11.1 SOURCE TERMS

The ultimate source of radioactivity for all plant systems and radioactive releases from the site is the reactor core. The fission product inventory in the reactor core is presented in Chapter 15. Additional sources of radioactivity considered in the design of the Radioactive Waste Management Systems and for the determination of plant releases are corrosion and activation products. These sources of radioactivity are generated by the unavoidable interaction of neutrons from the reactor core with corrosion particles in the coolant system and with air molecules in the Reactor Containment. Quantitative values for these radionuclides are taken from reactor operating experience and are considered as part of the overall source term.

11.1.1 Design Basis

The concept of radioactive waste management involves the examination of all potential pathways of radioactive release to the environment and the provision of appropriate processing and treatment equipment to ensure that release of radioactivity to the environment is kept as low as is reasonably achievable (ALARA) in compliance with Section 50.34a of 10 CFR Part 50. Appendix I to 10 CFR Part 50 provides numerical guides for those design objectives to meet the criterion ALARA. The plant operates within the limits of radiation levels set forth in 10 CFR Part 20.

On May 21, 1991, a complete revision to 10 CFR 20 was issued. Several design bases reference 10 CFR 20 and specific terms or parts of 10 CFR 20. Design bases information provides a historical perspective of the information used to formulate a particular design. References to 10 CFR 20 when used in a historical or design bases context have not been changed to reflect the revised 10 CFR 20.

The transport of radioactivity from the primary Reactor Coolant System to various parts of the plant during normal operation is traced and evaluated in order to determine the performance of each process interposed between the source of radioactivity and the subsequent pathways to the environment.

There are three radioactive waste treatment systems: the liquid radwaste systems (Liquid Waste System, Boron Recovery System and Steam Generator Blowdown System), the Radioactive Gas Waste System, and the Solid Radwaste Handling System. Potentially radioactive liquids, gases and solids are collected and processed according to physical and chemical properties and radioactive concentrations as deemed necessary. Care is taken in design to minimize the mechanical leakage paths in these systems in order to limit unprocessed leakage.

The calculation of potential offsite doses considers all possible pathways to the environment, including estimated leakage of radioactive process streams into the Containment, and into the auxiliary buildings, which are eventually released to the environment through building ventilation systems. The radioactive source terms used for design of shielding and building ventilation systems have been closely reviewed in conjunction with effluent releases to the environment to ensure that these source terms are not in excess of the acceptable release guidelines set forth in 10 CFR Part 50, Appendix I.

11.1.1.1 Failed Fuel

The concept of failed fuel determines what fraction (or percentage) of the reactor core fission product inventory is assumed to be released to and contained within the Primary Coolant System. Three sets of source terms (reactor coolant radionuclide concentrations) have been determined, using three different values for the assumed failed fuel fraction:

a. The coolant radionuclide concentrations based on a 1 percent failed fuel fraction to establish design basis values to be used for systems and shielding calculations

b. Reactor coolant radionuclide concentrations based on 0.25 percent failed fuel for design of the plant ventilation systems

c. Reactor coolant radionuclide concentrations based on 0.12 percent failed fuel to conform to the models and procedures described in Regulatory Guide 1.112 "Calculation of Releases of Radioactive Materials in Liquid and Gaseous Effluents from Pressurized Water Reactors (PWRs)." The inventories calculated in this manner represent "expected basis" activities and will be used for the evaluation of environmental impacts during normal operation.

With the exception of the Primary Coolant System source term based on 1% failed fuel, the analysis and parameters described in this section are historical and are the basis of the 10CFR Part 50 Appendix I program. It is noted that for an operating plant, the actual performance and operation of installed equipment, the reporting of offsite releases and doses, and compliance with the regulatory limits of 10CFR50 Appendix I, 10CFR20, and 40CFR190 are controlled by the Offsite Dose Calculation Manual.

11.1.1.2 Fission, Corrosion and Activation Product Activities

a. Activity in the Primary Coolant System

"Design basis" (1 percent and 0.25 percent failed fuel values) activity inventories in the reactor coolant are provided in Table 11.1-1. These inventories have been used for designing the waste management and ventilation systems components.

"Expected basis" (0.12 percent failed fuel values) activity inventories are also provided in Table 11.1-1.

b. Radioactivity in the Secondary Coolant System

Primary-to-secondary leakage of reactor coolant through the steam generator tubes will result in the contamination of steam generator secondary side liquid with radionuclides from the reactor coolant. The secondary side radioactivity levels are dependent on the reactor coolant radionuclide concentration, the primary-to-secondary leak rate, and the rate of steam generator secondary side blowdown. The anticipated values for these parameters for design and normal operating conditions are as follows:

1. Primary coolant radionuclide concentrations as given in Table 11.1-1 (0.12 percent values)

2. Primary-to-secondary leak rate of 100 lbm per day (total for 4 steam generators)

3. Steam generator blowdown rate of 75 gpm (total)

4. The carryover due to mechanical entrainment in the steam generators is 0.1 percent for all particulates except halogens, and 1 percent for halogens

5. Volatile chemistry is used with condensate demineralization available during startup and condenser leakage events.

These conditions combine to generate equilibrium steam generator secondary side activity concentrations as given in Table 11.1-4 and Table 11.1-5. Both design level activities (0.25 percent failed fuel) and expected levels of activity for normal operation (0.12 percent failed fuel) are presented.

11.1.1.3 Tritium

The liquid waste design principle of recycling all the reactor coolant from letdown and leakage that meets the reactor water chemistry specifications has potential long-term ramifications in regard to tritium levels within the plant systems and work areas. Tritium will be produced by ternary fission in the fuel, with subsequent escape from the fuel to the reactor coolant through the zircaloy cladding. Tritium will also be produced by neutron activation of boron and lithium present in the reactor coolant. Analysis shows that the tritium buildup in reactor coolant and associated liquids, as a result of recycling the intended amounts of reactor coolant, must be controlled so that containment accessibility is not unduly limited. This can be accomplished by periodic discharge or by feed and bleed of the reactor coolant.

The total tritium release through the combined liquid and vapor pathway is 0.40 Ci/yr/MWt, based on operational experience in PWRs. (NUREG-0017, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors," April 1976.)

The quantity of tritium released through the liquid pathway is based on the calculated liquid release volume, assuming a tritium concentration in the released liquid of 1.0 µCi/cc. Secondary system wastes are excluded. In accordance with Regulatory Guide 1.112, only a maximum of 50 percent of the total quantity of tritium calculated to be available for release can be assumed to be released via the liquid pathway. The remainder of the tritium is assumed to be released through the vapor pathway.

11.1.2 Mathematical Models and Parameters

11.1.2.1 Design Bases Activity Levels

The parameters used in the calculation of the design level inventories are summarized in Table 11.1-2. In these calculations, the defective fuel rods are assumed to be present at the initial core loading and to be uniformly distributed throughout the core; thus, the fission product escape rate coefficients are based upon average fuel temperature. It is also assumed that all isotopes are formed directly from fission and/or from the decay of parent isotopes that are also fission products. Two isotope decay chains are assumed throughout. These products are then released into the reactor coolant through small defects in the fuel cladding. Such release is assumed to be continuous and equal to some constant fraction of the core inventory of each individual isotope. The mechanisms by which radioactive material is removed from the coolant are:

- Radioactive decay
- Material flow through a purification system
- Leakage and letdown.

The differential equations governing the numbers of atoms of the various isotopes in the core and the coolant are as follows:

In the fuel

$$\frac{dA}{dt} = P_A - \lambda_1 A$$

$$\frac{dB}{dt} = P_B + \lambda_A A - \lambda_2 B$$

In the coolant

$$\frac{d\overline{A}}{dt} = f \alpha_A A - \lambda_3 A$$

$$\frac{d\overline{B}}{dt} = f \alpha_B B + \lambda_A A - \lambda_4 B$$

where:

A = total number of atoms of parent in core

B = total number of atoms of given isotope in core

A = total number of atoms of parent in coolant

B = total number of atoms of given isotope in coolant

P_A = production rate of parent in core = $(Y_2 - Y_1)F$ (atoms/sec)

P_B = production rate of isotope in core = Y_1 F (atoms/sec)

λ_A = radioactive decay constant of parent (sec^{-1})

λ_B = radioactive decay constant of isotope (sec^{-1})

Y_1 = direct fission yield of isotope in question

Y_2-Y_1 = direct fission yield of parent of give isotope

σ = thermal neutron absorption cross section of given isotope (cm^2)

$ø$ = average thermal neutron flux in core (n/cm^2 - sec)

α_A = fuel escape coefficient of parent (sec^{-1})

α_B = fuel escape coefficient of isotope (sec^{-1})

$ß_A$ = removal rate of parent from coolant (fractions/sec)

$ß_B$ = removal rate of isotope from coolant (fractions/sec)

f = failed fuel fraction

F = fissions per second in the core = $3.121 \times 10^{10} \times (P \times 10^6)$

$\lambda_1 = \lambda_A$

$\lambda_2 = \lambda_B + ø\sigma$

$\lambda_3 = \lambda_A + ß_A$

$\lambda_4 = \lambda_B + ß_B$

The activities of the various isotopes in the fuel and coolant are:

$D_f(t) = \lambda_B B(t)/3.7 \times 10^{10}$ (curies)

$D_c(t) = \lambda_B B(t)/(V_c \times 3.7 \times 10^{10})$ (curies/m^3)

where V_c is the coolant volume (m^3)

11.1.2.2 Expected Bases Activity Levels

Expected bases activity levels for primary and secondary coolant water are calculated based on Regulatory Guide 1.112, NUREG-0017 and the USNRC PWR GALE code. Input parameters used to derive coolant activity levels and plant liquid and gaseous releases are given in Appendix 11A, and are summarized in Table 11.1-3.

11.1.3 Source Terms for Shielding and Component Failures

See Section 12.2.

11.1.4 Radioactive Leakage and Estimated Contributions

11.1.4.1 Sources of Leakage

Westinghouse has surveyed various PWR facilities to identify design and operating problems influencing nonrecyclable reactor coolant leakage, and hence the load on the Waste Disposal System. Leakage sources have been identified in connection with pump shaft seals and valve stem leakage.

11.1.4.2 Estimated Contribution to Total

As discussed in Appendix 11A, Section 2d, the equipment leakage rate of primary coolant is estimated to be 300 gallons per day (0.21 gpm). This source of leakage is assumed to occur within the Primary Containment and is collected in the primary drain tank (PDT) before processing by the PDT Degassifier and Boron Recovery System.

Additional sources of recyclable primary grade leakage are expected to occur outside the Containment (within the Primary Auxiliary Building), and will be collected in the aerated waste recovery tank. The quantity of leakage assumed is 160 lbs. of primary coolant per day.

Leakage rates from the Reactor Coolant System and other fluid systems containing radioactivity into individual cubicles and areas that may require access are discussed in Section 12.2.

11.1.5 Effluent Releases from Other than Radioactive Waste Systems

This section discusses the effluent releases to the environment from other than radioactive waste systems during normal operations. Estimated releases from anticipated operational occurrences are discussed in Subsections 11.2.3 and 11.3.3.

11.1.5.1 Gaseous Effluent Releases

The Radioactive Gaseous Waste System has been designed to process the gaseous wastes generated in the plant, and is discussed in Section 11.3. Normally anticipated effluent releases from the system will be on a controlled basis. However, any leakage of primary coolant or the process stream either in the Containment or in the auxiliary buildings is collected in the buildings and vented through filtration systems to the environment. Any steam/water leakages in the Turbine Building are directly vented to the environment. The noncondensable gases will be also discharged through the main condenser vacuum system exhaust.

a. Containment Purges

Four purges are expected annually for shutdown, annual fuel loading, and planned maintenance. The duration of shutdown and pre-entry containment purges is expected to be 24 hours per purge. In addition, an online purge system is available for use during power operation. This system will be used to reduce containment airborne activity levels in anticipation of containment entry by plant personnel. To evaluate airborne releases from containment venting, a continuous 1000 scfm online purge rate is used.

b. Steam Generator Blowdown System

Steam Generator Blowdown System waste liquids are processed either through the blowdown demineralizer subsystem or the Liquid Waste Processing System. The normal method of processing blowdown fluids is through the blowdown demineralizer subsystem, which returns the liquid to the main condenser. This includes conditions of minor primary-to-secondary leakage. If necessary, this liquid may also be processed through the installed vendor system (WL-SKD-135). Blowdown flash tank bottoms would be transferred to the floor drains tanks and then through this skid for treatment prior to discharge. The flash tank steam can be either returned to the No. 3 feedwater heater, or processed through the flash steam condenser cooler to the waste test tanks, prior to discharge. Noncondensable gases contained within the Secondary Coolant System are released via the main condenser vent system and as such are considered as part of the gaseous releases source term for the main condenser evacuation system.

c. Primary Auxiliary Building Ventilation

The activity from leakage into the PAB is assumed to be released directly to the environment through the filtration system. The partition factors are 0.0075 for iodines and 1.0 for noble gases. The leakages are mostly equipment leakages, and take place at valves, flanges and pump seals. It is assumed that there is an inleakage of 160 lbs. of primary coolant per day into the Primary Auxiliary Building.

d. Turbine Building Ventilation

One hundred percent of the steam leakage is assumed to be released to the environment. Steam leakage from the Secondary Coolant System to the Turbine Building is assumed to be 1700 lbs. per hour.

e. Main Condenser Vacuum Pump

An iodine partition factor of 0.15 is used for volatile iodine species in the main condenser. All noble gases and a small part of the iodines in the condensing steam are considered to be released through the Primary Auxiliary Building charcoal filtration system. The main condenser vacuum pump is assumed to operate at 2 inches Hg and 60 scfm. Annual releases from the main condenser vacuum pump are calculated using an equilibrium model.

11.1.5.2 Liquid Effluent Releases

The Radioactive Liquid Waste System is designed to process liquid wastes generated in the plant, and is discussed in Section 11.2. In this section, the liquid effluent releases for tritium control in the Primary Coolant System and the condensate leakage in the Turbine Building are discussed.

a. Processed Reactor Coolant Releases for Tritium Control

It is assumed that 200,000 gallons of reactor coolant is released per year for tritium control. This liquid is treated by the Boron Recovery System prior to release. The process and discharge period is 120 hours.

b. Unprocessed Liquid Waste from the Turbine Building Floor Drain Sump and Water Treatment Liquid Effluent that includes Condensate Polishing System

It is assumed that the Turbine Building floor drains and Water Treatment Liquid Effluent that includes Condensate Polishing System will collect leakage/discharge at 7550 gallons per day at main steam activity. No credit for decay is taken in these calculations.

11.1.6 Operations Resulting in Radioactive Releases to the Environment

Operational activities resulting in radioactive liquid and gaseous releases to the environment are discussed in Subsections 11.2.3 and 11.3.3, respectively. Expected and design conditions are discussed, including anticipated releases and projected doses.

11.1.7 Operating Reactor Experience

Operating reactor experience has been incorporated into the development of Seabrook's source terms, by using the methodology described in NRC Regulatory Guide 1.112, NUREG-0017 and the PWR GALE computer program. These regulatory guides and computer codes are based on standardized primary and secondary coolant activity derived from American Nuclear Society (ANS) 18.1 Working Group recommendations. These recommendations, in turn, are based on several years of reactor operating experience.

11.1.7.1 Fuel Performance

Seabrook's "realistic" source terms have been developed using the ANS standard referenced above, and as such represent the radionuclide inventories and releases which are expected with 0.12 percent fuel cladding defects, and the escape and release rate coefficients given in Table 11.1-2. A value of 0.12 percent is the weighted average of the fuel performance data, based on the burnup rates, for operational experience with zircaloy-clad fuels in PWRs collected for the operating years of January 1970 to July 1973.

11.1.7.2 Leakage Sources and Pathways

Leakage sources and pathways for liquid and gaseous radioactive releases to the environment are discussed in Subsections 11.2.3 and 11.3.3, respectively.

11.1.7.3 Releases of Radioactivity to the Environment

Releases of liquid and gaseous radioactivity to the environment from planned and anticipated operational occurrences are presented in Subsections 11.2.3 and 11.3.3, respectively.

11.2 LIQUID WASTE SYSTEM

This section describes the manner in which radioactive liquid wastes are collected, processed and directed for either reuse or release from the site. The equipment involved and its operation is described, the amount of radioactivity in liquid effluents is estimated and, finally, the radiological environmental impact from such releases is evaluated.

11.2.1 Design Bases

The Liquid Waste System (WL) is nonnuclear safety class (NNS) and nonseismic Category I, in accordance with Regulatory Guides 1.26 and 1.29.

The Liquid Waste System is designed to meet applicable requirements specified in 10 CFR, Parts 20 and 50, as follows:

a. Provide a central collection point for radioactive liquid waste. This includes approximately 1200 gallons per week of reactor grade and nonreactor grade leakage from various systems (see Subsection 9.3.3) and approximately 400 gallons per week of floor drainage from area wash down.

b. Provide preliminary processing through the use of a strainer and filters.

c. Concentrate nonvolatile and, to some extent, volatile radioactive liquid contaminants, through evaporation, with a minimum decontamination factor (D.F. = Ratio of specific activity in the bottoms and distillate) of 10^4, at a bottoms concentration of 12 percent by weight.

d. Concentrate the residual contaminants (bottoms) up to 12 percent total dissolved solids (TDS) for transfer to the Waste Solidification System (Section 11.4).

e. Produce up to 25 gpm of distillate from the evaporator/condenser. The distillate is demineralized (if necessary) and tested in the WL waste test tank before disposal offsite.

f. Maintain, during normal operation, the radioactivity content of liquid effluents from the Seabrook site within the concentration limits expressed in 10 CFR 20, Appendix B, Table II, Column 2, on an instantaneous release basis and on an annual average release basis to maintain the radioactive liquid effluents so that the dose guidelines expressed in the Appendix I to 10 CFR 50 are not exceeded.

g. Provide processing equipment and capacity sufficient to maintain radioactivity in liquid effluents within the applicable flexibility provisions of Appendix I to 10 CFR 50 during anticipated operational occurrences.

11.2.2 System Description

11.2.2.1 Operation

The primary means of processing liquid radioactive water is through a vendor-supplied demineralizer system. Other systems or components may be used if necessary.

The Liquid Waste System, Figure 11.2-1, Figure 11.2-2, Figure 11.2-3 and Figure 11.2-4, is located in the Waste Processing Building, a seismic Category I structure. Backup processing capability is available from the Boron Recovery System (BRS) evaporator BRS-EV-3A (Subsection 9.3.5). Collection of the waste liquid is continual as the waste is generated. Processing is in batch mode. No feed occurs into the evaporator when it is discharging the concentrated liquid. The system can be used during normal operation, plant startup, shutdown, and refueling operations as long as electrical power and component cooling water systems do not fail. No emergency power or cooling water is available to this system. In the case of unavailability of main plant steam during shutdown, a limited amount of steam is available from the auxiliary boiler, depending upon other uses of steam at that time. In the event that steam and cooling water are not available during a shutdown, the waste evaporators will not be run. Residual waste liquid will be processed via a skid-mounted waste liquid processing system or transferred to the floor drain tanks for future processing.

The major system functions are discussed below.

a. Liquid Waste Storage

The following sources pass through a strainer and are stored in the floor drain tanks for further processing:

1. Liquid from the chemical drain treatment tanks

2. Effluent from the Resin Sluicing System

3. Liquid from the Steam Generator Blowdown System when that system requires additional processing capacity

4. Liquid from boron waste storage tanks when that liquid is unacceptable for reuse in the reactor plant

5. Recycled effluent from liquid waste evaporation and testing and demineralization when reprocessing is required
6. Demineralized water for system startup and flushing operations
7. Waste liquid from the spent fuel pool skimmer pump
8. All sumps in contaminated plant areas, and the Administration and Service Building sump and the RCA walkway B&C sumps.

Prior to further processing, the pH of the liquid can be adjusted with the liquid waste chemical addition pump by recirculating the tank contents with the floor drain tank pump.

From the floor drain tanks, liquid is transferred by one of the two floor drain tank pumps to one of the following:

1. The waste evaporator for processing
2. The waste test tanks for direct discharge offsite, if the quality and radioactivity levels are within design limits
3. The boron recovery system evaporator (Subsection 9.3.5.2)
4. The waste feed tanks of the Waste Solidification System (Section 11.4)
5. The skid-mounted Waste Liquid Processing System

Overflow and recirculation lines are provided on the floor drain tanks. The operation is manual, except for the automatic shutdown of the floor drain tank pump on low liquid level in the tank.

b. <u>Evaporation</u>

The evaporation equipment of the Liquid Waste System is identical to the Boron Recovery System (see Subsection 9.3.5.2) except that only one liquid waste evaporator is employed, versus two for the BRS, and the concentration in the liquid waste evaporator is generally up to 12 percent TDS, as mentioned in the design bases.

c. Testing and Demineralization

The liquid waste evaporator distillate and the skid-mounted Waste Liquid Processing System effluent are the normal sources of liquid to the testing and demineralization subsystem. Other sources of liquid which enter the waste test tanks are listed below.

1. Liquid directly from the floor drain tanks, when that liquid does not require processing in the evaporator

2. Distillate from the boron recovery evaporator, when that evaporator is substituting for the waste evaporator

3. Flashed steam or bottoms from the steam generator blowdown flash tank and flash steam condensers, when that system must discharge liquid off site (see Subsection 10.4.8)

4. Liquid from chemical drain liquid tanks, when that liquid does not require processing in the evaporator.

A radiation element monitors the liquid entering the waste test tank. Liquid collected in the waste test tanks is transferred by one of two waste test tank pumps to the Circulating Water System intake and discharge transition structures for final disposal. Radiation levels and flow rates are also monitored on this line. If purification is required prior to discharge, the liquid is circulated through the waste demineralizer and filter. If reprocessing is required, the waste test tank contents are pumped back to the floor drain tanks, the BRS recovery test tanks, or the boron waste storage tanks. Overflow and recirculation lines are provided on the waste test tanks.

Liquid from the BRS testing and demineralization subsystem (Subsection 9.3.5.2d) and blowdown from the steam generator blowdown system flash tank (Subsection 10.4.8) can be transferred directly to the Circulating Water System intake and discharge transition structures for final disposal when water quality permits. Wide-range flow control is achieved by two parallel control valves in 3" and ¾" lines. One valve always remains closed when the other is open. The flow of filtered demineralized water which is discharged to the Circulating Water System is recorded as well as totalized at the Waste Management System (WMS) control panel.

Operation is essentially manual, with only two protective functions being automatic, i.e., securing of the waste test tank pumps on low test tank level, and termination of offsite discharge on high radiation levels.

11.2.2.2 Component Description

The detailed data is given in Table 11.2-1.

11.2.3 Radioactive Liquid Release

The amount of radioactivity projected to be released in liquid effluents from normal operation, including anticipated operational occurrences, is described below. The radiological impact from such releases is evaluated based on the dose models in Regulatory Guide 1.109 (Revision 1).

The analysis and parameters described in this section are historical and are the basis of the facility's 10 CFR 50 Appendix I analysis, utilizing the assumptions and methodology of NUREG-0017 and a core power level of 3654 MWt. For the licensed core power level, the core power level for analysis purposes is 3659 MWt. Since 3659 MWt represents only an approximate 0.1% increase above 3654 MWt, use of the core power level of 3659 MWt in the Appendix I analysis discussed herein will have an insignificant impact on radiological releases.

Continued compliance with the annual dose limits to an individual in an unrestricted area set by 10 CFR 50 Appendix I and 40 CFR 190, resulting from gaseous and liquid effluents released to the environment following operation at the licensed core power level was also demonstrated by utilizing five (5) years (1998-2002) of effluent/dose impact data and using scaling factors to estimate the impact of operation at the licensed core power level.

It is noted that for an operating plant, the actual performance and operation of installed equipment, the reporting of actual offsite releases and doses, and compliance with the regulatory limits of 10 CFR 50 Appendix I, 10 CFR 20, and 40 CFR 190, are controlled by the Offsite Dose Calculation Manual.

11.2.3.1 Normal Operation Release

The sources of radioactive wastes that are to be released are as follows:

- Boron Recovery System discharges for tritium control
- Nonrecyclable liquid releases from the Liquid Waste System
- Secondary system condensate leakage
- Nonrecyclable liquid releases from the steam generator blowdown waste holdup sump

The list of potential sources of liquid to be discharged is reduced to the above because of the processing system design principle to segregate, process, and recycle as much of the liquid extracted from the Reactor Coolant System as possible. The systems provided to carry out this design principle are the Boron Recovery System (Subsection 9.3.5) and the Liquid Waste System, described above. These descriptions illustrate the manner in which the collecting and handling of liquid letdown and leakage from the Reactor Coolant System are segregated from nonrecyclable liquids and sent, after processing, to the reactor makeup water storage tanks for reuse within the Reactor Coolant System.

a. Release Assumptions

The main assumptions and parameters used in estimating the magnitude of radioactive liquid releases are as follows:

1. The radionuclides and their concentrations within the Reactor Coolant System are as listed in Table 11.1-1 under the heading of "0.12 percent cladding defects."

2. The radionuclides and their concentrations within the secondary side of the steam generators are as listed in Table 11.1-4. The feed and condensate system activities are equivalent to the steam activities, excluding noble gases.

3. The decontamination factors (DFs) within the Boron Recovery System and the Liquid Waste System are given in Appendix 11A.

4. The times of radioactive decay between collection, processing and discharge are listed in Appendix 11A for each stream of liquid waste.

5. The reactor is assumed to be operating at 3654 MWt, with an 80 percent capacity factor.

b. Releases

1. Releases from Boron Recovery System

As described in Subsection 11.1.1.3, tritium control considerations anticipate the need for discharging reactor coolant letdown after processing by the Boron Recovery System. The expected volume required for this measure is 200,000 gallons per year. With the input liquid containing radionuclides at Table 11.1-1 values (0.12 percent clad defect), the processing DFs and the radioactive decay times in Appendix 11A, the annual release from this source is 0.033 Ci/year, except tritium. This release is shown by radionuclide in Table 11.2-2.

2. Nonrecyclable Releases from Liquid Waste System

The estimated average volumetric generation rates of nonrecyclable primary system leakage are 40 gallons/day inside the Containment at primary coolant activity (PCA), and 200 gallons/day in the Auxiliary Building at 0.1 PCA. This liquid is collected in the floor drain tanks, which are at the head end of the floor drain portion of the Liquid Waste System. It is also estimated that there are 400 gpd of liquid waste from laboratory drains at 0.002 PCA, 15 gpd from sampling drains at 1.0 PCA, and 700 gpd from miscellaneous waste at 0.01 PCA released into the Liquid Waste System. (There will be no liquid waste generated from laundry operations.) This liquid is collected in the chemical drain treatment tanks (refer to Updated FSAR Subsection 9.3.3.2). The total input rate to the floor drain tank is less than 1355 gallons/day with the effective composite activity concentration of 0.061 PCA. With the processing DFs and radioactive decay times in Appendix 11A, the annual release from this source is less than 0.035 Ci/year, except tritium. This release is shown by radionuclide in Table 11.2-3.

3. Steam Generator Blowdown

With reactor coolant radionuclide concentrations as listed in Table 11.1-1 and an estimated average leakage of 100 lbs/day of reactor coolant through the steam generator tubes, equilibrium secondary side steam generator radionuclide concentrations are calculated as listed in Table 11.1-4. The steam generator secondary side blowdown rate associated with this leakage is 75 gpm, total from all four steam generators.

Blowdown from the four steam generators is processed by the blowdown flash tank, where approximately 30 percent of the blowdown volume is flashed to steam. This steam is normally routed to the No. 3 feedwater heater or to the main condenser. With no primary-to-secondary leakage, steam may be exhausted to the atmosphere if the heater and condenser are not available. The vent gases from the flash steam condenser are processed before discharge. Radioactive steam is not processed directly to the atmosphere. In the presence of a primary-to-secondary leak, flash tank steam may be sent to the No. 3 feedwater heaters or processed through the flash tank distillate coolers to the waste test tanks, prior to discharge.

The remaining volume of blowdown liquid (70 percent) can be released directly to the environment via the plant Circulating and Service Water System when no primary-to-secondary leakage exists or processed via the blowdown demineralizer subsystem. With significant primary-to-secondary contamination of the secondary side water, the primary method to process radioactive secondary liquid from the steam generators is to direct steam blowdown flash tank bottoms cooler discharge to the floor drain tanks. If no secondary pressure is available, the steam blowdown and wet lay-up pumps can be used. From the floor drain tanks, processing through the installed vendor system (WL-SKD-135) to the waste test tanks may be performed (reference Subsection 11.2.2.1). Additional methods that could be used, with station management approval and planning, include the liquid volume within the flash tank being processed by the Blowdown Evaporator System (Subsection 10.4.8). Two evaporators in parallel are available to process a maximum of 50 gpm of blowdown liquid. Distillate from the evaporators is condensed and directed to the waste test tanks of the Liquid Waste System. Further processing is available within the Liquid Waste System, if required, prior to discharge to the environment via the Plant Circulating

Water System. Bottoms from the blowdown evaporators is routinely released to the Solid Waste Processing System for processing and shipment offsite. No credit for collection and processing decay times has been assumed to calculate the liquid releases from this pathway. With the DFs presented in Appendix 11A for the installed vendor system, the annual release from this source is 0.02 Ci/year. This release by radionuclide is presented in Table 11.2-4.

4. Secondary System Condensate Leakage

The estimated average liquid leakage rate of the secondary system is 7200 gallons/day. The leakage is assumed from liquid sources at main steam activity. The concentrations of the main steam activity are listed in Table 11.1-4. This liquid is collected in the Turbine Building floor drain and then is discharged from the plant unprocessed, which results in the annual release of 0.00658 Ci/year, except tritium. This release is shown by radionuclide in Table 11.2-5.

5. Summary of Radioactive Liquid Release From Normal Operation

The total estimated radioactivity to be released due to the generation and release of the liquid streams described above is shown in Table 11.2-6. The total annual release of 0.08 Ci except tritium and 730 Ci of tritium are the expected discharge levels due to normal operation. The tritium releases are discussed in Section 11.1.

11.2.3.2 Releases from Anticipated Operational Occurrences and Design Basis Fuel Leakage

The additional unplanned liquid release due to anticipated operational occurrences is estimated to be 0.15 Ci/year based on reactor operating data over a 2.5 year period, January 1973 through June 1975, representing 102 reactor-years of operation (NUREG-0017). These releases are assumed to have the same isotopic distributions for the calculated source term of the liquid wastes. The annual release from the anticipated operational occurrences is shown by radionuclide in Table 11.2-7.

Table 11.2-8 shows the total annual release by radionuclide from normal operation, including anticipated operational occurrences. The discharge concentrations are compared with (MPC)w, the concentration limits of 10 CFR Part 20, Appendix B, Table II, Column 2. Table 11.2-9 shows the total annual release by radionuclide from design basis fuel leakage (i.e., 1 percent failed fuel).

11.2.3.3 Release Points

The release routes of radioactive liquids generated in operation are shown in simplified flow sheet form in Figure 11.2-5. Liquid processed for discharge by the Boron Recovery System ultimately accumulates in the waste or recovery test tanks, located adjacent to the Waste Processing Building, as shown in Figure 1.2-22 and Figure 1.2-25. After sampling for radioactivity analysis, the liquid is discharged from the waste or recovery test tank through a process radiation monitor to the Circulating Water System. The tie-in point with the Circulating Water System is as shown in Figure 11.2-4.

Nonrecyclable reactor coolant leakage is collected by the drain system within the Containment and Primary Auxiliary Building and is directed to the floor drain tank of the Liquid Waste Processing System. This tank is located within the Waste Processing Building as shown in Figure 1.2-29. This liquid is processed and directed to either of two waste test tanks. The liquid is then sampled and released as described above for boron recovery system releases.

Miscellaneous liquid wastes (decontamination water, laboratory, drains, etc.) are collected in the chemical drain treatment tank where they can be processed and directed to the waste or recovery test tanks for final sampling prior to discharge to the Circulating Water System.

Steam generator secondary side blowdown is directed to the flash tank. If steam generator blowdown activity is low, liquid from the flash tank may be transferred directly to the waste test tank or to the Circulating Water System. Otherwise, liquid from the flash tank is directed to the blowdown demineralizers or the installed vendor system for treatment.

Secondary system condensate leakage is collected in the Turbine Building sumps. From here the liquid is directed to an oil separator and transferred to the circulating water system discharge. A radiation monitor is located on the sump effluent line which automatically isolates the sump at a predetermined radioactive concentration.

All these waste liquids, once released to the Circulating Water System, experience the same release path to the Atlantic Ocean. The radioactive liquid wastes then reach the environment via the circulating water discharge line.

11.2.3.4 Dilution Factors

The discharge route of radioactive liquid to the environment, as described in the above section, provides onsite dilution by the flow of the Circulating Water System, conservatively assumed to be 390,000 gpm (for a discussion of CWS operation, see Subsection 10.4.5). With the assumption of 80 percent operating capacity of the Circulating Water System, this flow dilutes the normal liquid radioactivity releases from 0.24 Ci/year except tritium and 730 Ci/year of tritium, to 3.9×10^{-10} µCi/ml except tritium and 1.2×10^{-6} µCi/ml of tritium.

Further dilution of the circulating water discharge plume will occur after leaving the discharge pipe. A near-field dilution factor of 8 was used for calculating the estimated doses reported in the next section. This dilution factor estimate is based on a multi-port, deep water discharge concept and is conservatively calculated for the tidal cycle surface area above the discharge pipe.

The information above provides assurance that the original plant design would meet the requirements of 10 CFR Part 20 and 10 CFR Part 50. The Offsite Dose Calculation Manual (ODCM) contains actual plant data that is used for plant releases to comply with 10 CFR Part 20.

11.2.3.5 Estimated Doses

Radionuclides in liquid effluents from the site pose a potential environmental radiation source to certain individuals or segments of the general public. In general, many possible exposure pathways exist for liquid effluents, however, detailed consideration can be focused on the pathways that pose the greatest risk to public exposure - the "Critical Pathways." The critical pathways are considered to be the internal exposure by the ingestion of fin fish or other seafood harvested for the area affected by released radionuclides and the external exposures from recreational activities along the shoreline.

The radiological assessment of the ingestion pathways and the external exposure pathway is calculated based on the models, parameters, assumptions, and dose conversion factors in Regulatory Guide 1.109 (Revision 1). Table 11.2-10 shows the parameters used in the dose calculation.

The maximum annual doses from all pathways received by an individual in a particular age group due to the estimated liquid discharge radioactivities from normal operation (including anticipated operational occurrences) listed in Table 11.2-8 are shown in Table 11.2-11. Among three age groups, the highest maximum annual doses, 2.5×10^{-3} mrem/yr to the total body and 2.4×10^{-2} mrem/yr to the thyroid (the most critical organ), are small fractions of the numerical design dose objectives of Appendix I to 10 CFR 50.

11.2.4 Design Evaluation

a. General

The Liquid Waste System performs no safety function and is not required for the safe shutdown of the reactor. Accordingly, the system is designated NNS class. Also, because of the noncritical nature of this system, emergency electrical power is not provided.

All waste liquid generated within the plant is processed through the floor drainage system (Subsection 9.3.3), Boron Recovery System (Subsection 9.3.5) or Liquid Waste System, when necessary. Process discharge to the environment is only after testing of the effluent quality.

The evaporators are designed throughout with low-flow velocity and liquid vapor separators to reduce any entrainment of vapor. The evaporators are designed to yield a minimum decontamination factor of 10^4 for nonvolatiles. This will reduce the amount of radioactivity in the distillate to acceptable levels for disposal. The concentrated liquid is processed by the Solid Waste System before disposal offsite. Backup evaporator capacity from the BRS is available. All essential portions of the system are located away from any high energy lines. A negative pressure is always maintained in the vent header so that gases go into the vent header and not into the tank overflow lines. Relief valves and tank overflow lines have been provided to protect against overpressures. Should the high level point be exceeded, the tank overflow is channeled to the floor sump. Each floor sump includes one or more sump pumps which transfer the excess liquid to the floor drain tank. Pumps in the system have low level shutoffs. Filters and demineralizers have pressure indication upstream and downstream to indicate fouling.

Two sets of level instruments, one for process control and indication and the other for indication and alarm, are provided on all the essential equipment of the process. Moreover, the instrumentation and controls are located outside the boron concentrating areas to minimize radiation exposure to operating personnel.

The skid-mounted Waste Liquid Processing System is connected to permanent plant connections in the Waste Liquid System (located (-)3' elevation WPB).

Radiological consequences from postulated failures of components in the Liquid Waste System are bounded by the results of the liquid waste system failure analyses presented in Subsections 15.7.2 and 15.7.3.

b. Equipment Redundancies

Only one of the two floor drain tank pumps, waste test tanks, waste test tank pumps, and floor drain filters is required at a time. If the liquid waste evaporator subsystem is not available at any time, the liquid can be concentrated in one of the two boron recovery system evaporators, or it can be discharged directly to the waste feed tanks.

c. Faults of Moderate Frequency

The system is designed to handle the occurrence of equipment faults of moderate frequency such as:

1. Malfunctions in the Liquid Waste Processing System

Malfunctions in this system could include pump or valve failures or evaporator failure. Failure of a pump is acceptable since a standby is provided. Sufficient surge capacity is provided to accommodate waste in the event of evaporator failure.

2. Excessive Leakage in Reactor Coolant System Equipment

Excessive hydrogenated reactor coolant leakage can be accepted by the equipment and floor drainage system (Subsection 9.3.3) and processed by the BRS (Subsection 9.3.5).

Similarly, excessive rates of either aerated, recyclable or nonrecyclable reactor coolant leakage would be handled by the WL system by collection in the floor drain tanks (10,000 gallons each) and processing by the 25 gpm evaporator.

A design feature that enhances the surge processing capability of the BRS and WL system is the fact that the three evaporators are cross-connected to allow flexibility in evaporation capability during abnormal periods.

3. Excessive Leakage in Auxiliary System Equipment

   Leakage of this type could include water from steam side leaks and fan cooler leaks inside the Containment which are collected in the containment sump and sent to the floor drain tank. Other sources could be component cooling water leaks and service water leaks. This water will enter the floor drain tank and be processed and discharged as during normal operation.

4. Overflow or Rupture of a Tank in the Liquid Waste System

   Floor drains and curbing are provided around all tanks in the WL system to minimize the flow of radioactive liquid to uncontrolled areas in the event of tank rupture.

11.2.5 Testing and Inspection

Prior to startup, the WL system was tested to verify proper operation of system equipment, establish setpoints and test flow rates, temperatures, and pressures. During plant operation the system is inspected frequently to ensure proper performance and operability.

11.2.6 Instrumentation Requirements

The instrumentation and control for this system is similar to that for the BRS (Subsection 9.3.5.5). Control and remote instrumentation is located on the Waste Management System (WMS) panel in the Waste Processing Building. Pump trips, high/low levels, and temperatures are alarmed as in the BRS. High pressure differentials on filters and demineralizers are also alarmed, as is the high radiation of the final effluent from the waste test tank.

Discussed below is the instrumentation and control which differ from that of the BRS:

Radioactivity of the fluid passing through the line discharging to the Circulating Water System is continuously monitored. Upon high radiation, the discharge valves are automatically closed and an alarm is simultaneously actuated locally in the control room and at the WMS panel. Details of the radiation monitoring instrumentation are provided in Section 11.5.

Radioactivity and flow rate of the effluent are continuously monitored to assure that the effluent release is in compliance with the Technical Specification requirements.

When a predetermined high liquid level in any tank is reached, an alarm is sounded at the WMS panel alerting the operator of the potential for tank overflow. An annunciator on the waste management system control panel indicates the source of the alarm.

11.3 RADIOACTIVE GASEOUS WASTE SYSTEM

11.3.1 Design Basis

Hydrogenated fission product gases from the reactor coolant letdown stream and from the liquids collected in the primary drain tank and the reactor coolant drain tank are processed in the Radioactive Gaseous Waste System (RGWS). An iodine guard bed and a molecular sieve dryer reduce the contamination level of the gases before further processing by the carbon delay beds. The carbon delay beds provide a minimum of 60 days xenon delay and 85 hours krypton delay. Low activity aerated gas streams from the reactor plant aerated vent header (Subsection 9.3.6), and condenser vacuum pump units (Subsection 10.4.2) are filtered, monitored, and discharged to the plant unit vent.

The RGWS is designed to provide sufficient processing so that gaseous effluents are discharged to the environment at concentrations below the regulatory limits of 10 CFR 20 and within the "as low as is reasonably achievable" guidelines set forth in 10 CFR 50, Appendix I, (see Subsection 11.3.3 and Section 11.5). The RGWS also provides sufficient holdup and control of gaseous releases, as specified in 10 CFR 50, Appendix A, General Design Criterion 60. The RGWS can process a maximum surge flow of 1.2 scfm from the degasifiers, which is based on the maximum letdown flow of 120 gpm from the Reactor Coolant System to the Chemical and Volume Control System. This represents the most limiting plant operating condition for the RGWS.

The portion of the Waste Processing Building which houses the RGWS is seismic Category I.

The RGWS is designated NNS. Table 11.3-1 lists RGWS components and their design parameters. Redundant components are provided to minimize operator exposure and enhance system reliability. The RGWS is designed in conformance with Regulatory Guide 1.143.

Hydrogen concentration is monitored in cubicles containing RGWS components to detect a leak in the system. Monitoring of hydrogen concentration is not required while the RGWS is inerted with nitrogen. Subsection 15.7.1 discusses RGWS leak or failure. Dual oxygen monitors are provided to sample the process stream to monitor formation of explosive mixtures. An alarm is initiated at a predetermined setpoint prior to reaching a potentially explosive mixture. The RGWS is designed to withstand a H_2 explosion. A radiation monitor is provided on the line to the plant unit vent to detect an excessive release of radiation to the environment. An automatic isolation valve terminates flow in this line immediately upon receiving a high radiation signal. The RGWS monitors are discussed in Section 11.5. Monitoring of radioactive releases is provided in accordance with 10 CFR 50, Appendix A, General Design Criterion 64.

The Waste Processing Building Ventilation System is designed to discharge radioactive waste gases to the plant unit vent, and is discussed in Subsection 9.4.4.

The Primary Auxiliary Building Ventilation System is designed to discharge the radioactive waste gas to the atmosphere via the plant unit vent and is discussed in Subsection 9.4.3.

Shielding in separate cubicles is provided for the gas chillers, iodine guard beds, carbon delay beds, molecular sieve dryer, regeneration compressor package, hydrogen compressors, particulate filters, and hydrogen surge tank.

11.3.2 System Description

The Radioactive Gaseous Waste System, Figure 11.3-1, Figure 11.3-2, Figure 11.3-3, and Figure 11.3-4, processes noncondensable gases from the letdown degasifiers (Subsection 9.3.4), the primary drain tank degasifier (Subsection 9.3.5), the hydrogenated vent headers (Subsection 9.3.6), and the sample vessel purges (Subsection 9.3.2). Effluent gases from the operating letdown degasifiers are the major input to the RGWS. The gases from the hydrogenated vent header are processed as RGWS capacity is available. A pressure-regulating valve in the hydrogenated vent header maintains a constant pressure of 2 psig in the influent line of the RGWS, and serves to isolate this line from hydrogenated vent header pressure surges. Valves with special trim, seat and stem materials are used throughout the WG system to ensure tight shutoff and minimize leakage to the atmosphere. During normal operation, the expected influent flow rate from the degasifiers is 0.8 scfm which allows approximately 0.4 scfm for processing hydrogenated vent header gas. These influent gases consist primarily of hydrogen and water vapor with trace amounts of xenon, krypton, and iodine. Table 12.2-26 from Subsection 12.2.1 provides radioactivity content in the process stream. These gases enter the RGWS at the gas chiller compressor unit. A second chiller unit is provided for redundancy.

The chilled gas then enters the iodine guard bed which removes iodine from the gas stream (prior to further processing) to reduce radiation levels on downstream components. Since the guard beds will eventually become saturated with moisture and fission product iodines, it is anticipated that periodic replacement of the carbon in the guard beds will be necessary. In order to minimize operator exposure, "cartridge-type" carbon elements are used and the cartridges are changed remotely. Two guard beds are provided for redundancy. Each bed is capable of operating at maximum anticipated system flow.

The gas flows from the guard bed to the drying train. A three-bed dehydration system is provided. Each bed is capable of operating at maximum anticipated system flow. Normally, one bed is in operation, the second bed is heated for regeneration, and the third bed is in standby. A thermally regenerable molecular sieve is used as the drying agent. Instrumentation on the effluent side of the drying train automatically diverts the gas stream to the standby bed if an abnormally high dew point is experienced. The drying train is designed to remove vapor to -40°F dewpoint. Normal outlet dewpoint will be -80°F to -60°F.

Regeneration of a drying bed is accomplished in a closed loop system, consisting of a single-stage diaphragm compressor, an electric heater, and a purge gas condenser. The RGWS is designed with three waste gas dryers. One of them is in operation, the second is in regeneration, and the third is on standby and ready for operation. In case the purge gas condenser is inoperable, the regeneration loop will be shut off; on-stream time of the standby dryer is 96 hours which will be enough time for any maintenance work on the condenser. The regeneration compressor is also used to handle excess flow of up to 11.3 scfm when the RGWS is purged with hydrogen or nitrogen. Hydrogen from the hydrogen gas service system (see Figure 10.2-1) or nitrogen from the nitrogen gas service system (see Figure 11.3-3 and Figure 11.3-4) is directed to the hydrogenated vent header and enters the RGWS at the gas chiller. The gas stream bypasses the dryers and carbon delay beds and discharges to atmosphere via the equipment vent system.

The dried gas passes into a carbon delay bed. A total of five carbon beds are used, each containing 42.4 ft^3 (1600 lbs.) of 6x8 mesh type MBQ (or equivalent) carbon. Each carbon delay bed is approximately 3 feet in diameter by 8.5 feet in height (with 1.5 feet freeboard). Each bed provides 12 days of xenon delay and 17 hours of krypton delay based on conservatively estimated dynamic adsorption coefficients of 772.5 cc/gm. atm. for xenon and 45.4 cc/gm. atm. for krypton at a design flow rate of 1.2 scfm. Normal expected flow through the adsorbers is 0.8 scfm. The beds operate in series to provide a total delay of 60 days for xenon and 85 hours for krypton at a design flow rate of 1.2 scfm. In the event of an upset condition in the dryer, the first and second carbon beds can be operated in parallel, or the first bed can be bypassed. Bypassing of more than one bed is not permitted and would require shutoff of all input streams.

The particulate filter downstream of the carbon delay beds is designed to remove 99.97 percent of all particles 0.3 micron and larger released from the carbon delay beds. Two HEPA filters are provided for redundancy.

The waste gas stream is then compressed by a single stage diaphragm compressor. The compressor is furnished with an after-cooler to cool the discharge gas. A second compressor and after-cooler are provided for redundancy.

The compressor waste gas stream is either:

a. Returned directly to the Reactor Coolant System via the volume control tank, or the hydrogen injector,

b. Stored in the hydrogen surge tank,

c. Released to the environment via the equipment vent system, or

d. Recycled to the hydrogenated vent header as makeup gas.

The hydrogen gas compressor discharge and the hydrogen surge tank are common to a 150 psig header. Gas flows from this header through a pressure-reducing valve into 100 psig header which supplies the hydrogen injector.

Gas flow pressure is further reduced to 25 psig to supply the chemical and volume control tank. Additional gas flow pressure reduction is provided to supply gas to the hydrogenated vent header at 3 psig. Hydrogen from the hydrogen gas service system supplies makeup hydrogen to the 100 psig header if the RGWS is unable to supply the header demand, and supplies hydrogen directly to the volume control tank and hydrogen injection when RGWS hydrogen purity is below acceptable limits. A monitored line from the 150 psig header directs gas through a pressure-reducing valve and into a 3 psig header. This header then directs excess gas to the hydrogenated vent header where it is released to the atmosphere via the Primary Auxiliary Building normal ventilation cleanup exhaust unit. All releases comply with the "as low as is reasonably achievable" requirements of 10 CFR 50, Appendix I. The PAB exhaust unit is discussed in detail in Subsection 9.4.3.

The annual gaseous release due to lifting of relief valves on the evaporators or degasifiers is insignificant, since it is expected to occur less than once per year and will result in a small release to the environment for each occurrence.

Liquid drainage from the gas chillers, iodine guard beds, molecular sieve dryers, or the regeneration package is collected by a small drain pot and then pumped to the primary drain tank by the drain transfer pump.

Prior to operation, the entire system, including drain lines and the drain pot, is thoroughly purged with nitrogen to eliminate the possibility of obtaining a flammable mixture of hydrogen and oxygen.

The RGWS is taken out of operation during a "T" signal because primary component cooling water is lost to the regeneration and hydrogen gas compressors and the purge gas condenser.

The main condenser evacuation system and the turbine gland sealing system are described in Subsections 10.4.2 and 10.4.3, respectively.

11.3.2.1 Tests and Inspections

Periodic testing of the RGWS is not required as the system is in continuous operation. Inspection is performed in accordance with normal maintenance procedures. Samples can be drawn from lines downstream of the waste gas chiller dryers and downstream of the first and second carbon delay beds to check for oxygen, nitrogen and hydrogen concentrations, as desired, and for radioactivity.

11.3.2.2 Instrumentation

The following RGWS functions are instrumented and/or controlled:

- Measure concentration of oxygen in process stream
- Measure concentrations of hydrogen in cubicles
- Record and total flow of gas in main process path
- Measure the dryness at the outlet of molecular sieve dryer and hydrogen gas compressors
- Isolate the system in case of release of radioactive gases to the environment.

Further discussion on the instrumentation and control of the major components is presented below:

a. Waste Gas O_2 Monitors

The waste gas stream at the carbon delay bed inlet is sampled for trace oxygen to monitor explosive buildup. Two sensors are mounted in parallel. Oxygen concentration is indicated locally and high oxygen concentration is alarmed at the WMS panel.

b. Molecular Sieve Dryer

The three dryers are controlled from the Waste Management System (WMS) control panel. Normally the WMS is operated in auto with two drying towers alternating drying/regeneration cycles, with the third drying tower in standby. If failure occurs, the standby tower is auto-selected and it enters the cycle in place of the failed tower. This allows for a minimum of 72 hours to correct the malfunction. High humidity automatically shuts off the dryer. Humidity and radioactivity are monitored continuously at the outlet of the dryer bank. Humidity is recorded and radioactivity is indicated at the WMS panel. High humidity and radioactivity are alarmed at the WMS panel.

c. Regenerative Compressor

This compressor is a part of the dryer subsystem and maintains the flow of gas through the dryer column during regeneration. Starting of the compressor is always manually initiated from the WMS level. Deviant conditions such as low suction, lube oil and oil-gas leakage pressure automatically trip the compressor. Regenerative compressor gas flow and temperature are indicated at the WMS panel. Additionally, a trip signal from the dryer control logic stops the compressor on dryer trip.

d. Carbon Delay Bed

Temperature in each bed is recorded on a multi-point recorder at the WMS control panel. A radiation monitor is located at the outlet of the carbon delay beds. Radioactivity is indicated and high radioactivity is alarmed at the WMS panel.

e. Hydrogen Compressors

Starting of these compressors is always manual. Only one compressor runs at a time. Deviant operating conditions such as low lube oil level, generator cooler high temperature, etc., trip the compressor. The compressor discharge flow is recorded, totaled, and humidity is indicated at the WMS panel. High humidity is alarmed at the WMS panel.

f. Vent Isolation Valve

The release of radioactive waste gases is controlled by the vent isolation valve. A radiation monitor in this stream and another at the plant unit vent continuously monitor the stream for radioactivity. High radiation at the waste gas (WG) monitor automatically closes the vent isolation valve. This monitor also indicates radioactivity and alarms high radiation at the WMS panel. Flow can be re-established by the operator from the main control board. Should the WG radiation monitor malfunction for any reason, the radiation monitor at the plant unit vent alarms at the control room. The vent isolation valve is designed to fail closed on loss of air or power.

g. Monitoring of Leakage in Cubicles

Confined cubicles housing the primary drain tanks, the iodine guard beds, dryers, carbon delay beds, regenerative and hydrogen gas compressors, particulate filters and hydrogen surge tank are continuously ventilated and maintained at a slight negative pressure by the WPB normal ventilation system. These cubicles are also continuously monitored for hydrogen concentration except while the RGWS is inerted with nitrogen. The normal continuous ventilation is sufficient to maintain hydrogen concentration from normal minor leakage to a level well below 4 v/o. Should leakage increase and approach the 4 v/o concentration limit, the following will occur:

1. High hydrogen concentration will be alarmed at the WMS panel and at the main control board

2. Operator will take action to isolate and purge with nitrogen the leaking component

3. Operator will evacuate unnecessary personnel from the area.

In addition to the above, a supplementary exhaust fan will evacuate the hydrogen surge tank cubicle atmosphere when the hydrogen concentration reaches 2 v/o.

These steps will insure that the hydrogen concentrations will never exceed 4 v/o in the RGWS cubicles.

h. Hydrogen Gas Service

The makeup hydrogen flow from the hydrogen gas service system is recorded and totaled at the WMS panel.

i. Drain Transfer Subsystem

The drain transfer pump is operable from the WMS panel. In normal operation the pump cycles automatically on level switch actuation on a locally mounted drain pot. High pressure and time-delayed high level alarms are available at the WMS panel. The latter alerts the operator of the failure of the automatic start logic, so that corrective measures and/or manual starting of the pump can be initiated.

11.3.3 Radioactive Gaseous Release

Subsections 11.3.1 and 11.3.2 describe the Radioactive Gaseous Waste System. Since the system reduces fission product gas concentration in the reactor coolant during normal operation, it significantly reduces the escape of radioactive gases arising from any possible reactor coolant leakage.

Design is based on continuous operation with reactor coolant radioactivities associated with 1 percent failed fuel at rated thermal power. The estimated releases for normal operating conditions, however, are based on continuous operation with reactor coolant activities associated with 0.12 percent failed fuel at rated core thermal power. Provisions are also made to process gases from the reactor plant hydrogenated vent header. The radiological impact from such release is estimated by using the dose model in NRC Regulatory Guide 1.109 (Revision 1).

The analysis and parameters described in this section are historical and are the basis of the facility's 10 CFR 50 Appendix I analysis, utilizing the assumptions and methodology of NUREG-0017 and a core power level of 3654 MWt. For the licensed core power level, the core power level for analysis purposes is 3659 MWt. Since 3659 MWt represents only an approximate 0.1% increase above 3654 MWt, use of the core power level of 3659 MWt in the Appendix I analysis discussed herein will have an insignificant impact on radiological releases.

Continued compliance with the annual dose limits to an individual in an unrestricted area set by 10 CFR 50 Appendix I and 40 CFR 190, resulting from gaseous and liquid effluents released to the environment following operation at the licensed core power level was also demonstrated by utilizing five (5) years (1998-2002) of effluent/dose impact data and using scaling factors to estimate the impact of operation at the licensed core power level.

It is noted that for an operating plant, the actual performance and operation of installed equipment, the reporting of actual offsite releases and doses, and compliance with the regulatory limits of 10 CFR 50 Appendix I, 10 CFR 20, and 40 CFR 190, are controlled by the Offsite Dose Calculation Manual.

11.3.3.1 Normal Operation

The sources of radioactive gaseous releases (see Figure 11.3-5) are as follows:

a. Containment venting

b. Primary Auxiliary Building vent

c. Main condenser air evacuation pumps

d. Turbine Building leakage

e. Waste gas system release.

The release assumptions and parameters for each source are shown in Table 11.3-5. The estimated releases, by isotope, from each source, are shown in Table 11.3-2 for normal operation. This table is based on the source term information presented in Section 11.1, assumptions and parameters in Table 11.3-5, and an overall operating capacity factor of 80 percent.

11.3.3.2 Releases from Anticipated Operational Occurrences

The anticipated operational occurrences include the following:

a. Operation at 0.5 percent failed fuel for one year

b. Operation with 500 gallons/day of steam generator tube leakage for 90 days

c. Operations with 1 gallon/minute of hot reactor coolant leakage to the Containment for 12 days, followed by a containment purge

d. Operation with 200 gallons/day of reactor coolant leakage to the Primary Auxiliary Building for 90 days.

Each of the above anticipated operational occurrences has been evaluated by assuming that all the other parameters remain the same. The release rates from the anticipated operational occurrences are shown in Table 11.3-6, Table 11.3-7, Table 11.3-8 and Table 11.3-9.

11.3.3.3 Release Points

a. Primary Radioactive Gaseous Release Pathways

Primary gaseous release pathways for Seabrook Station are identified as releases from the following sources:

1. Waste Gas Processing System

2. Steam Generator Blowdown Processing System (included as part of main condenser releases)

3. Main Condenser Air Evacuation System

4. Primary Containment Purge Exhaust

5. Normal ventilation exhaust air from the Auxiliary and Turbine Buildings.

These sources are consistent with those presented in Reference 1. These primary release paths of potentially radioactive gaseous wastes are shown in Figure 11.3-5 and are discussed in Appendix 11A. With the exceptions of main condenser air evacuation pumps (hogging mode) and Turbine Building leakage, all significant releases to the environment are made through the primary vent stack. Releases from the main condenser air evacuation pumps are directed either to the Turbine Building vent or through the primary vent stack after filtration. Releases from the Turbine Building will be made through building ventilators.

For each primary release point, the height of release, inside dimensions of release point exit, effluent temperature and effluent exit velocity are given in Appendix 11A.

b. <u>Secondary Potentially Radioactive Gaseous Release Pathways</u>

The releases of radioactive materials in gaseous effluents from the following sources are considered as secondary sources and have not been included in the evaluation of station releases.

1. The releases of radioactivity materials in gaseous effluents from the following sources are calculated to be less than 1 Ci/yr of noble gases and 10^{-4} Ci/yr of iodine-131. Therefore, the following sources are considered negligible:

   (a) Steam releases due to atmospheric steam dump operation during startup and low power physics testing and

   (b) Ventilation air from buildings not covered in 11.3.3.3a.5 above (Reference 1).

2. Potentially radioactive materials in gaseous effluents from secondary pathways which are not routed to the plant unit vent include the following:

   (a) Radiochem lab fume hoods exhaust

   (b) Administration Building RCA exhaust

   (c) Thermostatically controlled room-type ventilators for equipment safeguards in PAB

   (d) Gland seal steam condenser exhaust.

   Releases from these sources are anticipated to result in offsite doses which are less than 1 percent of the dose limits of 10 CFR 50, Appendix I. Sampling and/or monitoring of these potential release pathways is discussed in Updated FSAR Section 11.5.

11.3.3.4 Dilution Factors

After the radioactive gases discharge from the release points, they are diffused and further diluted by the atmosphere. The dilution factors are evaluated by using annual average meteorological data. These dilution estimates are contained in Subsection 2.3.5. The maximum annual average X/Q at the site boundary is 3.90E-07 sec/m^3 for elevated stack releases and 5.65E-06 sec/m^3 for Turbine Building releases (ground level release point).

The information above provides assurance that the original plant design would meet the requirements of 10 CFR Part 20 and 10 CFR Part 50. The Offsite Dose Calculation Manual (ODCM) contains actual plant data that is used for plant releases to comply with 10 CFR Part 20.

11.3.3.5 Estimated Doses

The estimated doses given in the following subsections are based on the gaseous releases listed in Table 11.3-2 due to normal operation. The additional releases from the anticipated operational occurrences discussed in Subsection 11.3.3.2 would increase the maximum annual doses by a factor of 5 or less.

a. Estimated External Exposure From Noble Gaseous Releases

Radioactive noble gases released to the atmosphere will result in external radiation doses to individuals in the population. From Table 11.3-2, all the noble gases will be released from the plant stacks. The radiation from these gases consists of gamma rays, which result in a dose to deep tissues or total body dose, and beta particles which result only in a surface or skin dose, and which is effectively reduced by normal clothing. Total body dose refers to the gamma ray component of the dose, and skin dose refers to the beta plus gamma doses at the outer surface of the skin and assumes no dose reduction as a result of clothing.

The gamma and beta air doses from the stack releases are evaluated at the highest dose point on the site boundary, and the total body dose and the skin dose are evaluated at the highest residential dose location based on the models in Regulatory Guide 1.109. The highest dose point on the site boundary is 2999 feet from the station, in the ESE sector, with an annual average $(X/Q)_{\text{undepleted}}$ = 3.90E-07 sec/m^3. The highest residential dose location is approximately 7867 feet from the station, in the ESE sector, with an annual average (X/Q $_{\text{undepeleted}}$ = 2.08E-07 sec/m^3.

Residential doses take credit for a 0.7 shielding and occupancy factor as suggested in Regulatory Guide 1.109. Table 11.3-3 shows the estimated maximum air doses, total body dose and skin dose based on the noble gaseous releases in Table 11.3-2.

b. Estimated Doses from Radioiodines and Particulate Radionuclides

The releases of radioiodines and particulate radionuclides (not including noble gases) to the atmosphere will result in internal exposure through inhalation, ingestion pathways and external exposure to contaminated ground to individuals in the population. The maximum individual doses at the highest residential dose point are calculated based on the models in Regulatory Guide 1.109 for each age group and the following pathways:

1. Inhalation of air at the highest residential dose point

2. Ingestion of leafy vegetables and other food products grown at the highest residential dose point

3. Ingestion of milk and meat produced by the animals at the highest dairy farm dose point

4. External exposure to the contaminated ground plane at the highest residential dose point.

The release points and dilution factors are discussed in Subsections 11.3.3.3 and 11.3.3.4 above. All the parameters, usage factors for maximum individuals and dose conversion factors are taken from the suggested values in Regulatory Guide 1.109, except (1) fraction of year animals being on pasture = 0.5, and (2) absolute humidity = 8.0 g/m^3.

The calculated highest residential dose point from all release points is approximately 7867 feet from the station, in the ESE sector, with annual average dilutions of the following:

1. Stack releases:

   $(X/Q)_{\text{depleted}}$ = 2.0E-07 sec/m^3; D/Q = 1.11E-09/m^2.

2. Turbine Building releases:

   $(X/Q)_{\text{depleted}}$ = 1.06E-06 sec/m^3; D/Q = 3.17E-09/m^2.

The highest dairy farm dose point is approximately 2.4 miles from the station, in the NW sector, with an annual average D/Q = 2.64E-10/m^2 from the stack release, and 3.31E-10/m^2 from the Turbine Building vent release.

Table 11.3-4 shows the estimated maximum organ doses for each age group from all pathways due to the annual gaseous releases in Table 11.3-2. Among four age groups, the estimated maximum doses, 0.08 mrem/yr to total body, 0.16 mrem/yr to thyroid and 0.23 mrem/yr for the child bone pathway (the most critical organ), are small fractions of the numerical design dose objectives in Appendix I to 10 CFR 50.

11.3.4 References

1. USNRC NUREG-0017, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors (PWRs)," (PWR Gale Code) April 1976.

11.4 SOLID WASTE MANAGEMENT SYSTEM

11.4.1 Design Basis

11.4.1.1 Design Objective and Criteria

A Solid Waste Management System is provided to serve at the nuclear generating unit.

The Solid Waste Management System processes waste liquids, spent resins and dry wastes for onsite storage or disposal off site. The system is designed and operated to meet the limits for controlled releases of radioactive liquids from the site set forth in the Offsite Dose Calculation Manual (ODCM).

The Solid Waste Management System is designed in accordance with 10 CFR 20; Regulatory Guides 1.143, 8.8, and 8.10; Standard Review Plan 11.4; Branch Technical Position 11-3; and ANSI/ANS 55.1. A generic topical report discussing the WasteChem volume reduction and solidification system was accepted by the NRC on April 12, 1978. Consideration of operation, maintenance, and accident conditions have been factored into the system design to maintain radiation levels as low as reasonable achievable. Equipment layout and shielding are designed to limit radiation levels in areas accessible by the operator during a solidification operation to less than 15 mrem/hr, and in the radwaste control room levels are less than 2.5 mrem/hr. Handling, storage, and shipping of radioactive waste will be performed in conformance with 10 CFR 50, 61, and 71.

The containers used for solid waste storage and offsite shipment will meet the appropriate requirements of 49 CFR 171-179 (Department of Transportation Radioactive Material Regulations), and 10 CFR 71 (Packaging of Radioactive Materials for Transport).

System design incorporates backup processing capability for the WL, BRS and SB evaporators via the liquid waste volume reduction subsystem.

A 10 CFR 61 wasteform qualification program has been conducted (Reference 6) to demonstrate asphalt wasteform compliance with the NRC branch technical position for Class B and C wastes prior to using the asphalt system for waste processing. A process control plan (PCP) will be implemented to ensure compliance with 10 CFR 61 wasteform and burial site requirements. The wasteform qualification program (Reference 6) was submitted to, and accepted by, the NRC.

Waste will be classified pursuant to 10 CFR 61 classification requirements by use of isotopic inferral/correlation techniques in conjunction with periodic calibration programs.

The Solid Waste Management System is comprised of several subsystems as follows:

- Spent Resin Sluicing
- DAW Volume Reduction
- Alternate Solidification
- Volume Reduction and Solidification in Asphalt

Volume reduction and solidification in asphalt consists of:

- Waste concentrates handling
- Spent resin handling
- Liquid waste volume reduction
- Material handling

The system also provides the necessary instrumentation and connections for adequate control and monitoring of wet radwaste for delivery to contracted mobile solidification services equipment or to the WPB truck bay and/or adjacent shielded storage area for processing.

These subsystems are all interrelated with the exception of the DAW Volume Reduction which operates independently of any other subsystem. The various wet and dry solid radioactive wastes may be processed by either a permanently installed or mobile Solid Waste Management System.

a. <u>Waste Processing Subsystems Using Asphalt</u>

The permanently installed Solid Waste Management System, located in the Waste Processing Building (WPB), provides the following functions:

1. Collection and storage of spent resins from plant demineralizers or vendor-supplied systems
2. Collection and storage of concentrated wastes containing up to 12 wt% boric acid
3. Volume reduction to the maximum extent practical of all liquid concentrates and spent resins produced

4. Solidification of volume reduced waste, using an asphalt binder, as necessary, for offsite disposal in accordance with the requirements of 10 CFR 61

5. Encapsulation/solidification, using an asphalt binder, of noncompactible waste such as spent filter cartridges and similar items generated during plant operation and maintenance

6. Limited temporary onsite storage for solidified in asphalt low level wet waste, with a capacity of 15 months based on design volume production rates. Expected volume production rates yield capacities of 3.6 years.

The Solid Waste Management System is nonnuclear safety and nonseismic, and is designed in accordance with the requirements as set forth in NRC Regulatory Guide 1.143. The operation of the system has no effect on the capability to bring the plant to a safe shutdown condition.

b. <u>Alternate Waste Processing Via Mobile Contractor</u>

The processing of wet wastes and encapsulatable dry waste, in final preparation for onsite storage and eventual offsite shipment for disposal, may be accomplished using the services of a contractor using mobile equipment. Available contractor mobile equipment will utilize an approved solidification agent. The mobile solidification vendor wasteform qualification program has been placed on file. Permanently installed solid waste management system equipment needed for proper connections to and monitoring of waste inputs made to the mobile solidification services contractor equipment is listed in Table 11.4-1.

The station services and equipment necessary for the operation of the mobile solidification services to process liquid concentrates, spent resin or spent filter cartridge type waste are designed to:

1. Deliver liquid radwaste to an alternate Mobile Solidification System in the truck bay or in the adjacent shielded storage area.

2. Solidify completely all radioactive waste concentrates and chemical wastes.

3. Solidify noncompressible contaminated items such as spent filter cartridges and other items generated during plant operation and maintenance.

4. Package the solidified radioactive waste in containers suitable for transportation to a licensed burial site.

Temporary onsite storage of waste is available in the Waste Processing Building. Additional onsite storage will be provided as required.

The Mobile Solidification System has the capability to solidify an input volume of at least 16,000 ft^3/yr of waste consisting primarily of spent resin and 12 wt% boric acid concentrates, and to encapsulate and solidify 600 ft^3/yr of spent filter cartridges, contaminated and/or activated tools and other equipment. The resin sluice tanks (2) can accommodate no more than one year's supply. Spent resin and process filters can be dewatered with a vendor-developed dewatering process.

A Process Control Program (PCP) is in place that identifies the programs and procedures necessary for the processing, disposal and burial of dewatered resins and process filters.

c. <u>Portable Solidification System Interface</u>

1. Wet waste storage tanks are limited to the inplant installation.

2. Piping used to interface the plant's waste lines to the portable equipment complies with the requirements of Regulatory Guide 1.143.

3. The waste processing components of a portable system is appropriately arranged in the area bounded by lines A and D and columns 5 and 6 at elevation 25 feet of the Waste Processing Building. The area is provided with floor drains and monitored building ventilation exhaust ducts. Spill control arrangements will be evaluated on a case basis, since they would be component and equipment arrangement specific.

11.4.1.2 System Inputs

The Solid Waste Management System has several input sources. The following subsections discuss these inputs on a yearly operational basis, including maintenance and anticipated transients. Table 11.4-2 and Table 11.4-3 list the maximum and expected volumes and expected activity, respectively, for each source. The bases and assumptions used in determining the solid waste activities for offsite shipment for each waste type are listed in Table 11.4-4. The bases and assumptions used for determining the waste activities inside of each major subsystem component are presented in Section 12.2. Radionuclide concentrations and volumes are consistent with reactor operating experience presented in NUREG/CR-0144, ORNL-4924, NUREG/CR-1759, and NUREG/CR-1992. Process flow diagrams for the Solid Waste Management System are shown in Figure 11.4-11, sh1, and Figure 11.4-11, sh.2. Note that process flow diagrams, Figure 11.4-11, sh1, and Figure 11.4-11, sh.2 are provided for historical information only. Flow parameters are original plant design. P&IDs for the system are given in Figure 11.4-1, Figure 11.4-2, Figure 11.4-3, Figure 11.4-4, Figure 11.4-5, Figure 11.4-6, Figure 11.4-7, Figure 11.4-8, Figure 11.4-9 and Figure 11.4-10. Layouts identifying packaging, shipping, and storage areas are given by Figure 1.2-22, Figure 1.2-23, Figure 1.2-24, Figure 1.2-25, Figure 1.2-26, Figure 1.2-27, Figure 1.2-28, Figure 1.2-29 and Figure 1.2-30.

The estimated solid radwaste activity/volume values presented in the following sections represent data that were part of the original license application and is retained here for historical purposes.

Operation at the licensed core power level is expected to have minimal impact on installed equipment performance, system operation, and maintenance. Consequently, only minor, if any, changes are expected in waste generation volume. The activity levels for most of the solid waste will, however, increase proportionately to the increase in long half-life coolant activity, bounded by the percentage increase in power level to the licensed core power level.

a. Dry Active Wastes

Dry active wastes are classified into two categories. The two categories are (1) noncompactible and (2) compactible. Examples of noncompactible wastes would include small items such as used hand tools which cannot be economically decontaminated, electrical connectors, wood, et al., from contaminated areas. Examples of compactible wastes would include paper, polyethylene, tape, anti-contamination clothing, gloves, and shoe covers that are contaminated and/or beyond repair. The activity concentrations for offsite shipment, after processing, are listed in Table 11.4-5.

b. <u>Spent Demineralizer Resins</u>

The resin sluice tanks, located in the WPB at elevation (-)31' provide the collection point for spent resins. Normally spent resins are transferred to the shielded storage area or the WPB truck bay for processing, dewatering, and/or storage, as an alternative, spent resins may be solidified in asphalt using a permanent or alternate solidification system. The spent resin transfer pump takes suction on either of these tanks and transfers resin to the resin hopper at elevation 53'. After the resin hopper is filled, resin is processed by dewatering via the resin dewatering pump. The hopper fill operation is repeated until the proper resin slurry density is obtained in the resin hopper.

The resin process path is then to the resin centrifuge. Transport water is driven from the resin in the centrifuge and the resin free falls into the evaporator/extruder where it is homogeneously mixed with asphalt binder in an approximate one-to-one ratio, by weight, and is then discharged into an appropriate shipping container.

The function of the centrifuge is to remove transport water from the resin slurry, and thereby increase the system throughput by reducing the evaporative demand on the evaporator/extruder. As an alternative, resin slurry can also be fed directly to the evaporator/extruder via the concentrates metering pump. Resin slurries can also be pumped to the alternate solidification station located in the truck bay, via the resin centrifuge metering pump, for either solidification or dewatering in a disposable container.

At the alternate solidification station, a contractor with mobile waste processing equipment would then proceed to homogeneously mix the slurries with a solidification binder and then package them in an appropriate shipping container.

Spent resins from the following demineralizers are collected in the spent resin sluice tanks prior to processing:

1. Spent fuel pool demineralizer
2. Chemical volume control system demineralizers
3. Liquid waste system demineralizers
4. Boron recovery system demineralizers

The expected activity concentrations for offsite shipment, after processing, are listed in Table 11.4-6.

c. Evaporator Bottoms and Other

The waste concentrates tank at elevation (-)31' collects the evaporator bottoms and chemical drains from evaporators and drain tanks located throughout the plant. Concentrates are transferred to the waste feed tanks on elevation 53'. Once in these tanks, pH is adjusted and the concentrates can be fed to the crystallizer, or directly to the evaporator/extruder or to the alternate solidification station.

Chemically neutralized concentrates fed to the evaporator/extruder are homogeneously mixed with asphalt binder in a one-to-one ratio of solids to asphalt by weight, and discharged into appropriate approved containers. At the alternate solidification station, a contractor with mobile waste processing equipment would homogeneously mix the concentrates with an acceptable binder and package them into appropriate shipping containers.

The normal design operating mode is to process these concentrates through the crystallizer producing a concentrated bottoms of 35-50 percent total dissolved solids. These bottoms are then fed to the evaporator/extruder for further volume reduction by the removal of water, and solidification of the radsalts.

The concentrates collected for processing in the Solid Waste Management System on a batch basis, include evaporator bottoms and chemical drains from the following systems:

1. Liquid Waste
2. Boron Recovery
3. Steam Generator blowdown
4. Floor and Equipment Drains.

Other waste inputs handled by the system include the spent filters from various radioactive and/or potentially radioactive process systems from throughout the plant. Typically spent filter cartridges and other similar waste items are processed via a technique of encapsulation and solidification, using either cement or asphalt binder within a filter basket placed inside a 55-gallon drum or other container as needed. The expected activity concentrations of evaporator bottoms, chemical drains, and encapsulated and solidified waste for offsite shipment, after processing and binding with either asphalt or cement, are listed in Table 11.4-7.

11.4.2 System Description

The Solid Waste Management System is a plant system designed for the management of the final processing of wet or dry solid radwastes within the Waste Processing Building. Wet solid radwastes include spent demineralizer resins, concentrates from evaporator bottoms, chemical wastes, spent filter cartridges, and miscellaneous wastes from floor and equipment drains. Spent demineralizer resins are handled primarily by dewatering and storage prior to shipment offsite for burial. As an alternative, dewatered resins may be solidified using a permanent or alternate solidification system prior to disposal by one or more dewatering steps prior to being mixed with asphalt or other binder using a screw evaporator/extruder or fed directly to mobile equipment for dewatering and/or solidification with either cement or other approved binder. Concentrates at 12-50 wt% solids concentration are chemically adjusted, as necessary, then fed to the evaporator/extruder for the removal of water and mixing with the asphalt binder or fed directly to mobile equipment for mixing with an alternate binder. Dry solid waste includes both compactible and noncompactible trash. Transfer of wet wastes to contracted mobile services equipment is accomplished via use of the alternate solidification concentrates feed and the centrifugal metering pump and is controlled from a dedicated alternate solidification control panel located in the loading dock at elevation 25' of the Waste Processing Building.

The Solid Waste Management System is located in a separately shielded areas of the Waste Processing Building at the (-)31', 25', 42'-5", 53', and 55' elevations in the southwest corner of the building. Radwaste originating from sources from within the plant is controlled by this system for processing either by permanently installed equipment and or by contracted mobile solidification services. Wet radwastes for ultimate solidification and/or volume reduction into appropriate storage/shipping containers is collected into various tanks located at elevation (-)31' prior to transfer to the Solid Waste Management System for processing. Personnel exposure is kept as low as reasonably achievable by the use of shielding, by the use of closed circuit television cameras, by the provision of a separately shielded processed waste container storage area, and by the provision of a separately shielded loading dock area. Radioactive solid waste equipment parameters are summarized in Table 11.4-8.

11.4.2.1 Component Descriptions

The following are descriptions of major permanently installed mechanical system components.

a. Spent Resin Hopper (WS-TK-81)

The spent resin hopper is located at elevation 53' of the WPB. It has a working capacity of 934 gallons. The hopper accepts radwaste in the form of a spent resin slurry directly from the spent resin sluice tank. A volume reduction of this waste is accomplished in the hopper using an integral mixer, a dewatering screen, and decantation. A solids concentration of 15 wt% is expected. During recirculation pH is adjusted, if required. The normal disposition of the dewatered slurry produced by the hopper is feed to the centrifuge. Decant is returned to one of the resin sluice tanks. The hopper includes permanently installed internal sparging nozzles to permit remote decontamination after each use.

b. Resin Centrifuge (WS-MM-611)

The resin centrifuge is located above floor elevation 55' of the WPB. The unit has a maximum processing capacity of 4.7 gpm of resin slurry feed at a dry solids concentration of 10 wt%. It produces a processed effluent at a rate of 440 pounds per hour at a dry solids concentration of 50 wt% for feed to the evaporator/extruder and mixing with asphalt. This volume reduction is accomplished via the use of centrifugal forces to separate the liquid and solid phases of the resin slurry feed. Resin slurry feed is taken directly from the spent resin hopper recirculation for processing. Decant is returned to one of the resin sluice tanks.

c. Crystallizer (WS-EV-6)

The principal components of the crystallizer are located above floor elevation 55' of the WPB. The crystallizer is a low head, submerged tube, forced circulation package with a separate entrainment separator. The crystallizer has a working capacity of 1200 gallons; 600 gallons within the vapor body and another 600 gallons within its associated heater and recirculation piping. Its external two-pass horizontal heater utilizes the Plant Auxiliary Steam System. Steam is applied to the shell side of the heater where it condenses on the outside of the tubes and transfers heat to the liquor circulating inside the tubes. The steam condensate is then removed from the shell side of the heater and returned to the condensate storage tank. The liquor circulating through the tubes is not allowed to boil. After the liquor passes through the heater, it enters the vapor body where it releases water vapor to the entrainment separator, crystallizer condenser, and subcooler. The recirculation pump, vapor body, and heater are designed to process undissolved solids, with the capability to crystallize inorganic salts in typical PWR process waste concentrates.

d. Waste Feed Tanks (WS-TK-198A,B)

The two waste feed tanks are located at elevation 53' of the WPB. They each have a working capacity of 1,000 gallons for processing of radwaste originating from various plant evaporators and drain tanks. These tanks accept radwaste directly from the 6,000-gallon waste concentrates tank. When the waste feed tanks are filled to working level a sample is taken from their recirculation loop, their pH is adjusted, as necessary, and their contents are homogeneously mixed by a mixing eductor located inside the tank at the inlet nozzle. Their contents are then fed, normally on a batch basis, to the crystallizer system. Bottoms from the crystallizer are subsequently transferred to the concentrates bottoms tank prior to processing through the evaporator/extruder.

The contents of the waste feed tanks are kept heated in preparation for pre-concentration volume-reduction in the crystallizer vapor body, using electric strip heater elements wrapped around their girth. A dip pipe is provided in each tank for level indication. A permanently installed spray ball assembly is also included within the top of each tank to provide a remote decontamination capability after the processing of each batch of radwaste concentrates.

e. Concentrates Bottoms Tank (WS-TK-200)

The concentrates bottoms tank is located at elevation 53' of the WPB. It has a working capacity of 1,000 gallons. It receives bottoms directly from the crystallizer vapor body after the desired solids concentration has been attained. The tank is necessary for holdup prior to feed to the evaporator/extruder for final volume reduction and solidification via blending with asphalt. The holdup is required primarily because of the difference in processing rates of the evaporator/extruder and the crystallizer vapor body. It also allows final chemistry adjustments, as necessary, prior to processing through the evaporator/extruder.

The tank is kept electrically heated during processing. The tank is decontaminated remotely after each batch by using a built-in spray arrangement located in the top of the vessel. The contents of the tanks are kept homogeneously mixed by use of a mechanical agitator during feed to the evaporator/extruder.

f. Evaporator/Extruder (WS-EV-7)

The evaporator/extruder is located at elevation 42'-5" of the WPB. It combines volume reduction and solidification with asphalt (bitumen) of either spent resin slurries or waste concentrates. Both the radwastes and the asphalt (heated to 325°F) are fed simultaneously into the twin screw, steam heated, eight-section evaporator/extruder where the associated transport water and entrained moisture is evaporated and vented through the three steam domes during the blending process. Up to 99.5 percent of the associated free water is removed at the rate of 21 gallons per hour for a resin waste stream or at the rate of 32 gallons per hour from a concentrates waste stream.

Product discharge rate into shipping containers varies from about 9.25 to 27.5 gallons per hour, depending upon the speed of the screws and the feed stream solids concentration. Total residence time of material inside the evaporator/extruder is on the order of one minute and the total inventory of the unit when full of asphalt and waste is less than one gallon. The solids-to-asphalt weight ratio fed into the storage/shipping containers will vary up to one-to-one as determined by waste feed radioactivity concentration and weight percent solids loading. The evaporator/extruder is remotely controlled and has a variable speed motor drive. The unit can operate either continuously or in an on-off mode. If feed is stopped, it will continue to operate and self-clean within approximately one minute. Should power or steam failure occur allowing an asphalt mixture to harden inside of the evaporator/extruder, simple heating to the process temperature after repairs restores it to normal operation.

g. Asphalt Storage Tank (WS-TK-201)

The asphalt storage tank is located at elevation 20' of a separate asphalt storage building adjacent to the southwest corner of the WPB. It has a working capacity of 7000 gallons. This capacity is adequate to meet normal plant processing requirements for more than four months. The tank utilizes steam panels to permit heating using steam provided by the dedicated auxiliary boiler. The tank is designed to receive asphalt from a vendor's tank truck at a fill rate of 100 gpm, while simultaneously recirculating and straining the influx. The recirculation permits the rapid achievement of a homogeneous temperature distribution within the tank. Recirculation will be at a rate of 20 to 25 gpm. Fill strainers are designed to remove any foreign particles which could be present in the influx. Recirculation strainers are designed to remove foreign matter missed by the fill strainers.

h. Auxiliary Boiler Skid (WS-SKD-112)

The auxiliary boiler skid is located at elevation 53' of the WPB. The skid includes the electrically heated auxiliary boiler, the condensate return tank, two boiler feed pumps, blowdown tank and sample cooler plus associated valves and motors. The auxiliary boiler is designed to deliver steam at a rate of 5,000 pounds per hour at an operating pressure of 275 psig and an operating temperature of 410°F as needed to maintain asphalt within system piping and equipment in a molten condition for extended periods.

i. Chemical Feed Skid (WS-SKD-115)

The chemical feed skid is located at elevation 53' of the WPB. The skid includes the chemical feed tank and the chemical feed pump, plus associated valves and motors. The chemical feed tank is designed with a working capacity of 50 gallons and provides chemical feed to the auxiliary boiler.

j. Caustic Skid (WS-SKD-111)

The caustic skid is located at elevation 53' of the WPB. The skid includes the caustic day tank and the caustic metering pump, plus associated valves and motor. The caustic day tank is designed with a working capacity of 200 gallons and can supply caustic to the waste feed tanks, spent resin hopper, and the concentrates bottom tank.

k. Vent Hood Assembly (WS-MM-615)

The vent hood assembly is located at elevation 42'-5" of the WPB. The vent hood assembly is designed to fit over the container(s) to be filled, and exhaust any off-gassing of volatile vapors or steam carry-over at the evaporator/extruder discharge port or from the waste container being filled, to the WPB ventilation exhaust filter system. The vent hood assembly is equipped with redundant level probes and a camera for visual level inspection.

l. Compactor & Ancillary Equipment (WS-MM-722)

The compactor and its ancillary equipment is located on elevation 25' of the WPB within the shielded loading dock area due east of the truck bay. The compactor is a standard electro-mechanically driven device for hydraulic compression of dry active wastes into 100 ft^3 LSA boxes. Depending upon the type of dry material to be processed, volume reduction ratios ranging from 4:1 to 10:1 may be achieved. An expected average weight per box of compacted waste of approximately 3600 pounds is achieved with the use of anti-spring back devices. Ancillary equipment includes a two-horse power air vacuum pump, a furnace filter and absolute filter for the containment of contaminants generated during use, and a hydraulic unit.

m. Material Handling

A 30-ton bridge crane, WS-CR-35, 55-gallon drum conveyors, a 7½-ton self-propelled transporter cart, a process aisle overhead monorail, and fork lifts are provided for in-plant movement of solid waste containers. These items, in conjunction with the use of closed circuit television monitors, shielding and remote operation capability provide as low as reasonably achievable radiation protection for plant or contractor personnel.

11.4.2.2 Operating Procedures

The solid wastes listed in Subsection 11.4.1.2 are handled according to their physical properties. The system is used intermittently and requires one full-time operator in attendance during steady-state system operation. Control of the system is remote-manual.

The entire volume reduction, solidification and drum handling systems are controlled remotely from two control panels in the shielded control room of the Waste Processing Building (WPB).

Waste solidification processing is controlled from a single panel which integrates all the process subsystems. A programmable controller provides for automatic operation after manual startup of the system. Interlocks and permissives prohibit inadvertent or improper operation of subsystems and also form the basis of the solidification process control program. Details of the process control program for the use of permanently installed equipment utilizing an asphalt binder are given in a separate process control document. Details of a process control program for the use of contracted mobile solidification or dewatering services have been placed on file for NRC review.

The overall Radwaste Management System is designed to provide maximum flexibility in the processing of the various waste streams. Spent resins are processed by dewatering, volume reduction and/or solidification. Evaporator bottoms and chemical wastes are processed by volume reduction and solidification. Both waste types are capable of being pumped to an alternate solidification station in the truck bay area for solidification, or in the case of resin dewatering, into a dry form.

a. <u>Waste Concentrates Handling Subsystem</u>

The waste concentrates tank receives concentrated nonrecyclable wastes from the liquid waste evaporator, the two boron recovery evaporators, the three steam generator blowdown evaporators, the chemical drain treatment tanks, the floor drain tanks and return from the waste feed tanks. Concentrates in the tank are heated by electric tracing to prevent solidification of concentrated material. A waste concentrates transfer pump is provided for recirculation of the waste concentrates tank and transfer of the waste concentrates to the waste feed tanks. The overflow from the waste feed tanks is directed back to the waste concentrates tank. The overflow from the waste concentrates tank is directed to a floor drain.

The pre-concentrated liquid waste solution is pumped from the waste feed tanks to the crystallizer to remove excess water more quickly with the final waste volume reduction and solidification achieved upon subsequent delivery to either the permanently installed evaporator/extruder using an asphalt binder or alternately upon delivery to contracted mobile equipment for binding with an approved binder. The radioactive waste in the form of a slurry, containing

35-50 percent total dissolved solids by weight, is pumped to either the permanently installed asphalt evaporator/extruder by the concentrates metering pump or to the alternate solidification station using alternate solidification concentrates feed pump.

b. Spent Resin Handling Subsystem

The purpose of the spent resin dewatering subsystem is to remove the spent resins from the various demineralizers in the Radioactive Liquid Cleanup Systems, to store the resin for a time sufficient to reduce the radiation levels to an acceptable level, and then to pump the spent resin to the spent resin hopper where it is prepared for solidification and offsite disposal or to the WPB truck bay or adjacent shielded storage area for processing, storage, and/or shipment offsite.

The four basic operations performed by the system are:

1. Removal of spent resin from the demineralizers in various Radioactive Liquid Waste Cleanup Systems in the Primary Auxiliary and Waste Processing Buildings by a sluicing operation, and storage of these resins in the spent resin sluice tanks.

2. Post-sluicing cleanup of the resin sluice piping to lower the radiation levels, by recirculating the sluice water through the pipes and then through the resin sluice filter.

3. Recirculating the liquid in the sluice tanks and transferring the contents from one tank to the other, to equalize bulk or dose.

4. Spent resins are normally transferred to the WPB truck bay or to the adjacent shielded storage area. Alternatively, the spent resins may be transferred from the sluice tanks by the spent resin transfer pump for dewatering via the spent resin hopper and/or the centrifuge or directly to the shielded storage area or truck bay. The final waste volume reduction will be accomplished upon delivery to either the permanently installed evaporator/extruder using an asphalt binder or alternately upon delivery to mobile equipment.

   The spent resin hopper receives waste from the spent resin sluice tanks. The agitator in the hopper keeps solids and spent resin in suspension, and the spent resin dewatering pump and pump suction screen allow for removal of water from the resin slurry. The spent resin hopper is vented to the plant vent via the aerated vent header. Samples can be drawn from the recirculation line from the hopper for direct analysis of radionuclides, waste, and spent resin concentrations, as well as total activity level. Direct sampling also determines the process control requirements and container shield requirements.

The spent resin is pumped from the spent resin hopper in the form of a bead resin slurry by the resin recirculation and the resin centrifuge metering pumps.

c. Liquid Waste Volume Reduction Subsystem

The liquid waste volume reduction subsystem provides for the concentration of various liquid wastes from a nominal 5-12 wt% total dissolved solids to 35-50 wt% total dissolved solids. This concentration is accomplished by a 5-gpm forced circulation evaporator/crystallizer operating on plant auxiliary steam.

The prime function of the crystallizer is to quickly remove excess water from the liquid waste stream prior to feeding the asphalt evaporator/extruder, which has a comparatively low evaporation rate. The crystallizer also provides for a means of volume reduction independent of the evaporator/extruder operation and, in addition, allows for a reduced demand on the station's existing upstream evaporators.

Feed from the waste feed tanks enters the recirculation loop of the crystallizer and mixes with the recirculation flow prior to entering the heater. This stream then enters the vapor body where part of the flow flashes to steam. The flashed steam then enters the entrainment separator where it passes through distillation trays and a demister pad and then enters the condenser skid and ultimately returns to the plant liquid waste system or discharges to the Circulating Water System.

Bottoms from the crystallizer are pumped from the main recirculation loop to the crystallizer bottoms tank by the crystallizer drain pump.

System operation is dependent upon the amount of waste available for volume reduction. The system is capable of continuous operation with periodic bottoms blowdown. However, the crystallizer will normally be run on a batch basis, concentrating to the maximum extent practical and then shutting down.

The entire contents of the 6000-gallon waste concentrates storage tank can be volume reduced in the crystallizer and stored in the crystallizer bottoms tank prior to processing through the extruder/ evaporator.

d. Asphalt Volume Reduction and Solidification Subsystem

The evaporator/extruder is the main component of this subsystem and the heart of the entire volume reduction and solidification process. It is located in the WPB at elevation 42'-5". In the evaporator/extruder the waste stream (concentrates or resin) mixes with asphalt which is then heated by steam to evaporate the remaining water in the mix. The resulting matrix contains approximately 50 percent dried residual waste and 50 percent asphalt by weight and is deposited into either 55 gallon drums or 85 ft^3 containers. As the matrix cools from the operating temperature, solidification takes place due to the thermoplastic properties of asphalt. Water evaporated from the waste during volume reduction is condensed in the three evaporator/extruder steam domes and is gravity drained to the floor drain tanks.

Steam for this process is supplied by an electric auxiliary boiler at 410°F and 275 psig. Asphalt is supplied from the 7,000-gallon asphalt storage tank located in a separate building at grade just south of the WPB. Asphalt lines are steam traced using the high temperature steam from the solid waste auxiliary boiler. Plant auxiliary steam is provided as a backup system in case the solid waste system auxiliary boiler needs to be shut down for maintenance.

Normally, 55-gallon drums are used for disposal of solidified waste. However, the ability to utilize 85 ft^3 containers is provided. The following steps are required in order to align the system for usage with 85 ft^3 containers: (1) The drum capper station is relocated to the position required for functioning with 85 ft^3 containers. The cap stack unit is removed. (2) The 55-gallon drum turntable station is removed by unbolting the four flanged end connections. (3) The drum grab and rotator unit is changed out via the 30-ton overhead crane to enable use of the large container lifting rig. (4) A vent hood extension is added to properly interface the fill port to the large container. (5) The 85 ft^3 container will be loaded onto and off of the transfer cart via the overhead crane. (6) The waste solidification control panel settings are adjusted for 85 ft^3 container operation.

e. Material Handling Subsystem

The components of this subsystem are all located within the WPB at floor elevation 25', except for the radwaste crane which sits at elevation 73'. The empty drum conveyor is loaded from the solid waste control room with empty drums (6 minimum, 10 maximum) prior to the start of solidification operations. The drum hoist and grab lifts empty drums from the empty drum conveyor and places them on the turntable prior to fill operation.

The evaporator/extruder utilizes a one-step volume reduction and solidification process that operates continuously. The drum handling subsystem is designed to allow continuous flow of drums through the process area to take full advantage of the benefits that continuous system operation provides.

Operation of the evaporator/extruder will cause a drum to fill with the asphalt/waste mixture. After a drum has been filled, the turntable rotates an empty drum into the fill position and the filled drum into the pickup position on the turntable. Multiple fill passes on the turntable ensure maximum fill efficiency in each drum. The drum hoist and grab lifts the full drum and places it on the full drum conveyor. At this time an empty drum may be placed on the turntable if continuous operation beyond the six drum capacity of the turntable is anticipated. The drip pan mechanism is provided for product collection during drum indexing. The pan with drippings is deposited into the next empty drum after indexing.

All drum movement operations on the drum conveyors and turntable are manual start with automatic stop. All drum lifting and capping operations are manually controlled. Crane operation is manually controlled with specific lockouts to prevent crane travel with load in down position and to prevent collisions.

On the full drum conveyor the drum moves to the capping station where the capper is activated to pick up a cap from the cap stack and place it on the drum and crimp it in place. After capping, the swipe station turntable allows the drum to spin. A swipe may be taken, at this time, by the swipe manipulator to check for external contamination. The drum is then transferred to the end of the full drum conveyor where the overhead radwaste crane can pick it up and place it in the storage area.

All drum handling and process operations can be visually monitored via closed circuit television. A lead glass window on the fill aisle wall also provides direct visual observation of the capping and swipe operations.

As an alternative to filling drums, 85 ft^3 liners may be used. Liners will be located under the evaporator/extruder discharge by the transfer cart. The cart also moves liners to the capping and crane pickup positions. The use of liners or drums will be dictated by disposal economics.

f. Dry Active Wastes (DAW) Volume Reduction Subsystem

A subsystem for dry active waste is provided. The subsystem utilizes a 100 ft^3 box compactor with anti-spring back devices.

The subsystem handles typical trash collected at this operating plant with an average collected density of 7 to 10 lbs/ft^3 with a final waste density of approximately 30 to 40 lbs/ft^3.

This system combines reliable forty-year service life with maximum operator safety, compacting efficiency and ALARA exposure time.

The compactor exhaust system operates to prevent dust from escaping during trash handling and is complete with fan, fan motor, prefilter, absolute filter and connects to the duct in the filled drum storage area exhaust system.

g. Alternate Solidification

Waste concentrates or resin slurries prepared as described above will be delivered to the alternate solidification station, located in the truck bay area for processing via a mobile solidification or resin dewatering services contractor. Also provided will be flush water, vent lines, resin water return lines, isolation valves, and a control panel for the pumps and valves.

Two small panels in the truck bay supplement the main control and drum handling panels by providing functional control over waste flow to the alternate solidification station and control of crane movements in the truck bay area only.

11.4.2.3 System Controls, Protective Devices, and Instrumentation

The Solid Waste Management System has design provisions incorporated to reduce leakage, control the unplanned release of radioactive materials, and facilitate operation and maintenance in accordance with the guidelines of Regulatory Guide 1.143. Mechanical protective devices and/or instrumentation incorporated for the reduction of leakage potential and for control of the potential for uncontrolled releases of radioactive materials include: (a) a 100-micron filter array attached to the suction side of each of the two resin sluice pumps preventing resin movement except by the system; (b) system piping normally containing resin or concentrates has butt welds and five diameter bends; (c) a double mechanical seal for system pumps to prevent any leakage at the shaft; and (d) a vent hood assembly which is designed to collect off-gases, (e) seal water to most pumps is a closed system with individual seal water tanks. The remainder have a dead end seal water design.

Mechanical protective devices and/or instrumentation incorporated for the facilitation of operation and maintenance include: (a) capability to supply low pressure nitrogen to each of the two resin sluice tanks to prevent the possibility of the formation of an explosive hydrogen/oxygen mixture; (b) equipment within the system has high point vents and low point drain valves; (c) equipment layout is arranged so that radiation exposure during handling, volume reduction, or solidification operations is limited to less than 15 mrem/hr in operator occupied areas; (d) redundant disposal container level probes and television cameras provided for monitoring to minimize the potential for spills due to overflows during filling; (e) a 30-ton indexed radwaste crane which is remotely controlled from a shielded control station viewed through closed circuit television cameras; and (f) tank cubicles incorporate concrete curbs to further aid in controlling radioactive releases in the event of an overflow.

Radiation levels are monitored near the spent resin hopper cubicle and in the filling area by detectors permanently located in each of these areas. Local indication alarms are provided.

Contents of the hopper, the bottom tank, and the waste feed tanks are monitored for pH, with remote indication at the shielded waste solidification control panel.

Instrumentation is provided to monitor the pressure, level and temperature in various parts of the system to aid safe operation of the system from the main control panel.

Upon loss of power or air, the system reverts to the safe position.

In addition, due to the complexity of the Resin Sluice System and number of systems with which it interfaces, administrative procedures require a valve line-up inspection prior to any sluicing operation to insure that:

1. Operating systems under pressure do not become lined up to the sluice system.
2. Sluice water is not pumped into an operational system which is shut down and depressurized.
3. The sluice pump suction is not lined up with one sluice tank and the return flow to the other tank.

11.4.2.4 Maximum and Expected Processed Waste Volumes

The maximum and expected volumes of radioactive solid wastes available for offsite shipment annually, for each source, are presented in Table 11.4-9 considering the use of either an asphalt or a cement binder. The projected annual volumes are based on uniform one-to-one mixing ratio for a commercial asphalt with the maximum practical volume reduction of all wet solid plant wastes or a one-and-one-half-to-one mixing ratio using a cement binder without volume reduction. For spent demineralizer resins, the minimum expected applied volume reduction will be a factor of 1.85 and for evaporator bottoms and other, the minimum volume reduction will be 6.3. For compressible dry wastes, the minimum expected volume reduction will be 4. It is expected that, using an asphalt binder less than 22 containers with a 85 ft^3 capacity or 249 drums with a 55-gallon (7.35 ft^3) capacity will be needed for wet radwastes such as spent resins and evaporator bottoms and for encapsulation of other wastes such as spent filter cartridges, contaminated tools, etc. Less than seventy boxes with a 100 ft^3 capacity will be needed for all dry wastes such as swipes, rags, anti-contamination clothing, etc., on an annual basis.

It is expected that, using a cement binder, less than 118 containers with an 85 ft^3 capacity or 1360 drums with a 55-gallon capacity would be needed for wet radwaste and noncompressible wastes on an annual basis.

11.4.2.5 Packing

Spent resins, evaporator bottoms and miscellaneous chemical wastes are solidified in various-sized containers. The containers may be filled remote-manually from behind a shield wall using closed circuit television to avoid unnecessary exposure. Spent cartridge filters may be encapsulated or dewatered in large containers; dry low-level wastes are compacted to the maximum extent practical and shipped or stored in nominal 100 ft^3 boxes. Filled containers are shipped as required in appropriate overpacks and shields. Solid waste containers, shipping casks, and methods of packaging will meet applicable state and federal regulations, including 10 CFR 71.

11.4.2.6 Storage Facilities

Radioactive waste is stored in various locations in the Waste Process Building. The shielded storage area adjacent to the truck bay at elevation 25' is normally used to process and store resins and other packaged low level wastes prior to shipment offsite. Other areas in the WPB may be used to store radioactive material and wastes, as necessary in accordance with station procedures. The Unit 2 Cooling Tower is used to store DAW and other non-liquid radioactive material prior to re-use or shipment offsite. The Asphalt Storage Building is used to store radioactive material including low-level contaminated liquids. Additional storage capacity will be provided, as necessary, on a timely basis.

The shipment of solid radwaste does not disturb normal plant operations. Means of in-plant transport includes fork lifts, monorails and a bridge crane. Loaded trucks may stay overnight inside the enclosed radioactive material loading dock area, which is a restricted area with controlled access.

11.4.2.7 Shipment

Radwaste from the radioactive material storage areas will be loaded on trucks in the loading dock area of the Waste Processing Building or other locations as appropriate.

The containers are monitored for surface contamination, decontaminated if necessary, and released for transport to a licensed radioactive waste disposal site. The expected curie contents of these shipments are presented in Table 11.4-10. Wastes will be shipped in accordance with applicable NRC, Department of Transportation, and state regulations.

11.4.3 References

1. Kibbey, A.H. and Godbee, H.W., "A Critical Review of Solid Radioactive Waste Practices at Nuclear Power Plants," ORNL-4924, March 1974.

2. Kibbey, A.H., et al., "A Review of Solid Radioactive Waste Practices," NUREG/CR-0144, October 1978.

3. Wild, R.E., et al., "Data Base for Radioactive Waste Management/ Waste Source Options Report," NUREG/CR-1759, November 1981.

4. NWT Corporation, "Solid Radwaste Radionuclide Measurements," EPRI NP-2734, Project 1568-1, November 1982.

5. WasteChem (WPC), "Radwaste Volume Reduction and Solidification System," WPC-VRS-001, Rev. 1, May 1978.

6. WasteChem Topical Report, "10 CFR 61 Waste Form Conformance Program for Solidified Process Waste Products Produced by a WasteChem Corp. Volume Reduction and Solidification System," VRS-002, August 1987.

11.5 PROCESS AND EFFLUENT RADIOLOGICAL MONITORING AND SAMPLING SYSTEMS

11.5.1 Design Bases

The process and effluent radiological monitoring and sampling systems are designed to provide radiation measurements, records, alarms, and/or automatic line isolation required to handle, process, and/or release station liquid and gaseous radioactive effluents, in compliance with the requirements of 10 CFR Parts 20 and 50, General Design Criteria 60, 63 and 64, and Regulatory Guide 1.21. The systems are designed to be in general compliance with the guidelines of Regulatory Guides 4.15 and 1.97.

The systems are designed to continuously monitor and/or sample process and effluent streams wherever a potential for a significant release of radio-activity exists during normal operations, including anticipated operational occurrences, and during postulated accidents. For certain effluent streams for which the potential release of radioactivity is determined to be insignificant relative to the design objectives of 10 CFR 50 Appendix I, process monitoring/sampling and airborne sampling are utilized to conservatively assess the releases.

11.5.1.1 Performance Requirements

The functional performance requirements for the Radiation Monitoring Systems are to:

a. Warn of leakage from process systems containing radioactivity

b. Monitor the amount of radioactivity released in effluents

c. Isolate lines containing liquid and gaseous activity when activity levels reach a preset limit

d. Record the radioactivity present in various station systems and effluent streams

e. Provide a means for leakage detection

f. Provide information on failed fuel.

11.5.1.2 Design Provisions

The components of the process and effluent radiation monitoring and sampling systems are designed for the following environmental conditions:

a. Temperature: An ambient temperature range of 40°F to 120°F

b. Humidity: 0 to 95 percent relative humidity

c. Pressure: Components designed for normal atmospheric pressure.

Radiation monitors, utilizing G.M. tubes are of a nonsaturating design so that they register full scale if exposed to radiation levels up to 100 times full-scale indication.

Radiation monitoring equipment is designed and located so that radiation damage to electrical insulation and other materials will not affect their usefulness over the life of the plant. Where possible, electronic components beyond the detector are mounted near the detector in a background radiation level of less than 2.5 mR/hr.

Each radiation monitoring channel is designed so that it can be checked on a daily basis, tested monthly, and recalibrated at refueling intervals.

Access to each of the radiation monitoring channel alarm setpoints is under administrative control.

Process and effluent radiation monitors provide annunciation and indication in the main control room.

Process and effluent monitors continuously monitor radiation levels in the various process streams and effluent release points.

Process and effluent monitors provide instrument failure annunciation in the main control room.

All online process and effluent radiation monitors are implemented with the capability to replace or decontaminate these monitors without opening the process stream or losing the capability to isolate the effluent stream.

This system is non-Class 1E and nonsafety-related, with the exception of the monitors identified in Subsection 11.5.2.1n (Containment Online Purge). The containment online purge monitors are Class 1E, safety-related and supplied from Class 1E uninterruptible power supplies.

The resin sluicing monitors (RM-6560, -6561 and -6564) and the Storm Drainage System monitor (RM-6454) are the only monitors that do not interface with the RDMS host computer system (Subsection 11.5.2.1i)

11.5.2 System Description

Process and effluent radiological monitoring systems consist of multiple channels which monitor radiation levels in various plant operating systems. The digital computer-based Radiation Data Management System (RDMS) consists of local microprocessors for each channel interconnected by redundant communication loops to a redundant (two computers) host computer system. Either of the two computers can provide, by itself, the total computing capacity required for satisfactory operation of the RDMS. The host computer system, in turn, is connected to an operator display/control console in the control room, the health physics checkpoint, the RDMS computer room, the Main Plant Computer System (MPCS) computer room and the hot chemistry lab. The process and effluent radiation monitoring system instrument engineering diagram (Figure 11.5-1) shows an overview of the system, its components and location.

Table 11.5-1 lists the various processes and effluent radiation monitoring channels provided and their pertinent design information, such as detector type, ranges, reference isotopes which the detectors are keyed to, and sensitivities.

Ranges and sensitivities have been selected using the following bases:

a. Maximum calculated concentrations during normal operations, anticipated operational occurrences and postulated accidents

b. State-of-the-art limitations of the commercially available detectors

c. Minimum concentrations that must be detected to permit timely automatic or operator manual responses tabulated in Table 11.5-2 and to avoid exceeding Technical Specification limits.

Shielding is provided on all the monitors to reduce the effect of background radiation, so that the minimum sensitivities specified are met. Table 11.5-2 shows the automatic system response and the operator response to annunciated radioactivity level limits.

A modular assembly with a microprocessor is provided in a locally mounted cabinet for each channel. The assembly converts pulse rate from the detectors to engineering units suitable for indication and recording. The following functions and components are included:

a. Indication

Radiation level is indicated by digital readout. Units are microcuries per cubic centimeter, milliroentgens per hour, counts per minute or roentgens per hour.

b. Alarms

Alarms on (1) rising signal (the setpoint is adjustable to any point on the scale), and (2) loss of signal implying circuit failure are provided.

c. Functional Test and Calibration Requirements

Each radiation monitoring channel has the capability to expose the detector to a radiation check source by energizing a solenoid actuated device. This will cause an up-scale indication verifying the operability of the channel. The check source has a long half-life and an energy emission with the spectra of the radiation being monitored.

For both safety and nonsafety-related monitors, check sources can be activated locally from the RM-23s or the RM-80s. The RDMS Host Computer can initiate check source actuation for nonsafety-related monitors and is prevented from initiating check source actuation of safety-related monitors by a disabling switch at the RM-80s. On high background radiation the RDMS Host Computer will disable check source actuation from the workstations for non-safety related monitors.

Calibration test is accomplished by inputting a pre-calibrated pulse signal to the channel. Reading at the local meter will verify the calibration of the channel.

d. Indicating Lights

Indicating lights at the local radiation monitoring cabinets monitor individual channel high radiation alarms and circuit failures.

e. Power Supplies

Power supplies are mounted at the local radiation monitoring cabinets, and provide the voltages for the modular component circuitry, relays and alarm lights. The power supplies also supply high voltage for the detector. Internal battery backup is provided to prevent loss of stored information in the event of loss of AC power.

f. Fail-Safe

Fail-safe circuits in each monitoring channel indicate channel failure caused by signal or power failure.

11.5.2.1 Channel Descriptions

a. Waste Gas Processing Monitor - Channels 6502, 6503 and 6504

Radiogas monitors are located online at three points within the Radioactive Gaseous Waste System. Monitor 6502 is located up-stream of the carbon delay beds, 6503 is located downstream of the carbon delay beds, and 6504 is located downstream of the waste gas compressors. These monitors serve as indicators of carbon bed performance, with control room annunciation to alert station operators of abnormal operation or conditions. Remote indication and annunciation are provided on the control panel for the Radioactive Gaseous Waste System and in the control room.

A high radiation signal on 6504 terminates waste gas system discharges to the ventilation stack by automatic closure of the waste gas discharge valve.

b. Condenser Air Evacuator System Gas Monitor – Channel 6505

This channel monitors the discharge from the shell-side vacuum pump exhaust header of the condenser for gaseous radioactivity, which is indicative of a primary-to-secondary system leak. During normal plant operation, the gas discharge is routed to the Primary Auxiliary Building exhaust filter system. During startup operation (hogging), the gases removed by the evacuation system are discharged to the atmosphere via the Turbine Building vent. A beta scintillator is used to monitor the gaseous radioactivity level. Remote indication and annunciation are provided locally and in the control room.

c. Boron Recovery System Monitors – Channels 6500 and 6501

Radiation monitors are located at two points within the Boron Recovery System (BRS). Monitor 6500 is located downstream of the boron recovery filters and upstream of the boron waste storage tanks. Monitor 6501 is located between the distillate cooler and the recovery test tanks. These monitors serve as indicators of BRS processing performance, with control room annunciation and indication to alert station operators of abnormal operation or conditions. Remote indication and annunciation are provided on the control board for the BRS.

d. Primary Component Cooling Liquid Monitors – Channels 6515 and 6516

These two channels continuously monitor trains A and B of the Primary Component Cooling System for radioactivity indicative of a leak from the Reactor Coolant System or one of the other radioactive systems which exchange with the Primary Component Cooling System. Indication and annunciation are provided locally and in the control room.

The gamma scintillation detectors are located in offline liquid samplers.

e. Waste Processing System Liquid Effluent Monitor-Channel 6509

All discharges from the station via the WL Test Tank discharge header are monitored by an online gamma scintillation detector. This includes all discharges from the Test Tank itself, as well as steam generator blowdown demineralizer regenerant solution from the waste holdup sump or the bottom of the demineralizer beds. (See Subsection 10.4.8.2.)

Automatic valve closure action is initiated by this monitor to prevent further release after a high-radiation level is indicated and alarmed. Control room and remote indication and annunciation are provided.

f. Steam Generator Blowdown Liquid Sample Monitor – Channels 6510, 6511, 6512, 6513 and 6519

These channels monitor the liquid phase of the secondary side of the steam generator for radioactivity concentrations, which would indicate a primary-to-secondary system leak, providing backup information to that of the condenser air evacuation system gas monitor. A sample from the bottom of each steam generator is continuously monitored by a scintillation counter mounted in line with an offline type sample assembly.

Monitor 6519 is an offline detector with pumping system which monitors the flash tank discharge.

High activity alarm indications are displayed at the detector location and in the control room.

In the event of a high activity alarm from any monitor, the isolation valve in the blowdown flash tank discharge closes.

g. Reactor Coolant Letdown Gross Activity Monitor – Channels 6520-1, 6520-2

The reactor coolant letdown monitoring system is in service whenever normal letdown is in service and has no automatic functions. This monitor provides indication of primary coolant radioactivity concentration over a wide range of operating conditions and assists in the detection of failed fuel.

The design utilizes an adjacent-to-line detector (Geiger Mueller type) positioned in a shield that provides a collimated view of the letdown line. The detector is placed far enough from the RCS to allow for sufficient N-16 decay. The monitor readout indication is in mR/hr and has a range of 10^{-1} to 10^{4} mR/hr. This detector has a minimum sensitivity of about $1x10^{0}$ µCi/cc in a 15 mR/hr background depending upon the isotope of interest. The detector exhibits good energy linearity for gamma rays between 210 keV and 1333 keV and provides acceptable response to the isotopes listed in Table 11.5-1. The upper detection limit of the monitor is about $1x10^{3}$ µCi/cc depending upon the isotope of interest thereby providing a range of 1 to 1000 µCi/cc.

h. Liquid Waste From Evaporators to Waste Test Tanks – Channel 6514

A scintillation detector in an online sampler continuously monitors the waste liquid transferred to the waste test tanks. Increasing radioactivity concentrations indicate a potential problem upstream or a need to filter the tank contents or add water to the tanks to reduce the radioactivity concentration. Control room and remote indication and alarms are provided. A high alarm level closes the waste test tank inlet valves.

i. Resin Sluicing Operation Monitors - Channels 6560, 6561 and 6564

Geiger-Mueller tubes clamped to the process pipe monitor the resin sluicing operation. They provide indication and alarm locally and at the waste management panel. They do not interface with the RDMS host computer system. The function of these detectors is to monitor filter failure and to indicate completion of sluicing operation.

j. Main Steam Line Radiation Monitors - Channels 6481-1, 6481-2, 6482-1, and 6482-2

Online gamma-sensitive detectors are located on each main steam line upstream of the safety relief valves. As required by NUREG-0737, these monitors provide a method of quantifying high-level releases of radioactive noble gasses after an accident. Control room indications and alarms are provided.

The monitors display steam line dose rates in "mr/hr." The noble gas release rate is calculated from the dose rate by using a procedure.

k. Plant Vent Monitor - Channels 6528-1, 6528-2, 6528-3, and 6495

The monitoring capability associated with the main plant vent is described in Subsection 12.3.4.

l. Fuel Storage Building Exhaust Monitor – Channel 6562

The monitoring capability associated with air exhaust is described in Subsection 12.3.4.

m. Turbine Building Sump Liquid Radiation Monitor - Channel 6521

This online gamma scintillation detector is located on the Turbine Building sump effluent line. At a pre-determined radioactive concentration, this monitor will alarm and automatically terminate the discharge and isolate the sump.

n. Containment Online Purge Monitor - Channels 6527A, 6527B

These detectors monitor the air exhausted via the containment purge. They utilize GM tubes sensitive to Xe-133. These detectors provide measurement of the activity of the containment purge and provide isolation on a high signal. The detectors and their associated microprocessors are Class 1E. Each monitor utilizes a two-out-of-two detector logic such that two detectors must be in alarm before the monitor initiates an isolation signal.

o. Auxiliary Condensate Monitor - Channel 6490

An offline, skid-mounted monitor draws a sample from the auxiliary condensate return line. In the event that the auxiliary steam should become contaminated, this monitor will automatically terminate the condensate return to the auxiliary steam boiler and isolate the return piping.

p. Storm Drains Monitor - Channel 6454

This is an offline, skid-mounted monitor which continuously samples the storm drainage. The monitor design is identical to the steam generator blowdown liquid sample monitor. An auxiliary pump provides the necessary sample flow. Indication and alarm are provided only locally. This monitor does not interface with the RDMS host computer system. On a high alarm a composite grab sample is automatically obtained. An automatic continuous operating composite sampler is also provided.

q. Water Treatment Liquid Effluent Radiation Monitor – Channel 6473

This online gamma scintillation detector is located on the Water Treatment effluent line that receives water from the Water Treatment Neutralization Tank, Condensate Polishing Low Conductivity Tank or Condensate Polishing Resin Regeneration Megarinse Waste. High activity alarm indications are displayed at the detector location and in the Main Control Room. At a pre-determined radioactive concentration, this monitor will alarm and provide a signal to CPS PLC (1-CPS-CP-563) to automatically terminate the discharge.

11.5.2.2 Alarm Setpoints

The alarm setpoints for the process and effluent radiation monitoring system are provided in Table 11.5-1.

In establishing the site boundary concentration, it is assumed for continuous releases that average annual meteorology exists for gaseous discharges and average annual circulating water flow exists for diluting liquid discharges. For intermittent or off-normal releases, short-term meteorology and actual dilution water flow is assumed for gaseous and liquid discharges, respectively.

11.5.2.3 Design Evaluation

a. The reactor coolant letdown gross activity monitor (RM6520) serves as a failed fuel advisory and provides a function independent of the discharge monitoring system. This monitor is not required for detection of fuel cladding breach (see Appendix 7A, Deviation No. 8). The liquid and gaseous waste discharge monitoring system is employed to maintain surveillance over the release of radioactivity, and is provided with the following features:

1. The check source is operated by command from the display/control console or by command at the remote cabinet.

2. If the reading falls off scale at any time, an indicator visible to the operator in the control room will alarm.

3. Power failure is indicated by its own indicator, and does not alarm as high radiation failure.

b. An evaluation of instrumentation function, relative to monitoring and for controlling release of radioactivity from various plant systems, is discussed below.

1. Liquid and Gas Wastes

For ruptures or leaks in the Waste Processing System, station area monitors and the vent stack monitor (see Subsection 12.3.4) will alarm on an increase in radiation level over a preset level. For cases where leaks are involved, the operator may control activity release by system isolation. For more severe postulated accident cases, such as rupture of a carbon delay bed, activity release is not controlled. The environmental consequences of the postulated accidents are based on no instrument action. For inadvertent releases relative to violation of administrative procedures, monitors provide means for limiting radioactivity release as well as alarming functions. The waste gas vent monitor will trip the flow control valve in the discharge line when the radiation level exceeds a preset level. Where liquid waste releases are involved, the waste processing system liquid discharge monitor trips shut a valve in the liquid waste discharge line when the radiation level in the discharge line exceeds a preset level.

2. Liquid Waste Release Procedure

The release of liquid waste is under administrative control. The normal procedure for discharging liquid waste is:

(a) A batch of waste is collected in one waste test tank.

(b) The tank is isolated.

(c) The tank contents are recirculated to mix the liquid.

(d) Samples are taken for analysis before release.

(e) If analysis indicates that release can be made within the terms of the operating license, the quantity of activity to be released is recorded on the basis of the liquid volume in the tank and its activity concentration. Release is made when it is determined that the release will be within the operating license.

(f) To release the liquid, an operator must unlock and open the last stop valve in the discharge line (which is normally locked shut); open a second valve, which trips shut automatically on high radiation signal from the monitor (6509); start a test tank pump and establish the normal flow rate using the flow indicator provided; and finally, close the recirculation valve. Liquid is now being discharged.

11.5.2.4 Sampling

Sampling provisions are installed at the locations shown in Table 11.5-3. The samples are drawn from lines or tanks, and transported to the radiochemistry laboratory where the samples are analyzed for radioactivity content.

The sampling program is defined by a series of procedures for obtaining and analyzing representative samples. The administrative and procedural controls are in accordance with Regulatory Guide 4.15 (Position C) and Regulatory Guide 1.21 (Position C). Prior to sampling, large tanks of liquid waste are well mixed to assure uniform distribution of particulate solids. Sample lines are flushed for a sufficient period of time prior to sample extraction in order to remove sediment deposits and air and gas pockets. Sample collection techniques which preclude losses of radionuclides are employed.

Effluent ventilation release points are monitored continuously. Particulate sampling is done isokinetically for major release points.

11.5.2.5 Laboratory Analytical Instrumentation and Capabilities

Samples of process and effluent gases and liquids are analyzed in the laboratory. Laboratory instrumentation includes the appropriate means of detection for alpha and beta analysis, and gamma isotopic analysis.

Sample volume and counting time are chosen to yield the required sensitivities. Corrections are made for sample-detector geometry, sample self-absorption, and other parameters as necessary to assure accuracy. Gross alpha analysis of all liquid effluent samples is performed by liquid scintillation or by direct counting of evaporated deposits.

Gross alpha analysis of air particulate filters is performed by direct counting of the filters.

Alpha isotopic analysis is performed using the silicon surface barrier detector with the multichannel analyzer system, or by liquid scintillation.

Gamma spectrometry is used for isotopic analysis of liquid, gaseous and airborne particulate and iodine samples. A high efficiency, high-resolution HpGe detector is available, in conjunction with a multichannel analyzer, for resolving complex gamma spectra.

Effluent tritium samples are collected by various methods and analyzed by liquid scintillate.

11.5.2.6 Calibration and Maintenance of Effluent Radiation Monitors

A primary calibration is performed on a one-time basis, using typical isotopes of interest to determine proper detector response. Further primary calibrations are not required since the geometry cannot be significantly altered within the sampler. Calibration of samplers is then performed based on a known correlation between the detector responses and multiple secondary standards.

Secondary standard calibrations are performed with radiation sources of known activity. This calibration confirms the channel sensitivity. The secondary standard calibration is performed by placing the secondary standards on the sensitive area of the detector and comparing detector response to the detector response at the time of primary calibration.

The radiation monitoring system channels will be status checked at least daily and calibrated periodically. If a monitor functionally tested quarterly provides a control function on release, it will be functionally tested prior to that release.

Calibration of the indicating channels is performed following any equipment maintenance which could result in reducing the accuracy of the instrument indication. It is also done any time use of the ion chamber precalibrated pulse test signal or the radioactive check source indicates instrument drift.

A burn-in test, operational test and isotopic calibration of the Complete Radiation Monitoring System are performed at the factory. Field calibration after system installation will be performed using calibration sources and their decay curves provided with the system. The sample chambers will be decontaminated in situ periodically and, if required, are easily replaceable.

11.5.3 Effluent Monitoring and Sampling

General Design Criterion 64 requires monitoring of effluent discharge paths. Compliance with requirements is discussed in Subsection 11.5.2.

Airborne radioactivity monitoring is discussed in Subsection 12.3.4.

11.5.4 Process Monitoring and Sampling

Means are provided to control and monitor the release of radioactivity to the environment in accordance with the requirements of General Design Criteria 60 and 63 as discussed in Subsection 11.5.2.

TABLE 11.1-1 REACTOR COOLANT RADIONUCLIDE CONCENTRATIONS

Radionuclide			Concentration (µCi/gm) 0.12% Clad Defects [Historical]	1% Clad Defects	0.25% Clad Defects [Historical]
H	-	3	1.00E+00*	-	5.0E+00
N	-	16	4.00E+01	-	-
I	-	130	2.10E-03	5.6E-02	-
I	-	131	2.70E-01	2.5E+00	6.3E-01
I	-	132	1.00E-01	1.0E+00	2.3E-01
I	-	133	3.99E-01	3.8E+00	1.0E+00
I	-	134	4.70E-02	5.7E-01	1.5E-01
I	-	135	1.90E-01	2.2E+00	5.5E-01
Kr	-	83m	2.03E-02	3.7E-01	1.1E-01
Kr	-	85m	8.61E-02	1.6E+00	4.3E-01
Kr	-	85	2.03E-03	5.1E-01	3.3E-02
Kr	-	87	6.14E-02	9.3E-01	3.3E-01
Kr	-	88	1.78E-01	2.8E+00	8.5E-01
Kr	-	89	5.82E-03	-	-
Xe	-	131m	5.29E-03	4.6E-01	1.7E-02
Xe	-	133m	3.83E-02	1.8E+00	1.4E-01
Xe	-	133	1.59E+00	3.8E+00	6.3E+00
Xe	-	135m	1.48E-02	4.7E-01	2.1E-01
Xe	-	135	2.02E-01	7.5E+00	7.8E-01
Xe	-	137	1.05E-02	-	4.3E-02
Xe	-	138	4.99E-02	5.9E-01	1.8E-01
Br	-	83	4.80E-03	7.3E-02	-
Rb	-	86	8.50E-05	5.7E-01	-
Sr	-	89	3.50E-04	2.5E-03	1.0E-03
Sr	-	90	1.00E-05	1.6E-04	4.5E-05
Sr	-	91	6.50E-04*	1.3E-03	7.8E-03

* $1.00E+00 = 1.00 \times 10^{0}$
* $6.50E-04 = 6.50 \times 10^{-4}$

	Concentration (μCi/gm)		
Radionuclide	0.12% Clad Defects [Historical]	1% Clad Defects	0.25% Clad Defects [Historical]
Y - 90	1.20E-06	2.4E-04	5.5E-05
Y - 91	6.40E-05	5.8E-03	1.5E-03
Y - 92	-	7.9E-04	2.5E-04
Zr - 95	6.00E-05	5.5E-04	1.7E-04
Nb - 95	5.00E-05	5.6E-04	1.7E-04
Mo - 99	8.4E-02	3.8E+00	8.3E-01
Tc - 99m	4.80E-02	2.3E+00	-
Te - 127m	2.80E-04	2.7E-03	-
Te - 127	8.50E-04	1.2E-02	-
Te - 129m	1.40E-03	9.3E-03	-
Te - 129	1.60E-03*	1.3E-02	-
Te - 131m	2.50E-03	2.2E-02	-
Te - 132	1.7E-02	2.5E-01	6.5E-02
Cs - 134	2.5E-02	6.9E+00	1.1E-01
Cs - 136	1.3E-02	1.6E+00	5.5E-02
Cs - 137	1.8E-02	4.1E+00	5.5E-01
Ba - 137m	1.6E-02	3.9E+00	-
Ba - 140	2.2E-04	3.4E-03	1.1E-03
La - 140	1.5E-04	1.1E-03	3.5E-04
Ce - 144	3.3E-05	4.0E-04	1.1E-04
Np - 239	1.2E-03	-	-
All Others	2.5E-01	-	-

Note: With the exception of the Primary Coolant System source term based on 1% failed fuel, the analysis and parameters described in this table are historical and are the basis of the 10CFR Part 50 Appendix I program. It is noted that for an operating plant, the actual performance and operation of installed equipment, the reporting of offsite releases and doses and compliance with the regulatory limits of 10CFR50 Appendix I, 10CFR20, and 40CFR190 are controlled by the Offsite Dose Calculation Manual.

	Concentration (μCi/gm)		
Radionuclide	0.12% Clad Defects [Historical]	1% Clad Defects	0.25% Clad Defects [Historical]
Sr-92	-	6.5E-04	-
Y-93	-	5.4E-04	-
Zr-97	-	3.6E-04	-
Ru-103	-	4.8E-04	-
Ru-105	-	1.1E-04	-
Ru-106	-	1.6E-04	-
Rh-105	-	2.6E-04	-
Ba-139	-	5.1E-04	-
La-141	-	2.5E-04	-
La-142	-	8.0E-05	-
Ce-141	-	5.4E-04	-
Ce-143	-	4.0E-04	-
Pr-143	-	4.9E-04	-
Cs-138	-	8.9E-01	-
Cs-134m	-	4.4E-02	-
Rb-88	-	2.8E+00	-
Rb-89	-	7.3E-02	-
Te-131	-	1.3E-02	-
Te-133	-	7.6E-03	-
Te-134	-	2.6E-02	-
Te-125m	-	5.8E-04	-
Te-133m	-	1.5E-02	-
Ba-141	-	1.2E-04	-
Rh-106	-	1.6E-04	-
Rh-103m	-	4.7E-04	-
Tc-101	-	1.8E-02	-
La-143	-	1.3E-05	-
Nb-97	-	8.4E-05	-

	Concentration (μCi/gm)		
Nb-95m	-	4.0E-06	-
Radionuclide	0.12% Clad Defects [Historical]	1% Clad Defects	0.25% Clad Defects [Historical]
Pr-144	-	4.0E-04	-
Pr-144m	-	7.2E-06	-
Y-94	-	1.8E-05	-
Y-95	-	1.1E-05	-
Y-91m	-	7.7E-04	-
Br-82	-	1.2E-02	-
Br-84	-	3.4E-02	-

Corrosion Products

Radionuclide		Concentration μCi/gm**
Mn	-54	1.1E-03
Mn	-56	-
Co	-58	3.3E-03
Co	-60	3.8E-04
Fe	-59	2.2E-04
Cr	-51	2.2E-03
Fe	-55	8.6E-04

** Corrosion product activities based on values given in Table 2-12, NUREG-0017, April 1976

TABLE 11.1-2 PARAMETERS USED IN THE CALCULATION OF REACTOR COOLANT FISSION AND CORROSION PRODUCT ACTIVITIES

1.	Ultimate core thermal power, MWt		3,659
2.	Clad defects, as a percent of rated core thermal power being generated by rods with clad defects		1.0
3.	Reactor coolant liquid volume, lbm		492,200
4.	Reactor coolant full power average temperature, °F		571.0-589.1
5.	Purification flow rate (normal) gpm		120
6.	Effective cation demineralizer flow, gpm		7.5
7.	Fission product escape rate coefficients*		
	a.	Noble gas isotopes, sec^{-1}	$6.5x10^{-8}$
	b.	Br, Rb, I and Cs isotopes, sec^{-1}	$1.3x10^{-8}$
	c.	Te, Se, Tc, Sn and Sb isotopes, sec^{-1}	$1.0x10^{-9}$
	d.	Mo isotopes, sec^{-1}	$2.0x10^{-9}$
	e.	Sr and Ba isotopes, sec^{-1}	$1.0x10^{-11}$
	f.	Y, Zr, Nb, Ru, Rh, La, Ce, Pr, Nd and Pm isotopes, sec^{-1}	$1.6x10^{-12}$
8.	Mixed bed demineralizer decontamination factors:		
	a.	Noble gases and Cs, Y and Mo	1.0
	b.	All other isotopes including corrosion products	10.0
9.	Cation bed demineralizer decontamination factor for Cs, Y and Mo		10.0
10.	Degasifier noble gas stripping fractions:		
	a.	Kr83m	0.73
	b.	Kr85	$2.3x10^{-1}$
	c.	Kr85m	$2.90x10^{-1}$
	d.	Kr87	$6.00x10^{-1}$
	e.	Kr88	$4.30x10^{-1}$
	f.	Xe131m	$2.50x10^{-1}$
	g.	Xe133	$2.50x10^{-1}$
	h.	Xe133m	$2.60x10^{-1}$
	i.	Xe135	$2.80x10^{-1}$
	j.	Xe135m	$8.00x10^{-1}$
	k.	Xe138	1.0

* Escape rate coefficients are based on fuel defect tests performed at the Saxton reactor. Experience at two plants operating with fuel rod defects has verified the listed escape rate coefficients.

TABLE 11.1-3 PRINCIPAL PARAMETERS USED IN ESTIMATING REALISTIC RELEASES OF RADIOACTIVE MATERIAL IN EFFLUENTS FROM SEABROOK [HISTORICAL]

Reactor Power level, megawatts thermal	3,654
Plant Capacity Factor	0.80
Operating Power Fission Product Source Term, percent clad defects	0.12
Primary System	
Mass of coolant, pounds	505,000
Letdown Rate to chemical and Volume Control System, lbs/hr	4.01×10^4
Shim bleed rate and primary system equipment leakage, lbs/hr	4.16×10^2
leakage rate to secondary system, pounds per day	100
leakage rate to auxiliary area, pounds per day	160
Frequency of degassing (cold shutdown), times per year	2
Secondary System	
Steam flow rate, pounds per hour	15.14×10^6
Mass of steam in each generator, pounds	5,700
Mass of liquid in each generator, pounds	95,500
Mass of secondary coolant, pounds	1.8×10^6
Rate of steam leakage to Turbine Building, pounds per hour	1,700
Containment Building Volume, cubic feet	2.715×10^6
Frequency of Containment Purges, times per year	4 (refueling and maintenance)
Continuous Ventilation Rate, ft^3/min	1,000
Turbine Building Leak Rate, gallons per day	7,200
Iodine Partition Factors	
Steam generator internal partition	0.01
Primary coolant leak to auxiliary area	0.0075
Condenser/vacuum pump (volatile species)	0.15
Iodine Decontamination Factor for Ventilation Systems Charcoal absorbers	10
Particulate Decontamination Factors for Ventilation System HEPA Filters	100

Liquid Waste Processing Systems

System	Input Flow Rate, gallons per day	Decontamination Factors: Iodine	Decontamination Factors: Cesium, Rubidium	Decontamination Factors: Others
Miscellaneous Waste	1360	10^3	10^4	10^4
Equipment Drain	302	10^3	$2x10^3$	10^4
Turbine Building Sump Waste	7200	1	1	1
Boron Recovery	878	10^3	$2x10^3$	10^4
Steam Generator Blowdown (During a primary-to-secondary leak)	$1.1x10^5$	10^2	10^2	10^2

Note: The analysis and parameters described in this table are historical and are the basis of the 10 CFR Part50 Appendix I program. It is noted that for an operating plant, the actual performance and operation of installed equipment, the reporting of offsite releases and doses, and compliance with the regulatory limits of 10CFR50 Appendix I, 10CFR20, and 40CFR190 are controlled by the Offsite Dose Calculation Manual.

TABLE 11.1-4 STEAM GENERATOR SECONDARY SIDE EQUILIBRIUM RADIONUCLIDE CONCENTRATIONS [HISTORICAL]

Radionuclide Defects	Concentration (μCi/gm) Expected Values(1) 0.12% Clad Defects	Design Values(2) 0.25% Clad Defects
A. Water		
H-3	1.00E-03*	1.4E-03
N-16	1.00E-06	-
I-130	1.45E-07	-
I-131	3.33E-05	1.6E-04
I-132	1.04E-05	1.1E-05
I-133	3.51E-05	1.8E-04
I-134	6.12E-07	2.9E-06
I-135	1.01E-05	5.8E-05
Br-83	1.41E-07	-
Rb-86	1.02E-08	-
Sr-89	4.91E-08	3.1E-07
Sr-90	1.20E-09	9.4E-09
Sr-91	3.81E-08	1.0E-06
Y-90	6.66E-10	9.9E-09
Y-91	7.34E-09	4.1E-07
Y-92	-	1.6E-08
Zr-95	7.33E-09	4.5E-08
Nb-95	7.43E-09	4.9E-08
Mo-99	9.88E-06	2.0E-05
Tc-99m	2.24E-05	-
Te-127m	2.18E-08	-
Te-127	1.29E-07	-
Te-129m	1.49E-07	-
Te-129	6.28E-07	-
Te-131m	2.09E-07	-
Te-132	2.54E-06	1.6E-05
Cs-134	2.88E-06	2.8E-05

* 1.00E-03 = 1.00×10^{-3}

	Concentration (μCi/gm)	
Radionuclide Defects	Expected Values[1] 0.12% Clad Defects	Design Values[2] 0.25% Clad Defects
Cs-136	1.30E-06	1.4E-05
Cs-137	1.92E-06	1.3E-04
Ba-140	2.35E-08	2.4E-07
La-140	3.04E-08	7.0E-08
Ce-144	4.82E-09	2.9E-08
Mn-54	4.82E-08	2.6E-07
Mn-56	-	2.0E-06
Co-58	1.71E-06	8.8E-06
Co-60	2.16E-07	2.6E-07
Fe-59	1.23E-07	3.5E-08
Cr-51	2.00E-07	3.3E-07
Fe-55	1.68E-07	-
Np-239	1.03E-07	-
All others	1.1E-05	-
B. Steam		
H-3	1.00E-03	1.4E-03
N-16	1.00E-07	-
I-130	1.45E-09	-
I-131	3.33E-07	1.6E-06
I-132	1.04E-07	1.1E-07
I-133	3.51E-07	1.8E-06
I-134	6.12E-09	2.9E-08
I-135	1.01E-07	5.8E-07
Kr-83m	5.56E-09	5.1E-08
Kr-85m	2.41E-08	2.0E-07
Kr-85	5.63E-10	1.5E-08
Kr-87	1.62E-08	1.5E-07
Kr-88	4.85E-08	3.9E-07
Kr-89	1.61E-09	-
Xe-131m	1.48E-09	7.8E-09
Xe-133m	1.07E-08	6.4E-08

	Concentration (μCi/gm)	
Radionuclide Defects	Expected Values[(1)] 0.12% Clad Defects	Design Values[(2)] 0.25% Clad Defects
Xe-133	4.37E-07	2.9E-06
Xe-135m	4.06E-09	9.7E-08
Xe-135	5.54E-08	3.6E-07
Xe-137	2.88E-09	2.0E-08
Xe-138	1.35E-08	8.3E-08
Br-83	1.41E-09	-
Rb-86	1.02E-11	-
Sr-89	4.91E-12	3.1E-10
Sr-90	1.20E-12	9.4E-12
Sr-91	3.81E-11	1.0E-09
Y-90	6.66E-13	9.9E-13
Y-91	7.34E-12	4.1E-11
Y-92	-	1.6E-12
Zr-95	7.33E-12	4.5E-11
Nb-95	7.43E-12	4.9E-11
Mo-99	9.88E-09	2.0E-08
Te-99m	2.24E-08	-
Te-127m	2.18E-11	-
Te-127	1.29E-10	-
Te-129m	1.49E-10	-
Te-129	6.28E-10	-
Te-131m	2.09E-10	-
Te-132	2.54E-09	1.6E-08
Cs-134	2.88E-09	-
Cs-136	1.30E-09	-
Cs-137	1.92E-09	1.3E-08
Ba-140	2.35E-11	2.4E-10
La-140	3.09E-11	7.0E-11
Ce-144	4.82E-12	2.9E-11
Mn-54	4.82E-11	2.6E-10
Mn-56	-	2.0E-09

	Concentration (μCi/gm)	
Radionuclide Defects	Expected Values(1) 0.12% Clad Defects	Design Values(2) 0.25% Clad Defects
Co-58	1.71E-09	8.8E-09
Co-60	2.16E-10	2.6E-10
Fe-59	1.23E-10	3.5E-11
Cr-51	2.00E-10	3.3E-10
Fe-55	1.68E-10	-
Np-239	1.03E-10	-
All Others	1.1E-08	-

Notes

(1) Bases: 0.12% clad defects
100 lbs/day primary-to-secondary leak rate
75 gpm steam generator blow down rate
95,500 lbm per steam generator
3,654 MWt
0.1% moisture carryover for nonvolatiles
1.0% moisture carryover for halogens

(2) Bases: 0.25% clad defects
20 gal/day primary-to-secondary leak rate
50 gpm steam generator blowdown rate
97,000 lbm per steam generator
3,654 MWt
0.25% moisture carryover for nonvolatiles
1.0% moisture carryover for halogens

Note: The analysis and parameters described in this table are historical and are the basis of the 10CFR Part50 Appendix I program. It is noted that for an operating plant, the actual performance and operation of installed equipment, the reporting of offsite releases and doses, and compliance with the regulatory limits of 10CFR50 Appendix I, 10CFR20, and 40CFR190 are controlled by the Offsite Dose Calculation Manual.

TABLE 11.1-5 SECONDARY SYSTEM CONDENSATE RADIONUCLIDE CONCENTRATIONS [HISTORICAL]

	Concentration (µCi/gm)	
Radionuclide	Expected Values[(1)] 0.12% Clad Defects	Design Values[(2)] 0.25% Clad Defects
H-3	1.00E-03[*]	-
N-16	1.00E-07	-
I-130	1.45E-09	-
I-131	3.33E-07	1.6E-06
I-132	1.04E-07	1.1E-07
I-133	3.51E-07	1.8E-06
I-134	6.12E-09	2.9E-08
I-135	1.01E-07	5.8E-07
Br-83	1.41E-09	-
Rb-86	1.02E-11	-
Sr-89	4.91E-11	7.8E-10
Sr-90	1.20E-12	2.4E-11
Sr-91	3.81E-11	2.6E-09
Y-90	6.66E-13	2.5E-12
Y-91	7.34E-12	1.0E-10
Y-92	-	-
Zr-95	7.33E-12	1.1E-10
Nb-95	7.43E-12	1.2E-10
Mo-99	9.88E-09	5.0E-08
Tc-99m	2.24E-08	-
Te-127m	2.18E-11	-
Te-127	1.29E-10	-
Te-129m	1.49E-10	-
Te-129	6.28E-10	-
Te-131m	2.09E-10	-
Te-132	2.54E-09	4.00E-08

(1) Note 1 of Table 11.1-4 (Sheet 3 of 3)
(2) Note 2 of Table 11.1-4 (Sheet 3 of 3)
* 1.00E-03 = 1.00×10^{-3}

	Concentration (µCi/gm)	
Radionuclide	Expected Values(1) 0.12% Clad Defects	Design Values(2) 0.25% Clad Defects
Cs-134	2.88E-09	-
Cs-136	1.30E-09	-
Cs-137	1.92E-09	3.3E-08
Ba-140	2.35E-11	7.5E-10
La-140	3.09E-11	1.8E-10
Ce-144	4.82E-12	7.3E-11
Mn-54	4.82E-11	6.5E-10
Mn-56	-	5.0E-09
Co-58	1.71E-09	2.2E-08
Co-60	2.16E-10	6.5E-10
Fe-59	1.23E-10	8.8E-11
Cr-51	2.00E-10	8.3E-10
Fe-55	1.68E-10	-
Np-239	1.03E-10	-

Note: The analysis and parameters described in this table are historical and are the basis of the 10CFR Part50 Appendix I program. It is noted that for an operating plant, the actual performance and operation of installed equipment, the reporting of offsite releases and doses, and compliance with the regulatory limits of 10CFR50 Appendix I, 10CFR20, and 40CFR190 are controlled by the Offsite Dose Calculation Manual.

TABLE 11.2-1 LIQUID WASTE SYSTEM COMPONENTS

Floor Drain Tank

Number	2
Capacity each	10,000 gal.
Material	304 SS
Design Pressure	1 psig
Operating Pressure	Atmospheric
Design Temperature	250°F
Operating Temperature	Ambient
Design Code	ASME Sect. VIII, Div. 1

Floor Drain Tank Pump

Number	2
Design Flow	50 gpm
Design TDH	190 ft
Material	316 SS
Design Pressure	150 psig
Design Temperature	200°F
Design Code	Manufacturer's Standards
Type	Centrifugal frame-mounted
Pump Seals	Double Mechanical

Floor Drain Filter

Number	2
Capacity	50 gpm
Material	304 SS
Retention for 25 micron particles	98%
Design Pressure	200 psig
Design Temperature	250°F
Design Code	ASME Sect. VIII, Div. 1
Type	Cartridge

Duplex Strainer

Number	1
Capacity (Design)	120 gpm
Perforation Size	$\frac{1}{16}$ inch
Design Pressure	150 psig
Design Temperature	200°F
Material	316 SS
Design Code	Manufacturer's Standards
Type	Cartridge

Evaporator, Evaporator Distillate Condenser,
Evaporator Distillate Cooler

Same data as on BRS (Subsection 9.3.5)

Evaporator Distillate Pump

Number	1
Design Flow	30 gpm
Design TDH	170 ft
Material	316 SS
Design Pressure	150 psig
Design Temperature	300°F
Design Code	Manufacturer's Standards
Type	Centrifugal frame-mounted
Pump Seals	Single Mechanical

Evaporator Bottoms Pump

Number	1
Design Flow	15 gpm
Design TDH	40 ft
Material	Goulds Alloy 20
Design Pressure	150 psig
Design Temperature	300°F

Design Code	Manufacturer's Standards	
Type	Centrifugal frame-mounted	
Pump Seals	Double Mechanical	

Evaporator Bottoms Cooler

Number	1	
Design Code	ASME Sect. VIII, Div.1	
	Shell Side	Tube Side.
Design Temperature, °F	200	300
Design Pressure, psig	150	150
Operating Pressure, psig	85	50
Design Flow, gpm	47	15
Fluid	PCCW	Concentrate
Temperature in, °F	85	252
Temperature out, °F	105	180
Material	Carbon Steel	Incoloy 825

Waste Test Tanks

Number	2
Capacity	25,000 gal.
Design Pressure	1 psig
Operating Pressure	Atmospheric
Design Temperature	200°F
Operating Temperature	Ambient to 120°F
Material	304 SS
Design Code	ASME Sect. VIII, Div. 1

Waste Test Tank Pumps

Number	2
Material	316 SS
Design Flow	150 gpm
Design TDH	130 ft

Design Pressure	150 psig
Design Temperature	200°F
Design Code	Manufacturer's Standards
Type	Centrifugal frame-mounted
Pumps Seals	Single Mechanical

Waste Demineralizer

Number	1
Media Volume	75 ft^3
Material	304 SS
Design Pressure	150 psig
Design Temperature	300°F
Design Flow	150 gpm
Design Code	ASME Sect. VIII, Div. 1

Waste Demineralizer Filter

Number	1
Retention of 25 micron particles	98%
Material	304 SS
Design Pressure	150 psig
Design Temperature	200°F
Design Flow	150 gpm
Design Code	ASME Sect. VIII, Div. 1
Type	Cartridge

Piping and Valves

Material	316 SS, Al/Br, Cement-lined CS
Design Code	ANSI B31.1
Safety Class	NNS

Liquid Waste Chemical Addition Pump

Number	1
Design Flow	.75 gpm

Design TDH	40 psi
Material	316 SS
Design Pressure	300 psig
Design Temperature	130°F
Design Code	Manufacturer's Standards
Type	Positive Displacement, gear-type
Pump Seals	Magnetic coupling, no shaft seal

TABLE 11.2-2 BORON RECOVERY SYSTEM RELEASES [HISTORICAL]

Radionuclide	Annual Release (Ci/Year)
I-131	1.57E-02*
I-132	4.00E-05
I-133	8.00E-05
Rb-86	1.00E-05
Sr-89	1.00E-05
Mo-99	9.00E-05
Tc-99m	8.00E-05
Te-127m	1.00E-05
Te-127	1.00E-05
Te-129m	2.00E-05
Te-129	1.00E-05
Te-132	4.00E-05
Cs-134	6.33E-03
Cs-136	7.60E-04
Cs-137	4.78E-03
Ba-137m	4.48E-03
Cr-51	3.00E-05
Mn-54	1.00E-05
Fe-55	4.00E-05
Fe-59	2.00E-05
Co-58	3.20E-04
Co-60	5.00E-05
All Others	1.00E-05
TOTAL (Except Tritium)	3.30E-02

Note: The information presented in the above table represents the inputs/assumptions/results utilized in the Seabrook 10 CFR 50 Appendix I analysis (which is part of the original License Application), and is retained here for historical purposes. For the licensed core power level, the core power level for analysis purposes is 3659 MWt. Updating the Appendix I analysis addressed herein using the core power level of 3659 MWt (which includes margin for power uncertainty) will have an insignificant impact on the calculated effluent releases/doses, since 3659 MWt represents only an approximate 0.1% increase from the core power of 3654 MWt utilized in the original analysis.

* 1.57E-02 = 1.57×10^{-2}

TABLE 11.2-3 NONRECYCLABLE RELEASES FROM LIQUID WASTE SYSTEM [HISTORICAL]

Radionuclide	Annual Release (Ci/Year)
I-130	2.00E-05*
I-131	2.38E-02
I-132	5.20E-04
I-133	7.70E-03
I-135	8.40E-04
Mo-99	4.80E-04
Tc-99m	4.50E-04
Te-129m	2.00E-05
Te-129	1.00E-05
Te-131m	1.00E-05
Te-132	1.70E-04
Cs-134	2.90E-04
Cs-136	1.30E-04
Cs-137	2.10E-04
Ba-137m	1.90E-04
Cr-51	2.00E-05
Fe-55	2.00E-05
Fe-59	1.00E-05
Co-58	1.80E-04
Co-60	2.00E-05
Np-239	1.00E-05
All Others	1.00E-05
TOTAL (Except Tritium)	3.50E-02

Note: The information presented in the above table represents the inputs/assumptions/results utilized in the Seabrook 10 CFR 50 Appendix I analysis (which is part of the original License Application), and is retained here for historical purposes. For the licensed core power level, the core power level for analysis purposes is 3659 MWt. Updating the Appendix I analysis addressed herein using the core power level of 3659 MWt (which includes margin for power uncertainty) will have an insignificant impact on the calculated effluent releases/doses, since 3659 MWt represents only an approximate 0.1% increase from the core power of 3654 MWt utilized in the original analysis.

* 2.00E-05 = 2.00×10^{-5}

TABLE 11.2-4 STEAM GENERATOR BLOWDOWN SYSTEM RELEASES [HISTORICAL]**

Radionuclide	Annual Release (Ci/Year)
I-131	1.10E-3*
I-132	3.22E-3
I-133	3.23E-3
I-134	3.14E-3
I-135	5.27E-3
Mo-99	2.41E-4
Tc-99m	1.27E-4
Te-132	6.33E-5
Cs-134	3.65E-4
Cs-136	4.44E-5
Cs-137	4.86E-4
Ba-137m	9.68E-4
All Others	Negligible
TOTAL (Except Tritium)	1.83E-2

Note: The information presented in the above table represents the inputs/assumptions/results utilized in the Seabrook 10 CFR 50 Appendix I analysis (which is part of the original License Application), and is retained here for historical purposes. For the licensed core power level, the core power level for analysis purposes is 3659 MWt. Updating the Appendix I analysis addressed herein using the core power level of 3659 MWt (which includes margin for power uncertainty) will have an insignificant impact on the calculated effluent releases/doses, since 3659 MWt represents only an approximate 0.1% increase from the core power of 3654 MWt utilized in the original analysis.

** Bases:

- Primary coolant activity per USNRC PWR Gale Code, Rev. 1 (1986)
- 100 lbm/day primary-to-secondary leakage
- 75 gpm blowdown processed
- WL DF value of 100 for listed radionuclides

* 1.10E-3 = 1.10 X 10^{-3}

TABLE 11.2-5 NORMAL SECONDARY SYSTEM CONDENSATE LEAKAGE RELEASES [HISTORICAL]

Radionuclide	Annual Release (Ci/Year)
I-130	1.00E-05*
I-131	2.93E-03
I-132	1.90E-04
I-133	2.53E-03
I-135	5.40E-04
Mo-99	9.00E-05
Tc-99m	1.50E-04
Te-132	2.00E-05
Cs-134	3.00E-05
Cs-136	1.00E-05
Cs-137	2.00E-05
Ba-137m	2.00E-05
Co-58	2.00E-05
TOTAL (Except Tritium)	6.58E-03

Note: The information presented in the above table represents the inputs/assumptions/results utilized in the Seabrook 10 CFR 50 Appendix I analysis (which is part of the original License Application), and is retained here for historical purposes. For the licensed core power level, the core power level for analysis purposes is 3659 MWt. Updating the Appendix I analysis addressed herein using the core power level of 3659 MWt (which includes margin for power uncertainty) will have an insignificant impact on the calculated effluent releases/doses, since 3659 MWt represents only an approximate 0.1% increase from the core power of 3654 MWt utilized in the original analysis.

* 1.00E-05 = 1.00×10^{-5}

TABLE 11.2-6 SUMMARY OF NORMAL RADIOACTIVE LIQUID RELEASES [HISTORICAL]

Radionuclide	Annual Release (Ci/Year)
I-130	4.00E-05*
I-131	4.40E-02
I-132	1.30E-04
I-133	1.19E-02
I-134	3.00E-05
I-135	1.91E-03
Br-83	1.00E-05
Rb-86	1.00E-05
Sr-89	1.00E-05
Mo-99	7.10E-04
Tc-99m	8.00E-04
Te-127m	1.00E-05
Te-127	1.00E-05
Te-129m	4.00E-05
Te-129	3.00E-05
Te-131m	1.00E-05
Te-132	2.40E-04
Cs-134	6.66E-03
Cs-136	9.10E-04
Cs-137	5.02E-03
Ba-137m	4.74E-03
Cr-51	5.00E-05
Mn-54	1.00E-05
Fe-55	6.00E-05
Fe-59	3.00E-05
Co-58	5.20E-04
Co-60	8.00E-05
Np-239	1.00E-05
All Others	2.00E-05
Total (Except Tritium)	7.92E-02

Note: The information presented in the above table represents the inputs/assumptions/results utilized in the Seabrook 10 CFR 50 Appendix I analysis (which is part of the original License Application), and is retained here for historical purposes. For the licensed core power level, the core power level for analysis purposes is 3659 MWt. Updating the Appendix I analysis addressed herein using the core power level of 3659 MWt (which includes margin for power uncertainty) will have an insignificant impact on the calculated effluent releases/doses, since 3659 MWt represents only an approximate 0.1% increase from the core power of 3654 MWt utilized in the original analysis.

* 4.00E-05 = 4.00×10^{-5}

TABLE 11.2-7 RADIOACTIVE LIQUID RELEASES DUE TO ANTICIPATED OPERATIONAL OCCURRENCES [HISTORICAL]

Radionuclide	Annual Release (Ci/Year)
I-130	7.57E-05*
I-131	8.33E-02
I-132	2.46E-03
I-133	2.26E-02
I-134	5.68E-05
I-135	3.62E-03
Br-83	1.89E-05
Br-86	1.89E-05
Sr-89	1.89E-05
Mo-99	1.34E-03
Tc-99m	1.51E-03
Te-127m	1.89E-05
Te-127	1.89E-05
Te-129m	7.57E-05
Te-129	5.68E-05
Te-131m	1.89E-05
Te-132	4.54E-04
Cs-134	1.26E-02
Cs-136	1.72E-03
Cs-137	9.51E-03
Ba-137m	8.98E-03
Ba-140	1.00E-05
La-140	1.00E-05
Cr-51	2.84E-04
Mn-54	7.57E-05
Fe-55	3.60E-04
Fe-59	1.70E-04
Co-58	2.97E-03
Co-60	4.36E-04
Np-239	5.68E-05
All Others	3.79E-05
TOTAL (Except Tritium)	1.50E-01

Note: The information presented in the above table represents the inputs/assumptions/results utilized in the Seabrook 10 CFR 50 Appendix I analysis (which is part of the original License Application), and is retained here for historical purposes. For the licensed core power level, the core power level for analysis purposes is 3659 MWt. Updating the Appendix I analysis addressed herein using the core power level of 3659 MWt (which includes margin for power uncertainty) will have an insignificant impact on the calculated effluent releases/doses, since 3659 MWt represents only an approximate 0.1% increase from the core power of 3654 MWt utilized in the original analysis.

* 7.57E-05 = 7.57x10-5

TABLE 11.2-8 RADIONUCLIDE DISCHARGE CONCENTRATIONS NORMAL LIQUID RELEASES - INCLUDING ANTICIPATED OPERATIONAL OCCURRENCES [HISTORICAL]

Radio Nuclide	Total Annual Release (Ci/yr)	Discharge Concentration (μCi/ml)	(MPC)w (μCi/ml)	Fraction of (MPC)w
H-3	7.30E+02*	1.1E-06	3E-03	3.7E-04
I-130	1.2E-04	1.8E-13	3E-06	6.1E-08
I-131	1.3E-01	2.0E-10	3E-07	6.6E-04
I-132	3.9E-03	5.9E-12	8E-06	7.4E-07
I-133	3.6E-02	5.5E-11	1E-06	5.5E-05
I-134	1.0E-04	1.5E-13	2E-05	7.5E-09
I-135	5.8E-03	8.8E-12	4E-06	2.2E-06
Br-83	4.0E-05	6.1E-14	1E-07	6.1E-07
Rb-86	2.0E-05	3.1E-14	2E-05	1.5E-09
Rb-88	1.0E-05	1.5E-14	NA	NA
Sr-89	3.0E-05	4.6E-14	3E-06	1.5E-08
Mo-99	2.1E-03	3.2E-12	4E-05	8.0E-08
Tc-99m	2.4E-03	3.7E-12	3E-03	1.2E-09
Te-127m	3.0E-05	4.6E-14	5E-05	9.2E-10
Te-127	3.0E-05	4.6E-14	2E-04	2.3E-10
Te-129m	1.2E-04	1.8E-13	2E-05	9.2E-09
Te-129	9.0E-05	1.4E-13	8E-04	1.8E-10
Te-131m	3.0E-05	4.6E-14	4E-05	1.1E-09
Te-132	7.3E-04	1.1E-12	2E-05	5.6E-08
Cs-134	2.0E-02	3.0E-11	9E-06	3.4E-06
Cs-136	2.7E-03	4.1E-12	6E-05	6.9E-08
Cs-137	1.5E-02	2.3E-11	2E-05	1.2E-06
Ba-137m	1.4E-02	2.1E-11	1E-07	2.1E-04
Ba-140	1.0E-05	1.5E-14	2E-05	7.6E-10
La-140	1.0E-05	1.5E-14	2E-05	7.6E-10
Cr-51	1.5E-04	2.3E-13	2E-03	1.1E-10

* 7.30E+02 = 7.30×10^2

Radio Nuclide	Total Annual Release (Ci/yr)	Discharge Concentration (µCi/ml)	(MPC)w (µCi/ml)	Fraction of (MPC)w
Mn-54	4.0E-05	6.1E-14	1E-04	6.1E-10
Fe-55	1.9E-04	2.9E-13	8E-04	3.6E-10
Fe-59	9.0E-05	1.4E-13	5E-05	2.7E-10
Co-58	1.6E-03	2.4E-12	9E-05	2.7E-08
Co-60	2.3E-04	3.5E-13	3E-05	1.2E-08
Np-239	3.0E-05	4.6E-14	1E-04	4.6E-10
All Others	6.0E-05	9.2E-14	1E-07	9.2E-07
TOTAL (Except Tritium)	2.4E-01	3.6E-10	-	9.3E-04

Note: The information presented in the above table represents the inputs/assumptions/results utilized in the Seabrook 10 CFR 50 Appendix I analysis (which is part of the original License Application), and is retained here for historical purposes. For the licensed core power level, the core power level for analysis purposes is 3659 MWt. Updating the Appendix I analysis addressed herein using the core power level of 3659 MWt (which includes margin for power uncertainty) will have an insignificant impact on the calculated effluent releases/doses, since 3659 MWt represents only an approximate 0.1% increase from the core power of 3654 MWt utilized in the original analysis.

TABLE 11.2-9 ESTIMATED ANNUAL RADIOACTIVE LIQUID RELEASES - DESIGN BASES FUEL LEAKAGE (1% FAILED FUEL) [HISTORICAL]

Radionuclide	Total Annual Release (Ci/yr)	Discharge Concentration (μCi/ml)	(MPC)w (μCi/ml)	Fraction of (MPC)w
H-3	6.08E+03*	9.2E-06	3E-03	3.1E-03
I-130	8.33E-04	1.3E-12	3E-06	4.3E-07
I-131	1.08E+00	1.7E-09	3E-07	5.5E-03
I-132	1.92E-02	2.9E-11	8E-06	3.7E-06
I-133	2.67E-01	4.1E-10	1E-06	4.1E-04
I-135	3.50E-02	5.3E-11	4E-06	1.3E-05
Br-83	1.67E-04	2.6E-13	1E-07	2.6E-06
Rb-86	1.67E-04	2.6E-13	2E-05	1.3E-08
Sr-89	2.50E-04	3.8E-13	3E-06	1.3E-07
Mo-99	1.67E-02	2.6E-11	4E-05	6.3E-07
Tc-99m	1.75E-02	2.7E-11	3E-03	9.2E-08
Te-127m	2.50E-04	3.8E-13	5E-05	7.7E-09
Te-127	2.50E-04	3.8E-13	2E-04	1.9E-09
Te-129m	1.00E-03	1.5E-12	2E-05	7.7E-08
Te-129	6.67E-04	1.0E-12	8E-04	1.3E-09
Te-131m	2.50E-04	3.8E-13	4E-05	9.2E-09
Te-132	5.83E-03	9.2E-12	2E-05	4.4E-07
Cs-134	1.50E-01	2.3E-10	9E-06	2.6E-05
Cs-136	2.25E-02	3.4E-11	6E-05	5.8E-07
Cs-137	1.17E-01	1.8E-10	2E-05	9.2E-06
Ba-137m	1.08E-01	1.7E-10	1E-07	1.7E-03
Ba-140	8.33E-05	1.3E-13	2E-05	6.3E-09
La-140	8.33E-05	1.3E-13	2E-05	6.3E-09
Cr-51	1.25E-03	1.9E-12	2E-03	9.2E-10
Mn-54	2.50E-04	3.8E-13	1E-04	3.8E-09
Fe-55	1.50E-04	2.3E-12	8E-04	2.8E-09

* $6.08E{+}03 = 6.08 \times 10^3$

Radionuclide	Total Annual Release (Ci/yr)	Discharge Concentration (μCi/ml)	(MPC)w (μCi/ml)	Fraction of (MPC)w
Fe-59	7.50E-04	1.2E-12	5E-05	2.3E-09
Co-58	1.25E-02	1.9E-11	9E-05	2.1E-07
Co-60	1.92E-03	2.9E-12	3E-05	1.0E-07
Np-239	1.67E-04	2.6E-13	1E-04	2.6E-09
All Others	5.00E-04	7.7E-13	1E-07	7.7E-06
TOTAL (Except Tritium)	1.86E+00	2.8E-09	---	7.7E-03

Note: The information presented in the above table represents the inputs/assumptions/results utilized in the Seabrook 10 CFR 50 Appendix I analysis (which is part of the original License Application), and is retained here for historical purposes. For the licensed core power level, the core power level for analysis purposes is 3659 MWt. Updating the Appendix I analysis addressed herein using the core power level of 3659 MWt (which includes margin for power uncertainty) will have an insignificant impact on the calculated effluent releases/doses, since 3659 MWt represents only an approximate 0.1% increase from the core power of 3654 MWt utilized in the original analysis.

TABLE 11.2-10 PARAMETERS USED IN DOSE CALCULATION OF RADIOACTIVE LIQUID RELEASES [HISTORICAL]

Mixing Ratio (Dilution Factor)	8
Circulating Water Flow Rate (ft^3/sec)	735
Shoreline Width Factor	0.5
Environmental Transit Time (hours)	
Aquatic Foods	24
Shoreline Exposure	0
Usage Factors for Fish (kg/year)	
Adult	21.0
Teen	16.0
Child	6.9
Usage Factors for Other Seafoods (kg/year)	
Adult	5.0
Teen	3.8
Child	1.7
Usage Factors For Shoreline Recreation (hours/year)	
Adult (including recreational clam digging activities)	334
Teen	67
Child	14

Note: The information presented in the above table represents the inputs/assumptions/results utilized in the Seabrook 10 CFR 50 Appendix I analysis (which is part of the original License Application), and is retained here for historical purposes. For the licensed core power level, the core power level for analysis purposes is 3659 MWt. Updating the Appendix I analysis addressed herein using the core power level of 3659 MWt (which includes margin for power uncertainty) will have an insignificant impact on the calculated effluent releases/doses, since 3659 MWt represents only an approximate 0.1% increase from the core power of 3654 MWt utilized in the original analysis.

TABLE 11.2-11 MAXIMUM ANNUAL DOSES FROM ALL PATHWAYS DUE TO RADIOACTIVE LIQUID RELEASES [HISTORICAL]

Organ	Adult (mrem/yr)	Teen (mrem/yr)	Child (mrem/yr)
Total Body	2.5E-03*	1.1E-03	6.3E-04
Skin	1.5E-05	3.0E-04	6.3E-05
Bone	2.1E-03	1.1E-03	1.2E-03
Liver	2.8E-03	1.7E-03	1.3E-03
Kidney	3.6E-03	2.6E-03	2.1E-03
Lung	1.8E-03	6.6E-04	3.8E-04
Thyroid	2.4E-02	2.1E-02	2.2E-02
GI-LLI	7.2E-03	4.5E-03	1.6E-03

Note: The information presented in the above table represents the inputs/assumptions/results utilized in the Seabrook 10 CFR 50 Appendix I analysis (which is part of the original License Application), and is retained here for historical purposes. For the licensed core power level, the core power level for analysis purposes is 3659 MWt. Updating the Appendix I analysis addressed herein using the core power level of 3659 MWt (which includes margin for power uncertainty) will have an insignificant impact on the calculated effluent releases/doses, since 3659 MWt represents only an approximate 0.1% increase from the core power of 3654 MWt utilized in the original analysis.

---

* 2.5E-03 = 2.5×10^{-3}

Highest organ dose: 2.4 E-02 mrem/yr; fraction of Appendix I: 2.4×10^{-3}

TABLE 11.3-1 RADIOACTIVE GASEOUS WASTE SYSTEM COMPONENT DESIGN DATA

Iodine Guard Beds

Quantity	Two
Material	Stainless Steel, Type 304
Type	Vertical, activated carbon cartridge
Capacity	1.2 scfm, 2 psig
Design Pressure	350 psig
Safety Class	NNS*

Molecular Sieve Dryer

Quantity	One
Material	Stainless Steel, Type 304
Type	Vertical, three tower
Capacity	1.2 scfm, 2 psig
Gas Outlet Temperature	70°F max.
Outlet Moisture Content	-40°F dew point (min.)
Design Pressure	350 psig
Design Temperature	500°F
Regeneration flow	11.3 scfm
Safety Class	NNS*

Carbon Delay Beds

Quantity	Five
Material	Carbon Steel
Type	Vertical tower 6x8 mesh type MBQ carbon (1600 lb/vessel)
Capacity	1.2 scfm, 2 psig
Design Pressure	Full vacuum - 350 psig
Gas Temperature	70°F
Safety Class	NNS*

* Purchased as Safety Class 3

Particulate Filters

Quantity	Two
Material	Stainless Steel
Type	HEPA
Capacity	1.2 scfm, 1 psig
Design Pressure	350 psig
Safety Class	NNS

Compressors

Quantity	Three
Material	Stainless Steel
Type	Single-stage, diaphragm
Capacity	H_2 gas comp. - 1.2 scfm @150 psig Regen. comp. - 11.3 scfm @150 psig
Safety Class	NNS

Hydrogen Surge Tank

Quantity	One
Material	Carbon Steel
Type	Vertical
Capacity	44 ft^3
Design Pressure	300 psig
Safety Class	NNS

Drain Transfer Pump

Quantity	One
Type	Rotorgear
Flow Rate	1 gpm
Discharge Pressure	40 psig

<u>Drain Transfer Pump Motor</u>

Voltage	460 V
Frequency	60 Hz
Horsepower	0.5 H.P.
Speed	1725 rpm

TABLE 11.3-2 ANNUAL GASEOUS EFFLUENTS RELEASE (CI/YR) [HISTORICAL]

Radionuclide	Containment Purge	PAB Venting	Turbine Venting	Main Condenser Off-Gas System	Gaseous Waste System	Total
H-3	3.7E+02*	3.7E+02	c	c	c	7.4E+02
C-14	1.0E+00	a	a	a	7.0E+00	8.0E+00
Ar-41	2.5E+01	a	a	a	a	2.5E+01
Kr-83m	a	a	a	a	a	a
Kr-85m	7.0E+00	2.0E+00	a	1.0E+00	a	1.0E+01
Kr-85	1.0E+00	a	a	a	2.6E+02	2.6E+02
Kr-87	2.0E+00	1.0E+00	a	a	a	3.0E+00
Kr-88	1.0E+01	4.0E+00	a	2.0E+00	a	1.6E+01
Kr-89	a	a	a	a	a	a
Xe-131m	3.0E+00	a	a	a	2.0E+01	2.3E+01
Xe-133m	1.6E+01	a	a	a	a	1.6E+01
Xe-133	8.6E+02	3.4E+01	a	2.1E+01	7.7E+01	9.9E+02
Xe-135m	a	a	a	a	a	a
Xe-135	3.1E+01	4.0E+00	a	3.0E+00	a	3.8E+01
Xe-137	a	a	a	a	a	a
Xe-138	a	1.0E+00	a	a	a	1.0E+00
I-130	b	b	b	b	b	b
I-131	1.5E-02	4.3E-03	1.8E-03	2.7E-02	b	4.8E-02
I-132	b	b	b	b	b	b
I-133	1.1E-02	6.3E-03	1.9E-03	4.0E-02	b	5.9E-02
I-134	b	b	b	b	b	b

Radionuclide	Containment Purge	PAB Venting	Turbine Venting	Main Condenser Off-Gas System	Gaseous Waste System	Total
I-135	b	b	b	b	b	b
Mn-54	2.2E-04	1.8E-04	c	c	4.5E-05	4.4E-04
Fe-59	7.4E-05	6.0E-05	c	c	1.5E-05	1.5E-04
Co-58	7.4E-04	6.0E-04	c	c	1.5E-04	1.5E-03
Co-60	3.3E-04	2.7E-04	c	c	7.0E-05	6.7E-04
Sr-89	1.7E-05	1.3E-05	c	c	3.3E-06	3.3E-05
Sr-90	2.9E-06	2.4E-06	c	c	6.0E-07	5.9E-06
Cs-134	2.2E-04	1.8E-04	c	c	4.5E-05	4.4E-04
CS-137	3.7E-04	3.0E-04	c	c	7.5E-05	7.4E-04

* 3.7E+02 = 3.7×10^2

a. Less than 1.0 Ci/yr for Noble Gases and C-14

b. Less than 0.0001 Ci/yr for Iodine.

c. Less than 1.0 percent of total for this nuclide.

Note: The information presented in the above table represents the inputs/assumptions/results utilized in the Seabrook 10 CFR 50 Appendix I analysis (which is part of the original License Application), and is retained here for historical purposes. For the licensed core power level, the core power level for analysis purposes is 3659 MWt. Updating the Appendix I analysis addressed herein using the core power level of 3659 MWt (which includes margin for power uncertainty) will have an insignificant impact on the calculated effluent releases/doses, since 3659 MWt represents only an approximate 0.1% increase from the core power of 3654 MWt utilized in the original analysis.

TABLE 11.3-3 ESTIMATED MAXIMUM DOSES FROM NOBLE GASEOUS RELEASES [HISTORICAL]*

	Gamma Air Dose		Beta Air Dose	
Location	Annual Dose Rate (mrad/yr)	Fraction of Appendix I	Annual Dose Rate (mrad/yr)	Fraction of Appendix I
Highest Site Boundary Dose Point	1.2E-02	1.2E-03	2.3E-02	1.2E-03

	Total Body Dose		Skin Dose	
	Annual Dose Rate (mrad/yr)	Fraction of Appendix I	Annual Dose Rate (mrad/yr)	Fraction of Appendix I
Highest Residential Dose Point	4.4E-03	8.8E-04	1.0E-02	6.7E-04

* These values are for a two unit facility. Single unit values will be less.

Notes:

1. The numerical design objective of 10 CFR 50, Appendix I are:
   - Gamma air dose 10 mrad/yr, Beta air dose – 20 mrad/yr,
   - Total body dose 5 mrem/yr, skin dose – 15 mrem/yr
   - Organ dose due to radioiodines and other particulate radionuclides – 15 mrem/yr
2. The information presented in the above table represents the inputs/assumptions/results utilized in the Seabrook 10 CFR 50 Appendix I analysis (which is part of the original License Application), and is retained here for historical purposes. For the licensed core power level, the core power level for analysis purposes is 3659 MWt. Updating the Appendix I analysis addressed herein using the core power level of 3659 MWt (which includes margin for power uncertainty) will have an insignificant impact on the calculated effluent releases/doses, since 3659 MWt represents only an approximate 0.1% increase from the core power of 3654 MWt utilized in the original analysis.

TABLE 11.3-4 ESTIMATED MAXIMUM ANNUAL DOSE RATES (MREM/YR) FROM RADIOIODINES AND OTHER RADIONUCLIDES DUE TO STACK AND TURBINE VENT RELEASES [HISTORICAL]*

Age Group	Bone	Liver	Kidney	Lung	GI-LLI	Thyroid	Whole Body	Skin
Adult	6.3E-02**	3.4E-02	3.4E-02	3.4E-02	3.4E-02	7.2E-02	3.4E-02	1.1E-03
Teen	9.8E-02	4.4E-02	4.4E-02	4.3E-02	4.3E-02	9.0E-02	4.4E-02	1.1E-03
Child	2.3E-01	8.0E-02	7.9E-02	7.8E-02	7.8E-02	1.6E-01	7.9E-02	I.1E-03
Infant	7.0E-02	2.9E-02	2.9E-02	2.8E-02	2.8E-02	1.5E-01	2.8E-02	1.1E-03

Note: The information presented in the above table represents the inputs/assumptions/results utilized in the Seabrook 10 CFR 50 Appendix I analysis (which is part of the original License Application), and is retained here for historical purposes. For the licensed core power level, the core power level for analysis purposes is 3659 MWt. Updating the Appendix I analysis addressed herein using the core power level of 3659 MWt (which includes margin for power uncertainty) will have an insignificant impact on the calculated effluent releases/doses, since 3659 MWt represents only an approximate 0.1% increase from the core power of 3654 MWt utilized in the original analysis.

* These values are for a two unit facility. Single unit values will be less.

** 6.3E-02 = 6.3×10^{-2} = 0.063

The highest organ doses are 0.23 mrem/yr (child bone) and 0.16 mrem/yr (child thyroid), which are respectively 0.015 and 0.011 fractions of Appendix I dose objectives.

TABLE 11.3-5 GASEOUS WASTE RELEASE SOURCES AND ASSUMPTIONS [HISTORICAL]

1. Containment Venting

Reactor coolant leakage	
Noble gaseous leakage	1% of total reactor coolant daily
Iodine leakage	0.001% of total reactor coolant daily
Charcoal filter efficiency	90%
Containment free volume	$2.715x10^6$ ft^3
Containment vent	4 purges per year during shutdown, 1000 cfm onlinepurge system available during power operation

2. Primary Auxiliary Building

Reactor coolant leakage	160 lb/day
Noble gases released	100%
Iodine released	0.75%
Charcoal filter efficiency	90%
Venting mode	Instantaneous release through charcoal filter

3. Main Condenser

Steam generator tube leak	100 lb/day
Carry over in the steam generator	
Noble gases	100%
Iodine	1%
Other nuclides	0.1%
Fraction of main steam which reaches the main condenser	65%
Partition factor in the main condenser	
Noble gas	1.0
Iodine	0.15
Nonvolatile	0.0
Charcoal filter efficiency	90%
Vacuum pump vent rate	60 cfm

4. Turbine Building Leakage

Secondary steam leakage	1700 lb/hr
Noble gases released	100% of the leakage
Iodine released	100% of the leakage

5. Waste Gas System Release

Continuous stripping plus two reactor volumes degassed per year. All go through Gaseous Waste System.	
Delay time in the charcoal beds	60 days for Xe 85 hr for Kr

Note: The information presented in the above table represents the inputs/assumptions/results utilized in the Seabrook 10 CFR 50 Appendix I analysis (which is part of the original License Application), and is retained here for historical purposes. For the licensed core power level, the core power level for analysis purposes is 3659 MWt. Updating the Appendix I analysis addressed herein using the core power level of 3659 MWt (which includes margin for power uncertainty) will have an insignificant impact on the calculated effluent releases/doses, since 3659 MWt represents only an approximate 0.1% increase from the core power of 3654 MWt utilized in the original analysis.

TABLE 11.3-6 GASEOUS RELEASES WITH 0.5% FAILED FUEL [HISTORICAL]

Radionuclide	Total, Off-Normal Unit Release Rate (Ci/year)
Kr-85m	4.2E+01*
Kr-85	1.1E+03
Kr-87	1.3E+01
Kr-88	6.7E+01
Xe-131m	9.6E+01
Xe-133m	6.7E+01
Xe-133	4.1E+03
Xe-135	1.6E+02
Xe-138	4.2E+00
I-131	2.0E-01
I-133	2.5E-01

Note: The information presented in the above table represents the inputs/assumptions/results utilized in the Seabrook 10 CFR 50 Appendix I analysis (which is part of the original License Application), and is retained here for historical purposes. For the licensed core power level, the core power level for analysis purposes is 3659 MWt. Updating the Appendix I analysis addressed herein using the core power level of 3659 MWt (which includes margin for power uncertainty) will have an insignificant impact on the calculated effluent releases/doses, since 3659 MWt represents only an approximate 0.1% increase from the core power of 3654 MWt utilized in the original analysis.

* 4.2E+01 = 4.2×10^1

TABLE 11.3-7 GASEOUS RELEASES WITH 500 GAL/DAY STEAM GENERATOR TUBE LEAKAGE FOR 90 DAYS [HISTORICAL]

Radionuclide	Total, Off-Normal Unit Release Rate (Ci/year)
Kr-85m	1.6E+01*
Kr-85	2.6E+02
Kr-87	3.0E+00
Kr-88	2.8E+01
Xe-131m	2.3E+01
Xe-133m	1.6E+01
Xe-133	1.1E+03
Xe-135	5.6E+01
Xe-138	1.0E+00
I-131	2.2E-01
I-133	3.1E-01

Note: The information presented in the above table represents the inputs/assumptions/results utilized in the Seabrook 10 CFR 50 Appendix I analysis (which is part of the original License Application), and is retained here for historical purposes. For the licensed core power level, the core power level for analysis purposes is 3659 MWt. Updating the Appendix I analysis addressed herein using the core power level of 3659 MWt (which includes margin for power uncertainty) will have an insignificant impact on the calculated effluent releases/doses, since 3659 MWt represents only an approximate 0.1% increase from the core power of 3654 MWt utilized in the original analysis.

* $1.6E+01 = 1.6 \times 10^{1}$

TABLE 11.3-8 GASEOUS RELEASES WITH 1 GAL/MIN OF REACTOR COOLANT LEAKAGE TO CONTAINMENT FOR 12 DAYS, FOLLOWED BY A CONTAINMENT PURGE [HISTORICAL]

Radionuclide	Total, Off-Normal Unit Release Rate (Ci/year)[(1)]
Kr-85m	1.0E+01*
Kr-85	2.6E+02
Kr-87	3.0E+00
Kr-88	1.6E+01
Xe-131m	2.3E+01
Xe-133m	1.7E+01
Xe-133	1.0E+03
Xe-135	3.9E+01
Xe-138	1.0E+00
I-131	5.6E-02
I-133	6.1E-02

Note: The information presented in the above table represents the inputs/assumptions/results utilized in the Seabrook 10 CFR 50 Appendix I analysis (which is part of the original License Application), and is retained here for historical purposes. For the licensed core power level, the core power level for analysis purposes is 3659 MWt. Updating the Appendix I analysis addressed herein using the core power level of 3659 MWt (which includes margin for power uncertainty) will have an insignificant impact on the calculated effluent releases/doses, since 3659 MWt represents only an approximate 0.1% increase from the core power of 3654 MWt utilized in the original analysis.

---

(1) Total release from all sources with 1 gpm of reactor coolant leakage in the Containment for 12 days prior to purge and the following assumptions:

a. Iodine partition factor = 0.0075

b. Carbon filter efficiency = 90%

* 1.0E+01 = 1.0×10^1

TABLE 11.3-9 GASEOUS RELEASES WITH 200 GAL/DAY REACTOR COOLANT LEAKAGE TO PRIMARY AUXILIARY BUILDING FOR 90 DAYS [HISTORICAL]

Radionuclide	Total, Off-Normal Unit Release Rate (Ci/year)
Kr-85m	1.4E+01*
Kr-85	2.6E+02
Kr-87	5.2E+00
Kr-88	2.5E+01
Xe-131m	2.3E+01
Xe-133m	1.6E+01
Xe-133	1.1E+03
Xe-135	4.7E+01
Xe-138	3.2E+00
I-131	5.8E-02
I-133	7.3E-02

Note: The information presented in the above table represents the inputs/assumptions/results utilized in the Seabrook 10 CFR 50 Appendix I analysis (which is part of the original License Application), and is retained here for historical purposes. For the licensed core power level, the core power level for analysis purposes is 3659 MWt. Updating the Appendix I analysis addressed herein using the core power level of 3659 MWt (which includes margin for power uncertainty) will have an insignificant impact on the calculated effluent releases/doses, since 3659 MWt represents only an approximate 0.1% increase from the core power of 3654 MWt utilized in the original analysis.

* 1.4E+01 = 1.4×10^1

TABLE 11.4-1 PERMANENTLY INSTALLED EQUIPMENT NEEDED FOR PROCESSING WET WASTE VIA ALTERNATE MOBILE SOLIDIFICATION SYSTEM

Component	Quantity
Tanks:	
Spent resin hopper	1
Waste concentrates	1
Spent resin sluice	2
Waste feed	2
Pumps:	
Waste feed recirculation	2
Resin dewatering	1
Waste concentrates transfer	1
Spent resin transfer	1
Spent resin sluice	2
Spent resin recirculation	1
Resin centrifuge metering	1
Alternate solidification concentrates feed	1
Filters:	
Spent resin sluice	1
Panels:	
Crane control station	1
Crane power	1
Alternate solidification	1
Station feed control	1
Metering pumps SCR	1
Others:	
30 ton bridge crane	1
7½ ton transporter cart	1
Spent filter transfer cask	1

TABLE 11.4-2 DESIGN VOLUMETRIC INPUTS TO SOLID WASTE MANAGEMENT SYSTEM

Sources	Design Annual Volume (cf/yr)*	Expected Annual** Volume (cf/yr)
Dry active wastes		
(a) Noncompactible trash	8,800	4,400
(b) Compactible trash	20,400	10,200
Spent Demineralizer Resins	4,300	1,400
Evaporator Bottoms (at 12 w/o solids concentration) and Others, including filter cartridges	12,600	4,300
Totals	46,100	20,300

* The system design processing rate for evaporator bottoms and spent resins, using permanently installed equipment, is sufficient to process all of the design level quantities within 1664 hours of operation per year. This is equivalent to a 19% annual usage factor.

** These values are for a two unit facility. Single unit values will be less.

TABLE 11.4-3 DESIGN ACTIVITY INPUTS TO SOLID WASTE MANAGEMENT SYSTEM [HISTORICAL]

Sources		Expected Annual Activity (Ci/yr)*
Dry Active Wastes		7.2+01**
Spent Demineralizer Resins		3.6+03
Evaporator Bottoms & Other (at 12 w/o solids concentration)		3.1+02
Filter Cartridges		2.3+01
	Total	4.0+03

Note: The information presented in the above table represents the inputs/assumptions/results utilized in the Seabrook 10 CFR 50 Appendix I analysis (which is part of the original License Application), and is retained here for historical purposes. For the licensed core power level, the core power level for analysis purposes is 3659 MWt. Updating the Appendix I analysis addressed herein using the core power level of 3659 MWt (which includes margin for power uncertainty) will have an insignificant impact on the calculated effluent releases/doses, since 3659 MWt represents only an approximate 0.1% increase from the core power of 3654 MWt utilized in the original analysis.

* These values are for a two unit facility. Single unit values will be less.

** $7.2{+}01 = 7.2 \times 10^{1}$

TABLE 11.4-4 BASES AND ASSUMPTIONS FOR DETERMINING SOLID WASTES ACTIVITIES [HISTORICAL]

Volume Reduction Factors	
Compactible dry active waste	4 to 10
Bead resins in asphalt	1.85 minimum
Evaporator concentrates in asphalt	6.3 minimum
Radwaste in cement	1.0
Waste to Binder Mix Ratios, by Volume	
Asphalt	1 to 1 minimum
Cement	1.5 to 1 minimum
Waste Container Fill Fractions	
Bead resins in asphalt	91% average
Evaporator concentrates in asphalt	93% average
Radwaste in cement	93% average
Expected Failed Fuel Fraction	0.125% failed fuel
Decay of Filled Containers Awaiting Shipout	
Dry active wastes	none
Bead resins	30 days minimum
Evaporator concentrates	30 days minimum
Densities	
Noncompactible trash	10.0 lbs/cf
Compactible trash	37.4 lbs/cf
Encapsulated and solidified filter cartridges	37.4 lbs/cf
Depleted bead resins	56.8 lbs/cf
Evaporator concentrates	62.4 lbs/cf

Miscellaneous

Plant availability is eighty percent.

Each nonregenerable demineralizer is changed annually.

A maximum of thirty filter cartridges are expected to be shipped.

Time needed to fill each 55 gallon drum of waste for shipout, using asphalt binder, is 2 hours minimum; 6 hours maximum.

Time needed to fill each 85 cubic foot container of waste for shipout, using cement binder, is nominally 45 minutes.

Note: The information presented in the above table represents the inputs/assumptions/results utilized in the original License Application to determine solid waste activities and is retained here for historical purposes.

TABLE 11.4-5 EXPECTED ACTIVITY OF DRY ACTIVE WASTES AFTER PROCESSING [HISTORICAL][**]

Noncompactible Trash

Isotope	ìCi/cc	Isotope	ìCi/cc
I-131	7.03-04[*]	Fe-59	8.44-03
Cs-134	2.32-02	Cr-51	8.59-02
Cs-137	3.83-02	Zn-65	1.05-03
Mn-54	7.28-02	Others	4.23-03
Co-58	1.41-02		
Co-60	1.04-01	Total	3.53-01

Compactible Trash

Isotope	ìCi/cc	Isotope	ìCi/cc
I-131	3.05-05	Fe-59	3.67-04
Cs-134	1.01-03	Cr-51	3.73-03
Cs-137	1.66-03	Zn-65	4.58-05
Mn-54	3.16-03	Other	1.84-04
Co-58	6.12-04		
Co-60	4.50-03	Total	1.53-02

Note: The information presented in the above table represents the inputs/assumptions/results utilized in the Seabrook 10 CFR 50 Appendix I analysis (which is part of the original License Application), and is retained here for historical purposes. For the licensed core power level, the core power level for analysis purposes is 3659 MWt. Updating the Appendix I analysis addressed herein using the core power level of 3659 MWt (which includes margin for power uncertainty) will have an insignificant impact on the calculated effluent releases/doses, since 3659 MWt represents only an approximate 0.1% increase from the core power of 3654 MWt utilized in the original analysis.

** based on Table 11.4-4

* 7.03-04 = 7.03x10^{-4}

TABLE 11.4-6 EXPECTED ACTIVITY OF SPENT DEMINERALIZER RESINS [HISTORICAL]**

Isotope	ìCi/cc	Isotope	ìCi/cc
I-131	1.6+0*	Te-132	5.5-4
Sr-89	2.7-1	Ba-140	1.9+0
Sr-90	1.7-1	Ce-144	3.6+1
Y-91	1.2-2	Mn-54	1.5+1
Zr-95	1.0+0	Co-58	1.4+1
Nb-95	4.4-1	Co-60	7.2+0
Mo-99	5.4-4	Fe-59	4.8-1
Cs-134	1.8+1	Cr-51	2.4+0
Cs-136	1.9-1		
Cs-137	2.4+1	Total	1.2+2

Processed Using Cement Binder Via Alternate Mobile Equipment

Isotope	ìCi/cc	Isotope	ìCi/cc
I-131	5.2-1	Te-132	1.8-4
Sr-89	8.7-2	Ba-140	6.1-1
Sr-90	5.6-2	Ce-144	1.2+1
Y-91	3.9-3	Mn-54	4.9+0
Zr-95	3.3-1	Co-58	4.4+0
Nb-95	1.4-1	Co-60	2.3+0
Mo-99	1.7-4	Fe-59	1.6-1)
Cs-134	6.0+0	Cr-51	7.6-1
Cs-136	6.2-2		
Cs-137	7.8+0	Total	4.0+1

Note: The information presented in the above table represents the inputs/assumptions/results utilized in the Seabrook 10 CFR 50 Appendix I analysis (which is part of the original License Application), and is retained here for historical purposes. For the licensed core power level, the core power level for analysis purposes is 3659 MWt. Updating the Appendix I analysis addressed herein using the core power level of 3659 MWt (which includes margin for power uncertainty) will have an insignificant impact on the calculated effluent releases/doses, since 3659 MWt represents only an approximate 0.1% increase from the core power of 3654 MWt utilized in the original analysis.

** based on Table 11.4-4

* $1.6+01 = 1.6x10^0$

TABLE 11.4-7 EXPECTED ACTIVITY OF EVAPORATOR BOTTOMS AND OTHER [HISTORICAL]**

Evap Btms & Chemical Wastes Processed Using Asphalt Binder

Isotope	ìCi/cc	Isotope	ìCi/cc
I-131	7.1-1*	Te-132	6.3-4
Sr-89	1.6-2	Ba-140	2.5-3
Sr-90	1.6-3	Ce-144	3.2-3
Y-91	2.4-2	Mn-54	4.7-2
Zr-95	3.0-3	Co-58	1.7-1
Nb-95	2.9-3	Co-60	5.3-2
Mo-99	2.2-3	Fe-59	3.0-2
Cs-134	6.8-1	Cr-51	1.6-2
Cs-136	1.8-1		
Cs-137	8.7-1	Total	3.3+0

Evap Btms & Chemical Wastes Processed Using Cement Binder Via Alternate Mobile Equipment

Isotope	ìCi/cc	Isotope	ìCi/cc
I-131	6.8-2	Te-132	6.0-5
Sr-89	1.5-3	Ba-140	2.4-4
Sr-90	1.5-4	Ce-144	3.1-4
Y-91	2.3-3	Mn-54	4.5-3
Zr-95	2.9-4	Co-58	1.7-2
Nb-95	2.8-4	Co-60	5.1-3
Mo-99	2.1-4	Fe-59	2.8-3
Cs-134	6.4-2	Cr-51	1.5-3
Cs-136	1.7-2		
Cs-137	8.3-2	Total	2.4-1

Note: The information presented in the above table represents the inputs/assumptions/results utilized in the Seabrook 10 CFR 50 Appendix I analysis (which is part of the original License Application), and is retained here for historical purposes. For the licensed core power level, the core power level for analysis purposes is 3659 MWt. Updating the Appendix I analysis addressed herein using the core power level of 3659 MWt (which includes margin for power uncertainty) will have an insignificant impact on the calculated effluent releases/doses, since 3659 MWt represents only an approximate 0.1% increase from the core power of 3654 MWt utilized in the original analysis.

** based on Table 11.4-4

* 7.1-1 = 7.1×10^{-1}

TABLE 11.4-8 SOLID WASTE MANAGEMENT SYSTEM EQUIPMENT PARAMETERS

Tanks

Waste Concentrates Tank (1-WS-TK-76)

Capacity	6750 gallons – maximum 6000 gallons - working
Material	Incoloy 825
Design/Operating Pressure	15 psig/atmos
Design/Operating Temperature	250°F/180°F
Design Code	ASME VIII - Div. 1
Minimum Holdup Time	2.0 Hrs.

Waste Feed Tanks (1-WS-TK-198A & B)

Capacity (each)	1320 gallons – maximum 1000 gallons - working
Material	Incoloy 825
Design/Operating Pressure	14.7 psig/atmos
Design/Operating Temperature	265°F/180°F
Design Code	ASME VIII Div. 1 - Nonstamped
Minimum Holdup Time	23.8 Hrs.

Bottoms Collection Tank (1-WS-TK-200)

Capacity	1300 gallons – maximum 1000 gallons - working
Material	Incoloy 825
Design/Operating Pressure	5 psig/0.5 psig
Design/Operating Temperature	250°F/200°F
Design Code	ASME VIII - Div. 1
Minimum Holdup Time	23.8 Hrs.

Caustic Day Tank (1-WS-TK-199)

Capacity	250 gallons – maximum 200 gallons - working
Material	(Plastic)
Design/Operating Pressure	(Atmos/atmos)
Design/Operating Temperature	200°F/150°F
Design Code	Mfr. Std.
Safety Class	NNS

Resin Hopper (1-WS-TK-81)

Capacity	934 gallons
Design Operating Pressure	Atmos./atmos.
Material	Incoloy 825
Design Operating Temperature	200°F/200°F
Design Code	ASME VIII
Minimum Holdup Time	31.1 Hrs.

Crystallizer Distillate Tank (1-WS-TK-210)

Capacity	30 gallons
Material	304L Stainless Steel
Design/Operating Pressure	30 psig/Atmos.
Design Operating Temperature	220°F/130°F
Design Code	ASME VIII

Asphalt Storage Tank (1-WS-TK-201)

Capacity	7400 gallons – maximum 7000 gallons - working
Material	ASTM A285
Design/Operating Pressure	Atmos./atmos.
Design/Operating Temp.	425°F/325°F
Design Code	API 650

Auxiliary Boiler Condensate Return Tank
(1-WS-TK-202)

Capacity	40 gallons
Material	ASTM A36
Operating Pressure	Atmos.
Operating Temperature	210°F
Design Code	Mfr. Standards

Spent Resin Sluice Tanks (RS-RK-79A,B)

Capacity	9500 gal.
Material	304 SS
Design/Operating Press	150/5 psig
Design/Operating Temp.	150°F
Design Code	ASME VIII

Pumps

Concentrates Transfer Pump (1-WS-P-354)

Type	Centrifugal
Design Flow	50 gpm

Waste Feed Recirculation Pumps
(1-WS-P-332A & B)

Type	Centrifugal
Design Flow	50 gpm

Caustic Metering Pump (1-WS-P-333)

Type	Progressive Cavity
Design Flow	2 gpm

Alternate Station Concentrates Feed Pump
(1-WS-P-346)

Type	Progressive Cavity
Design Flow	18 gpm

Spent Resin Transfer Pump (1-WS-P-331)

Type	Progressive Cavity
Design Flow	100 gpm

Spent Resin Dewatering Pump (1-WS-P-353)

Type	In-line Centrifugal
Design Flow	10 gpm

Spent Resin Recirculation Pump (1-WS-P-342)

Type	Progressive Cavity
Design Flow	50 gpm

Resin Centrifuge Metering Pump (1-WS-P-338)

Type	Progressive Cavity
Design Flow	1-10 gpm

Crystallizer Recirculation Pump (1-WS-P-334)

Type	Centrifugal
Design Flow	1800 gpm

Crystallizer Distillate Pump (1-WS-P-335)

Type	Centrifugal
Design Flow	5 gpm

Crystallizer Reflux Pump (1-WS-P-352)

Type	Diaphragm
Design Flow	1 gpm

Crystallizer Drain Pump (1-WS-P-347)

Type	Centrifugal
Design Flow	1 gpm

Asphalt Recirculation Pump (1-WS-P-339)

Type	Progressive Cavity
Design Flow	20 gpm

Asphalt Metering Pump (1-WS-P-340)

Type	Progressive Cavity
Design Flow	3-30 gpm

Auxiliary Boiler Feed Pumps
(1-WS-P-341 A & B)

Type	Centrifugal
Design Flow	9 gpm

Spent Resin Sluice Pumps (WS-P-13 A, B)

Type	Centrifugal
Design Flow	250 gpm

Heat Exchangers, Coolers, Heaters, etc.

Crystallizer Condenser (1-WS-E-158)

Design Code	ASME VIII, Div. 1 TEMA R	
	Shell Side	Tube Side
Fluid	Process Vapor	Cooling Water
Design Flow	(2685 #/hr)	(150 gpm)
Design Temp	250°F	200°F
Operating Temperature	213°F	125°F
Design Pressure	30 psig	150 psig
Operating Pressure	Atmos.	60 psig
Material	304 S/S	Carbon Steel

Crystallizer Subcooler (1-WS-E-160)

Design Code	ASME VIII, Div. 1 TEMA R	
	Shell Side	Tube Side
Fluid	Distillate	Cooling Water
Design Flow	(2185 #/hr)	(150 gpm)
Design Temperature	250°F	200°F
Operating Temperature	212°F	85°F
Design Pressure	150 psig	150 psig
Operating Pressure	100 psig	100 psig
Material	304 S/S	Carbon Steel

Crystallizer Heater (1-WS-E-156)

Design Code	ASME VIII, Div. 1 TEMA R	
	Shell Side	Tube Side
Fluid	Steam	Concentrates
Design Flow	(3500 #/hr)	(1200 gpm)
Design Temperature	300°F	300°F
Operating Temperature	298°F	220°F
Design Pressure	150 psig	50 psig
Operating Pressure	50 psig	15 psig
Material	Carbon Steel	Inconel 625

Crystallizer Vapor Body (1-WS-EV-6)

Material	Inconel 625
Capacity	600 gallons - vapor body working 1200 gallons -vapor body, heater, recirculation pipe flooded
Design Code	ASME VIII, Div. 1
Design/Operating Pressure	30 psig/15 psia
Design/Operating Temperature	300°F/222°F

Entrainment Separator (1-WS-E-157)

Capacity	2695 #/hr.
Volume	20 gallons – working 300 gallons - flooded
Design Code	ASME VIII, Div. 1
Design Pressure	Full vacuum to 30 psig
Operating Pressure	Atmos.
Design Temperature	270°F
Operating Temperature	213°F

Auxiliary Boiler Vessel (1-WS-E-159)

Capacity - Steam	5000 #/Hr.
Operating Pressure	275 psig
Operating Temperature	410°F
Design Code	ASME VIII, Div. 1

Other Components

Resin Centrifuge (1-WS-MM-611)

Type	Horizontal
Capacity	4.7 gpm
Operating Pressure	Atmos.
Operating Temperature	Ambient
Design Code	Mfr. Std.

Evaporator/Extruder

Capacity Range	9.25 to 27.5 gph
Operating Pressure	Atmos.
Operating Temp.	280°F
Design Code	DIN

Shipping Containers

Types	55 gallon drums, 100 ft^3LSA boxes, and 85 ft^3 liners with associated shield

Resin Filter (RS-F-19)

Type	Backflushable, wedgewire
Design Code	ASME VIII, Div. 1

TABLE 11.4-9 DESIGN SHIPPING VOLUMES

	Processed Using Asphalt Binder Via Permanently Installed Equipment		Processed Using Cement Binder Via Alternate Mobile Equipment	
Sources	Design Annual Volume (Ft^3)	Expected* Annual Volume (Ft^3)	Design Annual Volume (ft^3)	Expected* Annual Volume (ft^3)
Dry Active Wastes				
a) Noncompactible Trash	8,800	4,400	8,800	4,400
b) Compactible Trash	5,100	2,600	5,100	2,600
Spent Demineralizer Resins	2,300	770	7,200	2,400
Evaporator Bottoms & Other	1,900	630	20,000	6,600
Encapsulated and Solidified Filter Cartridges	600	300	600	300
Totals	18,700	8,700	41,700	16,300

* These values are for a two unit facility. Single unit values will be less.

TABLE 11.4-10 EXPECTED ANNUAL ACTIVITY AVAILABLE FOR OFFSITE SHIPMENT (CI/YR) [HISTORICAL]*

Isotope	Dry Active Wastes	Spent Demin Resins[(1)]	Evap Bottoms[(1)]	Total
I-131	1.4-1**	3.5+1	1.3+1	4.7+1
Sr-89	-------	5.9+0	2.8-1	6.1+0
Sr-90	-------	3.8+0	2.8-2	3.8+0
Sr-91	-------	2.7-1	4.4-1	7.1-1
Zr-95	-------	2.2+1	5.4-2	2.2+1
Nb-95	-------	9.5+0	5.3-2	9.5+0
Mo-99	-------	1.2-2	4.0-2	6.2-2
Cs-134	4.8+0	4.0+2	1.2+1	4.2+2
Cs-136	---	4.2+0	3.2+0	7.9+0
Cs-137	7.8+0	5.3+2	1.6+1	5.5+2
Te-132	-------	1.2-2	1.1-2	2.3-2
Ba-140	-------	4.1+2	4.5-2	4.1+2
Ce-144	-------	7.8+2	5.8-2	7.8+2
Mn-54	1.5+1	3.3+2	8.5-1	3.4+2
Co-58	2.9+0	3.0+2	3.1+0	3.1+2
Co-60	2.1+1	1.6+2	9.5-1	1.8+2
Fe-59	1.7+0	1.0+1	5.3-1	1.2+1
Cr-51	1.8+1	5.1+1	2.8-1	7.0+1
Others	1.1+0	------	------	1.0+0
Totals	7.2+1	2.7+3	5.2+1	2.7+3

* These values are for a two-unit facility. Single unit values will be less.

(1) Thirty days decay minimum prior to shipout.

** 1.4-1 = 1.4×10^{-1}

Note: The information presented in the above table represents the inputs/assumptions/results utilized in the Seabrook 10 CFR 50 Appendix I analysis (which is part of the original License Application), and is retained here for historical purposes. For the licensed core power level, the core power level for analysis purposes is 3659 MWt. Updating the Appendix I analysis addressed herein using the core power level of 3659 MWt (which includes margin for power uncertainty) will have an insignificant impact on the calculated effluent releases/doses, since 3659 MWt represents only an approximate 0.1% increase from the core power of 3654 MWt utilized in the original analysis.

TABLE 11.5-1 PROCESS AND EFFLUENT RADIATION MONITORS

Instrument Tag No. Re-	Description	Detector Type	Det Back Grd mr/hr	Range Low-High (μci/cc)	(Note 5) Alarm Set Point (μci/cc)	Reference Isotope	Detector Qty	Safety Class	Energy* Level	Loop Diag. 1-NHY	P&Id 1-NHY
6454	Storm Drains	Gamma Scint	0.5	10^{-6} 10^{-2}		Co^{58},I^{131},CS^{137}	1	Non 1E	Note 2	506765	SD-20404
6502	Waste Gas Inlet to Carbon Delay Beds	Gamma Scint	15.0	10^{-2} 10^{+2}		Xe^{133}	1	Non 1E	Note 1	506897	20772
6503	Waste Gas Compressor Inlet	Gamma Scint	15.0	10^{-3} 10^{+1}		Kr^{85}	1	"	Note 1	506898	20770
6504	H_2 Gas Compressor Disch.	Gamma Scint	15.0	10^{-3} 10^{+1}		Kr^{85}	1	"	Note 1	506899	20773
6505	Condenser Air Evac	Beta Scint	0.5	Note 3		Xe^{133}	1	"	Note 1	506055	20774
6500	Boron Recovery Stor. Tank Inlet	Gamma Scint	1.0	$10^{-5}$$10^{-1}$		Co^{58},I^{131},CS^{137}	1	"	Note 2	506105	20856
6501	Boron Recovery Test Tank Inlet	Gamma Scint	2.5	10^{-6} 10^{-3}		Co^{58},I^{131},CS^{137}	1	"	Note 2	506113	20861
6515,6516	Primary Component Cooling Water	Gamma Scint	2.5	10^{-7} 10^{-3}		Co^{58},I^{131},CS^{137}	2	"	Note 2	506190, 506194	20211, 20205
6509	Liquid Waste Test Tk Disch to CWS	Gamma Scint	2.5	10^{-6} 10^{-2}		Co^{58},I^{131},CS^{137}	1	"	Note 2	506927	20831
6514	Waste Liquid From Evaporators	Gamma Scint	2.5	10^{-6} 10^{-2}		Co^{58},I^{131},CB^{137}	1	"	Note 2	506931	20831

* See Table 11.5-1 (Sheet 3) for notes.

Instrument Tag No. Re-	Description	Detector Type	Det Back Grd mr/hr	Range Low-High (µci/cc)	(Note 5) Alarm Set Point (µci/cc)	Reference Isotope	Detector Qty	Safety Class	Energy* Level	Loop Diag. 1-NHY	P&Id 1-NHY
6510, 6511, 6512, 6513	Steam Gen Blowdown Sample Loops 1,2,3,4	Gamma Scint	2.5	10^{-6} 10^{-2}		Co^{58},I^{131},CS^{137}	4	"	Note 2	506815	20521
6519	Steam Gen Blowdown Flash Tank Drain	Gamma Scint	2.5	10^{-7} 10^{-3}		Co^{58},I^{131},CS^{137}	1	"	Note 2	506734	20626
6520	Reactor Coolant Gross Activity Monitor	GM	15	10^{-1} 10^{+4} mr/hr		Co^{58},I^{131},CS^{137}		"	Note 2	506269	20722
6481-1, 6482-1 6481-2, 6482-2	Main Steam Line Monitor	Gamma Scint	2.5	10^{-0} 10^{+5}mr/hr		Xe^{133}		"	-	506551 -2, -3, -4	20580 20581
6490	Aux Steam Cond	Gamma Scint	0.5	$10^{-7}$$10^{-3}$		$Co^{58,}$ I^{131}, CS^{137}	1	Non 1E	Note 2	507165	20908
6521	Turb. Bldg. Sump Liq. Monitor	Gamma Scint	2.5	10^{-6} 10^{-2}		Co^{58}, I^{131}, CS^{137}	1	Non 1E	Note 2	506713, 506716	20195
6527A1, A2 B1, B2	COP Monitors	GM	2.5	10^{1} -10^{6}cpm		Xe^{133}	4	1E	Note 1	506211	20504
6560	Resin Sluice Line	GM	100	Note 4		Xe^{133}	1	Non 1E		506694	20252
6561	Resin Transfer Line	GM	100	Note 4		Xe^{133}	1	Non 1E		506694	20735
6564	Sluice Pump Line	GM	100	Note 4		Xe^{133}	1	Non 1E		586692	20252
6473	Water Treatment Liquid Effluent Radiation Monitor	Gamma Scint	2.5	10^{-6} 10^{-2}		Co^{58}, I^{131}, CS^{137}	1	Non 1E	Note 2	506976	20040

Note	Isotopes	Max. Beta Energy (Mev)	Predominant Gamma Energy (Mev)
1	Xe^{133}	0.346	0.081
	Xe^{135}	0.92	0.249
	Kr^{85}	0.67	0.514
	Kr^{85m}	0.82	0.150
2	I^{131}	0.606	0.364
	I^{133}	1.27	0.53
	Cs^{134}	0.662	0.604
	Cs^{137}	0.514	0.662
	Co^{58}	0.474	0.81
	Co^{60}	0.314	1.17, 1.33

3 Condenser Air Evacuation Monitor to have output in counts per min (cpm) (10^1 to 10^6).

4 Monitors 6560, 6561, 6564 have output in mr/hr ($10°$ to 10^5).

5 Radiation monitoring setpoints are varied during operation to follow station operating conditions. Setpoints are maintained within the bounds established in the Technical Specifications. The methodology for establishing the setpoints is found in the Station Offsite Dose Calculation Manual (ODCM) and/or station operating procedures.

TABLE 11.5-2 AUTOMATIC SYSTEM AND OPERATOR RESPONSES TO ANNUNCIATED RADIOACTIVITY LEVEL LIMITS

Monitor	Automatic Response to Radioactivity Level Limit	Operator Response to Radioactivity Level Limit
Waste Gas Processing (6504)	Closure of waste gas discharge valve	Request sampling and laboratory analysis
Condenser Air Evacuation System	No automatic response	Request sampling and laboratory analysis and/or observe steam generator liquid samplers.
Boron Recovery System	No automatic response	Request sampling and laboratory analysis - check operation of system.
Component Cooling Liquid	No automatic response	Request sampling and laboratory analysis
Waste Processing System Liquid	Close valve in effluent line	Terminate discharge of liquid effluents – Effluent request sampling and laboratory analysis of effluent.
Steam Generator	Isolation valve in the	Request sampling and Liquid Samples blowdown flash tank laboratory analysis discharge closes (observe Tech. Spec. 3/4.7.1.4 limit on secondary activity)
Reactor Coolant Gross Activity	No automatic response	Request sampling and laboratory analysis of reactor coolant samples -to detect failed fuel (observe Tech. Spec. 3/4.4.8 limit on reactor coolant activity)
Waste Test Tank Inlet	Close inlet valves	Correct cause of High Radioactivity Level in WL System, sample waste test tank
Turbine Bldg. Sump	Lock-out of sump pump	Request sampling and Liquid operation laboratory analysis (observe Tech. Spec. 3/4.3.3.9 limit on secondary activity)
Containment Online Purge	Close isolation valves on 8" atmospheric purge line	Request sampling and laboratory analysis of containment air and sump water.
Auxiliary Condensate Return	Terminate Condensate return and isolate return Piping	(later)
Water Treatment Liquid Effluent	Close valve in effluent line	Request sampling and laboratory analysis – check operation of system.

TABLE 11.5-3 RADIOLOGICAL SAMPLING MONITORING

Sampling Location	Basis for Selection of Location	Expected Process Flowrate	Sample Composition	Expected Concentrations ($\mu Ci/cm^3$)	Types Effluent Releases
Plant Vent	Determination of identity and quantity of radionuclides being released, calibration check of PVS monitor airborne radioactive	312,185 cfm (max) 272,185 cfm (normal)	Ventilation exhaust air	I-131: 1.18×10^{-11} Xe-133: 2×10^{-7} Cs-137: 1.82×10^{-13}	Continuous releases of fission and activation gases and tritium; releases of airborne radioactive particulates; releases of iodines
Condenser Air Evacuation	Determination of identity and quantity of radionuclides being released; calibration check of the condenser air evacuation monitor	10-20 cfm Condenser in leakage gases and t	Condenser gases	Xe-133: 1.6×10^{-5}	Continuous releases of fission and activation gases and tritium; release of airborne radioactive iodine
Turbine Gland Sealing System Exhaust	Determination of identity and quantity of radionuclides being released	<300 cfm	Condenser gases	I-131: 3×10^{-7}	Continuous releases of fission and activation gases and tritium; release of airborne radioactive iodine
Steam Generator Blowdown Flash Tank Drain Outlet calibration	Determination of identity and quantity of radionuclides released	200 gpm	Steam generator blowdown liquid	Co-58: 1.7×10^{-6} I-131: 3.3×10^{-5} Cs-134: 2.88×10^{-6}	Continuous releases of liquids

TABLE 11.5-3

Sampling Location	Basis for Selection of Location	Expected Process Flowrate	Sample Composition	Expected Concentrations ($\mu Ci/cm^3$)	Types Effluent Releases
Liquid Waste Test Tanks (pump discharge) waste demin. filter outlet and Recovery Test Tank Pump Discharge	Determination of identity and quantity of radio nuclides released; set trip point of discharge monitor	Variable, up to 150 gpm	Processed radwaste	Co-58: $1x10^{-6}$ I-131: $1x10^{-5}$ Cs-134: $4.1x10^{-9}$	Batch releases of liquids
Waste Evaporator Bottoms	Determination of identity and quantity of radio nuclides in concentrate	Variable	Concentrated liquid radwaste	Cs-134: $1x10^{-3}$ Cs-137: $1x10^{-3}$ Co-58: $1x10^{-1}$	Batch releases of concentrates
Hydrogen Surge Tank	Determination of identity and quantity of radionuclides stored in tank	Variable, depending on concentration of radionuclides	H_2, N_2 noble gases	Kr-85: 2.86 Xe-133: $6.97x10^{-2}$	Batch releases of fission product gases
Waste Gas System	Determination of identity and quantity of radio-nuclides upstream and downstream of carbon delay beds and downstream of waste gas compressors; calibration check of waste gas monitors, efficiency check of delay beds	0.4 cfm	H_2, N_2 noble gases	Xe-133: $1.92x10^{-2}$ Kr-85: 2.88 (input to system)	Continuous releases of fission gases and tritium

TABLE 11.5-3

Sampling Location	Basis for Selection of Location	Expected Process Flowrate	Sample Composition	Expected Concentrations (μCi/cm^3)	Types Effluent Releases
Turbine Building Sump	Determination that the sump is free of contamination	-	H_2O	$<1x10^{-9}$ (gross β - γ)	Continuous
Containment Purge	Determination of identity and quantity of radionuclides being released	1000 cfm	Ventilation exhaust air	Xe-133: $3.8x10^{-4}$	Continuous
PAB Ventilation System	Determination of identity and quantity of radionuclides being released	43,200 cfm (clean filter condition) 39,225 cfm (dirty filter conditions)	Ventilation exhaust air	I-131: $2.9x10^{-11}$ Xe-133: $3.9x10^{-7}$ CS-137: $1.2x10^{-12}$	Continuous
Fuel Storage Building	Determination of identity and quantity of radionuclides being released and present in building atmosphere	34,000 cfm	Ventilation exhaust air	$2.7x10^{-6}$ (gross β - γ)	Continuous
Waste Process Building	Determination of identity and quantity of radionuclides in building atmosphere	151,620 cfm	Ventilation exhaust air	I-131: $3.2x10^{-13}$ CS-137: $1.6x10^{-13}$	Continuous
Water Treatment System	Determination of identity and quantity of radionuclides being released	350 gal/day	H_2O	$<1x10^{-9}$ (gross β - γ)	Continuous

TABLE 11.5-3

Sampling Location	Basis for Selection of Location	Expected Process Flowrate	Sample Composition	Expected Concentrations ($\mu Ci/cm^3$)	Types Effluent Releases
Turbine Gland Steam Condenser	Determination of identity and quantity of radionuclides being released	--	Air with high moisture content	I-131: 2.6×10^{-10}	Continuous*
Evaporator System Distillate Cooler Vent	Determination of identity and quantity of radionuclides being processed	----	Evaporator gases	I-131: 5.5×10^{-8} Xe-133: 5.8×10^{-10}	Intermittent
Pressurizer and BRS Vent System	Determination of identity and quantity of radionuclides being released	----	Gases	I-131: 6.1×10^{-4} Xe-133: 1.6	Intermittent
Component Cooling Water System	Determination if cross-contamination with reactor coolant has occurred	8,986 gpm max	Demineralized water	Within the statistical background	None during normal operation
Service Water System	Determination if cross-contamination with reactor coolant has occurred	10,500 gpm per train	Sea water containing sodium chloride	Within the statistical background	Continuous but this system is not normally radioactive

*Only with a (primary to secondary) steam generator tube leak.

Note: Sampling frequencies and sensitivities shall be as specified in the Offsite Dose Calculation Manual (ODCM).

TABLE 11A-1 LIQUID WASTE PROCESSING SYSTEMS – SOURCES AND PROCESSING PARAMETERS

		Decontamination Factors				Holdup Times (days)		
System	**Input Flow Rate (gal. per day)**	**Iodine**	**Cesium, Rubidium**	**Others**	**Fraction of Primary Coolant Activity (PCA)**	**Processing & Collection**	**Discharge**	**Fraction Discharged**
Miscellaneous Waste	1,360	10^3	10^4	10^4	0.061	5.9	0.23	1.0
Equipment Drain	302	10^3	2×10^3	10^4	1.0	23.0	5.03	0.464
Turbine Building Sump Waste and Water Treatment Liquid Effluent that includes Condensate Polishing System	7,550	1.0	1.0	1.0	[a]	0.0	0.0	1.0
Boron Recovery System (Includes Shim Bleed)	878	10^3	2×10^3	10^4	1.0	152.8	5.03	0.560
Steam Generator Blowdown	108,000	10^3	10^4	10^4	-	0.0	0.0	0.7

(a) Activity levels are based on secondary side main steam inventories.

TABLE 11A-2 RADIONUCLIDE DISCHARGE CONCENTRATIONS NORMAL LIQUID RELEASES - INCLUDING ANTICIPATED OPERATIONAL OCCURRENCES

	Total Annual Release (Ci/yr)	Discharge Concentration (μCi/ml)	(MpC)w (μCi/ml)	Fraction of Nuclide (MPC)w
H-3	7.30E+02*	1.1E-06	3E-03	3.7E-04
I-130	1.2E-04	1.8E-13	3E-06	6.1E-08
I-131	1.3E-01	2.0E-10	3E-07	6.6E-04
I-132	3.9E-03	5.9E-12	8E-06	7.4E-07
I-133	3.6E-02	5.5E-11	1E-06	5.5E-05
I-134	1.0E-04	1.5E-13	2E-05	7.5E-09
I-135	5.8E-03	8.8E-12	4E-06	2.2E-06
Br-83	4.0E-05	6.1E-14	1E-07	6.1E-07
Rb-86	2.0E-05	3.1E-14	2E-05	1.5E-09
Sr-89	3.0E-05	4.6E-14	3E-06	1.5E-08
Mo-99	2.1E-03	3.2E-12	4E-05	8.0E-08
Tc-99m	2.4E-03	3.7E-12	3E-03	1.2E-09
Te-127m	3.0E-05	4.6E-14	5E-05	9.2E-10
Te-127	3.0E-05	4.6E-14	2E-04	2.3E-10
Te-129m	1.2E-04	1.8E-13	2E-05	9.2E-09
Te-129	9.0E-05	1.4E-13	8E-04	1.8E-10
Te-131m	3.0E-05	4.6E-14	4E-05	1.1E-09
Te-132	7.3E-04	1.1E-12	2E-05	5.6E-08
Cs-134	2.0E-02	3.0E-11	9E-06	3.4E-06
Cs-136	2.7E-03	4.1E-12	6E-05	6.9E-08
Cs-137	1.5E-02	2.3E-11	2E-05	1.2E-06

	Total Annual Release (Ci/yr)	Discharge Concentration (μCi/ml)	(MpC)w (μCi/ml)	Fraction of Nuclide (MPC)w
Ba-137m	1.4E-02	2.1E-11	1E-07	2.1E-04
Ba-140	1.0E-05	1.5E-14	2E-05	7.6E-10
La-140	1.0E-05	1.5E-14	2E-05	7.6E-10
Cr-51	1.5E-04	2.3E-13	2E-03	1.1E-10
Mn-54	4.0E-05	6.1E-14	1E-04	6.1E-10
Fe-55	1.9E-05	2.9E-14	8E-04	3.6E-11
Fe-59	9.0E-05	1.4E-13	5E-05	2.7E-10
Co-58	1.6E-03	2.4E-12	9E-05	2.7E-08
Co-60	2.3E-04	3.5E-13	3E-05	1.2E-08
Np-239	3.0E-05	4.6E-14	1E-04	4.6E-10
All Others	6.0E-05	9.2E-14	1E-07	9.2E-07
Total (Except Tritium)	2.4E-01	3.6E-10	-	9.3E-04
* 7.30E+02 = 7.30×10^2				

TABLE 11A-3 ANNUAL GASEOUS EFFLUENTS RELEASE (Ci/yr)

Radionuclide	Containment Purge	PAB Venting	Turbine Venting	Main Condenser Off-Gas System	Gaseous Waste System	Total
H-3	3.7E+02*	3.7E+02	c	c	c	7.4E+02
C-14	1.0E+00	a	a	a	7.0E+00	8.0E+00
Ar-41	2.5E+01	a	a	a	a	2.5E+01
Kr-83m	a	a	a	a	a	a
Kr-85m	7.0E+00	2.0E+00	a	1.0E+00	a	1.0E+01
Kr-85	1.0E+00	a	a	a	2.6E+02	2.6E+02
Kr-87	2.0E+00	1.0E+00	a	a	a	3.0E+00
Kr-88	1.0E+01	4.0E+00	a	2.0E+00	a	1.6E+01
Kr-89	a	a	a	a	a	a
Xe-131m	3.0E+00	a	a	a	2.0E+01	2.3E+01
Xe-133m	1.6E+01	a	a	a	a	1.6E+01
Xe-133	8.6E+02	3.4E+01	a	2.1E+01	7.7E+01	9.9E+02
Xe-135m	a	a	a	a	a	a

Radionuclide	Containment Purge	PAB Venting	Turbine Venting	Main Condenser Off-Gas System	Gaseous Waste System	Total
Xe-135	3.1E+01	4.0E+00	a	3.0E+00	a	3.8E+01
Xe-137	a	a	a	a	a	a
Xe-138	a	1.0E+00	a	a	a	1.0E+00

* 3.7E+02 = 3.7×10^2

[a] Less than 1.0 Ci/yr for Noble Gases and C-14

[b] Less than 0.0001 Ci/yr for Iodine

[c] Less than 1.0 percent of total for this nuclide

Radionuclide	Containment Purge	PAB Venting	Turbine Venting	Main Condenser Off-Gas System	Gaseous Waste System	Total
I-130	b	b	b	b	b	b
I-131	1.5E-02	4.3E-03	1.8E-03	2.7E-02	b	4.8E-02
I-132	b	b	b	b	b	b
I-133	1.1E-02	6.3E-03	1.9E-03	4.0E-02	b	5.9E-02
I-134	b	b	b	b	b	b
I-135	b	b	b	b	b	b
Mn-54	2.2E-04	1.8E-04	c	c	4.5E-05	4.4E-04
Fe-59	7.4E-05	6.0E-05	c	c	1.5E-05	1.5E-04
Co-58	7.4E-04	6.0E-04	c	c	1.5E-04	1.5E-03
Co-60	3.3E-04	2.7E-04	c	c	7.0E-05	6.7E-04
Sr-89	1.7E-05	1.3E-05	c	c	3.3E-06	3.3E-05
Sr-90	2.9E-06	2.4E-06	c	c	6.0E-07	5.9E-06
Cs-134	2.2E-04	1.8E-04	c	c	4.5E-05	4.4E-04
CS-137	3.7E-04	3.0E-04	c	c	7.5E-05	7.4E-04

Radionuclide	**Containment Purge**	**PAB Venting**	**Turbine Venting**	**Main Condenser Off-Gas System**	**Gaseous Waste System**	**Total**
[a] Less than 1.0 Ci/yr for Noble Gases and C-14 [b] Less than 0.0001 Ci/yr for Iodine [c] Less than 1.0 percent of total for this nuclide						

TABLE 11A-4 VENT RELEASE INFORMATION FOR GASEOUS RELEASES

	PLANT VENT*	**TURBINE BUILDING ROOF VENTS (10)**	**T.B. HEATER BAY ROOF VENTS (10)**
Height above grade (ft)	185	151	100
Height above adjacent structures	5.5	0	0
Exit temperature (°F) Summer/Winter	104°/50°	145°/100°	145°/100°
Exit flow rate (cfm)(max)	312,185	50,000/40,000	
Exit velocity (ft/min)	1,950	1,470	1,290
Vent size and shape	9'x12' diameter stack open to the environment(exit gas is deflected down towards the roof)	10 mushroom type vents 5.5.ft diameter (exit gas is deflected down toward the roof)	10 mushroom type vents5 ft diameter (exit gas is deflected down toward the roof)
Deflectors or diffusers:	No	Yes	Yes

* Two samples are drawn from the plant vent stack; one is routed through the wide range radiation detectors and the other through a portable air sampler and released to the atmosphere at the PAB roof. Since these two samples are from a monitored release path and their flow rates are negligible (L3 scfm each), these release points are not separately described.

See PID-1-WL-B20828

See PID-1-WL-B20829

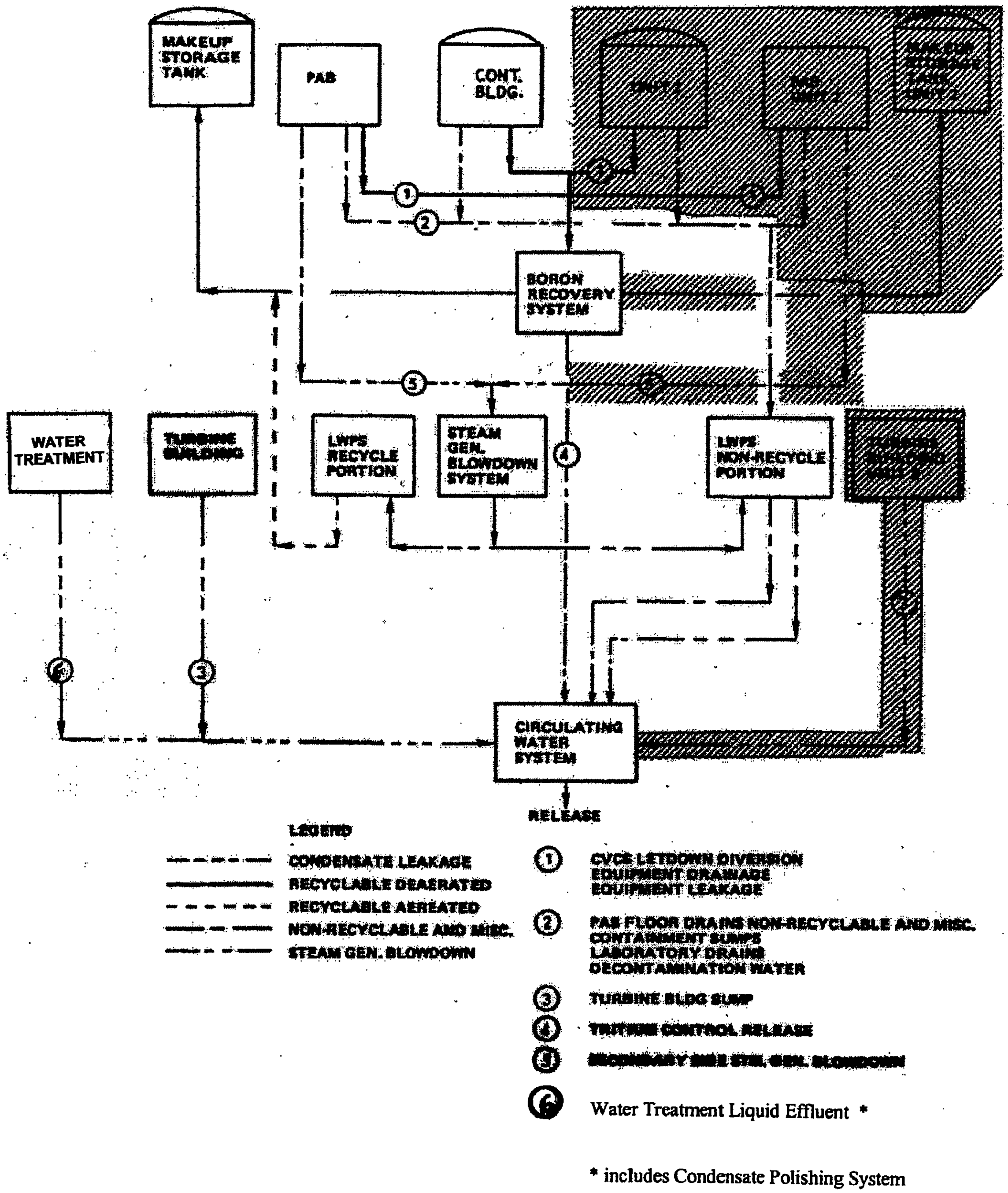
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Liquid Waste System Storage and Filtration Detail	
		Figure 11.2-2

See PID-1-WL-B20830

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Liquid Waste System Evaporator EV-4 Detail	
		Figure 11.2-3

See PID-1-WL-B20831

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Liquid Waste System Demineralization and Testing Detail	
		Figure 11.2-4



SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Radioactive Liquid Release Points	
		Figure 11-2-5

See PID-1-WG-B20768

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Waste Gas System Overview	
		Figure 11.3-1

See PID-1-WG-B20770

See PID-1-WG-B20771

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Waste Gas System Detail [4 Sheets]	
		Figure 11.3-2 Sh. 2 of 4

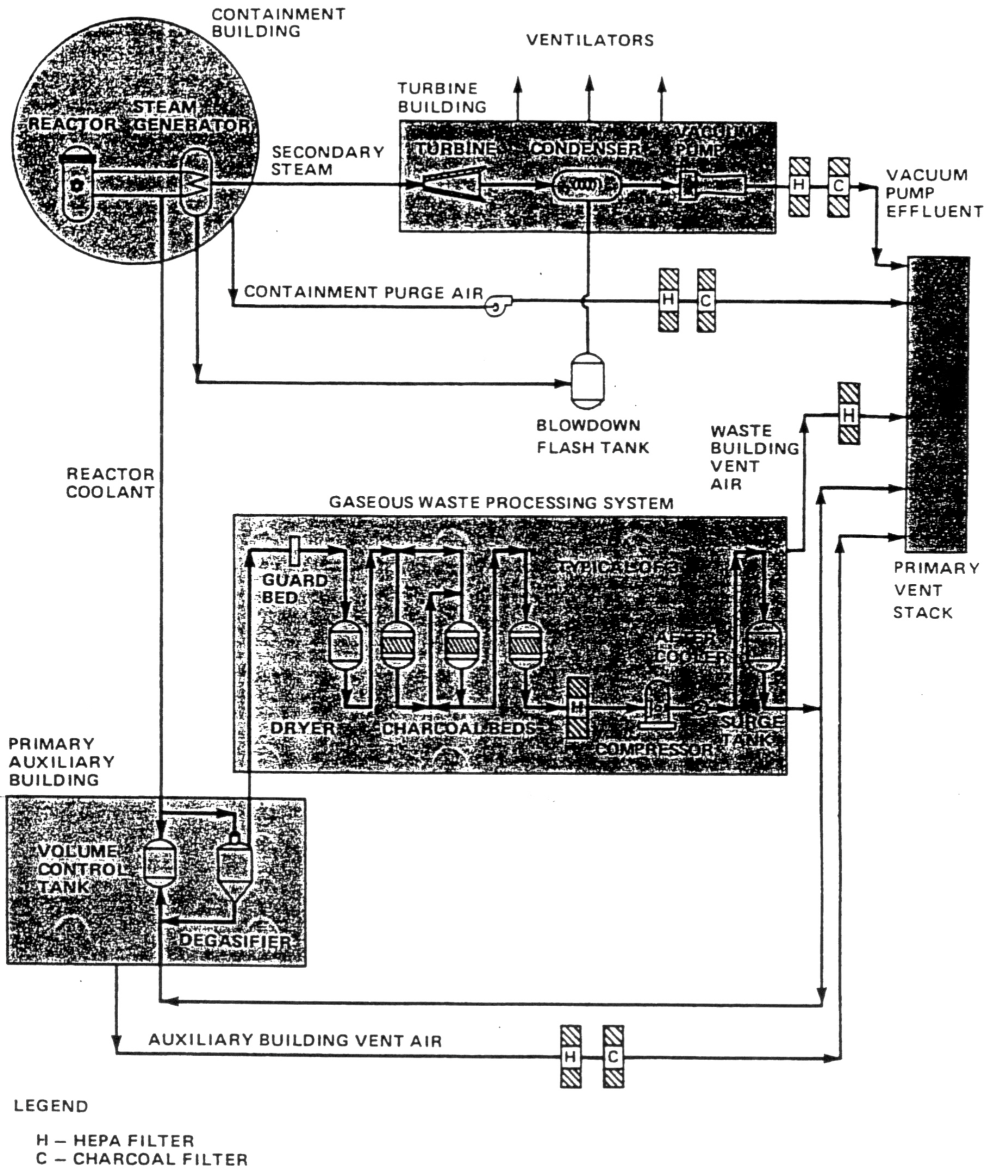
See PID-1-WG-B20772

See PID-1-WG-B20773

See PID-1-NG-B20132

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Nitrogen Gas Overview	
		Figure 11.3-3

See PID-1-NG-B20135



SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Sources of Gaseous Waste	
		Figure 11-3-5

See PID-1-WS-B20733

See PID-1-WS-B20734

See PID-1-WS-B20735

Solid Waste System Spent Resin Concentrates Detail

Figure 11.4-2

See PID-1-WS-B20736

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Solid Waste System Waste Feed and Bottom Detail	
		Figure 11.4-3

See PID-1-WS-B20737

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Solid Waste System Asphalt and Steam Detail	
		Figure 11.4-4

See PID-1-WS-B20738

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Solid Waste System Extruder Detail	
		Figure 11.4-5

See PID-1-WS-B20739

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Solid Waste System Crystallizer Detail	
		Figure 11.4-6

See PID-1-WS-B20740

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Solid Waste System Crystallizer Condenser Utilities Detail	
		Figure 11.4-7

See PID-1-WS-B20741

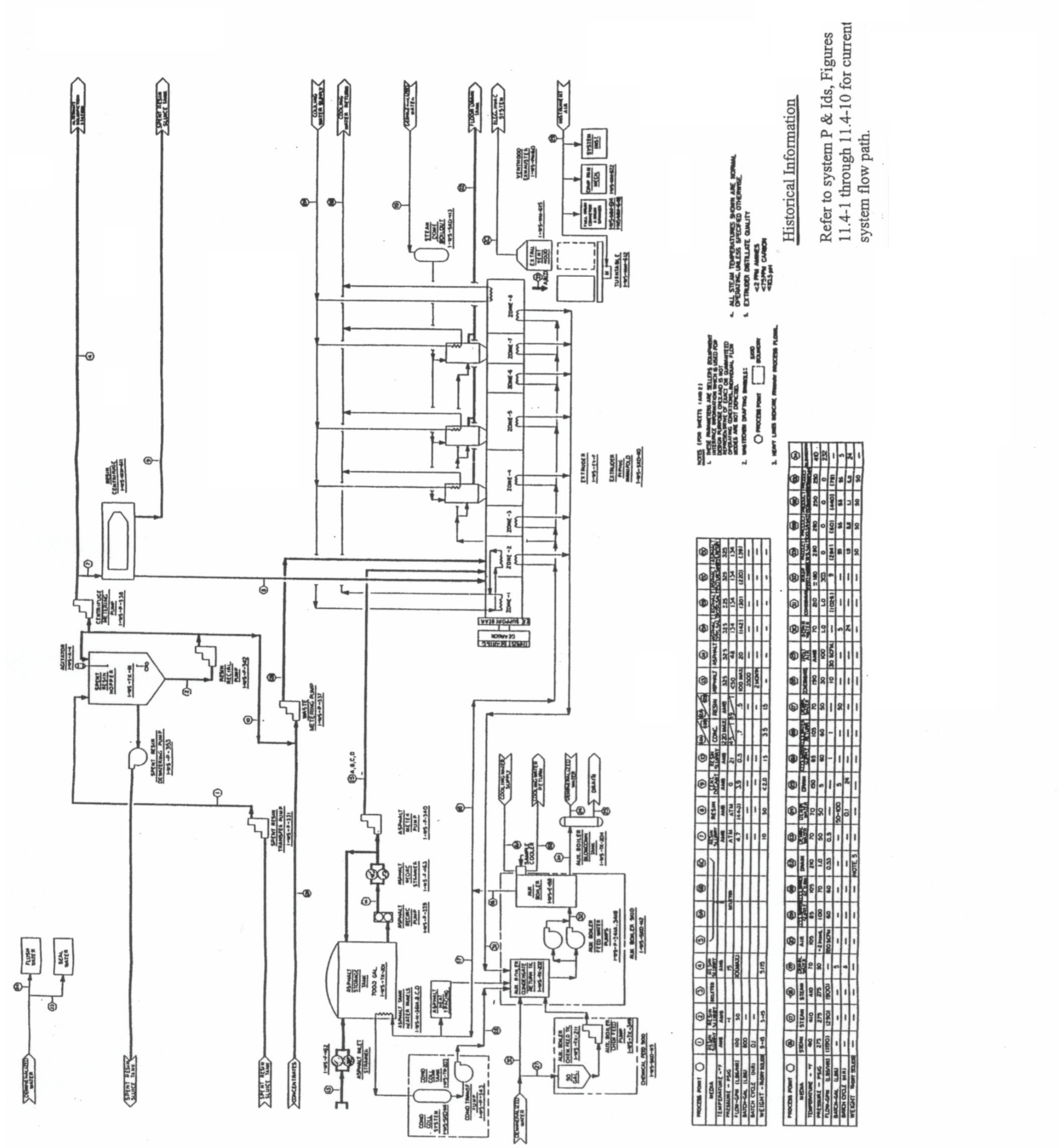
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Solid Waste System Caustic and Material Handling Detail	
		Figure 11.4-8

See PID-1-WS-B20742

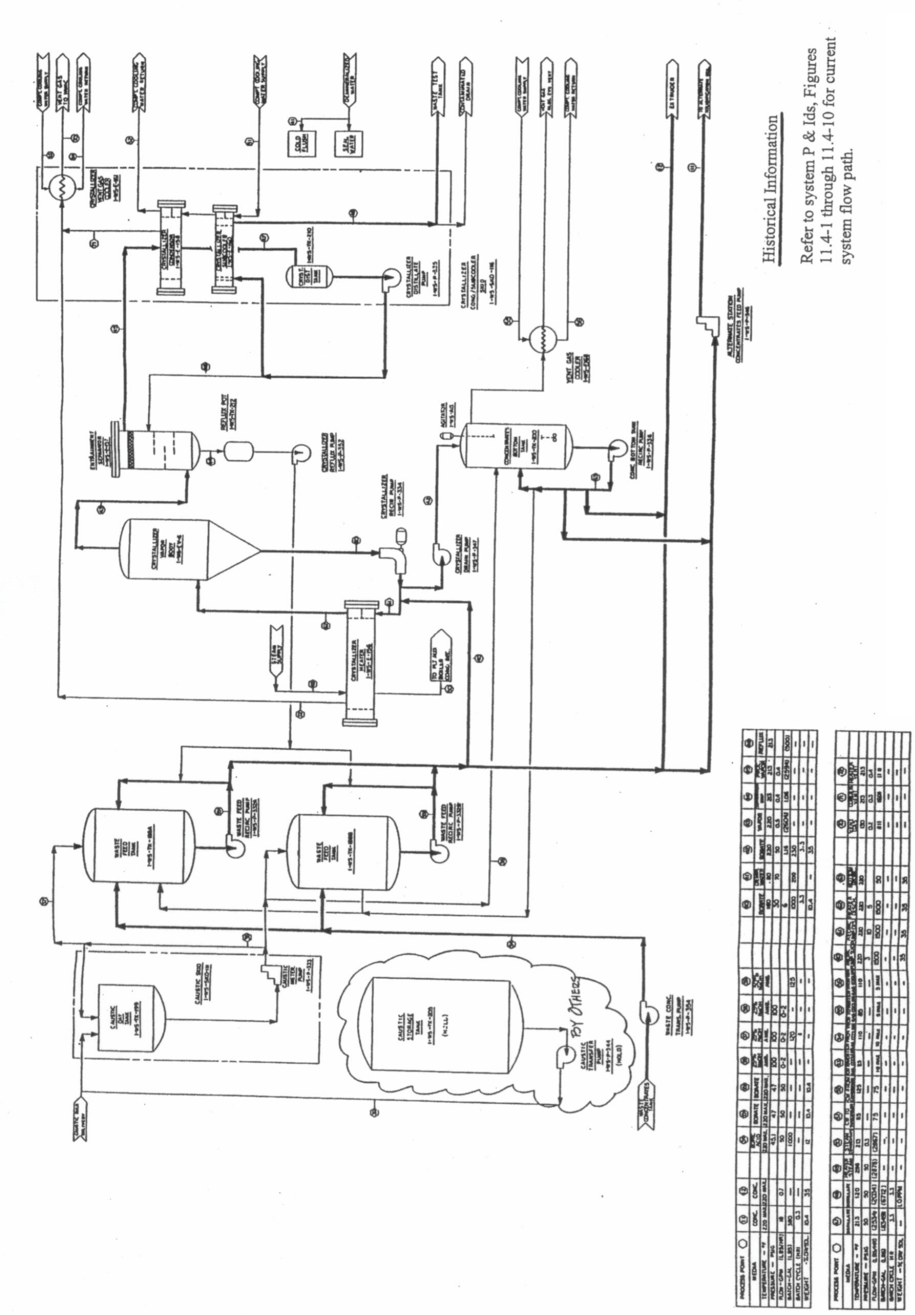
SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Solid Waste System Pump Seal Water Detail	
		Figure 11.4-9

See PID-1-RS-B20252

SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Spent Resin Sluicing System	
		Figure 11.4-10

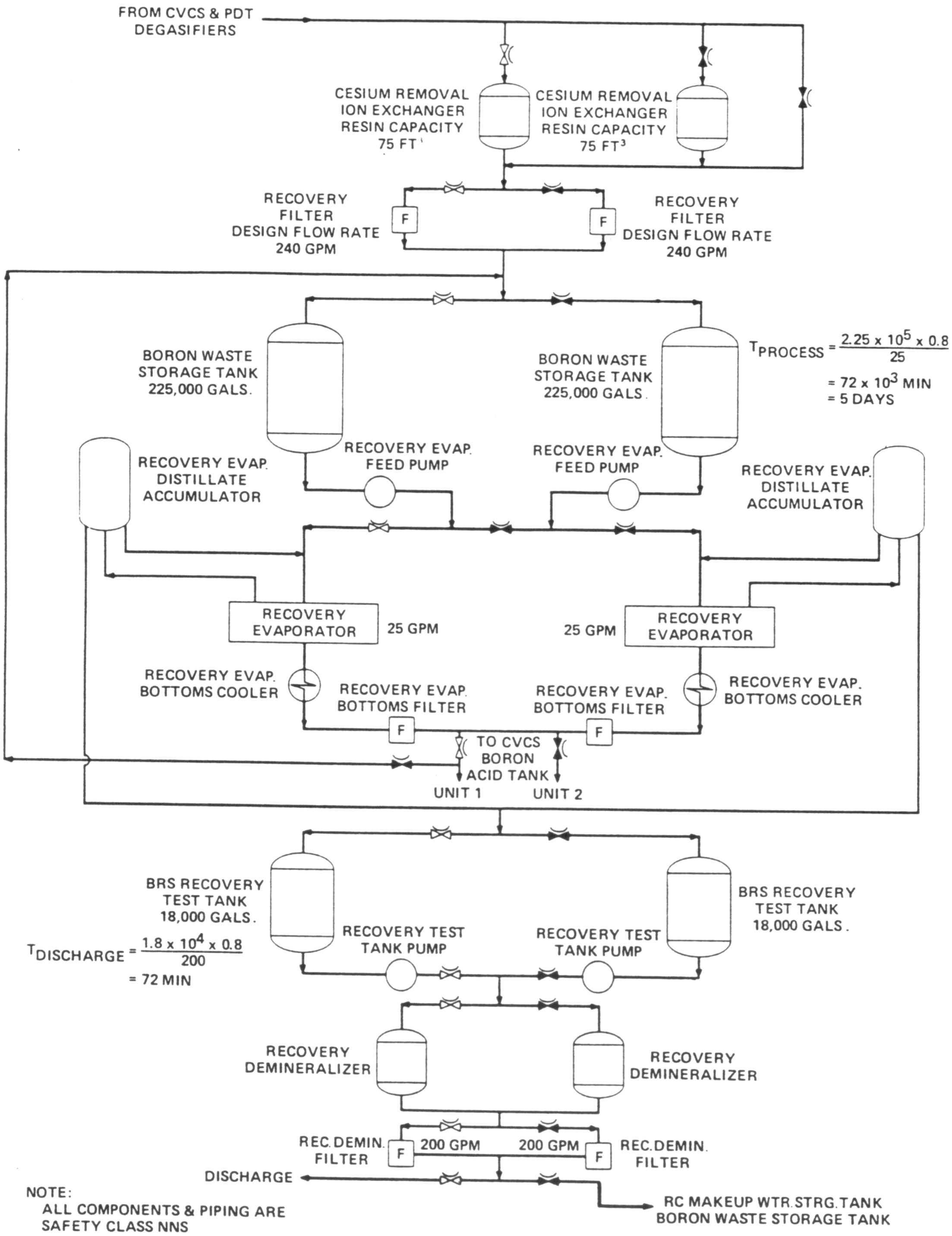


SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Solid Waste System Process Flow Diagram	
		Figure 11-4-11 Sh. 1 of 2

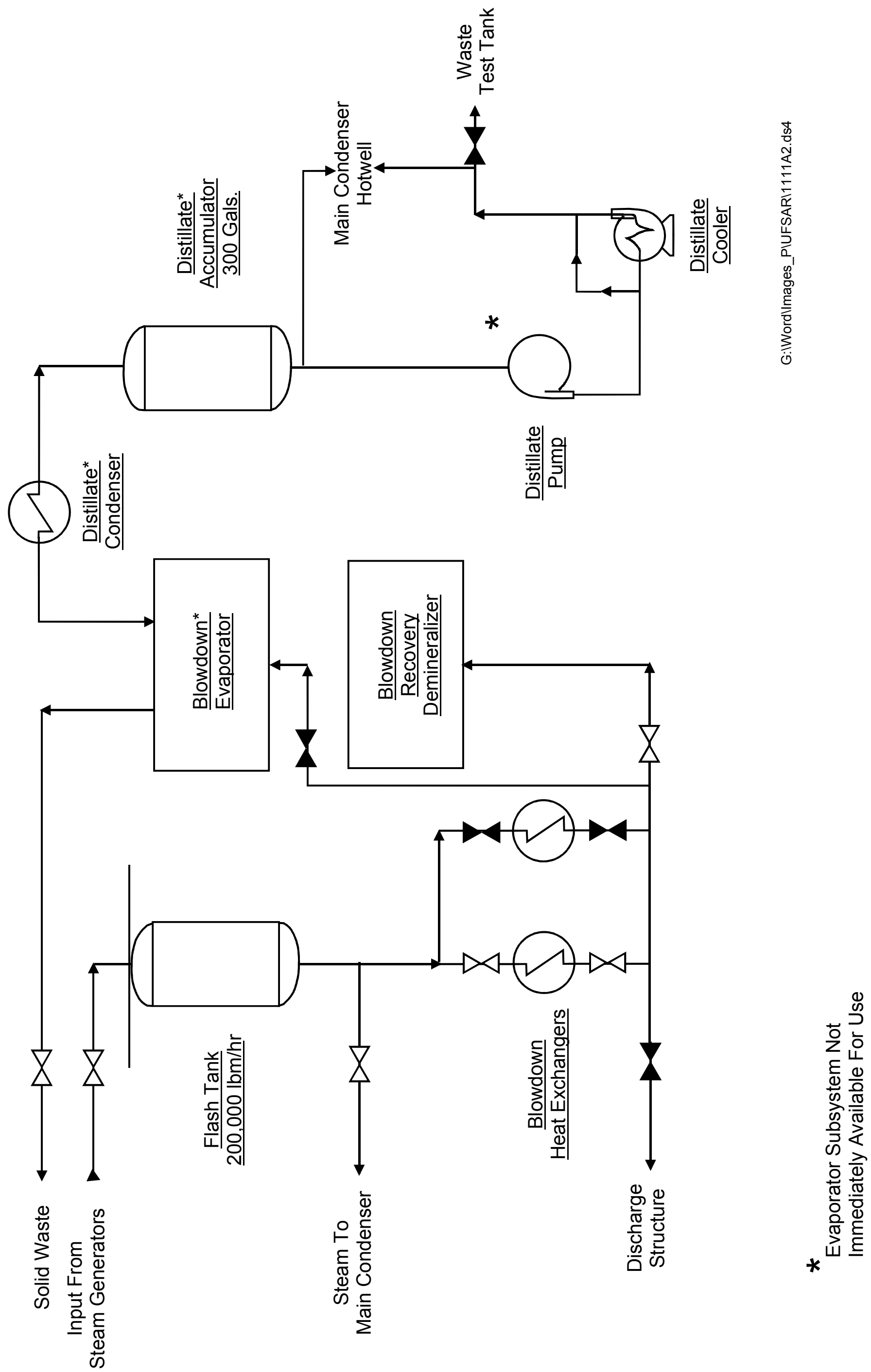


SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Solid Waste System Process Flow Diagram
	Figure 11-4-11 Sh. 2 of 2

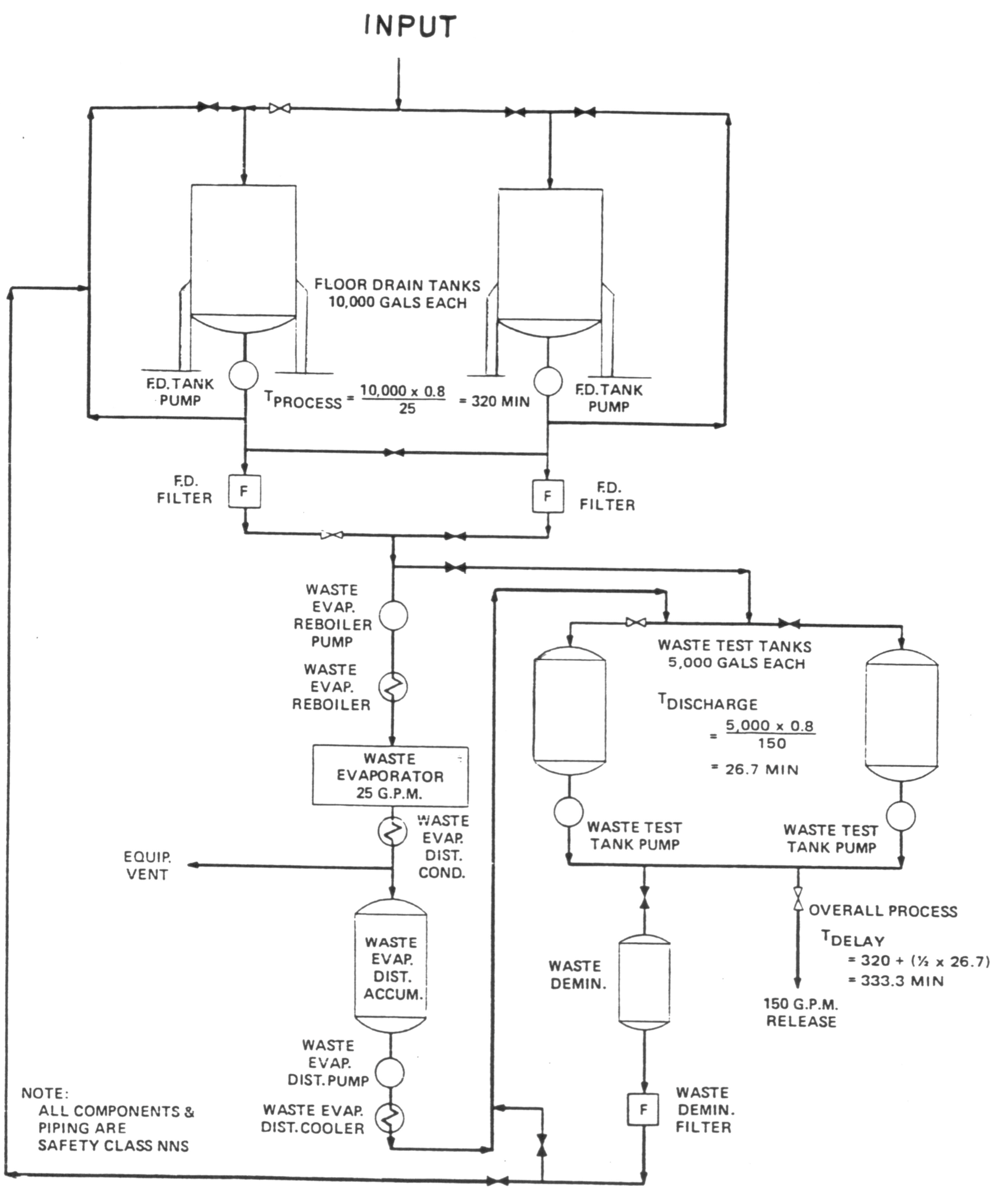
See 1-NHY-500015

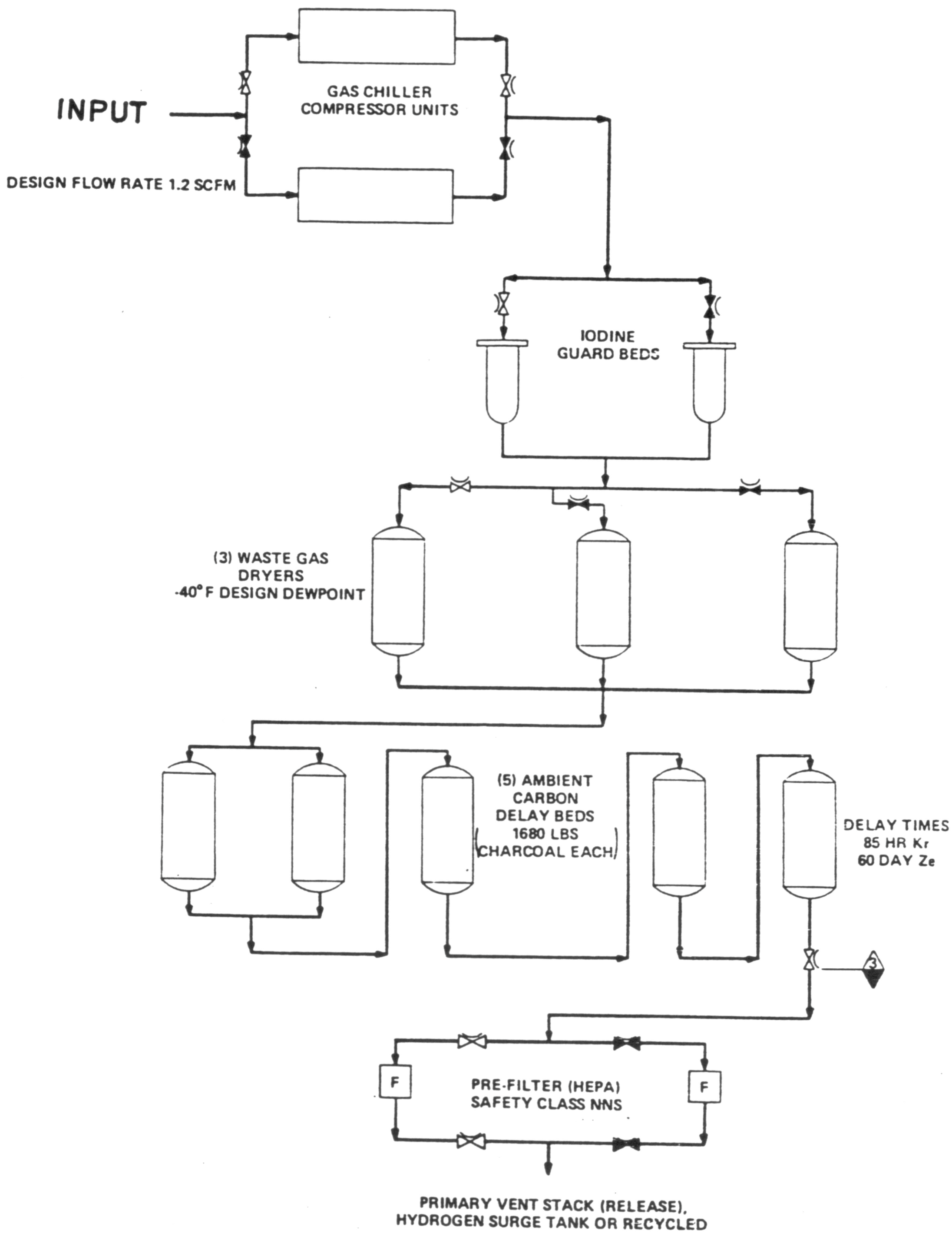


SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Boron Recovery System Schematic Figure 11A-1



SEABROOK STATION UPDATED FINAL SAFETY ANALYSIS REPORT	Steam Generator Blowdown System Schematic	
		Figure 11A-2


INPUT
FLOOR DRAIN TANKS
10,000 GALS EACH
F.D. TANK
PUMP
TPROCESS = 10,000 x 0.8 / 25 = 320 MIN
F.D. TANK
PUMP
F.D.
FILTER
F
F
F.D.
FILTER
WASTE
EVAP.
REBOILER
PUMP
WASTE
EVAP.
REBOILER
WASTE
EVAPORATOR
25 G.P.M.
WASTE
EVAP.
DIST.
COND.
EQUIP.
VENT
WASTE
EVAP.
DIST.
ACCUM.
WASTE
EVAP.
DIST. PUMP
WASTE EVAP.
DIST. COOLER
NOTE:
ALL COMPONENTS &
PIPING ARE
SAFETY CLASS NNS
WASTE TEST TANKS
5,000 GALS EACH
TDISCHARGE
= 5,000 x 0.8 / 150
= 26.7 MIN
WASTE TEST
TANK PUMP
WASTE TEST
TANK PUMP
OVERALL PROCESS
TDELAY
= 320 + (½ x 26.7)
= 333.3 MIN
150 G.P.M.
RELEASE
WASTE
DEMIN.
F
WASTE
DEMIN.
FILTER


INPUT
DESIGN FLOW RATE 1.2 SCFM
GAS CHILLER
COMPRESSOR UNITS
IODINE
GUARD BEDS
(3) WASTE GAS
DRYERS
-40° F DESIGN DEWPOINT
(5) AMBIENT
CARBON
DELAY BEDS
1680 LBS
(CHARCOAL EACH)
DELAY TIMES
85 HR Kr
60 DAY Ze
3
F
PRE-FILTER (HEPA)
SAFETY CLASS NNS
F
PRIMARY VENT STACK (RELEASE),
HYDROGEN SURGE TANK OR RECYCLED

APPENDIX 11A DATA BASE FOR SEABROOK 10 CFR 50, APPENDIX I (REALISTIC) SOURCE TERM

The analysis and parameters described in this section are historical and are the basis of the facility's 10 CFR 50 Appendix I analysis, utilizing the assumptions and methodology of NUREG-0017 and a core power level of 3654 MWt. For the licensed core power level, the core power level for analysis purposes is 3659 MWt. Since 3659 MWt represents only an approximate 0.1% increase above 3654 MWt, use of the core power level of 3659 MWt in the Appendix I analysis discussed herein will have an insignificant impact on radiological releases.

Continued compliance with the annual dose limits to an individual in an unrestricted area set by 10 CFR 50 Appendix I and 40 CFR 190, resulting from gaseous and liquid effluents released to the environment following operation at the licensed core power level was also demonstrated by utilizing five (5) years (1998-2002) of effluent/dose impact data and using scaling factors to estimate the impact of operation at the licensed core power level.

It is noted that for an operating plant, the actual performance and operation of installed equipment, the reporting of actual offsite releases and doses, and compliance with the regulatory limits of 10 CFR 50 Appendix I, 10 CFR 20, and 40 CFR 190, are controlled by the Offsite Dose Calculation Manual.

The following information is presented to comply with Appendix B of Regulatory Guide 1.112. Ventilation system flow rates used in the gaseous dose analysis are those values in effect at the time of the analysis.

The analytical methods described in NUREG-0017 are extensively used in the source term calculation. Radioactive concentrations in the Primary and Secondary Coolant Systems are evaluated on the basis of a pressurized water reactor (PWR) with recirculating U-tube steam generators. Volatile treatment is applied to control secondary system chemistry. A more detailed description is presented in Section 11.1 of the Updated FSAR.

1. General

a. The maximum core thermal power (MWt) evaluated for safety considerations in the Updated FSAR. (Note: All of the following responses are adjusted to this power level.)

Response 1a. Thermal power is 3654 MWt

b. The quantity of tritium released in liquid and gaseous effluents (Ci per year).

Response 1b. A total tritium release of 0.4 curies per MWt per year is recommended in NUREG-0017 for a PWR with moderate tritium control. Accordingly, 1462 curies of tritium are expected to be released per year. One-half of the total release is assumed to be through the liquid pathway and one-half through the gaseous pathway.

2. <u>Primary System</u>

a. The total mass (lb.) of coolant in the primary system, excluding the pressurizer and primary coolant purification system, at full power.

Response 2a. Total primary coolant mass is $5.05x10^5$ lbs.

b. The average primary system letdown rate (gal/min) to the primary coolant purification system.

Response 2b. 80 gal per min ($4.01x10^4$ lbs/hr)

c. The average flow rate (gal/min) through the primary coolant purification system cation demineralizers. (Note: The letdown rate should include the fraction of time the cation demineralizers are in service.)

Response 2c. Letdown flow through the primary coolant purification system cation demineralizers is used intermittently only when additional purification of the reactor coolant is required. No credit for cation demineralizer cleanup is assumed in the source term calculations.

d. The average shim bleed flow (gal/min)

Response 2d. The shim bleed and other clean recyclable waste (e.g., Primary Drain System) are processed through the Boron Recovery System. A detailed description and operational procedure are presented in Section 3.5 of the Seabrook Station Environmental Report and Subsection 9.3.5 of the Updated FSAR. A schematic flow diagram is shown in Figure 11A-1. The shim bleed is diverted from the normal chemical and volume control flow path (purification letdown) after the stream has been degasified. It has two components (computed on an annualized average) which are:

1) Reactor coolant diverted for boron recovery in the amount of 116 lb/hr (0.23 gpm).

2) Reactor coolant diverted for tritium control in the amount of 194 lb/hr (0.38 gpm).

The shim bleed is then routed to the cesium removal ion exchangers of the Boron Recovery System where it is treated through filtration and evaporation. Provisions for demineralization are included as shown in Figure 11A-1. However, this is an optional pathway used for the recycle mode of operation and as such is not included when calculating plant releases. Equipment leakages and valve stem leak-offs are collected through the primary drain tank in an estimated amount of 300 gpd (0.21 gpm). The primary drain tank inventory is processed through the primary drain tank degasifier and routed to the Boron Recovery System, joining with the shim bleed.

The radioactivity level for shim bleed and primary leakages is the same as the reactor coolant. Flow patterns for these sources are intermittent in nature. A combined flow rate of 4.16×10^2 lbs/hr (0.82 gpm) is estimated on an annual average basis.

To control the tritium level within the Primary Coolant System, 200,000 gallons of reactor coolant is expected to be discharged annually through the Boron Recovery System. Therefore, the release fraction amounts to 46 percent of reactor coolant processed through the BRS annually.

System decontamination factors (DF) are conservatively assumed to be 10^3 for iodines and 10^4 for other nuclides due to evaporation and demineralization.

Holdup time is calculated to be a minimum of 5 days on the basis of the capacities of two boron waste storage tanks (225,000 gals each) and two recovery test tanks (18,000 gals each).

3. **Secondary System**

a. The number and type of steam generators, the type of chemistry used and the carry-over factor used in the evaluation for iodine and nonvolatiles.

Response 3a. Four vertical, recirculating, inverted U-tube steam generators per unit. Volatile chemistry will be used to control secondary side chemistry.

Carry over factors: 1 percent for iodines

0.1 percent for nonvolatiles

b. The total steam flow (lbs/hr) in the secondary system.

Response 3b. 1.514×10^7 lbs/hr.

c. The mass of liquid in each steam generator (lb.) at full power.

Response 3c. 9.55×10^4 lbs.

d. The primary-to-secondary leakage rate (lb/day) used in the evaluation.

Response 3d. 100 lb. per day

e. Description of the Steam Generator Blowdown and Blowdown Purification Systems. The average steam generator blowdown rate (lb/hr) is used in the evaluation.

Response 3e. When primary-to-secondary leakage as described in response 3d exists, the primary method to process radioactive secondary liquid from the steam generators is to direct steam blowdown flash tank bottoms cooler discharge to the floor drain tanks. If no secondary pressure is available, the steam blowdown and wet lay-up pumps can be used. From the floor drain tanks, processing through the installed vendor system (WL-SKD-135) to the waste test tanks is the preferred method (reference Subsection 11.2.2.1). The DF assumed for all radionuclides by this method of treatment is 100. Approximately 30 percent of the blowdown volume flashes to steam in the flash tank. This steam is vented to the number 3 feedwater heater, or main condenser if they are available. With the number 3 feedwater heater or main condenser not available, steam from the flash tank will be directed to the flash tank condenser/cooler and then pumped to the waste test tanks in the Liquid Waste System (see Section 11.2). Further processing by the Liquid Waste System is available, if required, prior to release to the environment via the plant Service and Circulating Water System.

Average steam generator blowdown rate of 75 gpm ($3.75x10^4$ lbs/hr) is assumed for the analysis. The Steam Generator Blowdown System is described in 10.4.8 of the Updated FSAR. A schematic flow diagram of the Steam Generator Blowdown System is shown in Figure 11A-2.

f. The fraction of the steam generator feedwater processed through the condensate demineralizers and the decontamination factors used in the evaluation for the Condensate Demineralizer System.

Response 3f. Not applicable; Seabrook does not utilize a Condensate Demineralizer System.

g. Condensate demineralizers:

(1) Average flow rate (lb/hr)

(2) Demineralizer type (deep bed or powdered resin)

(3) Number and size (ft^3) of demineralizers

(4) Regeneration or replacement frequency

(5) Indicate whether ultrasonic resin cleaning is used and the waste liquid volume associated with its use

(6) Regenerant (backwash) volume (gal/event) and activity

Response 3g. Not applicable; Seabrook does not utilize a Condensate Demineralizer System.

4. Liquid Waste Processing Systems

a. For each Liquid Waste Processing System (including the shim bleed, steam generator blowdown, and detergent waste processing systems), provide the following information in tabular form:

(1) Sources, flow rates (gal/day), and expected activities (fraction of primary coolant activity (PCA) for all inputs to each system).

(2) Capacities of all tanks (gal) and processing equipment (gal/day) considered in calculating holdup times.

(3) Decontamination factors for each processing step.

(4) The fraction of each processing stream expected to be discharged over the life of the plant.

(5) For demineralizer regeneration, the time between regenerations, regenerant volumes and activities, treatment of regenerants, and the fraction of regenerant discharged. Include parameters used in making these determinations.

(6) Liquid source term by radionuclide (in Ci/yr) for normal operation, including anticipated operational occurrences.

Response 4a. Nonrecyclable dirty wastes are collected and processed through the Liquid Waste System. A schematic flow diagram of the Liquid Waste System is shown in Figure 11A-3. Sources are determined according to NUREG-0017.

They are:

		Fraction of Primary Coolant
Source	Flow Rate	Activity
Containment Building	40 gpd	1.0
Auxiliary Building Floor Drain	200 gpd	0.1
Laboratory Drains	400 gpd	0.002
Sampling Drains	15 gpd	1.0
Miscellaneous Sources	700 gpd	0.01

These sources and additional sources of liquid radwaste and processing parameters are presented in Table 11A-1.

Liquid source terms by radionuclide (in Ci/yr) for normal operation, including anticipated operational occurrences, are presented in Table 11A-2.

b. Provide piping and instrumentation diagrams and process flow diagrams for the Liquid Radwaste Systems and for all other systems influencing the source term calculations.

Response 4b. Piping and instrumentation diagrams and process flow diagrams for the Liquid Radwaste Systems are provided in Figure 11.2-1 and Figure 11.2-2 of the Updated FSAR.

5. Gaseous Waste Processing System

a. The volume (ft^3/yr) of gases stripped from the primary coolant.

Response 5a. 1.68×10^5 ft^3/yr.

b. A description of the process used to hold up gases stripped from the primary system during normal operations and reactor shutdown. If pressurized storage tanks are used, include a process flow diagram of the system indicating the capacities (ft^3), number, and design and operating storage pressures of the storage tanks.

Response 5b. Not applicable. The Seabrook Gaseous Waste Processing System is described under Item 5e below.

c. A description of the normal operation of the system, e.g., the number of tanks held in reserve for back-to-back shutdown, fill time for tanks. Indicate the minimum holdup time used in the evaluation and the basis for this number.

Response 5c. Not applicable. See Item 5e below.

d. If HEPA filters are used downstream of the pressurized storage tanks, the decontamination factor used in the evaluation.

Response 5d. Not applicable. See Item 5e below.

e. If a charcoal delay system is used, a description of this system indicating the minimum holdup times for each radionuclide considered in the evaluation. List all parameters including mass of charcoal (lb.), flow rate (ft^3/min), operating and dew point temperatures, and dynamic adsorption coefficients for Xe and Kr used in calculating holdup times.

Response 5e. Fission gases from the primary coolant are stripped through the letdown degasifier. Average letdown flow of 80 gpm (4.01×10^4 lbs/hr) is processed through the degasifier with a gas stripping fraction of 1. In addition to the above continuous process, two volumes of primary coolant are assumed to be degassed during cold shutdown. The reactor will operate in a base-load mode. Detailed descriptions are presented in Section 11.3 of the Updated FSAR.

Stripped gases from the primary coolant are processed through the Gaseous Waste Processing System (GWPS) during normal operation and shutdown. A detailed description of the GWPS and operational procedures are given in Section 3.5 of the Seabrook Station Environmental Report and Section 11.3 of the Updated FSAR. A schematic flow diagram is shown in Figure 11A-4. The GWPS consists of chillers, compressors, iodine guard beds, dryers, ambient carbon delay beds and filters. The Ambient Carbon Delay System includes five charcoal delay beds with 1680 lbs. of charcoal in each bed. Design flow rate through the adsorbers is 1.2 scfm. Normal expected flow is 0.8 scfm.

The minimum holdup time used for evaluation/dynamic adsorption coefficients:

Krypton isotopes: 85 hours/45.4 cc per gm. atm.

Xenon isotopes: 60 days/772.5 cc per gm. atm.

Operating and dew point temperatures are ambient (70°F) and -40°F, respectively.

f. Piping and instrumentation diagrams and process flow diagrams for the Gaseous Radwaste Systems and for other systems influencing the source term calculations.

Response 5f. Piping and instrumentation diagrams and process flow diagrams for the Gaseous Radwaste Systems are provided in Figure 11.3-1, Figure 11.3-2, Figure 11.3-3 and Figure 11.3-4 of the Updated FSAR.

6. Ventilation and Exhaust Systems

For each building that houses a steam generator blowdown system vent exhaust, a gaseous waste processing system vent, a main condenser air removal system, or a system that contains radioactive materials, provide the following:

a. Provisions to reduce radioactivity releases through the ventilation or exhaust systems.

Response 6a.1 Primary Auxiliary Building

The primary coolant leak rate to the Auxiliary Building is 160 lb/day. The temperature of the primary coolant in the letdown line as it enters the Auxiliary Building is 290°F. Release of 0.75 percent of the iodine is assumed.

The Auxiliary Building ventilation system pipes all air from potentially contaminated areas through charcoal filters at a flow rate of 36,000 cfm.

Response 6a.2 Waste Processing Building

The Waste Processing Building exhaust air is filtered by HEPA filters prior to release to the environment via the plant vent. No significant releases are anticipated from this building however, and it is not included as a source of gaseous release. Provisions are included in the Waste Processing Building ventilation system for the inclusion of carbon filters, if operational experience and releases indicate that they are required.

Response 6a.3 Turbine Building and Turbine Building Heater Bay Roof Vents

Turbine Building exhaust air is vented directly to the atmosphere, unfiltered, through roof vents.

Response 6a.4 Main Condenser Off-Gas System

Effluent from the main condenser during the normal mode of operation (holding mode) is routed through the Primary Auxiliary Building filter system which contains carbon filters to reduce potential iodine releases. Main condenser effluent during startup operations (hogging mode) is released directly to the atmosphere via the Turbine Building vents.

b. Decontamination factors assumed and the bases (include charcoal adsorbers, HEPA filters, and mechanical devices).

Response 6b. DF of 10 for iodine removal by charcoal adsorbers. DF of 100 for particulate removal by HEPA filtration.

Bases: NUREG-0017

c. Release rates for radioiodine, noble gases, and radioactive particulates and their bases.

Response 6c. See Table 11A-3

Bases: NUREG-0017 and PWR-Gale code:

d. Description of the release points, including height above grade, height above and location relative to adjacent structures, expected average temperature difference between gaseous effluents and ambient air, flow rate, exit velocity, and size and shape of flow orifice.

Response 6d. See Table 11A-4

e. For the Containment Building, the building free volume (ft^3) and a thorough description of the internal recirculation system (if provided), including the recirculation rate, charcoal bed depth, operating time assumed, and mixing efficiency. Indicate the expected purge and venting frequencies and duration and the continuous purge rate (if used).

Response 6e The containment free air volume used for the analysis is 2.704×10^6 ft^3.

The atmosphere inside the Containment is circulated through charcoal filters with a 4" bed depth and 90 percent efficiency for 16 hours prior to personnel entry or purge. A mixing efficiency of 70 percent is used. The recirculation flow is 4,000 cfm. A detailed description of the Containment internal recirculation system is given in Updated FSAR Subsection 9.4.5.

Experience with operating PWRs indicates a purge frequency of 4/year, during shutdown for a duration of 24 hours per purge. The purge flow of 15,000 cfm is filtered through 4" deep charcoal filter beds with iodine removal efficiency of 90 percent. Primary coolant leakage is assumed to be 1 percent per day for noble gases and 0.001 percent per day for iodine. An online purge system is available for use during power operation. The continuous purge rate used to evaluate plant releases is 1000 scfm, and is filtered through 4" deep charcoal filter beds with iodine removal efficiency of 90 percent.