Millstone Power Station Unit 2
Safety Analysis Report

Chapter 11

CHAPTER 11–RADIOACTIVE WASTE PROCESSING AND RADIATION PROTECTION SYSTEMS

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CHAPTER 11 – RADIOACTIVE WASTE PROCESSING AND RADIATION PROTECTION SYSTEMS

11.1 RADIOACTIVE WASTE PROCESSING SYSTEMS

11.1.1 GENERAL

The purpose of the radioactive waste processing systems is to provide controlled handling of all radioactive waste, proper discharging of radioactivity in liquid and gaseous effluents, and proper packaging for shipment of solid waste containing radioactivity from Millstone Unit 2.

On occasion, the Unit will generate liquid radioactive waste that cannot practicably be processed in the liquid radwaste system. The station may process this waste outside the Unit in compliance with state and federal regulations, and in accordance with the Radiological Effluent Control Program outlined in the Administrative Section of the Technical Specifications (e.g., Unit 1 evaporator or shipped off site for processing).

11.1.2 DESIGN BASES

11.1.2.1 Functional Requirements

These systems are required to ensure that the general public and plant personnel are protected against exposure to radioactive material in accordance with the regulations of 10 CFR Part 20, and the recommended guidelines of 10 CFR Part 50, Appendix I. Under normal plant operating conditions, the radioactive waste processing systems are required to limit radionuclide release concentrations to unrestricted areas to less than the maximum permissible concentrations as specified in 10 CFR Part 20, Section 1302, and Appendix B.

Interim onsite storage facilities accept waste from Millstone Units 1, 2 and 3. Information regarding facility design criteria is presented in Section 11.4 of the Millstone Unit 3 Final Safety Analysis Report.

11.1.2.2 Design Criteria

The radioactive waste processing system is designed in accordance with the following criteria:

a. To protect the general public and plant operating personnel against radiation from materials in accordance with 10 CFR Part 20 and 10 CFR Part 50, Appendix I.

b. To limit the levels of radioactivity of the effluents in unrestricted areas to as low as reasonably achievable (ALARA).

c. To provide suitable control of the release of radioactive materials in gaseous and liquid effluents during normal plant operation including anticipated operational occurrences.

d. To provide sufficient holdup capacity to retain gaseous and liquid wastes containing radioactive materials.

e. To ensure adequate safety under normal and postulated accident conditions.

f. To provide the capability to permit inspection and testing of appropriate components.

g. To provide suitable shielding for radiation protection.

h. To provide appropriate containment and confinement of radioactive materials.

i. To provide appropriate monitoring capability to detect excessive radiation levels and to monitor effluent discharge paths.

j. To provide suitable processing of liquid and gaseous radwaste generated in accordance with the following operating and design criteria:

    1. Normal operation with expected primary and secondary side activities calculated in accordance with NUREG-0017 Rev. 1 (Reference 11.1-1) using the PWR-GALE code.

    2. Normal operation with design basis reactor coolant activities based on 1% failed fuel in accordance with Technical Specifications.

The only portions of the radioactive waste processing system designated as Seismic Category I are the containment penetration piping and isolation valves. However, the high pressure (150 psig) portions of the gaseous waste system, consisting of the compressors, decay tanks, interconnecting piping and valves (see Figure 11.1–5), have been designed and analyzed for Seismic Category I requirements as given in Section 5.8.

An analysis had been performed, at the time of initial plant licensing, to determine the site boundary doses due to simultaneous failure of the entire radioactive waste processing system, excluding those portions of gaseous waste processing system that were analyzed to Seismic Category I requirements. That analysis was predicated on the original design and operation of the radioactive waste processing system. More recently, this chapter was updated to reflect changes to the radioactive waste processing system design and operation. The results of this update are bounded by those of the original analysis discussed in this subsection.

Failure the of the radioactive waste processing system may release gaseous and liquid wastes into the auxiliary building. The auxiliary building, a Seismic Category I structure, is designed to contain all liquids within the building.

Therefore, the total inventory of radioactivity within the gases contained in the radioactive waste processing system, excluding high pressure portion of gaseous waste system, are assumed to be released.

The total activity is conservatively assumed to be released to the auxiliary building at an instant followed by puff release at ground level to the turbulent wake downwind of plant structures. Under these conditions, the X/Q value of 9.6×10^{-5} sec/m^3 is applicable. The doses at 625 meters were calculated by the method of Safety Guides 3 and 4. Based on the results of the original plant licensing analysis, the site boundary doses due to simultaneous failure of entire radioactive waste processing system, excluding portions of gaseous waste system, are as follows:

Normal Operation Based on 0.1% Failed Fuel

Thyroid dose (rem) - 3.9×10^{-4}

Whole body dose (rem) - 5.04×10^{-2}

Normal Operation Based on 1.0% Failed Fuel

Thyroid dose (rem) - 4.35×10^{-3}

Whole body dose (rem) - 5.78×10^{-1}

Since the doses are approximately equal to or less than the limits given in 10 CFR Part 20, Sections 105 and 106 and Appendix B (version prior to January 1, 1994), a Seismic Noncategory 1 waste processing system is acceptable.

11.1.2.3 System Components

Descriptions of the radioactive waste processing system components are given in Table 11.1-1.

11.1.3 LIQUID WASTE PROCESSING SYSTEM

11.1.3.1 System Descriptions

Clean Liquid Waste Processing System:

Clean liquid waste is normally tritiated, non-aerated, low conductivity liquid waste consisting primarily of reactor coolant letdown and liquid waste collected from equipment leaks and drains and certain valve and pump seal leaks. The clean liquid radioactive waste processing system is shown schematically in Figures 11.1–1 and 11.1–2. The design of the clean liquid radioactive waste processing system is based upon processing of radioactive liquids postulated to be released from the reactor coolant system (RCS) during normal reactor operation with design basis reactor coolant activities. Operating experience of nuclear power plants indicates that Millstone Unit 2 can expect to continue operating with a percentage of fuel failure much less than the postulated design basis of one percent. Nevertheless, the performance of the clean liquid waste processing system during normal operation is based on both expected and design basis primary side activities. Discussed in Appendix 11.A are the methodologies used to determine the expected and design basis radionuclide activity concentrations in the reactor coolant.

The clean liquid waste processing system is designed to support the processing of 14 RCS volumes per year of reactor coolant waste. However, the quantity of wastes to be generated and

processed annually by this system is approximately 1,200,000 gallons, based on the assumptions used in the 10 CFR 50 Appendix I analysis. The liquid waste input streams are shown in Figure 11.B–1.

The clean liquid waste processing system is designed for the processing of reactor coolant wastes concurrently with the letdown flow from the chemical and volume control system (CVCS). This mode of operation, at the maximum clean liquid waste processing flow of 132 gpm, sets the maximum system flow rate.

Reactor coolant is diverted to the clean liquid radioactive waste processing system from the CVCS when changes in RCS inventory or boron concentration are necessitated by startups, shutdowns, fuel depletion, etc. Reactor coolant at a rate of 44 gpm to 132 gpm is let down from the CVCS through a filter to reduce insoluble particulate, after which it flows to the clean liquid radioactive waste processing system for further processing.

Sources of clean liquid waste in the containment are collected in the primary drain tank. A heat exchanger and pump are provided for cooling the primary drain tank content as well as the content of the pressurizer quench tank. The contents of these tanks are maintained or cooled to 120°F to minimize the carryover of radioactive moisture to the gaseous waste processing system due to tank venting.

Equipment drains, valve leak-offs, and relief valve discharges from components that are located in the auxiliary building and that contain liquids with dissolved fission gases are collected in the equipment drain sump tank via the closed drain system, as shown in Figure 11.1–3. This design minimizes the uncontrolled release of gaseous radioactivity to the atmosphere.

The liquid contents of both the primary drain tank and the equipment drain sump tank are pumped via the demineralizers to the coolant waste receiver tank, bypassing the degasifier. The demineralizers and degasifier are discussed below:

1. Degasifier

   Reactor coolant degassing is accomplished by diverting the letdown flow in the CVCS to the degasifier in the clean liquid waste system. Interconnecting piping and valves are provided for degassing the reactor coolant prior to cold shutdown of the reactor for refueling operations. The degasifier is placed in service, prior to cold shutdown, to remove hydrogen, fission product gases, and other dissolved gases from the reactor coolant system liquid and to discharge these gases to the waste gas surge tank in the gaseous waste processing system. (See Section 11.1.4) The degassed liquid is pumped through the degasifier effluent cooler, which is used to lower the temperature of this liquid to 120°F before it is passed through one of two primary demineralizers. This is done to protect liquid radwaste system demineralizer resins against the damage caused by high liquid temperatures. The degassed reactor coolant is returned to either the volume control tank inlet or to the clean liquid radwaste system primary demineralizers for processing and discharge to the environment.

The packed columnar type degasifier employs internally generated stripping steam for removal of dissolved gases. A nitrogen cover gas, slightly above atmospheric pressure, is maintained within the degasifier during the shutdown mode to prevent air in-leakage and the formation of a potentially explosive hydrogen/oxygen mixture.

The degasifier system is provided with cascading steam controls to minimize the adverse effects of varying feed flow rates and temperatures on the dissolved gas removal performance of the unit. The degasifier operates at a pressure of 5 psig and a temperature of 228°F, and its pumps (both operating at full capacity) are controlled automatically by the degasifier level controls.

2. Demineralizers

The two (2) primary and two (2) secondary demineralizers are of the mixed-bed non-regenerating type, the designs of which are based on an expected operating ion exchange capacity of 12,000 grains of $CaCO_3$. The resin beds for mixed-bed non-regenerating demineralizers are mixtures of cation and anion resins in the H-OH forms. These demineralizers are bypassed automatically upon detection of liquid waste temperatures above 135°F. An additional secondary demineralizer provides the capability to further polish the waste stream, with the capability of placing the two (2) secondary demineralizers in series operation.

The demineralizers of the mixed-bed nonregenerable type are sized for one-year operation between resin replacement. Resin replacement is accomplished remotely by N_2 pressure and/or water sluicing to the spent resin tank. All valves required for resin removal are located either outside the demineralizer compartments if non-radioactive, or inside shielded vaults with extension operators to provide protection of plant personnel if radioactive. The demineralizer effluent flows to one of two coolant waste receiver tanks.

The two receiver tanks provide storage for approximately two RCS volumes (120,000 gal. nominal) of liquid wastes. A nitrogen gas blanket in each of the tanks is automatically maintained above atmospheric pressure to prevent air in-leakage. The nitrogen cover gas is vented to the gaseous waste processing system or displaced into the other receiver tank or monitor tanks as liquid fills the first receiver tank. No flashing occurs in the receiver tanks, and any transfer of hydrogen or fission gases from the liquid to the cover gas occurs via the slow process of molecular diffusion.

The content of the coolant waste receiver tank is sampled prior to processing. The content of the coolant waste receiver tank is then pumped through the secondary demineralizer(s) to one of two coolant waste monitor tanks, which offers a final check on the liquid waste to be released. The total storage capacity of the monitor tanks is approximately one RCS volume (60,000 gal. nominal).

All liquid to be released will be sampled to ensure that the limits set forth in 10 CFR Part 20, Sections 1301 and 1302, and Appendix B are not exceeded. If the activity level is unacceptable after sampling the monitor tank contents, then the liquid is reprocessed through the demineralizers. If the activity level is within discharge limits, the liquid is pumped through a final filter, at a rate selectable over a range of 10 gpm to 132 gpm, to the circulating water system where it is diluted with the water in the discharge conduit. The proper rate of liquid release to the discharge conduit is determined by sampling the liquid to be released. The concentrations of the limiting isotopes for release are determined. Based on the circulating water flow rate, the discharge rate is selected such that the releases to unrestricted areas are within permissible concentrations as specified in 10 CFR Part 20, Appendix B. The second isolation valve in the waste discharge header to the circulating water system is provided with a panel mounted manual controller for setting the desired flow rate. The clean liquid waste processing system effluent enters the circulating water system discharge box that runs along the south wall of the auxiliary building, as shown in Figure 1.2–10. The routing of the discharge structure to the quarry is shown on the plot plan, Figure 1.2–2. The circulating water is further diluted by the Long Island Sound tidal flow.

The only liquid discharge path from the clean liquid waste processing system to the environment is through the discharge header, which contains a radiation monitor. The radiation monitor and redundant isolation valves are installed between the waste processing system and the circulating water system. This radiation monitor annunciates in the control room on high radioactivity level and instrument failure, and will automatically close the isolation valves to prevent further discharge. The single radiation monitor serves as a backup system to sampling of the liquid to be released. All liquid to be released is sampled to confirm that the regulations of 10 CFR Part 20, Sections 1301 and 1302, and Appendix B and 10 CFR Part 50, Appendix I are met. The radiation monitor is described in Section 7.5.6.

To prevent the addition of waste to a monitor tank, the content of which is being discharged to the circulating water system, a valve interlock system is provided. The pneumatic valves, located in the discharge piping from each tank, can be opened and kept open only if the corresponding valve in the inlet piping to each tank is closed. This arrangement prohibits the addition of wastes to a previously sampled tank whose content is being released.

Aerated Liquid Waste Processing System:

The design of the aerated liquid waste processing system, shown in Figure 11.1–4, is based upon the processing of radioactive liquids (other than those handled by the clean liquid waste processing system) postulated to be generated annually. However, the expected quantity of wastes to be generated and processed annually by this system is approximately 600,000 gallons, based on the assumptions used in the 10 CFR 50 Appendix I analysis.

The aerated liquid waste system design is based upon the processing of wastes with a radioactivity level resulting from normal reactor operation with design basis reactor coolant activity concentrations. The performance of the aerated liquid waste processing system during normal operation is evaluated based upon processing of wastes with radioactivity levels resulting

from expected reactor coolant source terms as well as from design basis reactor coolant source terms.

Aerated liquid wastes are collected by the open drains system (shown in Figure 11.1–3) that empties into one of two drain tanks. The system is designed for batch processing. The contents of the drain tanks are monitored by sampling and processed through a system that contains a filter, one hard-piped demineralizer, and three portable demineralizers.

The demineralizer effluent is collected in the aerated waste monitor tank, which offers a final check on the liquid to be released. If the radioactivity level is within discharge limits, as determined by sampling, the liquid is then pumped, at a rate selected over a range of 10 gpm to 100 gpm, to the circulating water system, where it is diluted with water in the discharge conduit. The second isolation valve in the waste discharge header to the circulating water system is provided with a panel mounted manual controller for setting the desired flow rate. The liquid released from the aerated liquid radioactive waste processing system is monitored continuously for radioactivity. The radiation monitor annunciates in the control room on high radioactivity and instrument failure, and will automatically close two isolation valves in the discharge piping to prevent further discharge. The radiation monitor is discussed in Section 7.5.6. Provisions are furnished for recycling the monitor tank's content for further processing.

The only liquid discharge path from the aerated liquid waste processing system to the environment is through the discharge pipe containing the radiation monitor. All system leakage, drains, and relief valve flows are collected in the drains system and returned to the aerated liquid waste processing system.

11.1.3.2 System Operation

Clean Liquid Waste Processing Systems:

1. Normal Operation

Letdown from the RCS is automatically diverted to the radioactive waste processing system on detection of a high volume control tank liquid level. The flow of wastes into the radioactive waste processing system, normally at a rate of 44 gpm, is intermittent due to changes in RCS inventory or boron concentration necessitated by startups, shutdowns, fuel depletion, etc.

All other liquid wastes input into the clean liquid waste processing system are intermittent due to plant operation. Liquid wastes are collected in either the equipment drain sump tank (EDST) or the primary drain tank. On high tank level, the equipment drain sump tank contents are pumped to the coolant waste receiver tank. During reactor shutdown for refueling, a greater quantity of waste is anticipated due to draining of system equipment for maintenance or inspection. The operation of the primary drain tank system, for collection of liquid wastes in the containment, is on a batch-type basis. On primary drain tank high tank liquid level, an alarm is annunciated in the main control room to alert the plant operator. The operation of the primary drain tank pumps and opening of the

containment isolation valves are initiated manually from the control room. The initiation of the operation from the main control room is required to maintain the tank normal operating temperature of 120°F.

The processing of EDST and letdown liquid wastes up to the coolant waste receiver tanks is on a fully automatic, unattended basis. Processing up to the coolant waste receiver tanks consists of filtration, degasification (when needed), and ion exchange. Degassing the RCS prior to shutdown facilitates subsequent opening of the system.

The processing of wastes downstream of the coolant waste receiver tanks occurs on a batch-type basis. The receiver tank content is sampled to determine the amount of further processing required. A remotely actuated recirculation system is provided for each tank for the purpose of taking a representative sample. Normal processing consists of demineralization.

The processed wastes are collected in the monitor tanks, which offer a final check on the liquid to be released. Sampling is employed to determine the required discharge rate to ensure releases are within 10 CFR Part 20, Sections 1301 and 1302 and Appendix B and 10 CFR Part 50, Appendix I limits.

2. Abnormal Operation

Abnormal operation of the clean liquid waste processing system may involve reactor operation with greater than expected failed fuel and larger than anticipated volume of liquid waste generated, such as for cold-shutdowns and startups late in the core cycle. Since the clean liquid radioactive waste processing system is designed for reactor operation with design basis reactor coolant activities for processing 14 RCS volumes of waste per year and with sufficient storage capacity, system operation for these abnormal occurrences will be the same as for normal operation described above.

Aerated Liquid Waste Processing Systems:

1. Normal Operation

The aerated liquid waste processing system is operated on a batch type basis. System processing will vary with the volume of waste generated. The typical processing cycle is expected to be 5 to 7 days.

Normal processing consists of filtration and demineralization. The aerated waste monitor tank provides a final check on the liquid to be released to ensure compliance with 10 CFR Part 20, Sections 1301 and 1302 and Appendix B, and 10 CFR 50 Appendix I limits. All potential releases are sampled prior to discharge to the environment.

2. Abnormal Operation

Abnormal operating occurrences considered are larger than expected volumes of wastes, such as for steam generator blowdown processing. If the waste generation rate exceeds the capacity of the system, interconnecting piping is provided for pumping the wastes to the clean liquid radioactive waste processing system for treatment.

11.1.4 GASEOUS WASTE PROCESSING SYSTEM

11.1.4.1 System Descriptions

The gaseous waste processing system, shown in Figure 11.1–5, processes potentially radioactive hydrogenated waste gases. The system design is based upon the processing of radioactive gases postulated to be released from the RCS during normal operation with design basis reactor coolant activities. The gaseous waste processing system is shown as one of the streams in Figure 11.B–2.

Waste gases flow to the waste gas header and are collected in the waste gas surge tank. When the surge tank pressure increases to approximately 3 psig, one of the two 25 scfm compressors is automatically started by pressure instrumentation located on the tank. The surge tank gases are compressed into one of the six waste gas decay tanks, where gases are stored at a maximum pressure of 150 psig.

The storage capacity of each waste gas decay tank is based on the storage of waste gases expected to enter the system during any two months of normal operation. For all the waste gases entering the system, including those contributed by operational occurrences, the six decay tanks provide adequate storage capacity for a decay time of 90 days. Gases are held in the decay tanks until the radioactivity level has been reduced by decay and the gases are suitable for release. Prior to release, the gases in the decay tanks are sampled to determine compliance with the regulations of 10 CFR Part 20, Sections 1301 and 1302 and Appendix B, and 10 CFR Part 50, Appendix I.

The decay tanks discharge through an absolute (i.e., HEPA) filter to the Millstone stack. The discharge pipe contains a radiation monitor and redundant automatic isolation valves. A radiation monitor annunciates in the control room on high radioactivity level and instrument failure and will automatically close the isolation valves to prevent any further release.

The release rate for waste gases is selectable over a range of 10 scfm to 50 scfm. The rate is determined by sampling the content of the decay tank to be discharged. The radionuclide concentrations are correlated with the potential site boundary dose and the 10 CFR Part 20, Appendix B Effluent Concentrations, and the release rate is determined.

The radiation monitor in the waste gas discharge header is provided as a backup system to the sampling of the decay tank contents. The radiation monitor is described in FSAR Section 7.5.6.

The pneumatic valve in the inlet to each decay tank is interlocked with the corresponding valve in the tank discharge piping. The discharge valve can be opened and remains open only when the

inlet valve is closed. This feature precludes the inadvertent release of waste gases that are not sampled.

11.1.4.2 System Operation

1. Normal Operation

Waste gases from the sources shown in Figure 11.B–2 are collected in the waste gas surge tank through the waste gas header. As the pressure in the surge tank increases to 3 psig, one compressor is automatically started by pressure instrumentation mounted on the surge tank. If the surge tank pressure continues to increase, the second compressor is automatically started at 5 psig. The compressor discharges the waste gases into one of the six decay tanks selected by the operator. When the decay tank being filled reaches approximately 140 psig pressure, an alarm is annunciated in the control room to alert the operating personnel. The inlet valve to the decay tank is closed remotely by the operator, and one of the five remaining decay tanks is selected to receive the gases.

After an appropriate storage period, and after sampling has confirmed that 10 CFR Part 20, Sections 1301 and 1302 and Appendix B, and 10 CFR 50 Appendix I limits are met, the gases are released on a controlled batch basis to the Millstone stack.

2. Abnormal Operation

The gaseous waste processing system, under abnormal conditions (e.g., unexpectedly large volumes of gases) will function in a fashion similar to that for normal conditions. Suitable storage capability is available to store the cover gas from the coolant waste receiver and monitor tanks resulting from back-to-back cold shutdowns and startups late in core life.

11.1.4.3 Gaseous Release Radiological Consequences

Annual design basis and expected releases of radioactive gaseous waste are presented in Table 11.A-9. Gaseous release radiological consequences are presented in Appendix 11.C.

11.1.4.4 Waste Gas System Failure

11.1.4.4.1 General

The limiting accident considered is the postulated and uncontrolled release to the auxiliary building of the radioactive xenon and krypton gases stored in one waste gas decay tank. The credibility of such an occurrence is low since the waste gas system is not subjected to pressures greater than 150 psig, or large stresses. The result of a rupture of a gas decay tank is analyzed in order that the maximum hazard, which would result from a malfunction in the radioactive waste process system, will be defined.

11.1.4.4.2 Method of Analysis

It is assumed that the tank contains the gaseous activity evolved from degassing one system volume of reactor coolant for refueling. The maximum activity would exist prior to cold shutdown at the end of an operating cycle during which extended operation with one percent defective fuel had occurred. Based on this and neglecting decay after degasification, the noble gas activity in the tank is given in Table 11.1-5.

11.1.4.4.3 Results of Analysis

	Dose (rems)	
Organ	**Exclusion Area Boundary (EAB)**	**LPZ**
Thyroid	--	--
Whole Body	6.4E-01 (1)	6.6E-02 (1)

(1) The current waste gas system failure analysis is based on updated reactor coolant design activity contained in Table 11.A-1. This analysis results in a whole body dose of 3.0E-01 rem at the EAB and 4.0E-02 rem at the LPZ. The current results are bounded by the licensed values listed above.

11.1.4.4.4 Conclusions

If a waste gas decay tank rupture did occur, the dose would be substantially below 10 CFR Part 100 guidelines.

11.1.5 SOLID WASTE PROCESSING SYSTEM

11.1.5.1 System Descriptions

The solid waste processing system is designed to provide controlled handling of spent resins, contaminated filter cartridges, and miscellaneous solid waste. The system is designed for handling solid waste with radioactivity levels resulting from reactor operation with design basis reactor coolant source terms. The sources and expected annual volumes of solid wastes are given in Table 11.1-2.

The estimated isotopic curie inventory for each source of solid waste is given in Table 11.1-3. The activity data in Table 11.1-3 are based on analyses done for the original licensing of Millstone Unit 2 and are not derived from the updated radionuclide activities shown in the Appendices to this FSAR chapter.

Design of the solid waste processing system for handling and disposal of each type of solid waste is as follows:

A. Spent Resins

Spent resins from the radioactive waste processing system demineralizers, CVCS ion exchangers, and spent fuel pool demineralizer are replaced in accordance with plant procedures. Resin replacement is accomplished by sluicing the resins from the hard-piped demineralizers and ion exchangers with nitrogen and/or demineralized water to the spent resin tank. The portable demineralizers may be sluiced with an air/demineralized water mixture to the spent resin tank (SRT), or sluiced directly to a shipping cask, bypassing the SRT.

Spent resins are accumulated and stored in the spent resin tank for radioactive decay prior to filling the disposable container located in a shipping cask at elevation (-)45 feet, 6 inches in the auxiliary building. The spent resin tank is sized for storing the total volume of resins resulting from one resin replacement per year per demineralizer. With the storage capacity available, minimum storage time of about six months is expected for resins in the tank. The resins are dewatered by the use of a pump after placement in the disposable container.

Solid waste containers, shipping casks, methods of packaging, and transportation meet applicable federal regulations 10 CFR Part 71and 10 CFR 171-178, and wastes are buried at a licensed burial site in accordance with applicable NRC 10 CFR Part 61. Solid waste treatment design is in compliance with the requirements of 10 CFR Part 20, Sections 1301 and 1302 as it relates to radioactivity in effluents to unrestricted areas.

The spent resin radwaste system is shown in Figure 11.1–7.

B. Contaminated Filter Cartridges

Filter cartridges for all radioactive service filters are normally replaced when the pressure drop across the filter unit exceeds 40 psi. All filters are located in concrete shielded compartments with access provided by a hatch located in the roof of the compartment. (See FSAR Section 11.2 for shielding design.) For the removal of contaminated cartridges, the concrete hatch plug is removed first. Then contaminated cartridges are removed in accordance with plant procedures.

The contaminated cartridges are transferred to shielded containers, which are then capped and stored in the drumming area for ultimate off-site disposal.

C. Miscellaneous Solid Wastes

Contaminated metallic materials and solid objects are placed in disposable shipping containers for transportation to an off site waste-processing facility or disposal site. Miscellaneous compressible wastes, such as contaminated clothing, rags, paper, etc., are transported to the Millstone Radwaste Reduction Facility for compacting.

A temporary storage facility for two high-integrity containers is provided in the container storage area of the auxiliary building. Temporary storage for various containers prior to further disposition can also be provided depending on curie content and container quantity.

Casks used for shipment of filter cartridges and miscellaneous solid wastes are rented as needed.

The containers are designed to prevent loss or dispersal of the contents and to maintain self-shielding properties under normal conditions of transport. Where surface dose rates exceed those allowed by the applicable Department of Transportation (DOT) regulations, the containers are provided with additional shielding. Where the curie content of the containers might exceed the regulatory limit applicable to this type of shipment, special casks licensed by DOT and by NRC, if required, would be used.

The shipment of radioactive waste materials is governed by NRC regulations as set forth in 10 CFR Part 71 and regulations of the U. S. DOT contained in 49 CFR Parts 170 through 178. The packaging and shipping of all waste from the Millstone site is in accordance with these regulations and any other applicable regulations that may come into effect. All loading of radioactive waste containers is performed by qualified personnel and monitored by radiation protection personnel. The transportation is provided by a carrier authorized by the disposal site operator in accordance with NRC and DOT regulations. The material to be shipped from the site is sent to a licensed waste disposal site or to a licensed waste-processing facility.

Table 11.1-4 gives the total estimated annual curie inventory of solid wastes to be shipped from the station for off-site burial. This estimated inventory is based on analyses done for the original licensing of Millstone Unit 2 and is not derived from the updated radionuclide activities shown in the Appendices to FSAR Chapter 11.

System Operation

1. Normal Operation

Spent resins from the sources listed in Table 11.1-3 are collected in the spent resin tank for radioactive decay. The spent resins are subsequently loaded into the shipping container and are dewatered by the spent resin shipping cask dewatering pump and portable dewatering pumps. After dewatering, the shipping container and cask are sealed and prepared for shipment.

Prior to a differential pressure of 40 psi being exceeded across any of the filters listed in Table 11.1-3, the filter is taken out of service, and allowed to decay. To change the filter cartridge, the concrete shield plug above the filter is removed. The cartridge is removed in accordance with plant procedures, placed in a shielded area and stored for shipment.

2. Abnormal Operation

No abnormal operations are anticipated.

11.1.6 SYSTEM RELIABILITY AND AVAILABILITY

11.1.6.1 Special Features

The radioactive waste processing system design is based upon the processing of wastes postulated to be generated from reactor operation with design basis reactor coolant activities and the release of these wastes in accordance with the requirements 10 CFR Part 20, Sections 1301 and 1302 and Appendix B. For expected radioactivity levels, sufficient processing capability is available to ensure all releases from the radioactive waste processing system are in accordance with 10 CFR Part 50, Appendix I.

The radioactive waste processing system is designed for as low as reasonably achievable (ALARA) radioactivity releases to the environment. All radioactive waste processing system vents, equipment drains, leakage, valve stem leadoffs, and relief valve discharges are collected by the drains system and reprocessed by the radioactive waste processing system. Those sources containing dissolved fission gases are collected in the equipment drain sump tank and processed as clean liquid waste. All other sources are collected and processed by the aerated liquid waste system. All vent gases containing fission gases are discharged to the gaseous waste system for storage and decay.

Each liquid and gaseous waste processing system is provided with a single discharge path to the environment. Each discharge header is provided with a radiation monitor and redundant isolation valves which are closed on high radiation level or instrument failure to prevent releases not in compliance with 10 CFR Part 20, Sections 1301 and 1302 and Appendix B.

All releases from the radioactive waste processing system to the environment are to be accomplished on a batch basis for suitable control. All wastes are first sampled to ensure compliance with 10 CFR Part 20, Sections 1301 and 1302 and Appendix B and 10 CFR Part 50, Appendix I. Adequate sample points are provided in the radioactive waste processing system for suitable control of the processing and to ensure that processing equipment performs as designed. All sampling points in the system are piped to the sampling room for analysis. Some local sampling points are utilized in the aerated and clean liquid waste systems.

To avoid inadvertent releases prior to sampling, the waste gas decay and coolant waste monitor tanks are provided with inlet and outlet valve interlocks. This control provision precludes the draining and filling of the tanks simultaneously.

All storage tanks in the liquid waste system are provided with piping and valves for recirculating the tank contents to obtain a representative sample. The valves are designed for remote operation to protect station operating personnel from radiation.

Sufficient storage capacity is provided for the retention of wastes in the radioactive waste processing system. The three RCS volume storage capacity in the clean liquid radioactive waste processing system is sufficient to allow for back-to-back cold shutdowns and startups up to 75 percent of the equilibrium core cycle. The six decay tanks allow for 60-day storage of all waste gases, including cover gases from the clean liquid radioactive waste processing system tanks.

The radioactive waste processing system is provided with suitable interconnections and sufficient flexibility to allow recirculation of waste for additional processing, if required. Suitable component redundancy is provided to ensure adequate processing.

11.1.6.2 Tests and Inspections

All components of the radioactive waste processing system are nondestructive tested in accordance with the applicable codes and standards as listed in Table 11.1-1. In addition to code requirements, additional testing is required for important components to ensure component integrity and performance.

All pumps in the radioactive waste processing system are manufacturer shop performance tested to demonstrate compliance with design head and capacity requirements.

Prototype filter cartridge testing was conducted by the filter manufacturer confirming a filter efficiency of 92 weight percent when tested with Fine Arizona Air Dust.

The assembled degasifier package was shop gas leak tested to ensure leak-tightness of all components. The degasifier package was subject to a shop performance test to confirm design calculations and performance. Testing to confirm the performance in the removing of dissolved gases was performed with the use of oxygen at feed rate of 60 gpm. Sufficient data were generated with oxygen testing to predict actual performance with respect to hydrogen, krypton and xenon removal. The results of these tests are shown in Figure 11.1–6. The testing also confirmed the capability of the degasifier to perform effective gas stripping under conditions of instantaneous feed rate changes and varying feed temperatures.

All system components are visually inspected and manually adjusted if necessary to ensure correct installation and arrangement. The completely installed system is subject to an acceptance test in accordance with design requirements. The acceptance test is to check and/or calibrate pumps, valves, instrumentation, interlocks, and system operation. In addition, the completely installed system will be checked for environmental considerations, such as correct routing of drains and vent connections, leakage, etc.

Individual system components are located so as to allow access for periodic inspection and testing after component decontamination. Major components are located in individual shielded rooms or compartments to maintain access control.

11.1.7 REFERENCES

11.1-1 NUREG-0017 Rev. 1, “Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors, PWR-GALE Code”.

TABLE 11.1-1 RADIOACTIVE WASTE PROCESSING SYSTEM COMPONENT DESCRIPTION

CLEAN LIQUID WASTE PROCESSING SYSTEM

Primary Drain Tank Pumps

Type	Inline horizontal centrifugal with mechanical seals
Quantity	2
Design temperature (°F)	250
Design head (TDH) (feet)	150
Design capacity (gpm)	50
NPSH available (feet)	3 to 33
Minimum NPSH required (feet)	3
Material:	
Case	ASTM A-351 Gr CF8M
Impeller	ASTM A-351 Gr CF8M
Shaft	ASTM A-276 TP 316
Motor	7.5 hp
Codes and Standards	ASME Section III Class 3 (1971)
Seismic design class	2
Design integrated radiation dosage (rad)	10^6

Degasifier Pumps

Type	Inline horizontal centrifugal with mechanical seals
Quantity	2
Design temperature (°F)	250
Design head (TDH) (feet)	115
Design capacity (gpm)	132
NPSH available (feet)	2
Minimum NPSH required (feet)	1.5
Material:	

Case	ASTM A-351 Gr CF8M
Impeller	ASTM A-351 Gr CF8M
Shaft	ASTM A-276 TP 316
Motor (hp)	7.5
Code	ASME Section III Class 3 (1971)
Seismic design class	2
Design integrated radiation dosage (rad)	10^6

Coolant Waste Receiver Tank and Monitor Tank Pumps

Type	Inline horizontal centrifugal with mechanical seals
Quantity	2
Design temperature (°F)	175
Design head (TDH) (feet)	115
Design capacity (gpm)	132
NPSH available (feet)	30
Minimum NPSH required (feet)	10
Material:	
Case	ASTM A-351 Gr CF8M
Impeller	ASTM A-351 Gr CF8M
Shaft	ASTM A-276 TP 316
Motor (hp)	7.5
Code	ASME Section III Class 3 (1971)
Seismic design class	2
Design integrated radiation dosage (rad)	10^6

Equipment Drain Sump Tank Pumps

Type	Vertical wet pit
Quantity	2
Design temperature (°F)	150
Design head (feet)	125
Design capacity (gpm)	50

Material:	
Case	Type 316 stainless steel
Impeller	Type 316 stainless steel
Shaft	Type 316 stainless steel
Motor (hp)	10
Code	Hydraulic Pump Institute
Seismic design class	2
Design integrated radiation dosage (rad)	10^6

Primary Drain Tank and Quench Tank Cooler Pump

Type	Inline horizontal centrifugal with mechanical seals
Quantity	1
Design temperature (°F)	325
Design head (TDH) (feet)	125
Design capacity (gpm)	100
NPSH available (feet)	3.3–50
Minimum NPSH required (feet)	3
Material:	
Case	ASTM A-351 Gr CF8M
Impeller	ASTM A-351 Gr CF8M
Shaft	ASTM A-276 TP 316
Motor (hp)	10
Code	ASME Section III Class 3 (1971)
Seismic design class	2
Design integrated radiation dosage (rad)	10^6

Primary Drain Tank

Type	Horizontal
Quantity	1
Volume (gallons net)	1,500
Design pressure (psig)	20

Design temperature (°F)	250
Operating pressure (psig)	3
Operating temperature (°F)	120–150
Material	ASTM A-240 TP 304
Design code	ASME Section III Class 3 (1971)
Seismic design classification	2

Coolant Waste Receiver Tanks

Type	Vertical
Quantity	2
Volume (gallons net)	60,000
Design pressure (psig)	15
Design temperature (°F)	150
Operating pressure (psig)	3
Operating temperature (°F)	120
Material	ASTM A-240 TP 304

Design code:

ASME Section III Class C (1968 Edition including addenda through Summer 1970)

Seismic design classification	2

Coolant Waste Monitor Tanks

Type	Vertical
Quantity	2
Volume (gallons net)	30,000
Design pressure (psig)	15
Design temperature (°F)	150
Operating pressure (psig)	3
Operating temperature (°F)	120
Material	ASTM A-240 TP 304

Design code:

ASME Section III Class C (1968 Edition including Addenda through Summer 1970)

Seismic design classification	2

Equipment Drain Sump Tank

Type	Vertical
Quantity	1
Volume (gallons net)	500
Design pressure (psig	20
Design temperature (°F)	150
Operating pressure (psig)	3
Operating temperature (°F)	120
Material	ASTM A-240 TP 304
Design code	ASME Section III Class 3 (1971)
Seismic design classification	2

Degasifier

Type	Packed column utilizing internally generated stripping steam.
Quantity	1
Design pressure (psig)	50
Design temperature (°F)	250
Operating pressure (psig)	5.3
Operating temperature (°F)	228
Design capacity (gpm)	40 to 132
Performance	See Figure 11.1–6

Design codes:

Components containing radioactive material:

ASME Section III Class C (1968 Edition including addenda through summer 1970), ANSI B31.7 Class III, TEMA R.

Other components:

ASME Section VIII, ANSI B31.1

Column:

Packing	One inch Rashig rings.
Height (feet)	4.25

Diameter (feet)	2
Materials:	
Components containing radioactive material:	ASTM A-240 TP 304
Other components:	Carbon steel
Seismic design classification	2

Coolant Waste– Demineralizers (Primary and Secondary)

Type	Mixed-bed, non-regenerative
Quantity	4 (See Note 3)
Design pressure (psig)	100
Design temperature (°F)	150
Operating pressure (psig)	60
Operating temperature (°F)	120
Design flow (gpm)	132
Design code	ASME Section III Class 3 (1971)
Material	ASTM A-240 TP 304
Seismic design classification	2

Coolant Waste–Filters

Type	Disposable cartridge
Quantity	2
Design pressure (psig)	200
Design temperature (°F)	250
Operating pressure (psig)	60
Operating temperature (°F)	120
Design flow rate (gpm)	132
Operating flow (gpm)	40–132
Filter rating (micron)	3
Design filter efficiency (%)	80
Design code:	

ASME Section III Class C (1968 Edition including Addenda through summer 1970).

Materials:

Vessel	ASTM A-312 TP 304
Internals	TP 304 SS, Micarta
Cartridges	Ethylene propylene
Seismic design classification	2

Coolant Waste–Piping and Valves

Piping:

Material	ASTM A-312 TP-304 or 316/316L
Design pressure (psig)	50, 100, 150
Design temperature (°F)	150, 250, 300, 350
Joints 2.5 inches and larger	Butt welded except at flanged equipment.
Joints 2 inches and smaller	Socket welded except at flanged equipment **

Codes:

Fabrication	ANSI B31.7 Class III *
Testing and Installation	ASME Section III Class 3 (1971) *, **

Valves: ASTM A-182 F304, F316; ASTM A-351 CF8, CF-8M

Ratings:

2.5 inches and larger	150 lb ANSI
2 inches and smaller	600 lb ANSI ***

Code:

ASME Draft for Pumps and Valves for Nuclear Service (1968) *, **

* Portions of the Clean Liquid Waste Processing System have been replaced with piping and piping components designed, constructed and tested to the ANSI B31.1 Power Piping Code with augmented requirements per the guidance in Regulatory Guide 1.143.

** Portions of the Clean Liquid Radwaste System have been designed, constructed and tested to the ANSI B31.1 Piping Code with augmented quality requirements per Regulatory Guide 1.143.

*** 600 pound ANSI rating represents minimum requirements. 800 pound ANSI rating valves are utilized on a case-by-case basis.

AERATED LIQUID WASTE PROCESSING SYSTEM

Aerated Waste Drain and Monitor Tanks

Type	Vertical
Quantity	3
Volume (gallons net)	5,000
Design pressure	Atmospheric
Design temperature (°F)	150
Operating pressure	Atmospheric
Operating temperature (°F)	120
Material	ASTM A-240 TP 304
Design codes	ASME Section III – ASME Section VIII (Original Fabrication)
Seismic design classification	2

Aerated Waste Demineralizer

Type	Mixed-bed, non-regenerative
Quantity	1
Design pressure (psig)	100
Design temperature (°F)	150
Operating pressure (psig)	60
Operating temperature (°F)	120
Design flow (gpm)	132
Normal operating flow (gpm)	50
Design code	ASME Section III Class 3 (1971)
Material	ASTM A-240 TP 304
Seismic design classification	2

Aerated Waste Portable Demineralizers

Type	Various, non-regenerative, sluicible
Quantity	3
Design Pressure	150 psig at 180°F

Volume	Approximately 15 ft^3 each

Filters

Type	Disposable cartridge
Quantity	2
Design pressure (psig)	200
Design temperature (°F)	250
Operating pressure (psig)	60
Operating temperature (°F)	120
Design flow rate (gpm)	132
Operating flow (gpm)	40–132
Filter rating (micron)	3
Design filter efficiency (%)	80

Design code:

ASME Section III Class C (1968 Edition including Addenda through summer 1970).

Seismic design	2

"A" Aerated Waste Drain Tank Pump

Type	Inline horizontal centrifugal with mechanical seals
Quantity	1
Design temperature (°F)	180
Design head (TDH) (feet)	238
Design capacity (gpm)	90
NPSH available (feet)	27
Minimum NPSH required (feet)	9
Horsepower (hp)	15

Material:

Case	ASTM A-351 Gr CF8M
Impeller	ASTM A-351 Gr CF8M
Shaft	ASTM A-276 TP 316
Codes and standards	ASME Section III Class 3 (1971)

Seismic design classification	2
Design integrated radiation dosage (rads)	10^6

"B" Aerated Waste Drain Tank and Monitor Tank Pumps

Type	Inline horizontal centrifugal with mechanical seals
Quality	2
Design temperature (°F)	150
Design head (TDH) (feet)	135
Design capacity (gpm)	50
NPSH available (feet)	27
Minimum NPSH required (feet)	5
Horsepower (hp)	5
Material:	
Case	ASTM A-351 Gr CF8M
Impeller	ASTM A-351 Gr CF8M
Shaft	ASTM A-276 TP 316
Codes and standards	ASME Section III Class 3 (1971)
Seismic design classification	2
Design integrated radiation dosage (rads)	10^6

Piping and Valves

Piping:	
Material	ASTM A-316 TP 304 or 316L
Design pressure (psig)	50 and 100
Design temperature (°F)	150
Joints:	
2.5 inches and larger	Butt welded except at flanged equipment
2 inches and smaller	Socket welded except at flanged equipment
Codes:	
Fabrication	ANSI B31.7 Class III *
Testing and Installation	ASME Section III Class 3 (1971) *

Valves	
Material	ASTM A-182 F 304, F316
Ratings:	
2.5 inches and larger	150 lb ANSI
2 inches and smaller	600 lb ANSI, 800 lb ANSI (Intermediate Class)
Code	ASME Draft for Pumps and Valves for Nuclear Service (1968) *

Portions of the Aerated Liquid Waste Processing System have been replaced with piping and piping components designed, constructed and tested to the ANSI B31.1 Power Piping Code with augmented requirements per the guidance in Regulatory Guide 1.143.

GASEOUS WASTE PROCESSING SYSTEM

Waste Gas Compressors

Type	Diaphragm, single state
Quantity	2
Capacity (scfm)	25 at 14.7 psia suction pressure
Design discharge pressure (psig)	150
Design pressure (psig)	165
Materials:	
Heads - ASTM A-105 Grade 2, with ASTM A-240 TP 304 in contact with waste gases.	
Diaphragm	TP 301 stainless steel or 316 stainless steel
Support cylinder	ASTM A-105 Grade 2
Motor (hp)	40
Seismic design classification	See Note 1
Design integrated radiation level (rads)	3.5×10^5
Code:	
ASME Section VIII, ASME Draft for Pump and Valves for Nuclear Service, ANSI B31.7.	

Waste Gas Surge Tank

Quantity	1
Type	Vertical
Design pressure (psig)	20
Design temperature (°F)	150
Normal operating pressure (psig)	3 to 5
Normal operating temp (°F)	150
Volume (ft^3)	582
Material	ASTM A-240 TP 304
Seismic design classification	2
Code:	
ASME Section III Class C (1968 Edition including Addenda through 1970).	

Waste Gas Decay Tank

Type	Vertical
Quantity	6
Design pressure (psig)	165
Design temperature (°F)	150
Normal operating pressure (psig)	5 to 150
Normal operating temp (°F)	120
Volume (ft^3)	582
Material	ASTM A-515 Grade 70
Seismic design classification	See Note 1
Code:	
ASME Section III Class C (1968 Edition including Addenda through summer 1970).	

Waste Gas Filter

Type	Disposable cartridge with HEPA filter and demister
Quantity	1
Design flow (scfm)	50
Normal operating flow (scfm)	5 - 25
Design pressure (psig)	165
Design temperature (°F)	150
Normal operating pressure (psig)	4
Normal operating temp (°F)	120
Design efficiency	99.97% of particles 0.3 micron and larger
Material:	
Filter housing	ASTM A-240 TP 304
Internals	TP 304 stainless steel
Cartridge	Glass fiber
Code	ASME Section III Class 3 (1971)
Seismic design classification	See Note 1

Piping and Valves

Piping:	
Material	ASTM A-106 Grade B
Design pressure (psig)	50, 150 and 200
Design temperature (°F)	150, 200 and 300
Joints:	
2.5 inches and larger	Butt welded except at flanged equipment.
2 inches and smaller	Socket welded except at flanged equipment.
Codes:	
Fabrication	ANSI B31.7 Class III
Testing and Installation	ASME Section III Class 3 (1971)
Design seismic classification	2 (See Note 1)
Valves:	
Materials	ASTM A-216, Grade WCB
Ratings:	
2.5 inches and larger	150 lb ANSI butt weld ends
2 inches and smaller	3,000 lb ANSI socket weld ends
Code:	ASME Draft for Pup and Valves for Nuclear Service (1968).

SOLID WASTE PROCESSING SYSTEM

Spent Resin Storage Tank

Type	Vertical
Quantity	1
Volume (ft^3)	380
Design pressure (psig)	75
Operating pressure (psig)	35
Design temperature (°F)	175
Operating temperature (°F)	120
Material	ASTM A-240 TP 304
Design code	ASME Section III Class 3 (1971)
Seismic design classification	2

Spent Resin Shipping Cask Dewatering Pump

Type	Horizontal Centrifugal
Quantity	1
Capacity (gpm)	10
Head (feet)	92
Material	Stainless steel

Note 1: The high pressure components of the gaseous waste processing system are designated as Seismic Class 2 equipment. However, the components are designed to meet Seismic Class 1 loadings due to the nature of the equipment service.

Note 2: The supply lines to the solidification concentrates pump were cut in accordance with PDCR 2-54-95.

Note 3: Secondary Demineralizer may be filled with either ion-specific resin or mixed bed resin.

TABLE 11.1-2 SOURCES AND EXPECTED VOLUMES OF SOLID WASTES

Source	Waste Generating Operation	Quantity Per Year	Per Year
Spent resins	One resin replacement per demineralizer/year	351 cu ft	7 *
Contaminated filter cartridges	One cartridge replacement per filter/year	12 cartridges	12 **
Miscellaneous solid wastes	One 55 gallon drum/week	-	52 **

* 50 cubic foot containers

** 55 gallon drums

Aerated Waste hardpiped demineralizer at 42 cu. ft. x 1 = 42 cubic feet

Aerated Waste sluicible portable demineralizers at 15 cubic feet/each x 3 = 45 cubic feet

Spent Fuel Pool demineralizer at 42 cubic feet x 1 = 42 cubic feet

Primary Liquid Radwaste demineralizers at 42 cubic feet x 2 = 84 cubic feet

Secondary Liquid Radwaste demineralizer at 42 cubic feet x 1 = 42 cubic feet

Letdown Ion Exchangers at 32 cubic feet x 3 = 96 cubic feet

351 cubic feet

Note: The data in this Table is not derived from the updated 10 CFR 50 Appendix I analysis. The best source of Solid Waste Volumes can be found in the Annual Radiological Effluent Release Report.

TABLE 11.1-3 RADIOACTIVITY LEVELS OF SOLID WASTES (SEE NOTE)

ION EXCHANGER AND DEMINERALIZER RESINS

	Chemical and Volume Control System Purification Ion Exchanger Resin		Chemical and Volume Control System Purification Ion Exchanger Resin		Chemical and Volume Control System Deborating Ion Exchanger Resin		Primary Clean Liquid Waste Demineralizer Demineralizer Resin		Aerated Liquid Waste Demineralizer Resin		Spent Fuel Pool Clean-up Demineralizer Resin	
Isotope	**(Curies)**		**(Li Removal) (Curies)**		**(Curies)**		**(Curies)**		**(Curies)**		**(Curies)**	
	Normal	**Maximum**	**Normal**	**Maximum**	**Normal**	**Maximum**	**Normal**	**Maximum**	**Normal**	**Maximum**	**Normal**	**Maximum**
Cr-51												
Mn-54												
Mn-56												
Co-58												
Fe-59												
Co-60	4.78×10^{-2}	0.478	9.6×10^{-3}	9.6×10^{-2}	1.2×10^{-3}	1.2×10^{-2}	1.12×10^{-4}	1.9×10^{-3}	8.72×10^{-5}	1.01×10^{-2}	1.23	12.3
Rb-88	1.366	13.66	0.246	2.46	--	--	3.1×10^{-3}	5.22×10^{-2}	2.4×10^{-3}	0.279	67.2	672
Rb-89	3.06×10^{-2}	0.306	5.5×10^{-3}	5.5×10^{-2}	--	--	6.9×10^{-5}	1.17×10^{-3}	5.38×10^{-5}	6.25×10^{-3}	1.69	16.9
Sr-89	13.8	138.0	2.398	23.98	--	--	3.11×10^{-2}	0.525	2.41×10^{-2}	2.80	0.134	1.34
Sr-90	3.052	30.52	0.471	4.71	--	--	6.77×10^{-3}	0.114	5.25×10^{-3}	0.610	6.88×10^{-3}	6.88×10^{-2}
Y-90	--	--	2.24×10^{-2}	0.224	--	--	4.31×10^{-5}	7.26×10^{-4}	3.34×10^{-5}	3.88×10^{-3}	2.69×10^{-2}	0.269
Sr-91	6.9×10^{-2}	0.691	1.26×10^{-2}	0.126	--	--	1.57×10^{-4}	2.65×10^{-3}	1.22×10^{-4}	1.41×10^{-2}	9.39×10^{-2}	0.939

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TABLE 11.1-3 RADIOACTIVITY LEVELS OF SOLID WASTES (SEE NOTE) (CONTINUED)

ION EXCHANGER AND DEMINERALIZER RESINS

	Chemical and Volume Control System Purification Ion Exchanger Resin		Chemical and Volume Control System Purification Ion Exchanger Resin		Chemical and Volume Control System Deborating Ion Exchanger Resin		Primary Clean Liquid Waste Demineralizer Demineralizer Resin		Aerated Liquid Waste Demineralizer Resin		Spent Fuel Pool Clean-up Demineralizer Resin	
Isotope	**(Curies)**		**(Li Removal) (Curies)**		**(Curies)**		**(Curies)**		**(Curies)**		**(Curies)**	
	Normal	**Maximum**	**Normal**	**Maximum**	**Normal**	**Maximum**	**Normal**	**Maximum**	**Normal**	**Maximum**	**Normal**	**Maximum**
Y-91	--	--	7.71×10^{-2}	0.771	--	--	1.48×10^{-4}	2.5×10^{-3}	1.15×10^{-4}	1.33×10^{-2}	2.93	29.3
Zr-95												
Mo-99	--	--	52.46	524.6	--	--	0.101	1.70	7.82×10^{-2}	9.08	53.5	535
Ru-103	7.84	78.4	1.57×10^{-2}	0.157	--	--	1.51×10^{-2}	0.255	1.17×10^{-2}	1.36	0.109	1.09
Ru-106	2.02	20.2	0.363	3.63	--	--	4.58×10^{-3}	7.72×10^{-2}	3.55×10^{-3}	0.412	6.54×10^{-3}	6.54×10^{-2}
I-129	9.36×10^{-4}	9.36×10^{-3}	1.56×10^{-4}	1.56×10^{-3}	--	--	2.1×10^{-6}	3.54×10^{-5}	1.63×10^{-6}	1.89×10^{-4}	1.9×10^{-6}	1.9×10^{-5}
Te-129	5.42×10^{-2}	0.542	1.09×10^{-2}	0.109	1.3×10^{-3}	1.3×10^{-2}	1.28×10^{-4}	2.15×10^{-3}	9.88×10^{-5}	1.15×10^{-2}	0.662	6.62
I-131	1,680.0	16,800.0	336.0	3,360.0	19.2	192.0	3.91	65.9	3.03	352	105	1,050
I-132	5.04	50.4	1.007	10.07	0.121	1.21	1.19×10^{-2}	0.20	9.19×10^{-3}	1.07	26.9	269
Te-132	49.3	493.0	9.86	98.6	0.944	9.44	0.116	1.95	8.96×10^{-2}	10.4	8.7	87.0
I-133	247.0	2,470.0	4.94	49.4	5.91	59.1	0.496	8.35	0.384	44.6	149	1,490
Cs-134	388.0	3,800.0	517.0	5,170.0	--	--	1.74	29.3	1.35	156	2.64	26.4

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TABLE 11.1-3 RADIOACTIVITY LEVELS OF SOLID WASTES (SEE NOTE) (CONTINUED)

ION EXCHANGER AND DEMINERALIZER RESINS

	Chemical and Volume Control System Purification Ion Exchanger Resin		Chemical and Volume Control System Purification Ion Exchanger Resin		Chemical and Volume Control System Deborating Ion Exchanger Resin		Primary Clean Liquid Waste Demineralizer Demineralizer Resin		Aerated Liquid Waste Demineralizer Resin		Spent Fuel Pool Clean-up Demineralizer Resin	
Isotope	**(Curies)**		**(Li Removal) (Curies)**		**(Curies)**		**(Curies)**		**(Curies)**		**(Curies)**	
	Normal	**Maximum**	**Normal**	**Maximum**	**Normal**	**Maximum**	**Normal**	**Maximum**	**Normal**	**Maximum**	**Normal**	**Maximum**
I-134	1.06	10.6	0.212	2.12	2.54×10^{-2}	0.254	2.49×10^{-3}	4.2×10^{-2}	1.93×10^{-3}	0.225	16.3	163
Te-134	3.19×10^{-2}	0.319	6.4×10^{-3}	6.4×10^{-2}	7.65×10^{-4}	7.65×10^{-4}	7.5×10^{-5}	1.26×10^{-3}	5.82×10^{-3}	6.76×10^{-3}	0.691	6.91
I-135	35.9	359.0	7.24	72.4	0.869	8.69	8.46×10^{-2}	1.43	6.56×10^{-2}	7.62	71.2	712
Cs-136	--	--	18.12	181.2	--	--	3.48×10^{-2}	0.587	2.7×10^{-2}	3.14	0.672	6.72
Cs-137	1,695.0	16,950.0	1,728.0	17,280.0	--	--	6.58	111	5.10	593	8.44	84.4
Cs-138	--	--	0.14	1.40	--	--	2.69×10^{-4}	4.54×10^{-3}	2.09×10^{-4}	2.42×10^{-2}	18.2	182
Ba-140	4.08	40.8	0.743	7.43	--	--	9.27×10^{-3}	0.156	7.19×10^{-3}	0.835	0.161	1.61
La-140	0.469	4.69	9.37×10^{-2}	0.937	--	--	1.08×10^{-3}	1.82×10^{-2}	8.38×10^{-4}	9.74×10^{-2}	0.154	1.54
Pr-143	3.49	34.9	0.697	6.97	--	--	8.05×10^{-3}	0.136	6.24×10^{-3}	0.725	0.154	1.54
Ce-144	26.2	262.0	5.24	52.4	--	--	6.04×10^{-2}	1.02	4.68×10^{-2}	5.44	0.109	1.09

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TABLE 11.1-3 RADIOACTIVITY LEVELS OF SOLID WASTES (SEE NOTE) FILTER CARTRIDGES

	Chemical and Volume Control System Filter Cartridges		Clean Liquid Waste Filter Cartridges		Aerated Liquid Waste Filter Cartridges		Spent Fuel Pool Clean-up Filter Cartridges		Reactor Vessel Head Decontamination System Filter Cartridges	
Isotope	(Curies/Cartridges)		(Curies/Cartridge)		(Curies/Cartridges)		(Curies/Cartridge)		(Curies/Cartridges)	
	Normal	Maximum	Normal	Maximum	Normal	Maximum	Normal	Maximum	Normal	Maximum
Cr-51	0.209	13.05	0.001	0.043	--	0.452	0.003	0.185	0.039	0.039
Mn-54	0.935	0.67	0.002	0.002	0.004	0.024	0.004	0.003	0.028	0.028
Mn-56	--	0.486	--	0.002	--	0.018	--	--	--	--
Co-58	64.0	107.0	0.123	0.347	0.192	3.702	0.50	0.84	4.78	4.78
Fe-59	0.192	0.069	0.001	0.001	--	0.002	0.002	0.001	0.022	0.022
Co-60	0.025	61.0	--	0.166	--	1.766	0.068	0.142	0.533	0.533
Br-84										
Rb-88										
Rb-89										
Sr-89										
Sr-90										
Y-90										
Sr-91										
Y-91										
Zr-95	--	--	--	--	--	--	--	--	--	--
Mo-99										
Ru-103										
Ru-106										

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TABLE 11.1-3 RADIOACTIVITY LEVELS OF SOLID WASTES (SEE NOTE) FILTER CARTRIDGES (CONTINUED)

	Chemical and Volume Control System Filter Cartridges		Clean Liquid Waste Filter Cartridges		Aerated Liquid Waste Filter Cartridges		Spent Fuel Pool Clean-up Filter Cartridges		Reactor Vessel Head Decontamination System Filter Cartridges	
Isotope	(Curies/Cartridges)		(Curies/Cartridge)		(Curies/Cartridges)		(Curies/Cartridge)		(Curies/Cartridges)	
	Normal	Maximum	Normal	Maximum	Normal	Maximum	Normal	Maximum	Normal	Maximum
I-129										
Te-129										
I-131										
I-132										
Te-132										
I-133										
Cs-134										
I-134										
Te-134										
I-135										
Cs-136										
Cs-137										
Cs-138										
Ba-140										
La-140										
Pr-143										
Ce-144										

Note:The activity data in this Table are based on analysis done for the original licensing of Millstone Unit 2 and are not derived from the updated radionuclide activities shown in the Appendices to FSAR Chapter 11.

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TABLE 11.1-4 CURIE INVENTORY OF SOLID WASTE SHIPPED FROM MILLSTONE UNIT 2 (SEE NOTE)

Source	Normal Curies per Year	Maximum Curies per Year
Resins	4,215	42,871
Filter Cartridges	35	149
Concentrates	Less than 1	5
TOTAL	4,250	43,025

Note: The activity data in this Table are based on analysis done for the original licensing of Millstone Unit 2 and are not derived from the updated radionuclide activities shown in the Appendices to FSAR Chapter 11. The best source of Solid Waste data can be found in the Annual Radiological Effluent Release Report.

TABLE 11.1-5 ASSUMPTIONS FOR WASTE GAS DECAY TANK ACCIDENT

Assumption (1) Maximum Noble Gas Activity in Waste Gas Decay Tank.

Isotope	Activity, curies
Kr-85m	278
Kr-85	942
Kr-87	160
Kr-88	494
Xe-131m	486
Xe-133m	708
Xe-133	45,600
Xe-135m	26.2
Xe-135	1,280
Xe-137	28.9
Xe-138	101

Basis: Maximum activity in tanks based on 1% degraded fuel, core power = 2700 MWt and degassing one system volume.

Assumption (2) 2 Hour Ground Level Release.

Basis: Regulatory Guide 1.24

Assumption (3) Ground Level X/Q:

EAB hr. = 3.66 E -04 (sec/m^3)

LPZ 4.80 E -05 (sec/m^3)

Basis: 95% maximum X/Q's during the years 1974 -1981.

FIGURE 11.1–1 P&ID CLEAN LIQUID RADWASTE SYSTEM

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 11.1–2 P&ID CLEAN LIQUID RADWASTE SYSTEM

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 11.1–3 P&ID DRAINS (CONTAINMENT & AUXILIARY BUILDING AND AUXILIARY YARD SUMP) (SHEET 1)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 11.1–3 P&ID DRAINS (CONTAINMENT & AUXILIARY BUILDING AND AUXILIARY YARD SUMP) (SHEET 2)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 11.1–3 P&ID DRAINS (CONTAINMENT & AUXILIARY BUILDING AND AUXILIARY YARD SUMP) (SHEET 3)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 11.1–4 P&ID AERATED LIQUID RADWASTE SYSTEM

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 11.1–5 P&ID DIAGRAM GASEOUS RADWASTE SYSTEM

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 11.1–6 DEGASIFIER PERFORMANCE CURVE

FIGURE 11.1–7 P&ID SPENT RESIN RADWASTE SYSTEM

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-2 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

11.2 RADIATION PROTECTION

11.2.1 FINAL SAFETY ANALYSIS REPORT UPDATE

It should be noted that the information in this section provided the basis and description of the shielding design prior to operation. As the station has operated, dose rates have fluctuated within the various areas as a function of time but for the most part have remained well within the zone designation limits presented below. Minor changes in equipment layout from that indicated in Figures 11.2–1 through 11.2–10 have been made, but since the values and figures given in this section provided the design basis for the major shielding structures, they are not being changed.

Many minor shielding changes (mostly through the use of portable shielding) have been incorporated in the interest of maintaining occupational exposures ALARA. The only major shielding change was the installation of a reactor cavity neutron shield.

The values and figures given in this section should not be used to estimate area dose rates. Actual health physics surveys are available and should be consulted to get a more accurate and up-to-date indication of radiological conditions.

Appendices 11.D and 11.E provide some of the design basis considerations on the control and expected levels of airborne radioactivity in the plant. Again, health physics survey data should be consulted for more realistic indications of actual airborne radioactivity levels. The occupational exposure due to airborne radioactivity has been insignificant in comparison to the direct radiation exposure.

11.2.2 DESIGN BASES

The shielding is designed to perform two primary functions: to ensure that during normal operation the radiation dose to operating personnel and to the general public is within the radiation exposure limits set forth in 10 CFR Part 20 and to ensure that operating personnel are adequately protected in the event of a reactor incident so that the incident can be terminated without undue hazard to the general public. The shielding design is based on operating at design power level of 2700 MWt with reactor coolant system activity levels corresponding to one percent failed fuel. The shielding design is governed by the limits for radiation levels as follows:

Zone Description	Maximum Dose Rate (mrem/hr)
Uncontrolled, unlimited access (Zone A)	0.5
Controlled, unlimited (40 hr/week) (Zone B)	1.0
Controlled, limited access (6-2/3 hr/week) (Zone C)	15
Controlled, limited access (1 hr/wk) (Zone D)	100
Normally inaccessible (Zone E)	> 100

The maximum external dose rates within the station are shown in the radiation zone diagrams, Figures 11.2–1 through 11.2–10, as described in this subsection and later in Section 11.2.2. These maximum external dose rates are based on one percent failed fuel. Since the expected amount of failed fuel is much less than this, the average external dose rates will be much lower. The unrestricted areas in the plant where construction workers and plant visitors may be are all radiation Zone A areas which have a maximum dose rate of 0.5 mrem/hr. Only plant workers are allowed in radiation zones with dose rates above this level, and their stay times are increasingly limited as the dose rate increases.

The anticipated average dose rates will vary within each zone and with operating conditions. In general, the dose rate normally around process equipment is expected to be at least an order of magnitude lower than indicated by the applicable zone. This is due to the extremely conservative assumptions made concerning the quantity of failed fuel. The direct dose at the site boundary is negligible both in the case of normal operation and under postulated incident conditions.

Shielding wall thicknesses, as shown in Figures 11.2–1 through 11.2–10, were found using the theory of the Reactor Shielding Design Manual by T. Rockwell, Reactor Physics Constants ANL-5800, Table of Isotopes by Perlman and Lederer, Engineering Compendium on Radiation Shielding and Nuclear Engineering Handbook by H. Etherington. The geometric considerations were determined by each physical situation, such as source configurations and distances to dose points.

Assumptions used included those of TID 14844 and the following:

1. If the source was liquid, then self-attenuation by water was used. If the source was gaseous, then no self-attenuation was assumed. If the source was solid, the volume percentages of each solid were used for self-attenuation (for example, solidified waste in drums).

2. Shields included the tank or pipe walls which were assumed to have an attenuation coefficient similar to that for iron.

3. The shielding materials were lead, iron, water, and concrete.

4. Sources were given in Mev/cm^3 sec for each of 7 standard energy bins: 0.4, 0.8, 1.3, 1.7, 2.2, 2.5, > 2.5 Mev, and 6.1 and 7.1 Mev for nitrogen 16. One percent failed fuel was assumed.

5. Flux to dose conversion factors used were those given by Henderson in XDC-59-8-179.

6. The shield thicknesses were determined by the amount of shielding required to achieve a dose rate less than the upper limit for the radiation zone in which the dose point was located.

The shielding design, therefore, is based on conservative assumptions ensuring adequate radiation protection to the general public and to operating personnel as well.

11.2.3 DESCRIPTION

Shielding throughout Millstone Unit 2 is designed in accordance with the criteria specified in Section 11.2.1. Figures 11.2–1 through 11.2–10 show the equipment placement, the quantity of shielding used and the appropriate radiation zones. All components containing radioactive fluids are shown on the figures except the refueling water storage tank (RWST), located as shown on Figure 1.2–2. The maximum direct dose rate from the content of the RWST is 5.6×10^{-3} mr/hr, based on reactor operation with one percent failed fuel. The maximum total site boundary dose due to the RWST is approximately 1.66 mrem/yr. For normal operating conditions, the site boundary doses will be approximately one-tenth of those values given above.

11.2.3.1 Containment Shielding

The containment shielding consisting of the primary shielding, the reactor cavity neutron shielding, the secondary shielding, and the containment wall, is shown on Figure 1.2–6.

Primary shielding is provided to limit radiation emanating from the reactor vessel. The primary shielding is designed to:

a. Attenuate the neutron flux to limit the activation of components and structural steel.

b. Limit the radiation level after shutdown to permit access to the reactor coolant system equipment.

c. To reduce, in conjunction with the secondary shield, the radiation level from sources within the reactor vessel to allow limited access to the containment during normal operation.

The primary shield consists of a minimum of five feet of reinforced concrete surrounding the reactor vessel. The cavity between the primary shield and the reactor vessel insulation is air-cooled to prevent overheating and dehydration of the concrete primary shield wall.

A reactor cavity neutron shield reduces the operating neutron and gamma dose rates in the containment. A dose reduction factor of 40 on the operating floor of the containment was the design goal for the shield. With the postulation of leak-before-break applied to the reactor coolant piping, the neutron shield design does not consider dynamic effects resulting from a design basis LOCA.

The permanent reactor cavity neutron shield consists of borated concrete blocks contained inside stainless steel containers. The concrete blocks are supported by the shielding support bars, which span between the reactor vessel flange and the embedment ring. The borated concrete blocks are 15 inches high and 23.25 inches wide, located all around the reactor vessel beneath the permanent

reactor cavity seal. The bottom and the sides of the shielding are insulated by NUKON blanket insulation to maintain an acceptable shielding temperature as well as limit heat loss from the reactor vessel (Figures 11.2–11 through 11.2–12). Thermal analysis has shown that the borated concrete blocks inside the stainless steel containers could reach a temperature of up to 455°F in some sections of the concrete blocks during normal operation.

The shielding is provided with personnel access openings, located directly above the neutron detector wells. During reactor operation these openings are plugged with borated concrete shield plugs to further minimize neutron flux. The shield plugs are removed through the access openings in the permanent cavity seal for maintenance of the neutron detector instrumentation.

The shielding in combination with the reactor cavity seal described in Section 4.3.10 will provide adequate airflow channels to maintain sufficient cavity cooling during reactor operation.

In addition to the borated concrete shielding, borated polyethylene layers are installed approximately at the reactor vessel flange elevation to reduce streaming between the borated concrete shielding and the primary shield wall. The thickness of the polyethylene shielding ranges from 1 to 3 inches contained in stainless steel liner between the ribs of the support members. The borated polyethylene layers are not insulated. This shielding is cooled by cavity cooling air passing under the shielding at temperatures less than 150°F. Together, the borated concrete blocks and the borated polyethylene layers are capable of reducing the neutron radiation by a dose factor of approximately 16 to 27.

Secondary shielding is provided to reduce the activity from the reactor coolant system to radiation levels which allow limited access to the containment during normal operation and to supplement primary shielding. Nitrogen 16 is the major source of radioactivity in the reactor coolant during operation and controls the thickness of the secondary shield. The secondary shielding consists of a minimum of 3.5 feet of reinforced concrete surrounding the reactor coolant piping, pumps, steam generators and pressurizer. The N-16 activity concentration at the reactor vessel outlet nozzles is 9.8×10^{-5} Ci/cc at full power.

The containment is a reinforced prestressed concrete structure with 3.75 feet thick cylindrical walls and a 3.25 feet thickness dome. In conjunction with the primary and secondary shield, it will limit the radiation level outside the enclosure building due to sources inside the structure to no more than 0.5 mrem/hr at full power operation. The structure is also designed to protect plant personnel from radiation sources inside the structure following a postulated incident.

11.2.3.2 Auxiliary Building Shielding

The function of the auxiliary building shielding is to protect personnel working near various system components, such as those in the CVCS, the radioactive waste processing system, sampling system, and the spent fuel pool cooling system. Controlled access to the auxiliary building is allowed during reactor operation. Each equipment compartment is individually shielded to reduce the radiation level in it and adjacent compartments as reflected by the zone designations. Source terms used in the design of shielding for major components throughout the

auxiliary building are listed in Table 11.2-1. Similar source terms for all components containing radioactive materials were developed and used in the shielding design.

11.2.3.3 Control Room Shielding

The layout of the control room is shown in Figure 1.2–7. In conformance with General Criterion Number 19 of 10 CFR Part 50, the control room shielding is designed to ensure that the dose will not exceed 5 rem for the duration of the incident (see Subsection 14.18.3.3). The walls of the control room are two feet thick. Under normal operating conditions, the control room is a Zone A region with expected dose rates well below the indicated limits.

11.2.3.4 Spent Fuel Pool Shielding and Fuel Handling Shielding

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

The refueling cavity above the reactor vessel is flooded to Elevation 36 feet 6 inches to provide a temporary water shield above the components being withdrawn from the reactor vessel.

The water height is approximately 24 feet above the reactor vessel flange. This height assures a minimum of 108 inches of water above the active portion of a withdrawn fuel assembly at its highest point of travel. Under these conditions, the dose rate from the spent fuel assembly is less than 1.0 mrem/hr at the water surface.

Upon removal of the fuel assembly from the reactor vessel, it is moved to the spent fuel pool by the fuel transfer mechanism, via the fuel transfer tube. Concrete shielding is provided around reactor internals storage and the steam generator for personnel protection during refueling. The spent fuel pool in the auxiliary building is permanently flooded to provide a minimum of 108 inches of water above the active portion of a fuel assembly when being withdrawn from the fuel transfer tube and raised by the fuel pool platform crane, prior to insertion in the spent fuel storage rack. The minimum water height above stored fuel assemblies is approximately 24.5 feet during operation of the spent fuel pool cooling system, described in Section 9.5.2.1, to avoid air entrainment at the pump suction intakes. Otherwise, a minimum height of 23 feet prevails in the spent fuel pool and also, while fuel is in movement, above the reactor pressure vessel, to satisfy fission product retention assumptions for fuel handling accident calculations (see Section 14.7.4.2). The sides of the spent fuel pool are six feet thick concrete to ensure a dose rate of less than 0.035 mrem/hr on the outer surface of the spent fuel pool.

11.2.3.5 Piping Systems Shielding

All piping systems containing radioactive material are routed and/or provided with shielding in accordance with the radiation zones given in Section 11.2.1. Consideration is given to maintenance and inspection requirements for components located in shielded compartments through which these lines are routed.

Isometrics of field run piping, two inches and smaller in diameter, were reviewed and approved by Bechtel Corporation Engineering Department.

11.2.4 HEALTH PHYSICS PROGRAM

Information regarding the health physics program organization is presented in Section 12.5 of Millstone 3 Final Safety Analysis Report (Reference 11.2-1). That information is contained herein by reference.

11.2.5 REFERENCES

11.2-1 Millstone Unit 3, Final Safety Analysis Report, Section 12.5 - Health Physics Program

TABLE 11.2-1 SOURCE TERMS FOR SHIELDING DESIGN

Isotope	Letdown Pre-Filter (Curies)	Purification Ion Exchanger (Curies)	Letdown Post-filter (Curies)	Volume Control Tank Demineralizer	Degasifier (Curies)
Br-84	-	1.23	-	840.6	4.24×10^{-3}
Kr-85m	-	-	-	34.0	0.0
Kr-85	-	-	-	87.1	0.0
Kr-87	-	-	-	18.4	0.0
Kr-88	-	-	-	58.2	0.0
Rb-88	-	38.20	-	4.63	0.23
Rb-89	-	0.82	-	0.11	4.02×10^{-3}
Sr-89	-	339.80	-	0.01	5.24×10^{-4}
Sr-90	-	85.90	-	5.0×10^{-4}	2.63×10^{-5}
Y-90	-	1.79	-	2.1×10^{-2}	1.05×10^{-3}
Sr-91	-	-	-	6.7×10^{-3}	3.32×10^{-4}
Y-91	-	-	-	5.22	0.26
Mo-99	-	-	-	40.86	2.06
Ru-103	-	222.0	-	8.4×10^{-3}	4.24×10^{-4}
Ru-106	-	59.4	-	4.9×10^{-4}	2.49x10–5
Te-129	-	1.54	-	5.0×10^{-2}	2.52×10^{-3}
I-129	-	2.4×10^{-2}	-	1.44×10^{7}	7.29×10^{-9}
I-131	-	4.26×10^{4}	-	8.01	4.04×10^{-1}
Xe-131m	-	-	-	50.8	0.0
Te-132	-	1.40×10^{3}	-	0.65	3.27×10^{-2}
I-132	-	129.0	-	2.03	1.03×10^{-1}
I-133	-	6.32×10^{2}	-	10.90	5.51×10^{-1}
Xe-133	-	-	-	5012.3	0.0
Te-134	-	0.92	-	4.7×10^{2}	2.39×10^{-3}

TABLE 11.2-1 SOURCE TERMS FOR SHIELDING DESIGN (CONTINUED)

Isotope	Letdown Pre-Filter (Curies)	Purification Ion Exchanger (Curies)	Letdown Post-filter (Curies)	Volume Control Tank Demineralizer	Degasifier (Curies)
I-134	-	27.55	-	1.13	5.72×10^{-2}
Cs-134	-	-	-	8.19	4.13×10^{-1}
I-135	-	933.4	-	5.04	2.54×10^{-1}
Xe-135	-	-	-	172.4	0.0
Cs-136	-	-	-	0.28	1.45×10^{-2}
Cs-137	-	-	-	39.78	2.0
Xe-138	-	-	-	8.16	0.0
Cs-138	-	-	-	1.24	3.27×10^{-2}
Ba-140	-	104.2	-	1.23×10^{-2}	6.23×10^{-4}
La-140	-	13.1	-	1.19×10^{-2}	6.00×10^{-4}
Pr-143	-	-	-	1.08×10^{-2}	5.45×10^{-3}
Ce-143	-	-	-	7.74×10^{-3}	3.91×10^{-3}
Co-60	263.0	-	2.63	1.98×10^{-4}	1.0×10^{-5}
Fe-59	0.45	-	4.5×10^{-3}	3.83×10^{-6}	2.09×10^{-7}
Co-58	690.0	-	6.90	1.42×10^{-3}	7.18×10^{-6}
Mn-56	3.27	-	3.27×10^{-2}	4.14×10^{-3}	2.09×10^{-5}
Mn-54	3.69	-	3.69×10^{-2}	4.95×10^{-6}	2.50×10^{-7}
Cr-51	88.10	-	0.88	4.32×10^{-4}	2.18×10^{-7}
Zr-95	1.54	-	0.15	3.42×10^{-6}	1.73×10^{-7}

TABLE 11.2-1 SOURCE TERMS FOR SHIELDING DESIGNS

Isotope	Waste Gas Surge Tank	Waste Gas Decay Tank	Aerated Waste Drain Tank	Aerated Waste Demineralizer	Aerated Waste Monitor Tank
Br-84	-	-	$8.8x10^{-3}$	$2.78x10^{-5}$	$8.8x10^{-4}$
Kr-85m	$4.09x10^{2}$	$2.80x10^{2}$	-	-	-
Kr-85	$3.13.10^{2}$	$2.61x10^{3}$	-	-	-
Kr-87	$6.97x10^{1}$	$1.41x10^{1}$	-	-	-
Kr-88	$4.80x10^{2}$	$2.10x10^{2}$	-	-	-
Rb-88	-	-	0.486	$8.63x10^{-4}$	0.243
Rb-89	-	-	$1.2x10^{-2}$	$1.85x10^{-5}$	$6x10^{-3}$
Sr-89	-	-	$1.08x10^{-3}$	$7.68x10^{-3}$	$5.4x10^{-4}$
Sr-90	-	-	$5.47x10^{-5}$	$1.94x10^{-3}$	$2.74x10^{-5}$
Y-90	-	-	$2.20x10^{-4}$	$4.05x10^{-5}$	$2.2x10^{-4}$
Sr-91	-	-	$7.04x10^{-4}$	-	$3.52x10^{-4}$
Y-91	-	-	$5.49x10^{-2}$	-	$5.49x10^{-2}$
Mo-99	-	-	0.430	-	0.430
Ru-103	-	-	$8.84x10^{-4}$	$5.02x10^{-3}$	$8.84x10^{-5}$
Ru-106	-	-	$5.19x10^{-5}$	$1.34x10^{-3}$	$5.19x10^{-6}$
Te-129	-	-	$5.24x10^{-3}$	$3.48x10^{-5}$	$5.24x10^{-4}$
I-129	-	-	$1.52x10^{-8}$	$5.42x10^{-7}$	$1.52x10^{-9}$
I-131	-	-	0.842	$9.63x10^{-1}$	$8.42x10^{-2}$
Xe-131m	$1.80x10^{3}$	$1.37x10^{4}$	-	-	-
Te-132	-	-	$6.81x10^{-2}$	$3.16x10^{-2}$	$6.81x10^{-3}$
I-132	-	-	0.214	$2.92x10^{-3}$	$2.14x10^{-2}$
I-133	-	-	1.15	$1.43x10^{-2}$	$1.15x10^{-1}$
Xe-133	$1.71x10^{5}$	$1.17x10^{6}$	-	-	-

TABLE 11.2-1 SOURCE TERMS FOR SHIELDING DESIGNS

Isotope	Waste Gas Surge Tank	Waste Gas Decay Tank	Aerated Waste Drain Tank	Aerated Waste Demineralizer	Aerated Waste Monitor Tank
Te-134	-	-	4.98×10^{-3}	2.08×10^{-5}	4.98×10^{-4}
I-134	-	-	0.119	6.23×10^{-4}	1.19×10^{-2}
Cs-134	-	-	0.861	-	0.430
I-135	-	-	0.530	2.11×10^{-2}	5.3×10^{-2}
Xe-135	3.41×10^{2}	4.86×10^{2}	-	-	-
Cs-136	-	-	3.03×10^{-2}	-	1.52×10^{-2}
Cs-137	-	-	4.18	-	2.09
Xe-138	6.75	2.99×10^{-1}	-	-	-
Cs-138	-	-	1.31×10^{-1}	-	6.6×10^{-2}
Ba-140	-	-	1.30×10^{-3}	2.35×10^{-3}	1.30×10^{-4}
La-140	-	-	1.25×10^{-3}	2.96×10^{-4}	1.25×10^{-4}
Pr-143	-	-	1.14×10^{-3}	-	1.14×10^{-4}
Ce-143	-	-	8.14×10^{-4}	-	8.14×10^{-5}
Co-60	-	-	2.08×10^{-4}	-	2.08×10^{-5}
Fe-59	-	-	4.03×10^{-6}	-	4.03×10^{-7}
Co-58	-	-	1.50×10^{-3}	-	1.50×10^{-4}
Mn-56	-	-	4.35×10^{-3}	-	4.35×10^{-4}
Mn-54	-	-	5.29×10^{-6}	-	5.20×10^{-7}
Cr-51	-	-	4.54×10^{-4}	-	4.54×10^{-3}
Zr-95	-	-	3.60×10^{-6}	-	3.6×10^{-7}

TABLE 11.2-1 SOURCE TERMS FOR SHIELDING DESIGNS

Isotope	Primary Demineralizer	Coolant Waste Receiver	Secondary Demineralizer	Coolant Waste Monitor Tanks	Primary Drain Tank
Br-84	4.87×10^{-1}	0.106	4.87×10^{-2}	5.30×10^{-3}	2.65×10^{-1}
Kr-85m	-	-	-	-	-
Kr-85	-	-	-	-	-
Kr-87	-	-	-	-	-
Kr-88	-	-	-	-	-
Rb-88	4.87×10^{-1}	5.84	4.87×10^{-2}	2.92×10^{-2}	$1.50 \times 10^{+1}$
Rb-89	1.04×10^{-2}	0.145	1.04×10^{-3}	7.24×10^{-3}	3.62×10^{-1}
Sr-89	4.34	1.30×10^{-2}	4.34×10^{-2}	6.48×10^{-4}	3.24×10^{-2}
Sr-90	1.10	6.56×10^{-4}	1.10×10^{-1}	3.28×10^{-5}	1.64×10^{-3}
Y-90	4.71×10^{-2}	2.63×10^{-2}	4.71×10^{-3}	1.32×10^{-3}	6.59×10^{-3}
Sr-91	2.28×10^{-2}	8.45×10^{-3}	2.28×10^{-3}	4.22×10^{-4}	2.11×10^{-2}
Y-91	2.50×10^{2}	6.59	2.50×10^{-1}	3.29×10^{-1}	1.65
Mo-99	9.47×10^{1}	5.16×10^{1}	9.47	2.58	1.29×10^{1}
Ru-103	2.83	1.06×10^{-2}	2.83×10^{-1}	5.30×10^{-4}	2.65×10^{-2}
Ru-106	7.59×10^{-1}	6.22×10^{-4}	7.59×10^{-2}	3.11×10^{-4}	1.56×10^{-3}
Te-129	1.96×10^{-2}	6.29×10^{-2}	1.96×10^{-3}	3.15×10^{-3}	1.57×10^{-1}
I-129	3.07×10^{-4}	-	3.04×10^{-5}	-	-
I-131	5.43×10^{2}	1.01×10^{1}	5.43×10^{1}	5.05×10^{-1}	2.53×10^{1}
Xe-131m	-	-	-	-	-
Te-132	1.77×10^{1}	0.818	1.77	4.09×10^{-2}	2.04
I-132	1.64	2.566	1.65×10^{1}	1.28×10^{-1}	6.42
I-133	8.06×10^{1}	1.378×10^{1}	8.06	6.89×10^{-1}	3.45×10^{1}
Xe-133	-	-	-	-	-
Te-134	1.16×10^{-2}	5.97×10^{-2}	1.16×10^{-3}	2.99×10^{-3}	1.49×10^{-1}

TABLE 11.2-1 SOURCE TERMS FOR SHIELDING DESIGNS

Isotope	Primary Demineralizer	Coolant Waste Receiver	Secondary Demineralizer	Coolant Waste Monitor Tanks	Primary Drain Tank
I-134	3.52×10^{-10}	1.43	3.52×10^{-2}	7.15×10^{-2}	3.58
Cs-134	1.50×10^{4}	1.03×10^{1}	1.5×10^{3}	5.17×10^{-1}	2.58×10^{1}
I-135	3.52×10^{-1}	6.36	3.52×10^{-2}	3.18×10^{-1}	1.59×10^{1}
Xe-135	-	-	-	-	-
Cs-136	3.06×10^{1}	3.63×10^{-1}	3.06	1.82×10^{-2}	9.08×10^{-1}
Cs-137	8.38×10^{4}	5.0×10^{1}	8.38×10^{3}	2.50	1.25^{2}
Xe-138	-	-	-	-	-
Cs-138	2.34×10^{-1}	8.18×10^{-1}	2.34×10^{-2}	4.09×10^{-2}	3.92
Ba-140	1.33	1.56×10^{-2}	1.33×10^{-1}	7.79×10^{-4}	3.89×10^{-2}
La-140	1.68×10^{-1}	1.5×10^{-2}	1.68×10^{-2}	7.51×10^{-4}	3.75×10^{-2}
Pr-143	1.24	1.24×10^{-1}	1.24×10^{-1}	6.81×10^{-3}	3.41×10^{-2}
Ce-143	1.09×10^{1}	9.77×10^{-2}	1.09	4.88×10^{-3}	2.44×10^{-2}
Co-60	-	-	-	-	6.25×10^{-3}
Fe-59	-	-	-	-	1.21×10^{-4}
Co-58	-	-	-	-	4.49×10^{-2}
Mn-56	-	-	-	-	1.31×10^{-1}
Mr-54	-	-	-	-	1.56×10^{-4}
Cr-51	-	-	-	-	1.36×10^{-2}
Zr-95	-	-	-	-	1.08×10^{-4}

Withheld under 10 CFR 2.390 (d)(1)

FIGURE 11.2–1 P&ID, RADIATION ZONES AND ACCESS CONTROL NORMAL OPERATION WITH 1.0% FAILED FUEL CONTAINMENT AND AUXILIARY BUILDING - ELEVATION (-) 45 FEET 6 INCHES

Withheld under 10 CFR 2.390 (d)(1)

FIGURE 11.2–2 P&ID, RADIATION ZONES AND ACCESS CONTROL NORMAL OPERATION WITH 1.0% FAILED FUEL CONTAINMENT AND AUXILIARY BUILDING - ELEVATION (-) 29 FEET 6 INCHES

Withheld under 10 CFR 2.390 (d)(1)

FIGURE 11.2–3 P&ID, RADIATION ZONES AND ACCESS CONTROL NORMAL OPERATION WITH 1.0% FAILED FUEL CONTAINMENT AND AUXILIARY BUILDING - ELEVATION (-) 5 FEET 0 INCHES

Withheld under 10 CFR 2.390 (d)(1)

FIGURE 11.2–4 P&ID, RADIATION ZONES AND ACCESS CONTROL NORMAL OPERATION WITH 1.0% FAILED FUEL CONTAINMENT BUILDING - ELEVATION 14 FEET 6 INCHES AND 38 FEET 6 INCHES

Withheld under 10 CFR 2.390 (d)(1)

FIGURE 11.2–5 P&ID, RADIATION ZONES AND ACCESS CONTROL NORMAL OPERATION WITH 1.0% FAILED FUEL AUXILIARY BUILDING - ELEVATION 14 FEET 6 INCHES AND 25 FEET 6 INCHES

Withheld under 10 CFR 2.390 (d)(1)

FIGURE 11.2–6 P&ID, RADIATION ZONES AND ACCESS CONTROL NORMAL OPERATION WITH 1.0% FAILED FUEL TURBINE BUILDING - ELEVATION 14 FEET 6 INCHES

Withheld under 10 CFR 2.390 (d)(1)

FIGURE 11.2–7 P&ID, RADIATION ZONES AND ACCESS CONTROL NORMAL OPERATION WITH 1.0% FAILED FUEL AUXILIARY BUILDING - ELEVATION 36 FEET 6 INCHES

Withheld under 10 CFR 2.390 (d)(1)

FIGURE 11.2–8 P&ID, RADIATION ZONES AND ACCESS CONTROL NORMAL OPERATION WITH 1.0% FAILED FUEL CONTAINMENT AND AUXILIARY BUILDING - SECTION A-A

Withheld under 10 CFR 2.390 (d)(1)

FIGURE 11.2–9 P&ID, RADIATION ZONES AND ACCESS CONTROL NORMAL OPERATION WITH 1.0% FAILED FUEL CONTAINMENT AND AUXILIARY BUILDING - SECTION B-B

Withheld under 10 CFR 2.390 (d)(1)

FIGURE 11.2–10 P&ID, RADIATION ZONES AND ACCESS CONTROL NORMAL OPERATION WITH 1.0% FAILED FUEL CONTAINMENT AND AUXILIARY

FIGURE 11.2-11 NEUTRON SHIELD SEGMENT

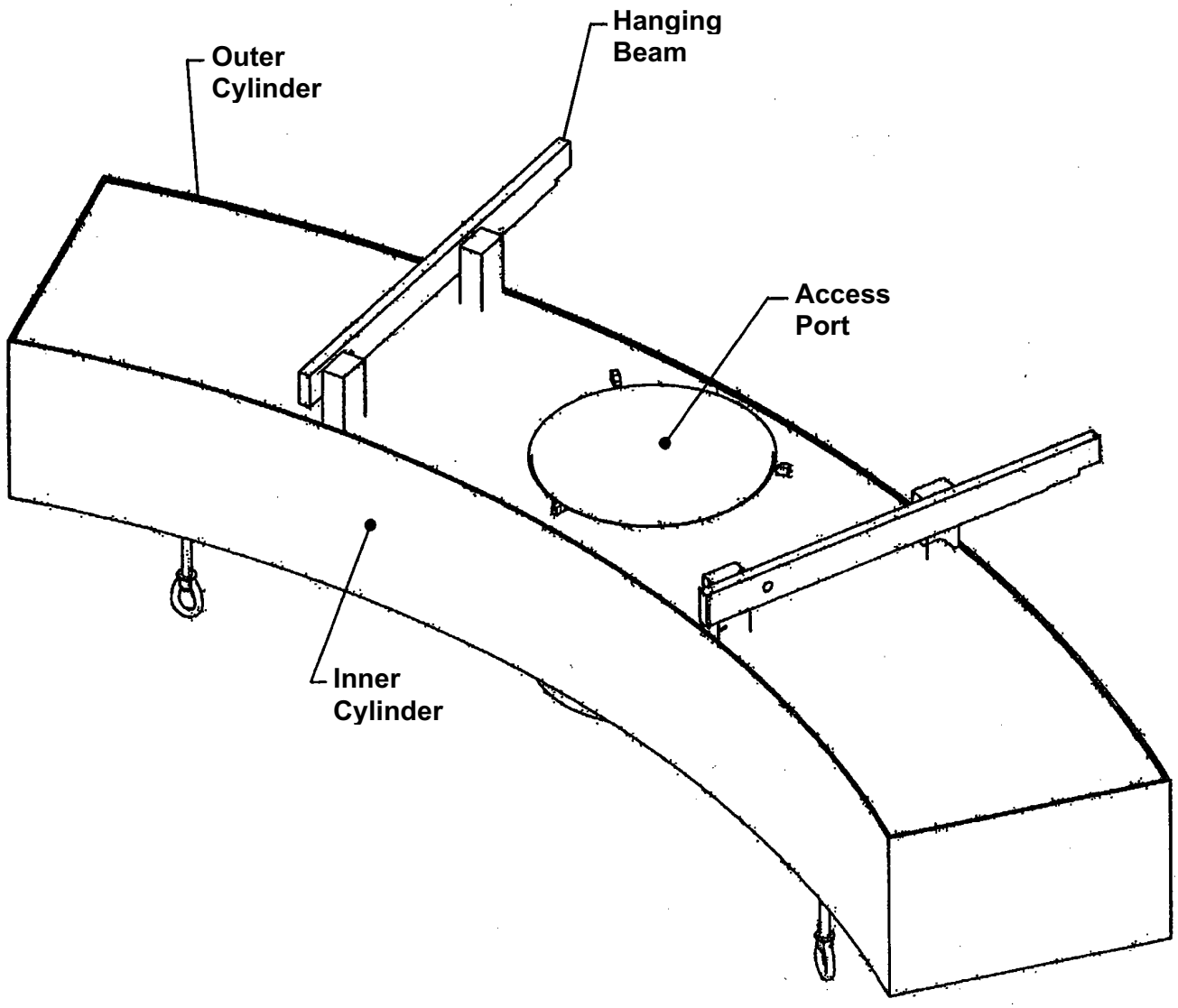


FIGURE 11.2-12 NEUTRON SHIELDING - SECTIONAL VIEW

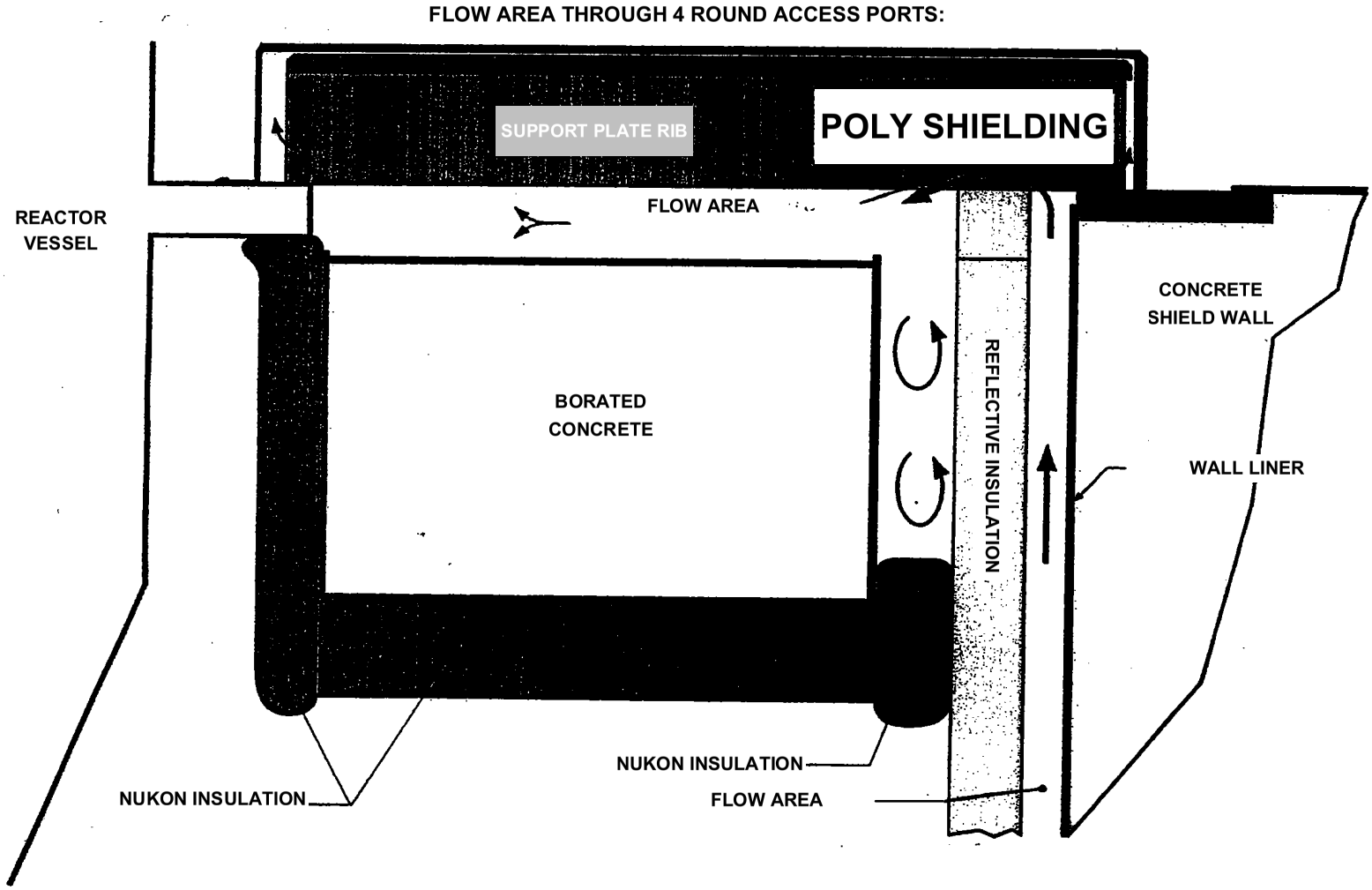


FIGURE 11.2–13 NOT USED

FIGURE 11.2–14 NOT USED

FIGURE 11.2-15 NEUTRON SHIELD - THERMAL LOADINGS (TYPICAL ANNULAR SECTION)

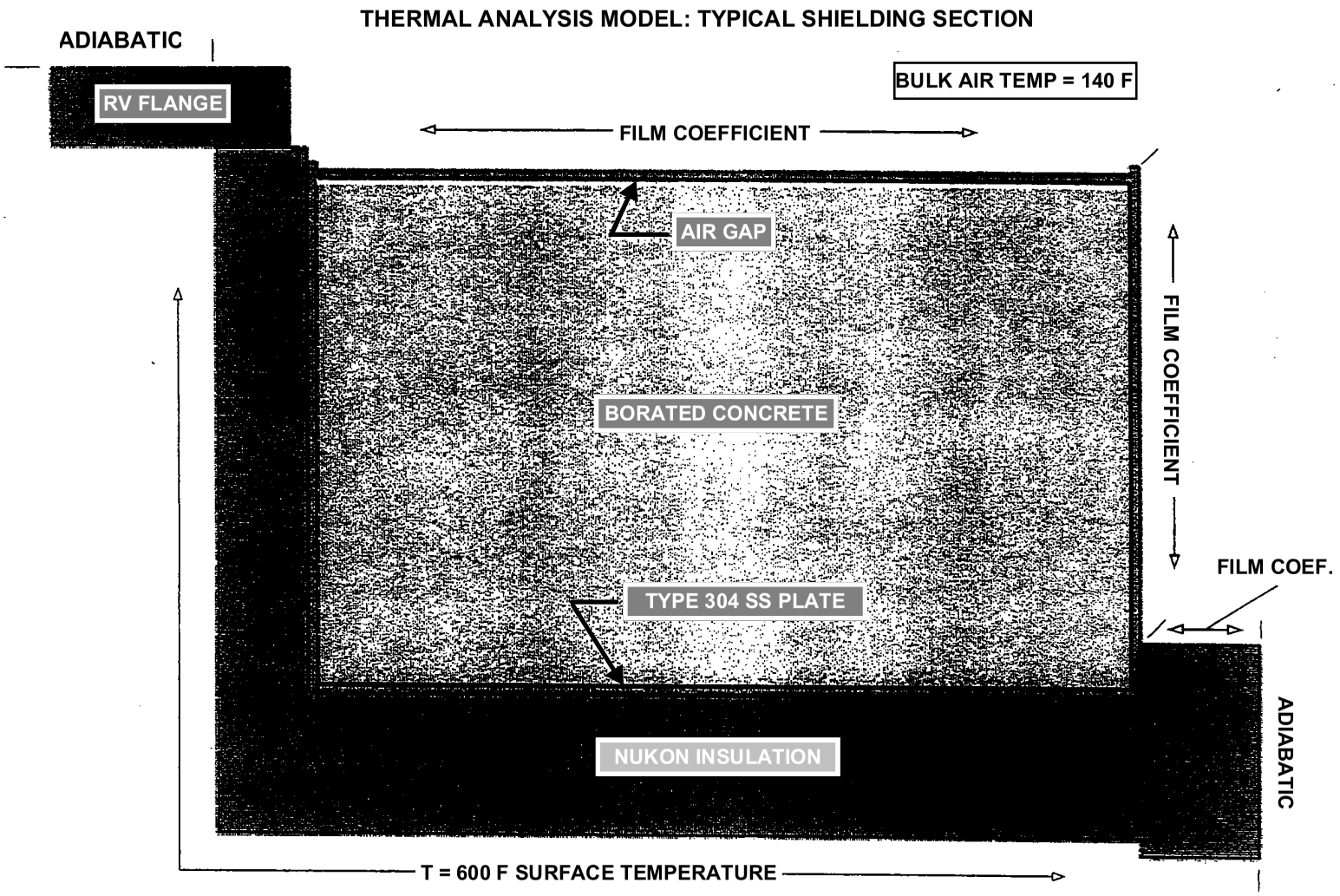
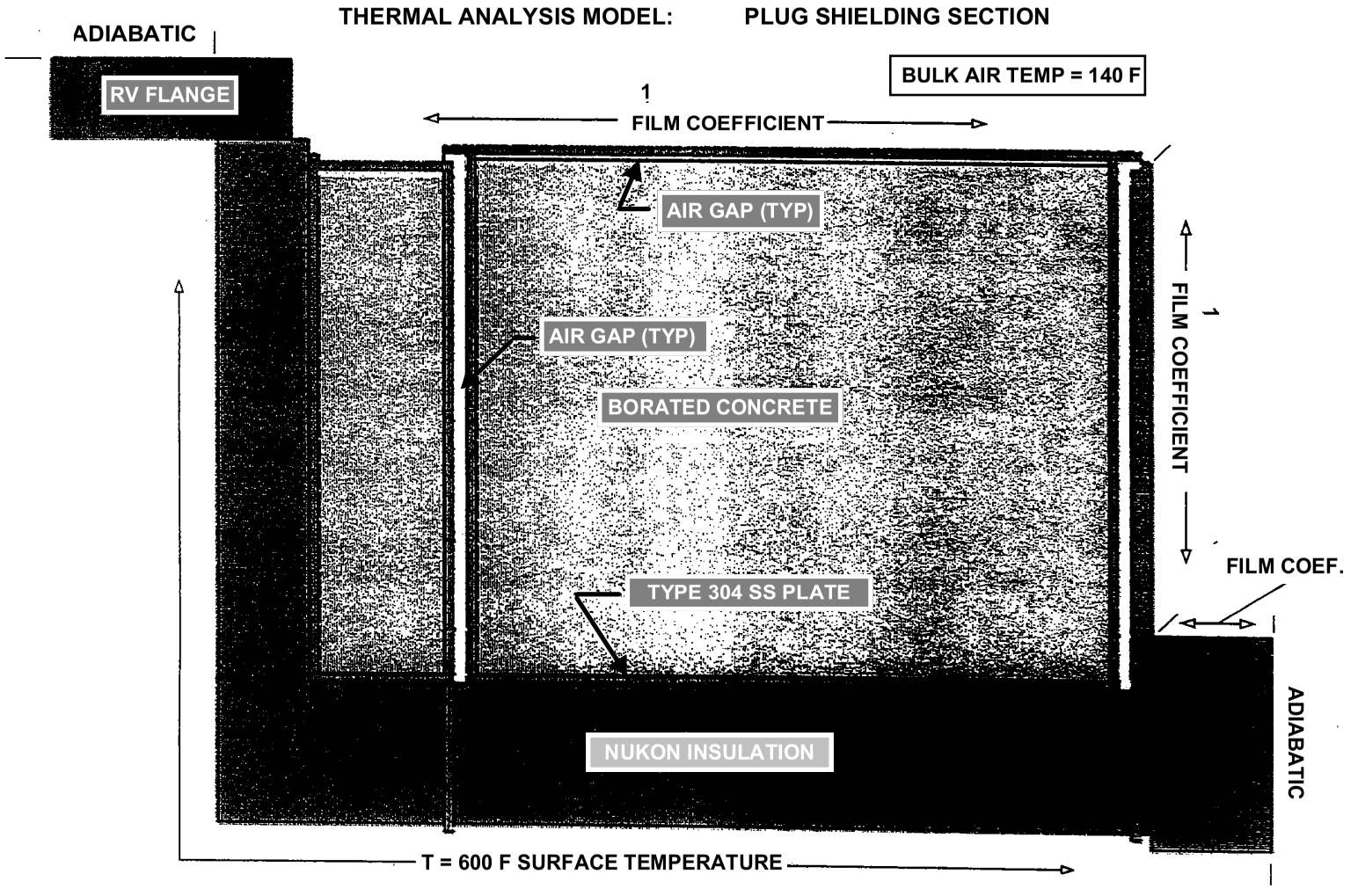


FIGURE 11.2-16 NEUTRON SHIELD - THERMAL LOADINGS (PLUG SHIELD SECTION)



11.A SOURCE TERMS FOR RADIOACTIVE WASTE PROCESSING AND RELEASES TO THE ENVIRONMENT

11.A.1 REACTOR COOLANT DESIGN BASIS RADIONUCLIDE ACTIVITIES

To calculate reactor coolant design-basis radionuclide activities, a calculation of reactor core source term (discussed in Section 11.A.1.1 below) was performed. Reactor coolant activities were calculated taking into account (a) leakage from the fuel pellets to the primary coolant due to fuel pin cladding failure, (b) removal by radioactive decay, (c) removal by the CVCS purification system, (d) removal of primary coolant for boron reduction, and (e) discharge of primary coolant to the liquid waste processing system. Included in the resultant reactor coolant design-basis radionuclide inventory are activated corrosion products (crud) and tritium (discussed in Sections 11.A.1.2 and 11.A.1.3, respectively). Table 11.A-1 provides a tabulation of the design-basis reactor coolant radionuclide activities. Bases for the calculation of these activities are presented in Table 11.A-10.

11.A.1.1 Development of Reactor Core Radionuclide Activities

For conservatism, the model used to calculate reactor core radionuclide activities employs an end-of-cycle equilibrium fuel cycle condition. Cycle characteristics are consistent with nominal two-year fuel cycles (with an 80% capacity factor included). The ORIGEN-S computer program is used to model the buildup and decay of core fission product radionuclides.

ORIGEN-S calculates the following three categories of core radionuclide activities:

1. Fission Products
2. Light Elements
3. Actinides

The light elements are neutron activation products of fuel assembly metallic structural components. The ORIGEN-S model used in the calculation of Millstone Unit 2 reactor core radionuclide activities is predicated on a standard 14-by-14 fuel pin arrangement. Actinides simulate the buildup, decay, and loss due to fissioning. Only the ^{239}Np actinide data are used.

The ORIGEN-S results are calculated in terms of gram-atoms per metric tonne of initial heavy metal. These results are converted, as necessary, to activities (in Curies) in the core using the following equation:

$$\text{Acore} = \frac{N_A \cdot \lambda}{\text{ACF}} \cdot \{M_{u1} \cdot (B1C1 + B1C2 + B1C3) + M_{u2} \cdot (B2C1 + B2C2)\}$$

where N_A is in atoms per gram-atom, λ is in seconds^{-1}, ACF is the activity conversion factor (equal to 3.7×10^{10} disintegrations/second per Curie), M_{u1} is equal to 19.208 metric tonnes of uranium in Batch 1, M_{u2} is equal to 13.72 metric tonnes of uranium in Batch 2, and where *B1C1, B1C2, B1C3, B2C1,* and *B2C2* are batch/cycle-specific values (in gram-atoms per metric tonnes of uranium) calculated by ORIGEN-S.

The reactor core radionuclide activities thus calculated and used as basic source terms are tabulated in Table 11.A-2. Also presented in Table 11.A-2 are the bases for the calculation of these reactor core radionuclide activities.

11.A.1.2 Corrosion Products

The activity concentrations of activated corrosion products (crud) in the reactor coolant have been calculated based on the model in the ANSI/ANS standard[1] dealing with the radioactive source term for normal operation of light water reactors. The starting point for this calculation was the reactor coolant radionuclide activity concentrations specified in the standard for a reference PWR with U-tube steam generators (such as those used at Millstone Unit 2). These activity concentrations were altered by adjustment factors that were prepared in accordance with the ANSI/ANS model to reflect the operating parameter differences between Millstone Unit 2 and the reference PWR, and were then further adjusted to the Technical Specification radionuclide concentration limits.

11.A.1.3 Tritium Production

Tritium may be produced in the coolant or enter the coolant from a number of sources. One source is from fissioning of uranium within the fuel, yielding tritium as a tertiary fission product. Since zircaloy fuel cladding reacts with tritium to form zircaloy hydride, no tritium diffuses through the cladding [2, 3]. Therefore, the tritium released to the coolant from the fuel originates only from defective fuel.

Tritium is also produced by the reaction of neutrons with boron in the control element assemblies (CEAs). Data from operating plants using B_4C control rods indicates that no tritium is released from the control rods. The tritium may combine with carbon to form hydrocarbons and/or with lithium to form lithium hydride, thereby preventing diffusion through the inconel cladding. Another possibility is that the low internal temperature of the B_4C control rods (relative to

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1 ANSI/ANS-18.1-1984, American National Standard — Radioactive Source Term for Normal Operation of Light Water Reactors, dated 12/31/1984

2 James M. Smith, Jr., The Significance of Tritium in Water Reactors, GE, APED, 9/19/67

3 Joseph W. Ray, et. al., Investigation of Tritium Generation and Release in PM Nuclear Power Plants, BMI-1787, 10/31/66

stainless clad fuel rods from which about 45% escape can be expected) may prohibit tritium diffusion. To account for possible control rod cladding defects, it is assumed that one percent of the tritium produced in the CEAs is released to the coolant.

Another source of tritium is the activation of boron, lithium, deuterium, and nitrogen within the reactor coolant. Boron in the form of boric acid is used in the coolant for reactivity control. Lithium is produced in the coolant as a result of neutron-boron reaction and may also be added as a pH control agent. Deuterium is a naturally occurring constituent of water. Nitrogen may be present due to aeration of the coolant during shutdown and due to aerated makeup water.

The expected tritium releases from the combined liquid and vapor pathways was assumed to be 0.4 Ci/year per MWt, based on review of the tritium release rates at a number of PWRs evaluated in NUREG-0017, Rev. 1. The quantity of tritium released in the liquid pathway is based on the calculated volume of liquid released, excluding secondary system wastes, with a primary coolant concentration of 1.0 μCi/ml. It was assumed that the remainder was released as a gas from building ventilation system exhaust.

11.A.1.4 Fuel Experience

Past operation of stainless steel-clad fuel rods in the Connecticut Yankee reactor showed fuel failure rates on the order of 0.01%.

Zircaloy-clad UO_2 fuel in the Obrigheim reactor in Germany sustained a fuel failure rate just over 0.1% in its first cycle, but this had fallen in the second cycle to essentially zero (0.001%). The fuel failure rate in the Dresden 1 reactor over a nine-year period had averaged < 0.1%, with the rate more recently being even lower. Fuel in the Mihama reactor in Japan and in the Point Beach reactor had exceeded the burn up at which failures in fuel of similar design were observed in Ginna, without exhibiting increases in coolant activity (indicative of fuel defects).

Fuel failure rates in the current generation of reactors can be controlled to very low levels. Widespread fuel defects in certain reactors have been observed, the cause having been attributed to fuel clad contamination. Appropriate corrective actions have been devised to ensure that the occurrence of such fuel defects will be greatly minimized in the future. Nevertheless, there is always the possibility, despite careful testing and manufacture, that other defects will become apparent in new fuel designs in the future that, because of statistical considerations or unrecognized or uncontrollable environmental differences, could not be foreseen. The design refinements continuously introduced in nuclear power reactors and their fuel as a natural outcome of a dynamic industry will, on rare occasions, introduce such defects. Existing licensing regulations limit coolant activity to that associated with 1% failed fuel, even during these transitory and infrequent periods.

The fuel failure rate attainable under more normal conditions has been demonstrated to be nearer 0.01%. Over the lifetime of an operating reactor, the latter rate is expected to predominate.

11.A.2 REACTOR COOLANT EXPECTED RADIONUCLIDE ACTIVITIES

Expected reactor coolant radionuclide activities are based on data generated from operating plants, field and laboratory tests, and plant-specific design considerations. These activities are built into the PWR-GALE Code (henceforth referred to as 'GALE'), which is a computerized mathematical model for calculating the expected releases of radioactive material in liquid and gaseous effluents from pressurized water reactors (PWRs). The expected reactor coolant radionuclide activities are tabulated in Table 11.A-1, juxtaposed with the design-basis reactor coolant radionuclide activities.

11.A.3 CALCULATION OF LIQUID AND GASEOUS EFFLUENT RELEASES

11.A.3.1 Expected Liquid and Gaseous Radioactive Effluent Releases

The Millstone Unit 2 expected liquid and gaseous radioactive effluent releases are calculated to determine compliance with 10 CFR Part 50 Appendix I. This calculation is performed using GALE. As previously stated, the calculation is based on data generated from operating reactors, field and laboratory tests, and plant-specific design considerations incorporated to reduce the quantity of radioactive materials that may be released to the environment during normal operation, including anticipated operational occurrences. The calculation performed by GALE is based on (a) American Nuclear Society (ANS) 18.1 Working Group recommendations for adjustment factors, (b) the release and transport mechanisms that result in the appearance of radioactive material in liquid and gaseous waste streams, (c) plant-specific design features used to reduce the quantities of radioactive materials that are ultimately released to the environment, and (d) information received on the operation of nuclear power plants. The principal mechanisms that affect the concentrations of radioactive materials in the primary coolant are the following:

- fission product leakage to the primary coolant from defects in the fuel cladding and fission product generation in tramp uranium,
- corrosion products activated in the core (i.e., crud),
- radioactivity removed in the reactor coolant treatment systems, and
- activity removed because of primary coolant leakage.

The descriptions of the Millstone Unit 2 liquid and gaseous waste processing systems, as well as the GALE input parameters unique to those systems, are presented in Appendix 11.B. The following tables provide tabulations of the GALE results:

- Table 11.A-3: Expected Annual Airborne Effluent Releases (Curies per Year), by Radionuclide, from Each Release Point
- Table 11.A-4: Expected Annual Liquid Activity Releases (Curies/Year), by Radionuclide, from Each Waste Stream

- Table 11.A-5: Expected Annual Liquid Effluent Concentrations (Diluted and Undiluted), by Radionuclide, from Each Liquid Waste Stream

- Table 11.A-8: Total Annual Design Basis And Expected Releases Of Radioactive Liquid Waste To The Environment From All Sources Combined, In Curies Per Year

- Table 11.A-9: Total Annual Design Basis And Expected Releases Of Airborne Radioactive Waste To The Environment From All Release Points Combined, In Curies Per Year

11.A.3.2 Design Basis Liquid and Gaseous Radioactive Effluent Releases

The Millstone Unit 2 design basis liquid and gaseous radioactive effluent releases are calculated to determine compliance with 10 CFR Part 20. To perform this calculation, the design basis primary coolant activity concentrations are used, in conjunction with their corresponding expected primary coolant activity concentrations, to determine isotopic scaling factors. These scaling factors are then applied to the expected liquid and gaseous effluent release inventory, as calculated by GALE, in order to determine the design basis liquid and gaseous radioactive effluent releases.

The calculated design basis liquid and gaseous radioactive effluent releases are provided in the following tables:

- Table 11.A-6: Design Basis Radionuclide Concentrations in Liquid Effluent, in Fractions of 10 CFR Part 20 Concentration Limits

- Table 11.A-7: Design Basis Radionuclide Airborne Concentrations at the Site Boundary from All Gaseous Effluent Release Points Combined, in Fractions of 10 CFR Part 20 Concentration Limits

- Table 11.A-8: Total Annual Design Basis And Expected Releases Of Radioactive Liquid Waste To The Environment From All Sources Combined, In Curies Per Year

- Table 11.A-9: Total Annual Design Basis And Expected Releases Of Airborne Radioactive Waste To The Environment From All Release Points Combined, In Curies Per Year

11.A.4 SOLID WASTE PROCESSING SYSTEM

The information in this section is based on analyses done for the original licensing of Millstone Unit 2 and is not derived from the updated radionuclide activities shown in the Appendices to FSAR Chapter 11.

11.A.4.1 Spent Resins

11.A.4.1.1 Spent Resins from CVCS Ion Exchanger

The radioactivity buildup on the CVCS ion exchangers is based on a continuous letdown flow of 40 gpm. One of the two purification ion exchangers and the deborating ion exchanger will be sluiced to the solid waste processing system at the end of each core cycle. Each of the two purification ion exchangers operates for two cycles, one cycle as a Li removal exchanger and the next cycle as the continuous purification exchanger, such that one of the two exchangers will be on the Li removal cycle while the other is for continuous purification.

The decontamination factor for ion exchangers is assumed to be 10 for all soluble isotopes except Y, Mo, and Cs. For Y, Mo, and Cs, a removal factor of 10 with a 20% usage factor is used for the Li removal cycle for each purification ion exchanger.

The buildup of activity on the CVCS ion exchangers' resins (32 ft^3 each) is given in Table 11.A-3.

11.A.4.1.2 Spent Resins from Clean Liquid Waste Processing System Demineralizers

Activity buildup on the clean liquid waste processing system demineralizers is based on the processing of 14 system volumes per year of reactor coolant wastes. An average processing rate based on annual volumes of liquid wastes is assumed, with a decontamination factor of 10^3 for soluble isotopes. Operation is assumed to be divided equally between the two primary demineralizers. The loading on the secondary demineralizer resin bed(s) will be much less than that on the primary demineralizers and has not been tabulated.

The activity buildup on the clean liquid waste processing system demineralizers' resins (42 ft^3 each) is given in Table 11.1-3.

11.A.4.1.3 Spent Resins from Aerated Liquid Waste Processing System Demineralizer

Activity buildup on the aerated liquid waste demineralizer resin is based on the processing of the volumes of aerated liquid radwaste generated annually. An average processing rate based on annual volumes of liquid waste is assumed, with a total DF of 500 taken for soluble isotopes.

The resin activity buildup on the hard piped demineralizer resin (42 ft^3) is given in Table 11.1-3.

The resin activity buildup on the portable demineralizers (15 ft^3 each) is not addressed, since this activity is bounded by the resin activity buildup on hard-piped demineralizer.

11.A.4.1.4 Spent Resins from Spent Fuel Pool Demineralizer

The activity buildup on resins from the spent fuel demineralizer is based on the processing of spent fuel pool water and refueling water with a radioactivity level corresponding to one-tenth of reactor coolant. The assumption is made that complete mixing of reactor coolant, spent fuel pool

water, and refueling water occurs during refueling operations, resulting in a dilution of the reactor coolant by a factor of approximately 10. An expected DF of 1000 is used for the demineralizer. The activity buildup on the demineralizer resin (42 ft^3) is given in Table 11.1-3.

11.A.4.1.5 Contaminated Filter Cartridges

Buildup of activity on filter cartridges is based on processing of liquid radioactive waste. A decontamination factor (DF) of 10 is taken for each filter. This DF is consistent with operating experience from nuclear power stations utilizing three micron filter cartridges.

The activity buildup on the filter cartridges is given in Table 11.1-3.

TABLE 11.A-1 DESIGN-BASIS AND EXPECTED PRIMARY COOLANT ACTIVITY CONCENTRATIONS

Noble Gases

NUCLIDE	DESIGN-BASIS PRIMARY COOLANT (μCi/gm)	EXPECTED PRIMARY COOLANT (μCi/gm)
Kr-85m	1.33E+00	1.40E-01
Kr-85	4.50E+00	2.12E-02
Kr-87	7.64E-01	1.41E-01
Kr-88	2.36E+00	2.53E-01
Xe-131m	2.32E+00	1.17E-01
Xe-133m	3.39E+00	3.03E-02
Xe-133	2.17E+02	6.88E-01
Xe-135m	1.25E-01	1.25E-01
Xe-35	6.13E+00	6.72E-01
Xe-137	1.38E-01	3.28E-02
Xe-138	4.81E-01	1.15E-01

Halogens

NUCLIDE	DESIGN-BASIS PRIMARY COOLANT (μCi/gm)	EXPECTED PRIMARY COOLANT (μCi/gm)
I-131	5.57E+00	4.51E-02
I-132	1.18E+00	2.04E-01
I-133	7.29E+00	1.39E-01
I-134	7.77E-01	3.30E-01
I-135	3.65E+00	2.55E-01
Br-84	4.84E-02	1.55E-02

Activated Corrosion Products (Crud)

NUCLIDE	DESIGN-BASIS PRIMARY COOLANT (μCi/gm)	EXPECTED PRIMARY COOLANT (μCi/gm)
Na-24	1.40E+00	4.63E-02
Cr-51	9.50E-02	3.09E-03
Mn-54	4.90E-02	1.60E-03
Fe-55	3.70E-02	1.20E-03
Fe-59	9.20E-03	2.99E-04
Co-58	1.40E-01	4.59E-03
Co-60	1.60E-02	5.29E-04
Zn-65	1.60E-02	5.09E-04
W-187	7.60E-02	2.47E-03
Np-239	6.70E-02	2.19E-03

Other Particulate Radionuclides

NUCLIDE	DESIGN-BASIS PRIMARY COOLANT (μCi/gm)	EXPECTED PRIMARY COOLANT (μCi/gm)
Rb-88	5.04E-02	1.84E-01
Sr-89	3.74E-03	1.40E-04
Sr-90	3.83E-04	1.20E-05
Sr-91	1.83E-03	9.42E-04
Y-91m	1.43E-04	4.46E-04
Y-91	4.83E-03	5.19E-06
Y-93	1.53E-03	4.12E-03
Zr-95	1.20E-02	3.89E-04
Nb-95	6.39E-03	2.79E-04
Mo-99	8.11E-01	6.36E-03
Tc-99m	1.77E-03	4.60E-03
Ru-103	5.57E-03	7.49E-03
Ru-106	1.94E-03	8.99E-02

Other Particulate Radionuclides

NUCLIDE	DESIGN-BASIS PRIMARY COOLANT (μCi/gm)	EXPECTED PRIMARY COOLANT (μCi/gm)
Ag-110m	1.28E-05	1.30E-03
Te-129m	2.20E-02	1.90E-04
Te-129	7.99E-03	2.33E-02
Te-131m	4.75E-02	1.48E-03
Te-131	8.51E-03	7.45E-03
Te-132	4.46E-01	1.69E-03
Cs-134	1.92E+00	7.56E-03
Cs-136	5.10E-01	9.21E-04
Cs-137	1.58E+00	1.00E-02
Ba-140	6.49E-03	1.30E-02
La-140	5.14E-03	2.48E-02
Ce-141	6.16E-03	1.50E-04
Ce-143	4.02E-03	2.77E-03
Ce-144	4.93E-03	3.89E-03

TABLE 11.A-2 CALCULATED REACTOR CORE ACTIVITIES

Nobel Gases

Nuclide	Reactor Core Activities (Curies) [1]
Kr-85m	1.859E+07
Kr-85	8.557E+05
Kr-87	3.736E+07
Kr-88	5.193E+07
Xe-131m	9.452E+05
Xe-133m	4.636E+06
Xe-133	1.482E+08
Xe-135m	3.021E+07
Xe-135	4.288E+07
Xe-137	1.344E+08
Xe-138	1.266E+08

Halogens

Nuclide	Reactor Core Activities (Curies) [1]
Br-84	1.658E+07
I-131	7.120E+07
I-132	1.036E+08
I-133	1.479E+08
I-134	1.644E+08
I-135	1.400E+08

Cesium and Rubidium

Nuclide	Reactor Core Activities (Curies) [1]
Rb-88	5.320E+07
Cs-134	1.202E+07
Cs-136	3.475E+06
Cs-137	9.888E+06

Other Nuclides

Nuclide	Reactor Core Activities (Curies) [1]
Cr-51	2.413E+06
Mn-54	9.754E+04
Fe-55	4.496E+05
Fe-59	3.382E+04
Co-58	8.739E+05
Co-60	8.888E+05
Zn-65	6.496E-02
Np-239	1.347E+09
Sr-89	7.321E+07
Sr-90	7.406E+06
Sr-91	9.077E+07
Y-91	9.435E+07
Y-91m	5.255E+07
Y-93	7.297E+07
Zr-95	1.250E+08
Nb-95	1.255E+08
Mo-99	1.343E+08
Tc-99m	1.188E+08
Ru-103	1.093E+08
Ru-106	3.752E+07

Other Nuclides

Nuclide	Reactor Core Activities (Curies) [1]
Ag-110m	2.479E+05
Te-129	2.136E+07
Te-129m	4.324E+06
Te-131	6.042E+07
Te-131m	1.373E+07
Te-132	1.025E+08
Ba-140	1.314E+08
La-140	1.359E+08
Ce-141	1.213E+08
Ce-143	1.127E+08
Ce-144	9.545E+07

1. The reactor core radionuclide activities tabulated above are based on the following: (a) a core power level of 2,700 MWt and (b) a three-region equilibrium cycle core, with an end-of-cycle core average burn-up of 36,142 MWD/MTU, the three regions having operated at a specific power of 31.74 MWt/MTU for 705, 1333 and 1550 EFPD, respectively.

TABLE 11.A-3 EXPECTED ANNUAL EFFLUENT RELEASES (CURIES PER YEAR), BY RADIONUCLIDE, FROM EACH RELEASE POINT

Release Points: Iodines

	Turbine Building (Ci/yr)	Unit 2 Vent (Ci/yr)	Millstone Stack (Ci/yr)
I-131	0.00E+00	1.77E-01	1.00E-03
I-133	1.80E-04	5.51E-01	1.00E-03
Totals	1.80E-04	7.28E-01	2.00E-03

Release Points: Noble Gases

	Turbine Building (Ci/yr)	Unit 2 Vent (Ci/yr)	Millstone Stack (Ci/yr)
Kr-85m	0.00E+00	3.00E+00	1.00E+00
K-r85	0.00E+00	3.80E+01	5.60E+02
Kr-87	0M.00E+00	3.00E+00	1.00E+00
Kr-88	0.00E+00	5.00E+00	4.00E+00
Xe-131m	0.00E+00	2.70E+01	2.30E+01
Xe-133m	0.00E+00	1.00E+00	2.00E+00
Xe-133	0.00E+00	8.10E+01	7.10E+01
Xe-135m	0.00E+00	3.00E+00	1.00E+00
Xe-135	0.00E+00	1.90E+01	1.20E+01
Xe-137	0.00E+00	0.00E+00	0.00E+00
Xe-138	0.00E+00	2.00E+00	1.00E+00
Totals	0.00E+00	1.82E+02	6.76E+02

Release Points: Others (Totals Allocated to Unit 2 Vent Release for Conservatism)

	Turbine Building (Ci/yr)	Unit 2 Vent (Ci/yr)	Millstone Stack (Ci/yr)
H-3	none	1.10E+02	none
C-14	none	7.30E+00	none
Ar-41	none	3.40E+01	none
Totals	none	1.51E+02	none

Release Points: Particulates

	Turbine Building (Ci/yr)	Unit 2 Vent (Ci/yr)	Millstone Stack (Ci/yr)
Cr-51	none	9.70E-05	1.40E-05
Mn-54	none	5.68E-05	2.10E-06
Co-57	none	8.20E-06	0.00E+00
Co-58	none	4.79E-04	8.70E-06
Co-60	none	1.13E-04	1.40E-05
Fe-59	none	2.75E-05	1.80E-06
Sr-89	none	1.59E-04	4.40E-05
Sr-90	none	6.29E-05	1.70E-05
Zr-95	none	1.00E-05	4.80E-06
Nb-95	none	4.23E-05	3.70E-06
Ru-103	none	1.66E-05	3.20E-06
Ru-106	none	7.50E-07	2.70E-06
Sb-125	none	6.09E-07	0.00E+00
Cs-134	none	4.74E-05	3.30E-05
Cs-136	none	3.25E-05	5.30E-06
Cs-137	none	8.92E-05	7.70E-05
Ba-140	none	4.00E-06	2.30E-05
Ce-141	none	1.33E-05	2.20E-06
Totals	none	1.26E-03	2.57E-04

TABLE 11.A-4 EXPECTED ANNUAL LIQUID EFFLUENT ACTIVITY RELEASES (CURIES/YEAR), BY RADIONUCLIDE, FROM EACH WASTE STREAM

NUCLIDE	BORON RS (Ci/yr)	MISC. WASTES (Ci/yr)	SECONDARY (Ci/yr)	TURB. BLDG. (Ci/yr)	DETERGENT (Ci/yr)
Corrosion and Activation Products:					
Na-24	2.46E-04	9.26E-03	4.62E-01	5.94E-05	0.00E+00
Cr-51	3.39E-04	1.96E-03	3.81E-02	6.23E-06	4.70E-04
Mn-54	2.11E-04	1.04E-03	1.94E-02	3.13E-06	3.80E-04
Fe-55	1.61E-04	7.84E-04	1.47E-02	2.36E-06	7.20E-04
Fe-59	3.55E-05	1.92E-04	3.54E-03	5.76E-07	2.20E-04
Co-58	5.71E-04	2.97E-03	5.63E-02	9.13E-06	7.90E-04
Co-60	7.12E-05	3.47E-04	6.59E-03	1.06E-06	1.40E-03
Zn-65	6.71E-05	3.32E-04	6.27E-03	1.01E-06	0.00E+00
W-187	2.36E-05	7.16E-04	2.61E-02	3.72E-06	0.00E+00
Np-239	5.48E-05	9.81E-04	2.45E-02	3.85E-06	0.00E+00
Fission Products:					
Br-84	4.64E-10	1.90E-05	3.21E-02	4.26E-09	0.00E+00
Rb-88	3.25E-10	1.44E-03	2.42E-01	3.36E-11	0.00E+00
Sr-89	1.69E-05	8.99E-05	1.68E-03	2.74E-07	8.80E-06
Sr-90	1.62E-06	7.85E-06	1.47E-04	2.36E-08	1.30E-06
Sr-91	2.71E-06	1.23E-04	8.95E-03	9.88E-07	0.00E+00
Y-91m	1.75E-06	7.86E-05	1.37E-03	6.33E-07	0.00E+00
Y-91	1.45E-06	6.86E-06	6.47E-05	1.35E-08	8.40E-06
Y-93	1.29E-05	5.68E-04	3.82E-02	4.31E-06	0.00E+00
Zr-95	4.80E-05	2.51E-04	4.74E-03	7.68E-07	1.10E-04
Nb-95	3.95E-05	1.85E-04	3.32E-03	5.31E-07	1.90E-04
Mo-99	1.90E-04	3.02E-03	7.26E-02	1.15E-05	6.00E-06
Tc-99m	1.80E-04	3.02E-03	3.73E-02	8.30E-06	0.00E+00
Ru-103	8.72E-04	4.79E-03	9.12E-02	1.49E-05	2.90E-05
Ru-106	1.19E-02	5.87E-02	1.11E+00	1.78E-04	8.90E-04
Ag-110m	1.71E-04	8.47E-04	1.58E-02	2.55E-06	1.20E-04
Te-29m	2.16E-05	1.21E-04	2.29E-03	3.74E-07	0.00E+00

TABLE 11.A-4 EXPECTED ANNUAL LIQUID EFFLUENT ACTIVITY RELEASES (CURIES/YEAR), BY RADIONUCLIDE, FROM EACH WASTE STREAM

NUCLIDE	BORON RS (Ci/yr)	MISC. WASTES (Ci/yr)	SECONDARY (Ci/yr)	TURB. BLDG. (Ci/yr)	DETERGENT (Ci/yr)
Te-29	1.40E-05	2.47E-04	9.12E-02	6.49E-07	0.00E+00
Te-131m	1.85E-05	4.97E-04	1.60E-02	2.36E-06	0.00E+00
Te-131	3.38E-06	9.50E-05	1.31E-02	4.32E-07	0.00E+00
I-131	2.91E-03	2.64E-02	4.51E-01	1.43E-04	1.60E-04
Te-132	5.87E-05	8.39E-04	1.91E-02	3.06E-06	0.00E+00
I-132	8.64E-05	5.45E-03	1.07E+00	6.24E-05	0.00E+00
I-133	1.04E-03	3.66E-02	1.24E+00	3.42E-04	0.00E+00
I-134	5.82E-07	1.41E-03	9.69E-01	2.92E-06	0.00E+00
Cs-134	1.14E-01	2.47E-01	9.36E-02	1.54E-05	1.10E-03
I-135	3.76E-04	2.22E-02	1.91E+00	3.48E-04	0.00E+00
Cs-136	9.16E-03	2.81E-02	1.11E-02	1.84E-06	3.70E-05
Cs-137	1.52E-01	3.28E-01	1.25E-01	2.05E-05	1.60E-03
Ba-140	1.14E-03	7.90E-03	1.51E-01	2.48E-05	9.10E-05
La-140	1.48E-03	1.29E-02	2.76E-01	4.42E-05	0.00E+00
Ce-141	1.69E-05	9.53E-05	1.79E-03	2.92E-07	2.30E-05
Ce-143	3.86E-05	9.81E-04	2.96E-02	4.42E-06	0.00E+00
Ce-144	5.15E-04	2.54E-03	4.78E-02	7.70E-06	3.90E-04
All Others Except Tritium	1.56E-01	3.73E-01	6.69E-02	2.20E-04	2.31E-04
TOTAL	4.54E-01	1.19E+00	8.90E+00	1.56E-03	8.98E-03

TABLE 11.A-5 EXPECTED ANNUAL LIQUID EFFLUENT CONCENTRATIONS (DILUTED AND UNDILUTED), BY RADIONUCLIDE, FROM EACH WASTE SYSTEM

NUCLIDE	BORON RS, UNDILUTED (μCi/ml)	MISC. WASTES, UNDILUTED (μCi/ml)	SECONDARY, UNDILUTED (μCi/ml)	TURB. BLDG., UNDILUTED (μCi/ml)	DETERGENT, UNDILUTED (μCi/ml)	BORON RS, DILUTED (μCi/ml)	MISC. WASTES, DILUTED (μCi/ml)	SECONDARY, DILUTED (μCi/ml)	TURB. BLDG., DILUTED (μCi/ml)	DETERGENT, DILUTED (μCi/ml)
Corrosion and Activation Products:										
Na-24	5.45E-08	6.04E-06	1.57E-06	5.97E-09	0.00E+00	2.64E-13	9.94E-12	4.96E-10	6.38E-14	0.00E+00
Cr-51	7.50E-08	1.28E-06	1.30E-07	6.26E-10	6.30E-07	3.64E-13	2.10E-12	4.09E-11	6.69E-15	5.05E-13
Mn-54	4.67E-08	6.78E-07	6.60E-08	3.15E-10	5.09E-07	2.27E-13	1.12E-12	2.08E-11	3.36E-15	4.08E-13
Fe-55	3.56E-08	5.11E-07	5.00E-08	2.37E-10	9.65E-07	1.73E-13	8.42E-13	1.58E-11	2.53E-15	7.73E-13
Fe-59	7.86E-09	1.25E-07	1.20E-08	5.79E-11	2.95E-07	3.81E-14	2.06E-13	3.80E-12	6.18E-16	2.36E-13
Co-58	1.26E-07	1.94E-06	1.91E-07	9.18E-10	1.06E-06	6.13E-13	3.19E-12	6.04E-11	9.80E-15	8.48E-13
Co-60	1.58E-08	2.26E-07	2.24E-08	1.07E-10	1.88E-06	7.65E-14	3.73E-13	7.07E-12	1.14E-15	1.50E-12
Zn-65	1.49E-08	2.16E-07	2.13E-08	1.02E-10	0.00E+00	7.20E-14	3.56E-13	6.73E-12	1.08E-15	0.00E+00
W-187	5.22E-09	4.67E-07	8.88E-08	3.74E-10	0.00E+00	2.53E-14	7.69E-13	2.80E-11	3.99E-15	0.00E+00
Np-239	1.21E-08	6.40E-07	8.33E-08	3.87E-10	0.00E+00	5.88E-14	1.05E-12	2.63E-11	4.13E-15	0.00E+00
Fission Products:										
Br-84	1.03E-13	1.24E-08	1.09E-07	4.28E-13	0.00E+00	4.98E-19	2.04E-14	3.45E-11	4.57E-18	0.00E+00
Rb-88	7.19E-14	9.39E-07	8.23E-07	3.38E-15	0.00E+00	3.49E-19	1.55E-12	2.60E-10	3.61E-20	0.00E+00
Sr-89	3.74E-09	5.86E-08	5.71E-09	2.75E-11	1.18E-08	1.81E-14	9.65E-14	1.80E-12	2.94E-16	9.45E-15
Sr-90	3.59E-10	5.12E-09	5.00E-10	2.37E-12	1.74E-09	1.74E-15	8.43E-15	1.58E-13	2.53E-17	1.40E-15
Sr-91	6.00E-10	8.02E-08	3.04E-08	9.93E-11	0.00E+00	2.91E-15	1.32E-13	9.61E-12	1.06E-15	0.00E+00
Y-91m	3.87E-10	5.13E-08	4.66E-09	6.36E-11	0.00E+00	1.88E-15	8.44E-14	1.47E-12	6.80E-16	0.00E+00
Y-91	3.21E-10	4.47E-09	2.20E-10	1.36E-12	1.13E-08	1.56E-15	7.37E-15	6.95E-14	1.45E-17	9.02E-15
Y-93	2.86E-09	3.70E-07	1.30E-07	4.33E-10	0.00E+00	1.39E-14	6.10E-13	4.10E-11	4.63E-15	0.00E+00
Zr-95	1.06E-08	1.64E-07	1.61E-08	7.72E-11	1.47E-07	5.15E-14	2.70E-13	5.09E-12	8.25E-16	1.18E-13
Nb-95	8.74E-09	1.21E-07	1.13E-08	5.34E-11	2.55E-07	4.24E-14	1.99E-13	3.56E-12	5.70E-16	2.04E-13

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TABLE 11.A-5 EXPECTED ANNUAL LIQUID EFFLUENT CONCENTRATIONS (DILUTED AND UNDILUTED), BY RADIONUCLIDE, FROM EACH WASTE SYSTEM

NUCLIDE	BORON RS, UNDILUTED (μCi/ml)	MISC. WASTES, UNDILUTED (μCi/ml)	SECONDARY, UNDILUTED (μCi/ml)	TURB. BLDG., UNDILUTED (μCi/ml)	DETERGENT, UNDILUTED (μCi/ml)	BORON RS, DILUTED (μCi/ml)	MISC. WASTES, DILUTED (μCi/ml)	SECONDARY, DILUTED (μCi/ml)	TURB. BLDG., DILUTED (μCi/ml)	DETERGENT, DILUTED (μCi/ml)
Mo-99	4.21E-08	1.97E-06	2.47E-07	1.16E-09	8.04E-09	2.04E-13	3.24E-12	7.79E-11	1.23E-14	6.44E-15
Tc-99m	3.98E-08	1.97E-06	1.27E-07	8.34E-10	0.00E+00	1.93E-13	3.24E-12	4.00E-11	8.91E-15	0.00E+00
Ru-103	1.93E-07	3.12E-06	3.10E-07	1.50E-09	3.89E-08	9.36E-13	5.14E-12	9.79E-11	1.60E-14	3.11E-14
Ru-106	2.63E-06	3.83E-05	3.77E-06	1.79E-08	1.19E-06	1.28E-11	6.30E-11	1.19E-09	1.91E-13	9.56E-13
Ag-110m	3.79E-08	5.52E-07	5.37E-08	2.56E-10	1.61E-07	1.84E-13	9.09E-13	1.70E-11	2.74E-15	1.29E-13
Te-129m	4.78E-09	7.89E-08	7.79E-09	3.76E-11	0.00E+00	2.32E-14	1.30E-13	2.46E-12	4.02E-16	0.00E+00
Te-129	3.10E-09	1.61E-07	3.10E-07	6.52E-11	0.00E+00	1.50E-14	2.65E-13	9.79E-11	6.97E-16	0.00E+00
Te-131m	4.10E-09	3.24E-07	5.44E-08	2.37E-10	0.00E+00	1.99E-14	5.34E-13	1.72E-11	2.53E-15	0.00E+00
Te-131	7.48E-10	6.20E-08	4.45E-08	4.34E-11	0.00E+00	3.63E-15	1.02E-13	1.41E-11	4.64E-16	0.00E+00
I-131	6.44E-07	1.72E-05	1.53E-06	1.44E-08	2.14E-07	3.12E-12	2.83E-11	4.84E-10	1.54E-13	1.72E-13
Te-132	1.30E-08	5.47E-07	6.50E-08	3.08E-10	0.00E+00	6.30E-14	9.01E-13	2.05E-11	3.29E-15	0.00E+00
I-132	1.91E-08	3.55E-06	3.64E-06	6.27E-09	0.00E+00	9.28E-14	5.85E-12	1.15E-09	6.70E-14	0.00E+00
I-133	2.30E-07	2.39E-05	4.22E-06	3.44E-08	0.00E+00	1.12E-12	3.93E-11	1.33E-09	3.67E-13	0.00E+00
I-134	1.29E-10	9.19E-07	3.30E-06	2.94E-10	0.00E+00	6.25E-16	1.51E-12	1.04E-09	3.14E-15	0.00E+00
Cs-134	2.52E-05	1.61E-04	3.18E-07	1.55E-09	1.47E-06	1.22E-10	2.65E-10	1.00E-10	1.65E-14	1.18E-12
I-135	8.32E-08	1.45E-05	6.50E-06	3.50E-08	0.00E+00	4.04E-13	2.38E-11	2.05E-09	3.74E-13	0.00E+00
Cs-136	2.03E-06	1.83E-05	3.77E-08	1.85E-10	4.96E-08	9.84E-12	3.02E-11	1.19E-11	1.98E-15	3.97E-14
Cs-137	3.36E-05	2.14E-04	4.25E-07	2.06E-09	2.14E-06	1.63E-10	3.52E-10	1.34E-10	2.20E-14	1.72E-12
Ba-140	2.52E-07	5.15E-06	5.14E-07	2.49E-09	1.22E-07	1.22E-12	8.48E-12	1.62E-10	2.66E-14	9.77E-14
La-140	3.28E-07	8.41E-06	9.39E-07	4.44E-09	0.00E+00	1.59E-12	1.39E-11	2.96E-10	4.75E-14	0.00E+00
Ce-141	3.74E-09	6.21E-08	6.09E-09	2.94E-11	3.08E-08	1.81E-14	1.02E-13	1.92E-12	3.14E-16	2.47E-14
Ce-143	8.54E-09	6.40E-07	1.01E-07	4.44E-10	0.00E+00	4.14E-14	1.05E-12	3.18E-11	4.75E-15	0.00E+00
Ce-144	1.14E-07	1.66E-06	1.63E-07	7.74E-10	5.23E-07	5.53E-13	2.73E-12	5.13E-11	8.27E-15	4.19E-13

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TABLE 11.A-5 EXPECTED ANNUAL LIQUID EFFLUENT CONCENTRATIONS (DILUTED AND UNDILUTED), BY RADIONUCLIDE, FROM EACH WASTE SYSTEM

NUCLIDE	BORON RS, UNDILUTED (μCi/ml)	MISC. WASTES, UNDILUTED (μCi/ml)	SECONDARY, UNDILUTED (μCi/ml)	TURB. BLDG., UNDILUTED (μCi/ml)	DETERGENT, UNDILUTED (μCi/ml)	BORON RS, DILUTED (μCi/ml)	MISC. WASTES, DILUTED (μCi/ml)	SECONDARY, DILUTED (μCi/ml)	TURB. BLDG., DILUTED (μCi/ml)	DETERGENT, DILUTED (μCi/ml)
All Others Except Tritium	3.45E-05	2.43E-04	2.28E-07	2.21E-08	3.10E-07	1.68E-10	4.01E-10	7.18E-11	2.36E-13	2.48E-13
TOTAL	1.01E-04	7.73E-04	3.03E-05	1.57E-07	1.20E-05	4.88E-10	1.27E-09	9.56E-09	1.67E-12	9.64E-12

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TABLE 11.A-6 DESIGN BASIS RADIONUCLIDE CONCENTRATIONS IN LIQUID EFFLUENT, IN FRACTIONS OF 10 CFR PART 20 CONCENTRATION LIMITS

NUCLIDE	FRACTION OF 10 CFR PART 20 MAXIMUM PERMISSIBLE CONCENTRATIONS [1]
Na-24	7.79E-05
Cr-51	6.77E-07
Mn-54	6.90E-06
Fe-55	7.03E-07
Fe-59	2.20E-06
Co-58	2.03E-08
Co-60	5.52E-06
Zn-65	2.29E-06
W-187	1.27E-05
Np-239	8.54E-06
Br-84	2.77E-07
Rb-88	1.84E-07
Sr-89	1.73E-05
Sr-90	1.83E-05
Sr-91	2.73E-07
Y-91m	1.72E-10
Y-91	2.76E-06
Y-93	5.18E-07
Zr-95	2.87E-06
Nb-95	9.34E-07
Mo-99	5.27E-05
Tc-99m	2.82E-09
Ru-103	9.79E-07
Ru-106	2.78E-06
Ag-110m	5.99E-09
Te-129m	1.03E-05
Te-129	4.28E-08

TABLE 11.A-6 DESIGN BASIS RADIONUCLIDE CONCENTRATIONS IN LIQUID EFFLUENT, IN FRACTIONS OF 10 CFR PART 20 CONCENTRATION LIMITS

NUCLIDE	FRACTION OF 10 CFR PART 20 MAXIMUM PERMISSIBLE CONCENTRATIONS [1]
Te-131m	9.77E-06
Te-131	1.99E-07
I-131	2.17E-01
Te-132	1.89E-04
I-132	8.52E-04
I-133	7.32E-02
I-134	1.25E-04
Cs-134	1.39E-02
I-135	7.69E-03
Cs-136	3.24E-04
Cs-137	5.26E-03
Ba-140	2.86E-06
La-140	3.23E-06
Ce-141	9.79E-07
Ce-143	1.21E-06
Ce-144	7.07E-06
H-3	3.54E-04
Sum of MPC Fractions [2]	3.19E-01

1. Based on 10 CFR Part 20, Appendix B, Table II, Column 2, prior to 1994.

2. In the course of plant operation, individual isotopic fractions of maximum permissible concentration (MPC) might vary from the values tabulated above. The purpose of this table is to demonstrate that the design of the liquid radioactive waste processing system is adequate at design conditions such that the sum of the fractions of MPCs in the pre-1994 version of 10 CFR Part 20 will not exceed the limit of 1.0.

TABLE 11.A-7 DESIGN-BASIS RADIONUCLIDE AIRBORNE CONCENTRATIONS AT THE SITE BOUNDARY FROM ALL GASEOUS EFFLUENT RELEASE POINTS COMBINED, IN FRACTIONS OF 10 CFR PART 20 CONCENTRATION LIMITS

Iodines

Nuclide	Fraction of 10 CFR Part 20 Maximum Permissible Concentration (1)
I-131	4.13E-04
I-133	7.65E-04

Noble Gases

Nuclide	Fraction of 10 CFR Part 20 Maximum Permissible Concentration (1)
Kr-85m	7.03E-05
Kr-85	7.87E-02
Kr-87	2.01E-04
Kr-88	7.77E-04
Xe-131m	4.60E-04
Xe-133m	2.07E-04
Xe-133	2.93E-02
Xe-135m	1.85E-05
Xe-135	5.23E-04
Xe-137	none
Xe-138	1.16E-04

Particulates

Nuclide	Fraction of 10 CFR Part 20 Maximum Permissible Concentration (1)
Cr-51	7.83E-09

Particulates

Nuclide	Fraction of 10 CFR Part 20 Maximum Permissible Concentration (1)
Mn-54	3.35E-07
Co-57	2.53E-10
Co-58	1.38E-06
Co-60	2.43E-06
Fe-59	8.26E-08
Sr-89	9.92E-07
Sr-90	2.37E-06
Zr-95	8.57E-08
Nb-95	6.50E-08
Ru-103	9.19E-10
Ru-106	6.99E-11
Sb-125	1.61E-10
Cs-134	9.39E-06
Cs-136	6.49E-07
Cs-137	9.94E-06
Ba-140	2.49E-09
Ce-141	2.28E-08
Ar-41	1.57E-04
C-14	1.35E-05
H-3	1.02E-04

Sum of MPC Fractions (2) 1.12E-01.

1. Based on 10 CFR Part 20, Appendix B, Table II, Column 1, prior to 1994.

2. In the course of plant operation, individual isotopic fractions of maximum permissible concentration (MPC) might vary from the values tabulated above. The purpose of this table is to demonstrate that the design of the gaseous radioactive waste processing system is adequate at design conditions such that the sum of the fractions of MPCs in the pre-1994 version of 10 CFR Part 20 will not exceed the limit of 1.0.

TABLE 11.A-8 TOTAL ANNUAL DESIGN-BASIS AND EXPECTED RELEASES OF RADIOACTIVE LIQUID WASTE TO THE ENVIRONMENT FROM ALL SOURCES COMBINED, IN CURIES PER YEAR

NUCLIDE	ANNUAL DESIGN BASIS LIQUID EFFLUENT RELEASE (Ci/yr)	ANNUAL EXPECTED LIQUID EFFLUENT RELEASE (Ci/yr)
Activated Corrosion Products (Crud):		
Na-24	1.45E+01	4.80E-01
Cr-51	1.26E+00	4.10E-02
Mn-54	6.43E-01	2.10E-02
Fe-55	5.24E-01	1.70E-02
Fe-59	1.23E-01	4.00E-03
Co-58	1.89E+00	6.20E-02
Co-60	2.57E-01	8.50E-03
Zn-65	2.14E-01	6.80E-03
W-187	8.31E-01	2.70E-02
Np-239	7.95E-01	2.60E-02
Fission Products:		
Br-84	1.03E-01	3.30E-02
Rb-88	6.85E-02	2.50E-01
Sr-89	4.81E-02	1.80E-03
Sr-90	5.11E-03	1.60E-04
Sr-91	1.79E-02	9.20E-03
Y-91m	4.82E-04	1.50E-03
Y-91	7.73E-02	8.30E-05
Y-93	1.45E-02	3.90E-02
Zr-95	1.60E-01	5.20E-03
Nb-95	8.70E-02	3.80E-03
Mo-99	9.81E+00	7.70E-02
Tc-99m	1.58E-02	4.10E-02
Ru-103	7.29E-02	9.80E-02
Ru-106	2.59E-02	1.20E+00
Ag-110m	1.67E-04	1.70E-02

TABLE 11.A-8 TOTAL ANNUAL DESIGN-BASIS AND EXPECTED RELEASES OF RADIOACTIVE LIQUID WASTE TO THE ENVIRONMENT FROM ALL SOURCES COMBINED, IN CURIES PER YEAR

NUCLIDE	ANNUAL DESIGN BASIS LIQUID EFFLUENT RELEASE (Ci/yr)	ANNUAL EXPECTED LIQUID EFFLUENT RELEASE (Ci/yr)
Te-129m	2.89E-01	2.50E-03
Te-129	3.19E-02	9.30E-02
Te-131m	5.46E-01	1.70E-02
Te-131	1.49E-02	1.30E-02
I-131	6.05E+01	4.90E-01
Te-132	5.28E+00	2.00E-02
I-132	6.35E+00	1.10E+00
I-133	6.82E+01	1.30E+00
I-134	2.33E+00	9.90E-01
Cs-134	1.17E+02	4.60E-01
I-135	2.86E+01	2.00E+00
Cs-136	2.71E+01	4.90E-02
Cs-137	9.79E+01	6.20E-01
Ba-140	7.98E-02	1.60E-01
La-140	6.00E-02	2.90E-01
Ce-141	8.22E-02	2.00E-03
Ce-143	4.50E-02	3.10E-02
Ce-144	6.59E-02	5.20E-02
H-3	9.90E+02	9.90E+02
TOTAL WITHOUT TRITIUM	4.25E+02	1.10E+01

TABLE 11.A-9 TOTAL ANNUAL DESIGN BASIS AND EXPECTED RELEASES OF AIRBORNE RADIOACTIVE WASTE TO THE ENVIRONMENT FROM ALL RELEASE POINTS COMBINED, IN CURIES PER YEAR

NUCLIDE	ANNUAL DESIGN BASIS AIRBORNE EFFLUENT RELEASE (Ci/yr)	ANNUAL EXPECTED AIRBORNE EFFLUENT RELEASE (Ci/yr)
I-131	2.23E+01	1.80E-01
I-133	2.89E+01	5.50E-01
Kr-85m	3.80E+01	4.00E+00
Kr-85	1.28E+05	6.00E+02
Kr-87	2.17E+01	4.00E+00
Kr-88	8.38E+01	9.00E+00
Xe-131m	9.94E+02	5.00E+01
Xe-133m	3.36E+02	3.00E+00
Xe-133	4.74E+04	1.50E+02
Xe-135m	4.00E+00	4.00E+00
Xe-135	2.83E+02	3.10E+01
Xe-137	0.00E+00	0.00E+00
Xe-138	1.25E+01	3.00E+00
Cr-51	3.38E-03	1.10E-04
Mn-54	1.81E-03	5.90E-05
Co-57	8.20E-06	8.20E-06
Co-58	1.49E-02	4.90E-04
Co-60	3.93E-03	1.30E-04
Fe-59	8.92E-04	2.90E-05
Sr-89	5.35E-03	2.00E-04
Sr-90	2.56E-03	8.00E-05
Zr-95	4.63E-04	1.50E-05

TABLE 11.A-9 TOTAL ANNUAL DESIGN BASIS AND EXPECTED RELEASES OF AIRBORNE RADIOACTIVE WASTE TO THE ENVIRONMENT FROM ALL RELEASE POINTS COMBINED, IN CURIES PER YEAR

NUCLIDE	ANNUAL DESIGN BASIS AIRBORNE EFFLUENT RELEASE (Ci/yr)	ANNUAL EXPECTED AIRBORNE EFFLUENT RELEASE (Ci/yr)
Nb-95	1.05E-03	4.60E-05
Ru-103	1.49E-05	2.00E-05
Ru-106	7.55E-08	3.50E-06
Sb-125	6.10E-07	6.10E-07
Cs-134	2.03E-02	8.00E-05
Cs-136	2.10E-02	3.80E-05
Cs-137	2.69E-02	1.70E-04
Ba-140	1.35E-05	2.70E-05
Ce-141	6.16E-04	1.50E-05
Ar-41	3.40E+01	3.40E+01
C-14	7.30E+00	7.30E+00
H-3	1.10E+02	1.10E+02

TABLE 11.A-10 BASIS FOR REACTOR COOLANT SYSTEM ACTIVITY NUREG-0017 GALE CODE INPUT

EPower Level, MWt	2754
Percent Failed Fuel	1.0
CVCS Purification Ion Exchanger	
Decontamination Factors:	
Iodine	100
Cs, Rb	2
others	100
Purification Flow Rate (CVCS Purification Ion Exchanger), gpm	60
Effective Purification Flow Rate for Lithium and Cesium Removal, gpm	0
Fission Product Escape Rate Coefficients, sec^{-1}	
Noble Gases	6.5×10^{-8}
Halogens, Cs	2.3×10^{-8}
Te, Mo	1.4×10^{-9}
All others	1.4×10^{-11}
Feed and Bleed Liquid Waste, gal./day	2670

FIGURE 11.A–1 ESCAPE RATE COEFFICIENTS

11.B RADIOACTIVE WASTE PROCESSING OF RELEASES TO ENVIRONMENT

11.B.1 BASES

Discussed in this section are the bases used for determining the expected and design-basis annual releases of radionuclides from the radioactive waste processing system to the environment. The description of radioactive waste processing system design and operation provided in this appendix is a representation of how the design and operation have been modeled in the 10 CFR 50 Appendix I, NUREG-0017, GALE Code analysis. GALE, the NRC computer program, was used to calculate expected releases of radioactive material in liquid, gaseous and airborne effluents. The inputs to GALE are summarized in Table 11.B-1. The total expected annual releases are given in Appendix 11.A, Tables 11.A-3 and 11.A-4.

Expected releases, with anticipated operational occurrences included, are calculated (using GALE) for the purpose of ascertaining that the radioactive waste processing systems and building ventilation systems have sufficient capability to ensure that annual releases will be within the limits specified by 10 CFR Part 50, Appendix I. Design basis releases are calculated (as discussed in Appendix 11.A) to demonstrate that the radioactive waste processing systems and building ventilation effluents are within the limits of 10 CFR Part 20, Sections 105 and 106 and Appendix B (version prior to January 1, 1994).

Actual system operation (discussed in FSAR Section 11.1) may differ from the model input to GALE. However, meeting the applicable criteria is ensured through the use of process and effluent radiation monitoring and sampling systems used in conjunction with the Radiological Effluent Monitoring & Offsite Dose Calculation Manual (REMODCM).

11.B.2 LIQUID WASTE PROCESSING SYSTEM

11.B.2.1 Processing of Clean Liquid Waste

The expected releases during normal operation from the clean liquid waste processing system (comprised of primary coolant waste and shim bleed) are based on processing and discharging of approximately 1,200,000 gallons per year of reactor coolant wastes with quantities as shown in Figure 11.B–1.

The clean liquid waste processing system is designed to process reactor coolant waste as well as the letdown flow from the CVCS. The following three sources of liquid waste contribute to the clean liquid waste processing system:

- The primary drain tank, containing clean liquid waste from containment
- The equipment drain sump tank, containing clean liquid waste from the auxiliary building
- Shim bleed from the primary coolant letdown that has been processed by a deborating or purification ion exchanger

Primary coolant letdown is processed by either the purification or deborating ion exchangers. A portion of what leaves these ion exchangers, referred to as shim bleed, is diverted to the clean liquid waste system for further processing. The remainder of the primary coolant letdown is passed on to the volume control tank (VCT). In the volume control tank, primary coolant liquid is degassed and the gases are purged and sent to the gaseous waste processing system with the remaining liquid returned to the primary coolant system.

The three liquid sources, listed above, are sent to a mixed-bed ion exchanger. Following processing in the mixed-bed ion exchangers, the liquid is sent to the coolant waste receiver tanks. From there, the liquid is then sent to a secondary mixed-bed ion exchanger for further processing, after which this liquid is passed into the coolant waste monitor tanks. From these tanks, the liquid is released to the environment via the discharge canal.

This process stream is shown schematically in Figure 11.B–1. Shown in that figure are the tank volumes, flow rates, and decontamination factors (DFs) associated with this liquid waste process stream that are inputs to GALE.

11.B.2.2 Processing of Aerated Liquid Waste

Contributions to the aerated liquid waste processing system include the following:

- Waste water from on-site laundry
- Drainage from hand wash sinks
- Equipment and area decontamination waste water
- Spent fuel pool liner drainage
- Primary coolant sampling system drainage
- Auxiliary building floor drainage
- Primary coolant leakage from miscellaneous sources

These contributions are directed into aerated waste drain tanks. The aerated liquid waste collected in these drain tanks is then sent to a mixed-bed ion exchanger for processing, after which it is sent to an aerated waste monitor tank. The liquid content of this tank is released to the environment via the discharge canal.

11.B.2.3 Processing of Secondary Side Liquid Waste

Secondary side liquid waste is comprised of the following three contributions:

- Steam generator blowdown

- Turbine building floor drainage

- Condensate demineralizer regenerant solutions from condensate polishing facility

Steam generator blowdown is sent to a blowdown flash tank, where one-third of the liquid blowdown flashes to steam and can be released as gaseous effluent. Turbine building floor drainage flows to the turbine building sump and is released to the environment via the discharge canal when radioactivity is detected. With respect to the liquid waste from the condensate polishing facility, the condensate demineralizer regenerant solutions are sent to the waste neutralization sumps. The liquid in the waste neutralization sumps is released to the environment via the discharge canal.

These three contributions to the secondary side liquid waste (identified above) are shown schematically in Figure 11.B–1.

11.B.3 GASEOUS WASTE PROCESSING SYSTEM

Airborne releases from the gaseous radioactive waste processing system, main condenser air ejector, and containment vents are discharged via the Millstone stack. Containment purges, as well as airborne releases from buildings such as the auxiliary and fuel building, are discharged via the Unit 2 enclosure building roof vent. Turbine building releases are discharged via the Unit 2 turbine building roof vent. Steam generator blowdown is discharged via a separate blowdown vent located on the Unit 2 enclosure building roof. For demonstrating compliance with the requirements of 10 CFR Part 20, Sections 105 and 106, and Appendix B (version prior to January 1, 1994) and 10 CFR Part 50, Appendix I, maximum dispersion factors derived from annual average meteorological data were assumed in the 10 CFR Part 50 Appendix I analysis.

The sources and pathways of airborne releases are the following:

- Auxiliary Building/Fuel Building Ventilation: Airborne releases from these sources are normally processed by HEPA filter and monitored prior to release to the environment via the Unit 2 enclosure building roof vent.

- Steam Generator Blowdown Tank Vent: Approximately one-third of the steam generator blowdown flashes to steam and is released to the environment at the Unit 2 enclosure building roof.

- Containment Building: Containment purge air is processed by a HEPA filter and monitored before being released to the environment via the Unit 2 enclosure building roof vent. Releases of routine venting of containment are processed by charcoal adsorber and HEPA filter and monitored before being released to the environment via the Millstone stack.

- Gaseous Waste Gas Processing System: Gases from liquid waste processing system tanks and gases stripped in the VCT from the primary coolant letdown flow are stored in six gas

decay tanks and monitored before being released to the environment via the Millstone stack. No credit is taken for the HEPA filter because it does not satisfy Reg. Guide 1.140.

- Main Condenser/Air Ejector: Non-condensable gases from steam flow to the main condenser are monitored and released to the environment via the Millstone stack.

- Turbine Building: Steam leakage into the turbine building is released to the environment at the Unit 2 turbine building roof vent.

The sources and pathways to the gaseous radioactive waste processing system (identified above) are shown schematically in Figure 11.B–2. These contributions are included in the inputs to GALE and are summarized in Table 11.B-1.

TABLE 11.B-1 INPUTS TO PWR-GALE CODE

DESCRIPTION	UNITS	VALUE
Thermal Power Level	MWt	2,754
Mass of Primary Coolant	10^3 lbm	461.1
Primary Coolant Letdown Rate	GPM	60
Letdown Cation Demineralizer Flow Rate	GPM	0
Number of Steam Generators		2
Total Steam Flow	10^6 lbm/hr	11.8
Mass of Liquid in Each Steam Generator	10^3 lbm	132.257
Total Blowdown Rate	10^3 lbm/hr	73.7
Blowdown Treatment Method	0, 1, or 2	2
Condensate Demineralizer Regeneration Time	days	56
Condensate Demineralizer Flow Fraction (equal to average flow for all condensate demineralizers ÷ total steam flow)	fraction	0.813
CLEAN LIQUID WASTE (see notes)		
Shim Bleed Rate:		
Shim Bleed Flow Rate	GPD	2,670
DF for Iodine		10,000
DF for Cs, Rb		200
DF for Other Nuclides		5,000
Collection Time	days	16.66
Process and Discharge Time	days	0.2865
Fraction Discharged		1
Equipment Drains (Coolant Waste) Inputs:		
Flow Rate	GPD	600
PCA (Fraction of Primary Coolant Activity)		0.145
DF for Iodine		1,000
DF for Cs, Rb		20
DF for Other Nuclides		1,000
Collection Time	days	16.66

TABLE 11.B-1 INPUTS TO PWR-GALE CODE (CONTINUED)

DESCRIPTION	UNITS	VALUE
Process and Discharge Time	days	0.2865
Fraction Discharged		1
Clean Wastes Inputs: None (see notes)		
AERATED LIQUID WASTE (see note)		
Dirty Wastes (Aerated Waste) Inputs:		
Flow Rate	GPD	1,110
PCA (Fraction of Primary Coolant Activity)		0.0427
DF for Iodine		100
DF for Cs, Rb		2.0
DF for Other Nuclides		100
Collection Time	days	2.61
Process and Discharge Time	days	0.0598
Fraction Discharged		1
MISC. other liquid waste inputs (see note)		
Steam Generator Blowdown:		
Blowdown Fraction Processed		1
DF for Iodine		1
DF for Cs, Rb		1
DF for Other Nuclides		1
Collection Time	days	0
Process and Discharge Time	days	0
Fraction Discharged		1
Regenerant Inputs:		
Regenerant Flow Rate	GPD	850
DF for Iodine		1
DF for Cs, Rb		1
DF for Other Nuclides		1
Collection Time	days	0
Process and Discharge Time	days	0
Fraction Discharged		1

TABLE 11.B-1 INPUTS TO PWR-GALE CODE (CONTINUED)

DESCRIPTION	UNITS	VALUE
Misc. other GALE Code INPUTS:		
Continuous Stripping of Full Letdown Flow		2
Hold-up Time For Xenon	days	90
Hold-up Time for Krypton	days	90
Fill Time for Decay Tanks for Gas Stripping	days	90
AIRBORNE INPUTS:		
Waste Gas System Particulate (HEPA) Filter Removal Efficiency		0%
Fuel Handling Building Charcoal Filter Removal Efficiency		0%
Fuel Handling Building HEPA Filter Removal Efficiency		99%
Auxiliary Building Charcoal Filter Removal Efficiency		0%
Auxiliary Building HEPA Filter Removal Efficiency		99%
Containment Volume	10^6 ft^3	1.899
Containment Atmosphere Clean-up Charcoal Removal Efficiency		0%
Containment Atmosphere Clean-up HEPA Removal Efficiency		0%
Containment Atmosphere Clean-up Rate	10^3 CFM	0
Containment High Volume (Shutdown) Purge Charcoal Removal Efficiency		0%
Containment High Volume (Shutdown) Purge HEPA Filter Removal Efficiency		99%
No. of Purges per Year (2 purges internal to GALE code, see notes)		0
Containment Low Volume Purge (Normal) Charcoal Efficiency (i.e.: containment venting, see note)		70%
Containment Low Volume Purge (Normal) Filter Efficiency (i.e.: containment venting, see note)		99%
Containment Low Volume Purge (Normal) Rate (i.e.: containment venting, see note)	CFM	10
Fraction of Iodine Released from Blowdown Tank Vent		0.05

TABLE 11.B-1 INPUTS TO PWR-GALE CODE (CONTINUED)

DESCRIPTION	UNITS	VALUE
Percent of Iodine Removed From (Main Condenser) Air Ejector Release		0
DETERGENT WASTES:		
Detergent Waste PF (Purification Factor), $(DF)^{-1}$		0.1

NOTES:

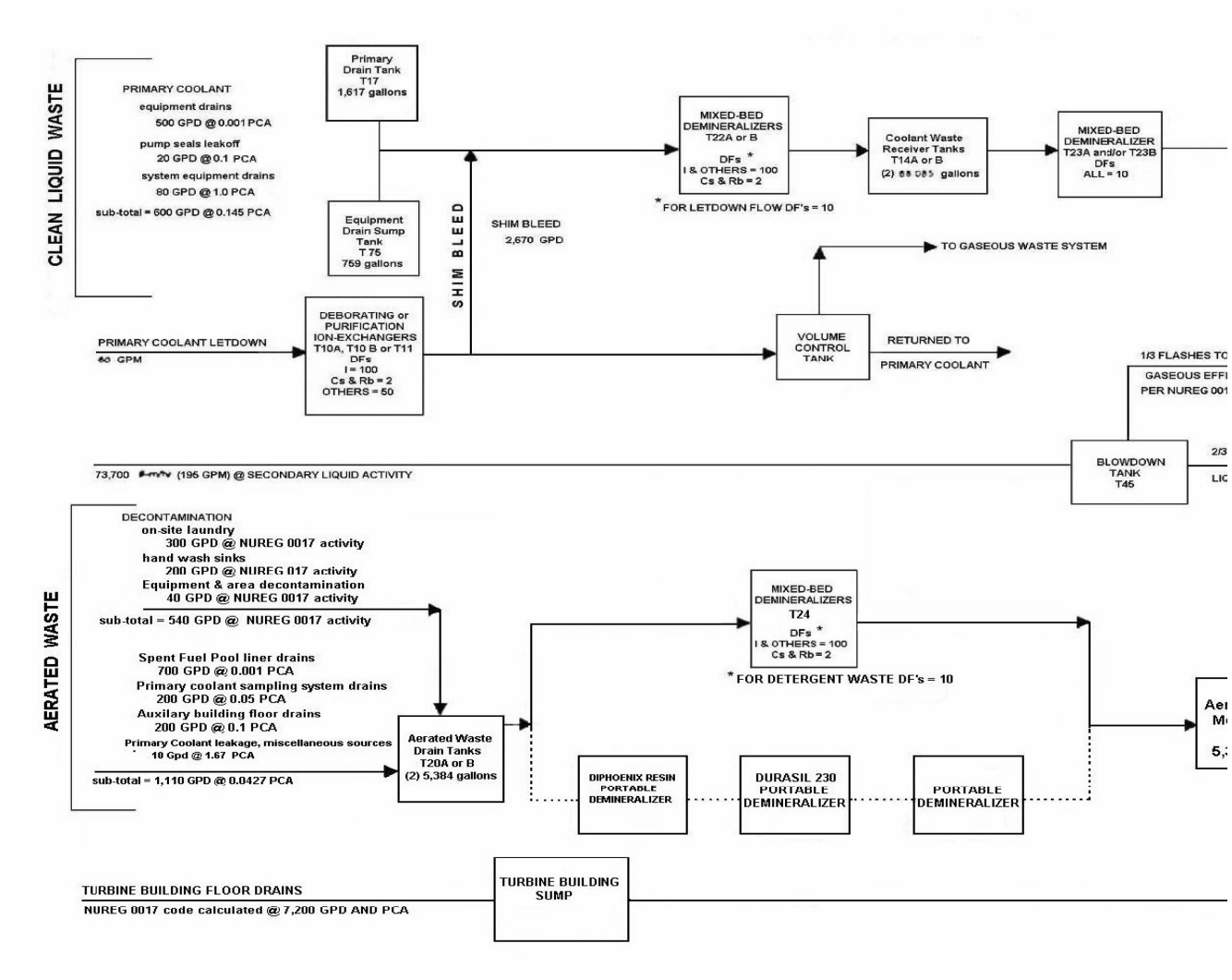
The component and system designations used in this table reflect the terminology used in NUREG-0017, GALE Code, and may differ from that used by Millstone Unit 2.

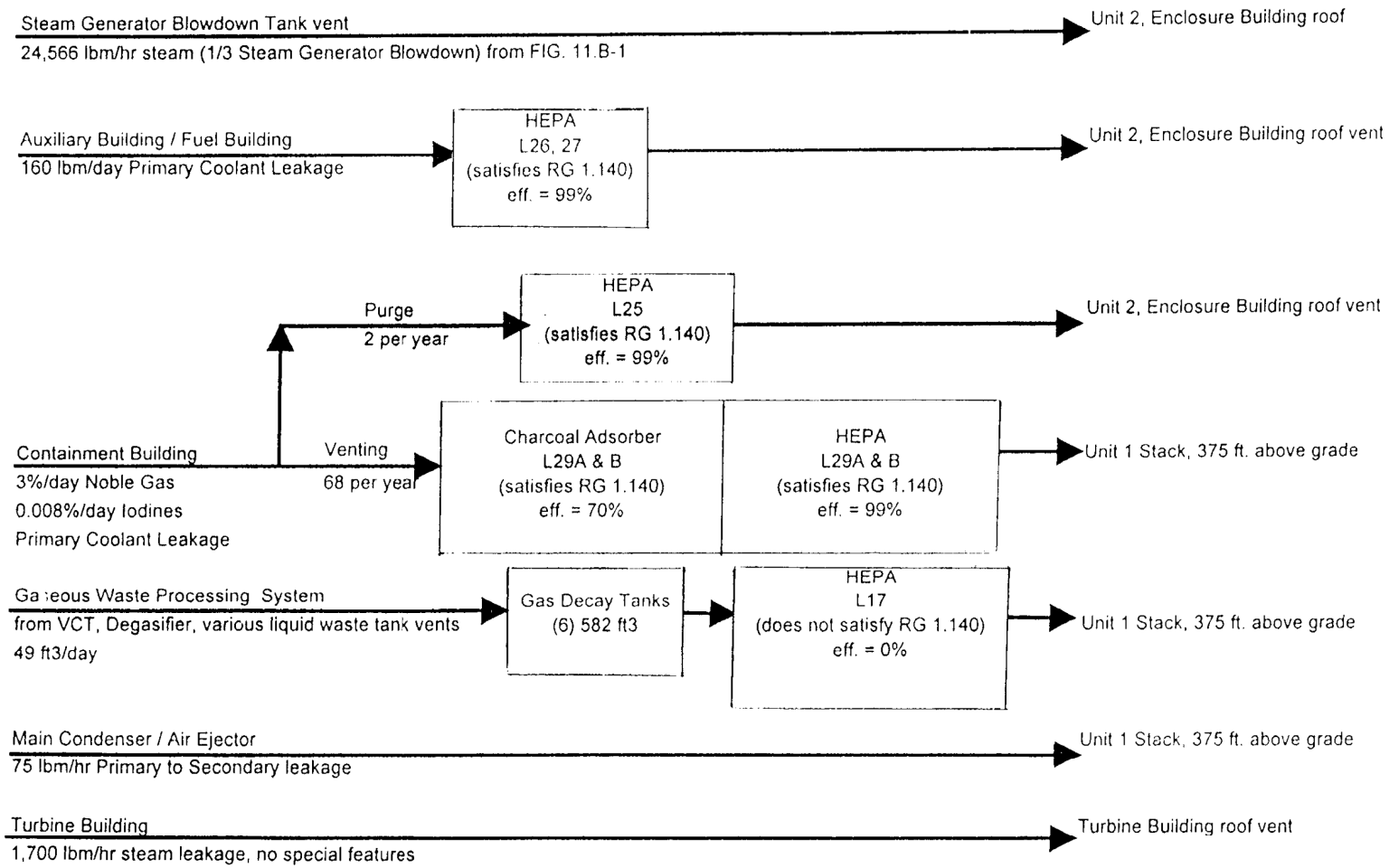
The GALE Code has internal calculations that impact the input data. The data provided as GALE Code inputs in this table and Figures 11.B–1 and 11.B–2 may differ from data listed in other tables in the FSAR. A specific example of this are the tank volumes listed on Figure 11.B–1 versus Table 11.1-1. The GALE Code applies a 0.8 or 0.4 factor to tank volumes to calculate collection, process, discharge and decay times.

FIGURE 11.B-1 NUREG 0017 GALE CODE INPUT DIAGRAM - LIQUID WASTE SYSTEML

MPS-2 FSAR

FIGURE 11.B-1 NUREG 0017 GALE CODE INPUT DIAGRAM - LIQUID WASTE SYSTEM





NOTE: THE VALUES IN THIS FIGURE ARE THE ASSUMPTIONS USED IN THE 10CFR50 APPENDIX I ANALYSIS. ACTUAL VALUES MAY VARY.

FIGURE 11.B-2
NUREG 0017 GALE CODE INPUT DIAGRAM - AIRBORNE

11.C DOSES FROM RADIOACTIVE RELEASES AND COST-BENEFIT ANALYSIS

11.C.1 DOSES TO HUMANS

The analysis of annual doses to the maximum individual and to the population residing within an 50 mile radius of Millstone Unit 2 are based on the methodology and equations presented in U. S. NRC Regulatory Guide 1.109 Rev. 1, *Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for The Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I*. To automate the Millstone Unit 2 pathways-to-man dose analyses, U.S. NRC Computer Code LADTAP II was used for the liquid effluent releases, and U.S. NRC Computer Code GASPAR was used for the gaseous effluent releases. The Regulatory Guide 1.109 equations germane to liquid and gaseous releases are presented in Sections 11.C.2 and 11.C.3, respectively. In the Millstone Unit 2 liquid and gaseous pathways-to-man dose analyses, the NRC default values provided in Regulatory Guide 1.109 were used as input to LADTAP II and GASPAR whenever site-specific data were unavailable.

The ensuing sections present the equations/methodologies employed in LADTAP II and GASPAR to calculate the dose contributions corresponding to the release/uptake pathways considered in the Millstone Unit 2 pathways-to-man dose analyses.

11.C.2 METHODS FOR CALCULATING DOSES FROM LIQUID RELEASES

LADTAP II was used to calculate the radiation exposure to man from ingestion of aquatic foods, shoreline deposits, swimming, and boating. Doses are calculated for both the maximum individual and for the population and are summarized for each pathway by age group and organ. LADTAP II also calculates the doses to certain representative biota other than man in the aquatic environment, such as fish, invertebrates, algae, muskrats, herons, and ducks, using the models presented in WASH-1258.

The equations and assumptions used in the LADTAP II analysis are presented in the ensuing subsections. Final dose results based on pathway doses calculated by LADTAP II are presented in Table 11.C-1.

11.C.2.1 Generalized Equation for Calculating Radiation Doses to Humans via Liquid Pathways

$$R_{aipj} = (C_{jp}) \cdot (U_{ap}) \cdot (D_{aipj})$$

where:

R_{aipj} = the annual dose to organ j of an individual of age group a from nuclide i via pathway p, in mrem/yr

C_{ip} = the concentration of nuclide i in the media of pathway p, in pCi/l, pCi/kg, or pCi/m^2

U_{ap} = the exposure time or intake rate (usage) associated with pathway p for age group a, in hr/yr, yr^{-1}, or kg/yr (as appropriate)

D_{aipj} = the dose factor, specific age group *a*, radionuclide *i*, pathway *p*, and organ *j*, in mrem/pCi ingested, mrem/hr per pCi/m^2 from exposure to deposited activity in sediment or on the ground, or mrem/hr per pCi/liter due to exposure from boating and swimming

11.C.2.2 Doses from Aquatic Foods

$$R_{apj} = (1{,}100)[(U_{ap})(M_p)/F] \cdot \Sigma_i Q_i l.\ B_{ip} \cdot D_{aipj} \cdot e^{-\lambda_i t_p}$$

where:

B_{ip} = the equilibrium bioaccumulation factor for nuclide *i* in pathway *p*, expressed as the ratio of the concentration in biota (in pCi/kg) to the radionuclide concentration in water (in pCi/l), in l/kg

M_p = the mixing ratio (reciprocal of the dilution factor) at the point of exposure or point of harvest of aquatic food, dimensionless

F = the flow rate of the liquid effluent in ft^3/sec

Q_i = the release rate of nuclide *i*, in Ci/yr

R_{apj} = the total annual dose to organ *j* of individuals of age group a from all of the nuclides *i* in pathway *p*, in mrem/yr

λ_i = the radioactive decay constant of nuclide *i*, in hours

t_p = the average transit time required for nuclides to reach the point of exposure. For internal dose, *t* is the total time elapsed between release of the nuclides and ingestion of food, in hours.

1,100 = the factor to convert from (Ci/yr)/(ft^3/sec) to pCi/l

All the other symbols are as previously defined.

11.C.2.3 Doses from Shoreline Deposits

$$R_{apj} = 110{,}000 \times [(U_{ap})(M_p)(W)/F] \times \Sigma_i Q_i T_i D_{aipj} \times e^{-\lambda_i t_p} \times (1 - e^{-\lambda_i t_b})$$

where:

W = the shoreline width factor that describes the geometry of the exposure, dimensionless

T_i = the radiological half-life of nuclide *i*, in days

t_b = the period of time for which sediment or soil is exposed to the contaminated water, in hours

110,000 = the factor to convert from Ci/yr per ft^3/sec to pCi/l and to account for the proportionality constant used in the sediment radioactivity model

All other symbols are as previously defined.

11.C.2.4 Doses from Swimming and Boating

The doses from swimming and boating were calculated using the methodology described in WASH 1258 (Atomic Energy Commission 1973).

The equation for calculation of the external dose to skin and the total body dose from swimming (water immersion) or boating (water surface) is:

$$R_{apj} = 1{,}000 \times [(U_{ap})(M_p)/(F)(K_p)] \times \Sigma_i Q_i D_{aipj} \times e^{-\lambda_i t_p}$$

where:

K_p = geometry correction factor equal to 1 for swimming and 2 for boating, dimensionless (no credit is taken for the shielding provided by the boat).

All other symbols are as previously defined.

11.C.3 METHOD FOR CALCULATING DOSES FROM GASEOUS RELEASES

FASPAR implements the air release dose models of the NRC Regulatory Guide 1.109 for noble gases (semi-infinite plume only) and for radioiodine and particulate emissions. GASPAR computes both population and individual doses using site data, meteorological data, and radionuclide release source terms as inputs. Location-dependent meteorological data for selected individuals are specified as input data. The site data include the population distribution and the quantities of milk, meat, and vegetation produced. The meteorological data include dispersion X/Q, X/Q decayed, X/Q decayed and depleted, and deposition D/Q. Population doses, individual doses, and cost benefit results are calculated.

There are two basic types of calculations, the individual dose calculation and the population dose calculation. Seven pathways by which the nuclides travel to man are considered. These are plume, ground, inhalation, vegetation, cows' milk, goats' milk, and meat. For the individual dose calculations, man is subdivided into the four age groups of infant (0 to 1 year), child (1 to 11 years), teenager (12 to 18 years), and adult (over 18 years). Each of these calculations takes into account eight body organs: total body, gastrointestinal (GI) tract, bone, liver, kidney, thyroid, lung, and skin.

The equations and assumptions used in the GASPAR analysis are presented in the ensuing subsections. Final dose results by GASPAR, based on activity releases tabulated in Table 11.A-3, are presented in Table 11.C-1.

11.C.3.1 Gamma and Beta Doses from Noble Gas Discharged to the Atmosphere

11.C.3.1.1 Annual Air Doses from Noble Gas Releases (Non-Elevated)

- Annual Gamma Air Dose Equation:

$$D^{\gamma}(r,\theta) = 3.17 \times 10^{4} \cdot \sum_{i} Q_i \cdot \left[X/Q\right](r,\theta) \cdot \left(DF_i^{\gamma}\right)$$

- Annual Beta Air Dose:

$$D^{\beta}(r,\theta) = 3.17 \times 10^{4} \cdot \sum_{i} Q_i \cdot \left[X/Q\right](r,\theta) \cdot \left(DF_i^{\beta}\right)$$

where:

$D^{\gamma}(r,\theta)$ = the annual gamma air doses at the distance r in the sector at angle θ from the discharge point in mrad/year

$D^{\beta}(r,\theta)$ = the annual beta air doses at the distance r in the sector at angle θ from the discharge point in mrad/year

Q_i = the release rate of the radionuclide i, in Ci/year

$X/Q(r,\theta)$ = the annual average gaseous dispersion factor at distance r in sector θ, in sec/m^3 (corrected for radioactive decay)

DF^{γ}_i = the gamma air dose factors for a uniform semi-infinite cloud of radionuclide i, in mrad·m^3/pCi·yr

DF^{β}_i = the beta air dose factors for a uniform semi-infinite cloud of radionuclide i, in mrad·m^3/pCi·yr

3.17×10^4 = the number of pCi per Ci divided by the number of seconds per year

11.C.3.1.2 Annual Total Body Dose from Noble Gas Releases (Non-Elevated)

$$D^{T}_{\infty}(r,\theta) = S_F \sum_i \chi_i(r,\theta) \cdot DFB_i$$

where:

$D^T\infty(r,\theta)$ = the total body dose due to immersion in a semi-infinite cloud at distance r in sector θ, in mrem/year

S_F = the attenuation factor that accounts for dose reduction due to shielding provided by residential structures, dimensionless

$\chi_i(r,\theta)$ = the annual average ground-level concentration of radionuclide i at distance r in sector θ, in pCi/ m^3

DFB_i = the total body dose factor for a semi-infinite cloud of the radionuclide i, which includes radiation attenuation through a depth of 5 cm into the body, in mrem·m^3/ pCi·yr

11.C.3.1.3 Annual Skin Dose from Noble Gas Releases (Non-Elevated)

$$D^{S}_{\infty}(r,\theta) = 1.11 \cdot S_F \cdot \sum_i \chi_i(r,\theta) \cdot DF^{\gamma}_i + \sum_i \chi_i(r,\theta) \cdot DFS_i$$

where:

$D^S\infty(r,\theta)$ = the annual skin dose due to immersion in a semi-infinite cloud at the distance r in sector θ, in mrem/yr

DFS_i = the beta skin dose factor for a semi-infinite cloud of radionuclide i, which includes the attenuation by the outer "dead" layer of the skin, in mrem·m^3/pCi·yr

1.11 = the average ratio of tissue to air energy absorption coefficients

All other parameters are as previously defined.

11.C.3.1.4 Annual Gamma Air Dose and Annual Total Body Dose Due to Noble Gas Releases from Free-Standing Stacks More Than 80 Meters Tall

Regulatory Guide 1.109 provides gamma air dose and total body dose equations that are applicable to elevated releases of noble gases from free-standing stacks that are more than 80 meters tall. However, GASPAR does not include these equations, and any attempt to use GASPAR to calculate cloud shine doses from elevated releases would lead to non-conservative results within approximately 1 kilometer from the point of release. What was done in the Millstone Unit 2 analysis was to simulate an elevated release by making a separate GASPAR run using the stack release radionuclide inventory as input, but with ground-level release X/Qs used in lieu of stack release X/Qs. This conservative approach was taken to calculate noble gas gamma shine doses (both gamma air doses and total body doses) to maximum individuals.

11.C.3.2 Doses from Radioiodines and Other Radionuclides, Exclusive of Noble Gases, Released to the Atmosphere

11.C.3.2.1 Annual Organ Dose Due to External Irradiation from Ground Deposition of Radionuclides

$$D_j^G(r,\theta) = 8{,}760 \cdot S_F \cdot \sum_i C^{G_i}(r,\theta) \cdot DFG_{ij}$$

where:

$D^G_j(r,\theta)$ = the annual dose to organ j at location (r,θ), in mrem/yr

S_F = a shielding factor that accounts for the dose reduction due to shielding provided by residential structures during occupancy (assumed to be 0.7), dimensionless

$C^G_i(r,\theta)$ = the ground plane concentration of radionuclide i at distance r in sector θ, in pCi/m^2

DFG_{ij} = the open field ground plane dose conversion factor for organ j from radionuclide i, in mrem·m^2/pCi·hr

8,760 = the number of hours in a year

11.C.3.2.2 Annual Organ Dose from Inhalation of Radionuclides in Air

$$D^A_{ja}(r,\theta) = R_a \cdot \sum_i \chi_i(r,\theta) \cdot DFA_{ija}$$

where:

$D^{A}_{ja}(r,\theta)$ = the annual dose to organ *j* of an individual in the age group *a* at location (r,θ) due to inhalation, in mrem/yr

R_a = the annual air intake for individuals in the age group *a*, in m^3/yr

$\chi_i(r,\theta)$ = the annual average concentration of radionuclide *i* in air at location (r,θ), in pCi/m^3

DFA_{ija} = the inhalation dose factor for radionuclide *i*, organ *j*, and age group *a*, in mrem/pCi

11.C.3.2.3 Annual Organ Dose from Ingestion of Atmospherically Released Radionuclides in Food

$$D^{D}_{ja}(r,\theta) = \sum_i DFI_{ija} \cdot \left[U^{V}_{a} \cdot f_g \cdot C^{V}_{i}(r,\theta) + U^{m}_{a} \cdot C^{m}_{i}(r,\theta) + U^{F}_{a} \cdot C^{F}_{i}(r,\theta) + U^{L}_{a} \cdot f_l \cdot C^{L}_{i}(r,\theta) \right]$$

where:

$C^{v}_{i}(r,\theta)$, $C^{m}_{i}(r,\theta)$, $C^{L}_{i}(r,\theta)$, $C^{F}_{i}(r,\theta)$ = the concentrations of radionuclide *i* in produce (non-leafy vegetables, leafy vegetables, fruits, and grains), milk, and meat, at location (r,θ), in pCi/kg or pCi/l

$D^{D}_{ja}(r,\theta)$ = the annual dose to the organ *i* of an individual in age group *a* from ingestion of produce, milk, leafy vegetables, and meat at location (r,θ), in mrem/year

DFI_{ija} = the ingestion dose factor for radionuclide *i*, organ *j*, and age group *a* in mrem/pCi

f_g, f_l = the respective fractions of the ingestion rates of produce and leafy vegetables that are produced in the garden of interest

U^{v}_{a}, U^{m}_{a}, U^{F}_{a}, U^{La} = the annual intake (usage) of produce, milk, meat, and leafy vegetables, respectively, for individuals in the age group *a*, in kg/yr or l/yr

11.C.4 COMPARISON OF CALCULATED ANNUAL MAXIMUM INDIVIDUAL DOSES WITH APPENDIX I DESIGN OBJECTIVES

A comparison of calculated annual maximum individual doses with 10 CFR Part 50 Appendix I design objectives is provided in Table 11.C-1.

11.C.5 GENERAL EXPRESSION FOR POPULATION DOSES

The general expression for calculating the annual population-integrated dose is:

$$D_j^P = 0.001 \sum_d P_d \cdot \sum_a D_{jda} \cdot f_{da}$$

where:

D^P_j = the annual population-integrated dose to organ *j* (total body or thyroid), in man-rem or thyroid man-rem

P_d = the population associated with sub-region *d*

D_{jda} = the annual dose to organ *j* (total body or thyroid) of an average individual of age group *a* in sub-region *d*, in mrem/yr

f_{da} = the fraction of the population in sub-region *d* that is in age group *a*

0.001 = the conversion factor from mrem to rem

The equation above, used in conjunction with the preceding equations with parameters adjusted for each age group, is used to calculate the population doses. The population doses due to annual releases of expected liquid and airborne radionuclide concentrations are presented in Table 11.C-2.

11.C.6 COST-BENEFIT ANALYSIS

Presented in this section are the cost-benefit analyses and their results, performed in accordance with the requirements set forth in 10 CFR Part 50, Appendix I, Section II.D. In these analyses, potential augments to the liquid and gaseous radioactive waste processing systems are examined for cost-effectiveness using the methodology and data provided in U. S. NRC Regulatory Guide 1.110 Rev. 0 (March 1976), *Cost-Benefit Analysis for Radwaste Systems for Light-Water-Cooled Nuclear Power Reactors*. The beneficial savings of each augment is calculated by multiplying the calculated dose reduction by $1,000 per man-rem. The cost of borrowed money is conservatively assumed to be 9%[1]. Provided in Appendix 11.B is information pertinent to the model input to GALE representing the design and configuration of the Millstone Unit 2 radioactive waste processing systems.

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[1] This is predicated on Northeast Utilities' June 1998 Cost of Capital Study, in which it was concluded that the weighted cost of borrowed money for NU, CL&P, WMECO, PSNH, and combined system is 9.3%. Based on this conclusion, it is conservative in the cost-benefit analysis to round down to 9%.

11.C.6.1 Procedure Used for Performing Cost-Benefit Analysis

The procedure employed for performing the cost-benefit analyses presented in Sections 11.C.6.2 and 11.C.6.3 below is taken directly from Regulatory Guide 1.110 Rev. 0. This procedure is summarized in the following steps:

1. Obtain the direct cost of equipment and materials from Table A-1 in Regulatory Guide 1.110.

2. Multiply the direct labor cost obtained from Table A-1 in Regulatory Guide 1.110 by the appropriate labor cost correction factor, obtained from Table A-4 in Regulatory Guide 1.110, to obtain the corrected labor cost for the geographical region in which the plant is located (Millstone Unit 2 being in Region I).

3. Add the costs obtained from the previous two steps to obtain the total direct cost (*TDC*).

4. Obtain the appropriate indirect cost factor (*ICF*) from Table A-5 in Regulatory Guide 1.110.

5. Determine the total capital cost (*TCC*) using the following equation:

   $TCC = TDC \times ICF$

6. Obtain the appropriate capital recovery factor (*CRF*) from Table A-6 in Regulatory Guide 1.110.

7. Determine the annual fixed cost (*AFC*) using the following equation:

   AFC = TCC x CRF

8. Obtain the annual operating cost (*AOC*) and the annual maintenance cost (*AMC*) from Regulatory Guide 1.110 Tables A-2 and A-3, respectively.

9. Determine the total annual cost (*TAC*) using the following equation:

   TAC = AFC + AOC + AMC

10. Determine the benefit of each augment by multiplying the dose reduction to be achieved by $1,000 per man-rem.

11. The benefit calculated in the previous step minus *TAC* provides the net benefit of adding the augment to the radwaste system. A positive net benefit means that adding the augment would be cost-beneficial; conversely, a negative net benefit means that it would not be cost-beneficial to add the augment to the radwaste system.

11.C.6.2 Augments to the Liquid Radioactive Waste Processing System

Table 11.C-2 presents the calculated base case annual total body dose (man-rem), annual maximum organ dose (man-rem), and annual thyroid dose (man-rem) associated with the operation of the Millstone Unit 2 liquid radioactive waste processing system for the population expected to live within a 50 mile radius of the plant. In the cost-benefit analysis performed for the liquid radioactive waste processing system, the augment that was chosen was a 50-gpm demineralizer on the blow-down waste stream. This demineralizer was selected because it was judged to be the least expensive augment that could provide any benefit in the reduction of population dose. Assuming that this augment is capable of reducing the population doses to zero (an extremely conservative assumption), the maximum annual benefit to be realized would be approximately $4,880. However, off-setting this benefit is a total annual cost associated with this augment that is estimated to be approximately $43,800.

Conclusion: It would not be cost-beneficial to add additional liquid radwaste processing equipment to the existing system.

The salient inputs to the liquid radioactive waste processing system cost-benefit analysis (taken from the tables in Regulatory Guide 1.110) are summarized below:

- Equipment/material cost: $43,000
- Labor cost: $29,000
- Labor cost correction factor: 1.6
- Indirect cost factor (ICF): 1.58, based on the fact that Millstone is a three unit site, with each unit having its own radwaste system
- Capital recovery factor (CRF): 0.0973
- Annual operating cost (AOC): $25,000
- Annual maintenance cost (AMC): $5,000
- Dose reduction goal: 4.88 man-rem (assuming that the goal is to reduce to zero the largest population dose resulting from liquid effluent released to the environment, that population dose having been calculated to be the maximum organ dose)

11.C.6.3 Augments to the Gaseous Radioactive Waste Processing System

Table 11.C-2 presents the calculated base case annual total body dose (man-rem) and annual thyroid dose (man-rem) associated with the operation of the Unit 2 gaseous radioactive waste processing system for the population expected to live within a 50 mile radius of the plant. In the cost-benefit analysis performed for the gaseous radioactive waste processing system, the augment that was chosen was a pair of 30,000 cfm charcoal/HEPA filtration systems on the Unit 2

auxiliary building vent. This choice was made because it was judged to be the least expensive augment that could provide any benefit in the reduction of population dose. Assuming that this augment is capable of reducing the population doses to zero (an extremely conservative assumption), the maximum annual benefit to be realized would be approximately $19,600. However, off-setting this benefit is a total annual cost associated with this augment that is estimated to be approximately $127,500.

Conclusion: It would not be cost-beneficial to add the two 30,000-cfm charcoal/HEPA filtration systems to the existing gaseous radioactive waste processing system. The salient inputs to the gaseous radioactive waste processing system cost-benefit analysis (taken from the tables in Regulatory Guide 1.110) are summarized below:

- Equipment/material cost: $314,000
- Labor cost: $102,000
- Labor cost correction factor: 1.6
- Indirect cost factor (ICF): 1.58, based on the fact that Millstone is a three-unit site, with each unit having its own radwaste system
- Capital recovery factor (CRF): 0.0973
- Annual operating cost (AOC): $18,000
- Annual maintenance cost (AMC): $36,000
- Dose reduction goal: 19.6 man-rem (assuming that the goal is to reduce to zero the largest population dose resulting from liquid effluent released to the environment, that population dose having been calculated to be the thyroid dose)

TABLE 11.C-1 COMPARISON OF CALCULATED ANNUAL MAXIMUM INDIVIDUAL DOSES WITH 10 CFR PART 50 APPENDIX I DESIGN OBJECTIVES

	10 CFR PART 50 APPENDIX I DESIGN OBJECTIVE DOSE [1]	CALCULATED ANNUAL DOSE
Airborne Effluent:		
• Gamma Air Dose	10 mrad	0.2 mrad
• Beta Air Dose	20 mrad	0.078 mrad
• Total Body Dose	5 mrem	0.15 mrem
• Skin Dose	15 mrem	0.25 mrem
• Maximum Organ Dose (Thyroid)	15 mrem	8.3 mrem
Liquid Effluent:		
• Total Body Dose	3 mrem	0.06 mrem
• Maximum Organ Dose (GI-LLI)	10 mrem	0.91 mrem

1 Per reactor.

TABLE 11.C-2 ANNUAL TOTAL BODY AND THYROID DOSES TO THE POPULATION WITHIN 50 MILES OF THE MILLSTONE SITE, IN MAN-REM, FROM EXPECTED LIQUID AND AIRBORNE EFFLUENT RELEASES

1. Annual Total Population Doses from Airborne Effluent Releases:
    - Total Body Dose: 0.607 man-rem
    - Thyroid Dose: 19.6 man-rem
2. Annual Total Population Doses from Liquid Effluent Releases:
    - Total Body Dose: 1.89 man-rem
    - Thyroid Dose: 2.86 man-rem
    - Maximum Organ Dose (GI-LLI): 4.88 man-rem

11.D EXPECTED ANNUAL INHALATION DOSES AND ESTIMATED AIR CONCENTRATIONS OF RADIOACTIVE ISOTOPES FOR MP2 FACILITIES

The expected annual inhalation doses to plant personnel and estimated air concentrations were computed as outlined below:

Equilibrium airborne concentrations:

Equilibrium concentration = (Production/Losses) = fA/λ

where: f = flow rate of activity source (Ci/sec)

A = activity of source (Ci/cc)

λ = loss rates = $\Sigma\lambda d = \lambda e + \lambda R$

λe = leakage or exhaust

λR = recirculation for iodines, if any

Doses:

The whole body dose is determined using a semi-infinite cloud model (Safety Guide 4). The thyroid inhalation dose is found using the inhalation model and dose conversion factors of TID 14844 and Safety Guide 4. The other organ doses are found by the fraction of allowable MPC, which yields the limiting doses.

Whole Body Doses

Lung Dose:

$$D_I Lung = \frac{C_I}{MPC_I Lung} \times MPD^{Lung}$$

where: C_I = Equilibrium concentration of the I^{th} isotope

$MPC_I Lung$ = Maximum permissible concentration for the I^{th} isotope for lung based on a 40 hour workweek taken from ICRP II for the airborne case

MPD^{Lung} = Maximum permissible yearly dose to lung

For Cs 138 and Rb 88, D_I^{Lung} is computed as follows:

$D^{Lung} = 2.81 \times 10^5 \times T \times E \times C$

where: Y = radiological half life

E = effective energy absorbed per disintegration in MeV/disintegration

C = Concentration in μCi/cc of CS^{138} or ^{88}Rb.

	Y Min.	E(MeV/dis)
CS 138	32.2	1.39
Rb 88	17.8	1.68

The total lung dose D^L is found as follows:

$$D^L = \sum_{I=1}^{n} D_I Lung$$

Thyroid Dose:

$D_I Thyroid = CI \times t \times BR \times DCF_I$

where: $D_I Thyroid$ = Thyroid dose in REMs

C_I = Concentration of the Iodine isotope I

t = Exposure time in day = 300 days

BR = Breathing rate in cc/day = 2×10^7 cc/day

DCF = Dose conversion factors for Iodine in REMs/Ci per TID 14844

The total thyroid dose is found as follows:

$$D^T = \Sigma_{I=1}^{n} D_I Thyroid$$

Bases and Results:

Source terms are based on reactor operation with 0.1 percent failed fuel.

Containment Building:

1. Containment volume - 1.92×10^6 ft^3

2. Purge containment for a minimum of five hours at 32,000 CFM prior to personnel entering the containment (see FSAR Section 9.9.2).

3. Activity source is conservatively assumed to be 40 GPD of reactor coolant leakage. The expected and calculated leakage of reactor coolant to the containment atmosphere will be a small fraction of the 40 GPD rate used in the analysis. The 40 GPD leakage rate was taken for "Twelfth AEC Air Cleaning Conference-Analysis of Power Reactor Gaseous Waste System."

4. Assume 75 days of activity buildup in the containment.

5. A decontamination factor of 10, used for iodines and particulates, was taken from "Twelfth AEC Air Cleaning Conference-Analysis of Power Reactor Gaseous Waste System."

The results indicate the whole body dose rate will be 0.85 mr/hr. After 10 hours, the dose rate decreases to 0.44 mrem/hr. Concentrations, given in Table 11.D-1, are below the 10 CFR Part 20 allowable airborne concentrations.

Auxiliary Building (Excludes nonradioactive areas and fuel handling area):

1. Assume 20 GPD leakage of 120°F fluid and 1 GPD leakage of 550°F fluid with reactor coolant concentrations of activity.

2. Ventilation air flow is 40,000 CFM for radioactive areas of auxiliary building. This corresponds to approximately five air changes per hour.

3. Iodine and particulate partition factors assumed as follows:

   Coolant at 120°F - 10^{-4}

   Coolant at 550°F - 10^{-1}

4. 2000 hours of exposure per year.

Doses based on 40 hours per week, 52 weeks per year exposure are as follows:

Lung Dose - 0.046 REM

Whole Body Dose - 0.468 REM

Thyroid Dose - 0.61 REM

Airborne concentrations are given in Table 11.D-2.

Auxiliary Building (Fuel handling area):

1. Sources arise due to evaporation from the spent fuel pool surface.

2. Sources assumed as one-tenth of reactor coolant concentrations (0.1% failed fuel). Only tritium is considered since separation factors for iodines and particulates in

spent fuel pool water will greatly reduce airborne concentrations. Contribution from noble gases will be insignificant.

3. Evaporation rate from spent fuel pool at 120°F is 0.37 gpm.

4. There will be 20 air changes per hour at average exhaust rate of 18,000 CFM (FSAR Section 9.9).

5. Only whole body dose is considered as the critical organ for tritium.

6. Exposure calculated for 2000 hrs/year.

The airborne equilibrium concentration for tritium is 1.48×10^{-8} µCi/cc. The whole body dose is 25.2 mr/yr.

Turbine Building

1. Sources arise due to leakage from the secondary systems into the turbine building.

2. Primary coolant system operating at 0.1% failed fuel.

3. Primary to secondary leakage of 100 gallons per day.

4. 5 gpm continuous blowdown per steam generator.

5. Processed blowdown liquid is discharged.

6. Feedwater cleanup factor is 0.

7. Steam leakage from the secondary side is 1700 lb/hr.

8. Liquid leakage from the secondary side is 15 gpm.

9. Volume of the turbine building 2,882,000 ft^3.

10. Exhaust flow rate is 200,000 cfm.

Doses based on 40 hours per week, 50 weeks per year exposure are as follows:

Lung Dose - 7.8×10^{-3} REM

Whole Body Dos - 2.4×10^{-5} REM

Thyroid Dose - 0.314 REM

Airborne concentrations are given in Table 11.D-3.

TABLE 11.D-1 CONTAINMENT BUILDING AIRBORNE CONCENTRATIONS

ISOTOPE	CONCENTRATION μCi/cc
I-131	1.31×10^{-8}
I-132	9.75×10^{-10}
I-133	7.46×10^{-9}
I-134	3.99×10^{-10}
I-135	2.95×10^{-9}
Xe-131M	6.35×10^{-8}
Xe-133M	--
Xe-133	4.60×10^{-6}
Xe-135M	--
Xe-135	8.61×10^{-8}
Xe-137	--
Xe-138	1.24×10^{-9}
Kr-83M	--
Kr-85M	1.52×10^{-8}
Kr-85	1.35×10^{-7}
Kr-87	6.07×10^{-9}
Kr-88	2.43×10^{-8}
Kr-89	--

TABLE 11.D-2 AUXILIARY BUILDING AIRBORNE CONCENTRATIONS

ISOTOPE	CONCENTRATION µCi/cc
Kr-85M	8.90×10^{-9}
Kr-85	5.45×10^{-9}
Kr-87	4.52×10^{-9}
Kr-88	1.53×10^{-8}
Xe-131M	9.10×10^{-9}
Xe-133	1.11×10^{-6}
Xe-135	4.56×10^{-8}
Xe-138	1.42×10^{-9}
I-129	2.96×10^{-18}
I-131	1.63×10^{-10}
I-132	4.23×10^{-11}
I-133	2.31×10^{-10}
I-134	2.21×10^{-11}
I-135	1.09×10^{-10}
Br-84	1.53×10^{-12}
Rb-88	7.24×10^{-11}
Rb-89	1.72×10^{-12}
Sr-89	2.08×10^{-13}
Sr-90	1.07×10^{-14}
Sr-91	1.44×10^{-13}
Te-129	9.24×10^{-13}
Te-132	1.35×10^{-11}
Te-134	9.08×10^{-13}
Be-140	2.50×10^{-13}
Ru-103	1.69×10^{-13}

TABLE 11.D-2 AUXILIARY BUILDING AIRBORNE CONCENTRATIONS

ISOTOPE	CONCENTRATION µCi/cc
Ru-106	1.02×10^{-15}
La-140	2.39×10^{-13}
Ce-144	1.69×10^{-13}
Pr-143	2.39×10^{-13}
Mo-99	8.31×10^{-11}
Y-90	4.17×10^{-14}
Y-91	4.51×10^{-12}
Cs-134	4.10×10^{-12}
Cs-136	1.05×10^{-12}
Cs-137	1.31×10^{-11}
Cs-138	2.27×10^{-11}
Cr-51	1.56×10^{-14}
Mn-54	1.31×10^{-14}
Fe-59	8.73×10^{-15}
Co-58	1.91×10^{-12}
Co-60	2.13×10^{-13}
Zr-95	3.83×10^{-16}

TABLE 11.D-3 TURBINE BUILDING AIRBORNE CONCENTRATIONS

ISOTOPE	CONCENTRATION Ci/cc
H-3	5.14E-10
Cr-51	1.59E-15
Mn-54	1.21E-15
Fe-59	9.08E-16
Co-58	2.01E-13
Co-60	2.29E-14
Br-84	1.09E-14
Kr-85M	4.61E-12
Kr-85	2.88E-12
Kr-87	2.19E-12
Kr-88	7.80E-12
Rb-88	5.74E-14
Rb-89	1.17E-15
Sr-89	2.17E-14
Sr-90	1.15E-15
Sr-91	3.24E-15
Y-90	2.85E-15
Y-91	4.72E-13
Zr-95	4.04E-17
Mo-99	5.75E-12
Ru-103	1.75E-14
Ru-106	1.09E-16
Te-129	2.96E-15
Te-132	9.90E-13
Te-134	1.80E-15
I-129	1.56E-18
I-131	7.22E-11
I-132	1.29E-12
I-133	4.32E-11

TABLE 11.D-3 TURBINE BUILDING AIRBORNE CONCENTRATIONS

ISOTOPE	CONCENTRATION Ci/cc
I-134	2.60E-13
I-135	8.81E-12
Xe-131M	4.83E-12
Xe-133	5.89E-10
Xe-135	2.41E-11
Xe-138	5.88E-13
Cs-134	4.40E-13
Cs-136	1.01E-13
Cs-137	1.41E-12
Cs-138	3.36E-14
Ba-140	2.40E-14
La-140	1.34E-14
Ce-144	1.81E-14
Fr-143	2.24E-14

Where E-14 = 10^{-14}, etc.

11.E AIRBORNE ACTIVITY SAMPLING SYSTEM FOR CONTAINMENT, SPENT FUEL AND RADWASTE ATMOSPHERES

In-plant control of personnel exposure from airborne radioactivity will be effected by a continuing program of sampling for airborne activity and administrative controls through radiation work permits.

1. Containment atmosphere is continuously monitored by two sample lines inside containment. The sample is pulled through a fixed particulate filter, a charcoal cartridge and a gas chamber before being pumped back to the containment. Particulate activity is measured by a beta scintillator through a derivative (rate of change) circuit. Any rapid changes in beta activity will be alarmed in the control room. Gaseous iodine is absorbed on the charcoal cartridge. The particulate filter is changed on a routine schedule to prevent excessive dust loading on the filter paper. The charcoal cartridge is normally changed in conjunction with the particulate filter. The removed filter and cartridge may be counted in accordance with approved station procedures, if needed, to evaluate the airborne radioactivity concentration during the collection period. Gaseous activity is measured with a beta scintillator type detector. Increased activity above pre-set levels will alarm in the control room.

   Upon indication of a significant increase in activity in the containment building, the Shift Supervisor will notify radiation protection. Radiation protection will obtain and analyze samples of containment air to determine the isotopes responsible for the increased activity. During power operation, if conditions warrant, radiation protection personnel may enter the containment and take portable air samples in selected areas in order to locate the source of the activity. Appropriate respiratory equipment will be indicated on the radiation work permits.

   During shutdown and maintenance conditions, air samples will be taken with the portable sampling equipment throughout the containment building on a scheduled basis and as required, to determine the presence of any unusual concentrations of airborne activity. Appropriate respiratory protection, if required, will be indicated on the radiation work permits.

   While work is being performed on the reactor vessel head and there is a potential for release of airborne activity, continuous radiation protection monitoring of the work area will be in effect. This monitoring will consist of particulate and iodine sampling. Workers will be instructed to leave the area if they suspect any unusual problem.

   The spent fuel pool exhaust monitor is a continuous sampling system that takes suction upstream to the HEPA filter. This sample passes through a particulate filter, charcoal cartridge and gaseous 4 pi beta, gamma detection chamber. The particulate filter is changed on a routine schedule to prevent excessive dust loading on the filter paper. The charcoal cartridge is normally changed in conjunction with

the particulate filter. The removed filter and cartridge may be counted in accordance with approved station procedures, if needed, to evaluate the airborne radioactivity concentration during the collection period. The gaseous channel is monitored and alarmed in the control room. An alarm condition will require notification of radiation protection to determine the source of the increased activity. Work area air sampling will be established consistent with spent fuel pool ventilation line up and work in progress. These samples will supplement the spent fuel pool exhaust monitor. During periods when the spent fuel pool exhaust or the containment atmosphere monitors are not available, air samples are taken with portable sampling equipment in these areas on a scheduled basis per the Millstone Effluent Control Program to determine the presence of any unusual concentrations of airborne activity.

2. The area with the highest potential for creating airborne activity is the sampling room. All other radwaste areas contain equipment that is closed and therefore of a low potential for creating airborne activity.

   The sampling room (primary sample sink area) has a separate ventilation system via the sample hood. The hood design is such that at least 100 linear feet per minute flow is maintained over the working area in the hood to ensure that there is no contamination or airborne activity spread into the sampling room. Visual indication is available to the personnel frequenting the sampling room to indicate the condition of the exhaust system. A shutdown exhaust system would require health physics evaluation of airborne conditions prior to entry.

3. The radwaste airborne radioactivity monitoring system consists of four radiation monitors, each capable of continuously monitoring airborne radioactive particulates. In addition, each monitor contains an iodine sampling assembly consisting of a replaceable charcoal cartridge mounted in series and downstream of the particulate monitor. The particulate filter is changed on a routine schedule to prevent excessive dust loading on the filter paper. The charcoal cartridge is normally changed in conjunction with the particulate filter. The removed filter and cartridge may be counted in accordance with approved station procedures, if needed, to evaluate the airborne radioactivity concentration during the collection period.

   Since the waste gas system of the radwaste areas has the potential for the release of gaseous activity, the airborne radiation monitor sampling the ventilation exhaust duct servicing that area also contains a gaseous radioactivity monitor in series with the particulate and iodine monitor. The gaseous monitor will be used to detect and measure significant noble gas releases from the waste gas system.

   Sampling for each monitor is accomplished by extracting a representative sample from the radwaste ventilation exhaust system by using an isokinetic nozzle designed in accordance with ANSI N13.1-1969, Guide to Sampling Airborne Radioactive Materials in Nuclear Facilities.

Since the airflow at each point selected represents the exhaust of several compartments, the effect of dilution on the compartment having the least flow was analyzed to demonstrate the capability of detection and measurement of airborne contaminants in the areas being served by that section of the ventilation exhaust system.

For this analysis, a concentration of $3x10^{-9}$ microcuries per cubic centimeter was used. To determine the radioactive concentration at the sampling point, this value was multiplied by the ratio of the flow for the compartment having the lowest airflow to the total flow at the sample point.

Point A (RM-8999 P&ID 25203-26029)

The airflow at this sampling point is 15,495 scfm. The ventilation exhaust system at this point services the storage area, the offices, the heat exchanger pump area, the coolant waste tank areas, and the evaporator rooms.

The airflow for the compartment having the least flow exhausting into this section of the ventilation exhaust system is 800 scfm. Assuming the concentration of $3x10^{-9}$ microcuries per cubic centimeter exists in this room with the least flow, the radioactive concentration C_A at this point is:

$$C_A = 3 \times 10^{-9} \ \mu Ci/cc \ \frac{800 \ scfm}{15,495 \ scfm}$$
$$= \underline{1.548 \times 10^{-10} \mu Ci/cc}$$

Manufacturer's data for the radiation monitoring equipment states that the monitoring equipment placed in a 1 mrem/hr background of 1 Mev gamma energies produces a count rate of 78 cpm. Concentrations of $1x10^{-11}$µCi/cc (limiting isotope Cs-137) produces a count rate 107 cpm. Therefore, the count rate for the concentration at point A is equal to 1656 cpm.

Point B (RM-8998 P&ID 25203-26029)

The airflow at this sampling point is 18,335 scfm. The ventilation exhaust system at this point services the sampling room, the letdown heat exchanger room, the coolant waste receiver area, and the volume control tank area.

The airflow for the compartment having the least flow exhausting into this section of the ventilation exhaust system is 160 scfm. Assuming the value of $3x10^{-9}$µCi/cc exists in this area, the radioactive concentration C_B at this point is:

$$C_B = 3 \times 10^{-9} \ \mu Ci/cc \left(\frac{160 \ scfm}{18,335 \ scfm}\right)$$

$$= \underline{2.61 \times 10^{-11} \mu Ci/cc}$$

Assuming the 78 cpm for the background counting rate and the 107 cpm for the concentration $1x10^{-11}$µCi/cc (Cs-137), the count rate for point B is 279 cpm.

Point C (RM-8997 P&ID 25203-26029)

The airflow at this sampling point is 10,170 scfm. The ventilation exhaust system at this point services the charging pump areas, the degasifier areas, and the engineered safety features areas. The airflow for the compartment having the least flow exhausting into this section of the ventilation exhaust system is 235 scfm. Assuming the value of $3x10^{-9}$ microcuries per cubic centimeter exists in this area, the radioactive concentration C_C, at this point is:

$$C_C = 3 \times 10^{-9} \ \mu Ci/cc \left(\frac{235 \ scfm}{10,170 \ scfm}\right)$$

$$= \underline{6.93 \times 10^{-11} \mu Ci/cc}$$

Assuming the 78 cpm for the background count rate and the 107 cpm for the concentration 1 x 10^{-11}µCi/cc (Cs-137), the count rate for the concentration at point C is 742 cpm.

Point D (RM-8434 A&B P&ID 25203-26029)

The air flow at this sampling point is 17,920 scfm. The ventilation system at this point services the equipment laydown area, the waste gas decay system areas, the aerated waste gas system area, and the RBCCW heat exchanger area. The airflow for the compartment having the least flow exhausting into this section of the ventilation exhaust system is 800 scfm. Assuming the value of $3x10^{-9}$µCi/cc exists in this area, the radioactive concentration, C_D, at this point D is:

$$C_D = 3 \times 10^{-9}\ \mu Ci/cc\ \frac{800\ scfm}{17,920\ scfm}$$
$$= \underline{1.34 \times 10^{-10} \mu Ci/cc}$$

Assuming the 78 cpm for the background count rate and 107 cpm for the concentration 1 x 10^{-11}µCi/cc (Cs-137), the count rate for the concentration at point D is 1434 cpm.

The calculations and analysis for the four sampling points shown above indicate a count rate ranging from 279 counts per minute to 1656 counts per minute. Since these count rates are greater than three times the background rate, as specified by the manufacturer, the concentration shown in the above calculations can be detected and alarmed.

The above analysis is conservative in that the value used is based on an exposure to the concentrations specified for forty hours in any period of seven consecutive days. However, since the normal occupancy for these areas is significantly less than this 40-hour period, the exposure of personnel to airborne radioactivity will be considerably less than that stated in the above calculations.

In conclusion, the above analysis clearly demonstrates the capability of the airborne radiation monitoring system to detect and measure appropriate concentrations in the radwaste ventilation exhaust system and thus to comply with the requirements for personnel safety stated in 10 CFR 20.103 and 10 CFR 20.203(d).