11.0 WASTE DISPOSAL AND RADIATION PROTECTION SYSTEM

11.1 WASTE DISPOSAL SYSTEM

11.1.1 Design Bases

Liquid, gaseous, and solid waste disposal facilities are designed so that discharge of effluents and off-site shipments are in accordance with applicable governmental regulations. The facilities are shown on Figures 11.1-1, 11.1-2, 11.1-2A, 11.1-2B, 11.1-3 and 11.1-4. The sizing of the various waste equipment was predicated on the volumes and flow rates originally expected to be handled.

Radioactive fluids entering the Waste Disposal System are collected in tanks until determination of subsequent treatment can be made. Provisions have been made for waste segregation to permit selective operation of the processing equipment to maintain radioactivity in the effluents as low as practicable. Fluids are sampled and analyzed to determine the quantity of radioactivity, with an isotopic breakdown if necessary. Liquid wastes are processed as required and then released under controlled conditions. The system design and operation are directed toward minimizing releases to unrestricted areas. Discharge streams are appropriately monitored and safety features are incorporated to preclude releases in excess of the limits of 10 CFR 50 Appendix I.

Some of the radioactive liquid discharge from the reactor coolant system can be processed and retained inside the plant by the chemical and volume control system (CVCS) recycle train. This minimizes liquid input to the waste disposal system, which processes relatively small quantities of generally low-activity level wastes. The processed water from the waste disposal system, from which most of the radioactive material has been removed, is discharged through a monitored line to the circulating water discharge.

Radioactive gases are pumped by compressors through a manifold to one of the gas decay tanks where they are held a suitable period of time for decay. The quantity of radioactive material in each gas decay tank is periodically determined to be within the Technical Specification limit whenever radioactive materials are added to the tank and during primary coolant system degassing operations. The radioactive material is quantified by analyzing the noble gas activity in the reactor coolant system or directly from samples of the contents of the gas decay tanks.

Cover gas is reused to minimize gaseous wastes. During normal operation, gases are discharged intermittently at a controlled rate from these tanks through the monitored plant vent.

The spent demineralizer resins, the filter cartridges, and the concentrates from the evaporators are packaged for temporary on-site storage or disposal.

11.1.2 General Description and Operation

The waste disposal system performance data are given in Table 11.1-1. With the exception of the reactor coolant drain tanks and drain tank pumps, the waste disposal system is common to Units 1 and 2. The system is capable of processing all wastes generated during continuous operation of the primary system assuming that fission products escape to the reactor coolant by diffusion through defects in the cladding of one percent of the fuel rods.

The waste disposal system collects and processes all potentially radioactive reactor plant wastes for removal from the plant site within limitations established by applicable governmental regulations. In addition, the system is capable of liquid waste segregation.

All planned releases may be either batch or continuous. Before a batch may be released, the tank is sampled, the sample analyzed in the laboratory, and determined not to exceed applicable regulatory limits prior to the tank release. Releases are made only if the release can be made without exceeding federal standards and lack of reserve holdup capacity requires such a release.

Radiation monitors are provided to maintain surveillance over the release operation, and a permanent record of activity released is provided by radiochemical analysis of known quantities of waste.

At least two valves must be manually opened to permit discharge of liquid or gaseous waste from the Waste Disposal System. One of these valves is normally locked or sealed closed. The other is a control valve which will trip closed on a high effluent radioactivity level signal.

As secondary functions, system components supply hydrogen and nitrogen to primary system components as required during normal operation, and provide facilities to transfer fluids from the containment to other systems outside the containment.

The system is controlled primarily from a central panel in the auxiliary building. Malfunction of the system is alarmed in the auxiliary building, and annunciated in the control room. All system equipment is located in or near the auxiliary building, except for the reactor coolant drain tanks which are located in the reactor containments.

11.1.2.1 System Description

11.1.2.1.1 Liquid Processing

Liquid waste is processed in accordance with the Radioactive Waste Process Control Program.

During normal plant operation the Waste Disposal System processes liquids from the following sources:

a. Equipment drains and leaks
b. Radioactive chemical laboratory drains
c. Radioactive laundry and hot shower drains
d. Decontamination area drains
e. CVCS demineralizer regeneration (is currently not used)
f. Sampling System

The system also collects and transfers liquids from the following sources in the containment for processing:

a. Reactor coolant loops
b. Pressurizer relief tank
c. Reactor coolant pump secondary seals
d. Excess letdown (during startup)
e. Accumulators
f. Valve and reactor vessel flange leakoffs
g. Refueling cavity drains

The liquids in the containment flow to the reactor coolant drain tank and are discharged by the reactor coolant drain tank pumps either directly to the CVCS holdup tanks or to the clean waste holdup tank. The pumps can be operated either automatically by a level controller in the tank or by manual control. These pumps also return water from the refueling cavity to the refueling water storage tank. The reactor coolant drain tank pumps are located inside the auxiliary building.

Where possible, waste liquids in the auxiliary building drain to the waste holdup tanks by gravity flow. Other waste liquids drain to the sump tanks and are discharged to the waste holdup tanks by pumps operated automatically by a level controller in the sump tanks.

The activity level of waste liquid from the laundry and hot shower area will usually be low enough to permit discharge from the plant without processing. If analysis indicates that the liquid is suitable for discharge, it is pumped to one of two CVCS monitor tanks which are both functioning as waste condensate tanks where the activity is determined before discharging through a line monitored for radiation to the circulating water. Otherwise, the liquid is pumped to the Dirty Waste Hold Up Tank for processing by the radwaste demineralization system. An analysis record is maintained for all releases. When the laundry and hot shower tanks are receiving CCW or ESW drainage, the tank contents can be transferred to the monitor tanks without recirculating and sampling.

Liquid radwaste is processed through a radwaste demineralization system. This system has the capability of processing all liquid radwaste prior to discharge and is designed in accordance with Regulatory Guide 1.143. The process decontaminates the water using filtration and ion exchange.

As a backup to the radwaste demineralization process, one of two CVCS boric acid evaporators was converted to function as a radwaste evaporator. A 15 gpm radwaste evaporator is available as backup to the 30 gpm boric acid/radwaste evaporator in case additional capacity is desired. However, neither of the evaporators is currently used. Liquids requiring cleanup before release can be processed in batches in this boric acid/radwaste evaporator. Processing liquid waste is similar to processing reactor coolant except for disposal of the processed liquids and vented gases. Liquid waste is pumped to the boric acid/radwaste evaporator via the waste evaporator feed pumps. The concentrates can be discharged to the waste evaporator bottoms storage tank for either draining back to waste for reprocessing or drumming prior to shipment to an offsite burial facility or temporary on-site storage.

Radwaste demineralizer effluent and evaporator distillate (condensate) which are to be released are routed to one of two CVCS monitor tanks which are both functioning as waste condensate tanks. When one tank is filled, it is isolated and sampled for analysis while the second tank is in service. If analysis confirms the activity level is suitable for discharge, the condensate is pumped through a flow meter and a line monitored for radiation to the condenser circulating water discharge. Condensate can also be released under the same administrative controls from the other two CVCS monitor tanks which serve the other boric acid evaporator. The releases are sampled and analyzed for both tritium and gamma emitting radionuclides and monitored by the same radiation monitor as that previously mentioned above before release into the circulating water discharge. If analysis indicates the activity level is not suitable for discharge, the condensate is returned to the station drainage waste holdup tank for reprocessing. Although

the radiochemical analysis forms the basis for recording activity released, the radiation monitor provides surveillance over the operation by closing the discharge valve if the liquid activity level exceeds a preset value.

Measures are taken to minimize the need to process fluids which contain foam causing substances. If possible, non foaming decontamination agents are used for equipment scrubdown where the decontamination agent must be processed through the evaporators. If foaming should occur a reagent tank is provided for charging the evaporator with an antifoaming reagent.

11.1.2.1.2 Gas Processing

During plant operations, gaseous wastes will originate from:

a. Degassing reactor coolant discharged to the Chemical and Volume Control System

b. Displacement of cover gases as liquids accumulate in various tanks

c. Miscellaneous equipment vents and relief valves

d. Sampling operations and automatic gas analysis for hydrogen and oxygen in cover gases

The waste disposal system includes nitrogen and hydrogen systems which supply these gases to primary plant components. The pressure regulator in the nitrogen system header is set at 75 psig. When the nitrogen header pressure drops below a preset pressure, an alarm alerts the operator.

Most of the gas received by the waste disposal system during normal operation is nitrogen cover gas displaced from the CVCS holdup tanks and boric acid reserve tank as they are filled with liquid. Since this gas must be replaced when the tanks are emptied during processing, facilities are provided to return gas from the decay tanks to the holdup tanks and boric acid reserve tank. A backup supply from the nitrogen header is provided for makeup if return flow from the gas decay tanks is not available. Since the hydrogen concentration may exceed the combustible limit during this type of operation, components discharging to the vent header system are restricted to those containing no air or no aerated liquids and the vent header itself is designed to operate at a slight positive pressure (but not high enough to cause overpressurization) including allowances for instrument uncertainties, process induced pressure changes, and any other special concerns that may be necessary to prevent oxygen in-leakage. To the extent practical, out-leakage from the system is minimized by using Saunders patent diaphragm valves, bellows seals, self contained pressure regulators and soft-seated packless valves throughout the system.

Gases vented to the vent header flow to the waste gas compressor suction header. One of the two compressors is in continuous operation with the second unit instrumented to act as backup for peak load conditions or failure of the first unit. From the compressors, gas flows to one of eight gas decay tanks. The control arrangement on the gas decay tank inlet header allows the operator to place one tank in service and to select another tank for backup. When the tank in service becomes pressurized to 100 psig, a pressure transmitter automatically closes the inlet valve to that tank, opens the inlet valve to the backup tank and sounds an alarm to alert the operator so he may select a new backup tank. Pressure indicators are provided to aid the operator in selecting the backup tank. The individual tank pressures are continuously recorded on the control panel in the auxiliary building.

Gas held in the decay tanks can either be returned to the CVCS holdup tanks or, if it has decayed sufficiently for release, discharged to the atmosphere. Generally, the last tank to receive gas will be the first tank recycled to the CVCS holdup tanks. This permits the maximum decay time before releasing gas to the environment. However, the header arrangement at the tank inlet gives the operator the option to fill, reuse, and discharge gas simultaneously. During degassing of the reactor coolant prior to a cold shutdown, for example, it may be desirable to pump the gas purged from the volume control tank into a particular gas decay tank and isolate that tank for decay rather than reuse the gas in it. This is done by opening the inlet valve to the desired tank and closing the outlet valve to the reuse header. Simultaneously, one of the other tanks can be opened to the reuse header if desired, while another is discharged to atmosphere.

Before a tank is discharged to the environment, it is sampled and analyzed to determine and record the activity to be released, and then is discharged to the plant vent at a controlled rate. The plant vent's radiation monitor enables the operator to monitor the radioactivity in the gas release. Samples of the gas to be released are taken in gas sampling vessels. During release a trip valve in the discharge line is closed automatically by a high radioactivity level indication in the plant vent.

During operation, gas samples are drawn automatically from the gas decay tanks and analyzed to determine their hydrogen and oxygen content. A second analyzer is used to monitor oxygen in the line from the discharge of the waste gas compressor in operation. There should be no significant oxygen content in the waste gas or in any of the gas decay tanks; an alarm sounds if either of the samples contains 2.5% or higher by volume of oxygen. Upon a "high-high" oxygen content of 2.7% by volume, the oxygen analyzer automatically isolates the tank being filled and places the standby gas decay tank in service. The operator then determines the source

of oxygen in-leakage and purges the affected component and Waste Gas System vent header piping as required with nitrogen. The isolated waste gas decay tank and standby tank can be diluted with nitrogen if they have high oxygen concentrations.

11.1.2.1.3 Solids Processing

Solid waste is processed in accordance with the Radioactive Waste Process Control Program.

The Waste Disposal System is designed to package solid wastes for removal to disposal facilities. Some solid waste is temporarily stored on-site.

The capability exists to either drain concentrates from the waste evaporator bottoms storage tank back to waste for reprocessing or pump the concentrates into shipping casks and mixed with the solidification agent. The casks are moved to a shielded storage area until removal to a burial site or temporary on-site storage.

Spent resins are either sluiced to the spent resin storage tank or pumped directly into shielded shipping casks within the Auxiliary Building. Resins in the storage tank can be sluiced by first bubbling nitrogen through the tank to the vent header to stir up the resin, then using water to transport the resin at a controlled rate into shipping casks within the Auxiliary Building. Resins are either dewatered and air dried or slurried with a solidification agent for shipment. The casks are handled and stored in a fashion identical to that for the concentrated bottoms.

Shielding is provided for each cask during filling and handling operations to reduce the dose rate in work areas. The basis for shield design and dose rate calculations is for one cycle of core operation with one percent defective fuel in each unit.

11.1.2.1.4 Components

Codes applying to components of the Waste Disposal System are shown in Table 11.1-2. Components summary data are shown in Table 11.1-3.

11.1.2.1.4.1 Laundry and Hot Shower Tanks

Two stainless steel tanks collect liquid wastes originating from the laundry and hot showers. When the tanks have filled, the contents are analyzed for gross activity. As dictated by the activity level, the tank contents are pumped to the Dirty Waste Hold Up tank for processing by the radwaste demineralization system or to the waste condensate tanks before discharging through a line monitored for radiation to the circulating water.

11.1.2.1.4.2 Chemical Drain Tank

A stainless steel chemical tank, collects drainage from the radiochemical laboratory. The tank contents are pumped to the station drainage waste holdup tank for processing.

11.1.2.1.4.3 Reactor Coolant Drain Tanks

The tanks serve as a drain collecting point for the Reactor Coolant Systems and other equipment located inside the reactor containments. The contents can be discharged to the waste holdup tanks, to the refueling water storage tanks, or to the CVCS holdup tanks. The tanks are of welded stainless steel construction.

11.1.2.1.4.4 Station Drainage Waste Holdup Tank and Pump

The station drainage waste holdup tank receives liquids from the station drainage sump tank, floor drains above elevation 587 feet, equipment drains, chemical drains and laundry and hot shower tanks. The tank contents are normally pumped to the radwaste demineralization system or to either the radwaste or converted boric acid/radwaste evaporators. The tanks are of welded stainless steel construction.

11.1.2.1.4.5 Clean Waste Holdup Tank and Pump

The clean waste holdup tank receives radioactive liquids from the clean sump tank, reactor coolant drain tanks, containment ventilation unit drains, ice condenser defrost drains, and equipment drains. The tank is of welded stainless steel construction. A pump transfers the contents of the tank to the radwaste demineralization system or to either the radwaste or converted boric acid/radwaste evaporators.

11.1.2.1.4.6 Station Drainage Sump Tank and Pump

The station drainage sump tank serves as a collecting point for waste discharged to the floor drain header below elevation 609 feet. A horizontal centrifugal pump transfers the tank's contents to the station drainage waste holdup tank. Wetted parts of the pump are stainless steel. The tank is welded stainless steel.

11.1.2.1.4.7 Clean Sump Tank and Pump

The clean sump tank serves as a collecting point for waste discharged to the lower elevation equipment drain header below elevation 609 feet. A horizontal centrifugal pump transfers the contents of the tank to the clean waste holdup tank. Wetted parts of the pump are stainless steel. The tank is welded stainless steel.

11.1.2.1.4.8 Spent Resin Storage Tank

The spent resin storage tank retains spent resin normally discharged from the reactor coolant system demineralizers. A layer of water is maintained over the resin surface to minimize resin degradation due to heat generated by decaying fission products. The tank is welded stainless steel.

11.1.2.1.4.9 Waste Evaporators

The two waste evaporators employed at Cook concentrate dissolved and suspended solids in liquid wastes. Each evaporator assembly consists of an evaporator, evaporator condenser, distillate cooler, concentrates pump and distillate pump.

The converted boric acid/radwaste evaporator is a submerged tube type with a stripping column to prevent bottoms carry over and remove any volatiles in the vapor.

The 15 gpm evaporator is a forced circulation flash type with an entrainment separator to prevent bottoms carry over and remove any volatiles in the vapor.

11.1.2.1.4.10 Waste Evaporator Bottoms Storage Tank and Transfer Pump

The waste evaporator bottoms storage tank receives concentrated wastes from the radwaste evaporator or the 30 gpm boric acid/radwaste evaporator. A transfer pump transfers the concentrates for drumming. The system is stainless steel and is heat traced. This equipment is not currently used.

11.1.2.1.4.11 Gas Decay Tanks

Eight welded carbon steel tanks are provided to contain compressed waste gases (hydrogen, nitrogen, and fission gases). After a period for radioactive decay, these gases are released at a controlled rate to the atmosphere through the auxiliary building ventilation system or recycled

to the CVCS. Discharges to the atmosphere are continuously monitored for radioactivity and flow.

11.1.2.1.4.12 Compressors

Two compressors are provided for compressing gases vented from equipment that contains or can contain radioactive gases. These compressors are of the water-sealed centrifugal displacement type. The operation of the compressors is automatically controlled by the gas manifold pressure. Construction is primarily of carbon steel. A mechanical seal is provided to minimize leakage of seal water.

11.1.2.1.4.13 Waste Evaporator Condensate Demineralizer

This mixed-bed demineralizer, located in the recycle line to the CVCS monitor tanks, can be used to remove dissolved solids that may be carried over from the waste evaporator. The demineralizer vessel is made of all-welded stainless steel and is equipped with a resin retention screen. This equipment is not currently used.

11.1.2.1.4.14 Waste Evaporator Condensate Filter

This filter is used to collect resin fines and particulates from the waste evaporator condensate stream downstream of the waste evaporator condensate demineralizer when it is in use. The vessel is made of stainless steel, and is provided with a connection for draining and venting. Disposable resin bonded glass medium filter type, cellulose filter type or equivalent filter elements are utilized. This equipment is not currently used.

11.1.2.1.4.15 Waste Evaporator Condensate Tanks

These tanks collect radwaste demineralization system effluent and evaporator condensate. The contents are sampled and analyzed for radioactivity before discharge. These tanks are of welded stainless steel construction.

11.1.2.1.4.16 Nitrogen Header

This header supplies nitrogen to purge hydrogen from the vapor space of various components or to provide cover gas when draining certain equipment. A pressure controller assures a constant gas pressure.

11.1.2.1.4.17 Gas Analyzers

An automatic gas analyzer is provided to monitor the concentrations of oxygen and hydrogen in the cover gas of primary side tanks and vessels which might accumulate a hazardous mixture of the two gases. Upon indication of a high oxygen level, provisions are made to purge the equipment to the gaseous waste system with nitrogen gas.

An automatic oxygen analyzer is provided to monitor the concentration of oxygen in the discharge of the waste gas compressor. Upon indication of a "high-high" oxygen content, the analyzer isolates the gas decay tank being filled and places the standby tank in operation. Provisions are made to purge the equipment with nitrogen gas.

11.1.2.1.4.18 Pumps

The wetted surfaces of all pumps are either stainless steel or other materials of equivalent corrosion resistance. All pumps are either the canned type or mechanically sealed to minimize leakage.

11.1.2.1.4.19 Piping

Piping carrying liquid wastes is stainless steel while process gas piping is carbon steel. Piping connections are welded except where flanged connections are necessary to facilitate equipment maintenance.

11.1.2.1.4.20 Valves

Process valves exposed to waste gases are carbon steel. Those exposed to liquid wastes are stainless steel.

Isolation valves are provided to isolate each piece of equipment for maintenance, to direct the flow of waste through the system, and to isolate storage tanks for radioactive decay.

Relief valves are provided on all tanks not vented to the atmosphere and which contain radioactive waste liquid or gas in case the tanks are overpressurized by improper operation or component malfunction. Tanks containing liquid waste which are normally free of gaseous activity are vented to the atmosphere.

11.1.3 Design Evaluation

11.1.3.1 Liquid Wastes

Liquid waste is processed in accordance with the Radioactive Waste Process Control Program.

Liquid Wastes are generated primarily by plant maintenance and service operations, and consequently, the quantities and activity concentrations relating to the system, Tables 11.1-4, 11.1-5 and 11.1-6, are estimated values. There is considerable operational margin between the design capability and the estimated system load as indicated by Table 11.1-4, Estimated Liquid Discharges To Waste Disposal. A conservative estimate of dissolved and suspended activity released in the liquid phase is summarized in Table 11.1-5, Estimated Liquid Release by Isotope. This tabulation is based on the following assumptions and parameters:

1. All liquid waste is assumed to be initially at peak reactor coolant activity concentrations based on continuous full power operation with 1% fuel cladding defects.
2. A total of 1200 minutes is allowed for decay. This is the time required to process a 18,000 gallon batch at 15 gallons per minute.
3. The waste is concentrated to an activity of 40 μCi/ml in the bottoms.
4. The waste evaporator Decontamination Factor (DF) of 10^6 yields an activity of 4×10^{-5} μCi/ml in the waste condensate. The DF is the ratio of activity concentration in the evaporator bottoms to activity concentration in the condensate (distillate), and is based on sodium tracer tests.
5. The total estimated annual release by isotope (Table 11.1-5) in μCi/yr, excluding tritium, is obtained by multiplying the quantity released per year (Table 11.1-4) by the computed activity for each isotope in μCi/ml.
6. To this is added the activity released through waste disposal by the CVCS monitor tanks and is based on processing for release four reactor coolant volumes per year. This activity is estimated to be less than 1 μCi/yr.
7. The design basis for tritium release given in Table 11.1-1 assumes that 30 percent of the tritium that is formed in the fuel (the predominant source) diffuses through the zircaloy clad and eventually becomes available for dispersal to the environment. This is believed to be an overestimate for zircaloy clad fuel. The expected release of ternary produced tritium is one percent. All of the sources of

tritium accumulating in the reactor coolant, are included in the annual design release.

The liquid in the clean and station drainage holdup tanks is processed for discharge through the radwaste demineralizers or the waste evaporators, to the waste condensate tanks, except when sampling indicates the waste can be discharged without processing. However, neither waste evaporator is currently used. This reduces the total activity discharged in the liquid waste which contains both tritium and other sources of radioactivity.

11.1.3.2 Gaseous Wastes

Gaseous wastes consist primarily of hydrogen stripped from coolant discharged to the CVCS holdup tanks during boron dilution, nitrogen and hydrogen gases purged from the CVCS volume control when degassing the reactor coolant, and nitrogen from the closed gas blanketing system.

Assuming normal operation and one refueling outage per year per unit, the decay capacity of the gas decay tanks should permit a hold up time of 45 days before discharging the waste gases.

Hold up of gases for 60 days on a routine basis should be possible for the first 35 to 40 weeks of a core cycle. At this point in time, the gas volumes due to any daily load follow cycle begin to reach their peak. A cold shutdown during this period (week 35 through the end of the core cycle) would require release of some gases earlier than 60 days if degassing of the Reactor Coolant System was necessary.

Waste gases discharged to the environment must be sampled and analyzed to determine and record the activity to be released. Releases of waste gases must not exceed the federal standards of 10 CFR 50 Appendix I.

Table 11.1-6 contains an estimate of annual noble gas activity release based on 1% defective fuel clad, and 3250 MWt power with daily load reductions to 50% power for several hours.

11.1.3.3 Solid Wastes

Solid waste is processed in accordance with the Radioactive Waste Process Control Program.

Solid wastes consist of waste liquid concentrates, spent resins and miscellaneous materials, such as paper and glass ware. All solid wastes are packaged for shipment to an off-site processor or a

disposal facility or temporary on-site storage. Estimated yearly quantities are given in Table 11.1-1.

11.2 PLANT RADIATION SHIELDING

11.2.1 Design Basis

Auxiliary shielding for the Waste Disposal System and its storage components is designed to limit exposure rates to levels not exceeding 1 mrem/hr in normally occupied areas, to levels not exceeding 2.5 mrem/hr in periodically occupied areas and to levels not exceeding 15 mrem/hr in controlled occupancy areas.

All radiation shielding is designed for operation at maximum calculated thermal power and limits the normal operation radiation levels at the site boundary to below those levels allowed for continuous unrestricted exposure as set forth in 10 CFR 20. The plant is capable of continued safe operation with 1% fuel element defects.

In addition, the shielding provided ensures that in the unlikely event of a maximum design accident, the contained activity does not result in off-site radiation doses in excess of those given in Regulatory Guide 1.183 and 10 CFR 50.67.

Operating personnel at the plant are protected by adequate shielding, monitoring, and operating procedures. Each area in the plant is classified according to the exposure rate allowable in that area, based on the expected frequency and duration of occupancy. All plant areas capable of personnel occupancy are classified as one of the five zones of radiation level listed in Table 11.2-1. Typical Zone 1 areas are the offices, Control Room, turbine building and turbine plant service areas. Typical Zone 2 areas are the ground areas immediately around the Containment and Auxiliary Buildings, the corridors and local control spaces in the Auxiliary Building and the operating deck of the containment during reactor shutdown. Areas designated Zone 3 include the sampling room, fuel handling areas, and intermittently occupied work areas. Typical Zone 4 areas are valve galleries, and areas outside the crane wall in the containment during power operation. Zone 5 areas include shielded compartments in the Auxiliary Building, and the primary loop compartments during power operation.

All radiation and high radiation areas are appropriately marked and isolated in accordance with 10 CFR 20.

The shielding is divided into six categories according to function. These functions are the primary shielding, the secondary shielding, the accident shielding, the fuel transfer shielding, the auxiliary shielding, and the pump room shielding.

11.2.1.1.1 Primary Shield

The primary shield is designed to:

1. Reduce the neutron fluxes incident on the reactor vessel to limit the radiation induced increase in nil ductility transition temperature.
2. Attenuate the neutron flux sufficiently to prevent excessive activation of plant components.
3. Limit the gamma flux in the reactor vessel and the primary concrete shield to avoid excessive temperature gradients or dehydration of the primary shield.
4. Reduce the residual radiation from the core, reactor internals and reactor vessel to levels which will permit access to the region between the primary and secondary shields after plant shutdown.
5. Reduce the contribution of radiation leakage to obtain optimum division of the shielding between the primary and secondary shields.

11.2.1.1.2 Secondary Shielding

The main function of the secondary shielding is to attenuate the radiation originating in the reactor and the reactor coolant. The major source in the reactor coolant is the Nitrogen -16 activity, which is produced by neutron activation of oxygen during passage of the coolant through the core. The secondary shielding will limit the full power exposure rate outside the containment building from radioactivity inside the containment to less than 1 mrem/hr during normal plant operation.

11.2.1.1.3 Fuel Handling Shield

The fuel handling shield permits the safe removal and transfer of spent fuel assemblies and control rod clusters from the reactor vessel to the spent fuel pool. It is designed to attenuate radiation from spent fuel, control clusters, and reactor vessel internals to less than 2.5 mrem/hr at the refueling cavity water surface and less than 1.0 mrem/hr in the auxiliary building.

11.2.1.1.4 Accident Shield

The main purpose of the accident shield is to achieve the following after a maximum design accident:

a. prevent off-site radiation exposure in excess of Regulatory Guide 1.183 and 10 CFR 50.67;

b. to limit exposure to control room operators.

11.2.1.1.5 Auxiliary Shielding

The function of the auxiliary shielding is to protect personnel working near various system components in the Chemical and Volume Control System, the Residual Heat Removal System, the Waste Disposal System and the Sampling System. The shielding provided in the auxiliary building is designed to limit the exposure rate to less than the 2.5 mrem/hr listed for Occupational Access Zone 2 (see Table 11.2-1). In addition, those areas more frequently occupied are designed to limit the exposure rate to less than 1 mrem/hr.

11.2.1.1.6 Pump Room Shielding

The pump room shielding design makes it possible to uncover equipment openings in the Emergency Core Cooling System Pump Rooms quickly.

11.2.1.2 Shielding Design

11.2.1.2.1 Primary Shield

The primary shield consists of the core baffle, water annuli, barrel, thermal shield (all of which are within the reactor vessel), the reactor vessel wall, and a concrete structure surrounding the reactor vessel.

The primary shield immediately surrounding the reactor vessel consists of an annular reinforced concrete structure extending from the base of the containment to the operating deck. The lower portion of the shield is a minimum thickness of 7.0 feet of regular concrete and is an integral part of the main structural concrete support for the reactor vessel; it extends upward to the Reactor Vessel Flange. The upper portion of the shield is 4½ feet thick and forms an integral portion of the reactor cavity, extending upward to the operating deck.

A removable bulkhead, consisting of a metal shell with a concrete fill, approximately 20-3/4 inches thick joins the sidewalls of the refueling cavity and serves to isolate the reactor cavity from the upper containment during power operation. The section is removed when refueling.

The primary concrete shield is air cooled to prevent overheating and dehydration from the heat generated by radiation absorption in the concrete. Eight "windows" have been provided in the primary shield for insertion of the out-of-core nuclear instrumentation.

The primary shield design parameters and the calculated neutron and gamma fluxes are listed in Table 11.2-2.

11.2.1.2.2 Secondary Shield

The secondary shield surrounds the reactor coolant loops and the primary shield. It consists of the crane wall, the operating floor, and the reactor containment structure. The containment structure also serves as the accident shield.

The secondary shield includes the following: containment cylinder, 3½ ft; dome, 2½ ft; crane wall, 3 ft; and operating floor, 2½ ft. The crane wall will attenuate the radiation levels in the primary loop compartment from a value of 25 rem/hr to a level of approximately 25 mrem/hr in the annular region during full power operation. This permits limited access in the annulus while the reactor is at power. Radiation levels outside the Containment Building are less than 1 mrem/hr.

The secondary shield design parameters are listed in Table 11.2-3.

11.2.1.2.3 Accident Shield

The accident shield consists of the 3 ft. - 6 in. reinforced concrete cylinder capped by a hemispherical reinforced concrete dome 2 ft. -6 in. thick. Supplemental shielding has been provided for containment penetrations, which includes a 3 ft. concrete shield for the 609' elevation personnel lock. Smaller penetrations associated with piping, cables, and ventilation are directed into pipe tunnels, which are shielded with 24 inches of concrete. The control room is protected with concrete roof, floor and sidewalls at least 18 inches thick.

The accident shield design parameters are listed in Table 11.2-4.

11.2.1.2.4 Pump Room Shielding

The pump room shielding is unmortared high density concrete blocks seismically supported by a steel grating on each side of the opening. The shielding varies from 13" to 19" in thickness depending on the type of pump.

11.2.1.2.5 Fuel Handling Shield

The refueling cavity is irregularly shaped, formed by the upper portions of the primary shield concrete, and other sidewalls of varying thickness.

A portion of the cavity is used for storing the upper and lower internals during refueling periods which are shielded with concrete walls, 3 and 4 ft. thick, respectively. The remaining walls are a minimum thickness of 4½ ft. of concrete, and provide the shielding required during transfer of spent fuel.

The refueling cavity and refueling canal, flooded with borated water to an elevation of approximately 645' during refueling operations, provide a temporary water shield above the components being withdrawn from the reactor vessel. The water height during refueling is approximately 24 ft. above the reactor vessel flange. This height ensures that there will be sufficient water depth above the active fuel of a withdrawn fuel assembly to maintain exposures as low as reasonably achievable (ALARA).

The spent fuel assemblies and control rod clusters are remotely removed from the reactor containment through the horizontal fuel transfer tube and placed in the spent fuel pit. The transfer tube is shielded with a minimum of 5'-2" of concrete in all areas except a small piping area located under the transfer tube in the containment. The piping area is shielded with 2'-5" of concrete. It is posted as radiological conditions dictate and is protected with locked gates to ensure that personnel cannot enter this area while spent fuel is being transferred.

Fuel is stored in the spent fuel pool portion of the Auxiliary Building. Shielding for the spent fuel storage pool is provided by a minimum of 5’-2” thick concrete walls, and the pool is flooded to a level such that the water height is at least 23 feet above the top of the irradiated fuel rod assemblies seated in the storage racks. During spent fuel handling, sufficient water depth is maintained above the fuel assembly being handled to maintain exposures ALARA.

The original refueling shield design parameters are listed in Table 11.2-5.

11.2.1.2.6 Auxiliary Shielding

The auxiliary shield consists of concrete walls around certain components and piping which process reactor coolant. In some cases, the concrete block walls are removable to allow personnel access to equipment during maintenance periods. Access to the auxiliary building is allowed during reactor operation. Each equipment compartment is individually shielded so a compartment may be entered without having to shutdown and, possibly, to decontaminate equipment in an adjacent compartment. The shield material provided throughout the auxiliary building is normal density concrete (ρ= 2.3 g/cc). The principal auxiliary shielding provided and the design parameters are tabulated in Table 11.2-6.

11.2.1.3 Shielding Design Evaluation

The total core activity calculation is consistent with TID-14844 (Reference 1) and data from ORNL-2127 (Reference 2). Numerical values for isotopes which are important as health hazards are given in Table 11.2-7. The integrated exposures beyond the control room under accident conditions are calculated assuming the release of the following sources to the reactor containment (Per TID-14844):

a. 100% of the noble gases

b. 50% of the halogens

c. 1% of the remaining fission product inventory.

These sources, tabulated in Table 11.2-8, are assumed to be homogeneously distributed within the free volume of the reactor containment. The source intensity as a function of time after the accident is conservatively determined by considering decay only; no credit is taken for washdown or spray or ice condenser removal of iodine.

The results of the combined direct and scattered radiation dose calculations from activity confined in the containment following a TID-14884 release are plotted in Figure 11.2-1. This integrated exposure is shown as a function of distance from the containment for a various time intervals after the maximum hypothetical loss of coolant accident. Integrated exposures for time periods longer than one month are virtually equal to the 30 day integrated exposure. Use of the TID-14844 release represents the upper limits of radiation exposure. The combined direct and scattered radiation dose from the activity confined in the containment following a TID-14844 release is less than 20 mrem for two hours and less than 60 mrem for the duration of

the accident at distances beyond the site boundary. These values have not been included in the doses in Section 14.3.5 because they are relatively minor.

The exposure inside the control room, which is shielded with at least 18 inches of concrete, is negligible.

The exposures at the site boundary and in the control room would be reduced further by containment spray iodine removal and for the additional radiation shielding provided by the auxiliary building concrete structure and partitions located between the containment and the control room.

The radiation sources used for the design of the accident shielding in the Auxiliary Building are based upon a gap release accident. This would result in only the fission products which are in the fuel rod gaps being released to the containment.

The computed gap activities are based on buildup in the fuel from the fission process and diffusion to the fuel rod gap at rates dependent on the operating temperature. For the purposes of this analysis, the fuel pellets were divided into five concentric rings each with a release rate dependent on the mean fuel temperature within that ring. The diffusing isotope is assumed present in the gas gap when it has diffused to the boundary of its ring. The diffusion coefficient, D', for Xe and Kr in UO_2, varies with temperature through the following expression:

$$D'(T) = D'(1673)\exp\left(-\frac{E}{R}\right)\left(\frac{1}{T} - \frac{1}{1673}\right)$$

where:	E	is the activation energy
	$D'(1673)$	is the diffusion coefficient at 1673°K = 1 x 10^{-11} sec^{-1}
	T	is the temperature in °K
	R	is the gas constant

The above expression is valid for temperatures above 1100°C. Below 1100°C fission gas release occurs mainly by two temperature independent phenomena, recoil and knock-out, and is predicted by using D' at 1100°C. The value used for D' (1673°K), based on data at burnups greater than 10^{19} fissions/cc, accounts for possible fission gas release by other mechanisms and pellet cracking during irradiation.

The diffusion coefficient for iodine isotopes is assumed to be the same as for Xe and Kr. Toner and Scott (Reference 3) observed that iodine diffuses in UO_2 at about the same rate as Xe and

Kr and has about the same activation energy. Data surveyed and reported by Belle (Reference 4) indicates that iodine diffuses at slightly slower rates than do Xe and Kr.

For a full core cycle at 3391 MWt, the above analysis results in a pellet-clad gap activity of less than 3% of the dose equivalent equilibrium core iodine inventory. The noble gas activity present in the pellet-clad gap and assumed release to the containment is about 2.5% of the core inventory. The percentage of the total core activity present in the gap for each isotope is also listed in Table 11.2-7.

The core temperature distribution used in this analysis, based on hot channel factors, $F_{\Delta H} = 1.70$ and $F_q = 2.82$, is presented in Table 11.2-9.

The non-gaseous activity would be absorbed in the sump water, which flows in the residual heat removal loop and associated equipment. Table 11.2-10 lists the concentration of iodine isotopes in the recirculation loop at time zero after the loss-of-coolant accident, based on a realistic analysis of the gap inventories shown in Table 11.2-7, and the total sump water volume listed below. The energy release rate in the sump water at various times after the accident is found in Table 11.2-11. Calculations were based on the parameters listed below.

Parameters -- Gap-release Accident:	
Power level	3391 MWt
Equivalent percent fuel clad failure	100%
Percent fission product gap inventory circulating in sump water:	
Noble gases	0.0%
All others	100%
Sump Water Volume:	
Reactor Coolant System	93,960 gals.
Accumulators	30,000
Refueling Water Storage Tank	350,000
Total	473,960 gals.

It is assumed that following a loss of coolant accident, personnel access to vital areas will not be unduly limited nor will vital safety equipment be unduly degraded by the post-accident radiation fields. The results of the surface dose rates outside the concrete wall to each of the post-accident recirculating-system equipment compartments are listed in Table 11.2-12.

Each Residual Heat Removal Pump and Containment Spray Pump can be isolated from the system, drained and flushed of radioactivity to permit personnel access should post-accident maintenance of the pumps be required. Shielding for each pump compartment is sufficient to limit the integrated exposure from adjacent operating pumps and piping to less than 2 rem in any 8-hour period following the accident.

11.2.1.4 References

1. DiNunno, J. J., et. al., "Calculation of Distance Factors for Power and Test Reactor Sites," TID 14844, March 1962.
2. Bloncke, J. O. and Todd, Mary F., "Uranium-235 Fission Product Production as a Function of Thermal Neutron Flux, Irradiation Time, and Decay Time," (4 volumes) August and December 1957.
3. D. F. Toner and J. L. Scott, "Fission Product Release for UO2," Nuclear Safety Volume III No. 2, December 1961.
4. J. Belle, Uranium Dioxide: Properties and Nuclear Applications, Naval Reactors Division of Reactor Development, USAEC, 1961.

11.3 RADIATION MONITORING SYSTEM

11.3.1 Application of Design Criteria

Liquid release pathways are normally monitored by installed radiation detection instruments or composite sampling and analysis. Releases may continue when installed radiation detection equipment is inoperable for short periods of time by compensatory sampling and analysis. Liquid effluents of the plant are released to the circulating water system or the turbine room sump.

Gaseous releases are monitored by radiation detection instruments. The gaseous effluent from the steam generator blowdown tank vent is normally routed to the main condenser, except during startup, shutdown, and other periods of condenser inleakage or off normal chemistry when it may be vented to the atmosphere. In addition, any time there are non-condensible radioactive gases which may be released from the blowdown flash tank, such gases would also be present in the condensate system where they would be removed by the steam jet air ejectors and be detected by the applicable radiation monitor.

Accidental spills of radioactive liquids are maintained within the auxiliary building and collected in a drain tank. Any contaminated liquid effluent discharged to the condenser circulating water is monitored. Gaseous effluent from possible sources of accidental releases of radioactivity external to the reactor containment (e.g., the spent fuel pool and waste handling equipment) is exhausted from the unit vent and monitored by a radiation monitor. Gaseous batch releases are permitted only if the release can be made without exceeding federal standards and lack of reserve hold-up capacity requires such a release.

Monitoring and alarm instrumentation are provided for fuel and waste storage and handling areas to detect excessive radiation levels. Radiation monitors are provided to maintain surveillance over the release operation, but the permanent record of activity releases is provided by radiochemical analysis of known quantities of waste.

A controlled ventilation system removes gaseous radioactivity from the fuel storage and waste treating areas of the auxiliary building and discharges it to the atmosphere via the unit vent. Radiation area monitors are in continuous service in these areas to actuate high-activity alarms in the Control Room.

Waste handling and storage facilities are contained and equipment designed so that accidental releases directly to the atmosphere are monitored and will result in offsite doses below the limits of Regulatory Guide 1.183 and 10 CFR 50.67, as discussed in Section 11.1.1 and Chapter 14.

11.3.2 Design Basis

The Radiation Monitoring System is designed to perform two basic functions:

a. Warn of any radiation hazard which might develop, and

b. Give early warning of conditions which might lead to a radiation hazard or plant damage.

Instruments are located at selected points in and around the plant to detect, compute, and record the radiation levels. In the event the radiation level should rise above a desired setpoint, an alarm is initiated in the Control Room. The Radiation Monitoring System operates in conjunction with regular and special radiation surveys and with chemical and radiochemical analyses performed by the plant staff. Adequate information and warning is thereby provided for the continued safe operation of the plant and assurance that personnel exposure does not exceed 10 CFR 20 limits.

The components of the Radiation Monitoring System are designed to operate during all expected environmental conditions for normal operation. Specific components are designed to operate during adverse or accident plant conditions. In addition, process and area radiation monitors are of a nonsaturating design so that they "peg" full scale if exposed to radiation levels up to 100 times full scale indication.

11.3.2.1 Gaseous Release Pathways

1. Containment and Instrument Room Exhaust - Releases are through the Unit Vent. Noble gas activity and release rates are monitored and recorded. Releases are on an intermittent basis as the containment is purged only periodically. The containment atmosphere is sampled prior to release and containment purge and exhaust isolation valves will close on a Containment Ventilation Isolation (CVI) actuation. CVI can be actuated by either the containment airborne monitors or by the normal range containment area monitors. The Containment Airborne and Unit Vent monitor systems also sample iodine and particulate activity. Operation of the containment purge and exhaust system is controlled by Plant Technical Specifications.

2. Auxiliary Building Ventilation - The activity in the exhaust depends on leakage into the Auxiliary Building atmosphere from the primary systems, and it is expected to be very low.

   Releases are through the unit vent. Activity is measured prior to the release point of the unit vent. High radioactivity will be alarmed in the control room.

3. Steam Jet Air Ejector - A continuous release of activity can exist only during periods when the ejectors are in service, coincident with steam generator primary to secondary leakage. The steam jet air ejector exhaust is continuously monitored when the ejectors are in service.

4. Gland Seal Condenser Exhaust - A continuous release of activity can exist only during periods when the exhauster is in service, coincident with steam generator primary to secondary leakage. The gland seal condenser exhaust is continuously monitored when the exhauster is in service .

5. Steam Generator Blowdown Exhaust - The blowdown is normally sent to the normal blowdown flash tank from which the non-condensibles are routed to and released through the main condenser. During off normal chemistry conditions, unit start-up, or unit shutdown the non-condensibles are released to the atmosphere via the S/G blowdown flash tank vent. The steam generator blowdown is continuously monitored at a point representative of its activity prior to the non-condensibles being separated.

6. Main Steam PORV/Safety Relief Valves - The main steam power operated relief valves and safety valves provide pressure relief on each steam lead if steam

pressure exceeds normal operating values. They also allow plant cooldown by steam discharge to the atmosphere if the turbine by-pass system is not available. The PORV discharge lines are continuously monitored.

7. Waste Gas Decay Tanks - These tanks are batch released through the unit vent. Their total activity and release rates are monitored by a radiation monitor and a flow meter and both are recorded. Isolation valves on the discharge header from the tanks close on a high radiation signal. The contents of the tanks are sampled prior to release and analyzed to determine isotopic concentrations.
8. Miscellaneous Ventilation - Releases are through the unit vent from ventilation systems such as Spent Fuel pool, nuclear sampling room, etc. Noble gas activity and release rates are monitored and recorded. High radioactivity will be alarmed in the Control Room.

Meteorological conditions during periods of release from the above systems will be obtained from the meteorological program.

The methods and formulas for computation of doses and radiation monitor setpoints associated with the liquid and gaseous releases are given in the Cook Nuclear Plant's Off-site Dose Calculations Manual (ODCM).

11.3.2.2 Liquid Release Pathways

Radioactive liquids are released through the waste disposal system monitor tanks, steam generator blowdown, essential service water system and turbine room sump. Radioactive releases are classified as either batch or continuous. Release pathways are monitored by radiation detection instruments or in the case of the turbine room sumps, a composite sampler is used.

1. Waste Disposal System-Radioactive effluent releases from the WDS are batch releases. Tanks from which a batch originates are re-circulated, sampled and analyzed. Releases are monitored by the liquid waste disposal radiation monitor, or if inoperable, by collecting and analyzing two separate and independent samples. The radiation monitor terminates the release on high radiation alarm or fail alarms. Releases from the WDS are directed to the discharge tunnels of the circulating water system.

2. Steam Generator Blowdown-Radioactive effluents released by the steam generator blowdown system are discharged to the forebay of the circulating water system and are considered continuous releases. The process stream is monitored by one of two installed radiation monitors. Releases are terminated automatically on a high radiation alarm or low flow alarm.
3. Essential Service Water System–Radioactive effluents released by the essential service water system are discharged to the circulating water system. The process streams are monitored by installed radiation monitors. It is not equipped with automatic isolation. This is a normally non-radioactive effluent release path and is considered a continuous release.
4. Turbine Room Sump-The Turbine Room Sump receives secondary drain waste. Only during periods of primary to secondary leakage does the waste stream contain radioactive material. The sump discharges automatically to the Absorption pond. The discharge of the sump is sampled by a auto-compositor.

Before a batch may be released, the tank is sampled and the sample analyzed. If the radioactivity level of the sample is found to be within acceptable limits, the liquid wastes will be released, monitored, and recorded. At the same time, the rate of the liquid release is measured by a flow meter. By using the rate of liquid waste releases, the rate of flow of the condenser cooling water, the activity of the liquid waste released, the rate of activity release and the concentration of activity in the condenser cooling water can be determined.

Continuous effluent releases are sampled and analyzed in accordance with established plant programs.

11.3.3 General Description and Operation

The Radiation Monitoring System is divided into the following sub-systems:

a. The Process Radiation Monitoring System monitors various fluid streams for indication of increasing radiation levels.
b. The Area Radiation Monitoring System monitors radiation in certain areas of the plant.
c. Environmental radiation monitoring program monitors radiation in the area surrounding the plant as described in Sub-Chapter 2.7.

The original radiation monitoring channel equipment, including chassis with signal conditioning equipment, controls, power supplies, indicators and alarms is centralized in cabinets located in the control rooms for convenient operator access. The Plant Process Computer (PPC) is used to sequentially record each Unit 1 and Unit 2 monitoring channel.

This equipment has been supplemented and partially replaced by a system of distributed, multi-channel field data acquisition units. Each field unit services one or more detector channels. It measures and records the channel readings, performs alarm and other status checks and initiates trip functions (if applicable). Each field unit is connected via isolation devices to two data communication lines. Each line is connected via isolation devices to associated RadServ System computers. The RadServ System computers poll the field units and make channel readings, status, and history information available to the control room operators through displays and annunciators. Channel status changes are reported and recorded as they occur. Printers can print a hard copy of historical data on demand.

Typical sensitivity ranges of the various radiation monitor channels are given in Table 11.3-1 and are based on the first isotope listed in the last column of the table.

The monitor channels are response checked using a radioactive source, tested electronically, and calibrated by temporarily lowering their setpoints below their background radiation levels. The detectors are calibrated using appropriate calibrated sources at a frequency listed in the Technical Specifications and/or the ODCM. Effluent monitor setpoints are determined in accordance with the ODCM and are designed to aid in maintaining ODCM limits.

11.3.3.1 Process Radiation Monitoring System

This system consists of (original and newer) channels which monitor radiation levels in various plant operating systems. High radiation level alarms are annunciated and identified in the control room.

The radiation monitoring channels employ instrument failure alarms at the radiation monitoring cabinets, control board annunciator, and at local indicators (where provided). Instrument failure alarms are initiated upon failure of the radiation monitor, loss of detector signal or loss of power.

Gaseous effluents are scanned, alarmed, and recorded thereby providing a complete history of abnormal occurrences for evaluation.

These radiation monitoring channels are shown in Table 11.3-3.

11.3.3.2 Area Radiation Monitoring System

This system consists of channels, which monitor radiation levels in various plant areas. Certain of these monitors have been upgraded. Both the original and newer monitors are shown in Table 11.3-1.

Each original monitor consists of a fixed position Geiger-Mueller detector with local indicator and check source. An associated readout drawer in the control room provides high radiation and failure alarms, and initiates trips (if required). The channel readings are logged via the Plant Process Computer (PPC).

Newer monitors consist of either Geiger-Mueller or ion chamber detectors, with check sources, connected to multi-channel field-mounted data acquisition units. Selected channels are provided with individual indicator/alarm units near the detector.

Two high range ion chamber detectors monitor each containment. One is located in the upper containment while the second is in the lower containment about 180^{o} apart from the first. These accident monitors are separate from the other area channels. Each has a dedicated readout module in the control room with a multi-range indicator, status lights, and test circuits. An isolated output from each module is sent to an associated field unit to provide for recording and supplemental data access via the RadServ system.

11.3.4 Reactor Coolant Activity Monitoring

Refueling shutdown programs at operating Westinghouse PWRs indicate that, during cooldown and depressurization of the Reactor Coolant System (RCS), a release of activated corrosion products and fission products from defective fuel has been found to increase the coolant activity level above that experienced during steady state operation. However, high coolant activity is avoided by implementing established shutdown procedures. These procedures include purification of the RCS through the cation and mixed bed demineralizers and system degasification.

Table 11.3-2 illustrates the calculated coolant activity increases of several isotopes for the Donald C. Cook Plant. This table lists the calculated activities during steady state operation before refueling shutdown outage and calculated peak activities during plant cooldown operations. These data are based on measurements from an operating PWR which is similar in

design to the Cook Nuclear Plant and has operated with fuel defects. The measured activity levels are also included in Table 11.3-2.

The dominant non-gaseous fission product released to the coolant during system depressurization is found to be Iodine-131. The activity level in the coolant was observed to be higher than the normal operating level for nearly a week following initial plant shutdown. Although lesser in magnitude, the other fission product particulates (e.g., cesium isotopes) exhibited a similar pattern of release and removal by purification. It is reasonable to project this data to the Cook Nuclear Plant since the purification constants are similar and as it is standard operating procedure to purify the coolant through the demineralizers during plant cooldown. Fission gas data from operating plants indicate a maximum increase of approximately 1.5 over the normal coolant gas activity concentration. However, the system degasification procedures are implemented prior to and during shutdown, and have proven to be an effective means for reducing the gaseous activity concentration and controlling the activity to levels lower than the steady state value during the entire cooldown and depressurization procedure.

The corrosion product activity releases have been determined to be predominantly dissolved Cobalt-58. From Table 11.3-2, it is noted that this contribution is less than 1% of the total expected coolant activity and is hence considered to be a minor contribution.

Since continued operation of the purification system is standard operating procedure during plant cooldown and since means for system degasification are available for fission gas removal, the total activity concentration in the coolant can be maintained within Technical Specification limits throughout the plant shutdown, while considering the additional activity inventory released during system cooldown and depressurization. The coolant activity concentrations and inventories during the shutdown and prior to plant startup are established by chemical analysis of samples for the Reactor Coolant System.

11.3.5 Improved Inplant Iodine Instrumentation under Accident Conditions

Mobile continuous air monitors are available for use in emergencies involving airborne radioactivity concerns. This equipment includes particulate, radioiodine and noble gas monitors. Regulated air samples are also available which require sample collection and laboratory analysis.

Emergency response equipment is located in the basement assembly area for counting radioactive samples.

11.4 PLANT HEALTH PHYSICS PROGRAM

An extensive health physics program under the supervision of trained professional and technical personnel is established for the plant and for the Independent Spent Fuel Storage Installation (ISFSI). Appropriate administrative controls are developed to ensure that procedures and other requirements involving radiological safety considerations are strictly adhered to.

11.4.1 Facilities

Facilities for radiation protection, personnel and equipment decontamination, and chemical and radiochemical analysis are located in the auxiliary and turbine buildings.

a. Access Control Facilities

Access control facilities are located at the entrances to the radiologically restricted areas of the plant. These facilities are used to control normal access of personnel into posted radiologically controlled areas where the use of protective clothing and/or special radiation monitoring equipment might be necessary. Protective clothing and calibrated radiation monitoring instrumentation are available for personnel to use as needed.

Radiation protection offices are located at the access control facilities.

Personnel decontamination facilities are located at the access control facilities where appropriate measures may be taken to decontaminate personnel if needed. The personnel decontamination facilities contain a wash basin, shower (east facility only), and a radiation survey instrument for monitoring personnel for residual levels of contamination following decontamination efforts.

b. Decontamination Facilities

Personnel decontamination facilities are located at the access control facilities as described above. The liquid wastes collected are normally sent to the waste disposal system prior to release.

Tools and equipment are normally decontaminated in the hot tool decontamination facility located on the 633' elevation.

Protective clothing is processed by an off-site vendor. Soiled protective clothing is periodically collected, packaged, and transported to a vendor for cleaning or disposal. Sufficient protective clothing is maintained on-site.

c. Chemistry Facilities

A sampling room where radioactive and potentially radioactive samples are collected is located in the auxiliary building.

A chemical laboratory where radioactive samples are analyzed is located in the radiologically restricted area of the auxiliary building near the auxiliary building access control facility.

A chemical laboratory, also used for analyzing non-radioactive samples, is located in the turbine building.

A chemistry counting room where samples are analyzed for radioactivity is located in the radiologically restricted area near the auxiliary building access control facility.

Chemical Supervisor's offices are located near the turbine side (east) access control facility.

Eyewash stations and, when appropriate, safety showers are located near all chemical handling and analysis areas.

d. Hot Laboratory

This laboratory can be used for radiochemical work, such as chemical separations, etc., in addition to routine water chemistry. The hot laboratory includes three enclosed ventilated hoods and a ventilated storage cabinet. The flow from all these vents, as well as the rest of the ventilation flow from this area, is filtered and monitored by the auxiliary building ventilation system as described in Subchapter 9.9. Liquid wastes from the sinks in this laboratory are directed to the waste disposal system for processing as described in Subchapter 11.1. The waste liquid from the deluge shower, the face-eye wash and the floor drains are also treated as contaminated liquid waste as described in Section 11.1 of the FSAR.

e. Counting Room

This room is provided for the measurement of radioactivity in contained liquid, solid, gaseous, or particulate collection samples. No liquid wastes are disposed of in this room. Solid wastes are handled as described in Subchapter 11.1.

The ventilation for this room is part of the auxiliary building ventilation system.

f. Radiation Protection Calibration Facility

The calibration facility is located off the 609' Auxiliary Building Crane Bay. This provides an area for storage and calibration of instruments and storage of higher activity radioactive calibration sources. The room is also used as a repair facility for radiation protection equipment.

Access to the facility is normally controlled using the computer controlled keycard system. Methods are available for locking the facility should the computer system fail.

11.4.2 Radiation Control

Personnel exposure to radiation is maintained as low as reasonably achievable (ALARA) by controlling access, through radiation work permits and by the use of shielding when appropriate.

11.4.3 Access Control

Access is controlled to the radiologically restricted area of the plant and the ISFSI on the basis of radiation levels and/or the presence of radioactive materials or contamination.

Any area in which radioactive material is stored, handled or processed, or in which the radiation dose from external sources exceeds 2.0 mR in any one hour is designated a radiologically restricted area. These areas are designated by signs such as RESTRICTED AREA, RADIOACTIVE MATERIALS, etc.

Within the radiologically restricted area access is further controlled based on radiation and contamination levels. The entrances to all areas within the restricted area are posted with signs stating; CAUTION, DANGER, OR GRAVE DANGER, and the appropriate area designation(s): radiation area, high radiation area, locked high radiation area, very high radiation area, contamination area, high contamination area, and / or airborne radioactivity area.

Radiation protection personnel make routine surveys of accessible areas of the plant and the ISFSI to establish current status of the radiation levels in these areas. Radiological information is posted showing radiation levels (and significant radiation sources) in the area.

11.4.4 Contamination Control

The spread of contamination from one area to another is minimized by the use of step-off pads. Bags are used to carry contaminated tools and equipment.

Personnel monitoring devices such as, count rate meters with Geiger-Mueller detectors, hand and foot monitors and whole body contamination monitors are located throughout the plant's radiological restricted area for use by personnel to monitor themselves for contamination. Personnel are also monitored for contamination in an access control facility upon leaving the auxiliary building or a posted contamination area.

All personnel are monitored for contamination when leaving the plant's Protected Area through the Security access control area.

Radiation protection personnel conduct routine contamination surveys of accessible areas of the plant and the ISFSI. Any area contaminated above set procedural levels is posted appropriately and decontaminated as soon as, and if, practical. Radiological information is available showing the contamination and radiation levels. Appropriate protective clothing to be worn when entering the radiologically restricted area is specified on radiation work permits or by radiation protection personnel.

11.4.5 Personnel Contamination Control

The potential for personnel contamination is minimized by the use of several types of protective clothing. The type and number of each specific piece of protective clothing to be worn is specified by the radiation work permit or by radiation protection personnel based on known or suspected contamination levels.

Normally, most of the plant and the ISFSI are accessible to personnel in street or conventional clothing.

11.4.6 Airborne Contamination Control

Airborne contamination is minimized by maintaining loose surface contamination at a low level. Efforts are made to install temporary process ventilation control devices to avoid approaching or exceeding 10 CFR 20 levels. If the use of this equipment is not feasible or calculations show that the total dose would be lower, a provision may be made for personnel to use respiratory protection equipment to minimize personnel exposure to airborne radioactivity. Allowances are

made for determining if personnel in radiologically restricted areas are subjected to concentrations in excess of Appendix B, Table 1, Column 3 of 10 CFR 20.

Several types of respiratory protection are available for personnel use:

a. Full-Face Cartridge Mask

b. Supplied

1. Air
2. Full-Face Mask, Constant Airflow
3. Polyethylene Hood

c. Self-Contained Breathing Apparatus

11.4.7 External Radiation Dose Determination

Personnel expected to receive occupational dose while on site are issued self-reading dosimeters and/or thermoluminescent dosimeters prior to entering the radiologically restricted areas. Thermoluminescent dosimeters are normally used to measure personnel radiation doses and are normally the primary basis for determining the dose of record. Self-reading dosimeters are normally used to provide a continuous readout of occupational dose accumulated between thermoluminescent dosimeter processing and also provide a backup to the thermoluminescent dosimeter data and may be used either concurrently with or in lieu of TLDs in some circumstances. Dose records are maintained on personnel that receive dose while on site.

Extremity dosimeters are issued as required. Neutron exposure monitoring is accomplished through dosimeters or the use of measured neutron dose equivalent rates and stay times.

Provisions are in place for the determination of skin dose in the event of significant skin contamination or upon repeated entry into areas having a significant airborne noble gas concentration.

An annual tabulation of the number of plant, utility, and other personnel receiving dose greater than 100 mrem in a calendar year and the associated collective dose according to work and job function is included in the annual operating report. In addition, all doses are reported as required by 10 CFR 20.

Plant supervisors are kept informed of plant personnel doses as results are received from the computerized radiation protection system.

11.4.8 Internal Radiation Dose Determination

Passive internal monitoring is routinely performed for all workers in order to detect external and/or internal deposition of radioactive materials. Additionally, investigational whole body counts are also conducted when a potential intake of radioactive material may have occurred. Any committed effective dose equivalent determination results are maintained with other dose records consistent with records retention guidelines. The determination of dose will be based upon approved radiological protection methodology.

11.4.9 Radiation Protection / Radiochemistry Instrumentation

a. Counting Room Instrumentation

   Counting room instrumentation includes gamma spectroscopy equipment, tritium analysis equipment and alpha and beta counting equipment (low background).

b. Portable Radiation Detection Instrumentation

   Portable instrumentation includes devices for measuring thermal and fast neutrons; alpha contamination; low, mid, and high range gamma exposure rates; and personnel contamination (e.g., friskers). A transfer standard ionization chamber is available.

c. Air Sampling Instrumentation

   Air sampling instrumentation includes low and high volume air samplers and continuous air samplers which measure for particulate, radioiodines and noble gases.

d. Personnel Monitoring Instrumentation

   Personnel monitoring instrumentation includes devices appropriate for monitoring deep, shallow, and lens dose equivalents are used.

e. Emergency Instrumentation

   Emergency instrumentation located throughout the plant and offsite include low and high range exposure rate meters, air samplers and personnel monitoring devices.

f. Self-reading dosimeters of appropriate range.

11.4.10 Tests and Inspections

a. Shielding

To assure shielding integrity is maintained, radiation surveys of plant and ISFSI areas are performed routinely.

b. Area and Process Radiation Monitors

Each technical specification area and process monitor is regularly tested to assure that the:

1. Calibration of the monitor is correct.
2. Alarm and trip points function properly.

In addition, each non-technical specification radiation monitor is regularly tested to ensure the calibration of the monitor is correct.

c. Portable and Semi-Portable Radiation Monitors

This equipment is regularly tested to assure correct calibration and function.

11.4.10.1 Methods, Frequency, and Standards Used in Calibrating Instruments

1. Method

Beta-gamma portable survey instruments and portable count rate instruments are calibrated as described in Subchapter 11.4, using written procedures. Neutron survey instruments are typically calibrated offsite. Counting and measuring instruments are calibrated using low level calibration sources.

Small check sources are available for checking the operation and response of survey instruments, portal monitors, and contamination monitoring instruments.

2. Frequency

Radiation protection instruments are periodically checked, repaired, and/or calibrated by qualified personnel.

3. Standards

The calibration sources in the calibration facility are themselves calibrated using an instrument for which the standardization is traceable to the National Institute of Standards and Technology.

11.5 STEAM GENERATOR BLOWDOWN TREATMENT SYSTEM

11.5.1 Design Basis

Should fuel clad defects and steam generator tube leakage occur simultaneously, fission products would have a transfer path to the environment via the steam generator blowdown. This would result in immediate isolation of the blowdown system and eventual forced shutdown of the unit because the common drain line to the screenhouse from the normal and start-up steam generator blowdown flash tanks would be isolated upon receipt of a high radiation signal.

The results of an analysis of the radioactive materials present, their relative concentrations and their release rate were incorporated into the design of the Steam Generator Blowdown Treatment (SGBT) system to bring the effluent within allowable release limits and thus avoid an immediate plant shutdown. Table 11.5-1 provides a comparison of the design basis coolant fission product concentrations for the D. C. Cook Plant with those measured at other operational PWRs. Discharge streams are appropriately monitored and safety features are incorporated to preclude releases in excess of the limits of 10 CFR 20.

The treatment system conditions the blowdown liquid to acceptable release quality on a once through basis. It is designed to permit continuing operation for a sufficient period to prepare for a planned maintenance shutdown. Included in the general design basis is operating with a reactor coolant system activity level equivalent to that resulting from 1% fuel defects, with a 2 gpm reactor coolant system leak through the steam generators tubes and a unit normal blowdown rate of 50 gpm.

The blowdown conditioning consists of the removal of radioactive ions and particulates. The effluent is discharged to the circulating water through the screenhouse forebay.

11.5.2 System Design and Operation

The steam generator blowdown treatment system is shown on Figure 11.5-1. The system is integrated into the unit blowdown and release system in that the blowdown control valves, blowdown tank and vent, and the drain piping to the turbine room sump overflow into the screenhouse forebay are utilized. It forms a flow path through the treatment system, discharging through the normal blowdown discharge into the forebay. The treatment system consists of a pump, a cooler, three mixed bed demineralizers in series, valves, and instrumentation.

The blowdown fluid goes from the normal blowdown tank to the blowdown treatment pump suction. It passes through a cooler on the pump discharge where its temperature is reduced to a level compatible with the demineralizer resins. The bulk of the cooled fluid is passed through a flow control valve to the demineralizer. A small portion is recirculated to the pump suction.

Liquid effluent radiation monitors, R-24 (1-DRS-3201, 2-DRS-4201), are positioned after the second demineralizer in the steam generator blowdown treatment system, to monitor the SGBT liquid effluent before it is discharged to the third demineralizer. This monitor is discussed in Section 11.3.

This third resin bed prevents inadvertent release should the first two demineralizer resin beds become exhausted. In normal treatment system operation this backup bed has no duty since full treatment takes place through the first two demineralizers. Therefore, no credit is taken for the last bed in the analysis. This arrangement provides for detection of radioactivity prior to the last resin bed. When a high radiation level is detected, an alarm is sounded in the control room, the isolation valves in the blowdown lines are automatically closed, and the blowdown treatment pump is automatically stopped. Periodic manual sampling provides the back-up to the automatic monitoring. Sample points are provided for this purpose.

The signals from radiation monitors R-24 (1-DRS-3201, 2-DRS-4201), are recorded continuously when the blowdown treatment system is in operation. This data, supplemented by the blowdown treatment system flow measurements and data from samples which may be taken of the discharge of the third resin bed, may be used to quantify the radioactive releases by this pathway.

Because of the ability to replace spent resins in a matter of hours, and because of the presence of a backup resin bed, it will be possible to treat all normal blowdown fluid when radioactivity is detected. If necessary, blowdown can be temporarily shut off while resins are being changed.

This system is not a safeguard system since failure of the system does not affect the reactor or public safety. It is an operational aid to avoid a reduction in plant availability and as such did not require safeguard design standards.

Component Parameters

For Blowdown Treatment System Component parameters, see Table 11.5-2.

11.6 RADIOACTIVE MATERIALS SAFETY

11.6.1 Materials Safety Program

Licensed material is used, handled by, or under direction of one of those designated as an individual user in the license. Each individual using such radioactive material is familiar with the restrictions and limitations placed upon that particular source.

An inventory of licensed material on site is maintained and periodically updated. The inventory record contains, as a minimum, the use and locations of licensed material, and the receipt date and final disposition of material no longer in use.

11.6.1.1 Security

Sealed sources, with the exception of those installed in or on equipment and small check or calibration sources, are kept under lock and key when not in use.

11.6.1.2 Source Handling

Whenever radioactive sources are handled or used, care is taken to avoid unnecessary exposure, the spread of contamination, or damage to the source. In addition, a Radiation Work Permit is required for the following situations dealing with licensed sources:

1. Any time a sealed source capable of giving an exposure greater than 5 mrem/hr at 30 centimeters is used or handled.
2. Any time a sealed source is installed in, or removed from, equipment on which it is normally installed.

11.6.1.3 Material Handling of Special Nuclear Material (SNM)

A Nuclear Materials Management Group has the overall responsibility for SNM and the associated inventory records. The Radiation Protection Manager is responsible for radioactive sources used by Radiation Protection and associated records.

There are four Item Control Areas (ICA) designated for the plant. The custodian for each of these areas is responsible for all SNM entering, leaving, or being stored in this area. (except radioactive sources used by Radiation Protection and controlled by Radiation Protection

Manager). All material transfer documents for this ICA are signed by the ICA Custodian or alternate. The chain of responsibility between the Custodian and the Nuclear Materials Management Group is shown in Figure 11.6-1.

Figure 11.6-2 presents a diagram of the flow of SNM except radioactive sources used by Radiation Protection through the plant and at the ISFSI. Responsibility for the control and accountability of each physical unit of SNM begins with the on-site receipt. The responsibility for this control and accountability terminates for each physical unit when the physical unit of SNM is shipped off-site.

Each time fuel is transferred into or out of the ICA, the transfer is documented by appropriate entries on an ICA Transfer Form.

Inventory of all SNM must be taken on a periodic basis.

11.6.1.4 SNM Transfer Procedure

Fuel assembly, fission chamber detector, and moveable miniature neutron flux detector (MMNFD) movement from one ICA to another ICA is not to be initiated until an ICA Transfer Form has been approved by the Reactor Engineering Supervisor or alternate*. The ICA Transfer Form shows the fuel assembly and/or fission chamber detectors and/or MMNFD involved, their origin, destination, and approval by the Reactor Engineering Supervisor or alternate; and permits transfer of the fuel assembly, fission chamber detectors or MMNFD across the boundaries of the ICAs involved.

Additionally, SNM movement within an ICA is administratively controlled by use of the ICA Internal Transfer Form. Each Internal Transfer Form must be signed by the Reactor Engineering Supervisor or alternate before it is considered approved. Each ICA Internal Transfer Form and ICA Transfer Form has a limited lifetime of ten calendar days as delineated on the form with the exception of new fuel receipt.

The detailed procedures for material handling and transfer of all Special Nuclear Material are described in the SNM Accountability Manual of the Donald C. Cook Nuclear Plant. and Radiation Protection procedures for RP sources, which are considered SNM

* During refueling, new fuel receipt, fuel exams and fuel shuffles, the approved fuel shuffle sequence may be substituted for the ICA transfer and internal transfer forms.

11.6.2 Personnel and Procedures

The key personnel responsible for handling and monitoring the Special Nuclear Material (except radioactive sources used by Radiation Protection and controlled by Radiation Protection Manager) are identified by title in the Special Nuclear Materials Accountability Manual.

Radiation Safety Instructions

The Radiation Protection Plan presents the philosophy and guidelines to be used to control the exposure to radiation and radioactive materials, and to effectively restrict the exposure of personnel within the plant (including the ISFSI) and members of the general public to ionizing radiation resulting from the operation of the plant and the ISFSI.

Safe handling and usage of radioactive materials are also described in Radiation Protection Procedures, Laboratory Procedures, Fuel Handling Procedures, and the Special Nuclear Material Accountability Manual.

11.6.3 Required Materials

The isotope, quantity, form and use for all required by-product, source and special nuclear material for the Donald C. Cook Nuclear Plant are identified in the Facility Operating License and ISFSI General License and subject to the conditions identified in the Technical Specifications for the Plant and ISFSI.

11.6.4 Radioactive Waste Storage

The Radioactive Material Building was designed and constructed with the primary purpose of storing low level radioactive waste. The building is located behind the Training Building about one half mile from the auxiliary and containment buildings. The building has two waste storage areas: the cell area and the dry active waste (DAW) area.

The cell area is used for storing higher activity wastes. There are twelve cells, each with a pair of two-foot thick covers. High activity wastes such as resins, primary system filters, or high activity DAW is packaged in disposal containers prior to placement into one of the storage cells. The number of disposal containers that can be stored in a cell is dependent on the size of the containers and the amount of activity. Accident analysis involving a fire within a cell has resulted in a limit of 2,000 curies per cell. This limited curie content if released would result in

a dose that is a small fraction of the 10CFR100 limits. Calculated doserates on the exterior of the building based on the 2,000 curie per cell limit results in a doserate of less than 1 mR/year at the nearest site boundary.

The DAW area is used for the storage of lower activity dry active waste. The DAW is packaged in steel boxes or drums prior to being put into storage in the DAW area. Due to the low activity nature of DAW, the capacity of the DAW area is limited by the physical space and not the amount of radioactivity being stored.

A surveillance program of waste in storage has been established. On a predetermined frequency visual inspection of the stored waste is conducted to ensure there is no leakage or deterioration of the waste packaging. Additionally, routine radiological surveys are performed in the waste storage areas.

11.6.5 Spent Fuel Dry Storage

The Donald C. Cook Nuclear Plant has established an Independent Spent Fuel Storage Installation (ISFSI) on an approximately 4.5 acre elevated site located about 1/3 mile east of the Unit 1 and Unit 2 Containment Buildings near the center of the Owner Controlled Area. The ISFSI site is being implemented in two phases. The Phase 1 ISFSI concrete pad has a capacity for 94 vertical spent fuel storage casks. The ISFSI area includes space for a Phase 2 expansion that would accommodate an additional 80 storage casks, if needed.

The ISFSI site boundary has a security fence with a 20 ft wide isolation zone-on each side of the fence to establish a separate protected area. A nuisance fence is included on the perimeter of the outside isolation zone to minimize intrusion detection alarms. The ISFSI has one access point on the northwest side of the site, facing the access road from the plant.

The D.C. Cook ISFSI uses the Holtec HI-STORM 100S Version B vertical cask storage overpack and the Holtec MPC-32 multi-purpose canister (MPC).

The MPC provides the confinement boundary for the stored fuel. MPC-32 is a welded, cylindrical canister with a honeycombed fuel basket. All MPC confinement boundary components are made entirely of stainless steel. The honeycombed basket, which is equipped with neutron absorbers, provides criticality control. The MPC numbered suffix indicates the maximum number of fuel assemblies permitted to be loaded in the MPC.

The HI-STORM 100S Version B storage overpack provides shielding and structural protection of the MPC during storage. The HI-STORM 100S Version B overpack design includes a lid

which incorporates the air outlet ducts. The overpack is a heavy-walled steel and concrete, cylindrical vessel. Its side wall consists of plain (un-reinforced) concrete that is enclosed-between inner and outer carbon steel shells. The overpack has four air inlets at the bottom and four air outlets at the top to allow air to circulate naturally through the cavity to cool the MPC inside. The inner shell has supports attached to its interior surface to guide the MPC during insertion and removal, and allow cooling air to circulate through the overpack. A loaded MPC is stored within the HI-STORM 100S Version B storage overpack in a vertical orientation.

WASTE DISPOSAL SYSTEM PERFORMANCE DATA

Plant Design Life	40 years
Normal process capacity, liquids	15 gpm
Evaporator load factor	32%
Annual approximate liquid discharge [1]	
Volume (2 units)	2,415,000 gal.
Tritium Activity [2] (2 units)	2.0×10^3 curies
Other (2 units)	1.125 curies/year
Annual gaseous discharge	
Activity (2 units)	11,957 curies/year
Annual drummed solids shipped for burial [3]	27,624 ft^3/year

[1] Estimate based on Table 11.1-4, equilibrium cycle.
[2] Volume is an annual average based on actual shipments for two units from 1979 through 1988.
[3] Quantity based on approximate actual discharge for 1986.

WASTE DISPOSAL COMPONENTS CODE REQUIREMENTS[1]

COMPONENT	CODE
Chemical Drain Tank	No code
Reactor Coolant Drain Tanks	ASME III, [2] Class C
Sump Tanks	No code
Waste Holdup Tank	No code
Waste Evaporator Condensate Tank	No code
Laundry and Hot Shower Tank	No code
Waste Evaporator(s)	No code
Waste Filters	ASME III,(2) Class C
Piping and Valves	USAS-B31.1[3], Section 1 ASME III Appendix F [4]
Gas Decay Tank	ASME III,(2) Class C
Spent Resin Storage Tank	ASME III,(2) Class C
Waste Evaporator Condensate Demineralizer	ASME III,(2) Class C
Waste Evaporator Condensate Filter	ASME III,(2) Class C
Waste Evaporator Bottoms Storage Tank	No code

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

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

3 USAS-B31.1 American Standards Association Code for pressure piping and special nuclear cases where applicable.

4 The evaluation criteria of ASME III Appendix F (faulted conditions) is applicable to 1) piping from normally closed PRT drain line isolation valve and the RCDT drain line check valve inside containment to the normally closed isolation valve outside containment (U-1 & U-2); and 2) piping between containment sump pump discharge check valves inside containment and discharge isolation valve outside containment (U-1 only).

COMPONENT SUMMARY DATA

TANKS	QUANTITY	TYPE	VOLUME	DESIGN PRESSURE	DESIGN TEMP.°F	MATERIAL [1]
Reactor Coolant Drain (per unit)	1	Horiz	350 gal	25 psig	267	ss
Laundry & Hot Shower	2 [2]	Vert	600 gal	Atm	180	ss
Chemical Drain	1 [2]	Vert	600 gal	Atm	180	ss
Clean Sump	1 [2]	Vert	600 gal	Atm	180	ss
Station Drainage Sump	1 [2]	Vert	525 gal	Atm	180	ss
Waste Holdup	2 [2]	Horiz	24,700 gal	Atm	180	ss
Waste Condensate	2 [2]	Vert	6,450 gal	Atm	180	ss
Gas Decay	8 [2]	Vert	600 ft^3	150 psig	180	cs
Waste Evaporator Bottoms Storage	1 [2]	Vert	4,000 gal	Atm	250	ss
Spent Resin Storage	1 [2]	Vert	300 ft^3	100 psig	180	ss

PUMPS	QUANTITY	TYPE	FLOW gpm	HEAD ft.	DESIGN PRESSURE psig	DESIGN TEMP °F	MATERIAL [1]
Reactor Coolant Drain (A) (per unit)	1	Horiz canned	50	175	150	300	ss
Reactor Coolant Drain (B) (per unit)	1	Horiz canned	150	175	150	300	ss
Chemical Drain	1 [2]	Horiz	20	100	150	180	ss
Laundry & Hot Shower	1 [2]	Horiz [3]	20	100	150	180	ss
Sump Tank	2 [2]	Horiz [3]	20	100	150	180	ss
Waste Evaporator	2 [2]	Horiz	20	100	150	180	ss
Waste Condensate	2 [2]	Horiz [3]	150	200	150	180	ss
Waste Evaporator Bottoms	1 [2]	Horiz [3]	20	60	150	180	ss

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[1] Material contacting fluid
[2] Shared by Units 1 and 2
[3] Mechanical seal provided
[2] Shared by Units 1 and 2
[3] Mechanical seal provided

Storage Tank							

MISCELLANEOUS EQUIPMENT	QUANTITY	CAPACITY	TYPE
Waste Evaporator	1(2)	15 gpm	Forced Circulation Flash (Incoloy - 825 tubes)
Boric Acid/Waste Evaporator	1(2)	15 gpm	Submerged Tube (Incoloy - 825 tubes)
Waste Gas Compressors	2(2)	40 CFM	Liquid piston rotary (3)

	QUANTITY	TYPE	CAPACITY	DESIGN PRESSURE psig	DESIGN TEMP. ^{o}F	MATERIAL (1)
Waste Evaporator Condensate Filter	1(2)	Disposable cartridge	20 gpm	150	180	ss
Waste Evaporator Condensate Demineralizer	1(2)	Flushable	30 ft^3	100	250	ss

ESTIMATED LIQUID DISCHARGE TO WASTE DISPOSAL SYSTEM

SOURCE	TOTAL ANNUAL (Gal)
Laundry and Shower	390,000
Equipment drains, leaks, laboratory	1,950,000
Decontamination	75,000
Totals	2,415,000
Load Factor[1]	32%

[1] Based on 15 gpm Radwaste Evaporator Capability.

ESTIMATED LIQUID RELEASE BY ISOTOPE

(Two Units)

ISOTOPE	ANNUAL RELEASE μCi	ISOTOPE	ANNUAL RELEASE μCi
Sr 89	6.12E2	Cs 134	2.78E4
Sr 90	1.54E1	Cs 136	5.36E3
Y 90	1.49E1	Cs 137	1.69E5
Sr 91	3.19E1	Cs 138	3.24E-12
Y 91	1.13E3	Te 132	3.10E4
Sr 92	5.93E-2	I 132	1.72E1
Y 92	4.52E-1	Te 134	6.36E-10
Zr 95	1.29E2	Ba 140	5.92E2
Nb 95	1.25E2	La 140	1.49E2
Mo 99	3.38E5	Ce 144	7.55E1
I 133	2.11E5	I 134	5.04E-6
I 131	3.27E5		
I 135	1.31E4		

Notes: Other waste disposal 11.25E5 μCi/yr.
All Isotopes with total activity per year <1.0E-12 were ignored.)

ESTIMATED ANNUAL GASEOUS RELEASE BY ISOTOPE

ISOTOPE	ACTIVITY TO ENVIRONMENT (Ci/Yr)
Kr 85	10,808
Kr 85m, 87, 88	Negligible
Xe 133	1149
Xe 133m, 135, 135m, 138	Negligible
Total	**11,957**

PLANT ZONE CLASSIFICATIONS

ZONE	ACCESS CONDITIONS	MAXIMUM EXPOSURE RATE (1% failed fuel) mrem/hr.
1	Unlimited	<0.25
2	Occupational	0.25 - 2.499
3	Periodic	2.5 - 4.999
4	Limited	5.0 - 100
5	Restricted	>100

PRIMARY SHIELD DESIGN PARAMETERS, NEUTRON AND GAMMA FLUXES

DESIGN PARAMETERS	
Core thermal power	3391 MW
Active core height	144 in.
Effective core diameter	132.7 in.
Baffle wall thickness	1.125 in.
Barrel wall thickness	2.25 in.
Thermal shield wall thickness	2.75 in.
Reactor vessel I.D.	173.0 in.
Reactor vessel wall thickness	8.625 in.
Reactor coolant cold leg temperature	536°F
Reactor coolant hot leg temperature	600°F
Maximum thermal neutron flux exiting primary concrete	8.4×10^3 n/cm^{2}sec.
Reactor shutdown dose exiting primary concrete	<15 mrem/hr

CALCULATED NEUTRON FLUXES		
ENERGY GROUP	**INCIDENT FLUXES (n/cm^2 - sec)**	**LEAKAGE FLUXES (n/cm^2 - sec)**
E 1 Mev	7.7×10^8	2.5×10^1
5.3 Kev $\leq$ E $\leq$ 1 Mev	1.3×10^{10}	5.6×10^1
.625 ev $\leq$ E $\leq$ 5.3 Kev	7.8×10^9	9.5×10^1
E < .625 ev	2.0×10^9	8.4×10^3

CALCULATED GAMMA FLUXES		
ENERGY GROUP	**INCIDENT FLUXES (γ/cm^2 - sec)**	**LEAKAGE FLUXES (γ/cm^2 - sec)**
E = 7.5 Mev	4.5×10^9	4.4×10^5
E = 4.0 Mev	1.2×10^9	3.1×10^5
E = 2.5 Mev	2.2×10^9	3.4×10^5
E = 0.8 Mev	7.6×10^8	2.8×10^4

SECONDARY SHIELD DESIGN PARAMETERS

Core power density @ 3391 MWt	103.9 w/cc
Reactor coolant liquid volume	12,600 ft^3 [1]
Reactor coolant transit times:	
Core	0.8 sec.
Core exit to steam generator inlet	2.1 sec.
Steam generator inlet channel	0.7 sec.
Steam generator tubes	3.7 sec.
Steam generator tubes to vessel inlet	2.1 sec.
Vessel inlet to core	2.2 sec.
Total Out of Core	10.8 sec.
Total power dose rate outside secondary shield	<1 mrem/hr

[1] This value has been conservatively chosen for the purpose of shield design. Actual best-estimated reactor coolant systems volumes can be obtained from the current Westinghouse IMP databases.

ACCIDENT SHIELD DESIGN PARAMETERS

TID-14844 RELEASE	
Core thermal power	3391 MW
Minimum full power operating time	650 days
Equivalent fraction of core melting	1.0
Fission product fractional releases:	
Noble gases	1.0
Halogens	0.5
Remaining fission product inventory	0.01
Clean-up rate following accident	0
Maximum integrated dose (infinite exposure) in the control room	<1 rem
GAP ACTIVITY RELEASE	
Core Thermal Power, MW	3391
Minimum full power operating time, days	650
Equivalent fraction fuel rod failure	1.0
Fraction of gap activity absorbed by sump water:	
Noble Gases	0.0
All Other	1.0
Cleanup rate following accident	0.0
Sump water volume, ft^3: Reactor Coolant 12,560 Refueling Water 46,800 Accumulators 4,000 **Total** **63,360 ft^3**	

ORIGINAL REFUELING SHIELD DESIGN PARAMETERS [1]

Total number of fuel assemblies	193
Minimum full power exposure	1000 days
Minimum time between shutdown and fuel handling	100 hours
Maximum exposure rate adjacent to spent fuel pit	1.0 mrem/hr
Maximum exposure rate at water surface	2.5 mrem/hr

[1] These parameters are kept for historical reasons. The dose rates are no longer applicable since the design of the spent fuel pit has been changed.

PRINCIPAL AUXILIARY SHIELDING

Design parameters for the auxiliary shielding include:	
Core thermal power	3391 MWt
Fraction of fuel rods containing small clad defects	0.01
Reactor coolant liquid volume	12600 $ft.^3$ [1]
Letdown flow (normal purification)	75 gpm
Cesium purification flow (intermittent)	75 gpm
Cut-in concentration deborating demineralizer	100 ppm
Dose rate outside auxiliary building	<1 mrem/hr
Dose rate in the building outside shield walls	<2.5 mrem/hr

COMPONENT	CONCRETE SHIELD THICKNESS Ft. - In.
Mixed Bed Demineralizers	4 - 0
Charging pumps	2 - 6
Liquid holdup tanks	2 - 8
Volume control tank	3 - 9
Reactor Coolant filter	2 - 6
Boric Acid Evaporator	2 - 4
Gas decay tanks	3 - 3
Waste Gas Compressors	2 - 8
Waste Evaporator	2 - 0
Liquid Waste Holdup Tank	2 - 0
Spent Resin Storage Tank	4 - 0

[1] This value has been conservatively chosen for the purpose of shield design. Actual best-estimated reactor coolant systems volumes can be obtained from the current Westinghouse IMP databases.

CORE AND GAP ACTIVITIES

Assumptions: Operation at 3391 MWt for 650 days.
Temperature Distribution Specified in Table 11.2-9

Isotope	Curies in the Core (x 10^7)	Percent of Core Activity in the Gap	Curies in the Gap (x 10^5)
I-131	8.26	2.3	19.0
I-132	12.65	0.26	3.29
I-133	18.76	0.79	14.82
I-134	21.92	0.16	3.51
I-135	17.02	0.43	7.32
Xe-133	18.00	1.85	33.30
Xe-133m	0.45	1.27	0.57
Xe-135	5.31	0.54	2.87
Xe-135m	5.22	0.086	0.45
Kr-85	0.095	21.57	2.05
Kr-85m	4.30	0.29	1.25
Kr-87	7.79	0.20	1.56
Kr-88	10.60	0.29	3.07

INSTANTANEOUS RADIATION SOURCES RELEASED TO THE CONTAINMENT

FOLLOWING TID-14844 ACCIDENT RELEASE - MEV/SEC

	GAMMA ENERGY (MEV/PHOTON)						
TIME AFTER RELEASE	**0.4**	**0.8**	**1.3**	**1.7**	**2.2**	**2.5**	**3.5**
0 HR	2.94×10^{18}	1.42×10^{19}	3.29×10^{18}	1.51×10^{19}	1.24×10^{19}	6.24×10^{18}	6.31×10^{18}
0.5 HR	2.82×10^{18}	1.17×10^{19}	2.51×10^{18}	1.57×10^{18}	8.10×10^{18}	5.09×10^{18}	2.34×10^{17}
1 HR	2.74×10^{18}	9.97×10^{18}	2.18×10^{18}	1.32×10^{18}	6.48×10^{18}	4.24×10^{18}	1.20×10^{17}
2 HR	2.61×10^{18}	7.46×10^{18}	1.68×10^{18}	1.01×10^{18}	5.15×10^{18}	3.01×10^{18}	3.56×10^{16}
8 HR	2.04×10^{18}	2.76×10^{18}	5.70×10^{17}	3.16×10^{17}	2.21×10^{18}	5.53×10^{17}	1.19×10^{15}
1 DY	1.15×10^{18}	1.28×10^{18}	1.00×10^{17}	1.30×10^{17}	3.63×10^{17}	3.08×10^{16}	4.27×10^{14}
1 WK	4.41×10^{17}	2.15×10^{17}	6.07×10^{15}	8.04×10^{16}	1.66×10^{15}	7.39×10^{15}	3.29×10^{14}
1 MO	2.76×10^{17}	1.41×10^{17}	2.25×10^{15}	2.63×10^{16}	1.58×10^{15}	2.41×10^{15}	1.13×10^{14}

CORE TEMPERATURE DISTRIBUTION

% of Core Fuel Volume Above the Given Temperature	Local Temperature, °F
0.0	4100
0.2	3700
1.8	3300
7.0	2900
14.5	2500

CONCENTRATION OF IODINE ISOTOPES IN THE RECIRCULATION LOOP

ISOTOPES	RECIRCULATION LOOP CONCENTRATION (c/cc)
I-131	$1.06X10^{3}$
I-132	$1.83X10^{2}$
I-133	$8.26X10^{2}$
I-134	$1.96X10^{2}$
I-135	$4.08X10^{2}$

The radiation sources circulating in the residual heat removal loop are shown in Table 11.2-11 and are used for whole body radiation doses in the auxiliary building.

The radioactivity in the containment also would be additional source of radiation to the auxiliary building following a loss-of-coolant accident.

GAP ACTIVITY CIRCULATING IN RESIDUAL HEAT REMOVAL LOOP, MEV/CC-SEC

GAMMA ENERGY (MEV/PHOTON)							
TIME AFTER RELEASE	**0.4**	**0.8**	**1.3**	**1.7**	**2.2**	**2.5**	**3.5**
0 HR	1.63×10^{7}	1.31×10^{8}	8.54×10^{6}	4.90×10^{6}	4.61×10^{6}	1.70×10^{6}	4.50×10^{5}
0.5 HR	1.51×10^{7}	1.23×10^{8}	7.56×10^{6}	4.16×10^{6}	4.16×10^{6}	1.61×10^{6}	3.78×10^{5}
1 HR	1.39×10^{7}	1.14×10^{8}	6.18×10^{6}	3.46×10^{6}	3.67×10^{6}	1.20×10^{6}	2.78×10^{5}
2 HR	1.28×10^{7}	1.03×10^{8}	4.59×10^{6}	2.53×10^{6}	3.01×10^{6}	8.24×10^{5}	2.00×10^{5}
8 HR	1.11×10^{7}	7.75×10^{7}	7.16×10^{5}	4.16×10^{5}	5.61×10^{5}	1.30×10^{5}	2.51×10^{4}
1 DY	1.03×10^{7}	6.99×10^{7}	4.84×10^{4}	1.82×10^{4}	1.75×10^{5}	7.07×10^{3}	9.96×10^{1}
1 WK	9.54×10^{6}	4.88×10^{7}	1.16×10^{2}	2.93×10^{2}			
1 MO	1.21×10^{6}	4.69×10^{7}					
6 MO	4.16×10^{4}	1.56×10^{7}					
1 YR	1.22×10^{3}	1.31×10^{7}					

DOSE RATE (REM/HR) RHR OR CONTAINMENT SPRAY

Time	Pump Room	Heat Exchanger Room	Safety Injection Pump Room
0	2.8	22.7	37.6
0 - 5 hr	2.3	18.6	32.2
1 hr.	2.0	16.3	27.4
2 hr.	1.6	13.4	22.1
8 hr.	0.83	6.6	10.5
1 day	0.18	1.5	2.6
1 week	0.02	0.2	0.41
1 month	0.008	0.08	0.18

Under the assumptions of:

(1) increased sump dilution by melted ice,

(2) core and halogen releases in accordance with Safety Guide No. 4 (in effect on September 1971), and

(3) washdown of 50% of the core halogens to the sump occurs as a result of the action of the containment sprays.

Radiation Monitoring System Channel Sensitivities and Detecting Medium

Monitor Name	Channel Number	Medium	Typical Range	Detected Isotopes
U1 Containment-Air Particulate U2 Containment-Air Particulate	ERS-1301, 1401 ERS-2301, 2401	Air	1×10^{-4} to 10 μCi	Cs^{137}, Radioactive Particulates
U1 Containment-Air Iodines U2 Containment-Air Iodines	ERS-1303, 1403 ERS-2303, 2403	Air	2×10^{-4} to 3 μCi	I^{131}, Radioiodine
U1 Containment Low Range Noble-Gas U2 Containment Low Range Noble-Gas U1 Containment Medium Range Noble-Gas U2 Containment Medium Range Noble-Gas U1 Containment High Range Noble-Gas U2 Containment High Range Noble-Gas	ERS-1305, 1405 ERS-2305, 2405	Air	9×10^{-7} to 5×10^{-2} μCi/cc	Xe^{133}, Noble Gases
	ERS-1307, 1407 ERS-2307, 2407	Air	2×10^{-3} to 2×10^{3} μCi/cc	Xe^{133}, Noble Gases
	ERS-1309, 1409 ERS-2309, 2409	Air	1×10^{-1} to 9×10^{4} μCi/cc	Xe^{133}, Noble Gases
U1 Steam Jet Air Ejector Gas Low Range Gas U2 Steam Jet Air Ejector Gas Low Range Gas U1 Steam Jet Air Ejector Gas Medium Range Gas U2 Steam Jet Air Ejector Gas Medium Range Gas U1 Steam Jet Air Ejector Gas High Range Gas U2 Steam Jet Air Ejector Gas High Range Gas	SRA-1905 SRA-2905	Air	9×10^{-7} to 5×10^{-2} μCi/cc	Xe^{133}, Noble Gases
	SRA-1907 SRA-2907	Air	2×10^{-3} to 2×10^{3} μCi/cc	Xe^{133}, Noble Gases
	SRA-1909 SRA-2909	Air	1×10^{-1} to 9×10^{4} μCi/cc	Xe^{133}, Noble Gases

Radiation Monitoring System Channel Sensitivities and Detecting Medium

Monitor Name	Channel Number	Medium	Typical Range	Detected Isotopes
U1 Component Cooling Loop Liquid U2 Component Cooling Loop Liquid	1-R-17A & 1-R17B [1] CRS-3301, 3401	Water	$1x10^{-5}$ to $1x10^{-2}$ μCi/cc [1] 5 to 500,000 cpm	Co^{60}, Mixed Fission Products
	2-R17A & 2-R-17B [1] CRS-4301, 4401	Water	$1x10^{-5}$ to $1x10^{-2}$ μCi/cc[(1)] 5 to 500,000 cpm	Co^{60}, Mixed Fission Products
Waste Disposal System Liquid Effluent	RRS-1001	Water	5 to 500,000 cpm	Co^{60}, Mixed Fission Products
U1 Steam Generator Blowdown Liquid U2 Steam Generator Blowdown Liquid	1-R-19 [1] 1-DRS-3101 2-R-19 [1] 2-DRS-4101	Water	$1x10^{-5}$ to 2 μCi/cc [1] 5 to 500,000 cpm	Cs^{137},Mixed Fission Products Co^{60}, Mixed Fission Products
U1 Essential Service Water Liquid U2 Essential Service Water Liquid	1-R-20	Water	$3x10^{-5}$ to $4x10^{-1}$ μCi/cc [1] 5 to 500,000 cpm	Cs^{137},Mixed Fission Products Co^{60}, Mixed Fission Products

[1] Old Westinghouse channel to be removed from this list.

Radiation Monitoring System Channel Sensitivities and Detecting Medium

Monitor Name	Channel Number	Medium	Typical Range	Detected Isotopes
	2-R-20	Water	$3x10^{-5}$ to $4x10^{-1}$ µCi/cc [1] 5 to 500,000 cpm	Cs^{137},Mixed Fission Products Co^{60}, Mixed Fission Products
Turbine Room Sump	Compositor	Water	Not Applicable	Not Applicable
U1 Steam Generator Blowdown Treatment System Liquid U2 Steam Generator Blowdown Treatment System Liquid	1-R-24 [1] 1-DRS-3201 2-R-24 [1] 2-DRS-4201	Water	$1x10^{-6}$ to $2x10^{-1}$ µCi/cc [1] 5 to 500,000 cpm	Co^{60}, Mixed Fission Products
U1 Unit Vent Air Particulate U2 Unit Vent Air Particulate	VRA-1501 VRA-2501	Air	$1x10^{-4}$ to 10 µCi	Cs^{137}, Radioactive Particulates
U1 Unit Vent Radioiodine U2 Unit Vent Radioiodine	VRA-1503 VRA-2503	Air	$2x10^{-4}$ to 3 µCi	I^{131}, Radioiodine
Unit Vent Low Noble Gas	VRS-1505, 2505	Air	$9x10^{-7}$ to $5x10^{-2}$ µCi/cc	Xe^{133}, Noble Gas
	VRS-1507, 2507	Air	$2x10^{-3}$ to $2x10^{3}$ µCi/cc	Xe^{133} Noble Gas

Radiation Monitoring System Channel Sensitivities and Detecting Medium

Monitor Name	Channel Number	Medium	Typical Range	Detected Isotopes
Unit Vent Hi-Level Noble Gas	VRS-1509, 2509	Air	$1x10^{-1}$ to $9x10^{4}$ µCi/cc	Xe^{133}, Noble Gas
Gland Seal Condenser Exhaust Monitor	SRA-1805, 2805	Air	$9x10^{-7}$ to $5x10^{-2}$ µCi/cc	Xe^{133}, Noble Gas
	SRA-1807, 2807	Air	$2x10^{-3}$ to $2x10^{3}$ µCi/cc	Xe^{133}, Noble Gas
	SRA-1809, 2809	Air	$1x10^{-1}$ to $9x10^{4}$ µCi/cc	Xe^{133}, Noble Gas
U1 Essential Service Water Liquid	1-R-28	Water	$1x10^{-5}$ to $1x10^{-2}$ µCi/cc [1] 5 to 500,000 cpm	Co^{60}, Mixed Fission Products Co^{60}, Mixed Fission Products
U2 Essential Service Water Liquid	2-R-28	Water	$1x10^{-5}$ to $1x10^{-2}$ µCi/cc [1] 5 to 500,000 cpm	Co^{60}, Mixed Fission Products Co^{60}, Mixed Fission Products
Containment Area at Personnel Lock	VRS-1101, 2101	Air	$1x10^{-1}$ to $1x10^{4}$ mR/hr	
Upper Containment Area Monitor	VRS-1201, 2201	Air	$1x10^{-1}$ to $1x10^{4}$ mR/hr	

Radiation Monitoring System Channel Sensitivities and Detecting Medium

Monitor Name	Channel Number	Medium	Typical Range	Detected Isotopes
Steam Generator Power Operated Relief Valve Monitor	MRA-1600, 2600 1700, 2700	Vapor	$1x10^{-2}$ to $1x10^{+2}$ μCi/cc	Xe^{133}, Noble Gas
Sampling Room Particulate	ERA-7001	Air	$1x10^{4}$ to 10 μCi	C_S^{137}, Radioactive
Sampling Room Iodine	ERA-7003	Air	$2x10^{4}$ to 3 μCi	I^{131}, Radioiodine
Sampling Room Low Range Noble Gas	ERA-7005	Air	$9x10^{-7}$ to $5x10^{-2}$ μCi/cc	Xe^{133}, Noble Gas
Sampling Room Med. Range Noble Gas	ERA-7007	Air	$2x10^{-3}$ to $2x10^{-3}$ μCi/cc	Xe^{133}, Noble Gas
Spent Fuel Area	R-5	Air	$1x10^{-1}$ to $1x10^{4}$ mR/hr	
	VRS-5006	Air	$1x10^{-2}$ to $1x10^{2}$ mR/hr	
Sampling Room Area	R-6	Air	$1x10^{-1}$ to $1x10^{4}$ mR/hr $1x10^{-1}$ to $1x10^{7}$ mR/hr	

Radiation Monitoring System Channel Sensitivities and Detecting Medium

Monitor Name	Channel Number	Medium	Typical Range	Detected Isotopes
In-Core Instrumentation Room Area	ERA-7402 (Unit 1)	Air	$1x10^{-4}$ to $1x10^{4}$ R/hr [2]	
	ERA-8402 (Unit 2)	Air	$1x10^{-4}$ to $1x10^{4}$ R/hr [2]	
Drumming Station Area	R-8 [1] 12-ERA-7505	Air	$1x10^{-1}$ to $1x10^{7}$ mR/hr [1]	
High Range Containment Area Monitor	VRA-1310, VRA-2310, VRA-1410, VRA-2410	Air	1 to $1x10^{7}$ R/HR	
Vestibule Elevation 591'	ERA-1306, -2306	Air	$1x10^{-3}$ to $1x10^{2}$ mR/hr	
Outside Containment Spray Pump Rooms Elevation 573'	ERA-1406, -2406	Air	$1x10^{-3}$ to $1x10^{2}$ mR/hr	
West of Equipment Hatch Elevation 650'	VRA-1506, -2506	Air	$1x10^{-3}$ to $1x10^{2}$ mR/hr	

[2] These monitors are calibrated to the appropriate range for the expected radiation levels in a particular area.

Radiation Monitoring System Channel Sensitivities and Detecting Medium

Monitor Name	Channel Number	Medium	Typical Range	Detected Isotopes
Turbine Building, Elevation 609'	SRA-1806, -1906, -2906	Air	$1x10^{-3}$ to $1x10^{2}$ mR/hr	
Turbine Building, Elevation 591'	SRA-2806	Air	$1x10^{-3}$ to $1x10^{2}$ mR/hr	
North of Boric Acid Tanks Elevation 587'	RRA-1003	Air	$1x10^{-1}$ to $1x10^{4}$ mR/hr	
Unit 1 E CCP Room	ERA-7303	Air	$1x10^{-4}$ to $1x10^{4}$ R/hr [2]	
Unit 1 W CCP Room	ERA-7304	Air	$1x10^{-4}$ to $1x10^{4}$ R/hr [2]	
Unit 1 E RHR Pump Room	ERA-7305	Air	$1x10^{-4}$ to $1x10^{4}$ R/hr [2]	
Unit 1 W RHR Pump Room	ERA-7306	Air	$1x10^{-4}$ to $1x10^{4}$ R/hr [2]	
Unit 1 N SIS Pump Room	ERA-7307	Air	$1x10^{-4}$ to $1x10^{4}$ R/hr [2]	
Unit 1 S SIS Pump Room	ERA-7308	Air	$1x10^{-4}$ to $1x10^{4}$ R/hr [2]	
Unit 1 Reactor Coolant Filter Cubicle	ERA-7309	Air	$1x10^{-4}$ to $1x10^{4}$ R/hr [2]	

Radiation Monitoring System Channel Sensitivities and Detecting Medium

Monitor Name	Channel Number	Medium	Typical Range	Detected Isotopes
Unit 2 E CCP Room	ERA-8303	Air	$1x10^{-4}$ to $1x10^{4}$ R/hr [2]	
Unit 2 W CCP Room	ERA-8304	Air	$1x10^{-4}$ to $1x10^{4}$ R/hr [2]	
Unit 2 E RHR Pump Room	ERA-8305	Air	$1x10^{-4}$ to $1x10^{4}$ R/hr [2]	
Unit 2 W RHR Pump Room	ERA-8306	Air	$1x10^{-4}$ to $1x10^{4}$ R/hr [2]	
Unit 2 N SIS Pump Room	ERA-8307	Air	$1x10^{-4}$ to $1x10^{4}$ R/hr [2]	
Unit 2 S SIS Pump Room	ERA-8308	Air	$1x10^{-4}$ to $1x10^{4}$ R/hr [2]	
Unit 2 Reactor Coolant Filter Cubicle	ERA-8309	Air	$1x10^{-2}$ to $1x10^{4}$ R/hr [2]	
Unit 1 Control Room	ERS-7401	Air	$1x10^{-1}$ to $1x10^{4}$ mR/hr	
Access Control Facility	ERA-7403	Air	$1x10^{-1}$ to $1x10^{4}$ mR/hr	
Radio Chemistry Lab	ERA-7404	Air	$1x10^{-1}$ to $1x10^{4}$ mR/hr	

Radiation Monitoring System Channel Sensitivities and Detecting Medium

Monitor Name	Channel Number	Medium	Typical Range	Detected Isotopes
Unit 1 N Seal Water Injection Filter Cubicle	ERA-7407	Air	$1x10^{-4}$ to $1x10^{4}$ R/hr [2]	
Unit 1 S Seal Water Injection Filter Cubicle	ERA-7408	Air	$1x10^{-4}$ to $1x10^{4}$ R/hr [2]	
Unit 1 Seal Water Filter Cubicle	ERA-7409	Air	$1x10^{-4}$ to $1x10^{4}$ R/hr [2]	
Unit 2 Control Room	ERS-8401	Air	$1x10^{-1}$ to $1x10^{4}$ mR/hr	
609' Elevation Passageway	ERA-8403	Air	$1x10^{-4}$ to 1x10 R/hr [2]	
Unit 2 N Seal Water Injection Filter Cubicle	ERA-8407	Air	$1x10^{-4}$ to $1x10^{4}$ R/hr [2]	
Unit 2 S Seal Water Injection Filter Cubicle	ERA-8408	Air	$1x10^{-4}$ to $1x10^{4}$ R/hr [2]	
Unit 2 Seal Water Injection Filter, Filter Cubicle	ERA-8409	Air	$1x10^{-4}$ to $1x10^{4}$ R/hr [2]	
587' Elevation Passageway	ERA-7504	Air	$1x10^{-4}$ to $1x10^{4}$ R/hr [2]	
Emergency Sampling Location	ERA-7507	Air	$1x10^{-1}$ to $1x10^{4}$ mR/hr	

Radiation Monitoring System Channel Sensitivities and Detecting Medium

Monitor Name	Channel Number	Medium	Typical Range	Detected Isotopes
573' Elevation Passageway	ERA-7508	Air	$1x10^{-4}$ to $1x10^{4}$ R/hr [2]	
Refueling Water Purification Filter Cubicle	ERA-7509	Air	$1x10^{-4}$ to $1x10^{4}$ R/hr [2]	
Unit 1 Vent Sampling Area	ERA-7601	Air	$1x10^{-1}$ to $1x10^{4}$ mR/hr	
Unit 1 Vent Sampling Flow Adjacent Area	ERA-7602	Air	$1x10^{-4}$ to $1x10^{4}$ R/hr [2]	
Unit 2 Vent Sampling Area	ERA-7603	Air	$1x10^{-1}$ to $1x10^{4}$ mR/hr	
Unit 2 Vent Sampling Flow Adjacent Area	ERA-7604	Air	$1x10^{-4}$ to $1x10^{4}$ R/hr [2]	
633' Elevation Passageway	ERA-7605	Air	$1x10^{-1}$ to $1x10^{4}$ mR/hr	

REACTOR COOLANT FISSION AND CORROSION PRODUCT ACTIVITIES DURING STEADY STATE OPERATION AND PLANT SHUTDOWN OPERATION

	OPERATING PWR PLANT		DONALD C. COOK PLANT - 1% FUEL DEFECTS	
ISOTOPE	MEASURED ACTIVITY BEFORE SHUTDOWN µCi/gm	MEASURED PEAK SHUTDOWN ACTIVITY µCi/gp	CALCULATED ACTIVITY BEFORE SHUTDOWN µCi/gm	EXPECTED PEAK SHUTDOWN ACTIVITY µCi/gm
I-131	0.83	14.9	2.4	43.0
Xe-133	127.0	65.0 [1]	254.0	130.0 [(1)]
Cs-134	1.29	1.7	0.19	0.25
Cs-137	1.67	2.14	1.1	1.4
Cs-144	0.00068	0.0058	0.00051	0.0044
Sr-89	0.0033	0.40	0.0042	0.51
Sr-90	0.00057	0.013	0.0001	0.0023
Co-58	---	0.95	0.025	1.0

[1] Activity reduced from steady state level by approximately one day of system degassification prior to plant shutdown.

RADIATION MONITORING SYSTEM CHANNELS

CHANNEL	PURPOSE	ASSOCIATED TRIP FUNCTION OVERVIEW
ERS-1301, 1401, 2301, 2401	Containment Airborne Particulates - Detection	Containment ventilation isolation, prevent further release
ERS-1303, 1403, 2303, 2403	Containment Radioiodine - Detection	Containment ventilation isolation, prevent further release
ERS-1305, 1405, 2305, 2405	Containment Low Range Noble Gas - Detection	Containment ventilation isolation, prevent further release
ERS-1307, 1407, 2307, 2407	Containment Mid Range Noble Gas - Detection	Containment ventilation isolation, prevent further release
ERS-1309, 1409, 2309, 2409	Containment High Range Noble Gas - Detection	Containment ventilation isolation, prevent further release
ERS-7401, 8401	Control Room Area Monitor	Isolate Control Room Ventilation
SRA- 1905, 2905	Steam Jet Air Ejector low Range Noble Gas - Detect primary to secondary leakage	None
SRA-1907, 2907	Steam Jet Air Ejector Mid Range Noble Gas - Detect primary to secondary leakage	None
SRA- 1909, 2909	Steam Jet Air Ejector High Range Noble Gas - Detect primary to secondary leakage	None
R-17 A, R-17 B, CRS-3301, 3401, 4301, 4401	Component Cooling Water Loop Liquid Monitor - Detect leaks from RCS or RHR into the CCW system	Isolate CCW surge tank vent
RRS-1001	Waste Disposal System Liquid Effluent Monitor	Automatic valve closure to prevent further release
R-5	SFP Area Monitor	Place SFP ventilation into service
VRS-5006	SFP Area Monitor	Place SFP ventilation into service
R-19	Steam Generator Blowdown Liquid Monitor - detect primary to secondary leakage via common blowdown header	Isolate steam generator blowdown system.

RADIATION MONITORING SYSTEM CHANNELS

CHANNEL	PURPOSE	ASSOCIATED TRIP FUNCTION OVERVIEW
R-20, R-28	Essential Service Water Liquid Monitor – Detect leakage in the containment spray heat exchangers, (post LOCA)	None
R-24	Steam Generator Blowdown Treatment System Liquid Monitor - measure activity in the blowdown liquid after it passes the treatment demineralizer	Isolate steam generator blowdown system
VRA-1501, 2501	Unit Vent Airborne Particulates - Detection	None
VRA-1503, 2503	Unit Vent Radioiodines - Detection	None
VRS-1505, 2505	Unit Vent Low Range Noble Gas - Detection	Gas decay tank isolation valves [1]
VRS-1507, 2507	Unit Vent Mid Range Noble Gas - Detection	None
VRS-1509, 2509	Unit Vent High Range Noble Gas - Detection of accidental release	Sample pathway bypass of channels 1, 3, 5, 7 to sample pallet
SRA-1805, 2805	Gland Seal Condenser Exhaust - Low range detection	None
SRA 1807, 2807	Gland Seal Condenser Exhaust - Mid range detection	None
SRA 1809, 2809	Gland Seal Condenser Exhaust - High range detection	None
Unit Vent Continuous air flow sampler	Tritium sampling	None

[1] Available setpoint is used to accommodate: 1) normal operation, and 2) gas decay tank release.

DESIGN AND MEASURED EQUILIBRIUM REACTOR COOLANT FISSION PRODUCT

ACTIVITIES FOR OPERATING PWR'S AND CALCULATED VALUES FOR THE D. C. COOK STATIONS

		GINNA STATION [1]			BEZNAU STATION (1)		COOK STATION
	DESIGN VALUE [2] μc/cc	MEASURED VALUE μc/cc	RATIO MEASURED (Design)	DESIGN VALUE (2) μc/cc	MEASURED VALUE μc/cc	RATIO MEASURED (DESIGN)	DESIGN VALUE (2) μc/cc
Total Activity	216	71	0.33	299	168	0.73	207
Isotopic Activity (Key Isotopes)							
I-131	1.53	0.56	0.37	0.96	0.75	0.78	1.7
I-133	2.55	1.7	0.67	1.74	2.0	1.16	2.6
Xe-133	184	45	0.24	200	119	0.60	178
Cs-134	0.19	0.06	0.32	0.22	0.075	0.35	0.13
Cs-137	0.94	0.37	0.40	1.53	0.22	0.15	0.8

[1] Based on an assumed 1% defect level.
[2] Amendment 20 to original FSAR (Mar, 1972).

Blowdown Treatment System Components

Pump	
Number	1 per unit
Fluid	Steam generator blowdown
Pressure, Suction	Atmospheric
Temperature	200°F
Head	125 ft.
Flow	60 gpm
Type	Horizontal centrifugal
Material, Casing	Stainless Steel
Impeller	Stainless Steel
NPSH, minimum Ft. H_2O	2.5
Heat Exchanger[1]	
Number	1 per unit
Shell Side (blowdown liquid)	
Inlet Temperature	200°F
Outlet Temperature	120°F
Max. pressure	70 psi
Operating pressure	50 psi
Flow	60 gpm
Material	304 Stainless Steel
Pressure drop, normal	4 psi
maximum allowable	15 psi
Tube Side (non-essential service water)	
Inlet Temperature	76°F

[1] The system has been evaluated for a NESW pump discharge temperature of 88.9°F.

Blowdown Treatment System Components

Outlet Temperature	106°F
Max. pressure	150 psig
Operating pressure	75 psig
Flow	160 gpm
Material 304 Stainless Steel	
Pressure drop, normal	5 psi
maximum allowable	9 psi
Mixed Bed Demineralizers	
Number	3 per unit
Type	Flushable
Vessel, design pressure, psig	200
Operating pressure, psig	50
Vessel, design temperature, °F	250
Operating temperature, °F	120
Resin volume, ft^3	56 (Nos. 1 & 2) 20 (No. 3)
Design flow rate, gpm	50
Resin type	Cation and anion
Material of construction	Austenitic stainless steel


DWG. NO. FSAR FIG. 11.1-1
DUPLICATED COMPONENTS UNIT 1 OR 2
SHARED COMPONENTS UNITS 1 AND 2
FROM ACCUMULATOR TANK DRAINS SEE FIG. 9.3-1
Nº3 Nº2
INSIDE REACTOR CONTAINMENT UNIT 1 OR 2
OUTSIDE REACTOR CONTAINMENT UNIT 1 OR 2
FROM ACCUMULATOR TANK DRAINS SEE FIG. 6.2-1A
Nº1 Nº4
FROM EXCESS LETDOWN HEAT EXCHANGER SEE FIG. 9.2-1
FROM REGENERATIVE HEAT EXCHANGER SEE FIG. 9.2-1
REACTOR COOLANT PUMPS SECONDARY SEAL LEAKAGE SEE FIG. 4.2-1A
Nº3 Nº2
REACTOR VESSEL FLANGE LEAK-OFF SEE FIG. 4.2-1
REACTOR COOLANT PUMPS SECONDARY SEAL LEAKAGE SEE FIG. 4.2-1A
FROM REACTOR COOLANT SYSTEM SEE FIG. 4.2-1
LOOP Nº3
LOOP Nº2
Nº1 Nº4
IMB OMB
OMB IMB
FROM DEBORATING DEMINERALIZERS (REGENERATION) SEE FIG. 9.2-2
UNIT Nº 2 REACTOR COOLANT DRAIN TANK VENT
MISCELLANEOUS DRAINS (VALVE LEAK-OFFS, ETC.)
LOOP Nº1
LOOP Nº4
FROM REACTOR COOLANT SYSTEM SEE FIG. 4.2-1
FROM NITROGEN SUPPLY MANIFOLD SEE FIG. 11.1-4
F.C.
FROM REFUELING CAVITY DRAIN SEE FIG. 9.4-1
F.C. F.C.
FROM REACTOR COOLANT SYSTEM PRESSURIZER RELIEF TANK, SEE FIG. 4.2-1A
TO GAS ANALYZER SEE FIG. 11.1-4
TO PIPE TUNNEL SUMP
DSI
DPA
F.C. F.C.
L.C. L.C.
TO REFUELING WATER PURIFICATION PUMPS SEE FIG. 9.4-1
REACTOR COOLANT DRAIN TANK
DTA
DLC
FROM REACTOR COOLANT SYSTEM PRESSURIZER RELIEF TANK, SEE FIG. 4.2-1A
FROM REACTOR COOLANT PUMPS NO. 3 LEAK-OFF SEE FIG. 4.2-1A
EMERGENCY DRAIN TO PIPE TUNNEL SUMP
FROM SPENT RESIN STORAGE TANK SEE FIG. 11.1-2
F.C.
F.C.
TO REFUELING WATER STORAGE TANK SEE FIG. 6.3-1
TO CHEMICAL & VOLUME CONTROL SYSTEM HOLD UP TANKS SEE FIG. 9.2-4
TO WASTE HOLDUP TANK THRU DRAIN HEADER SEE THIS DWG. N/4
REACTOR COOLANT DRAIN PUMPS
FROM MIXED BED, CATION BED AND DEBORATING DEMINERALIZERS SEE FIG. 9.2-2
FROM EVAPORATOR FEED ION EXCHANGERS SEE FIG. 9.2-4
FROM EVAPORATOR CONDENSATE DEMIN'S (REGENERATION) SEE FIG. 9.2-6
FROM EVAPORATOR CONDENSATE DEMINS SEE FIG. 9.2-6
FROM SPENT FUEL PIT DEMINERALIZER SEE FIG. 9.4-1
VENTS FROM BLOWDOWN DEMINERALIZERS SEE FIG. 11.5-1
FROM SPENT FUEL PIT SKIMMER FILTER SEE FIG. 9.4-1
FROM SPENT FUEL PIT FILTER SEE FIG. 9.4-1
FROM HOT TOOL DECONTAMINATION FACILITY EQUIPMENT
FROM SPENT RESIN STORAGE TANK SEE FIG. 11.1-2
FROM COND. FILTER SEE FIG. 9.2-6
UNIT Nº1
FROM SPENT RESIN STORAGE TANK SEE FIG. 11.1-2
FROM AUTOMATIC GAS ANALYZER SEE FIG. 11.1-4
FROM UNIT NO. 1 & 2 VOLUME CONTROL TANK GAS SAMPLES SEE FIG. 9.6-1
STRAINER
DLA
FROM BORIC ACID EVAPORATORS SEE FIG. 9.2-5
LOOP SEAL
DLA
STRAINER
FROM REACTOR COOLANT FILTERS SEE FIG. 9.2-1
FROM AUTOMATIC GAS ANALYZER SEE FIG. 11.1-4
FROM BORIC ACID EVAPORATOR SEE FIG. 9.2-5
FROM SEAL WATER INJECTION FILTERS SEE FIG. 9.2-1
FROM SEAL WATER FILTER SEE FIG. 9.2-1
FROM GAS DECAY TANK DRAINS
FROM ION EXCHANGER FILTERS SEE FIG. 9.2-5
UNIT Nº1
UNIT Nº2
UNIT Nº1
UNIT Nº2
UNIT Nº1
UNIT Nº2
TO WASTE GAS COMPRESSOR PACKAGES SEE FIG. 11.1-3
UNIT Nº1
UNIT Nº2
FROM VOLUME CONTROL TANK SEE FIG. 9.2-1
FROM VOLUME CONTROL TANK SEE FIG. 9.2-1
FROM GAS COMPRESSOR MOISTURE SEPARATORS SEE FIG. 11.1-3
UNIT Nº1
UNIT Nº2
CONTAINMENT VENTILATION UNIT DRAINS
F.C. F.C.
ICE CONDENSER FAN COOLERS UNIT DRAINS
F.C. F.C.
FROM REACTOR COOLANT DRAIN TANK PUMPS SEE THIS DWG. G/8
TO WASTE HOLD-UP TANKS SEE FIG. 11.1-2
21
DRAIN FROM SAMPLING RACK "SRC" SEE FIG. 9.6-1
FROM REFUELING WATER PURIFICATION PUMP SEE FIG. 9.4-1
FROM MISC. LOW POINT FLOOR DRAINS
FROM SPRAY ADDITIVE TANK VAULT
FROM REFUELING WATER PURIFICATION FILTER SEE FIG. 9.4-1
TO SUMP TANK SEE FIG. 11.1-2A
FROM HOLD UP TANKS SEE FIG. 9.2-4
NOTE:
REACTOR COOLANT DRAIN TANK PUMPS ALIGNED TO CVCS HOLD UP TANK, OTHER LINEUPS ALLOWED, AS DESIGNED.
VALVES POSITIONS ARE NOT GOVERNED BY THE UFSAR FIGURES BUT RATHER BY STATION PROCEDURES (EXCEPT VALVES WITH LOCKED OR SEALED, AND FAILURE POSITIONS AS INDICATED).
REFERENCES:
21 REVISED PER UCR-1770.
DATE NO. DESCRIPTION APPD.
REVISIONS
FILENAME: fsra-fig-11.1-1 21.dgn (raster file): fsar-fig-11.1-1 21.cit
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INDIANA MICHIGAN POWER COMPANY
DONALD C. COOK
NUCLEAR PLANT
BRIDGMAN MICHIGAN
VENTS AND DRAINS UNITS 1 & 2
DWG. NO. FSAR FIG. 11.1-1
ARCH ELEC MECH STR
SCALE: DR:
DATE: CH:
DESIGN ENGINEERING DIVISION:
AEP AMERICAN ELECTRIC POWER
AEP SERVICE CORP. 1 RIVERSIDE PLAZA COLUMBUS, OH 43215
DUPLICATED COMPONENTS UNIT 1 OR 2
SHARED COMPONENTS UNITS 1 AND 2
IGS0694

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

REFERENCES:

OP-12-5138

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

REVISIONS

FILENAME: FSAR-FIG-11-1-2.027 (raster file): FSAR-FIG-11-1-2_27.TIF

INDIANA MICHIGAN POWER COMPANY

DONALD C. COOK

NUCLEAR PLANT

BRIDGMAN MICHIGAN

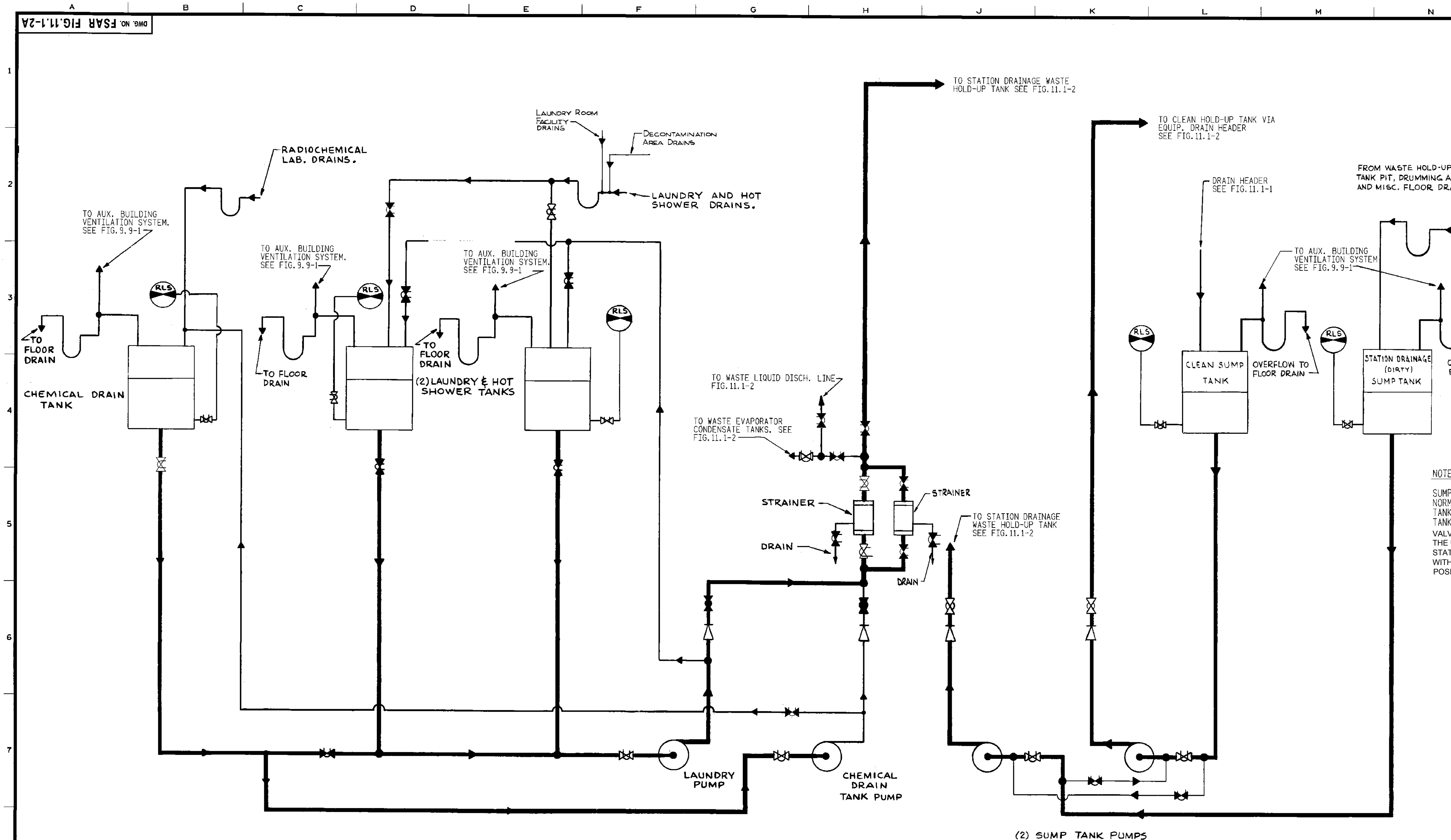
WDS-LIQUIDS & SOLIDS
UNITS NO. 1 & 2
SHEET 1 OF 3

DWG. NO. **FSAR FIG. 11.1-2**

ARCH	ELEC	MECH	STR

SCALE: DR:
DATE: CH:
DESIGN ENGINEERING DIVISION

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



NOTE:

SUMP TANK PUMPS LINED UP FOR NORMAL OPERATION. CHEMICAL DRAIN TANK AND LAUNDRY AND HOT SHOWER TANK SHOWN IN STANDBY.

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

REFERENCES:

DATE	NO.	DESCRIPTION	APPD.
	16.3	REVISED PER UCR 98-UFSAR-0588, 99-UFSAR-1182, 99-UFSAR-1185	

REVISIONS

FILENAME: fsar-fig-11-1-2a.163
(raster file): fsar-fig-11-1-2a 163.cit

INDIANA MICHIGAN POWER COMPANY

DONALD C. COOK

NUCLEAR PLANT

BRIDGMAN MICHIGAN

WDS LIQUIDS & SOLIDS
UNITS NO. 1 & 2
SHEET 2 OF 3

DWG. NO. **FSAR FIG. 11.1 - 2A**

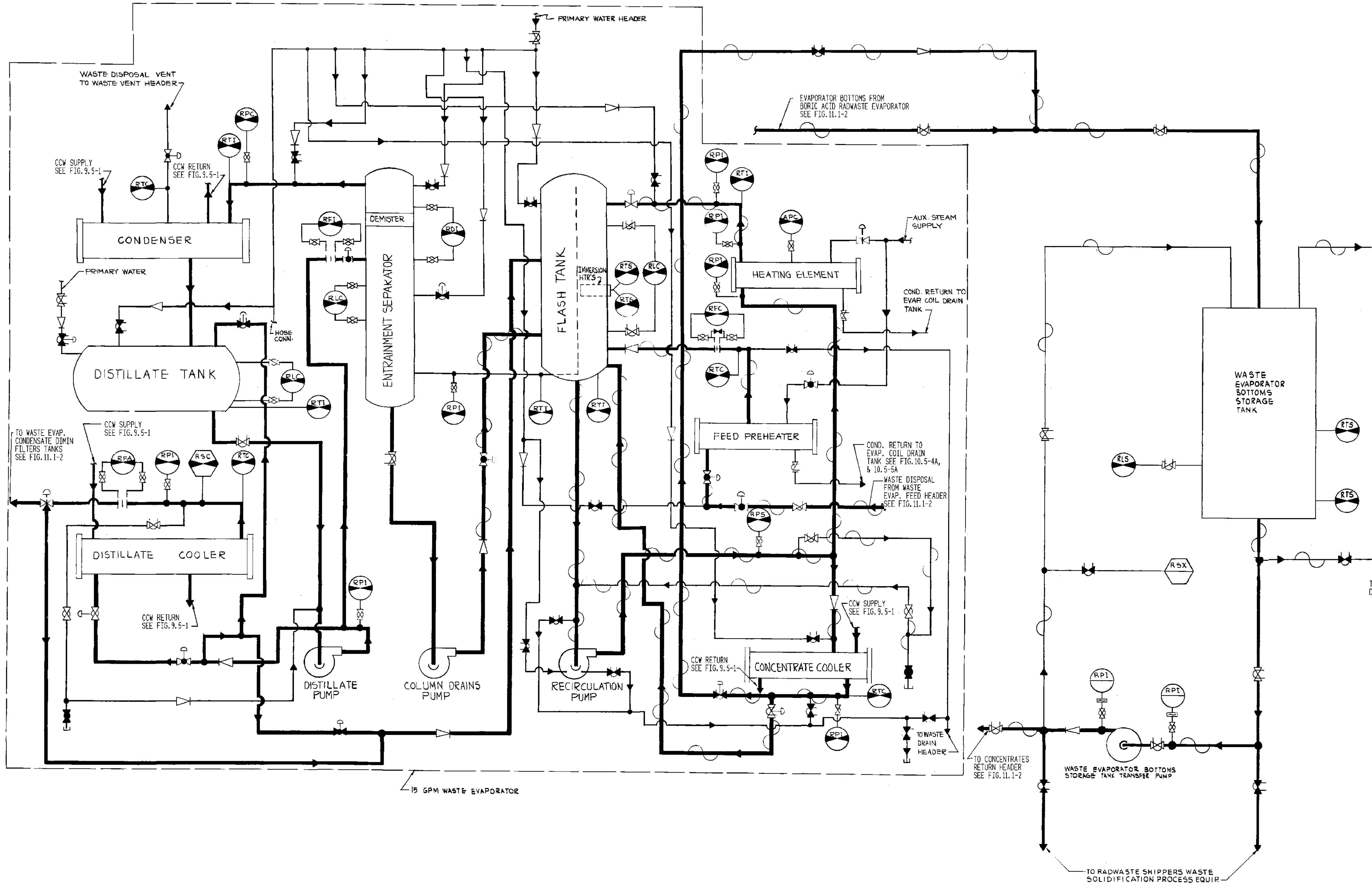
ARCH | ELEC | MECH | STR

SCALE: DR:
DATE: CH:

DESIGN ENGINEERING DIVISION:

AMERICAN ELECTRIC POWER

AEP SERVICE CORP.
1 RIVERSIDE PLAZA
COLUMBUS, OH 43215



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

REFERENCES:

DATE	NO.	DESCRIPTION	APPD.
	16.3	REVISED PER UCR 99-UFSAR-1182 & 99-UFSAR-1185	

REVISIONS

FILENAME: fsar-fig-11-1-2b.163
(raster file): fsar-fig-11-1-2b 163.cit

INDIANA MICHIGAN POWER COMPANY

DONALD C. COOK

NUCLEAR PLANT

BRIDGMAN MICHIGAN

WDS LIQUIDS & SOLIDS
UNITS NO. 1 & 2
SHEET 3 OF 3

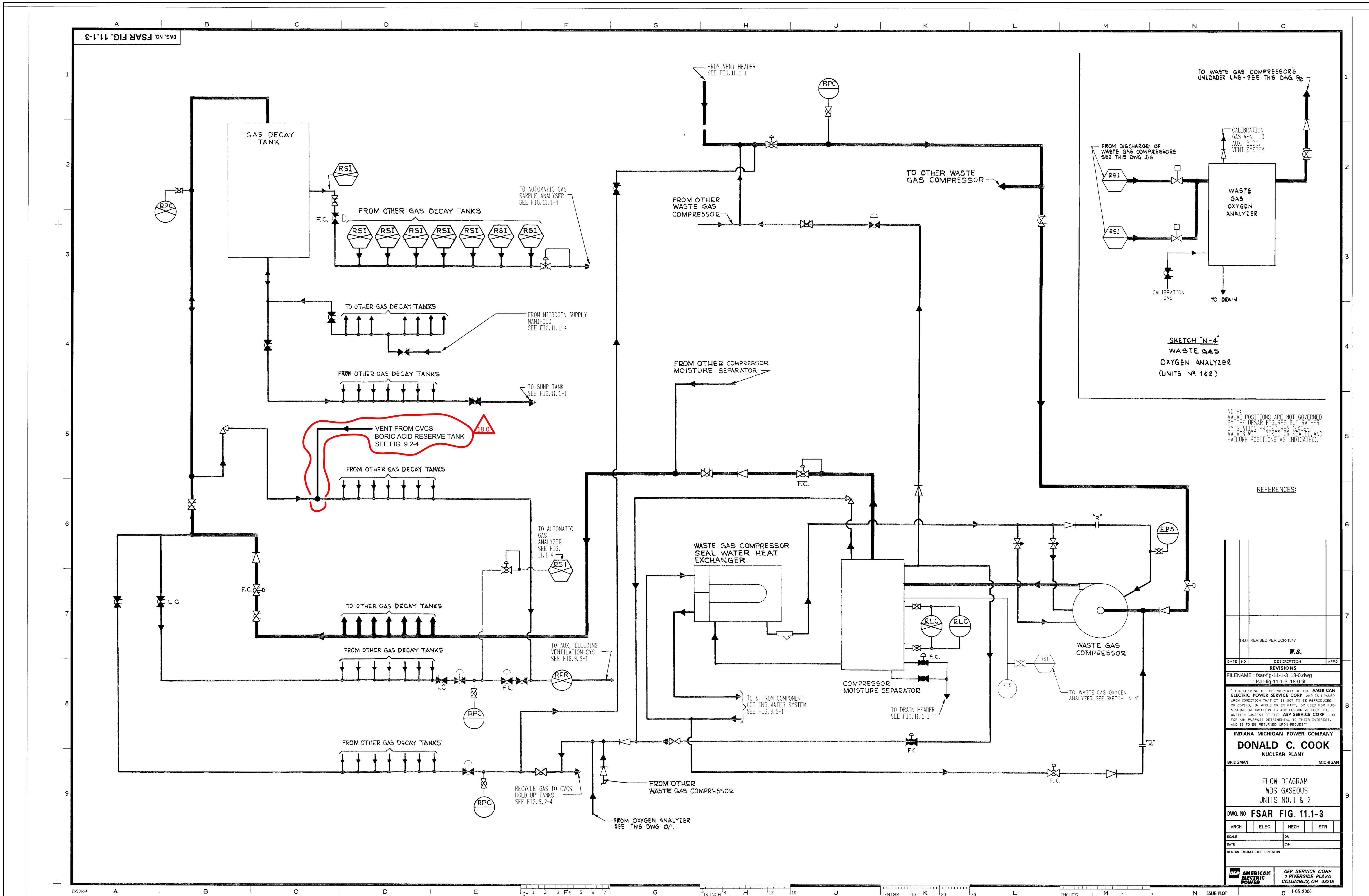
DWG. NO. **FSAR FIG. 11.1 - 2B**

ARCH | ELEC | MECH | STR

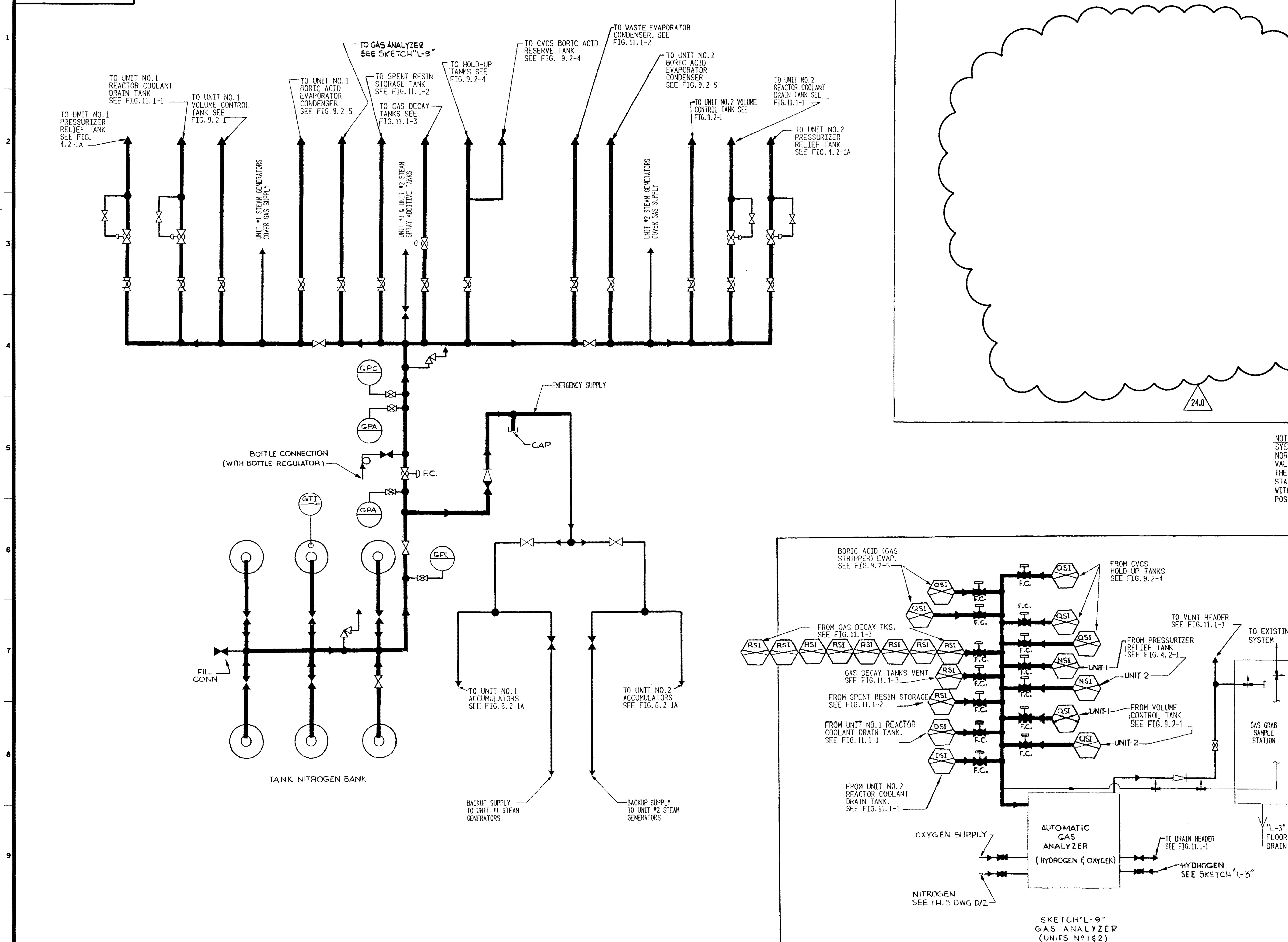
SCALE: DR:
DATE: CH:
DESIGN ENGINEERING DIVISION:

AMERICAN ELECTRIC POWER

AEP SERVICE CORP.
1 RIVERSIDE PLAZA
COLUMBUS, OH 43215



DWG. NO. FSAR FIG. 11.1-3
GAS DECAY TANK
RSI
RPC
F.C.
FROM OTHER GAS DECAY TANKS
TO AUTOMATIC GAS SAMPLE ANALYSER SEE FIG. 11.1-4
TO OTHER GAS DECAY TANKS
FROM NITROGEN SUPPLY MANIFOLD SEE FIG. 11.1-4
TO SUMP TANK SEE FIG. 11.1-1
VENT FROM CVCS BORIC ACID RESERVE TANK SEE FIG. 9.2-4
18.0
TO AUTOMATIC GAS ANALYZER SEE FIG. 11.1-4
LC
TO AUX. BUILDING VENTILATION SYS SEE FIG. 9.3-1
RFR
RECYCLE GAS TO CVCS HOLD-UP TANKS SEE FIG. 9.2-4
FROM OTHER WASTE GAS COMPRESSOR
FROM OXYGEN ANALYZER SEE THIS DWG O/1.
FROM VENT HEADER SEE FIG. 11.1-1
FROM OTHER WASTE GAS COMPRESSOR
TO OTHER WASTE GAS COMPRESSOR
FROM OTHER COMPRESSOR MOISTURE SEPARATOR
WASTE GAS COMPRESSOR SEAL WATER HEAT EXCHANGER
COMPRESSOR MOISTURE SEPARATOR
TO & FROM COMPONENT COOLING WATER SYSTEM SEE FIG. 9.5-1
TO DRAIN HEADER SEE FIG. 11.1-1
RLC
RPS
"R"
WASTE GAS COMPRESSOR
TO WASTE GAS OXYGEN ANALYZER SEE SKETCH "N-4"
TO WASTE GAS COMPRESSOR'S UNLOADER LINE - SEE THIS DWG. F/6
CALIBRATION GAS VENT TO AUX. BLDG. VENT SYSTEM
FROM DISCHARGE OF WASTE GAS COMPRESSORS SEE THIS DWG. J/5
WASTE GAS OXYGEN ANALYZER
CALIBRATION GAS
TO DRAIN
SKETCH "N-4"
WASTE GAS OXYGEN ANALYZER (UNITS Nº 1&2)
NOTE: VALVE POSITIONS ARE NOT GOVERNED BY THE UFSAR FIGURES BUT RATHER BY STATION PROCEDURES (EXCEPT VALVES WITH LOCKED OR SEALED AND FAILURE POSITIONS AS INDICATED).
REFERENCES:
18.0 REVISED PER UCR-1347
REVISIONS
FILENAME: fsar-fig-11-1-3_18-0.dwg
INDIANA MICHIGAN POWER COMPANY
DONALD C. COOK NUCLEAR PLANT
BRIDGMAN MICHIGAN
FLOW DIAGRAM WDS GASEOUS UNITS NO. 1 & 2
DWG. NO FSAR FIG. 11.1-3
AMERICAN ELECTRIC POWER
AEP SERVICE CORP 1 RIVERSIDE PLAZA COLUMBUS, OH 43215
1-05-2000



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

REFERENCES:
OP-1(2)-5141

DATE	NO.	DESCRIPTION	APPD.
	24.0	REVISED PER UCR-1989	

REVISIONS

FILENAME: fsar-fig-11-1-4 24.dgn
(raster file): fsar-fig-11-1-4 24.tif

INDIANA MICHIGAN POWER COMPANY

DONALD C. COOK
NUCLEAR PLANT
BRIDGMAN MICHIGAN

GAS SUPPLY AND ANALYSIS UNITS NO.1 & 2

DWG. NO. FSAR FIG. 11.1-4

ARCH	ELEC	MECH	STR

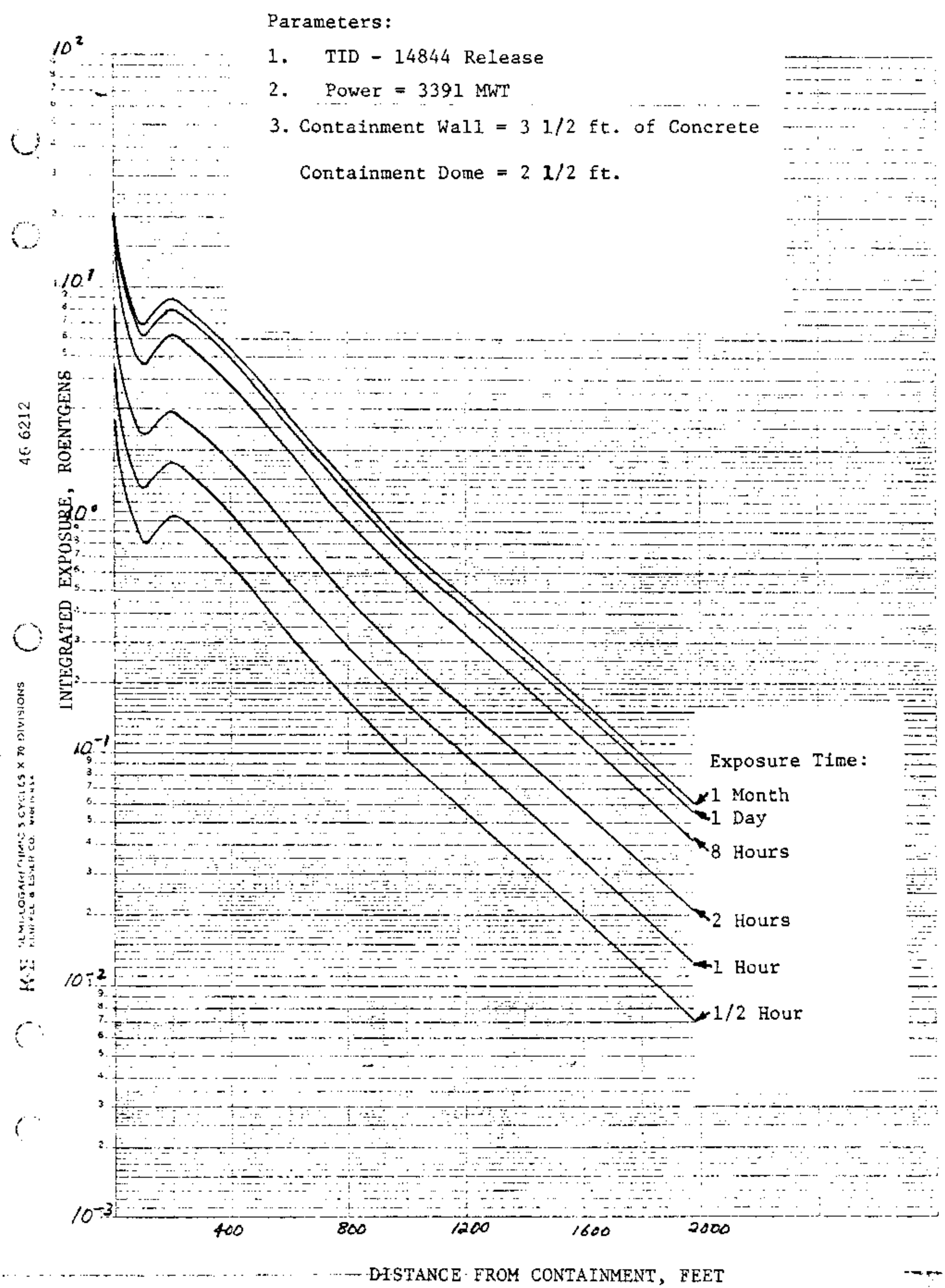
SCALE: DR:
DATE: CH:
DESIGN ENGINEERING DIVISION

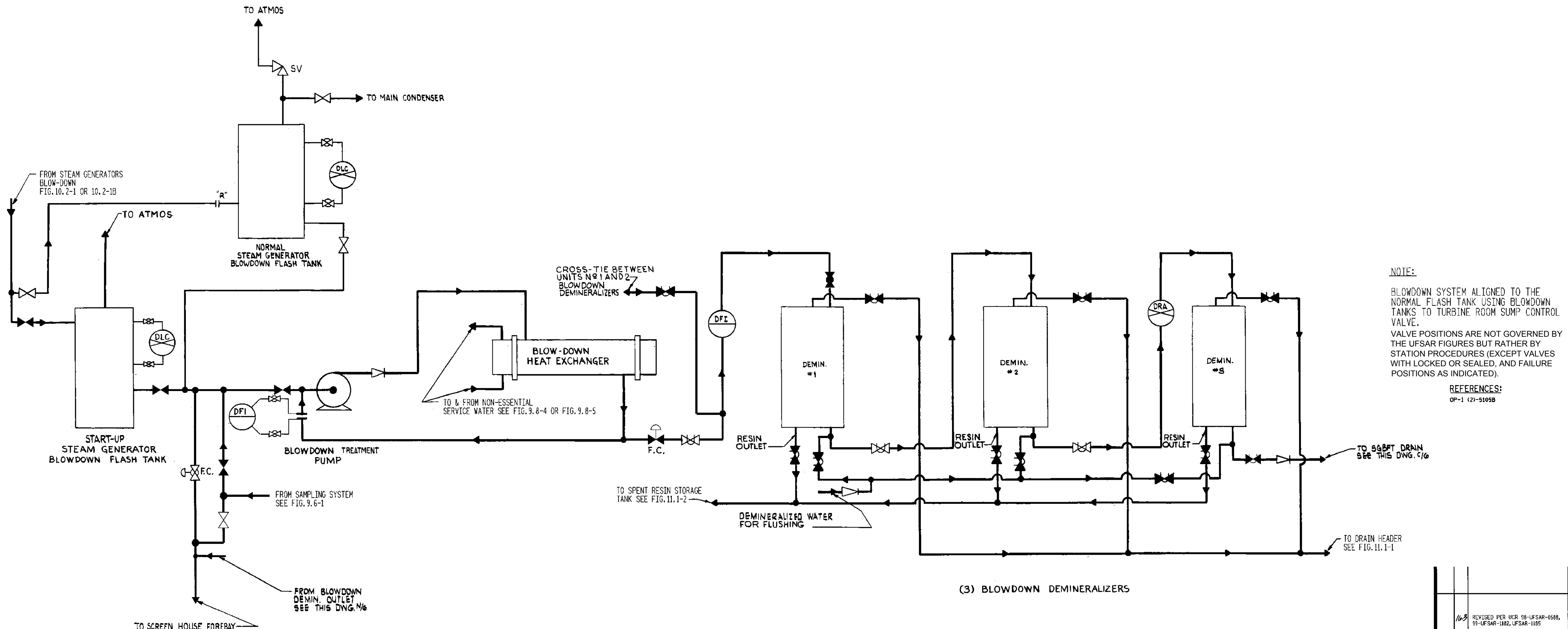
AMERICAN ELECTRIC POWER
AEP SERVICE CORP. 1 RIVERSIDE PLAZA COLUMBUS, OH 43215

FIGURE 11.2-1

INTEGRATED EXPOSURE AS A FUNCTION OF

DISTANCE FROM CONTAINMENT BUILDING





DATE	NO.	DESCRIPTION	APPD.
	163	REVISED PER UCR 98-UFSAR-0588, 99-UFSAR-1182, UFSAR-1185	

REVISIONS

FILENAME: fsar-fig-11-5-1.163
(raster file): fsar-fig-11-5-1 163.cit

INDIANA MICHIGAN POWER COMPANY

DONALD C. COOK

NUCLEAR PLANT

BRIDGMAN MICHIGAN

STEAM GENERATOR
BLOW-DOWN SYSTEM
UNIT NO. 1 & 2

DWG. NO. **FSAR FIG. 11.5 - 1**

ARCH | ELEC | MECH | STR

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

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

1090694

ADMINISTRATIVE CONTROL

AEPNG VICE PRESIDENT

NUCLEAR GENERATION

NUCLEAR MATERIAL MANAGEMENT GROUP

PLANT MANAGER

REACTOR ENGINEERING SUPERVISOR

NUCLEAR FUELS SUPERVISOR

NUCLEAR MATERIALS MANAGER

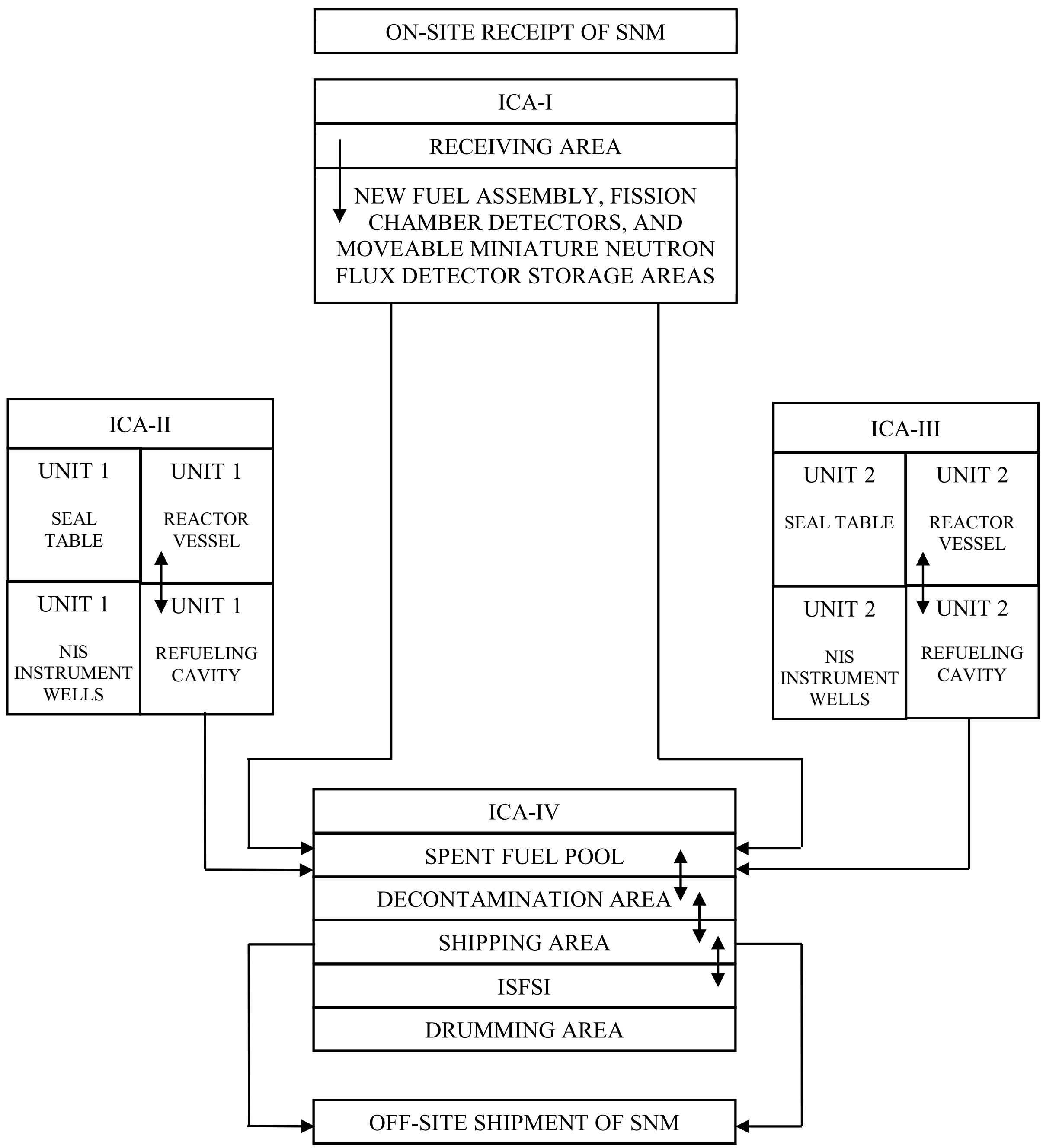
ACCOUNTABILITY

NUCLEAR MATERIAL MANAGER

ICA CUSTODIANS

ORGANIZATION AND FUNCTIONAL STRUCTURE

FIGURE 11.6-1



Revision: **25.0**	Change Description: **UCR-2010, Rev. 0**
AMERICAN ELECTRIC POWER **COOK NUCLAR PLANT** **NUCLEAR GENERATION GROUP** **BRIDGMAN, MICHIGAN**	Title: **Fuel Assembly Flow Chart**
	UFSAR Figure: **11.6-2A** — Sheet 1 of 1

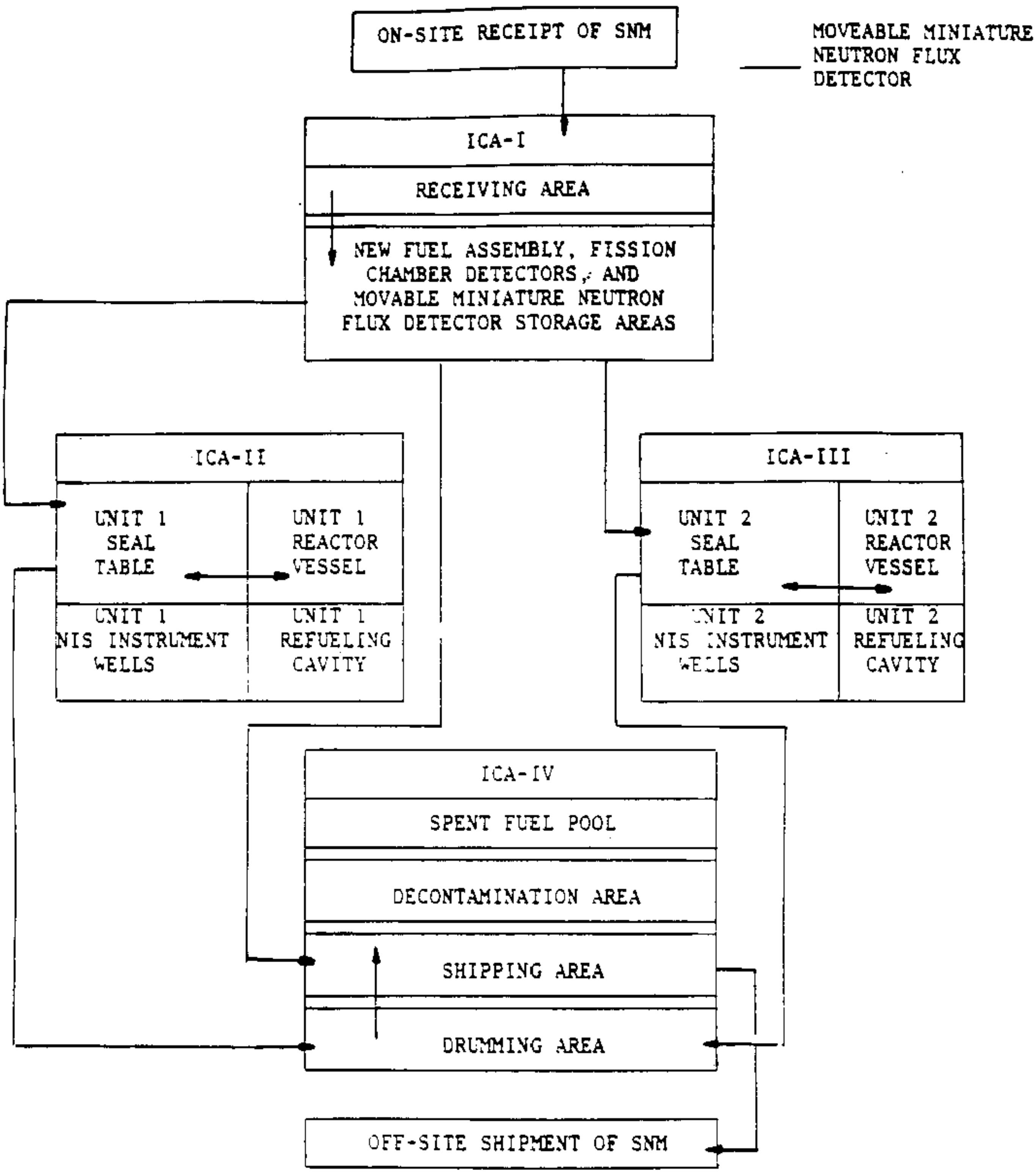


Figure 11.6-2b MOVEABLE MINIATURE NEUTRON FLUX DETECTOR FLOW CHART

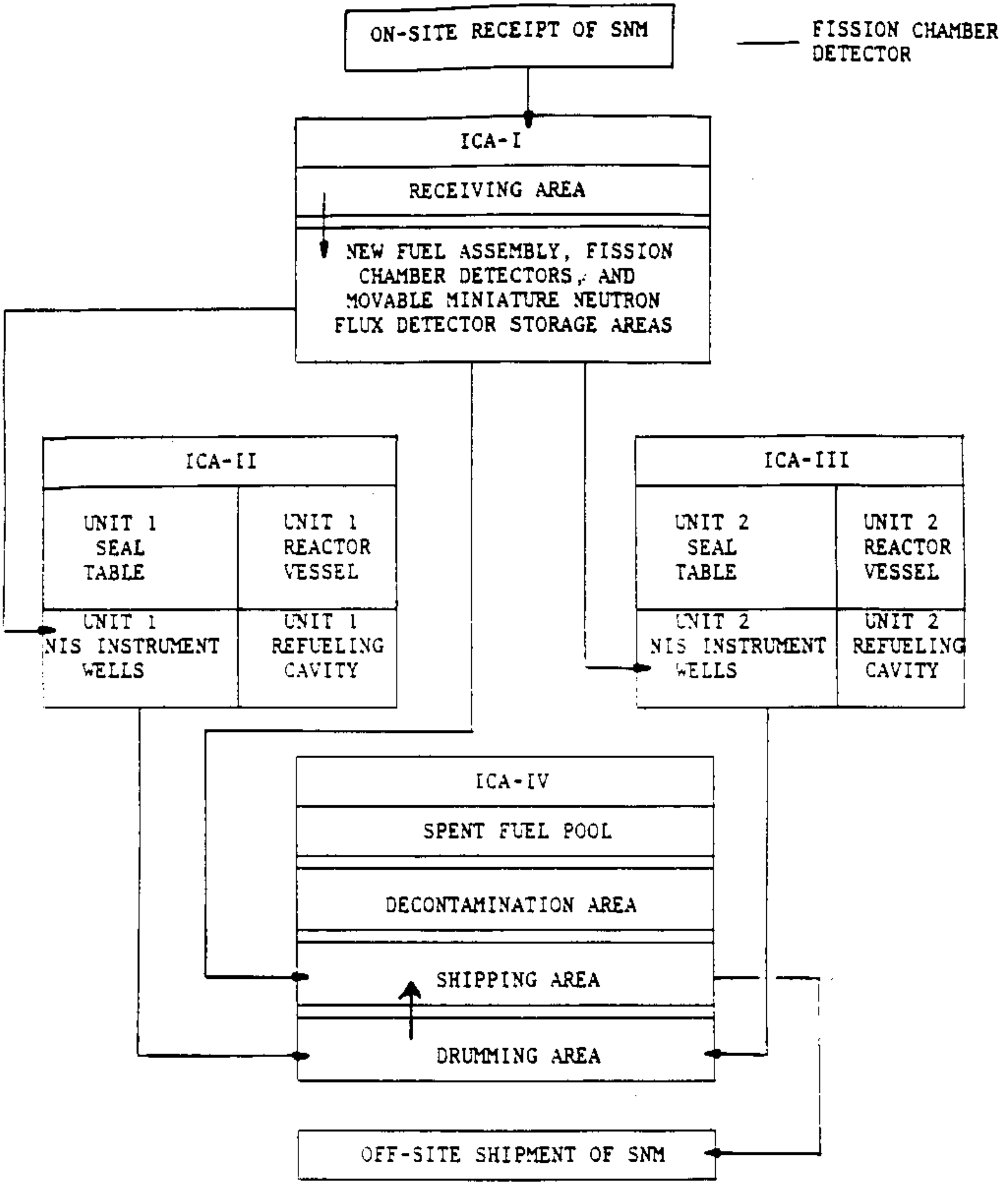


Figure 11.6-2c FISSION CHAMBER DETECTOR FLOW CHART