0.1313×10^{-5}	0.50×10^{-5}	
$^{137}Ba^{m}$	$0.2547 \times 10^{+0}$	0.62×10^{-5}	
^{139}Ba	0.3901×10^{-7}	0.35×10^{-6}	
^{140}Ba	0.9061×10^{-4}	0.36×10^{-7}	
^{149}La	0.1980×10^{-3}	0.13×10^{-7}	
^{144}Ce	0.2487×10^{-3}	0.29×10^{-8}	
^{51}Cr	0.5828×10^{-3}	0.49×10^{-7}	
^{54}Mn	0.9741×10^{-4}	0.56×10^{-8}	
^{59}Fe	0.7817×10^{-4}	0.56×10^{-8}	
^{58}Co	0.4475×10^{-2}	0.29×10^{-6}	
^{60}Co	0.2934×10^{-2}	0.16×10^{-6}	
^{95}Zr	0.5848×10^{-2}	0.39×10^{-6}	
Total	$0.9300 \times 10^{+0}$	$0.38261 \times 10^{+4}$	
Tritium	$0.3000 \times 10^{+3}$	$0.3000 \times 10^{+3}$	



TRITIUM BUILDUP IN PRIMARY AND REFUELING VOLUMES WITH A LIMIT OF 2.5 μ CI/CC IN PRIMARY VOLUME

FIGURE 11.1-1

REVISED BY AMMEND. 3 SEPT. 4 1973

11.2. Liquid Waste Systems

11.2.1. Design Objectives

The systems handling liquid wastes are designed such that the estimated annual releases in liquid effluents comply with the following requirements of 10 CFR Part 20 and 10 CFR Part 50:

1. The radioactive isotopic concentrations in liquid effluents at the site boundary shall not exceed the limits for releases to unrestricted areas given in Appendix B of 10 CFR Part 20.

2. The releases of radioactivity from the station shall comply with the design objectives for the "as low as practicable" standard set forth in 10 CFR Part 50.

11.2.2. System Description

Figures 11.2-1, 11.2-2, and 11.2-3 are process flow diagrams which illustrate the liquid waste disposal system with its subsystems, principal flow paths, vents, drains, secondary flow paths, and interconnections with other systems. These figures also show the sources of liquid wastes and estimated quantities of the liquid waste process streams. The estimated specific activity of each nuclide in various lines and components is shown in Table 11.1-9. Table 11.2-1 is a summary of the liquid waste quantities, and Table 11.2-2 gives design data for the individual components in the system. Portions of the system are shared by both units, but since the system performs no emergency functions, no safety function is compromised by sharing this system. Tritated and nontritiated liquid wastes, chemical wastes, and detergent wastes are collected separately, and they are stored, processed, and discharged or recycled by the liquid waste disposal system, on a batch basis. Processed liquid of high chemical purity is reused in the reactor coolant system or in the secondary coolant system.

The liquid waste disposal system is arranged to collect and handle tritiated and nontritiated wastes separately. In the reactor building, equipment drains from the reactor coolant system and related tritiated sources are collected in the reactor coolant drain tank. The liquid in this tank, which normally is primarily reactor coolant pump seal leakage, is transferred with the component drain pumps to the boron recovery system for reprocessing.

For refueling, the reactor coolant system is drained to below the reactor vessel flange. This water is also transferred, using the component drain pumps, to the boron recovery system for temporary storage and is then used to refill the reactor coolant system after refueling is completed. The liquid from the component drain pump discharge can also be routed to the tritiated waste holdup tank in the auxiliary building.

Liquid collected in the reactor building normal sump may come from both tritiated and nontritiated sources and can be routed to either the tritiated or nontritiated waste holdup tank. During reactor operation this liquid, made up primarily of uncontrolled reactor building floor drains, is routed to the tritiated waste holdup tank with the reactor building sump pumps. During refueling, the liquid is routed according to its tritium content.

The tritiated liquid from auxiliary building tritiated drain header is collected in the tritiated waste holdup tank or in the tritiated auxiliary building sump tank (except for the liquids collected in the spent resin storage tank as discussed below). The tritiated liquids in the sump tank are, in turn, normally discharged to the tritiated waste holdup tank. The main sources of tritiated liquids collected in these tanks during normal operations are (1) rinse and flush water from radioactive demineralizers, (2) sample and laboratory drains, (3) tritiated water from flushing of concentrated boric acid addition lines after additions to the RC system, (4) regeneration solutions from the deborating demineralizers, (5) tritiated liquids collected in the reactor building normal sump, and (6) excess water collected in spent resin storage tank. Tritiated wastes are processed from the tritiated waste holdup tank through the waste evaporator system. The evaporator distillate is collected in distillate test tanks and checked for water quality. The distillate in the evaporator distillate test tank can be reprocessed through the evaporator until it has the desired water quality. The distillate is then transferred through the waste evaporator distillate test tank demineralizer to the distillate storage tanks in the chemical addition and boron recovery system for reuse in the plant. The evaporator concentrate is sent to the solid waste drumming station for packaging and disposal.

Radioactive spent resins sluiced from the demineralizers are collected and stored in the spent resin storage tank. The sluicing operation is performed in a closed loop between the tank and the demineralizer. The tank also collects tritiated water used to backflush filters and strainers. The excess water from these operations is collected in the tank and is routed to the tritiated auxiliary building sump tank. After the activity on the resins and other suspended matter has decayed, the resins are sluiced to the solid waste packaging station.

Nontritiated liquid from the auxiliary building nontritiated drains is collected in the nontritiated waste holdup tank, the nontritiated auxiliary building sump tank, or the chemical drain tank. The main sources of liquids collected in the waste holdup tank and the auxiliary building sump tank during normal operations are from (1) nontritiated floor drains, (2) nontritiated sample drains, and (3) control rod drive cooling water system and component cooling water system blowdown for chemical treatment and control. The main sources of liquids routed to the chemical drain tank are nontritiated chemical laboratory drains and solutions resulting from the decontamination of small items.

Liquids collected in the nontritiated waste holdup tank, the nontritiated auxiliary building sump tank, and the chemical drain tank are treated on the basis of their radioactivity content. These liquids are normally expected to have very low specific activity, so that they can be pumped to the plan discharge via the plant discharge filter. If the specific activity of the liquids in these tanks is above the discharge limits, the liquids are processed through the auxiliary waste evaporator at a time when it is not being used for processing spent regenerant solutions. The evaporator distillate is sent to a distillate test tank, sampled, analyzed, and discharged via the plant discharge line. Alternatively, it may be recycled to the secondary system. Nontritiated water used to decontaminate the spent fuel shipping cask is received in the cask decontamination waste collector tank. This waste is expected to have a very low specific activity, so after sampling and analysis, the water is routed through the waste filter to the plant discharge.

Hot laundry and shower water, which is nontritiated, is collected in a drain tank and then normally routed to the plant discharge. However, these liquids can also be processed in the auxiliary waste evaportor system, as previously described for other nontritiated wastes, if high levels of radioactivity are encountered.

Waste regenerant solutions from the condensate polishing demineralizers are collected in the regenerants neutralization tank. The regenerant solutions are then neutralized to a pH of about 7 by the addition of sulfuric acid or sodium hydroxide. Effluent from the regenerants neutralization tank is normally pumped to the plant discharge line for release from the plant. If the specific activity of this effluent is above the discharge limits, as may occur with a primary to secondary steam generator leak, the effluent is routed directly to the auxiliary waste evaporator system for additional processing.

Distillate from the auxiliary waste evaporator is normally pumped to the plant cooling tower blowdown line for release from the plant. Alternatively, the distillate may be recycled for reuse in the secondary system. The water is sent to the condenser hotwells and then through the condensate demineralizers.

11.2.3. System Design

Liquid waste disposal system components, including design parameters, are listed in Table 11.2-2. The ASME Code classification and the ANS safety class are given for each component. The equipment is similar to that used in industrial applications and to that used in the nuclear plants cited in this section. The quality classification of the equipment is in accordance with Safety Guide 26. Tanks, evaporators, and other equipment for handling tritiated liquids are provided with nitrogen gas cover systems maintained at a pressure above atmospheric to preclude the possibility of dissolved hydrogen causing a flammable or explosive hydrogen-air mixture.

The auxiliary building is designed to comply with seismic Category I. The following equipment is designed to comply with seismic Category I:

1. Reactor coolant drain tank.
2. Tritiated waste holdup tank.
3. Tritiated auxiliary building sump tank.
4. Spent resin storage tank.

5. Waste evaporator feed tank.
6. Waste evaporator (those portions that may contain radioactive gases).
7. Waste evaporator distillate test tanks.
8. Waste gas compressors.
9. Waste gas decay tanks.
10. Waste gas filter system.

The failure of other components will not result in the release of significant quantities of radioactive material.

Process instrumentation and radiation instrumentation are shown in Figures 11.2-1, 11.2-2, and 11.2-3. Radiation monitors are located at appropriate points in the system, and all lines that normally carry radioactive materials are shielded to assure that the exposure of operating personnel to radiation is as low as practicable.

11.2.4. Operating Procedures

All equipment installed to reduce radioactive effluents to the minimum practicable level will be maintained in good operating order and will be operated to the maximum extent practicable. In order to assure that these conditions are met, administrative controls will be exercised on overall operation of the system, preventive maintenance will be utilized to maintain equipment in good condition, and experience available from similar plants will be used in planning operations at this plant.

Administrative controls will be exercised through the use of detailed instructions covering valve alignment for various operations, equipment operating instructions, and other instructions pertinent to the proper operation of the processing equipment. Discharge permit forms will be utilized to assure proper procedures are followed in sampling and analyzing any radioactive liquid to be discharged and to assure proper valve alignments and other operating conditions before a release. These forms will be signed and verified by those personnel responsible for performing the analyses and approving the release.

The operating procedures and administrative controls used at the plant will be written using as guidelines current proven procedures and controls developed by operational PWR plants which have similar waste management equipment. This will ensure that the liquid waste management operations reflect field experience gained at operational plants. Two plants which have similar equipment and have, with experience, developed workable administrative controls and procedures are H.B. Robinson and R.E. Ginna.

11.2.5. Performance Tests

All tanks and process lines and equipment in the liquid waste disposal system are provided with sample connections. Analyses of samples taken from these sample points are used to determine the actual operating DF of any component for any or all of the various radionuclides of concern. The sampling and analyses of various points in the waste processing cycle will give a good indication of the performance of the individual components.

11.2.6. Estimated Releases

The estimated quantities of radioactive liquid waste released are listed in Table 11.2-1. The estimated amounts of radioactivity in these releases are given in Tables 11.1-11 and 11.1-2. These releases cover those due to both | 11
normal operation and anticipated operational occurrences. The releases are considered to be within the "as low as practicable" standard set forth in 10 CFR Part 50.

11.2.7. Release Points

All radioactive liquid wastes are released from the plant through the cooling tower blowdown (plant discharge) line. The connections from the waste disposal system to this line are shown in Figure 11.2-3. These connections are downstream of the waste evaporator distillate demineralizer and the auxiliary waste evaporator distillate demineralizer. The discharge point from the waste disposal system to the cooling tower blowdown line is shown on Figure 10.4-2, the condensate circulating water system. The location of the cooling towers with relation to the site boundary is shown on the site plot plan, Figure 1.2-1.

11.2.8. Dilution Factors

The cooling tower blowdown rate averages 30,000 gpm for two units. The maximum (design) discharge flow rate from the waste to evaporator distillate test tanks to the cooling tower blowdown line is 20 gpm. The resulting dilution factor is 30,000/20 or 1500. The maximum discharge flow rate from the auxiliary waste evaporator distillate test tank is 80 gpm and the resulting diultion factor is 375. The activity level of the contents of each distillate test tank is determined before discharge to the cooling tower blowdown.

The dosimetry calculations for drinking water are based on the conservative assumption that the liquid effluent will be mixed with 50% of the river flow between the discharge and the first public water supply intake (Scottsboro, TRM 385.8) approximately 6 miles downstream. This assumption is considered to be reasonable regardless of the final design of a diffuser system. Although further mixing will occur, 50% dilution is assumed to be maintained for approximately 43 miles until Guntersville Dam (TRM 358.0) is reached where 100% dilution is assumed to occur. Reconcentration in aquatic biota is estimated by multiplying the radionuclide concentration in the water by the factors listed in Table 11.2-3. |11

11.2.9. Estimated Doses from Radionuclides in Liquid Effluents

The following doses are calculated for exposure to radionuclides routinely released in liquid effluents.

1. Doses to man

   a. From the ingestion of water

   b. From the consumption of fish

   c. From water sports

2. Doses to terrestrial vertebrates from the consumption of aquatic plants

3. Doses to aquatic plants, aquatic invertebrates, and fish

The organisms and pathways that are considered in this report are those that are judged to be the most significant because of species, habitat, diet, or patterns of living. Conservative assumptions are applied in these analyses which should result in overestimation of the doses.

11.2.9.1. Assumptions and Calculational Methods

Internal doses are calculated using methods outlined by the International Commission on Radiological Protection which describe internal retention of radionuclides with a single-exponential model. This model is used for estimating the doses to bone, G.I. tract, thyroid, and total body of man from ingestion of water and consumption of fish and for estimating the doses to terrestrial vertebrates from the consumption of green algae. For calculating the internal doses to aquatic organisms it is assumed an equilibrium exists between the activity concentrations in the water and those inside the organisms.

External doses are estimated using either an infinite or semi-infinite, homogeneous-medium approximation depending on whether the organism is considered to be immersed in or floating on the water.

Tritium doses are considered separately and are based on a normalized release of one curie per year. The tritium dose can be computed by multiplying this normalized value by the annual tritium release estimated to be approximately 300 curies.

Population doses are estimated for the year 2020 based on the current populations multiplied by 1.95. The factor 1.95 is the increase projected for a 125-county area in the Tennessee River basin.

1. Doses to Man From the Ingestion of Water

   Data listed in Table 11.2-4 for public and industrial water systems is used to calculate dose commitments from the consumption of Tennessee River water. It is assumed that the plant effluent is mixed with one-half of the riverflow in the 6-mile reach between the nuclear plant site and the first water supply intake. Although natural water turbulence will continue to increase the dispersion downstream, it is assumed that half-dilution is maintained as far as Guntersville Dam past which full-dilution is assumed.

   Dilution is calculated using average annual flow data for the Tennessee River as measured during the 69-year period 1899-1968. The average flow ranges from approximately 39,000 ft^3/s at the nuclear plant site to 65,000 ft^3/s at the mouth of the river near Paducah, Kentucky.

   Radioactive decay and the buildup of daughter activity are based on estimates of the transport time calculated from data for water velocities which vary between 0.1 and 3.5 ft/s. No radioactive decay is considered between the time of intake in a water system and the time of consumption. It is assumed that each individual consumes 2200 ml of water per day (the average daily adult ingestion from all sources including drinking water, food, bottled drinks, etc.).

Due to lack of definitive data, no credit is taken for removal of activity from the water through adsorption on solids and sedimentation, by deposition in the biomass, or by processing within water treatment systems.

Internal doses, D_{ij}, for the jth organ from the ith radionuclide are calculated using the relation

$$D_{ij} = (DCF)_{ij} \times I_i, \tag{1}$$

where $(DCF)_{ij}$ = the dose commitment factor for the jth organ from the ith radionuclide for an average adult assuming that the dose can be accumulated over a 50-year interval, (mRem/μCi),

I_i = the activity of the ith radionuclide taken into the body annually via ingestion, (μCi).

The dose commitment factors are derived from data given in references 1 through 4 and are defined in units of (mRem/μCi) by the equation

$$(DCF)_{ij} = \frac{51.2 \times 10^3 \, f_{wij} \, \varepsilon_{ij} \left[1-\exp\,(\lambda_{ij}T)\right]}{m_j \, \lambda_{ij}}, \tag{2}$$

where $51.2 \times 10^3 = \left(1.60 \times 10^{-8} \dfrac{\text{g-rad}}{\text{MeV}}\right)\left(3.20 \times 10^9 \dfrac{\text{dis}}{\mu\text{Ci-day}}\right)\left(10^3 \dfrac{\text{mRem}}{\text{Rem}}\right)$

f_{wij} = fraction of the ith radionuclide taken into the body by ingestion that is retained in the jth organ, (dimensionless),

ε_{ij} = effective energy absorbed in the jth organ per disintegration of the ith radionuclide including daughter products, (MeV-Rem/dis-rad),

λ_{ij} = the effective decay constant of the ith radionuclide in the jth organ, (day $^{-1}$),

T = integration time, (18,250 days),

n_j = mass of the jth organ, (g).

In the absence of a detailed knowledge regarding solubility characteristics of the radionuclides, the dose for the G.I. tract is overestimated using the assumption that none of the radionuclides is removed from the G.I. tract by absorption. Estimates of the doses to the bone, thyroid, and total body are based on fractional uptakes given by the International Commission on Radiological Protection.[1] A detailed breakdown of the dose commitments at each public water supply intake is shown in Tables 11.2-5 and 11.2-6.

For comparison, dose commitments are also calculated for a hypothetical individual whose entire yearly water supply is obtained from the plant discharge conduit prior to dilution. These estimates are upper limits based on a continuous discharge flow rate of 30,000 gpm which corresponds to the average effluent flow rate. Average-annual concentrations of radionuclides in the liquid effluent can be estimated by dividing the releases by the annual discharge flow.

Doses to humans from ingestion of Tennessee River water affected by slug releases can be estimated using the data in section A of Tables 11.2-5 and 11.2-6 provided: (1) the distribution of activity is essentially the same as that given in Table 11.1-12, (2) the total activity of the slug release is known, and (3) the river velocities and dilution factors are not grossly different from the average values on which the routine dose estimates are based. A conservative estimate of the doses to humans from a slug of radioactivity released during low-flow conditions can be obtained by multiplying the doses in Table 11.2-5 by: (1) the ratio of activity released to 0.93 Ci, and (2) by the ratio of the average flow rate to the actual flow rate. For example, a slug of 1.0 Ci activity released during a 5%* flow condition could result in doses that are higher than those in Table 11.2-6 by the factor

$$F = \frac{1.0\ \text{Ci}}{0.93\ \text{Ci}} \times \frac{\text{(average flow rate)}}{0.37 \times \text{(average flow rate)}} = 2.9$$

2. Doses to Man From the Consumption of Fish

Current estimates of Tennessee River annual fish harvests are 15.2/lb/acre sport fish[5] and 13.7 lb/acre edible commercial fish.[6] It is assumed that these rates will increase with the population expansion, so that the dose calculations are based on harvests of 30 lb/acre sport fish and 27 lb/acre commercial fish in the year 2020. The Tennessee River is segmented into 10 reaches in order to facilitate the calculations of fish harvests and radioactivity concentrations. The radioactivity levels in the fish from each reach are estimated by the product of an average activity concentration in the reach and a concentration factor for each radionuclide.[3,4,] It is assumed that the maximum annual consumption of fish by an individual is 45 pounds. The population dose 11

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*A 5% flow rate is that which is equaled or exceeded 95% of the time. This flow rate is approximately 37% of the annual-average flow rate based on daily discharge data during 1960-1970 for Nickajack, Guntersville, Wheeler, Pickwick Landing, and Kentucky Dams.

is calculated using the assumption that all of the edible fish harvested are consumed by humans. Radioactive decay is not considered between the time the fish is removed from the water and the time of consumption, and the entire mass of the fish is assumed to be eaten.

Dose commitments are calculated with equations 1 and 2 which are discussed for water ingestion in the previous section, and the results are shown in Tables 11.2-5 and 11.2-6.

Calculations indicate that there would be no significant radiological impact from human utilization of shellfish. Shellfish are not currently being harvested commercially in the Tennessee River, and consumption of shellfish by humans is assumed to be negligible.

3. Doses to Man Due to Water Sports

Estimates of the doses from immersion in the Tennessee River are calculated for each radionuclide using the following relations. For the dose rate to the skin,

$$R_i \left(\frac{\text{mRem}}{\text{day}}\right) = 51.2 \times 10^3 \, C_{wi} \left(E_\beta/2 + E_{\gamma\, i}\right) \tag{3}$$

For the dose rate to the total body,

$$R_i \left(\frac{\text{mRem}}{\text{day}}\right) = 51.2 \times 10^3 \, C_{wi} \, E_{\gamma i} \tag{4}$$

where

$$51.2 \times 10^3 = \left(1.60 \times 10^{-8} \, \frac{\text{g-rad}}{\text{MeV}}\right) \left(3.20 \times 10^9 \, \frac{\text{dis}}{\mu\text{Ci-day}}\right) \left(10^3 \, \frac{\text{mRem}}{\text{Rem}}\right)$$

C_{wi} = water concentration for the ith radionuclide, (μCi/g),

$E_{\gamma i}$ or

$(E_\beta/2 + E_\gamma)_i$ = average effective energy emitted by the ith radionuclide per disintegration, (MeV-Rem/dis-rad).

Dose rates for above-water activities such as boating are assumed to be given by equations 3 and 4 divided by 2. Water concentrations are calculated for 10 reaches between the nuclear plant site and Kentucky Dam (TRM 22.4). Doses to the population are calculated using estimates for above-water visits and in-water visits for the respective reaches based on current information given in reference 9 multiplied by the predicted population growth factor of 1.95.

The maximum individual doses for above-water use of the river are estimated for a commercial fisherman who is not a water sport enthusiast but who might be exposed for 300 days per year at 5 hours per day. The maximum individual doses for in-water activities are estimated for a person who swims 918 hours per year (6 hours per day for the 5 warm months) at a location just below the Bellefonte site. In order to estimate the maximum tritium dose to a swimmer, continuous immersion for 5 months in the Tennessee River just below the Bellefonte site is assumed.

4. Doses to Organisms Other Than Man

A comprehensive analysis of the radiation doses to species other than humans would require many man-years of effort that could be justified only if a significant radiological impact on a particular species were anticipated. After consultation with professionals in the health physics and radioecology fields, a decision was made by TVA to restrict the analyses to those organisms living on or near the Bellefonte site that would most likely receive the greatest doses. These include terrestrial vertebrates, aquatic plants, aquatic invertebrates, and fish.

a. Terrestrial vertebrates. Radioactivity contained in nuclear plant liquid effluents is concentrated in fish, invertebrates, and plants by factors that range from less than 1 to greater than 10^5 depending on inter-related physical, chemical, and biological factors. Terrestrial vertebrates will receive a radiation dose from liquid effluents if their food chain includes aquatic organisms that have concentrated radionuclides. In general, aquatic plants such as green algae concentrate trace elements to a greater extent than do fish and invertebrates.[3] Therefore, internal dose estimates have been made for ducks and muskrats with the conservative assumption that their diet consists entirely of green algae from algal masses growing near the Bellefonte discharge. Equations 1 and 2 from section 11.2.9.1.1. are used for estimating the annual internal total body 11
dose. It is assumed that the duck or muskrat has a mass m of 1000 g, and an effective radius of 10 cm, and consumes 333 g of green algae per day. Long-lived radionuclides such as Sr-90 can deliver significant portions of the total dose commitment long after the time of ingestion. Therefore, a period of 5 years was chosen for the integration interval T. In the absence of data specifically applicable to ducks or muskrats, ICRP data[1] are used for the fractional uptake in the total body and for the biological half-life of parent radionuclides. The use of human data for the biological half-lives is considered to be conservative because, in general, warm-blooded vertebrates that are smaller than man exhibit more rapid elimination rates.[4] Equation 5 is a combination of the above assumptions with equations 1 and 2.

$$D_i(\text{mrad}) = 51.2 \times 10^3\ I_i\ f_{wi}\ \varepsilon_i\ (1-\exp(-\lambda_i T))/\lambda_i m \qquad (5)$$

where

I_i = 333 g/d × C_{wi} F_{pi} × 365 d/y, (μCi/y),

C_{wi} = water concentration, (μCi/g),

F_{pi} = concentration factor(7,8) for aquatic plants, (dimensionless).

T = 1825 days

m = 1000 g

External doses are estimated with equation 4 using the conservative assumption that the duck and muskrat are exposed continuously by full immersion in the water.

Estimates of the doses to ducks and muskrats living near the Bellefonte Nuclear Plant are shown in Table 11.2-7.

b. Aquatic plants, invertebrates, and fish. Radionuclide activity internally deposited in these organisms is estimated from the concentration in the water in the Tennessee River just below the liquid effluent discharge, assuming mixing with one-half the average river flow, multiplied by the applicable concentration factors.(3,4) Doses are estimated for organisms having effective radii of 3 cm and 30 cm. Although estimates for both geometries are reported, an effective radius of 30 cm could represent organisms weighing up to 250 pounds. This geometry probably results in overestimates of the doses. In the absence of a detailed knowledge of the dynamic behavior of daughter products that are produced from internally deposited parents, the conservative assumption is made that all daughter products are permanently bound in the organisms and every daughter in a decay chain contributes energy at an equilibrium disintegration rate for each disintegration of the parent. The annual doses from the ith radionuclide are calculated using the relation:

$$D_i\ (\text{mrad}) = 51.2 \times 10^3\ C_{fi}\ \varepsilon_i \times 365 \qquad (6)$$

where

C_{fi} = radioactivity concentration in the organism

= C_{wi} × F_i, (μCi/g),

C_{wi} = water concentration, (μCi/g),

F_i = concentration factor, (dimensionless)

External doses for organisms surrounded by water are calculated using equation 4. Benthic organisms such as mussels, worms, and fish eggs may receive higher external doses if significant radioactivity is associated with bottom sediments. Accurate prediction of the accumulation of activity in sediment requires a detailed knowledge of a number of physicochemical factors including mineralogy, particle size, exchangeable calcium in the sediment, channel geometry, water-flow patterns, and the chemical forms of the radiocompounds. Many of these factors must be obtained from extensive field experiments. In the absence of detailed knowledge, the doses are calculated using the following assumptions.

(1) Two-tenths of the activity in the liquid effluent is deposited uniformly in a sediment bed having dimensions of 10 cm × 100 m × 10 km.

(2) The radioactivity concentration in the sediment is calculated assuming a buildup over the plant life of 35 years at a constant rate of deposition.

(3) Beta doses are based on a 4-π geometry and gamma doses assume a 2-π geometry.

The doses calculated using these assumptions are probably overestimated. Periodic surveillance of the sediment downstream from the nuclear plant will detect a buildup of radionuclides in the sediment, should it occur. If a gradual buildup of radionuclides in the sediment does occur, correction action will be taken prior to its becoming a significant environmental hazard.

Estimates of the doses to aquatic plants, invertebrates, and fish living near the Bellefonte Nuclear Plant are shown in Table 11.2-8.

11.2.9.2. Summary of Dose From Radionuclides in Liquid Effluents

Radiation doses calculated for releases of radionuclides in liquid effluents during normal operation of the Bellefonte Nuclear Plant are summarized in Table 11.2-9. Thyroid tissues are expected to receive the greatest average doses for both the maximum individual dose and the Tennessee Valley population dose. The maximum individual doses are less than 0.03% of natural background doses, and the average doses to humans from drinking water and eating fish from the Tennessee River downstream from the Bellefonte Nuclear Plant are less than 0.01% of natural background.* |11

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*This conclusion is based on the assumption that the population drinking water from the Tennessee River (630,000 in the year 2020) also consumes all of the fish harvested from the river downstream from the Bellefonte Nuclear Plant.

REFERENCES

(Section 11.2)

(1) C. M. Lederer, et al., Table of Isotopes, John Wiley & Sons, New York, (1968).

(2) ICRP Publication 2, Pergamon Press, New York, (1959).

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(4) I. S. Eve, "A Review of the Physiology of the Gastrointestinal Tract in Relation to Radiation Doses from Radioactive Materials," Health Physics, Vol 12 (1966), p 131.

(5) "Kentucky Lake Fishing," Tennessee Game and Fish Commission, Tennessee 11
Valley Authority, and Kentucky Department of Fish and Wildlife Resources.

(6) "Estimated Commercial Fish Harvest, Tennessee Valley, 1969," U.S. Department of the Interior, Bureau of Commercial Fisheries.

(7) W.H. Chapman, et al., Concentration Factors of Chemical Elements in Edible Aquatic Organisms, UCRL-50564, Lawrence Radiation Laboratory (1968).

(8) D. E. Reichle, et al., "Turnover and Concentration of Radionuclides in Food Chains," Nuclear Safety, Vol 11, No. 1, January-February 1970.

(9) Tennessee Department of Conservation, "Statistical Summary — State Demand, Supply, and Comparisons," Tennessee Statewide Comprehensive Outdoor Recreation Plan - 1969, Final Report, Appendix IV, Table 27, (1969), p 31.

Table 11.2-1 Summary of Liquid Waste Quantities

Source	Quantity (dual plant), ft^3/yr	Assumptions and comments
Tritiated Wastes		
Misc system leakage	7,735	Total system leakage, 10 gph with ~2/3 classed as tritiated
Charging ion exchange resins	3,156	Draining and rinsing demineralizers after resin sluicing and during resin fill
Regeneration of deborating demineralizers	4,000	Four regenerators per year
Filter backwash and drain	1,022	Backflush 5 min per operation
Strainer flush	837	Flush 5 min per flush
Batch controller flush	2,050	Flush once/day during daily load changes
Sampling and lab drains	4,700	20 samples/day @ 5 gal. sample
Refueling canal drain	324	Remove remaining water from canal
Subtotal	23,824	
Non-Tritiated Wastes		
Misc system leakage	3,867	Total system leakage, 10 gph with ~1/3 classed as non-tritiated
Spent fuel cask decontamination	50,000	30 decontaminations/yr @ 1600 ft^3 each
Sample drains	1,100	9 samples/day @ 5 gal./sample
CW systems blowdown, filter backflush, and non-tritiated filter and stainer drains	1,874	--
Spent regenerant solutions from condensate demineralizers	330,000	
Subtotal	386,841	
Chemical Wastes		
Non-tritiated chemical lab drains	5,800	
Decontamination drains	1,000	500 items @ 2 ft^3 each

Table 11.2-1 (Cont'd)

Source	Quantity (dual plant), ft^3/yr	Assumptions and comments
Detergent Wastes		
Laundry drains	28,800	600 gal./day
Shower and sink drains	28,800	20 showers/day @ 30 gal. each
Subtotal	57,600	
Total liquid discharged (sum of non-tritiated, chemical, and detergent wastes)	121,241	

Table 11.2-2. Liquid Waste Disposal System Component

Reactor coolant drain tank	
Number per unit	1
Volume, ft^3	1600
Material	SS
Design press., psig	100
Design temp, F	340
Design code	ASME III, Class 3
Reactor coolant drain tank cooler	
Number per unit	1
Type	shell and tube
Heat transferred, Btu/h $\times 10^6$	1.0
Shell/tube flow, lb/h $\times 10^4$	6.25/6.25
Tube side temp change, F	150 to 134
Shell side temp change, F	105 to 121
Material shell/tube	CS/SS
Design press., psig	150
Design temp, F	200
Design code	ASME III, class 3 (tube side) ASME VIII (shell side)
Non-tritiated waste holdup tank	
Number	1
Volume, ft^3	3300
Material	SS
Design press., psig	4
Design temp, F	200
Design code	ASME VIII (no code stamp)
Tritiated waste holdup tank	
Number	1
Volume, ft^3	3300
Material	SS
Design press., psig	4
Design temp, F	200
Design code	ASME III, class 3

Table 11.2-2. (Cont'd)

Spent resin storage tank	
Number	1
Volume, ft^3	1300
Material	SS
Design press., psig	4
Design temp, F	200
Design code	ASME III, class 3
Waste evaporator feed tank	
Number	1
Volume, ft^3	300
Material	SS
Design press., psig	4
Design temp, F	200
Design code	ASME III, class 3
Waste evaporator distillate test tank	
Number	2
Volume, ft^3	300
Material	SS
Design press., psig	4
Design temp, F	200
Design code	ASME III, class 3
Waste evaporator unit	
Number	1
Process rate, gpm	2
Material	SS
Design code	ASME III, class 3*
Waste evaporator distillate test tank pump	
Number	2
Capacity/head, gpm/ft	10/60
Material	SS
Design press., psig	150
Design temp, F	200
Design code	ASME III, class 3

Table 11.2-2. (Cont'd)

Spent resin transfer pump	
Number	1
Capacity/head, gpm/ft	10/100
Material	SS
Design press., psig	150
Design temp, F	200
Design code	ASME III, class 3
Spent resin liquid sluicing pump	
Number	1
Capacity/head, gpm/ft	50/50
Material	SS
Design press., psig	150
Design temp, F	200
Design code	ASME III, class 3
Component drain pump	
Number	2
Capacity/head, gpm/ft	150/100
Material	SS
Design press., psig	150
Design temp, F	300
Design code	ASME III, class 3
Waste transfer pump	
Number	3
Capacity/head, gpm/ft	50/50
Material	SS
Design press., psig	150
Design temp, F	200
Design code	ASME III, class 3
Non-tritiated auxiliary building sump tank	
Number	1
Volume, ft^3	500
Material	SS

Table 11.2-2. (Cont'd)

Design press., psig	4
Design temp, F	200
Design code	ASME VIII (no code stamp)
Tritiated auxiliary building sump tank	
Number	1
Volume, ft^3	500
Material	SS
Design press., psig	4
Design temp, F	200
ASME VII, Class 3	Design code
Laundry and hot shower drain tank pump	
Number	1
Capacity/head, gpm/ft	20/50
Material	SS
Design press., psig	150
Design temp, F	200
Design code	ASME III, class 3
Non-tritiated auxiliary building sump pump	
Number	2
Capacity/head, gpm/ft	50/50
Material	SS
Design press., psig	150
Design temp, F	200
Design code	ASME III, class 3
Tritiated auxiliary building sump pump	
Number	2
Capacity/head, gpm/ft	50/50
Material	SS
Design press., psig	150
Design temp, F	200
Design code	ASME III, class 3

Table 11.2-2. (Cont'd)

Laundry and hot shower drain tank	
Number	1
Volume, ft^3	500
Material	SS
Design press., psig	4
Design temp, F	200
Design code	ASME VIII (no code stamp)
Chemical drain tank	
Number	1
Volume, ft^3	80
Material	SS
Design press., psig	4
Design temp, F	200
Design code	ASME VIII (no code stamp)
Chemical tank drain pump	
Number	1
Capacity/head, gpm/ft	20/50
Material	SS
Design press., psig	150
Design temp, F	200
Design code	ASME III, class 3
Waste evaporator distillate test tank demineralizer	
Number	1
Type	Non-regenerative mixed bed
Material	SS
Resin volume, ft^3	6
Flow, gpm	20
Design press., psig	150
Design temp, F	200
Design code	Not applicable

Table 11.2-2. (Cont'd)

Plant discharge filter	
Number	1
Type	Disposable element
Material	SS
Flow, gpm/ΔP, psi	100/20
Design press., psig	150
Design temp, F	200
Design code	Not applicable
Reactor building normal sump pumps	
Number	2
Capacity/head, gpm/ft	50/100
Material	SS
Design press., psig	150
Design temp, F	200
Design code	ASME III, class 3
Cask decontamination waste collector tank	
Number	1
Volume, ft^3	2000
Material	CS
Design press., psig	atm
Design temp, F	200
Cask decontamination waste pump	
Number	2
Capacity/head, gpm/ft	40/150
Material	SS
Design press., psig	150
Design temp, F	200
Cask decontamination waste filter	
Number	2
Type	Disposable element
Material	SS
Flow, gpm/(ΔP, psi normal/max)	40/(5/20)

Table 11.2-2. (Cont'd)

Design press., psig	150
Design temp, F	200
Auxiliary waste evaporator unit	
Number	1
Process rate, gpm	30
Material	SS
Design code	ASME VIII (vend. std)
Auxiliary waste evaporator distillate test tank	
Number	2
Volume, ft^3	3000
Design press., psig	4
Design temp, F	200
Design code	ASME VIII (vend.std)
Auxiliary waste evaporator distillate test tank pumps	
Number	2
Capacity/head, gpm/ft	80/60
Material	SS
Design press., psig	150
Design temp, F	200
Design code	ASME VIII (vend. std)
Auxiliary waste evaporator distillate test tank demineralizer	
Number	2
Resin volume, ft^3	30
Capacity, gpm	80
Material	SS
Design press., psig	150
Design temp, F	200
Design code	ASME VIII (vend. std)

*Includes portions that contain radiogases

Table 11.2-3. Concentration Factors for Aquatic Organisms

Nuclide	Half-Life (d)	Concentration Factors: Fish	Concentration Factors: Invertebrates	Concentration Factors: Plants	
^{3}H	4.5 (+3)	1.0 [a]	1.0 [a]	1.0 [a]	
^{51}Cr	2.8 (+1)	2.0 (+2)[a]	2.0 (+3)[a]	4.0 (+3)[a]	
^{54}Mn	3.0 (+2)	2.5 (+1)[a]	1.4 (+5)[b]	3.5 (+4)[b]	
^{59}Fe	4.6 (+1)	3.0 (+2)[a]	3.2 (+3)[a]	5.0 (+3)[a]	
^{58}Co	7.1 (+1)	2.1 (+1)[b]	1.3 (+3)[a]	6.2 (+3)[b]	
^{60}Co	1.9 (+3)	4.8 (+1)[b]	1.5 (+3)[a]	6.2 (+3)[b]	
^{84}Br	2.2 (-2)	1.3 (+2)[a]	1.0 (+2)[a]	7.5 (+2)[a]	
^{88}Rb	1.2 (-2)	2.0 (+3)[a]	2.0 (+3)[a]	1.0 (3)[a]	
^{89}Sr	5.3 (+1)	3.5 [c]	4.0 (+3)[b]	3.0 (+3)[b]	
^{90}Sr	1.0 (+4)	9.9 [c]	4.0 (+3)[b]	3.0 (+3)[b]	
^{91}Sr	4.0 (-1)	4.0 (-2)[c]	3.2 (+3)[b]	3.0 (+3)[b]	
^{92}Sr	1.1 (-1)	1.1 (-2)[c]	2.1 (+3)[b]	3.0 (+3)[b]	
^{90}Y	2.7	1.0 (+2)[a]	1.0 (+3)[a]	1.0 (+4)[a]	
^{91}Y	5.9 (+1)	1.0 (+2)[a]	1.0 (+3)[a]	1.0 (+4)[a]	
^{95}Zr	6.6 (+1)	1.0 (+2)[a]	1.0 (+3)[a]	1.0 (+4)[a]	
^{95}Nb	3.5 (+1)	3.0 (+4)[a]	1.0 (+2)[a]	1.0 (+3)[a]	
^{99}Mo	2.8	1.0 (+2)[a]	1.0 (+2)[a]	1.0 (+2)[a]	
^{106}Ru	3.7 (+2)	1.0 (+2)[a]	2.0 (+3)[a]	2.0 (+3)[b]	11
^{129}I	6.2 (+9)	5.0 (+1)[b]	1.0 (+3)[b]	2.0 (+2)[b]	
^{131}I	8.1	4.5 (+1)[b]	1.0 (+3)[b]	2.0 (+2)[b]	
^{132}I	9.4 (-2)	4.3 [b]	1.0 (+3)[b]	2.0 (+2)[b]	
^{133}I	8.5 (-1)	2.3 (+1)[b]	1.0 (+3)[b]	2.0 (+2)[b]	
^{134}I	3.6 (-2)	1.7 [b]	1.0 (+3)[b]	2.0 (+2)[b]	
^{135}I	2.8 (-1)	1.1 (+1)[b]	1.0 (+3)[b]	2.0 (+2)[b]	
^{134}Cs	7.5 (+2)	1.0 (+3)[a]	9.9 (+3)[b]	2.5 (+4)[b]	
^{136}Cs	1.4 (+1)	9.3 (+2)[a]	5.8 (+3)[b]	2.5 (+4)[b]	
^{137}Cs	1.1 (+4)	1.0 (+3)[a]	1.0 (+4)[b]	2.5 (+4)[b]	
^{138}Cs	2.2 (-2)	2.2 (+1)[a]	2.2 (+1)[b]	2.5 (+4)[b]	
^{140}Ba	1.3 (+1)	1.0 (+1)[a]	2.0 (+2)[a]	5.0 (+2)[a]	
^{140}La	1.7	1.0 (+2)[a]	1.0 (+3)[a]	1.0 (+4)[a]	

Table 11.2-3. (Cont'd)

Nuclide	Half-Life (d)	Concentration Factors Fish	Invertebrates	Plants
^{144}Ce	2.8 (+2)	1.0 (+2)[a]	1.0 (+3)[a]	1.0 (+4)[a]
^{144}Pr	1.2 (-2)	1.0 (+2)[a]	1.0 (+3)[a]	1.0 (+4)[a]

a. W. H. Chapman, L. H. Fisher, and M.W. Pratt, "Concentration Factors of Chemical Elements in Edible Aquatic Organisms," Lawrence Livermore Laboratory Report, UCRL-50564 (1968).

b. D. E. Reichle, P. B. Dunaway, and D. J. Nelson, "Turnover and Concentration of Radionuclides in Food Chains," Nuclear Safety, 11, (1) (January-February, 1970).

c. Personal Communication D. J. Nelson, Oak Ridge National Laboratory, to W. H. Wilkie, 1972.

d. Personal Communication S. V. Kaye, Oak Ridge National Laboratory, to W. H. Wilkie, 1972.

Table 11.2-4 Tennessee River Drinking Water Supply Intakes Downstream From the Bellefonte Nuclear Plant

System	Location (TRM)	Distance (Miles)	Populations Served 1970	2020
Bellefonte Nuclear Plant	392.0	0.0	0	0
Scottsboro	385.8	6.5	11,000	21,000
Sand Mountain Water Authority	382.1	9.9	8,200	16,000
Christian Youth Camp	368.2	23.8	130	240
Guntersville	358.0	34.0	6,600	13,000
N.E. Morgan Co. Water and Fire	334.4	57.6	3,600	7,000
Huntsville	334.2	57.8	150,000	290,000 |11
Decatur	306.0	86.0	41,000	80,000
U.S. Plywood - Champion Papers	283.0	109.0	500	1,000
Wheeler Dam	274.9	117.1	50	100
Reynolds Metals	260.0	132.0	5,000	10,000
Muscle Shoals	259.6	132.4	7,500	15,000
Wilson Dam	259.5	132.5	2,500	4,900
Sheffield	254.3	137.7	14,000	27,000
Colbert Steam Plant	245.0	147.0	350	680
Cherokee	239.3	152.7	2,700	5,300
Tri-County Utility District	193.5	198.5	1,700	3,200
Clifton	158.0	234.0	1,000	2,000
New Johnsonville	100.5	291.5	950	1,900
Camden	100.4	291.6	3,100	6,000
Foote Mineral	100.0	292.0	170	320
Johnsonville Steam Plant	100.0	292.0	380	730
Bass Bay Resort	79.5	312.5	120	230
Paris Landing State Park	66.3	325.7	100	200
Grand Rivers	24.0	368.0	640	1,200
Paducah	0.1	391.9	63,000	120,000

Table 11.2-5. Doses[(a)] to Humans From Water Containing a Mixture[(b)] of Radionuclides

A. Ingestion of Tennessee River Water[(c)]

Location	Bone	G.I. Tract	Thyroid	Total Body
Bellefonte Site (for comparison)	1.7 (-3)[d]	2.1 (-3)	1.5 (-2)	9.8 (-4) mRem
Scottsboro	1.7 (-3)	2.1 (-3)	1.4 (-2)	9.8 (-4) mRem
	3.6 (-2)	4.4 (-2)	3.0 (-1)	2.1 (-2) man-Rem
Sand Mountain Authority	1.7 (-3)	2.0 (-3)	1.3 (-2)	9.7 (-4) mRem
	2.7 (-2)	3.3 (-2)	2.1 (-1)	1.6 (-2) man-Rem
Christian Youth Camp	1.6 (-3)	2.0 (-3)	1.2 (-2)	9.6 (-4) mRem
	4.0 (-4)	4.8 (-4)	2.8 (-3)	2.3 (-4) man-Rem
Guntersville	1.6 (-3)	1.9 (-3)	1.1 (-2)	9.3 (-4) mRem
	2.1 (-2)	2.4 (-2)	1.4 (-1)	1.2 (-2) man-Rem
N.E. Morgan Co., Water and Fire	7.6 (-4)	8.8 (-4)	4.5 (-3)	4.4 (-4) mRem
	5.3 (-3)	6.2 (-3)	3.2 (-2)	3.1 (-3) man-Rem
Huntsville	7.6 (-4)	8.8 (-4)	4.5 (-3)	4.4 (-4) mRem
	2.2 (-1)	2.5 (-1)	1.3 (0)	1.3 (-1) man-Rem
Decatur	7.3 (-4)	8.5 (-4)	4.0 (-3)	4.3 (-4) mRem
	5.9 (-2)	6.8 (-2)	3.2 (-1)	3.4 (-2) man-Rem
U.S. Plywood - Champion Papers	6.6 (-4)	7.5 (-4)	3.0 (-3)	3.8 (-4) mRem
	6.4 (-4)	7.3 (-4)	2.9 (-3)	3.7 (-4) man-Rem
Wheeler Dam	6.5 (-4)	7.3 (-4)	2.5 (-3)	3.8 (-4) mRem
	6.3 (-5)	7.1 (-5)	2.4 (-4)	3.7 (-5) man-Rem
Reynolds Metals	6.3 (-4)	7.0 (-4)	1.9 (-3)	3.7 (-4) mRem
	6.1 (-3)	6.8 (-3)	1.8 (-2)	3.6 (-3) man-Rem
Muscle Shoals	6.3 (-4)	7.0 (-4)	1.8 (-3)	3.7 (-4) mRem
	9.2 (-3)	1.0 (-2)	2.7 (-2)	5.3 (-3) man-Rem
Wilson Dam	6.3 (-4)	7.0 (-4)	1.8 (-3)	3.7 (-4) mRem
	3.1 (-3)	3.4 (-3)	9.0 (-3)	1.8 (-3) mRem
Sheffield	6.3 (-4)	6.9 (-4)	1.8 (-3)	3.6 (-4) mRem
	1.7 (-2)	1.9 (-2)	4.9 (-2)	9.9 (-3) man-Rem
Colbert Steam Plant	6.2 (-4)	6.9 (-4)	1.8 (-3)	3.6 (-4) mRem
	4.3 (-4)	4.7 (-4)	1.2 (-3)	2.5 (-4) man-Rem
Cherokee	6.2 (-4)	6.9 (-4)	1.7 (-3)	3.6 (-4) mRem
	3.3 (-3)	3.6 (-3)	9.1 (-3)	1.9 (-3) man-Rem
Tri-County Utility District	5.9 (-4)	6.5 (-4)	1.3 (-3)	3.4 (-4) mRem
	1.9 (-3)	2.1 (-3)	4.2 (-3)	1.1 (-3) man-Rem
Clifton	5.8 (-4)	6.4 (-4)	1.2 (-3)	3.4 (-4) mRem
	1.1 (-3)	1.2 (-3)	2.4 (-3)	6.6 (-4) man-Rem

a. Estimates for parts A, B, and C are internal dose commitments for each annual intake of radioactivity. Estimates for part D are external doses for each annual exposure.

b. Excluding tritium.

c. Based on the estimated population in the year 2020.

d. 1.7×10^{-3}.

Table 11.2-5. (Cont'd)

Location	Bone	G.I. Tract	Thyroid	Total Body	
New Johnsonville	5.2 (-4) 9.6 (-4)	5.6 (-4) 1.0 (-3)	9.5 (-4) 1.8 (-3)	3.0 (-4) mRem 5.5 (-4) man-Rem	
Camden	5.2 (-4) 3.1 (-3)	5.6 (-4) 3.3 (-3)	9.5 (-4) 5.7 (-3)	3.0 (-4) mRem 1.8 (-3) man-Rem	11
Foote Mineral	5.2 (-4) 1.7 (-4)	5.6 (-4) 1.8 (-4)	9.5 (-4) 3.1 (-4)	3.0 (-4) mRem 9.6 (-5) man-Rem	
Johnsonville Steam Plant	5.2 (-4) 3.8 (-4)	5.6 (-4) 4.1 (-4)	9.5 (-4) 7.0 (-4)	3.0 (-4) mRem 2.2 (-4) man-Rem	
Bass Bay Resort	5.1 (-4) 1.2 (-4)	5.5 (-4) 1.3 (-4)	8.6 (-4) 2.0 (-4)	3.0 (-4) mRem 6.9 (-5) man-Rem	
Paris Landing State Park	5.0 (-4) 9.8 (-5)	5.4 (-4) 1.1 (-4)	7.9 (-4) 1.5 (-4)	2.9 (-4) mRem 5.7 (-5) man-Rem	
Grand Rivers	5.0 (-4) 6.2 (-4)	5.3 (-4) 6.6 (-4)	5.4 (-4) 6.8 (-4)	2.9 (-4) mRem 3.6 (-4) man-Rem	
Paducah	4.9 (-4) 6.0 (-2)	5.2 (-4) 6.4 (-2)	5.1 (-4) 6.2 (-2)	2.8 (-4) mRem 3.5 (-2) man-Rem	
Total Population Dose Commitments	4.7 (-1)	5.5 (-1)	2.5	2.8 (-1) man-Rem	

B. Ingestion of Nuclear Plant Effluent[e] Prior to Dilution in the Tennessee River

	Bone	G.I. Tract	Thyroid	Total Body
Individual Dose Commitments	4.9 (-1)	6.2 (-1)	4.2	2.9 (-1) mRem

C. Eating Fish Taken from the Tennessee River

	Bone	G.I. Tract	Thyroid	Total Body
Maximum Individual Dose Commitment	1.9 (-2)	1.2 (-2)	1.9 (-2)	1.2 (-2) mRem
Population Dose Commitment	6.1	4.1	4.9	3.9 man-Rem

D. Use of the Tennessee River for Water Sports

	Above Water[f]		In Water[g]	
	Skin	Total Body	Skin	Total Body
Maximum Individual Dose	2.6 (-5)	2.0 (-5)	6.6 (-5)	5.0 (-5) mRem
Population Dose	1.5 (-3)	1.1 (-3)	5.6 (-4)	4.3 (-4) man-Rem

e. Assuming a continuous discharge of 30,000 gpm.

f. Boating and fishing, for example.

g. Swimming and water skiing, for example.

Table 11.2-6 Doses[a] to Humans From Water Containing Tritium[b]

A. Ingestion of Tennessee River Water[c]

	Individual (mRem)	Population (man-Rem)
Bellefonte Nuclear Plant (for comparison)	5.2 (-6)[d]	-
Scottsboro	5.2 (-6)	1.1 (-4)
Sand Mountain Water Authority	5.2 (-6)	8.3 (-5)
Christian Youth Camp	5.1 (-6)	1.2 (-6)
Guntersville	5.0 (-6)	6.4 (-5)
N.E. Morgan Co. Water and Fire	2.4 (-6)	1.7 (-5)
Huntsville	2.4 (-6)	6.8 (-4)
Decatur	2.3 (-6)	1.8 (-4)
U.S. Plywood - Champion Papers	2.1 (-6)	2.0 (-6)
Wheeler Dam	2.0 (-6)	2.0 (-7)
Reynolds Metals	2.0 (-6)	1.9 (-5)
Muscle Shoals	2.0 (-6)	2.9 (-5)
Wilson Dam	2.0 (-6)	9.6 (-6)
Sheffield	2.0 (-6)	5.4 (-5)
Colbert Steam Plant	2.0 (-6)	1.3 (-6)
Cherokee	1.9 (-6)	1.0 (-5)
Tri-County Utility District	1.9 (-6)	6.0 (-6)
Clifton	1.8 (-6)	3.6 (-6)
New Johnsonville	1.6 (-6)	3.0 (-6)
Camden	1.6 (-6)	9.6 (-6)
Foote Mineral	1.6 (-6)	5.2 (-7)
Johnsonville Steam Plant	1.6 (-6)	1.2 (-6)
Bass Bay Resort	1.6 (-6)	3.8 (-7)
Paris Landing State Park	1.6 (-6)	3.1 (-7)
Grand Rivers	1.6 (-6)	2.0 (-6)
Paducah	1.6 (-6)	1.9 (-4)
Population Total		1.5 (-3) man-Rem

11

Table 11.2-6 (Cont'd)

B. Ingestion of Nuclear Plant Effluent[e] Prior to Dilution in the Tennessee River	
Individual Dose Commitment	1.5 (-3) mRem
C. Eating Fish Taken From the Tennessee River	
Maximum Individual Dose Commitment	6.1 (-8) mRem
Population Dose Commitment	2.1 (-5) man-Rem
D. Use of the Tennessee River for Water Sports	
Maximum Individual Dose[f]	4.7 (-6) mRem
Population Dose	7.3 (-1) man-Rem

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a. Estimates are internal dose commitments for each annual intake of tritium.

b. Normalized to 1.0 Ci total annual release.

c. Based on the estimated population in the year 2020.

d. 5.2×10^{-6}

e. Assuming a continuous discharge of 30,000 gpm.

f. Assuming continuous immersion for five months.

Table 11.2-7 Doses[a] to Ducks and Muskrats Living Near The Bellefonte Nuclear Plant

	0.93 Ci Mixture	1.0 Ci Tritium
Internal	1.6 (2) mrad	5.1 (-5)[b] mrad
External	2.4 (-4) mrad	0
Total	1.6 (2) mrad	5.1 (-5) mrad

----

a. Internal dose commitments for each annual intake and external doses from each annual exposure.

b. 5.1×10^{-5}

Table 11.2-8 Doses to Aquatic Organisms Living in the Tennessee River Near the Bellefonte Nuclear Plant

A. Doses from Annual Release of 0.93 Ci Radionuclide Mixture[a]

	Internal (mrad)		External (mrad)
	3-cm	30-cm	
Plants	3.6	8.5	6.3 (-4)[b]
Invertebrates	1.6	3.5	6.3 (-4) suspended 120 benthic
Fish	0.1	0.3	6.3 (-4)

B. Doses from an Annual Release of 1.0 Ci Tritium

Plants, invertebrates, and fish	1.1 (-5) mrad (internal)

----

a. Excluding tritium

b. 6.3×10^{-4}

Table 11.2-9 Summary of Annual Radiological Impact From Liquid Effluents[a,b]

	Normal Operation	Proposed 10 CFR 50 Appendix I Guides
Liquid Effluents		
Activity released	0.93	10 Ci
Average concentration before dilution in the Tennessee River	1.5 (-8)[c] μCi/cm^3	2.0 (-8) μCi/cm^3
Maximum human organ doses		
1. bone	2.1 (-2) mRem	5 mRem
2. G.I. tract	1.4 (-2) mRem	5 mRem
3. thyroid	3.3 (-2) mRem	5 mRem
4. skin	1.3 (-2) mRem	5 mRem
5. total body	1.3 (-2) mRem	5 mRem
Human population doses within the Tennessee Valley region		
1. bone	6.6 man-Rem	
2. G.I. tract	4.7 man-Rem	
3. thyroid	7.4 man-Rem	
4. skin	4.2 man-Rem	
5. total body	4.2 man-Rem	
Maximum dose to terresterial vertebrates	160 mrad	
Maximum doses to aquatic organisms		
1. plants	8.5 mrad	
2. invertebrates	3.5 mrad suspended 120 mrad benthic	
3. fish	0.4 mrad	

a. Table excludes tritium. Doses due to released of tritium in liquid effluents are 3.0×10^{-3} mRem and 0.67 man-Rem.

b. Releases for two units operating at full power with 0.25% failed fuel.

c. 1.5×10^{-8}.

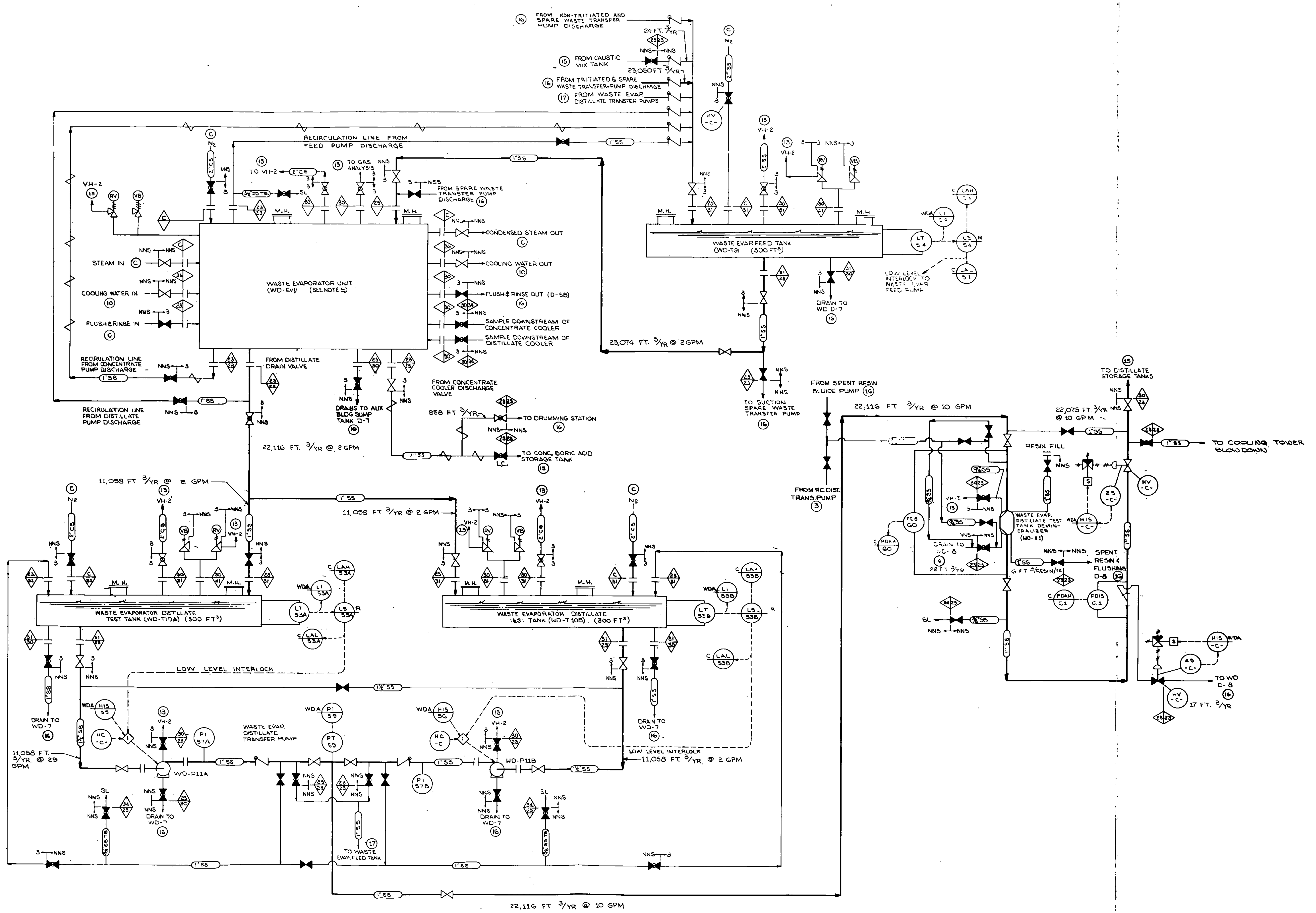
LIQUID WASTE DISPOSAL SYSTEM (DRAIN TANKS AND SUMPS)

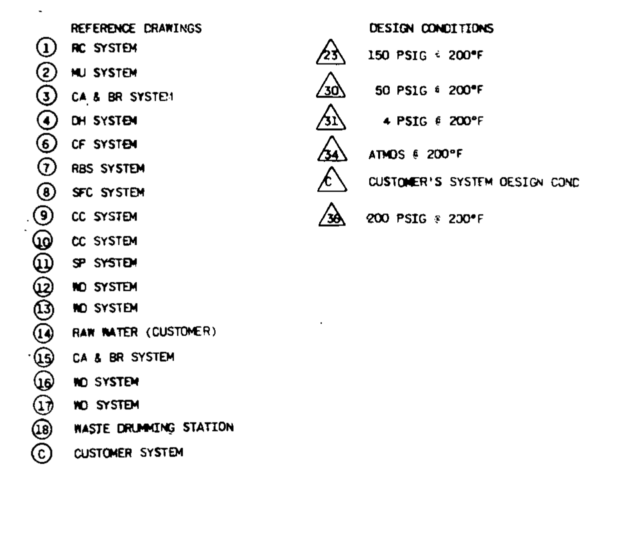
FIGURE 11.2-1

Bellefonte Nuclear Plant
Preliminary Safety Analysis Report

LIQUID WASTE DISPOSAL SYSTEM
(TRITIATED AND NONTRITIATED, ALSO
CHEMICAL DRAIN TANKS)

FIGURE 11.2-2





Bellefonte Nuclear Plant
Preliminary Safety Analysis Report

LIQUID WASTE DISPOSAL SYSTEM
(WASTE EVAPORATOR)

FIGURE 11.2-3

11.3. Gaseous Waste System

11.3.1. Design Objectives

The system is designed such that estimated annual releases of gaseous effluents from the station comply with the following requirements of 10 CFR Part 20 and 10 CFR Part 50:

1. The individual isotopic concentrations in gaseous effluents at the site boundary shall not exceed the limits for releases to unrestricted areas given in Appendix B of 10 CFR Part 20.

2. The releases of radioactivity from the station shall comply with the design objectives for "as low as practicable" standard set forth in 10 CFR Part 50.

3. The annual average exposure rate due to noble gases at any location on the site boundary, or in the offsite environment, shall not exceed 10 mRem.

11.3.2. System Description

The gaseous waste system is shown on the process flow diagram (Figure 11.3-1). The gases from tritiated sources are collected in a common vent header. Vents from the reactor coolant system and related equipment inside the reactor building are first routed to the reactor coolant drain tank and then to the vent header.

Gaseous waste will originate from:

1. Degassing reactor coolant bleed for lifetime shim and xenon override reactivity control.

2. Degassing of reactor coolant during startup expansion and dilution.

3. Pressurizer venting.

4. Degassing RC system at end of fuel cycle.

5. Reactor coolant system venting after refueling.

6. Displacement gas at startup of fuel cycle.

7. Equipment vents, relief valves and vents on liquid waste system.

One of two compressors is in normal service with the second unit on standby for peak load conditions or failure of the first compressor. The pressure in the vent header is controlled between 0.5 and 2.0 psig. On high-header pressure, the compressor compresses the gases into a decay tank. On low-header pressure, compressed gases are released back to the vent header. Nitrogen is used as a cover gas in the vent header and in the tanks that are open to the vent header. Explosions are precluded by maintaining the system under positive pressure so that oxygen cannot enter. Any leakage from the system is diluted with ventilation air such that the resulting hydrogen concentrations are below 4% by volume. An oxygen-hydrogen analyzer monitors the gas spaces in the tanks and in the evaporators in the waste disposal system and in the chemical addition and boron recovery system where hydrogen is most likely to concentrate. The analyzer is set to warn the operator if the oxygen content approaches 2.0% by volume.

When a decay tank is pressurized to 85 psig, the tank is isolated, and the gas flow is directed to the empty tank. The full tank is held for a minimum of 60 days for decay. After the 60-day period, the gas is sampled and analyzed to determine the amount of activity to be released. The gases are then released at a controlled rate through a waste gas filter system containing a prefilter, an HEPA filter, and a charcoal adsorber. The HEPA filter will be moisture- and fire-resistant. The gases discharged from the waste disposal system to the station vents will be made through one line equipped with a radiation monitor. If the radioactivity level is above the discharge limits, the radiation monitor automatically closes the valve on the discharge to stop the release. The volume of waste gas shown on the process flow diagram (Figure 11.3-1) is for both units.

Components that handle nontritiated liquids of negligible or low specific activity, such as those associated with the component water cooling water system, are vented directly to the auxiliary building atmosphere. Vents from components in the waste disposal system that handle nontritiated liquids that can contain significant radioactivity are connected directly to the auxiliary building ventilation exhaust ducts. The auxiliary building ventilation exhaust system is a trained system, with flow in each train normally passing through HEPA filters and charcoal adsorbers in series before discharge through the station vents, one atop each of the two containment structures. Each of the two vents has redundant particulate, gas, and iodine radiation detection monitors.

11.3.3. System Design

The components for the gaseous waste disposal system are listed in Table 11.3-1. The size, capacity, flow rate, design pressure, and design temperature are given, where applicable, for each component. The safety classification of this equipment is in accordance with Safety Guide 26. The possibility of a hydrogen explosion is greatly reduced as described in section 11.3.2.

The auxiliary building is designed to comply with seismic Category I. Gaseous waste disposal system components are also designed to comply with seismic Category I.

Process instrumentation and radiation instrumentation are shown in Figure 11.3-1. Radiation monitors are located at appropriate points in the system, and all equipment that processes or stores radioactive gases is shielded or located in such a manner to assure that exposure of operating personnel to radiation is as low as practicable.

The plant ventilation systems are described in 9.4.

11.3.4. Operating Procedures

All equipment installed to reduce radioactive effluents to the minimum practicable level will be maintained in good operating order and will be operated to the maximum extent practicable. In order to assure that these conditions are met, administrative controls will be exercised on overall operations of the system, preventive maintenance will be utilized to maintain equipment in peak condition, and experience available from similar plants will be used in planning for operations at the plant.

Administrative controls will be exercised through the use of detailed instructions covering valve alignment for various operations, equipment operating instructions, and other instructions pertinent to the proper operation of the processing equipment. Discharge permit forms will be utilized to assure proper procedures are followed in sampling and analyzing any radioactive gases to be discharged and to assure proper valve alignments and other operating conditions before a release. These forms will be signed and verified by those personnel responsible for performing the analysis and approving the release.

The operating procedures and administrative controls used at the plant will be written using as guidelines current proven procedures and controls developed by operational PWR plants which have similar waste management equipment. This will ensure the gaseous waste management operations reflect field experience gained by the operational plants. Two plants which have similar equipment and have, with experience, developed workable administrative controls and procedures are H. B. Robinson and R. E. Ginna.

11.3.5. Performance Tests

The two gas decay tanks will be filled, isolated, and allowed to decay. The tanks are equipped with pressure gauges and controls to fill to the proper pressure. A sample system is provided to allow for gas analysis. The tanks, after a specified decay period, will be discharged at a controlled rate through a filter charcoal adsorber system to the atmosphere. An estimated 95,000 standard cubic feet of gas will be discharged per fuel cycle from the two reactor systems. A radiation monitor is installed on the discharge line to allow cut-off of gas discharge if activity is above the setpoint.

11.3.6. Estimated Releases

The estimated amounts of radioactivity in gaseous releases from each source are given in Table 11.1-10 in terms of iodine-131 and noble gases. Table 11.1-12 gives the isotopic distribution of the total gaseous release. These releases are within the limits outlined in the technical specification and 11
in the design objective. They include those due to normal operation as well as those due to anticipated operational occurrences.

11.3.7. Release Points

There are three plant exhaust vents that have the potential to release radioactivity to the environment. These are the two station vents, one located at the roof of each secondary containment structure, and the turbine room exhaust vent. The locations of the vents are shown on Figure 1.2-2. The location of the secondary containment structures and the turbine building with relation to the site boundary are shown on Figure 1.2-1.

Radioactivity is released from the station vents as a result of containment and instrument room purging, auxiliary building ventilation, and discharges from the gas decay tanks. Normal turbine building ventilation flow, main condenser mechanical vacuum pump exhausts, and gases vented from the gland seal condensers are released from the turbine room exhaust vent.

11.3.8. Dilution Factors

TVA generates dilution factors as an integral part of performing dose calculations for gaseous releases from nuclear power plants. Radionuclides will be released from the Bellefonte Nuclear Plant through vents located near the top of various plant buildings. It is assumed that the gaseous effluents are initially diluted in the turbulent wake downwind of the building. To calculate downwind, ground-level air concentrations of these radionuclides, a ground-level, sector-average, volume-source dispersion equation, as described by Davidson[1,2], is used (equation 1):

$$X_{kmn} = \sum_{i} \sum_{j} \frac{\sqrt{2}\, Q_n f_{ijk}}{\sqrt{\pi}\,\Sigma_{zim}\, u_j\, \theta x_m} \exp\left(\frac{-\lambda_n x_m}{u_j}\right) \qquad (1)$$

and

$$\Sigma_{zim} = \left(\sigma^2_{zim} + \frac{cA}{\pi}\right)^{1/2}$$

where

X_{kmn} = average-annual, ground-level concentration of radionuclide n in sector k at distance x_m, (Ci/m^3),

Q_n = release rate of radionuclide n, (Ci/s),

f_{ijk} = fraction of the release period during which the wind blows in direction k, with speed j, and atmospheric stability condition i,

σ_{zim} = vertical standard deviation of the plume for stability condition i at distance x_m, (m),

Σ_{zim} = vertical standard deviation of the plume (modified for the effect of building wake dilution) for stability condition i at distance x_m, (m),

c = a parameter which relates the cross-sectional area of the building to the size of a turbulent wake caused by the building,

A = cross-sectional area of the reactor building, (m^2), 11

x_m = downwind distance at which the radionuclide concentration is calculated, (m),

u_j = wind speed j, (m/s),

θ = sector width, (radians),

λ_n = radioactive decay constant for radionuclide n, (s^{-1}).

It is necessary to use this sector-average, volume-source dispersion equation to account for vertical perturbation of the plume due to the building turbulent wake. The vertical plume perturbation is not accounted for in a sector-average, point-source dispersion equation.

In equation 1, the sector width, θ, is assumed to be 22-1/2 degrees (0.39 radians), c is assumed to be 0.5, and A is assumed to be 2,450 m^2, which is the minimum cross-sectional area of the reactor building. Pasquill vertical plume standard deviations[1] are used. Values for the joint meteorological frequency, f_{ijk}, are determined by methods discussed in section 2.3. In this section, the joint meteorological frequencies for the seven Pasquill stability conditions A through G are presented as a function of wind direction and wind speed in Tables 2.3-57 through 2.3-71. The data are grouped for five wind speed ranges (0-0.5, 0.6-3.4, 3.5-7.4, 7.5-12.4, $\geq$ 12.5 mph) and for 16 standard wind directions, (N, NNE, NE, ---, NW, NNW).

11.3.9. Estimated Doses from Radionuclides in Gaseous Effluents

The following doses to humans living in the vicinity of the Bellefonte Nuclear Plant are calculated for routine releases of radioactive gases:

1. External beta doses.

2. External gamma doses.

3. Thyroid doses due to inhalation of radioactive iodine.

4. Thyroid doses due to concentration of radioactive iodine in milk produced near the site.

11.3.9.1. Assumptions and Calculational Methods

The doses and radioiodine concentrations which appear in this section are calculated assuming operation of two units for one year at full power with 0.25% failed fuel. Doses are calculated for routine releases with a waste treatment system with 60-day holdup. The method for generating the dilution factors used in these dose calculations is presented in 11.3.8.

1. External beta doses - Beta doses to individuals are computed using an immersion dose model described by the equation

$$D_\beta = \sum_n 4.64 \times 10^9 \, \overline{E}_{\beta n} X_{kmn} \qquad (1)$$

where

D_β = external beta dose due to immersion in a cloud, (mRem/yr),

4.64×10^9 = conversion constant for external beta dose calculations, mRem/yr/ Ci-MeV/dis-m^3,

$\overline{E}_{\beta n}$ = average beta energy of nuclide n, (MeV/dis),

X_{kmn} = (see section 11.3.8).

In this equation, a correction factor of 0.64 is included to account for cloud geometry and a correction factor of 0.5 is included to account for self-shielding by the human body. The average beta energies for the nuclides are calculated from information contained in reference 3 and are listed in Table 11.3-2.

In computing the beta dose to the population within 50 miles of the Bellefonte Nuclear Plant, the area is divided uniformly into 16 directional sectors and 10 concentric rings, i.e., 160 area elements. A beta dose computed at the center of each element is multiplied by the number of people residing in that element. A summation of these products over all elements gives the total population dose within 50 miles of the plant. The projected population for the year 2020, as listed in Table 11.3-3, is used in calculating population dose.

The individual and population external beta doses for gaseous effluents are reported in Table 11.3-4.

2. External gamma doses — Gamma doses to individuals are computed using an immersion dose model described by the equation

$$D_\gamma = \sum_n 7.21 \times 10^9 \, \overline{E}_{\gamma n} \, X_{kmn} \qquad (2)$$

where

D_γ = external gamma dose due to immersion in a cloud, (mRem/yr),

7.21×10^9 = conversion constant for external gamma dose calculations, mRem/yr / Ci-MeV/dis-m^3,

$\overline{E}_{\gamma n}$ = average gamma energy of nuclide n, (MeV/dis),

X_{kmn} = (see section 11.3.8).

Equation 2 includes a correction factor of 0.5 to account for cloud geometry. The average gamma energies used in calculating external gamma doses are computed from data contained in reference 3 and are listed in Table 11.3.2.

The total population gamma dose within 50 miles of the Bellefonte Nuclear Plant is calculated using the method described for the population beta dose. The annual individual and population external dose for gaseous effluents are reported in Table 11.3-4.

3. Thyroid doses due to iodine inhalation — The equation used in calculating inhalation doses for routine releases of radioiodine from the Bellefonte Nuclear Plant is

$$D = \sum_n 8.76 \times 10^3 \ X_{kmn} \ (BR) \ (DCF_n) \qquad (3)$$

where

D = thyroid dose committed, (mRem committed/yr),

8.76×10^3 = hours per year,

X_{kmn} = (see section 11.3.8),

BR = breathing rate, (m^3/h),

DCF_n = dose commitment factor for inhalation of iodine isotope n, (mRem/Ci inhaled).

Maximum individual thyroid doses due to intake of radioiodine are calculated for a one-year-old child in accordance with the recommendations of
the Federal Radiation Council.(4) Population doses are calculated using 11
adult parameters and the same method described for calculating population beta doses.

The breathing rate assumed for a one-year-old child(5) is 0.29 m^3/h, and
for an adult(6) is 0.83 m^3/h. The iodine inhalation dose commitment fac- 11
tors for the one-year-old child and for the adult are obtained from reference 7.

The calculated annual individual and population iodine inhalation doses for gaseous effluents are reported in Table 11.3-5.

4. Thyroid doses due to iodine ingestion — The equation used in calculating the thyroid doses due to iodine ingestion through the milk food chain is

$$D = \sum_n 3.15 \times 10^7 \ (\chi_{kmn}) \ (v_g) \ (M_n) \ (CR) \ (DCF_n) \qquad (4)$$

where

D = thyroid dose committed, (mRem committed/yr),

3.15×10^7 = seconds per year,

χ_{kmn} = (see section 11.3.8),

v_g = radioiodine deposition velocity, (m/s),

M_n = empirically determined value for concentration of iodine isotope n in milk per unit deposition rate, Ci/liter / Ci/m^2-day,

CR = milk consumption rate, (liter/day),

DCF_n = dose commitment factor for ingestion of an iodine isotope n, (mRem/Ci ingested).

Only Iodine-131 and -133 are considered in calculating milk ingestion doses due to routine releases of radioiodine. Iodine-132, -134, and -135 have short half-lives (< 7 hours) and will have essentially disappeared due to decay before significant concentration in milk occurs.

The one-year-old child is assumed to be the critical receptor in calculating the maximum dose to an individual drinking milk produced at the nearest dairy farm (11 miles SSW of the plant). Population doses to persons within 50 miles of the plant are calculated using adult parameters. The assumption is made that all milk produced within 50 miles of the Bellefonte Nuclear Plant is consumed within this area, and cows are assumed to graze the pastures during the entire year. County milk production data[8-11] are used in computing milk ingestion population doses. 11
The population dose for the year 2020 is estimated assuming that the population dose increases in direct proportion to the increase in the population.

The numerical values used for the parameters, v_g, M, CR, and DCF are taken from references 7, 12, 13, 14, and 15. 11

The individual and population milk ingestion doses are reported in Table 11.3-5.

5. Maximum average-annual radioiodine concentration — The maximum average-annual radioiodine concentrations occur in the NNE sector at the site boundary (950 m). The maximum iodine concentrations for routine gaseous releases are calculated using the methods for generation of dilution factors discussed in section 11.3.8 and are reported in Table 11.3-6.

11.3.9.2. Summary of Doses From Radionuclides in Gaseous Effluents

Radiation doses calculated for releases of radionuclides in gaseous effluents during normal operation of the Bellefonte Nuclear Plant are summarized in Table 11.3-7.

A comparison of doses resulting from the operation of the Bellefonte Nuclear Plant to those occurring from natural radioactivity assists in placing the doses from Bellefonte in perspective. Near the plant site the average annual dose from naturally occurring external sources of radiation is 125 mRem (Table 11.3-8). An individual receives an additional dose of approximately 20 mRem per year from naturally occurring internal sources. Therefore, the average total dose from natural radioactivity in the vicinity of the Bellefonte plant is approximately 145 mRem per year. Individual doses vary widely around this average value because of local differences in the concentrations of terrestrial radioactivity and because of variations in dose rates within different types of buildings. Large variations are also observed between different areas within the United States because of the dependence of cosmic ray dose rates on altitude and geomagnetic latitude. Due to these variations, the annual total-body doses to individuals in the United States from natural radioactivity range from approximately 110 to 240 mRem.

A hypothetical individual at the site boundary would receive a maximum annual dose of about 2 mRem from the normal operation of the Bellefonte Nuclear Plant.

For this individual to receive the maximum dose he would have to stand in the open at the highest dose point on the site boundary for 24 hours a day, 365 days per year. The maximum dose to the hypothetical individual is about 1% of the dose from natural background radiation. The maximum dose to an actual individual should be significantly less than the dose to the hypothetical individual.

The population dose within 50 miles of the Bellefonte site from naturally occurring radioactivity is estimated to be approximately 240,000 man-Rems in the year 2020 (Table 11.3-8). The population dose in the year 2020 due to gaseous effluents released during the normal operation of the Bellefonte Nuclear Plant is calculated to be 7.9 man-Rems (Table 11.3-7), which is less than 0.004% of | 11
the dose to the population within 50 miles from natural background radiation.

The external beta and gamma doses to terrestrial plants and animals are considered to be of the same magnitude as the doses estimated for humans.

11.3.10. Noncondensable Waste Gases From the Secondary System

Main condenser vacuum is maintained with three vacuum pumps. One of these pumps is normally on standby. The exhaust from the vacuum pumps will be routed through a common header and vented to a single turbine building exhaust. The exhaust from the vacuum pumps is monitored with the condenser vacuum pump exhaust gas monitors. The exhaust is passed through a HEPA filter and charcoal adsorber train before being discharged through the single turbine building exhaust vent. A bypass around the filter train is used only during maintenance and when vacuum pump exhaust flow is in excess of 50 cfm as during unit startup. The single turbine building exhaust vent is monitored with redundant particulate, radiogas, and iodine monitoring systems. The feedwater heaters will be vented to the main condenser. The gland seal steam noncondensibles are vented to the single turbine building exhaust. They are thus monitored before release to the atmosphere with the redundant particulate gas and iodine radiation monitoring systems. All other components in the secondary system will be vented to the turbine building atmosphere.

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(Section 11.3)

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Table 11.3-1. Gaseous Waste Disposal System Components

Waste gas compressors	
Number	2
Capacity/discharge pressure, cfm/psig	30/85
Material	CS
Design code	ASME III class 3
Waste gas decay tanks	
Number	2
Volume, ft^3	3000
Material	CS
Design press., psig	100
Design temp, F	200
Design code	ASME III class 3
Waste gas filter system	
Number	1
Type	Pre-absolute, and charcoal filter combination
Capacity/discharge pressure, cfm/in. H_2O	200/6
Design code	ASME III class 3

Table 11.3-2. Average Gamma and Beta Energies Used to Estimate External Doses From Nuclides Released in Gaseous Effluents

Isotope	Average gamma energy, MeV/dis	Average beta energy, MeV/dis
^{131}I	3.8(-1)	2.0(-1)
^{132}I	2.5	5.0(-1)
^{133}I	6.7(-1)	4.4(-1)
^{134}I	2.4	5.2(-1)
^{135}I	1.7	3.3(-1)
$^{83}Kr^{m}$	9.0(-3)	0
$^{85}Kr^{m}$	1.5(-1)	2.5(-1)
^{85}Kr	2.0(-3)	2.4(-1)
^{87}Kr	1.5	1.3
^{88}Kr	1.7	3.9(-1)
^{89}Kr	3.9	1.7
$^{131}Xe^{m}$	2.5(-2)	1.2(-1)
$^{133}Xe^{m}$	5.3(-2)	1.6(-1)
^{133}Xe	4.9(-2)	1.2(-1)
$^{135}Xe^{m}$	4.3(-1)	9.9(-2)
^{135}Xe	2.3(-1)	3.3(-1)
^{137}Xe	3.2(-1)	1.7
^{138}Xe	2.9	9.4(-1)

Table 11.3-3. Projected 2020 Population Distribution Within 50 Miles of the Bellefonte Nuclear Plant

Direction from plant	Distance from plant, miles									
	0-1	1-2	2-3	3-4	4-5	5-10	10-20	20-30	30-40	40-50
N	--	15	100	10	--	200	350	450	6,915	3,795
NNE	--	--	30	115	45	640	5,030	13,735	2,910	4,295
NE	--	--	--	--	30	100	3,035	13,525	25,015	148,535
ENE	--	--	5	10	30	50	2,755	10,700	80,960	308,365
E	--	--	40	50	105	705	3,080	2,550	17,480	5,195
ESE	--	--	35	40	305	680	2,015	1,570	19,565	6,645
SE	--	5	25	--	40	540	2,335	12,275	4,370	13,705
SSE	--	--	--	10	10	495	11,020	17,515	4,820	16,880
S	--	--	--	5	30	840	4,530	3,345	2,495	86,775
SSW	--	--	--	5	15	795	1,750	4,010	62,230	12,245
SW	--	55	610	305	1,355	18,385	6,970	4,625	20,565	24,475
WSW	5	50	600	1,610	5,835	22,800	1,150	4,490	4,680	7,465
W	--	--	2,150	2,225	1,475	2,875	1,500	3,090	365,865	74,145
WNW	--	1,320	2,810	1,385	35	210	1,030	785	8,985	12,550
NW	5	25	70	10	25	185	555	485	5,755	18,365
NNW	5	30	20	10	15	225	155	170	26,360	34,995

Table 11.3-4. Estimated Annual External Gamma and Beta Doses from Nuclides Released in Gaseous Effluents[a]

	Total routine releases including 60-day holdup
Maximum individual gamma dose at site boundary (mRem)	5.6(-1)[b]
Maximum individual beta dose at site boundary (mRem)	1.1
Total population gamma dose within 50 miles (man-Rem)	1.8
Total population beta dose within 50 miles (man-Rem)	6.1

[a] For operation of two units at full power with 0.25% failed fuel.

[b] 5.6×10^{-1}.

Table 11.3-5. Estimated Annual Thyroid Dose Commitments From Radioiodine Released in Gaseous Effluents[a]

	Total routine releases including 60-day holdup
Iodine inhalation	
Maximum individual thyroid dose at site boundary (mRem)	1.7(-2)[b]
Total population thyroid dose within 50 miles (man-Rem)	4.2(-2)
Iodine ingestion via milk	
Maximum individual thyroid dose at nearest dairy farm (mRem)	4.5(-2)
Total population thyroid dose within 50 miles (man-Rem)	3.3(-1)

[a] For operation of two units at full power with 0.25% failed fuel.

[b] 1.7×10^{-2}.

Table 11.3-6. Estimated Maximum Annual Iodine Concentrations From Releases in Gaseous Effluents[a]

	Total routine releases including 60-day holdup
Maximum annual concentration of ^{131}I, μCi/cc	4.4(-16)[b]
Maximum annual concentration of ^{132}I, μCi/cc	1.3(-16)
Maximum annual concentration of ^{133}I, μCi/cc	4.0(-16)
Maximum annual concentration of ^{134}I, μCi/cc	3.8(-17)
Maximum annual concentration of ^{135}I, μCi/cc	2.0(-16)

[a] For operation of two units at full power with 0.25% failed fuel.

[b] 4.4×10^{-16}.

Table 11.3-7. Summary of Annual Radiological Impact From Gaseous Effluents[a,b]

	Normal operation	Proposed 10 CFR 50 Appendix I guides
A. Gaseous effluents		
^{131}I concentration at site boundary	4.4(-16)[c] μCi/cc	1.0(-15) μCi/cc
Maximum individual doses		
1. Inhalation at site boundary (thyroid)	1.7(-2) mRem	5 mRem
2. Consumption of milk from nearest dairy farm (thyroid)	4.5(-2) mRem	5 mRem
3. External exposure at site boundary (β&γ)	1.7 m/Rem	10 mRem
Population doses within a 50-mile radius		
1. Inhalation (thyroid)	4.2(-2) man-Rem	
2. Consumption of milk (thyroid)	3.3(-1) man-Rem	
3. External exposure (β&γ)	7.9 man-Rem	
B. Maximum annual dose[d] to any individual	1.7 mRem	
C. Maximum population dose[d]	7.9 man-Rem	

[a] Table excludes tritium. Doses due to releases of tritium in gaseous effluents are 0.16 mRem and 1.0 man-Rem.

[b] Releases for two units operating at full power with 0.25% failed fuel.

[c] 4.4×10^{-16}. 11

[d] Skin dose. Thyroid dose is of about the same magnitude as skin dose.

Table 11.3-8. Doses From Naturally Occurring Background Radiation

Individual doses (mRem)

External[a]		125
Internal[b]		20
	Total	145 mRem

Population dose (man-Rem)

0.145 Rem × 1,650,000[c] people = 240,000 man-Rem

[a] Measured by TVA personnel.

[b] Principles of Radiation Protection. K. Z. Morgan and J. E. Turner, eds., New York: John Wiley and Sons, Inc., (1967), p. 10.

[c] Estimated population within a 50-mile radius of the Bellefonte Nuclear Plant in the year 2020.

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WASTE GAS DISPOSAL SYSTEM

FIGURE 11.3-1

11.4. Process and Effluent Radiological Monitoring System

Means are provided for monitoring during normal operation including anticipated operational occurrences, the reactor containment atmosphere, various process streams, process effluent and ventilation flow discharge paths, and control room ventilation intake air. Some of the monitors, including the monitors on the control room intake, perform a function during accident conditions. The systems provided comply with AEC Safety Guide 21.

11.4.1. Design Objectives

The process and effluent radiological monitoring systems are designed to perform these basic functions:

1. Give warning of a condition which might lead to radioactivity releases (and concentrations in liquid releases) that could result in exceeding the limits set forth in 10 CFR 20 and the proposed Appendix I of 10 CFR 50.

2. Warn plant personnel of increasing radiation levels which might result in a radiation health hazard.

3. Rapidly provide information on fuel-clad and equipment failures or malfunctions.

4. Provide a means of leakage detection.

5. Perform, during accident conditions, functions detailed below.

All of the monitoring systems provided to meet these objectives function during normal operation.

Several of the monitors have a function during loss-of-coolant-accident conditions. Any one of four independent detectors for each unit (two radiogas detector units that sample the containment purge exhaust and the radiogas detector in each of the two containment particulate, radioiodine, and radiogas detection units) will isolate the purge system. Isolation of the purge system is necessary to ensure that if a LOCA condition occurs during purging, pressure relief through the ventilation system does not prevent containment pressure from reaching the setpoint for a containment isolation signal.

The ERCW return flow from the reactor building coolers is monitored to detect air inleakage into the ERCW system during the accident.

Water which is recirculated through the DHR coolers in the accident situation is monitored as is the component cooling system water which removes heat from the recirculation water.

The main control room air inlet monitors have a function described below in assuring compliance with General Design Criterion No. 19. When environmental conditions inside the containment permit, the containment particulate, radioiodine, and radiogas detection units are placed on line to monitor the containment atmosphere activity levels.

Additional information regarding activity and exposure levels in the plant during accident conditions is provided by some of the area and airborne activity monitors (see 12.1.4 and 12.2).

11.4.2. Continuous Monitoring

The various process and effluent radiological monitoring systems are described in the following sections. Each monitor has its power supply, ratemeter, indicator, continuous strip chart recording, and alarms provided in the main control room. High activity in addition to instrument malfunction is alarmed in the main control room. Each monitor has a local indicator at the detector location to aid in calibration of the equipment. Any control functions associated with the monitors are described below and also outlined in Tables 11.4-1 and 11.4-2. Figures 11.4-1 through 11.4-4 are schematic drawings of the radiation monitor system.

11.4.2.1. Gaseous Systems

11.4.2.1.1. Containment Air Particulate, Radioiodine, and Radiogas Monitors

There are two containment air activity monitoring systems per unit, each drawing sample air from four locations in the primary containment. Each containment air activity monitoring system consists of an air particulate monitor, a radiogas monitor, and a radioiodine monitor.

One system takes a composite sample from the area adjacent to RC pump P1A1 seal, the area adjacent to P1A2 seal, the area near the top of steam generator 1SG1A, and the reactor upper cavity. The other system takes a combined sample from the area adjacent to RC pump P1B1 seal, the area adjacent to P1B2 seal, the area near the top of steam generator 1SG1B, and the cavity under the reactor vessel. Individual sample lines may be isolated from the control room in order to assist in identifying leakage.

There is a cross-connection line with a valve (normally closed) which joins the two detector assemblies and will enable either assembly to monitor the areas served by both, if one monitor experiences instrument malfunction.

11.4.2.1.1.1. Primary Containment Air Particulate Monitor

The composite sample is continuously drawn from the containment atmosphere to determine the airborne particulate activity levels within the containment. The sample is pumped to a filter paper assembly located outside the containment where the sample is collected and viewed by a detector. The filter paper, which is viewed by the detector or perhaps a fixed filter in the system may, upon replacement be analyzed in the radiochemistry laboratory. The containment air particulate monitor also has a leakage detection capability and is further described in section 5.2.7.

11.4.2.1.1.2. Primary Containment Radioactive Gas Monitor

The sample effluent from the air particulate monitor is used as the influent for this monitor. The sample is constantly mixed in a fixed shielded volume where it is viewed by the detector and ultimately returned to the containment. In the accident situation, the containment radioactive gas monitors will isolate the purge system as described in section 11.4.1. The reactor coolant leakage detection capability of this monitor is described in section 5.2.7.

11.4.2.1.1.3. Primary Containment Radioiodine Monitor

The air streams pumped from the containment atmosphere are also monitored for radioiodine. One of two designs will be used: (1) part of the containment air pumped from the containment for each of the particulate, radioiodine and radiogas detector assemblies will be directed through the radioiodine assembly in parallel with the flow through the particulate and radiogas detector assemblies and will then rejoin the effluent from these detector assemblies before return to the containment, or (2) flow through the radioiodine detector assembly will be in series with the flow through the particulate and radiogas detector assemblies. The former possibility is schematically shown on Figure 11.4-2.

A detector views radioiodine removed from the airflow by a charcoal adsorber. The charcoal adsorbers are replaced periodically, and the exposed cartridges can be analyzed in the radiochemical laboratory.

11.4.2.1.2. Station Vent Gas Monitors

There are three vents from which plant gaseous releases emanate, one located atop the secondary containment of Unit 1, one located atop the secondary containment of Unit 2, and one located atop the turbine building. Each vent has two monitoring systems (one redundant), which continuously sample the gaseous activity passing through the vent to the atmosphere. Both station vent 1 monitoring systems, both station vent 2 monitoring systems, and both turbine building exhaust vent monitoring systems consist of air particulate, radioiodine, and radioactive gas detectors.

A sample is pumped from a nozzle (or nozzles) in the exhaust duct to the cabinet containing the particulate, radioiodine, and radiogas detection assemblies. The detection assemblies and the arrangement and operation are identical to those of the containment particulate, radioiodine, and radiogas systems. The recorded activity levels, in conjunction with the recorded flow rates out of the vents, provide a record of gaseous activity release. Isotopic compositions of activity collected by the filter paper and charcoal adsorbers are determined at intervals consistent with Safety Guide 21.

11.4.2.1.3. Condenser Vacuum Pump Exhaust Gas Monitor

A continuous sample is drawn from the condenser vacuum pump air exhaust into a shielded compartment and monitored by a detector for gaseous activity indicative of a primary to secondary system leakage. There are two monitoring systems for the 2-unit plant. One system monitors gases from the Unit 1 condenser vacuum pump exhausts while the other system monitors the discharge from the Unit 2 condenser vacuum pumps.

11.4.2.1.4. Waste Disposal System Gas Effluent Monitor

This instrumentation monitors the radiogas activity in the effluent from the waste gas decay tanks. Upon receipt of a high radiation signal, the valve on the discharge line is automatically closed. One radiogas monitor is provided in the discharge line and is capable of monitoring the effluent from either tank.

11.4.2.1.5. Main Control Room Inlet Air Monitoring

The main control room inlet air monitors detect radiogas activity in the air that is being drawn into the main control room from the outside atmosphere. Redundant off-line monitoring systems which employ beta-sensitive detectors are provided for the intake duct. Upon receipt of a high-radiation signal, either control room inlet air monitoring system automatically isolates the control room from the normal airflow and also actuates one of the emergency pressurizing fans. (The pressurizing air fans draw air through one of two HEPA filter charcoal adsorber trains into the control room. The control room air is continuously recirculated through this cleanup system.)

11.4.2.1.6. Containment Purge Exhaust Radiogas Monitor

These monitoring systems aid in ensuring that pressure relief through the purge exhaust ductwork does not occur in the event that a LOCA condition exists coincident with containment purging (as described in section 11.4.1). There are two containment purge exhaust radiogas monitoring systems per unit. Upon receipt of high-radiogas activities either monitoring system will isolate the entire unit purge system.

11.4.2.2. Liquid Systems

11.4.2.2.1. Component Cooling Liquid Monitor

This instrumentation continuously monitors the component cooling system for activity indicative of a leak from one of the systems cooled by the component cooling water. There are two component cooling liquid monitors per unit. One monitor is located downstream of the train A component cooler, and the other is located downstream of the train B cooler. Since the systems which the component cooling system services include the reactor coolant system (via the decay heat removal system during shutdown), these monitors also serve as reactor coolant pressure boundary leakage detection devices. Their function in this respect is described in section 5.2.7. The component cooling liquid monitors also function in the accident situation by detecting any recirculation water leakage from the DHR system.

11.4.2.2.2. Waste Disposal System Liquid Effluent Monitor

This system monitors the waste disposal system liquids that are released from the plant. Minimum cooling tower blowdown flow rate available for liquid waste dilution and compliance with Proposed 10 CFR 50 Appendix I are considerations in establishing a high radiation setpoint for this monitor. Upon a high radiation indication, the valve on the liquid waste discharge line is automatically closed.

11.4.2.2.3. Essential Raw Cooling Water Discharge Header Liquid Monitor

The flow through each of the two ERCW plant discharge headers is monitored in order to detect leakage of radioactive fluids into the system. Except in the accident condition, the presence of activity in the ERCW system can occur only when two barriers are breeched (e.g., a letdown heat exchanger and a component cooling heat exchanger). Under accident conditions the ERCW system may become contaminated via leakage in a component cooling heat exchanger or as a result of inleakage from the containment atmosphere into the reactor building coolers.

11.4.2.2.4. Reactor Building Cooler ERCW Discharge Monitor

This monitor (one per unit) takes a common sample from the three reactor building cooler ERCW discharge lines. In the loss-of-coolant-accident situation, this monitor detects any leakage from the containment atmosphere into the ERCW system. If activity is detected, a valving arrangement allows sampling of each line individually to determine the leaking cooler.

11.4.2.2.5. Boric Acid Evaporator Distillate Monitor

This monitor (two per plant) samples the distillate flow during evaporator operation. If high activity indicative of a decrease in evaporator efficiency is detected, the operator may stop the feed flow to the evaporator or take other action to avoid a high activity inventory in the distillate test tank.

11.4.2.2.6. Reactor Coolant Letdown Liquid Monitor

The reactor coolant letdown liquid monitor allows determination of the gross activity in the reactor coolant system. These monitors (one per unit) are located in the auxiliary building upstream of the purification components on the letdown lines.

11.4.2.2.7. Decay Heat Removal System Monitor

A decay heat removal system monitor is mounted on the outside of the effluent pipe from each DHR cooler (two coolers per unit). This instrumentation monitors the activity in the system during shutdown and during recirculation of containment emergency cooling water following a loss-of-coolant accident. Additional information regarding accident severity is thus obtained.

11.4.2.2.8. Plant Liquid Effluent Monitor

This monitor samples the plant liquid discharge just prior to release to the river. The total flow (cooling tower blowdown plus any liquid wastes being discharged) is monitored with this detector to ensure that the liquid release limits as set forth in the Proposed Appendix I of 10 CFR 50 are not violated. This monitor provides backup for the waste disposal system liquid effluent monitor.

11.4.2.2.9. Holding Pond Discharge Monitor

This monitor samples the liquid flowing from the holding pond to the river. The holding pond is a part of the yard drainage system (described in section 9.3.3) and normally contains extremely low levels of radioactivity. The source of any activity in the holding pond will normally be turbine building sump water. This water, which is drainage from all turbine building areas except the condensate demineralizer area, may contain low level activity resulting from steam line and feedwater line leakage when primary to secondary leakage has occurred.

11.4.3. Sampling

The points subject to periodic sampling are all liquid and gaseous releases to the environment. All analytical radiation sampling instrumentation, sampling frequency and procedures meet the AEC Safety Guide 21 standards. Tables 11.4-3 and 11.4-4 list the operation characteristics (sampling procedure, frequency, type of analysis, etc.) and the effects on plant operation of each gaseous and liquid sample. The plant liquid effluent integrated sample meets the Safety Guide 21 and Proposed Appendix I, 10 CFR 50 guidelines.

11.4.4. Calibration and Maintenance

The calibration and maintenance procedures for the process monitors are described below.

Each detector has a built-in check source which is remotely actuated from the main control room. Each detector is checked daily using its built-in check source. A monthly response check will be performed on each monitor. The response check includes:

1. Check control room ratemeter and recorder response to one activity level from a portable calibration source which will trip the ratemeter setpoint.

2. Trip the upscale (by calibration source) and downscale ratemeter setpoints and check the respective annunciation (high radiation or instrument malfunction) and interlock functions.

A calibration check will be performed quarterly on each monitor. The calibration procedure includes:

1. Recalibration of each detector using a portable calibration unit.

2. Electronically recalibrate the control room ratemeters and recorders.

3. Verify that "instrument malfunction" annunciation is initiated on downscale ratemeter trip, loss of power, low monitor flow, etc.

4. Verify that "high radiation" annunciation and interlock functions are initiated on upscale ratemeter trip.

11.4.4.1. Inspection of Radiological Monitoring Instruments

Process and effluent radiological monitoring instruments performing a continuous monitoring function will be inspected and tested periodically according to predetermined instructions to assure proper operation. Initial calibration of these instruments will be performed before plant startup, and their calibrations will be periodically verified and documented according to predetermined maintenance instructions. Instrument maintenance, as required, will be performed and documented according to predetermined maintenance instructions. An independent verification of the adequacy and accuracy of the instrument calibrations will be made periodically by the Plant Engineering Branch.

11.4.4.2. Testing and Inspection of Laboratory Instruments

Periodic tests and inspections to assure proper operation will be conducted on the laboratory instruments used to analyze radiological effluents. These tests and inspections will be performed and documented according to predetermined instructions and schedules. The calibrations of these laboratory instruments will be verified and documented at periodic intervals according to predetermined calibration instructions. Maintenance, as required, will be performed according to appropriate manufacturer instructions and documented. An independent verification of the adequacy and accuracy of the instrument calibrations will be made by the Plant Engineering Branch.

Table 11.4-1. Gaseous Monitoring Systems

TVA instrument No.	Function	Automatic actions taken on high radiation indication	Detector location	Principal isotopes
1-RE-90-9, -12 2-RE-90-9, -12	Primary containment air monitor	Primary containment purge exhaust inlet and outlet valves are closed	Outside primary containment	
	Particulate			^{131}I, ^{137}Cs, ^{60}Co
	Gas			^{85}Kr, ^{133}Xe
	Iodine			^{131}I
1-RE-90-3, -10 2-RE-90-6, -11	Station vent monitor	None	Auxiliary building	
	Particulate			^{131}I, ^{137}Cs, ^{60}Co
	Gas			^{85}Kr, ^{133}Xe
	Iodine			^{131}I
0-RE-90-7, -13	Turbine building exhaust vent monitor	None	Turbine building	
	Particulate			^{131}I, ^{137}Cs, ^{60}Co
	Gas			^{133}Xe, ^{85}Kr
	Iodine			^{131}I
1-RE-90-5 2-RE-90-5	Condenser vacuum pump air exhaust monitor	None	Turbine building	^{85}Kr, ^{133}Xe
0-RE-90-8	Waste disposal system gas effluent monitor	Valve is closed in discharge line of waste gas decay tank	Auxiliary building	^{85}Kr, ^{133}Xe

Table 11.4-1. (Cont'd)

TVA instrument No.	Function	Automatic actions taken on high radiation indication	Detector location	Principal isotopes
0-RE-90-4 0-RE-90-24	Main control room inlet air monitor	Control room isolated from normal air supply and emergency pressurizing fan activated	Control room	^{85}Kr, ^{133}Xe
1-RE-90-1, -2 2-RE-90-1, -2	Containment purge exhaust radiogas monitor	Purge inlet and outlet valves are closed	Auxiliary building	^{85}Kr, ^{133}Xe

Table 11.4-2. Liquid Monitoring Systems

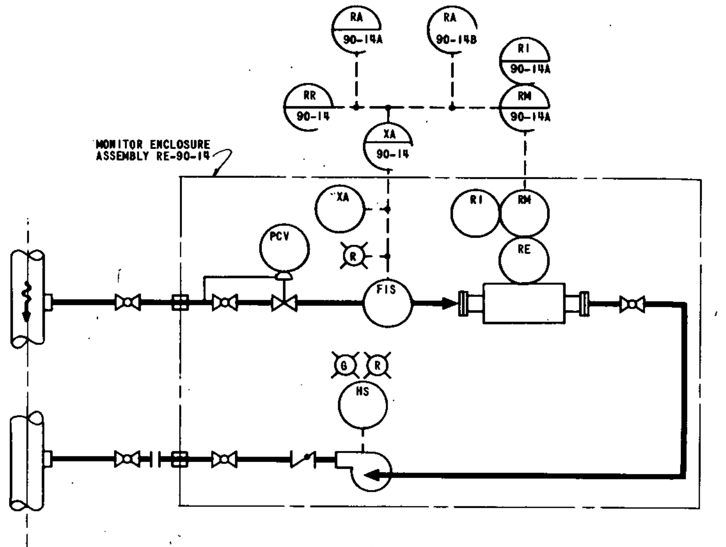
TVA instrument No.	Function	Automatic actions taken on high radiation indication	Detector location	Principal isotopes
1-RE-90-18, -19 2-RE-90-18, -19	Component cooling system liquid monitor	None	Auxiliary building	^{60}Co, ^{137}Cs, ^{131}I
0-RE-90-17	Waste disposal system liquid effluent monitor	Valve is closed in WDS discharge line	Auxiliary building	^{60}Co, ^{137}Cs, ^{131}I
0-RE-90-15 0-RE-90-16	Essential raw cooling water discharge header monitor	None	Diesel generator building	^{60}Co, ^{137}Cs, ^{131}I
1-RE-90-22 2-RE-90-22	Boric acid evaporator distillate monitor	None	Auxiliary building	^{60}Co, ^{137}Cs, ^{131}I
1-RE-90-14 2-RE-90-14	Reactor coolant letdown liquid monitor	None	Auxiliary building	^{58}Co, ^{60}Co, ^{137}Cs, ^{131}I
1-RE-90-33, -34 2-RE-90-33, -34	Decay heat removal system monitor	None	Auxiliary building	^{58}Co, ^{60}Co, ^{137}Cs, ^{131}I
1-RE-90-20 2-RE-90-20	Reactor building coolers ERCW discharge monitor	None	Auxiliary building	^{133}Xe, ^{85}Kr
0-RE-90-27	Plant liquid effluent monitor	None	Outside	^{60}Co, ^{137}Cs, ^{131}I
0-RE-90-25	Holding pond discharge monitor	None	Outside	^{137}Cs, ^{131}I

Table 11.4-3. Liquid Radiation Sample Points

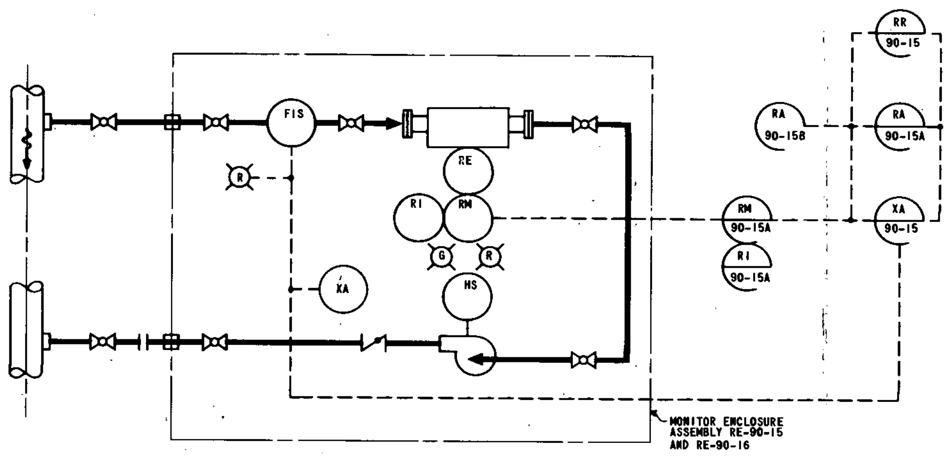
Sample location	Type of Analysis	Frequency	Procedure	Sample method	Effects on plant operations
Waste disposal system liquid effluent	pH	Each batch	pH meter	Grab	High activity (1) WDS discharge valve closes. (2) WDS discharge is recycled and reprocessed or discharge is held until it meets release requirements. (3) Check sample analysis and monitor calibration.
	Gross β, γ activity	Each batch	GM counting system	Grab	
	Dissolved fission gases and activation gases	One batch/2 weeks	Gamma spectrometer	Grab	
	^{140}Ba - ^{140}La, ^{131}I activity	Weekly composite	Well counting system	Grab	
	γ emitting fission & activation products	Monthly composite	Gamma spectrometer	Grab	
	^{3}H activity	Monthly composite	All counting systems	Grab	
	^{89}Sr activity	Monthly composite	All counting systems	Grab	
	Gross α activity	Monthly composite	All counting systems	Grab	
	^{90}Sr activity	Quarterly composite	All counting systems	Grab	
Plant liquid effluent (Downstream of dilution of liquid waste with cooling tower blowdown)	pH	1 to 2 weeks	pH meter	Grab	Same as above.
	Gross β, γ activity	1 to 2 weeks	GM counting system	Grab	
Holding pond	pH	Before any possible release	pH meter	Grab	Same as above.
	Gross β, γ activity	Before any possible release	GM counting system	Grab	

Table 11.4-4. Gaseous Radiation Sample Points

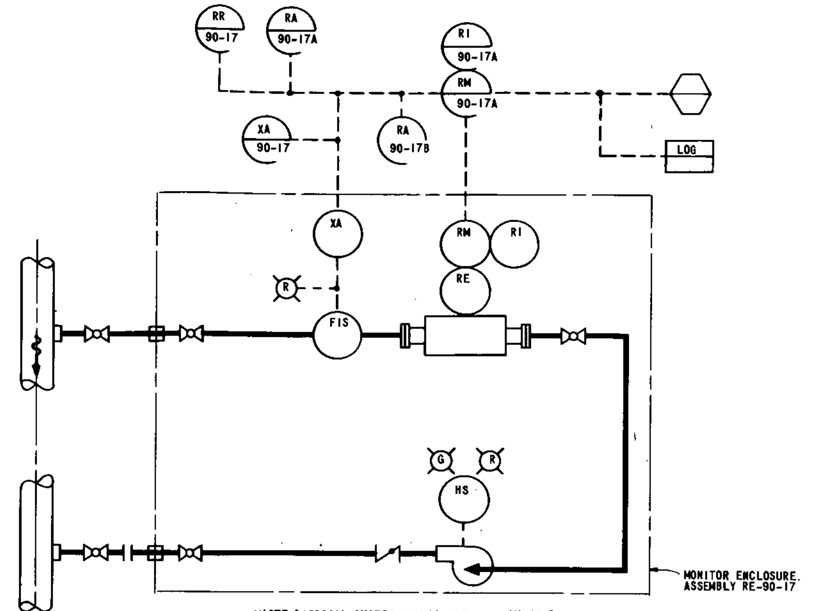
Sample location	Type of analysis	Frequency	Procedure	Sample method	Effects on Plant operation
Gas decay tanks	Noble gases isotopic analysis	Each batch release	Gamma spectrometer	Grab	High activity (1) Release automatically terminated. (2) Holdup time increased so release will meet allowable limits.
	^{3}H concentration	Each batch release	Liquid scintillation system	Grab	
Plant exhaust monitors (1) Station vent unit 1 (2) Station vent unit 2 (3) Turbine Building Exhaust (4) Condenser Vacuum pump Air exhaust	Noble gas conc	Monthly calibration	Well counting system	Samples are sample filter cartridges and sample cylinders removed from monitors or from monitored flow and taken to lab for check.	High activity (1) Initiates investigation of source. (2) Starts examination of Tech Spec limits on reactor coolant leakage.
	^{131}I activity	Weekly/each monitor	Gamma spectrometer		
	^{131}I, ^{135}I activity	Quarterly on weekly sample/each monitor	Gamma spectrometer		
	Gross β, γ activity	Weekly/each monitor	GM counting system and gamma spectrometer		
	^{140}Ba - ^{140}La, ^{131}I	Monthly composite (on weekly samples/each monitor)	Gamma spectrometer		
	Principle gamma emitting nuclides	Weekly/each monitor	Gamma spectrometer		
	^{89}Sr, ^{90}Sr activity	Quarterly on a monthly composite sample/each monitor	Proportion counting system		
	Gross α activity	Quarterly on a weekly sample/each monitor	α counting system		



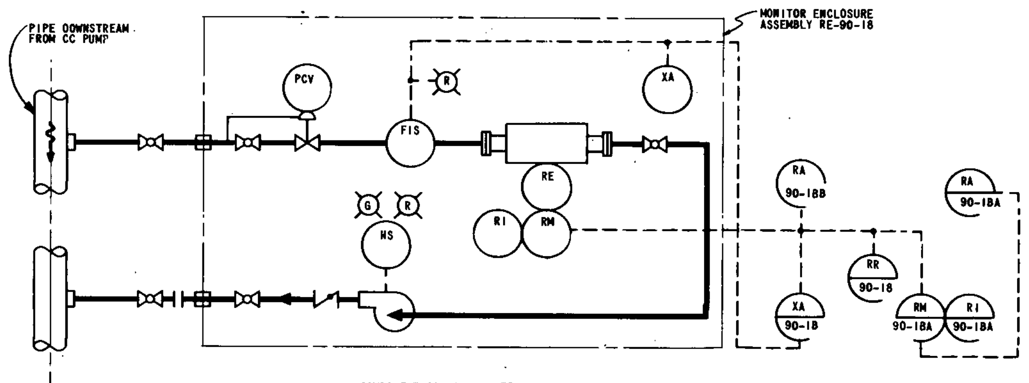
REACTOR COOLANT LETDOWN LIQUID MONITOR
1 DETECTOR PER UNIT, 2 DETECTORS PER PLANT



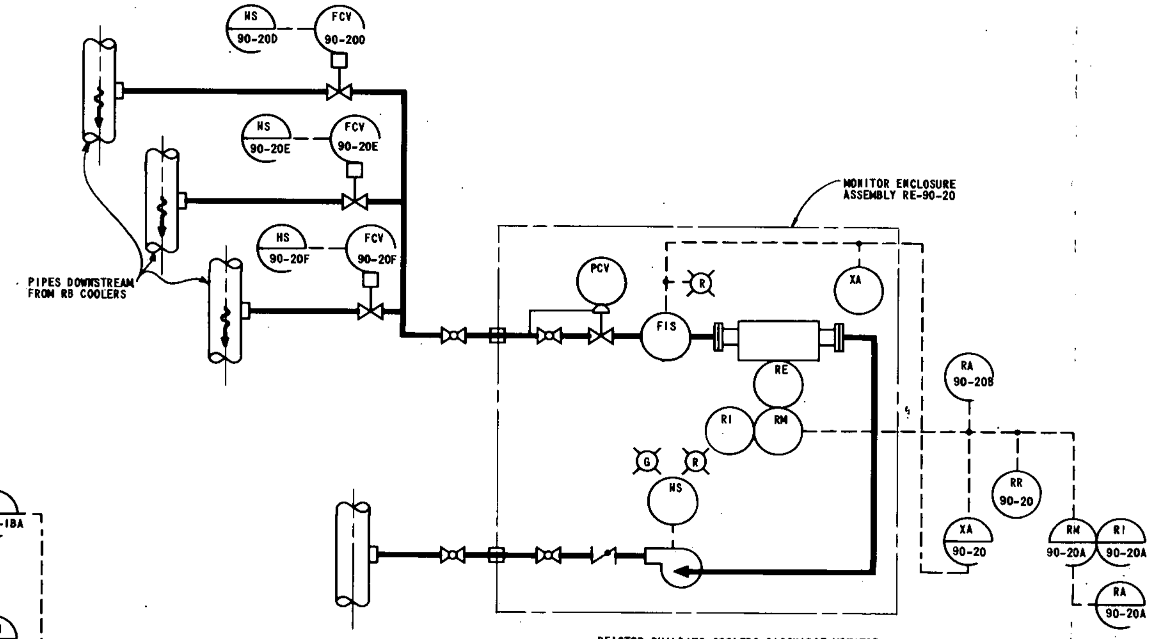
ESSENTIAL RAW COOLING WATER LIQUID MONITOR
2 MONITORS REQUIRED PER 2 UNIT PLANT, BOTH COMMON
ONE SHOWN



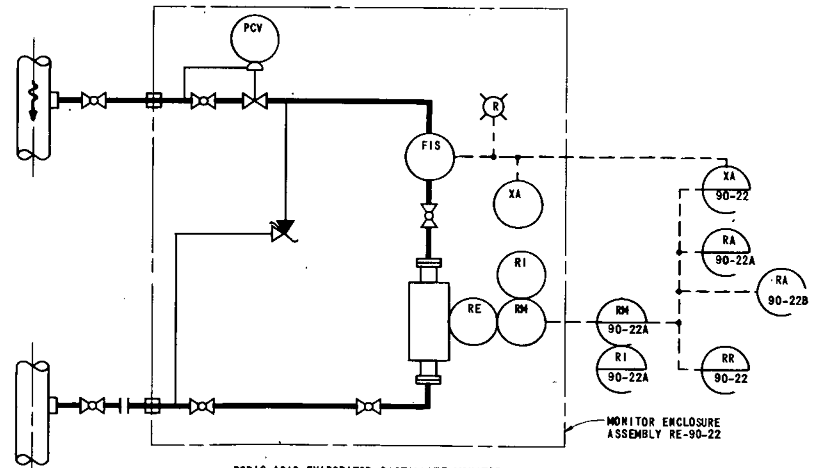
WASTE DISPOSAL SYSTEM LIQUID EFFLUENT MONITOR
AS SHOWN FOR 2 UNIT PLANT



COMPONENT COOLING SYSTEM LIQUID MONITOR
1 MONITOR PER TRAIN
RE-90-18 TRAIN A, RE-90-19 TRAIN B
4 DETECTORS PER PLANT



REACTOR BUILDING COOLERS DISCHARGE MONITOR
ONE MONITOR PER UNIT



BORIC ACID EVAPORATOR DISTILLATE MONITOR
AS SHOWN FOR UNIT 1, UNIT 2 SAME

NOTE:
SEE NOTES ON DRAWING NUMBER 47W610-90-1.

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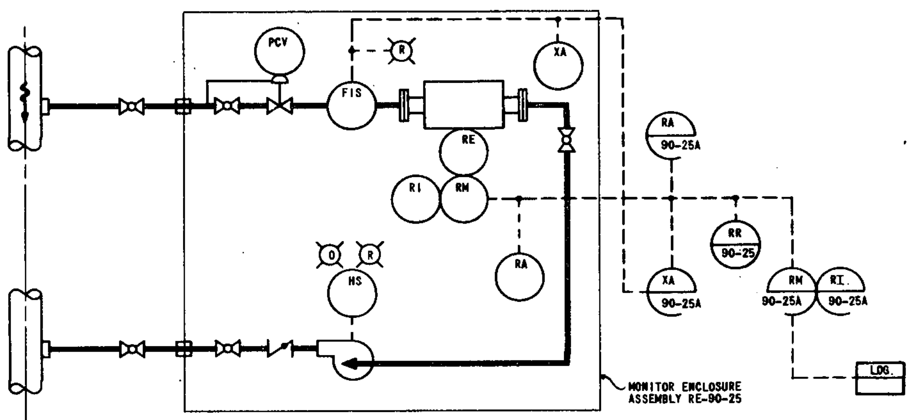
RADIATION MONITORING SYSTEM
CONTROL DIAGRAM FOR ACTIVITIES
CARRIED BY LIQUID FLOWS

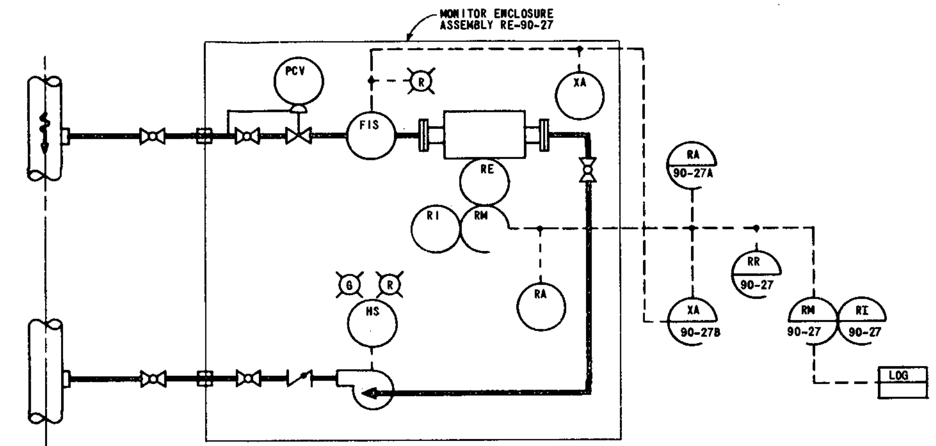
SHEET 1 OF 2

FIGURE 11.4-1

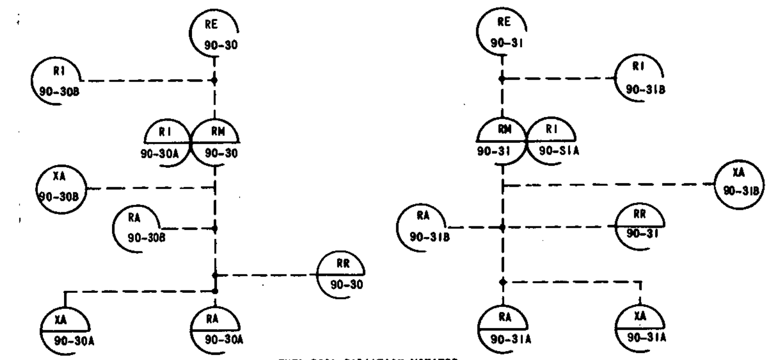
TVA DWG.NO. 47W610-90-3 R0



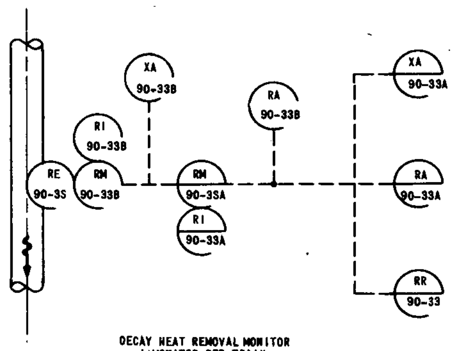
HOLDING POND DISCHARGE MONITOR
1 DETECTOR PER 2 UNIT PLANT



PLANT LIQUID EFFLUENT MONITOR
1 DETECTOR PER 2 UNIT PLANT



FUEL POOL RADIATION MONITOR
AS SHOWN FOR 2 UNIT PLANT



DECAY HEAT REMOVAL MONITOR
1 MONITOR PER TRAIN
RE-90-33 TRAIN A, RE-90-34 TRAIN B
4 MONITORS PER PLANT

NOTE:
SEE NOTES ON DRAWING NUMBER 47W610-90-1.

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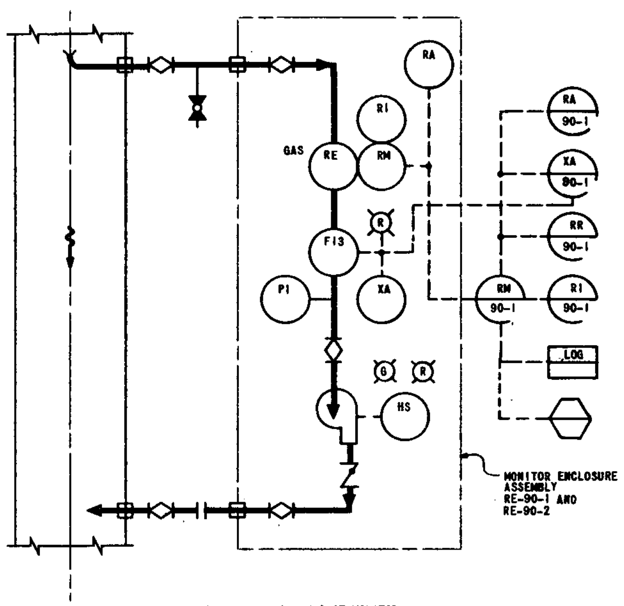
RADIATION MONITORING SYSTEM
CONTROL DIAGRAM FOR ACTIVITIES
CARRIED BY LIQUID FLOWS

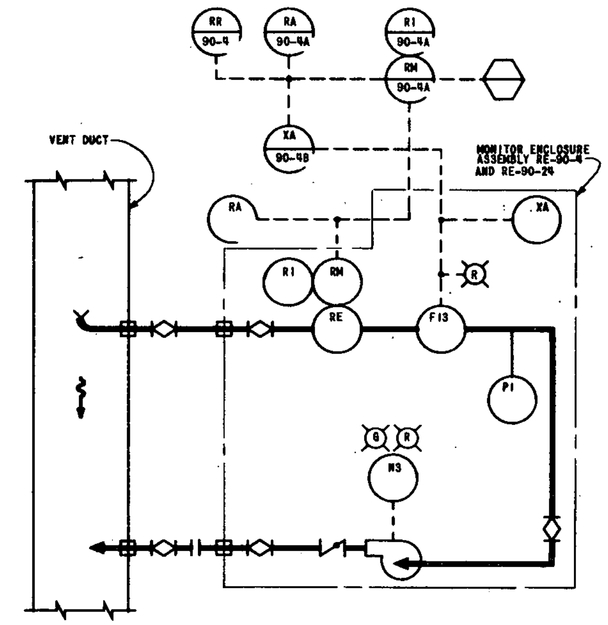
SHEET 2 OF 2

FIGURE 11.4-2

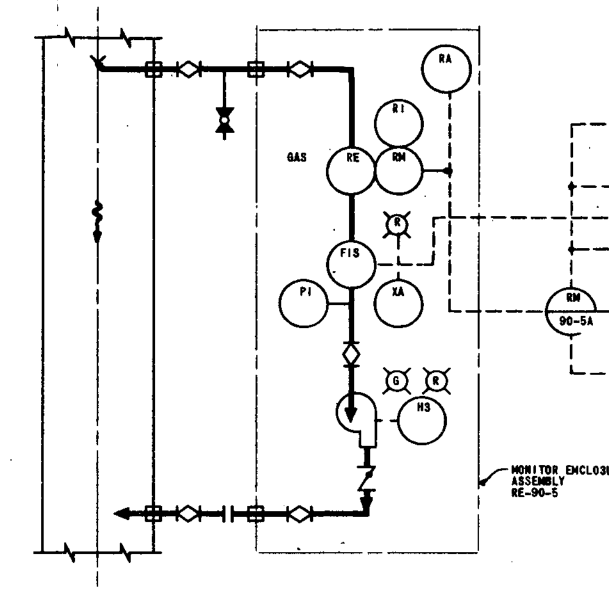
TVA DWG.NO. 47W610-90-4 R0



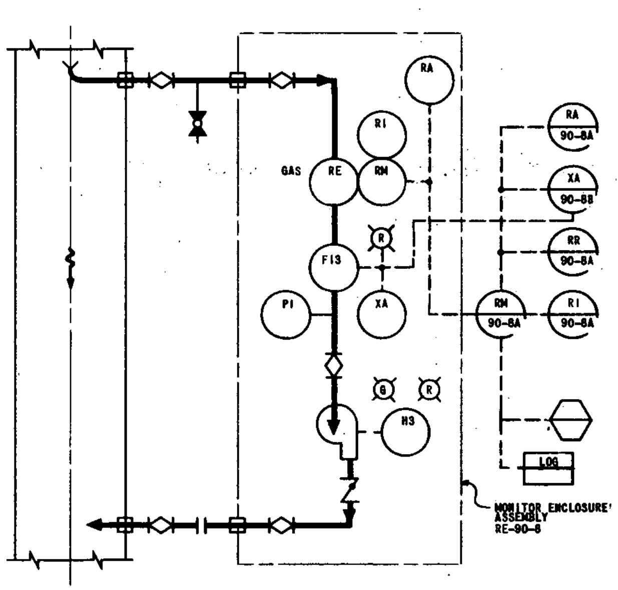
CONTAINMENT PURGE EXHAUST MONITOR
TWO MONITORS PER UNIT



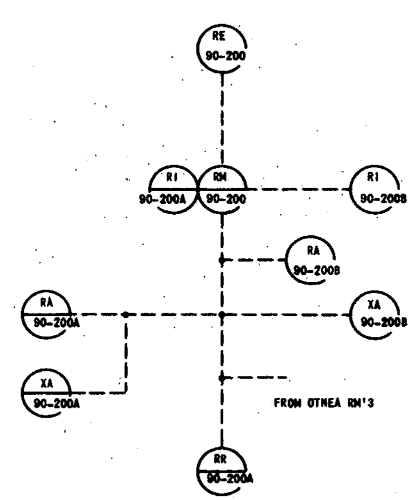
MAIN CONTROL ROOM INLET AIR MONITOR
2 MONITORS FOR 2 UNIT PLANT



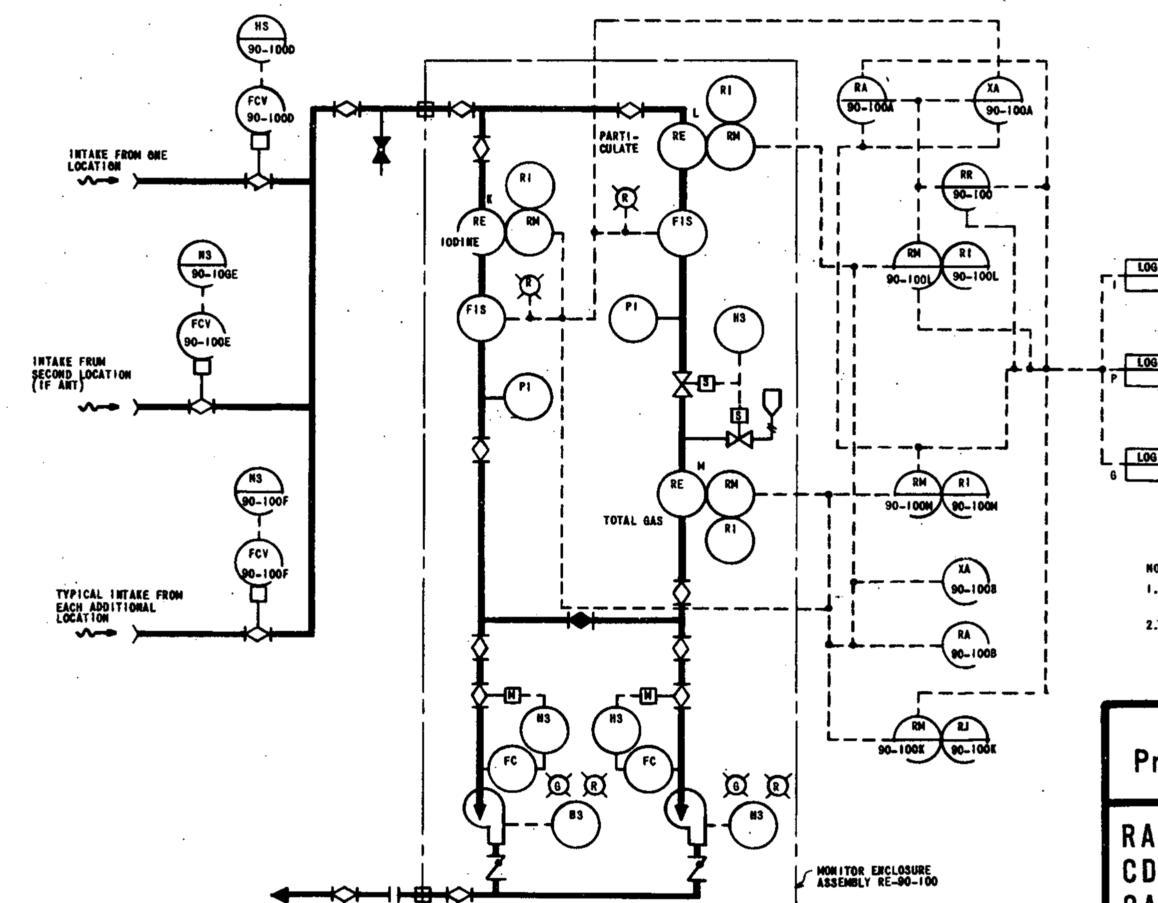
CONDENSER VACUUM PUMP AIR EXHAUST MONITOR
AS SHOWN FOR UNIT 1, UNIT 2 SAME



WASTE DISPOSAL SYSTEM GAS EFFLUENT MONITOR
AS SHOWN FOR 2 UNIT PLANT



TYPICAL AREA RADIATION MONITORING SUBSYSTEM



TYPICAL AIRBORNE MONITORING SUBSYSTEM

NOTES:

1. THE INSTRUMENTS SHOWN IN EACH MONITOR ENCLOSURE ASSEMBLY IS A REPRESENTATION OF THE INSTRUMENTATION THE CONTRACTOR WILL FURNISH.
2. WHERE A RA AND XA ALARM FUNCTION IS SHOWN. THE RA IS A HIGH RADIATION ALARM AND THE XA IS AN EQUIPMENT MALFUNCTION (LOSS OF POWER, FLOW, ETC.) ALARM.

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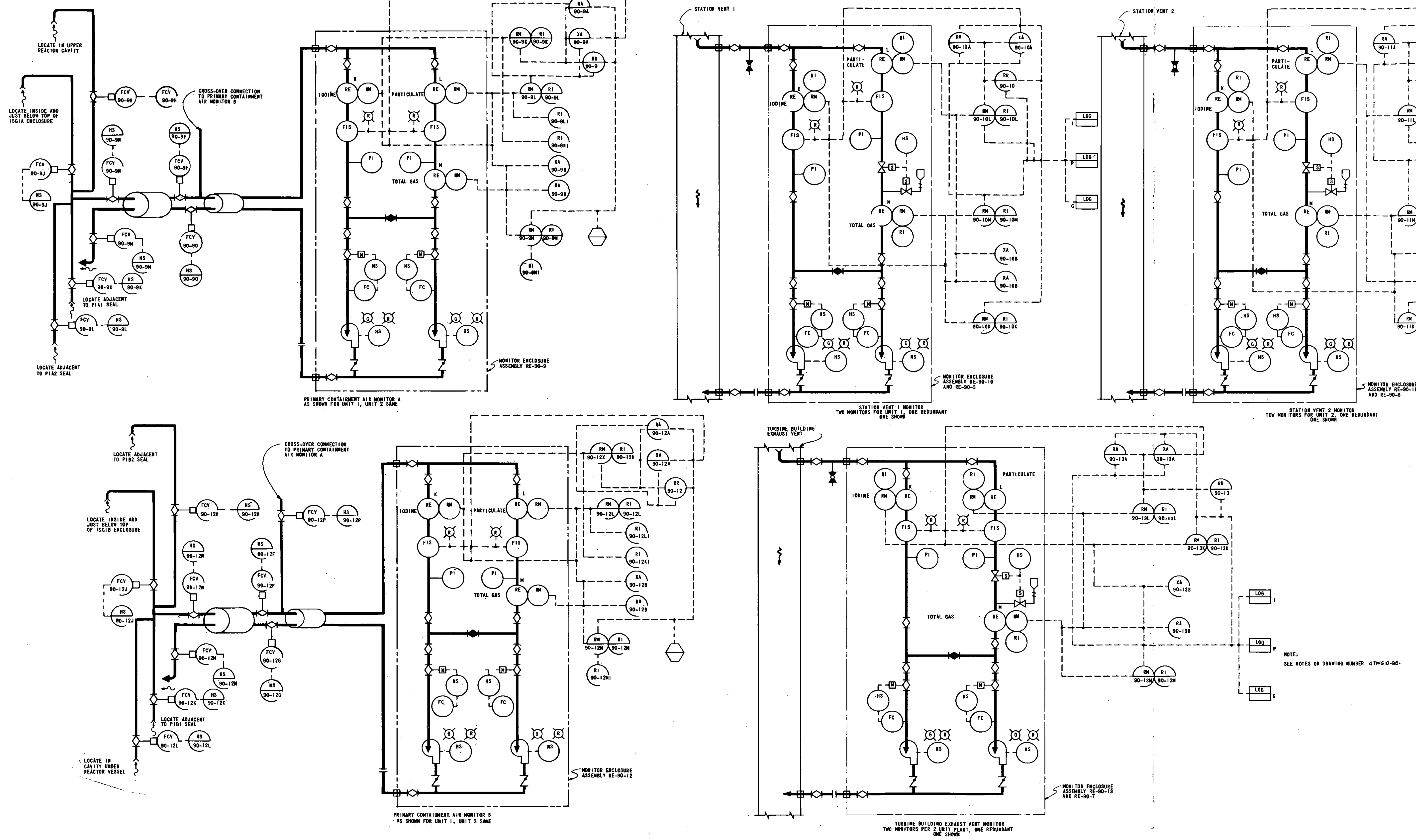
RADIATION MONITORING SYSTEM
CONTROL DIAGRAM FOR ACTIVITIES
CARRED BY GASEOUS FLOWS

SHEET 1 OF 2

FIGURE 11.4-3

TVA DWG.NO. 47W610-90-1 R0



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RADIATION MONITORING SYSTEM
CONTROL DIAGRAM FOR ACTIVITIES
CARRIED BY GASEOUS FLOWS

SHEET 2 OF 2

FIGURE 11.4-4

TVA DWG.NO. 47W610-90-2 R0

11.5. Solid Waste System

11.5.1. Design Objectives

The sources of solid wastes are spent resins, waste evaporator concentrate, auxiliary waste evaporator concentrate, and miscellaneous solids made up of filter elements, paper, rags, plastic sheeting, glass, and contaminated equipment and maintenance parts. The solid waste disposal facilities are designed to operate as required by the need to package radioactive waste. The rate of accumulation varies, with the maximum occurring during refueling periods and the minimum during normal operation. The system is designed so that shipments are in accordance with 10 CFR 20, 10 CFR 71, 10 CFR 73, and Department of Transportation regulations 49 CFR 173 and 49 CFR 178. The volumes and activities of the radioactive wastes generated during each fuel cycle are listed in Tables 11.5-1 and 11.5-2, respectively.

The spent resin storage tank provides for the accumulation of spent resins prior to packaging and also allows for the decay of the short half-life isotopes. The spent resin is sluiced from the demineralizers by circulating water from the storage tank through the demineralizers using the spent resin liquid sluicing pump. This pump is also used to pump the excess water from the resin tank to the tritiated waste holdup tank.

The system that fills, seals, and stores tanks or drums containing solid wastes is designed to protect operating personnel and contain all radioactive material for offsite disposal. The solid waste system will contribute no pollution to the environment.

11.5.2 System Inputs

Table 11.5-1 lists the expected amounts of spent resin, waste evaporator bottoms, and miscellaneous wastes. The sources of solid waste and the storage, processing, and packaging system are shown on the process flow diagram, Figure 11.5-1. The activity of the spent resin and the evaporator bottoms is shown in Table 11.5-2.

11.5.3 Equipment Description

The solid waste handling system comprises several components required for packaging the waste and handling the packaged waste; these components are (1) a conveying system, (2) a filling system, (3) a package sealing system, and (4) a baling system to package common solid wastes (e.g., paper, rags, plastic sheeting, etc.) used during the maintenance and operation of the units. This system is shared by both units.

Concentrates from the waste evaporator and the auxiliary waste evaporator are pumped to the packaging area. A distribution header feeds the concentrates to several containers, either singly or in multiples. The associated instrumentation prevents overfilling the containers; level indicator controllers have semi-automatic control of the flow control valve(s) and vent valve(s), if required, and close these valves when the container contents reach a preset level.

After reaching this predetermined level, the fill connection line and/or equipment is rinsed with water, again a predetermined quantity to complete filling the container. The containers are now disengaged from the filling station(s) and plugged or capped. After capping or plugging, the containers are conveyed to the area for shielding removal (if required) and then conveyed to the spray wash or decontamination station to ensure that the containers' exteriors are free of contamination.

The containers are then conveyed to either the radwaste storage area or loaded in shielded containers on a carrier for offsite disposal. The waste material is mixed with a solidification agent to form a solid at the plant before shipment.

The spent ion exchange resins are pumped to the packaging station from the spent resin storage tank. A distribution header feeds the spent resin to several containers, either singly or in multiples. The associated instrumentation prevents overfilling the containers; level indicator controllers have semi-automatic control of the flow control valve(s) and vent valve(s), if required, and close these valves when the container contents reach a preset level.

After reaching this predetermined level, the fill connection line and/or equipment is rinsed with a predetermined quantity of water to complete filling the container. The containers are then disengaged from the filling station(s) and plugged or capped. After capping or plugging, the containers are conveyed to the area for possible removal of temporary shielding and then conveyed to the spray wash or decontamination station to ensure that the containers' exteriors are free from contamination.

The containers are then conveyed to either the radwaste storage area or loaded on the carrier in shielded containers for offsite disposal. The waste material is mixed with a solidification material to form a solid at the plant before shipment.

Solid wastes generated by maintenance and operation consist of low-level radioactively contaminated material such as rags, paper, plastic sheeting, small hand tools, and expendable or consumable maintenance supplies (e.g., flange gaskets, valve-stem packing, pump packing, disposable filter cartridges, and HEPA filter cells). These material wastes are compacted in the waste packaging station baling unit. These compacted wastes are baled and packaged in suitable containers for transporting and burial offsite. The packaged wastes are accumulated and stored in the radwaste storage area until removed for scheduled shipment to the appropriate disposal site.

The disposable filter cartridges are packaged so that the radiation levels as required by 49 CFR 173 are not exceeded. The mode of packaging the filter cartridges and flange gaskets depends on the radioactivity level of the items. If the radiation levels are within the 10 CFR 20 limits, the filters and gaskets can be packaged with the maintenance and operational wastes. However, should the radiation levels exceed the 49 CFR 178 limits when packaged as described above, the filter cartridges and gaskets would be packaged in containers with solidification materials providing some shielding. Additional shielding would be installed (if required) for transfer to the disposal site.

11.5.4. Expected Activities

The associated curie content and principle nuclides in solid waste to be shipped from the site are given in Table 11.5-2. The activities were determined on the following basis:

1. A constant flow equal to the average daily flow rate at 90% core life (95 gpm) through the makeup and purification system is used. Also used are the maximum nongaseous fission product activities as listed in Table 11.1-4 (based upon 0.25 % failed fuel) and corrosion product activities listed in Table 11.1-6. (In more refined calculations credit can be taken for the following: (1) lower concentrations in the coolant during much of the year (cf. Table 11.1-4) and (2) a lower letdown flow during much of the year.)

2. For determining demineralizer resin activity inventories, the resins are conservatively assumed to remove all of the nongaseous fission product activity in the letdown flow.

3. The nongaseous fission products in the letdown stream are assumed to accumulate on these demineralizers for 6 months, at which time the resins are transferred to a storage location and allowed to decay for 6 months prior to shipment.

4. The resultant inventories on demineralizer resins are increased by a few percent to account for activity in shipped evaporator bottoms and on other demineralizers.

5. For determining corrosion product inventories, all corrosion product activity in the letdown stream is assumed to be removed over the period of 1 year's operation. No credit is taken for decay of corrosion products in the interval between collection on the prefilter for 1 year and shipment.

6. The resultant corrosion product inventories are conservatively increased by 20% to account for shipped evaporator concentrates activity resulting from the processing of reactor coolant system leakage and to account for activity accumulated on spent fuel coolant filters.

11.5.5. Packaging

The primary method of waste packaging is to be a system using standard 30- and 55-gallon drums. Larger containers may also be used. This system is designed to encapsulate liquid radioactive wastes and compressible solid wastes within the limitations specified by Federal Regulations (10 CFR 71 and 49 CFR 170-178) applicable to packaging, handling, and transportation of radioactive materials.

Two processes are employed within the facility. One, for use with solid compressible materials, is a baling process. The other, for use with evaporator concentrates, chemical drain tank effluents, and spent resin, is an encapsulation process.

Solid compressible wastes of low radiation level will be compressed in standard 55-gallon drums (DOT 17H) or other DOT approved containers. After compaction it is closed and transported to the storage area. The drum is then transferred to a commercial vehicle for final disposal at a commercial burial site.

For encapsulating spent resin and evaporator bottoms, 30- and 55-gallon drums, respectively, will be used. The prepared drum packages, filled with a blend of vermiculite and cement, are placed in the drumming room at the filling positions. If required by the activity level of the material to be dispensed, the drums are enclosed in steel-jacketed lead shields. At their respective fillings stations, the drum packages (which already contain an inner injector assembly) are connected to the dispensing manifold.

Process preparation for each drum consists of evacuating the drum package to a prescribed level. Once the desired vacuum is achieved, the package is checked for leakage. If there are no component malfunction indications, the filling process can begin.

The multiple and simultaneous filling process is initiated by the operator at the control panel. Termination of the process is effected automatically. Piping connections are manually removed from the drum package and the packages are moved by an overhead crane to the shield stripping area. The evaporator bottom shields are removed and the drums are stored in the accumulation area. The 30-gallon drum may be stored with the spent resin shield in place. This shield fits around the 30-gallon drum in such a manner that the temporary shield of the type used for the evaporator bottoms drums will fit around the outside of it for additional shielding. The spent resin shield has been granted DOT special permit No. 6405 for the transport of type B quantities of radioactive materials in special form.

Radioactive plant filter cartridges will also be handled in the DOT 17H drums. The filter cartridge elements will normally be removed in a cluster from the housing and placed in a drum which contains cement. Filters with a high activity level will be placed in the waste package storage area for a suitable period before shipping to allow for decay.

11.5.6. Storage Facilites

The solid waste container storage area is located adjacent to the waste packaging area at floor El 649 of the auxiliary building, see Figure 11.5-2. The usable storage area is approximately 45 x 25 ft with a storage area equivalent to 110 55-gallon drums (unstacked).

The entire floor area is serviced by a 6-ton overhead crane. The crane is equipped with an automatic lifting device for remote handling of containers.

Shielded containers are moved from the packaging area to the temporary shield removal area using the overhead crane and a spreader bar. After the temporary shield has been removed, the container is moved to a selected storage position using the overhead crane and the automatic mechanical lifting device.

Assuming that the contents of each container consists of a mixture of 50% solidification reagents and 50% solid waste, each container can be stored for a period of 6 months or longer. Based on the most conservative sources, the activity of a container of radioactive waste will decrease approximately 20% during a 6-month storage period.

Containers designated for offsite shipment are moved from the storage area to a conveyor using the overhead crane and the automatic lifting device. The containers are then moved into the adjacent radioactive waste shipping area with the conveyor.

The shipping and receiving area is shown in Figure 11.5-2. The trailer unit with the shipping shield or container installed is backed into the shipping area. The area is serviced with a 6-ton overhead crane. The shipping container cover is removed and placed in a designated area with the overhead crane. Waste containers are loaded from the conveyor into the shipping container using the overhead crane, equipped with an automatic lifting device. The crane is controlled from the floor above the shipping area. The operation may be viewed by the operator through the hatch located in the floor above the shipping area. Mirrors may be used, if required, to avoid radiation exposure to operating personnel. After the shipping container cover is replaced, the shipping container is checked for surface contamination. If any surface contamination is detected, it is removed using appropriate cleaning methods.

The shipping container cover is bolted in place and additional tiedowns are installed as required. The trailer transport unit is then moved from the shipping area.

11.5.7 Shipment

The 30- and 55-gallon drums are both qualified to transport type A quantities of special or normal form materials in accordance with 49 CFR paragraphs 173.394 and 173.395. Normally, the activity of material in the 55-gallon evaporator bottom drums will not exceed type A quantities as specified in 49 CFR 173.389. This is also the case for the 30-gallon spent resin drums. Should the 30-gallon drum contents exceed the type A quantity limits, the spent resin shield can be left in place for transportation as it is qualified for transport of type B quantities of radioactive materials in special form. For any of these cases, the drums will be transported from the plant to the burial facility in a truck shielded such that the dose rates do not exceed those listed in 49 CFR paragraph 173.393. If additional shielding should be required, these drums would be transported in a cask similar to the all-steel low-level cask to be used at the Browns Ferry Nuclear Plant.

Compacted wastes will be of an extremely low activity level and will be transported by truck to the burial site.

Table 11.5-1. Radioactive Wastes — Expected Volumes

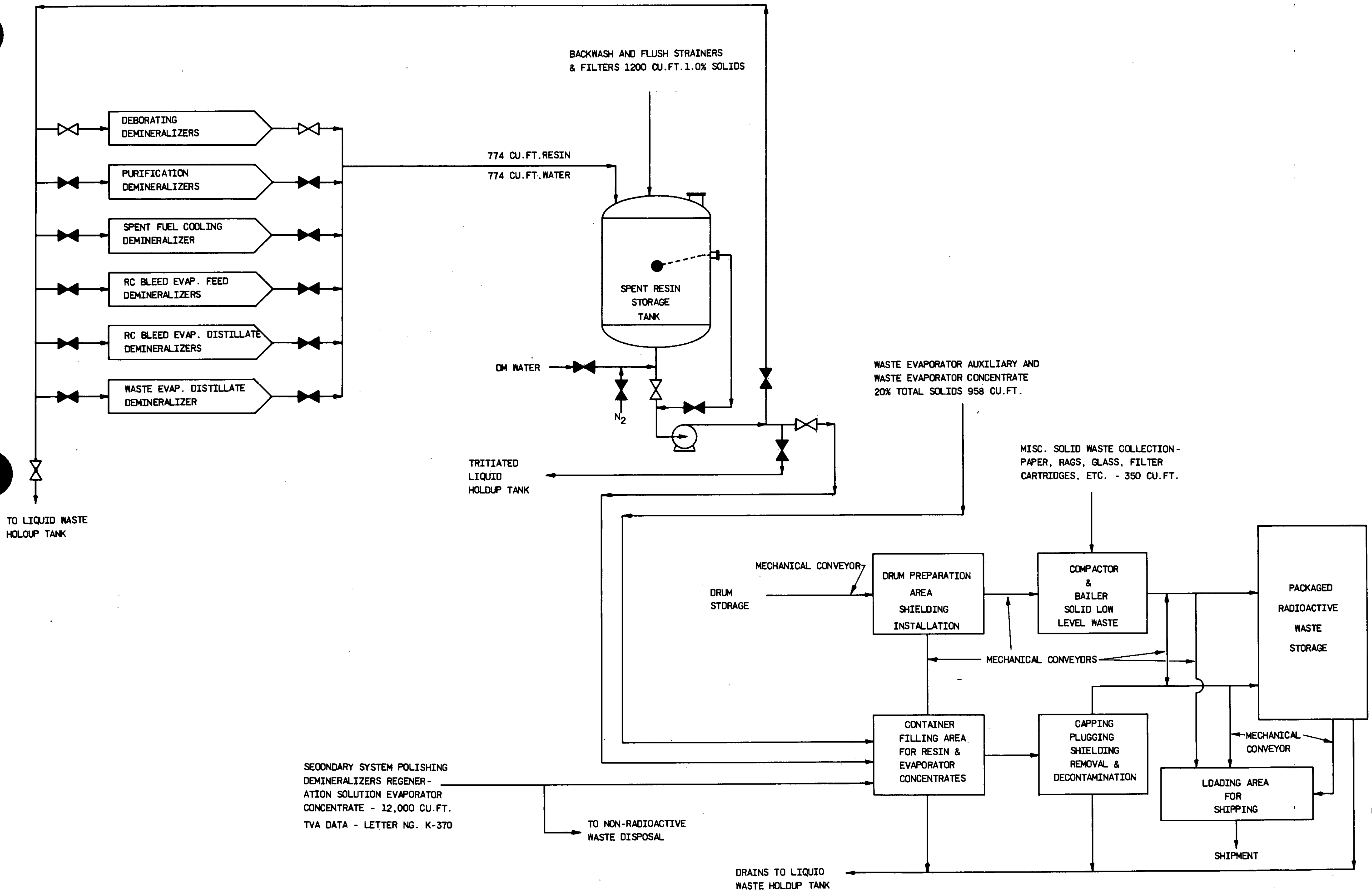
	Volume, ft^3/yr
Spent resin (1.0 ft^3 water/ft^3 resin)	1,550
Waste evaporator bottoms	960
Miscellaneous solids --filter cartridges, paper, glassware, rags, equipment	350
Spent HEPA and charcoal filters	2,100
Total	4,960
Secondary system--auxiliary evaporator, condensate polishing demineralizer regeneration solution, evaporator bottoms (25% solids)	12,000

Solid wastes will be packaged for offsite disposal.

Table 11.5-2. Estimated Solid Waste Activity Shipped From Plant (Two-Units)

Isotope	Curies shipped per year
^{3}H	4.2×10^{2}
^{89}Sr	1.9×10^{1}
^{90}Sr	1.8×10^{1}
^{90}Y	1.8×10^{1}
^{91}Y	3.1×10^{1}
^{129}I	6.0×10^{-3}
^{131}I	6.0×10^{-3}
^{134}Cs	3.7×10^{4}
^{136}Cs	1.8×10^{-1}
^{137}Cs	1.0×10^{5}
$^{137}Ba^{m}$	9.3×10^{4}
^{140}Ba	4.8×10^{-3}
^{140}La	5.7×10^{-3}
^{144}Ce	3.4×10^{1}
^{144}Pr	3.4×10^{1}
^{58}Co	3.5×10^{3}
^{60}Co	5.5×10^{1}
^{54}Mn	1.5×10^{2}
^{51}Cr	2.5×10^{2}
^{55}Fe	6.8×10^{3}
^{59}Fe	4.5×10^{1}
^{95}Zr	4.2×10^{3}
^{95}Nb	4.0×10^{3}
	2.5×10^{5}

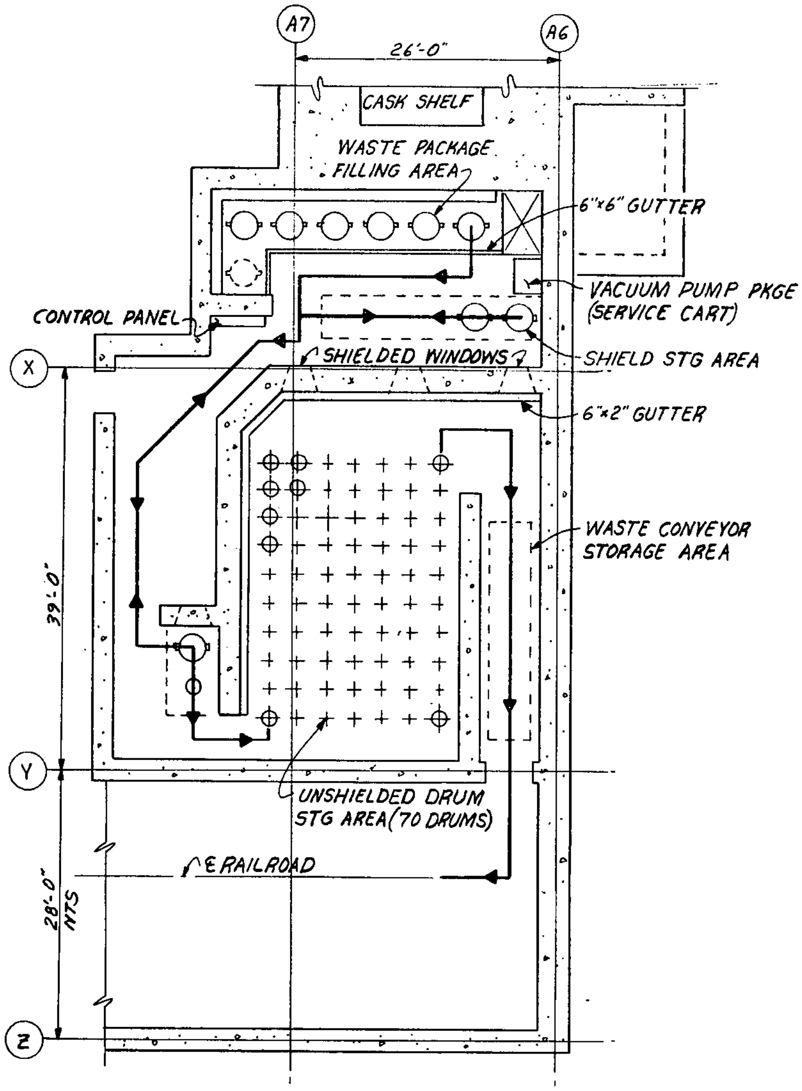
4



Bellefonte Nuclear Plant
Preliminary Safety Analysis Report

RADIOACTIVE SOLID
WASTE DISPOSAL

FIGURE 11.5-1



Bellefonte Nuclear Plant
Preliminary Safety Analysis Report

SOLID WASTE HANDLING AND STORAGE IN THE AUXILIARY BUILDING

FIGURE 11.5-2

11.6 Offsite Radiological Monitoring Program

The preoperational environmental monitoring program has the objective of establishing a baseline of data on the distribution of natural and man-made radioactivity in the environment near the plant site. The preoperational 13 environmental monitoring program will be initiated approximately two years prior to receipt of radioactive material at the site. The program will remain essentially unchanged throughout the preoperational period and through the first several years of operation.

Evaluations after plant startup will be made on the basis of the baselines established in the preoperational program, considering geography and the time of the year where these factors are applicable, and by comparisons to control stations where the concentrations of station effluents is expected to be negligible. In those cases where a statistically significant increase in the radioactivity level is seen in a particular sampling vector but not in the control station, meteorology and specific nuclide analysis will be used to identify the source of the increase.

The planned sampling frequencies will ensure that changes in the environmental radioactivity can be detected. The vectors which would first indicate increases in radioactivity are sampled most frequently. Those which are less effected by transient changes but show long-term accumulations are sampled less frequently. However, specific sampling dates are not cricial and adverse weather conditions or equipment failure may on occasion prevent collection of specific samples.

The capability of the environmental monitoring program to detect design-level releases from plant effluents in uncertain because of the insignificant quantities which will be released. The program will however provide the capability of detecting any significant buildup of radioactive material in the environment above and beyond that which is already present. Those vectors which are most sensitive to reconcentration of specific isotopes are sampled. If any increase in radioactivity levels is detected in these vectors, the program will be evaluated and broadened if deemed necessary.

From the data obtained from the radioanalytical and radiochemical analyses of the vectors sampled, dose estimates can be made for an individual or the population living near the plant site.

11.6.1 Expected Background

For a number of years measurements of background radiation have been made at various locations throughout the Tennessee Valley Region. TVA has conducted environmental monitoring programs in the vicinity of Oak Ridge and Chattanooga, Tennessee, and Decatur, Alabama. Over periods of not less than 2 years, the measurements made in these areas have indicated only very slight variations from location to location. The measurements obtained utilizing film badges of thermoluminescent dosimeters have revealed the following background radiation levels: Oak Ridge - 110 mR/year, Chattanooga - 130 mR/year, and Decatur - 120 mR/year. It is estimated that the expected background levels 13 in the vicinity of the Bellefonte Nuclear Plant will be between 115 and 130 mR/year.

Measurements will be made in the immediate vicinity of the Bellefonte Nuclear Plant site and will provide the baseline data necessary for comparison of background radiation levels prior to and after startup of the plant. 13

11.6.2 Critical Pathways to Man

Although the amounts of radioactivity added to the environment from plant operations are small, critical exposure pathways to man have been identified in order to estimate the maximum dose to the individual and to establish the sampling requirements for the environmental radioactivity monitoring program. The six principal pathways which can result in radiation exposure to him are as follows:

1. External exposures and inhalation 13

2. Drinking water from the Tennessee River and from wells in the immediate vicinity of the plant

3. External exposure from swimming, boating, and fishing in the Tennessee River

4. Eating fish from the Tennessee River

5. Consuming animal flesh and other animal products which may be affected by plant operations 13

6. Eating foods grown in areas adjacent to the plant site 13

The environmental monitoring program, as outlined, provides sampling of critical detectors necessary to evaluate the dose received through the critical pathways in items 1 to 6 above. The following items indicate the samples collected in order to make the critical pathway-dose correlation:

1. Data from readings of the thermoluminescent dosimeters will be utilized to estimate the total body dose received from the gaseous effluents.

2. Analysis of water samples collected will be used to estimate the dose that might be received from drinking water from the Tennessee River or from wills in the vicinity of the plant.

3. Analysis of water samples will also be used to estimate the dose an individual might receive while swimming, boating, or fishing on the lake in the vicinity of the plant.

4. Analysis of samples of three species of fish will be correlated to estimate the dose that might be received by an individual who eats fish from the Tennessee River. 13

5. Analysis of samples of sediment will be correlated to estimate the dose an individual might receive from shoreline activities.

6. & 7. Analysis of samples of air particulate matter, soil, vegetation, food crops, and milk will be used to estimate the dose to the surrounding population through the consumption of food or dairy products. All samples referenced will be analysed for the most biologically-significant gamma-emitting radionuclides found in the waste stream of the plant. In addition, an analysis for ^{89}Sr, ^{90}Sr and tritium will also bo performed. 13

The environmental monitoring program to be conducted throughout operation of the plant provides the necessary means of evaluating the dose to man through critical exposure pathways. 13

11.6.2.1 Doses From Gaseous Effluents

The following doses to humans living in the vicinity of the Bellefonte Nuclear Plant will be calculated for the releases of radioactive gases:

1. External beta and gamma doses from airborne radioactivity
2. External beta and gamma doses from ground contamination
3. Internal doses from inhalation 13
4. Internal doses from ingestion

The basic assumptions and calculational methods that will be used in computing these doses are described in Section 11.3.9.

Continual review of the data resulting from the offsite monitoring program and reevaluations of the adequacy of the dose models will ensure that the actual doses received by individuals and the population as a whole remain as low as practicable and within the applicable Federal regulations.

11.6.2.2 Doses From Liquid Effluents

The following doses will be calculated for exposures to radionuclides routinely relaased in liquid effluents:

1. Internal doses from the ingestion of water
2. Internal doses from the consumption of fish
3. External doses from water sports
4. External doses from shoreline activities 13

A detailed description of the basic assumptions and calculational methods that will be used in calculating the doses is given in Section 11.2.9.

The dose models that are employed will be continually reevaluated in light of the data resulting from the offsite monitoring program to ensure that all significant pathways are included in the calculations and to ensure that the actual doses received by individuals and the population as a whole remain as low as practicable and within the applicable Federal regulations.

11.6.3 Sampling Media, Locations, and Frequency

The sampling media, the locations from which the samples are collected, and the frequency on which the samples are collected are presented in Table 11.6-1 and Figures 11.6-1 and 11.6-2. The media selected were chosen on two bases: first, those vectors which would readily indicate releases from the plant, and second, those vectors which would indicate long-term buildup of radioactivity. Consideration was also given to the pathways which would result in exposure to man, such as milk and food crops. Locations for sampling stations were chosen after considering meterological factors and population density around the site. Frequencies for sampling the various vectors were established so that seasonal variations in radioactivity levels might be determined. In addition, samples are collected during the season in which the major growth occurs to ascertain radioactivity uptake by the vectors during their most susceptible period of growth. The monitoring program outlined herein is subject to change based upon continued evaluation of programs now being conducted in the Tennessee Valley Region and upon other available data. 13

11.6.4 Analytical Sensitivity

Samples will be collected routinely following established procedures so that uniformity in sampling methods will always be assured. The samples will be transported to a central laboratory facility for preparation and analysis. All the radioanalytical and radiochemical analyses will be conducted in the central laboratory. In performing the analyses, three pulse height analyzers with two 4 x 4 solid NaI crystals, two well NaI crystals, and a Ge(Li) detector, four low background beta counters, one liquid scintillation system, two scalers with GM tube detectors, and one internal proportional counter will be utilized. Data will be coded and punched on IBM cards or automatically printed on paper tape for computer processing specific to the analysis conducted. An IBM 370 Model 165 Computer, employing a least squares code, will be used to solve multimatrix problems associated with identification of gamma-emitting isotopes. 13

TVA assumes that minimum sensitivities are those concentration values where the precent counting error is 100 percent, calculated assuming a 3-sigma counting error, when standard sample sizes and counting times are employed. The minimum sensitivities are therefore those concentration values below which it is impossible to state, at the 99 percent confidence level, that any amount of radioactivity above background exists in the sample. 13

The minimum sensitivities, based wholly on counting statistics, for gross alpha, gross beta, radiostrontium, tritium, and radioiodine are given in Table 11.6-2. Minimum sensitivities have not been calculated for specific gamma-emitting radionuclides determined by a least squares computer code (Alpha-M).

The counting errors (and therefore the MDC's) for each radionuclide vary from sample to sample depending upon the concentrations of all the other radionuclides present in the sample. TVA is analyzing gamma analyses data to determine minimum 13
sensitivities for each radionuclide under the varying conditions.

11.6.5 Data Analysis and Presentation

Data measured at each indicator and control station will be averaged for the 6-month reporting period. In order to describe the distribution of control station data, a mean, standard deviation, and 3-sigma value will be calculated. One can expect, with 99 percent confidence, that background concentrations would be distributed within these limits. This provides a basis for comparing control and indicator data. If the indicator data falls within the limits 13
defined for control data, one cay say, with 99 percent confidence, that the indicator was not significantly affected by radioactivity released from the plant. If the data does not fall within the limits, TVA will perform further analyses to determine if the difference is attributable to the nuclear plant.

A quality control program has been established with the Tennessee Department of Public Health Radiological Laboratory and the Eastern Environmental Radiation Facility, Environmental Protection Agency, Montgomery, Alabama. Samples of air, water, milk, and vegetation will be collected and forwarded to these laboratories for analysis. Results will be exchanged and compared to insure that the laboratories involved are maintaining the desired degree of accuracy. For several years TVA has submitted semiannual environmental radioactivity monitoring reports to the NRC for the Browns Ferry and Sequoyah Nuclear Plant 13
The results from individual samples for each station are combined to obtain a semiannual average. This format for those documents and the results contained therein have been satisfacotry. Therefore, this format will be used until such time that revisions become necessary.

11.6.6 Program Statistical Sensitivity

As previously noted, because of the small quantities of radioactivity expected to be released to the environment from the Bellefonte Nuclear Plant, it is uncertain as to what extent the results from the environmental monitoring program will be used to estimate the radiation exposures to humans. TVA is in the process of devoloping procedures to determine the overall statistical sensitivity of the TVA's environmental monitoring programs. These procedures will be applied to the preoperational environmental monitoring data for the 13
Bellefonte Nuclear Plant to estimate the minimum incremental doses that can be detected. It is expected that the overall statistical sensitivity of the environmental monitoring program will be sufficient to detect potential doses well below 10CFR20 limits.

Results from the analyses of effluent samples, which contain higher concentrations of radionuclides than environmental samples, will be used in the TVA models similar to those given in Sections 11.2 and 11.3 to estimate the possible exposure to man. TVA will control the releases of radioactive materials to the environment 13
such that the releases will be less than the limits described in Title 10 Code of Federal Regulations Parts 20 and 50.

Table 11.6-1

Environmental Radiological Surveillance Program

Bellefonte Nuclear Plant

	Criteria and Sampling Locations	Collection Frequency	Analysis/Counting	
I. Atmospheric				
A. Air				
1. Particulate	Filter paper at 10 locations	Weekly	Gross beta (gamma scan monthly)	
2. Radioiodine	Charcoal filter at 10 locations	Weekly	^{131}I	
B. Fallout	Gummed acetate at 10 locations	Monthly	Gross beta	
C. Rainwater	Rainwater collection trays at 10 locations.	Monthly	Gross beta, gamma scan,[a] ^{89}Sr, ^{90}Sr, ^{3}H	13
II. Reservoir				
A. Water				
1. Municipal (public supplies)	All public water supply intakes within 10 miles downstream of the plant.	Monthly	Gross beta, gamma scan, ^{3}H	13
2. Reservoir	Automatically sampled from three locations; one above, one below, and one at plant site	Analyzed Monthly	Gross beta, gamma scan, ^{89}Sr, ^{90}Sr, ^{3}H	
B. Aquatic Biota				
1. Fish (buffalo, crappie, and catfish)	Nickajack, Guntersville, and Wheeler reservoirs	Quarterly	Gross beta, gross alpha,[b] gamma scan, ^{89}Sr, ^{90}Sr	
2. Shellfish (Asiatic clams)	Three locations: TRM 396.8, 391.2, and 388.0	Quarterly	Gross beta, gross alpha, gamma scan, (^{89}Sr, ^{90}Sr shells only)	
3. Aquatic Macrophytes	Three locations: TRM 396.8, 391.2, and 388.0	Quarterly	Gross beta, gross alpha, gamma scan, ^{89}Sr, ^{90}Sr	13
4. Plankton	Three locations: TRM 396.8, 391.2, and 388.0	Quarterly	Gross beta, gross alpha, gamma scan, ^{89}Sr, ^{90}Sr	
C. Sediment	Three locations: TRM 396.8, 391.2, and 388.0	Quarterly	Gross beta, gross alpha, gamma scan, ^{89}Sr, ^{90}Sr	
III. Terrestrial				
A. Soil	Atmospheric monitoring locations	Annually	Gross beta, gamma scan	
B. Vegetation				
1. Pasturage and grass	Selected dairy farms within 10-mile radius of plant	Monthly	Gross beta, gamma scan, ^{89}Sr, ^{90}Sr	
	10 atmospheric monitoring stations	Quarterly	Gross beta, total alpha,[c] gamma scan, ^{89}Sr, ^{90}Sr	
2. Food crops	Within 10-mile radius of plant	Annually	Gross beta, total alpha, gamma scan, ^{89}Sr, ^{90}Sr	
C. Milk	Selected dairy farms within 10-mile radius of plant	Monthly[d]	Gamma scan, ^{89}Sr, ^{90}Sr, ^{131}I	
D. Well water	Selected wells within 5 miles of plant	Quarterly	Gross beta, gamma scan	
E. Direct radiation	TLD's onsite and at remote and perimeter monitors	Quarterly	Dose determination	

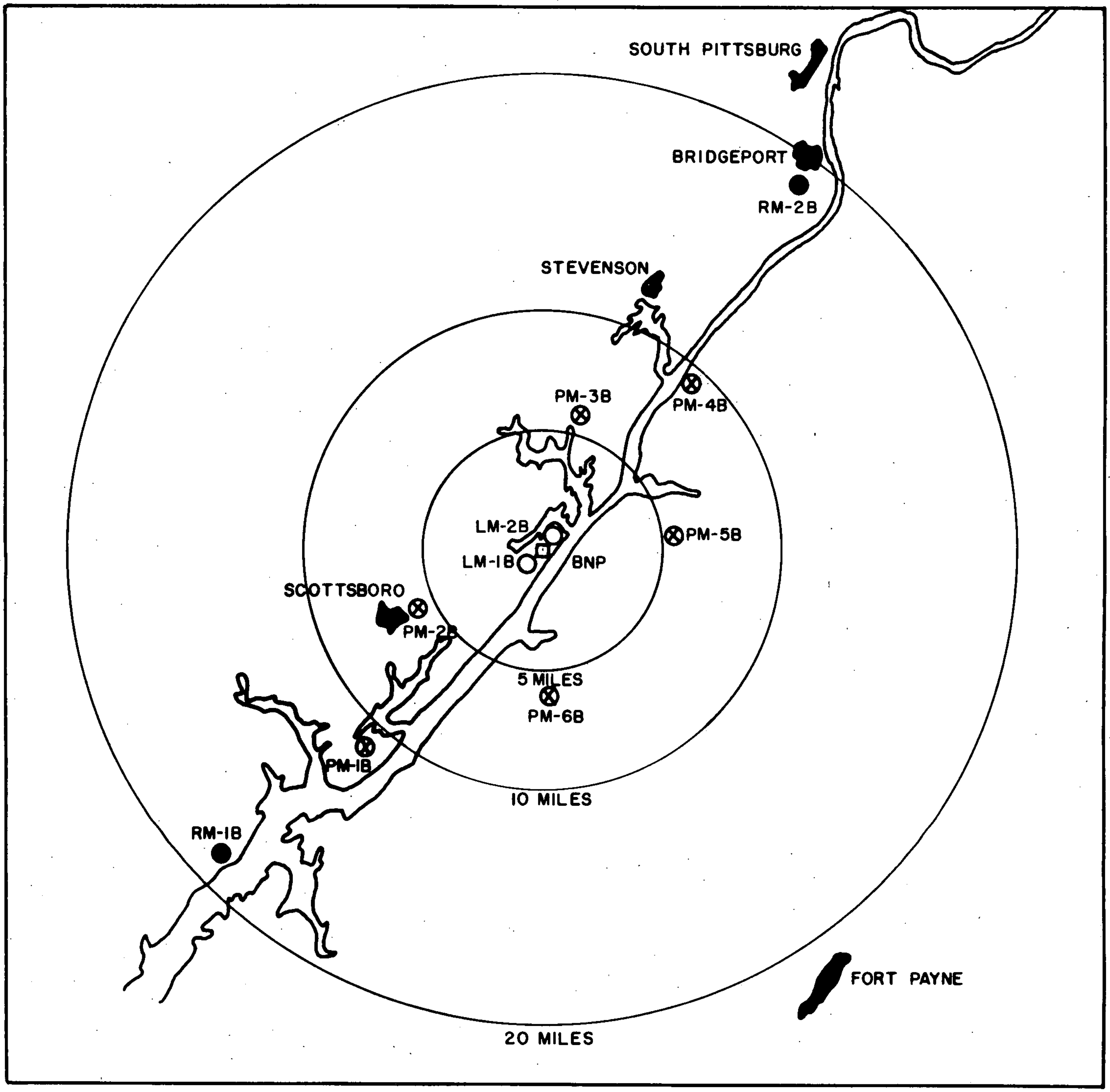
a. The gamma scan will include specific analyses of at least 10 gamma-emitting isotopes except for milk samples which will be analyzed for four isotopes.
b. Aliquot of prepared sample counted directly for alpha.
c. Heavy metals separated as a part of the ^{89}Sr, ^{90}Sr separation process are precipitated, filtered, and counted for alpha.
d. After the plant begins operation, milk samples will be taken weekly and analyzed for ^{131}I during seasons cows are on pasture.

TABLE 11.6-2

MINIMUM SENSITIVITIES

Type Sample	Units	Gross α	Total α	Gross β	Sr	^3H	^{131}I
Air Filter	pCi/m^3	0.002		0.004	0.0005		
Charcoal	pCi/m^3						0.022
Fallout	mCi/km^2			0.01			
Water	pCi/l	1.2		1.6	1.0	400	
Vegetation	pCi/gm, Dry		0.01	0.1	0.002		
Soil	pCi/gm, "			0.15	0.2		
Fish	pCi/gm, "	0.02		0.03	0.03		
Sediment	pCi/gm, "	0.11		0.15	0.2		
Clam Flesh	pCi/gm, "	0.03		0.04			
Clam Shell	pCi/gm, "	0.11		0.14	0.31		
Food	pCi/gm, "		0.02	0.1	0.1		
Plankton	pCi/gm, "	0.03		0.04	0.05		
Milk	pCi/l				1.0		0.5

ATMOSPHERIC AND
TERRESTRIAL MONITORING NETWORK



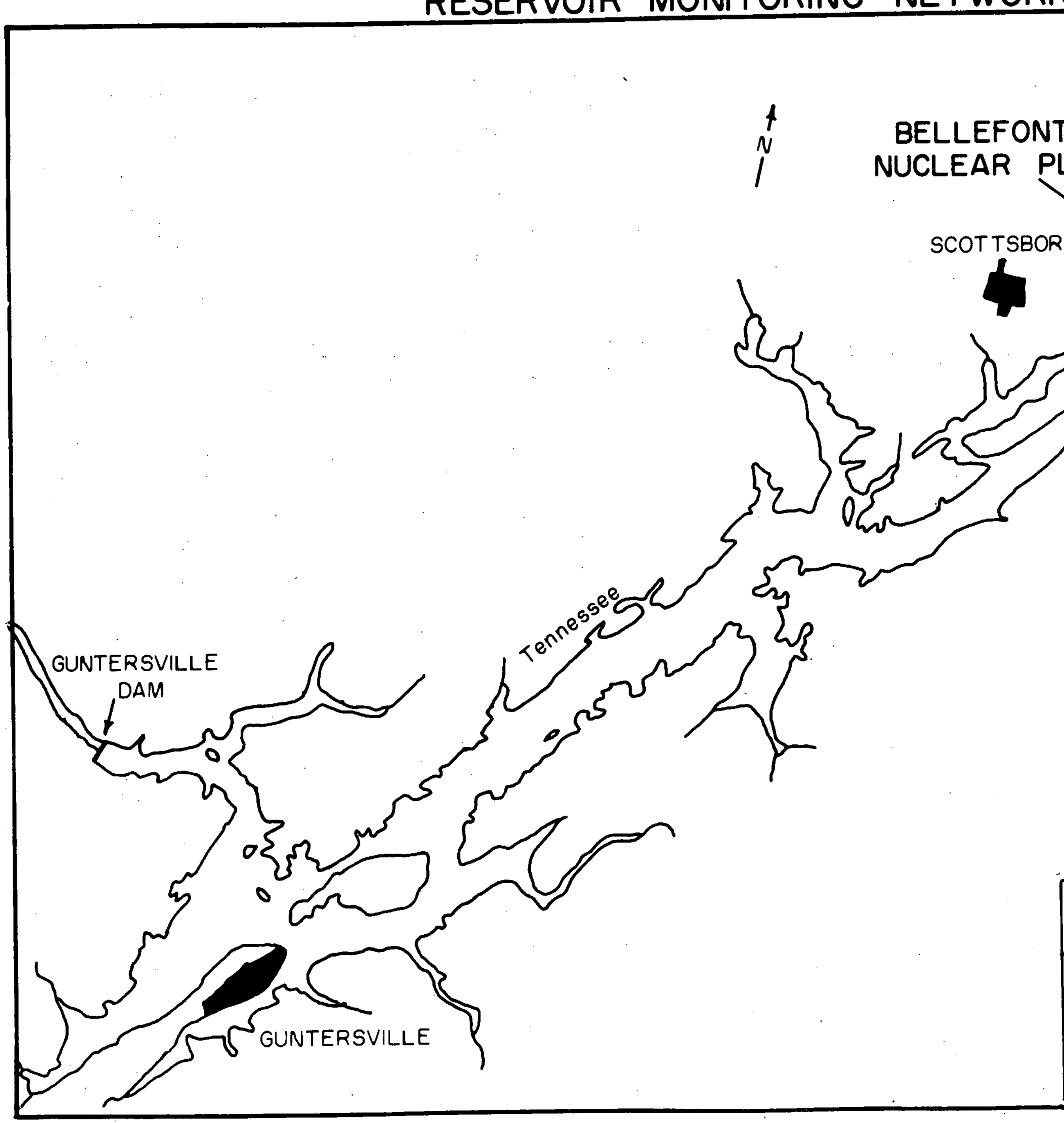
NOTE: THE FOLLOWING SAMPLES ARE COLLECTED FROM EACH STATION:

AIR PARTICULATES	RAINWATER
RADIOIODINE	SOIL
HEAVY PARTICLE FALLOUT	VEGETATION

BELLEFONTE NUCLEAR PLANT
PRELIMINARY SAFETY
ANALYSIS REPORT

ATMOSPHERIC AND TERRESTRIAL
MONITORING NETWORK
FIGURE 11.6-1
REVISED BY AMENDMENT 13
Oct. 3, 1975

RESERVOIR MONITORING NETWORK



BELLEFONTE NUCLEAR PLANT
PRELIMINARY SAFETY
ANALYSIS REPORT

RESERVOIR MONITORING
NETWORK
FIGURE 11.6-2
REVISED BY AMENDMENT 13
Oct. 3, 1975