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Chapter 11

RADIOACTIVE WASTE MANAGEMENT

The radioactive waste systems are designed to collect, process, and dispose of potentially radioactive wastes produced during the operation of the plant. These wastes are grouped as liquid, gaseous, or solid.

The liquid radwaste system is designed to process and return the collected liquid waste to the condensate storage tanks to the maximum extent practicable. The gaseous wastes are processed through a recombiner/charcoal delay system, monitored, and released to the atmosphere via the offgas stack. Solid wastes are packaged in suitable containers for offsite shipment and burial.

The liquid and gaseous effluents from the treatment systems are continuously monitored, and the discharges are terminated if the effluents exceed preset radioactivity levels.

The radioactive waste treatment system design discussed in this chapter will limit the radioactive releases to the environment form the DAEC to the as low as is reasonably achievable (ALARA) level.

11.1 SOURCE TERMS

11.1.1 BASIC DATA FOR SOURCE-TERM CALCULATIONS

The basic data for source-term calculations are as follows:

1. Operating power at which impact is to be analyzed is 1912 Mwt.

2. Weight of uranium loaded (first loading and equilibrium cycle): first core ~152,000-lb uranium. Approximately the same total weight will be maintained through reloading cycles.

3. Isotopic ratio in fresh fuel (first loading and equilibrium cycle): first core will contain 1.90% of uranium-235. Reload fuel contains up to 4.4% uranium-235.

4. The DAEC Design Basis offgas rate is 100,000 µCi/sec after 30 min delay, the typical rate is 5000 to 25,000 µCi/sec after 30-min delay.

5. The curie decontamination factor (DF) used is 61 and the krypton and xenon offgas discharge is approximately 410 µCi/sec. (See Table 11.3-6 for a detailed isotopic release rate.)

6. Mass of primary coolant in system

   a. Mass of primary coolant in reactor is 289,540-lb water, 9460-lb steam.

   b. Mass of primary coolant in recirculating system is 29,649 lb.

7. Steam conditions at turbine

Temperature	542.5°F
Pressure	983 psia
Flow	8,223,041 lb/hr (maximum)

8. Normal recirculation flow rate is 22,400,000 lb/hr.

9. The normal cleanup system flow rate is 70,000 lb/hr. The filter-demineralizer units are pressure precoat-type filters that use resinous fibers (Solka-floc) and finely ground mixed ion-exchange resins (Epifloc). The fiber-to-resin ratio can vary from 1:1 up to 5:1. Specific decontamination factors are not available for particular isotopes. However, an average decontamination factor of 10 is expected for particulate filtration and 100 for ion demineralization.

10. The expected performance of the expanded offgas system is described in Section 11.3.2. The main condenser design air inleakage is 18.5 cfm at 130°F. The condenser air ejector is a two-stage air ejector discharging to the offgas system. There are two condenser shells. The ejector discharge will pass through 37 tons of charcoal at 77°F.

11. The expected leak rate of primary coolant to the drywell is 0.5-gpm steam and 0.5-gpm unidentified reactor water. The drywell is planned to be purged once a year; however, the impact has been analyzed using four purges per year. The purge is passed through high-efficiency particulate absorber (HEPA) and deep-bed charcoal filters in the standby gas treatment system.

12. The expected leak rate of primary coolant to the reactor building is 1.0-gpm reactor water. The ventilation air flow through the reactor building is 69,500 cfm. There is an additional 3000 cfm from infiltration. The air flow is discharged through three reactor building discharge stacks. The air is not filtered or otherwise treated before discharge; however, it is monitored.

13. The expected leak rate of steam to the turbine building is 5.0 gpm. The ventilation air flow through the turbine building is 37,500 cfm in the winter and 109,500 cfm in the summer. There is an additional 3500 cfm from infiltration. The air is discharged through the reactor building stacks in the winter and through the reactor building stacks as well as through eight turbine building roof vents during the summer. The air is not filtered or treated before discharge; however, it is monitored.

14. Primary steam is the source of steam used in the turbine gland seals.

    Effluent steam from the gland seals is discharged to the gland steam condenser. The condensate drains to the main condenser and the noncondensibles are discharged to a short delay in and then to the 100-m stack. Approximately a 2-min delay time exists between steam leaving the reactor vessel and subsequent release to the environment.

15. The estimated average gallons per day and microcuries per cubic centimeter are listed below for these categories of liquid waste: high-level wastes (e.g. "clean" or low-conductivity waste and equipment drains): "dirty" wastes (e.g., floor drain wastes, high-conductivity wastes, and laboratory wastes; chemical wastes; and laundry, decontamination, and washdown wastes:

	Liquid Discharges	
Waste	Gal/day	$\mu Ci/cm^3$
Equipment drains	0	--
Floor drains	2040	$8x10^{-6}$
Chemical wastes	500	$4x10^{-5}$
Detergent wastes	300	$1x10^{-5}$

For the above-listed wastes, the following are provided:

a. Number and capacity of collector tanks.

b. Fraction of water to be recycled or factors controlling decision.

c. Treatment steps, including number, capacity, and process decontamination factor for each principal nuclide for each step. If step is optional, factors controlling decision are stated.

d. Decay time from primary loop to discharge.

e. How waste concentrate (filter cake, demineralizer resin, evaporator bottoms) is handled. Total volume or weight and curies per day or year is given.

Equipment Drains

a. Waste collector tank (one), 10,000 gal. Radwste surge tanks, 40,000 gal, and 70,000 gal.

b. One-hundred percent to be recycled.

c. Filtration and demineralization. The overall decontamination factor for the filter an demineralizer is approximately 100. The decontamination factor for individual radionuclides is unkown. The decontamination factor will vary in actual operation since it is a function of inlet concentration of the soluble and insoluble species present.

d. Twelve hours.

e. Waste concentrate is dewatered or solidified. The estimated weight and volume are 63,000 lb/yr and 2200 ft^3/yr, respectively, for all sludges and resins for the plant. The total isotopic inventory of these solids is expected to be about 1000 Ci/yr.

Floor Drains

a. Floor drain collector tank (one), 10,000 gal.

b. Seventy percent recycled. The impact of discharges has been made assuming 30% floor drain, 100% chemical waste, and 100% detergent drains discharged.

c. Filtration and demineralization. The overall decontamination factor for the filter and demineralizer is approximately 100. The decontamination factor for individual radionculides is unknown. This factor will vary in actual operation since it is a function of inlet concentration of the soluble and insoluble species present.

d. Twelve hours

e. Waste concentrate is dewatered or solidified. The estimated weight and volume are 63,000 lb/yr and 2200 ft^3/yr, respectively, for all sludges and resins for the plant. The total isotopic inventory of these solids is expected to be about 1000 Ci/yr.

Chemical Wastes

a. Chemical waste tank (one), 4000 gal.

b. None recycled. The impact of discharges has been made assuming 30% floor drain, 100% chemical waste, and 100% detergent drains discharged.

c. Normal method of processing is through the floor drain system. However, high-conductivity chemical wastes are received by the chemical waste tank and are processed by filtration.

d. Twelve hours.

e. The water concentrate is processed with spent resin in high integrity containers (HICs) or steel liners. Weight, volume, and isotopic inventory included in total cited for equipment and floor drains.

Detergent Wastes

a. Detergent drain tanks (two), 1000 gal each.

b. None recycled. The impact of discharges has been made assuming 30% floor drain, 100% chemical waste, and 100% detergent drains discharged.

c. Detergent wastes will be treated in the same manner as chemical wastes to the maximum extent practicable.

d. Twelve hours.

e. Same as chemical wastes above.

16. For the condensate demineralizers, the flow rate, type of resin used, and expected backwash and regeneration frequency, and expected decontamination factor for each principal nuclide are as follows:

Flow rate	3370 gpm (For 5 demineralizers in service) 4213 gpm (For 4 demineralizers in service)
Type of resin	Powdered Resin
Expected backwash frequency	As needed, to maintain system differential pressure and effluent water chemistry
Regeneration frequency	No regeneration
Expected decontamination factor for each nuclide	Specific decontamination factors are not available for particular isotopes. However, an average decontamination factor of 10 is expected for particulate filtration and a df of 100 for ion exchange.

17. The normal dilution flow rate for liquid effluents is 6000 to 24,000 gpm as necessary and 3.16×10^9 to 1.2×10^{10} gal/yr as necessary.

11.1.2 PRIMARY RADIOACTIVE GAS SOURCES

The four primary potentially radioactive gas sources are described below.

11.1.2.1 Process Offgas (Steam Jet Air Ejector)

Noncondensible radioactive offgas is continuously removed from the main condenser by the air ejector during plant operation. This is the major source and is larger than all other sources combined. The air ejector offgas will normally contain activation gases, principally Nitrogen-16, Oxygen-19, and Nitrogen-13. The Nitrogen-16 and Oxygen-19 isotopes have short half-lives and quickly decay. The 10-min half-life Nitrogen-13 isotope is present in small amounts, which are further reduced by decay. The air ejector offgas also contains the radioactive noble gas parents of biologically significant Strontium-89, Strontium-90, Barium-140, and Cesium-137 isotopes. The concentration of these noble gases depends on the usually extremely small amount of tramp uranium in the coolant and on the cladding surfaces, and the number and size of fuel cladding leaks.

Radioactive-particulate daughters are retained on the HEPA filters and on the charcoal. The offgas is discharged to the environs by the plant stack. The activity of the gas entering and leaving the offgas treatment system is continuously monitored. Thus, the system performance is known to the operator at all times.

11.1.2.2 Mechanical Vacuum Pump Offgas

During unit startup, air is removed from the main condenser by a mechanical vacuum pump. The mechanical vacuum pump exhaust is discharged to the off gas stack. The mechanical vacuum pump will normally be in service only during startup and shutdown when little or no radioactive gas is present. It will be manually isolated on the receipt of a high-radiation signal from the offgas stack radiation monitors.

11.1.2.3 Drywell Ventilation Gas

The drywell air is exposed to neutron fluxes around the reactor vessel, which results in some activation products. Activity is also introduced into the drywell atmosphere by the drywell sumps and by the primary system relief valves when they vent to the suppression chamber. The drywell forms a closed system that may be purged with normal reactor building air, if necessary, when access is required. The drywell can also be vented during the plant startup to accommodate the expansion of air with increasing temperature. This air is discharged through the standby gas treatment system.

11.1.2.4 Gland Seal Condenser Offgas

The gland seal condenser exhauster discharges into a separate holdup piping system. There is a holdup of approximately 2 min to permit the decay of the short-lived radioactive gases present. These are principally Nitrogen-13, Nitrogen-16, Nitrogen-17, and Oxygen-19. The release rate of radioactive gas is less than 0.1% of that from the air ejector offgas system. The gland seal exhaust gas flows past the elevated release point radiation monitors before release so that its contribution to the release rate is included in the measured total.

11.1.2.5 Other Potentially Radioactive Gases

At times, it is desirable to vent certain tanks and discharge gases from specific laboratories and building service areas to the roof vent after allowing suitable delay time for radionuclide decay. These additions have low-activity levels and add small increments to the total quantity of radioactive gas requiring treatment. In all cases, maximum activities from these sources are examined to ensure that the vent discharge is safely below the established limits.

11.2 LIQUID WASTE MANAGEMENT SYSTEM

11.2.1 DESIGN BASES

11.2.1.1 Power Generation Objectives

The objectives of the liquid waste management systems are the following:

1. Liquids that potentially contain radioactive materials are collected and processed in the radwaste system.

2. Waste concentrations before release from the liquid radwaste system will permit compliance with limits established for the plant before discharge to the river.

3. The various types of liquid wastes are treated to recycle water to the plant for reuse, within the limitations of water inventory balance and reactor water-quality specifications.

4. Plant-contributed radiation doses to individuals resulting from liquid waste releases to the environment will be as low as reasonably achievable not to exceed 3.0 mrem to the total body during any calendar year or 10.0 mrem to any organ during any calendar year.

5. Wastes containing a high content of solids are processed so that they can be packaged for safe transport and offsite disposal as solids containing no free liquid.

11.2.1.2 Power Generation Design Bases

The design bases of the liquid waste management system are the following:

1. The liquid radwaste system is designed so that for normal operation the estimated annual average radiation dose to the whole body or to any organ of an individual at any point on the site boundary from radioactive materials in the liquid wastes discharged from the plant does not exceed 3.0 mrem to the total body during any calendar year or 10.0 mrem to any organ during any calendar year.

2. The system is able to process the expected quantities of liquid wastes without impairing operation or availability of the plant during expected operational occurrence conditions.

11.2.1.3 Safety Design Basis

The liquid radwaste system is designed so that any quantities of liquid radwastes inadvertently released result in radiation levels within annual exposure limits of 10 CFR 20.

11.2.1.4 Decontamination Factors

The design decontamination factors used for some of the DAEC liquid radwaste equipment to predict equipment and system performance are as follows:

Equipment	Predicted Decontamination Factor
Filter	
Corrosion products	2
Fission products	1
Demineralizer	
Corrosion products	50
Fission products	100

11.2.1.5 Codes and Standards

The liquid radwaste system equipment is designed to codes and standards given in Tables 3.2-2 and 3.2-4 for "Components Ordered Before January 1, 1970." In addition, the waste collector filter and waste demineralizer vessels are designed per ASME Code, Section III.C. Further information on codes and standards commitments are discussed in UFSAR Section 1.8.33.

11.2.2 SYSTEM DESCRIPTION

The liquid radwaste system collects, monitors, processes, stores, and disposes of radioactive liquid wastes. The liquid radwaste system equipment and flow paths are shown in Figure 11.2-1. The following are included in the system:

1. Piping and equipment drains carrying potentially radioactive wastes.

2. Floor drain systems in controlled access areas that may contain potentially radioactive wastes.

3. Tanks and sumps used to collect potentially radioactive wastes.

4. Tanks, sumps, piping, pumps, process equipment, instrumentation, and auxiliaries necessary to collect, process, store, and dispose of potentially radioactive wastes.

Expected annual liquid volume total for floor drain, detergent, and chemical wastes is 2,873,000 gal.

Figures 11.2-2, 11.2-3, 11.2-4 and 11.2-5 are piping and instrumentation diagrams for the liquid radwaste system.

Current operating procedures provide for both chemical and detergent wastes to be processed through the floor drain system or solidified and disposed of as solid waste.

The normal method of processing chemical waste is through the radwaste floor drain system. If for any reason the waste cannot be handled by the floor drain system, chemical waste is processed through the chemical waste filter, pumped to the chemical waste sample tank, and sent to the discharge pipe by the route described below. 2015-010

Equipment is selected, arranged, and shielded to permit operation, inspection and maintenance with acceptable personnel exposures. For example, sumps, pumps, valves and instruments that may contain radioactivity are located in controlled access areas. Tanks and

processing equipment that may contain significant quantities of radioactive material are shielded.
2015-010 | The operation of the radwaste system is essentially manual start/automatic stop.

Protection against accidental discharge is provided by instrumentation for the detection and alarm of abnormal conditions and procedural controls. The radwaste facility arrangement and the methods of waste processing provide a substantial degree of immobility of the wastes within the plant. These provisions ensure that, in the event of a failure of the liquid waste system equipment or errors in the operation of the system, the potential for inadvertent release of liquids is small. The immobility of wastes is further accomplished by collecting solids on filters and demineralizer resins.

The liquid radwaste system is divided into several subsystems so that the liquid wastes from various sources can be kept segregated and processed separately. Cross-connections between the subsystems provide additional flexibility for the processing of the wastes by alternative methods. The liquid radwastes are classified, collected, and treated as high purity, low purity, chemical, detergent, sludge, or spent resins. The terms "high purity" and "low purity" refer to the conductivity and not the radioactivity.

The liquid radwaste system design provides for the filtration and demineralization of both waste collector (high purity) and floor drain (low purity) effluents. Radioactive liquids are recycled within the plant to the extent practicable. The liquid radwaste systems are used to ensure that levels of radioactive materials in liquid effluents are as low as reasonably achievable as discussed in Section 11.2.2.7.

Organics in the radioactive liquids may be processed by Ultra Violet Ozone (UVO_3) Treatment System capable of reducing 1000 ppb total organics contents (TOC) ethylene glycol in radioactive liquids to less than 100 ppb TOC at a combined system flow rate of greater than 50 GPM. The UVO_3 Treatment System consists of two 500 gram per hour ozone generators and three 3795 watts total ultra-violet (and ozone) contact chambers.

DAEC operating experience has revealed that from time to time it is necessary to employ temporary filtration or processing equipment to supplement the processing capability of the permanent Liquid Radwaste System. Such systems may include the use of equipment designed to address intrusions of liquid waste streams high in the levels of organics, conductivity, turbidity or other waterborne chemical agent. In circumstances where temporary equipment is utilized, such equipment is either designed in a manner to be consistent with the pressure rating of the Liquid Radwaste System or pressure regulating devices will be used to ensure that the pressure does not exceed the design pressure of the temporary equipment. The effluents from the temporary equipment are returned to the Radwaste System for final processing prior to transfer to the Condensate Storage System or environmental release.

The release of all liquid radwaste to the environment is via the [redacted] 2-in. discharge line beginning in Figure 11.2-4, teeing into the [redacted] 24-in. radwaste dilution flow line shown in Figure 11.2-6, and continuing in Figure 11.2-7. The radwaste dilution line terminates at the dilution structure. The dilution line, dilution structure, and discharge from the dilution structure to the river via the discharge pipe are shown in detail in Figures 11.2-8, 11.2-9 and 11.2-10. 2015-010

The auxiliary turbine building floor drain sump is connected to the radwaste drain system, which can be discharged to the river. Discharge lines coming into the auxiliary turbine building floor drain sump are the general floor area drain from the south end of the turbine building. Discharge lines coming into the chemical waste sump are from the makeup demineralizer floor drain (low-curb area), backwash line from the makeup demineralizer, floor drain from the neutralizing tank (high-curb area), flush line from acid and caustic tanks (by way of neutralizing tank), and the acid and caustic tank sump (by way of neutralizing tank/high-curb area). In addition, the auxiliary turbine building floor drain and chemical waste sumps are connected by a cross-connection with a shear valve near the bottoms of the sumps.

To preclude an accidental release, the system is designed as follows:

1. The cross-connection between the chemical waste sump and the auxiliary turbine building floor drain sump is eliminated by filling the cross-connection with grout.

2. The auxiliary turbine building floor drain sump is isolated from the normal waste drain system by rerouting the piping so that the sump pumps into the turbine building radwaste sump system.

3 The chemical sump pumps are connected to the normal waste drains by the addition of necessary piping and valves.

4. The drain line from the neutralizing tank to the auxiliary turbine building floor drain sump has been capped. It now drains only to the chemical waste sump.

5. A lock is installed on the discharge transfer valves used to transfer waste from the chemical waste sump to the normal drain system and the breakers on the pumps are turned off. The key for this lock is under the control of the Control Room Supervisor who will call the Radiation Protection Department for sampling for solids and pH before release to the normal drain system. This is also covered by a written procedure.

6. A two foot high retaining wall surrounds the auxiliary turbine building floor drain sump to prevent overflow from either sump from entering the other sump.

7. A lock has been installed on the Turbine Building waste sump pumps discharge valves to prevent pumping into the normal waste system. These will remain permanently locked closed and the sump will be emptied by the use of barrels or transferred to Radwaste system floor drains. This is covered by a written procedure. The Turbine Building oil sump pump discharge has been capped and a hose fitting installed to allow contents of this sump to be transferred into barrels or to Radwaste system floor drains.

It is not a normal operating practice to release any effluent containing radioactivity to the river by way of the auxiliary turbine building floor drain sump. Chemical waste can be discharged to the river, but only after it has been properly sampled by the Radiation Protection Department and permission given by the Control Room Supervisor to close the sump pump breakers and unlock the discharge transfer line sumps from the chemical waste sump to the normal drain system. Discharges from the oily waste sumps are accomplished as described in Item 7 above.

The design of the liquid radwaste system incorporates floor drain demineralization, the filtration of chemical wastes, and the ability to recycle liquids to the maximum extent practicable.

The design of the DAEC liquid radwaste system fully satisfies the requirement that releases of radioactivity be reduced to the lowest practicable level.

All process piping designated to carry radioactive materials is routed in shielded pathways to meet the criteria for radiation zones as specified in Section 12.3.1.1.

11.2.2.1 High-Purity Wastes

High-purity (low conductivity) liquid wastes are collected in the waste collector tank from the following sources:

1. Drywell equipment drain sump.

2. Reactor building equipment drain sump.

3. Radwaste building equipment drain sump.

4. Turbine building equipment drain sump.

5. Reactor water cleanup system.

6. Residual heat removal (RHR) system.

7. Decantate from cleanup phase separators.

8. Spent-fuel-pool cooling and demineralizer system.

9. Decantate from condensate phase separators.

10. Condensate storage tank drain and overflow.

11. Decantate from waste sludge tank.

The high-purity wastes are processed by filtration and ion exchange through the radwaste filters and demineralizers. The UVO_3 treatment system may be used for TOC reduction. After processing, the liquid is received in the tank where it is sampled. Then, if it is satisfactory for reuse, it is transferred to the condensate storage tank as makeup water.

If the analysis of the sample reveals water not meeting specification for reuse, it is returned to the system for additional processing by the filter-demineralizer train, or to the radwaste surge tanks if processing capacity is not immediately available. Infrequently, treated high-purity wastewater meeting 10 CFR 20 limits for disposal after dilution may be discharged from the plant because of excess water inventories or expected operational occurrences.

11.2.2.2 Low-Purity Wastes

Low-purity (moderate conductivity) liquid wastes are collected in the floor drain collector tank from the following sources:

1. Drywell floor drain sump.

2. Reactor building floor drain sump.

3. Radwaste building floor drain sump.

4. Turbine building floor drain sump.

5. Decantate from floor drain sludge tank.

6. Waste sample station drain.

7. RHR system drains.

8. LLRPSF processing area floor drain sumps.

9. LLRPSF storage area sump (routed to the floor drain sludge tank).

These wastes generally have low concentrations of radioactive impurities. Processing consists of filtration, ion exchange, or UV/O_3 Treatment. After processing the liquid is transferred to a sample tank for sampling and analysis. Alternately, these wastes may be transferred to various spent resin tanks where unused ion exchange capacity of the spent resin is utilized to pre-treat these wastes prior to filtration and ion exchange through either the high purity or low purity filtration trains. Some small volumes of liquid waste collected in drums is desiccated within the LLRPSF.

Normally, treated low-purity wastes will meet the specifications of water quality used in the plant and, if the water inventory of the plant permits, they are returned to the condensate storage tank for reuse. The wastes that do not meet the plant water-quality specifications on the basis of activity and/or conductivity are recycled to the collector tank or recycled through the ion-exchange unit. Infrequently, when the water inventory of the plant does not permit, treated low-purity waste is sampled and diluted before disposal from the plant to ensure that 10 CFR 20 limits are met.

11.2.2.3 Chemical Wastes

Chemical wastes are collected in the chemical waste tank and come from the following sources:

1. Cask cleaning drains.

2. Reactor and turbine building decontamination drains.

3. Laboratory drains.

4. Shop decontamination solutions.

5. Demineralizer decontamination solutions.

The normal method of processing chemical wastes is through the radwaste floor drain system. If it is determined that the treated waste meets the discharge criteria after dilution, it maybe discharged to the effluent line. The waste that does not meet the discharge criteria is recycled for reprocessing. Solidified wastes are processed as solid radwaste.

11.2.2.4 Detergent Wastes

Detergent wastes that are collected in the detergent drain tanks are from the following sources:

1. Shop regulated drain.

2. Personnel decontamination.

3. Turbine washdown area drain.

Detergent wastes are of low radioactivity concentration ($<10^{-5}$ μCi/ml). These wastes are treated in the same manner as chemical wastes to the maximum extent practicable. Plant laundry may be done at either an onsite facility or shipped to an offsite facility.

The detergent wastes and treated chemical wastes represent excess water in the plant water balance, and as such may be wastes discharged to the effluent line.

11.2.2.5 Sludges

Expended filter-demineralizer ion-exchange resins from the cleanup and condensate filter-demineralizers are removed when necessary by backwashing. Cleanup system sludges and sludge from the condensate system are collected in their respective phase separator where excess clear water is decanted and the sludge is accumulated. The fuel pool filter-demineralizer and waste collector filter are backwashed to the waste sludge tank. The floor drain filter is backwashed to the floor drain sludge tank. The accumulated resins and sludges are processed through the solid radwaste system.

11.2.2.6 Spent Resins

Expended ion-exchange resins from the waste and floor drain demineralizers are backwashed to the spent resin tank or the waste sludge tank of the solid radwaste system. The resins are then stored until transferred within the solid radwaste system for processing.

11.2.2.7 Means for Keeping Radioactive Discharges As Low As Reasonably Achievable

11.2.2.7.1 System Features

Liquid radwastes are received and processed in the subsystems described above. To ensure the operability of each of these systems, thus ensuring that wastes are processed by the treatment methods provided, the following system features are included:

1. Processing equipment is designed and selected to minimize maintenance requirements and the equipment is shielded so that it can be maintained without interfering with the operations of other subsystems.

2. Low-purity and high-purity waste filters and demineralizers are cross-connected so that each filter or demineralizer can be used in place of the other, if necessary, to maintain process continuity.

3. Major liquid subsystem pumps are cross-connected for maintainability and so that the outage of a pump does not prevent subsystem continuity.

4. Because the subsystems are batch systems rather than continuous systems and are preceded by collection tanks, time is available to accumulate wastes during the maintenance of downstream equipment or during filter backwashing and resin replacement. The radwaste surge tanks are available to accumulate certain wastes and thus provide time for maintenance.

5. Certain operations are also subject to scheduling and can be delayed in the event of mechanical problems. Examples are as follows:

    a. Transfer of liquid from cleanup phase separators to waste collector tank and transfer of solids for solid waste processing.

    b. Transfer to and from the waste sludge phase separator and spent-resin tank.

    c. Loading of sludges into containers can be delayed; however, the packaging of solid wastes is out of the path of usual wastewater processing.

6. Filter backwashing and precoating are part of the normal operating procedure for which cycle time has been allowed in the design.

7. Waste collector and floor drain demineralizer resin replacement is an infrequent operation. The essential factor in reducing outage time is to maintain an appropriate resin inventory at the plant for resin replacement. Resins can be replaced in less than one shift.

8. Radwaste system design can accommodate abnormal volumes of wastes resulting from refueling and maintenance activities, or from certain kinds of malfunctions such as increased valve seal and pump seal leakages, and opening of system relief valves (not reactor relief valves which are routed to the Torus). The system design will also handle periodic testing of emergency core cooling systems. From a radioactivity content viewpoint, the system is designed to accommodate design-basis fuel leaks.

The principal administrative areas involved in maintaining an operational system are daily planning of radwaste processing, the control of the reactor water inventory, and carrying out a preventive maintenance program.

Radwaste system planning ensures that wastes are processed in a timely manner. Timeliness ensures that reactor operations and maintenance activities (draining, flushing, decontamination, etc.) are coordinated so as not to impose unusual, unexpected quantities of water on the radwaste system.

The reactor water inventory is controlled to minimize the need for discharging wastewater because of excessive inputs via the makeup system. The planning and water inventory control activities are also useful in detecting abnormal inputs to radwaste and thus revealing causes of such inputs for correction.

The preventive maintenance program has the obvious objective of minimizing unplanned equipment conditions that would affect radwaste performance. The cross-connections (noted above) accommodate such outages in critical flow paths.

11.2.2.7.2 Procedural Controls

The following primary procedural controls have been implemented to ensure that all liquid radwaste management equipment performs as designed:

1. Daily planning of radwaste processing. The liquid radwaste processing requirements are reviewed on a daily basis to ensure that an overloading of processing components of this radwaste system does not occur.

2. Reactor water inventory control. Procedures for providing makeup water to the reactor coolant system via the makeup water system include control measures for precluding excessive inputs to reactor coolant. In this manner, the need for discharging radioactive liquid waste to the radwaste system can be minimized, thus ensuring that the use of liquid radwaste management equipment is not extended beyond design limits, and a more effective maintenance program can be implemented.

3. Preventive maintenance program. A preventive maintenance program has been implemented to ensure that malfunctions or a breakdown of components of the liquid radwaste system are minimized.

In conjunction with the above primary procedural controls, the following specific procedures have been implemented:

1. The liquid radwaste system is designed for the cross-connection of filters, demineralizers, pumps, etc. These cross-connections to alternate components are used while necessary maintenance is being performed on a malfunctioning unit.

2. Batch-sampling procedures are implemented to analyze the radioactivity content of liquid waste before and after treatment in a processing component to verify the effectiveness of the component.

3. Liquid waste solutions are retained in collection tanks, if necessary, until proper maintenance on a malfunctioning downstream processing component is completed. This maintenance also includes filter backwashing and demineralizer resin replacement.

4. Process radiation monitor operation, calibration, and alarm setpoints are thoroughly checked and maintained to ensure proper functioning. These monitors, which annunciate in the control room, provide an additional check (following batch sampling and analyses) before discharge to the environment.

5. Operators are instructed to closely observe the annunciators on the radwaste control panels during batch processing and discharging of liquid effluents. Operator corrective action is required following any alarm. Local instrumentation is periodically checked during these periods to determine any abnormalities.

11.2.2.8 Power Generation Evaluation

Treated high-purity and low-purity radwastes are normally routed to condensate storage to the maximum extent practicable, consistent with reactor water inventory and reactor water-quality requirements. Chemical wastes are normally processed in the same manner. However, chemical wastes may be processed through the filter and discharged to the effluent line after sampling and analysis to ensure that the discharge is within 10 CFR 20 limits. Detergent wastes and excess treated floor drain wastes are discharged to the effluent line in the same manner as chemical wastes.

The effluent discharge from the sample tank to the discharge pipe is monitored by taking batch samples; records of the volumes and concentration levels are retained. A process monitoring system is provided to indicate high radiation levels in the release to the discharge pipe. On the annunciation of the high-level alarm, the release of the liquid radwastes is terminated.

The processing equipment is located within a concrete building to provide secondary enclosures for the wastes in the event of leaks or overflows. Tanks and equipment that may contain significant quantities of radioactive material are shielded. Process lines that penetrate shield walls are routed to prevent a direct radiation path from the tanks or equipment for which shielding is required. The waste system is controlled from a local panel in the radwaste control room.

The radioactivity concentrations in the discharge system are as low as reasonably achievable and well within the guideline limits of 10 CFR 20. The components of the liquid radwaste system are sized to collect and process the volume of liquid radwaste generated from the reactor under normal power operation and expected operational occurrences.

11.2.2.9 Safety Evaluation

The concrete reactor and radwaste buildings and the LLRPSF retain and return to the system any spills or leaks from the liquid radwaste system for additional processing. These buildings are able to handle a major leak in the largest tank without permitting significant quantities of the liquid to escape off the site. A major leak in the radwaste system, such as a tank rupture, would result in a dose to an individual at the plant perimeter not exceeding the annual limits of 10 CFR 20.

Because leaks or spills from the liquid radwaste system will go into the radwaste building, the LLRPSF, and/or the reactor building, they will not cause doses at the plant boundary exceeding the annual limits of 10 CFR 50, Appendix I. Also, the system is monitored for inadvertent discharge of radioactive waste (see Section 11.5).

11.2.2.10 Inspection and Testing

The liquid radwaste system normally operates on an as-required basis during operation of the nuclear plant and thereby demonstrates operability without special inspections or testing.

11.2.2.11 Instrumentation and Control

System operation is controlled from a local control panel in the radwaste building. Instrumentation including alarms is provided for both process control and for the detection and alarm of abnormal conditions. The various alarms located at the local control panel provide signals of specific conditions. Indications and general alarms are also provided in the plant control room.

Offsite discharge is under operator control. Two console-operated parallel-flow control valves control flow at fast and slow rates. Console-operated, fail-closed shutoff valves are provided in the sample tank discharge and in the discharge to the effluent line and are closed if the radioactivity being discharged exceeds a preset limit. Discharge to the effluent line is prevented if there is not sufficient dilution water flow available. This is done by interlocking the diluent flow with the shutoff valves.

11.2.2.12 Design Pressures, Temperatures, and Material

Design pressures, temperatures, and materials for all major liquid radwaste system components are as follows:

1. Waste Demineralizer and Floor Drain Demineralizer

   Construction materials: stainless steel.
   Design pressure: 150 psig.
   Design temperature: 150°F.
   Capacity 35 ft^3

2. Condensate Phase Separator
   Construction materials: carbon steel phenolic-lined tank; stainless steel internal piping and eductor.
   Design pressure: atmospheric.
   Design temperature: 250°F.
   Capacity 12,500 Gal (2)

3. Cleanup Phase Separator

   Construction material: stainless steel tank, internal piping and eductor.
   Design pressure: atmospheric.
   Design temperature: 250°F.
   Capacity 4,500 Gal (2)

4. Chemical Waste Filter

   Construction material: stainless steel used for shell, tube shell, and nozzles.
   Design pressure: 190 psig.
   Design temperature: 175°F.

5. Reagent Addition Pump

   Construction material: stainless steel used for all wetted parts.
   Design pressure: 15 psig.
   Design temperature: 100°F.

6. Detergent Drain Filter

   Construction materials: carbon steel vessel and nozzle flanges and stainless steel internals.
   Design pressure: 75 psig.
   Design temperature: 150°F.

7. Waste Collector Filter and Floor Drain Filter
   Construction materials: carbon steel Plasite-lined vessel and stainless steel filter elements.
   Design pressure: 150 psig.
   Design temperature: 150°F filter element.
   Design temperature: 200°F vessel.
   Capacity 120 ft^3

8. Waste Precoat Tank and Filter Aid Tank

   Construction material: carbon steel Plasite-lined tank.
   Design pressure: atmospheric.
   Design temperature: 150°F.

9. Radwaste Filter and Floor Drain Filter Holding Pump Coolers

   Construction material: carbon steel.
   Design pressure: 150 psig.
   Design temperature: 200°F.

10. Radwaste Filter and Floor Drain Filter Holding Pumps

    Construction material: carbon steel.
    Design pressure: 150 psig.
    Hydrotest pressure: 300 psig.
    Design temperature: 200°F.

11. Waste Precoat Pump

    Construction material: carbon steel.
    Design pressure: 150 psig.
    Hydrotest pressure: 300 psig.
    Design temperature: 150°F.

12. Waste Filter Aid Pump and Floor Drain Filter Aid Pump

    Construction material: cast steel.
    Design pressure: 150 psig.
    Design temperature: 200°F.

13. Filter Aid Agitator

    Construction material: stainless steel.
    Design pressure: atmospheric.
    Design temperature: 150°F.

14. Waste Precoat Agitator

    Construction material: stainless steel.
    Design pressure: atmospheric.
    Design temperature: 150°F.

15. Dust Evacuators

    Construction material: carbon steel collection drums.
    Design pressure: atmospheric.
    Design temperature: ambient.

16. Waste Collector Pump

    Construction material: carbon steel.
    Design pressure: 200 psig.
    Design temperature: 150°F.

17. Floor Drain Collector Pump

    Construction material: carbon steel.
    Design pressure: 175 psig.
    Design temperature: 150°F.

18. Chemical Waste Pump

    Construction material: Type 316 stainless steel.
    Design pressure: 175 psig.
    Design temperature: 150°F.

19. Waste Sample Pumps, Floor Drain Sample Pump, Condensate Backwash Transfer Pump, and Condensate Sludge Mixing Pump

    Construction material: stainless steel.
    Design pressure: 150 psig.
    Design temperature: 150°F.

20. Spent-Resin Pump, Floor Drain Sludge Pump, Waste Sludge Pump, Cleanup Sludge Pump, Detergent Drain Pump, Chemical Waste Sample Pump, Waste Surge Pump, and Detergent Holding Tank Transfer Pump

    Construction material: carbon steel.

Design pressure: 150 psig.
Design temperature: 150°F.

21. Waste Collector Tank, Floor Drain Collector Tank, Radwaste Surge Tank, LLRPSF Surge Tank, Waste Sample Tanks, Chemical Waste Sample Tank, Floor Drain Sample Tank, Detergent Drain Tanks, Waste Sludge Tank, Floor Drain Sludge Tank, Detergent Holding Tank, Condensate Backwash Receiver Tank

Construction materials: carbon steel, phenolic lined with stainless steel internals.
Design pressure: atmospheric.
Design temperature: 150°F.

Capacities:

Waste Collector Tank	10,000 Gal
Floor Drain Collector Tank	10,000 Gal
Radwaste Surge Tank	40,000 Gal
LLRPSF Surge Tank	70,000 Gal
Waste Sample Tanks	10,000 Gal (2)
Chemical Waste Sample Tank	4,000 Gal
Floor Drain Sample Tank	10,000 Gal
Detergent Drain Tanks	1,000 Gal (2)
Waste Sludge Tank	4,000 Gal
Floor Drain Sludge Tank	4,000 Gal
Detergent Holding Tank	500 Gal
Condensate Backwash Receiving Tank	8,500 Gal
Chemical Waste Neutralizing Tank	4,000 Gal
Spent Resin Tank	1,200 Gal

22. Chemical Waste Tank, Spent-Resin Tank

Construction material: stainless steel with stainless steel internals.
Design pressure: atmospheric.
Design temperature: 150°F.

23. LLRPSF Sample Tank

Construction material: SA-516-70.
Design Pressure: Atmospheric.
Design Temperature: 150°F.

11.2.3 RADIOACTIVE RELEASES

11.2.3.1 Principal Radionuclides

Estimating the radionuclide content of wastes that may be discharged to the environment is subject to uncertainty, especially as increased treatment and recycle of wastes is applied in response to the requirement to keep releases of radioactive materials as low as reasonably achievable. Since there is no operating experience on plants having the exact radioactive waste treatment equipment configuration as the DAEC, a precise estimate of total annual activity released is not available. However, it is possible to provide conservative estimates of liquid radioactive releases for various fuel-defect operational conditions.

The primary source of radioactivity is that of the reactor coolant. With no fuel leaks, fission product activity is minor and activation product activities predominate. Fission product content is approximately linear with leak rate from defective fuel. The design basis for the liquid radwaste system is an annual offgas rate of 100,000 μCi/sec of a diffusion mixture of noble gases referenced to 30-min decay.

From the correlation of data from operating plants experiencing various degrees of fuel leakage [redacted] reactor water concentrations of activation products and fission products are predictable with an accuracy of approximately plus 100% to minus 50% for a given fuel leakage rate. Because the wastes arrive in the waste system from various sources within the reactor facility and under a variety of conditions, the best conservative estimate of radionuclide composition of the wastes is to assume that they have the same composition (not concentration) as reactor water. Table 11.2-1 shows annual average amounts of biologically significant radionuclides that may be present in wastes discharged to the environment during operation with design-basis fuel leakage. The total activity release rate per year shown in this table is somewhat greater than the expected sum of discharges from various subsystems under normal operating conditions, and it includes releases from expected operational occurrences. All isotopes will not necessarily be measurable in the effluent waste or in the environment.

11.2.3.2 Effluent Concentration

2015-010 When the liquid wastes are released to the discharge pipe, the limit of effluent concentration is established as 10^{-7} μCi/cm^3 above background for an unidentified mixture of radioisotopes. If analyses are to be made and radioisotopes are to be identified, the discharge of concentrations higher than 10^{-7} μCi/cm^3 is permissible under limits of 10 CFR 20. Tritium is present in the effluent and is discussed in Section 11.2.3.4.

Estimated activity concentrations in liquid wastes discharged from the radwaste facility to the effluent line (before introduction into the dilution flow) are given in Table 11.2-2. Relative concentrations of the major radioisotopes in the effluent line after dilution to 10^{-7} μCi/cm^3 in Table 11.2-3. Table 11.2-1 shows the estimated annual release rate and the 10-yr activity inventory for each of the radioisotopes on the basis of an offgas rate of 100,000 μCi/sec annual release rate after 30-min decay.

11.2.3.3 Effect on the Environment

Table 11.2-4 shows estimated annual activity discharge and dose rates as a function of various operating conditions, and Table 11.2-5 shows the quantities of the principal radioisotopes estimated in liquid wastes subject to release, based on an offgas rate of 25,000 μCi/sec aged to 30 min.

The dose under expected normal operating conditions is less than 3.0 mrem/yr total body or 10.0 mrem/yr to any organ. The doses to the public, under any of the operating conditions shown in Table 11.2-4, are less than the dose limits of 10 CFR 50, Appendix I, and are within the normal annual variation in natural background (~5 mrem/yr). Thus, potential radiation exposure to the public from liquid waste discharge is deemed to be as low as is reasonably achievable.

11.2.3.4 Tritium

Because of its relatively long half-life (12.26 years) and its ability to exchange with the hydrogen atom in water, tritium is frequently a subject of public interest in liquid wastes that are discharged.

Tritium is formed in the reactor in three ways: (1) n, γ reactions with deuterium in the water, (2) from fission at the rate of one tritium atom for each 10^4 fissions, and (3) from fast neutron reactions with the boron in the control rods. At most, 0.1% of the tritium formed in the fuel diffuses or leaks out to the reactor coolant where an exchange with the hydrogen in the coolant takes place. The formation rate in the coolant is 0.27 μCi/sec assuming a leakage of 0.1% from the fuel and a reactor power of 1912 MWt.

Assuming a continuous release of tritium to the river, uniform mixing in the river, and an average river flow rate of 1.38×10^6 gpm, the tritium concentration in the river is 3.0×10^{-9} $\mu Ci/cm^3$. The annual average tritium concentration limit for plant liquid effluent has been set at 1×10^{-3} $\mu Ci/ml$ in 10 CFR 20. The tritium concentration in the river from the plant effluent is a factor of 10^6 below the 10 CFR 20 limit. Assuming the minimum recorded daily average flow in the river of 9.53×10^4 gpm, the resulting tritium concentration is 4.4×10^{-8} $\mu Ci/cm^3$, a factor of 10^5 below the limit. Accordingly, the environmental impact of tritium releases are insignificant. The tritium releases from the plant are expected to approach 10 Ci/yr. The distribution between gaseous and liquid wastes will depend on actual amounts of water leaving by each route.

Laboratory analyses are performed periodically on reactor coolant and liquid radwaste batches to establish the presence of tritium and the activity level. These analyses are used to determine the buildup of tritium in the reactor water and to confirm that the tritium levels in the liquid radwaste discharges are within the appropriate maximum permissible concentration.

11.2.3.5 Release of Accumulated Gaseous and Liquid Radwastes

Analyses have been performed for the whole-body and critical organ doses that could result from the unlikely releases to unrestricted areas of the accumulated gaseous and liquid radwastes from the radwaste building, the offgas system charcoal delay beds and the LLRSPF sample tank as a result of tornado, flood, or earthquake damage.

A nonmechanistic failure of all charcoal delay tanks and the instantaneous loss of the complete equilibrium accumulation of noble gas and halogen isotopes was conservatively assumed for the offgas system evaluation. Failed fuel equivalent to a 100,000 µCi/sec offgas rate at 30-min decay and a finite-cloud ground-level release meteorology model with Pasquill Type F diffusion parameters and a 1.0 m/sec wind speed were used to determine the site boundary whole-body and thyroid doses. The resulting whole-body and thyroid doses were 71 and 500 mrem, respectively, well below the annual limits of 10 CFR 20.

A nonmechanistic failure of all liquid radwaste surge, sample, and storage tanks within the radwaste building together with the instantaneous release of their contents to the soil was conservatively assumed for the liquid radwaste system evaluation. The percolation of the released liquids through the soil with subsequent movement in the ground water to the river is the most likely pathway to unrestricted areas. A transit time to the river of 1000 days was conservatively calculated on the basis of a permeability of 10^{-2} cm/sec for the sand and clay soil mixture, an effective porosity of 15%, a flow gradient of 5/650 ft/ft, and a distance of 1500 ft to the river. Data was obtained from well-production studies done at the site. No credit was taken for isotope demineralization of the flowing liquid radwaste although considerable retention can be expected in the clay-rich sandy soil.[1] Also, no credit was taken for dilution in the ground water before reaching the river.

Fission-product concentrations in the tanks were based on failed fuel equivalent to a 100,000 μCi/sec offgas rate at 30-min decay. Tank isotope concentrations and gross activity accumulations are shown in Table 11.2-6.

The released liquid radwastes were conservatively assumed to reach the river as a slug (following the 1000-day transit time), and dispersion in the river under average flow conditions was calculated by the method described in Reference 2 using conservative turbulence coefficients. The resulting calculated total fraction of the maximum permissible concentration at the first municipal water intake (Cedar Rapids intake) was 0.04. Noble gas releases to the air from the radwaste tanks were found to be insignificant in comparison to the offgas system releases discussed above.

The 4000 gallon LLRPSF sample tank receives liquids only from the floor drains in the processing area of the LLRPSF. This area is not expected to contain significant levels of radioactivity. Any radioactivity is expected to be well below DAEC release limits. The sample tank contents will normally be transferred to the liquid radwaste system, or released to the environment, depending on activity level.

On the basis of the conservative approach and evaluations presented above and the resulting doses and fraction of the maximum permissible concentration, it is not necessary to design the structures housing the liquid radwaste system, or offgas system or LLRPSF sample tank for the design-basis earthquake.

A storm drain system has been installed in the yard area since the time of the FSAR radwaste tank failure analysis. The storm drain system discharges to a retention pond south of the security fence. The pond connects to a drainage ditch which runs to the Cedar River via the plant discharge canal. A sluice gate structure was installed at the outlet of the retention pond to prevent the contents of a radwaste tank spill from reaching the river via surface runoff. The gate is normally closed and in the event of a spill the tank contents would be retained in the pond. Because the distance between the sluice gate and the river is 1500 feet, the calculated 1000 day transport time is still valid. Therefore, the analysis of the consequences of liquid radwaste tank failure has not been affected by the presence of the yard storm drain system.

REFERENCES FOR SECTION 11.2

1. International Atomic Energy Agency, Disposal of Radioactive Wastes II, STI/Pub. 18, 1960.

2. W. W. Sayre, Dispersion of Mass in Open-Channel Flow, Hydraulics Papers, Colorado State University, 1968.

Table 11.2-1

ESTIMATED ANNUAL RADIOACTIVITY RELEASE IN LIQUID EFFLUENTS[a,b]

Isotope[c]	Release Rate[d] (Ci/yr)	10-Year Inventory[e] (Ci)
Sr-89[f]	3.1×10^{-3}	6.3×10^{-4}
Sr-90[f]	2.4×10^{-4}	2.1×10^{-3}
Sr-91	3.0×10^{-2}	4.8×10^{-5}
Mo-99[f]	2.0×10^{-2}	2.2×10^{-4}
I-131	1.4×10^{-2}	4.4×10^{-4}
I-133	6.4×10^{-2}	2.2×10^{-4}
I-135	4.1×10^{-2}	4.5×10^{-5}
Cs-134	1.7×10^{-4}	4.7×10^{-4}
Cs-137	2.5×10^{-4}	2.2×10^{-3}
Ba-140[f]	8.9×10^{-3}	4.5×10^{-4}
Ce-144[f]	3.6×10^{-5}	4.0×10^{-5}
Np-239	2.2×10^{-1}	2.0×10^{-3}
Co-58	2.7×10^{-3}	7.5×10^{-4}
Co-60	2.7×10^{-4}	1.5×10^{-3}
Total	[g]4×10^{-1}	1.1×10^{-2}

See notes on the following page.

ESTIMATED ANNUAL RADIOACTIVITY RELEASE IN LIQUID EFFLUENTS[a,b]

[a] The radioactivity release does not include tritium.

[b] Based on an offgas rate of 100,000 μCi/sec annual release rate after 30-min decay.

[c] Isotopes having a half-life less than 2.3 hr were excluded because the holdup in the plant generally would be sufficient to result in negligible concentrations in released wastes. Other isotopes of the elements listed were considered. The radionuclides Zr-95, Nb-95, Ru-103, Ru-106, Te-129m, Te-132, Nd-147, Na-24, S-35, P-32, Cr-51, Mn-54, Mn-56, Fe-55, Fe-59, Cu-64, Ni-65, Zn-65, Zn-69m, Ag-110m, Ta-182, W-187 were also considered, but if present will be negligible relative to those isotopes listed.

[d] Although two significant numbers are used to express release rates as a convenience for making further calculations, only one significant figure is warranted by source data.

[e] Quantity present in the environment at the end of 10 year as a result of continous discharge at release rate shown.

[f] Daughter isotopes yttrium, technetium, lanthanum, and praseodymium may be observed in waste samples in equilibrium with, or approaching equilibrum with their parent depending on sample and analysis timing and procedure.

[g] Based on 30% floor drain, 100% detergent drain, and 100% chemical waste to discharge.

Additional Notes:

The following assumptions were used in the preparation of the isotopes list and the selection of listed isotopes:

1. Radioisotopes with half-lives < 2.3 hr were excluded.
2. Radioisotopes observed in BWR water having half-lives > 2.3 hrs were further evaluated.
3. An initial list was prepared taking into account fission product noble gas release rates, reactor water fractional cleanup rates, in-plant decay, and in-plant decontamination.

An isotope was included in the final list if three or four of the following criteria were applicable:

a.	Half-life	≥ 24 hr
b.	Maximum permissible concentration of isotope	≤ 2 x 10^{-5} μCi/cm^3
c.	Percent of maximum permissible concentration	≥ 0.01%
d.	Percent of total activity	≥ 0.5%

ESTIMATED ANNUAL RADIOACTIVITY RELEASE IN LIQUID EFFLUENTS[a,b]

4. Exceptions

   a. Sr-91 with only two criteria applicable was included because its daughter Y-91 would have three of the criteria applicable.

   b. Co-60 with only one of the criteria applicable was included because it would be expected to be of greater environmental significance than Co-58, which has three of the criteria applicable.

Table 11.2-2

ESTIMATED ACTIVITY CONCENTRATIONS IN WASTES BEFORE INTRODUCTION INTO THE EFFLUENT LINE

Source of Waste	Tank Volume (gal)	Daily Discharge Volume (avg. gpd)	Expected Annual Average Concentration[a] ($\mu Ci/cm^3$)	Concentration With Design-Basis Fuel Leak[b] ($\mu Ci/cm^3$)
Floor drain	10,000	2040[c]	8×10^{-6}	1×10^{-5}
Detergent	1,000	300	1×10^{-5}	1×10^{-5}
Chemical waste	4,000	500	4×10^{-5}	2×10^{-4}

[a] Based on a fuel leakage rate that results in a an annual average stack release rate of 25,000 µCi/sec referenced to 30-min decay.
[b] Based on design-basis fuel leak that results in an annual average stack release rate of 1.0×10^5 µCi/sec of a noble gas diffusion mixture referenced to 30-min decay.
[c] Assuming 70% recycled and 30% discharged.

Table 11.2-3

RELEASE CONCENTRATIONS OF INDIVIDUAL ISOTOPES DURING DISCHARGE IN EFFLUENT LINE[a]

Isotope	Discharge Concentration Diluted to 1 x 10^{-7} µCi/cm^3
Sr-89	7.8 x 10^{-10}
Sr-90	6.0 x 10^{-11}
Sr-91	7.5 x 10^{-9}
Mo-99	5.1 x 10^{-9}
I-131	3.4 x 10^{-9}
I-133	1.6 x 10^{-8}
I-135	1.0 x 10^{-8}
Cs-134	4.1 x 10^{-11}
Cs-137	6.2 x 10^{-11}
Ba-140	2.2 x 10^{-9}
Ce-144	9.0 x 10^{-12}
Np-239	5.4 x 10^{-8}
Co-58	6.7 x 10^{-10}
Co-60	6.8 x 10^{-11}
Total	1.0 x 10^{-7}

[a] After mixture has been diluted to 10^{-7} µCi/cm^3.

Table 11.2-4

ESTIMATE RADIOACTIVE LIQUID EFFLUENT QUANTITIES AND DOSE RATES FOR VARIOUS OFFGAS RELEASE RATES

Operational Release Basis	Activity Discharge (Ci/yr)	Annual Whole-Body Dose Rate[a] (mrem/yr)
Offgas release rate of 250,000 μCi/sec at 30 min	2.0	0.2
Offgas release rate of 100,000 μCi/sec at 30 min	0.4	0.03
Offgas release rate of 25,000 μCi/sec at 30 min	0.1	0.006

[a] Based on annual average river flow of 3000 cfs.

Table 11.2-5

QUANTITIES OF PRINCIPAL RADIOSOTOPES ESTIMATED IN LIQUID WASTES SUBJECT TO RELEASE[a]
(Offgas rate of 25,000 μCi/sec aged to 30 min)

Isotopes	Daily Discharge (μCi/day)
Sr-89	1.0
Sr-90	0.1
Sr-91	10.0
Mo-99	7.0
I-131	6.0
I-133	30.0
I-135	20.0
Cs-134	0.07
Cs-137	0.1
Ba-140	4.0
Ce-144	0.01
Np-239	70.0
Co-58	7.0
Co-60	0.7
Total	151.98 (round to 150)

[a] Based on 30% floor drain, 100% detergent drain, and 100% chemical waste to discharge.

Table 11.2-6 Page 1 of 2

ISOTOPIC ACTIVITY INVENTORY (μCi)

Isotope	Detergent Drain Tank (each tank)	Waste Sludge Tank (each tank)	Waste Sample Tank (each tank)	Floor Drain Sample Tank	Waste Surge Tank (40,000 Gal)	WasteSurge Tank (70,000 Gal)
Br-83	-	19.5	71.8	0.35	195,000	341,250
Br-84	-	36.0	6.4×10^{-4}	3.1×10^{-6}	360,000	630,000
Br-85	-	21.0	-	-	210,000	367,500
I-131	-	18.0	1,965	9.8	180,000	315,000
I-132	-	165	491	2.5	1,650,000	2,887,500
I-133	-	120	9,072	45.4	1,200,000	2,100,000
I-134	-	315	2.6	0.013	3,150,000	5,512,500
I-135	-	180	5,670	29.1	1,800,000	3,150,000
Sr-89	-	3.9	454	2.19	39,000	68,250
Sr-90	-	0.3	33	0.17	3,000	5,250
Sr-91	-	90	4,158	21.5	900,000	1,575,000
Sr-92	-	142.5	756	3.7	1,425,000	2,493,750
Zr-95	-	0.05	5.67	0.03	525	919
Zr-97	-	0.04	0.29	0.014	405	709
Nb-95	-	0.05	0.06	0.03	525	919
Mo-99	-	28.3	2,873	14.4	285,000	498,750
Tc-99m	-	360	10,584	52.9	3,600,000	6,300,000
Tc-101	-	180	1.06×10^{-11}	5.3×10^{-14}	1,800,000	3,150,000
Ru-103	-	0.026	2.8	0.014	255	446
Ru-106	-	0.003	37.8	0.002	33	58
Te-129m	-	0.05	0.57	0.029	510	893
Te-132	-	63	6,426	32.1	630,000	1,102,500
Cs-134	7.56	0.21	23.4	0.117	2,100	3,675
Cs-136	-	0.135	15.1	0.076	1,350	2,363
Cs-137	22.68	0.315	35.5	0.178	3,150	5,513
Cs-138	-	240	.005	2.53×10^{-5}	2,400,000	4,200,000
Ba-139	-	210	56.7	0.29	2,100,000	3,675,000
Ba-140	-	11.55	1,285	6.4	115,500	202,125
Ba-141	-	225	3.8×10^{-8}	1.89×10^{-10}	2,250,000	3,937,500
Ba-142	-	225	1.36×10^{-16}	6.8×10^{-19}	2,250,000	3,937,500
Ce-141	-	0.05	5.67	0.028	510	893
Ce-143	-	0.045	4.2	0.02	450	788
Ce-144	-	0.045	5.3	0.026	450	788
Pr-143	-	0.05	5.3	0.027	495	866

Table 11.2-6 Page 2 of 2

ISOTOPIC ACTIVITY INVENTORY (μcI)

Isotope	Detergent Drain Tank (each tank)	Waste Sludge Tank (each tank)	Waste Sample Tank (each tank)	Floor Drain Sample Tank	Waste Surge Tank (40,000 Gal)	Waste Surge Tank (70,000 Gal)
Nd-147		0.018	1.97	0.0098	180	315
Np-239	-	315	31,000	155	3,150,000	5,512,500
Na-24	-	1.35	90.7	0.45	13,500	23,625
P-32	-	0.014	1.5	0.0076	135	236
Cr-51	-	0.35	38	0.19	3,450	6,038
Mn-54	-	0.027	3.1	0.155	270	473
Mn-56	-	34.5	151	0.76	345,000	603,750
Co-58	3.78	3.45	378	1.93	34,500	60,375
Co-60	3.78	0.345	38	0.193	3,450	6,038
Fe-59	-	0.054	6.04	0.03	540	945
Ni-65	-	2.1	9.1	0.045	21,000	36,750
Zn-65	-	6.8×10^{-4}	0.076	3.8×10^{-4}	6.75	11.81
Zn-69m	-	0.021	1.29	0.006	210	368
Ag-110m	-	0.04	4.5	0.023	405	709
W-187	-	2.1	174	0.87	21,000	36,750

11.3 GASEOUS WASTE MANAGEMENT SYSTEM

11.3.1 DESIGN BASES

11.3.1.1 Power Generation Objectives

The power generation objectives of the gaseous radioactive waste system are to process and control the release of noble radiogases to the environs so as to limit the annual average exposure to an individual at any point on the site boundary to a maximum of 10 mRad/yr to the total body. This design objective for the DAEC is as low as reasonably achievable.

The design objective for radioactive halogens and particulates with half-lives longer than 8 days is stated in Section 11A.3.3.2.

The corresponding release rates are as stated in the Technical Specifications.

11.3.1.2 Power Generation Design Basis

The gaseous radioactive waste system is designed to limit offsite doses from routine plant releases to the lowest practicable level. The offgas system is designed to provide adequate time for corrective action to limit the activity release rates should they approach established limits.

The design basis for this system is a noble gas input equivalent to an annual average offgas rate (based on 30-min decay) of 25,000 μCi/sec with a diffusion mixture. Table 11.3-1 indicates the noble gas activity from the steam jet air ejector system referenced to 30 min after exiting from the reactor for an offgas rate of 100,000 μCi/sec.

Normally air inleakage is expected to be approximately 7 cfm (at 130°F, 1 atm) per condenser shell. Leakage from two condenser shells corrected to standard conditions gives 12.3 scfm. However, the design leakage of the plant is conservatively taken as 18.5 scfm. Additonally, air or oxygen is injected to the offgas system upstream of the catalytic recombiner to completely recombine excess hydrogen from the Hydrogen Water Chemistry System.

The effect of air inleakage and injection on system performance is shown below:

Condenser Air Inleakage & Injection (scfm)	Curie Reduction Factor Relative to 100,000 μCi/Sec at 30 Min (12 Beds)
18.5	61
20	50
30	22
40	14
50	10.5
60	8.5

There is an oil-free air supply that bleeds into the system during startup. Its flow rate is 56.7 lb/hr, which is stopped after the recombiner comes up to temperature.

The isotopic release from the gland seal exhauster to the environment is presented in Table 11.3-2. The values in the table are based on an effective decay time of 1.75 min and take into consideration that an average of 0.1% of the total steam flow is routed to the gland seals. The values in the table are also based on the assumption that the offgas release rate from the main steam turbine condenser is equivalent to a release rate of 100,000 μCi/sec after an effective holdup time of 30 min. The calculated annual dose rate from the gland seal noble gas effluent corresponding to an offgas rate of 100,000 μCi/sec (reference 30-min holdup) is 0.42 mrem/yr at the site boundary.

The gland seal noble gas particulate daughters produced as a consequence of radioactive decay are presented in Table 11.3-3. Because of different half-lives of the parents and daughters, the location of maximum daughter concentration will vary for each isotope. The radiological effects of the daughter products are expressed in terms of X_i/MPC_i for each isotope. As noted in Table 11.3-3, there is no daughter product that exceeds 1.9×10^{-5} of the allowable MPC_{air}. Therefore, it is unnecessary to provide additional holdup or filtration of the gland seal noble gas effluent.

Shielding is provided as necessary for process piping and equipment.

11.3.1.3 Safety Design Basis

The gaseous radwaste system is designed so that any quantities of gaseous radwastes inadvertently released result in radiation levels within annual exposure limits of 10 CFR 20.

11.3.2 SYSTEM DESCRIPTION

11.3.2.1 Process Description

The offgas treatment system shown in Figure 11.3-1 uses a high-temperature catalytic recombiner to recombine radiolytically dissociated hydrogen and oxygen from the air ejector system. After chilling to strip the condensibles and reduce the volume, the remaining noncondensibles (principally kryptons, xenons, and air) are delayed in a 30-min holdup system cooled to a dewpoint of 45°F with a chilled glycol cooler, passed through a de-entrainer, heated to 74°F (relative humidity of 35%), and passed through a HEPA filter before reaching the adsorption bed. The charcoal adsorption bed, operating in a constant temperature vault, selectively adsorbs and delays the xenons and kryptons from the bulk carrier gas (principally air). This delay on the charcoal permits the xenon and krypton to decay in place. This system results in a reduction of the offgas activity released by a factor of approximately 61 relative to a 30-min holdup system and based on a diffusion mixture.

The design of the offgas system incorporates an automatic loop seal isolation system that monitors system pressure at the 37-sec holdup volume ahead of the recombiner and in the 30-min holdup volume downstream of the recombiner. These isolation setpoints are set at 4.0 psig and 4.5 psig, respectively, with associated control room indication.

In the event of a loop seal liquid loss, leaking gas mixtures would be alarmed in the control room through the reactor building ventilation stack gaseous monitors, reactor building ventilation radiation monitors, ventilation shaft radiation monitors, offgas post-treatment radiation monitors, and offgas stack radiation monitors.

On the isolation of the offgas loop seals, the normal operating procedure is to keep loop seals isolated until they naturally fill through condensation buildup.

In the event of an inadvertent offgas system explosion, the site preparedness plan and implementing procedures provide adequate guidance for proper response.

During the review of off-normal offgas system operation, one condition was identified that could lead to the accumulation of an explosive mixture. During normal operation, offgas is diluted in jet compressor [redacted]. In the event of the loss of dilution steam, flow valve [redacted] is automatically shut, isolating the jet compressor. This causes a pressure transient in the system that actuates several pressure switches, which in turn automatically close the isolation valves on the system loop seals.

Two loop seals are located upstream of [redacted]. In the event that insufficient water is available in one of the loop seals upstream of [redacted], the loop seal may blow before sufficient pressure to actuate the system pressure switches develops. Under these conditions, undiluted offgas would be discharged to the turbine building equipment drain sump or the offgas retention building equipment drain sump. Considering normal ventilation flow, the average hydrogen concentration in each area would be 16.7% and 13.0% by volume, respectively.

No other conditions have been identified that could result in the accumulation of sufficient hydrogen to form an explosive mixture. The review considered loss of ventilation flow, loss of dilution steam, lost loop seals, blown rupture disks, and leakage of offgas into isolated portions of the system. Bypassing the recombiners was not considered as this is not applicable to the DAEC design.

To ensure that the conditions identified above do not occur, the system has been designed in the following manner:

1. Water fill lines are installed to all loop seals that do not presently have fill capability.

2. System logics are designed to automatically isolate the two loop seals upstream of [redacted] on the closure of [redacted].

The adsorption of noble gases on charcoal depends on gas flow rate, holdup time, mass of charcoal, and a gas-unique coefficient known as the dynamic adsorption coefficient. The parametric interrelationships and governing equations are well proven from 3 yr of operation of a similar unit at [redacted].

The basis for these coefficients and supporting experimental data are discussed in a proprietary document submitted with Amendment 1, May 1972, in response to an AEC question.

11.3.2.2 Equipment Description

The design of the DAEC gaseous radwaste system incorporates a catalytic recombiner and a 12-bed charcoal adsorber system. The design fully satisfies the requirements that releases of radioactivity be reduced to the lowest practicable level.

The piping and instrumentation drawings for the gaseous waste systems are included as Figures 11.3-2 and 11.3-3. The design codes for piping in these systems are provided by the line classification numbers given in the drawings. The line classification numbers are defined in Section 3.2.2.2.

System components and equipment that serve as pressure boundaries for the offgas system are fabricated in accordance with ASME Code, Section III, Class 3.

The description of offgas system major equipment items is as follows:

1. Offgas Preheaters

   Construction: Stainless steel tubes and carbon steel shell.
   Design pressure: 350-psig shell, 1000-psig tube.
   Design temperature: 400°F shell, 575°F tube.

2. Catalytic Recombiners

   Construction: Stainless steel cartridge, low-alloy steel shell. Catalyst cartridge containing a precious metal catalyst on nichrome strips of porous, nondusting ceramic. Catalyst cartridge to be replaceable without removing vessel.
   Design pressure: 350 psig.
   Design temperature: 900°F.

3. Offgas Condenser

   Construction: Low-alloy steel shell, stainless steel tubes.
   Design pressure: 350-psig shell, 250-psig tube.
   Design temperature: 900°F.

4. Water Separator

   Construction: Carbon steel shell, stainless steel wire mesh.
   Design pressure: 350 psig.
   Design temperature: 250°F.

5. Cooler Condenser

   Construction: Stainless steel shell, stainless steel tubes.
   Design pressure: 100-psig tube, 350-psig shell.
   Design temperature: 50°F tube, 150°F shell.

6. Moisture Separators (Downstream of Cooler Condenser)

   Construction: Carbon steel shell, stainless steel wire mesh.
   Design pressure: 350 psig.
   Design temperature: 150°F.

7. Offgas Reheater

   Construction: Carbon steel pipe. Electrically heated.

8. Glycol Storage Tank

   Construction: Carbon steel. Capacity is 3000 gal.
   Water-filled hydrostatic design pressure.
   Design temperature: 0°F.

9. Glycol Solution Refrigerators and Motor Drives

   Construction: Conventional refrigeration units.
   Glycol exit solution temperature: 35°F.

10. Glycol Pumps and Motor Drives

    Construction: Cast iron, 3-in. connections, 50 ft.
    Design temperature: 0°F.

11. Prefilters and After Filters

    Construction: Carbon steel shell. High-efficiency, moisture-resistant filter element.
    Flanged shell.
    Design pressure: 350 psig.
    Design temperature: 130°F.

12. Carbon Bed Adsorbers (12 Beds)

Construction: Carbon steel. Four ft outside diameter x 21 ft vessels, each with a 19-ft packed section containing ~3 tons of 8-14 mesh carbon (~200 ft^3 of charcoal) Columbia G or equivalent.
Design pressure: 350 psig.
Design temperature: 130°F.

Flow channeling and bed settling are minimized in the charcoal adsorber vessels by the following design considerations:

a. Charcoal adsorber beds are installed for vertical flow of the process gas stream.

b. The first three charcoal beds in each parallel pathway have piping arrangements that cause up-flow from vessel bottom to vessel top of the process gas.

c. As shown in Figures 11.3-4 and 5, toroidal-shaped flow-distribution rings are positioned at the bottom and top ends of each charcoal vessel to enhance the process gas flow pattern. For the first three charcoal vessels in the process stream, the gas enters the distribution ring nozzle at the bottom of the vessel and flows through a distribution torus and out through 251 one-in. holes and 3 layers of screen on the bottom of the torus; the gas then flows upward around both outside walls of the distribution torus and through the charcoal to the upper torus region. In the upper vessel region, the charcoal-filtered gas enters the distribution torus through 3 layers of screen and 251 holes on the top of this torus and proceeds out of the distribution ring nozzle to the next charcoal vessel in the process flow path.

   Process gas enters the top distribution ring and exits out the lower ring nozzle for charcoal vessels other than the first three in each parallel path.

13. Offgas Jet Compressor

Construction: Carbon steel body.
Design pressure: 2150 psig.
Design temperature: 400°F.
Flow rate: 4624 lb/hr.

The ventilation system for each DAEC building that can be expected to contain radioactive materials is described in Section 9.4.

The main condenser gas removal system is described in Section 10.4.2. The main steam line isolation valve leakage treatment path is described in Section 6.7.

11.3.2.3 Instrumentation and Control

The radiation levels at the air ejector offgas discharge line and after the offgas treatment system are continuously monitored by pairs of detectors. This system is also monitored by flow and temperature instrumentation and hydrogen analyzers to ensure proper operation and control and to ensure that hydrogen concentration is maintained below the flammable limit. Process radiation instrumentation is described in Section 11.5. Table 11.3-4 lists process instrument alarms.

11.3.2.4 Safety Evaluation

The decay time provided by the 30-min holdup pipe and the long-delay charcoal adsorbers is established to provide for radioactive decay of the activation gases and fission gases in the main condenser offgas. The adsorbers provide a 15-day xenon and a 19-hr krypton holdup. The daughter products that are solids are removed by filtration following the 30-min holdup and/or are retained on the charcoal. Final filtration of the charcoal adsorber effluent precludes the escape of charcoal fines that contain radioactive materials. Thus, there is virtually no particulate activity release.

Iodine input into the offgas system is small because of its retention in the reactor water and condensate. The charcoal effectively removes the iodine entering the system by adsorption and prevents its release.

Radiation monitors at the recombiner outlet continuously monitor radioactivity releases from the reactor and, therefore, continuously monitor the degrees of fuel leakage and input to the charcoal adsorbers. Radiation monitors are used to provide an alarm on high radiation in the offgas. Two radiation monitors are provided at the outlet of the charcoal adsorbers to continuously monitor the release rate from the adsorber beds. The radiation monitors are further described in Section 11.5.

Shielding is provided for offgas system equipment to maintain safe radiation exposure levels for plant personnel. The equipment is principally operated from the control room.

The charcoal adsorbers operate at essentially room temperature so that upon system shutdown, radioactive gases in the adsorbers are subject to the same holdup time as during normal operation, even in the presence of continued air flow. Therefore, the radioactive materials are not subject to an accidental release evaluation. The charcoal adsorbers are designed to limit the temperature of the charcoal to well below the charcoal ignition temperature, precluding overheating or fire and consequent escape of radioactive materials. The adsorbers are located in a shielded room and maintained at a constant temperature by an air conditioning system, which removes the decay heat generated in the adsorbers. The failure of the air conditioning system will cause an alarm in the control room. In addition, a radiation monitor is provided to monitor the radiation level in the charcoal bed vault. High radiation causes an alarm in the control room.

The hydrogen concentration of the gases from the air ejector is maintained below the flammable limit by providing adequate steam flow for dilution at all times. This steam flow rate is monitored and alarmed. The preheaters are heated by steam rather than electricity to eliminate the presence of potential ignition sources and to limit the temperature of the gases in the event of cessation of gas flow. The recombiner temperatures are monitored and alarmed to indicate any deterioration of performance. A hydrogen analyzer downstream of the recombiners provides an additional check.

In addition, this system is designed to be explosion resistant in the unlikely event a combustible mixture exists.

The gaseous radwaste system piping and equipment is designed to be explosion resistant by employing design methods for circular-section steel systems to contain explosions of near stoichiometric mixtures of gaseous hydrogen and oxygen.

The design method used a static analysis with dynamic materials properties. More exact rigorous dynamic analyses were conducted on selected designs with the results confirming that the static method was sufficiently conservative to use for offgas system design. Ratios of maximum pressure to initial pressure (before an explosion) ranging from 17 to 170 were used to determine the maximum peak pressure (Pm) in the component under analysis. Wall thicknesses for the particular component were then computed using Pm as the pressure load.

An equivalent detonation-containing-static-pressure was then derived for which the component could be "rated" on the basis of the wall thickness calculated per the above procedure.

This equivalent pressure was computed from one of various code equations on the basis of the purchase order date of the given equipment and the applicable codes in effect at that time.

The air ejector offgas system operates at a pressure of about 5 psig or less so that the differential pressure available to cause leakage of radioactive gases is small. To reduce the possibility of leakage of radioactive gases, the system is welded wherever possible and bellows seal valve stems or equivalent are used.

Operational control is maintained by the use of radiation monitors to ensure that the release rate is within the established limits. Provision is also made for sampling and periodic analysis of the influent and effluent gases for purposes of determining their composition. This information is used in the calibration of the monitors and in relating the release to the environmental
dose. The operator is in full control of the system at all times.

Table 11.3-5 contains a detailed malfunction analysis indicating consequences of the failure of various components of the system and design precautions taken to prevent such failures.

11.3.2.5 Inspection and Testing

The gaseous waste disposal systems are used on a routine basis and do not require specific testing to ensure operability. Calibration and maintenance of monitoring equipment is done on a specific schedule and on indication of malfunction.

The particulate filters are tested when they are changed using a dioctylpthalate (DOP) smoke test or equivalent.

Experience with BWRs has shown that the calibration of the offgas and effluent monitors changes with isotopic content. Isotopic content can change depending on the presence or absence of fuel cladding leaks in the reactor and the nature of the leaks. Because of this, the monitors are calibrated using grab samples periodically and at any time there appears to be a significant change.

11.3.3 RADIOACTIVE RELEASES

See Appendix 11A for gaseous release rate limit calculations.

All potential sources of radioactivity released to the environment because of gaseous waste and ventilation will be via the offgas stack, [redacted] These release points are shown in Figures 1.2-5, 6, 7, 8, 12, and 14, and Figure 11.3-6.

Isotopic release rates for expected normal operating conditions and design-basis conditions have been estimated for air ejector offgas, gland seal, high-pressure coolant injection (HPCI) testing, miscellaneous plant vent losses (including turbine building steam leakage and reactor building primary coolant leakage), and primary containment purge. The assumed fuel failure during normal operation is associated with a 25,000-μCi/sec offgas release rate at 30-min decay. Design-basis estimates are based on a 250,000-μCi/sec offgas rate at 30-min decay. On the basis of operating BWR fuel performance, maximum expected release rates would be less than the design basis of 250,000 μCi/sec annual average (reference 30-min decay).

The total amount of radioactive noble gases estimated to be stored in the first charcoal bed in the filter train is 5.2×10^8 μCi after 10 yr of operation. This estimate is based on residence time assumptions of 1.62 hr for krypton, 1.22 days for xenon, and 4.42×10^2 sec for N_{13} and an offgas average release rate of 25,000 μCi/sec at 30-min holdup time.

Steam Jet Air Ejector Offgas

The steam jet air ejector offgas exhausts through the catalytic recombiner and 12-bed charcoal adsorption system before exiting from the main stack.

Table 11.3-1 gives the estimated inventory of radioactivity in the main condenser offgas stream corresponding to 100,000 μCi/sec from the 30-min holdup line. As the expected normal release is approximately 25,000 μCi/sec from the 30-min holdup line, the inventory will be approximately 25% of the values given in Table 11.3-1.

Table 11.3-6 lists the noble gas radioisotopes discharged from the air ejector offgas system during normal operation. Design-basis estimates would be a factor of 10 greater.

The maximum calculated dose to a point on the site boundary for the isotopic release shown in Table 11.3-6, assuming a 365-day continuous exposure, is 0.24 mrem/yr, which occurs at a distance of [redacted]. This result is based on a condenser air inleakage rate of 18.5 scfm, discharge from the 100-m stack, and DAEC onsite meteorology data taken at the 156-ft level. The gamma dose model used is described in Section 11A.1.

Radioiodine was assumed to be completely captured in the charcoal adsorption system.

Gland Seal

Process steam is supplied to the shaft seals of the main turbine for the purpose of providing a positive seal on this system. The exhaust steam from the seal system is routed to the gland seal condenser with the noncondensibles being discharged to the environment via the gland seal exhauster and a short delay line.

Hypothetical radiological effects for this system are based on the following assumptions.

Table 11.3-7 lists the radioisotopes discharged from the gland seal system during normal operation. Design-basis estimates would be a factor of 10 greater.

The maximum calculated dose to a point on the site boundary for the isotopic release shown in Table 11.3-7 is 0.12 mrem/yr, which occurs at a distance of [redacted]. This result is based on a steam flow to the gland seal system of 0.1% of the process steam flow, an approximately 2-min delay time between steam leaving the reactor vessel and subsequent release to the environment via the 100-m stack, and DAEC onsite meteorology data taken at the 156-ft level. The gamma dose model used is described in Section 11A.1.

Radioiodine carryover is assumed to be approximately 1%, of which approximately 2% enters the gland seal system and releases to the environment.

HPCI Testing

Periodic testing of the HPCI system may also result in minor releases of radioactivity to the environment. The only activity that would be released to the environment would be that associated with the HPCI gland seal steam. The process steam used to drive the high-pressure coolant injection is routed to the pressure suppression pool and is considered in the containment purge calculation. The HPCI gland seal steam is discharged to the standby gas treatment system (SGTS) and then to the environment via the plant stack.

Table 11.3-8 lists the noble gas radioisotopes discharged from the HPCI gland seal system during HPCI operation with expected normal operation fuel defects. Design-basis estimates would be a factor of 10 greater.

The maximum calculated dose to a point on the site boundary for the isotopic release shown in Table 11.3-8 assuming an HPCI test once every 3 months of 1-hr duration is 2.3×10^{-5} mrem per release period. Steam flow to the HPCI turbine gland seals was assumed to be 500 lb/hr. The SGTS filter efficiency for noble gases was assumed to be 0% with an approximately 2-min delay time between steam leaving the reactor vessel and subsequent release to the environment. The meteorology at the time of release was conservatively assumed to be unstable with a 2 m/sec wind speed. The exposure is a maximum in the [redacted] at a distance of [redacted].

Radioiodine release rate would be insignificant because of the 99.9% SGTS filter efficiency.

Building Ventilation

Reactor building fission product releases to ventilation air are based on a continuous unidentified 1.0-gpm reactor water leak, for which a halogen decontamination factor of 10^3 is assumed. The escaped fission products are assumed to pass directly to the building vent without filtration. Meteorological assumptions for building ventilation releases were treated as described in Section 11A.3.2.1.1 to determine fission product dispersion to the site boundary.

Turbine building fission product releases to ventilation air are based on a continuous 5-gpm steam leak. An iodine decontamination factor of 80 is assumed for internal steam generation, while an iodine decontamination factor of 1.0 is assumed for the turbine building. The escaped fission products are assumed to pass directly to the building vent without filtration. Meteorological assumptions for building ventilation releases were treated as described in Section 11A.3.2.1.1 to determine fission product dispersion to the site boundary.

Table 11.3-9 gives the combined reactor building and turbine building isotope release rates corresponding to normal operation. It is assumed that the release rate from the LLRPSF is negligible in comparison. Design-basis estimates would be a factor of 10 greater. (Reactor building primary water leakage contributes negligible noble gas losses to the overall noble gas release and less than 10% to the total iodine release.) The resulting annual average exposure at the worst site boundary location is 0.11 mrem/yr.

Fission product releases from primary containment purge are based on an assumed primary system leakage of 0.5-gpm steam and 0.5-gpm unidentified reactor water. Radiological equilibrium (except for Kr-85) is assumed to occur in containment. An iodine decontamination factor of 80 is assumed for internal steam generation, and an iodine decontamination factor of 10 is used for reactor water leakage. Four controlled purges per year are assumed to occur, with the exhaust passing through the standby gas treatment system (deep-bed charcoal filter 99.9% efficient for iodine) before being released from the 100-m-high plant main stack. A conservative stack c/Q of 2.3×10^{-7} sec/m^3 (at the site boundary) for short-term releases was determined for the 90% probability level. Table 11.3-10 gives the total noble gas and iodine releases, in microcuries, for four containment purges corresponding to normal operation fuel defects. Design-basis estimates would be a factor of 10 greater. The calculated site boundary whole-body exposure for the isotopic release shown in Table 11.3-10 is less than 1×10^{-3} mrem/yr.

Table 11.3-1

ESTIMATED ISOTOPIC RELEASE RATES[a] FROM THE
STEAM JET AIR EJECTOR SYSTEM
(Corresponding to 100,000 µCi/sec release from 30-min holdup line)

Isotope	Discharge from 30-Min Holdup Line (µCi/sec)	Discharge from Charcoal Adsorbers (12 beds) (µCi/sec)
Kr-83m	2,900	2.0
Kr-85m	5,600	266
Kr-85	10-20[b]	10-20[b]
Kr-87	15,000	0.04
Kr-88	18,000	144
Kr-89	180	0
Xe-131m	15	6.5
Xe-133m	280	3.0
Xe-133	8,200	1190
Xe-135m	6,900	0
Xe-135	22,000	0
Xe-137	670	0
Xe-138	21,000	0
Halides	- -	Insignificant
Total (approx.)	100,000	1630

[a] Based on diffusion mixture.
[b] Estimated from experimental observations.

Table 11.3-2

NOBLE GAS RELEASE RATE TO ENVIRONS FROM TURBINE GLAND SEAL EXHAUSTER SYSTEM[a]

Isotope	Release Rate to Environment (μCi/sec)
Kr-83m	3.4
Kr-85m	6.1
Kr-85	0.02
Kr-87	19.0
Kr-88	20.0
Kr-89	87.0
Kr-90	30.0
Kr-91	0.07
Xe-131	0.015
Xe-133m	0.28
Xe-133	8.2
Xe-135m	24.0
Xe-135	22.0
Xe-137	111.0
Xe-138	83.0
Xe-139	46.0
Xe-140	17.0

[a] Based on 100,000 μCi/sec at 30-min decay diffusion mixture, 0.1% to gland seal system.

Table 11.3-3

RADIOACTIVE PARTICULATE DAUGHTER BUILDUP FROM GLAND SEAL NOBLE GAS EFFLUENT[a] RATIO OF X_i/MPC_i (Annual average) VERSUS DISTANCE IN WORST SECTOR

	Distance[b] (m)							
Isotope	100	300	800	1200	3000	5000	8000	12000
Rb-87	6.0-30	3.5-22	1.1-20	2.3-30	2.6-20	2.2-20	2.1-20	1.9-20
Rb-88	5.3-15	2.7-7	6.9-6	1.2-5	8.1-6	4.8-6	2.6-6	1.5-6
Rb-89	1.5-14	5.9-7	9.4-6	1.5-5	8.8-6	3.5-6	1.1-6	3.6-7
Rb-90	2.6-15	2.9-7	2.2-6	3.1-6	5.8-7	7.9-8	8.4-9	9.9-10
Rb-91	1.1-18	3.0-10	1.7-9	1.8-9	1.6-10	1.6-11	1.1-12	7.8-14
Sr-89	5.2-17	4.4-9	5.1-8	2.2-7	1.9-7	1.3-7	1.1-7	9.9-8
Sr-90	4.0-15	1.3-7	3.3-6	5.4-6	3.8-6	2.1-6	1.4-6	1.1-6
Sr-91	3.7-20	1.4-12	2.8-11	4.4-11	2.8-11	1.4-11	7.6-12	3.8-12
Y-91m	2.6-23	2.1-15	6.3-14	1.4-13	1.4-13	1.0-13	8.2-14	6.1-14
Y-91	8.8-24	7.7-16	2.6-14	6.2-14	1.0-13	1.1-13	1.5-13	1.8-14
Cs-135	1.7-25	1.0-17	3.3-16	7.7-16	1.1-15	1.2-15	1.6-15	1.9-16
Cs-137	3.5-19	1.3-11	3.1-10	5.0-10	3.5-10	2.0-10	1.4-10	1.1-10
Cs-138	1.2-14	5.3-7	1.2-5	1.9-5	1.2-5	6.8-6	3.4-6	1.8-6
Cs-139	3.3-15	2.1-7	2.1-6	3.3-6	1.3-6	3.8-7	9.3-8	2.2-8
Cs-140	2.5-17	1.4-8	6.7-8	6.0-8	2.9-9	2.0-10	8.3-12	2.7-13
Ba-137m	1.2-20	6.2-13	7.1-12	2.4-11	1.8-11	1.2-11	8.6-12	7.1-12
Ba-139	1.4-16	6.7-9	1.5-7	2.5-7	1.7-7	9.7-8	4.8-8	2.6-8
Ba-140	2.5-19	1.0-11	1.9-10	3.0-10	1.8-10	9.4-11	6.2-11	5.0-11
La-140	2.1-22	1.7-14	5.6-13	7.5-13	2.3-12	2.5-12	3.6-12	4.7-12

[a] Based on annual average meterology from DAEC site, January 8, 1971 to January 8, 1972.

[b] Second number in each column indicates exponent (e.g., 2.9-17 = 2.9×10^{-17}).

Table 11.3-4

PROCESS INSTRUMENT ALARMS[a]

Parameter	Main Control Room		
	Indicated	Recorded	Alarms
Preheater discharge temperature - low	x		x
Offgas line pressure - high	x		x
Recombiner catalyst temperature - high/low		x	x
Offgas condenser drain well (dual) level - high/low			x
Offgas condenser gas discharge temperature - high			x
Hydrogen analyzer (Condenser discharge) (dual) - high		x	x
Gas flow (offgas condenser discharge) - high/low		x	x
Cooler condenser discharge temperature - high/low		x	x
Glycol solution temperature - high/low	x	x	x
Prefilter P - high	x		x
Carbon bed temperature - high		x	x
Carbon vault temperature - high/low	x	x	x
Postfilter P - high	x		x

[a] Instrumentation elements
Temperature: thermocouple
Level: differential pressure diaphragm
Hydrogen: thermal conductivity
Gas flow: flow orifice
Differential pressure: differential pressure diaphragm

Table 11.3-5

EQUIPMENT MALFUNCTION ANALYSIS

Equipment Items	Malfunction	Consequences	Design Precautions
Preheaters	1. Steam leak	Would further dilute process offgas. Steam compumption would increase.	Space preheated.
	2. Low-pressure steam supply	Recombiner performance would fall off at low-power level and hydrogen content of recombiner gas discharge would increase, eventually to a combustible mixture.	Low-temperature alarms on preheater exit and recombiner inlet. Recombiner hydrogen analyzer.
Recombiner	1. Catalyst gradually deactivates	Temperature profile changes through catalyst. Eventually excess hydrogen would be detected by meter. Eventually, the gas could be become combustible.	Temperature probes in recombiner and hydrogen analyzer recombiner.
	2. Catalyst gets wet at startup	Hydrogen conversion falls off and hydrogen is detected by downstream analyzers. Eventually the gas could become combustible.	Condensate drains, temperature probes in recombiner. Air bleed system at startup. Recombiner thermal blanket. Spare recombiner and heater. Hydrogen analyer.
Recombiner condenser	Cooling water leak	The coolant (service water) would leak to the process gas (shell) side. This would be detected if drain well liquid level increases. Moderate leakage would be of no concern from a process standpoint.	None.

Table 11.3-5

EQUIPMENT MALFUNCTION ANALYSIS

Equipment Items	Malfunction	Consequences	Design Precautions
Drain Well	Liquid level instruments fail	If both drain valves fail to open, water will build up in the condenser and pressure drop will increase. The high Δp, if not detected by instrumentation, could cause pressure buildup in the main condenser and eventually a reactor trip. If a drain valve fails to close, gas will recycle to the main condenser increasing the load on the steam jet air ejector causing back-pressure on the main condenser, eventually causing a reactor trip.	Two separate drain systems each provided with high- and low- level alarms.
Water separator	Corrosion of wire mesh element	Higher quanity of water collected in 30-min holdup line androuted to radwaste.	Stainless steel mesh specified.
30-min holdup line	Corrosion of line	Leakage to soil of gaseous and liquid fission products.	Outside of pipe dipped and wrapped.
Cooler condenser	1. Corrosion of bare tubes	Glycol-water solution would leak into process (shell) side and be discharged to clean radwaste. If not detected at radwaste, the glycol solution would discharge to the reactor condensate system.	Stainless steel bare tube specified. The inventory of glycol-water can be observed in tank A002. Spare cooler provided.

Table 11.3-5

EQUIPMENT MALFUNCTION ANALYSIS

Equipment Items	Malfunction	Consequences	Design Precautions
	2. Icing-up of bare tube	Shell side of cooler could plug with ice, gradually building up pressure drop. If this happens, the spare unit could be activated. Complete blockage of both units would increase main condenser pressure leading to a reactor trip.	Design glycol-water solution temperature of 34-40°F. Spare unit provided. Redundant temperature indication and alarm system.
Moisture separators	Corrosion of wire mesh element	Increased moisture would be retained in process gas routed to charcoal adsorbers. Over a long period of time, the charcoal performance could deteriorate owing to moisture pickup.	Stainless steal mesh specified. Spare unit provided.
Prefilters	Hole in filter media	More radioactivity would deposit on the charcoal in the first adsorber vessel of the train. This would increase the radiation level in the charcoal vault, making maintenance more difficult.	ΔP instrumentation provided. Spare unit provided.
Charcoal adsorbers	Charcoal gets wet	Charcoal performance will deteriorate gradually as charcoal gets wet. Holdup times for krypton and xenon will decrease and plant emissions will increase.	Highly instrumented mechanically simple gas dehumidification system with redundant equipment.

Table 11.3-5

EQUIPMENT MALFUNCTION ANALYSIS

Equipment Items	Malfunction	Consequences	Design Precautions
Vault air conditioning units	Mechanical failure	If ambient temperature exceeds about 80°F, increased emission could occur.	Spare refrigerator unit provided.
		If ambient temperature is below about 60°F, charcoal could pick up additional moisture.	Vault temperature alarms provided.
After filters	Hole in filter media	Probably of no real consequence. The charcoal media themselves should be a good filter at the low air velocity.	ΔP instrumentation provided. Spare unit provided.
Glycol refrig. machines	Mechanical failure	If spare unit fails to operate, the glycol solution temperature will rise and the dehumidfication system performance will deteriorate. This will cause gradual buildup of moisture on the charcoal, with increased plant emissions.	Spare refrigerator provided. Glycol solution temperature alarms provided.

Table 11.3-5

EQUIPMENT MALFUNCTION ANALYSIS

Equipment Items	Malfunction	Consequences	Design Precautions
Steam jet ejectors	1. Low flow of motive high-pressure steam	When the hydrogen and oxygen concentrations exceed 4 and 6 volume percent, respectively, the process gas becomes combustible. Inadequate steam flow will cause overheating and deterioration of the catalyst.	Alarms provided for low steam flow and low steam pressure. Steam flow to be held at constant maximum flow regardless of plant power level.
	2. Wear of steam supply nozzle of ejector	Increased steam flow to recombiner. This could reduce degree of recombination at low power levels.	

Table 11.3-6

STEAM JET AIR EJECTOR ISOTOPIC RELEASE RATE
(Corresponding to 25,000 μCi/sec at 30-min decay offgas rate)

Isotope	Release Rate (μCi/sec)
Kr-83m	0.54
Kr-85m	66.5
Kr-85	5.95
Kr-87	0.092
Kr-88	36
Kr-89	- -
Kr-90	- -
Kr-91	- -
Xe-131m	1.615
Xe-133m	0.77
Xe-133	294.5
Xe-135m	- -
Xe-135	- -
Xe-137	- -
Xe-138	- -
Xe-139	- -
Xe-140	- -

Table 11.3-7

TURBINE GLAND SEAL ISOTOPIC RELEASE RATE
(Corresponding to 25,000 μCi/sec at 30-min decay offgas rate)

Isotope	Release Rate (μCi/sec)
Kr-83m	0.86
Kr-85m	1.54
Kr-85	0.006
Kr-87	4.85
Kr-88	5.
Kr-89	21.8
Kr-90	7.9
Kr-91	0.0175
Xe-131m	0.0038
Xe-133m	0.0695
Xe-133	2.045
Xe-135m	6.1
Xe-135	5.55
Xe-137	27.
Xe-138	20.65
Xe-139	11.35
Xe-140	0.435
I-131	0.00135
I-132	0.012
I-133	0.009
I-134	0.0235
I-135	0.013

Table 11.3-8

HPCI GLAND ISOTOPIC RELEASE RATE
(Corresponding to 25,000 μCi/sec at 30-min decay offgas rate)

Isotope	Release Rate[a] (μCi/sec)
Kr-83m	0.059
Kr-85m	0.106
Kr-85	- -
Kr-87	0.334
Kr-88	0.344
Kr-89	1.5
Kr-90	0.51
Kr-91	0.0014
Xe-131m	- -
Xe-133m	0.0048
Xe-133	0.141
Xe-135m	0.42
Xe-135	0.382
Xe-137	1.86
Xe-138	1.41
Xe-139	0.78
Xe-140	0.03

[a] Release rate during operation of HPCI turbine.

Table 11.3-9

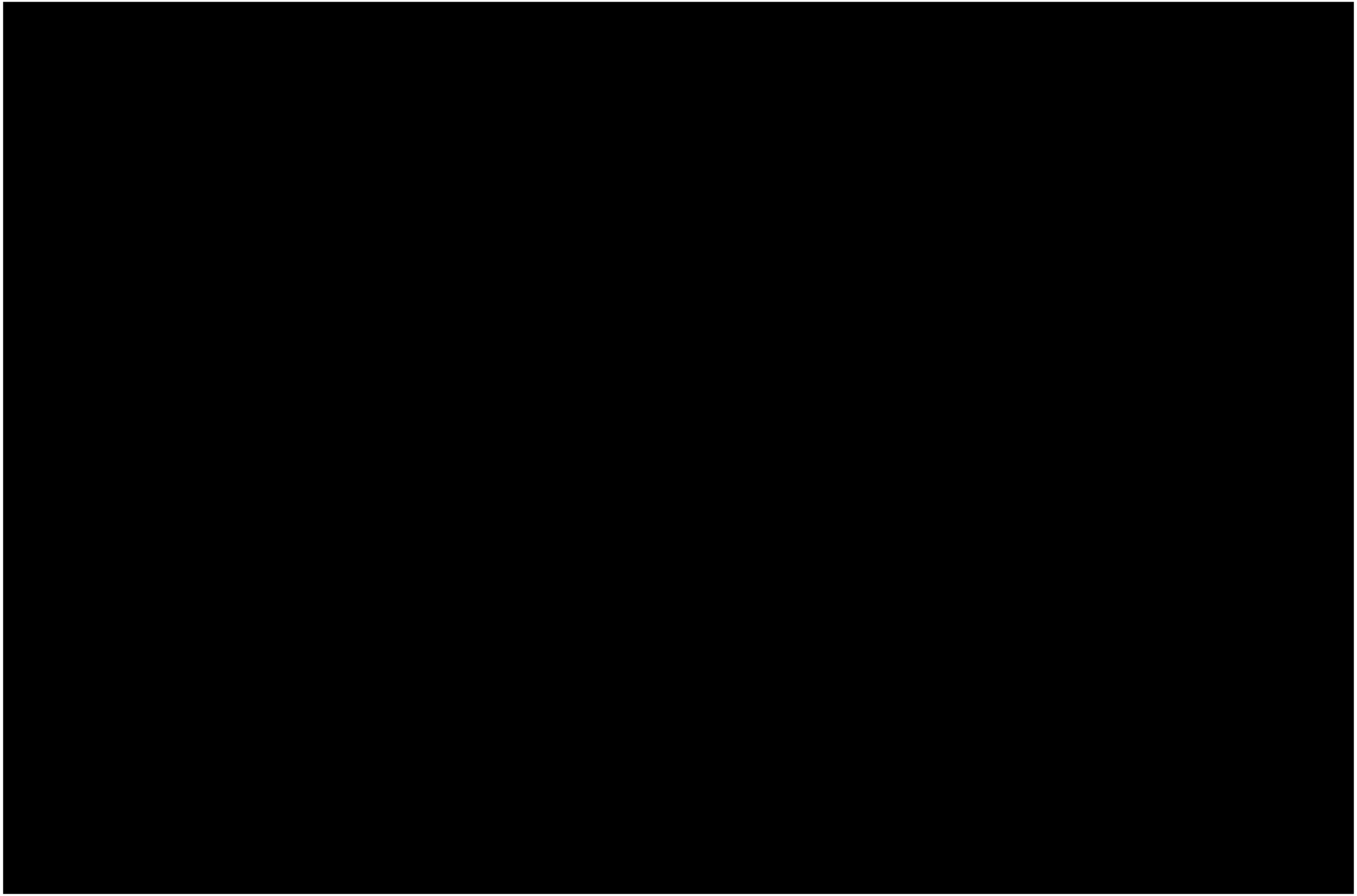
REACTOR BUILDING AND TURBINE BUILDING
COMBINED VENTILATION RELEASE RATE
(Corresponding to 25,000 µCi/sec at 30-min decay offgas rate)

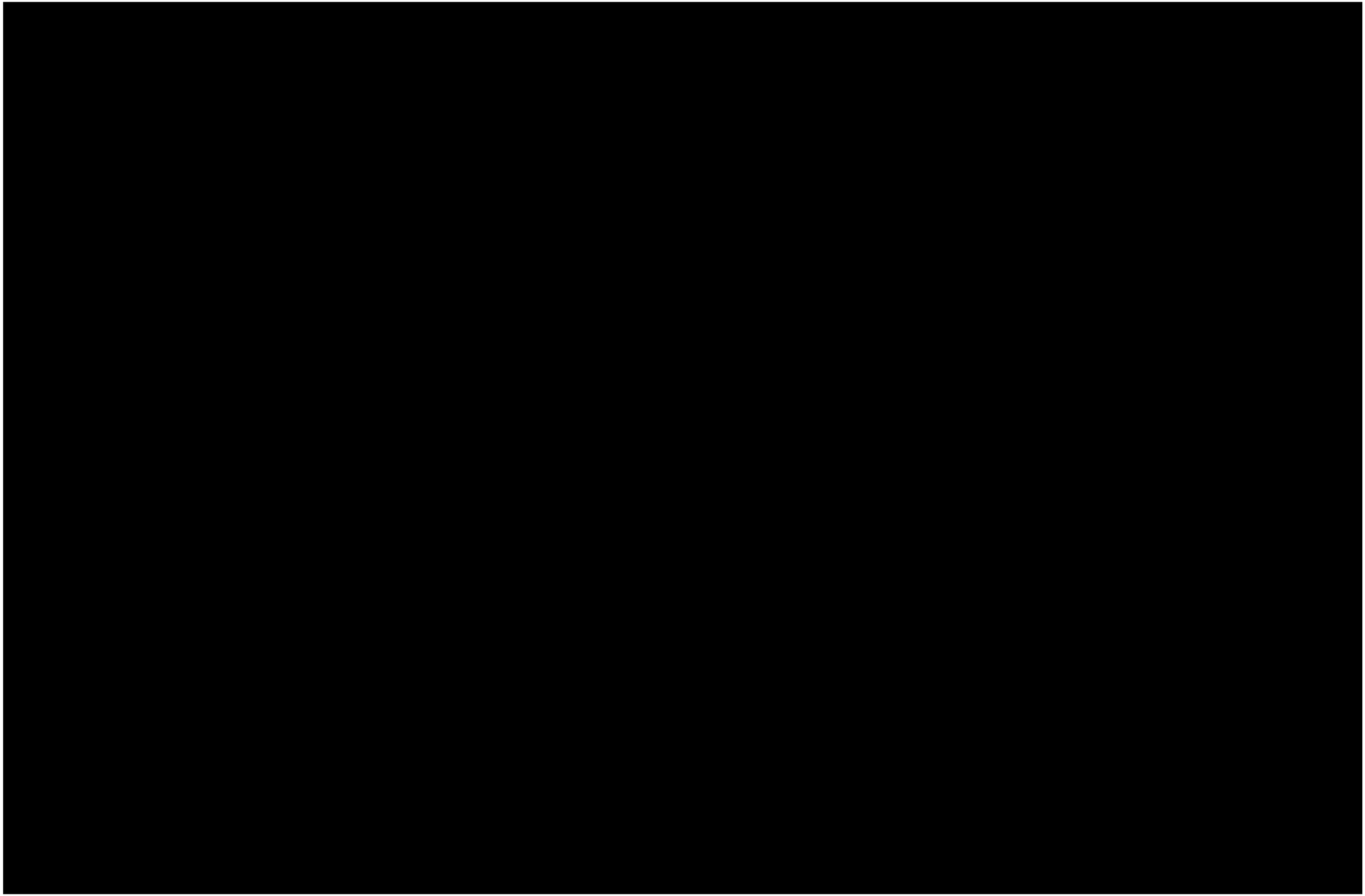
Isotope	Release Rate (µCi/sec)
Kr-83m	0.34
Kr-85m	0.6
Kr-85	0.0024
Kr-87	1.9
Kr-88	4.3
Kr-89	2.0
Xe-131m	0.0015
Xe-133m	0.028
Xe-133	0.82
Xe-135m	2.1ú
Xe-135	2.2
Xe-137	6.3
Xe-138	7.1
I-131	0.0025
I-132	0.023
I-133	0.015
I-134	0.048
I-135	0.025

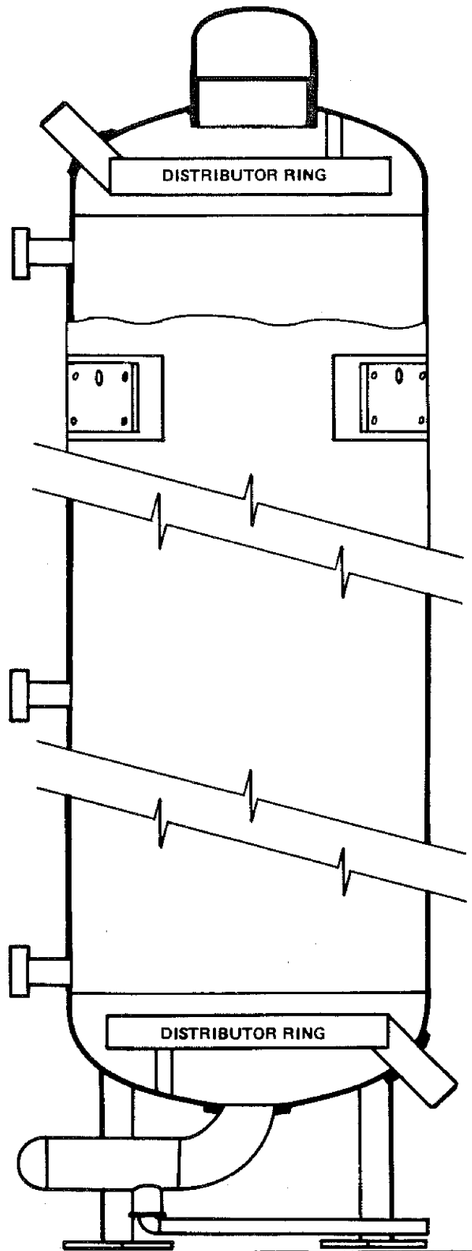
Table 11.3-10

DRYWELL PURGE ISOTOPIC RELEASE
(Four purge/per year, corresponding to 25,000 μCi/sec at 30-min decay offgas rate)

Isotope	Release (μCi/sec)
Kr-83m	3.1×10^2
Kr-85m	1.3×10^3
Kr-85	1.4×10^3
Kr-87	1.3×10^3
Kr-88	2.8×10^3
Xe-131m	2.1×10^2
Xe-133m	7.7×10^2
Xe-133	5.1×10^4
Xe-135m	3.3×10^2
Xe-135	9.9×10^3
Xe-138	1.3×10^3
I-131	1.0×10^4
I-132	1.1×10^3
I-133	7.5×10^3
I-134	8.4×10^2
I-135	4.3×10^2







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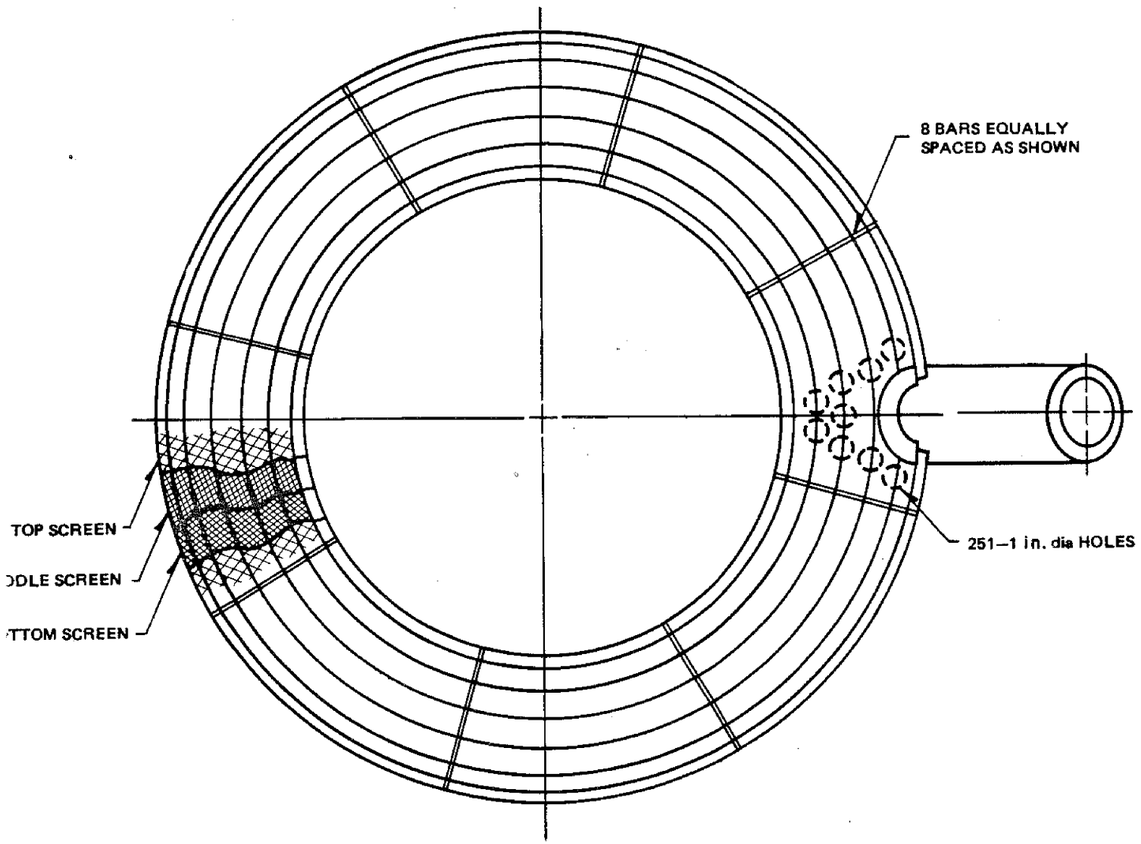
UPDATED FINAL SAFETY ANALYSIS REPORT

Charcoal Adsorber Vessel Assembly

Figure 11.3-4



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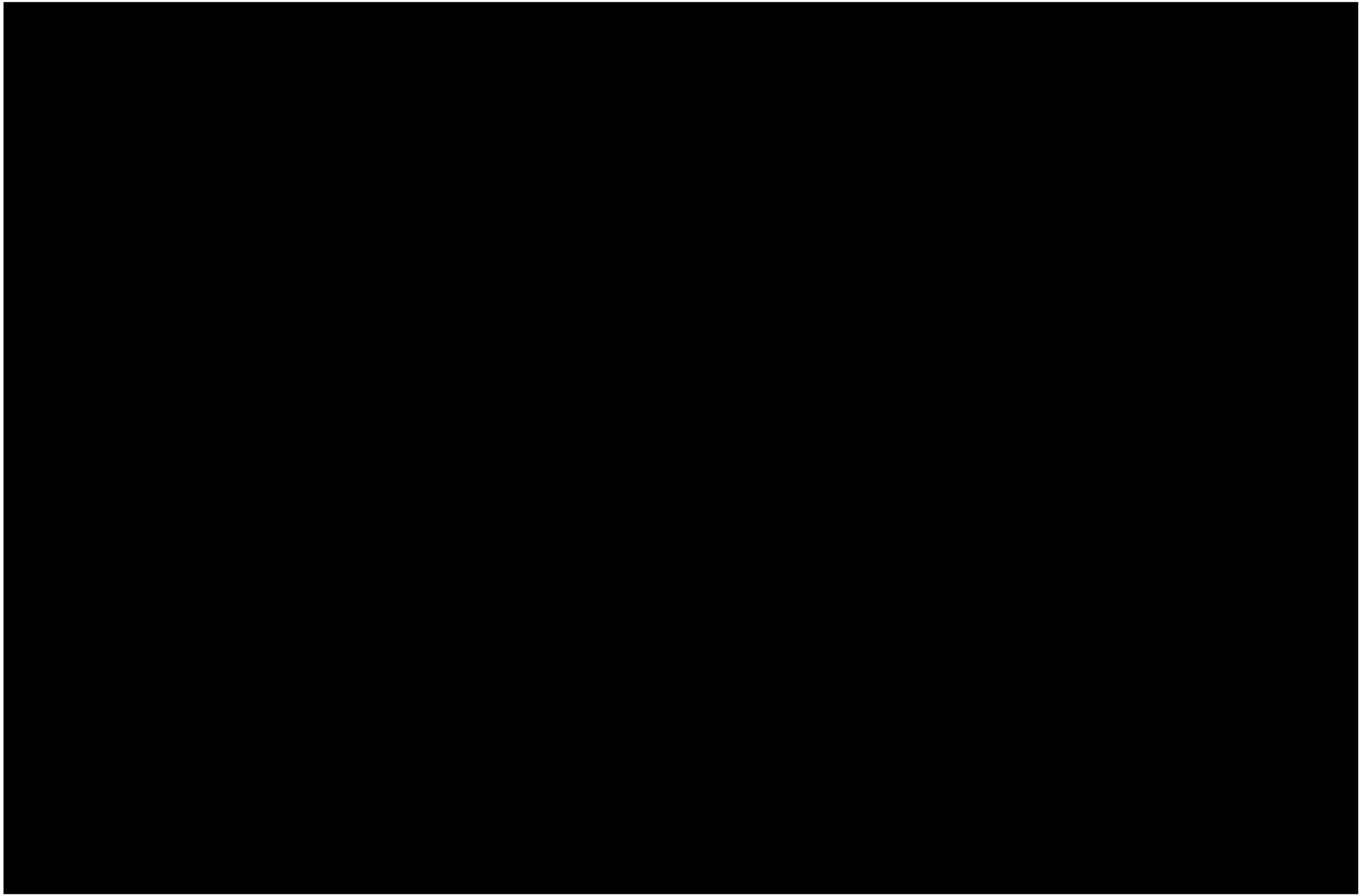
Distributor Ring

Figure 11.3-5



11.4 SOLID WASTE MANAGEMENT SYSTEM

11.4.1 DESIGN BASES

11.4.1.1 Power Generation Objectives

The power generation objectives of the solid radwaste system are to collect, process, and package potentially radioactive solid wastes in a safe manner for offsite shipment and permanent disposal.

11.4.1.2 Power Generation Design Bases

Solid radwaste material is packaged for on-site storage in the Radwaste Building or the low-level radwaste processing and storage facility (LLRPSF) or for shipment and offsite disposal in accordance with applicable published regulations.

The maximum expected isotopic curie inventories in solid wastes per year of operation are given in Table 11.4-1. The average isotopic curie inventories are expected to be approximately one-fourth of the expected maximum listed.

The decay periods assumed for the calculation of the solid waste isotopic activity inventories are listed below:

1. Spent resin - no decay.
2. Cleanup filter-demineralizer backwash - 60 days decay.
3. Fuel pool filter-demineralizer - no decay.
4. Floor drain filter backwash - no decay.
5. Condensate filter-demineralizer - no decay.
6. Waste filter - no decay.

The total average expected curies per year for various waste sources are as follows:

Waste Source	Calculated Total Yearly Activity (ci)
Fuel pool filter-demineralizer	340
Floor drain demineralizer spent resins	0.325
Floor drain filter	7.5×10^{-4}
Waste demineralizer spent resins	125
Waste filter	0.4
Condensate filter-demineralizer	60
Cleanup filter-demineralizer	4.5×10^{2}
Chemical and detergent	6.5
Subtotal	~1000
Dry waste	90

The estimated maximum weight and volume of solid waste processed in the radwaste system are 63,000 lb/yr and 2200 ft^3/yr, respectively. These estimates are based on fuel defects corresponding to an offgas rate (30-min decay) of 100,000 µCi/sec and consist of filter-demineralizer backwash sludges (condensate, cleanup, and fuel pool), radwaste filter sludges, and radwaste demineralizer spent resins. However, variations in the amount of solid radioactive waste generated and shipped from year to year are expected based on the overall performance of the plant and the number and scope of outages and maintenance activities.

11.4.1.3 Safety Design Basis

The solid radwaste system is designed so that any quantities of solid radwastes inadvertently released do not result in concentrations in excess of the guideline values of 10 CFR 20.

11.4.2 SYSTEM DESCRIPTION

11.4.2.1 General

The solid waste areas are located in the radwaste building, the LLRPSF, and the offgas retention building. The system processes wet and dry solid radwastes. The wet solid wastes are the spent demineralizer resins and filter sludges that are byproducts of plant water-treatment processes including radwaste. The dry solid radwastes consist of other miscellaneous radioactive or contaminated solid wastes. Because of differences in radioactivity or contamination levels of the many wastes, various methods are employed for processing and packaging.

Dry Active Waste (DAW) process on-site is packaged for shipment to an off-site processor and/or packaged for shipment to a licensed disposal facility. Packaging and shipping is performed in accordance with applicable NRC and DOT regulations. Waste form is determined by the applicable disposal criteria of the licensed disposal facility.

Wet Wastes are processed in order to achieve a stable form in accordance with 10CFR61, and applicable site disposal criteria for the licensed disposal facility.

11.4.2.2 Wet Wastes

Wet wastes consist of spent demineralizer resins and filters and filter-demineralizer sludge wastes.

Sludge wastes are removed from filters and demineralizers as these components are backwashed. Sludge wastes from the reactor water cleanup system and condensate treatment system are collected in phase separators. Sludge wastes from the fuel pool cooling and cleanup system are collected in the waste sludge tank. Sludge from the radwaste system waste demineralizer and floor drain demineralizer is collected in the spent-resin tank or the waste sludge tank. Sludge from the waste collector filter and floor drain filter is collected in the waste sludge tank and floor drain sludge tank, respectively.

The sludge wastes consist of spent ion-exchange resins, corrosion products, fission products, and other insoluble material removed from the various systems. The reactor water cleanup system sludges are kept separate from the condensate and fuel pool system sludges because of the variation in radioactive material content. This reduces shielding requirements for the storage and shipping of the lower activity solid wastes.

The excess backwash water from the sludge wastes collected in the phase separators is decanted and transferred to the collection tanks. The concentrated sludge that remains after each batch is decanted is held until the resin volume reaches a predetermined level. The sludge is processed to remove the excess water, the solids are prepared for disposal, and the extracted water is routed back to the collection tanks.

Solidification of waste sludge is a method of waste stabalization. Solidification of the sludge is achieved by first removing a portion of the water from the sludge. Cement is added to the waste. The waste will then become a solid mass. The waste is then prepared for loading and offsite shipment. Dewatering of sludges is a method of waste stabalization.

The dewatering equipment is located in the LLRPSF. The waste is transferred from the waste holding tank to a HIC or liner located in the dewatering pit in the storage portion of the LLRPSF. The dewatering system uses DAEC service air which exhausts into the LLRPSF ventilation system. Water removed from the HIC or liner is returned to the radwaste system via the conveyor floor drain sump.

Dewatering is performed by pulling a suction on an underdrain manifold in the HIC. The dewatering process achieves a residual of less than 1% freestanding water in the waste, which meets the requirements of 10 CFR 61.56(a)(3) and (b)(2).

When dewatering is complete, the liner or HIC can be stored in the LLRPSF storage area or moved to the truck bay for loading and offsite shipment by a licensed carrier to a licensed disposal site in accordance with applicable regulations of the Department of Transportation and the NRC.

HICs are used for packaging of solid waste during temporary onsite storage, shipment, and permanent offsite storage because of ready availability, ease of handling, and conformance with present shipping practices and disposal site requirements.

Loading of HICs and drums for offsite shipment is done within the radwaste building or the LLRPSF.

11.4.2.3 Dry Wastes

Miscellaneous solid wastes result from operation and maintenance, and a means for handling and disposal are necessary to ensure proper control and prevent the spread of contamination. Typical of these wastes are air filters; miscellaneous paper, rags, etc., from contaminated areas; contaminated clothing, tools, and equipment parts, which cannot be economically decontaminated; solid laboratory wastes; used reactor equipment such as spent control rod blades, fuel channels, and incore ion chambers; and large pieces of equipment.

The disposition of a particular item of waste is determined by its radiation level, type, and the availability of disposal space. Because of high activation and contamination levels, used reactor components are stored in the spent-fuel pool for sufficient time to obtain optimum radioactive decay before removal to either in-plant, on-site or offsite storage and final disposal. Otherwise, the wastes need to be held on the site only until quantities large enough for economical shipment are accumulated.

Many solid radwastes can be handled manually because of low-radioactivity content or minimal contamination levels. Material that can be compressed is compacted into 55-gal drums
2016-009 | by a hydraulic press in the radwaste building (Figure 11.4-1). Compressable material may also be directly loaded into a shipping container for shipment to a waste processor.

11.4.2.4 Storage Facilities

Solid radwaste storage facilities are located in the radwaste building and the LLRPSF. The LLRPSF has a storage section and a processing section.

The storage section is a steel and concrete structure whose major structural components are:

1. A pile-supported reinforced concrete foundation mat;

2. Reinforced concrete on metal deck floor systems, supported by structural steel framing;

3. Reinforced concrete exterior and interior walls; and

4. A reinforced concrete slab on metal deck roof system, supported by steel framing.

The top of the foundation mat is at [redacted], and the building extends up to [redacted].

The process section of the LLRPSF is a braced-frame structural steel and concrete structure, consisting of the following major components:

1. A reinforced concrete footing foundation supported on piles;

2. Reinforced concrete on metal deck floor systems;

3. Concrete block interior walls and insulated metal siding exterior walls; and

4. A roof system consisting of elastomeric sheet roofing with rigid insulation on metal deck, supported by steel framing.

Seismic loads used in the design of the LLRPSF conform to the criteria of the Uniform Building Code, Zone 1.

Truck loading in the radwaste building is accomplished with an overhead, motorized bridge crane that is pendant controlled by an operator in the truck loading area or remote controlled from the crane control room.

The available storage capacity for compressible dry wastes in the radwaste building is approximately 75 drums (about 560 ft^3). Allowance is also made for noncompressible dry wastes to be stored in this area. These wastes are manually loaded when quantities large enough for economical shipment have accumulated with no particular delay time anticipated.

Storage capacity is also provided for resin and DAW in the LLRPSF. The available storage capacity is based on a stacking configuration for both the resin and DAW vaults. Approximately 12,750 cu ft of resin and 23,520 cu ft of DAW can be stored in the LLRPSF. It is planned that oil will also be stored in the storage vaults and the amount is included in the above totals.

A remotely-operated bridge crane is provided in the storage portion of the LLRPSF to transfer containers from either the resin or DAW storage vaults to the truck bay to ship waste off-site. Although the crane can be used to handle DAW, it is anticipated that a fork-lift will be used to transfer these waste containers to the truck bay for off-site shipment.

11.4.2.5 Safety Evaluation

The shielding design in the solid radwaste system working area limits radiation exposure to personnel to within the guideline values of 10 CFR 20. The operation of the system is by semi-remote means in the radwaste building. Solid wastes are handled by the remotely-operated bridge crane or by lift truck in the LLRPSF. The operating procedures and containment and storage facilities of the system limit the concentrations that could result from any inadvertent or accidental release of radioactive materials to within the guideline values of 10 CFR 20.

11.4.2.6 Inspection and Testing

The solid radwaste system is used on a routine basis and does not require specific testing of its operability.

Table 11.4-1

MAXIMUM ISOTOPIC ACTIVITY INVENTORIES IN SOLID WASTES

Initial Design		Average Isotopic Curies of Shipped Waste: 2000-2009	
Isotope	Activity (Ci/yr)	Isotope	Activity (Ci/yr)
Sr-89	1.9E+02	Co-60	7.2E+02
Sr-90	1.3E+02	Cs-137	1.0E+00
Sr-91	1.4E+01	Fe-55	9.2E+02
Zr-95	9.5E+00	Mn-54	3.5E+01
Nb-95	3.5E+00	Ni-63	9.6E+01
Mo-99	1.3E+02	Zn-65	2.3E+01
Tc-99m	1.2E+01		
Ru-103	2.8E+00		
Ru-106	1.2E+00		
Te-129	4.7E+00		
Te-132	3.4E+02		
Cs-134	8.3E+01		
Cs-136	2.9E+00		
Cs-137	1.3E+02		
Ba-140	2.6E+02		
Ce-141	4.4E+00		
Ce-144	2.6E+01		
Np-239	1.2E+03		
I-131	2.0E+02		
I-133	8.9E+01		
I-135	8.0E+00		
Mn-54	1.3E+01		
Cr-51	2.6E+01		
Fe-59	8.4E+00		
Co-58	8.9E+02		
Ag-110m	1.8E+01		
Co-60	2.0E+02		
Total	4.0E+03	Total	1.8E+03

Notes:

1. Isotopic curies of initial design were estimated on the basis of plant operation at the stack release rate of 100,000 μCi/sec (after 30-min holdup) annual average throughout the year.
2. Decay is applied for the duration of the design storage within the systems but does not account for decay occurring after drumming and before shipment.
3. Those isotopes with annual curies less than 1 have been omitted from the list.
4. The average isotopic curies of shipped waste for 2000-2009 includes a fuel pool clean up campaign, which greatly increases the average curies shipped in this time period.

[redacted]

11.5 PROCESS AND EFFLUENT RADIATION MONITORING AND SAMPLING SYSTEMS

A number of radiation monitors and monitoring systems are provided on process liquid and gas lines that may serve as discharge routes for radioactive materials. These include the following:

1. Main steam line radiation monitoring system.

2. Air ejector offgas radiation monitoring system.

3. Offgas stack radiation monitoring system.

4. Process liquid radiation monitors.

5. Reactor building ventilation radiation monitoring system.

6. Extended range airborne effluent radiation monitoring system.

7. Turbine building ventilation radiation monitoring system.

8. Liquid radwaste discharge radiation monitoring system.

The DAEC Offsite Dose Assessment Manual[1] contains the methodology and parameters used in the calculation of offsite doses due to radioactive gaseous and liquid effluents and in the calculation of radioactive gaseous and liquid effluent monitoring instrumentation alarm and trip setpoints. The Technical Specifications implement operating procedures and limiting conditions for operation to keep levels of radioactive materials in gaseous and liquid effluents as low as is reasonably achievable.

11.5.1 MAIN STEAM LINE RADIATION MONITORING SYSTEM

11.5.1.1 Safety Objective

As approved in Amendment 261, this system no longer has a safety function.

11.5.1.2 Design Bases

1. The main steam line radiation monitoring system is designed to give prompt indication of a gross release of fission products from the fuel.

2. The main steam line radiation monitoring system is able to detect a gross release of fission products from the fuel under any anticipated operating combination of main steam lines.

11.5.1.3 System Description

Four gamma-sensitive instrumentation channels monitor the gross gamma radiation from the main steam lines. The detectors are physically located near the main steam lines just downstream of the outboard main steam line isolation valves in the space between the primary containment and secondary containment walls. The detectors are geometrically arranged so that the system is able to detect significant increases in radiation level with any number of main steam lines in operation. Their locations along the main steam lines allow the earliest practical detection of a gross fuel failure. Two of the channels are powered from one reactor protection system bus, and the other two channels are powered from the other reactor protection system bus.

On receipt of the high-high radiation alarms from one channel on each trip system bus, trip signals are transmitted to the primary containment isolation (PCI) system and the offgas system. The mechanical vacuum pump is tripped. The trip of the mechanical vacuum pump in turn closes its suction valve from the low pressure and high pressure condenser. The main steam line drain valves and recirculation loop sample valves also close on high radiation. As approved in Amendment 261, credit is no longer taken for these trip functions in the accident analysis.

Each monitoring channel consists of a gamma-sensitive ion chamber and a logarithmic (log) radiation monitor as shown in Figure 11.5-1. Monitoring channel capabilities are listed in Table 11.5-1. Each log radiation monitor has three trip circuits. One trip circuit comprises the upscale trip high-high setting or instrument inoperative trip, which actuates an alarm in the main control room along with the trip signals discussed above. The second trip circuit is a downscale (low) trip that actuates an instrument trouble alarm in the main control room. An upscale (high) alarm setting, lower than the log radiation monitor upscale (high high) trip setting, actuates an alarm in the main control room. The output from each log radiation monitor is displayed on a six-decade meter in the main control room.

A two-pen recorder is used to record the outputs from two of the four monitoring channels. Manual selector switches allow the output of one of the channels in each trip system to be recorded.

The trip circuits for each monitoring channel operate normally energized, so that interruptions in the power to monitoring components result in a trip signal. The environmental capabilities of the components of each monitoring channel are selected in consideration of the component locations.

11.5.1.4 Safety Evaluation

As approved in Amendment 261, this system no longer has a safety function. Chapter 15.2.4 contains the updated analysis of the design basis rod drop accident.

11.5.1.5 Inspection and Testing

A built-in, adjustable current source is provided with each log radiation monitor for test purposes. Routine verification of the operability of each monitoring channel can be made by comparing the outputs of the channels during power operation.

11.5.2 AIR EJECTOR OFFGAS RADIATION MONITORING SYSTEM

11.5.2.1 Power Generation Objectives

The power generation objectives of the air ejector offgas radiation monitoring system are to indicate when limits for the release of radioactive material to the environs are approached and to effect appropriate control of the offgas so that the limits are not exceeded.

11.5.2.2 Power Generation Design Bases

1. The air ejector offgas pre-treatment radiation monitoring system provides an alarm to operations personnel when the radioactivity discharge rate exceeds 1 Curie/second after 30 minutes of delay and decay.

2. The air ejector offgas radiation monitoring system provides a record of the radioactivity released via the air ejector offgas line.

3. The air ejector offgas post-treatment radiation monitoring system initiates appropriate action in time to prevent exceeding the maximum instantaneous release rate limit of radioactive materials to the environs from the air ejector offgas.

11.5.2.3 System Description

The air ejector offgas radiation monitor system is shown in Figures 11.5-1 and 11.3-2, and specifications are given in Table 11.5-1. The offgas is monitored both before and after the recombiner/carbon-bed treatment. The monitoring system used before treatment is comprised of one instrument channel monitoring the gases passing through a vertical section of stainless steel pipe designed to minimize plateout. A sample is drawn from the offgas line through the sample chamber by the main condenser suction. The sample system is arranged to give at least a 2-min time delay before the sample is monitored to allow nitrogen-16 and oxygen-19 activity decay. This reduces the background radiation that the detector would otherwise measure. The channel consists of a gamma-sensitive ion chamber, a log radiation monitor that includes a power supply and a meter, and a paperless chart recorder. The monitor and the paperless recorder are located in the main control room.

The monitor has two upscale trip circuits, one downscale trip circuit and an instrument inoperative trip. The upscale trips indicate high and high-high radiation, the downscale trip (low) indicates instrument trouble or a dose rate below the downscale setpoint and the instrument inoperative trip indicates instrument trouble. Any one trip will give an alarm in the main control room.

The monitoring system used after the recombiner/carbon-bed treatment is comprised of two independent channels monitoring gases passing through a sample chamber mounted on a sample rack along with pump, flow measuring and control equipment, check sources, purge equipment, scintillation detectors, and pre-amplifiers. Each channel is comprised of a detector, a pre-amplifier, a log count-rate monitor including power supply and meter, and one channel of a paperless two channel recorder. The detectors monitoring the process after treatment are gamma-sensitive scintillation detectors. The monitors for these channels are seven-decade log count rate monitors located in the control room with three adjustable upscale trip circuits, one downscale trip circuit, and an instrument inoperative trip. An instrument failure gives a downscale trip or an inoperative trip. If either channel experiences the lower level upscale trip (high), the carbon bed bypass line will close, the treatment line isolation valve will open and an alarm will be received. The intermediate upscale trip (high-high) is used to alarm only. If an inoperative, downscale or upscale trip (high-high-high) occurs on one channel along with any of the same trips on the other channel, an alarm will occur. The offgas post-treatment radiation monitor system is shown schematically in Figure 11.5-2.

The electronic signals from the post-treatment radiation monitors (Figure 11.5-1) feed the "Treat, Auto, and Bypass" remote manual switch, which controls valves [redacted](treat) and [redacted] (bypass) (Figure 11.3-2). The automatic provisions of shifting from bypass to treat are only applicable in the auto mode.

The following are examples of system operation using the above-described logic:

1. If the offgas system is operating with valve [redacted] open and valve [redacted] closed (bypass mode configuration), and the mode switch is in the auto position, on the receipt of a low-radiation alarm signal, no change in valve positions would occur. On the receipt of a high-radiation alarm signal, valve [redacted] would close and valve [redacted] would open, thereby directing the offgas process stream through the charcoal beds for treatment. If the high-radiation alarm is cleared and it is determined that the operator desires to return the system to bypass, the operator may do this by resetting the alarm.

2. If the offgas system is operating with valve [redacted] open (charcoal bed process valve) and valve [redacted] (bypass valve) closed, and the mode switch is in the treat mode, on the receipt of a low-radiation signal or a high-radiation signal, no change in valve positions would occur. The offgas system mode switch will normally be maintained in the treat position. The auto and bypass provisions are retained for system flexibility so that plant reliability is not compromised.

The operation of the offgas system with the offgas system mode switch in the bypass position is not normally permitted.

Small changes in the offgas gross fission product concentration can be detected by the continuous use of the linear (flux tilt) radiation monitor. The linear radiation monitor is not a process monitor such as the channels described above but is used as an expanded scale device for aiding in the measurement of small changes in the offgas radiation level. The detector is a gamma-sensitive ionization chamber that monitors the same sample as the air ejector offgas detectors monitoring the process before treatment. The system uses a linear readout with a range switch instead of a logarithmic readout. The output from the monitor is recorded on a one-pen recorder. (Improved fuel sipping technology and the poor sensitivity of the flux tilt monitoring process have made the use of this monitor obsolete. This equipment has been abandoned in place except for the recorder which has been removed from the plant.)

The carbon vault is monitored for gamma activity with a single instrument channel. The channel includes a sensor and converter, an indicator and trip unit, and a locally mounted auxiliary unit. The power source is one of the power supplies associated with the refuel pool ventilation exhaust radiation monitors. The indicator and trip unit is located in the main control room. The channel provides for sensing and readout, both local and remote, of gamma radiation over a range of six logarithmic decades (1 to 10^6 mR/hr).

The indicator and trip unit has one adjustable upscale trip circuit for alarm and one downscale trip circuit for instrument trouble. The trip circuits provide convenient operational verification by means of test signals or through the use of portable gamma sources. All components are self-monitoring to the extent that power failure to any component operates the trip circuits.

The following applies to effluent and in-plant gaseous radwaste sampling locations subject to periodic sampling:

1. There are four general areas where periodic samplings are taken:

   a. Last stage of steam jet air ejectors before dilution steam to obtain data on "raw" offgas process steam.

   b. Just upstream of 30-min holdup pipe to acquire data on recombined offgas stream.

c. Just before entry to charcoal bed vault to obtain data before gas treatment through charcoal bed adsorbers.

d. Two sample points in vault downstream of first bed in each adsorber train to monitor performance of first bed.

2. Expected composition and respective concentrations are one-fourth of those contained in Tables 11.3-2 and 11.3-6.

3. Quantity measurements for atmospheric releases are discussed in Section 1.8.2.1.

4. Frequency of measurements for atmospheric releases are also discussed in Section 1.8.

11.5.2.4 Power Generation Evaluation

The air ejector offgas radiation monitors have been selected with characteristics sufficient to provide plant operations personnel with accurate indication of radioactivity in the air ejector offgas. The system thus provides the operator with enough information to easily control the activity release rate. Sufficient redundancy is provided to allow maintenance on one channel without losing the indications provided by the system.

11.5.2.5 Inspection and Testing

Each channel can be calibrated by the analysis of a grab sample.

11.5.3 OFFGAS STACK RADIATION MONITORING SYSTEM

The offgas stack radiation monitoring system described herein is in addition to the offgas stack radiation monitoring subsystem of the extended range airborne effluent radiation monitoring system described in Section 11.5.9.

11.5.3.1 Power Generation Objectives

The power generation objectives of the offgas stack radiation monitoring system are to indicate when limits on the release of radioactive material to the environs are reached or exceeded and to indicate the rate of radioactive material released during planned operation.

11.5.3.2 Power Generation Design Bases

1. The offgas stack radiation monitoring system provides a clear indication to operations personnel when limits on the release of radioactive material to the environs are reached or exceeded.

2. The offgas stack radiation monitoring system indicates the rate of release of radioactive material from values above release rate limits down to the release rates normally encountered during high-power operation.

3. The offgas stack radiation monitoring system records the rate of release of radioactive material to the environs, so that the determination of the total amounts of activity released is possible.

11.5.3.3 System Description

The offgas stack radiation monitoring system is shown in Figures 11.5-1 and 11.3-2, and specifications are given in Table 11.5-1. The system consists of two individual channels. Each channel consists of a gamma-sensitive scintillation detector, a pulse preamplifier, a log count rate monitor that includes a power supply and a meter, and a strip chart recorder pen. The monitors and the two-pen recorder are located in the main control room.

Each monitor has two upscale trip circuits and one downscale trip circuit (including instrument in operation). Each trip circuit initiates an alarm in the main control room. The upscale alarms indicate high and high-high radiation, and the downscale alarm (low) indicates instrument trouble. The high-high upscale trip circuits also close containment isolation valves in the lines listed in Section 7.3.1.1.1.8, item 15.

To monitor the offgas stack stream, a gas sample is drawn through an isokinetic probe that is located high enough in the vent pipe stream to ensure representative sampling. The sample passes through two shielded chambers where the radiation level of the stack gas is measured by two scintillation detectors, one located in each shielded chamber.

The system also monitors iodine and particulates by using filters in the gas sample stream. The filters are routinely analyzed in a laboratory.

Gaseous radwaste monitoring is described below:

1. The following two continuous monitors are provided for the offgas system, as indicated in Figure 11.3-2:

   a. Offgas stack monitor with isokinetic probe to monitor gaseous discharges just before release to the atmosphere.

   b. Posttreatment sample point to monitor the stream after treatment and before entry to stack.

2. Concentrations at these detectors are one-fourth of those presented in Table 11.3-1.

3. Gross concentration measurements are made continuously at both detectors.

4. Both monitors are scintillation detectors with a range of 0.1 to 10^6 counts per second (cps). The sensitivity and the corresponding detectable level is 28 cps and approximately 60 μCi/sec.

The offgas stack probe is near the bottom of the offgas stack, and a gas sample is collected at this point. A particulate filter and an iodine filter are near this probe. A similar set of filters is at the offgas sample rack, which contains dual-gas-sample chambers. The counts between the two filter set locations are used to determine sample losses. In operating BWR plants, gas sample losses have been found to be negligible to date. The detectors are in this sample rack.

5. Exact setpoints are compatible with the Technical Specification release rates.

6. All annunciators and alarms for the gaseous radwaste system in the main control room are made by Lundell.

11.5.3.4 Power Generation Evaluation

The offgas stack radiation monitors have been selected with characteristics sufficient to provide plant operations personnel with accurate indication of radioactivity being released to the environs via the offgas stack. The system thus enables the operator to easily control the activity release rate. Sufficient redundancy allows maintenance on one channel without losing the indication provided by the system.

11.5.3.5 Inspection and Testing

Each individual channel includes a built-in check source and a purge line to purge the stack gas from the sampling chamber.

Both the purge valve and the check source are operated from the main control room.

The offgas stack radiation monitors are calibrated and tested as prescribed in the Technical Specifications.

11.5.4 LIQUID PROCESS RADIATION MONITORS

11.5.4.1 Power Generation Objectives

1. On process streams that do not discharge to the environs, process liquid radiation monitors are provided to indicate process system malfunctions by detecting the accumulation of radioactive material in a normally uncontaminated system.

2. On process streams that normally discharge to the environs, liquid process radiation monitors are provided to indicate when preestablished limits for the release of radioactive material are exceeded.

11.5.4.2 Power Generation Design Bases

1. Liquid process radiation monitors located in streams that do not discharge to the environs provide a clear indication to operations personnel whenever the radioactivity level in the

stream reaches or exceeds a preestablished limit above the normal radiation level of the stream.

2. Liquid process radiation monitors located in streams that normally discharge from the liquid radwaste system to the environs provide a clear indication to operations personnel whenever the radioactivity level in the stream reaches or exceeds preestablished limits for the discharge of radioactive material. Actual control of liquid release is via sampling, analysis, and controlled flow.

11.5.4.3 System Description

The liquid process radiation monitor specifications are given in Table 11.5-1. Five individual channels are provided. One channel monitors the plant service water stream, another channel monitors the reactor building closed cooling water system, the third monitors the discharge from the liquid radwaste system, the fourth monitors the residual heat removal service water and emergency service water systems, and the fifth monitors the residual heat removal service water system rupture disk.

2011-020 Each channel has a scintillation detector, a pulse pre-amplifier, a radiation monitor, and a recorder. Each detector is located in a shielded sampler that is positioned in the process liquid piping so as to minimize the effect of background radiation. The monitors and the recorders are all in the main control room except for the radwaste system radiation monitor recorder, which is on the radwaste main control panel.

Each channel has an upscale trip to indicate high-radiation level and one downscale trip (low) to indicate instrument trouble. The trips give an alarm but no control action except in the case of [redacted], which closes the radwaste discharge valves on high radiation.

The process radiation monitoring system and the radwaste system terminate discharge on the receipt of a signal that the radioactivity concentration in the effluent line equals or exceeds a predetermined value.

Plant service water cools normally nonradioactive areas such as air compressors, turbine auxiliary systems, and pump bearings. It also cools the reactor building closed cooling water system via a heat exchanger. An increase in the radiation level of the service water stream may indicate that a leak into the system from a contaminated stream has occurred.

The reactor building closed cooling water system is used to provide cooling for potentially contaminated areas such as the nonregenerative heat exchanger, recirculation pumps, and various sample coolers. Changes in the normal radiation level could indicate leaks of radioactive water into the system.

The RHR service water system provides coolant to the RHR heat exchangers. The emergency service water system provides coolant for all emergency equipment except the RHR heat exchangers (see Section 9.2.3.2.2). Changes in the normal radiation level could indicate leaks of radioactive water into the system.

The liquid radwaste system provides for the collection of waste liquids through various drainage systems. An analysis is made of each radwaste batch before release. This analysis determines if the batch may be released, and if it is to be released, at what flow rate. In addition, a process liquid radiation monitor, such as shown in Figure 11.5-1, is located in the sample tank discharge line upstream of the junction with the effluent line. This monitor indicates discharge radiation levels.

2015-010 The control of the radioactivity concentration in the liquid wastes released to the environment through the dilution structure to the discharge pipe is dependent on a knowledge of the concentration and volume of the radioactive materials in the tank to be discharged. The volume of liquid in the tank is known from tank level instrumentation. Before the discharge of a given batch, the radioactivity concentration is measured by taking a sample from the batch in the tank and analyzing it for radioactivity in the laboratory.

2015-010 The rate of release of the liquid waste to the discharge pipe that will produce a concentration in the pipe after dilution equal to the allowable concentration limit is then calculated. A rate of release below the calculated maximum allowable rate is then selected and used for the particular batch of liquid waste to be discharged to the environment.

When the proper release rate has been determined for a given batch of liquid radwaste, the desired flow rate is set on one of the two flow controllers, and the high-radiation monitor alarm and trip point are correspondingly set such that the radioactivity concentration after dilution will be in conformance with the requirements of the Technical Specifications.

During the release, the downstream radiation monitor in the discharge line continuously measures, indicates, and records the radioactivity level in the liquid being discharged. The liquid waste discharge flow rate is also indicated and recorded to ensure that proper dilution of discharged radioactive waste has been attained.

If the preset high activity level of the liquid radwaste effluent is exceeded, the discharge valve will automatically close to terminate the release of liquid radwaste, and an alarm will sound in both the main and radwaste control rooms.

1. The detector is located in the effluent pipe before dilution. It is positioned as an aid to station operating personnel in checking waste discharge concentrations from the batch discharge process.

2. Depending on the source of waste discharge at any given time, the concentration can vary from 2 x 10^{-3} μCi/cm^3 to 10^{-5} μCi/cm^3, as shown in Figure 11.2-1.

3. The quantities measured by this monitor are gross concentration only.

4. The detector type, sensitivity, and range are given below. As stated in preceding item 1, the detector is positioned inside the liquid radwaste pipe. Since the effluent is discharged further downstream, the detector will show higher activities at its actual location.

System	Detector Type	Range	Sensitivity (Corresponding Detectable Level)
Liquid discharge	Scintillation detector	0.1-10^6 cps	220 cps (2.67 x 10^{-6} μCi/cm^3)

5. Liquid radwaste monitor setpoint and basis for selection is as follows:

Setpoint	Basis
Field set dependent on batch concentration and dilution flow such that concentration after dilution is less than 10^{-7} μCi/cm^3 for gross beta/gamma analysis or 10 times the water effluent concentrations in 10 CFR 20, Appendix B.	In conformance with the Offsite Dose Assessment Manual and Technical Specifications.

6. All annunciators and alarms for the liquid radwaste system located in the control room are made by Lundell.

The radioactive wastes released from the plant are controlled in such a manner that the monitoring of the effluent after mixing with the dilution water is unnecessary.

Locations of liquid radwaste sample points subject to periodic sampling, both effluent and in the plant, are as indicated in Figures 11.2-3, 11.2-4, and 11.2-5. The radwaste sample station is presented in Figure 11.5-3.

1. The locations for liquid sampling points are chosen to enable operating personnel to decide on what treatment process and what routes will be selected to handle a given batch to be processed. They are also used as a check on how the processing is going once the treatment process has been selected for the given batch.

2. Expected compositions and concentrations for 1 yr of operation are one-fourth of those presented in Table 11.2-6.

3. For liquid releases, isotopic concentrations will be determined for the sample tanks as detailed in Section 1.8, Response to AEC Safety Guide No. 21. Collector tanks will be sampled for gross concentrations to determine the necessary batch process. Sample tanks will be sampled for gross activity to determine the effectiveness of the process and to determine if further processing is necessary.

4. Liquid releases will be sampled before release. Other samples will be taken before and after processing. Maximum tank activity and sampling frequency shall be in accordance with the plant Technical Specifications.

11.5.4.4 Power Generation Evaluation

The liquid process radiation monitors for the service water and liquid radwaste discharges possess radiation detection and monitoring characteristics sufficient to inform plant operations personnel whenever radiation levels in the discharges rise above preset limits.

11.5.4.5 Inspection and Testing

All alarm trip circuits can be tested by using test signals or portable gamma sources.

Radioactive liquid waste sampling and activity analyses are performed in accordance with the Technical Specifications. The liquid radwaste monitor is calibrated and tested in accordance with the Technical Specifications.

11.5.5 REACTOR BUILDING VENTILATION RADIATION MONITORING SYSTEM

11.5.5.1 Power Generation and Safety Objectives

The power generation and safety objectives of the reactor building ventilation radiation monitoring system are to indicate whenever abnormal amounts of radioactive material exist in the reactor building ventilation exhaust and to effect appropriate action so that the release of radioactive material to the environs is controlled.

11.5.5.2 Design Bases

1. The reactor building ventilation radiation monitoring system provides a clear indication to operations personnel whenever abnormal amounts of radioactivity exist in the reactor building ventilation exhaust.

2. The reactor building ventilation radiation monitoring system initiates appropriate action to control the release of radioactive material to the environs when abnormal amounts of radioactive material exist in the reactor building ventilation exhaust.

11.5.5.3 System Description

The reactor building ventilation radiation monitoring system is shown in Figure 11.5-4 and specifications are given in Table 11.5-1. The system monitors two separate locations in the reactor building ventilation system. One set of detectors monitors the air flow in the reactor building vent shaft while the other set monitors air flow from the fuel storage pool, reactor well pool, and dryer/separator pool.

The reactor building vent shaft monitors consist of two independent channels. Characteristics are included in Table 11.5-1.Each channel includes a Gieger-Mueller type detector and a monitor. Except for detectors, all other components along with remote annunciator provided for alarm and failure/downscale indications are located in the control room. Each channel has two trip circuits, one upscale and one downscale. The upscale trip indicates high radiation while the downscale trip indicates instrument trouble. If either an upscale or downscale trip occurs, the main reactor building heating and ventilation system is shut down (secondary containment isolates) and the standby gas treatment system is started. In addition, the trip signal is sent to the PCI/NSS shutoff system to initiate closure of the primary containment purge and vent valves. Each radiation monitor is powered from a different instrument AC bus.

The refuel pool exhaust radiation monitoring system monitors air exhausted from the refuel pool area. Characteristics are included in Table 11.5-1. The system consists of two independent channels. Each channel includes a local Geiger-Mueller type detector, trip/control unit, and annunciator indication in the control room. Both channels supply a single 2-pen recorder. On a high-high radiation signal or if the monitor is taken out of the operate mode, the trip/control unit will provide a signal to isolate secondary containment, start the standby gas treatment system, and close the primary containment purge and vent valves. The DNSCL/INOP annunciator will also activate with the mode switch out of operate. Each channel also provides two additional alarms; a high radiation alarm off of the recorder output and a downscale alarm. Each monitor is powered from a different reactor protection system bus through its respective DC power supply.

The environmental and power supply design conditions are given in Table 11.5-2.

Ventilation air from the reactor building, turbine building, radwaste building, and recombiner building is exhausted through the three reactor building ventilation exhaust stacks, and the air from each of these buildings is monitored for radioactivity. These monitors are used to check for noble gases, particulate, and iodine and are part of the extended range airborne effluent radiation monitoring system described in Section 11.5.9.

A reactor building auxiliary sampling station has been installed [redacted] consists of an external connection for grab samples from each of the three reactor building ventilation exhaust stacks. This station permits effluent sampling when the local panel in the reactor building is inaccessible following a design-basis LOCA.

In addition to these systems in the reactor building, there are [redacted] Air from the space in the turbine building above the operating floor is exhausted through these openings by three 50% capacity vaneaxial exhaust fans as described in Section 9.4. The extended range airborne effluent radiation monitoring system described in Section 11.5.9 extracts a sample from the main exhaust ductwork to monitor the turbine building effluent radiation releases during normal operation and after a postulated accident.

Ventilation from the remainder of the turbine building is as described in Section 9.4.

11.5.5.4 Safety Evaluation

2017-013 The response of the reactor building ventilation radiation monitoring system to the refueling accident is presented in Section 15.2.5.

11.5.5.5 Inspection and Testing

The monitors are installed so as to be readily accessible for inspection, calibration, and testing. The reactor building ventilation radiation monitoring system and the response of the reactor building heating and ventilation system and the standby gas treatment system are routinely tested. The test and calibration schedules are set forth in the Technical Specifications.

11.5.6 SITE ENVIRONS RADIATION MONITORS

11.5.6.1 Power Generation Objectives

The power generation objectives of the site environs radiation monitors are the following:

1. To provide integrated measurements of direct radiation exposure at the boundary of the unrestricted area to confirm that the operation of the plant is in accordance with the requirements of 10 CFR 20.

2. To provide a record of onsite air particulate radiation levels.

3. To provide measurements of direct radioactivity levels at the boundary of the unrestricted area in the unlikely event of an abnormal release of radioactivity from the plant.

11.5.6.2 System Description

The site environmental radiation monitoring system consists of over 40 TLDs located within each of the 16 standard meteorological sectors (i.e., 22.5 degrees apart) both at and beyond the boundary of the unrestricted area.

2013-002 An airborne particulate radiation sampling network has been established for the plant and is discussed in Section 11.5.7.5. Two of these air samplers are designated as onsite units. One is located within the owner controlled area to the south of the plant and the other is located on the north edge of the owner controlled area. Filters from these units are changed weekly and the exposed filters are forwarded to a vendor laboratory for analysis.

11.5.7 PREOPERATIONAL AND POSTOPERATIONAL ENVIRONMENTAL RADIOACTIVITY MONITORING PROGRAMS

11.5.7.1 General

Environmental radioactivity monitoring around the DAEC was initiated in the second quarter of 1971 and will continue through the plant operating lifetime.

This monitoring program is designed to detect, measure, and document radioactivity concentrations and their fluctuations in various environmental media at the DAEC site and surrounding areas. The program will provide data on environmental levels of radioactivity for several purposes. The preoperational program was designed to measure and document existing radioactivity levels in the environment and demonstrate the adequacy of the monitoring methods including the training of personnel and the sensitivity and suitability of the sampling program and equipment.

The postoperational program is designed to document that environmental radioactivity levels do not reflect significant change from those levels measured during the preoperational program, confirming that plant control of effluent releases has been maintained as low as reasonably achievable in accordance with 10 CFR 50. The program description below is representative of the purpose and scope of the program. However, some details will vary from time to time. Complete descriptions are contained in the Offsite Dose Assessment Manual and Annual Environmental Operating Reports.

Any increase in the amount of radioactive materials in the environment would be promptly identified by the postoperational program.

During the initial or confirmatory stage of the postoperational radiation monitoring program, a positive correlation was made between effluent release concentrations and the corresponding levels in the various environmental media. After observing this relationship for a

sufficient period of time to ensure confirmation that there was no significant increase in environmental radioactivity levels, the number of sampling stations associated with each type of environmental medium was reevaluated and changes to the program made as appropriate.

Both the preoperational and postoperational programs provide a means for gathering data for state, federal, and local agencies, as well as for the general public. The DAEC will continue to cooperate with these agencies in discussing the program specified herein, and the results of the monitoring program is reported to these agencies at periodic intervals.

11.5.7.2 Technical Discussion

An evaluation of critical exposure pathways combined with an estimate of expected radionuclides released to the environment was used to develop the monitoring program with respect to sample locations, sample frequency, and necessary analyses. Air and water are obvious pathways through which man may be exposed; therefore, the analysis of radioactivity in these media is an essential part of this monitoring program. In addition, in order to ensure that no potential hazard to the health of the general public exists from radioactivity released from the DAEC, the radiation exposure from all potential vectors has been considered. These include food and any reconcentrating media that exist in man's food chain or contribute, in any way, to man's total intake of radioactive material. The DAEC Radiological Environmental Monitoring Program meets these requirements and is described below. The number, types, locations, and the frequency of analyses for various environmental media are included in the Offsite Dose Assessment Manual. The locations of the sampling points with respect to the plant site are included in the Offsite Dose Assessment Manual.

Radioactivity in the environmental samples may be analyzed either at the DAEC or at an offsite laboratory. In order to provide a comparative check on the accuracy and precision of these analyses, the laboratory participates in an Interlaboratory Comparison Program by analyzing radioactive samples distributed for the purpose.

The lower limit of detection for radiological environmental samples is calculated as defined in the Offsite Dose Assessment Manual.

Results of environmental sample analyses confirmed to exceed the appropriate activity concentration stated in the Offsite Dose Assessment Manual, when averaged over any calendar quarter, are reported to the NRC as required by the Offsite Dose Assessment Manual reporting requirements, if attributable to station operations.

11.5.7.3 Aquatic Environment

Surface Water

The factors that establish the basis for determining the number, type, frequency, and location of sampling points include the following:

1. Downstream uses of receiving water (i.e., public water supply, industry, recreation, or fishing).

2. Characteristics of the receiving waters.

2013-002 Water samples are collected for monthly activity concentration analysis from the Cedar River upstream and downstream of the DAEC and immediately downstream of the discharge
2015-010 point. Sampling during atypical environmental conditions is avoided.

Water samples are analyzed for tritium and for specific radionuclides by gamma spectrometry.

For the surface water sampling point below the plant discharge, the monthly sample will be coordinated, where possible, with a radioactive liquid release.

Ground Water

2013-002 Site well water and water from nearby offsite wells used for domestic water supplies are sampled quarterly and are analyzed for specific radionuclides by gamma spectrometry. A tritium analysis is performed using liquid scintillation counting techniques. For those samples where there is a positive identification of reactor by-product radionuclides or tritium, analyses for Sr-89 and Sr-90 are also performed.

The DAEC does not use proportional compositing (sample volume proportional to river flow) because the integrated concentration of the water treatment plant discharge is the dosimetrically correct value and is not subject to correction for river and/or municipal water system flow rate.

Drinking Water

Since the Cedar Rapids potable water supply comes from shallow wells that are recharged from the Cedar River, samples of raw and treated water at the municipal water treatment plant are taken. These samples are analyzed in the same manner as ground water samples.

Bottom Sediments

Because of a combination of a number of mechanisms, including chemical precipitation, settling, and ion exchange with colloidal particles in the water, certain radionuclides may accumulate in bottom sediments.

These samples are analyzed by a gamma isotopic analysis for individual nuclides.

11.5.7.4 Terrestrial Environment

Vegetation

2013-002 Samples of forage, grain and green leafy vegetation are collected at the time of harvest. The samples are analyzed for specific radionuclides by gamma spectrometry including I-131.

Milk

2013-002 Milk is sampled from dairy farms near the DAEC and from farms distant from the station influence. Iodine present in the milk is extracted for analysis.

Milk is collected monthly and analyzed for specific radionuclides by gamma spectrometry and radiochemical separations. It is recognized that for efficient monitoring of I-131 in milk, the sample analysis frequency would need to be increased. If an indication of I-131 in a sample occurs, the monitoring program will be revised to permit more frequent I-131 monitoring. Also, this frequency will be increased during episodes of weapon test fallout should any occur.

Land Use Census

A land use census is conducted annually for the purpose of identifying changes in the use of the offsite area surrounding the DAEC, which may substantially affect the radiological dose assessment or which may indicate needed adjustments to the program.

The census includes land within 3 miles of the DAEC and will be conducted by door-to-door survey, aerial survey, by consulting local agricultural authorities, or a combination thereof.

11.5.7.5 Atmospheric Environment

Air Samples

Continuous air sampling for particulates is done with filter paper and for halides with potassium iodide or "teda" impregnated charcoal or with silver zeolite cartridges with the object of retaining all particulates and halides from the air sampled. Appropriate analyses of particulate filters or halide sampling media are performed.

Air particulate samples are collected at five or more locations with low-volume (approximately 1 cfm) air samplers equipped with high-efficiency (99.9% efficient for particles 0.3 μm in diameter or greater) filters. The basic sample interval is 1 week. The air samplers are fitted with charcoal traps placed behind the absolute filter. Locations for these units have been selected on the basis of proximity to the reactor and also from the standpoint of providing distributed sampling around the wind rose.

Air samples are held for at least 72 hr before analysis to permit the decay of most of the natural radon daughter radioactivity that may have been collected. The samples are then analyzed for gross beta. Air samples with a gross beta activity in excess of 10 times the yearly mean of control samples are analyzed by gamma spectrometry. The activated charcoal filters are also analyzed by gamma spectrometry for I-131. 2013-002

Air Radiation Dosimeters

The normal airborne emissions from the plant consist predominantly of noble gases that are not sampled by the air samplers previously discussed.

Integral beta and gamma radiation measurements are made with TLDs placed at multiple locations. The locations are specified in the Offsite Dose Assessment Manual. 2013-002

11.5.8 POSTACCIDENT RADIOLOGICAL MONITORING AND SAMPLING SYSTEMS

The postaccident liquid and gas sampling systems are discussed in Section 12.3.4.

Containment postaccident radiation monitors are discussed in Section 12.3.3.3.4.

11.5.9 EXTENDED RANGE AIRBORNE EFFLUENT RADIATION MONITORING SYSTEM

11.5.9.1 System Description

The extended range airborne radiation monitor system was installed to satisfy the requirements of NUREG-0737, Item II.F.1, Attachments 1 and 2, and Regulatory Guide 1.21. The system is a Kaman Sciences Corporation digital radiation monitoring system (DRMS) which acquires and displays radiation data from the turbine building and reactor building vents and the offgas stack. This system is in addition to the offgas stack radiation monitoring system

described in Section 11.5.3. It also includes a monitor for the low-level radwaste processing and storage facility (LLRPSF) ventilation exhaust. The DRMS consists of two major systems, a data acquisition system and a display and control system. The data acquisition system consists of a serial chain of 11 radiation monitor units. Each monitor unit consists of a multichannel radiation monitor and an Intel microprocessor acting as an interface control between the monitors and the display and control system. The chain of 11 monitor units or data loop terminates at the plant process computer. Asynchronous communications multiplexers interface the ends of the data loop to the plant process computer. The display and control system interrogates all monitors from one end of the data loop and then alternately from the other end to ensure the integrity of the data links between monitor units. Alternating the direction of interrogation also ensures that all monitors are still in communication with the display and control system even after a single failure of the communications data loop. The interrogation message is sent to each monitor at least once every 2 sec to obtain the current radiation level and at time intervals to acquire 1-min, 10-min, 1-hr, and 1-day averages of the radiation level. The operator has the ability to review the status of all monitors through color-coded displays generated on the CRT. A set of keyboard commands allows the operator to call up displays, initiate certain monitor activities, and enter monitor parameters and other data base changes. Critical conditions reported by the monitors cause an audible alarm on control room panel [redacted]. Alarm setpoints are derived in the manner described in the DAEC Offsite Dose Assessment Manual (Reference 1). All events will be recorded in a data file and available for operator initiated display on the operator's console or one of the plant process computer system printers. Display and control units are located in the control room and the radiochemistry laboratory for interactive data retrieval.

There are 11 monitor units, a normal range monitor for the LLRPSF ventilation exhaust stack, and a normal and an accident range monitor in each of the following: turbine building vent, reactor building vent numbers 1, 2, and 3, and the offgas stack.

The Extended Range Gaseous Effluent Monitoring system consists of two units (the LLRPSF has a normal range monitor only): the normal range monitor (KMG-HRN) and the high range monitor (KMG-HRH). Each unit has its own microcomputer, and the two are interconnected to provide appropriate signals to each other for proper flow during an accident condition. The high range monitor is actuated by a handover signal from the normal range monitor when the activity increases to a pre-set level.

Under normal operating conditions, the normal range monitor is operating and the high range monitor is in a standby mode. Flow through the normal range monitor involves passing the sample through unshielded particulate and iodine collectors and through a noble gas sampler chamber. This chamber houses a beta scintillator detector for normal range activity monitoring. During an accident condition, however, the normal range unit bypasses the sample flow around the collectors and sampler chamber since they are not equipped to handle the higher levels of activity. The high range unit takes the "accident" flow from the normal range unit and passes it through shielded collectors and a noble gas sampler chamber containing mid- and high-range GM detectors.

The high range monitor uses three sequentially controlled particulate-iodine collectors to collect radioactive particulates and iodines for laboratory analysis. Each collector has an associated GM tube detector located above it monitoring the activity of the collector. Only one collector is used at any instant, while the remaining two are held in reserve. When the activity level of a channel reaches a predetermined setpoint, the high range unit will switch to the next particulate-iodine collector.

The noble gas samplers on the normal and high range units send the information they gather to the plant process computer. The particulate and iodine information is obtained from processing the particulate and iodine collector cartridges from the normal range units and the high range units (if they were collecting).

The DRMS is supplied by the 120-V ac distribution system. If primary power is lost, a battery supplies 10 to 12 V to the micro-processors so that memory and calibration data are not lost while primary power is off.

11.5.9.2 Monitor Characteristics

The radiation monitors have up to two types of channels for detecting and measuring radiation as follows:

1. Gas detector - measures the gross beta activity level of radioisotopes in gaseous form present in air streams.

2. Combined particulate-iodine sampler - collects particulate or iodine radioactive substance from an air stream for laboratory analysis.

The characteristics of the particulate, iodine, and noble gas detectors are given in Table 11.5-3 for each of the five effluent vents and the offgas stack.

In addition to the monitoring units, the Kaman system has radiation recorders for the turbine building vent and reactor building vents 1, 2, and 3. There is also a flow recorder associated with the turbine building monitoring units.

REFERENCES FOR SECTION 11.5

1. Duane Arnold Energy Center, Offsite Dose Assessment Manual for Gaseous and Liquid Effluents, March 1983. (Submitted by letter from R. W. McGaughy, Iowa Electric, to Harold Denton, NRC, December 2, 1985, NG-85-4166.)

Table 11.5-1

CHARACTERISTICS OF PROCESS RADIATION MONITORING SYSTEMS

Monitoring System	Instrument Range[a]	Instrument Scale	Upscale Trips per Channel	Downscale Trips per Channel	Inoperative Trips per Channel
Main steam line	1.0 to 10^6 mR/hr	6-decade log	1 1[b]	1[b]	1
Air ejector offgas (before treatment)	1.0 to 10^6 mR/hr	6-decade log	2[b]	1[b]	1[b]
Air ejector offgas (after treatment)	10^{-1} to 10^6 counts per sec[c]	7-decade log	2 1[b]	1	1
Carbon bed vault area radiation	1.0 to 10^6 mR/hr	6-decade log	1[b]	1[b]	--
Offgas vent pipe	10^{-1} to 10^6 counts per sec[c]	7-decade log	1 1[b]	1[b]	1[b]
Liquid processes[d]	10^{-1} to 10^6 counts per sec[c]	7-decade log	1[b,e]	1[b]	1[b]
Reactor building vent shaft	0.01 to 10^4 mR/hr	6-decade log	1	1	--
Refuel pool ventilation exhaust	0.01 to 100 mR/hr	4-decade log	1[b]	1[b]	1[f]

[a] Range of measurement is dependent on items such as the source of geometry, background radiation, shielding, energy levels, and method of sampling.
[b] Alarms only.
[c] Readout is dependent on the pulse height discriminator setting.
[d] Five process steam monitored.
[e] Upscale trip for Radwaste Effluent monitor only.
[f] Mode switch out of operate.

Table 11.5-2

ENVIRONMENTAL AND POWER SUPPLY DESIGN CONDITIONS FOR PROCESS RADIATION MONITORING SYSTEM

	Sensor Location		Control Room	
Parameter	Design Center	Range	Design Center	Range
Temperature	25°C	0°C to +60°C	25°C	5°C to +50°C
Relative humidity	50%	20% to 98%	50%	20% to 90%
Power, ac	115V	±10%	115V	±10%
	60 Hz	±5%	60 Hz	±5%
Power, dc	+24V	+22V to +29V	+24V	+22V to +29V
	-24V	-22V to -29V	-24V	-22V to -29V

CHARACTERISTICS OF EXTENDED RANGE AIRBORNE EFFLUENT MONITOR SYSTEM

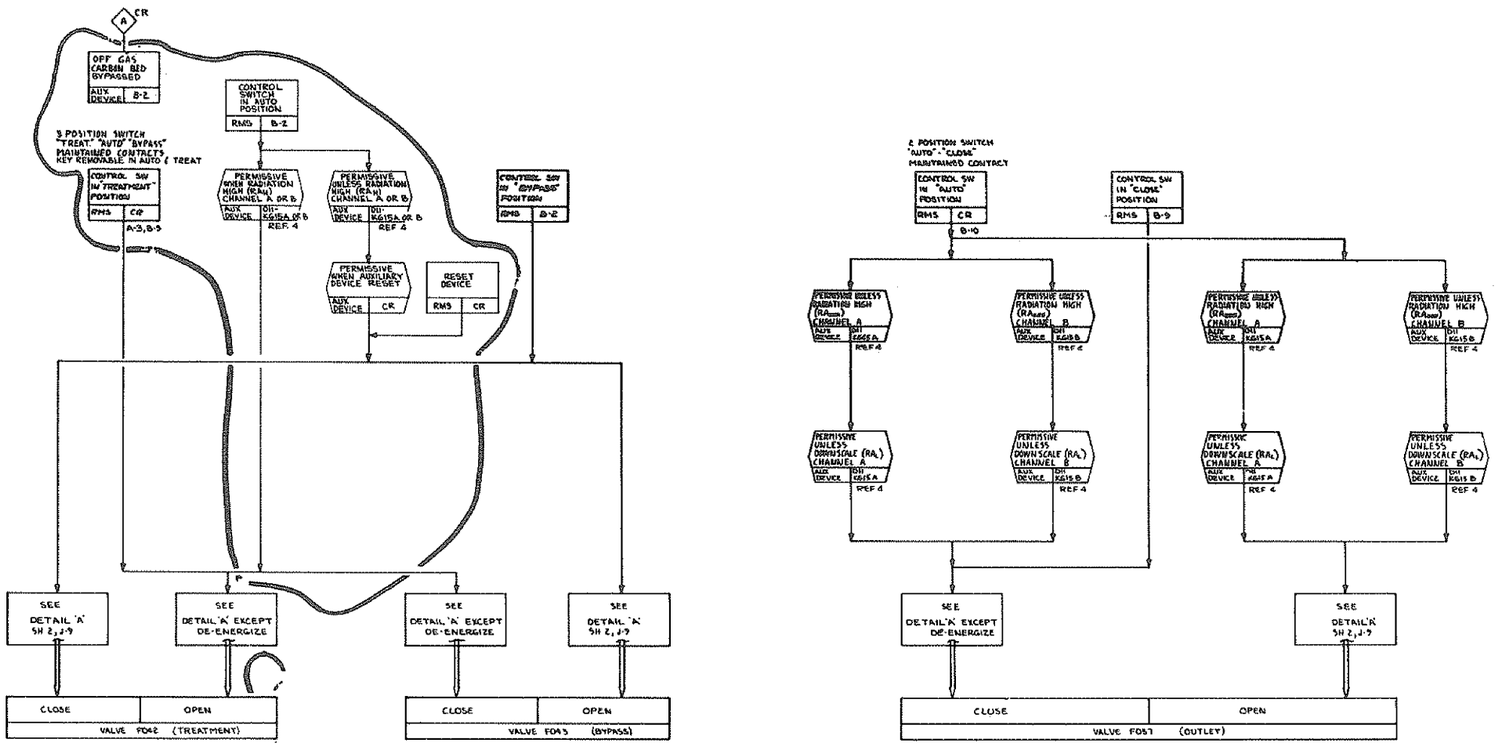
Characteristics	Particulate	Iodine	Noble Gas
REACTOR BUILDING VENTS 1, 2, AND 3			
Measurement type	Collection only	Collection only	Gross beta
Detector type	Not applicable	Not applicable	Beta scintillator Geiger-Mueller counter (2 required)
Dynamic range, $\mu Ci/cm^3$	Not applicable	Not applicable	10^{-7} to 10^5
Minimum detectable concentration (MDC), $\mu Ci/cm^3$	Not applicable	Not applicable	2×10^{-7}
Calibration Isotope	Not applicable	Not applicable	Mixed noble gases[a] Xe^{133}
Maximum response time at MDC	Not applicable	Not applicable	1 min.
Minimum filter efficiency	99% > 0.3μ	95%	Not applicable
Filter type	HV-LB 5211	SAI	Not applicable
Analog type	--	--	2 required
Remote Indication	--	--	1 required
TURBINE BUILDING VENT			
Measurement type	Collection only	Collection only	Gross beta
Detector type	Not applicable	Not applicable	Beta scintillator Geiger-Mueller counter (2 required)
Dynamic range, $\mu Ci/cm^3$	Not applicable	Not applicable	10^{-7} to 10^5
MDC, $\mu Ci/cm^3$	Not applicable	Not applicable	2×10^{-7}
Calibration Isotope	Not applicable	Not applicable	Mixed noble gasses[a] Xe^{133}
Maximum response time at MDC	Not applicable	Not applicable	1 min.
Minimum filter efficiency	99% > 0.3μ	95%	Not applicable
Filter type	HV-LB 5211	SAI	Not applicable
Analog outputs	--	--	2 required
Remote Indication	--	--	1 required

[a] Mixed noble gases for beta scintillator and Xe^{133} for Geiger-Mueller counter.

CHARACTERISTICS OF EXTENDED RANGE AIRBORNE EFFLUENT MONITOR SYSTEM

Characteristics	Particulate	Iodine	Noble Gas
OFFGAS STACK			
Measurement type	Collection only	Collection only	Gross beta
Detector type	Not applicable	Not applicable	Beta scintillator Geiger-Mueller counter (2 required)
Dynamic range, $\mu Ci/cm^3$	Not applicable	Not applicable	10^{-7} to 10^{5}
MDC, $\mu Ci/cm^3$	Not applicable	Not applicable	2×10^{-7}
Calibration Isotope	Not applicable	Not applicable	Mixed noble gases[a] Xe^{133}
Maximum response time at MDC	Not applicable	Not applicable	1 min.
Minimum filter efficiency	99% > 0.3 μ	95%	Not applicable
Filter type	HV-LB 5211	SAI	Not applicable
Analog outputs	--	--	2 required
Remote Indication	--	--	1 required
LLRPSF EXHAUST STACK			
Measurement Type	Collection only	Collection only	Gross beta
Detector Type	Not applicable	Not applicable	Beta Scintillator Gieger-Mueller counter
Dynamic Range, $\mu Ci/cm^3$	Not applicable	Not applicable	10^{-7} to 10^{-1}
MDC, $\mu Ci/cm^3$	Not applicable	Not applicable	2×10^{-7}
Calibration Isotope	Not applicable	Not applicable	Xe^{133}
Maximum response time at MDC	Not applicable	Not applicable	1 min.
Minimum filter efficiency	99% > 0.3 μ	95%	NA
Filter type	HV-LB 5211	SAI	NA
Analog outputs	--	--	1 required
Remote indication	--	--	1 required

[a] Mixed noble gases for beta scintillator and Xe^{133} for Geiger-Mueller counter.



APED-N62-7(3)-2 Rev 1

DUANE ARNOLD ENERGY CENTER
IOWA ELECTRIC LIGHT & POWER COMPANY
UPDATED FINAL SAFETY ANALYSIS REPORT

Air Ejector Offgas Radiation Monitoring System - FCD

Figure 11.5-2

[redacted]

[redacted]

[redacted]

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APPENDIX 11A

GASEOUS RELEASE RATE LIMIT CALCULATIONS

11A.1 ANALYTICAL MODEL

The analytical model is primarily concerned with calculating the annual gamma dose at ground level resulting from a continuous release of radioactive materials. As a direct consequence, a method is also obtained for calculating the annual average concentration at ground level.

In essence, the gamma dose model considers the integrated dose rate from a continuously distributed gaseous source (the plume). The source distribution is treated by a standard dispersion model that relates the dispersion of airborne particles to downwind distances and to the meteorological conditions that exist during the release intervals. The annual gamma dose is obtained by weighting the gamma dose rate associated with a given meteorological condition by the frequency of occurrence of that condition.

11A.1.1 METEOROLOGICAL FACTORS

The air concentration per unit amount released at a point (x, y, z) in the cloud at any instant is given by Equation 11A.1-1, which is Sutton's equation corrected by Cramer[1] for depletion by ground deposition and radioactive decay.

$$(X) = \frac{Q_o}{2\pi\sigma_y\sigma_z\bar{u}_h} \exp - \left[\frac{y^2}{2\sigma_y^2} + \frac{z^2}{2\sigma_z^2} \right] \left[\frac{Q}{Q_o} \right] \exp(-\lambda t) \qquad \text{(11A.1-1)}$$

where

(X) = average air concentration, Ci/m^3 or $\mu Ci/cm^3$

Q_o = release rate, Ci/sec

$\bar{u}_h$ = average wind speed at height of release, m/sec

$\sigma_z\sigma_y$ = standard deviation of cloud width in vertical and horizontal direction, respectively, m

t = time after release, sec

λ = radioactive decay constant, sec^{-1}

The factor $\frac{Q}{Q_o}$ is the correction for cloud depletion due to deposition and is equal to the fraction of the initial amount released, which is present at a downwind distance x. According to Watson and Gamertsfelder[2], $\frac{Q}{Q_o}$ is given by

$$\frac{Q}{Q_o} = \exp\left[-\frac{V_d\left(\frac{\bar{u}_o}{\bar{u}_h}\right)}{\bar{u}_o\sqrt{\frac{\pi}{2}}} \int_o^{t_{\bar{u}}} \sigma_z^h \exp\frac{-z^2}{2\sigma_z^2} dt \right] \qquad (11A.1\text{-}3)$$

Values of the deposition velocity, V_d, are obtained from Table 11A.1-1.

It is a reasonable approximation to assume that throughout the year all the plumes that travel anywhere within a given sector direction do not have a skewed frequency distribution within the sector. Then, the average cloud concentration in the sector is found by integrating Equation 11A.1-1 in the crosswind direction and dividing by the sector width.

$$(X)_{ave} = \frac{\int_{-\infty}^{\infty} X dy}{\theta X} \qquad (11A.1\text{-}3)$$

where

θX = sector width

Equation 11A.1-3 cannot be integrated because the interrelationships among the variables σ_y, σ_z and $\bar{u}_h$ with respect to their average values is not generally known. However, for any specific combination of wind speed and stability at a given downwind distance, all these variables are known and can be treated as constants, and the integration can then be performed. Thus, the average concentration in the sector for all occurrences of any specific condition is given by

$$(X)_{ave}^{ij} = \frac{Q_o \frac{Q}{Q_o}}{\sqrt{2\pi}\theta x \sigma_z \bar{u}_h} \exp\left[\frac{-z^2}{2\sigma_z^2}\right] \exp[-\lambda\tau] \qquad (11A.1\text{-}4)$$

where

θ = sector angle ($\pi/8$ or 22.5 degrees is used in this appendix)
x = downwind distance and is equal to $\bar{u}_h t$
σ_z = a function of stability, wind speed, and downwind distance, x

Thus, the average cloud is seen to have a uniform concentration distribution vertically, which is of the Gaussian form.

The standard deviation in the vertical direction is described by Watson and Gamertsfelder[2] as

$$\sigma_z^2 = a\left[1 - \exp\left(-k^2 t^2\right)\right] + bt \qquad \text{stable condition} \qquad (11A.1\text{-}5)$$

$$\sigma_z^2 = \frac{C_z^2 x^{(2-n)}}{2} \qquad \text{neutral, unstable conditions} \qquad (11A.1\text{-}6)$$

The expression for σ_z, in Equation 11A.1-6 is easily recognized as the standard Sutton equation. The expression for σ_z in Equation 11A.1-5 was derived from Hanford field measurements of the vertical concentration taken at several downwind locations under stable conditions. The constants for Equation 11A.1-5 and 11A.1-6 were evaluated from the Hanford measurements for a source height of 200 ft and correlated with vertical temperature gradients at the point of emission.

Since the concentration measurements were averaged over 30- to 60-min intervals, the constants used to evaluate σ_z are considered to be more appropriate for long-term releases rather than the shorter term or "puff" releases. Figures 11A.1-1 through 11A.1-4 show vertical cloud width, σ_z as a function of distance for each stability category.

The following stability classification is used along with vertical temperature lapse rates for each:

Very stable	ΔT	$\geq$ 1.5°C/100 m
Moderately stable	-0.5 $\leq \Delta T$	< 1.5
Neutral	-1.5 $\leq \Delta T$	<-0.5
Unstable	-1.5 < ΔT	

Table 11A.1-1 shows the deposition velocity coefficients for each stability category. Table 11A.1-2 shows the appropriate values of a, b, k^2, C_z, and n used with each stability condition and wind speed. Such values are used to calculate the vertical dimensions of the plume, σ_z, and as stated earlier, were constants derived from the Hanford field measurements.

The conventional reflection factor of 2 usually applied for releases is not included. For the passing cloud that is primarily a gamma dose, the entire plume volume is integrated as an infinite number of point sources to plus and minus infinity in the z direction. This ignores the interception by the ground so that the entire cloud volume is included.

Inhalation doses are a function of concentration at the ground and subject to reflection effects if they exist. Since the materials of interest in inhalation effects deposit on the ground, it is doubtful that "perfect" reflection will occur, but rather that the cloud will expand distorting the Gaussian mass distribution of the cloud resulting in, at most, a small increase in concentration. In addition, no account was taken of the better diffusion at the ground (effective on the portion of the cloud near the ground) compared to the stack exit elevation used. Meteorology and Atomic Energy (AECU 3066) shows that compared to an elevation of 200 m, ground-level diffusion coefficients are larger by about a factor of 2 plus proportionally increasing dispersion. In any event, an increase by a factor of slightly more than 1.0 but less than 2 would account for this reflection effect.

11A.1.2 RADIOLOGICAL FACTORS

The ground-level gamma dose rate from an elevated plume of radioactive materials having a spatial distribution as given in Equation 11A.1-3 may be considered as the sum of the dose rates from all, the points in the plume. The source strength of each point is (X)dV and the total source is

$$S = \int_{-\infty}^{\infty} (X)dV \qquad \text{(11A.1-7)}$$

where

dV = dxdydz

The flux from a point source, considering buildup in the air, is given by Glasstone[3] as

$$\varnothing = \frac{SBe^{-\mu R}}{4\pi R^2} \text{ photons per m}^2\text{/sec} \qquad \text{(11A.1-8)}$$

where

S = source strength
B = buildup factor = $1 + K\mu R$ (Figure 11A.1-5)
K = $\dfrac{\mu - \mu_a}{\mu_a}$
μ = total (linear) attenuation coefficient, m^{-1}
μ_a = energy absorption coefficient, m^{-1}
R = distance from source equal to $\left(x^2 + y^2 + z^2\right)^{1/2}$
x_1, y_1, z_1 = coordinates of dose point at ground level relative to the incremental volume, dV

The gamma dose rate from a flux of a given energy (E) from Glasstone is

$$(DR)_\gamma = 5 \times 10^{-3} \varnothing E \mu_a \qquad \text{(units of mR/hr)} \qquad \text{(11A.1-9)}$$

so that the total dose rate from the plume at any point is found by combining Equations 11A.1-7, 11A.1-8 and 11A.1-9. Hence, the gamma dose rate

$$(DR)_\gamma = \frac{5 \times 10^{-3}}{4\pi} E \mu_a \int_{-\infty}^{\infty} \frac{(X)_{ave}^{Be^{-\mu R}}}{R^2} dV \qquad \text{(mR/hr)} \qquad \text{(11A.1-10)}$$

As Equation 11A.1-10 is written, it assumes a monoenergetic source. For a mixture of isotopes, it is proper to perform the calculation for each gamma energy present considering its abundance. Since μ and μ_a are energy dependent and appear in an exponential term, care must be exercised if an average energy is to be used. A listing of each of the noble gas isotopes and significant particulate daughter products is shown in Tables 11A.1-3 and 11A.1-4. Also shown are the gamma energies, total attenuation, and linear absorption coefficients.

In general, Equation 11A.1-10 cannot be solved analytically and must be solved numerically. While integration to infinity is indicated, in practice finite bounds are placed on the cloud. Integrating Equation 11A.1-10 to $\pm 3\sigma_z$ includes more than 99.97% of the entire matter per unit length; hence, the dose contributions from points in the cloud when vertical displacement is more than three standard deviations from the plume center line can be ignored. Likewise, because of the geometric and material attenuation shown in Equation 11A.1-8, one can usually ignore the dose contribution from source points that are more than 400 to 500 m downwind or upwind of the receptor point without significant error. The integration proceeds by reducing the distributed source (the plume) into a large array of point sources. This is done by dividing the cloud into cubical

volume elements. The assumption is made that the concentration at the center of the cube is average for the volume sheets.

The total source strength is preserved by multiplying the concentration at the center ($\mu Ci/cm^3$) by the volume of the element (cm^3). The dose rate from each point source is calculated by Equation 11A.1-10 and summed over all points. Equation 11A.1-10 then becomes a finite series.

Mathematically the numerical integration can be expressed as

$$DR^{ij}(P) = \sum_{P'} \sum_{I} G^{ij}(I, P'; P) \qquad (11A.1\text{-}11)$$

where $G^{ij}(I, P'; P)$ is the dose rate contribution from isotope I to point P from a source at P' as described by Equation 11A.1-10.

Equations11A.1-10 and 11A.1-11 give the average dose rate for the (ij) $\underline{\text{th}}$ meteorological condition for a point P, which may be immersed in the cloud or at some point outside the cloud. This is a significant item since the gamma dose at ground level from a stack plume is not merely existent when the receptor is immersed in the plume. Dose is also received when the plume is traveling in some other sector than the one in which the receptor point is located. The effect is particularly important at points close to the stack where the receptor remains at a nearly constant distance from the plume regardless of angular separation.

11A.1.3 ENGINEERING FACTORS

From Equations 11A.1-4 and 11A.1-10, it is evident that the dose rate is significantly affected by the height of the plume above ground level. This height is made up of the physical stack height plus plume rise due to exit velocity and buoyancy. Many formulas are available to calculate plume rise. The method used here is the Holland formula as modified by Moses and others.[4]

$$\Delta H = \frac{K\left(1.5 V_s d + 4 \times 10^{-5} Q_h\right)}{\overline{u}_h} \qquad (11A.1\text{-}12)$$

where

V_s = exit velocity, m/sec
d = stack diameter, m
Q_h = heat emission of effluent, cal/sec
$\overline{u}_h$ = wind speed at stack exit, m/sec

K = correction factor for stack diameter [4] (Stümke regression coefficient)

In proposing the correction factor K in the plume rise formula, Moses used data from an experimental stack at Argonne with a diameter of about 0.46 m and from a stack at Duisburg, Germany, which has a diameter of 3.5 m. His conclusions are that a value of 3 for the correction factor is proper for large stacks with appreciable buoyancy whereas a factor of 2 is recommended for small stacks with modest buoyancy. In applying the Moses correction to individual situations, a linear interpretation is made from the actual stack diameter compared to those from which data were obtained.

The AEC documents[5] points out similar results in Section 5.2 discussed by Gary A. Briggs. He states that "both the Stümke formula and Holland formula times a factor of 3 seem to give good agreement (calculated versus observed plume rise) for the moderate-sized sources (heat emissions of about 10^6 cal/sec), but grossly underestimate rise in the case of the large Colbert plant (heat emissions of about 7×10^6 cal/sec)."

11A.1.4 AVERAGING TECHNIQUES

One is usually interested in the cumulative dose over some appropriate time interval, such as a year. To compute the annual gamma dose, the gamma dose rate for a given meteorological condition must be weighted by the frequency distribution F^{ijk}. F^{ijk} describes the frequency of the i^{th} stability condition with j^{th} wind speed occurring in direction sector k. The average annual gamma dose rate in section k is given

$$DR_{\gamma}^{ij}(P) = C\sum_{k'}\sum_{k} DR_{\gamma}^{ij;k'}(P)F^{ijk} \qquad \text{(11A.1-13)}$$

where

$DR_{\gamma}^{ijk;k'}(P)$ = the gamma dose rate to a point P in sector k from a plume traveling in sector k'

C = 8760 hr/yr

Equation 11A.1-13 indicates a finite summation over the variables of stability, wind speed, and direction. For stability and direction it has already been indicated how these variables can be grouped into 4 stability classes and 16 directions. The spectrum of wind speeds can also be grouped into representative ranges. One such grouping that has proven useful, especially when using U.S. Weather Bureau summaries, is as follows:

Wind Speed Range (mph)	Average Wind Speed (m/sec)
0-3	1
4-7	2
8-12	5
13-18	7
19-24	10
>24	>13

Also included above is the average wind speed that is representative of each speed range.

11A.1.5 AVERAGE AIR CONCENTRATION

For doses other than the whole-body gamma dose, the annual average concentration at ground level is of interest. This is easily obtained from the preceding material presented by substituting plume height for z. The air concentration during any meteorological condition has been described by Equation 11A.1-4. However, for materials other than noble gases, the depletion factor $\frac{Q}{Q_o}$ is not 1 and must be accounted for. For the calculations made in this appendix, the deposition rates shown in Table 11A.1-1 were used.

Using the joint frequency distribution F^{ijk} defined previously, computations of the annual average concentration at the ground can be made from:

$$(X)_{gr}^{k} = \sum_{ij} (X)_{gr}^{ij} F^{ijk} \qquad (11A.1\text{-}14)$$

11A.1.6 SHIELDING AND OCCUPANCY FACTORS

Radiation doses are usually calculated for certain distances from the point of release and often are calculated for locations where no actual dose would be received by a human receptor. In fact, it is not too uncommon to see radiation doses from the passing cloud calculated as if the dose receptors were outdoors day and night. This is certainly possible, but it does not lead to particularly accurate dose estimates for most people. For this reason, occupancy by individuals should be considered in arriving at reasonable dose estimates. Credit for this is allowed by 10 CFR 20.

In addition, it seems rather incongruous to assume that a person would stay in one place all of the time without being inside some type of shelter. For this reason, the shielding effect for various types of structures was evaluated. The shielding value[6] of such typical structures is shown in Table 11A.1-5.

It is easily seen that the error introduced by omitting this effect can be a factor of 2 or more. Where larger urban complexes are concerned, such an error may be far greater.

11A.2 VERIFICATION OF ANALYTICAL MODEL

The methods described previously that are used for calculating offgas limits use standard, available numerical techniques. Certain portions of the calculations, however, involve assumptions and/or extrapolation of existing data. Consequently, while the calculations are conservative, experimental verification of the results has been undertaken. Two pieces of information have been used in this verification effort. One quantity of data comes from a monitoring program at the Brookhaven National Laboratory; the other was accumulated at the [redacted] site monitoring the effluent from Unit 1. In the case of Brookhaven, the research reactor emitted argon-41 (A-41) in small but easily measurable quantities and in a rather uniform pattern. Meteorological and offgas data taken from this location were used to predict environmental doses[7] and compared to measured data.

11A.2.1 BROOKHAVEN NATIONAL LABORATORY

11A.2.1.1 Meteorology Data

Micrometeorological data for 1963 were obtained from Brookhaven National Laboratory. The data were in the form of computer input cards containing hourly observations of average wind speed and direction at levels of 37, 150, and 355 ft and the air temperatures at levels of 37, 75, 150, 300, and 410 ft. The measurements at the 355-ft level were summarized in terms of frequency of occurrence according to wind speed and direction and atmospheric stability. The stability was determined according to the method described in Section 11A.1 by using the temperature gradient measured between the 410- and 37-ft levels.

The summaries are presented in Tables 11A.2-1 through 11A.2-6. The frequency of occurrence was based on 6464 hr of good observations. Of the missing 2296 hr of 1963, August and September account for 1464 missing hr, the rest being scattered throughout the year. A total of 12 hr was observed to have a wind speed less than 0.5 mph. These “calm” conditions were included in the wind speed category (0 to 3 mph).

11A.2.1.2 Radiological Data

As is discussed by Hull[8], the radiation dose was measured at several stations around the Brookhaven Graphite Research Reactor (BGRR) in 1963 using 6-liter, atmospheric ion chambers. The dose rate from the release of A-41 was determined from the total dose measurement by subtracting from it the contribution from natural background and operation of the forest ecology station. The resultant dose rate is shown in Table 11A.2-7.

It was necessary to adjust the measured values of annual gamma dose to account for the absence of meteorological data during August and September. The average dose rate (millirads per week) was averaged over the 10 months for which meteorological data were available and multiplied by 52 to get annual dose (millirads per year). The

exception to this is station E-2, which was moved in December. For this station, a 9-month period was used to determine the annual dose. These normalized values are shown in comparison with calculated values in Table 11A.2-8.

11A.2.1.3 Gamma Dose Calculations

The methods described in Section 11A.1 were used to analyze the effects of the BGRR stack effluent in the Brookhaven environs. The following is a discussin of the calculaions leading to the gamma dose rate matrix, $DR_{\gamma}^{ijk,k'}$.

11A.2.1.3.1 Plume Rise

The BGRR has a 350-ft stack (107 m) with an exit velocity of 6 m/sec and an effluent temperature difference of 50°C above ambient. For use in Equation 11A.1-12, these values correspond to

Q_h = 1.62 x 10^6 cal/sec, heat rate
d = 5.18, stack exit diameter
K = 3.47, correction factor in equation

Using these values in Equation 11A.1-12 the plume rise formula becomes

$$\Delta H = \frac{387}{\bar{u}_h} m$$

Using the six standard wind speed groups described earlier, the effective stack heights were computed and are shown below:

BGRR Plume Rises for Various Wind Speeds

Wind Speed Range (mph)	Average Speed (m/sec)	Plume Rise (m)	Effective Height (m)
0-3	1	387	494
4-7	2	194	301
8-12	5	77	184
13-18	7	55	162
19-24	10	39	146
>24	>13	30	137

11A.2.1.3.2 Isotopic Data

The BGRR during full-power operation released about 12,960 Ci of A-41 per day (0.15 Ci/sec). However, the actual average release rate during 1963 was 0.127 Ci/sec as determined from communications with the BGRR staff, which represents an 85% operation factor.

Pertinent radiological properties of A-41 are as follows:

E = 1.29 MeV, gamma energy
μ = $6.93 \times 10^{-3} M^{-1}$, total attenuation coefficient
μ_a = $3.3 \times 10^{-3} M^{-1}$, energy absorption coefficient
γ = 1.1×10^{-4} sec^{-1}, decay constant

11A.2.1.3.3 Dose Rate Calculations

From the above information, the gamma dose rate as given by Equations 11A.1-10 and 11A.1-11 was evaluated using a digital computer program to evaluate Equation 11A.1-10. The dose rate was evaluated for downwind distances of 10, 100, 400, 1400, 2400, 3200, and 6400 meters, using the six wind speeds shown earlier and all four stability conditions. The results are shown in Figures 11A.2-1-through 11A.2-3. The dose rates for the very stable and moderately stable conditions are essentially identical, because for the distances used here the vertical spread of the plume is small in each case. Hence, the difference in cloud dimensions between the two stable conditions is not great compared to the attenuation distances involved.

Another important feature is that there is very little variation in dose rate between any of the stability classes for the plume height considered here. Figure 11A.2-2 illustrates this point more clearly by showing the dose rate for a 5-m/sec wind speed for each of the stability conditions. The variation of dose rate between stability conditions is very small for downwind distances less than 400 m, and is less than a factor of 2 even to a distance of 6 miles. From the shape of the dose rate curves, the maximum usually occurs within 1000 m and decreases rapidly thereafter.

The dose rates shown in Figures 11A.2-1 through 11A.2-3 are for points on the ground directly below the centerline of the sector-averaged plume. As previously mentioned, significant dose contributions can also occur in sections other than the one in which the plume is traveling. Because of symmetry, there are only nine unique sectors for which dose rate calculations can be made.

If the sector in which the plume is traveling is designated as sector 1 (Figure 11A.2-4), then the dose to sector 16 from the plume is equal to the dose to sector 2; the dose to sector 15 is the same as the dose to sector 3, and so on. In terms of the dose rate matrix, the following equalities can be listed:

$$DR_{\gamma}^{ij1,1} = DR_{\gamma}^{ij1,1}$$
$$DR_{\gamma}^{ij16,1} = DR_{\gamma}^{ij2,1}$$
$$DR_{\gamma}^{ij15,1} = DR_{\gamma}^{ij3,1}$$
$$\vdots \qquad \vdots$$
$$DR_{\gamma}^{ij9,1} = DR_{\gamma}^{ij9,1}$$

However, for distances greater than 100 m, the dose rate to adjacent sectors is very small because of the large separation distances. This is illustrated by Figure 11A.2-3, which shows the sector variation of dose rate with distance for one particular meteorological condition. In practice, the dose rate to a point in sector k is not calculated if the dose rate is less than 0.1% of the dose rate to a point at the same downwind distance in sector 1.

Figures 11A.2-1 through 11A.2-3 indicate how the dose rate matrix $DR_{\gamma}^{ijk,k'}$ is constructed. One must then find the joint frequency distribution F^{ijk} to calculate the annual dose rate.

11A.2.1.3.4 Conclusions About Gamma Dose Calculations

From the data presented in Figure 11A.2-5, it is concluded that the analytical model provides a fairly precise correlation between stack release rate and ground-level gamma radiation dose. The maximum dose is at the closest point to the stack. This is expected because at the base of the stack, for example, the dose rate is continuous and independent of plume direction travel. The dose rate curves presented in Figures 11A.2-1 through 11A.2-3 indicate this.

Further examination of Figures 11A.2-1 through 11A.2-3, which show dose rates during each meteorological condition, leads to additional conclusions. The dose rate does not seem to be very sensitive to the atmospheric stability condition. This is markedly in contrast to the air concentration differences at ground level during the various stability regimes. It is widely known that, during very stable conditions near zero, air concentration exists at ground level from an elevated plume since it remains very narrow and highly concentrated aloft. On the other hand, unstable conditions promote rapid effluent growth and dispersion and highest ground-level air concentrations.

It appears that, while the gamma dose rate is quite insensitive to atmospheric stability, it is quite dependent on plume height and wind speed. This is to be expected intuitively from Equation 11A.1-4 in which the average concentration that is used to obtain the dose rate is inversely proportional to wind speed and the attenuation distances increase with plume height. In practice, buoyant effluents are typical (although not universal), so that effluent buoyancy enters the calculations (i.e, plume height is made up of stack height plus plume rise due to buoyancy). The latter is greatest for slowest wind speeds. Thus, the slowest wind speed conditions do not, a priori, yield the largest dose rates. Experience with calculations using this analytical model verifies this.

Calculations have also shown that most of the dose over a long period of time comes from the conditions where the wind speed is about at the average speed of 4 to 7 m/sec (9 to 16 mph), which most locations are observed to have. The calculation for Brookhaven is no exception. This can partially be explained by the fact that for elevations considered here (300 to 400 ft), low wind speeds, for example, are rather infrequent accounting for about 3% of the time.

A final conclusion drawn from the comparison of calculated and measured doses refers to the dose pattern depicted in Figure 11A.2-5. It is observed that for distances out to about 0.5 mile (typical large reactor site) the iso-dose contours exhibit a smooth rather than a peaked pattern. This is quite different from the wind direction distribution (wind rose, see Table 11A.2-5) where total direction frequency is indicated. However, the smooth gamma dose pattern, as indicated in Figure 11A.2-5, is attributed to the fact that the total dose at each point is made up of the dose from plumes traveling in all directions. At distances of 2 miles and beyond, the gamma dose contours exhibit a peaked pattern similar to the wind rose. At these distances, only plumes traveling in the direction of a dose point contribute significantly to the gamma dose at the point.

11A.2.1.3.5 Ground-Level Air Concentration Calculation

For some kinds of radiation doses, only the ground-level air concentration is of interest. Examples of these are dose from inhalation, external beta dose, and deposition. In each of these, concentration at the dose point determines the dose regardless of the concentration at other points in the plume. This method of calculating the correlation between stack emission rate and ground-level air concentration is also of interest in assessing environmental effects of a stack effluent.

Some limited air concentration measurements are also made at Brookhaven.[8] These are measurements of small quantities of iodine released from the BGRR. Three monitoring stations were operated in 1963, although since then the scope of this program has been augmented. The release of I-131 from the BGRR was about 0.1 μCi/sec continuously.

11A.2.2 [redacted] NUCLEAR POWER STATION UNIT 1

In the case of the [redacted], data were used for a period when offgas release was measurable in the environment. The values used in these determinations were obtained from paired 10-mR gamma dosimeters in each of the environmental stations at [redacted]. All chambers were read weekly with results averaged on a monthly basis. The calculated iso-dose contours relating doses in the environment to [redacted] Unit 1 stack emission were compared to the field data. A 7-month period when the release rate averaged about 51,500 μCi/sec (September 1964 to March 1965) was selected as the test period. Since the environmental background was known to be decreasing during the period of interest, comparable 7-month periods the year before and the year following were averaged to minimize seasonal meteorological variables. The background obtained was subtracted from the gross reading during the test period. Results were compared to predicted values obtained from iso-dose curves at a release rate of 700,000 μCi/sec ([redacted] license limit)* related to 51,500 μCi/sec. Statistical analyses have shown that the ion chambers lowest level of detection at the 95% confidence level was approximately 0.4 to 0.5 mR/week above background. Results of these comparisons are shown in Figure 11A.2-6. In this case, the measurements are quite near the sensitivity of the instruments so that more uncertainty exists in the data compared with the Brookhaven data.

However, agreement between predicted and observed is still quite good. In the direction of predicted maximum dose, predicted is greater than observed. However, in one direction, predicted is less than observed.

On the basis of the above experimental data supporting the analytical methods, it is concluded that the proposed offgas limits are reasonably realistic, conservative values.

* [redacted] Unit 1 license limit was recently changed by the NRC to 500,000 μCi/sec.

11A.3 GASEOUS RELEASE RATE CALCULATIONS FOR DAEC

The analytical model described in Sections 11A.1 and 11A.2 was used in the following calculations for the DAEC plant site.

11A.3.1 NOBLE RADIOGAS RELEASE COMPOSITION

11A.3.1.1 Steam Jet Air Ejector

The offgas treatment system used in the DAEC employs a catalytic recombiner and a 12-bed charcoal adsorption unit as described in Section 11.3.2.1. Table 11A.3-1 lists the radiogases and their respective release rates as they leave the recombiner/ charcoal system to the environment for an assumed fuel defect operational condition corresponding to 100,000 μCi/sec (referenced to 30-min decay) input activity and a condenser inleakage of 18.5 scfm.

11A.3.1.2 Gland Seal Exhaust

An additional source of radioactive noble gases released to the environment is the gland seal steam condenser. Approximately 0.1% of the total steam flow is routed through the gland seal system to the gland seal condenser. The result is a release of approximately 0.1% of the total noble gases through the gland seal system. These gases are routed to the stack and then released to the atmosphere. Based on a total noble gas generation of 100,000 μCi/sec at 30-min decay, release from the gland seal system will be 100 μCi/sec at 30-min decay. However, these gases are approximately 1.75-min old when released from the stack. As a result, 464 μCi/sec is released to the environment. The isotopic composition of this release is shown in Table 11A.3-2.

11A.3.1.3 Building Ventilation Exhaust

Ventilation exhaust from plant structures is an additional source of noble gas activity that has previously been considered negligible but takes on greater significance when compared to the offgas discharge from a plant employing a recombiner/charcoal system such as the DAEC. The noble gases contained in the ventilation exhaust result primarily from small steam leaks from reactor coolant pressure boundary system piping external to primary containment.

11A.3.2 WHOLE-BODY DOSE CALCULATIONS

11A.3.2.1 Gamma Dose

11A.3.2.1.1 Meteorology

The procedure for calculating annual gamma dose consists of calculating the dose rate at various points during each different meteorological condition, weighting the dose

rate by the frequency of occurrence, and summing over the year to determine the total dose.

Gamma dose rate calculations were done for a number of downwind dose points. Terrain height of these dose points was considered in the calculation. These dose rates were weighted by the frequency of occurrence of wind speed and direction and atmospheric stability in accordance with the DAEC onsite meteorological data and summed to give a total "air dose" for the year. Onsite meteorological data were collected from January 1971 through January 1972 at both 156-ft and 33-ft elevations as described in Section 2.3 and summarized into appropriate distributions to allow the calculation of both stack and building ventilation releases.

For the building ventilation releases, aerodynamic downwash of the effluent plume was assumed to occur whenever wind speeds were greater than 3 m/sec. A building dilution factor of 1.4 was applied for downwash conditions for distances out to 1500 m. For wind speeds less than 3 m/sec, a release height of 40 m was assumed with no downwash or building wake effect.

The characteristics of the DAEC stack offgas exhaust system design are such that vertical momentum and buoyancy effects result in a plume rise that is negligible when compared with the stack height. Accordingly, the release elevation used in the calculations is 100 m for the offgas and gland seal discharge.

11A.3.2.1.2 Shielding and Occupancy Considerations

All doses calculated in this section are doses a detector would receive. However, doses to persons would be lower for two reasons. First, at least part of the time persons would be inside structures, such as their homes, which provide shielding. Second, most people would not stay at a given location all the time. While away from this location, they would not be exposed to plant releases. Thus their annual exposure would be lower. Conservatively, all doses could be reduced by a factor of 2 to give a dose to people. In many cases, the dose would be even more, anywhere from a factor of 3 to a factor of 8 lower.

11A.3.2.1.3 Steam Jet Air Ejector Doses

Table 11A.3-3 shows offsite gamma doses for the air ejector offgas system corresponding to the isotopic composition and release rate shown in Table 11A.3-1. The maximum calculated dose of 0.97 mrem/yr occurs in the north sector. Figure 11A.3-1 shows the calculated exposures for each direction sector on the site boundary.

11A.3.2.1.4 Gland Seal Exhaust Doses

Gamma dose corresponding to the gland seal release indicated in Table 11A.3-2 was calculated in the same manner as the air ejector offgas doses. For an annual average

release of 464 μCi/sec, the dose at the site boundary in the north sector is 0.47 mrem/yr. Table 11A.3-4 lists gamma doses calculated for the gland seal stack release.

11A.3.2.1.5 Building Ventilation Exhaust Doses

Gamma doses were calculated at site boundary locations in each sector as indicated in Table 11A.3-5 using an arbitrarily assumed annual average release from ventilation exhausts of 278 μCi/sec using a 5-min decay mix. Table 11A.3-5 indicates that the maximum dose of 1.07 mrem/yr occurs in the south-southeast sector.

11A.3.2.1.6 Dose Variation with Release Rate and Distance From Plant Stack

Table 11A.3-6 presents the variation of calculated annual average gamma dose as a function of distance from the plant stack from the combined gland seal exhaust and air ejector offgas contribution of 2104 μCi/sec assuming an operational condition corresponding to 100,000 μCi/sec at 30-min decay.

Table 11A.3-7 presents the calculated annual average gamma dose at the maximum site boundary location from the combined air ejector offgas and gland seal exhaust contribution for several different levels of fuel defect operation.

11A.3.2.1.7 Design Objective Release Rate

11A.3.2.1.7.1 Main Stack

As indicated in Section 11.3.1.1, the design objective of the gaseous radioactive waste system is to limit the annual average exposure to a point on the site boundary to a maximum of 10 mrem/yr. This objective for the DAEC meets the intent of as low as reasonably achieveable as defined by 10 CFR 20.

The annual average release rate corresponding to the design objective site boundary exposure of 10 mrem/yr-is 11,800 μCi/sec-release of air ejector offgas from the recombiner/charcoal system, which contributes 7.0 mrem/yr, and a 3,300 μCi/sec-release of 1.75-min-old gland seal exhaust, which contributes 3.0 mrem/yr.

Thus, the 10 mrem/yr design objective release rate for the main stack is 15,100 μCi/sec on an annual average basis.

11A.3.2.1.7.2 Building Ventilation

As indicated in Section 11A.3.2.1.5, the maximum dose from the ventilation exhaust was calculated to be 1.07 mrem/yr for an arbitrarily assumed annual average release rate of 278 μCi/sec and occurs in the south-southeast sector. If this source alone is considered, the design objective site boundary dose of 10 mrem/yr results from an annual average ventilation exhaust release rate of 2,608 μCi/sec.

11A.3.2.1.7.3 Correlation of Ventilation and Stack Effluent Releases

Since the overall objective is to maintain the total site boundary dose at less than 10 mrem/yr, a relationship must be used to relate stack and ventilation releases such that the sum of the exposure contributions is within the 10 mrem/yr design objective. This relationship is as follows:

$$\frac{Q_{vent}}{2{,}608} + \frac{Q_{stack}}{15{,}000} \leq 1.0$$

where

Q_{vent} = annual average ventilation release rate, μCi/sec.
Q_{stack} = annual average stack release rate, μCi/sec.

This combination technique is conservative since the stack and building vent release rates are predicated on exposures affecting two different sectors.

11A.3.2.1.8 Population Dose Calculation

In addition to calculating the doses to offsite locations, calculations have also been performed to determine the general population dose. The results are shown in Table 11A.3-8.

The population dose is in units of man-rem per year. The assumed annual average release considered corresponds to 100,000 μCi/sec at 30-min decay delayed to the holdup time appropriate for the 12-bed recombiner/charcoal system used for the DAEC giving a net release rate of 2104 μCi/sec. The population distribution used was the estimated population around the DAEC site in the year 2010.

The exposure calculated is 10.2 man-rem/yr. Natural background radiation would result in 139,000 man-rem/yr. Thus, the population exposure from this source would be 0.007% of natural background per year. Even at the design objective release rate of 15,100 μCi/sec, the plant increment would only be about 0.12% of background.

The background assumption is 150 mrem/yr. Assuming that natural background varied only 1 mrem/yr, this would change the natural population dose around the DAEC by about 925 man-rem/yr. This is more than the plant contribution by nearly a factor of 100 when the release averages 2104 μCi/sec. Actually, natural background would vary more than 1 mrem/yr (more like 10) so that the contribution from the plant would be virtually undetectable.

11A.3.2.2 Beta Dose

The range of beta particles in air is only a few meters. Hence, for beta calculations, a cloud of material released via a stack and which expands to large dimensions at downwind distances where the cloud has reached ground level is frequently considered an "infinite" cloud. In such a cloud, the air dose rate is calculated assuming that the rate of energy release per unit volume in the cloud is equal to the rate of absorption in that volume (no buildup). The body is considered a small volume within the flux of the cloud and causes no perturbation in the flux.

Beta flux incident on the human body comes from one direction only, so that the air dose rate at the surface of the body is only one-half of that in the air. In addition, the cloud is not infinite since the ground represents a boundary to the cloud, such that at the ground the cloud is a hemisphere of infinite radius. It approaches the infinite cloud at some height above the ground equal to the range of the betas in air. There will be a variation in the dose rate from the head to the foot of an individual, with the highest dose rate at the head. This factor varies from 0.5 at the ground to 1 at heights greater than the range of betas in air. Taylor[9] has computed this effect to show that the average dose to the body of a person 1.8 m tall is about 0.64 times the semi-infinite cloud dose. This factor applies for mixed fission products with maximum energies of about 1 to 2 MeV.

The following beta dose equation[10] is used and modified:

$$D_B = 0.457\overline{E}X \qquad \text{(11A.3-1)}$$

This equation is multiplied by 0.5 for the beta flux factor discussed above and by 0.64 to account for the average dose to the body. Converting Equation 11A.3-1 into a dose rate yields the equation used in this analysis:

$$(DR)_B = 0.53\times10^6\,\overline{E}(X) \qquad \text{(11A.3-2)}$$

Substituting X^i_{avg} for (X) gives the average beta dose rate avg for the i^{th} meteorological condition. Since the range of betas in air is quite short, the annual total beta dose in a given direction is the sum of the dose rates (in millirad per hour) during each i^{th} condition accompanied by wind blowing in that direction weighted by the annual frequency (in hours) of occurrence. The conversion of this dose into a dose delivered to an individual requires adjustments to take the shielding effect of clothing into account.

In addition, for the beta dose to be truly additive with the whole-body gamma dose, the betas must penetrate the skin. Several discussions are found in the literature. Even radiation protection regulations differentiate between whole-body doses and the skin of the whole-body dose. Hendrickson[11] indicates that about a factor of 70 difference exists between whole-body and skin doses. Dunster[l2] also indicated that care should be exercised when discussing doses by properly identifying whether one is considering external dose to the whole body from penetrating radiation, skin dose from beta radiation,

and dose to the thyroid from internal radiation from inhaling or ingesting I-131. In discussing the inadequacy of some references to dose from Kr-85, Dunster indicated that the whole-body dose and gonad dose from immersion in Kr-85 (a beta emitter) is only about 1% of the skin dose. Some references seem to add or equate skin dose with the whole-body or gonad dose. Bond[13] basically supports the position of Hendrickson and Dunster but arrives at about a factor of 1000 difference between skin dose and whole-body dose.

Since the beta dose to the skin typically contributes a small fraction of the whole-body dose, it was not added to the whole-body dose. Whole-body dose is far more limiting than external skin dose.

11A.3.3 INTERNAL DOSE CALCULATIONS

11A.3.3.1 Introduction

During normal plant operation, small quantities of radioiodine may be discharged from the gland seal exhaust system via the main stack and from the building ventilation system. Radioiodine discharged from the main stack comes primarily from the gland seal exhaust. Although radioiodine may be present in gases leaving the steam jet air ejector, it will undergo essentially complete adsorption on the charcoal delay beds before being discharged from the main stack. Small leaks that could occur in process piping and equipment may result in discharges of radioiodine from the building ventilation system.

Inhalation or ingestion of radioiodine will result in ultimate deposition of the material in the thyroid gland because of natural metabolic processes. This, of course, presumes that such exposure pathways actually exist. Ingestion, rather than inhalation, is the controlling path for thyroid exposure from radioiodine at the DAEC site because of concentration.effects unique to the air-vegetation-cow-milk-infant food chain. This concentration effect is accounted for by reducing the annual average maximum permissible air concentrations of I-131 at grazing locations by a factor of 700.

11A.3.3.2 Design Objectives for Radioiodine Releases

To maintain levels of radioiodine in gaseous effluents released from the DAEC as low as reasonably achievable, DAEC proposes the following design objectives:

1. Annual average concentrations at any location on the boundary of the site or in the offsite environment of radioactive iodines or radioactive material in particulate form with a half-life-greater than 8 days shall be less than the concentrations in air specified in Appendix B, Table 2, Column I of 10 CFR 20 divided by 100.

2. Where there are grazing animals providing fresh milk for human consumption, the 10 CFR 20, Appendix B, Table 2, Column I concentration of I-131 will be divided by 7000. (The factor of 7000 results from a design objective of 10% of the Appendix B, Table 2 value divided by 700 to account for concentration effects in the milk food

chain.) However, this number may be reduced by considerations such as the following:

a. The fraction of the year during which grazing is impossible.

b. The fraction of dilution provided by pooling at a central dairy.

11A.3.3.3 Inventory of Dairy Farms Surrounding DAEC Site

In order to establish a design objective release rate for I-131 from the DAEC, the grazing season average air concentration at the nearest pasture supporting a dairy herd in each meteorological sector was required. Accordingly, an inventory was made to determine the location of the nearest dairy farm in each sector, and the results of this inventory are summarized in Table 11A.3-9. This table includes such information as name and location of dairy farm, distance from plant stack, number of dairy cows, months on pasture, and name and location of distributor. Table 11A.3-9 shows that for most of the farms interviewed, the grazing season was identified as the 6-month period, May through October. However, for conservatism the calculations of this section assumed a 7-month grazing period extending from April 15 to November 15. Table 11A.3-10 shows the monthly production of dairy farms versus the monthly production of their distributors for a typical month (June 1971) felt to be representative for milk production and distribution. This table also delineates Class B dairies. Class B dairies are those whose milk product is not distributed for direct liquid consumption but rather for cheese and other dairy products that would not have the potential of radioiodine ingestion.

11A.3.3.4 Calculation of Design Objective I-131 Release Rate from Main Stack

The average ground-level air concentration of I-131 was determined for each farm listed in Table 11A.3-9 for the April 15 through November 15 grazing season using onsite data from the 156-ft level of the DAEC meteorological tower. The I-131 release rate from the main stack was arbitrarily assumed to be 0.01 μCi/sec. Table 11A.3-11 presents the results of these calculations. The highest calculated concentration is 2.69 x 10^{-16} μCi/cml at the farm located in the west-northwest sector. Since the average annual I-131 concentration that would be equal to MPCa/7000 is 1.43 x 10^{-14} μCi/cm^3, the design objective release rate from the stack would be 0.53 μCi/sec. However, as discussed previously the farm is used for grazing only 7 months of the year. The consideration of this factor would establish a design objective main stack release rate for I-131 of 0.91 μCi/sec on an annual average basis.

As Table 11A.3-9 shows, the farm located in the west-northwest sector is listed as a Class B dairy, meaning that the milk product is not commercially distributed for direct liquid consumption.

This would appear to indicate that the farm in the south-southwest sector, which is a Class A dairy, should be used as the basis for release rate calculations, with a

resultant increase in the design objective release rate. However, for conservatism it has been assumed that family consumption of liquid milk at the Class B farm is possible. In addition, infant consumption of the liquid milk is conservatively assumed.

Further reference to Table 11A.3-9 indicates that the population exposure from Class B dairy products is essentially zero, and the dilution afforded by the pooling of Class A liquid products at the central dairy is such that population exposure from this source is totally negligible. The monthly production of Class A dairy farms surrounding the DAEC site is on the order of 0.1% of the total product distributed by the central dairy.

11A.3.3.5 Calculation of Design Objective I-131 Release Rate from Building Ventilation Exhaust

The meteorological assumptions used in this calculation varied slightly from those used for noble gas ventilation releases in that aerodynamic downwash occurs for wind speeds greater than 3 m/sec, but a release height of 0 m was conservatively assumed instead of using a building dilution factor. For wind speeds equal to or less than 3 m/sec, a release height of 40 m was used.

An annual average release rate of 0.001 μCi/sec of I-131 was arbitrarily assumed as the source term. The resultant grazing season average concentration at the dairy farms surrounding the DAEC is in Table 11A.3-12. As before, the highest concentration occurs at the farm in the west-northwest sector. Normalizing the ground-level concentration to MPCa/7000 (1.43×10^{-14}) results in a release rate of 0.044 μCi/sec. However, since the farm is used for grazing only 7 months of the year, the consideration of this factor would establish a design objective annual average ventilation release rate for I-131 of 0.075 μCi/sec.

11A.3.3.6 Design Objective Release Rate for Combined Ventilation and Stack Effluent Releases

The correlation for ventilation and stack effluent releases for I-131 is similar to that developed in Section 11A.3.2.1.7.3 for noble gases.
The design objective release rate is

$$\frac{Q_{vent}}{0.075} + \frac{Q_{stack}}{0.91} \leq 1.0$$

where

Q_{vent} = annual average ventilation release rate, μCi/sec
Q_{stack} = annual average stack release rate, μCi/sec

11A.4 SIGNIFICANCE OF RADIATION EXPOSURE

11A.4.1 NATURAL RADIATION BACKGROUND

Every day people receive radiation from the sky, the ground, the air, and food. The magnitude of this radiation level is strongly influenced by where people live, what they do, and even in what kind of house they live. For most locations around the United States, this natural radiation level averages about 140 mrem/yr. This typical value can be discussed in its various component contributions.

11A.4.1.1 Cosmic Radiation

Cosmic radiation is one of the more significant sources of natural radiation. This radiation is to some extent dependent on latitude and to a large extent dependent on altitude.

In the mid-latitudes, where most people live, the cosmic radiation varies from about 50 mrem/yr at sea level to about 3800 mrem/yr at altitudes that jet aircraft fly (35,000 ft). This does not mean that all commercial jet-airliner crews receive 3800 mrem/yr, since this would assume that the crews were continuously airborne. Assume, for instance, that these crews stay aloft a tenth of the year; thus their occupational radiation exposure due to cosmic radiation alone would be in the range of 300 to 400 mrem/yr. Even one transcontinental round trip would give the business man or vacationer about 4 mrem/yr.

The average cosmic radiation of 50 mrem/yr will increase to about 150 mrem/yr for some mile-high locations such as Denver and Salt Lake City. Even with this, 50 mrem/yr seems to be a good average.

11A.4.1.2 Radiation from Ground

Another source of radiation in nature is the ground itself because it contains many radioactive minerals, particularly the uranium and thorium series, together with the important uranium decay product, radium. Another significant radioisotope in the ground is potassium-40, the naturally radioactive isotope of the element potassium. This incidence of radioactive material in the ground causes the earth to act as a large plane radiation source. This produces an average radiation exposure in the continental United States of about 45 mrem/yr. Assuming that the average person spends about one-fourth of the time walking on the ground outside of buildings, this 45 mrem/yr would reduce to 15 mrem/yr.

There are a number of locations in the world where the radiation exposure from the ground is actually much higher. In various locations in Brazil, India, and in the French mountains, the exposure may range from 180 to as high as 1600 mrem/yr. This is largely because of the presence of deposits of thorium near the surface of the ground. There also have been reports of exposures higher than these.

11A.4.1.3 Radiation from Air

The radioisotopes in the ground give rise to a secondary source of radiation, since the natural decay of the uranium and thorium series each contains a natural radiogas. These radiogases evolve from the ground at a fairly constant rate and thus cause concentrations of natural radiogases in the air.

The principal constituent of this source of exposure of radiation in nature is the radiogas radon, which has a 3.8-day radioactive half-life. This element, together with its daughter decay products, causes a world average of about 5 mrem/yr full-body, external radiation exposure. Actually, the inhalation of these radiogases and the deposition of their radioactive daughters in the lung may cause a lung dose of about 200 mrem/yr.

11A.4.1.4 Radiation from Structures

Since man uses materials from the ground for buildings, natural radioisotopes from the ground are transferred to these structures. A significant variation will result from the use of different building materials. A wooden structure may give a radiation dose rate of about 50 mrem/yr, while concrete may give 70, and brick as high as 100. Even these may vary within a particular material on the basis of where the material originated.

11A.4.1.5 Radiation from Food and Water

Another source of radiation in nature is food and water, both of which contain naturally radioactive materials. The general average radiation exposure from food and water is about 25 mrem/yr from the deposition and retention of these radioactive materials within the body. In a typical case, about 20 mrem/yr of this exposure comes from the natural radioisotope potassium-40, which is found particularly in protein-type foods.

11A.4.1.6 Total Radiation from Nature

The following summarizes the various contributions in arriving at the average natural background radiation of 140 mrem/yr.

Source	Radiation (mrem/yr)
Cosmic rays	50
Ground (1/4 time)	15
Buildings (3/4 time)	45
Air	5
Food and water	25
Total	140

11A.4.2 MANMADE RADIATION

Man has added to the radiation exposure from nature in a number of ways. The largest contribution has been from use of medical and dental X-rays. Typically, an average of 55 mrem/yr is received by the average person. Recent reports indicate that 100 mrem is applicable. In addition, radiation from luminous watch dials (2 mrem/yr) and television viewing (1 to 10 mrem/yr) also contribute to manmade exposure. The result is that about 50 to 100 mrem/yr is added to the natural background radiation exposure. Therefore, 200 to 250 mrem/yr is the exposure received by the average U.S. resident.

11A.4.3 RADIATION IN PERSPECTIVE

Reference 14 gives a better perspective on the significance to health of various radiation doses and the experience forming the basis for permissible doses.

Most of the permissible exposure limits for persons have originated from the accidental radiation exposure that occurred in the 1920s and 1930s, largely in the state of New Jersey, to radium-dial painters. In this industry at that time, workers had the habit of shaping their paint brushes with their tongue when painting radium on instrument dials. A considerable quantity of radium was ingested by a number of women. Although many of them have died as a result of this radiation exposure, many have shown no significant radiation effects in the subsequent 30 to 40 yr. The dividing line seems to be that if the deposition of radium was less than about 0.5 μCi fixed in the bone marrow, there has been no significant subsequent radiation effect.

The radium deposition of 0.5 μCi in the bone will give an annual radiation exposure of about 150,000 mrem/yr.

The basic permissible doses have been determined largely from this radium exposure experience. Because there has been relatively little effect from 1 μCi of radium fixed in the bone and none from about 0.5 μCi, the permissible dose to bone has been established at 0.1 μCi deposited, which will result in a radiation exposure of 29,000 mrem/yr. Thus, there is a built-in safety factor of 5 to 10 included in any permissible limits based on this consideration as a result of this arbitrary reduction in the permissible dose to bone.

The permissible occupational exposure of 5000 mrem/yr is determined largely from the permissible dose to bone, but there is an additional reduction of about 6 between the permissible dose to bone and the permissible full-body occupational dose.

When exposure to the neighbors of nuclear power plants is considered, the permissible occupational dose of 5000 mrem/yr contains an additional reduction factor of 10 resulting in the permissible general public dose of 500 mrem/yr. Therefore, it can be seen that between a dose per year that might cause injury, and the permissible dose to the general public, there is an overall safety factor of several hundred. This is not to suggest that the permissible limits are too low, but merely to point out the substantial safety factors included in the current permissible radiation exposure limits. Such safety factors are perhaps greater than will be found in any other limits based on industrial hygiene or public health considerations.

These are the permissible limits that apply to licensed facilities such as nuclear power plants. There is, however, no expectation that the actual dose to the general public from the operation of DAEC will come anywhere near the permissible dose of 500 mrem/yr. In fact, the design objective for the DAEC is that the effects on neighbors will be far below any permissible radiation dose consideration and will be as low as reasonably achievable.

The actual expectancy is that the typical neighbor in the vicinity of the DAEC will receive a whole-body exposure that may be on the order of 1 mrem/yr from typical waste disposal operations, as averaged over the operating life of the plant. The further away from the plant, the estimated dose would be even lower because whole-body exposures from elevated stacks decrease with distance.

Radiation exposure to man from any one source must be viewed in proper perspective. One essential perspective is provided by considering the full spectrum of other sources of ionizing radiation that people are exposed to every day. Radioactivity released from nuclear power plants makes only a small contribution to this broad spectrum of natural background radiation.

The National Academy of Sciences has stated that radiation is by far the best understood environmental hazard. This scientific understanding has been translated into permissible levels of radiation exposure to man. These in turn have been incorporated into the limits set by the NRC, limits set in the interest of public health and safety.

11A.5 SUMMARY

The method of calculating a stack release limit is given along with partial verification of the method using data from Brookhaven National Laboratory. The whole-body gamma dose calculations are quite close to that observed at Brookhaven. The ground-level integrated air concentration calculations give an order-of-magnitude type of verification because of the lack of sensitive field measurements.

Release rate calculations have been performed for the DAEC site using the onsite meteorological data.

Design objective release rates have been established for noble gases and radioiodines and particulates having half-lives greater than 8 days.

It is expected that it should generally be feasible to keep average annual releases of radioactivity in airborne effluents within their design objective levels.

At the same time, DAEC is permitted the flexibility of operation, compatible with considerations of health and safety, to ensure that the public is provided a dependable source of power even under unusual operating conditions that may temporarily result in releases in excess of the design objective but still within the limits specified in 10 CFR 20. These limiting conditions of operation are presented in the Technical Specifications.

It is recognized that a precise determination of dose from a certain emission from the stack is only possible by direct measurement. Such information will be provided by the environmental monitoring program conducted at and around the site. If the stack emission ever reaches a level such that it is measurable in the environment, such measurements will provide a basis for adjusting the proposed stack limit long before the effect in the environment is of any safety concern.

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Table 11A.1-1

DEPOSITION VELOCITY COEFFICIENTS[a]

Stability Condition	$v_d \overline{u}_o$	
	Particulates	Halogens
Very stable	1.5×10^{-4}	2.4×10^{-3}
Moderately stable	2.2×10^{-4}	3.4×10^{-3}
Neutral	3.0×10^{-4}	4.6×10^{-3}
Unstable	6.0×10^{-4}	8.0×10^{-3}

[a] To obtain the deposition velocity, multiply this ratio of deposition velocity to surface wind speed by the surface speed $\left(\overline{u}_o\right)$.

Table 11A.1-2

DIFFUSION COEFFICIENTS

Constants	Stability Condition			
	Very Stable	Moderately Stable	Neutral	Unstable
$a(m^2)$	34	97	- -	- -
$b(m^2/sec)$	0.025	0.33	- -	- -
$K^2(sec^{-2})$	8.8×10^{-4}	2.5×10^{-4}	- -	- -
C_z(u=1m/sec)	- -	- -	0.15	0.30
C_z(u=5m/sec)	- -	- -	0.12	0.26
C_z(u=10m/sec)	- -	- -	0.11	0.24
	- -	- -	0.25	0.20

Table 11A.1-3

NOBEL GAS ISOTOPES CONSTITUTING MIXTURE

Isotope	Half-life	$E\gamma(MeV)$	$N(No.\gamma' s / dis)$	μ per Meter	μ_a per Meter
Kr-83m	1.86 hr	1.3-2[a]	1.6-1	1.3-1	1.3-1
Kr-85m	4.4 hr	1.3-2	5.2-2	1.3-1	1.3-1
		1.5-1	7.7-1	1.8-1	3.3-3
		3.1-1	1.4-1	1.4-1	3.7-3
Kr-85	10.76 yr	5.1-1	4.4-3	1.1-2	3.9-3
Kr-87	76 min	4.0-1	5.9-1	1.2-2	3.8-3
		6.7-1	2.5-2	9.8-3	3.8-3
		8.4-1	8.0-3	88.9-3	3.7-3
		8.5-1	8.1-2	8.8-3	3.6-3
		1.2+0	1.4-2	7.6-3	3.5-3
		1.3+0	7.5-3	7.0-3	3.4-3
		1.4+0	5.5-3	6.9-3	3.4-3
		1.4+0	4.6-2	5.7-3	3.2-3
		2.6+0	1.5-2	4.4-3	2.9-3
		3.1+0	1.0-2	4.5-3	2.6-3
Kr-88	2.8 hr	1.7-1	6.9-2	1.7-2	3.4-3
		2.0-1	3.8-1	1.6-2	3.5-3
		3.6-1	3.0-2	1.3-2	3.8-3
		4.3-1	1.2-2	1.2-2	3.9-3
		8.3-1	1.3-1	9.0-3	3.7-3
		1.1+0	3.4-2	8.0-3	3.5-3
		1.5+0	1.5-1	6.8-3	3.3-3
		2.0+0	9.6-2	5.6-3	3.0-3
		2.3+0	5.7-1	5.4-3	2.9-3
Kr-89	3.2 min	2.2-1	2.8-1	1.6-2	3.5-3
		3.2-1	1.4-1	1.3-2	3.8-3
		4.7-1	1.5-1	1.2-2	3.9-3
		5.9-1	3.2-1	1.2-2	3.9-3
		7.3-1	7.5-2	9.6-3	3.8-3
		9.1-1	3.0-1	8.6-3	3.6-3
		1.8+0	5.5-1	6.1-3	3.2-3
		3.4+0	1.1-1	4.6-3	2.6-3
		4.4+0	1.3-2	3.7-3	2.4-3
		2.9+0	4.7-2	4.7-3	2.7-3

[a] 1.3-2 = 1.3×10^{-2}

Table 11A.1-3

NOBEL GAS ISOTOPES CONSTITUTING MIXTURE

Isotope	Half-life	$E\gamma(MeV)$	$N(No.\gamma' s / dis)$	μ per Meter	μ_a per Meter
Kr-90	33 sec	1.1-1	1.2-1	2.6-2	3.0-3
		1.2-1	5.4-1	1.9-2	3.1-3
		1.7-1	4.1-2	1.7-2	3.4-3
		2.4-1	1.2-1	1.6-2	3.6-3
		4.3-1	2.1-1	1.2-2	3.8-3
		5.9-1	5.6-1	1.1-2	3.9-3
		1.1+0	5.3-1	7.6-3	3.5-3
		1.7+0	3.2-1	6.2-3	3.2-3
		2.3+0	7.3-2	5.4-3	2.9-3
		2.9+0	8.9-2	4.7-3	2.6-3
Kr-91	10 sec	1.1-1	7.5-1	2.0-2	3.1-3
		5.1-1	1.5-1	1.1-2	3.8-3
Kr-93	2 sec	2.0-1	5.0-1	1.6-2	3.6-3
Xe-131m	11.7 days	3.0-2	5.9-1	4.6-2	2.0-2
		1.6-1	2.3-2	1.7-2	3.4-3
Xe-133m	2.3 days	3.0-2	1.4-1	4.0-2	2.0-2
		3.4-2	3.2-2	3.9-2	1.6-2
		2.3-1	8.0-2	1.5-2	3.6-3
Xe-133	5.27 days	3.1-2	3.8-1	4.0-2	1.6-2
		3.5-1	8.6-2	3.8-2	1.4-2
		8.0-2	6.0-3	2.1-2	3.8-3
		8.1-2	3.7-1	2.0-2	4.5-3
		1.6-1	6.4-4	1.7-2	3.4-3
		2.2-1	2.4-6	1.6-2	3.6-3
		3.0-1	5.1-5	1.4-2	3.7-3
		3.8-1	2.3-5	1.3-2	3.8-3
Xe-135m	15.6 min	3.0-2	1.4-1	4.0-2	1.6-2
		5.3-1	8.2-1	1.1-2	3.8-3
Xe-135	9.2 hr	3.1-3	4.5-2	4.0-2	1.6-2
		1.6-1	2.1-3	1.7-2	3.3-3
		2.0-1	2.6-4	1.6-2	3.5-3
		2.5-1	9.2-1	1.5-2	3.6-3
		3.6-1	2.2-3	1.4-2	3.8-3
		3.7-1	1.1-4	1.3-2	3.8-3
		4.1-1	3.1-1	1.2-2	3.8-3
		5.7-1	5.0-5	1.1-2	3.8-3
		6.1-1	2.4-2	1.1-2	3.8-3
		7.6-1	1.3-3	4.4-3	3.7-3

Table 11A.1-3 Page 3 of 3

NOBEL GAS ISOTOPES CONSTITUTING MIXTURE

Isotope	Half-life	$E\gamma(MeV)$	$N(No.\gamma' s / dis)$	μ per Meter	μ_a per Meter
Xe-137		3.6-2	7.0-2	1.8-2	3.3-3
		1.5-1	5.6-2	1.3-2	3.8-3
		3.7-1	1.7-3	1.3-2	3.8-3
		4.6-1	3.0-1	1.2-2	3.9-3
		6.6-1	4.0-4	1.1-2	3.9-3
		8.5-1	6.5-3	8.9-3	3.7-3
		1.0+0	4.4-3	8.1-3	3.6-3
		1.2+0	6.0-3	7.4-3	3.5-3
		1.7+0	8.9-3	6.2-3	3.3-3
		2.5+0	4.7-3	5.0-3	2.9-3
Xe-138	14 min	3.0-2	3.0-2	4.0-2	2.0-2
		1.6-1	7.8-2	1.8-2	3.3-3
		2.4-1	3.6-2	1.5-2	3.6-3
		2.6-1	3.7-1	1.4-2	3.7-3
		4.0-1	7.4-1	1.3-2	3.8-3
		4.1-1	2.8-2	1.2-2	3.8-3
		4.3-1	2.3-1	1.2-2	3.9-3
		1.8-1	2.0-1	6.1-3	3.2-3
		2.0-1	1.6-1	5.7-3	3.1-3
Xe-139	41 sec	1.2-1	6.0-3	2.0-2	3.1-3
		1.8-1	2.4-1	1.7-2	3.4-3
		2.2-1	7.2-1	1.6-2	3.5-3
		2.3-1	2.8-2	1.5-2	3.6-3
		2.4-1	1.6-1	1.4-2	3.7-3
		3.5-1	1.1-2	1.3-2	3.8-3
		4.1-1	1.1-1	1.3-2	3.8-3
		5.8-1	8.9-2	1.1-2	3.9-3
Xe-140	13.7 sec	8.3-2	1.5-1	2.1-2	3.7-3
		1.2-1	1.7-1	1.9-2	3.1-3
		2.1-1	8.0-2	1.6-2	3.6-3
		2.9-1	9.0-2	1.4-2	3.7-3
		4.2-1	1.3-1	1.2-2	3.8-3
		4.8-1	7.8-2	1.2-2	3.9-3
		5.9-1	3.4-1	1.1-2	3.8-3
		7.0-1	2.3-1	9.8-3	3.8-3
		8.6-1	4.9-1	8.8-3	3.7-3
		1.4+0	4.4-1	7.1-3	3.4-3

Table 11A.1-4

PARTICULATE DUAGHTER PRODUCTS

Isotope	Half-life	E (MeV)	μ per Meter	μ_a per Meter
Rb-88	18 min	9.91	0.0085	0.0037
		1.28	0.0072	0.0034
		1.85	0.0060	0.0032
		2.18	0.0050	0.0030
		4.2	0.0038	0.0024
CS-138	32.2 min	0.14	0.018	0.0033
		0.19	0.016	0.0035
		0.23	0.015	0.0037
		0.41	0.0122	0.0037
		0.46	0.0116	0.0038
		0.55	0.0108	0.0038
		0.87	0.0088	0.0037
		1.01	0.0082	0.0036
		1.43	0.0068	0.0034
		2.21	0.0055	0.003
		2.63	0.0050	0.0039
		3.34	0.0043	0.0026

Table 11A.1-5

DOSE REDUCTION FACTORS FOR SHIELDING

Structure	Location	Dose Reduction Factor
One-story frame	First floor, middle	0.4
	Basement, middle	0.07
One-story brick veneer	First floor, middle	0.3
	Basement, middle	0.06
Two-story brick veneer	First floor, middle	0.23
	Basement, middle	0.03
Multistory reinforced concrete	Upper floors (excluding top floor)	0.02
	Basement	0.001

Table 11A.2-1

FREQUENCY OF OCCURRENCE, IN PERCENT, OF VARIOUS WIND SPEEDS AND DIRECTIONS OBSERVED AT THE BROOKHAVEN NATIONAL LABORATORY SITE UNDER VERY STABLE ATMOSPHERIC CONDITIONS[a,b]

Wind Direction	Wind Speed[c] (mph)						All Wind Speeds
	0-3	4-7	8-12	13-18	19-24	>24	
N	0.0619	0.108	0.433	0.449	0	0	1.05
NNE	0.0619	0.139	0.155	0.124	0	0	0.48
NE	0.0309	0.0464	0.155	0.0309	0	0	0.26
ENE	0	0.0928	0.124	0.0309	0	0	0.25
E	0.0464	0.0774	0.155	0.0774	0.0619	0	0.42
ESE	0.108	0.201	0.263	0.0155	0	0	0.59
SE	0.0464	0.0309	0.0464	0.0155	0.0155	0	0.15
SSE	0.0619	0.139	0.201	0.124	0	0	0.53
S	0.0464	0.201	0.248	0.356	0.232	0.0309	1.11
SSW	0.0619	0.294	0.665	1.13	1.01	0.0928	3.25
SW	0.108	0.170	0.433	1.22	0.897	0	2.83
WSW	0.0928	0.124	0.340	1.11	0.557	0.0155	2.24
W	0.0464	0.186	0.804	0.712	0.541	0.309	2.32
WNW	0.0774	0.186	0.572	0.433	0.139	0	1.41
NW	0.0309	0.186	0.433	0.603	0.186	0	1.44
NNW	0.0155	0.124	0.433	0.789	0.0774	0	1.44
All	0.90	2.31	5.46	7.22	3.71	0.17	19.77

[a] Frequency of occurrence is based on 6464 hr of good observation during 1963.
[b] Stability-based temperature difference measured at the 410-ft and 37-ft levels.
[c] Wind speed measured at the 355-ft level.

Table 11A.2-2

FREQUENCY OF OCCURRENCE, IN PERCENT, OF VARIOUS WIND SPEEDS AND DIRECTIONS OBSERVED AT THE BROOKHAVEN NATIONAL LABORATORY SITE UNDER MODERATELY STABLE ATMOSPHERIC CONDITIONS[a,b]

Wind Direction	Wind Speed[c] (mph)						All Wind Speeds
	0-3	4-7	8-12	13-18	19-24	>24	
N	0	0.186	0.433	0.433	0.232	0.0464	1.33
NNE	0.0464	0.278	0.557	0.464	0.186	0.0309	1.56
NE	0.0774	0.232	0.557	0.124	0.0928	0	1.08
ENE	0.0309	0.139	0.139	0.201	0.155	0	0.66
E	0.0309	0.186	0.263	0.278	0.232	0.0619	1.05
ESE	0.0928	0.139	0.371	0.278	0.186	0.0619	1.13
SE	0.0309	0.0774	0.155	0.387	0.0619	0	0.71
SSE	0.0309	0.0619	0.495	0.696	0.418	0.449	2.15
S	0.0928	0.108	0.402	1.08	0.804	0.201	2.69
SSW	0.139	0.232	0.913	1.90	1.36	0.108	4.66
SW	0.0464	0.232	0.495	1.53	0.572	0.0619	2.94
WSW	0.0464	0.108	0.449	1.44	0.480	0.0774	2.60
W	0	0.139	0.387	1.42	1.01	0.170	3.12
WNW	0.0464	0.139	0.371	0.727	1.07	0.0464	2.40
NW	0.0464	0.139	0.371	0.743	0.727	0.0309	2.06
NNW	0.0464	0.155	0.655	1.25	0.309	0.0619	2.49
All	0.80	2.55	7.02	12.96	7.89	1.41	32.63

[a] Frequency of occurrence is based on 6464 hr of good observation during 1963.
[b] Stability-based temperature difference measured at the 410-ft and 37-ft levels.
[c] Wind speed measured at the 355-ft level.

Table 11A.2-3

FREQUENCY OF OCCURRENCE, IN PERCENT, OF VARIOUS WIND SPEEDS AND DIRECTIONS OBSERVED AT THE BROOKHAVEN NATIONAL LABORATORY SITE UNDER NEUTRAL ATMOSPHERIC CONDITIONS[a,b]

Wind Direction	Wind Speed[c] (mph)						All Wind Speeds
	0-3	4-7	8-12	13-18	19-24	>24	
N	0.0619	0.217	0.526	0.325	0.170	0.0309	1.331
NNE	0.0928	0.340	0.495	0.774	0.464	0.0155	1.764
NE	0.0774	0.480	0.526	0.217	0.0155	0	1.316
ENE	0.0928	0.186	0.371	0.325	0.155	0.0155	1.145
E	0.0464	0.0928	0.263	0.155	0	0	0.557
ESE	0.139	0.449	0.743	0.278	0.0774	0.0155	1.702
SE	0.0309	0.217	0.464	0.124	0.0309	0.0155	0.882
SSE	0.0309	0.248	1.45	0.665	0.232	0.201	2.827
S	0.155	0.402	1.58	1.42	0.603	0.0774	4.237
SSW	0.186	0.172	1.67	0.93	0.774	0.0619	5.334
SW	0.139	0.294	0.712	0.804	0.433	0.139	2.521
WSW	0.124	0.263	0.882	1.18	0.990	0.325	3.764
W	0.155	0.248	0.789	1.39	1.73	0.975	5.287
WNW	0.124	0.248	0.851	1.01	1.30	0.619	4.152
NW	0.0464	0.294	0.511	0.851	0.866	0.263	2.831
NNW	0.0619	0.464	0.619	0.990	0.402	0.928	2.630
All	1.56	5.15	12.45	12.44	7.83	2.85	42.28

[a] Frequency of occurrence is based on 6464 hr of good observation during 1963.
[b] Stability-based temperature difference measured at the 410-ft and 37-ft levels.
[c] Wind speed measured at the 355-ft level.

Table 11A.2-4

FREQUENCY OF OCCURRENCE, IN PERCENT, OF VARIOUS WIND SPEEDS AND DIRECTIONS OBSERVED AT THE BROOKHAVEN NATIONAL LABORATORY SITE UNDER UNSTABLE ATMOSPHERIC CONDITIONS[a,b]

Wind Direction	Wind Speed[c] (mph)						All Wind Speeds
	0-3	4-7	8-12	13-18	19-24	>24	
N	0	0	0.0928	0.0464	0	0	0.1392
NNE	0	0	0.0464	0.0155	0	0	0.0619
NE	0	0	0.0619	0	0	0	0.0619
ENE	0	0	0	0	0.0619	0.0155	0.0774
E	0	0	0	0.0155	0.0155	0	0.031
ESE	0	0	0.0155	0	0	0	0.0155
SE	0	0	0	0	0	0	0
SSE	0	0	0.0155	0.0309	0	0	0.0464
S	0	0	0.0928	0.248	0.232	0.0464	0.6192
SSW	0	0	0.0155	0.217	0.0774	0	0.3099
SW	0	0	0.0619	0.0619	0	0.0155	0.1393
WSW	0	0	0.0309	0.155	0.170	0.0928	0.4487
W	0	0	0.0619	0.402	0.541	0186	0.1909
WNW	0	0.0309	0.0619	0.495	0.402	0.201	1.1908
NW	0	0	0.186	0.309	0.139	0.0155	0.6495
NNW	0	0.0155	0.0928	0.155	0.0619	0	0.3252
All	0.0	0.05	0.84	2.15	1.70	0.57	5.32

[a] Frequency of occurrence is based on 6464 hr of good observation during 1963.
[b] Stability-based temperature difference measured at the 410-ft and 37-ft levels.
[c] Wind speed measured at the 355-ft level.

Table 11A.2-5

FREQUENCY OF OCCURRENCE, IN PERCENT, OF ALL WIND SPEEDS AND DIRECTIONS OBSERVED AT THE BROOKHAVEN NATIONAL LABORATORY SITE UNDER ALL ATMOSPHERIC CONDITIONS[a,b]

Wind Direction	Wind Speed[c,d] (mph)				All Stability Conditions
	VS	MS	N	U	
N	1.05	1.33	1.32	0.14	3.85
NNE	0.48	1.56	1.76	0.06	3.86
NE	0.26	1.08	1.31	0.06	2.72
ENE	0.25	0.66	1.15	0.08	2.14
E	0.42	1.05	0.56	0.03	2.06
ESE	0.59	1.13	1.70	0.02	3.44
SE	0.15	0.71	0.88	0.00	1.74
SSE	0.53	2.15	2.83	0.05	5.56
S	1.11	2.69	4.24	0.62	8.66
SSW	3.25	4.66	5.33	0.31	13.55
SW	2.83	2.94	2.52	0.14	8.43
WSW	2.24	2.60	3.76	0.45	9.05
W	2.32	3.12	5.29	1.19	11.92
WNW	1.41	2.40	4.15	1.19	9.15
NW	1.44	2.06	2.83	0.65	6.98
NNW	1.44	2.49	2.63	0.33	6.89
All	19.77	32.63	42.28	5.32	100.00

[a] Frequency of occurrence is based on 6464 hr of good observation during 1963; 12 hr of calm (wind speed less than 0.5 mph).
[b] Stability-based temperature difference measured at the 410-ft and 37-ft levels.
[c] Wind speed measured at the 355-ft level.
[d] Key: VS = very stable; MS = moderately stable; N = neutral; U = unstable.

Table 11A.2-6

FREQUENCY OF OCCURRENCE, IN PERCENT, OF VARIOUS WIND SPEEDS AND DIRECTIONS OBSERVED AT THE BROOKHAVEN NATIONAL LABORATORY SITE UNDER ALL ATMOSPHERIC CONDITIONS[a,b]

Wind Direction	Wind Speed[c] (mph)						All Wind Speeds
	0-3	4-7	8-12	13-18	19-24	>24	
N	0.124	0.510	1.48	1.25	0.402	0.077	3.85
NNE	0.201	0.758	1.25	1.38	0.232	0.046	3.87
NE	0.186	0.758	1.30	0.371	0.108	0	2.72
ENE	0.124	0.418	0.634	0.557	0.371	0.031	2.13
E	0.124	0.356	0.681	0.526	0.303	0.062	2.06
ESE	0.340	0.789	1.39	0.572	0.263	0.077	3.43
SE	0.108	0.325	0.665	0.526	0.108	0.015	1.75
SSE	0.124	0.449	2.16	1.52	0.650	0.650	5.55
S	0.294	0.712	2.32	3.11	1.87	0.356	8.66
SSW	0.387	1.24	3.26	5.18	3.22	0.263	13.55
SW	0.294	0.696	1.70	3.62	1.90	0.216	8.43
WSW	0.263	0.495	1.70	3.98	2.20	0.510	9.05
W	0.201	0.572	2.04	3.93	3.82	1.36	11.93
WNW	0.247	0.603	1.86	2.66	2.91	0.866	9.14
NW	0.124	0.619	1.50	2.51	1.92	0.309	6.98
NNW	0.124	0.758	1.81	3.19	0.851	0.155	6.88
All	3.42	10.05	25.77	34.78	21.13	5.00	100.00

[a] Frequency of occurrence is based on 6464 hr of good observation during 1963.
[b] Stability-based temperature difference measured at the 410-ft and 37-ft levels.
[c] Wind speed measured at the 355-ft level.

Table 11A.2-7

1963MONTHLY AVERAGE GAMMA DOSE FOR MONITORING STATIONS AROUND THE BROOKHAVEN GRAPHITE RESEARCH REACTOR SITE[a,b]

(in mR/wk of AR-41)

	Onsite Stations			Perimeter Stations				Off-site Station
Month	E-10	E-11	E-12	E-2	E-4	E-7	E-9	0-6
January	1.46	2.08	2.59	0.45	0.26	0.28	0.76	0
February	0.06	2.22	2.92	0.06	0.11	0.76	0.76	0.03
March	0.68	2.58	2.25	0.58	0.05	0.57	0.57	0.03
April	0.78	1.94	2.59	0.14	0.19	1.08	0.74	0.01
May	0.44	6.55	5.19	0.43	0.24	0.41	1.86	0.01
June	0.85	2.31	2.43	0.82	0.32	0.57	0.74	0.02
July	0.35	2.56	4.30	0.47	0.25	0.42	1.49	0.03
August	0.64	3.18	5.02	0.17	0.01	0.48	1.02	0
September	1.63	3.07	3.83	0.21	0.70	0.27	0.55	0.03
October	1.51	2.68	3.46	0.41	0.57	0.53	0.80	0.02
November	0.90	2.16	3.40	0.31	0.39	0.45	0.58	0.04
December	0.58	1.60	1.17	0.19	0.25	0.39	0.35	0.04
Average	0.82	2.74	3.26	0.35	0.28	0.52	0.85	0.02
Peak weekly average	3.23	12.91	7.57	1.97	1.94	1.63	2.29	1.08

[a] From Brookhaven National Laboratory publication BNL 915.
[b] Estimated error at 90% confidence level, ⇧ 0.25 mR/wk.

Table 11A.2-8

1963 AVERAGE ANNUAL MEASURED AND CALCULATED GAMMA DOSE RATES FOR MONITORING STATIONS AROUND THE BROOKHAVEN GRAPHITE RESEARCH REACTOR SITE

Station	Direction Sector	Distance from Stack (m)	Dose (mR/yr)	
			Measured[a]	Calculated[b]
E-2	NW	1100	21[c]	20
E-4	WSW	2200	14	13
E-7	SE	2500	28	30
E-9	NE	2750	45	34
E-10	W	520	40	42
E-11	S	420	140	122
E-12	NNW	460	158	156

[a] Based on a 10-month average.
[b] Based on an 85% operation factor giving a release rate of 0.127 Ci/sec.
[c] Based on a 9-month average.

Table 11A.3-1

NOBLE RADIOGAS RELEASE RATE TO ENVIRONS
FROM 12-BED RECOMBINER/CHARCOAL SYSTEM[a]

Noble Radiogas	Release Rate (μCi/sec)
Kr-83m	2.16
Kr-85m	266.0
Kr-85	23.8
Kr-87	0.366
Kr-88	144.0
Xe-131m	6.46
Xe-133m	3.08
Xe-133	1190.0
All radiogases	~1640.0

[a] Based on 100,000 μCi/sec at 30-minute-decay diffusion mixture and a condenser inleakage of 18.5 scfm.

Table 11A.3-2

NOBLE RADIOGAS RELEASE RATE TO ENVIRONS
FROM GLAND SEAL SYSTEM[a]

Noble Radiogas	Release Rate (μCi/sec)
Kr-83m	3.44
Kr-85m	6.14
Kr-85	0.0238
Kr-87	19.4
Kr-88	20.0
Kr-89	87.0
Kr-90	29.6
Kr-91	0.0698
Xe-131m	0.0151
Xe-133m	0.278
Xe-133	8.18
Xe-135m	24.4
Xe-135	22.2
Xe-137	108.0
Xe-138	82.6
Xe-139	45.4
Xe-140	1.74
All radiogases	~464.0

[a] Based on 100,000 μCi/sec at 30-minute-decay diffusion mixture; 0.1% to gland seal system.

Table 11A.3-3

ANNUAL AVERAGE GAMMA DETECTOR DOSE AT GROUND LEVEL FROM RECOMBINER/CHARCOAL SYSTEM[a]
(Based on a continuous release rate of 1640 μCi/sec)

	Site Boundary		Dose (mrem/yr) as Function of Distance from Stack (km)				
Direction Sector	Distance From Stack	Dose (mrem/yr)	3.218	6.436	12.067	40.225	72.405
N	1176	0.967	0.420	0.255	0.089	0.008	0.001
NNE	1203	0.429	0.226	0.137	0.046	0.004	0.001
NE	695	0.434	0.191	0.112	0.037	0.003	0.001
ENE	642	0.325	0.164	0.097	0.033	0.003	0.001
E	535	0.474	0.270	0.154	0.050	0.004	0.001
ESE	455	0.607	0.342	0.193	0.063	0.006	0.001
SE	588	0.601	0.353	0.207	0.067	0.005	0.001
SSE	481	0.843	0.296	0.171	0.054	0.004	0.001
S	455	0.647	0.237	0.140	0.046	0.004	0.001
SSW	535	0.475	0.169	0.094	0.029	0.002	[b]
SW	668	0.361	0.138	0.079	0.025	0.002	[b]
WSW	749	0.317	0.158	0.091	0.028	0.002	[b]
W	668	0.343	0.136	0.081	0.027	0.002	[b]
WNW	722	0.657	0.270	0.156	0.049	0.004	0.001
NW	936	0.626	0.209	0.125	0.041	0.003	0.001
NNW	1257	0.687	0.247	0.150	0.051	0.004	0.001

[a] Thirty-minute-decay diffusion mixture of noble radiogases would be 100,00 μCi/sec.
[b] Negligible dose.

Table 11A.3-4

ANNUAL AVERAGE GAMMA DETECTOR DOSE AT GROUND LEVEL FROM GLAND SYSTEM[a]
(Based on a continuous release rate of 464 μCi/sec)

Direction Sector	Distance from Stack (m)	Dose (mrem/yr)
N	1176	0.425
NNE	1203	0.194
NE	695	0.218
ENE	642	0.161
E	535	0.240
ESE	455	0.305
SE	588	0.331
SSE	481	0.472
S	455	0.383
SSW	535	0.221
SW	668	0.159
WSW	749	0.145
W	668	0.178
WNW	722	0.306
NW	936	0.285
NNW	1257	0.290

[a] Thirty-minute-decay diffusion mixture of noble gases would be 100 μCi/sec.

Table 11A.3-5

ANNUAL AVERAGE GAMMA DETECTOR DOSE AT GROUND LEVEL FROM BUILDING VENTILATION SYSTEM[a]
(Based on a continuous release rate of 278 μCi/sec)

Direction Sector	Distance from Stack (m)	Dose (mrem/yr)
N	1176	0.881
NNE	1203	0.502
NE	695	0.764
ENE	642	0.518
E	535	0.624
ESE	455	0.772
SE	588	0.799
SSE	481	1.066
S	455	1.061
SSW	535	1.005
SW	668	0.900
WSW	749	0.783
W	668	0.842
WNW	722	0.818
NW	936	0.471
NNW	1257	0.545

[a] Five-minute-old mix, thirty-minute-decay diffusion mixture of noble gases would be 100 μCi/sec.

Table 11A.3-6

ANNUAL AVERAGE GAMMA DETECTOR DOSE VERSUS DISTANCE FOR MAXIMUM DIRECTION FROM COMBINED CONTRIBUTIONS OF GLAND SEAL AND AIR EJECTOR OFFGAS SYSTEMS[a]

Distance from Stack		Dose
km	miles	(mrem/yr)
1.2	0.73	1.39
3.2	2	0.537
6.4	4	0.308
12.1	7.5	0.104
40.2	25	0.009
72.4	45	0.002

[a] 2104 μCi/sec release from stack.

Table 11A.3-7

ANNUAL AVERAGE GAMMA DOSE FROM COMBINED CONTRIBUTIONS OF GLAND SEAL AND AIR EJECTOR OFFGAS SYSTEMS AT POINT ON SITE BOUNDARY FURTHEST FROM STACK FOR DIFFERENT LEVELS OF FUEL-DEFECT OPERATION

Annual Average 30-Min-Decay Diffusion Mixture (μCi/sec)	Release Rate (μCi/sec)		Total Fixed-Point or Detector Dose (mrem/yr)
	Recombiner/ Charcoal System	1.75-Min Gland Seal System	
25,000[a]	410	116	0.35
100,00[b]	1,640	464	1.39
718,400[c]	11,800	3,300	10.00

[a] Typical release rate corresponding to normal operation.
[b] Release rate that is expected only infrequently.
[c] Design-objective release rate.

Table 11A.3-8

AVERAGE WHOLE-BODY DOSES FOR POPULATION DISTRIBUTION FROM COMBINED CONTRIBUTIONS OF STEAM JET AIR EJECTOR AND GLAND SEAL SYSTEMS[a,b]

(Anuual Doses in man-rem; based on a continuous release rate of 2104 μCi/sec))

Distance	Distance from Stack (miles)							
Sector	SB-3[c]	3-5	5-10	10-20	20-30	30-40	40-50	SB-50[c]
N	1.6-2[d]	2.6-2	1.4-1	3.0-2	6.4-3	6.9-3	7.6-3	2.3-1
NNE	1.1-2	1.1-2	1.2-2	3.5-2	5.1-3	5.1-3	3.4-3	8.3-2
NE	9.0-3	8.7-3	9.6-3	2.7-2	3.9-3	4.0-3	2.7-3	6.5-2
ENE	8.1-3	9.2-3	2.0-2	3.8-2	8.9-3	3.4-3	2.1-3	8.9-2
E	2.2-2	1.8-2	2.5-2	5.9-2	1.4-2	5.4-3	3.4-3	1.5-1
ESE	2.2-2	3.7-2	1.2+0	9.3-1	2.2-2	3.7-2	4.7-3	2.2+0
SE	6.2-2	6.8-2	3.0+0	9.9-1	2.2-2	3.5-2	4.4-3	4.2+0
SSE	5.5-2	3.9-2	2.2+0	5.2-2	1.4-2	8.1-3	2.9-3	2.3+0
S	9.0-3	7.1-3	1.7-1	4.6-2	1.2-2	7.2-3	2.5-3	2.6-1
SSW	4.5-2	1.9-2	2.4-2	9.7-3	5.1-3	1.4-3	7.2-4	1.0-1
SW	5.0-3	3.7-3	6.9-3	8.4-3	4.3-3	1.2-3	6.0-4	3.0-2
WSW	4.7-3	1.1-2	7.2-3	1.6-2	2.6-3	1.0-3	1.2-3	4.5-2
W	7.9-3	3.5-2	1.1-2	1.7-2	2.8-3	1.1-3	1.4-3	7.6-2
WNW	1.2-2	1.2-2	1.3-2	2.0-2	6.6-3	1.7-2	3.5-2	1.1-1
NW	1.1-2	5.6-3	1.1-2	1.7-2	5.9-3	1.5-2	3.0-2	9.6-2
NNW	4.0-3	6.3-3	4.2-2	1.7-2	3.7-3	4.0-3	4.3-3	8.1-2
							Total:	10.2
							Natural background total:	139,000

[a] Whole-body doses are doses to individuals. A combined occupancy and shielding factor of 2 has been assumed.
[b] Population estimate for year 2010.
[c] SB = site boundary.
[d] 1.6-2 = 1.6×10^{-2}.

Table 11A.3-9

INFORMATION ON DAIRY FARMS SURROUNDING THE DAEC SITE IN 1971, BY DIRECTION SECTOR

Direction Sector	Distance from Stack (miles)	Azimuth (from true north)	Farm	Location	Number of Dairy Cows	Months on Pasture	Distributor	Location of Distributor
N	3.5	010°	August Holub	Section 27, Washington Township, Linn Co., Center Point, IA	20	May-October	Ryan Cooperative	Ryan, IA
NNE	5.0	029°	Melvin Cress	Section 13, Creek, Township, Linn Co., Center Point, IA	50	May-October	Mid-America Dairymen Inc.	Marion, IA
NE	6.7	048°	Bernice Haehlen	Section 17, Creek Township, Linn Co., Alburnett, IA	22	May-October	Maquoketa Valley Cooperative	Ryan, IA
ENE[a]								
E[a]								
ESE[a]								
SE[a]								
SSE[a]								

[a] No dairy farms in this sector within 10 miles of DAEC site.

Table 11A.3-9

INFORMATION ON DAIRY FARMS SURROUNDING THE DAEC SITE IN 1971, BY DIRECTION SECTOR

Direction Sector	Distance from Stack (miles)	Azimuth (from true north)	Farm	Location	Number of Dairy Cows	Months on Pasture	Distributor	Location of Distributor
S	6.0	185°	George Young	Section 9, Palo, IA	53	[b]	Mid-America Dairymen, Inc.	Marion, IA
SSW	2.0	216°	M. Van Note	Section 20, Palo, IA	40	May - October	Mid- America Dairymen, Inc.	Marion, IA
SW	3.4	219°	Ronald Beatty	Section 19, Palo, IA	40	May - October	Mid- America Dairymen, Inc.	Marion, IA
WSW [a]								
W	2.6	272°	Rose Myers	Section 7, Shellsburg, IA	5	April - November	Walker Creamery	Walker, IA
WNW	1.6	298°	W. W. Andrews	Section 5,	32	April - November	Ryan Cooperative	Ryan, IA
NW	7.5	317°	Raymond Lerch	Section 14, Vinton, IA	30	May - October	Wapsie Vally Cooperative	Maquoketa, IA
NNW	4.6	346°	Cliff Gott	Section 20, Center Point, IA	12	May - October	Maquoketa Valley Cooperative	Ryan, IA
W	7.2	260°	Donald Beatty	Section 17, Shellsburg, IA	75	4 months	Mid- America Dairymen, Inc.	Marion, IA

[b] Not applicable; feed local chopped alfalfa to confined animals.

[a] No dairy farms in this sector within 10 miles of DAEC site.

Table 11A.3-10

MONTHLY PRODUCTION OF INDIVIDUAL FARMS VERSUS MONTHLY PRODUCTION OF DISTRIBUTOR, JUNE 1971

Distributor		Farm	
Name	Production (lb)	Name	Production (lb)
Maquoketa Valley Cooperative	8,623,969	A. Holub	17,696[a]
		B. Haehlen	9,726[a]
		C. Gott	6,968[a]
		W. Andrews	28,360[a]
Walker Creamery	3,366	R. Myers	54[b]
Wapsie Valley Cooperative	8,882,187	R. Lerch	22,710[a]
Mid-America Dairymen, Inc.	13,008,275	M. Cress	16,312
		G. Young	29,867
		M. Van Note	53,221
		R. Beatty	40,237
		D. Beatty	46,716[c]

[a] Class B dairy milk production is not commercially distributed for direct consumption but rather is used for cheese, dried milk, etc.
[b] Product is butterfat.
[c] Product is distributed in Des Moines area. Monthly production of distributor is 14,453,485 lb.

Table 11A.3-11

GRAZING-SEASON INTEGRATED GROUND-LEVEL AIR CONCENTRATIONS FROM MAIN STACK RELEASE[a,b]

Direction Sector	Distance from Stack (m)	Concentration ($\mu Ci/cm^3$)
N	5,631	1.15×10^{-16}
NNE	8,040	5.08×10^{-17}
NE	10,780	3.23×10^{-17}
ENE[c]	- -	- -
E[c]	- -	- -
ESE[c]	- -	- -
SE[c]	- -	- -
SSE[c]	- -	- -
S	9,650	3.54×10^{-17}
SSW	3,220	1.13×10^{-16}
SW	5,470	4.28×10^{-17}
WSW[c]	- -	- -
W	4,180	7.66×10^{-17}
WNW	2,575	2.69×10^{-16}
NW	12,067	2.36×10^{-17}
NNW	7,401	5.16×10^{-17}

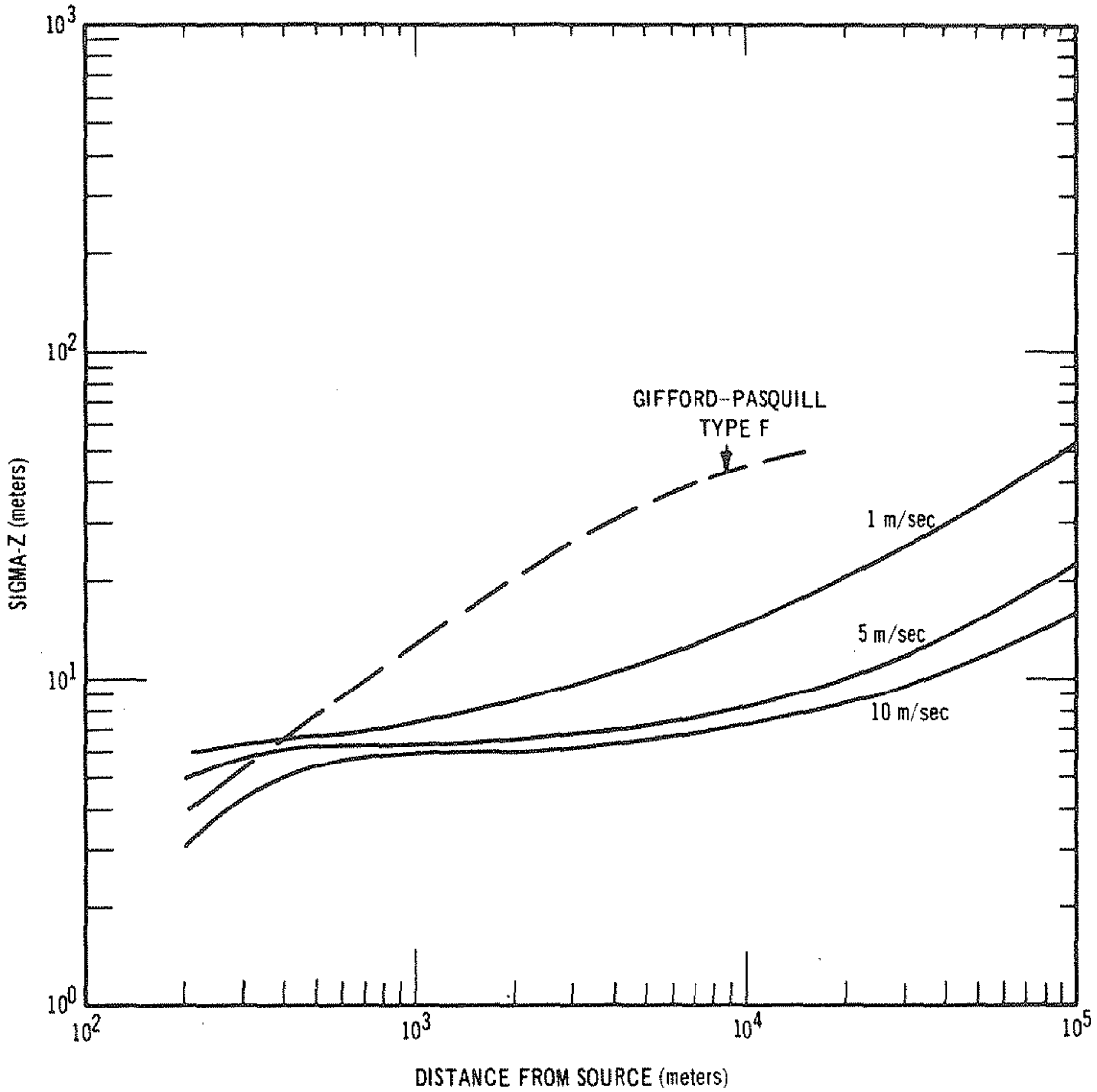
[a] Grazing season is April 15 through November 15.
[b] Assumed I-131 release rate of 10^{-2} µCi/sec.
[c] No dairy farms in this sector within 10 miles of DAEC site.

Table 11A.3-12

GRAZING-SEASON INTEGRATED-GROUND LEVEL AIR CONCENTRATIONS FROM BUILDING VENTILATION RELEASE[a,b]

Direction Sector	Distance from Stack (m)	Concentration ($\mu Ci/cm^3$)
N	5,631	1.28×10^{-16}
NNE	8,040	6.06×10^{-17}
NE	10,780	2.77×10^{-17}
ENE[c]	- -	- -
E[c]	- -	- -
ESE[c]	- -	- -
SE[c]	- -	- -
SSE[c]	- -	- -
S	7,650	1.06×10^{-17}
SSW	3,220	5.22×10^{-17}
SW	5,470	3.27×10^{-17}
WSW[c]	- -	- -
W	4,180	4.61×10^{-17}
WNW	2,575	3.24×10^{-16}
NW	12,067	1.36×10^{-17}
NNW	7,401	4.46×10^{-17}

[a] Grazing season is April 15 through November 15.
[b] Assumed I-131 release rate of 10^{-3} µCi/sec.
[c] No dairy farms in this sector within 10 miles of DAEC site.



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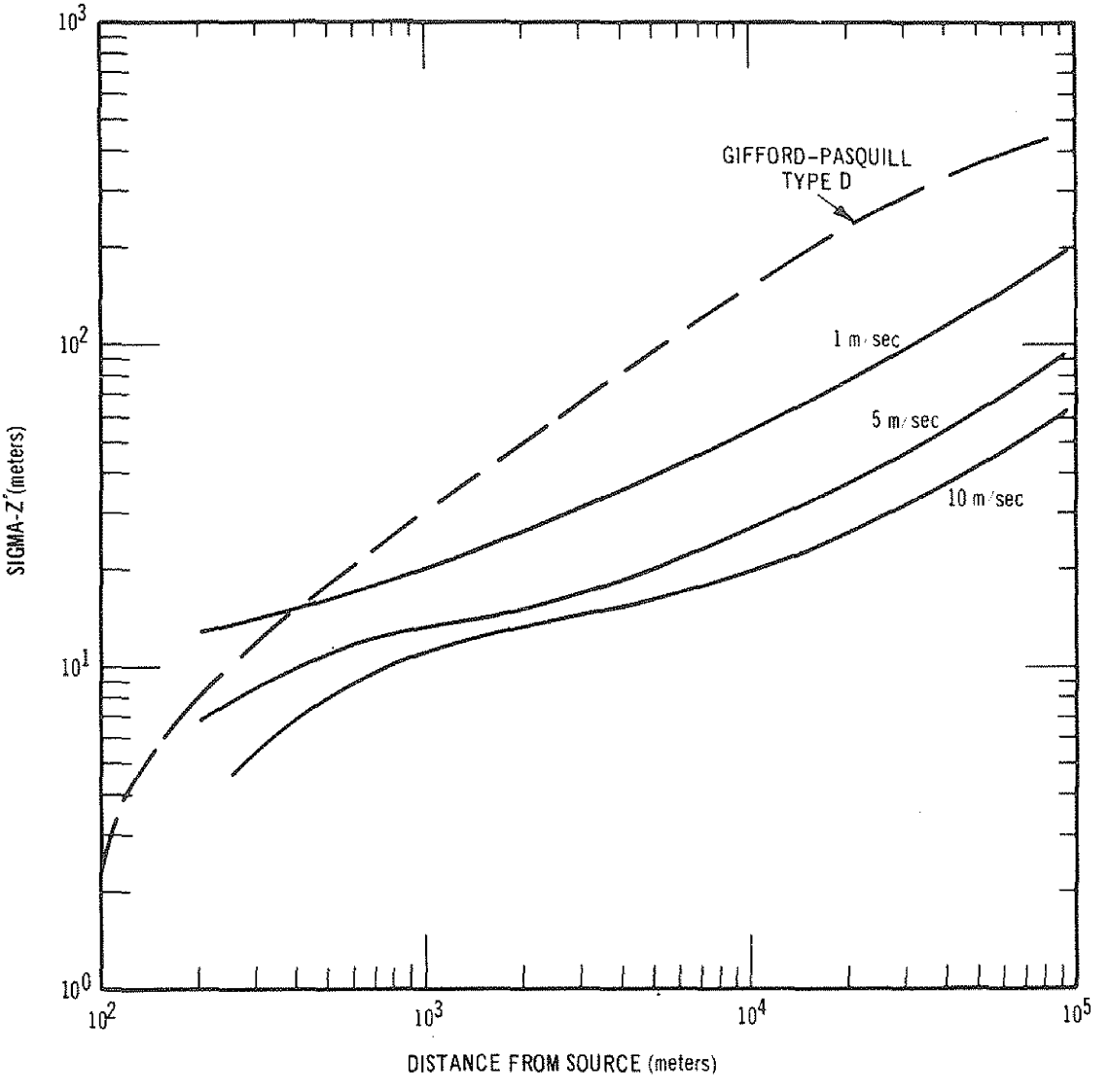
Vertical Cloud Width
versus Distance - Very Stable

Figure 11A.1-1



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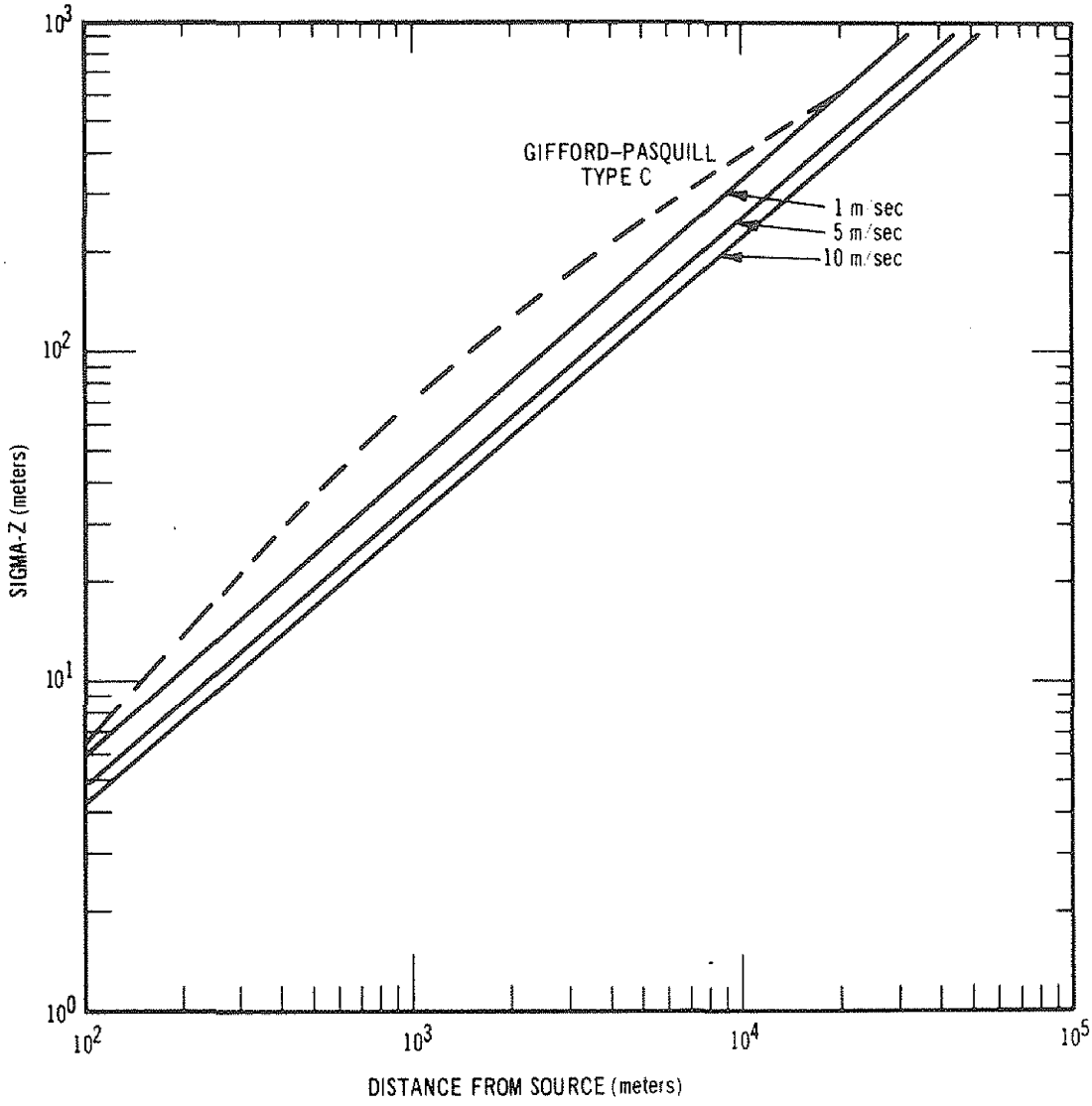
Vertical Cloud Width
versus Distance - Moderately Stable

Figure 11A.1-2



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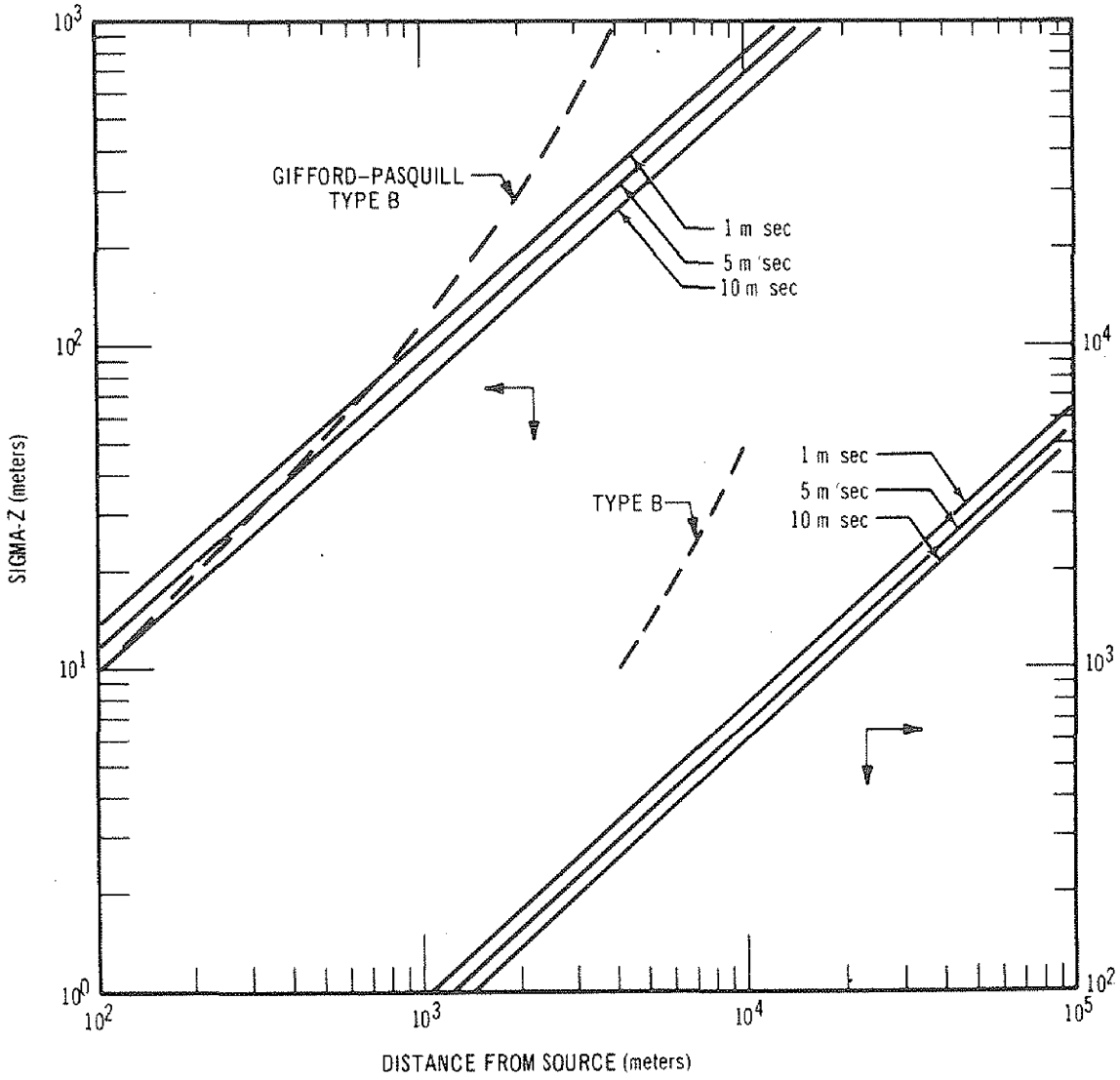
Vertical Cloud Width
versus Distance - Neutral

Figure 11A.1-3



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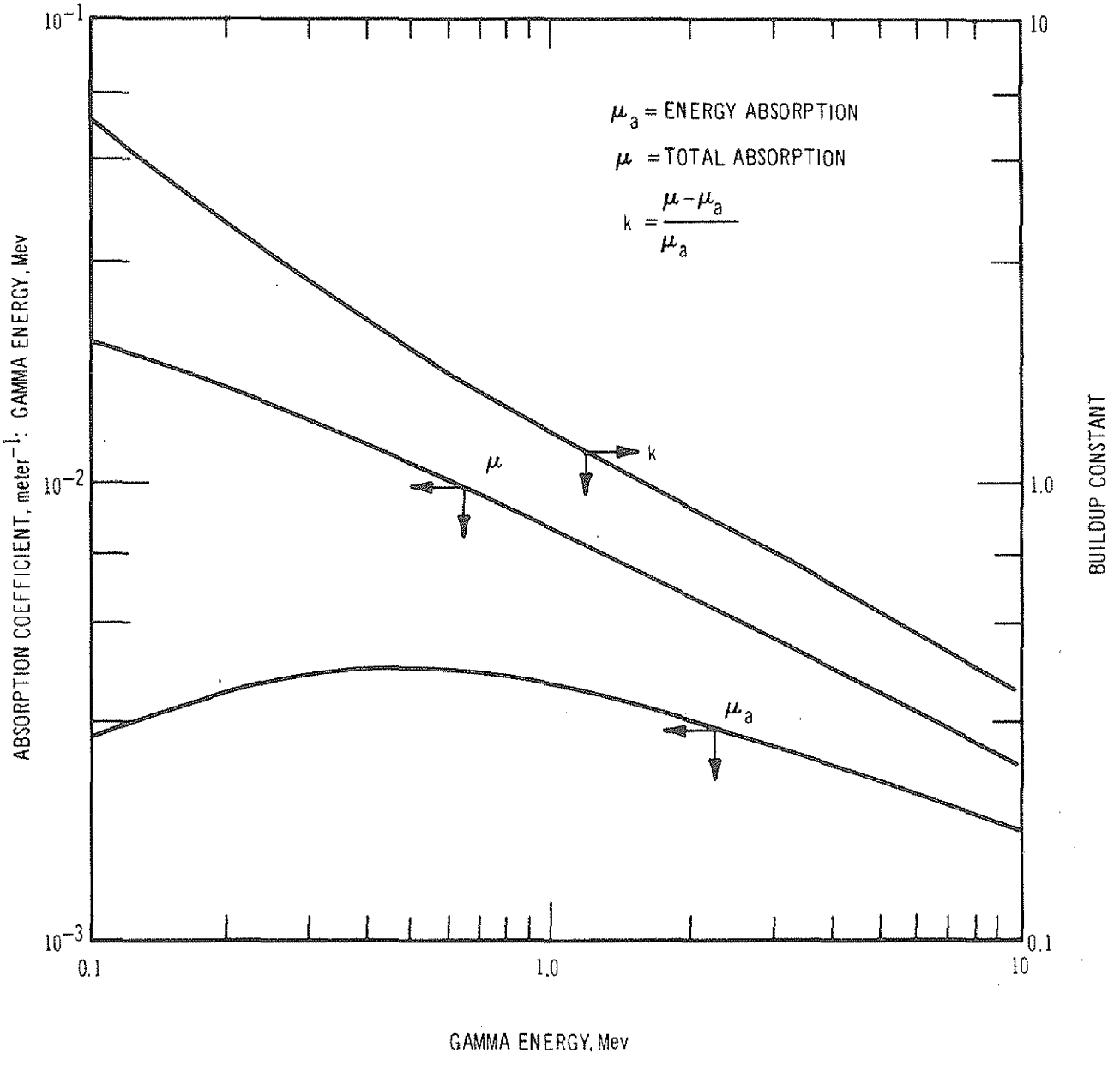
Vertical Cloud Width
versus Distance - Unstable

Figure 11A.1-4



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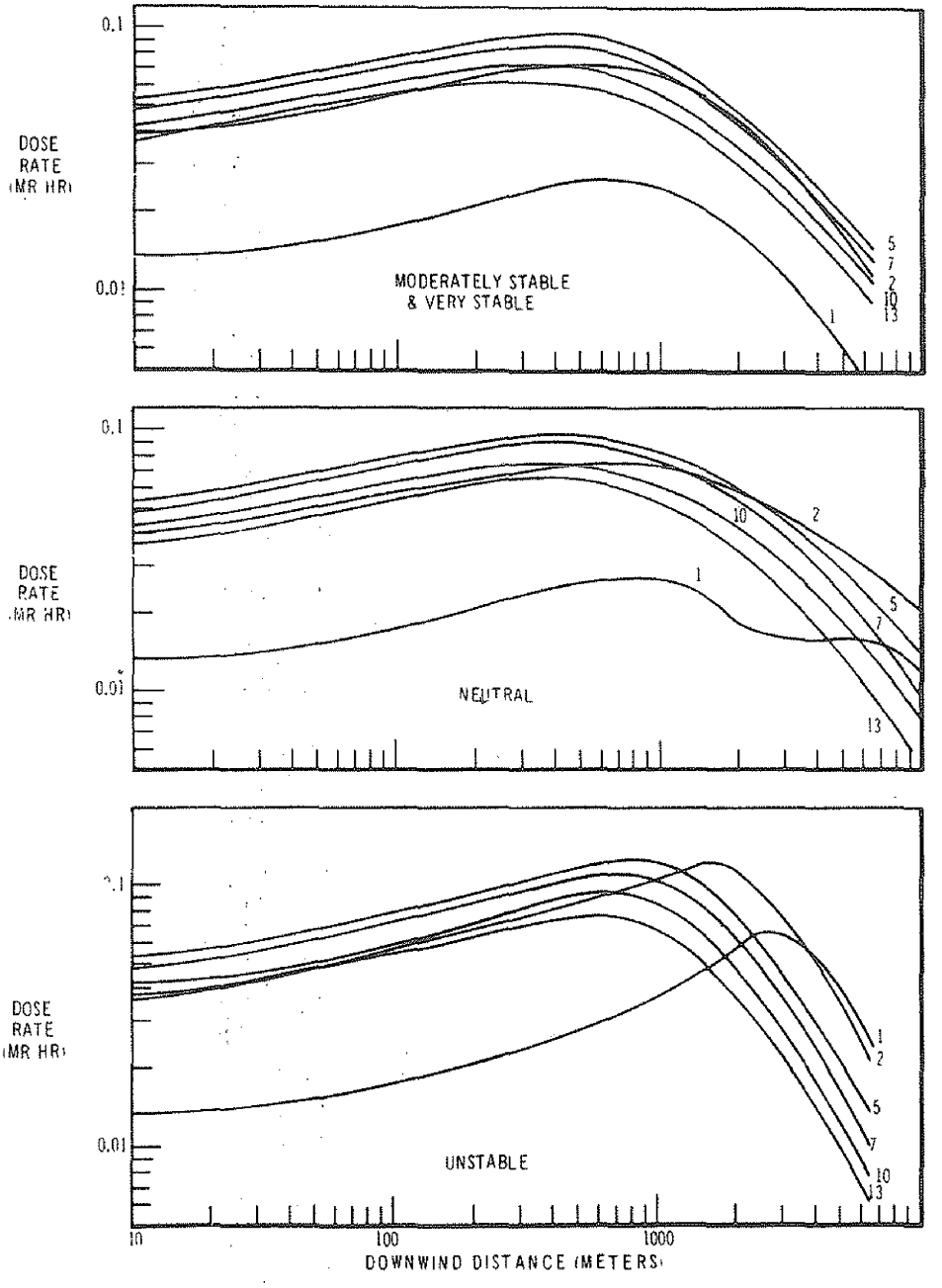
Gamma Radiation Absorption Coefficients and Buildup Constants for Air, STP

Figure 11A.1-5



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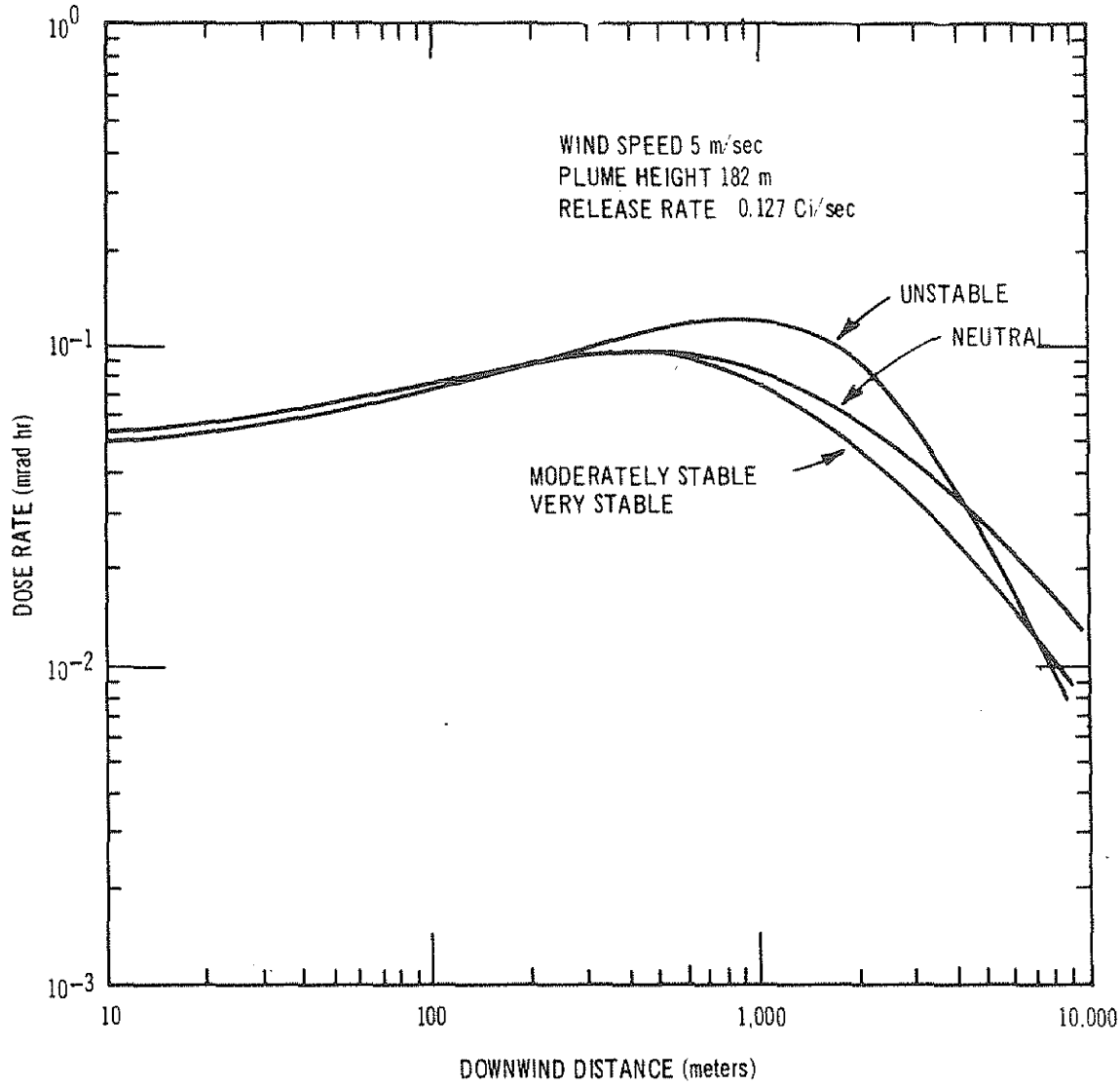
Gamma Dose Rate for Various Wind Speeds and Stabilities for BGRR Stack (Release Rate 0.127 Ci/sec)

Figure 11A.2-1



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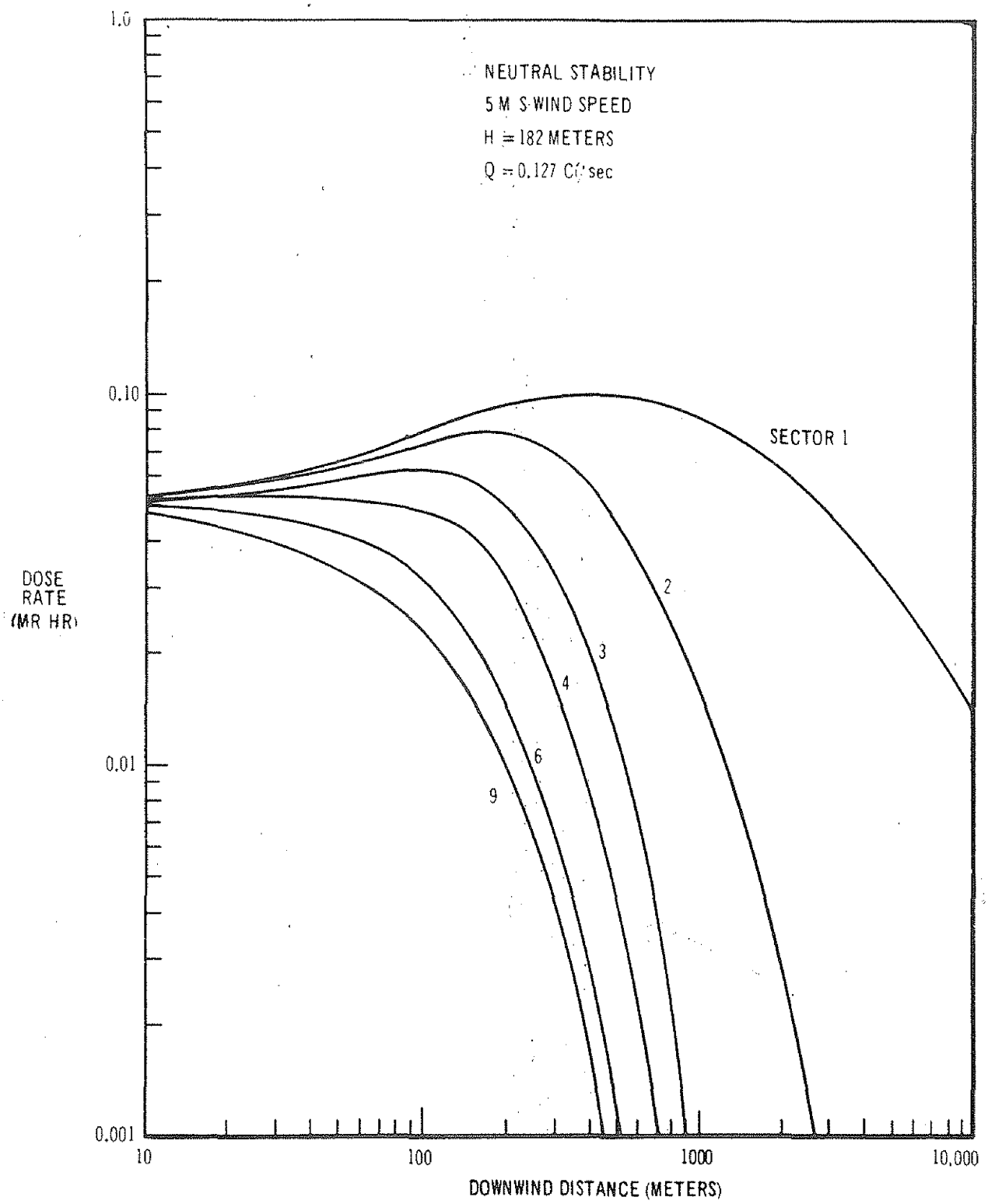
Gamma Dose Rate in Air for Various Stability Conditions

Figure 11A.2-2



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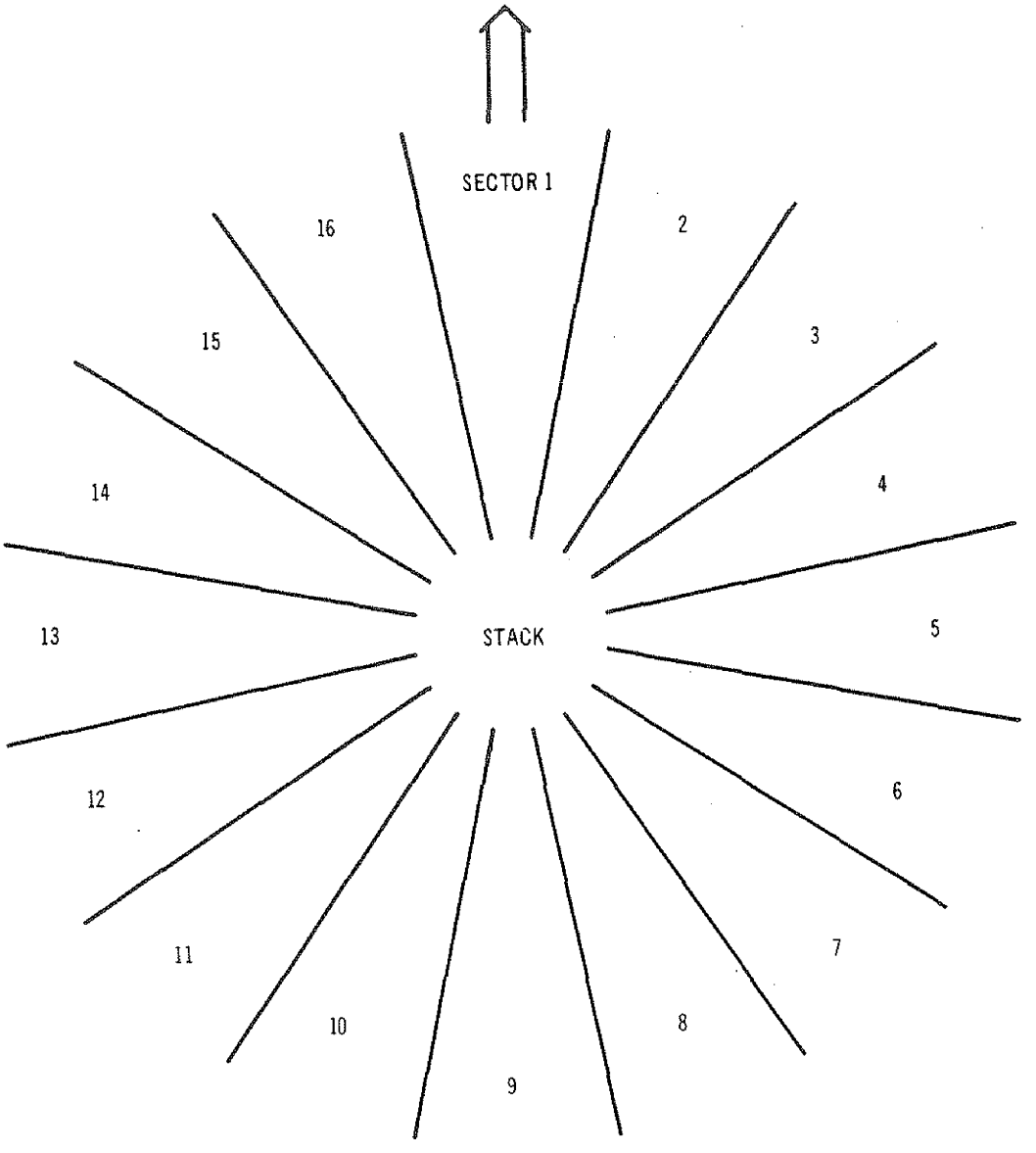
Dose Rate in Each Sector

Figure 11A.2-3



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Nomenclature of Sector Used for Averaging
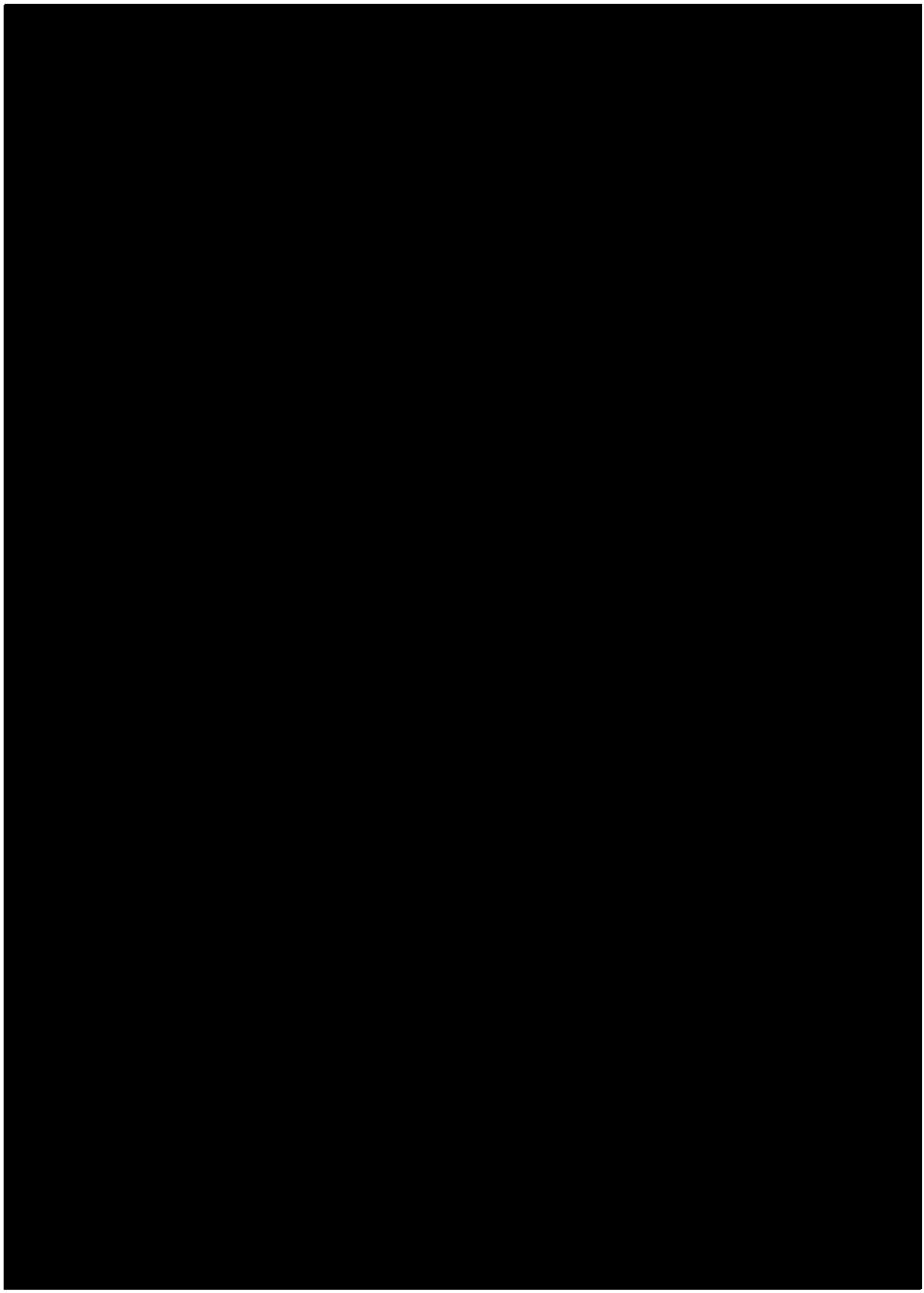
Figure 11A.2-4