Chapter 11: Radioactive Wastes and Radiation Protection

Table of Contents

Chapter 11: Radioactive Wastes and Radiation Protection

Table of Contents (continued)

Chapter 11: Radioactive Wastes and Radiation Protection

Table of Contents (continued)

Chapter 11: Radioactive Wastes and Radiation Protection

List of Tables

Chapter 11: Radioactive Wastes and Radiation Protection

List of Figures

Intentionally Blank

Chapter 11
RADIOACTIVE WASTES AND RADIATION PROTECTION

11.1 GENERAL DESCRIPTION

Note: As required by the Renewed Operating Licenses for Surry Units 1 and 2, issued March 20, 2003, various systems, structures, and components discussed within this chapter are subject to aging management. The programs and activities necessary to manage the aging of these systems, structures, and components are discussed in Chapter 18.

Waste disposal systems are provided to separate, treat, and dispose of radioactive liquid, gaseous, and solid waste materials. The liquid, solid, and gaseous waste disposal systems are common to both reactor units and designed to serve both units simultaneously. These systems incorporate one or more of the following basic processes:

1. Filtration, to remove particulate matter.
2. Evaporation, to concentrate and remove contaminants.
3. Demineralization, to remove dissolved material.
4. Compaction, to reduce the volume of compressible wastes.
5. Natural decay of radioactive isotopes.
6. Dilution, to reduce concentration.

Liquid, gaseous, and solid waste materials originate in the reactor coolant system, the auxiliary and emergency systems, the waste disposal system, and as a result of operation and maintenance procedures. Waste materials enter the waste disposal system directly from their source or via the vent and drain system (Section 9.7).

Adequate sampling, analysis, and monitoring of the waste disposal system are provided to comply with the design criteria. Process radiation monitors and flow-measuring equipment are provided for the surveillance of various station and radwaste effluents and process streams to ensure compliance with applicable regulations and to provide early indications of possible malfunctions and hazardous conditions.

Sufficient shielding is provided to reduce radiation to acceptable levels for normal operation and incident conditions. Allowable dose rates are based on applicable regulations, expected frequency, and the duration of exposure to radiation.

Area radiation monitoring equipment, health physics facilities, environmental programs, and administrative controls are provided for the surveillance and control of radiation exposure levels. These ensure radiation protection for plant personnel and the general public in accordance with applicable criteria.

Radiological and chemical respiratory protection equipment approved by the National Institute for Occupational Safety and Health/Mine Safety and Health Administration (NIOSH/MSHA) is provided. Equipment not tested and certified by NIOSH/MSHA requires specific authorization by the NRC and an approved exemption from 10 CFR 20.1703(a)(1), 10 CFR 20.1703(c), and certain parts of 10 CFR Part 20, Appendix A, Protection Factors for Respirators, Footnote d.2.(d) before use. Authorization has been received and appropriate exemption granted for the use of MSA Model 401 (brass or aluminum parts), Ultralite, and Custom 4500 Dual-Purpose SCBAs charged with 35% oxygen and 65% nitrogen. All units are to be equipped with silicone face-pieces. Regulator use is not to be initiated at temperatures >135°F. Units may be used in areas where temperatures exceed 135°F if regulator use is initiated prior to entry into those areas. Breathing gas quality and composition, including hydrocarbon exclusion, are insured by strict controls and maintained in accordance with the latest revision of the United Stated Pharacopeia (USP) - The National Formulary (NF). Due to regulator flow rate restrictions, MSA 401 SCBAs will not be used for in-containment fire fighting.

Prior to Unit 1 operation, a radiological study of the environs was performed (Section 11.3.5). It included an investigation of the background radiology relating to various forms of the aquatic and terrestrial environment. The nature and extent of the postoperational environmental survey were determined from the results of the preoperational study.

11.1 References

1. Letter from G.E. Edison, NRC, to J. P. O'Hanlon, Virginia Power, September 3, 1998, *Surry Power Station, Units 1 and 2 - Exemption From 10 CFR 20.1703(a)(1), 10 CFR 20.1703(c), and 10 CFR Part 20, Appendix A, Protection Factors for Respirators, Footnote d.2.(d), and Authorization to Use Certain Respirators for Worker Protection Inside Containment (TAC Nos M98382 and M98383),* Serial No. 98-565.

11.2 RADIOACTIVE WASTE SYSTEMS

11.2.1 Design Bases

It is Vepco's waste management policy to maintain radioactive waste effluent from the Surry Power Station at the lowest practical level. In keeping with this policy, the Radioactive waste disposal system is designed, to the extent possible in accordance with maintenance practices, to maintain releases of radioactive material and radiation exposures to unrestricted areas as far below the limits of 10 CFR 20[1] as is practical. Normally, no radioactive waste stream will be discharged from the station without having first been processed through the waste disposal system.

The liquid, solid, and gaseous waste disposal systems are common to both reactor units. Each waste disposal system is designed to accommodate radioactive wastes produced during simultaneous operation of the two units. Both units are assumed to be operating on a daily load follow cycle using boric acid between 100% and 50% power.

The systems are also designed to accommodate the corrosion products originating in the reactor coolant system, and not removed in other systems.

11.2.2 System Design

The waste disposal system and radiation monitoring system are designed to satisfy the applicable sections of the general design criteria of Section 1.4. In addition, these systems are designed to limit the discharge of radioactive materials from the station so as not to exceed the limits of 10 CFR 20 or the suggested criteria of 10 CFR 100, and so as not to endanger the health of station operating personnel. The transportation of radioactive materials from the station is carried out in such a manner as to conform with applicable Federal, state, and local ordinances. Design data are given in Table 11.2-1. An evaluation of the waste disposal systems in accordance with the requirements of 10 CFR 50, Appendix I, is provided in Appendix 11A.

The liquid waste disposal system is described in detail in Section 11.2.3. This system has been designed to ensure that the release of radioactivity to the environment will be kept at the lowest practical level.

All normally radioactive waste gases from the gaseous waste disposal system, the gas stripper in the boron recovery system, the vent and drain system, various pressure relief valves, and the containment vacuum system are regulated before discharge by the process vent subsystem, as described in Section 11.2.5.1. All these sources of gaseous effluent are, before discharge collected; diluted; filtered through charcoal filters; monitored for flow rate, pressure, temperature, and particulates and gaseous activity; and then released through the process vents. Gas stripped from liquids entering the boron recovery system is stored in decay tanks as discussed

1. Virginia Power implemented the revised 10 CFR 20 on January 1, 1994. However, as allowed by the NRC, the calculational methodology used for the design analyses is based on the revision of 10 CFR 20 to which the plant was originally licensed.

in Section 11.2.5. The process vents and the process vent blowers are sized such that the minimum exit velocity is approximately 100 fps, which prevents any significant downdrafting of the effluent.

Radioactive waste gases may also be present in the Radwaste Facility (RF) ventilation system. Minor amounts of noble gases may be entrained in the liquid waste being processed by the RF. If evaporation of the liquid waste is the process method used, these gases will be released to the RF ventilation system. The RF ventilation system is designed to process and monitor these gases and any other airborne activity produced in the facility process areas. All tanks and process equipment vents which have radioactive contents are connected to the tank vent system. This system has a demister, a charcoal filter, and high-efficiency particulate air (HEPA) filter bank. The general area ventilation for the radiological controlled areas passes through a HEPA filter bank prior to discharge. Both the general area and the tank vent ventilation systems discharge through the RF vent stack and are continuously monitored. The exit velocity of the RF vent stack is in excess of 3200 fpm.

Other gaseous effluents will normally not be radioactive during operation; however, ventilation exhaust from some primary plant areas is subject to comparatively slight radioactive contamination from such limited sources as pump gland or pipe weepage. Features are incorporated in these exhaust systems to protect the environment from these relatively remote contamination possibilities. Three filter banks (two common and one stand alone) are capable of handling the largest possible exhaust ventilation flow rate that can be aligned through the individual filter bank(s). Each filter bank consists of roughing, HEPA, and charcoal filters. Exhaust bypass arrangements for selective filtration of any exhaust add to the flexibility of the system. All bypass and filter dampers are remote manually operated from the control room as required. Ventilation exhaust trains for the safeguards areas and charging pump cubicles automatically realign on a safety injection signal as discussed in Section 9.13.4.1.

The process vent, ventilation vents, and the RF vent are continuously monitored so that the effluent activity release rates result in concentrations considerably less than those limits provided by 10 CFR 20 at the site boundary. The gaseous waste disposal system is designed to provide adequate radioactive decay storage time for the waste gases prior to discharge through the process vent and, in addition, provides sufficient capacity to allow adequate holdup of these gases even when high-flow letdown is required.

The estimated releases from the gaseous waste disposal system are based on assumptions, discussed in Section 11.2.5, regarding the operation of the system. These assumptions were developed during the original plant design in order to generate estimates of gaseous effluent releases. These estimates were used to demonstrate compliance with effluent release regulations as part of the original licensing basis. Adherence to the gaseous waste effluent requirements is monitored by procedures in accordance with the Offsite Dose Calculation Manual (ODCM). Monitoring gaseous effluents in accordance with the ODCM ensures that the composite results of the variations in gaseous waste inputs and processing on the actual releases are within the accepted current licensing basis for the gaseous waste disposal system as specified in the acceptance criteria of the ODCM.

An analysis of the estimated curies of each radionuclide released from the station gaseous waste disposal system via the waste gas decay tanks has been made to demonstrate that 10 CFR 20 will be met. This analysis is presented in Section 11.2.5, and, as can be seen from Tables 11.2-2 and 11.2-3, the yearly dose at the site boundary with the recombiner operating is about 0.007 rem and is about 0.019 rem without the recombiner. These values are 1.4% and 3.8% (with and without the recombiner, respectively) of the unrestricted area dose of 0.5 rem specified in the revision of 10 CFR 20 to which the plant was originally licensed.

11.2.3 Liquid Waste Disposal System

The liquid waste disposal system for Units 1 and 2 is shared, except for the primary drain transfer tanks and the gaseous drain system in each containment. Two systems currently exist for treating liquid wastes. These are the boron recovery system and the liquid waste disposal system. The boron recovery system, which is described in detail in Section 9.2, treats effluents collected in the primary drain tank from the vents and drains system, as well as letdown from the primary coolant that is diverted from the chemical and volume control system (CVCS). The liquid waste disposal system treats the liquid wastes originating from containment, auxiliary building, fuel building, safeguards, radwaste facility and decontamination building sumps, and from laboratory drains. Steam generator blowdown may be transferred via the component cooling heat exchanger pit sump to the liquid waste disposal system as discussed in Section 11.2.3.2.2. Liquid waste originating from the containment, auxiliary building, fuel building, safeguards, component cooling water heat exchanger, and decontamination sumps and from the laboratory drains are collected in either the low-level waste drain tank or the high-level waste drain tank depending on the valve lineup in the primary vent and drain system. Liquid wastes are normally transferred to one of the two 30,000-gallon RF liquid waste collection tanks for processing through the RF evaporator system or are processed in the RF liquid waste reverse osmosis and demineralizer system. Liquid waste originating in the radwaste facility itself is collected in sumps and pumped directly to the RF liquid waste collection tanks or through filters to the laundry drain monitor tanks. An additional 60,000 gallons of transfer/storage capacity is available via the RF liquid waste surge tanks. The RF evaporator system is available for use during high liquid waste generation periods or as a backup to the RF liquid waste reverse osmosis and demineralizer system. Processed liquid waste is sent to the liquid waste monitor tanks. An inclined plate suspended solids/oil separator is in-line for either the evaporation or reverse osmosis and demineralization process options.

Laundry waste and personnel decontamination showers and sink wastes are collected in the contaminated waste drain tanks. These tanks are pumped to the RF for processing. Some large particulate liquid wastes originating in the radwaste facility are collected in the sumps and pumped to the laundry waste system for processing. Waste is processed with a laundry pre-filter and the laundry waste filter, then collected in the laundry drains monitor tank. Processed laundry waste and RF waste can be sampled and released or mixed with other station liquid waste.

Liquid and laundry wastes discharged to the circulating water system via the RF are monitored. The liquid effluent radiation monitor is an on-line monitor with automatic isolation of the effluent discharge when a high radiation alarm is received.

Table 11.2-4 presents information regarding the originally licensed Surry liquid waste treatment system. This information provides parameters used as input into the calculation of radiation exposure to the public presented in Appendix 11A. With respect to the present Radwaste Facility, the information in Table 11.2-4 is conservative when compared to the RF radwaste volumes, DFs, and hold-up capacity. Therefore, the original evaluation of radiation exposure to the public presented in Appendix 11A bounds the design of the RF. Reference Drawings 1 through 3 and Figures 11.2-1, 11.2-2, 11.2-3, and 11.2-4 depict the liquid and laundry waste systems in the RF and their tie-ins to the station collection points.

11.2.3.1 **Components**

11.2.3.1.1 High-Level Waste Drain Tanks

Two high-level waste drain tanks are provided. Each tank has a usable capacity of approximately 2000 gallons. Level indicators are provided. These are stainless steel tanks designed according to Section III.C of the ASME Boiler and Pressure Vessel Code.

11.2.3.1.2 Low-Level Waste Drain Tanks

Two low-level waste drain tanks are provided. Each tank has a usable capacity of approximately 1785 gallons. Level indicators are provided. These are stainless steel tanks designed according to Section III.C of the ASME Code.

11.2.3.1.3 Contaminated Drain Tanks

Two contaminated drain tanks are provided. Each tank has a usable capacity of approximately 1045 gallons. Level indications are provided. These are stainless steel tanks designed to Section VIII of the ASME Boiler and Pressure Vessel Code.

11.2.3.1.4 Waste Disposal Evaporator and Auxiliaries (Installed But No Longer Used)

One forced-circulation evaporator with a feed capacity of 6 gpm is provided. The evaporator shell is fabricated from a high-nickel alloy in accordance with Section III.C of the ASME Code. Internals are fabricated from an austenitic stainless steel not susceptible to stress cracking.

The external heat source is a shell and tube steam reboiler fabricated on the tube side from a high-nickel alloy and on the shell side from carbon steel. Distillate is condensed in a water-cooled shell and tube condenser fabricated from austenitic stainless steel. The reboiler, shell, and tube condenser are all fabricated in accordance with Section III.C of the ASME Code, and TEMA

Standards. (The external heat source steam lines have been cut and capped to preclude steam and/or water leakage.)

The condensed distillate is held in the distillate accumulator. This tank is fabricated from austenitic stainless steel in accordance with Section III.C of the ASME Code.

A distillate cooler is provided to further cool the distillate. The tube side of the distillate cooler is fabricated from austenitic stainless steel and the shell side from carbon steel, in accordance with Section III.C of the ASME Code.

11.2.3.1.5 Waste Disposal Evaporator Test Tanks (Installed but no longer used)

Two waste disposal evaporator test tanks, each of 3000-gallon capacity, with level indicators, are provided. These tanks are stainless steel and designed according to Section VIII of the ASME Code.

11.2.3.1.6 Pumps

Centrifugal frame-mounted pumps with single or double mechanical seals are provided. The waste disposal evaporator bottoms pump is a canned pump. One pump is provided for each tank with cross ties where appropriate, such as on high-level waste drain tank pumps. External cooling and seal water is supplied to radioactive pump seals as required.

11.2.3.1.7 RF Liquid Waste Evaporator System

The evaporator system consists of a 30-gpm forced circulation evaporator system. The evaporator is designed to concentrate waste up to a boron concentration of 24,500 ±5% ppm or the total solids concentration of 25 weight percent.

Normal feed to the evaporator is from a liquid waste transfer pump after the SPI oil/SS remover. Clean liquid effluent from the evaporator is transferred to the evaporator distillate demineralizer for further processing prior to transfer to the liquid waste monitoring tanks. The evaporator concentrates are forwarded to the bitumen solidification system for volume reduction, solidification, and packaging, or stored for later shipment in liquid form.

The evaporator is a forced circulation system utilizing a mechanical vapor recompression (MVR) system. The MVR evaporator operates on a heat pump principle. The process vapors are compressed to a higher pressure so they will condense at a higher temperature. The hot compressed vapors condense in the heater. The liberated heat causes boiling in the evaporator vapor body. Desuperheating water is added to the vapors to recover the superheat as sensible heat.

The liquor entering the vapor body flash boils to release heat in the form of water vapor. As the water is driven from the system in to vapor phase, the liquor contained in the vapor body is further concentrated. When the boron concentration reaches 24,500 ±5% ppm or the total solid concentration of 25 weight percent, a portion is removed by gravity to the Evaporator Bottoms

Tank. The amount of concentrates removed is replaced by an increase in the feed rate, thus the solids concentration in the evaporator is decreased again.

The vapor leaving the vapor body passes through an entrainment separator. In the separator, the vapor flows upward through percolated trays and mesh pads. These remove droplets entrained in the vapor to protect the compressor wheel from erosion and to insure clean condensate.

The vapor from the separator is compressed by a high speed centrifugal compressor. The compressed vapors exit the compressor with a large amount of superheat, which is desuperheated to bring the vapor temperature close to saturation temperature and recover the superheat as sensible heat.

The vapor from the compressor is condensed on the shell side of the heater. The distillate flows by gravity to the distillate flash tank, where it is first flashed to the entrainment separator to recover heat. Then it is pumped through a distillate subcooler to the distillate demineralizer.

Concentrated waste from the evaporator is periodically discharged to the evaporator bottoms tank. The bottoms tank vent is connected to the tank vent system after passing through a vent cooler.

The major components of this subsystem are the vapor body, the heater, the recirculation pump, the entrainment separator, the vent gas cooler, a motor driven vapor compressor, a bottoms tank and bottoms tank pump. Design information on these components are given in Table 11.2-1.

11.2.3.1.8 RF Liquid Waste Reverse Osmosis and Demineralizer System

The RF liquid waste reverse osmosis (RO) and demineralizer system is designed to remove radioactivity and dissolved solids from the liquid waste process prior to collection in the liquid waste monitor tanks where liquids are sampled and discharged or reused. The RF liquid waste reverse osmosis and demineralizer system is normally in service to process liquid waste streams.

The RF liquid waste reverse osmosis and demineralizer system is designed to remove total suspended solids to < 25 ppm, and oil and grease to < 15 ppm prior to entering the liquid waste monitor tanks.

The system consists of demineralizer vessels and a Thermex RO unit. The RO concentrates are recirculated while the permeate is directed to the liquid waste monitoring tanks. The content of the process feed tank is directed to a collection tank when concentrate limits have been met.

The demineralizer vessels are designed and constructed per ASME VIII. The reverse osmosis skid was designed per ANSI B31.1.

11.2.3.1.9 Postaccident Radiation Waste Connection

The capability for processing highly radioactive postaccident liquid waste has been incorporated into the liquid waste disposal system. A flanged connection (Reference Drawing 1) is located in the vicinity of the boron recovery tanks for the purpose of discharging radioactive liquids to an external process system without requiring personnel to enter high radiation areas. The external process system would be brought onsite, if needed, following an accident. The postaccident radiation waste connection also has an isolation valve that can be operated by reach rod to further minimize personnel exposure.

11.2.3.1.10 RF Laundry Waste System

The RF laundry waste system receives waste from the contaminated drain tanks and from the RF building drain system sump pumps via a cross-connect line. The RF laundry systems consist of laundry drain prefilters and the main laundry drain filter.

The two laundry drain prefilters are installed in parallel with one operational to remove large solid matter such as cotton fibers to extend the service cycle of the downstream laundry drain filter. The filters are designed to operate at 50 gpm. The filtration media element is a bag type constructed of artificial fiber cloth. The filter media will be periodically removed manually and transferred to the dry activated waste (DAW) area for drying and volume reduction at the Radwaste Facility or packaged for offsite processing. The filter housing is constructed of 304 stainless steel.

The laundry drain filter is designed to remove particulate matter from the incoming laundry drain stream. The filter is designed to operate at a rate of 50 gpm. It is designed to remove suspended solids with a filtration media of polyethylene fiber balls. The expended filter media will be removed manually and transferred to the DAW compaction area for drying and volume reduction at the Radwaste Facility or packaged for offsite processing.

The filter vessel is designed and constructed in accordance with ASME VIII. The principal material of construction is 304 stainless steel.

Laundry waste and RF building drain system waste processed by this system is sent to the laundry waste monitor tanks for sampling prior to monitored discharge.

11.2.3.1.11 RF Monitor Tanks

The RF liquid waste evaporator, reverse osmosis, and/or demineralization process systems send their processed waste to one of two liquid waste monitor tanks. These tanks are 15,000 gallons, vertical tanks. Each tank has level indication and recirculation capability. They are constructed of 304 stainless steel and have an atmospheric pressure design. The tanks are designed and constructed to ASME Section III.

Laundry waste processed through the RF is sent to one of two laundry waste monitor tanks. These tanks are 7500 gallons, vertical tanks. Each tank has level indication and recirculation capability. They are constructed of 304 stainless steel and have an atmospheric design. The tanks are designed and constructed to ASME Section III.

11.2.3.2 **Processing Steam Generator Blowdown**

The steam generator blowdown system is described in Section 10.3.1.2. A review of the effects of the power uprate to a core power of 2546 MWt was conducted and the steam generator blowdown system was found to be adequate.

A steam generator blowdown treatment system, located in the Condensate Polisher Building, was utilized to remove impurities from the blowdown stream. The system contained prefilters, demineralizers, and postfilters. The blowdown treatment system was designed to be used during normal operation and following steam generator tube leakage. Subsequently, the system was determined to be incompatible with changes made in secondary water chemistry. As a result, the blowdown treatment system is no longer used for blowdown treatment and blowdown is untreated, except as described below. Plant modifications are being implemented to abandon-in-place and physically isolate each unit's blowdown treatment system. This plant modification has been partially implemented on Unit 1.

11.2.3.2.1 Normal Operation

During startup and power operations, blowdown is either released to the discharge canal or returned to the condenser hotwell as described in Section 10.3.1.2. During outages, the steam generators may be gravity drained through the blowdown lines to a waste neutralization sump located in the condensate polishing building. Water in the waste neutralization sump can be treated, recirculated, sampled and discharged to either the settling pond or the circulating water discharge. During discharge, the water may be directed through a filter or the filter can be bypassed.

11.2.3.2.2 Operation Following Steam Generator Tube Leak

If a steam generator tube leak occurs and shutdown is desired, station procedures provide guidance on evaluating contamination potential and determining appropriate actions for processing blowdown. Depending on activity levels, blowdown may be directed to the condenser hotwell or may need to be processed through the Surry Radwase Facility (SRF). If it is decided to process the blowdown through the SRF, procedures direct that the inventory from the affected steam generator be transferred to the component cooling (CC) heat exchanger (HX) pit sump using blowdown hose connections. Steam generator pressure will provide the motive force for transfer from the affected steam generator to the sump, but gravity transfer is possible. The CC HX pit sump pump transfers water in the sump to the combined containment and safeguard area sump pump discharge header where it can be processed by the liquid waste disposal system (Section 11.2.3). The flow rate into the sump from the affected steam generator will be limited to less than 25 gpm so that it will not exceed the pumping capacity of the CC HX pit sump pump. The flow rate can be controlled from the main control room by use of an HCV in the blowdown line.

Controls are in place to minimize the airborne activity levels in the vicinity of the CC HX pit. Hose connections will be monitored for leakage. Water entering the sump from the blowdown lines will be cooled by the steam generator blowdown coolers to a subcooled condition to preclude flashing. Flow rates into the CC HX pit sump will be limited to below the capacity of the CC HX pit sump pump so that the water level remains in the sump and does not enter the pit. This minimizes the liquid surface area and would allow for an exhaust hood to be installed over the sump if needed to reduce airborne activity levels. Health Physics will monitor the radiation levels in the areas of the routed hose and the radiation and airborne levels around the CC HX pit.

11.2.4 Solid Waste Disposal System

The solid waste disposal system provides logging, packaging, and storage facilities for scheduled shipment off the site and ultimate disposal of radioactive waste material. Materials handled as solid waste include concentrated liquid sludge, water, spent resin, spent filter cartridges, solid noncompactible and compactible trash, and other miscellaneous materials resulting from station and RF operation and maintenance. The operation of the system is described below.

11.2.4.1 **Solid Waste Handling Operations**

11.2.4.1.1 Expended Filter-Cartridge Handling Operations

Radioactive liquid service filters are removed from the system when the pressure drop across the filters becomes excessive or when the radiation level exceeds a predetermined maximum. The filter housing is surveyed prior to any opening or removal. After this is completed, the filter cover is remotely opened and removed by personnel using appropriate tools and protected by a filter removal shield, when required. A lead cask is placed over the filter housing and the filter is then moved upward into the lead cask. A drip pan is then secured to the bottom of the cask and the entire assembly is transported to the Surry Radioactive Waste Facility (SRF). The filter is placed in an approved container. The filter and container remain in the SRF until shipment to the burial site.

Surveys are conducted on the filter when the filter housing is opened, in the shielded casks, and when the container is transported to the Radwaste Facility or removed for shipment. The transport vehicle undergoes a complete radiological survey before leaving the site.

11.2.4.1.2 Spent Resin Handling Operations

A spent resin catch tank and spent resin blend tank are provided to receive spent resin from the station's ion exchangers located in the Auxiliary Building. A transfer pump is associated with each tank. Spent resin is transferred from the blend tank to a high-integrity container for shipment to a burial facility. A shipping container may be sent to the Radwaste Facility for staging prior to shipment offsite.

Primary plant resins are directed to the spent resin catch tank. Resins from the catch tank are slurried to the blend tank to produce a mixture of resin whose contents may be shipped in a high integrity container. As an option, resin from the spent resin catch tank can be sluiced to a mobil resin transfer vessel (MRTV) for shielded transport to the Radwaste Facility. Subsequent to the processing operations, the lines are flushed with primary grade water.

Spent low-activity resins from the condensate polishing system are typically dewatered to acceptable strong tight containers and sent offsite for disposal. If the activity in these resins becomes high enough that the disposal site would not accept them in drums, the resins would be slurried to high integrity containers (HICs) and then dewatered prior to shipment to an offsite radwaste burial site. There is also an option to transport the spent condensate polishing resins to the Radwaste Facility. From the Radwaste Facility, condensate polishing resins can be sent to the bitumen solidification system, sent to a high integrity container filling and dewatering station, or proportionately blended with other resins for the purpose of lowering the dose rate of some higher activity resins.

11.2.4.1.3 Evaporator Concentrate Operations

Solids that are concentrated in the evaporator are discharged to the evaporator bottoms tank. These concentrates are at 25% by weight solids or at 24,500 ±5% ppm boron. Additionally, sludge from the suspended solids separator is periodically pumped to the bottoms tank.

The concentrates and sludges in the bottoms tank may be pumped to the waste batch tanks through heat traced lines where the concentrate waste is pretreated for processing by the solidification system.

In order to preclude plugging of the concentrates transfer piping, redundant heat tracing circuits are installed. Clean, hot water flushing connections are included to clean each line following concentrates transfer.

11.2.4.1.4 Solidification Operations (Installed but no longer used)

The bitumen solidification system incorporates a chemical and physical process for reducing the volume of radwaste and for incorporating the radwaste into a solidified bitumen matrix. The process uses a LUWA thin-film evaporator operating at a waste product outlet temperature of approximately 320°F. This results in the evaporation of free water from waste effluents and the remaining solids are incorporated in a bitumen matrix. Solidification of the end product occurs upon the natural cooling of the binder.

The system is capable of processing waste which includes evaporator concentrates and spent bead resin. Waste to be processed is collected in one of two waste batch tanks. The waste is sampled and chemically pretreated to prepare it for processing.

The conditioned waste is fed at a controlled rate to a thin film evaporator. Molten bitumen is simultaneously fed into the evaporator through a second feed nozzle. The evaporator is heated by means of hot thermal fluid flowing through an external jacket. As the waste flows downward through the evaporator, the water is evaporated and the water vapor flows counter-currently upward and out of the evaporator. The waste solids are mixed with molten bitumen and exit the bottom of the evaporator, flowing into a waste container. Upon cooling, the waste/bitumen mixture solidifies into a freestanding, monolithic solid with free liquids less than 0.5 percent by volume of the waste form.

The water vapor leaving the thin film evaporator is condensed in a shell and tube condenser. The condensate flows into the distillate oil separator. When this tank is filled, the distillate is pumped to a liquid waste collection tank.

The bituminized waste product flows from the discharge valve of the thin film evaporator into a 55-gallon steel drum. Once filled and cooled, drums are inspected for free liquid, capped, smeared and surveyed. The drums are then transferred to the RF storage area to await shipment to a licensed disposal contractor.

Figure 11.2-5 depicts the solidification process flow.

11.2.4.1.5 Ultimate Disposal Operations

All packages containing radioactive nonfissionable material, and the procedures used to prepare these for offsite shipment, are in accordance with U. S. Department of Transportation regulations. The Radwaste Facility, Low-Level Waste Storage Facility and Sea Van Storage Pad are facilities used for the storage of radioactive material. All waste material is transferred either to a licensed disposal or processing contractor or to common carrier for delivery to a licensed disposal or processing contractor. Radwaste shipments fall under the purview of Vepco's procedures and quality assurance program.

11.2.4.2 Components

All components listed below except the spent resin catch tank, the spent resin blend tank, and their associated pumps are located within the Radwaste Facility.

11.2.4.2.1 Spent Resin Catch Tank
Spent Resin Blend Tank

One of each tank is provided. Each tank is installed in a separate cubicle on Elevation 6 ft. 10 in. in the Decontamination Building. The normal operating volume (high level to low level) is approximately 214 ft^3. Total usable volume is approximately 245 ft^3. Vessels are designed to ASME Section VIII.

11.2.4.2.2 Spent Resin Catch Tank Transfer Pump
Spent Resin Blend Tank Transfer Pump

One of each pump is provided. Pumps are of the progressive cavity design. Each pump is designed to deliver 27.7 gal/100 rpm at 0 psi.

11.2.4.2.3 Evaporator Bottoms Tank

The evaporator bottoms tank is a 5000-gallon, 11-foot diameter, vertical tank with a dished bottom and a flat top. The tank is constructed of Inconel 625 and is equipped with a mixing eductor and a demineralizer water flush header, level indication, and heat tracing. The heat tracing prevents concentrates from solidifying in the tank.

11.2.4.2.4 Spent Resin Collection Tanks

There are four spent resin collection tanks in the Radwaste Facility. These tanks are 1020-ft^3 capacity, vertical tanks with a 10-foot diameter. Each tank is constructed of 304 stainless steel and is designed for atmospheric pressure. Each tank is equipped with mixing lines, flush lines, decant lines and an overflow. Each of the two mixing lines is attached to internal mixing eductors.

11.2.4.2.5 Spent Resin Collection Tank Pumps

There are two spent resin collection tank pumps each of which is capable of pumping from any of the four spent resin collection tanks. Each pump is rated at 440 gpm at a total dynamic head of 86 psi. Parts of the pumps in contact with the radioactive resin/water slurry are made of stainless steel.

11.2.4.2.6 Waste Batch Tanks

There are two 1000-gallon waste batch tanks. Each tank is constructed of 316L stainless steel and is equipped with external heating elements to prevent the solidification of evaporator concentrates. Each tank has mechanical agitators for mixing.

11.2.4.2.7 Bitumen Storage Tank

The bitumen storage tank is a horizontal 6000-gallon carbon steel tank. The tank has an internal electric heater and is heavily insulated. The tank is equipped with instrumentation for tank level and temperature.

11.2.4.2.8 Bitumen Metering and Transfer

Bitumen is metered by a gear type metering pump capable of accurate control of the bitumen feed between 0.04-0.92 gpm. The piping from the storage tank to the thin film evaporator are not heat traced. Instead, transfer lines are jacketed pipes using the heating oil from the thin film evaporator heating system to maintain flow of the bitumen feed.

11.2.4.2.9 Thin Film Evaporator

The thin film evaporator is a LUWA design capable of a 52-gal/hr evaporation rate. The body of the evaporator is made of 316L stainless steel. The internal paddle assembly in the evaporator continuously spreads the feed material into a thin film along the vessel walls to assist in the evaporation. The constant action of the paddle assembly also assures adequate and uniform mixing of the waste and the bitumen binder. The paddle assembly is turned by a 15 hp electric motor.

11.2.4.2.10 Distillate Oil Separator Tank

Water vapor from the thin film evaporator is condensed and collected in the distillate oil separation tank. Due to the use of bitumen in this process, small amounts of light weight oils are volatilized during the evaporator process. These oils are separated in the distillate oil separation tank and are skimmed off for separate treatment. The condensed water in the tank sent to the liquid waste collection tanks via the distillate transfer pump.

11.2.5 Gaseous Waste Disposal System

The process vent subsystem regulates the discharge of potentially high-activity waste gases to the atmosphere. The ventilation vent subsystem described in Section 9.13 and the RF vent described in 11.2.2 regulate the discharge of potentially low-activity air streams to the atmosphere. Radioactive waste discharges from these subsystems are monitored by particulate and gas monitors that are part of the process radiation monitoring system described in Section 11.3.3. Limitations on gaseous releases, and associated reporting requirements, are included in the Technical Specifications and the Offsite Dose Calculation Manual.

Waste gases, primarily hydrogen, nitrogen, and minor amounts of fission product gases, such as xenon and krypton, are removed from reactor coolant letdown by the stripper in the boron recovery system. The stripped gases are processed in the gaseous waste disposal system.

The gaseous waste disposal system is designed to provide adequate radioactive decay storage time for the waste gases and, in addition, provide long-term holdup of these gases when high-flow letdown is required.

Gases pass from the stripper to the stripper surge tank, where they are compressed. From the surge tank, the gases are bled off to the waste gas surge drum. At a pressure of approximately 1 atm, the waste gas diaphragm compressor transfers the gases to one of two waste gas decay tanks.

When released, effluent from the waste gas decay tanks is mixed with dilution air, effluent from the containment vacuum system, and the aerated vents from the vent and drain system. The combined gaseous waste is filtered through charcoal and high-efficiency particulate air (HEPA) filters before being released to the atmosphere. The process vent blowers maintain a small vacuum in the charcoal filters to prevent outleakage from the filter assembly. The decay tank contents are sampled before any release to the process vent.

11.2.5.1 **Process Vent Subsystem**

Gaseous wastes enter the process vent subsystem from the gaseous waste disposal system, the stripper in the boron recovery system, the vent and drain system, various pressure relief valves, and the containment vacuum system, as shown in Reference Drawings 4 and 5.

A catalytic recombiner system is installed (but not used) as part of the gaseous waste disposal system. A summary description of the catalytic recombiner is provided in Section 11.2.5.3.1.

Two double-walled waste gas decay tanks are provided. Each tank is buried for tornado protection. The inner tank is fabricated from austenitic stainless steel in accordance with Section III.C of the ASME Code, and the outer tank from carbon steel, in accordance with Section VIII of the ASME Code. Sampling connections are provided for the tank contents and for leakoff in the annular intercept space between the tanks. The decay tanks have piping connections for parallel operation with alternate feed and bleed.

Overpressure relief protection is provided at the waste gas decay tanks in accordance with Section III.C of the ASME Code. The protective devices consist of bellows-sealed pressure relief valves followed by rupture disk assemblies. The use of bellow seals and rupture disks precludes the leakage of the waste gas to the environment during normal operation of the gaseous waste disposal system. The piping downstream of the protective devices relieves to the process vent through the radiation monitor station.

Effluent from the waste gas decay tanks is mixed with dilution air, effluent from the containment vacuum system, and the aerated vents from the vent and drain system. The combined gaseous waste is filtered through charcoal filters before being released to the atmosphere. The process vent blowers maintain a small vacuum in the charcoal filters to prevent outleakage from the filter assembly. The decay tank contents are sampled before any release to the process vent.

The entire discharge stream of radioactive letdown gas and dilution air is monitored for flow rate, pressure, temperature, and particulate and gaseous activity before release through the process vents. The total flow is regulated by a flow control valve on the process vent blower. The ratio of dilution air to waste gas letdown flow is such that the mixed streams never enter the flammability region of the air-steam-hydrogen phase diagram.

The process vent and the process vent blowers are sized such that the minimum exit velocity is approximately 100 fps. This exit velocity prevents any significant downdrafting of the effluent. The process vent terminates at an elevation approximately 22 feet above the top of one of the containment structures.

The process vent monitors are set such that the effluent activity release rate results in concentrations less than those limits provided in the revision of 10 CFR 20 to which the plant was originally licensed. In the event that the activity of the effluent stream exceeds the setting of the monitors, the process vent control station automatically terminates the release of waste effluents from the waste gas decay tanks and isolates the containment vacuum system from the process vent sub system. The monitor also activates an alarm in the control room before valve closure if the activity approaches a preset value. Subsequent restart of the system is manual, in accordance with procedures. The discharge of gases from the waste gas decay tanks is initiated and controlled separately.

The gaseous waste disposal system is designed to provide adequate radioactive decay storage time for the waste gases and, in addition, to provide long-term holdup of these gases when high-flow letdown is required.

The combined volume of the two waste gas decay tanks is sized to process the gas stripped from the estimated annual average letdown flow of 17 gpm, based on simultaneous operation of two reactor units.

The average gas stripping rate is a function of the average letdown flow rate, and this flow rate is dependent on the assumed plan of operation as described in Section 9.2.

On this basis, the total annual letdown volume for two units is 8.94×10^6 gal, and the average annual letdown flow rate for two units is 17 gpm.

If the hydrogen volume is assumed to be 35 cm^3/kg and 90% of the total gas volume, then the hydrogen stripping rate at an annual average of 17-gpm letdown is 0.0792 scfm and the total gas stripping rate is 0.088 scfm.

The gas decay tanks are sized so that a 17-gpm letdown rate with the recombiner not operating gives an average holdup time equivalent to approximately 5 half-lives of Xe-133 (30 days).

Assuming 1% failed fuel, the estimated curies of each radionuclide released from the station via the gaseous waste disposal system are listed in Tables 11.2-2 and 11.2-3. Table 11.2-2 is for a waste gas cycle with the recombiner not operating, which is the design basis for the system, and Table 11.2-3 is for a waste gas cycle with the recombiner operating. With the recombiner not operating, the gas cycle is 30 days of feed/20 days of decay/10 days of bleed; most of the gas is hydrogen. With the recombiner operating, the feed portion of the waste gas cycle can vary between 30 days and approximately 300 days, the time needed to reach maximum design pressure in the tank. To be conservative, 300 days of feed was chosen as the basis for Table 11.2-3, and the waste gas cycle considered was 300 days of feed/20 days of decay/10 days of bleed.

In each case, it is assumed that all of the gases and 0.1% of the iodines are removed at the gas stripper and sent to the waste gas decay tanks except for hydrogen in the case with the recombiner operating. The system is operated so that one tank is on the feed portion of the cycle while the other is on the decay and bleed portion.

The equilibrium reactor coolant activity is a function of the waste gas removal rate by the gas stripper. Using the parameters listed in Table 9.1-5 and a 17-gpm letdown rate to the gas stripper, the equilibrium coolant activity for each radionuclide was calculated. These are also listed in Tables 11.2-2 and 11.2-3.

As can be seen from these tables, the yearly dose at the site boundary is about 0.019 rem for continuous operation with a 30 days of feed/20 days of decay/10 days of bleed cycle, and about 0.007 rem for the 300 days of feed/20 days of decay/10 days of bleed cycle. Both of these values are well below the member of the public dose limit of 0.5 rem/yr set forth in the revision of 10 CFR 20 to which the plant was originally licensed.

11.2.5.2 **Ventilation Vent Subsystem**

The ventilation vent subsystem is considered to be a portion of the gaseous waste disposal system only for purposes of radiological surveillance, and it is designed on this basis. However, since it handles air streams of very low activity levels, and since the gases to be handled are predominantly of nonradioactive origin, this subsystem has been considered as an auxiliary system for the purpose of this report. A full description of this subsystem is included in Section 9.13.

11.2.5.3 **Components**

The major components of the gaseous waste disposal system are described below.

11.2.5.3.1 Catalytic Recombiner (Installed But Not Usable)

One skid-mounted catalytic recombiner system is provided. The system includes duplicate full-capacity recycle compressors, duplicate full-capacity electrical preheaters, duplicate full-capacity catalytic recombiners, one aftercooler condenser, one moisture separator, one electrical reheater, duplicate hydrogen analyzers of the thermal conductivity type for the recombiner influent and effluent, duplicate oxygen analyzers of the paramagnetic type on the recombiner effluent, a single oxygen analyzer on the recombiner influent, and one bleed stream cooler. The recombiner system operates at approximately 22 psia and has a feed capacity of approximately 1.14 scfm. The diluent is nitrogen. The catalytic recombiner system is designed according to Section III.C of the ASME Code.

The recycle compressors are rotary positive blowers designed to circulate 40 cfm at 8 psig discharge pressure. They are of gas-tight construction, with three mechanical shaft seals in series between the circulating gas and the outside atmosphere. The end bell of the compressor is pressurized with nitrogen as a further precaution against outward leakage.

The preheaters are stainless steel pipes with external electrical heating elements and are used to raise the temperature of the recycle stream to 300°F before the recycle stream enters the catalyst bed.

The catalytic recombiners are all-metal, low-halogen catalysts. Each bed contains miles of crimped-nickel alloy ribbon coated with catalytically activated precious metals, mainly palladium and platinum.

The aftercooler condenser is a pipe-within-a-pipe heat exchanger, with the recycle gas flowing through the inner pipe and component cooling water flowing through the outer pipe. The aftercooler condenser condenses the water vapor by cooling the recycle gas stream and lowering the water vapor in it to a dewpoint of 75°F.

The moisture separator is a centrifugal separator with an automatic drain operated by a level controller and has high- and low-level alarms. Moisture from the moisture separator drains to a high-level waste drain tank.

The reheater is similar to the preheaters, and raises the temperature of the recycle stream to 120°F.

The hydrogen analyzers are of the thermal conductivity type; the oxygen analyzers are of the paramagnetic type.

11.2.5.3.2 Waste Gas Surge Tank

One waste gas surge tank with a 15.7 ft^3 capacity is provided. This tank is operated at a pressure of approximately 10 to 20 psia. The tank is fabricated from austenitic stainless steel in accordance with Section III.C of the ASME Code.

11.2.5.3.3 Waste Gas Compressor

Two waste gas compressors of the diaphragm type are provided. Each has a rated capacity of 1.5 scfm at a discharge pressure of 120 psig. The compressor heads are leak tested to ensure that the leakage does not exceed a predetermined amount.

11.2.5.3.4 Waste Gas Decay Tank

Two buried waste gas decay tanks are provided. These tanks have double-wall construction with feed and bleed lines; sample, nitrogen purge, drain, and relief valve lines to the inner tank; and sample, nitrogen purge, drain, and relief valve lines from the outer tank. An access opening is provided to the inner tank. In addition, adequate grounding and corrosion protection are provided. The inner tank is fabricated from stainless steel in accordance with Section III.C of the ASME Code, and the outer tank from carbon steel in accordance with Section VIII of the ASME Code.

To ensure an explosive gas mixture does not develop in the tanks, samples are taken via the oxygen analyzer. Compressors have been installed as sample pumps but are normally isolated and bypassed. The differential pressure between the waste gas decay tank and the waste gas surge drum is used to induce flow through the oxygen analyzer.

11.2.5.3.5 Process Vent Blowers

Two full-capacity dilution air blowers of 300 cfm capacity at 2 psia are provided. The blowers are of a centrifugal type, located in a field fabricated box with the blower suction from the box's interior. Some inleakage is tolerated.

11.2.5.3.6 Charcoal Filters

Two charcoal filter beds are provided to service approximately 300-scfm radioactive gas. The filters are maintained at a subatmospheric pressure.

11.2.5.3.7 Postaccident Radiation Waste Connection

The capability for processing highly radioactive postaccident gaseous waste has been incorporated into the gaseous waste disposal system. A flanged connection (Reference Drawings 4 & 5) is located in the vicinity of the boron recovery tanks for the purpose of discharging radioactive gases to an external process system without requiring personnel to enter high radiation areas. The external process system would be brought onsite, if needed, following an accident. The postaccident radiation waste connection also has an isolation valve that can be operated by reach rod to further minimize personnel exposure.

11.2.6 Tests and Inspections

11.2.6.1 Construction and Fabrication

During the manufacturing period, Vepco's inspectors inspected all equipment periodically, as required, to ensure that all equipment had been provided in strict accordance with specifications. Shop hydrostatic and performance tests of principal equipment were witnessed by Vepco's inspectors. Certified code inspection data sheets were provided by manufacturers for all equipment covered by ASME or other applicable codes.

During the construction period, all pressure systems were subjected to field hydrostatic or pneumatic tests to verify the integrity of welded connections and to ensure that the system as a whole functioned as intended.

During the preliminary operation period, all equipment in the waste disposal system was tested to verify conformance with specification performance requirements. All control systems and interlocks were tested and operated to ensure satisfactory functional performance and reliability.

The following information is HISTORICAL and is not intended or expected to be updated for the life of the plant.

On February 9 and 10, 1971, a performance test of the waste disposal system catalytic recombiner was conducted to demonstrate the actual catalyst performance over a range of operating conditions. Tests were rerun at three hydrogen inlet concentrations: test 1, 2.0% hydrogen; test 2, 0.5% hydrogen; and test 3, 3.2% hydrogen.

The specified test requirements at each test condition were to maintain the outlet oxygen concentration at no more than 1% and to have less than 100 ppm of hydrogen in the outlet.

The catalyst efficiency was greater than 99.5% over the range of hydrogen flow rates and concentrations tested. The outlet hydrogen and oxygen concentrations were well within the stated limits. The tests confirmed the capability of the recombiner to operate over the range of operating conditions expected during normal operations.

11.2.6.2 **Operation**

The basic function of the waste disposal system is to release controlled amounts of radioactivity to the environment with no undue effects on the health and safety of the general public. This is accomplished by ensuring that all releases from the station are at less than the maximum levels of radioactivity set by applicable regulatory agencies, as given in Section 11.2.1.

The following is a list of the types and areas monitored to ensure the proper functioning of the waste disposal system:

1. Continuous Process Monitoring: As described in Section 11.3.3, process radiation monitors continuously monitor certain key systems where radioactive material may exist. These monitors give an indication of the waste-processing requirements of certain systems.
2. Batch Sample Process Monitoring: As described in Section 9.6, batch samples, obtained from certain subsystems, provide information on the effectiveness of ion exchangers, filters, and evaporators. This gives an indication of the effectiveness of the various waste processing subsystems. The monitoring of the gas in the waste gas holdup tanks avoids the storage of excessive activity.
3. Continuous Monitoring of Discharge Effluents: As described in Section 11.3.3, radiation monitors continuously monitor discharges from the process and ventilation vent systems and RF systems and the liquid waste disposal and service water systems. These monitors give an indication of liquid and gaseous radiation discharges to the environment and provide alarms with automatic valve closure when radiation levels exceed a preset level, thus terminating discharge.
4. Radiation Survey of Radioactive Waste/Material Containers: Radiation surveys and smear samples are taken of shipping casks, drums, etc., that contain radioactive waste/material to ensure that such waste/material is properly contained and meets transportation regulations.
5. Environmental Monitoring: As described in Section 11.3.5, environmental samples are taken to indicate the effect of liquid and gas discharges on the environment and the compliance of these discharges with applicable regulations.

To ensure that the performance of the waste disposal systems is meeting design criteria, the following checks are made:

1. Standardized laboratory radiochemical analytical procedures are used to verify decontamination factors.
2. Radiation monitors are periodically checked with remotely operated check sources. The Local Processing Units (LPUs) of the MGP Instruments (MGPI) monitors perform various self checks automatically. Their electrical self check introduces a known and fixed level of pulses into the electronics excluding the detector and verifies that the response is correct, otherwise a fault is generated. Additionally, the electronics continuously monitor the detector for a minimum countrate, otherwise a fault alarm is generated. In addition, samples are

withdrawn from the process streams being monitored and analyzed to ensure compliance with regulatory limits. Those monitors that actuate control valves by a high radiation signal are periodically tested to ensure the valves activate on alarm signal.

3. Portable survey instruments and laboratory analytical instruments are periodically calibrated with known radiation sources in accordance with station health physics procedures.

4. Radiation levels on the outside of components, pumps, valves, and piping in the waste disposal systems are monitored periodically to avoid inadvertent discharge of activity that may accumulate with time.

11.2 REFERENCE DRAWINGS

The list of Station Drawings below is provided for information only. The referenced drawings are not part of the UFSAR. This is not intended to be a complete listing of all Station Drawings referenced from this section of the UFSAR. The contents of Station Drawings are controlled by station procedure.

	Drawing Number	Description
1.	11448-FM-083C	Flow/Valve Operating Numbers Diagram: Vent and Drain System, Unit 1
2.	11448-FM-30B	Flow Diagram: Waste Disposal System
3.	11448-FM-30C	Flow Diagram: Waste Disposal System
4.	11448-FM-090A	Flow/Valve Operating Numbers Diagram: Gaseous Waste Disposal System, Unit 1
5.	11448-FM-090B	Flow/Valve Operating Numbers Diagram: Gaseous Waste Disposal System, Unit 1

Table 11.2-1
WASTE DISPOSAL SYSTEMS DESIGN DATA

Spent Resin Catch Tank (1-LW-TK-14)	
Number	1
Capacity	1955 gal
Operating capacity	214 ft^3
Design pressure	150 psig
Design temperature	200°F
Operating pressure	Atmospheric
Operating temperature	60-120°F
Material	SS, SA-240, 316L
Design code	ASME VIII, D.V.
Spent Resin Blend Tank (1-LW-TK-15)	
Number	1
Capacity	1955 gal
Operating capacity	214 ft^3
Design pressure	150 psig
Design temperature	200°F
Operating pressure	Atmospheric
Operating temperature	60-120°F
Material	SS, SA-240, 316L
Design code	ASME VIII, D.V.
High-level Waste Drain Tank	
Number	2
Capacity (each)	2390 gal
Operating capacity (each)	2000 gal
Design pressure	25 psig
Design temperature	200°F
Operating pressure	Atmospheric
Operating temperature	120°F
Material	SS Type 316
Design code	ASME III.C
Low-Level Waste Drain Tank	
Number	2
Capacity (each)	2874 gal
Operating capacity (each)	1785 gal
Design pressure	25 psig
Design temperature	200°F
Operating pressure	Atmospheric
Operating temperature	120°F

Table 11.2-1 (continued)
WASTE DISPOSAL SYSTEMS DESIGN DATA

Low-Level Waste Drain Tank (continued)			
Material	SS Type 316		
Design code	ASME III.C		
Contaminated Drains Collection Tanks			
Number	2		
Capacity (each)	1230 gal		
Operating Capacity (each)	1045 gal		
Design pressure	25 psig		
Design temperature	200°F		
Operating pressure	Atmospheric		
Operating temperature	120°F		
Material	SS Type 304		
Design code	ASME VIII		
Liquid Waste Test Tanks (installed but no longer used)			
Number	2		
Capacity (each)	3000 (usable) gal		
Design pressure	25 psig		
Design temperature	212°F		
Operating pressure	Atmospheric		
Operating temperature	140°F		
Material	SS Type 304		
Design code	ASME VIII		
Waste Gas Catalytic Recombiner (installed but no longer used)			
Number	1		
Capacity, feed	1.31 scfm		
Design feed pressure	22 psia		
Design feed temperature	70-120°F		
Feed composition, scfm	Max.	Avg.	Min.
H_2	1.14	0.0805	0
H_20	0.04	0.026	0
Xe	Trace		
Kr	Trace		
N_2	0.130	0.00922	0
Design bleed pressure	14.0 psia		
Design H_2 bleed concentration	0.1% max.		
Design bleed volume	10% of feed		
Design code	ASME III.C		

Table 11.2-1 (continued)
WASTE DISPOSAL SYSTEMS DESIGN DATA

Waste Gas Decay Tanks		
Number	2	
Capacity (each)	434 ft^3	
Design pressure	Outer Tank	Inner Tank
	From 30 in. Hg vacuum to 150 psig	From 30 in. Hg vacuum to 150 psig
Design temperature	200°F	200°F
Operating pressure	Atmospheric	115 psig
Operating temperature	120°F	140°F
Material	Carbon	SS Type 304L
Design code	ASME VIII	ASME III.C
Earthquake design	Complies with Class I requirements	
Waste Gas Surge Tank		
Number	1	
Capacity	15.7 ft^3	
Design pressure	From 30 in. Hg vacuum to 30 psig	
Design temperature	300°F	
Operating pressure	15 psig	
Operating temperature	120°F	
Material	SS Type 304	
Design code	ASME III.C	
Low-level Waste Drain Pumps		
Number	2 (1 required)	
Type	Horizontal centrifugal	
Motor horsepower	7.5 hp	
Seal type	Double mechanical	
Capacity (each)	120 gpm	
Head at rated capacity	94 ft	
Design pressure	150 psig	
Materials		
Casing	SS Type 316	
Shaft	A 5564, Type 630	
Impeller	SS Type 316	
High-level Waste Drain Pumps		
Number	2 (1 required)	
Type	Horizontal centrifugal	
Motor horsepower	7.5 hp	
Seal type	Double mechanical	
Capacity (each)	120 gpm	

Table 11.2-1 (continued)
WASTE DISPOSAL SYSTEMS DESIGN DATA

High-level Waste Drain Pumps (continued)	
Head at rated capacity	86 ft
Design pressure	150 psig
Materials	
Pump casing	SS Type 316
Shaft	A564, Type 630
Impeller	SS Type 316
Contaminated Drains Transfer Pumps	
Number	2 (1 required)
Type	Horizontal centrifugal
Motor horsepower	10 hp
Seal type	Mechanical
Capacity (each)	75 gpm
Head at rated capacity	166 ft
Design pressure	150 psig
Materials	
Pump casing	SS Type 316
Shaft	A564, Type 630
Impeller	SS Type 316
Spent Resin Catch Tank Transfer Pump (1-LW-P-12)	
Number	1
Type	Progressive cavity
Capacity	27.7 gal/100 rpm @ 0 psi
Design pressure	100 psig
Design temperature	150°F
Material	SS Type 316
Design Code	None
Spent Resin Blend Tank Transfer Pump (1-LW-P-13)	
Number	1
Type	Progressive cavity
Capacity	27.7 gal/100 rpm @ 0 psi
Design pressure	100 psig
Design temperature	150°F
Material	SS Type 316
Design Code	None

Table 11.2-1 (continued)
WASTE DISPOSAL SYSTEMS DESIGN DATA

Process Vent Blower	
Number	2 (1 required)
Type	Multistage centrifugal
Motor horsepower	7.5 hp
Capacity (each)	300 scfm
Differential pressure	2 psi
Suction pressure	14.0 psia
Design pressure	15 psig
Materials	
Casing	Cast iron
Impeller	Aluminum
Shaft	SS Type 304
Waste Gas Compressor	
Number	2 (1 required)
Type	Diaphragm
Motor horsepower	1.5 hp
Capacity (each)	1.5 scfm
Discharge pressure at rated capacity	120 psig
Design pressure	220 psig
Materials	
Cylinder	Carbon steel
Piston rod	Forged steel
Piston	Nodular iron
Diaphragms and parts contacting waste gas	SS Types 304/316
Low-Level Waste Drain Filter (installed but no longer used)	
Number	1
Retention size, microns	5
Filter element material	Fibre
Capacity normal	50 gpm
Capacity maximum	75 gpm
Material	SS Type 304
Design pressure	150 psig
Design temperature	250°F
Design code	ASME III.C
High-level Waste Drain Filter (installed but no longer used)	
Number	1
Retention size, microns	5
Filter element material	Fibre

Table 11.2-1 (continued)
WASTE DISPOSAL SYSTEMS DESIGN DATA

High-level Waste Drain Filter (installed but no longer used) (continued)	
Capacity normal	50 gpm
Capacity maximum	75 gpm
Material	SS Type 304
Design pressure	150 psig
Design temperature	250°F
Design code	ASME III.C
Contaminated Drains Filters (installed but no longer used)	
Number	2 (1 required)
Filter element material	Porous stone media
Capacity normal	50 gpm
Material	SS Type 304
Design pressure	150 psig
Design temperature	120°F
Design code	ASME VIII
Liquid Waste Collection Tank	
Number	2
Capacity	30,000 gal
Design pressure	Atmospheric Plus Content
Design temperature	150°F
Operating pressure	Full of water
Operating temperature	104°F
Material	SS 316L
Design code	ASME III
Liquid Waste Surge Tanks	
Number	2
Capacity	30,000 gal
Design pressure	Atmospheric Plus Content
Design temperature	150°F
Operating pressure	Full of water
Operating temperature	104°F
Material	SS 316L
Design code	ASME III
Liquid Waste Collection Tank Pumps	
Number	2
Type	Centrifugal
Motor horsepower	15 hp
Seal type	Double Mechanical

Table 11.2-1 (continued)
WASTE DISPOSAL SYSTEMS DESIGN DATA

Liquid Waste Collection Tank Pumps (continued)	
Capacity (each)	300 gpm
Head at rated capacity	74 ft
Design pressure	142.2 psig
Materials	
Casing	SS 316
Shaft	SS 316
Impeller	SS 316
Liquid Waste Surge Tank Pumps	
Number	2
Type	Centrifugal
Motor horsepower	15 hp
Seal type	Double Mechanical
Capacity (each)	300 gpm
Head at rated capacity	74 ft
Design pressure	142.2 psig
Materials	
Casing	SS 316
Shaft	SS 316
Impeller	SS 316
SPI Suspended Solids/Oil Separator	
Number	2
Design capacity	30 gpm (each)
Design pressure	Atmospheric
Design temperature	150°F
Material	SS 316L
Separation area, equiv.	8 ft^2 (each)
Design code	ASME III
Liquid Waste Transfer Pumps	
Number	2
Type	Centrifugal
Motor horsepower	10 hp
Seal type	Double Mechanical
Capacity (each)	40 gpm
Heat at rated capacity	200 ft
Design pressure	142.2 psig
Materials	
Casing	SS 316

Table 11.2-1 (continued)
WASTE DISPOSAL SYSTEMS DESIGN DATA

Liquid Waste Transfer Pumps (continued)	
Materials (continued)	
Shaft	SS 316
Impeller	SS 316
Liquid Waste Filter/liquid Waste Oil Filter	
Number	1
Retention size, microns	N/A
Filter element material	Charcoal
Capacity normal	60 gpm
Capacity maximum	60 gpm
Material	SS 316L
Design pressure	200 psig
Design temperature	150°F
Design code	ASME VIII
Evaporator Recirculation Pump	
Number	1
Type	Axial Flow
Motor horsepower	75 hp
Seal type	Double Mechanical
Capacity (each)	9000 gpm
Head at rated capacity	15 ft
Design pressure	60 psig
Materials	
Casing	Alloy 20
Shaft	Alloy 20
Impeller	Alloy 20
Oil Drain Tank	
Number	1
Capacity	1070 gal
Design pressure	Atmospheric Plus Water Full
Design temperature	150°F
Operating pressure	Full of Content
Operating temperature	104°F
Material	SS 316L
Design Code	ASME III
Vapor Recompressor	
Number	1
Type	Centrifugal
Motor horsepower	600 hp

Table 11.2-1 (continued)
WASTE DISPOSAL SYSTEMS DESIGN DATA

Vapor Recompressor (continued)		
Seal type	Labyrinth	
Capacity (each)	7904 ACFM	
Compression Ratio	1.97	
Design pressure	150 psig	
Materials		
Casing	SS 316L	
Shaft	SS 316L	
Impeller	SS 316L	
Distillate Flash Tank		
Number	1	
Capacity	150 gal	
Design pressure	30 psig	
Design temperature	300°F	
Operating pressure	0.5 psig	
Operating temperature	212°F	
Material	SS 316L	
Design Code	ASME VIII	
Distillate Subcooler		
Number	1	
Total duty	1,589,994 Btu/hr	
	Shell	Tube
Design pressure	150 psig	75 psig
Design temperature	250°F	300°F
Operating pressure	60 psig	52 psig
Operating temperature, in/out	80/92.5°F	212/120°F
Material	SS 304/Tube Side Carbon Steel/Sheet Side	
Fluid	Cooling Water	Distillate
Design code	ASME VIII	
Evaporator Heater		
Number	1	
Total duty	17,787,245 Btu/hr	
	Shell	Tube
Design pressure	50 psig	45 psig
Design temperature	300°F	300°F
Operating pressure	15 psig	5 psig
Operating temperature, in/out	247.8/247.8°F	218.3/222.3°F
Material	SS 316L/Alloy 20	Inconel 625

Table 11.2-1 (continued)
WASTE DISPOSAL SYSTEMS DESIGN DATA

Evaporator Heater (continued)		
	Shell	Tube
Fluid	Steam	Liquid Waste
Design code	ASME VIII	
Liquid Waste Monitor Tanks		
Number	2	
Capacity	15,000 gal	
Design pressure	Atmospheric Plus Contents	
Design temperature	150°F	
Operating pressure	Full of Water	
Operating temperature	104°F	
Material	SS 304	
Design Code	ASME III	
Liquid Waste Monitor Tank Pumps		
Number	2	
Type	Centrifugal	
Motor horsepower	60 hp	
Seal type	Single Mechanical	
Capacity (each)	330 gpm	
Head at rated capacity	326 ft	
Design pressure	275 psig	
Materials		
Casing	316 SS	
Shaft	Steel	
Impeller	316 SS	
Laundry Drain Pre-filters		
Number	2	
Rentention size	50 microns	
Filter element material	Bag	
Capacity, normal	50 gpm	
Capacity maximum	50 gpm	
Material	SS 304	
Design pressure	150 psig	
Design temperature	150°F	
Design Code	ASME VIII	
Laundry Drain Filter		
Number	1	
Rentention size, microns	N/A	
Filter element material	Polyester	

Table 11.2-1 (continued)
WASTE DISPOSAL SYSTEMS DESIGN DATA

Laundry Drain Filter (continued)	
Capacity, normal	50 gpm
Capacity maximum	50 gpm
Material	SS 304
Design pressure	150 psig
Design temperature	150°F
Design Code	ASME VIII
Laundry Drain Monitor Tank	
Number	2
Capacity	7500 gal
Design pressure	Atmospheric Plus Content
Design temperature	150°F
Operating pressure	Full of Water
Operating temperature	104°F
Material	SS 304
Design Code	ASME III
Laundry Drain Monitor Pump	
Number	2
Type	Centrifugal
Motor horsepower	40 hp
Seal type	Single Mechanical
Capacity (each)	140 gpm
Head at rated capacity	310 ft
Design pressure	225 psig
Materials	
Casing	Ductile Iron
Shaft	Steel
Impeller	Cast Iron
Evaporator Bottoms Tank	
Number	1
Capacity	5000 gal
Design pressure	Atmospheric Plus Contents
Design temperature	275°F
Operating pressure	Full of Water
Operating temperature	212°F
Material	Inconel 625
Design Code	ASME III

Table 11.2-1 (continued)
WASTE DISPOSAL SYSTEMS DESIGN DATA

Evaporator Bottoms Tank Pump	
Number	1
Type	Centrifugal
Motor horsepower	7.5 hp
Seal type	Double Mechanical
Capacity (each)	165 gpm
Head at rated capacity	60 ft
Design pressure	142.2 psig
Materials	
Casing	SS 316
Shaft	SS 316
Impeller	SS 316
Evaporator Bottoms Tank	
Number	1
Capacity	5000 gal
Design pressure	Atmospheric Plus Contents
Design temperature	275°F
Operating pressure	Full of Water
Operating temperature	180°F
Material	Inconel 625
Design code	ASME III
Liquid Waste Demineralizer Vessels	
Number	3
Capacity	29 ft^3
Design flow	30 gpm
Design pressure	150 psig
Material	SS 304
Design code	ASME VIII
RF Liquid Waste Reverse Osmosis Unit	
Number	1
Capacity	25 gpm
Materials	SS, PVC
Pressure rating	150 psig (low pressure portion) 500 psig (high pressure portion)
Design code	ANSI B31.1
Distillate Demineralizers	
Number	1
Capacity	50 ft^3
Design flow	30 gpm

Table 11.2-1 (continued)
WASTE DISPOSAL SYSTEMS DESIGN DATA

Distillate Demineralizers (continued)	
Design pressure	150 psig
Material	SS 304
Design code	ASME VIII

Table 11.2-2

ESTIMATED WASTE GAS RELEASE WITH RECOMBINER NOT OPERATING

Based on gaseous activity from:

a. Waste gas cycle: 30 days of feed/20 days of decay/10 days of bleed.

b. Two units, 2546 MWt each.

c. 1% failed fuel.

d. 17-gpm total letdown flow rate to gas stripper from both units.

e. 100% of gases and 0.1% of iodines removed in gas stripper.

f. X/Q - 7.5×10^{-6} sec/ m^3 for calculating dose at site boundary.

Nuclide	Equilibrium Coolant Activity ($\mu Ci/cm^3$)	Curies In Tank At End Of Feed Cycle	Curies In Tank After 20 Days Decay	Discharge Rate (Ci/sec)		Curies Discharged In One Cycle	Total Curies Discharged Per Year	Dose At Site Boundary (rem/yr)
				Initial	Final			
Kr-85m	1.22	3.63×10^{1}	2.05×10^{-35}	2.14×10^{-41}	1.61×10^{-59}	4.44×10^{-37}	5.40×10^{-36}	6.4×10^{-42}
Kr-85	3.25×10^{-1}	1.25×10^{3}	1.25×10^{3}	1.30×10^{-3}	1.30×10^{-3}	1.12×10^{3}	1.36×10^{4}	5.4×10^{-3}
Kr-87	8.27×10^{-1}	7.96	0	0	0	0	0	0
Kr-88	2.36	4.92×10^{1}	0	0	0	0	0	0
Xe-131m	4.62×10^{-2}	1.95×10^{2}	6.16×10^{1}	6.43×10^{-5}	3.61×10^{-5}	4.22×10^{1}	5.13×10^{2}	1.6×10^{-4}
Xe-133m	1.33	5.41×10^{2}	1.30	1.36×10^{-6}	6.66×10^{-8}	3.70×10^{-1}	4.50	1.8×10^{-6}
Xe-133	80.9	7.52×10^{4}	5.47×10^{3}	5.71×10^{-3}	1.54×10^{-3}	2.75×10^{-3}	3.35×10^{4}	1.3×10^{-2}
Xe-135m	7.13×10^{-1}	1.40	5.70×10^{-25}	5.95×10^{-31}	1.02×10^{-41}	2.07×10^{-26}	2.52×10^{-25}	3.0×10^{-31}
Xe-135	3.39	2.27×10^{2}	2.80×10^{-14}	2.92×10^{-20}	3.24×10^{-28}	1.38×10^{-15}	1.68×10^{-14}	1.9×10^{-20}
Xe-138	4.50×10^{-1}	9.59×10^{-1}	0	0	0	0	0	0
1-131	1.72	2.29×10^{-1}	4.11×10^{-2}	4.28×10^{-8}	1.81×10^{-8}	2.48×10^{-2}	3.01×10^{-1}	1.2×10^{-3}
1-132	5.85×10^{-1}	1.00×10^{-3}	0	0	0	0	0	0
1-133	2.65	4.15×10^{-2}	5.44×10^{-9}	5.68×10^{-15}	2.06×10^{-18}	6.19×10^{-10}	7.53×10^{-9}	7.9×10^{-12}
1-134	3.61×10^{-1}	2.35×10^{-4}	0	0	0	0	0	0
1-135	1.40	7.01×10^{-3}	2.03×10^{-24}	2.12×10^{-30}	3.60×10^{-41}	7.38×10^{-26}	8.98×10^{-25}	2.9×10^{-25}
Total		7.76×10^{4}	6.78×10^{3}	7.07×10^{-3}	2.87×10^{-3}	3.91×10^{3}	4.76×10^{4}	1.9×10^{-2}

Table 11.2-3
ESTIMATED WASTE GAS RELEASE WITH RECOMBINER OPERATING

Based on gaseous activity from:

a. Waste gas cycle: 300 days of feed/20 days of decay/10 days of bleed.

b. Two units, 2546 MWt each.

c. 1% failed fuel.

d. 17-gpm total letdown flow rate to gas stripper from both units.

e. 100% of gases and 0.1% of iodines removed in gas stripper.

f. X/Q - 7.5×10^{-6} sec/ m^3 for calculating dose at site boundary.

Nuclide	Equilibrium Coolant Activity ($\mu Ci/cm^3$)	Curies In Tank At End Of Feed Cycle	Curies In Tank After 20 Days Decay	Discharge Rate (Ci/sec)		Curies Discharged In One Cycle	Total Curies Discharged Per Year	Dose At Site Boundary (rem/yr)
				Initial	Final			
Kr-85m	1.22	3.63×10^{1}	2.05×10^{-35}	2.14×10^{-4}	1.61×10^{-59}	4.44×10^{-37}	4.44×10^{-37}	5.3×10^{-43}
Kr-85	3.25×10^{-1}	1.63×10^{4}	1.63×10^{4}	1.70×10^{-3}	1.69×10^{-2}	1.46×10^{4}	1.46×10^{4}	5.8×10^{-3}
Kr-87	8.27×10^{-1}	7.96	0	0	0	0	0	0
Kr-88	2.36	4.92×10^{1}	0	0	0	0	0	0
Xe-131m	4.62×10^{-2}	1.95×10^{2}	6.16×10^{1}	6.43×10^{-5}	3.61×10^{-5}	4.22×10^{1}	4.22×10^{1}	1.3×10^{-5}
Xe-133m	1.33	5.42×10^{2}	1.30	1.36×10^{-6}	6.67×10^{-8}	3.71×10^{-1}	3.71×10^{-1}	1.5×10^{-7}
Xe-133	80.9	7.68×10^{4}	5.58×10^{3}	5.82×10^{-3}	1.57×10^{-3}	2.80×10^{-3}	2.80×10^{3}	1.1×10^{-3}
Xe-135m	7.13×10^{-1}	1.40	5.75×10^{-25}	6.00×10^{-31}	1.02×10^{-41}	2.09×10^{-26}	2.09×10^{-26}	2.5×10^{-32}
Xe-135	3.39	2.32×10^{2}	8.52×10^{-14}	2.98×10^{-20}	3.30×10^{-28}	1.40×10^{-15}	1.40×10^{-15}	1.7×10^{-21}
Xe-138	4.50×10^{-1}	9.59×10^{-1}	0	0	0	0	0	0
1-131	1.72	2.48×10^{-1}	4.44×10^{-1}	4.64×10^{-8}	1.96×10^{-8}	2.69×10^{-2}	2.69×10^{-2}	1.0×10^{-4}
1-132	5.85×10^{-1}	1.00×10^{-3}	0	0	0	0	0	0
1-133	2.65	4.23×10^{-2}	5.50×10^{-9}	5.79×10^{-15}	2.10×10^{-18}	6.31×10^{-10}	6.31×10^{-10}	6.6×10^{-13}
1-134	3.61×10^{-1}	2.35×10^{-4}	0	0	0	0	0	0
1-135	1.40	7.07×10^{-3}	2.05×10^{-24}	2.14×10^{-30}	3.63×10^{-41}	7.44×10^{-26}	7.44×10^{-26}	2.4×10^{-29}
Total		9.41×10^{4}	2.19×10^{4}	2.28×10^{-2}	1.85×10^{-2}	1.75×10^{4}	1.75×10^{4}	7.0×10^{-3}

Table 11.2-4
LIQUID WASTE DISPOSAL SYSTEM

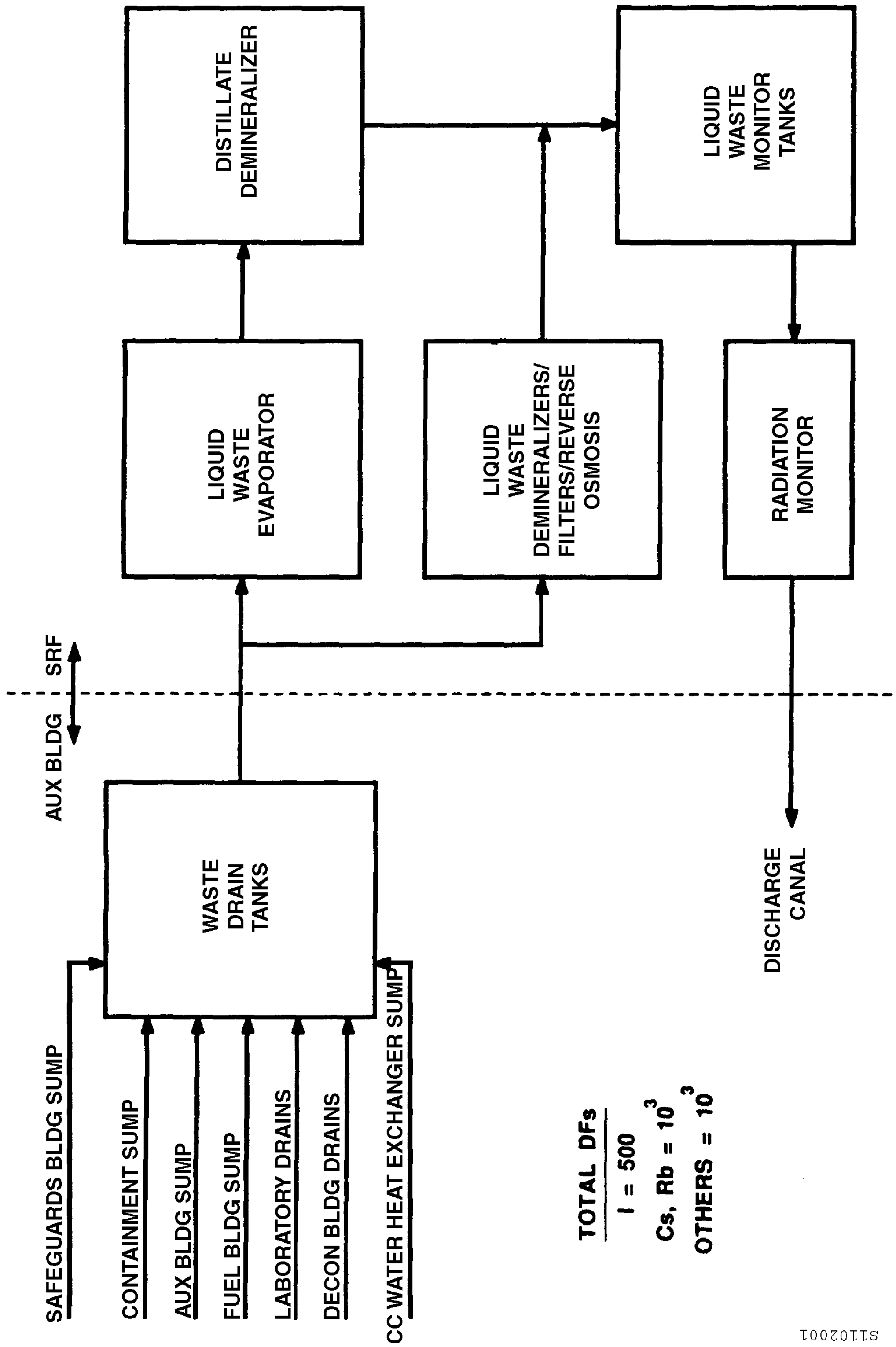
	Shim Bleed	Equipment Drains	Clean and Dirty Wastes	Steam Generator Blowdown	Detergent Wastes
Sources	Reactor coolant letdown	Primary drain tanks	Waste drain tanks		Laundry wastes
Flow rate, gpd	2500	1150	10,150	90,600	1250
Activity, FPCA	1.0	1.0	0.36		
Collection tank volume, gal	120,000	120,000	5804		1230
Collection rate, gpd[a]	7,300[b]	7,300[b]	20,300		2500
Collection time days	13.2	13.2	0.23	0	0.39
Processing rate, gpd	31,700	31,700	34,560		72,000
Processing time, days	3.03	3.03	0.14	0	0.01
Discharge tank volume, gal	30,000	30,000	3548		1, 230
Discharge rate, gpd	7, 300	7, 300	72,000		72,000
Discharge time, days	3.28	3.28	0.04	0	0.01
Fraction of processed steam released	1.0	1.0	1.0	1.0	1.0

		Anion Ion Exch.	Evap.	Same as shim bleed	Mixed-Bed Demin	Mixed-Bed Demin. Case 1	Case 2	
DFs	I	10^2	10^2		10^2(10)	Not treated	10^2(10)	None
	Cs, Rb	1	10^3		2 (10)		10 (10)	
	Others	1	10^3		10^2 (10)		10^2 (10)	
Regenerant time, days		Not regenerated		Not regenerated	Not regenerated	Not regenerated		NA
Source terms		See Tables 11A-9 and 11A-10		See Tables 11A-9 and 11A-10	See Tables 11A-9 and 11A-10	See Tables 11A-9 and 11A-10		See Tables 11A-9 and 11A-10

a. Reflects shared system.

b. Sum of equipment drains and shim bleed for both units.

Figure 11.2-1
LIQUID WASTE DISPOSAL SYSTEM
PROCESS FLOW DIAGRAM, SHEET 1



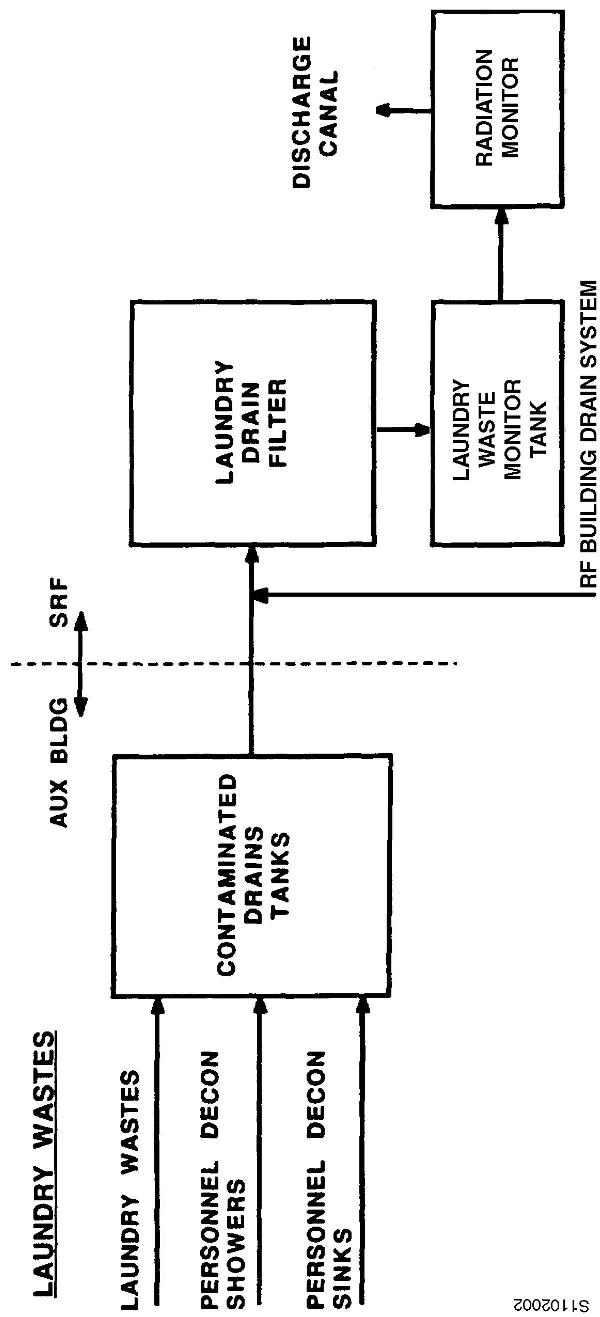


Figure 11.2-2
LIQUID WASTE DISPOSAL SYSTEM; PROCESS FLOW DIAGRAM, SHEET 2

Figure 11.2-3
LIQUID WASTE SYSTEM PROCESS FLOW DIAGRAM

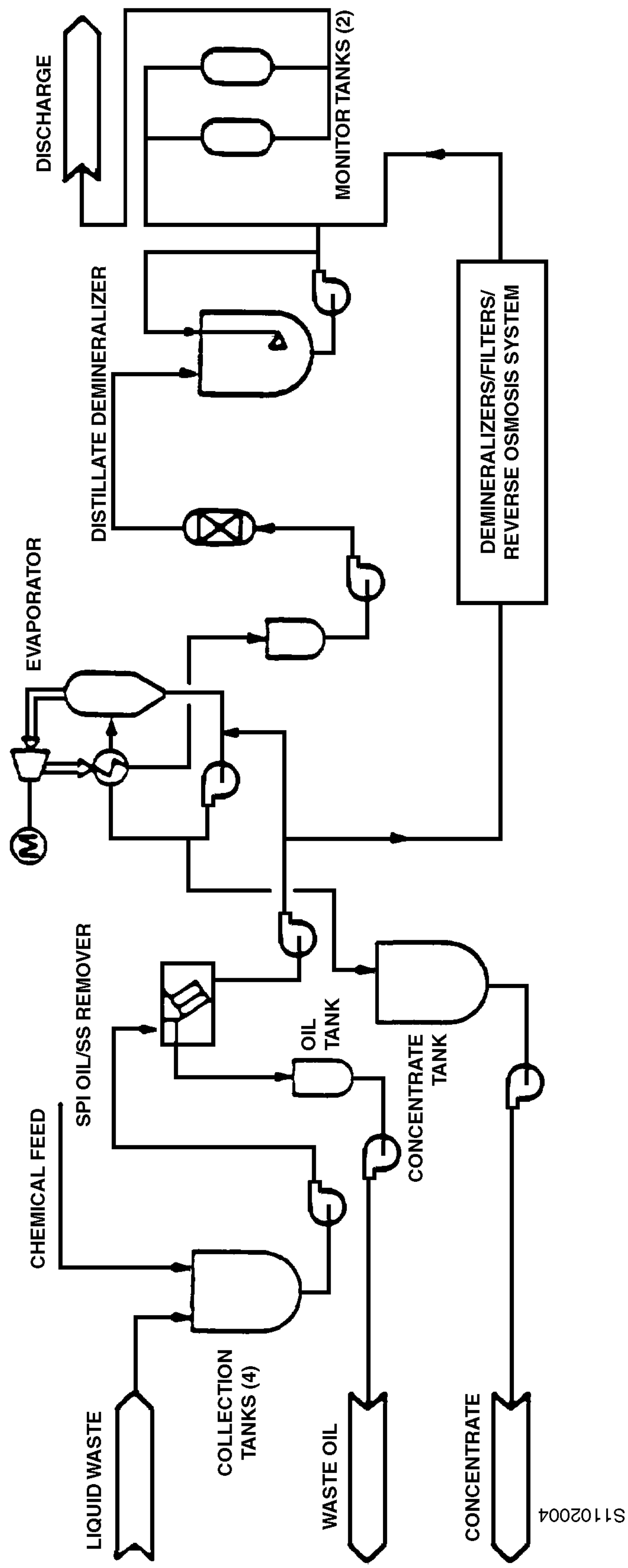


Figure 11.2-4
LAUNDRY WASTE SYSTEM PROCESS FLOW DIAGRAM

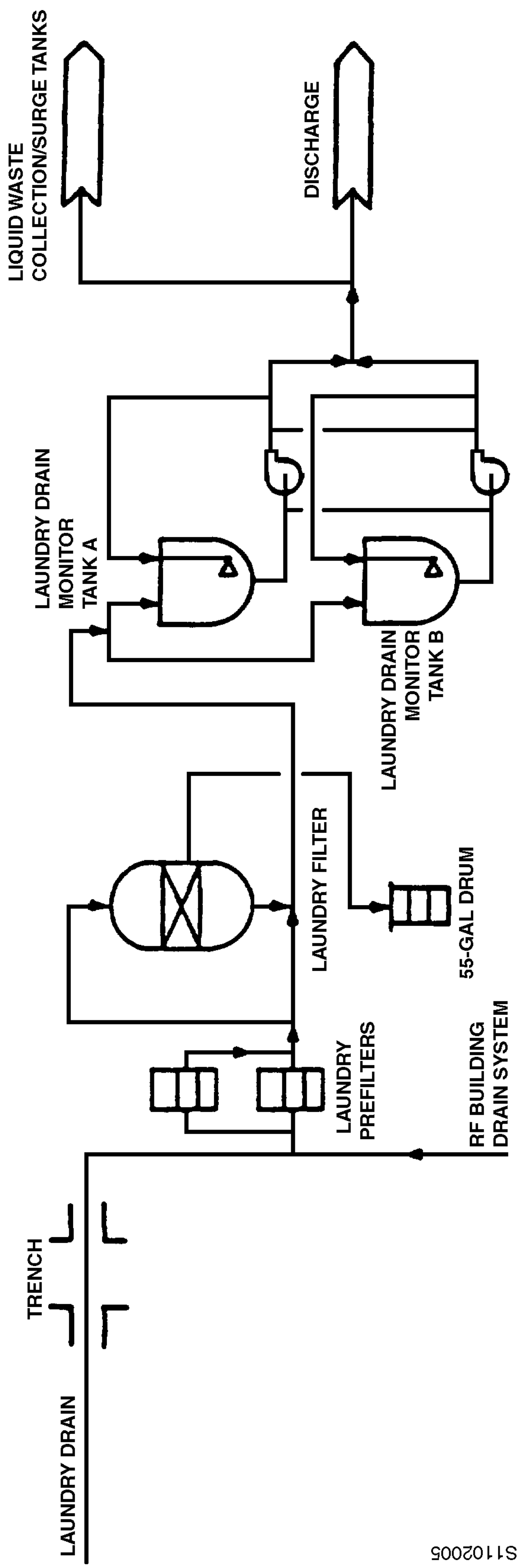
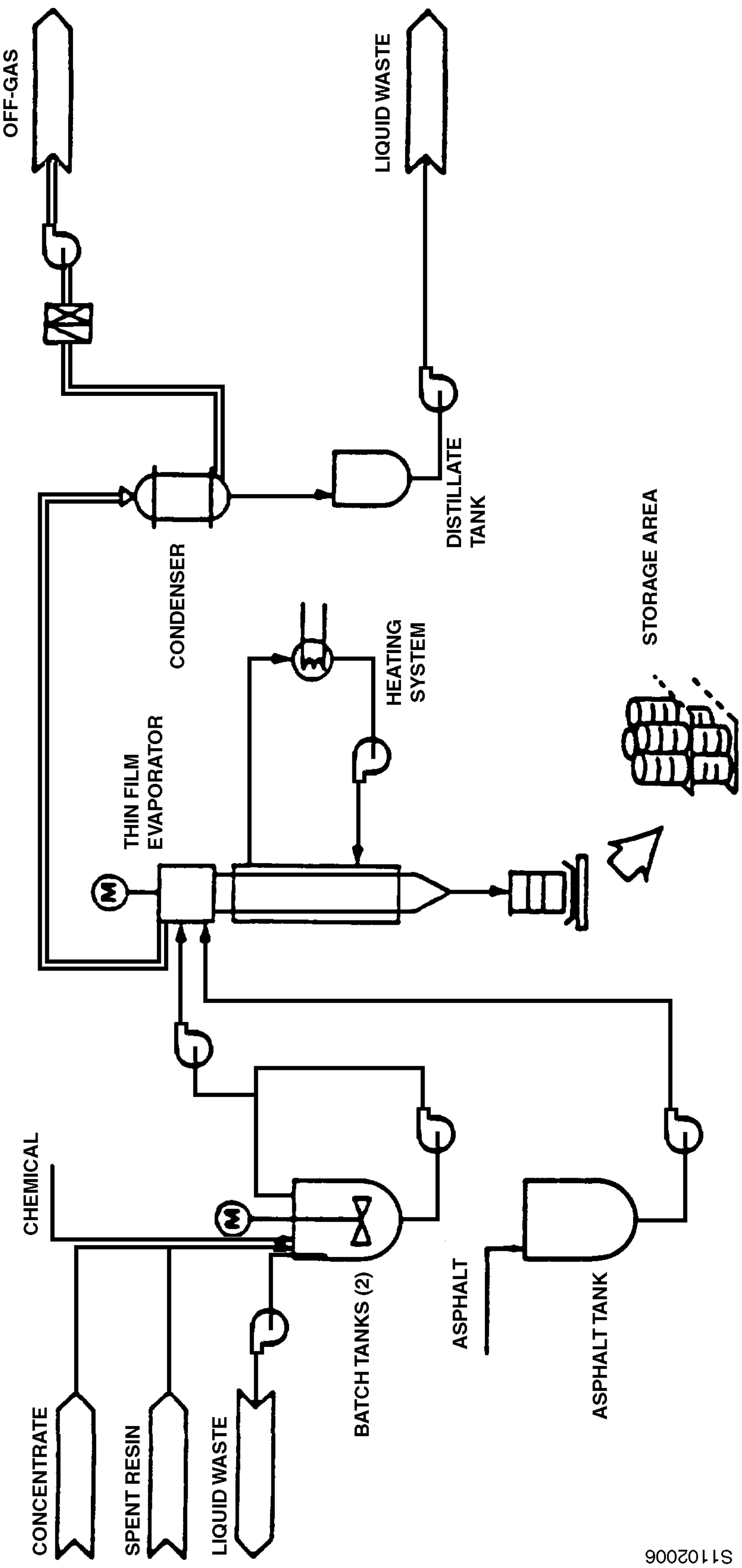


Figure 11.2-5
BITUMEN SOLIDIFICATION SYSTEM PROCESS FLOW DIAGRAM



Intentionally Blank

11.3 RADIATION PROTECTION

11.3.1 Design Bases

Radiation protection, including radiation shielding, was designed to ensure that the criteria specified in 10 CFR 20[1] are met during normal operation and that the guidelines in 10 CFR 50.67 are met in the event of the design-basis loss of coolant accident (Section 14.5.5). Maximum dose limits for design-basis accident conditions are given in Table 14.5-11.

Allowable dose rates are based on the expected frequency and duration of occupancy. Occupancy time and dose rates are such that no personnel shall receive in excess of those doses recommended in 10 CFR 20. All dose rate calculations are based on 1% failed fuel elements. Allowable dose rates for typical locations are given in Table 11.3-1.

11.3.1.1 Leak Reduction Program

A leak reduction program has been established to minimize leakage from systems outside containment that would or could contain highly radioactive fluids during transients or accidents. The following systems, which are described in detail elsewhere in the FSAR, are at least partially located outside containment and are expected to contain potentially radioactive fluids immediately following an accident:

1. Safety injection system.
2. Containment and recirculation spray systems.
3. CVCS (those portions associated with safety injection).
4. Boron recovery system.
5. Resin waste disposal system.
6. Sampling system.
7. Containment vacuum system.
8. Containment purge system.

Several plant systems have been excluded from the leak reduction program. Their exclusion is justified because their unavailability would not eliminate any of the options for cooling the reactor core, nor would it prevent the use of any safety system. The excluded systems are the following:

1. CVCS (those portions not associated with safety injection).
2. Purification system.

---

1. Virginia Power implemented the revised 10 CFR 20 on January 1, 1994. However, as allowed by the NRC, the calculational methodology used for the design analyses is based on the revision of 10 CFR 20 to which the plant was originally licensed.

3. Gas stripper system.
4. Vent and drain systems.
5. Liquid waste disposal system.
6. Reactor cavity purification system.
7. Spent-fuel pool cooling and purification systems.

The letdown portion of the CVCS is normally used for reactor coolant system inventory control, reactor coolant pump seal injection, and reactor coolant system purification. After an accident, the safety injection system using the refueling water storage tank or containment sump would provide the necessary inventory control and seal injection functions. Coolant purification would be deferred until some time after the accident and would probably be performed using a temporary system. Since CVCS letdown will not be used after an accident, the gas stripper system and the vent and drain systems are not needed to support the associated vent. The containment sump would be used for liquid waste storage in place of the liquid waste storage system. The reactor cavity purification system is used during refueling outages. The spent-fuel pool cooling and purification system is not connected to the reactor coolant system or to containment.

A preventive maintenance program, including periodic leak tests, has been established for the systems in the leak reduction program. This program uses administrative controls and procedures as outlined by the plant quality assurance program.

11.3.2 Shielding Design and Evaluation

11.3.2.1 Primary Shielding

Primary shielding is provided to limit radiation emanating from the reactor vessel. The radiation consists of neutrons diffusing from the core, prompt fission gammas, fission product gammas, and gammas resulting from the slowing down and capture of neutrons.

The primary shielding is designed to:

1. Attenuate neutron flux to prevent excessive activation of unit components and structures.
2. Reduce the contribution of radiation from the reactor to obtain a reasonable division of the shielding function between primary and secondary shields.
3. Reduce residual radiation from the core to a level that does not limit access to the region between the primary and secondary shields at a reasonable time after shutdown.
4. Postaccident shielding considerations are discussed in Section 11.3.2.9.

The primary shield consists of a water-filled neutron shield tank with a radial dimension of approximately 3 feet, surrounded by 4.5 feet of reinforced concrete. The neutron shield tank is designed to prevent overheating and dehydration of the concrete primary shield wall and to prevent activation of the plant components within the reactor containment. A thermosiphon cooling system is provided for cooling the water in the shield tank.

A 2-inch-thick cylindrical lead shield that is approximately 15-foot-high is located beneath the neutron shield tank to protect station personnel servicing the neutron detectors during reactor shutdown.

For Unit 2 only, a 3-1/2-inch thick stainless steel radiation shield is provided at the 12-inch diameter incore sump room drain to protect station personnel during normal power operation and refueling outage. The drain is designed to convey the held up water, in excess of the invert elevation of the Incore Sump Room drain, from the Incore Sump Room to the containment sump strainer. This additional water facilitates submergence of the containment sump strainer for RS and LHSI pumps for post-LOCA operation.

To maintain the integrity of the primary shield, streaming shields fabricated from both masonite Benelex 70 and steel are provided in the annular gap between reactor vessel flange and the primary shield concrete. In addition, masonite Benelex 401 and steel streaming shields located outside the primary concrete shield are provided around all of the reactor coolant pipe penetrations.

The primary shield arrangement is shown in Figures 11.3-1 and 11.3-2. The shield materials and thicknesses are listed in Table 11.3-2.

11.3.2.2 **Secondary Shielding**

Secondary shielding consists of reactor coolant loop shielding, reactor containment shielding, fuel handling shielding, auxiliary equipment shielding, and waste storage shielding.

Nitrogen-16 is the major source of radioactivity in the reactor coolant during normal operation and establishes the combined thickness of the crane and containment walls. Activated corrosion and fission products in the reactor coolant system establish the shutdown radiation levels in the reactor coolant loop areas. Tables 9.1-5 and 11.3-3 list the activities that were used in designing the containment secondary shielding. Table 9.1-5 lists the fission product activities in the reactor coolant system with 1% failed fuel. Table 11.3-3 lists the activated corrosion product activities and the N-16 activity at the reactor vessel outlet nozzle.

Activated corrosion and fission products from the reactor coolant system are the radioactive sources for which shielding is required in the auxiliary and waste disposal systems.

Auxiliary steam used for space heating and other purposes throughout the station may become contaminated due to primary-to-secondary leakage. The estimated dose rate at the surface of a 6-inch, 15-psig auxiliary steam supply header is 2×10^{-6} mrem/hr. The estimated dose rate at 3 feet is 1.25×10^{-7} mrem/hr. The dose rate estimates are based on 1% failed fuel activities (Table 9.1-5), 25% steam flow, 1000 cm^3/hr total steam generator leakage, and zero decay time. All noble gases and 1% of the halogens are assumed to leak into the steam generator. Based on these assumptions, the dose received by an individual in the vicinity of the auxiliary steam piping is insignificant.

11.3.2.3 **Reactor Coolant Loop Shielding**

Interior shield walls separate reactor coolant loop, pressurizer, incore instrumentation, and containment access sectors. This shielding allows access to the incore instrument sector during normal operation and facilitates maintenance in all sectors during shutdown. The crane support wall provides limited access protection in the annulus between the crane wall and the reactor containment wall and provides part of the exterior shielding required during power operation. Shield walls are provided around each steam generator above the charging floor to a height required for personnel protection. The shielding arrangement is shown in Figures 11.3-1 and 11.3-2. The shield materials and thicknesses are listed in Table 11.3-2.

11.3.2.4 **Reactor Containment Shielding**

The containment shielding consists of the steel-lined, steel-reinforced concrete cylinder and hemispherical dome, as further described in Chapter 5. This shielding, together with the crane support wall, attenuates radiation during full-power operation at the outside surface of the containment to less than 0.75 mrem/hr. In addition, it attenuates the dose rate from the design-basis accident to design levels.

11.3.2.5 **Fuel Handling Shielding**

Fuel handling shielding is designed to facilitate the removal and transfer of spent fuel assemblies from the reactor vessel to the spent-fuel pool. It is designed to protect personnel against the radiation emitted from the spent fuel and control rod assemblies.

The refueling cavity above the reactor vessel is flooded to Elevation +45-1/3 ft. to provide a temporary water shield above the components being withdrawn from the reactor vessel. The water height is thus approximately 27 feet above the reactor vessel flange. This height ensures that more than 84 inches of water is present above a withdrawn fuel assembly at its highest point of travel. Under these conditions, the dose rate is less than 50 mrem/hr at the water surface.

The fuel is removed from the reactor vessel to the spent-fuel pool by the fuel transfer mechanism via the refueling canal.

The spent-fuel pool in the fuel storage building is permanently flooded to provide more than 84 inches of water above a fuel assembly when it is being withdrawn from the fuel assembly transfer basket. Water height above stored fuel assemblies is at least 24 feet. The sides of the spent-fuel pool, three of which also form part of the fuel storage building exterior walls, are 6-foot-thick concrete to ensure a dose rate of no more than 2.5 mrem/hr outside the building.

Sixteen feet of earth shielding is provided above the fuel transfer tube between the reactor containment and the fuel storage pool wall.

11.3.2.6 **Auxiliary Equipment Shielding**

The auxiliary components exhibit varying degrees of radioactive contamination due to the handling of various fluids. The function of the auxiliary shielding is to protect personnel working near the various auxiliary system components, such as those in the CVCS, the boron recovery system, the waste disposal system, and the sampling system. Controlled access to the auxiliary building is allowed during reactor operation. Each equipment compartment is individually shielded so that compartments may be entered without having to shut down and, possibly, decontaminate the entire system. Ilmenite concrete is used in certain areas where substantial shielding is required and space is at a premium, such as the primary drain tank compartment and the mixed-bed demineralizer compartments.

Ion exchangers and the most highly contaminated filters are located in the ion-exchange structure along the north wall of the auxiliary building. Each ion exchanger or filter is enclosed in a separate, shielded compartment. The concrete thicknesses provided around the shielded compartments are sufficient to reduce the surrounding area dose rate to less than 2.5 mrem/hr, and the dose rate of any adjacent cubicle to less than 100 mrem/hr. The shielding thicknesses around the mixed-bed demineralizers are based on a saturation activity that gives a contact radiation level of nearly 11,000 rem/hr.

Ion exchangers and potentially contaminated filters are also associated with the steam generator blowdown treatment system (no longer used) located in the condensate polisher building.

In many areas, tornado missile protection in the form of thick concrete affords more shielding than that required for radiation protection.

11.3.2.7 **Waste Storage Shielding**

The waste storage and processing facilities in the auxiliary building and decontamination building, the RF, and the waste storage tanks are shielded to provide protection of operating personnel in accordance with the radiation protection design bases set forth in Section 11.3.1.

Periodic surveys by health physics personnel using portable radiation detectors ensure that radiation levels outside the shield walls meet design specifications, and establish access limitations within the shielded cubicles. In addition, continuous surveillance is provided at the Radwaste Facility drumming area and control area and the RF compactor area by area radiation monitors (Section 11.3.4).

Area and process monitoring also ensure that any accidental radioactivity release would be detected within a reasonable period of time. The largest accidental radioactivity release from the waste disposal system would be the rupture of one of the waste gas decay tanks. An analysis of this accident is provided in Section 14.4.2. Furthermore, periodic samples of the gas in the waste gas decay tanks are analyzed by health physics personnel to ensure that the activity level in these tanks is never above the design level used in the accident analysis.

11.3.2.8 **Accident Shielding**

Accident shielding is provided by the reactor containment, which is a reinforced concrete structure lined with steel. For structural reasons, the thicknesses of the cylindrical walls and dome are 54 and 30 inches, respectively. These thicknesses are more than adequate to meet the requirements of 10 CFR 100 at the exclusion boundary.

Additional shielding is provided for the control room. This, together with the shielding afforded by its physical separation from the containment structure, ensures that an operator would be able to remain in the control room for 30 days after an accident and not receive an integrated whole-body dose in excess of 5 rem. The calculational methods and radiation sources used in designing the control room shielding are discussed in Section 11.3.6.

In addition, the control room will serve as a fallout shelter with a protection factor of better than 500, as defined by the Office of Civil Defense.

11.3.2.9 **Postaccident Shielding Review**

A postaccident shielding review was conducted during the 1979 to 1980 time period using the Stone & Webster GAMTRAN1 computer code with inputs developed from the ACTIVITY-2 and RADIOISOTOPE computer codes. NRC-specified source terms were used. The review assumed that the postaccident period was divided into two phases: the mitigation phase and the recovery phase.

The mitigation phase, which was assumed to last 6 months, experiences radiation levels resulting from the operation of the recirculation portion of the safety injection and recirculation spray systems and the postaccident sampling system. This phase also experiences radiation levels from the auxiliary building sump and from the drain lines from the discharge of the auxiliary building and safeguard building sump pumps to the low-level liquid waste tank, as well as to the containment.

The recovery phase has been identified as the period beginning 6 months after the accident, when cleanup and plant recovery is performed. The recovery phase is a controlled evolution that will be planned and carried out to meet the specific recovery requirements of the particular accident.

All essential system piping and equipment that are required to mitigate the effects of a loss-of-coolant accident (LOCA), and that contain or could contain highly radioactive fluids, were considered as sources in the shielding review. These systems included the high-head safety injection system, the low-head safety injection system, the recirculation spray system, the sample system, and the containment atmosphere cleanup (hydrogen recombiner) system. In addition, other systems that are not required to mitigate a LOCA, but that could contain significant radioactivity, were considered, such as drain lines and standing water in sumps and waste tanks. All branch connections to and from these systems were considered as sources to the first isolation valve. Other sources, such as the shine from the containment dome, shine through containment penetrations, and shine through the personnel hatch, were considered. The location of field run pipe, which is part of the systems listed above, was considered in this analysis. The routing and location of radioactive piping is such that the piping is in shielded areas. The exact routing of field run pipe is not critical in the production of radiation zone maps. The highest activity level in each zone is calculated, and that level is considered for the entire zone. For instance, the highest activity may be 12 inches from a pipe, regardless of its exact location within the zone.

Indirect radiation was not considered as a source. Buildup factors in shield walls are considered, but scatter over walls or through labyrinth doorways was not considered. Airborne activity was not considered as a source in the shielding review.

All vital areas were also considered during the review. Vital areas for personnel exposure are defined as those areas that require continuous or frequent occupancy in order to control, monitor, and evaluate the accident. These areas include the control room, technical support center, the counting lab/health physics area, the operational support center, and the security control center. In addition, any area to which access is required to perform manual operation of equipment in systems that are used to mitigate the accident was considered. Vital areas for equipment qualification include all areas in which mitigating equipment is located. Nonvital areas include the entire auxiliary building, main steam valve house, quench spray pump house, safeguards building, service building, and selected areas in the yard.

11.3.2.9.1 Mitigation Phase

The integrated radiation dose calculated for this phase is comprised of the original license period of 40-year normal dose and a 6-month mitigation phase dose. The safety equipment required to operate during the mitigation phase is the same as that equipment tabulated in Chapter 7 for NRC I&E Bulletin 79-01 (Reference 1). The source term developed to calculate the 40-year normal operating dose is based on the assumptions in Chapter 11. The source terms assumed to calculate the 6-month mitigation phase dose are based on TID-14844 (Reference 2) and Regulatory Guide 1.4 (Reference 3), and are listed in Table 11.3-4. The impact of increased integrated dose associated with an additional 20 years of normal operation is small and is accounted for in the environmental qualification of equipment applied to the mitigation phase.

The exact course of an accident is unpredictable. It is impractical to determine the dose rate and shielding requirements for every possible location and time duration associated with each possible failure, or incident, that requires personnel access. Therefore, radiation “zone maps” have been developed for use as postaccident administrative guidelines. These maps show estimated worst-case gamma rates in various areas of the plant as a function of time. The zone maps are used to help plant operators plan access and egress routes, and to help evaluate the relative benefits of delaying certain actions to allow for radioactive decay. The gamma dose rates on the zone maps are based on worst-case source terms, but do not consider an airborne source term. Depending on the severity of the situation, actual dose rates may be smaller. The zone maps are used only as guidelines, and actual radiation levels will be determined through actual postaccident surveys. In addition, the dose rates on the zone maps are based on the highest, or one of the highest, dose rates in that area. The dose rates at other locations within that area may be lower.

11.3.2.9.2 Recovery Phase

The design basis for certain systems and their associated shielding did not consider postaccident recovery operations, that is, postaccident cleanup of highly radioactive fluids. These are the waste disposal system, the boron recovery system, the containment purge system, and the letdown and charging portions of the CVCS. As a result, these systems will not be used for postaccident cleanup operations.

The activity levels (based on Regulatory Guide 1.4 and TID-14844) of the influent to the liquid waste disposal system or to the boron recovery system are approximately 2×10^3 μCi/cm^3 after 6 months of radioactive decay. The area radiation dose rates from concentrated waste and from waste storage tanks would severely limit access to parts of the auxiliary building and would hinder the operation of both units. Since the radioactive waste disposal systems are common to both Units 1 and 2, the use of these systems for the cleanup of waste in the accident-affected unit would preclude the normal use of the radioactive systems for the non-accident-affected unit.

There is extensive piping for the above-listed systems throughout the auxiliary building. The resulting dose rate if all these systems operated simultaneously would severely limit freedom of access for required operations. Shielding for the piping and components would be very difficult, and in some cases impossible, to install, because of the physical arrangement of the piping and components.

11.3.2.9.3 Postaccident Sampling Capability

The Surry Power Station has the capability to sample the reactor coolant, containment sump, and the containment atmosphere. The reactor coolant sample can be taken and analyzed within approximately 3 hours of the decision to sample. A containment atmosphere sample can also be taken with the high radiation sample system. Provisions are included for personnel exposure control. A detailed discussion of the postaccident sampling system is contained in Chapter 9.

11.3.3 Process Radiation Monitoring System

The process radiation monitoring system continuously monitors selected lines containing, or possibly containing, radioactive effluents. Lines through which waste liquids and gases are discharged to the environment are also monitored. The function of this monitoring system is to warn personnel of increasing radiation levels that could result in a radiation health hazard and to give early warning of a system malfunction. An audible alarm in the control room has been incorporated to indicate the loss of power to the monitor cabinet. The process radiation monitoring system serving both units is comprised of the channels listed in Table 11.3-5.

Each radiation monitoring channel is designed to provide continuous information about the process or effluent stream being monitored. Continuous, as used to describe the operation of the process and effluent radiation monitoring systems, means that a monitor provides the required information at all times with the following exceptions: (1) the system is not required to be in operation because of specified plant conditions per the Technical Specifications, or (2) the system is out of service for testing or maintenance and approved alternate monitoring, sampling, or recording methods are in place.

With the exception of the monitoring channels from the Laundry and Radwaste Facilities, each channel has a readout in the control room, and selected channels, as indicated in Table 11.3-5, have a readout at the detector location. The Laundry Facility monitors all readout locally and the Radwaste Facility monitors readout in the Radwaste Facility control room as well as locally. In addition, each channel has an audible and visual alarm for radiation levels in excess of preset values, as well as a visual alarm for detector malfunction. The output from all channels is recorded on recorders that produce a continuous record of radiation levels and radioactive discharges from the station. Each channel has its own power supply and check source thus making it completely independent of any other channel. Each channel check source is remotely operated from the main control room except the process monitors in the Radwaste Facility and the Laundry Facility. The normal/high range Process Vent and Vent Stack No. 2 (MGPI) effluent monitors do not have a check source. The Radwaste Facility and the Laundry Facility check sources are remotely operated from local control panels. The MGPI equipment is continually self-checking that, should the detector output signal drop below some prescribed value, effectively indicating that there is no background signal, the detector will be considered faulty. This monitoring, in conjunction with continual testing of the electronic signal processing circuits by generation and confirmed receipt of test signals, provides assurance that the circuits remain in good health. The adjustment of alarm setpoints, voltage, power, and other variables is made from the control room for all radiation monitors except for the following: 1) monitors in the Radwaste, and 2) monitors in the Laundry Facilities. Adjustments to the Radwaste and Laundry Facility monitors are performed locally. The normal/high range Process Vent and Vent Stack No. 2 MGPI effluent monitors can be adjusted either at the local display unit (LDU) or at the remote display unit (RDU). The entire system is designed with emphasis on system reliability and availability. Certain channels, as indicated in the following text, actuate control valves on a high-activity alarm signal. In the event of a loss of power to these detectors, the system is designed to provide an alarm of the failed condition. Any control functions associated with a high radiation alarm are also initiated.

The expected concentrations of radionuclides in the process streams monitored by the ventilation vent monitors, component cooling water monitors, component cooling heat exchanger service water monitors, condenser air ejector monitors, steam generator blowdown monitors, and recirculation spray cooler heat exchanger service water monitors are natural background radioactivity. The sensitivity of these detectors ensures that abnormal plant conditions will be detected before they cause a hazard to the operators or to the general public.

The use of a single detector is justified in lines used for normal releases from the plant. The surveillance requirements for each of the liquid effluent monitors and gaseous effluent monitors are given in the Offsite Dose Calculation Manual. The liquid and gas waste tanks are sampled and analyzed before and during discharges. Effluent source terms are discussed in Appendix 11A.

Channels monitoring Unit 1 are supplied from the emergency bus for Unit 1. Channels monitoring Unit 2 are supplied from the emergency bus for Unit 2. Channels monitoring systems or areas common to both units can be supplied from the emergency bus for either Unit 1 or Unit 2.

The type of detector, sensitivity, range, background radiation, and other information for each channel are listed in Table 11.3-5. Counting rates are given in Table 11.3-6. A description of each channel is included in the following text.

11.3.3.1 **Process Vent Particulate Monitor**

This channel continuously withdraws a sample from the process vent and passes the sample through a moving filter paper with a collection efficiency of 99% for particle sizes greater than 1.0μ. The amount of deposited activity is continuously scanned by a silicon diode type. A high-activity alarm automatically initiates the closure of the process vent discharge line valves.

A separate isokinetic sampling nozzle used for Health Physics accountability is provided for each unit, as shown in Reference Drawing 2. The nozzles sample the process vent fluid to ensure that a representative sample is taken. Isokinetic sampling is achieved by locating the nozzles in a straight unobstructed piping run of at least five pipe diameters (30 inches). The sampling systems include a pump and a mass flow meter, and supply 1-cfm flow to a sample filter.

11.3.3.2 **Process Vent Gas Monitor**

This channel takes the continuous process vent sample, after it has passed through the particulate filter paper, and draws it through a sealed system to the process vent gas monitor assembly, which is a fixed lead-shielded sampler containing a silicon diode type detector. The sample activity is measured, and is then returned to the process vent. A high-activity alarm automatically initiates the closure of the process vent discharge line valves. A purge system is integral with the gas monitoring system for flushing the sampler with clean air for purposes of calibration.

11.3.3.3 **Ventilation Vent No. 1 Gas Monitor**

This channel withdraws a sample from ventilation vent no. 1 and passes the sample through a particulate and iodine filter assembly. The sample then enters a gas monitor assembly, which is a fixed lead-shielded sampler enclosing a beta scintillation detector. The sample activity is measured, and is then returned to the ventilation vent. A purge system is integral with the gas monitoring system for flushing the sampler with clean air for purposes of calibration. A multi-probe isokinetic sampling nozzle is provided to obtain a representative sample in the duct.

11.3.3.4 **RF Vent Particulate and Gas Monitors**

These two channels continuously sample the RF ventilation stack particulate and gas. The monitors process a sample (approximately 2 scfm) through a replacable fixed 0.3-micron filter. Noble gases are then monitored in a pressure compensated chamber. Sample flows are supplied by an isokinetic probe. This monitor has provisions for a removable silver zeolite iodine sampler cartridge and for a grab sample for tritium analysis.

11.3.3.5 **Component Cooling Water Monitors**

These two channels continuously monitor the component cooling water by means of a gamma scintillation detector enclosed in lead shielding and mounted on the component cooling piping. The complex piping arrangement of this system dictates that two detectors are required to ensure that the system is properly monitored. Activity is indicative of a leak into the component cooling system (Section 9.4) from one of the radioactive systems that exchange heat to the component cooling system.

11.3.3.6 **Component Cooling Heat Exchanger Service Water Monitor**

These four channels continuously monitor the service water effluent from the four component cooling water heat exchangers. Each channel consists of an inline gamma scintillation detector located in a pipe well on the discharge side of the heat exchanger.

11.3.3.7 **Liquid Waste Disposal System Monitor**

The RF has a liquid waste discharge monitor. This channel continuously monitors processed liquid and laundry waste leaving the RF by means of a gamma scintillation detector mounted on a 3-inch discharge pipe. The unit is shielded to ensure the required detector sensitivity. A high-activity alarm automatically initiates closure of a valve that terminates discharges from the RF.

11.3.3.8 **Condenser Air Ejector Monitors**

There are two identical radiation detection channels (1-SV-RM-111 and 2-SV-RM-211 for Units 1 and 2, respectively) for continuously monitoring the normal gaseous effluent from the condenser air ejectors to the atmosphere. The detectors are gamma scintilators mounted in an in-line sampler surrounded by lead shielding. Activity is indicative of a primary-to-secondary system leak. On a high-activity alarm, the flow is automatically diverted to the containment.

If a condition exists such that the normal radiation monitor alarms, but the containment is under Phase 1 isolation (Section 5.2.2), isolation valves 1-SV-TV-102, 102A, and 103 (Unit 1) and 2-SV-TV-202, 202A, and 203 (Unit 2) would shut (Reference Drawing 1) to stop all flow from the air ejector. However, the operator can maintain condenser vacuum by manually establishing air ejector flow through a discharge line to a point upstream of the ventilation vent no. 2 high-range noble gas effluent radiation monitor (Section 11.3.3.14). This monitor provides both normal and high-range effluent radiation detection. If ultimately required, the air ejector effluent can be isolated remotely from the control room.

11.3.3.9 Steam Generator Blowdown Sample Monitors

Each of these channels (two channels per unit) monitors the liquid phase of the steam generators for radioactivity indicative of a primary-to-secondary system leak. The three steam generator blowdowns are combined and continuously monitored by the detectors. Upon indication of radioactivity, a valving arrangement enables the steam generators to be individually sampled, in turn, to determine the source of the activity. Once it has been established which steam generator is leaking, one of the detectors monitors only the blowdown from that steam generator, while the other detector monitors the combined blowdown from the other two steam generators. The detectors are gamma scintillation detectors, mounted in liquid samplers surrounded by lead shielding.

11.3.3.10 Recirculation Spray Cooler Service Water Outlet Monitors

The recirculation spray coolers, as part of the recirculation spray system (Section 6.3.1), operate only when containment pressure increases to the Hi Hi Consequence Limiting Safeguards (CLS) setpoint.

There are four recirculation spray coolers per unit, and each service water outlet line from the coolers is monitored, thus giving a total of eight channels. Each of these channels is identical. If the recirculation spray system is placed in service, a 5-gpm to 10-gpm sample is drawn out of each service water outlet line by a small pump with a 2-hp motor and passed through an offline liquid sampler, where it is monitored for activity indicative of a leak in the respective recirculation spray cooler. After passing through the liquid sampler, which is located outside the containment, the sample is returned to the service water line. Each monitor consists of a gamma scintillation detector mounted in a standard offline sampler surrounded by 3 inches of lead.

To ensure low background radiation in the event of an accident, the Unit 1 monitors are located in the Unit 2 safeguards building and the Unit 2 monitors are located in the Unit 1 safeguards building.

11.3.3.11 Reactor Coolant Letdown Gross Activity Monitors

Each of the units has its reactor coolant continuously monitored by means of a sample taken from the letdown line to the CVCS (Section 9.1). In this system, large variations in activity level are possible in the event of fuel assembly failure. This is a two-stage monitoring system consisting of a low-range channel and a high-range channel. There is one such system for each unit. After being withdrawn from the letdown line, the sample is passed through a delay line to allow N-16 to decay, then enters a sampler consisting of two gamma scintillation detectors surrounded by lead shielding, and is finally discharged to the volume control tank. Both detectors sit on a 1/2-inch removable stainless steel tube, providing flow through the sampler. Shielded lead plugs are used to convert the two detectors into either high- or low-range letdown monitors. Normally, the low-range detector will be sitting on the 1/2-inch tubing and the high-range detector will be sitting on the shielded lead plug. In the event of a fuel element failure, the activity released could be sufficient to raise the coolant activity level above 1.0 $\mu Ci/cm^3$ gross fission products. This causes the high-range monitor to begin to indicate activity level at 10^{-1} $\mu Ci/cm^3$, providing a one-decade overlap. At this point, the high-range channel provides the activity data, and the low-range monitor can be converted into a high-range monitor by inserting a shielded lead plug.

11.3.3.12 **Circulating Water Discharge Tunnel Monitors**

Each of these identical channels (one per unit) monitors the effluent (service water, condenser circulating water, and liquid waste) in the circulating water discharge tunnel beyond the last point of possible radioactive material addition. A gamma scintillation detector slides into a capped pipe, which is then inserted directly into the discharge tunnel and acts as a well. At the top of the pipe is a waterproof support assembly that encloses a check source. The entire device is waterproof.

11.3.3.13 **Ventilation Vent No. 2 Particulate and Gas Monitors**

The two channels in Ventilation Stack No. 2 continuously sample for particulate and gas in the same way that the two Process Vent channels monitor the process vent sample, except that multi-probe samplers are provided to obtain a representative sample in the duct and both channels are equipped with silicon diode type detectors. In addition the post-accident noble gas and particulate monitor is used to obtain grab samples of the ventilation flow stream. Finally, the operability of the ventilation vent No. 2 particulate and gas monitors is relied upon in conjunction with the fuel pit bridge area monitor and communications to provide a timely and valid indication of a fuel handling accident in the spent fuel pool.

11.3.3.14 **High-Range Postaccident Radiation Monitors**

Methods for monitoring high-level releases of noble gases, iodine, and particulates have been developed and implemented. All potential releases are monitored by instrumenting ventilation vent no. 2, the process vent stack, main steam safety valve and power operated relief valve header, and the auxiliary feedwater pump turbine exhaust. The waste gas decay tank and hydrogen purge exhaust are discharged through the process vent stack. The auxiliary building, decontamination building, fuel building, and safeguards area exhausts are discharged through the auxiliary building ventilation vent no. 2. The containment purge system, which is common to both units, discharges through the ventilation vent no. 2 during outages. The main condenser air ejectors normally discharge to the atmosphere, but flow is diverted to containment if the set radiation level limit is exceeded.

The high-range noble gas radiation monitors on the ventilation and process vents are listed in Table 11.3-8. These monitors have a range of 10^{-7} to 10^{5} $\mu Ci/cm^3$ (Xe-133) under normal background conditions (less than 1 mR/hr). Due to shielding around the detector, the reduction of effluent detector sensitivity under maximum background conditions will not exceed the normal effluent instrument range. A multidetector system with detectors enclosed in lead shielding and sufficient range overlap is provided to ensure complete coverage for all expected background conditions. Shielded effluent detectors are needed to obtain the required sensitivities.

Accident particulate and iodine releases are determined by retrieving fixed filters for laboratory analysis. The filters are shielded to provide personnel protection during removal and reinstallation. Several filters in parallel provide for continuous sampling during filter removal.

High-range monitors, with a usable range of 0.01 mrem/hr to 10,000 R/hr, are installed on all main steam lines and the exhaust of the turbine driven AFW pump. These monitors are listed in Table 11.3-8. The 10,000 R/hr maximum reading corresponds to a noble gas concentration which is significantly greater than the maximum anticipated value, as determined by analysis, for noble gas concentrations following any design basis accident. The main steam monitors will be used in conjunction with secondary system sampling and offsite radiation monitoring.

The control units for the monitors contain the electronics necessary to interpret and display detector readings. A total of eight control units, one for each monitor, are located in the emergency switchgear room. The units provide visual alarms for failure, alert, and high radiation. Controls for a check source and a digital readout of radiation level are also provided.

The detectors are powered from the control units, which are powered from local 120V ac power sources. A backup 120V ac power source is provided.

The Surry Power Station has the capability of monitoring all vital areas through the process vent and ventilation vent stack samples or local grab samples, as appropriate. Accurate monitoring of iodine in the presence of high noble gas concentrations is accomplished by the use of silver zeolite sampling cartridges. The cartridges can be analyzed by the multichannel analyzer system (MCA) that is located in a concrete walled count room for shielding purposes. The MCA detectors are located within lead-lined shields that also contribute to background reduction. This combination of shielding provided in the count room is adequate to provide a low background analysis location under most emergency conditions. If background becomes too high for accurate analysis, samples can be taken to the Radwaste Facility or shipped offsite. Procedures for iodine sampling and analysis are available in the Health Physics office.

Particulate sampling is accomplished in conjunction with radioiodine sampling by using fiber filter patches positioned upstream of the iodine filter. Particulate analysis is accomplished by using a multichannel analyzer.

11.3.3.15 **Storm Drain Radiation Sample System**

The storm drain radiation sample system is used to monitor for radioactive contamination of the effluent from the storm drain system prior to discharge into the James River. Recording flow meters and automatic wastewater samplers are installed at the four final release points of the storm drain system. Access to the equipment is through precast equipment manholes into the buried storm water drainage lines adjacent to the discharge canal. Electrical service to the equipment is provided by 120V ac power outlets in weatherproof enclosures inside the manholes.

Samples are automatically drawn on a periodic basis to be sampled by Health Physics personnel for radioactive material content.

11.3.4 Area Radiation Monitoring System

11.3.4.1 General

The area radiation monitors are designed for continuous operation. Continuous, as used to describe the operation of an area radiation monitor, means that the monitor provides the required information at all times with the following exceptions: (1) the monitor is not required to be in operation because of specified plant conditions per the Technical Specifications, or (2) the system is out of service for testing or maintenance.

The area radiation monitoring system reads out and records the radiation levels in selected areas throughout the station and activates alarms (audible and visible) if these levels exceed a preset value or if the detector malfunctions. With the exception of the containment gas and particulate monitors, each detector reads out and activates alarms, both in the control room and at its station location. Each channel is equipped with a check source remotely operated from the control room. The recorders provide a continuous permanent record of radiation levels while the detectors are functioning. Detectors monitoring Unit 1 are supplied with power from the emergency bus for Unit 1. Detectors monitoring Unit 2 are supplied with power from the emergency bus for Unit 2. Detectors monitoring areas common to both units have the capability of being supplied with power from the emergency bus for either Unit 1 or Unit 2.

Additions subsequent to the original station area radiation monitoring system include the spent resin handling area, laundry, and radwaste facilities. These systems are powered from reliable power supplies and are indicated and source tested locally.

The alarm setpoint of each area monitor is variable and is set at a level slightly above the normal background radiation level in the respective area.

The area radiation monitoring system consists of the detectors listed in Table 11.3-7, plus each unit's containment particulate and gas monitors, described below.

Criticality monitors are not required in the spent fuel and new fuel storage and handling areas. An exemption from the criticality monitoring requirements specified in 10 CFR 70.24(a) was received from the NRC for the storage and handling of fuel assemblies enriched up to 4.3 weight percent U-235 (Reference 8). The exemption was based on station design features and procedural controls that are in place to preclude an inadvertent criticality. Area radiation monitors are provided in these areas which would alert personnel to excessive radiation levels and would initiate appropriate response actions.

11.3.4.2 Containment Particulate Monitors

This channel continuously withdraws a sample from the containment atmosphere into a closed, shielded system exterior to the containment. The sample is passed through a moving filter paper with a collection efficiency of 99% for particle sizes greater than 1.0 micron. The amount of deposited activity is continuously scanned by a lead-shielded beta scintillation detector with a sensitivity of 1×10^{-11} μCi/cm^3 for particulates in a background of 2.5 mR/hr. The sample system, which is common to both the particulate and gas monitors, includes a pump with a 0.75-hp motor, a flow meter, automatic pressure protecting valves, a flow regulating valve, and isolation valves. The pump and motor are located outside the containment. A sample point is available for taking a sample of the containment atmosphere after an incident for spectrum analysis in the laboratory. During refueling a high-activity alarm automatically trips the containment purge air supply fans and closes the purge system butterfly valves, thus isolating the purge system. This automatic function is not credited in the fuel handling accident nor is it required to be functional. The operability of the containment particulate monitor is relied upon in conjunction with the containment gas monitor, the manipulator crane area monitor, and communications to provide a timely and valid indication of a fuel handling accident in the containment. The counting rate of the limiting isotopes, I-131 and Cs-137, is nominally 2.6×10^{12} cpm/μCi/cm^3 (2 scfm and 1"/hour tape speed).

11.3.4.3 **Containment Gas Monitors**

This channel takes the continuous containment atmosphere sample, after it has passed through the particulate filter paper, and draws it through an in-line, easily removable, charcoal cartridge arrangement to the containment gas monitor assembly, which is a fixed-volume, lead-shielded sampler enclosing a beta scintillation detector. The sensitivity of this detector is 1×10^{-6} μCi/cm^3 for noble gases in a background of 2.5 mR/hr. The sample activity is measured, and then the sample is returned to the containment.

During refueling, a high-activity alarm automatically trips the containment purge air supply fans and closes the purge system butterfly valves, thus isolating the purge system. This automatic function is not credited in the fuel handling accident nor is it required to be functional. The operability of the containment gas monitor is relied upon in conjunction with the containment particulate monitor, the manipulator crane area monitor, and communications to provide a timely and valid indication of a fuel handling accident in the containment.

A purge valve arrangement blocks the normal sample flow to permit purging the detector with a clean sample for calibration. Purged gases are discharged to the containment. Protection and isolation are provided as described in Section 11.3.4.2.

The counting rates of the limiting isotopes Xe-133, and Kr-85 are nominally 3.145×10^7, and 1.09×10^8 cpm/μCi/cm^3, respectively.

11.3.4.4 **Other Area Radiation Monitoring Equipment**

This equipment consists of fixed-position, ion-chamber-type gamma detectors and associated electronic equipment. These channels warn personnel of any increase in radiation level at locations where personnel may be expected to remain for extended periods of time. The channels and their ranges are listed in Table 11.3-7.

In addition, if the dose rate at the manipulator crane area monitor exceeds the alarm setpoint during refueling, the alarm automatically trips the containment's purge air supply fans and closes the purge system butterfly valves, thus isolating the purge system. This automatic function is not credited in the fuel handling accident nor is it required to be functional. The operability of the manipulator crane area monitor is relied upon in conjunction with the containment gas and particulate monitors, and communications to provide a timely and valid indication of a fuel handling accident in the containment.

Finally, the operability of the fuel pit bridge area monitor is relied upon in conjunction with the ventilation vent No. 2 particulate and gas monitors and communications to provide a timely and valid indication of a fuel handling accident in the spent fuel pool.

11.3.4.5 **High-Range Postaccident Containment Monitors**

A high-range reactor containment area monitor is located outside the containment structure. The detector is permanently mounted and aimed at the personnel hatch. The monitor has a range of 0.1 to 10^7 mR/hr to measure the expected high gamma dose rate in the containment following a LOCA.

An additional set of two high-range containment radiation monitors is installed at separate locations on the crane wall above the operating deck level inside containment. The monitors are single ion chamber detectors that measure photons over the range of 1 to 10^7 R/hr. The system is sensitive to photon energies from 60 keV to 3 MeV, with ±20% accuracy for 0.1 to 3 MeV photons.

The readout for the in-containment monitors is located in the control room and consists of a rate meter and recorder that starts at a present value. Each redundant monitor is powered by a separate vital instrument bus.

The in-containment monitors meet the Seismic Class 1 requirements of Regulatory Guide 1.100 (Reference 5), will withstand the LOCA conditions specified by Regulatory Guide 1.89 (Reference 6), and have been environmentally qualified in accordance with the requirements of IEEE 323-1974 and Regulatory Guide 1.97, Revision 3.

11.3.4.6 **Technical Support Center Area Monitor**

The Technical Support Center (TSC) radiation monitoring system is a localized system and satisfies the guidelines established in NUREG-0696. The radiation monitoring system components consist of one particulate, iodine, and noble gas monitor, two area monitors and a remote alarm panel.

This monitoring system provides continuous indication of the dose rate and airborne activity in the TSC during an emergency, as well as alerting personnel of adverse conditions. It is totally contained within the TSC and is in no way connected to the control room or any safety-related systems.

11.3.4.7 **Local Emergency Operations Facility Area Monitors**

The Local Emergency Operations Facility (LEOF) radiation monitoring system is a localized system and satisfies the guidelines established in NUREG-0696. The radiation monitoring system components consist of one particulate, iodine, and noble gas monitor; two area monitors; and a remote alarm panel.

This monitoring system provides continuous indication of the dose rate and airborne activity in the LEOF during an emergency, as well as alerting personnel of adverse conditions. It is totally contained within the LEOF and is in no way connected to the control room or any safety-related systems.

11.3.5 Environmental Survey Program

The information in this section gives the programmatic elements of the Radiological Environmental Monitoring Program (REMP) for Surry Power Station. Historical data is provided on the pre-operational radiological surveillance performed in support of the Applicant's Environmental Report. Current requirements of the REMP program contained in Technical Specifications are implemented through the Offsite Dose Calculation Manual (ODCM), which describes the specific elements of the present radiological environmental surveillance program at Surry Power Station.

The Surry Power Station Applicant's Environmental Report contains the preoperational radiological surveillance program covering the period from May 1968 through June 1970.

Comments made by the U.S. Fish and Wildlife Service concerning the preoperational phase of the station were taken into account by:

1. Holding a conference on June 21, 1968, to discuss the pre- and postoperational surveys for the Surry Power Station. Representatives from the following agencies were present:
   a. Federal Water Pollution Control Authority.
   b. Bureau of Commission of Fisheries, Radiobiological Laboratory.
   c. Virginia State Water Control Board.
   d. Commonwealth of Virginia, Bureau of Industrial Hygiene.
   e. Virginia Commission of Game and Inland Fisheries.
   f. U.S. Fish and Wildlife Service, Bureau of Sport Fisheries and Wildlife.
   g. Virginia Institute of Marine Science.

The meeting adjourned with the understanding that all agencies represented at the meeting were completely satisfied with the Company's program for pre- and postoperational radiological and ecological surveillance programs.

2. Conducting the preoperational radiological surveillance program in such a manner that indigenous species that concentrate radionuclides are routinely sampled and analyzed.
3. Preparing reports on the radiological program for distribution to interested agencies before station operation.
4. Holding discussions with appropriate state officials and personnel from the U.S. Fish and Wildlife Service regarding thermal pollution.
5. Installing of seven platforms in the James River upon which are seven instruments continuously measuring water temperature. (Instruments to measure river salinity were also installed on those same towers.)
6. Working in conjunction with personnel from the Virginia Institute of Marine Science regarding their grant from the AEC concerning thermal pollution of the river.
7. Designing the intake water facility with a 1.0-fps intake velocity at the screen surface so as to prevent significant damage to fishery resources.

A postoperational radiation surveillance program was developed using the knowledge and information obtained from the preoperational surveillance program. The latter was in effect over a 2-year period and served to train plant personnel in sampling and analytical techniques; aid in identifying those "indicator samples" that may be an indication of a slow buildup of radioactivity in the environment; establish the degree of variability between measurements resulting from seasonal changes in the weather (since fluctuations do occur and are expected); generate meaningful environmental data based primarily on scientific and technical requirements; and establish a correlation of data between the consulting service and the station's laboratory group.

The ultimate objective of the postoperational surveillance program is the verification of the adequacy of radiation source control. Therefore, analytical efforts are directed toward those samples that have the ability to concentrate the radioelements of concern and afford an integrated and sensitive sampling mechanism. Milk, shellfish (oysters, crabs, and clams), and silt are considered indicator samples and are indicative of radioactivity levels in the environment. Commercial and/or recreationally important species of fish (catfish, eel and perch) are selected for sampling in the vicinity of the discharge point. The consumption of fish present a direct ingestion pathway to man. Because certain species of fish, such as bottom feeders, concentrate radionuclides which may be taken up from the water and aquatic sediments or may accumulate in the fish directly via internal deposition, it is important to include these organisms in the radiological environment sampling program. Samples are collected from the environment surrounding the station at various intervals throughout the year. Radioanalysis of these samples indicates conditions both in time and space, and thus a slow buildup of radioactivity can be determined. In addition, comparison in the trends of radioactivity levels are more meaningful, since the biological variation caused by using many different media has been eliminated.

Since the program's origination, sampling has been expanded to include obtaining representative samples of the sediment, shoreline silt, food products, oysters, fish and clams from the surrounding area. These samples are analyzed for radioisotopes.

11.3.5.1 **Air Monitoring**

To meet the surveillance objective previously stated, it is desirable for the surveillance methods to indicate changes in radioactivity above-background levels. Therefore, continuous-duty air particulate samplers are also used to confirm that the station presents no hazard to the public. An air-sampling station is located at different locations including a control location which provides background data. The establishment of the air particulate network takes into account the following four general considerations:

1. The average meteorological conditions in the vicinity of the site.
2. The current and projected population densities near the station.
3. The proximity of other nuclear facilities.
4. The proximity of the site boundary location of the highest calculated annual average ground level D/Q.

Air particulate matter is accumulated for a 1-week period on appropriate filter media using a low-volume air sampler. The particulate filters are analyzed in accordance with Radiological Environmental Monitoring Program requirements.

11.3.5.2 **Milk**

The dose consequence to an individual is from both a direct and indirect exposure pathway. The direct exposure pathway is from the inhalation of radioactive material and the indirect exposure pathway is from the grass-cow-milk pathway. In this pathway radioactive material is deposited on the plants consumed by the dairy animals. The radioactive material is them passed on to the individual via the milk.

It has been estimated that, since a cow grazes over an area of 160 m^2 per day (Reference 7), cow's milk affords a good integrated sample. Since milk is one of the best and most direct biosamplers for determining the radiocesium, radiostrontium, and radioiodine levels in the environment, samples are collected from the local dairy farms in the vicinity of the station.

11.3.5.3 **Shellfish, Crabs, and Fish**

Shellfish have the ability to concentrate certain stable elements and radionuclides far above the normal concentrations found in their saline water environment. Oysters and clams, *Crasostrea virginica* and *Mercenaria mercenaria* respectively, found in the James River are thus one of the more sensitive mechanisms for the determination of radioactivity released from the station.

Oysters and clams are collected in the vicinity of the station. Crab and fish samples are obtained in the vicinity of the station. Radioanalysis of these sample media are in accordance with the Radiological Environmental Monitoring Program.

Additional aquatic vectors useful as integrating samples have been investigated, but none have proven to be as sensitive or indicative as shellfish.

11.3.5.4 **Silt and Sediment**

Since shellfish do not concentrate all radionuclides, river bottom sediment samples are collected from downstream areas which have or potentially may have recreational value. Silt samples are obtained in the vicinity of the station. Because of the interaction of a number of mechanisms, radionuclides accumulate in silt and bottom sediments. Because of this, silt is one of the few environmental media in which radioactive effluents from nuclear power stations are usually detected. These samples thus afford an integrated indication of average water concentrations.

11.3.5.5 **Water Samples**

Water samples are collected for analysis from the following sources: surface water samples from two different locations—one upstream and one downstream from the station and ground (well) water samples, which are collected from various wells in the vicinity of the station. The well water and surface water samples are analyzed in accordance with Radiological Environmental Monitoring Program requirements.

11.3.5.6 **Food Products**

The dose consequence to an individual from food products is via the ingestion exposure pathway. Samples of food products are obtained from farms in the vicinity of the station and analyzed in accordance with Radiological Environmental Monitoring Program requirements.

11.3.5.7 **Equipment**

Analytical equipment used for the Radiological Environmental Monitoring Program includes the following:

1. Gas-flow proportional counter (alpha-beta counter).
2. Multichannel gamma spectroscopy analyzer.
3. Liquid scintillation counter.
4. Thermoluminescent Dosimeter (TLD) readers.
5. Other analytical equipment which meet the Lower Limit of Detection (LLD) requirements set forth in the Offsite Dose Calculation Manual.

11.3.5.8 **Environmental Dosimetry**

Normally, the gaseous wastes discharged from the station consist almost entirely of the noble gases, xenon and krypton. The radiation hazard from these gases, due to their inertness, is external radiation exposure. Therefore, radiation surveillance can be maintained by using devices to measure total external body radiation levels in the station environs. The TLDs measure external radiation exposure from several sources including naturally occurring radionuclides in the air, radiation from cosmic origin, fallout from atomic weapons testing as well as potential radioactive airborne releases from Surry Power Station. An inner ring of TLDs are located in the five mile range from the site with a station in each of the 16 sectors of each ring. Other TLDs are located in special program interest areas (residences, schools, etc.) and in control locations. The TLDs are collected and analyzed for gamma radiation at a frequency which optimizes statistical sensitivity and characterizes seasonal fluctuations.

11.3.5.9 **Interlaboratory Comparison Program**

Surry Power Station participates in an Interlaboratory Comparison Program to ensure that independent checks on the precision and accuracy of measurements of radioactive material in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring. The program is applicable for the radioanalyses performed in support of the Radiological Environmental Monitoring Program.

11.3.6 Control Room

The control area is described in Section 7.7. The design basis for radiation protection in the control room under accident conditions is that the whole body radiation dose to personnel accessing and occupying the control room is limited to less than or equal to 5 rem TEDE for the duration of the accident. This dose includes the 30-day dose from ECCS leakage (including RWST backleakage) and the contribution from the postulated radioactive plume leaking from the containment (as discussed in Section 14.5.5) until engineered safeguards equipment returns the containment to subatmospheric pressure and terminates the leakage.

The control room doses for the design basis LOCA are discussed in Section 14.5.5.3.

The control room walls, which are a minimum of 24-inch thick for tornado missile protection, provide more than adequate shielding from radiation. Special consideration has been given to the design of penetrations and structural details of the control room so as to establish an acceptable condition of leaktightness.

The air conditioning systems are installed within the spaces served and are designed to provide uninterrupted service under accident conditions. Upon a safety injection signal, the normal replenishment air and exhaust systems are isolated automatically from the control room by tight closures in the ductwork. Breathing-quality compressed air is supplied to the control room area for a period exceeding the containment leakage period. This outflow can be verified by means of a pressure gauge reading the inside and outside pressure difference in inches of water.

The control and relay rooms are also provided with an emergency ventilation system fitted with particulate and impregnated charcoal filters to introduce cleaned outside air into the protected spaces upon depletion of the high-pressure air. This can continue indefinitely to hold the area pressure above atmospheric to ensure outflow leakage.

The radiation level in the control room is measured by a gamma monitor to verify safe operating conditions.

As a secondary precaution, personnel air-packs are available in the control area.

11.3 REFERENCES

1. IE Bulletin 79-01, *Environmental Qualification of Class 1E Equipment*, 1979.
2. J. J. Dinunno et al., *Calculation of Distance Factors for Power and Test Reactors*, TID - 14844, U.S. Atomic Energy Commission, 1962.
3. Regulatory Guide 1.4, *Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors*, 1974.
4. NUREG-0578, *TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations*, 1979.
5. Regulatory Guide 1.100, *Seismic Qualification of Electric Equipment for Nuclear Power Plants*, 1977.
6. Regulatory Guide 1.89, *Qualification of Class 1E Equipment for Nuclear Power Plants*, 1974.
7. G. Wortley, *Contamination of Milk with Radionuclides Dispersed into the Biosphere*, a report submitted to the Joint FAO/WHO Expert Committee on Milk Hygiene, WHO Headquarters, Geneva, Switzerland, 1969.
8. Letter from G. Edison of the USNRC to J.P. O'Hanlon of Virginia Electric and Power Company dated July 15, 1998 (Serial No. 98-427), *Issuance of Revised Exemption from the Requirements of 10 CFR 70.24(a) - Surry Power Station, Units 1 and 2 (TAC Nos. MA0657 and MA0658).*

11.3 REFERENCE DRAWINGS

The list of Station Drawings below is provided for information only. The referenced drawings are not part of the UFSAR. This is not intended to be a complete listing of all Station Drawings referenced from this section of the UFSAR. The contents of Station Drawings are controlled by station procedure.

	Drawing Number	Description
1.	11448-FM-066A	Flow/Valve Operating Numbers Diagram: Auxiliary Steam and Air Removal System, Unit 1
	11548-FM-066A	Flow/Valve Operating Numbers Diagram: Auxiliary Steam and Air Removal System, Unit 2
2.	11448-FM-090B	Flow/Valve Operating Numbers Diagram: Gaseous Waste Disposal System, Unit 1

Table 11.3-1
ALLOWABLE DOSE RATES

Zone Description	Maximum Dose Rate (mrem/hr)	Typical Locations
Full-power operation		
Continuous access (Zone I)	0.75	Control room, outside surface of containment, and all turbine plant and administration areas
Periodic access (Zone II)	2.5	Auxiliary and fuel building passageways in general and inside reactor containment personnel lock
Limited access (Zone III)	15	Outside surface of shielded tank shields
Controlled access (Zone IV)	15	Inside shielded equipment compartments
Access to incore instrumentation	100	Annulus between crane wall and containment wall
Access to incore instrumentation	40	Vicinity of incore instrumentation transfer devices
Access to incore instrumentation	20	Vicinity of incore instrumentation motors
Hot shutdown (after 15-min decay)		
Limited access (Zone III)	15	Reactor containment above charging floor and outside of crane wall
Controlled access (Zone IV)	15	Inside shielded equipment compartments
Cold shutdown for maintenance (after 8-hr decay)		
Periodic access (Zone II)	2.5	Reactor containment above charging floor and outside of crane wall
Controlled access (Zone IV)	15	Inside shielded equipment compartments
Cold shutdown for refueling (after 4-day decay)		
Periodic access (Zone II)	2.5	Reactor containment above charging floor, outside of crane wall, and adjacent to fuel transfer canal near incore instrumentation devices
Controlled access (Zone IV)	15	Inside shielded equipment compartments
Surface of water over raised fuel elements	50	Fuel element above up-ender, and above other fuel elements in fuel building

Table 11.3-2
CONTAINMENT SHIELDING SUMMARY

Symbol	Figure	Shield Description	Material[a]	Thickness (inches)
A	11.3-2	Neutron shield tank	Water	34
			Steel	3
B	11.3-2	Primary shield	Concrete	54
C	11.3-2	Supplementary neutron shield	Benelex 70	14
D	11.3-2	Streaming shield ring	Benelex 401	5
E	11.3-2	Neutron shield tank support	Steel	1-1/2
			Lead	2
F	11.3-1 & 11.3-2	Cubicle - crane support wall	Concrete	33
G	11.3-2	Crane support wall	Concrete	24
H	11.3-1 & 11.3-2	Containment wall	Concrete	54
I	11.3-2	Containment dome	Concrete	30
J	11.3-2	Floor elevation, 3' 6"	Concrete	24
K	11.3-2	Charging floor	Concrete	24
L	11.3-1 & 11.3-2	Refueling cavity wall	Concrete	36
M	11.3-2	Missile shield	Concrete (Unit 1)	24
	11.3-3		Steel (Unit 2)	2
N	11.3-2	Refueling cavity water	Water	-
O	11.3-2	Removable block wall	Concrete (Ilmenite)	12
P	11.3-1	Fuel trans. canal wall	Concrete	54
Q	11.3-1	Fuel trans. canal wall	Concrete	72
R	11.3-1	Fuel trans. tube shielding	Concrete	36
S	11.3-1	Fuel trans. canal wall	Concrete	72
T	11.3-1	Incore inst. cubicle wall	Concrete	42
U	11.3-1	Cubicle wall	Concrete	30
V	11.3-1	Regen. heat exchanger wall	Concrete	24
W	11.3-1	Cable vault wall	Concrete	24
X	11.3-1	Auxiliary feed pump cubicle wall	Concrete	36
Y	11.3-1	Safeguards area wall	Concrete	24
Z	11.3-3	Incore Sump Room Drain (Unit 2 only)	Steel	3.5

a. All concrete is reinforced with steel.

Table 11.3-3
N-16 AND ACTIVATED CORROSION PRODUCT ACTIVITY

Isotope	Activity ($\mu Ci/cm^3$ at 574°F)
Mn-54	5.6×10^{-4}
Mn-56	2.1×10^{-2}
Fe-59	7.5×10^{-4}
Co-58	1.8×10^{-2}
Co-60	5.4×10^{-4}
N-16[a]	64.0

a. At the reactor vessel outlet nozzle at 2546 Mwt.

Table 11.3-4
POSTACCIDENT MITIGATION PHASE SOURCE TERMS

Source	Source Term	Basis
Sump water	0% noble gas	Sump water is degassed
	50% halogens	TID-14844
	1% solid fission products	TID-14844
Primary coolant sample	100% noble gas	R.G. 1.4/TID-14844
	50% halogens	TID-14844
	1% solid fission products	TID-14844
Containment atmosphere	100% noble gas	R.G. 1.4/TID-14844
	25% halogens	R.G. 1.4
	0% solid fission products	R.G. 1.4

Table 11.3-5
PROCESS RADIATION MONITORING SYSTEM [a]

Monitor	Number	Type of Detector	Medium	Limiting Isotopes	Sensitivity (μci/cm^3)	Range (decades)	Maximum Background (mR/hr)
Process Vent Particulate (1-GW-RM-130A)	1	Silicon Diode	Gas, air	1-131	1×10^{-9} (I-131)	3	0.75
Process Vent Gas (1-GW-RM-130B)	1	Silicon Diode	Gas, air	I-131, Xe-133, Kr-85	5×10^{-6} (Kr-85)	4	0.75
Ventilation Vent No. 1 - Gas (1-VG-RM-104)	1	Beta scint.	Air	1-131. Xe-133, Kr-85	5×10^{-6} (Kr-85)	3	0.75
RF Vent Particulate (1-RRM-RE-100)	1	Beta scint.	Gas, air	Cs-137	1×10^{-11} (Cs-137)	5	<1
RF Vent Gas (1-RRM-RE-101)	1	Beta scint.	Gas, air	Xe-133	1×10^{-7} (Xe-133)	5	<1
Component Cooling Water (1-CC-RM-105, 106)	2	Gamma scint.	Water	Co-60, Cs-137	1×10^{-5} (Cs-137)	3	0.75
Comp. Cooling Hx Service Water (1-SW-RM-107A, B, C, D)	4	Gamma scint.	Water	Co-60, Cs-137	1×10^{-5} (Cs-137)	3	0.75
RF Liquid Waste Disposal[b] (1-RRM-RE-131)	1	Gamma scint.	Water	CS-137	1×10^{-7} (Cs-137)	5	<1
Condenser Air Ejector (1/2-SV-RM-111/211)	2	Gamma scint.	Vapor	I-131, Xe-133, Kr-85	1.74×10^{-5} (Kr-85)	5	0.75
Steam Generator Blowdown[b] (1/2-SS-RM-112/212 &113/213)	4	Gamma scint.	Water	Co-60, Cs-137	4×10^{-6} (Cs-137)	3	2.5
Recirculation Spray Cooler (1/2-SW-RM-114/214, 115/215, 116/216, & 117/217)	8	Gamma scint.	Water	Co-60, Cs-137	3×10^{-4} (Cs-137)	2	5.0

a. This table contains information based on the requirements stated in the various specifications for the listed monitors. The actual monitors meet or exceed these minimum requirements.

b. These channels also have local readout and alarm.

Table 11.3-5
PROCESS RADIATION MONITORING SYSTEM [a]

Monitor	Number	Type of Detector	Medium	Limiting Isotopes	Sensitivity (µci/cm^3)	Range (decades)	Maximum Background (mR/hr)
R.C. Letdown High Range[b] (1/2-CH-RM-118/218)	2	Gamma scint.	Water	Co-60, mixed fission products	1×10^{-1} (Co-60)	4	2.5
R.C. Letdown Low Range[b] (1/2-CH-RM-119/219)	2	Gamma scint.	Water	Co-60, mixed fission products	1×10^{-4} (Co-60)	4	2.5
C.W. Discharge Tunnel (1/2-SW-RM-120/220)	2	Gamma scint.	Water	Co-60, Cs-137	2×10^{-7} (Cs-137)	3	0.05
Ventilation Vent No. 2 - Particulate (1-VG-RM-131A)	1	Silicon Diode	Air	I-131	1×10^{-9} (I-131)	3	0.75
Ventilation Vent No. 2 - Gas (1-VG-RM-131B)	1	Silicon Diode	Air	I-131, Xe-133, Kr-85	5×10^{-6} (Kr-85)	3	0.75
Laundry Facility							
Continuous Effluent - Particulate (1-RM-RMS-RIC3)	1	Beta scint.	Air	I-131, Cs-137	1×10^{-10} (I-131)	5	0.5
Continuous Effluent - Iodine (1-RM-RMS-RIC4)	1	Beta scint.	Air	I-131	1×10^{-10} (I-131)	5	0.5

a. This table contains information based on the requirements stated in the various specifications for the listed monitors. The actual monitors meet or exceed these minimum requirements.

b. These channels also have local readout and alarm.

Table 11.3-6
PROCESS RADIATION MONITORING SYSTEM COUNTING RATES OF LIMITING ISOTOPES (cpm/μCi/cm^3)

Monitor	I-131	Xe-133	Kr-85	Cs-137	Co-60
Process Vent Particulate	1.04E+9	-	-	1.15E+9	7.1E+8
Process Vent Gas	2.28E-6	1.7E-6	2.92E-6	-	-
Ventilation Vent No. 1 - Gas	-	2.2×10^7	9.5×10^7	-	-
RF Vent Particulate	-	-	-	5.0×10^{11}	-
RF Vent Gas	-	1.9×10^7	4.1×10^7	-	-
Ventilation Vent No. 2 - Part.	1.04E+9	-	-	1.15E+9	7.1E+8
Ventilation Vent No. 2 - Gas	-	1.7E-6	2.92E-6	-	-
Component Cooling Water	-	-	-	1.0×10^8	2.0×10^8
Component Cooling HX Service Water	-	-	-	2.8×10^8	7.6×10^8
RF Liquid Waste Disposal	-	-	-	3.4×10^8	9.9×10^8
Condenser Air Ejector	1.79×10^9	2.32×10^7	3.22×10^6	-	-
Steam Generator Blowdown	-	-	-	$1.0 \text{ x } 10^8$	2.0×10^8
Recirculating Spray Cooler	-	-	-	1.0×10^8	2.0×10^8
R.C. Letdown High Range	-	-	-	-	9.9×10^2
R.C. Letdown Low Range	-	-	-	-	9.6×10^5
C.W. Discharge Tunnel	-	-	-	4.2×10^8	1.1×10^9
Laundry Facility					
Continuous Effluent Part.	3.6×10^{10}	-	-	3.6×10^{10}	-
Continuous Effluent Iodine	4.1×10^9	-	-	-	-

Table 11.3-7

AREA RADIATION MONITORING LOCATIONS, NUMBER, AND RANGES

Channel Location (Number)	Range (mR/hr)
Containment personnel hatch area (2) 1/2-RM-RMS-161/261	10^{-1}—10^{7}
Manipulator crane (2) 1/2-RM-RMS-162/262	10^{-1}—10^{7}
Reactor containment area (2) 1/2-RM-RMS-163/263	10^{-1}—10^{7}
Incore instrument transfer area (2) 1/2-RM-RMS-164/264	10^{-1}—10^{7}
New fuel storage area (1) 1-RM-RMS-152	10^{-1}—10^{7}
Fuel pit bridge (1) 1-RM-RMS-153	10^{-1}—10^{7}
Auxiliary Building control area (1) 1-RM-RMS-154	10^{-1}—10^{7}
Solid waste drum storage and handling area (1) 1-RM-RMS-155	10^{-1}—10^{7}
Sample room (1) 1-RM-RMS-156	10^{-1}—10^{7}
Main control room (1) 1-RM-RMS-157	10^{-1}—10^{7}
Laboratory (1) 1-RM-RMS-158	10^{-1}—10^{7}
Decontamination area (1) 1-RM-RMS-151	10^{-1}—10^{7}
Spent resin handling area (2) 1-RM-RMS-138, 139	10^{-2}—10^{3}
Laundry Facility (2) 1-RM-RMS-RIC8, RIC9	10^{-2}—10^{3}
Radwaste Facility (10) 1-RRM-RE-121 (RF control room) 1-RRM-RE-122 (RF laboratory) 1-RRM-RE-123 (RF DAW truck area) 1-RRM-RE-124 (RF DAW sorting/compactor area) 1-RRM-RE-125 (RF LSA box storage area) 1-RRM-RE-126 (RF HIC storage and handling Area) 1-RRM-RE-127 (RF hot machine shop truck bay) 1-RRM-RE-128 (RF hot machine shop area) 1-RRM-RE-129 (RF local control panel area) 1-RRM-RE-130 (RF bitumen control room)	10^{-1}—10^{4}
Containment high range radiation monitor system (4) 1/2-RM-RMS-127/227 & 128/228	10^{0}—10^{7} R/hr
Technical support center (2) 1-RM-RMS-136, 137	10^{-1}—10^{4}
LEOF (2) (No mark nos.)	10^{-1}—10^{4}

Table 11.3-8
HIGH-RANGE POST-ACCIDENT RADIATION MONITORS

Normal Range Noble Gas Effluent Monitors

- Process Vent (1-GW-RM-130B)
- Ventilation Vent No. 2 (1-VG-RM-131B)

High Range Noble Gas Effluent Monitors

- Process Vent (1-GW-RM-130C)
- Ventilation Vent No. 2 (1-VG-RM-131C)

High Range Effluent Monitors

Main Steam Lines

- 1-MS-RM-124 (1A)
- 1-MS-RM-125 (1B)
- 1-MS-RM-126 (1C)
- 2-MS-RM-224 (2A)
- 2-MS-RM-225 (2B)
- 2-MS-RM-226 (2C)

Auxiliary Feedwater Turbine Exhaust

- 1-MS-RM-129
- 2-MS-RM-229

Figure 11.3-1
SHIELD ARRANGEMENT, PLAN

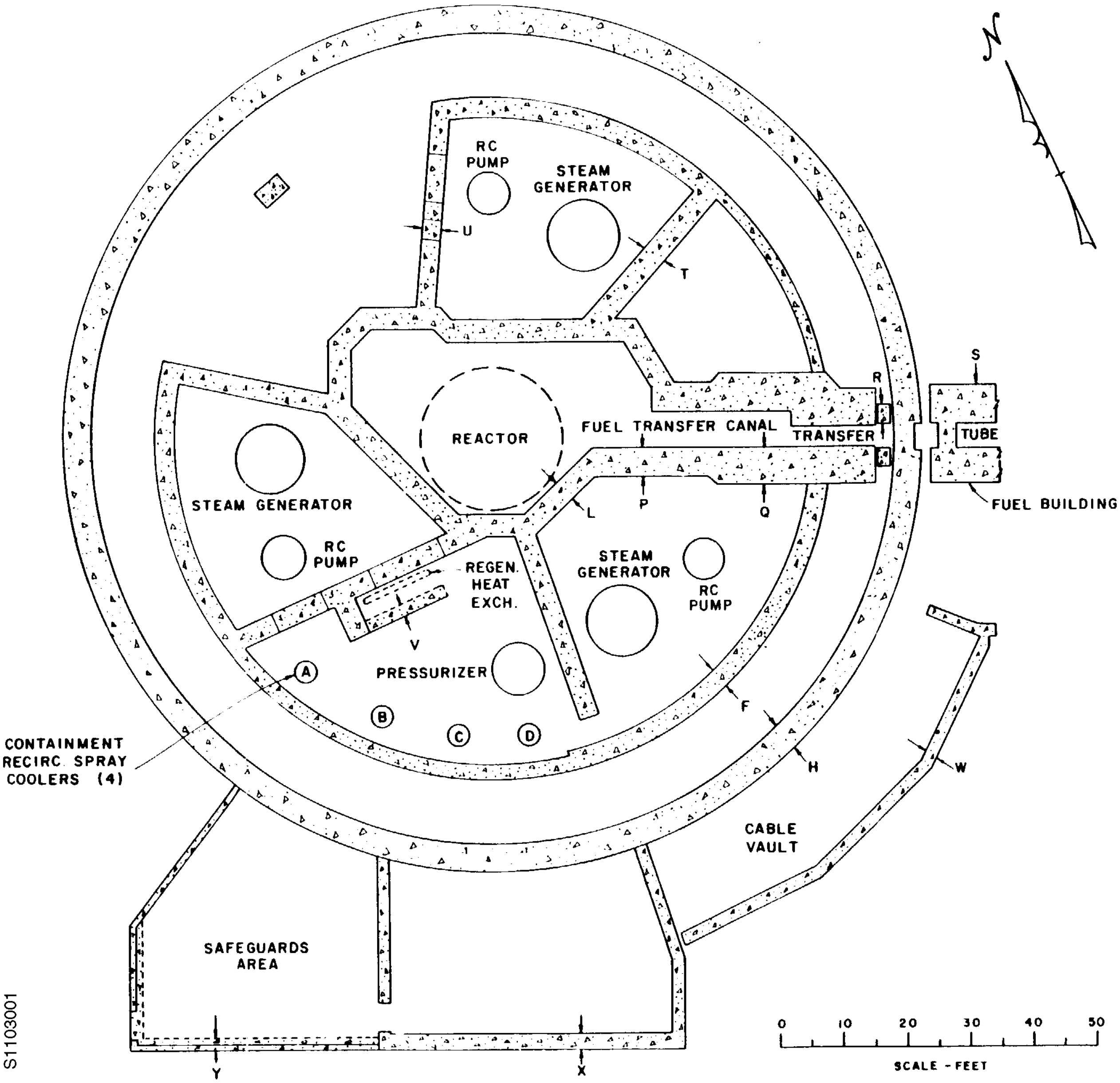
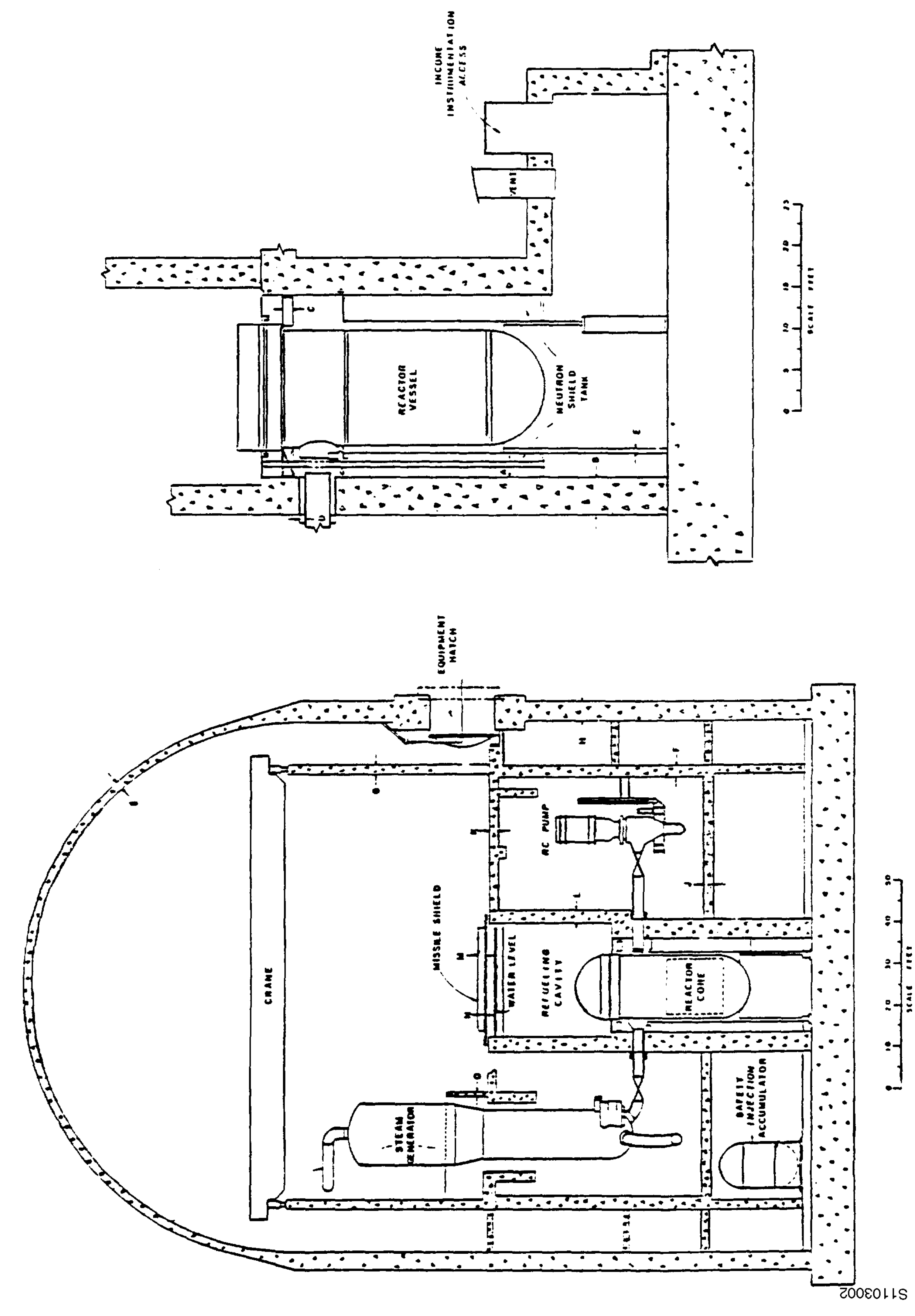


Figure 11.3-2
SHIELD ARRANGEMENT, ELEVATION, UNIT 1



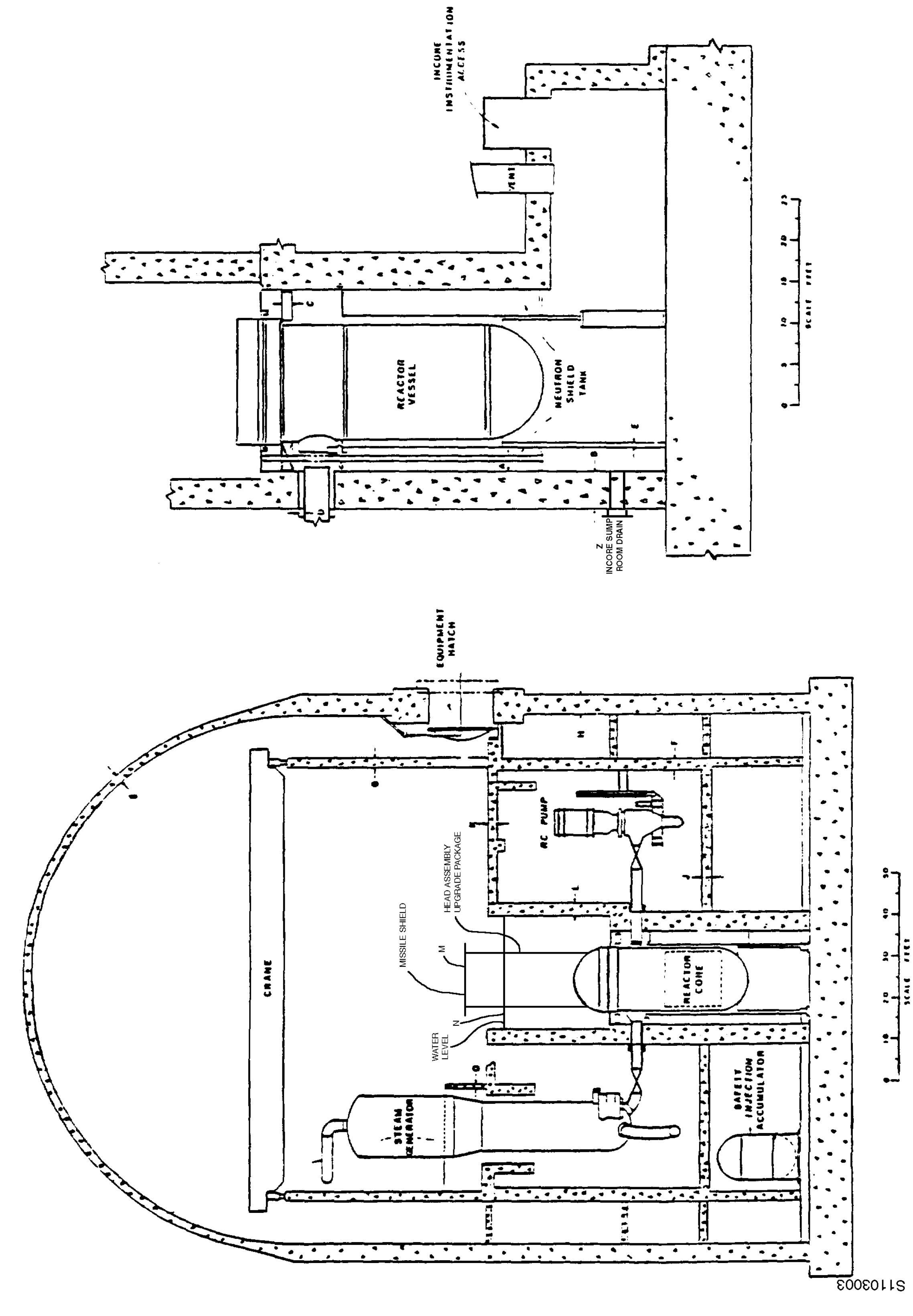


Figure 11.3-3
SHIELD ARRANGEMENT, ELEVATION, UNIT 2

Intentionally Blank

Appendix 11A
Radiation Exposure Evaluation for Expected Radioactive Effluents

Intentionally Blank

Appendix 11A: Radiation Exposure Evaluation for Expected Radioactive Effluents
Table of Contents

Appendix 11A: Radiation Exposure Evaluation for Expected Radioactive Effluents
List of Tables

Appendix 11A
RADIATION EXPOSURE EVALUATION FOR EXPECTED RADIOACTIVE EFFLUENTS

11A.1 ANALYTICAL BASIS

Surry Units 1 and 2 were analyzed and evaluated using the parameters and methodology set forth in Regulatory Guides 1.109 (Reference 1), 1.111 (Reference 2), and 1.112 (Reference 3), and NUREG-0017 (Reference 4). Maximum individual doses resulting from gaseous and liquid effluents were calculated.

Radioactive source terms, both liquid and gaseous, were calculated in a manner consistent with Regulatory Guide 1.112 and NUREG-0017. Specific data used for the generation of the sources terms are presented in Section 11A.2.1.

Meteorological information used in the calculation of doses was developed consistent with the methodology described in Regulatory Guide 1.111. Information related to the meteorological inputs is contained in Section 11A.2.2. The dispersion factors (χ/Q) and ground depositions factors (D/Q) from the release points at Surry Units 1 and 2 to the various receptions are contained in Section 11A.2.3.

A plant and animal census is provided in Section 11A.3.

Dose calculations were performed in a manner consistent with Regulatory Guide 1.109. The NRC computer codes LADTAP and GASPAR were used to perform the calculations. The results of the analyses presented in Section 11A.4 support the Surry Power Station's capability of keeping the levels of radioactivity in effluents as low as reasonably achievable.

The liquid waste disposal system described Sections 11.2.3 reflects changes in the liquid waste design used in the original Appendix I evaluation. Table 11.2-4, however, maintains the parameters used in the evaluation. This table was unchanged because the new liquid waste design was to be, at a minimum, equal to the previous system. In reality, improved performance was expected. Therefore, this analysis represents a conservative evaluation of offsite doses.

The gaseous waste disposal system is as described in 11.2.5.

11A.2 INPUT INFORMATION

General plant information, meteorological information, dispersion factors (χ/Q), and ground deposition factors (D/Q) are given in the following sections.

11A.2.1 General Plant Information

Plant information required by Appendix B of Regulatory Guide 1.112 is contained in Table 11A-1.

11A.2.2 Meteorological Information

Information concerning the onsite meteorological measurements program is found in Section 2.2.1.2.

Joint frequency distributions of wind speed and wind direction by atmospheric stability class are prepared monthly and annually, based on the format of Table 1 in Regulatory Guide 1.23 (Reference 5). Monthly and annual joint frequency distributions of 35-foot wind and $\Delta T_{150\text{ ft.} - 35\text{ ft.}}$ data are used as input for χ/Q and D/Q calculations for ground-level releases. Monthly and annual joint frequency distributions of 150-foot wind and $\Delta T_{150\text{ ft.} - 35\text{ ft.}}$ data are used as input for qualifying elevated release calculations of χ/Q and D/Q. The 2-year (1974-1976) data set was chosen as the most recent representative data set.

The meteorological data for the period 1974-1976 are considered to be representative of atmospheric transport and diffusion conditions of the site region on a long-term basis. The stability distribution based on $\Delta T_{150\text{ ft.} - 35\text{ ft.}}$ for the period March 3, 1974, to March 2, 1975, is consistent with the 2-year data period as used in this report (Table 11A-2). Comparison of annual wind roses for both the 35- and 150-foot levels indicates that the annual 2-year wind roses are consistent with the first year of data and are in general agreement with Richmond and Norfolk wind roses for the period January 1, 1969, to December 31, 1973 (Reference 6). The representativeness of the first year of the 2-year data set to the long-term meteorological conditions is discussed in Reference 6.

11A.2.3 Dispersion Factors and Ground Deposition Factors

Table 11A-3 provides χ/Q and D/Q values for ground-level and mixed-mode releases for the special appropriate distances as indicated in Section 11A.3 for each downwind sector. Tables 11A-4 and 11A-5 provide χ/Q and D/Q values associated with surface-level releases from the containment (considered as entrained in the building wake, and therefore a ground-level release) for the standard population distances. Tables 11A-6 and 11A-7 provide χ/Q and D/Q values associated with a process vent release from 3.048 m above one of the containment structures, or approximately 43 m above grade (considered as a mixed-mode release) for the standard population distances.

Dispersions factors (χ/Q) were calculated using a sector-average, straight-line model specified in Regulatory Guide 1.111. Ground deposition (D/Q) values were calculated according to Regulatory Guide 1.111. The mixed release mode was used as applicable for a release height of 3.05 m above the 40.1-m containment from a 0.08-m-diameter vent at an exit velocity of 30.5 m/s. Qualifying elevation release heights were adjusted for momentum stack downwash, and terrain rise, as described in Regulatory Guide 1.111.

The open terrain correction factor for χ/Q and D/Q values was applied in accordance with Figure 2 of Regulatory Guide 1.111. As described in References 2 and 3 and in Table 11A-8, the terrain is flat and rises to less than about 170 feet out to a distance of 50 miles near Richmond. Therefore, straight-line airflow trajectory regimes are considered to reasonably represent dispersion conditions as related to annual χ/Q values in the vicinity of the Surry Power Station.

The calculation χ/Q and D/Q values were based on onsite meteorological data during the period March 3, 1974, to March 2, 1975, and May 1, 1975, to April 30, 1976. Representative joint frequency distributions were developed for ground-level or elevated releases from the station as follows:

1. Ground-level release χ/Q and D/Q calculations were based on meteorological tower observations of wind speed and direction at the 35-foot level and of temperature differential (delta T) between the 150- and 35-foot levels. These levels were selected to conservatively represent the transport and diffusion of surface releases in the vicinity of the plant, or for vent releases entrained in the building wake. The σ_z diffusion parameter was based on the curves in Figure 1 of Regulatory Guide 1.111.
2. Qualifying elevated release χ/Q and D/Q calculations were based on meteorological tower observations of wind speed and direction at the 150-foot level and of the same temperature differential (delta T) between the 150- and 35-foot levels, as representing the environment of the plume between its release height and the ground.

11A.3 PLANT AND ANIMAL CENSUS

The plant and animal census is conducted annually in order to determine the current land use of the area surrounding Surry Power Station. The purpose of the census is to locate the distance to the nearest milk cow or other bovine, milk goat, vegetable garden, and residence within 5 miles of the Surry Power Station. The annual land use is detailed in the current radiological environmental monitoring report.

11A.4 DOSE CALCULATIONS

11A.4.1 Doses From Liquid Effluents

Liquid source terms were calculated for two specific cases using the GALE Code: the liquid radwaste system as presently operating and as the system operated with the blowdown treatment system, which is no longer used. These cases are indicated below:

1. Dirty wastes treated by a system consisting of two mixed-bed demineralizers and no treatment of steam generator blowdown.
2. Same as 1 above, only steam generator blowdown treated by two mixed-bed demineralizers. (This equipment is no longer used.)

Inputs to the GALE Code were based on (1) station operating experience, (2) information supplied previously in Chapter 11 and the Environmental Report (ER) (References 8 & 9), and (3) NUREG-0017. Source terms for each of the two cases outlined above are presented in Tables 11A-9 and 11A-10.

Liquid radioactive wastes from the units are released to the James River via the discharge canal. Possible pathways of exposure for release from the station include ingestion of fish and invertebrates and shoreline activities. The irrigated food pathway does not exist at this location, nor does the potable water pathway. For all pathways, a river dilution factor of 5 was assumed as appropriate per Regulatory Guide 1.109.

Doses from liquid pathways were calculated for the maximum individual, based on the models given in Regulatory Guide 1.109, using the computer code LADTAP. Dose factors, bioaccumulation factors, and shorewidth factors given in Regulatory Guide 1.109 and in the LADTAP code were used, as were usage terms for shoreline activities and ingestion of fish and invertebrates.

Tables 11A-11 and 11A-12 present the LADTAP input data and the maximum individual doses for both cases indicated above.

During normal station operations, doses from liquid effluents are calculated according to the Offsite Dose Calculation Manual (ODCM). Calculations from the ODCM demonstrate compliance with this section.

11A.4.2 Doses From Gaseous Effluents

Gaseous source terms were calculated using the GALE Code, and are presented in Table 11A-13. Inputs to the GALE Code were based on (1) plant operating experience, (2) information supplied in Chapter 11 and the ER, and (3) NUREG-0017.

Doses to the maximum individual from gaseous effluents were calculated by the NRC GASPAR Code, using the models of Regulatory Guide 1.109. Dose factors, annual air intake, intakes of food products, and parameters for calculating radionuclide concentrations in food products as given in Regulatory Guide 1.109 and in the GASPAR Code were used.

Dose contributions from the following pathways were calculated and analyzed in the assessment of the maximally exposed individual:

1. Immersion in the plume.
2. Ground contamination.
3. Inhalation.
4. Consumption of vegetables, meat, and milk.

For dose calculation purposes, the source terms of Table 11A-2 were divided into mixed-mode releases (i.e., those released from the Surry process vent that could be considered to be elevated at certain times and at ground level at others) and ground-level releases (i.e., those released from the ventilation vents, steam generator flash tank, and turbine building). The sources of releases for the Surry process vent are the gaseous waste and containment vacuum systems. For dose calculation purposes, these releases were considered mixed mode, and the χ/Q and D/Q values, as presented in Table 11A-3, reflect this.

The sources of releases from the ventilation vents of the Surry units are the auxiliary, decontamination, and spent fuel buildings, safeguards areas, condenser air ejector, and containment purge systems. For dose calculation purposes, these releases were considered ground level, and the χ/Q and D/Q values, as presented in Table 11A-3, reflect this.

Based on the χ/Q and D/Q values in Table 11A-3, specific locations were analyzed for the location of the maximally exposed individual. When the principal locations were determined, dose calculations were performed, incorporating the pathways specific to each location. By adding the doses resulting from both mixed-mode and ground-level releases for each of the pathways existing at these locations, the location of the maximum individual was determined.

After evaluating the special locations, the maximum organ dose occurred to an infant who resides 3.75 miles north-northwest of the power station and drinks milk from a cow raised at this location. The maximum total body dose occurred to an individual 1.53 miles south of the Surry Power Station. Table 11A-14 presents the doses at the location of the maximally exposed individuals for Surry Units 1 and 2.

Table 11A-15 presents the doses to the above-mentioned individual based on the cooling of steam generator blowdown below 212°F. Operation in this manner eliminates the gaseous releases of I-131 and I-133 from the blowdown vent offgas and results in a reduction of the maximum organ dose by a factor of 5.

During normal station operations, doses from gaseous effluents are calculated according to the Offsite Dose Calculation Manual (ODCM). Calculations from the ODCM demonstrate compliance with this section.

11A REFERENCES

1. Regulatory Guide 1.109, *Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I*, 1976.
2. Regulatory Guide 1.111, *Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors*, 1976.
3. Regulatory Guide 1.112, *Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Power Reactors*, 1976.
4. NUREG-0017, *Calculations of Releases of Radioactive Materials in Gaseous and Liquid Effluents for Pressurized Water Reactors (PWR-GALE Code),* 1976.
5. Regulatory Guide 1.23, *Onsite Meteorological Program (Safety Guide 23)*, 1972.
6. Virginia Electric and Power Company, *Surry 3 and 4 Annual Meteorological Report*, Docket Nos. 50-434 and 40-435, 1975.
7. U.S. Atomic Energy Commission, *Surry Power Station Units 3 and 4 Final Environmental Statement*, Docket Nos. 50-434 and 50-435, 1974.
8. U.S. Atomic Energy Commission, *Surry Power Station Unit 1 Final Environmental Statement*, Docket No. 50-280, 1972.
9. U.S. Atomic Energy Commission, *Surry Power Station Unit 2 Final Environmental Statement*, Docket No. 50-281, 1972.

Table 11A-1
GENERAL PLANT INFORMATION

	Units	Value	Source
General			
Maximum core thermal power evaluated for safety considerations in SAR	MWt	2546	Section 1.1
Quantity of liquid tritium released	Ci/yr	480	GALE Code calculations
Quantity of gaseous tritium released	Ci/yr	490	GALE Code calculations
Primary system			
Mass of coolant in primary system, excluding pressurizer and primary coolant purification system, at full power	10^3 lb	367	Calculations based on information in Tables 4.1-2, 4.1-3, 4.1-4, and 4.1-6
Average primary system letdown rate to primary coolant purification system	gpm	60	Table 9.1-3
Average flow rate through the primary coolant purification system demineralizer	gpm	6	Table 9.1-5
Average shim bleed flow	gpm	1.8	Unit operating experience
Secondary system			
Number of steam generators		3	Table 4.1-4
Type of steam generators		U-tube	Table 4.1-4
Carryover factor used for evaluation of iodine and non-volatiles		1% iodine 0.1% nonvolatiles	NUREG-0017
Total steam flow in secondary system	10^6 lb/hr	11.2	Figure 10.2-1
Mass of liquid in each steam generator at full power	10^3 lb	90.7	Calculations based on information in Table 4.1-4
Primary-to-secondary system leakage rate used in evaluation	lb/day	100	NUREG-0017
Average steam generator blowdown rate used in evaluation total	10^3 lb/hr	30.3	Unit operating experience

Table 11A-2
MONTHLY $\Delta T_{150\ ft\ -\ 35\ ft}$ STABILITY DISTRIBUTION (%)

Month	A	B	C	D	E	F	G
January 1975[a]	2.13	2.63	2.79	36.29	41.22	7.88	7.06
2-yr	10.07	2.19	2.42	32.08	39.73	5.93	7.57
February 1975	7.54	3.28	4.10	35.57	31.97	9.02	8.52
2-yr	11.88	3.83	3.35	25.68	34.53	9.65	11.08
March 1974-1975	20.68	4.46	5.36	32.29	28.12	4.02	5.06
2-yr	17.04	4.12	4.12	29.75	32.42	4.77	7.80
April 1974	14.92	2.44	4.26	21.77	40.79	8.37	7.46
2-yr	20.35	2.59	3.50	18.18	33.78	9.65	11.96
May 1974	21.26	4.42	4.42	24.25	32.24	7.42	5.99
2-yr	15.06	3.35	4.28	21.98	34.33	9.92	11.06
June 1974	17.26	4.99	3.12	25.57	34.51	5.41	9.15
2-yr	12.85	3.68	3.37	17.95	30.56	9.72	21.87
July 1974	15.47	5.56	6.91	19.67	31.83	8.56	12.01
2-yr	7.67	3.33	4.27	21.49	38.13	9.84	15.27
August 1974	10.01	3.65	5.28	29.50	37.89	6.77	6.90
2-yr	6.73	3.05	4.58	28.36	34.88	11.10	11.30
September 1974	14.12	5.24	4.37	26.93	33.77	7.57	8.01
2-yr	7.77	3.78	3.63	23.46	41.03	7.63	12.71
October 1974	15.80	2.66	2.24	17.62	25.73	14.13	21.82
2-yr	10.98	3.08	2.87	18.72	31.71	12.12	20.52
November 1974	16.74	4.08	3.52	21.38	29.11	13.22	11.95
2-yr	11.72	3.07	3.97	21.48	31.94	13.88	13.95
December 1974	16.95	4.89	3.02	27.01	34.48	7.61	6.03
2-yr	13.27	3.76	2.77	29.03	36.48	1.95	6.74

a. The first-year data period is March 3, 1974, to March 2, 1975, and the 2-year data period is the combined years of March 3, 1974, to March 2, 1975, and May 1, 1975, to April 30, 1976.

Table 11A-3
χ/Q AND D/Q VALUES AT SPECIAL DISTANCES
AND RELEASE MODES FOR A 2-YEAR DATA PERIOD[a]

		Ground-Level Release		Mixed-Mode Release (joint ground-level and elevated release)	
Receptor Direction	Distance (m)	χ/Q (sec/m^3)	D/Q (m^{-2})	χ/Q (sec/m^3)	D/Q (m^{-2})
NNE	2414	4.7 (-06)[b]	1.0 (-08)	3.9 (-07)	2.2 (-09)
NNE	3058	2.9 (-06)	5.9 (-09)	2.8 (-07)	1.3 (-09)
NE	2333	5.2 (-06)	1.1 (-08)	4.2 (-07)	3.3 (-09)
S	503	3.1 (-05)	1.7 (-07)	8.0 (-07)	3.5 (-08)
S	2470	1.5 (-06)	5.5 (-09)	3.1 (-07)	2.5 (-09)
SSW	3492	3.8 (-07)	1.4 (-09)	1.2 (-07)	7.7 (-10)
SW	2881	6.7 (-07)	2.0 (-09)	1.2 (-07)	9.7 (-10)
SW	3379	4.8 (-07)	1.4 (-09)	1.1 (-07)	6.6 (-10)
WSW	4828	2.0 (-07)	5.9 (-10)	6.8 (-08)	3.3 (-10)
NNW	6034	5.9 (-07)	5.9 (-10)	5.5 (-08)	1.2 (-10)
N	274	2.7 (-04)	5.9 (-07)	5.8 (-07)	2.6 (-08)
N	503	9.5 (-05)	2.2 (-07)	4.5 (-07)	1.7 (-08)
SSE	4747	4.1 (-07)	1.3 (-09)	7.5 (-08)	4.8 (-10)

a. Data period is March 3, 1974, to March 2, 1975, and May 1, 1975, to April 30, 1976. Open terrain corrective factors of Regulatory Guide 1.111 are incorporated.

b. 4.7 (-06) = 4.7×10^{-6}.

Table 11A-4

ANNUAL AVERAGE χ/Q (sec/m^3) VALUES BASED ON A GROUND-LEVEL RELEASE FOR A 2-YEAR DATA PERIOD[a]

Receptor	Distance in Meters									
Direction	805	2414	4023	5633	7242	12070	24140	40234	56327	72420
N	4.4 (-5)[b]	4.7 (-6)	1.7 (-6)	9.4 (-7)	6.2 (-7)	2.6 (-7)	1.1 (-7)	5.5 (-8)	3.6 (-8)	2.7 (-8)
NNE	4.4 (-5)	4.7 (-6)	1.7 (-6)	9.2 (-7)	6.1 (-7)	2.6 (-7)	1.0 (-7)	5.4 (-8)	3.6 (-8)	2.6 (-8)
NE	4.5 (-5)	4.8 (-6)	1.7 (-6)	9.5 (-7)	6.2 (-7)	2.6 (-7)	1.1 (-7)	5.6 (-8)	3.7 (-8)	2.7 (-8)
ENE	2.0 (-5)	2.1 (-6)	7.5 (-7)	4.1 (-7)	2.7 (-7)	1.1 (-7)	4.6 (-8)	2.4 (-8)	1.6 (-8)	1.1 (-8)
E	1.8 (-5)	1.9 (-6)	6.6 (-7)	3.6 (-7)	2.4 (-7)	9.9 (-8)	4.0 (-8)	2.1 (-8)	1.4 (-8)	1.0 (-8)
ESE	1.5 (-5)	1.5 (-6)	5.3 (-7)	2.9 (-7)	1.9 (-7)	7.8 (-8)	3.1 (-8)	1.6 (-8)	1.1 (-0)	7.8 (-9)
SE	1.5 (-5)	1.6 (-6)	5.5 (-7)	3.0 (-7)	1.9 (-7)	8.1 (-8)	3.3 (-8)	1.7 (-8)	1.1 (-8)	8.1 (-9)
SSE	1.6 (-5)	1.6 (-6)	5.6 (-7)	3.0 (-7)	1.9 (-7)	8.0 (-8)	3.2 (-8)	1.6 (-8)	1.1 (-8)	7.8 (-9)
S	1.5 (-5)	1.6 (-6)	5.3 (-7)	2.9 (-7)	1.8 (-7)	7.6 (-8)	3.0 (-8)	1.5 (-8)	1.0 (-8)	7.3 (-9)
SSW	8.1 (-6)	8.2 (-7)	2.8 (-7)	1.5 (-7)	9.4 (-8)	3.8 (-8)	1.5 (-8)	7.6 (-9)	4.9 (-9)	3.6 (-9)
SW	9.4 (-6)	9.5 (-7)	3.3 (-7)	1.0 (-7)	1.2 (-7)	4.8 (-8)	1.9 (-8)	9.7 (-9)	6.4 (-9)	4.6 (-9)
WSW	8.2 (-6)	8.2 (-7)	2.8 (-7)	1.5 (-7)	9.5 (-8)	3.9 (-8)	1.5 (-8)	7.8 (-9)	5.0 (-9)	3.7 (-9)
W	1.3 (-5)	1.3 (-6)	4.4 (-7)	2.4 (-7)	1.5 (-7)	6.3 (-8)	2.5 (-8)	1.3 (-8)	8.2 (-9)	6.0 (-9)
WNW	1.8 (-5)	1.9 (-6)	6.4 (-7)	3.5 (-7)	2.3 (-7)	9.4 (-8)	3.8 (-8)	2.0 (-8)	1.3 (-8)	9.3 (-9)
NW	2.1 (-5)	2.2 (-6)	7.8 (-7)	4.2 (-7)	2.8 (-7)	1.2 (-7)	4.6 (-8)	2.4 (-8)	1.6 (-8)	1.2 (-8)
NNW	3.3 (-5)	3.6 (-6)	1.3 (-6)	7.0 (-7)	4.6 (-7)	2.0 (-7)	7.9 (-8)	4.1 (-8)	2.7 (-8)	2.0 (-8)
	0.5 ml	1.5 ml	2.5 ml	3.5 ml	4.5 ml	7.5 ml	15.0 ml	25.0 ml	35.0 ml	45.0 ml

a. Data period is March 3, 1974 to March 2, 1975, and May 1, 1975, to April 30, 1976. Open terrain correction factors of Regulatory Guide 1.111 are incorporated.

b. 4.4 (-05) = 4.4×10^{-5}.

Table 11A-5

ANNUAL AVERAGE D/Q (M^{-2}) VALUES BASED ON A GROUND-LEVEL RELEASE FOR A 2-YEAR DATA PERIOD[a]

Receptor Direction	Distance in Meters									
	805	2414	4023	5633	7242	12070	24140	40234	56327	72420
N	9.7 (-08)[b]	7.3 (-09)	2.1 (-09)	9.9 (-10)	5.9 (-10)	2.0 (-10)	5.4 (-11)	2.0 (-11)	1.0 (-11)	6.2 (-12)
NNE	1.4 (-07)	1.0 (-08)	3.0 (-09)	1.4 (-09)	8.4 (-10)	2.8 (-10)	7.8 (-11)	2.9 (-11)	1.5 (-11)	8.9 (-12)
NE	1.4 (-07)	1.0 (-08)	3.0 (-09)	1.4 (-09)	8.3 (-10)	2.8 (-10)	7.6 (-11)	2.8 (-11)	1.4 (-11)	8.8 (-12)
ENE	6.4 (-08)	4.8 (-09)	1.4 (-09)	6.5 (-10)	3.9 (-10)	1.3 (-10)	3.6 (-11)	1.3 (-11)	6.7 (-12)	4.1 (-12)
E	6.0 (-08)	4.5 (-09)	1.3 (-09)	6.1 (-10)	3.6 (-10)	1.2 (-10)	3.3 (-11)	1.2 (-11)	6.2 (-12)	3.8 (-12)
ESE	6.0 (-08)	4.5 (-09)	1.3 (-09)	6.2 (-10)	3.7 (-10)	1.2 (-10)	3.4 (-11)	1.2 (-11)	6.3 (-12)	3.9 (-12)
SE	7.4 (-08)	5.5 (-09)	1.6 (-09)	7.6 (-10)	4.5 (-10)	1.5 (-10)	4.1 (-11)	1.5 (-11)	7.7 (-12)	4.7 (-12)
SSE	8.4 (-08)	6.3 (-09)	1.8 (-09)	8.6 (-10)	5.1 (-10)	1.7 (-10)	4.7 (-11)	1.7 (-11)	8.8 (-12)	5.4 (-12)
S	7.7 (-08)	5.7 (-09)	1.7 (-09)	7.9 (-10)	4.6 (-10)	1.5 (-10)	4.3 (-11)	1.6 (-11)	8.0 (-12)	4.9 (-12)
SSW	4.5 (-08)	3.3 (-09)	9.6 (-10)	4.6 (-10)	2.7 (-10)	9.0 (-11)	2.5 (-11)	9.1 (-12)	4.6 (-12)	2.9 (-12)
SW	4.0 (-08)	3.0 (-09)	8.8 (-10)	4.2 (-10)	2.5 (-10)	8.2 (-11)	2.3 (-11)	8.3 (-12)	4.2 (-12)	2.6 (-12)
WSW	4.1 (-08)	3.1 (-09)	8.8 (-10)	4.2 (-10)	2.5 (-10)	8.2 (-11)	2.3 (-11)	8.3 (-12)	4.3 (-12)	2.6 (-12)
W	6.4 (-08)	4.8 (-09)	1.4 (-09)	6.5 (-10)	3.9 (-10)	1.3 (-10)	3.6 (-11)	1.3 (-11)	6.7 (-12)	4.1 (-12)
WNW	7.0 (-08)	5.2 (-09)	1.5 (-09)	7.1 (-10)	4.2 (-10)	1.4 (-10)	3.9 (-11)	1.4 (-11)	7.3 (-12)	4.5 (-12)
NW	7.2 (-08)	5.4 (-09)	1.6 (-09)	7.3 (-10)	4.3 (-10)	1.4 (-10)	4.0 (-11)	1.5 (-11)	7.5 (-12)	4.6 (-12)
NNW	7.0 (-08)	5.3 (-09)	1.5 (-09)	7.2 (-10)	4.3 (-10)	1.4 (-10)	3.9 (-11)	1.4 (-11)	7.3 (-12)	4.5 (-12)
	0.5 ml	1.5 ml	2.5 ml	3.5 ml	4.5 ml	7.5 ml	15.0 ml	25.0 ml	35.0 ml	45.0 ml

a. Data period is March 3, 1974, to March 2, 1975, and May 1, 1915, to April 30, 1976. Open terrain correction factors of Regulatory Guide 1.111 are incorporated.

b. 9.7 (-08) = 9.7×10^{-8}.

Table 11A-6

ANNUAL AVERAGE χ/Q (SEC/M^3) VALUES BASED ON A MIXED-MODE RELEASE FOR A 2-YEAR DATA PERIOD[a]

Receptor Direction	Distance in Meters									
	805	2414	4023	5633	7242	12070	24140	40234	56327	72420
N	5.6 (-7)[b]	2.8 (-7)	1.4 (-7)	8.9 (-8)	6.8 (-8)	3.5 (-8)	1.2 (-8)	6.3 (-9)	4.1 (-9)	3.0 (-9)
NNE	8.8 (-7)	3.9 (-7)	1.8 (-7)	1.2 (-7)	8.9 (-8)	4.4 (-8)	1.4 (-8)	7.3 (-9)	4.8 (-9)	3.5 (-9)
NE	1.0 (-6)	3.9 (-7)	1.8 (-7)	1.1 (-7)	8.2 (-8)	4.1 (-8)	1.4 (-8)	6.9 (-9)	4.5 (-9)	3.3 (-9)
ENE	6.6 (-7)	2.3 (-7)	1.1 (-7)	6.6 (-8)	4.7 (-8)	2.4 (-8)	8.1 (-9)	4.1 (-9)	2.7 (-9)	2.0 (-9)
E	6.8 (-7)	2.3 (-7)	1.0 (-7)	6.4 (-8)	4.6 (-8)	2.3 (-8)	8.0 (-9)	4.1 (-9)	2.6 (-9)	1.9 (-9)
ESE	6.7 (-7)	1.9 (-7)	8.2 (-8)	5.3 (-8)	3.7 (-8)	1.8 (-8)	7.2 (-9)	3.7 (-9)	2.4 (-9)	1.7 (-9)
SE	7.5 (-7)	2.1 (-7)	9.1 (-8)	5.6 (-8)	4.0 (-8)	1.9 (-8)	7.1 (-9)	3.6 (-9)	2.3 (-9)	1.7 (-9)
SSE	7.3 (-7)	2.2 (-7)	9.5 (-8)	5.8 (-8)	4.0 (-8)	1.9 (-8)	6.7 (-9)	3.4 (-9)	2.2 (-9)	1.6 (-9)
S	8.7 (-7)	3.1 (-7)	1.4 (-7)	8.4 (-8)	5.9 (-8)	2.8 (-8)	8.6 (-9)	4.4 (-9)	2.8 (-9)	2.1 (-9)
SSW	6.2 (-7)	2.0 (-7)	8.9 (-8)	5.9 (-8)	4.3 (-8)	2.0 (-8)	6.6 (-9)	3.4 (-9)	2.2 (-9)	1.6 (-9)
SW	5.4 (-7)	1.6 (-7)	7.8 (-8)	4.8 (-8)	3.5 (-8)	1.7 (-8)	5.6 (-9)	2.9 (-9)	1.9 (-9)	1.4 (-9)
WSW	6.0 (-7)	1.7 (-7)	7.8 (-8)	5.3 (-8)	3.7 (-8)	1.9 (-8)	6.1 (-9)	3.1 (-9)	2.0 (-9)	1.5 (-9)
W	7.1 (-7)	2.3 (-7)	1.0 (-7)	6.7 (-8)	5.0 (-8)	2.3 (-8)	7.5 (-9)	3.8 (-9)	2.5 (-9)	1.8 (-9)
WNW	6.0 (-7)	2.3 (-7)	1.1 (-7)	6.6 (-8)	4.7 (-8)	2.3 (-8)	8.9 (-9)	4.5 (-9)	3.0 (-9)	2.2 (-9)
NW	7.5 (-7)	2.2 (-7)	1.0 (-7)	6.4 (-8)	4.6 (-8)	2.5 (-8)	9.2 (-9)	4.7 (-9)	3.1 (-9)	2.2 (-9)
NNW	4.4 (-7)	1.9 (-7)	9.6 (-8)	6.3 (-8)	4.7 (-8)	2.8 (-8)	9.8 (-9)	5.0 (-9)	3.3 (-9)	2.4 (-9)
	0.5 ml	1.5 ml	2.5 ml	3.5 ml	4.5 ml	7.5 ml	15.0 ml	25.0 ml	35.0 ml	45.0 ml

a. Data period is March 3, 1974, to March 2, 1975, and May 1, 1915, to April 30, 1976. Open terrain correction factors of Regulatory Guide 1.111 are incorporated.

b. 5.6 (-7) = 5.6×10^7.

Table 11A-7

ANNUAL AVERAGE D/Q (M^{-2}) VALUES BASED ON A MIXED-MODE RELEASE FOR A 2-YEAR DATA PERIOD[a]

Receptor Direction	Distance in Meters									
	805	2414	4023	5633	7242	12070	24140	40234	56327	72420
N	1.0 (-08)[b]	1.1 (-09)	3.4 (-10)	1.8 (-10)	1.2 (-10)	5.8 (-11)	2.4 (-11)	1.0 (-11)	5.9 (-12)	4.1 (-12)
NNE	2.0 (-08)	2.2 (-09)	6.7 (-10)	3.4 (-10)	2.2 (-10)	9.8 (-11)	3.8 (-11)	1.6 (-11)	9.3 (-12)	6.4 (-12)
NE	2.9 (-08)	3.0 (-09)	9.0 (-10)	4.5 (-10)	2.9 (-10)	1.2 (-10)	4.3 (-11)	1.9 (-11)	1.1 (-11)	7.0 (-12)
ENE	1.6 (-08)	1.7 (-09)	5.2 (-10)	2.6 (-10)	1.6 (-10)	6.4 (-11)	2.3 (-11)	9.9 (-12)	5.6 (-12)	3.7 (-12)
E	1.7 (-08)	1.8 (-09)	5.3 (-10)	2.6 (-10)	1.6 (-10)	6.4 (-11)	2.3 (-11)	9.7 (-12)	5.5 (-12)	3.6 (-12)
ESE	2.1 (-08)	2.2 (-09)	6.5 (-10)	3.2 (-10)	1.9 (-10)	7.2 (-11)	2.4 (-11)	1.0 (-11)	5.7 (-12)	3.7 (-12)
SE	2.5 (-08)	2.6 (-09)	7.6 (-10)	3.7 (-10)	2.3 (-10)	8.5 (-11)	2.9 (-11)	1.2 (-11)	6.7 (-12)	4.4 (-12)
SSE	2.1 (-08)	2.3 (-09)	6.9 (-10)	3.4 (-10)	2.1 (-10)	7.8 (-11)	2.7 (-11)	1.2 (-11)	6.5 (-12)	4.3 (-12)
S	2.3 (-08)	2.6 (-09)	7.9 (-10)	3.9 (-10)	2.4 (-10)	9.1 (-11)	3.2 (-11)	1.4 (-11)	7.9 (-12)	5.2 (-12)
SSW	1.5 (-08)	1.8 (-09)	5.3 (-10)	2.6 (-10)	1.6 (-10)	6.1 (-11)	2.1 (-11)	9.2 (-12)	5.2 (-12)	3.4 (-12)
SW	1.3 (-08)	1.4 (-09)	4.2 (-10)	2.1 (-10)	1.3 (-10)	4.8 (-11)	1.5 (-11)	7.0 (-12)	3.9 (-12)	2.6 (-12)
WSW	1.5 (-08)	1.6 (-09)	4.9 (-10)	2.4 (-10)	1.5 (-10)	5.4 (-11)	1.9 (-11)	8.0 (-12)	4.5 (-12)	2.9 (-12)
W	1.7 (-08)	1.9 (-09)	5.7 (-10)	2.8 (-10)	1.7 (-10)	6.5 (-11)	2.3 (-11)	9.7 (-12)	5.5 (-12)	3.6 (-12)
WNW	1.2 (-08)	1.3 (-09)	4.0 (-10)	2.0 (-10)	1.3 (-10)	5.2 (-11)	1.9 (-11)	8.2 (-12)	4.7 (-12)	3.2 (-12)
NW	2.5 (-08)	2.4 (-09)	7.1 (-10)	3.5 (-10)	2.2 (-10)	8.2 (-11)	2.8 (-11)	1.2 (-11)	6.5 (-12)	4.3 (-12)
NNW	9.3 (-08)	9.6 (-10)	2.9 (-10)	1.5 (-10)	9.9 (-11)	4.5 (-11)	1.8 (-11)	7.5 (-12)	4.3 (-12)	3.0 (-12)
	0.5 ml	1.5 ml	2.5 ml	3.5 ml	4.5 ml	7.5 ml	15.0 ml	25.0 ml	35.0 ml	45.0 ml

a. Data period is March 3, 1974 to March 2, 1975, and May 1, 1975, to April 30, 1976. Open terrain correction factors of Regulatory Guide 1.111 are incorporated

b. $1.0\ (-08) = 1.0 \times 10^{-8}$.

Table 11A-8
0- TO 5-MILE HIGHPOINTS BY MILE AND 5- TO 10-MILE HIGHPOINTS FOR 16 CARDINAL POINTS FROM SURRY POWER STATION[a]

Section	0-1	1-2	2-3	3-4	4-5	5-10
N	37	-	-	12	55	80
NNE	37	5	2	-	71	100
NE	38	4	-	-	52	80
ENE	43	10	-	-	30	80
E	38	3	-	-	-	60
ESE	38	37	-	-	11	30
SE	39	36	-	-	-	-
SSE	39	33	39	37	37	50
S	37	39	50	51	60	80
SSW	42	38	42	70	85	90
SW	39	34	70	72	84	95
WSW	40	-	55	81	83	130
W	38	-	-	70	88	90
WNW	39	-	-	-	-	10
NW	38	-	-	6	5	80
NNW	39	-	-	12	47	110

a. All highpoints are measured in feet. Dash (-) denotes sea level.

Sources: Surry Power Station Units 3 and 4 Environment Report, Figures 2.6-8 through 2.6-11; and U.S. Geological Survey 7.5 minute topographic maps.

Table 11A-9
LIQUID SOURCE TERMS FROM SURRY UNITS 1 & 2 (PER UNIT BASIS)
WITHOUT STEAM GENERATOR BLOWDOWN PROCESSING

		Coolant Concentrations		Annual Releases To Discharge Canal							
Nuclide	Half-life (Days)	Primary (Micro Ci/ml)	Secondary (Micro Ci/ml)	Boron Rs (Curies)	Misc. Wastes (Curies)	Secondary (Curies)	Turb Bldg (Curies)	Total LWS (Curies)	Adjusted Total (Ci/yr)	Detergent Wastes (Ci/yr)	Total (Ci/yr)
Corrosion And Activation Products											
CR51	2.78E+01	1.68E-03	2.11E-07	0.00257	0.00844	0.02643	0.00000	0.03744	0.03757	0.00000	0.03800
MN54	3.03E+02	2.74E-04	5.06E-08	0.00052	0.00138	0.00634	0.00000	0.00824	0.00827	0.00036	0.00860
FE55	9.50E+02	1.41E-03	1.77E-07	0.00271	0.00713	0.02214	0.00000	0.03199	0.03210	0.00000	0.03200
FE59	4.50E+01	8.84E-04	1.30E-07	0.00148	0.00445	0.01624	0.00000	0.02217	0.02225	0.00000	0.02200
CO58	7.13E+01	1.41E-02	1.80E-06	0.02492	0.07121	0.22509	0.00000	0.32124	0.32239	0.00144	0.32000
C060	1.92E+03	1.76E-03	2.27E-07	0.00340	0.00891	0.02845	0.00000	0.04077	0.04091	0.00313	0.04400
NP239	2.35E+00	1.09E-03	1.14E-07	0.00022	0.00512	0.01426	0.00000	0.01960	0.01967	0.00000	0.02000
Fission Products											
BR83	1.00E-01	4.97E-03	1.96E-07	0.00000	0.00498	0.02458	0.00000	0.02956	0.02967	0.00000	0.03000
BR84	2.21E-02	2.77E-03	3.29E-08	0.00000	0.00003	0.00412	0.00000	0.00415	0.00416	0.00000	0.00420
BR85	2.08E-03	3.22E-04	3.53E-10	0.00000	0.00000	0.00004	0.00000	0.00004	0.00004	0.00000	0.00004
RB86	1.87E+01	7.47E-05	1.07E-08	0.00018	0.01868	0.00134	0.00000	0.02019	0.02027	0.00000	0.02000
RB88	1.24E-02	2.14E-01	1.36E-06	0.00000	0.00218	0.17036	0.00000	0.17304	0.17367	0.00000	0.17000
SR89	5.20E+01	3.09E-04	5.17E-08	0.00053	0.00156	0.00647	0.00000	0.00856	0.00859	0.00000	0.00860
SR90	1.03E+04	8.82E-06	1.26E-09	0.00002	0.00004	0.00016	0.00000	0.00022	0.00022	0.00000	0.00022
Y90	2.67E+00	1.09E-06	7.31E-10	0.00002	0.00001	0.00009	0.00000	0.00011	0.00012	0.00000	0.00012
SR91	4.03E-01	6.35E-04	4.74E-08	0.00000	0.00210	0.00594	0.00000	0.00805	0.00808	0.00000	0.00810

Table 11A-9 (continued)
LIQUID SOURCE TERMS FROM SURRY UNITS 1 & 2 (PER UNIT BASIS)
WITHOUT STEAM GENERATOR BLOWDOWN PROCESSING

		Coolant Concentrations		Annual Releases To Discharge Canal							
Nuclide	Half-life (Days)	Primary (Micro Ci/ml)	Secondary (Micro Ci/ml)	Boron Rs (Curies)	Misc. Wastes (Curies)	Secondary (Curies)	Turb Bldg (Curies)	Total LWS (Curies)	Adjusted Total (Ci/yr)	Detergent Wastes (Ci/yr)	Total (Ci/yr)
Fission Products											
Y91M	3.47E-02	3.82E-04	4.73E-08	0.00000	0.00136	0.00593	0.00000	0.00728	0.00731	0.00000	0.00730
Y91	5.88E+01	5.66E-05	7.73E-09	0.00011	0.00029	0.00097	0.00000	0.00137	0.00137	0.00000	0.00140
Y93	4.25E-01	3.31E-05	3.09E-09	0.00000	0.00011	0.00039	0.00000	0.00050	0.00050	0.00000	0.00050
ZR95	6.50E+01	5.30E-05	7.71E-09	0.00009	0.00027	0.00097	0.00000	0.00133	0.00133	0.00000	0.00130
NB95	3.50E+01	4.42E-05	7.84E-09	0.00009	0.00022	0.00098	0.00000	0.00129	0.00130	0.00000	0.00130
MO99	2.79E+00	7.61E-02	1.08E-05	0.02037	0.36131	1.35526	0.00010	1.73705	1.74331	0.00000	1.70000
TC99M	2.50E-01	4.80E-02	2.91E-05	0.01948	0.28162	3.64351	0.00019	3.94479	3.95902	0.00000	4.00000
RU103	3.96E+01	3.98E-05	5.20E-09	0.00007	0.00020	0.00065	0.00000	0.00092	0.00092	0.00005	0.00097
RH103M	3.96E-02	4.76E-05	3.10E-08	0.00007	0.00020	0.00389	0.00000	0.00416	0.00417	0.00000	0.00420
RU106	3.67E+02	8.63E-06	1.26E-09	0.00002	0.00004	0.00016	0.00000	0.00022	0.00022	0.00086	0.00110
RH106	3.47E-04	1.08E-05	8.89E-09	0.00002	0.00004	0.00111	0.00000	0.00118	0.00118	0.00000	0.00120
TE125M	5.80E+01	2.56E-05	2.32E-09	0.00004	0.00013	0.00029	0.00000	0.00046	0.00047	0.00000	0.00047
TE127M	1.09E+02	2.47E-04	2.30E-08	0.00045	0.00125	0.00288	0.00000	0.00457	0.00459	0.00000	0.00460
TE127	3.92E-01	8.32E-04	1.60E-07	0.00045	0.00315	0.02007	0.00000	0.02367	0.02375	0.00000	0.02400
TE129M	3.40E+01	1.24E-03	1.57E-07	0.00197	0.00622	0.01966	0.00000	0.02786	0.02796	0.00000	0.02800
TE129	4.79E-02	1.69E-03	9.07E-07	0.00127	0.00417	0.11370	0.00000	0.11914	0.11957	0.00000	0.12000

Table 11A-9 (continued)
LIQUID SOURCE TERMS FROM SURRY UNITS 1 & 2 (PER UNIT BASIS)
WITHOUT STEAM GENERATOR BLOWDOWN PROCESSING

		Coolant Concentrations		Annual Releases To Discharge Canal							
Nuclide	Half-life (Days)	Primary (Micro Ci/ml)	Secondary (Micro Ci/ml)	Boron Rs (Curies)	Misc. Wastes (Curies)	Secondary (Curies)	Turb Bldg (Curies)	Total LWS (Curies)	Adjusted Total (Ci/yr)	Detergent Wastes (Ci/yr)	Total (Ci/yr)
Corrosion and Activation Products											
I130	5.17E-01	2.03E-03	1.77E-07	0.00000	0.00736	0.022112	0.00001	0.02950	0.02961	0.00000	0.03000
TE131M	1.25E+00	2.32E-03	2.38E-07	0.00011	0.01021	0.02985	0.00000	0.04018	0.04033	0.00000	0.04000
TE131	1.74E-02	1.17E-03	8.36E-07	0.00002	0.00187	0.10471	0.00000	0.10660	0.10699	0.00000	0.11000
I131	8.05E+00	2.41E-01	3.20E-05	0.02153	1.18990	4.01574	0.00312	5.23029	5.24915	0.00002	5.20000
TE132	3.25E+00	2.44E-02	2.77E-06	0.00825	0.11676	0.34691	0.00003	0.47196	0.47366	0.00000	0.47000
I132	9.58E-02	1.04E-01	1.46E-05	0.00851	0.19360	1.82326	0.00026	2.02562	2.03292	0.00000	2.00000
I133	8.75E-01	3.58E-01	3.61E-05	0.00060	1.48638	4.52219	0.00295	6.01212	6.03380	0.00000	6.00000
I134	3.67E-02	4.98E-02	8.93E-07	0.00000	0.00449	0.11191	0.00000	0.11640	0.11640	0.00000	0.12000
CS134	7.49E+02	2.18E-02	2.99E-06	0.07167	5.50343	0.37523	0.00003	5.95036	5.97182	0.00468	6.00000
I135	2.79E-01	1.89E-01	1.30E-05	0.00000	0.52153	1.62652	0.00069	2.14874	2.15649	0.00000	2.20000
CS136	1.30E+01	1.15E-02	1.37E-06	0.02321	2.85474	0.17198	0.00001	3.04995	3.06094	0.00000	3.10000
CS137	1.10E+04	1.57E-02	1.99E-06	0.05202	3.96259	0.24974	0.00002	4.26437	4.27975	0.00864	4.30000
BA137M	1.77E-03	1.72E-02	1.41E-05	0.04864	3.70502	1.77272	0.00002	5.52641	5.54634	0.00000	5.50000
BA140	1.28E+01	1.95E-04	2.49E-08	0.00023	0.00097	0.00312	0.00000	0.00433	0.00434	0.00000	0.00430
LA140	1.68E+00	1.38E-04	3.40E-08	0.00025	0.00072	0.00426	0.00000	0.00524	0.00526	0.00000	0.00530
CE141	3.24E+01	6.19E-05	7.86E-09	0.00010	0.00031	0.00098	0.00000	0.00139	0.00140	0.00000	0.00140
CE143	1.38E+00	3.70E-05	4.13E- 09	0.00000	0.00016	0.00052	0.00000	0.00068	0.00069	0.00000	0.00069

Table 11A-9 (continued)
LIQUID SOURCE TERMS FROM SURRY UNITS 1 & 2 (PER UNIT BASIS)
WITHOUT STEAM GENERATOR BLOWDOWN PROCESSING

		Coolant Concentrations		Annual Releases To Discharge Canal							
Nuclide	Half-life (Days)	Primary (Micro Ci/ml)	Secondary (Micro Ci/ml)	Boron Rs (Curies)	Misc. Wastes (Curies)	Secondary (Curies)	Turb Bldg (Curies)	Total LWS (Curies)	Adjusted Total (Ci/yr)	Detergent Wastes (Ci/yr)	Total (Ci/yr)
Corrosion and Activation Products											
PR143	1.37E+01	4.44E-05	5.51 E-09	0.00006	0.00022	0.00069	0.00000	0.00097	0.00098	0.00000	0.00098
CE144	2.84E+02	2.91E-05	5.06E-09	0.00006	0.00015	0.00063	0.00000	0.00084	0.00084	0.00187	0.00270
PR144	1.20E-02	3.53E-05	3.41E-08	0.00006	0.00015	0.00427	0.00000	0.00447	0.00449	0.00000	0.00450
All Others		0.00	0.00	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000
Total (Except Tritium)		1.41E+00	1.67E-04	0.31637	20.35739	20.91034	0.00748	41.59157	41.74157	0.02244	42.00000
Tritium Release		480	Curies Per Year								

Table 11A-10
LIQUID SOURCE TERMS FROM SURRY UNITS 1 & 2 (PER UNIT BASIS)
WITH STEAM GENERATOR BLOWDOWN PROCESSING [a]

		Coolant Concentrations		Annual Releases To Discharge Canal							
Nuclide	Half-life (Days)	Primary (Micro Ci/ml)	Secondary (Micro Ci/ml)	Boron Rs (Curies)	Misc. Wastes (Curies)	Secondary (Curies)	Turb Bldg (Curies)	Total LWS (Curies)	Adjusted Total (Ci/yr)	Detergent Wastes (Ci/yr)	Total (Ci/yr)
Corrosion And Activation Products											
CR51	2.78E+01	1.68E-03	2.11E-07	0.00257	0.00044	0.00003	0.00000	0.01104	0.01118	0.00000	0.01100
MN54	3.03E+02	2.74E-04	5.06E-08	0.00058	0.00138	0.00001	0.00000	0.00191	0.00198	0.00036	0.00230
FE55	9.50E+02	1.41E-03	1.77E-07	0.00271	0.00713	0.00002	0.00000	0.00987	0.000994	0.00000	0.00990
FE59	4.50E+01	8.84E-04	1.30E-07	0.00148	0.00448	0.00002	0.00000	0.00594	0.00599	0.00000	0.00600
CO58	7.13E+01	1.41E-02	1.80E-06	0.02492	0.07121	0.00023	0.00002	0.09639	0.09707	0.00144	0.09900
C060	1.98E+03	1.76E-03	2.27E-07	0.00340	0.00891	0.00003	0.00000	0.01235	0.01243	0.00313	0.01600
NP239	2.35E+00	1.09E-03	1.14E-07	0.00022	0.00512	0.00001	0.00000	0.00536	0.00540	0.00000	0.00540
Fission Products											
BR83	1.00E-01	4.97E-03	1.96E-07	0.00000	0.00498	0.00002	0.00000	0.00501	0.00505	0.00000	0.05000
BR84	2.21E-02	2.77E-03	3.29E-08	0.00000	0.00003	0.00000	0.00000	0.00003	0.00003	0.00000	0.00003
RB86	1.87E+01	7.47E-03	1.07E-08	0.00018	0.01848	0.00001	0.00000	0.01887	0.01901	0.00000	0.01900
RB88	1.24E-02	2.14E-01	1.38E-06	0.00000	0.00218	0.00171	0.00000	0.00389	0.00398	0.00000	0.00390
SR89	5.40E+01	3.09E-04	5.17E-08	0.00053	0.00156	0.00001	0.00000	0.00209	0.00211	0.00000	0.00210
SR90	1.03E+04	6.82E-06	1.26E-09	0.00002	0.00004	0.00000	0.00000	0.00006	0.00006	0.00000	0.00006
Y90	2.67E+00	1.09E-06	7.31E-10	0.00002	0.00001	0.00000	0.00000	0.00002	0.00002	0.00000	0.00002
SR91	4.03E-01	6.35E-04	4.74E-08	0.00000	0.00210	0.00001	0.00000	0.00211	0.00213	0.00000	0.00210

a. Steam generator blowdown treatment system is no longer used.

Table 11A-10 (continued)
LIQUID SOURCE TERMS FROM SURRY UNITS 1 & 2 (PER UNIT BASIS)
WITH STEAM GENERATOR BLOWDOWN PROCESSING [a]

		Coolant Concentrations		Annual Releases To Discharge Canal							
Nuclide	Half-life (Days)	Primary (Micro Ci/ml)	Secondary (Micro Ci/ml)	Boron Rs (Curies)	Misc. Wastes (Curies)	Secondary (Curies)	Turb Bldg (Curies)	Total LWS (Curies)	Adjusted Total (Ci/yr)	Detergent Wastes (Ci/yr)	Total (Ci/yr)
Fission Products											
Y91M	1.47E-02	3.82E-04	4.73E-08	0.00000	0.00136	0.00001	0.00000	0.00136	0.00137	0.00000	0.00140
Y91	5.88E+01	5.66E-05	7.73E-09	0.00011	0.00029	0.00000	0.00000	0.00040	0.00040	0.00000	0.00040
Y95	4.45E-01	3.31 E-05	3.09E-09	0.00000	0.00011	0.00000	0.00000	0.00011	0.00011	0.00000	0.00011
ZR95	6.50E+01	5.30E-05	7.71E-09	0.00009	0.00027	0.00000	0.00000	0.00036	0.00036	0.00000	0.00036
NB95	3.50E+01	4.42E-05	7.84E-09	0.00009	0.00022	0.00000	0.00000	0.00031	0.00031	0.00000	0.00031
MO99	2.79E+00	7.61E-02	1.08E-05	0.02037	0.36131	0.00136	0.00010	0.38314	0.38591	0.00000	0.39000
TC99M	2.50E-01	4.80E-02	2.91E-05	0.01948	0.28162	0.00364	0.00019	0.30493	0.30713	0.00000	0.31000
RU103	3.96E+01	3.98E-05	5.20E-09	0.00007	0.00020	0.00000	0.00000	0.00027	0.00027	0.00005	0.00032
RH103M	3.96E-02	4.76E-05	3.10E-08	0.00007	0.00020	0.00000	0.00000	0.00027	0.00027	0.00000	0.00027
RH106	3.67E+02	8.83E-06	1.26E-09	0.00008	0.00004	0.00000	0.00000	0.00006	0.00006	0.00086	0.00093
RH106	3.47E-04	1.08E-05	8.89E-09	0.00002	0.00004	0.00000	0.00000	0.00006	0.00006	0.00000	0.00027
TE125M	5.80E+01	4.56E-05	4.32E-09	0.00004	0.00013	0.00000	0.00000	0.00017	0.00017	0.00000	0.00170
TE127M	1.09E+02	2.47E-04	2.30E-08	0.00045	0.00125	0.00000	0.00000	0.00170	0.00171	0.00000	0.00360
TE127	4.92E-01	8.32E-04	1.60E-07	0.00045	0.00315	0.00004	0.00000	0.00362	0.00365	0.00000	0.00830
TE129M	3.40E+01	1.24E-03	1.57E-07	0.00197	0.00622	0.00004	0.00000	0.00822	0.00828	0.00000	0.00560
TE130	5.17E-01	2.03E-03	1.77E-07	0.00000	0.00736	0.00002	0.00001	0.00740	0.00745	0.00000	0.00750
TE131M	1.25E+00	2.32E-03	2.38E-07	0.00011	0.01021	0.00003	0.00000	0.01036	0.01043	0.00000	0.01000
TE131	1.74E-02	1.17E-03	8.36E-07	0.00004	0.00187	0.00010	0.00000	0.00199	0.00401	0.00000	0.00200

a. Steam generator blowdown treatment system is no longer used.

Table 11A-10 (continued)
LIQUID SOURCE TERMS FROM SURRY UNITS 1 & 2 (PER UNIT BASIS)
WITH STEAM GENERATOR BLOWDOWN PROCESSING [a]

		Coolant Concentrations		Annual Releases To Discharge Canal							
Nuclide	Half-life (Days)	Primary (Micro Ci/ml)	Secondary (Micro Ci/ml)	Boron Rs (Curies)	Misc. Wastes (Curies)	Secondary (Curies)	Turb Bldg (Curies)	Total LWS (Curies)	Adjusted Total (Ci/yr)	Detergent Wastes (Ci/yr)	Total (Ci/yr)
Corrosion and Activation Products											
T131	8.05E+00	2.41E-01	3.20E-07	0.02153	1.18990	0.00402	0.00318	1.21857	1.28739	0.00002	1.20000
TE132	3.25E+00	2.44E-02	2.77E-06	0.00025	0.11876	0.00035	0.00003	0.12539	0.12630	0.00000	0.13000
T132	4.58E-04	1.04E-01	1.46E-05	0.00851	0.19360	0.00182	0.00026	0.20419	0.20567	0.00000	0.21000
T135	8.75E-01	3.58E-01	3.61E-05	0.00060	1.48538	0.00452	0.00295	1.49445	1.50328	0.00000	1.50000
T134	3.67E-02	4.98E-02	8.93E-07	0.00000	0.00449	0.00011	0.00000	0.00460	0.00463	0.00000	0.00460
CS134	7.49E+02	2.18E- 02	2.99E-06	0.07167	5.50343	0.00175	0.00003	5.57888	5.61929	0.00468	5.60000
T135	2.79E-01	1.892E-01	1.30E-05	0.00000	0.52153	0.00165	0.00089	0.52385	0.52764	0.00000	0.53000
CS136	1.40E+01	1.15E-02	1.47E-06	0.02321	2.85474	0.00172	0.00001	2.87968	2.90054	0.00000	2.90000
CS137	1.10E+04	1.57E-02	1.99E-06	0.05202	3.98459	0.00250	0.00004	4.01713	4.04625	0.00864	4.10000
BA137M	1.77E-03	1.78E-02	1.41E-05	0.04864	3.70502	0.00177	0.00002	3.75546	3.78266	0.00000	3.80000
BA140	1.28E+01	1.95E-04	2.49E-08	0.00023	0.00097	0.00000	0.00000	0.00121	0.00121	0.00000	0.00120
LA140	1.68E+00	1.38E-04	3.40E-08	0.00023	0.00072	0.00000	0.00000	0.00098	0.00099	0.00000	0.00099
CE 41	3.24E+01	6.19E-05	7.86E-09	0.00010	0.00031	0.00000	0.00000	0.00041	0.00041	0.00000	0.00041
CE143	1.38E+00	3.70E-05	4.13E- 09	0.00000	0.00016	0.00000	0.00000	0.00017	0.00017	0.00000	0.00017
PR143	1.37+01	4.44E-05	5.51E-09	0.00006	0.00022	0.00000	0.00000	0.00028	0.00028	0.00000	0.00028
CE144	2.84E+04	2.91E-05	5.06E-09	0.00006	0.00015	0.00000	0.00000	0.00020	0.00024	0.00187	0.00210
PR144	1.20E-02	3.53E-05	3.41E-08	0.00006	0.00015	0.00000	0.00000	0.00021	0.00021	0.00000	0.00021

a. Steam generator blowdown treatment system is no longer used.

Table 11A-10 (continued)
LIQUID SOURCE TERMS FROM SURRY UNITS 1 & 2 (PER UNIT BASIS)
WITH STEAM GENERATOR BLOWDOWN PROCESSING [a]

		Coolant Concentrations		Annual Releases To Discharge Canal							
Nuclide	Half-life (Days)	Primary (Micro Ci/ml)	Secondary (Micro Ci/ml)	Boron Rs (Curies)	Misc. Wastes (Curies)	Secondary (Curies)	Turb Bldg (Curies)	Total LWS (Curies)	Adjusted Total (Ci/yr)	Detergent Wastes (Ci/yr)	Total (Ci/yr)
All Others		3.22E-04	3.53E-10	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000
Total (Except Tritium)		1.41E+00	1.67E-04	0.31537	20.35739	0.02963	0.00748	20.71086	20.86086	0.02244	21.00000
Tritium Release		480	Curies Per Year								

a. Steam generator blowdown treatment system is no longer used.

Table 11A-11
LADTAP INPUT DATA AND RESULTS -
MAXIMALLY EXPOSED INDIVIDUAL DOSE CALCULATIONS FOR SURRY UNITS 1 AND 2
(DEMINERALIZER RADWASTE SYSTEM WITHOUT
STEAM GENERATOR BLOWDOWN TREATMENT)

			Usage Rates (kg/yr or hr/yr)		
Exposure Pathway	Dilution Factor	Transit Time (hr)	Adult	Teen	Child
Fish ingestion	5	24	21.0	16.0	6.9
Invertebrate ingestion	5	24	5.0	3.8	1.7
Shoreline use	5	0	12.0	67.0	14.0

	Dose Results (mrem/yr per unit)[a]					
	Adults			Teenagers		
Exposure Pathway	Total Body	GI-LLI	Skin	Total Body	GI-LLI	Skin
Fish ingestion	1.15 (-1)[b]	1.74 (-2)	-	6.63 (-2)	1.27 (-2)	-
Invertebrate ingestion	6.12 (-2)	2.02	-	5.47 (-2)	1.59	-
Shoreline use	3.28 (-3)	3.28 (-3)	3.82 (-3)	1.83 (-2)	1.83 (-2)	2.13 (-2)
	1.79 (-1)	2.04	3.82 (-3)	1.39 (-1)	1.62	2.13 (-2)

a. Doses to other individuals and organs are smaller than those presented.

b. 1.15 (-1) = 1.15×10^{-1}.

Table 11A-12
LADTAP INPUT DATA AND RESULTS -
MAXIMALLY EXPOSED INDIVIDUAL DOSE CALCULATIONS FOR SURRY UNITS 1 AND 2
(MODIFIED LIQUID RADWASTE SYSTEM WITH
STEAM GENERATOR BLOWDOWN TREATMENT) [a]

			Usage Rates (kg/yr or hr/yr)		
Exposure Pathway	Dilution Factor	Transit Time (hr)	Adult	Teen	Child
Fish ingestion	5	24	21.0	16.0	6.9
Invertebrate ingestion	5	24	5.0	3.8	1.7
Shoreline use	5	0	12.0	67.0	14.0

	Dose Results (mrem/yr per unit)[b]					
	Adults			Teenagers		
Exposure Pathway	Total Body	GI-LLI	Skin	Total Body	GI-LLI	Skin
Fish ingestion	1.06 (-1)[c]	7.22 (-3)	-	6.06 (-2)	5.16 (-3)	-
Invertebrate ingestion	2.81 (-2)	5.60 (-1)	-	2.15 (-2)	4.40 (-1)	-
Shoreline use	3.06 (-3)	3.06 (-3)	3.57 (-3)	1.71 (-2)	1.71 (-2)	2.00 (-2)
	1.37 (-1)	5.70 (-1)	3.57 (-3)	9.92 (-2)	4.62 (-1)	2.00 (-2)

a. Steam generator blowdown treatment system is no longer used.

b. Doses to other individuals and organs are smaller than those presented.

c. $1.06\ (-1) = 1.06 \times 10^{-1}$.

Table 11A-13

GASEOUS SOURCE TERMS FROM SURRY UNITS 1 AND 2 (PER UNIT)

GASEOUS RELEASE RATE - CURIES PER YEAR

	PRIMARY COOLANT (MICROCI/GM)	SECONDARY COOLANT (MICROCI/GM	GAS STRIPPING SHUTDOWN	GAS STRIPPING CONTINUOUS	BUILDING VENTILATION REACTOR	BUILDING VENTILATION AUXILIARY	BUILDING VENTILATION TURBINE	BLOWDOWN VENT OFFGAS	AIR EJECTOR EXHAUST	TOTAL
KR-83M	2.251E-02	8.325E-09	7.0E+00	0.	0.	0.	0.	0.	0.	7.0E+00
KR-85M	1.173E-01	4.426E-08	3.9E+01	7.0E+00	1.0E+00	2.0E+00	0.	0.	2.0E+00	5.1E+01
KR-85	6.231E-02	2.337E-08	2.1E+01	2.5E+02	3.0E+01	1.0E+00	0.	0.	0.	3.0E+02
KR-87	6.438E-02	2.299E-08	2.1E+01	1.0E+00	0.	1.0E+00	0.	0.	0.	2.3E+01
KR-88	2.139E-01	7.874E-08	7.1E+01	6.0E+00	2.0E+00	5.0E+00	0.	0.	3.0E+00	8.9E+01
KR-89	5.179E-03	2.817E-09	2.0E+00	0.	0.	0.	0.	0.	0.	2.0E+00
XE-131M	8.252E-02	3.115E-08	2.7E+01	2.1E+02	2.9E+01	2.0E+00	0.	0.	1.0E+00	2.7E+02
XE-133M	2.139E-01	8.874E-08	7.1E+01	1.6E+02	2.0E+01	5.0E+00	0.	0.	3.0E+00	2.7E+02
XE-133	1.577E+01	5.865E-06	5.1E+03	2.5E+04	3.9E+03	3.3E+02	0.	0.	2.1E+02	3.3E+04
XE-135M	1.399E-02	5.185E-09	5.0E+00	0.	0.	0.	0.	0.	0.	5.0E+00
XE-135	3.695E-01	1.371E-07	1.2E+02	4.7E+01	9.0E+00	8.0E+00	0.	0.	5.0E+00	1.9E+02
XE-137	9.602E-03	3.662E-09	3.0E+00	0.	0.	0.	0.	0.	0.	3.0E+00
XE-138	4.731E-02	1.728E-08	1.6E+01	0.	0.	1.0E+00	0.	0.	0.	1.7E+01
TOTAL NOBLE GASES										3.6E+04
I-131	2.408E-01	3.207E-05	0.	0.	2.3E-03	3.0E-03	1.7E-03	1.6E-01	2.4E-02	1.9E-01
I-133	3.582E-01	3.611E-05	0.	0.	7.9E-04	5.7E-03	2.0E-03	1.8E-01	3.6E-02	2.2E-01
TRITIUM GASEOUS RELEASE			490	CURIES/YR						

0.0 APPEARING IN THE TABLE INDICATES RELEASE IS LESS THAN 1.0 CI/YR FOR NOBLE GAS, 0.0001 CI/YR FOR I

AR-41 25 CURIES/YR

C-14 8 CURIES/YR

S11A0001

Table 11A-13 (continued)
GASEOUS SOURCE TERMS FROM SURRY UNITS 1 AND 2 (PER UNIT)

AIRBORNE PARTICULATE RELEASE RATE-CURIES PER YEAR

NUCLIDE	WASTE GAS SYSTEM	BUILDING VENTILATION REACTOR	BUILDING VENTILATION AUXILIARY	TOTAL
MN-54	4.5E-05	6.0E-05	1.8E-04	2.9E-04
FE-59	1.5E-05	2.1E-05	6.0E-05	9.6E-05
CO-58	1.5E-04	2.1E-04	6.0E-04	9.6E-04
CO-60	7.0E-05	9.3E-05	2.7E-04	4.3E-04
SR-89	3.3E-06	4.7E-06	1.3E-05	2.1E-05
SR-90	6.0E-07	8.2E-07	2.4E-06	3.8E-06
CS-134	4.5E-05	6.0E-05	1.8E-04	2.9E-04
CS-137	7.5E-05	1.0E-04	3.0E-04	4.8E-04

S11A0002

Table 11A-14

MAXIMUM DOSES TO AN INDIVIDUAL RESULTING FROM GASEOUS EFFLUENTS FROM SURRY UNITS 1 AND 2 WITH STEAM GENERATOR FLASH TANK (MREM/YR PER UNIT

	Location 2.17 miles SSW	Location 3.75 miles NNW
	Total Body	Organ Dose (thyroid)
A. Radioiodines and particulates[a]		
Ground	7.80 (-4)[b]	3.11 (-4)
Ingestion of vegetables	5.01 (-2)	-
Inhalation	3.31 (-3)	9.96 (-2)
Milk	-	1.67
	5.42 (-2)	1.77
	Total Body	Skin
B. Noble gases		
Plume (1.53 miles S)	1.32 (-1)	3.53 (-1)
C. Air doses		
(1.53 miles S)	Annual beta 5.67 (-1) mrad/yr	Annual gamma 2.20 (-1) mrad/yr
(Site boundary 0.31 miles N)	Annual beta 15.4 mrad/yr	Annual gamma 6.26 mrad/yr

a. Maximum organ dose occurs to an infant.

b. 7.80 (-4) = 7.80×10^{-4}

Table 11A-15

MAXIMUM DOSES TO AN INDIVIDUAL RESULTING FROM GASEOUS EFFLUENTS FROM SURRY UNITS 1 AND 2 WITH STEAM GENERATOR BLOWDOWN COOLED (MREM/YR PER UNIT)

	Location 2.17 miles SSW	Location 3.75 miles NNW
	Total Body	Organ Dose (thyroid)
A. Radioiodines and particulates[a]		
Ground	7.09 (-4)[b]	2.81 (-4)
Ingestion of vegetables	4.97 (-2)	-
Inhalation	3.25 (-3)	2.08 (-2)
Milk	-	3.00 (-1)
	5.37 (-2)	3.21 (-1)
	Total Body	Skin
B. Noble gases		
Plume (1.53 miles S)	1.32 (-1)	3.53 (-1)
C. Air doses		
(1.53 miles S)	Annual beta 5.67 (-1) mrad/yr	Annual gamma 2.20 (-1) mrad/yr
(Site boundary 0.31 miles N)	Annual beta 15.4 mrad/yr	Annual gamma 6.26 mrad/yr

a. Maximum organ dose occurs to an infant.

b. 7.09 (-4) = 7.09×10^{-4}