Millstone Power Station Unit 3
Safety Analysis Report

Chapter 11: Radioactive Waste Management

CHAPTER 11—RADIOACTIVE WASTE MANAGEMENT

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11.0 BACKGROUND

An estimate of the radioactive effluents and public dose is provided that documents projected public dose consequences are within 10 CFR 50 Appendix I radioactive release criteria. The estimates were based on nominal assumptions and generic models and based on plant operations at a core power level of 3636 MWt. These dose estimates were developed in support of the original license and updated during MPS-3 restart. These dose estimates are historical and not subject to future update. This information is retained to avoid loss of original design basis.

The Radiological Effluent Monitoring and Offsite Dose Calculation Manual (REMODCM) provide guidance requirements for system operation, dose calculations, and monitoring requirements, and to ensure compliance with effluent limits. Actual measured concentrations of radioactivity released and real time dilution or dispersion estimates are required to verify compliance with effluent limits. Therefore, it is operation within the requirements of the REMODCM that ensures compliance with effluent limits, rather than operation to the nominal assumptions in this chapter.

The Annual Radioactive Effluent Release Report and Annual Radiological Environmental Operating Report, which are required to be submitted in May of each year, document that operation of the plant complies with effluent limits and appropriate regulations. Technical Specifications requires implementation of the REMODCM and Radiation Environmental Monitoring Program.

11.1 SOURCE TERMS

The source of all radioactivity occurring in the process streams of the various radioactive systems is the radionuclides generated in the reactor core and neutron activation of nuclides in the reactor coolant system (RCS) and the air surrounding the reactor vessel.

Radioactive liquid and gaseous releases are described in Sections 11.2 and 11.3, respectively. All reduction factors and decontamination factors associated with these systems are discussed in the appropriate sections.

11.1.1 RADIONUCLIDE INVENTORY IN THE CORE

The discussion on core inventory presented below represents information used by the original license application to establish plant shielding and radwaste effluent assessments. It is retained for historical purposes to avoid loss of original design basis and is not subject to future update. The core inventory used for accident analyses is presented in Chapter 15.

The specific activity of fission products in the core is calculated using the computer program ACTIVITY2. This program calculates the contribution from "parent," "daughter," and "granddaughter" isotopes by solving the following differential equations:

1. First order nuclides:

$$\frac{dN_{ci}(t)}{dt} = F\alpha_i - (\lambda_i + h\gamma_i + \beta_i)N_{ci}(t) \qquad (11.1\text{-}1)$$

2. Second order nuclides:

$$\frac{dN_{cj}(t)}{dt} = F\alpha_j + (\lambda_i\ f_{ij}\ N_{ci}\ (t)) - (\lambda_j + h\gamma_j + \beta_j)N_{cj}(t) \qquad (11.1\text{-}2)$$

3. Third order nuclides:

$$\frac{dN_{ck}(t)}{dt} = F\alpha_k + \lambda_i\ f_{ik}\ N_{ci}\ (t) + \lambda_j\ f_{jk}\ N_{cj}\ (t) - (\lambda_k + h\gamma_k + \beta_k)N_{ck}(t) \qquad (11.1\text{-}3)$$

where:

- i, j, k = indicate first, second, and third order nuclide parameters
- $N_{c_i}(t)$ = concentration of nuclide i per fuel region at time t (atoms/region)
- t = time (sec)
- F = fission rate (fissions/sec in fuel region)
- α_i = fission yield for isotope i (atoms/fission)
- λ_i = decay constant for isotope i (sec^{-1})
- γ_i = escape rate coefficient (sec^{-1})
- $\beta_i = \Gamma_{a_i} \phi_{th} \beta_i = r_{a_i} \phi_{th}$ = burnup rate (sec^{-1})
- f_{ij} = branching fraction from i to j
- h = fraction of failed fuel

The program has a basic library of 167 nuclides with a capability of 200 nuclides. Library data include decay scheme information, production information, purification factors for typical demineralizers, and fuel escape rate coefficients. The library also contains decay gamma spectra in seven energy groups. Input data include time intervals, initial source inventory in the fuel, neutron flux, power level, fraction of fuel defects, and density of reactor coolant. The program describes the system analyzed, as well as the operating history, the activities, and associated gamma spectral information as a function of time. Fuel assembly source terms for shielding design, given in Chapter 12, are calculated by this method.

The calculation of the core iodine fission product inventory is consistent with the inventories given in TID-14844 (DiNunno et al., 1962). The core iodine and noble gas fission product inventories in Table 11.1-1 are based on continuous operation of the unit at 3,636 MWt (design core output plus 2.0 percent instrumentation error). The core inventory source terms used for Chapter 15 radiological accident analyses have been revised to reflect requirements as described by Regulatory Guide 1.183 and is discussed in Section 15.0.9.1.

The core inventory information listed in Table 11.1-1 is being retained for historical purposes because that was the basis for original plant shielding design as described in Section 12.2.1.

Fuel element heat loadings and stresses as well as fuel operating experience are presented in Chapter 4.

11.1.2 RADIONUCLIDE INVENTORY IN FUEL ELEMENT GAP

The gap activity is that fraction of the gaseous activity in the core that diffuses to the fuel gaps. The fuel element gap source terms used for Chapter 15 radiological accident analyses have been revised to reflect requirements as described by Regulatory Guide 1.183 and is discussed in Section 15.0.9.2.

11.1.3 PRIMARY COOLANT EQUILIBRIUM ACTIVITIES

11.1.3.1 Fission Product Activities

The design basis fission product activities in the reactor coolant resulting from fuel defects associated with 1-percent power are also calculated with the ACTIVITY2 program. The following differential equations are used:

a. First order nuclides:

$$\frac{dN_{wi}}{dt} = \frac{hn\gamma_i}{V_w} N_{ci}(t) - \left(\lambda_i + \frac{PF_{EQ_i}Q_1}{V_w} + \beta_i \frac{T_1}{T_2}\right) N_{wi}(t) \qquad (11.1\text{-}4)$$

b. Second order nuclides:

$$\frac{dN_{wj}}{dt} = \frac{hn\gamma_j}{V_w} N_{cj}(t) + \lambda_i f_{ij} N_{wi}(t) - \left(\lambda_j + \frac{PF_{EQ_j}Q_1}{V_w} + \beta_j \frac{T_1}{T_2}\right) N_{wj}(t) \qquad (11.1\text{-}5)$$

c. Third order nuclides:

$$\frac{dN_{wk}}{dt} = \frac{hn\gamma_k}{V_w} N_{ck}(t) + \lambda_i f_{ik} N_{wi}(t) + \lambda_j f_{jk} N_{wj}(t) - \left(\lambda_k + \frac{PF_{EQ_k}Q_1}{V_w} + \beta_k \frac{T_1}{T_2}\right) N_{wk}(t) \qquad (11.1\text{-}6)$$

where:

$N_{w_i}(t)$ = concentration of nuclide i in the main coolant at time t (atoms/cm^3)

n = total number of fuel regions

V_w = volume of main coolant (cm^3)

PF_{EQ} = equivalent purification factor (fraction) for i

$\beta_i = \sigma_{a_i} \phi_{th}$ = Burnup rate (sec^{-1})

Q_1 = equivalent flow into purification stream (cm^3/sec)

T_1 = coolant residence time in core (sec)

T_2 = coolant circulation time (sec)

The reactor coolant system (RCS) design basis equilibrium activities in Table 11.1-2 are based on the parameters listed in Table 11.1-3 and represent coolant data used by the original license application to establish plant shielding and radwaste effluent assessments. This data is retained for historical purposes to avoid loss of original design basis and is not subject to future update.

In these calculations, the defective fuel rods were assumed to be present in the initial core and uniformly distributed throughout the core. Thus, the fission product escape rate coefficients were based upon average fuel temperature. Calculations are performed using the average temperature of the reactor coolant. The reactor coolant density correction of 1.4 was made in order to obtain the correct radionuclides concentration downstream of the letdown heat exchanger.

Also included in Table 11.1-2 are the expected equilibrium concentrations for the RCS. These results were based on measured and calculated concentrations given in NUREG-0017 (USNRC 1976) and the parameters listed in Table 11.1-3.

The expected reactor coolant activities were used to develop the source terms for gaseous and liquid effluents in Sections 11.2 and 11.3.

Both expected and design reactor coolant activities were used to evaluate the ventilation design in Chapter 12.

The methodology to calculate the current design basis primary coolant activity concentrations is similar to that used during original licensing basis with a minor modification. The design basis core inventory is calculated by industry computer code ORIGEN instead of ACTIVITY2. Since the source of primary coolant fission product activity is the leakage of core activity via the defective fuels, the primary coolant activity concentrations calculated by ACTIVITY2 are adjusted by the ratio of the core inventory developed by ORIGEN, and presented in Chapter 15, to the core inventory calculated by ACTIVITY2. The coolant concentration for a given isotope depends on the core inventory, the escape coefficients of the isotope and its precursor isotopes, and the depletion rate of the isotopes. For each isotope in the primary coolant, the decay chain and the escape coefficients are examined, and those isotopes that have significant impact on the concentration of the referenced isotope in the coolant are identified. Amongst these major contributors, the ratios of the ORIGEN core activity to the ACTIVITY2 core activity are generally very close to each other, and the maximum scaling factor is used to adjust the ACTIVITY2 based coolant concentration to reflect the core developed by ORIGEN.

The current design basis reactor coolant equilibrium activities presented in Table 11.1-2A, are based on parameters listed in Table 11.1–3A.

11.1.3.2 Tritium Activity

There are two principal contributors to tritium production within the pressurized water reactor system: the ternary fission source and the dissolved boron in the reactor coolant. Additional contributions are made by Li-6, Li-7, and deuterium in the reactor water. Tritium is also produced by nuclear reactions with boron contained in burnable poison rods. Tritium production from different sources is shown in Table 11.1-4.

Fission Source

This tritium is formed within the fuel material and may:

1. Remain in the fuel rod uranium matrix
2. Diffuse into the cladding and become fixed there, as zirconium tritide
3. Diffuse through the cladding and be released into the primary coolant, or
4. Be released to the coolant through microscopic cracks or failures in the fuel cladding

Previous WNES fuel design has conservatively assumed that the ratio of fission tritium released into the coolant to the total fission tritium formed was approximately 0.30 for Zircaloy clad fuel. The operating experience at the R.E. Ginna Plant of the Rochester Gas and Electric Company, and at other operating pressurized water reactors using zircaloy clad fuel, has shown that the tritium release through the Zircaloy fuel cladding is less than the earlier estimates. Consequently, the release fraction has been revised downward from 30 to 10 percent based on these data (WCAP-8253 1974).

Boric Acid Source

A direct contribution to the reactor coolant tritium concentration is made by neutron reaction with the boron in solution. The concentration of boric acid varies with core life and load follow so that this is a steadily decreasing source during core life. The principal boron reactions are B-10(n,2α) H-3 and B-10(n,α)Li-7(n,nα)H-3 reactions. The Li-7 reaction is controlled by limiting the overall lithium concentration to approximately 2 ppm during operation.

Burnable Poison Rod Source

These rods are in the core only during the first operating cycle and their potential tritium contribution is only during this period.

Lithium and Deuterium

Lithium and deuterium reactions contribute only minor quantities to the tritium inventory (Table 11.1-4). These sources are due to the activation of the lithium and deuterium in the RCS as they pass through the reactor. Lithium-6 is essentially excluded from the system by using 99.9 percent Li-7.

Design Bases

The design intent is to reduce the tritium sources in the RCS to a practical minimum to permit longer retention of the reactor coolant within the plant without compromising operator exposures. Reduction of source terms is provided by using hafnium or Ag-In-Cd control rods instead of B4C and the determination that the quantity of tritium released from the fuel rods fabricated with zirconium alloy cladding is less than originally expected.

Design Evaluation

Table 11.1-4 compares a typical design basis tritium production which has been used in the past to establish system and operational requirements of the plant and present expected values. There are two principal contributors to the expected tritium release to the RCS: ternary fission source and the dissolved boron in the reactor coolant.

Because of the importance of the ternary fission source on the operation of the plant, WNES has been closely following operating plant data at the R. E. Ginna Plant. The R. E. Ginna Plant has a Zircaloy clad core with silver-indium-cadmium control rods. The operating levels of boron concentration during the startup of the plant are approximately 1,100 to 1,200 ppm of boron. In addition, burnable poison rods in the core contain boron which contributes some tritium to the coolant, but only during the first cycle. Data during the operation of the plant have very clearly indicated that the present design sources were conservative. The tritium released is essentially from the boron dissolved in the coolant and a ternary fission source which is less than 10 percent. In addition to these data, other operating plants with Zircaloy clad cores have also reported low tritium concentrations in the RCS after considerable periods of operation.

This quantity of tritium becomes uniformly distributed in the RCS, the primary grade water system, and the boron recovery system. During refueling operations, the tritium is further diluted when the refueling canal is filled from the borated refueling water storage tank.

The expected tritium concentration in the primary coolant is based on the value provided in NUREG-0017 (USNRC 1976). For radioactive liquid waste analysis, it is assumed that 50 percent of the activity produced and entering into the coolant in 1 year is released in the radioactive liquid waste system effluent. The remaining 50 percent is released in the radioactive gaseous waste system and ventilation effluents.

The design tritium concentration in the primary coolant is selected to allow limited access to the containment during normal operation. The water management plan controls tritium concentrations to design levels and also allows for continuous containment access during refueling with operation of the containment ventilation system.

Based on the above, the following conclusions have been reached:

1. The tritium levels in plants operating with Zircaloy clad cores are lower than previous design predictions.

2. The tritium source at full power operation is reduced by using hafnium control rods.

11.1.3.3 Corrosion Products

Corrosion products in the reactor coolant become activated when they pass through the core. The most important corrosion products are Cr-51, Mn-54, Fe-55, Fe-59, Co-58, and Co-60. The corrosion product activity is dependent on many factors, including the type of plant and the materials of construction. The mass transport process is complex and stochastic, and calculational methods to predict corrosion product activity accurately have not been successfully correlated with operational data (Bartlett 1969). Analytical predictions of the corrosion product activity levels are approximations. Therefore, design corrosion product levels are assumed to be the values measured at operating reactors. The design corrosion product levels are based on NUREG-0017 (USNRC 1976) values modified by the appropriate adjustment factor described therein.

The corrosion product activities in the reactor coolant are given in Table 11.1-2, for the original license and Table 11.1–2A for current conditions.

11.1.3.4 Nitrogen-16 Activity

Nitrogen-16 is a concern only during reactor operation because of its short half-life (7.1 seconds). Nitrogen-16 is produced in circulating primary coolant entering the core region and irradiated by neutrons. Reactions with all three oxygen isotopes 0-16 (99.76 percent), 0-17 (0.037 percent), and 0-18 (0.204 percent) result in the production of N-16.

Nitrogen-16 emits high energy gammas in 75-percent of the disintegrations (70-percent at 6.13 MeV and 5 percent at 7.11 MeV).

The N-16 activity at various points in the RCS is given in Table 11.1-5.

11.1.4 RADIOACTIVITY IN THE SECONDARY SIDE

The concentrations of principal radioisotopes in the secondary side of the steam generators are listed for both the design and expected cases in Table 11.1-6 for liquid and Table 11.1-7 for steam. These tables present secondary coolant and steam data used by the original license application to establish plant shielding and radwaste effluent assessments. This data is retained for historical purposes to avoid loss of original design basis and is not subject to future update. The design results for fission and activation products, based on parameters in Table 11.1-3, were calculated with the computer program IONEXCHANGER which solves the following differential equations for secondary liquid activities.

1. First order nuclides:

$$\frac{dN_i}{dt} = R_i - \left(\lambda_i + \frac{Q_B}{V}\right)N_i(t) \qquad (11.1\text{-}7)$$

2. Second order nuclides:

$$dN_i = R_j + \lambda_i\ f_{ij}\ N_j\,(t) - \left(\lambda_j + \frac{Q_B}{V}\right)N_j(t) \qquad (11.1\text{-}8)$$

3. Third order nuclides:

$$\frac{dN_k}{dt} = R_k + \lambda_i\ f_{ik}\ N_i(t) + \lambda_j\ f_{jk}\ N_j(t) - \left(\lambda_k + \frac{Q_B}{V}\right)N_k(t) \qquad (11.1\text{-}9)$$

where:

N_i = number of atoms of nuclide i (atoms)

R_i = feed rate of nuclide i (atoms/sec)

λ_i = radioactive decay constant for nuclide i (sec^{-1})

f_{ij} = branching fraction from i to j

Q_B = steam generator radioactivity removal rate (cm^3/sec)

V = volume of steam generator liquid (cm^3)

t = time in seconds

Secondary side steam activities are obtained by using the following relationship:

$$A_i = P_i A_{oi} \qquad (11.1\text{-}10)$$

where:

A_i = steam equilibrium activity for isotope i (μCi/gm)

A_{oi} = liquid equilibrium activity for isotope i (μCi/gm)

The expected secondary liquid and steam activities are based on the concentrations reported in NUREG-0017 and the parameters listed in Table 11.1-3.

11.1.5 REFERENCES FOR SECTION 11.1

11.1-1 Barlett, J.W. 1969. Stochastics of Coolant Crud. In: ANS Transactions, Vol 12.

11.1-2 DiNunno, J.J.; Anderson, F.D.; Baker, R.E.; and Waterfield, R.L. 1962. Calculation of Distance Factors for Power and Test Reactor Sites. TID 14844, U.S. Atomic Energy Commission, U.S. National Technical Information Service, Springfield, Va.

11.1-3 U.S. Nuclear Regulatory Commission 1976. Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors (PWR-GALE code), NUREG-0017. U.S. National Technical Information Service, Springfield, Va.

11.1-4 WCAP-8253 1974, “Source Term Data for Westinghouse Pressurized Water Reactors,” Westinghouse Electric Corporation, Pittsburgh, Penn.

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

TABLE 11.1-1 IODINE AND NOBLE GAS INVENTORY IN REACTOR CORE - ORIGINAL LICENSE BASIS [1]

Isotope	Core (Ci)
I-131	9.1E+07
I-132	1.3E+08
I-133	2.0E+08
I-134	2.4E+08
I-135	1.9E+08
Kr-83m	1.6E+07
Kr-85m	4.0E+07
Kr-85	8.8E+05
Kr-87	7.7E+07
Kr-88	1.1E+08
Kr-89	1.4E+08
Xe-131m	8.0E+04
Xe-133m	4.9E+06
Xe-33	2.0E+08
Xe-135m	5.5E+07
Xe-135	5.4E+07
Xe-137	1.8E+08
Xe-138	1.8E+08

NOTES:

(1) Based on 650 days of operation at 3,636 MWt.

Historical, not subject to future updating. This table has been retained to preserve original license basis.

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

TABLE 11.1-2 REACTOR COOLANT EQUILIBRIUM CONCENTRATIONS - ORIGINAL LICENSE BASIS

Nuclide		Design (μCi/g)	Expected (μCi/g)
Noble Gases			
Kr	83m	4.5E-01 (1)	2.2E-02
Kr	85m	1.7E+00	9.2E-02
Kr	85	3.4E-02	2.1E-03
Kr	87	1.2E+00	6.6E-02
Kr	88	3.4E+00	1.9E-01
Kr	89	1.1E-01	6.2E-03
Xe	131m	1.1E-02	5.5E-03
Xe	133m	6.1E-01	4.1E-02
Xe	133	2.6E+01	1.7E+00
Xe	135m	1.2E+00	1.6E-02
Xe	135	5.1E+00	2.2E-01
Xe	137	1.7E-01	1.1E-02
Xe	138	6.0E-01	5.4E-02
Subtotal:		4.1E+01	2.4E+00
Halogens			
Br	83	9.0E-02	5.8E-03
Br	84	4.1E-02	3.2E-03
Br	85	5.7E-03	3.8E-04
I	130	7.2E-03	2.4E-03
I	131	2.6E+00	2.9E-01
I	132	9.3E-01	1.2E-01
I	133	4.2E+00	4.3E-01
I	134	5.8E-01	5.8E-02
I	135	2.2E+00	2.2E-01
Subtotal:		1.1E+01	1.1E+00

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

TABLE 11.1-2 REACTOR COOLANT EQUILIBRIUM CONCENTRATIONS - ORIGINAL LICENSE BASIS (CONTINUED)

Nuclide		Design (μCi/g)	Expected (μCi/g)
Corrosion Products			
Cr	51	2.0E-03	2.0E-03
Mn	54	3.3E-04	3.3E-04
Fe	55	1.7E-03	1.7E-03
Fe	59	1.1E-03	1.1E-03
Co	58	1.7E-02	1.7E-02
Co	60	2.1E-03	2.1E-03
Subtotal:		2.4E-02	2.4E-02
Other Nuclides			
Rb	86	2.7E-04	9.1E-05
Rb	88	3.4E+00	2.5E-01
Sr	89	4.3E-03	3.7E-04
Sr	90	1.7E-04	1.0E-05
Sr	91	2.0E-03	7.5E-04
Y	90	2.1E-04	1.3E-06
Y	91m	1.1E-03	4.5E-04
Y	91	6.9E-04	6.7E-05
Y	93	3.4E-04	3.9E-05
Zr	95	7.1E-04	6.3E-05
Nb	95	7.4E-04	5.3E-05
Mo	99	3.4E+00	9.0E-02
Tc	99m	1.9E+00	5.6E-02
Ru	103	3.4E-04	4.7E-05
Ru	106	3.3E-05	1.0E-05
Rh	103m	3.4E-04	5.6E-05
Rh	106	3.3E-05	1.3E-05
Te	125m	9.0E-05	3.0E-05
Te	127m	2.1E-03	2.9E-04
Te	127	1.1E-03	9.8E-04

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

TABLE 11.1-2 REACTOR COOLANT EQUILIBRIUM CONCENTRATIONS - ORIGINAL LICENSE BASIS (CONTINUED)

Nuclide		Design (μCi/g)	Expected (μCi/g)
Te	129m	3.9E-02	1.5E-03
Te	129	2.2E-02	2.0E-03
Te	131m	2.3E-02	2.7E-03
Te	131	1.1E-02	1.4E-03
Te	132	2.7E-01	2.9E-02
Cs	134	3.3E-01	2.7E-02
Cs	136	1.7E-01	1.4E-02
Cs	137	1.7E+00	1.9E-02
Ba	137m	1.5E+00	2.0E-02
Ba	140	4.4E-03	2.3E-04
La	140	1.5E-03	1.6E-04
Ce	141	7.0E-04	7.4E-05
Ce	143	5.2E-04	4.4E-05
Ce	144	5.0E-04	3.5E-05
Pr	143	6.8E-04	5.3E-05
Pr	144	5.0E-04	4.1E-05
Np	239	3.9E-03	1.3E-03
Subtotal:		1.4E+01	5.2E-01
H-3		3.5E+00	1.0E+00
Total (Excluding H-3)		5.8E+01	4.0E+00
Total (Including H-3)		6.2E+01	5.0E+00

NOTE:

(1) 4.5E-01 = 4.5×10^{-1}

Historical, not subject to future updating. This table has been retained to preserve original license basis.

TABLE 11.1–2A DESIGN REACTOR COOLANT EQUILIBRIUM CONCENTRATIONS AT 3723 MWt

Nuclide	Primary Coolant Activity Concentration (μCi/g)
Kr-83m	3.53E-01
Kr-85m	1.09E+00
Kr-85	1.33E-01
Kr-87	8.49E-01
Kr-88	2.22E+00
Kr-89	6.96E-02
Xe-131m	1.93E-01
Xe-133m	7.68E-01
Xe-133	2.54E+01
Xe-135m	9.45E-01
Xe-135	5.50E+00
Xe-137	1.94E-01
Xe-138	6.54E-01
Br-83	7.05E-02
Br-84	3.50E-02
Br-85	3.68E-03
Br-87	1.89E-03
I-129	1.79E-07
I-130	4.39E-02
I-131	2.67E+00
I-132	1.09E+00
I-133	4.06E+00
I-134	6.19E-01
I-135	2.39E+00
I-136	6.73E-03
Se-81	5.84E-07
Se-83	7.94E-07
Se-84	4.60E-07
Rb-86	1.46E-01

TABLE 11.1–2A DESIGN REACTOR COOLANT EQUILIBRIUM CONCENTRATIONS AT 3723 MWt (CONTINUED)

Nuclide	Primary Coolant Activity Concentration (μCi/g)
Rb-88	2.32E+00
Rb-89	1.45E-01
Rb-90	1.12E-02
Rb-91	5.70E-03
Rb-92	3.88E-04
Sr-89	3.02E-03
Sr-90	1.96E-04
Sr-91	1.30E-03
Sr-92	9.45E-04
Sr-93	4.41E-05
Sr-94	7.49E-06
Y-90	3.84E-04
Y-91m	7.77E-04
Y-91	1.43E-02
Y-92	1.12E-03
Y-93	6.13E-04
Y-94	2.72E-05
Y-95	1.14E-05
Zr-95	5.72E-04
Zr-97	3.67E-04
Nb-95m	6.58E-06
Nb-95	5.78E-04
Nb-97m	3.48E-04
Nb-97	3.91E-04
Mo-99	5.27E+00
Mo-101	2.10E-02
Mo-102	1.52E-02
Mo-105	7.22E-04
Tc-99m	2.73E+00

TABLE 11.1–2A DESIGN REACTOR COOLANT EQUILIBRIUM CONCENTRATIONS AT 3723 MWt (CONTINUED)

Nuclide	Primary Coolant Activity Concentration (μCi/g)
Tc-101	2.05E-02
Tc-102	1.53E-02
Tc-105	7.66E-04
Ru-103	5.49E-04
Ru-105	1.34E-04
Ru-106	2.02E-04
Ru-107	1.90E-06
Rh-103m	5.50E-04
Rh-105m	3.82E-05
Rh-105	3.44E-04
Rh-106	2.25E-04
Rh-107	1.14E-05
Sn-127	2.41E-06
Sn-128	5.04E-06
Sn-130	8.31E-07
Sb-127	2.77E-05
Sb-128	2.66E-06
Sb-129	3.73E-05
Sb-130	2.67E-06
Sb-131	1.16E-05
Sb-132	8.85E-07
Sb-133	1.04E-06
Te-125m	3.98E-04
Te-127m	3.11E-03
Te-127	1.10E-02
Te-129m	1.32E-02
Te-129	1.35E-02
Te-131m	3.34E-02
Te-131	1.26E-02

TABLE 11.1–2A DESIGN REACTOR COOLANT EQUILIBRIUM CONCENTRATIONS AT 3723 MWt (CONTINUED)

Nuclide	Primary Coolant Activity Concentration (μCi/g)
Te-132	2.77E-01
Te-133m	1.91E-02
Te-133	8.60E-03
Te-134	2.91E-02
Cs-134m	4.72E-02
Cs-134	2.31E+01
Cs-136	3.52E+00
Cs-137	1.62E+01
Cs-138	1.00E+00
Cs-139	8.99E-02
Cs-140	9.09E-03
Cs-142	1.08E-04
Ba-137m	1.52E+01
Ba-139	7.79E-02
Ba-140	3.75E-03
Ba-141	1.21E-04
Ba-142	1.66E-04
La-140	1.26E-03
La-141	2.61E-04
La-142	2.44E-04
La-143	1.38E-05
Ce-141	5.65E-04
Ce-143	4.19E-04
Ce-144	4.46E-04
Ce-145	2.09E-06
Ce-146	7.74E-06
Pr-143	5.20E-04
Pr-144	4.50E-04
Pr-145	1.49E-04

TABLE 11.1–2A DESIGN REACTOR COOLANT EQUILIBRIUM CONCENTRATIONS AT 3723 MWt (CONTINUED)

Nuclide	Primary Coolant Activity Concentration (μCi/g)
Pr-146	2.04E-05
Nd-147	2.22E-04
Nd-149	2.28E-05
Nd-151	1.66E-06
Pm-147	1.13E-04
Pm-149	1.97E-04
Pm-151	5.48E-05
Sm-151	7.50E-07
Sm-153	1.55E-04
Cr-51	2.00E-03
Mn-54	3.30E-04
Fe-55	1.70E-03
Fe-59	1.10E-03
Co-58	1.70E-02
Co-60	2.10E-03
Np-239	1.98E-02
H-3	3.5E+00

TABLE 11.1-3 PARAMETERS USED IN THE CALCULATION OF REACTOR COOLANT, SECONDARY SIDE LIQUID, AND SECONDARY SIDE STEAM FISSION AND ACTIVATION PRODUCT ACTIVITY - ORIGINAL LICENSE BASIS

Parameter	Value (1)	
Core thermal power (includes 2% instrumentation error)		3,636 MWt
Percent fuel defects (design case)		1.0
Percent fuel defects (expected case)		0.12
Fission product escape rate coefficients		
Noble gas nuclides		6.5×10^{-8} (sec^{-1})
Br, Rb, I, and Cs nuclides		1.3×10^{-8} (sec^{-1})
Te nuclides		1.0×10^{-9} (sec^{-1})
Mo nuclides		2.0×10^{-12} (sec^{-1})
Sr and Ba nuclides		1.0×10^{-12} (sec^{-1})
Y, La, Ce, Pr nuclides		1.6×10^{-12} (sec^{-1})
Reactor coolant liquid mass (without pressurizer)		4.70×10^{5} lb
Reactor coolant liquid volume (without pressurizer)		10,920 ft^3
Reactor coolant liquid volume (with pressurizer)		12,000 ft^3
Reactor coolant full power average temperature		590°F
Purification flow rate, normal		75 gpm
Mixed bed demineralizer decontamination factors		
Noble Gases, N-16, H-3		1.0
Cations		
Design: Cs, Mo, Y		1.0
Expected: Cs, Rb		2.0
All other nuclides including activation products		10.0

TABLE 11.1-3 PARAMETERS USED IN THE CALCULATION OF REACTOR COOLANT, SECONDARY SIDE LIQUID, AND SECONDARY SIDE STEAM FISSION AND ACTIVATION PRODUCT ACTIVITY - ORIGINAL LICENSE BASIS (CONTINUED)

Parameter	Value (1)	
Cation bed demineralizer decontamination factors		
Noble Gases, N-16, Halogens, H-3		1.0
Cations		
Design: Cs, Y, Mo		10.0
Expected: Cs, Rb		10.0
All other nuclides including activation products		
Design		1.0
Expected		10.0
Expected halogens		10.0
Ratio of cation bed demineralizer flow to purification bed demineralizer flow		0.1
Reactor coolant letdown discharged via boron recovery system		
Design		500 lb/hr
Expected		500 lb/hr
Steam flow rate		1.589×10^7 lb/hr
Primary to secondary leak rate		
Design		1,370 lb/day
Expected		100 lb/day
Steam generator partition factor (recirculating U-tube)		
Noble Gases: N-16, H-3		1.0
Halogens		0.01
Cations		
Design: Cs, Rb		0.0025
Expected: Cs, Rb		0.001
Others		

TABLE 11.1-3 PARAMETERS USED IN THE CALCULATION OF REACTOR COOLANT, SECONDARY SIDE LIQUID, AND SECONDARY SIDE STEAM FISSION AND ACTIVATION PRODUCT ACTIVITY - ORIGINAL LICENSE BASIS (CONTINUED)

Parameter		Value (1)
Design		0.0025
Expected		0.001
Condensate polishing demineralizer decontamination factors		
Cations		
Design: Cs, Y, Mo		2.0
Expected: Cs, Rb		2.0
All other nuclides including activations products		
Design		10.0
Expected		10.0
Condensate polishing flow rate		9.89×10^6 lb/hr
Thermal neutron flux		2.0×10^{13} n/cm^2-sec
Operating time (650 EFPD)		15,600 hr
Coolant cycle time		9.9 sec
Coolant in core time		0.7 sec
Degasification factor		1.0
Secondary side equilibrium time		10^4 hr
Number of condensate demineralizers		7 plus 1 spare
Volume of each condensate demineralizer		197 ft^3
Type of condensate demineralizers		Deep bed
Regeneration time for each demineralizer		
Design		3.5 days (estimate)
Expected		7.0 days
Number of regenerations/yr for condensate polishing demineralizer		
Design		100 (estimate)
Expected		56

TABLE 11.1-3 PARAMETERS USED IN THE CALCULATION OF REACTOR COOLANT, SECONDARY SIDE LIQUID, AND SECONDARY SIDE STEAM FISSION AND ACTIVATION PRODUCT ACTIVITY - ORIGINAL LICENSE BASIS (CONTINUED)

Parameter	Value (1)	
Volume control tank volumes		
Vapor		240 ft^3
Liquid		160 ft^3
Total secondary fluid per steam generator	Expected	Design
Liquid	99,253 lb	103,000 lb
Steam	8,437 lb	8,000 lb
Total	107,690 lb	111,000 lb
Hypothetical Design flowrates for the steam generator blowdown system (see Section 11.2.2.3 for time periods for each release)		
Hot Standby:		150,520 lb/hr
1% MSR from each steam generator (37,630 lb/hr per steam generator)		
Intermittent Blowdown:		263,410 lb/hr
1% MSR from three steam generators (37,630 lb/hr per steam generator)		
4% MSR from one steam generator (150,520 lb/hr)		
Hypothetical Expected flowrates for the steam generator blowdown system		
1% MSR from each steam generator (37,000 lb/hr per steam generator)		148,000 lb/hr
Fraction removed from steam generator blowdown (purification factors for design and expected cases)		
Noble Gases		0.0
Halogens		0.8505
Cs, Rb		0.4975
Others		0.8955

TABLE 11.1-3 PARAMETERS USED IN THE CALCULATION OF REACTOR COOLANT, SECONDARY SIDE LIQUID, AND SECONDARY SIDE STEAM FISSION AND ACTIVATION PRODUCT ACTIVITY - ORIGINAL LICENSE BASIS (CONTINUED)

Parameter	Value [(1)]	
Tritium		0.0
Ratio of condensate demineralizer flow rate to the total steam flow rate		0.6224

NOTES:

(1) Values represent assumptions used to estimate liquid radiological effluents prior to initial plant licensing and are retained for historical purposes only. The Radiological Effluent Monitoring and Offsite Dose Calculation Manual (REMODCM) provides requirements for system operation, dose calculations and monitoring to ensure compliance with 10 CFR 20 Appendix B, Table II, Column 2 effluent limits.

TABLE 11.1–3A PARAMETERS USED IN THE CALCULATION OF DESIGN REACTOR FISSION AND ACTIVATION PRODUCT ACTIVITY

Parameter	Value
Core thermal power (includes 2% instrumentation error)	3,723 MWt
Percent fuel defects	1.0
Fission product escape rate coefficients	
Noble gas nuclides	6.5×10^{-8} (sec^{-1})
Br, Rb, I, and Cs nuclides	1.3×10^{-8} (sec^{-1})
Te nuclides	1.0×10^{-9} (sec^{-1})
Mo nuclides	2.0×10^{-12} (sec^{-1})
Sr and Ba nuclides	1.0×10^{-11} (sec^{-1})
Y, La, Ce, Pr nuclides	1.6×10^{-12} (sec^{-1})
Reactor coolant average density	44.39 lbs/ft^3
Reactor coolant liquid volume (without pressurizer and surge line)	10,100 ft^3
Reactor coolant full power average temperature	594.5°F
Purification flow rate, normal	82 gpm
Mixed bed demineralizer decontamination factors	
Noble Gases, N 16, H 3	1.0
Cations	
Cs, Mo, Y	1.0
Halogens	10.0
All other nuclides including activation products	10.0
Cation bed demineralizer decontamination factors	
Noble Gases, N 16, H 3	1.0
Cations	
Cs, Mo, Y	10.0
Halogens	1.0
All other nuclides including activation products	1.0

TABLE 11.1–3A PARAMETERS USED IN THE CALCULATION OF DESIGN REACTOR FISSION AND ACTIVATION PRODUCT ACTIVITY (CONTINUED)

Parameter	Value
Ratio of cation bed demineralizer flow to purification bed demineralizer flow	0.01
Reactor coolant letdown discharged via boron recovery system	500 lb/hr
Thermal neutron flux - core / coolant	3.83×10^{13} n/cm^{2}sec
Reactor operating time (assumed 2 cycles)	28,800 hr
Coolant cycle time	9.74 sec
Coolant in core time	0.721 sec
Degasification factor	1.0
Design corrosion product concentrations in RCS	
Cr 51	2.0E-3 µCi/gm
Mn 54	3.3E-4 µCi/gm
Fe 55	1.7E-3 µCi/gm
Fe 59	1.1E-3 µCi/gm
Co 58	1.7E-2 µCi/gm
Co –60	2.1E-3 µCi/gm
Np 239	2.2E-3 µCi/gm
Design Tritium concentration	3.5 µCi/gm

TABLE 11.1-4 TRITIUM PRODUCTION

Tritium Source	Total Produced (Ci/yr)	Release Expected to Reactor Coolant (Ci/yr)
Ternary fissions		
Initial cycle	14,000	1,400
Equilibrium cycle	10,900	1,090
Burnable poison rods		
Initial cycle	1,950	195
Coolant (soluble boron)		
Initial cycle	388	388
Equilibrium cycle	285	285
Coolant lithium, deuterium		
Initial cycle	141	141
Equilibrium cycle	109	109
Total initial cycle	16,470	2,124
Total equilibrium cycle	11,294	1,484

NOTES:

Values presented are for a typical 3,565 MWt PWR. However, calculated tritium releases are based on 102 percent of this value (3,636 MWt) in order to comply with Regulatory Guide 1.49 (Section 1.8.1.49). Calculation of tritium releases are based on methodology presented in NUREG-0017.

Release fraction from fuel, 10 percent

Release fraction from burnable poison rods, 10 percent

Weight of boron-10 in burnable poison rods, 6,160 gm

Initial cycle boron, 900 ppm

Equilibrium cycle boron, 1,100 ppm

Lithium concentration (99.9 atom percent lithium-7), 2.2 ppm

Initial cycle operating time, 9,240 effective full-power hours

Equilibrium cycle operating time, 7,200 effective full-power hours

Production in control rods is based on continuous daily load follow (12, 3, 6, 3 cycle). During base load full power operation, the production would be negligible.

TABLE 11.1-5 REACTOR COOLANT N-16 ACTIVITY [1]

Position in Loop	Loop Transit Time (sec)	N-16 Activity (μCi/g)
Leaving core	0.0	189
Leaving reactor vessel	1.1	170
Entering steam generator	1.4	164
Leaving steam generator	5.4	112
Entering reactor coolant pump	6.0	106
Entering reactor vessel	6.8	98
Entering core	9.0	86
Leaving core	9.7	189

NOTE:

(1) These values are based on a typical 3,565 Mwt PWR.

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

TABLE 11.1-6 SECONDARY SIDE LIQUID EQUILIBRIUM CONCENTRATIONS ORIGINAL LICENSE BASIS

Nuclide	Design (1) (μCi/g)	Expected (2) (μCi/g)
Br 83	1.5E-05 (3)	7.3E-08
Br 84	3.1E-06	1.9E-08
Br 85	5.2E-08	2.7E-10
I 130	1.7E-06	4.1E-08
I 131	6.9E-04	5.4E-06
I 132	2.0E-04	2.0E-06
I 133	1.0E-03	7.7E-06
I 134	6.3E-05	4.5E-07
I 135	4.9E-04	3.6E-06
Cr 51	7.3E-07	5.6E-08
Mn 54	1.2E-07	1.2E-08
Fe 55	6.2E-07	5.0E-08
Fe 59	4.0E-07	3.8E-08
Co 58	6.2E-06	5.0E-07
Co 60	7.6E-07	5.6E-08
Rb 86	1.8E-07	4.6E-09
Rb 88	1.8E-04	1.0E-06
Sr 89	1.6E-06	1.3E-08
Sr 90	6.2E-08	2.5E-10
Sr 91	6.1E-06	1.6E-08
Y 90	7.5E-08	5.3E-11
Y 91m	3.7E-07	1.2E-08
Y 91	2.5E-07	1.9E-09
Y 93	1.1E-07	7.8E-10
Zr 95	2.6E-07	2.5E-09
Nb 95	2.7E-07	2.5E-09
Mo 99	1.2E-03	2.6E-06

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

TABLE 11.1-6 SECONDARY SIDE LIQUID EQUILIBRIUM CONCENTRATIONS ORIGINAL LICENSE BASIS (CONTINUED)

Nuclide	Design (1) (μCi/g)	Expected (2) (μCi/g)
Tc 99m	7.9E-04	2.5E-06
Ru 103	1.3E-07	1.3E-09
Ru 106	1.2E-08	2.5E-10
Ru 103m	1.2E-07	2.3E-09
Rh 106	1.2E-08	0.0
Te 125m	3.3E-08	6.3E-10
Te 127m	7.7E-07	6.3E-09
Te 127	4.5E-07	2.4E-08
Te 129m	1.4E-05	3.8E-08
Te 129	8.8E-06	6.7E-08
Te 131m	8.1E-06	6.9E-08
Te 131	2.2E-06	2.5E-08
Te 132	9.6E-05	6.5E-07
Cs 134	2.2E-04	1.4E-06
Cs 136	1.1E-04	7.3E-07
Cs 137	1.1E-03	1.0E-06
Ba 137m	5.7E-04	1.2E-06
Ba 140	1.6E-06	6.3E-09v
La 140	5.9E-07	4.7E-09
Ce 141	2.5E-07	2.5E-09
Ce 143	1.8E-07	6.8E-10
Ce 144	1.8E-07	1.2E-09
Pr 143	2.5E-07	1.3E-09
Pr 144	1.8E-07	2.6E-09
Np 239	1.4E-06	4.0E-08
H 3	1.3E-03	1.0E-03
Total (excluding H-3)	6.8E-03	3.1E-05

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

TABLE 11.1-6 SECONDARY SIDE LIQUID EQUILIBRIUM CONCENTRATIONS ORIGINAL LICENSE BASIS (CONTINUED)

Nuclide	Design (1) (μCi/g)	Expected (2) (μCi/g)
Total (including H-3)	8.0E-03	1.0E-03

NOTES:

(1) Based on 1,370 lb/day primary to secondary leak rate.

(2) Based on 100 lb/day primary to secondary leak rate.

(3) 1.5E-05 = 1.5 x 10^{-5}.

Historical, not subject to future updating. This table has been retained to preserve original license basis.

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

TABLE 11.1-7 SECONDARY SIDE STEAM EQUILIBRIUM CONCENTRATIONS - ORIGINAL LICENSE BASIS

Nuclide	Design (1) (μCi/g)	Expected (2) (μCi/g)
Kr-83m	1.6E-06 (3)	5.7E-09
Kr-85m	6.2E-06	2.4E-08
Kr-85	1.2E-07	5.6E-10
Kr-87	4.4E-06	1.7E-08
Kr-88	1.2E-05	4.9E-08
Kr-89	3.8E-07	1.6E-09
Xe-131m	4.0E-08	1.5E-09
Xe-133m	2.2E-06	1.1E-08
Xe-133	9.4E-05	4.4E-07
Xe-135m	4.1E-06	4.1E-09
Xe-135	1.8E-05	5.6E-08
Xe-137	6.0E-07	2.9E-09
Xe-138	2.1E-06	1.4E-08
Br-83	1.5E-07	7.3E-10
Br-84	3.1E-08	1.9E-10
Br-85	5.2E-10	2.7E-12
I-130	1.7E-08	4.1E-10
I-131	6.9E-06	5.4E-08
I-132	2.0E-06	2.0E-08
I-133	1.0E-05	7.7E-08
I-134	6.3E-07	4.5E-09
I-135	4.9E-06	3.6E-08
Cr-51	1.8E-09	5.6E-11
Mn-54	3.0E-10	1.2E-11
Fe-55	1.6E-09	5.0E-11
Fe-59	9.9E-10	3.8E-11
Co-58	1.5E-08	5.0E-10

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

TABLE 11.1-7 SECONDARY SIDE STEAM EQUILIBRIUM CONCENTRATIONS - ORIGINAL LICENSE BASIS (CONTINUED)

Nuclide	Design (1) (μCi/g)	Expected (2) (μCi/g)
Co-60	1.9E-09	5.6E-11
Rb-86	4.4E-10	4.8E-12
Rb-88	4.6E-07	1.0E-09
Sr-89	3.9E-09	1.3E-12
Sr-90	2.5E-10	2.5E-13
Sr-91	1.5E-09	1.6E-11
Y-90	1.9E-10	5.6E-14
Y-91m	9.3E-10	1.2E-11
Y-91	6.3E-10	1.9E-12
Y-93	2.7E-10	7.8E-13
Zr-95	6.5E-10	2.5E-12
Nb-95	6.7E-10	2.5E-12
Mo-99	3.0E-06	2.6E-09
Tc-99m	2.0E-06	2.5E-09
Ru-103	1.1E-10	1.3E-12
Ru-106	3.0E-11	2.5E-13
Rh-103m	2.9E-10	2.3E-12
Rh-106	3.0E-11	0.0
Te-125m	8.2E-11	6.3E-13
Te-127m	1.9E-09	6.3E-12
Te-127	1.1E-09	2.4E-11
Te-129m	3.6E-08	3.8E-11
Te-129	2.2E-08	6.7E-11
Te-131m	2.0E-08	6.9E-11
Te-131	5.5E-09	2.5E-11
Te-12	2.4E-07	6.5E-10
Cs-134	5.4E-07	1.4E-09

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

TABLE 11.1-7 SECONDARY SIDE STEAM EQUILIBRIUM CONCENTRATIONS - ORIGINAL LICENSE BASIS (CONTINUED)

Nuclide	Design (1) (μCi/g)	Expected (2) (μCi/g)
Cs-136	2.8E-07	7.3E-10
Cs-137	2.7E-06	1.0E-09
Ba-137m	1.4E-06	1.2E-09
Ba-140	4.0E-09	6.3E-12
La-140	4.5E-09	4.7E-12
Ce-141	6.4E-10	2.5E-12
Ce-143	4.5E-10	6.8E-13
Ce-144	4.5E-10	1.2E-12
P-143	6.2E-10	1.3E-12
P-144	4.5E-10	2.6E-12
Np-239	3.4E-09	4.0E-11
H-3	1.3E-03	1.0E-03
Total (Excluding H 3):	1.8E-04	8.3E-07
Total (Including H 3):	1.5E-03	1.0E-03

NOTES:

(1) Based on 1,370 lb/day primary to secondary leak rate.

(2) Based on 100 lb/day primary to secondary leak rate.

(3) 1.6E-06 = 1.6×10^{-6}

Historical, not subject to future updating. This table has been retained to preserve original license basis.

11.2 LIQUID WASTE MANAGEMENT SYSTEMS

In accordance with General Design Criterion 60, liquid waste management systems are provided to control, collect, process, store, recycle, and dispose of liquid radioactive waste generated as the result of normal plant operation, including anticipated operational occurrences. The liquid waste management systems include the radioactive liquid waste system (LWS) and condensate demineralizer liquid waste system (LWC) (which has been removed from service). Figures 11.2–1 and 11.2–2 are the piping and instrumentation diagrams of the radioactive liquid waste system and condensate demineralizer liquid waste systems, respectively.

The boron recovery system (Section 9.3.5) also processes radioactive fluid for ultimate discharge from the plant. The radioactive waste handling aspects of this system are described in this section.

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

The radioactivity values provided in this section are the design basis values used for the design of the liquid waste system. As such, they are considered historical and not subject to future updating. The information is retained to avoid loss of the original design bases. Actual liquid radioactive release quantities can be found in the annual radioactive effluent release reports as submitted to the NRC.

11.2.1 DESIGN BASES

1. The design objectives of the liquid waste management systems are:

    a. To control the releases of radioactive materials within the limits set forth in 10 CFR 20 and to meet the numerical design objectives of Appendix I to 10 CFR 50.

    b. To meet the anticipated processing requirements of the plant. Adequate storage capacity is provided to hold liquid wastes during periods when major processing equipment may be down for maintenance or during periods of excessive waste generation.

2. Table 11.2-1 gives the daily input, in terms of average and peak flows, to the waste management systems. These values are based on the values in NUREG-0017, April 1976.

3. Listings of expected and design case concentrations and annual quantities of radionuclides released to the plant discharge are given in Tables 11.2-6 and 11.2-9.

4. Design data for components in the radioactive liquid waste system and the condensate demineralizer liquid waste system (removed from service) are given in Table 11.2-2.

5. The radioactive liquid waste (LWS) and condensate demineralizer liquid waste systems (LWC) (removed from service) are designated nonnuclear safety (NNS). Equipment is designed and fabricated in conformance with codes and standards identified in Regulatory Guide 1.143 (Section 1.8).

6. The foundation and walls of the radioactive liquid waste building and the condensate polishing facility building are seismically designed in conformance with Regulatory Guide 1.143.

7. A cost-benefit analysis for reducing cumulative dose to the population by using available technology has been performed in accordance with Regulatory Guide 1.110 and is included in Appendix 11A.

8. General Design Criterion 61 applies with regard to provisions for suitable shielding for radiation protection of personnel under normal and postulated accident conditions. Radiation protection criteria for the radioactive liquid waste system are given in Section 12.2.

9. Releases to the environment are monitored prior to discharge. Process and effluent radiological monitoring systems are described in Section 11.5.

10. The following design features are incorporated to reduce maintenance, equipment downtime, liquid leakage, and gaseous releases to the building atmosphere, or otherwise improve radwaste operations:

    a. Dished, sloped, and conical bottoms are used in vessels and tanks with a potential for high activity and suspended solids to minimize buildup of radioactive sludge and facilitate cleaning.

    b. Pressure-retaining components of the system utilize welded construction to the maximum practicable extent. Flanged joints or suitable quick-disconnect fittings are used only where maintenance or operational requirements clearly indicate that such construction is preferable. Screwed connections in which threads provide the only seals are not used except for instrumentation connections where welded connections are not suitable. Process lines are not less than 3/4-inch. Screwed connections backed up by seal welding, socket welding, or mechanical joints are used on lines greater than 3/4-inch but less than 2.5 inch nominal size. For lines of 2.5 inches and above, piping is butt-welded (except as noted by FSAR Table 1.8-1). Backing rings are not used in lines carrying resins or other particulate material. All welding constituting the pressure boundary of pressure-retaining components is performed in accordance with ANSI B31.1.

c. The piping systems are hydrostatically tested. Testing of piping systems is performed in accordance with ANSI B31.1.

d. Pumps handling radioactive liquids are fitted with mechanical seals and outboard restriction bushings to minimize leakage. In the event of a seal failure, the leakage is directed to a radioactive sump through a drain connection.

e. Piping is designed and valves are selected to minimize crud pockets where activity could accumulate.

f. Tanks that are expected to contain liquids of high radioactivity are vented to the aerated vent system (Section 9.3.3) to minimize the potential for gaseous releases into working areas.

g. A centralized control panel is provided to allow system operation and monitoring from one location.

h. Components handling highly radioactive liquids are separated by shield walls to minimize exposure to operators and maintenance personnel.

i. Plastic pipes are not used for radioactive service.

11. The design provisions to control radioactive releases due to overflows from all liquid tanks containing potentially radioactive materials are shown in Table 11.2-3.

11.2.2 SYSTEM DESCRIPTION

11.2.2.1 Radioactive Liquid Waste System (LWS)

The radioactive liquid waste system consists of two separate, but interconnected, portions: The high-level waste portion and the low-level waste portion.

<u>High-Level Waste Portion</u>

Two 26,000-gallon high-level waste drain tanks accept and store high-level radioactive liquid waste from the sources identified in Table 11.2-1. Tank capacity allows time for recirculating and sampling of one tank while the other tank is being filled from any of the above sources. Two waste evaporator feed pumps service either tank. The tanks are cross-connected to the low-level waste drain tanks (described below) at the discharge of the pumps.

Two filters, located downstream of each waste evaporator feed pump, are available to pre-filter high-level waste drain tank contents. These filters may be operated in parallel, in series, or bypassed when recirculating or processing tank contents of a high-level waste drain tank. Subsequent to this flowpath, the effluent from the high-level waste drain tank is processed in one of the following paths: (1) through the high-level radioactive waste filter and demineralizer, (2) in the waste evaporator subsystem, or (3) through a liquid radioactive waste processing system, which can be connected to the Liquid Radioactive Waste system connection points located in the Solid Waste Area of the Waste Disposal Building. The waste evaporator subsystem has been removed from service.

The waste evaporator, which was designed as an alternate path for processing high level waste has been demonstrated by analysis to not be required to operate in order to meet 10 CFR 50 Appendix I Release Criteria. The waste evaporator subsystem has been removed from service.

The waste evaporator is designed with an external reboiler, a large liquid disengaging space, a vapor-liquid separator, and a tray section. These features combine to form a system with extremely high separation factors for nonvolatile nuclides. A decontamination factor of greater than 10^4 for nonvolatile nuclides is expected. The waste evaporator subsystem has been removed from service.

Waste evaporator bottoms are allowed to concentrate until either approximately 15 percent total dissolved and undissolved solids by weight have concentrated or an activity level to be determined by the characteristics of the container used to ship the evaporator bottoms offsite is accumulated. Evaporator bottoms are pumped to the radioactive solid waste system (Section 11.4) via the waste bottoms holding tank. The waste evaporator subsystem has been removed from service.

Effluent from the high level radioactive waste demineralizer, distillate from the waste evaporator, or effluent from a liquid radioactive waste processing system is collected in the waste test tanks. Samples of the liquid are analyzed for radioactivity and chemistry parameters. Depending on analysis results, the liquid is discharged to the circulating water discharge tunnel (Section 10.4.5), or is capable of being recycled to the primary grade water system. The waste evaporator subsystem has been removed from service.

If samples indicate that the distillate is unacceptable for reuse or discharge, it is either passed through a second waste demineralizer and resampled, or sent back to the high-level waste drain tanks for reprocessing. The second waste demineralizer is a mixed bed of ion exchange resins in the H and OH form. The resin is replaced when analysis of influent and effluent samples indicate that the decontamination factor becomes unacceptable or when the radiation level exceeds a predetermined limit.

It is expected that liquid from the waste test tanks is totally discharged. For the purpose of evaluating the radiological impact on the environment, one hundred percent of the input flow is assumed to be discharged. Assurance that waste above activity limits is not inadvertently discharged to the environment is provided through sampling of the waste test tank effluent and by the radiation monitor in the discharge line. This monitor provides audible and visual alarms if activity levels in the effluent exceed limits. An air-operated valve in the discharge line is actuated to terminate the release.

Each batch is isotopically analyzed prior to release and the total activity discharged is recorded. Composite samples are retained in accordance with the procedures outlined in Regulatory Guide 1.21. Detailed administrative records of all radioactive liquid releases are maintained.

Minimum expected decontamination factors are shown on Figure 11.2–3.

Low-Level Waste Portion

Two 4,000-gallon low-level waste drain tanks accept and store low level radioactive liquid waste from the reactor plant aerated drains system (Section 9.3.3), as identified in Table 11.2-1. These tanks and their respective pumps are arranged in the same manner as the high-level waste drain tanks.

Turbine building floor and equipment drains are normally discharged directly to the environment. When there is high radioactivity in the discharge line, flow is diverted to the low-level waste system for further processing.

System design provides for conveying the contents of the low-level waste drain tanks to the high-level waste drain tanks if the activity level of the liquid in the low-level tanks is greater than a predetermined level. Normally, the low-level waste is sampled, analyzed and discharged to the circulating water discharge tunnel. If the particulate concentration is above permit discharge levels, the contents of the low-level waste drain tanks will be recirculated through filter assemblies using the low-level waste drain pump(s) until the particulate concentration levels are acceptable. These filters are cartridge-type filters provided to remove particulate matter from the effluent. Filter elements are changed when the radioactivity level at the filter surface or the pressure drop across the filter exceeds a predetermined value.

Assurance that high-level radioactive waste is not inadvertently discharged to the environment is provided through the analysis of low-level waste drain tank liquid samples and the radiation monitor in the discharge line.

Minimum expected decontamination factors are shown on Figure 11.2–3.

11.2.2.2 Condensate Demineralizer Liquid Waste System (LWC)

The LWC has been removed from service and is no longer used.

This system has been isolated from the plant via locked closed valves, where possible. The system piping remains intact, such that if any leakage occurs across the locked, closed valves, it will be contained within the existing system boundary. The system is interconnected to the Unit 2 LWC at the regenerant evaporator feed tanks, regenerant evaporator feed pump suction and discharge lines, and at the system discharge downstream of the regenerant demineralizer (originally designed to provide added system availability, capacity, and redundancy). These interconnections are isolated also, with locked closed valves, where possible. Electrical power to the system components is administratively controlled by plant procedures.

The LWC was originally designed and installed as an evaporator system to receive and process potentially radioactive liquid waste from the condensate demineralizer-mixed bed system (Section 10.4.6). Evaporator bottoms were designed to be pumped to Unit 2 for processing.

An evaluation has been performed which has determined that the LWC is not needed. This evaluation is documented by change to the Radiological Effluent Monitoring Offsite Dose Calculation Manual (ref. REMODCM CR# 95-7).

Analysis has shown that LWC is not required to be operated to meet 10 CFR 50 Appendix I release criteria.

For the purpose of evaluating the radiological impact on the environment, it is assumed that 100 percent of the regenerant chemical waste is discharged to the environment.

11.2.2.3 Other Systems Discharging Radioactive Liquid Waste

The boron recovery system (Section 9.3.5) receives degasified reactor coolant letdown and reactor plant gaseous drains flow, as identified on Figure 11.2–3. For the purpose of evaluating the radiological impact on the environment, one hundred percent of the input flow is assumed to be discharged. Provisions made to protect against inadvertent discharge are similar to those for high-level radioactive liquid waste, as the boron recovery system distillate is discharged to the liquid waste system upstream of the radiation monitor in the liquid waste system. Minimum expected decontamination factors are shown on Figure 11.2–3.

During open-cycle blowdown, the steam generator blowdown (Section 10.4.8) intermittently discharges directly to the circulating water discharge tunnel as part of the secondary water chemistry control program. For the purpose of evaluating the radiological impact on the environment, the following hypothetical cases have been developed.

For design base case:

- 100 percent discharge to the circulating water discharge tunnel.
- Four steam generators blow down at 1 percent maximum steaming rate (MSR) for 14 days per year during hot standby.
- One steam generator blows down intermittently at 4 percent MSR, and the remaining three steam generators blow down at 1 percent MSR for 5 minutes per week per steam generator.

For expected case:

- 10 percent discharge to the circulating water discharge tunnel.
- Four steam generators each blowdown at 37,000 lb/hr for a total of 148,000 lb/hr.

The actual radioactive releases will be reported annually.

11.2.3 RADIOACTIVE RELEASES

Models and assumptions contained in NUREG-0017 are used to calculate the expected radioactivity concentrations in the liquid discharge. The concentration from each of the parent liquid waste streams following treatment are presented in Tables 11.2-4 and 11.2-7 for the expected and design conditions, respectively. Liquid releases to the environment are listed in Tables 11.2-5 and 11.2-6 for the expected nuclide concentrations prior and subsequent to dilution with the circulating water discharge system. A similar evaluation shown in Tables 11.2-8 and 11.2-9 are for design nuclide concentrations prior and subsequent to dilution with the circulating water discharge system. The diluted release for the expected nuclide concentrations are further analyzed for the environmental impact as described in Appendix I of 10 CFR 50 and in Appendix 11A of this FSAR. Table 11.2-10 presents the design nuclide concentration releases to the unrestricted area in terms of fraction of maximum permissible concentration (MPC) limits described in 10 CFR 20, Appendix B, Table II, Column 2. The results indicate that the sum of the fractions of MPC values does not exceed the limits in 10 CFR 20.

11.2.3.1 Radioactive Liquid Waste System Leak or Failure (Atmospheric Release)

The accident is defined as an unexpected and uncontrolled atmospheric release from the postulated rupture of a boron recovery tank. This tank is the highest potential atmospheric release source term because of its large volume, the relatively high potential for activity in streams feeding the tank, and its location in the yard area.

The atmospheric release due to the postulated rupture of this tank is minimized by prior degasification and demineralization of letdown, which removes essentially all xenon and krypton and most of the iodine from the tank feed. The liquid released from the tank is held in a dike until drained or pumped, under administrative control, to the liquid radioactive waste system for further processing.

The radiological consequences of a postulated radioactive liquid waste system failure resulting atmospheric release is reported in Table 15.0-8 based on design release assumptions in Table 11.2-11, boron recovery tank concentrations in Table 11.2-12 and the X/Q values in Table 15.0-11. The resulting releases are listed in Table 11.2-13. The dose methodology is discussed in Appendix 15A.

In order to bound the reactor power to 3723 MWt (including calorimetric uncertainty), each of the iodine and noble gas isotopes released in Table 11.2-13 were scaled based on factors determined from the ratio of the primary activity concentrations (RCS concentration from Table 15.0-10 and Design concentration from Table 11.1-2) to determine the updated doses for the operating condition.

The radiological consequences are consistent with the guidelines of the pre-1991 version of 10 CFR 20, i.e., the whole body dose does not exceed 500 mRem to an individual at the nearest exclusion area boundary, and is substantially below the guidelines of 10 CFR 100.

11.2.3.2 Liquid Containing Tank Failure

This accident is defined as an unexpected and uncontrolled postulated rupture of the boron recovery tank. The boron recovery tanks are located in the yard area northeast of the containment structure (Figure 1.2–2). This area is provided with dikes to retain any liquid released from a tank rupture. This analysis assumes a combined rupture of a boron recovery tank and leakage into the groundwater.

Description of the analysis of this event is provided in Section 2.4.13.3.

The concentration of radionuclides in Niantic Bay resulting from a liquid containing tank failure are given in Table 11.2-14 based upon assumptions in Table 11.2-11.

In order to bound the reactor power to 3723 MWt (including calorimetric uncertainty), each of the isotopes in Table 11.2-14 were scaled based on factors determined from the ratio of the primary activity concentrations (RCS concentration from Table 15.0-10 and Design concentration from Table 11.1-2) to determine the updated concentrations for the operating condition.

Concentrations in Niantic Bay are within the concentration of 10 CFR 20, Appendix B, Table II as they existed prior to the 1991 revision to 10 CFR 20 (see Section 2.4.13).

11.2.4 REFERENCE FOR SECTION 11.2

TABLE 11.2-1 LIQUID WASTE MANAGEMENT SYSTEM DAILY INPUT FLOWS

NUREG-0017 Liquid Waste Stream Category	Source	Peak Flow Rate (gal/day)	Average Flow Rate (gal/day)
Misc. Liquid Waste System			
Clean Wastes	High Level Waste Drain Tanks		
	Containment Building Sump	300	40
	Auxiliary Building Sump	1,500	200
	Sample Fluids	200	35
	Laboratory Waste	500	400
	Misc. High Level Waste	800	660
Dirty Wastes	Low Level Waste Drain Tanks		
	Misc. Low Level Waste	650	40
Turbine Plant Floor Drains	Turbine Plant Leakage		7,200
Boron Recovery System			
Shim Bleed	Boron Recovery Tank		1,440
Equipment Drains	Boron Recovery Tank		300
Secondary Waste System			
Steam Generator Blowdown	Steam Generators		426,411
Regenerant Chemicals	Regenerant Line		3,400
Detergent Wastes	Laundry Facility		450

TABLE 11.2-2 LIQUID WASTE MANAGEMENT SYSTEM DESIGN DATA

Radioactive Liquid Waste System		
Waste Evaporator Feed Pumps		
Number		2
Capacity (gpm)		35
Design pressure (psig)		350
Design temperature (°F)		250
Material of construction		316 SS
Waste Evaporator Reboiler Pump (removed from service)		
Number		1
Capacity (gpm)		4,000
Design pressure (psig)		600
Design temperature (°F)		350
Material of construction		Alloy 20
Waste Evaporator Bottoms Pump (removed from service)		
Number		1
Capacity (gpm)		15
Design pressure (psig)		350
Design temperature (°F)		250
Material of construction		Incoloy
Waste Distillate Pump (removed from service)		
Number		1
Capacity (gpm)		50
Design pressure (psig)		350
Design temperature (°F)		250
Material of construction		316 SS
Waste Test Tank Pumps		
Number		2
Capacity (gpm)		150
Design pressure (psig)		350
Design temperature (°F)		250

TABLE 11.2-2 LIQUID WASTE MANAGEMENT SYSTEM DESIGN DATA

Material of construction		316 SS
Low-Level Waste Drain Pumps		
Number		2
Capacity (gpm)		50
Design pressure (psig)		350
Design temperature (°F)		250
Material of construction		316 SS
Waste Bottoms Coolant Pump (removed from service)		
Number		1
Capacity (gpm)		120
Design pressure (psig)		350
Design temperature (°F)		250
Material of construction		316 SS
Waste Test Heating Pumps		
Number		2
Capacity (gpm)		60
Design pressure (psig)		350
Design temperature (°F)		250
Material of construction		316 SS
Waste Bottom Holding Tank Pump (removed from service)		
Number		1
Capacity (gpm)		50
Design pressure (psig)		250
Design temperature (°F)		170
Material of construction		Alloy 20
Waste Demineralizer		
Number		1
Capacity (ft^3)		35
Design pressure (psig)		150
Design temperature (°F)		140
Material of construction		304 SS

TABLE 11.2-2 LIQUID WASTE MANAGEMENT SYSTEM DESIGN DATA

High-Level Waste Demineralizer		
Number		1
Capacity (ft^3)		35
Design pressure (psig)		140
Design temperature (°F)		200
Material of construction		304 SS
Waste Evaporator Reboiler (removed from service)		
Number		1
Total duty (Btu/hr)		26,200,000
	Shell Side	Tube Side
Total fluid entering	29,900	1,930,000
Design pressure (psig)	180	100
Operating pressure inlet (psig)	100	25
Temperature in/out (°F)	338 / 338	251.2 / 265.1
Material of construction	Carbon steel	Incoloy 825
Waste Evaporator Bottoms Cooler (3LWS-E2) (removed from service)		
Total duty (Btu/hr)		637,500
	Shell Side	Tube Side
Total fluid entering (lb/hr)	63,750	7,500
Temperature in/out (°F)	140 / 150	255 / 170
Design pressure (psig)	210	170
Operating pressure inlet (psig)	125	50
Material of construction	Carbon steel	Carpenter 20
Waste Bottoms Coolant Preheater (3LWS-E3) (removed from service)		
Total duty (kW)		90
Flow rate (gpm)		150
Design pressure (psig)		210
Temperature in/out (°F)		140 / 185
Material of construction		Carbon steel
Waste Evaporator Condenser (removed from service)		
Total duty (Btu/hr)		22,717,947

TABLE 11.2-2 LIQUID WASTE MANAGEMENT SYSTEM DESIGN DATA

	Shell Side	Tube Side
Total liquid entering (lb/hr)	24,040	1,135,898
Temperature in/out (°F)	250 / 250	95 / 115
Design pressure (psig)	100	150
Material of construction	304 SS	304 SS
Waste Distillate Cooler (removed from service)		
Total duty (Btu/hr)		2,285,500
	Shell Side	Tube Side
Total liquid entering (lb/hr)	17,500	114,275
Temperature in/out (°F	250 / 120	95 / 115
Design pressure (psig)	150	200
Material of construction	304 SS	304 SS
Waste Test Tank Heaters		
Number		2
Total duty (Btu/hr)		10,236
Operating flow (gpm)		60
Design pressure (psig)		50
Material of construction		304 SS
Waste Bottoms Holding Tank Heaters (removed from service)		
Number		2
Design temperature (°F)		200
Design pressure (psig)		10
Total load (kW)		18
Material of construction		Incoloy 825
Waste Evaporator (removed from service)		
Capacity (gpm)		35
Design pressure (psig)		100 and full vacuum
Design temperature (°F)		350
Material of construction		Top, 316 SS Bottom, Incoloy 825
Waste Demineralizer Filter		

TABLE 11.2-2 LIQUID WASTE MANAGEMENT SYSTEM DESIGN DATA

Design flow rate (gpm)		160
Design pressure (psig)		165
Design temperature (°F)		200
Material of construction		Internals, 304 SS
Effluent Filters		
Number		2
Design Flow rate (gpm)		N/A
Design pressure (psig)		150
Design temperature (°F)		N/A
Design pressure drop, clean (psi)		1
Material of construction		Internals, 304 SS
High-Level Waste Recirc Filters		
Number		4
Design Flow rate (gpm)		75
Design pressure (psig)		150
Design temperature (°F)		200
Material of construction		304 SS
High Waste Demineralizer Filter		
Flow rate (gpm)		75
Design pressure (psig)		110
Design temperature (°F)		200
Material of construction		Internal, 304 SS
Low-Level Waste Recirc Filters		
Number		2
Design flow rate (gpm)		60
Design pressure (psig)		150
Design Temperature (°F)		150
Design Pressure drop, clean (psi)		1
Material of construction		Housing 304 L
High-Level Waste Drain Tanks		
Number		2

TABLE 11.2-2 LIQUID WASTE MANAGEMENT SYSTEM DESIGN DATA

Capacity (gal)		26,000
Design pressure (psig)		25
Design temperature (°F)		200
Material of construction		Shell 304 SS
Waste Distillate Tank (removed from service)		
Capacity (gal)		500
Design pressure (psig)		100
Design temperature (°F)		340
Material of construction		Shell, 304 SS
Waste Test Tanks		
Number		2
Capacity (gal)		24,000
Design pressure (psig)		atm and full liquid
Design temperature (°F)		200
Material of construction		Shell, 304 SS
Low-Level Waste Drain Tanks		
Number		2
Capacity (gal)		4,000
Design pressure (psig)		25
Design temperature (°F)		212
Material of construction		Shell, 304 SS
Waste Bottoms Holding Tank (removed from service)		
Number		1
Capacity (gal)		3,000
Design pressure (psig)		atm
Design temperature (°F)		212
Material of construction		Incoloy 825
Condensate Demineralizer Liquid Waste System (removed from service)		
Regenerant Evaporator Feed Tanks (removed from service)		
Number		2
Capacity (gal)		13,000

TABLE 11.2-2 LIQUID WASTE MANAGEMENT SYSTEM DESIGN DATA

Design pressure (psig)		atm
Design temperature (°F)		212
Material of construction		Fiberglass
Regenerant Distillate Tank (removed from service)		
Number		1
Capacity (gal)		500
Design pressure (psig)		100
Design temperature (°F)		340
Material of construction		304 SS
Regenerant Evaporator Feed Pump (removed from service)		
Capacity (gal)		50
Design pressure (psig)		150
Design temperature (°F)		150
Material of construction		316 SS
Regenerant Evaporator Bottoms Pump (removed from service)		
Capacity (gal)		15
Design pressure (psig)		150
Design temperature (°F)		274
Material of construction		A-20 SS
Regenerant Bottoms Coolant Pump (removed from service)		
Capacity (gal)		120
Design pressure (psig)		150
Design temperature (°F)		185
Material of construction		316 SS
Regenerant Distillate Pump (removed from service)		
Capacity (gal)		50
Design pressure (psig)		150
Design temperature (°F)		274
Material of construction		316 SS
Regenerant Evaporator Reboiler Pump (removed from service)		
Capacity (gal)		4,000

TABLE 11.2-2 LIQUID WASTE MANAGEMENT SYSTEM DESIGN DATA

Design pressure (psig)		600
Design temperature (°F)		350
Material of construction		Alloy 20
Regenerant Distillate Cooler (removed from service)		
	Shell Side	Tube Side
Total fluid entering (lb/hr)	17,500	114,275
Design pressure (psig)	150	150
Design temperature (°F)	274	274
Total duty (Btu/hr)	2,285,500	2,285,500
Material of construction	304SS	304SS
Regenerant Evaporator Reboiler (removed from service)		
	Shell Side	Tube Side
Total fluid entering (lb/hr)	23,700	1,930,000
Design pressure (psig)	180	100
Design temperature (°F)	380	350
Total duty (Btu/hr)	21,600,000	21,600,000
Material of construction	Carbon steel	Incoloy 825
Regenerant Evaporator Bottoms Cooler (removed from service)		
	Shell Side	Tube Side
Total fluid entering (lb/hr)	63,750	7,500
Design pressure (psig)	150	150
Design temperature (°F)	274	274
Total duty (Btu/hr)	637,500	637,500
Material of construction	Carbon steel	Carpenter 20
Regenerant Evaporator Condenser (removed from service)		
	Shell Side	Tube Side
Total fluid entering (lb/hr)	24,040	1,135,898
Design pressure (psig)	100	150
Design temperature (°F)	350	350
Total duty (Btu/hr)	22,717,947	22,717,947
Material of construction	304 SS	304 SS

TABLE 11.2-2 LIQUID WASTE MANAGEMENT SYSTEM DESIGN DATA

Regenerant Bottoms Coolant Reheater (removed from service)		
Design flow (gpm)		150
Duty (kW)		90
Design pressure (psig)		175
Design temperature (°F)		185
Material of construction		SA 106
Regenerant Demineralizer (removed from service)		
Design flow rate (gpm)		50
Design pressure (psig)		150
Design temperature (°F)		250
Material of construction		304 SS
Regenerant Evaporator (removed from service)		
Capacity (gpm)		50
Design pressure (psig)		100 and full vacuum
Design temperature (°F)		300
Material of construction		Top, 316 SS Bottom, Incoloy 825
Regenerant Demineralizer Filter (removed from service)		
Flow rate (gpm)		50
Design pressure (psig)		150
Design temperature (°F)		120
Material of construction		304 SS

TABLE 11.2-3 TANK OVERFLOW PROTECTION

Tanks	Mark No.	Level Monitoring and Alarms	Monitoring or Alarm Location [1]	Overflow Provisions	Processing of Overflow
Containment Drains Transfer Tank	3DGS-TK1	Indicator	MCB	Overflows to containment sump	In RLWS
Primary Drain Transfer Tank	3DGS-TK2	Indicator High	MCB	Overflows to auxiliary building sump	In RLWS
Boron Recovery Tanks	3BRS-TK1A/1B	Indicator High	BRP	Overflows to second tank then to diked area	In RLWS if it overflows to dike
Boron Test Tanks	3BRS-TK2A/2B	Indicator High	BRP	Overflows to the waste disposal building sump	In RLWS if it overflows to dike
Boron Distillate Tank	3BRS-TK3	Indicator High	BRP	Closed tank	None
Cation Regeneration Tank	3CND-TK	Indicator	Local	Located in diked area. Overflows to waste neutralization sump	Contents of sumps are discharged to the recirculating water discharge tunnel or, if radioactive, processed in the condensate demineralizer liquid waste system
Anion Regeneration Tank	3CND-TK2	Indicator	Local	Located in diked area. Overflows to waste neutralization sump	Contents of sumps are discharged to the recirculating water discharge tunnel or, if radioactive, processed in the condensate demineralizer liquid waste system

TABLE 11.2-3 TANK OVERFLOW PROTECTION (CONTINUED)

Tanks	Mark No.	Level Monitoring and Alarms	Monitoring or Alarm Location [(1)]	Overflow Provisions	Processing of Overflow
Resin Mix and Storage Tank	3CND-TK3	Indicator	Local	Located in diked area. Overflows to waste neutralization sump	Contents of sumps are discharged to the recirculating water discharge tunnel or, if radioactive, processed in the condensate demineralizer liquid waste system
Recovered Caustic Tank	3CND-TK8	Indicator High	CD	Located in diked area. Overflows to waste neutralization sump	Contents of sumps are discharged to the recirculating water discharge tunnel or, if radioactive, processed in the condensate demineralizer liquid waste system
Recovered Water Tank	3CND-TK9	Indicator High	CD	Located in diked area. Overflows to waste neutralization sump	Contents of sumps are discharged to the recirculating water discharge tunnel or, if radioactive, processed in the condensate demineralizer liquid waste system
Regenerant Evaporator Feed Tanks (removed from service)	3LWC-TK1A/B	Indicator High	LWC	Located in diked area. Overflows to contaminated floor drains	
High-Level Waste Drain Tanks	3LWS-TK1A/1B	Indicator High	LWP	Overflows to the waste disposal building sump	In RLWS
Low-Level Waste Drain Tanks	3LWS-TK4A/4B	Indicator High	LWP	Overflows to the waste disposal building sump	In RLWS

TABLE 11.2-3 TANK OVERFLOW PROTECTION (CONTINUED)

Tanks	Mark No.	Level Monitoring and Alarms	Monitoring or Alarm Location (1)	Overflow Provisions	Processing of Overflow
Waste Distillate Tank (removed from service)	3LWS-TK2	Indicator High	LWP	Closed tank	None
Regenerant Distillate Tank (removed from service)	3LWC-TK3	Indicator High	LWC	Closed tank	None
Waste Test Tanks	3LWS-TK3A/3B	Indicator High	LWP	Overflows to the waste disposal building sump	In RLWS
Spent Resin Dewatering Tank	3WSS-TK1	Indicator High	SWP	To solid waste building sump	In RLWS
Spent Resin Hold Tank	3WSS-TK3	Indicator	SWP	Overflows to spent resin dewatering tank	In RLWS
Waste Bottoms Holding Tank (removed from service)	3LWS-TK5	Indicator High	SWP	Overflows to the waste disposal building sump	
Condensate Surge Tank	3CNS-T2	Indicator High	MCB	Overflows to turbine building floor drains sump	Contents of sump are routed to the radioactive liquid waste system

NOTES:

(1) Location Symbols
BRP = Boron Recovery Panel
MCB = Main Control Board
LWP = Radioactive Waste Panel

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

TABLE 11.2-4 EXPECTED RADIOACTIVE LIQUID CONCENTRATIONS FROM EACH LIQUID RELEASE STREAM (μCI/ML) [1] FOLLOWING TREATMENT

ISOTOPE	BORON RECOVERY (μCi/ml)	MISC. WASTES (μCi/ml)	SECONDARY [2] SYSTEM WASTES (μCi/ml)	TURB BLDG DRAINS (μCi/ml)	TOTAL LWS (μCi/ml)	ADJUSTED [3] TOTAL LWS (μCi/ml)	DETERGENT [4] WASTES (μCi/ml)
CORROSION AND ACTIVATION PRODUCTS							
Cr-51	1.66E-08 [5]	7.05E-07	6.68E-09	0.00E+00	8.77E-09	9.17E-09	0.00E+00
Mn-54	4.16E-09	1.21E-07	1.75E-09	0.00E+00	2.10E-09	2.19E-09	1.61E-06
Fe-55	2.91E-08	6.15E-07	7.08E-09	0.00E+00	8.97E-09	9.36E-09	0.00E+00
F-59	1.25E-08	3.73E-07	4.71E-09	0.00E+00	5.82E-09	6.08E-09	0.00E+00
Co-58	2.16E-07	6.06E-06	6.54E-08	0.00E+00	8.37E-08	8.74E-08	6.43E-06
Co-60	3.74E-08	7.68E-07	8.01E-09	0.00E+00	1.04E-08	1.08E-08	1.40E-05
Zr-95	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	2.25E-06
Nb-95	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	3.21E-06
Np-239	0.00E+00	3.68E-07	3.68E-09	0.00E+00	4.75E-09	4.96E-09	0.00E+00
FISSION PRODUCTS							
Br-83	0.00E+00	3.89E-07	7.00E-09	0.00E+00	8.05E-09	8.41E-09	0.00E+00
Br-84	0.00E+00	1.58E-08	1.83E-09	0.00E+00	1.84E-09	1.94E-09	0.00E+00
Br-85	0.00E+00	0.00E+00	3.36E-11	0.00E+00	3.29E-11	3.29E-11	0.00E+00
Rb-86	4.16E-09	2.79E-07	2.69E-10	0.00E+00	1.17E-09	1.22E-09	0.00E+00
Rb-88	0.00E+00	2.47E-07	9.58E-08	0.00E+00	9.43E-08	9.85E-08	0.00E+00

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

TABLE 11.2-4 EXPECTED RADIOACTIVE LIQUID CONCENTRATIONS FROM EACH LIQUID RELEASE STREAM (μCI/ML) [1] FOLLOWING TREATMENT (CONTINUED)

ISOTOPE	BORON RECOVERY (μCi/ml)	MISC. WASTES (μCi/ml)	SECONDARY [2] SYSTEM WASTES (μCi/ml)	TURB BLDG DRAINS (μCi/ml)	TOTAL LWS (μCi/ml)	ADJUSTED [3] TOTAL LWS (μCi/ml)	DETERGENT [4] WASTES (μCi/ml)
Sr-89	4.16E-09	1.31E-07	1.60E-09	0.00E+00	1.99E-09	2.07E-09	0.00E+00
S-90	0.00E+00	5.26E-09	3.36E-11	0.00E+00	4.93E-11	4.93E-11	0.00E+00
S-91	0.00E+00	1.42E-07	1.43E-09	0.00E+00	1.84E-09	1.94E-09	0.00E+00
Y-90	0.00E+00	0.00E+00	1.68E-11	0.00E+00	1.64E-11	1.64E-11	0.00E+00
Y-91m	0.00E+00	9.47E-08	1.16E-09	0.00E+00	1.43E-09	1.50E-09	0.00E+00
Y-91	0.00E+00	2.63E-08	2.52E-10	0.00E+00	3.29E-10	3.29E-10	0.00E+00
Y-93	0.00E+00	5.26E-09	6.73E-11	0.00E+00	9.86E-11	9.86E-11	0.00E+00
Zr-95	0.00E+00	2.10E-08	3.20E-10	0.00E+00	3.94E-10	4.11E-10	0.00E+00
Nb-95	0.00E+00	2.10E-08	3.53E-10	0.00E+00	4.11E-10	4.27E-10	0.00E+00
Mo-99	4.16E-08	2.62E-05	2.45E-07	2.01E-09	3.21E-07	3.35E-07	0.00E+00
Tc-99m	4.16E-08	2.06E-05	2.44E-07	2.01E-09	3.02E-07	3.16E-07	0.00E+00
Ru-103	0.00E+00	1.58E-08	1.51E-10	0.00E+00	2.14E-10	2.14E-10	2.25E-07
Ru-106	0.00E+00	5.26E-09	3.36E-11	0.00E+00	4.93E-11	4.93E-11	3.86E-06
Rh-103m	0.00E+00	1.58E-08	2.69E-10	0.00E+00	3.12E-10	3.29E-10	0.00E+00
Rh-106	0.00E+00	5.26E-09	6.73E-11	0.00E+00	8.21E-11	8.21E-11	0.00E+00
Ag-110m	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	7.07E-07
Te-125m	0.00E+00	1.05E-08	8.41E-11	0.00E+00	1.15E-10	1.15E-10	0.00E+00

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

TABLE 11.2-4 EXPECTED RADIOACTIVE LIQUID CONCENTRATIONS FROM EACH LIQUID RELEASE STREAM (μCI/ML) [1] FOLLOWING TREATMENT (CONTINUED)

ISOTOPE	BORON RECOVERY (μCi/ml)	MISC. WASTES (μCi/ml)	SECONDARY [2] SYSTEM WASTES (μCi/ml)	TURB BLDG DRAINS (μCi/ml)	TOTAL LWS (μCi/ml)	ADJUSTED [3] TOTAL LWS (μCi/ml)	DETERGENT [4] WASTES (μCi/ml)
Te-127m	4.16E-09	1.05E-07	8.41E-10	0.00E+00	1.17E-09	1.22E-09	0.00E+00
Te-127	4.16E-09	2.37E-07	2.44E-09	0.00E+00	3.14E-09	3.29E-09	0.00E+00
Te-129m	1.25E-08	5.21E-07	4.56E-09	0.00E+00	6.14E-09	6.41E-09	0.00E+00
Te-129	8.31E-09	3.58E-07	7.34E-09	0.00E+00	8.31E-09	8.67E-09	0.00E+00
Te-131m	0.00E+00	7.10E-07	6.31E-09	0.00E+00	8.38E-09	8.76E-09	0.00E+00
Te-131	0.00E+00	1.31E-07	2.52E-09	0.00E+00	2.89E-09	3.02E-09	0.00E+00
Te-132	1.66E-08	8.55E-06	6.12E-08	1.00E-09	8.66E-08	9.04E-08	0.00E+00
I-130	0.00E+00	5.00E-07	4.07E-09	0.00E+00	5.54E-09	5.78E-09	0.00E+00
I-131	6.54E-06	9.30E-05	1.11E-06	4.92E-08	1.40E-06	1.46E-06	9.64E-08
I-132	1.66E-08	1.46E-05	1.97E-07	4.02E-09	2.38E-07	2.49E-07	0.00E+00
I-133	9.97E-08	1.02E-04	7.93E-07	5.83E-08	1.10E-06	1.14E-06	0.00E+00
I-134	0.00E+00	8.78E-07	4.40E-08	0.00E+00	4.57E-08	4.77E-08	0.00E+00
I 135	0.00E+00	3.53E-05	3.46E-07	1.81E-08	4.48E-07	4.69E-07	0.00E+00
Cs-134	4.82E-06	1.05E-04	9.40E-08	1.00E-09	4.40E-07	4.60E-07	2.09E-05
Cs-136	5.82E-07	3.83E-05	4.10E-08	0.00E+00	1.62E-07	1.69E-07	0.00E+00
Cs-137	3.59E-06	7.63E-05	6.84E-08	1.00E-09	3.19E-07	3.34E-07	3.86E-05
Ba-137m	3.35E-06	7.14E-05	1.41E-07	0.00E+00	3.74E-07	3.90E-07	0.00E+00

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

TABLE 11.2-4 EXPECTED RADIOACTIVE LIQUID CONCENTRATIONS FROM EACH LIQUID RELEASE STREAM (μCI/ML) [1] FOLLOWING TREATMENT (CONTINUED)

ISOTOPE	BORON RECOVERY (μCi/ml)	MISC. WASTES (μCi/ml)	SECONDARY [2] SYSTEM WASTES (μCi/ml)	TURB BLDG DRAINS (μCi/ml)	TOTAL LWS (μCi/ml)	ADJUSTED [3] TOTAL LWS (μCi/ml)	DETERGENT [4] WASTES (μCi/ml)
Ba-140	0.00E+00	7.89E-08	6.73E-10	0.00E+00	9.04E-10	9.36E-10	0.00E+00
La-140	0.00E+00	6.31E-08	5.38E-10	0.00E+00	7.23E-10	7.56E-10	0.00E+00
Ce-141	0.00E+00	2.63E-08	3.03E-10	0.00E+00	3.78E-10	3.94E-10	0.00E+00
Ce-143	0.00E+00	1.05E-08	6.73E-11	0.00E+00	9.86E-11	9.86E-11	0.00E+00
C-144	0.00E+00	1.05E-08	1.68E-08	0.00E+00	2.14E-10	2.14E-10	8.36E-06
Pr-143	0.00E+00	1.58E-08	1.35E-10	0.00E+00	1.97E-10	1.97E-10	0.00E+00
Pr-144	0.00E+00	1.05E-08	3.20E-10	0.00E+00	3.61E-10	3.78E-10	0.00E+00

NOTES:

(1) Refer to Figure 11.2–3 for respective steam processing.
(2) Includes steam generator blowdown during open cycle blowdown and regenerate chemical wastes.
(3) Adjusted for releases from anticipated operational occurrences of 0.15 Ci/yr (NUREG-0017).
(4) Detergent waste is conservatively included for flexibility in the event that a laundry is installed in the future.
(5) 1.66E-08 = 1.66×10^{-08}; values less than 1.0E-15 are reported as zero.

Historical, not subject to future updating. This table has been retained to preserve original license basis.

TABLE 11.2-5 EXPECTED ANNUAL RADIOACTIVE LIQUID RELEASES PRIOR TO DILUTION IN THE CIRCULATING WATER DISCHARGE SYSTEM AND PRIOR TO INCLUSION OF ANTICIPATED OPERATIONAL OCCURRENCES (1) (HISTORICAL)

Isotope	Radioactivity Concentration (μCi/gm)	Radioactivity Released (Ci/yr)
CORROSION AND ACTIVATION PRODUCTS		
Cr-51	6.85E-08 (2)	5.34E-03
Mn-54	1.64E-08	1.28E-03
Fe-55	7.01E-08	5.46E-03
Fe-59	4.54E-08	3.54E-03
Co-58	6.53E-07	5.09E-02
Co-60	8.10E-08	6.31E-03
Zr-95	0.00E+00	0.00E+00
Nb-95	0.00E+00	0.00E+00
Np-239	3.71E-08	2.89E-03
FISSION PRODUCTS		
Br-83	6.29E-08	4.90E-03
B-84	1.44E-08	1.12E-03
Br-85	2.57E-10	2.00E-05
Rb-86	9.11E-09	7.10E-04
Rb-88	7.36E-07	5.74E-02
Sr-89	1.55E-08	1.21E-03
Sr-90	3.85E-10	3.00E-05
Sr-91	1.44E-08	1.12E-03
Y-90	1.28E-10	1.00E-05
Y-91m	1.12E-08	8.70E-04
Y-91	2.57E-09	2.00E-04
Y-93	7.70E-10	6.00E-05
Zr-95	3.08E-09	2.40E-04
Nb-95	3.21E-09	2.50E-04
Mo-99	2.51E-06	1.95E-01

TABLE 11.2-5 EXPECTED ANNUAL RADIOACTIVE LIQUID RELEASES PRIOR TO DILUTION IN THE CIRCULATING WATER DISCHARGE SYSTEM AND PRIOR TO INCLUSION OF ANTICIPATED OPERATIONAL OCCURRENCES [1] (HISTORICAL) (CONTINUED)

Isotope	Radioactivity Concentration (μCi/gm)	Radioactivity Released (Ci/yr)
Tc-99m	2.36E-06	1.84E-01
Ru-103	1.67E-09	1.30E-04
Ru-106	3.85E-10	3.00E-05
Rh-103m	2.44E-09	1.90E-04
Rh-106	6.42E-10	5.00E-05
Ag-110m	0.00E+00	0.00E+00
Te-125m	8.98E-10	7.00E-05
Te-127m	9.11E-09	7.10E-04
Te-127	2.45E-08	1.91E-03
Te-129m	4.80E-08	3.74E-03
Te-129	6.49E-08	5.06E-03
Te-131m	6.54E-08	5.10E-03
Te-131	2.26E-08	1.76E-03
Te-132	6.76E-07	5.27E-02
I-130	4.32E-08	3.37E-03
I-131	1.09E-05	8.50E-01
I-132	1.86E-06	1.45E-01
I-133	8.56E-06	6.67E-01
I-134	3.57E-07	2.78E-02
I-135	3.50E-06	2.73E-01
Cs-134	3.44E-06	2.68E-01
Cs-136	1.26E-06	9.86E-02
Cs-137	2.49E-06	1.94E-01
Ba-137m	2.92E-06	2.27E-01
Ba-140	7.06E-09	5.50E-04
La-140	5.65E-09	4.40E-04
Ce-141	2.95E-09	2.30E-04

TABLE 11.2-5 EXPECTED ANNUAL RADIOACTIVE LIQUID RELEASES PRIOR TO DILUTION IN THE CIRCULATING WATER DISCHARGE SYSTEM AND PRIOR TO INCLUSION OF ANTICIPATED OPERATIONAL OCCURRENCES (1) (HISTORICAL) (CONTINUED)

Isotope	Radioactivity Concentration (μCi/gm)	Radioactivity Released (Ci/yr)
Ce-143	7.70E-10	6.00E-05
Ce-144	1.67E-09	1.30E-04
Pr-143	1.54E-09	1.20E-04
Pr-144	2.82E-09	2.20E-04
Total (excluding H-3)	4.30E-05	3.35E+00
H-3	9.24E-03	7.20E+02
Total (including H-3)	9.28E-03	7.23E+02

NOTES:

(1) Detergent wastes not included in this table

(2) 6.85E-08 = 6.85×10^{-8}

(3) Total annual liquid waste release volume (excluding detergent wastes) is 7.79E+10 ml

Historical, not subject to future updating. This table has been retained to preserve original license basis.

TABLE 11.2-6 EXPECTED ANNUAL RADIOACTIVE LIQUID RELEASES AFTER DILUTION IN THE CIRCULATING WATER DISCHARGE SYSTEM AND INCLUSION OF ANTICIPATED OPERATIONAL OCCURRENCES [1] (HISTORICAL)

Isotope	Radioactivity Released (μCi/gm)	Radioactivity Released (Ci/yr)
CORROSION AND ACTIVATION PRODUCTS		
Cr-51	3.41E-12 [2]	5.60E-03
Mn-54	1.40E-12	2.30E-03
Fe-55	3.47E-12	5.70E-03
Fe-59	2.25E-12	3.70E-03
Co-58	3.47E-11	5.70E-02
Co-60	9.12E-12	1.50E-02
Zr-95	8.51E-13	1.40E-03
Nb-95	1.22E-12	2.00E-03
Np-239	1.82E-12	3.00E-03
FISSION PRODUCTS		
Br-83	3.10E-12	5.10E-03
Br-84	7.30E-13	1.20E-03
Br-85	1.22E-14	2.00E-05
Rb-86	4.50E-13	7.40E-04
Rb-88	3.65E-11	6.00E-02
Sr-89	7.91E-13	1.30E-03
Sr-90	1.82E-14	3.00E-05
Sr-91	7.30E-13	1.20E-03
Y-90	6.08E-15	1.00E-05
Y-91m	5.53E-13	9.10E-04
Y-91	1.22E-13	2.00E-04
Y-93	3.65E-14	6.00E-05
Zr-95	1.52E-13	2.50E-04
Nb-95	1.58E-13	2.60E-04
Mo-99	1.22E-10	2.00E-01

TABLE 11.2-6 EXPECTED ANNUAL RADIOACTIVE LIQUID RELEASES AFTER DILUTION IN THE CIRCULATING WATER DISCHARGE SYSTEM AND INCLUSION OF ANTICIPATED OPERATIONAL OCCURRENCES [(1)] (HISTORICAL) (CONTINUED)

Isotope	Radioactivity Released (μCi/gm)	Radioactivity Released (Ci/yr)
Tc-99m	1.16E-10	1.90E-01
Ru-103	1.64E-13	2.70E-04
Ru-106	1.46E-12	2.40E-03
Rh-103m	1.22E-13	2.00E-04
Rh-106	3.04E-14	5.00E-05
Ag-110m	2.68E-13	4.40E-04
Te-125m	4.26E-14	7.00E-05
Te-127m	4.50E-13	7.40E-04
Te-127	1.22E-12	2.00E-03
Te-129m	2.37E-12	3.90E-03
Te-129	3.22E-12	5.30E-03
Te-131m	3.22E-12	5.30E-03
Te-131	1.09E-12	1.80E-03
Te-132	3.34E-11	5.50E-02
I-130	2.13E-12	3.50E-03
I-131	5.41E-10	8.90E-01
I-132	9.12E-11	1.50E-01
I-133	4.26E-10	7.00E-01
I-134	1.76E-11	2.90E-02
I-135	1.76E-10	2.90E-01
Cs-134	1.76E-10	2.90E-01
Cs-136	6.08E-11	1.00E-01
Cs-137	1.40E-10	2.30E-01
Ba-137m	1.46E-10	2.40E-01
Ba-140	3.47E-13	5.70E-04
La-140	2.80E-13	4.60E-04

TABLE 11.2-6 EXPECTED ANNUAL RADIOACTIVE LIQUID RELEASES AFTER DILUTION IN THE CIRCULATING WATER DISCHARGE SYSTEM AND INCLUSION OF ANTICIPATED OPERATIONAL OCCURRENCES (1) (HISTORICAL) (CONTINUED)

Isotope	Radioactivity Released (μCi/gm)	Radioactivity Released (Ci/yr)
Ce-141	1.46E-13	2.40E-04
Ce-143	3.65E-14	6.00E-05
Ce-144	3.22E-12	5.30E-03
Pr-143	7.30E-14	1.20E-04
Pr-144	1.40E-13	2.30E-04
Total (excluding H-3)	2.19E-09	3.60E+00
H 3	4.38E-07	7.20E+02
Total (including H-3)	4.40E-07	7.24E+02

(1) Detergent wastes are included in this table

(2) 3.41E-12 = 3.41 x 10^{12}

(3) Includes source term adjustment for anticipated operational occurrences 1.50E-01 Ci/yr

(4) Total annual liquid waste release volume is 7.86E+10 ml/yr

(5) MP3 average dilution flow of 1840 ft^3/sec or 1.64E+15 ml/yr

Historical, not subject to future updating. This table has been retained to preserve original license basis.

TABLE 11.2-7 DESIGN [1] RADIOACTIVE LIQUID CONCENTRATIONS FROM EACH LIQUID RELEASE STREAM (μCI/GM) FOLLOWING TREATMENT [2]

Isotope	Cont.Bldg. Sump	Aux. Bldg. Sump	Laboratory Waste	Reactor Plant Samples	Misc. Low-Level Waste	Misc. High-Level Waste	Reactor Coolant Bleed	Reactor Plant Gaseous Drains	Regenerant Chemicals	Turbine Bldg. Drains	Steam Generator Blowdown [3]
Cr-51	1.7E-07 [4]	1.7E-08	3.3E-10	1.7E-07	8.7E-09	1.7E-09	8.8E-09	8.8E-08	6.9E-11	1.8E-09	7.2E-07
Mn-54	3.2E-08	3.2E-09	6.5E-11	3.2E-08	3.0E-06	3.2E-10	3.0E-09	3.0E-08	1.2E-11	3.0E-10	1.2E-07
Fe-55	1.7E-07	1.7E-08	3.4E-10	1.7E-07	1.7E-05	1.7E-09	1.7E-08	1.7E-07	6.2E-11	1.5E-09	6.2E-07
Fe-59	9.8E-08	9.8E-09	2.0E-10	9.8E-08	6.3E-06	9.8E-10	6.4E-09	6.4E-08	3.9E-11	1.0E-09	4.0E-07
Co-58	1.6E-06	1.6E-07	3.2E-09	1.6E-06	1.2E-04	1.6E-08	1.2E-07	1.2E-06	6.1E-10	1.5E-08	6.2E-06
Co-60	2.1E-07	2.1E-08	4.2E-10	2.1E-07	2.1E-05	2.1E-09	2.1E-08	2.1E-07	7.6E-11	1.9E-09	7.6E-07
Sr-89	3.9E-07	3.9E-08	7.8E-10	3.9E-07	2.6E-05	3.9E-09	2.7E-08	2.7E-07	1.5E-10	3.9E-09	1.6E-06
Sr-90	1.7E-08	1.7E-09	3.4E-11	1.7E-08	1.7E-06	1.7E-10	1.7E-09	1.7E-08	6.2E-12	1.6E-10	6.2E-08
Sr-91	3.8E-09	3.8E-10	7.6E-12	3.8E-09	1.4E-07	3.8E-11	4.8E-13	4.8E-12	7.9E-12	1.5E-09	6.1E-07
Y-90	1.8E-08	1.8E-09	3.6E-11	1.8E-08	1.7E-06	1.8E-10	1.7E-09	1.7E-08	7.1E-12	1.9E-10	7.5E-08
Y-91M	2.6E-09	2.6E-10	5.2E-12	2.6E-09	9.5E-08	2.6E-11	3.2E-13	3.2E-12	5.4E-12	9.3E-10	3.7E-07
Y-91	6.4E-08	6.4E-09	1.3E-10	6.4E-08	4.5E-06	6.4E-10	4.6E-09	4.6E-08	2.5E-11	6.3E-10	2.5E-07
Y-93	7.4E-10	7.4E-11	1.5E-12	7.4E-10	2.6E-08	7.4E-12	1.3E-13	1.3E-12	1.5E-12	2.6E-10	1.1E-07
Zr-95	6.6E-08	6.6E-09	1.3E-10	6.6E-08	4.8E-06	6.6E-10	4.9E-09	4.9E-08	2.5E-11	6.5E-10	2.6E-07
Nb-95	7.3E-08	7.3E-09	1.5E-10	7.3E-08	6.3E-06	7.3E-10	6.4E-09	6.4E-08	2.7E-11	6.7E-10	2.7E-07
Mo-99	8.0E-05	8.0E-06	1.6E-07	8.0E-05	1.7E-03	8.0E-07	8.5E-07	8.5E-06	8.0E-08	3.0E-06	1.2E-03
Tc-99M	7.6E-05	7.6E-06	1.5E-07	7.6E-05	1.6E-03	7.6E-07	8.2E-07	8.2E-06	7.5E-08	2.0E-06	7.8E-04
Ru-103	3.0E-08	3.0E-09	6.0E-11	3.0E-08	1.8E-06	3.0E-10	1.9E-09	1.9E-08	1.2E-11	3.1E-10	1.2E-07
Ru-106	3.2E-09	3.2E-10	6.4E-12	3.2E-09	3.0E-07	3.2E-11	3.0E-10	3.0E-09	1.2E-12	3.0E-11	1.2E-08
Rh-103M	3.0E-08	3.0E-09	6.0E-11	3.0E-08	1.8E-06	3.0E-10	1.9E-09	1.9E-08	1.2E-11	2.9E-10	1.2E-07
Rh-106	3.2E-09	3.2E-10	6.4E-12	3.2E-09	3.0E-07	3.2E-11	3.0E-10	3.0E-09	1.2E-12	2.7E-11	1.1E-08
Te-125M	8.2E-09	8.2E-10	1.6E-11	8.2E-09	5.8E-07	8.3E-11	5.9E-10	5.9E-09	3.2E-12	8.2E-11	3.3E-08

TABLE 11.2-7 DESIGN [1] RADIOACTIVE LIQUID CONCENTRATIONS FROM EACH LIQUID RELEASE STREAM (μCI/GM) FOLLOWING TREATMENT [2] (CONTINUED)

Isotope	Cont.Bldg. Sump	Aux. Bldg. Sump	Laboratory Waste	Reactor Plant Samples	Misc. Low-Level Waste	Misc. High-Level Waste	Reactor Coolant Bleed	Reactor Plant Gaseous Drains	Regenerant Chemicals	Turbine Bldg. Drains	Steam Generator Blowdown [3]
Te-127M	2.0E-07	2.0E-08	4.0E-10	2.0E-07	1.7E-05	2.0E-09	1.7E-08	1.7E-07	7.6E-11	1.9E-09	7.7E-07
Te-127	1.9E-07	1.9E-08	3.9E-10	1.9E-07	1.6E-05	1.9E-09	1.6E-08	1.6E-07	7.0E-11	1.1E-09	4.5E-07
Te-129M	3.3E-06	3.3E-07	6.7E-09	3.3E-06	1.5E-04	3.3E-08	1.9E-07	1.9E-06	1.4E-09	3.5E-08	1.4E-05
Te-129	2.1E-06	2.1E-07	4.3E-09	2.1E-06	1.2E-04	2.1E-08	1.2E-07	1.2E-06	8.7E-10	2.2E-08	8.7E-06
Te-131M	2.3E-07	2.3E-08	4.5E-10	2.3E-07	5.3E-06	2.3E-09	9.7E-10	9.7E-09	3.5E-10	2.0E-08	8.0E-06
Te-131	4.1E-08	4.1E-09	8.3E-11	4.1E-08	9.9E-07	4.1E-10	1.8E-10	1.8E-09	6.3E-11	5.5E-09	2.2E-06
Te-132	7.5E-06	7.5E-07	1.5E-08	7.5E-06	1.6E-04	7.5E-08	9.0E-08	9.0E-07	6.8E-09	2.4E-07	9.6E-05
Ba-137M	1.6E-04	1.6E-05	3.1E-07	1.6E-04	1.6E-02	1.6E-06	3.9E-04	7.8E-04	1.0E-07	1.4E-06	5.7E-04
Ba-140	3.0E-07	3.0E-08	5.9E-10	3.0E-07	1.0E-05	3.0E-09	9.6E-09	9.6E-08	1.4E-10	4.0E-09	1.6E-06
La-140	2.9E-07	2.9E-08	5.8E-10	2.9E-07	1.0E-05	2.9E-09	1.1E-08	1.1E-07	1.0E-10	1.5E-09	5.9E-07
Ce-141	5.9E-08	5.9E-09	1.2E-10	5.9E-08	3.4E-06	5.9E-10	3.4E-09	3.4E-08	2.4E-11	6.3E-10	2.5E-07
Ce-143	5.6E-09	5.6E-10	1.1E-11	5.6E-09	1.3E-07	5.6E-11	2.8E-11	2.8E-10	8.3E-12	4.5E-10	1.8E-07
Ce-144	4.9E-08	4.9E-09	9.8E-11	4.9E-08	4.5E-06	4.9E-10	4.5E-09	4.5E-08	1.8E-11	4.5E-10	1.8E-07
Pr-143	5.0E-08	5.0E-09	1.0E-10	5.0E-08	1.8E-06	5.0E-10	1.7E-09	1.7E-08	2.3E-11	6.1E-10	2.5E-07
Pr-144	4.9E-08	4.9E-09	9.7E-11	4.9E-08	4.5E-06	4.9E-10	4.5E-09	4.5E-08	1.8E-11	4.5E-10	1.8E-07
Np-239	7.8E-08	7.8E-09	1.6E-10	7.8E-08	1.7E-06	7.8E-10	7.2E-10	7.2E-09	8.5E-11	3.4E-09	1.4E-06
Br-83	5.5E-08	5.5E-09	1.1E-10	5.5E-08	1.6E-06	5.5E-10	0.0	0.0	1.7E-10	1.5E-07	1.5E-05
Br-84	3.1E-13	3.1E-14	0.0	3.1E-13	1.6E-07	3.1E-15	0.0	0.0	4.7E-14	3.1E-08	3.1E-06
Br-85	0.0	0.0	0.0	0.0	2.0E-09	0.0	0.0	0.0	0.0	5.1E-10	5.2E-08
I-130	2.1E-07	2.1E-08	4.2E-10	2.1E-07	6.7E-07	2.1E-09	9.0E-11	9.0E-10	3.1E-10	1.7E-08	1.7E-06
I-131	1.4E-03	1.4E-04	2.9E-06	1.4E-03	3.8E-03	1.4E-05	3.3E-05	3.3E-04	6.0E-07	6.9E-06	6.9E-04
I-132	1.2E-05	1.2E-06	2.4E-08	1.2E-05	1.7E-04	1.2E-07	9.3E-08	9.3E-07	2.3E-08	2.0E-06	2.0E-04
I-133	2.5E-04	2.5E-05	5.0E-07	2.5E-04	6.5E-04	2.5E-06	5.2E-07	5.2E-06	3.3E-07	1.0E-05	1.0E-03

TABLE 11.2-7 DESIGN [1] RADIOACTIVE LIQUID CONCENTRATIONS FROM EACH LIQUID RELEASE STREAM (μCI/GM) FOLLOWING TREATMENT [2] (CONTINUED)

Isotope	Cont.Bldg. Sump	Aux. Bldg. Sump	Laboratory Waste	Reactor Plant Samples	Misc. Low-Level Waste	Misc. High-Level Waste	Reactor Coolant Bleed	Reactor Plant Gaseous Drains	Regenerant Chemicals	Turbine Bldg. Drains	Steam Generator Blowdown [3]
I-134	1.1E-09	1.1E-10	2.2E-12	1.1E-09	3.8E-06	1.1E-11	0.0	0.0	2.2E-11	6.3E-07	6.3E-05
I-135	2.2E-05	2.2E-06	4.3E-08	2.2E-05	1.1E-04	2.2E-07	2.9E-10	2.9E-09	3.8E-08	4.9E-06	4.9E-04
Rb-86	2.0E-08	2.0E-09	4.1E-11	2.0E-08	8.6E-07	2.0E-10	2.1E-08	4.3E-08	1.6E-11	4.4E-10	1.7E-07
Rb-88	0.0	0.0	0.0	0.0	7.5E-06	0.0	0.0	0.0	0.0	4.6E-07	1.8E-04
Cs-134	3.3E-05	3.3E-06	6.6E-08	3.3E-05	3.2E-03	3.3E-07	8.0E-05	1.6E-04	2.2E-08	5.4E-07	2.2E-04
Cs-136	1.1E-05	1.1E-06	2.3E-08	1.1E-05	3.9E-04	1.1E-07	9.4E-06	1.9E-05	1.0E-08	2.7E-07	1.1E-04
Cs-137	1.7E-04	1.7E-05	3.3E-07	1.7E-04	1.7E-02	1.7E-06	4.1E-04	8.3E-04	1.1E-07	2.7E-06	1.1E-03

NOTES:

(1) Design values represent assumptions used to estimate liquid radiological effluents prior to initial plant licensing and are retained for historical purposes only. The Radiological Effluent Monitoring and Offsite Dose Calculation Manual (REMODCM) provide requirements for system operation, dose calculations and monitoring to ensure compliance with 10 CFR 20 Appendix B, Table II, Column 2 effluent limits.

(2) Refer to Figure 11.2–3 for respective stream processing.
Values less than 1.0E-15 are reported as zero.

(3) During open cycle blowdown.

(4) 1.7E-07 = 1.7×10^{-7}

TABLE 11.2-8 DESIGN [1] ANNUAL RADIOACTIVE LIQUID RELEASES PRIOR TO ADDITION OF ANTICIPATED OPERATIONAL OCCURRENCES AND DILUTION IN THE CIRCULATING WATER DISCHARGE SYSTEM

Isotope	Activity Released (μCi/gm)	Activity Released (Ci/yr)
Cr-51	5.5E-07 [2]	1.9E-02
Mn-54	9.3E-08	3.2E-03
Fe-55	4.8E-07	1.6E-02
Fe-59	3.0E-07	1.0E-02
Co-58	4.7E-06	1.6E-01
Co-60	5.9E-07	2.0E-02
Sr-89	1.2E-06	4.1E-02
Sr-90	4.8E-08	1.6E-03
Sr-91	4.5E-07	1.5E-02
Y-90	5.8E-08	2.0E-03
Y-91M	2.8E-07	9.4E-03
Y-91	1.9E-07	6.5E-03
Y-93	7.9E-08	2.7E-03
Zr-95	2.0E-07	6.7E-03
Nb-95	2.1E-07	7.0E-03
Mo-99	9.0E-04	3.0E+01
Tc-99M	5.9E-04	2.0E+01
Ru-103	9.5E-08	3.2E-03
Ru-106	9.2E-09	3.1E-04
Rh-103M	8.9E-08	3.0E-03
Rh-106	8.7E-09	3.0E-04
Te-125M	2.5E-08	8.5E-04
Te-127M	5.9E-07	2.0E-02
Te-127	3.6E-07	1.2E-02
Te-129M	1.1E-05	3.7E-01
Te-129	6.7E-06	2.3E-01
Te-131M	6.0E-06	2.0E-01
Te-131	1.6E-06	5.5E-02
Te-132	7.2E-05	2.4E+00
Ba-137M	4.4E-04	1.5E+01
Ba-140	1.2E-06	4.0E-02

TABLE 11.2-8 DESIGN (1) ANNUAL RADIOACTIVE LIQUID RELEASES PRIOR TO ADDITION OF ANTICIPATED OPERATIONAL OCCURRENCES AND DILUTION IN THE CIRCULATING WATER DISCHARGE SYSTEM (CONTINUED)

Isotope	Activity Released (μCi/gm)	Activity Released (Ci/yr)
La-140	4.6E-07	1.5E-02
Ce-141	1.9E-07	6.5E-03
Ce-143	1.3E-07	4.5E-03
Ce-144	1.4E-07	4.7E-03
Pr-143	1.9E-07	6.3E-03
Pr-144	1.4E-07	4.7E-03
Np-239	1.0E-06	3.4E-02
Br-83	1.1E-05	3.8E-01
Br-84	2.3E-06	7.9E-02
Br-85	3.9E-08	1.3E-03
I-130	1.3E-06	4.3E-02
I-131	5.2E-04	1.8E+01
I-132	1.5E-04	5.0E+00
I-133	7.7E-04	2.6E+01
I-134	4.7E-05	1.6E+00
I-135	3.6E-04	1.2E+01
Rb-86	1.3E-07	4.4E-03
Rb-88	1.4E-04	4.6E+00
Cs-134	1.7E-04	5.6E+00
Cs-136	8.2E-05	2.8E+00
Cs-137	8.3E-04	2.8E+01
H-3	7.5E-02	2.5E+03

NOTES:

(1) Design values represent assumptions used to estimate liquid radiological effluents prior to initial plant licensing and are retained for historical purposes only. The Radiological Effluent Monitoring and Offsite Dose Calculation Manual (REMODCM) provide requirements for system operation, dose calculations and monitoring to ensure compliance with 10 CFR 20 Appendix B, Table II, Column 2 effluent limits.

(2) 5.5E-07 = 5.5×10^{-7}.

Total liquid waste release is 3.4E+10 gm/yr.
Total release (excluding tritium) is 1.7+02 Ci/yr.
Total release concentration (excluding tritium) is 5.1E-03 mCi/gm.

TABLE 11.2-9 DESIGN (1) ANNUAL RADIOACTIVE LIQUID RELEASES FOLLOWING ADDITION OF ANTICIPATED OPERATIONAL OCCURRENCES AND DILUTION IN THE CIRCULATING WATER DISCHARGE SYSTEM

Isotope	Activity Released (μCi/gm)	Activity Released (Ci/yr)
Cr-51	6.3E-12 (2)	1.9E-02
Mn-54	1.1E-12	3.2E-03
Fe-55	5.5E-12	1.6E-02
Fe-59	3.5E-12	1.0E-02
Co-58	5.4E-11	1.6E-01
Co-60	6.8E-12	2.0E-02
Sr-89	1.4E-11	4.1E-02
Sr-90	5.5E-13	1.6E-03
Sr-91	5.2E-12	1.5E-02
Y-90	6.6E-13	2.0E-03
Y-91M	3.2E-12	9.4E-03
Y-91	2.2E-12	6.5E-03
Y-93	9.0E-13	2.7E-03
Zr-95	2.3E-12	6.7E-03
Nb-95	2.4E-12	7.0E-03
Mo-99	1.0E-08	3.0E+01
Tc-99M	6.7E-09	2.0E+01
Ru-103	1.1E-12	3.2E-03
Ru-106	1.1E-13	3.1E-04
Rh-103M	1.0E-12	3.0E-03
Rh-106	1.0E-13	3.0E-04
Te-125M	2.9E-13	8.5E-04
Te-127M	6.7E-12	2.0E-02
Te-127	4.1E-12	1.2E-02
Te-129M	1.2E-10	3.7E-01
Te-129	7.6E-11	2.3E-01
Te-131M	6.8E-11	2.0E-01
Te-131	1.9E-11	5.5E-02
Te-132	8.2E-10	2.4E+00
Ba-137M	5.0E-09	1.5E+01
Ba-140	1.4E-11	4.0E-02
La-140	5.2E-12	1.5E-02

TABLE 11.2-9 DESIGN [1] ANNUAL RADIOACTIVE LIQUID RELEASES FOLLOWING ADDITION OF ANTICIPATED OPERATIONAL OCCURRENCES AND DILUTION IN THE CIRCULATING WATER DISCHARGE SYSTEM (CONTINUED)

Isotope	Activity Released (μCi/gm)	Activity Released (Ci/yr)
Ce-141	2.2E-12	6.5E-03
Ce-143	1.5E-12	4.5E-03
Ce-144	1.6E-12	4.8E-03
Pr-143	2.1E-12	6.3E-03
Pr-144	1.6E-12	4.7E-03
Np-239	1.2E-11	3.5E-02
Br-83	1.3E-10	3.8E-01
Br-84	2.6E-11	7.9E-02
Br-85	4.4E-13	1.3E-03
I-130	1.5E-11	4.3E-02
I-131	5.9E-09	1.8E+01
I-132	1.7E-09	5.0E+00
I-133	8.7E-09	2.6E+01
I-134	5.3E-10	1.6E+00
I-135	4.1E-09	1.2E+01
Rb-86	1.5E-12	4.4E-03
Rb-88	1.6E-09	4.6E+00
Cs-134	1.9E-09	5.6E+00
Cs-136	9.4E-10	2.8E+00
Cs-137	9.4E-09	2.8E+01
H-3	8.6E-07	2.5E+03

NOTES:

(1) Design values represent assumptions used to estimate liquid radiological effluents prior to initial plant licensing and are retained for historical purposes only. The Radiological Effluent Monitoring and Offsite Dose Calculation Manual (REMODCM) provide requirements for system operation, dose calculations and monitoring to ensure compliance with 10CFR20 Appendix B, Table II, Column 2 effluent limits.

(2) 6.3E-12 = 6.3×10^{-12}

Anticipated operational occurrences = 1.5E-01 Ci/yr.
Dilution release rate is 2.97E+15 gm/yr.
Total release (excluding tritium) is 1.7E+02 Ci/yr.
Total release concentration (excluding tritium) is 5.8E-08 μCi/gm.

TABLE 11.2-10 FRACTION OF MPC RELEASED - DESIGN CASE (1) (HISTORICAL)

Isotope	Conc (μCi/cc)	MPC (μCi/cc)	Fraction MPC Released
Cr-51	6.3E-12 (1)	2.0E-03	3.1E-09
Mn-54	1.1E-12	1.0E-04	1.1E-08
Fe-55	5.5E-12	8.0E-04	6.9E-09
Fe-59	3.5E-12	5.0E-05	7.0E-08
Co-58	5.4E-11	9.0E-05	6.0E-07
Co-60	6.8E-12	3.0E-05	2.3E-07
Sr-89	1.4E-11	3.0E-06	4.7E-06
Sr-90	5.5E-13	3.0E-07	1.8E-06
Sr-91	5.2E-12	5.0E-05	1.0E-07
Y-90	6.6E-13	2.0E-05	3.3E-08
Y-91M	3.2E-12	3.0E-03	1.1E-09
Y-91	2.2E-12	3.0E-05	7.3E-08
Y-93	9.0E-13	3.0E-05	3.0E-08
Zr-95	2.3E-12	6.0E-05	3.8E-08
Nb-95	2.4E-12	1.0E-04	2.4E-08
Mo-99	1.0E-08	4.0E-05	2.5E-04
Tc-99M	6.7E-09	3.0E-03	2.2E-06
Ru-103	1.1E-12	8.0E-05	1.4E-08
Ru-106	1.1E-13	1.0E-05	1.1E-08
Rh-103M	1.0E-12	1.0E-02	1.0E-10
Te-125M	2.9E-13	1.0E-04	2.9E-09
Te-127M	6.7E-12	5.0E-05	1.3E-07
Te-127	4.1E-12	2.0E-04	2.0E-08
Te-129M	1.2E-10	2.0E-05	6.0E-06
Te-129	7.6E-11	8.0E-04	9.5E-08
Te-131M	6.8E-11	4.0E-05	1.7E-06
Te-132	8.2E-10	2.0E-05	4.1E-05
Ba-140	1.4E-11	2.0E-05	7.0E-07

TABLE 11.2-10 FRACTION OF MPC RELEASED - DESIGN CASE (1) (HISTORICAL)

Isotope	Conc (μCi/cc)	MPC (μCi/cc)	Fraction MPC Released
La-140	5.2E-12	2.0E-05	2.6E-07
Ce-141	2.2E-12	9.0E-05	2.4E-08
Ce-143	1.5E-12	4.0E-05	3.7E-08
Ce-144	1.6E-12	1.0E-05	1.6E-07
Pr-143	2.1E-12	5.0E-05	4.2E-08
Np-239	1.2E-11	1.0E-04	1.2E-07
Br-83	1.3E-10	3.0E-06	4.3E-05
I-130	1.5E-11	3.0E-06	5.0E-06
I-131	5.9E-09	3.0E-07	2.0E-02
I-132	1.7E-09	8.0E-06	2.1E-04
I-133	8.7E-09	1.0E-06	8.7E-03
I-134	5.3E-10	2.0E-05	2.6E-05
I-135	4.1E-09	4.0E-06	1.0E-03
Rb-86	1.5E-12	2.0E-05	7.5E-08
Cs-134	1.9E-09	9.0E-06	2.1E-04
Cs-136	9.4E-10	6.0E-05	1.6E-05
Cs-137	9.4E-09	2.0E-05	4.7E-04
H-3	8.6E-07	3.0E-03	2.9E-04
Totals	9.1E-07	2.4E-02	3.1E-02

NOTE

(1) 1. 6.3E-12 = 6.3×10^{-12}

Total concentration (excluding tritium) is 5.1E-08 μCi/gm.

Total fraction of MPC released (excluding tritium) is 3.07E-02.

Historical, not subject to future updating. This table has been retained to preserve original license basis.

TABLE 11.2-11 ASSUMPTIONS USED FOR THE RADIOACTIVE LIQUID WASTE SYSTEM FAILURE (RELEASE TO ATMOSPHERE) AND FOR THE LIQUID CONTAINING TANK FAILURE

	Design
Tank with Assumed Highest Radionuclide Inventory	Boron Recovery Tank
Tank Volume (gal)	150,000
Fill Fraction of Tank	0.8
Reactor Coolant Feed Rate to Tank (gpm)	75
Degasification Fraction Upstream to Tank	1.0
Source Stream Feeding Tank	Reactor Coolant (Table 11.1-2)
Fraction of Volatile Nuclides Released to Atmosphere	
Noble Gases	1.0
Halogens	0.1
Dilution Factor of Ground Water to Niantic Bay	73
Transition Time to Niantic Bay (years) (1)	6.64
Dilution Factor at Point of Entry to Niantic Bay	13,052
Dilution Factor at 1,000 Feet from Point of Entry to Bay	32,151

NOTE:

1. Transit time to Niantic Bay is the travel time of the ground water. Credit for sorption in soil was not assumed to this analysis.

TABLE 11.2-12 BORON RECOVERY TANK CONCENTRATIONS (μCI/CC)

Isotope	Design (μCi/cc)	Isotope	Design (μCi/cc)	Isotope	Design (μCi/cc)
Kr-83m	0.13E-03 (1)	Y-94	0.39E-08	Te-127	0.19E-04
Xe-131m	0.90E-03	Y-95	0.12E-08	Te-129m	0.43E-03
Xe-133m	0.13E-03	Zr-95	0.79E-05	Te-129	0.42E-03
Xe-133	0.24E-02	Zr-97	0.28E-05	Te-131m	0.20E-03
Xe-135m	0.25E-02	Nb-95m	0.16E-06	Te-131	0.41E-04
Xe-135	0.61E-02	Nb-95	0.83E-05	Te-132	0.27E-02
Br-83	0.13E-03	Nb-97m	0.27E-05	Te-133m	0.14E-04
Br-84	0.13E-04	Nb-97	0.31E-05	Te-133	0.15E-04
Br-85	0.17E-06	Mo-99	0.33E-01	Te-134	0.14E-04
Br-87	0.32E-07	Mo-101	0.24E-05	Cs-134	0.20E+00
I-129	0.53E-09	Mo-102	0.12E-05	Cs-136	0.10E+00
I-131	0.28E-01	Mo-105	0.95E-09	Cs-137	0.10E+01
I-132	0.37E-02	Tc-99m	0.27E-01	Cs-138	0.16E-01
I-133	0.31E-01	Tc-101	0.46E-05	Cs-139	0.45E-03
I-134	0.32E-03	Tc-102	0.16E-05	Cs-140	0.54E-05
I-135	0.85E-02	Tc-105	0.12E-07	Cs-142	0.22E-08
I-136	0.92E-07	Ru-103	0.38E-05	Ba-137m	0.93E+00
Se-81	0.11E-09	Ru-105	0.81E-07	Ba-139	0.52E-03
Se-83	0.24E-09	Ru-106	0.37E-06	Ba-140	0.48E-04
Se-84	0.17E-10	Ru-107	0.61E-11	Ba-141	0.23E-07
Rb-88	0.33E-01	Rh-103m	0.38E-05	Ba-142	0.22E-07
Rb-89	0.85E-03	Rh-105m	0.81E-07	La-140	0.23E-04
Rb-90	0.40E-03	Rh-105	0.76E-06	La-141	0.67E-06
Rb-91	0.64E-05	Rh-106	0.37E-06	La-142	0.79E-09
Rb-92	0.28E-07	Rh-107	0.21E-09	La-143	0.21E-08
Sr-89	0.61E-04	Sn-127	0.21E-08	Ce-141	0.77E-05
Sr-90	0.19E-05	Sn-128	0.21E-08	Ce-143	0.45E-05
Sr-91	0.10E-04	Sn-130	0.14E-10	Ce-144	0.56E-05
Sr-92	0.13E-05	Sb-127	0.15E-06	Ce-145	0.59E-10

TABLE 11.2-12 BORON RECOVERY TANK CONCENTRATIONS (μCI/CC) (CONTINUED)

Isotope	Design (μCi/cc)	Isotope	Design (μCi/cc)	Isotope	Design (μCi/cc)
Sr-93	0.46E-08	Sb-128	0.97E-11	Ce-146	0.10E-08
Sr-94	0.87E-10	Sb-129	0.87E-07	Pr-143	0.75E-05
Y-90	0.23E-05	Sb-130	0.29E-09	Pr-144	0.56E-05
Y-91m	0.65E-05	Sb-131	0.26E-08	Pr-145	0.56E-06
Y-91	0.78E-05	Sb-132	0.27E-10	Pr-146	0.58E-08
Y-92	0.28E-05	Sb-133	0.45E-10	Nd-147	0.26E-05
Y-93	0.18E-05	Te-127m	0.24E-04	Nd-149	0.18E-07
Nd-151	0.12E-09	Sm-151	0.44E-09	Mn-56	0.47E-04
Pm-147	0.93E-06	Sm-153	0.13E-06	Fe-55	0.57E-04
Pm-149	0.98E-06	Cr-51	0.11E-04	Fe-59	0.12E-04
Pm-151	0.29E-06	Mn-54	0.88E-05	Co-58	0.30E-03
				Co-60	0.86E-05

NOTE:

(1) 0.13E-03 = 0.13×10^{-3}

TABLE 11.2-13 ACTIVITY RELEASED TO ATMOSPHERE FROM A RADIOACTIVE LIQUID CONTAINING TANK FAILURE (BORON RECOVERY TANK)

Isotope	Radioactivity Released (Ci)
Kr-83m	5.90E-02 (1)
Xe-131m	4.08E-01
Xe-133m	5.81E-02
Xe-133	1.08E+00
Xe-135m	1.15E+00
Xe-135	2.77E+00
I-131	1.28E+00
I-132	1.67E-01
I-133	1.40E+00
I-134	1.45E-02
I-135	3.84E-01

NOTE:

(1) 5.90E-02 = 5.90×10^{-2}

TABLE 11.2-14 RADIOACTIVE CONCENTRATIONS IN GROUNDWATER ENTERING NIANTIC BAY FOLLOWING A RUPTURE OF BORON RECOVERY TANK

Isotope	Concentration at Entry Point (μCi/cc)	Concentration 1000 Feet from Entry Point (μCi/cc)
I-129	*	*
Sr-90	1.72E-12**	*
Y-90	1.72E-12	*
Ru-106	3.99E-15	*
Rh-106	3.99E-15	*
Te-127m	*	*
Te-127	*	*
Cs-134	2.27E-08	7.07E-13
Cs-137	9.22E-07	2.86E-11
Ba-137m	8.48E-07	2.64E-11
Ce-144	1.60E-14	*
Pr-144	1.60E-14	*
Pm-147	1.75E-13	*
Sm-151	*	*
Mn-54	4.34E-14	*
Fe-55	1.09E-11	*
Co-58	*	*
Co-60	3.77E-12	*
H-3	2.54E-06	7.90E-11

NOTES:

* Nuclide concentration less than 1.0E-15, μCi/cc

** 1.72E-12 = 1.72×10^{-12}

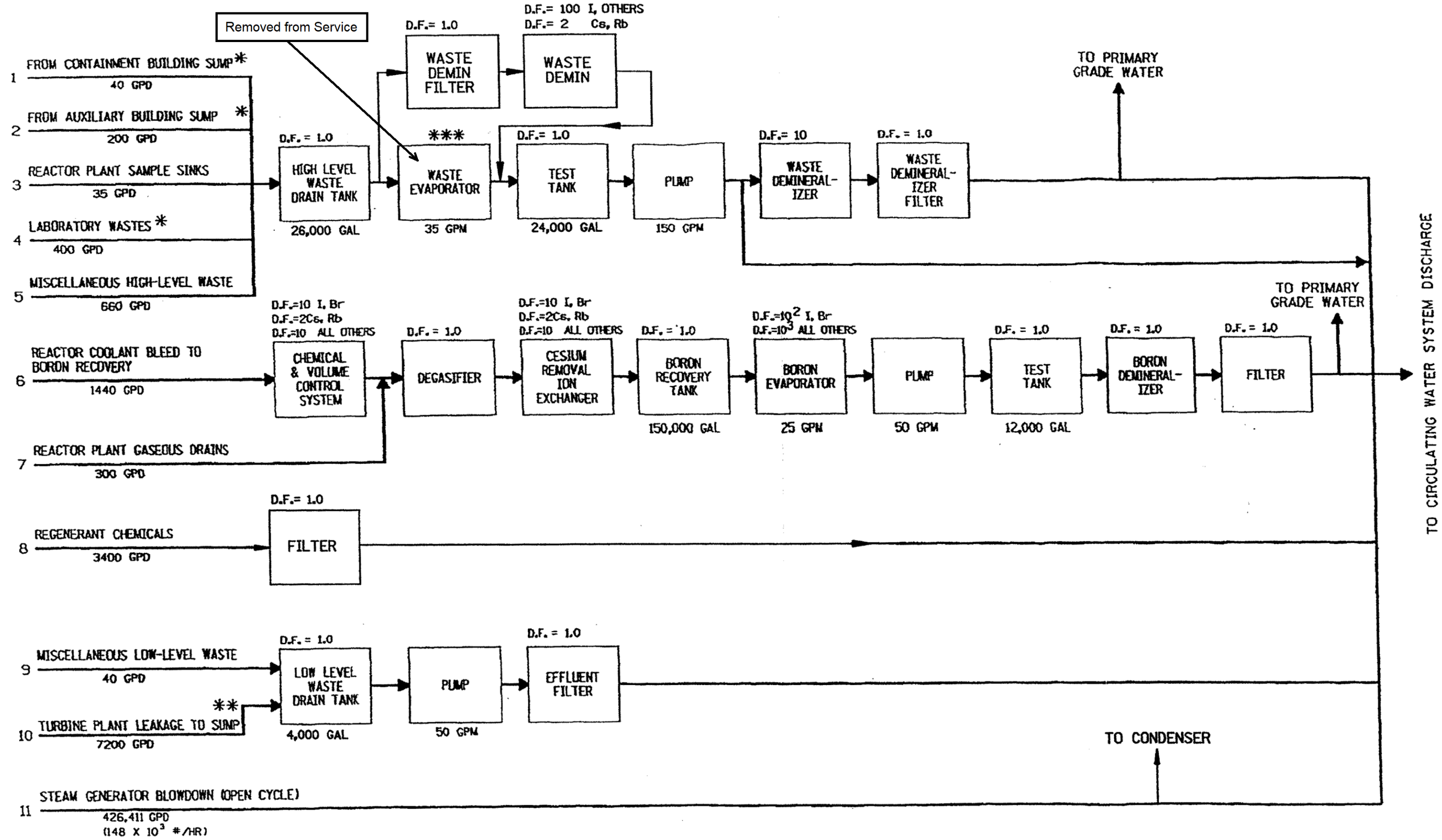
FIGURE 11.2–1 P&ID RADIOACTIVE LIQUID WASTE AND AERATED DRAINS (SHEETS 1-3)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-3 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 11.2–2 CONDENSATE DEMINERALIZER LIQUID WASTE

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-3 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 11.2–3 EXPECTED RADIOACTIVE LIQUID WASTE SOURCE AND DISCHARGE PATHS



NOTES

* — VIA REACTOR PLANT AERATED DRAINS SYSTEM.

** — NORMALLY DISCHARGED DIRECTLY TO ENVIRONMENT.

*** — NOT REQUIRED TO MEET 10 CFR 50 APPENDIX I GUIDELINES.

D.F. — DECONAMINATION FACTORS ARE CONSISTENT WITH NUREG-0017, SECTION 2.2.21.

GPD — ESTIMATED OR EXPECTED FLOWS ARE PROVIDED FOR ILLUSTRATION OF PROCESSED WASTES. FLOW ARE NOT NECESSARILY CONSISTENT WITH ACTUAL OPERATING FLOW DATA.

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11.3 GASEOUS WASTE MANAGEMENT SYSTEMS

This section describes the capabilities of Millstone 3 to control, collect, process, store, and dispose of gaseous radioactive waste generated from normal operation and anticipated operational occurrences. Section 11.5 describes the process and effluent radiation monitoring systems. The reactor plant gaseous and aerated vents systems are described in Section 9.3.3. The gaseous waste management systems include the radioactive gaseous waste system and ventilation systems.

The radioactive gaseous waste system (Figure 11.3–1, Sheets 1 and 2) consists of three subsystems: the degasification subsystem, the process gas (hydrogenated) subsystem, and low activity process vent (aerated) subsystem.

In the degasification subsystem, the fluid from the reactor coolant letdown stream (CHS), or alternately, from the reactor plant gaseous drains system (DGS), is sent to a degasifier, where noncondensable fission product gases are removed. The remaining liquid may be transferred to the volume control tank (CHS) or to the boron recovery system (BRS). The normal flowpath is from the reactor coolant letdown to the volume control tank. The gases are forwarded to the process gas portion of GWS.

In the process gas (hydrogenated) subsystem, the noncondensable fission product gas stream is first dehydrated. Then, radioactive iodine is removed and the activity of the radioactive xenon and krypton is reduced. Finally, the gas is released into the process vent portion of GWS.

In the low activity process vent (aerated) subsystem, aerated and hydrogenated gas streams from various plant inputs (including the process gas portion of GWS) are collected, dehydrated, and discharged to the reactor plant ventilation system (HVR) for release to the environment via the Millstone stack. The gas streams are monitored for radioactivity prior to release.

In a separate flow path, relief effluents from the degasifier, the waste evaporator, and the boron evaporator are collected and discharged to the reactor plant ventilation system (HVR) for release to the environment via the turbine building stack. The waste evaporator is removed from service.

The radioactivity values provided in this section are the design basis values used for the design of the Gaseous Waste System. As such, they are considered historical and not subject to future updating. The information is retained to avoid loss of the original design bases. Actual airborne radioactivity release quantities can be found in the annual radioactive effluent release reports as submitted to the NRC.

11.3.1 DESIGN BASES

11.3.1.1 Design Objective

The objective of the gaseous waste management system is to process and control the release of radioactive gaseous effluents to the site environs so as to maintain the exposure to radioactive gaseous effluents of persons in unrestricted areas to as low a level as is reasonably achievable (Appendix I to 10 CFR 50, May 5, 1975). This is to be accomplished while also maintaining occupational exposure as low as reasonably achievable and without limiting plant operation or availability.

11.3.1.2 Design Criteria

The design of the radioactive gaseous waste system and the ventilation systems meet the following criteria.

1. Section 11.1 discusses the design basis source terms. Expected radioactive gaseous effluents from all sources (Table 11.3-1) have been calculated using the data shown in Table 11.3-2. These values are consistent with NUREG-0017, April 1976.

2. The systems have the capability to meet the requirements of 10 CFR 20 and the dose design objectives specified in Appendix I to 10 CFR 50, including provisions to treat gaseous radioactive wastes such that:

    a. The calculated annual total quantity of all radioactive material released from Millstone 3 to the atmosphere does not result in an estimated annual external dose from gaseous effluents to any individual in unrestricted areas in excess of 5 millirems to the total body or 15 millirems to the skin.

    b. The calculated annual total quantity of all radioactive iodine and radioactive material in particulate form released from Millstone 3 to the atmosphere does not result in an estimated annual dose or dose commitment from such radioactive iodine and radioactive material in particulate form for any individual in an unrestricted area from all pathways of exposure in excess of 15 mRem to any organ.

    c. The concentrations of radioactive materials in gaseous effluents released to an unrestricted area do not exceed the limits in 10 CFR 20, Appendix B, Table 2, Column 1.

3. The radioactive gaseous waste system is designed to meet the anticipated processing requirements of the plant. Adequate capacity is provided to process gaseous wastes during periods when major processing equipment may be down for maintenance (single failures) and during periods of excessive waste generation.

4. The system design contains provisions to control leakage and to facilitate operation and maintenance in accordance with the guidelines of Regulatory Guide 1.143 (formerly Branch Technical Position ETSB 11-1, Rev. 1) (Section 1.8).

5. The radioactive gaseous waste system meets General Design Criteria 60 and 64 of Appendix A to 10 CFR 50 as discussed in Sections 3.1.2.60 and 3.1.2.64.

11.3.1.3 Cost Benefit Evaluation

Appendix 11A shows that the systems contain all items of reasonably demonstrated technology that affect a reduction in dose to the population reasonably expected to be within a 50-mile radius of the plant with a favorable cost benefit.

11.3.1.4 Equipment Design Criteria

Table 11.3-3 lists the radioactive gaseous waste system major equipment items. This list includes materials, rates process conditions, and number of units supplied. Equipment and piping are designed and constructed in accordance with the requirements of the applicable codes (Table 11.3-4).

Table 3.2–1 shows the safety classes of the various systems. Seismic category, safety class, quality assurance requirements, and principal construction codes information is contained in Section 3.2. The system is designed to Safety Classification NNS.

The design of the system precludes an explosive mixture from accumulating. Since the system operates above atmospheric pressure, in-leakage cannot occur. Instrumentation with automatic alarm functions monitors the concentrations of hydrogen and oxygen in portions of the system having the potential for containing explosive mixtures.

The radioactive gaseous waste system processes letdown from the chemical and volume control system (CHS) or reactor plant gaseous drain through the degasifier at a maximum rate of 150 gpm. Maximum letdown from the CHS is 120 gpm and the maximum flow rate from the gaseous drains system is no more than 150 gpm.

The following design features are incorporated to minimize maintenance, equipment downtime, leakage, and radioactive gaseous releases. These features facilitate radwaste operation, and assist in maintaining occupational exposures as low as is reasonably achievable.

1. Redundant degasifier recirculation pumps prevent degasifier outage due to pump failure.

2. Components requiring servicing are placed in individual shielded cubicles to minimize personnel exposure during maintenance.

3. Leakage from pumps is piped to sumps.

4. The radioactive gaseous waste system can be operated locally from the radioactive gaseous waste and process gas treatment control panels.

Conservative analyses of the radioactive gaseous waste system, presented in Section 15.2, demonstrate that equipment failure results in doses well within the guidelines of 10 CFR 100.

11.3.1.5 Building Ventilation Systems

Figure 11.3–2 gives a composite diagram of ventilation systems which may release radioactivity during normal operations. These systems are:

1. Fuel building ventilation (Section 9.4.1)
2. Auxiliary building ventilation (Section 9.4.2)
3. Turbine building ventilation (Section 9.4.3)
4. Containment structure ventilation (Section 9.4.6)
5. Engineered safety features building ventilation (Section 9.4.4)
6. Waste disposal building ventilation (Section 9.4.8)
7. Service building ventilation (Section 9.4.11)

11.3.2 SYSTEM DESCRIPTIONS

Figure 11.3–1, Sheets 1 & 2 show the piping and instrumentation drawings (P&IDs) of the radioactive gaseous waste system and Figure 9.3–5 shows reactor plant gaseous drains.

The appropriate subsections of Section 9.4 provide specific component data with P&IDs for ventilation systems subject to radioactive release.

11.3.2.1 Radioactivity Inputs and Release Points

Radioactivity in process streams is processed by the radioactive gaseous waste system. Ventilation releases for building housing systems which could potentially be radioactive during normal operations are:

1. Ventilation vent
    a. Containment
    b. Auxiliary building*
    c. Fuel building*
    d. Waste disposal building*
    e. Service building*

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*For calculation purposes, releases from the fuel building, waste disposal building, service building, and engineered safety features building are combined with auxiliary building releases, in accordance with NUREG-0017, April 1976.

2. Engineered safety features building*

3. Millstone stack

   a. Radioactive gaseous waste system

   b. Main condenser air ejector

4. Turbine Building

   a. Roof exhausters

   b. Steam generator blowdown flash tank vent

5. Condensate polishing building

   a. Turbine gland sealing system exhaust

Radioactivity releases are provided in Tables 11.3-5 through 11.3-10.

Building volumes and expected flow rates are provided in Section 9.4.

11.3.2.2 Degasifier Subsystem of Radioactive Gaseous Waste System

Reactor coolant letdown, containing dissolved hydrogen and fission gases, is normally directed to the degasifier from the letdown line upstream of the volume control tank in the chemical and volume control system. Alternately, liquid collected by the reactor plant gaseous drains system (Section 9.3.3) may be also directed to either the degasifier or the boron recovery system. Dissolved gases are separated from the liquid in the degasifier.

The degasifier processes reactor coolant letdown continuously except as needed to process gaseous (hydrogenated) drains. However, reactor coolant letdown may bypass the degasifier, if desired. The degasifier design flow of 150 gpm exceeds the maximum expected throughput for the liquid portion of the process gas subsystem. Separation of dissolved gases at all reactor coolant letdown flow rates is thus ensured. The degasifier operates at approximately 2 psig. If the degasifier is not operating, it may be placed in either the standby mode or in a shutdown condition.

11.3.2.3 Process Gas Subsystem of Radioactive Gaseous Waste System

The process gas (hydrogenated) portion of the radioactive gaseous waste system is designed to treat gases stripped from in the reactor coolant letdown and the reactor plant gaseous (hydrogenated) drains (Figure 11.3–1).

Effluent gases from the degasifier contain primarily hydrogen and water vapor. A small amount of nitrogen and traces of xenon, krypton, argon, carbon, and iodine are also present in the effluent gases. These gases and any hydrogenated gas stream from the reactor plant gaseous vent header are dehumidified (dew point 35°F) in one of the two redundant process gas refrigerant dryers.

Condensation effluent from the dryers is returned to the suction of the degasifier recirculation pump. The dry stream is passed through and filtered by the ambient temperature process gas charcoal bed adsorbers and one of two redundant HEPA filters. The heat due to radioactive decay is small and does not affect the adsorption of noble gases on the charcoal. The charcoal bed adsorbers are designed to provide holdup of most krypton and xenon isotopes long enough in comparison with their half-lives so that passage through the beds will result in the effective removal of these isotopes. In addition, decontamination of iodine to negligible levels is obtained during passage through the charcoal beds. The charcoal is divided evenly between two vertical tanks in series. The tanks are piped so that either one may be bypassed, if necessary, with a corresponding decrease in decay time; however, bypass is not anticipated. The only radioisotope present in any quantity in the predominantly hydrogen stream after the decay period is krypton-85. This processed hydrogenated stream is monitored by the supplementary leak collection monitor and released to the environment through the Millstone stack in accordance with technical specifications. The normal flow path for the processed stream is to the Millstone stack. When a test of the gas at the vent dampers at the discharge of the process vent fans is required, then the process gas flow is monitored and released through the reactor plant ventilation vent.

Liquid seals are provided in the drain lines from the process gas precoolers and the process gas water traps to prevent backflow of vapor from the degasifier. The process gas precooler drains continuously at approximately 3 pounds per hour of water at 120°F. The process gas water trap drains at approximately 0.3 pound per hour at 35°F. In the event of a sudden vacuum condition at the suction of the degasifier recirculation pump, the seals are temporarily lost. However, the process gas precooler drain seal refills in approximately 10 minutes and the process gas water trap seal refills in approximately 100 minutes.

A process gas monitor is provided for the gaseous releases from the process gas portion to monitor radioactivity release to the environment and to automatically isolate the flow from the process gas receiver when a predetermined level is exceeded (Section 11.5).

11.3.2.4 Process Vent Portion of Radioactive Gaseous Waste System

The process vent (aerated) portion of the radioactive gaseous waste system is designed to collect the low activity aerated gas stream from the following sources:

1. Reactor plant aerated vents system
2. Reactor plant gaseous vent system
3. Condenser air removal system
4. Containment vacuum system
5. Low activity effluent from the process gas (hydrogenated) portion of the radioactive gaseous waste system
6. Radioactive gaseous waste system component purges

7. Boron recovery system relief valve discharge
8. Liquid waste system relief valve discharges
9. Degasifier relief valve discharge

The process vents, with the exception of the relief valve vent header, are monitored by the supplementary leak collection monitor and discharged to the Millstone stack. The process vent portion is operated continuously, unless required to be shutdown for maintenance. The relief valve vent header is discharged to the ventilation vent where it is monitored and released without filtration.

11.3.2.5 Steam and Power Conversion System

The main condenser is evacuated by steam jet air ejectors (Section 10.4.2). Air ejector exhaust is monitored and discharged to the Millstone stack.

The turbine gland seal steam condenser (Section 10.4.3) exhausts directly to the atmosphere. These releases are given in Tables 11.3-1 and 11.3-11 for expected and design cases, respectively.

During the intermittent and hot standby blowdown (open cycle), the steam generator flash tank is vented directly to the atmosphere through a vent on the turbine building roof. Releases are given in Tables 11.3-7 and 11.3-10.

The quantity of steam released during steam dumps to the atmosphere is provided in Section 10.3. Actual unit trips are expected to be less than those assumed in NUREG-0017 (2 turbine trips per year).

11.3.2.6 System Instrumentation Requirements

11.3.2.6.1 Radioactive Gaseous Waste System

The process gas radiation monitor is located downstream of the process gas charcoal bed adsorbers prior to discharge into the process vent portion. This monitor automatically isolates flow from the process gas receiver. Radioactive releases from the process vent portion of the radioactive gaseous waste system are monitored by the supplementary leak collection monitor. The radiation monitoring system is discussed in Section 11.5.

Hydrogen and oxygen analyzers are provided to measure the process gas composition and to alarm before a dangerous concentration exists.

Temperature and moisture detectors and alarms are provided downstream of the process gas refrigerant dryer to assure proper moisture control in the process gas charcoal bed adsorbers.

11.3.2.6.2 Ventilation Systems

Radiation monitoring of ventilation system effluents is discussed in Section 11.5. Instrumentation requirements are discussed in Section 9.4.

11.3.2.7 Seismic Design Provisions of the Radioactive Gaseous Waste System

The radioactive gaseous waste system has not been provided with special seismic design. However, the entire portion of the system that provides for treatment of gases stripped from the reactor coolant is located in the auxiliary building, a Seismic Category I building (Section 3.8). The ventilation in the auxiliary building system (Section 9.4.2) has the capability of detecting radioactive gas leakage and filtering prior to release. This is an alternate method of meeting the design guidance given in Regulatory Guide 1.143.

11.3.2.8 Quality Control

A program is established to ensure that the design, construction, and testing requirements are met. The following areas are included in the program.

1. Design and Procurement Document Control - Procedures are established to ensure that requirements are specified and included in design and procurement documents and that deviations therefrom are controlled.

2. Inspection - A program for inspection of activities affecting quality is established and executed by or for the organization performing the activity to verify conformance with the documented instructions, procedures, and drawings for accomplishing the activity.

3. Handling, Storage, and Shipping - Procedures are established to control the handling, storage, shipping, cleaning, and preservation of material and equipment in accordance with work and inspection instructions to prevent damage or deterioration.

4. Inspection, Test, and Operating Status - Procedures are established to provide for the identifications of items which have satisfactorily passed required inspections and tests.

5. Corrective Action - Procedures are established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, deviations, defective material and equipment, and nonconformances, are promptly identified and corrected.

11.3.2.9 Welding

All welding constituting the pressure boundary of pressure retaining components is performed by qualified welders employing qualified welding procedures according to Table 11.3-4. Nonconsumable weld inserts are prohibited in process lines unless they are ground out after the weld is completed.

11.3.2.10 Materials

Materials for pressure retaining components of process systems are selected from those covered by the material specifications listed in Section II, Part A of the ASME Boiler and Pressure Vessel Code, except that malleable, wrought, cast iron materials, or plastic pipe is not used. The components meet all of the mandatory requirements of the material specifications with regard to manufacture, examination, repair, testing, identification, and certification.

A description of the major process equipment, including the design temperature and pressure and the materials of construction, is given in Table 11.3-3.

11.3.2.11 Construction of Process Systems

Pressure retaining components of process systems use welded construction to the maximum practicable extent. Process piping systems include the first root valve on sample and instrument lines. Process lines are not less than 3/4-inch nominal pipe size. Sample and instrument lines are not considered as portions of the process systems. Flanged joints or suitable rapid disconnect fittings are not used except where maintenance requirements clearly indicate that such construction is preferable. Screwed connections in which threads provide the only seal are not used. Screwed connections backed up by seal welding or mechanical joints are used only on lines of 3/4-inch nominal pipe size.

In lines 3/4-inch or greater, but less than 2 1/2-inch nominal pipe size, socket type welds are used. In lines 2 1/2-inch nominal pipe size and larger, pipe welds are of the butt joint type.

11.3.2.12 System Integrity Testing

Completed process systems are pressure tested to the maximum practicable extent. Piping systems are hydrostatically tested in their entirety, using available valves for pressure test boundaries (lines vented to atmosphere do not require a pressure test). Hydrostatic testing of piping systems is performed at a pressure 1.2 times the design pressure and held for a minimum of 30 minutes with no leakage indicated. Pneumatic testing may be substituted for hydrostatic testing in accordance with the applicable codes.

11.3.3 RADIOACTIVE RELEASES

Tables 11.3-5 thru 11.3-10 give the calculated sources of radioactive nuclide inventory released via gaseous effluents. Table 11.3-1 gives the expected radioactive gaseous isotope releases from each release point assumed in terms of curies per year per nuclide. Table 11.3-11 provides the design releases for these release points.

Table 11.3-2 lists the parameters in these calculations.

A summary of the estimated expected annual radioactivity doses is presented in Appendix 11A. A summary of design release concentrations at the site boundary, maximum permissible concentration (MPC) and Fraction of MPCs is presented in Tables 11.3-8, 11.3-9 and 11.3-10.

The “design” releases are within the limits of 10 CFR 20. The doses from “expected” releases are within the numerical design objectives of Appendix I of 10 CFR 50. Atmospheric diffusion and ground deposition factors used in the dose calculations are discussed in Section 2.3.5.

Gaseous effluents are discharged to the environment through the following release points:

1. The reactor plant ventilation vent
2. The turbine building roof exhausters
3. The Millstone stack
4. The engineered safety features building exhaust
5. The steam generator blowdown flash tank vent
6. The turbine gland seal steam condenser exhaust on the condensate polishing building roof

Release points are identified on Figure 11.3–2.

Exhausts from the auxiliary building, the fuel building, the waste disposal building, the containment purge, the service building, and the gaseous waste process vent are released from the reactor plant ventilation vent. This vent is located on the turbine building 133 feet above grade and 157 feet-0 inches above sea level. The base elevation is 75 feet above grade. The square vent cross-sectional dimensions are 10 feet by 10 feet and the discharge velocity is 3,000 feet per minute. The maximum discharge temperature is 104°F. Containment purge is considered an intermittent release. Others are assumed as continuous.

The turbine building ventilation system exhausts through the turbine building roof exhausters, located on the turbine building roof. The exhausters are approximately 114 feet above grade, 138 feet above sea level, and have a base elevation of 107 feet above grade. The system exhausts through roof-mounted, mushroom-type hoods with dimensions of 14 feet-6 inches by 14 feet-6 inches. The discharge velocity is approximately 900 feet per minute and the maximum temperature is 104°F. These releases are assumed to be continuous releases.

During the open cycle blowdown, the steam generator blowdown flash tank is vented directly to the atmosphere through a vent on the turbine building roof. The vent is a 10-inch diameter pipe attached to a roof-mounted silencer. The discharge elevation is approximately 113 feet above sea level. Exhaust flow velocity is 8,450 feet per minute.

The turbine gland sealing system exhaust is vented to the atmosphere at a point above the condensate polishing building. The vent is a 10-inch diameter pipe at an elevation of 72 feet above sea level.

The reactor plant aerated vents, the reactor plant gaseous vents, condenser air removal effluent, radioactive gaseous waste system discharges, steam generator blowdown tank condenser vent, and containment vacuum pump discharge are released through the Millstone stack. The Millstone stack is 375 feet above grade, 389 feet above sea level, and has a circular orifice with a 7 foot inside diameter. The stack discharges at a maximum velocity of 332 feet per minute for accident condition and 40 feet per minute for normal condition (MP2 & MP3) at a maximum temperature of 104°F. These discharges are assumed as continuous releases. The portion of the discharge vent between the ESF building and the stack is underground.

The engineered safety features building ventilation system exhausts through the ESF building vent located on the east wall. The vent is 12 feet-9 inches above grade, 36 feet-9 inches above sea level. The vent cross sectional dimensions are 3 feet-0 inches by 10 feet-0 inches and the discharge velocity is 350 feet per minute. The maximum discharge temperature is 104°F. The release from the ESF building is considered continuous and is included as part of the continuous release from the auxiliary building in Table 11.3-5.

11.3.3.1 Radioactive Gaseous Waste System Failure

A gaseous waste system failure is postulated to produce a unique unplanned release by a pathway not normally used for planned releases and requiring a reasonable time to detect and take remedial action to terminate the release. An inadvertent bypass of the process gas charcoal bed adsorbers with continued normal operation of the gaseous waste system for one hour is assumed. This source is then assumed to be continually released to the auxiliary building and then directly to the environment without holdup.

The gaseous waste system design, described in Section 11.3, precludes the single failure of an active component from causing this event, but it is postulated as the design basis accident as prescribed by Branch Technical Position ETSB 11-5. For this event to occur the two manual bypass valves must fail or be improperly aligned, and the piping system downstream of the adsorbers must experience a passive failure to result in a release directly into the auxiliary building.

The process gas charcoal bed adsorber tanks are designed in accordance with ASME III, Class 3, and have a 335 psig design pressure. Because the process gas charcoal bed adsorber normal internal pressure is approximately 1 psig, a gross rupture of the tanks is not considered credible.

Radiation monitors are provided for the ventilation system in order to detect the release of noble gases to the environs from the building ventilation system.

However, no credit is taken for normal operating plant systems, instrumentation or controls, nor operation of any engineered safety feature systems to mitigate the consequences of this event.

Normally, all noncondensable gases are removed from the reactor coolant letdown stream in the degasifier. For this analysis it is assumed that the activity released following the bypass of process gas charcoal bed adsorbers consists of noble gas activities discharged from the degasifier for 60 minutes and released without benefit of delay in charcoal beds. It is also assumed that a fraction of the noble gases adsorbed in the charcoal beds is released. The fractions released were calculated using the model developed for the Fast Flux Test Facility (Underhill). Table 11.3-13 lists the following:

1. Maximum radioisotope inventory in the process gas charcoal bed adsorbers and associated piping.

2. Fraction of the charcoal bed radioisotope inventory released.

3. Sixty-minute discharge from the degasifier.

4. Total radioisotope release.

Table 11.3-12 gives the assumptions used for analyzing the postulated bypass of the process gas charcoal bed adsorbers.

The radiological consequences of bypassing the charcoal bed adsorbers are listed in Table 15.0-8 based on design release assumptions in Table 11.3-12, the releases in Table 11.3-13 and the X/Q values in Table 15.0-11. The dose methodology of Appendix 15A is used here.

In order to bound the reactor power to 3723 MWt (including calorimetric uncertainty), each of the noble gas isotopes released in Table 11.3-13 were scaled based on factors determined from the ratio of the primary activity concentrations (RCS concentration from Table 15.0-10 and Design concentration from Table 11.1-2) to determine the updated doses for the operating condition.

This event will not cause a Condition IV event as defined in Section 15.0.1. The radiological consequences are well within the guidelines of 10 CFR 100.

11.3.4 REFERENCE FOR SECTION 11.3

TABLE 11.3-1 TOTAL EXPECTED RADIOACTIVE GASEOUS RELEASED TO ATMOSPHERE FROM MILLSTONE 3 (HISTORICAL)

Nuclide	Millstone Stack (Ci/yr)	Ventilation Vent (Ci/yr)	Turbine Building (Ci/yr)	Condensate Polishing Building (1) (Ci/yr)	Total (Ci/yr)
Kr-83m	2.9E-01 (2)	4.7E-01	3.1E-05	1.5E-04	7.6E-01
K-85m	1.2E+00	2.0E+00	1.3E-04	6.5E-04	3.2E+00
Kr-85	2.4E+02	1.5E+00	3.0E-06	1.5E-05	2.4E+02
Kr-87	8.7E-01	1.4E+00	9.2E-05	4.6E-04	2.3E+00
Kr-88	2.5E+00	4.0E+00	2.7E-04	1.3E-03	6.5E+00
Kr-89	8.2E-02	1.3E-01	8.7E-06	4.3E-05	2.1E-01
Xe-131m	2.4E-01	9.6E-01	8.1E-06	4.0E-05	1.2E+00
Xe-133m	5.4E-01	1.9E+00	5.9E-05	3.0E-04	2.4E+00
Xe-133	2.3E+01	1.5E+02	2.4E-03	1.2E-02	1.7E+02
Xe-135m	2.1E-01	3.4E-01	2.2E-05	1.1E-04	5.5E-01
Xe-135	2.9E+00	5.0E+00	3.0E-04	1.5E-03	7.9+00
Xe-137	1.5E-01	2.3E-01	1.6E-05	7.8E-05	3.8-01
Xe-138	7.2E-01	1.1E+00	7.6E-05	3.8E-04	1.8E+00
I-131	1.9E-02	4.6E-02	7.3E-03	2.2E-04	7.3E-02
I-133	2.8E-02	6.8E-02	9.9E-03	3.1E-04	1.1E-01
Co-58	0.0	6.4E-02	0.0	0.0	6.4E-02
Co-60	0.0	2.9E-02	0.0	0.0	2.9E-02
Mn-54	0.0	1.9E-02	0.0	0.0	1.9E-02
Fe-59	0.0	6.4E-03	0.0	0.0	6.4E-03
Sr-89	0.0	1.4E-03	0.0	0.0	1.4E-03
Sr-90	0.0	2.1E-04	0.0	0.0	2.1E-04
Cs-134	0.0	1.9E-02	0.0	0.0	1.9E-02
Cs-137	0.0	3.2E-02	0.0	0.0	3.2E-02
C-14	7.0E+00	1.0E+00	0.0	0.0	8.0E+00
Ar-41	0.0	2.5E+01	0.0	0.0	2.5E+01
H-3	0.0	7.3E+02	0.0	0.0	7.3E+02

NOTES:

(1) Releases from turbine gland sealing system exhaust.

(2) 2.9E-01 = 2.9×10^{-1}.

Historical, not subject to future updating. Has been retained to preserve original license basis.

TABLE 11.3-2 RADIOACTIVE GASEOUS SOURCE TERM PARAMETERS (HISTORICAL)

Plant Capacity Factor	0.8
Fuel Defects (%)	0.12 (Expected) 1 (Design)
Containment Building:	
1. Noble gas release to containment building (fraction/day of primary coolant activity)	0.01
2. Iodine release to containment building (fraction/day of primary coolant activity)	10^{-5}
3. Purge exhaust ventilation rate (cfm)	35,000
4. Purge exhaust ventilation time per purge (hr)	8
5. Containment air filtration subsystem recirculation rate during purge (cfm)	12,000 (1)
6. Charcoal iodine adsorber depth (in)	4
7. Iodine exhaust filter efficiency (%)	95
8. Particulate exhaust filter efficiency (%)	95
9. Number of cold purges/year	4
10. Continuous ventilation exhaust rate (cfm)	0
11. Free containment volume (cu ft)	2.32×10^6
Containment Internal Cleanup System:	
1. Operates prior to purging cold shutdown (hr)	16
2. Operates prior to purging hot shutdown (hr)	16
3. Mixing efficiency (%)	70
4. Containment air filtration subsystem recirculation rate prior to purge (cfm)	12,000 (1)
5. Charcoal iodine adsorber depth (in)	4
6. Iodine filter efficiency (%)	90
7. Particulate filter efficiency (%)	90
Auxiliary Building:	
1. Iodine exhaust filter efficiency (%) (2)	0
2. Particulate exhaust filter efficiency (%) (2)	0
3. Primary coolant leakage rate into building (lb/day)	160

TABLE 11.3-2 RADIOACTIVE GASEOUS SOURCE TERM PARAMETERS (HISTORICAL) (CONTINUED)

4. Iodine partition factor	0.0075
Turbine Building:	
1. No special design to collect valve leakage)	
2. Steam leakage (lb/hr)	1,700
Main Condenser/Air Ejector:	
1. Volatile iodine/total iodine in primary system	0.05
2. Volatile iodine is treated as noble gas in steam generator	
3. Primary to secondary leak rate (lb/day)	1370 (Design) 100 (Expected)
4. MC/AE volatile iodine partition factor	0.15 (Design)
5. Charcoal iodine adsorber depth (in)	0.0
6. Iodine exhaust filter efficiency (%)	0.0
7. Particulate exhaust filter efficiency (%)	0.0
8. Volatile iodine condenser bypass fraction	0.35
Steam Generator Blowdown:	
1. Flash tank vented to the atmosphere during open cycle blowdown	
2. Flash tank iodine partition factor	0.05
3. Hypothetical flow assumptions (see Section 11.2.2.3 for time periods for each release)	
Hot Standby:	150,520 lb/hr
1% MSR from four steam generators (37,630 lb/hr per steam generator)	
Intermittent Blowdown:	263,410 lb/hr
1% MSR from three steam generators (37,630 lb/hr per steam generator)	
4% MSR from one steam generator (150,520 lb/hr)	
Turbine Gland Sealing System Exhaust:	
1. Total steam flow rate	8,460 lb/hr
2. Iodine partition factor	0.15
Radioactive Gaseous Waste System (Process Gas System):	
1. Letdown flow to degasifier (lb/hr)	35,900
2. Holdup time prior to charcoal beds (minutes)	7.41

TABLE 11.3-2 RADIOACTIVE GASEOUS SOURCE TERM PARAMETERS (HISTORICAL) (CONTINUED)

3. Krypton dynamic adsorption coefficient (cm^3/gm)	6.3
4. Xenon dynamic adsorption coefficient (cm^3/gm)	146
5. System flow rate (scfm) expected/maximum	0.3/3
6. Total mass of charcoal in beds (lb x 1,000)	27
(No iodine or particulates are released from system)	
7. Krypton holdup time in delay bed (hr)	147
8. Xenon holdup time in delay bed (hr)	3,410
(Complete degasification is handled by the same equipment as normal operation)	

NOTES:

(1) This is the flow rate for one fan unit. The containment air filtration subsystem includes two fan units, which operate simultaneously.

(2) Filters installed but not used during normal operation. Capability exists to filter exhaust upon high activity in building.

Data and Assumptions from NUREG-0017, April 1976.

Historical, not subject to future updating. This table has been retained to preserve original license basis.

TABLE 11.3-3 RADIOACTIVE GASEOUS WASTE SYSTEM

			Parameters	
Process Gas Charcoal Bed Absorbers				
Number			2	
Vessel pressure				
Operating (psig)			1	
Design (psig)			335	
Vessel temperature				
Operating (psig)			104	
Design (psig)			150	
Total weight - one unit				
Empty			16,650	
Operating			30,150	
Construction material			Stainless steel	
Degasifier				
Number			1	
Capacity (gpm)			150	
Pressure				
Operating (psig)			2	
Design (psig)			150 + full vac.	
Temperature				
Operating (°F)			219	
Design (°F)			366 and 100	
Construction material			Stainless steel	
Process Gas Receiver				
Number			1	
Pressure				
Operating (°F)			Approximately Atmospheric	

TABLE 11.3-3 RADIOACTIVE GASEOUS WASTE SYSTEM (CONTINUED)

Design (°F)			1,800	
Temperature				
Operating (°F)			120	
Design (°F)			450	
Total weight				
Empty (lb)			1,755	
Full (lb)			1,762	
Construction material			Stainless steel	
Degasifier Recovery Exchangers		Shell Side		Tube Side
Number			2	
Total duty (Btu/hr)			6,300,000	
Total liquid entering (lb/hr)		75,000		75,000
Pressure				
Operating - inlet (psig)		100		150
Design (psig)		300		300
Temperature				
In (°F		219		115
Out (°F)		135		199
Construction material		Stainless steel		Stainless steel
Degasifier Feed Preheater		Shell Side		Tube Side
Number			1	
Total duty (Btu/hr)			4,600,000	
Total fluid entering (lb/hr)		5,400		75,000
Pressure				
Operating - inlet (psig)		145		50

TABLE 11.3-3 RADIOACTIVE GASEOUS WASTE SYSTEM (CONTINUED)

Design (psig)		300 + full vac.		300 + full vac.
Temperature				
In (°F		363		199
Out (°F)		363		260
Construction material		Carbon steel		Stainless steel
Degasifier Condenser		Shell Side		Tube Side
Number			1	
Total duty (Btu/hr)			3,219,000	
Pressure				
Operating - inlet (psig)		125		2
Design (psig)		185		50 + full vac.
Temperature				
In (°F		95		219
Out (°F)		116		190
Construction material		Carbon steel		Stainless steel
Process Gas Compressor (Abandoned in Place, Currently Bypassed)				
Number		2		
Capacity (scfm)		3		
Pressure - suction				
Minimum (psig)		14.7		
Maximum (psig)		240		
Pressure				
Discharge (psig)		75		
Design (psia)		1,800		

TABLE 11.3-3 RADIOACTIVE GASEOUS WASTE SYSTEM (CONTINUED)

Discharge temperature Leaving aftercooler, max (°F)		110		
Process Gas Compressor Aftercooler				
Number		2		
Gas flow (cfm)		2.4		
Gas temperature				
Inlet (°F		408		
Outlet (°F)		105		
Cooling water flow (gpm)		2		
Cooling water temperature				
Inlet (°F		95		
Outlet (°F)		100		
Pressure				
Operating (psig)		75		
Design (psia)		1,800		
Process Gas Prefilter	Max	Design	Refueling	Purge
Number			2	
Pressure				
Operating (psig)			1	
Design (psia)			335	
Temperature				
Operating (psig)			104	
Design (psia)			150	
Filtration efficiency (%) (based on DOP test)	99.97	99.97	99.97	99.97
Process Gas Prefilter	Max	Design	Refueling	Purge
Flow rate (scfm)	8	9	3	20
Pressure drop (in H_20)	0.5	0.5	4	1

TABLE 11.3-3 RADIOACTIVE GASEOUS WASTE SYSTEM (CONTINUED)

Process Gas H2/02 Analyzers	H2 Analyzer		02 Analyzer	
Pressure				
Operating - inlet (psig)	12		12	
Minimum operating (psig)	2		2	
Maximum operating (psig)	40		40	
Design (psig)	40		40	
Relief valves set (psig)	40		40	
Design Temperature				
In (°F)	104		104	
Out (°F)	104		104	
Analyzers Required Flow (cc/min)				
normal	100-125		100-125	
maximum	150		150	
Degasifier Trim Cooler		Shell Side		Tube Side
Number			1	
Total duty (Btu/hr)			1,500,000	
Temperature				
In (°F)		95		135
Out (°F)		110		115
Pressure				
Operating inlet (psig)		125		85
Design (psig)		175		300
Construction material		Carbon steel		Stainless steel

TABLE 11.3-3 RADIOACTIVE GASEOUS WASTE SYSTEM (CONTINUED)

Process Gas Precooler		Shell Side		Tube Side
Number			2	
Total duty (Btu/hr)			12,900	
Total fluid entering (lb/hr)		27.1		2,600
Pressure				
Operating inlet (psig)		1		125
Design (psig)		335		200
Temperature				
In (°F)		190		95
Out (°F)		120		100
Design (°F)		190		120
Number of passes		1		2
Construction material		Stainless steel		Stainless steel
Process Gas Glycol Chiller		Refrigerant Tube Side		Process Gas Tube Side
Number		2		
Total fluid entering (lb/hr)				50.7
Pressure				
Inlet (psig)		20		1
Design (psig)		150		335
Temperature				
In (°F)		34		120
Out (°F)		30		35
Construction material		Stainless steel		Stainless steel
Process Gas Water Trap				
Number		2		
Pressure				

TABLE 11.3-3 RADIOACTIVE GASEOUS WASTE SYSTEM (CONTINUED)

Operating (psig)		1		
Design (psig)		335		
Temperature				
Operating (°F)		35		
Design (°F)		150		
Construction material		Stainless steel		
Degasifier Recirculation Pump				
Number		2		
Capacity (gpm) @ 386 ft head		150		
Pressure				
Operating-discharge (psig)		186		
Design (psig)		200		
Temperature				
Operating (°F)		218		
Design (°F)		250		
Construction material		Stainless steel		
Process Vent Fans				
Number		2		
Capacity (cfm) @ 20 in water		180		
Operations pressure - discharge (psig)		0.4		
Operating temperature (°F)		50		
Outlet air velocity (fpm)		1,939		
Process Vent Cooler		Shell Side		Tube Side
Number		1		
Total duty (Btu/hr)		148,828		
Total fluid entering (lb/hr)		14,828		752.5
Pressure				

TABLE 11.3-3 RADIOACTIVE GASEOUS WASTE SYSTEM (CONTINUED)

Operating - inlet (psig)		125		14.6
Design (psig)		150		75
Temperature				
In (°F)		45		180
Out (°F)		55		50
Construction material		Carbon steel		Stainless steel

TABLE 11.3-4 CODES AND STANDARDS

The radioactive gaseous waste, the reactor plant aerated vents, and the reactor plant gaseous vents system shall be designed and constructed in accordance with the requirements of the following codes and standards:

System Component	Safety Class	Codes and Standards (1) (2)
Piping, fittings	NNS	ASME III, Class 3
		ANSI B31.1
Valves	NNS	ASME III, Class 3
		ANSI B16.5
		ANSI B16.34
Adsorbers, degasifier, heat exchangers, filters, water removal equipment	NNS	ASME VIII, Division 1
Pumps and compressors	NNS	ASME III, Class 3
Fans	NNS	Manufacturer's standards
Filter assemblies	NNS	ASME VIII, Division 1
Instrumentation and controls		N/A
Motors		NEMA MG.1

NOTES:

(1) Portions of the system were procured to ASME Section III requirements and are defined in the procurement specification.

(2) The overall system classification is NNS.

TABLE 11.3-5 EXPECTED RADIOACTIVE GASEOUS RELEASES TO ATMOSPHERE VIA VENTILATION VENT (HISTORICAL)

Nuclide	Containment Building (Ci/yr)	Auxiliary Building (Ci/yr) (1)	Total (Ci/yr)
Kr-83m	5.7E-05	4.7E-01	4.7E-01
Kr-85m	2.0E-02	2.0E+00	2.0E+00
Kr-85	1.4E+00	4.5E-02	1.5E+00
Kr-87	7.4E-06	1.4E+00	1.4E+00
Kr-88	5.7E-03	4.0E+00	4.0E+00
Kr-89	0.0	1.3E-01	1.3E-01
Xe-131m	8.5E-01	1.2E-01	9.6E-01
Xe-133m	1.0E+00	8.7E-01	1.9E+00
Xe-133	1.1E+02	3.6E+01	1.5E+02
X-135m	0.0	3.4E-01	3.4E-01
Xe-135	3.4E-01	4.7E+00	5.0E+00
Xe-137	0.0	2.3E-01	2.3E-01
Xe-138	0.0	1.1E+00	1.1E+00
I-131	2.8E-06	4.6E-02	4.6E-02
I-133	2.8E-07	6.8E-02	6.8E-02
Co-58	3.8E-03	6.0E-02	6.4E-02
Co-60	1.7E-03	2.7E-02	2.9E-02
Mn-54	1.1E-03	1.8E-02	1.9E-02
Fe-59	3.8E-04	6.0E-03	6.4E-03
Sr-89	8.5E-05	1.3E-03	1.4E-03
Sr-90	1.5E-05	2.0E-04	2.1E-04
Cs-134	1.1E-03	1.8E-02	1.9E-02
Cs-137	1.9E-03	3.0E-02	3.2E-02
C-14	1.0E+00	0.0	1.0E+00
Ar-41	2.5E+01	0.0	2.5E+01
H-3	1.3E+02	6.0E+02	7.3E+02

NOTE:

(2) In accordance with NUREG-0017, April 1976, releases from the fuel building, waste disposal building, service building, and engineered safety features building are combined with auxiliary building releases.

Historical, not subject to future updating. This table has been retained to preserve original license basis.

TABLE 11.3-6 EXPECTED RADIOACTIVE GASEOUS RELEASES TO ATMOSPHERE FROM MILLSTONE 3 VIA MILLSTONE STACK (HISTORICAL)

Nuclide	Main Condenser/ Air Ejector (Ci/yr	Radioactive Gaseous Waste System (Ci/yr)	Total (Ci/yr)
Kr-83m	2.9E-01	0.0	2.9E-01
Kr-85m	1.2E+00	0.0	1.2E+00
Kr-85	2.8E-02	2.4E+02	2.4E+02
Kr-87	8.7E-01	0.0	8.7E-01
Kr-88	2.5E+00	0.0	2.5E+00
Kr-89	8.2E-02	0.0	8.2E-02
Xe-131m	7.3E-02	1.7E-01	2.4E-01
Xe-133m	5.4E-01	0.0	5.4E-01
Xe-133	2.3E+01	0.0	2.3E+01
Xe-135m	2.1E-01	0.0	2.1E-01
Xe-135	2.9E+00	0.0	2.9E+00
Xe-137	1.5E-01	0.0	1.5E-01
Xe-138	7.2E-01	0.0	7.2E-01
I-131	1.9E-02	0.0	1.9E-02
I-133	2.8E-02	0.0	2.8E-02
Co-58	0.0	0.0	0.0
Co-60	0.0	0.0	0.0
Mn-54	0.0	0.0	0.0
Fe-59	0.0	0.0	0.0
Sr-89	0.0	0.0	0.0
Sr-90	0.0	0.0	0.0
Cs-134	0.0	0.0	0.0
Cs-137	0.0	0.0	0.0
C-14	0.0	7.0E+00	7.0E+00
Ar-41	0.0	0.0	0.0
H-3	0.0	0.0	0.0

Historical, not subject to future updating. This table has been retained to preserve original license basis.

TABLE 11.3-7 EXPECTED RADIOACTIVE GASEOUS RELEASES TO ATMOSPHERE VIA TURBINE BUILDING (HISTORICAL)

Nuclide	Roof Exhausters (Ci/yr)	Steam Generator Blowdown Flash Tank Vent (2) (Ci/yr)	Total (Ci/yr)
Kr-83m	3.1E-05 (3)	0.0	3.1E-05
Kr-85m	1.3E-0	0.0	1.3E-04
Kr-85	3.0E-06	0.0	3.0E-06
Kr-87	9.2E-05	0.0	9.2E-05
Kr-88	2.7E-04	0.0	2.7E-04
Kr-89	8.7E-06	0.0	8.7E-06
Xe-131m	8.1E-06	0.0	8.1E-06
Xe-133m	5.9E-05	0.0	5.9E-05
Xe-133	2.4E-03	0.0	2.4E-03
Xe-135m	2.2E-05	0.0	2.2E-05
Xe-135	3.0E-04	0.0	3.0E-04
Xe-137	1.6E-05	0.0	1.6E-05
Xe-138	7.6E-05	0.0	7.6E-05
I-131	2.9E-04	7.0E-03	7.3E-03
I-133	4.2E-04	9.5E-03	9.9E-03
Co-58			0.0
Co-60			0.0
Mn-54			0.0
Fe-59			0.0
Sr-89			0.0
Sr-90			0.0
Cs-134			0.0
Cs-137			0.0
C-14			0.0
Ar-41			0.0
H-3			0.0

NOTES:

(2) Open cycle blowdown.

(3) 3.1E-05 = 3.1×10^{-5}.

Historical, not subject to future updating. This table has been retained to preserve original license basis.

TABLE 11.3-8 DESIGN RADIOACTIVE GASEOUS RELEASES TO ATMOSPHERE VIA VENTILATION VENT (HISTORICAL)

A. CURIES PER YEAR RELEASED

Nuclide	Containment Building (Ci/yr)	Auxiliary Building (Ci/yr) (1)	Total (Ci/yr)
Kr-83m	1.2E-03	9.5E+00	9.5E+00
Kr-85m	3.6E-01	3.6E+01	3.6E+01
Kr-85	2.3E+01	7.3E-01	2.4E+01
Kr-87	1.4E-04	2.6E+01	2.6E+01
Kr-88	1.0E-01	7.2E+01	7.2E+01
Kr-89	0.0	2.2E+00	2.2E+00
Xe-131m	1.7E+00	2.4E-01	1.9E+00
Xe-133m	1.5E+01	1.3E+01	2.8E+01
Xe-133	1.7E+03	5.6E+02	2.3E+03
Xe-135m	0.0	2.4E+01	2.4E+01
Xe-135	7.9E+00	1.1E+02	1.2E+02
Xe-137	0.0	3.5E+00	3.5E+00
Xe-138	0.0	1.3E+01	1.3E+01
I-131	2.6E-05	4.2E-01	4.2E-01
I-133	2.7E-06	6.6E-01	6.6E-01
Co-58	3.8E-03	6.0E-02	6.4E-02
Co-60	1.7E-03	2.7E-02	2.9E-02
Mn-54	1.1E-03	1.8E-02	1.9E-02
Fe-59	3.8E-04	6.0E-03	6.4E-03
Sr-89	9.9E-04	1.5E-02	1.6E-02
Sr-90	2.6E-04	3.4E-03	3.7E-03
Cs-34	1.3E-02	2.2E-01	2.3E-01
Cs-137	1.7E-01	2.6+00	2.8E+00
C-14	1.0E+00	0.0	1.0E+00
Ar-41	2.5E+01	0.0	2.5E+01
H-3	4.6E+02	2.1E+03	2.6E+03

TABLE 11.3-8 (CONTINUED) DESIGN RADIOACTIVE GASEOUS RELEASES TO ATMOSPHERE VIA VENTILATION VENT (HISTORICAL)

B. CONCENTRATION AND FRACTION OF MPC (2) AT SITE BOUNDARY DUE TO RELEASES FROM CONTAINMENT PURGE (3) VIA VENTILATION VENT:

Nuclide	Concentration (μCi/cc)	MPC Value (μCi/cc)	Fraction of MPC
Kr-83m	1.02E-15 (4)	3.0E-08	3.40E-08
Kr-85m	3.06E-13	1.0E-07	3.06E-06
Kr-85	1.95E-11	3.0E-07	6.52E-05
Kr-87	1.19E-16	2.0E-08	5.95E-09
Kr-88	8.50E-14	2.0E-08	4.25E-06
Kr-89	0.0	3.0E-08	0.0
Xe-131m	1.44E-12	4.0E-07	3.61E-06
Xe-133m	1.27E-11	3.0E-07	4.25E-05
Xe-133	1.44E-9	3.0E-07	4.82E-03
Xe-135m	0.0	3.0E-08	0.0
Xe-135	6.71E-12	1.0E-07	6.71E-05
Xe-137	0.0	3.0E-08	0.0
Xe-138	0.0	3.0E-08	0.0
I-131	2.21E-17	1.0E-10	2.21E-07
I-133	2.29E-18	4.0E-10	5.74E-09
Co-58	3.23E-15	2.0E-09	1.61E-06
Co-60	1.44E-15	3.0E-10	4.82E-06
Mn-54	9.35E-16	1.0E-09	9.35E-07
Fe-59	3.23E-16	2.0E-09	1.61E-07
Sr-89	8.41E-16	3.0E-10	2.80E-06
Sr-90	2.21E-16	3.0E-11	7.37E-06
Cs-134	1.10E-14	4.0E-10	2.76E-05
Cs-137	1.44E-13	5.0E-10	2.89E-04
C-14	8.50E-13	1.0E-07	8.50E-06
Ar-41	2.12E-11	4.0E-08	5.31E-04
H-3	3.91E-10	2.0E-07	1.95E-03

TABLE 11.3-8 (CONTINUED) DESIGN RADIOACTIVE GASEOUS RELEASES TO ATMOSPHERE VIA VENTILATION VENT (HISTORICAL)

C. CONCENTRATION AND FRACTION OF MPC AT SITE BOUNDARY DUE TO RELEASES FROM AUXILIARY BUILDING (1), (5) VIA VENTILATION VENT:

Nuclide	Concentration (μCi/cc)	MPC Value (μCi/cc)	Fraction of MPC
Kr-83m	1.20E-12	3.0E-08	4.01E-05
Kr-85m	4.55E-12	1.0E-07	4.55E-05
Kr-85	9.24E-14	3.0E-07	3.08E-07
Kr-87	3.29E-12	2.0E-08	1.64E-04
Kr-88	9.11E-12	2.0E-08	4.55E-04
Kr-89	2.78E-13	3.0E-08	9.28E-06
Xe-131m	3.04E-14	4.0E-07	7.59E-08
Xe-133m	1.64E-12	3.0E-07	5.48E-06
Xe-133	7.09E-11	3.0E-07	2.36E-04
Xe-135m	3.04E-12	3.0E-08	1.01E-04
Xe-135	1.39E-11	1.0E-07	1.39E-04
Xe-137	4.43E-13	3.0E-08	1.48E-05
Xe-138	1.64E-12	3.0E-08	5.48E-05
I-131	5.31E-14	1.0E-10	5.31E-04
I-133	8.35E-14	4.0E-10	2.09E-04
Co-58	7.59E-15	2.0E-09	3.80E-06
Co-60	3.42E-15	3.0E-10	1.14E-05
Mn-54	2.28E-15	1.0E-09	2.28E-06
Fe-59	7.59E-16	2.0E-09	3.80E-07
Sr-89	1.90E-15	3.0E-10	6.33E-06
Sr-90	4.30E-16	3.0E-11	1.43E-05
Cs-134	2.78E-14	4.0E-10	6.96E-05
Cs-137	3.29E-13	5.0E-10	6.58E-04
C-14	0.0	1.0E-07	0.0
Ar-41	0.0	4.0E-08	0.0
H-3	2.66E-10	2.0E-07	1.33E-03

NOTES:

(1) Releases from the fuel building, waste disposal building, service building, and engineered safety features building, are combined with auxiliary building releases.

(2) MPC values are taken from 1 0 CFR 20, Appendix B, Table II, Column I. When releasing radioactivity in air or water, instantaneous release rate limits are based on the values in Appendix B of the version of 10 CFR 20 prior to January 1, 1994 (Reference 11.3-1).

(3) Containment purge is an intermittent release from ventilation vent, and the applicable X/Q is 2.68×10^{-5} sec/m^3, which is the historical value used in the original estimates of radioactive concentrations at the Site Boundary.

(4) 1.02E-15 = 1.02×10^{-15}.

(5) Auxiliary building releases are continuous, and the applicable X/Q is 3.99×10^{-6} sec/m^3, which is the historical value used in the original estimates of radioactive concentrations at the Site Boundary.

Historical, not subject to future updating. Has been retained to preserve original license basis.

TABLE 11.3-9 DESIGN RADIOACTIVE GASEOUS RELEASES TO ATMOSPHERE FROM MILLSTONE 3 VIA MILLSTONE STACK (HISTORICAL)

A. CURIES PER YEAR RELEASED

Nuclide	Main Condenser / Air Ejector (Ci/yr)	Radioactive Gaseous Waste System (Ci/yr)	Total (Ci/yr)
Kr-83m	8.2E+01	0.0	8.2E+01
Kr-85m	3.2E+02	0.0	3.2E+02
Kr-85	6.3E+00	4.0E+03	4.0E+03
Kr-87	2.2E+02	0.0	2.2E+02
Kr-88	6.2E+02	0.0	6.2E+02
Kr-89	1.9E+01	0.0	1.9E+01
Xe-131m	2.1E+00	3.5E-01	2.5E+00
Xe-133m	1.1E+02	0.0	1.1E+02
Xe-133	4.8E+03	2.4E-02	4.8E+03
Xe-135m	2.1E+02	0.0	2.1E+02
Xe-135	9.3E+02	0.0	9.3E+02
Xe-137	3.0E+01	0.0	3.0E+01
Xe-138	1.1E+02	0.0	1.1E+02
I-131	2.3E+00	0.0	2.3E+00
I-133	3.7E+00	0.0	3.7E+00
Co-58	0.0	0.0	0.0
Co-60	0.0	0.0	0.0
Mn-54	0.0	0.0	0.0
Fe-59	0.0	0.0	0.0
Sr-89	0.0	0.0	0.0
Sr-90	0.0	0.0	0.0
Cs-134	0.0	0.0	0.0
Cs-137	0.0	0.0	0.0
C-14	0.0	7.0E+00	7.0E+00
Ar-41	0.0	0.0	0.0
H-3	0.0	0.0	0.0

TABLE 11.3-9 (CONTINUED) DESIGN RADIOACTIVE GASEOUS RELEASES TO ATMOSPHERE FROM MILLSTONE 3 VIA MILLSTONE STACK (HISTORICAL)

B. CONCENTRATIONS AND FRACTION OF MPC (1) AT SITE BOUNDARY DUE TO RELEASES VIA MILLSTONE STACK (2):

Nuclide	Concentration (μCi/cc)	MPC Value (μCi/cc)	Fraction of MPC
Kr-83m	3.06E-14 (4)	3.0E-08	1.02E-06
Kr-85m	1.20E-13	1.0E-07	1.20E-06
K-85	1.50E-12	3.0E-07	4.99E-06
Kr-87	8.23E-14	2.0E-08	4.12E-06
Kr-88	2.32E-13	2.0E-08	1.16E-05
Kr-89	7.11E-15	3.0E-08	2.37E-07
Xe-131m	9.35E-16	4.0E-07	2.34E-09
Xe-133m	4.12E-14	3.0E-07	1.37E-07
Xe-133	1.80E-12	3.0E-07	5.99E-06
Xe-135m	7.86E-14	3.0E-08	2.62E-06
Xe-135	3.48E-13	1.0E-07	3.48E-06
Xe-137	1.12E-14	3.0E-08	3.74E-07
Xe-138	4.12E-14	3.0E-08	1.37E-06
I-131	8.61E-16	1.0E-10	8.61E-06
I-133	1.38E-15	4.0E-10	3.46E-06
Co-58	0.0	2.0E-09	0.0
Co-60	0.0	3.0E-10	0.0
Mn-54	0.0	1.0E-09	0.0
Fe-59	0.0	2.0E-09	0.0
Sr-89	0.0	3.0E-10	0.0
Sr-90	0.0	3.0E-11	0.0
Cs-134	0.0	4.0E-10	0.0
Cs-137	0.0	5.0E-10	0.0
C-14	2.62E-15	1.0E-07	2.62E-08
Ar-41	0.0	4.0E-08	0.0
H-3	0.0	2.0E-07	0.0

NOTES:

(1) MPC values are taken from 10 CFR 20, Appendix B, Table II, Column I. When releasing radioactivity in air or water, instantaneous release rate limits are based on the values in Appendix B of the version of 10 CFR 20 prior to January 1, 1994 (Reference 11.3-1)

(2) This is continuous, and the applicable X/Q is 1.18×10^{-8} sec/m^3, which is the historical value used in the original estimates of radioactive concentrations at the Site Boundary.

(3) 3.06E-14 = 3.06×10^{-14}.

Historical, not subject to future updating. This table has been retained to preserve original license basis.

TABLE 11.3-10 DESIGN RADIOACTIVE GASEOUS RELEASES TO ATMOSPHERE VIA TURBINE BUILDING ROOF (HISTORICAL)

A. CURIES PER YEAR RELEASED

Nuclide	Roof Exhausters (Ci/yr)	Steam Generator Blowdown Flash Tank Vent (1) (Ci/yr)	Total (Ci/yr)
Kr-83m	8.7E-03	0.0	8.7E-03
Kr-85m	3.3E-02	0.0	3.3E-02
Kr-85	6.7E-04	0.0	6.7E-04
Kr-87	2.4E-02	0.0	2.4E-02
Kr-88	6.5E-02	0.0	6.5E-02
Kr-89	2.1E-03	0.0	2.1E-03
Xe-131m	2.2E-04	0.0	2.2E-04
Xe-133m	1.2E-02	0.0	1.2E-02
Xe-133	5.1E-01	0.0	5.1E-01
Xe-135m	2.2E-02	0.0	2.2E-02
Xe-135	1.0E-01	0.0	1.0E-01
Xe-137	3.2E-03	0.0	3.2E-03
Xe-138	1.2E-02	0.0	1.2E-02
I-131	3.7E-02	8.65E-01	9.0E-01
I-133	5.6E-02	1.30E+00	1.4E+00
Co-58	0.0	0.0	0.0
Co-60	0.0	0.0	0.0
Mn-54	0.0	0.0	0.0
Fe-59	0.0	0.0	0.0
Sr-89	0.0	0.0	0.0
Sr-90	0.0	0.0	0.0
Cs-134	0.0	0.0	0.0
Cs-137	0.0	0.0	0.0
C-14	0.0	0.0	0.0
Ar-41	0.0	0.0	0.0
H-3	0.0	0.0	0.0

TABLE 11.3-10 (CONTINUED) DESIGN RADIOACTIVE GASEOUS RELEASES TO ATMOSPHERE VIA TURBINE BUILDING ROOF (HISTORICAL)

B. CONCENTRATIONS AND FRACTION OF MPC (2) AT SITE BOUNDARY DUE TO RELEASES FROM TURBINE BUILDING VENTILATION (3):

Nuclide	Concentration (μCi/cc)	MPC Value (μCi/cc)	Fraction of MPC
Kr-83m	2.79E-15 (4)	3.0E-08	9.29E-08
Kr-85m	1.06E-14	1.0E-07	1.06E-07
Kr-85	2.15E-16	3.0E-07	7.15E-10
Kr-87	7.69E-15	2.0E-08	3.84E-07
Kr-88	2.08E-14	2.0E-08	1.04E-06
Kr-89	6.73E-16	3.0E-08	2.24E-08
Xe-131m	7.05E-17	4.0E-07	1.76E-10
Xe-133m	3.84E-15	3.0E-07	1.28E-08
Xe-133	1.63E-13	3.0E-07	5.44E-07
Xe-135m	7.05E-15	3.0E-08	2.35E-07
Xe-135	3.20E-14	1.0E-07	3.20E-07
Xe-137	1.02E-15	3.0E-08	3.42E-08
Xe-138	3.84E-15	3.0E-08	1.28E-07
I-131	1.18E-14	1.0E-10	1.18E-04
I-133	1.79E-14	4.0E-10	4.48E-05
Co-58	0.0	2.0E-09	0.0
Co-60	0.0	3.0E-10	0.0
Mn-54	0.0	1.0E-09	0.0
Fe-59	0.0	2.0E-09	0.0
Sr-89	0.0	3.0E-10	0.0
Sr-90	0.0	3.0E-11	0.0
Cs-134	0.0	4.0E-10	0.0
Cs-137	0.0	5.0E-10	0.0
C-14	0.0	1.0E-07	0.0
Ar-41	0.0	4.0E-08	0.0
H-3	0.0	2.0E-07	0.0

TABLE 11.3-10 (CONTINUED) DESIGN RADIOACTIVE GASEOUS RELEASES TO ATMOSPHERE VIA TURBINE BUILDING ROOF (HISTORICAL)

C. CONCENTRATIONS AND FRACTION OF MPC AT SITE BOUNDARY DUE TO RELEASES FROM STEAM GENERATOR BLOWDOWN [(5)]:

Nuclide	Concentration (μCi/cc)	MPC Value (μCi/cc)	Fraction of MPC
Kr-83m	0.0	3.0E-08	0.0
Kr-85m	0.0	1.0E-07	0.0
Kr-85	0.0	3.0E-07	0.0
Kr-87	0.0	2.0E-08	0.0
Kr-88	0.0	2.0E-08	0.0
Kr-89	0.0	3.0E-08	0.0
Xe-131m	0.0	4.0E-07	0.0
Xe-133m	0.0	3.0E-07	0.0
Xe-133	0.0	3.0E-07	0.0
Xe-135m	0.0	3.0E-08	0.0
Xe-135	0.0	1.0E-07	0.0
Xe-137	0.0	3.0E-08	0.0
Xe-138	0.0	3.0E-08	0.0
I-131	1.33E-12	1.0E-10	1.33E-02
I-133	2.00E-12	4.0E-10	4.99E-03
Co-58	0.0	2.0E-09	0.0
Co-60	0.0	3.0E-10	0.0
Mn-54	0.0	1.0E-09	0.0
Fe-59	0.0	2.0E-09	0.0
Sr-89	0.0	3.0E-10	0.0
Sr-90	0.0	3.0E-11	0.0
Cs-134	0.0	4.0E-10	0.0
Cs-137	0.0	5.0E-10	0.0
C-14	0.0	1.0E-07	0.0
Ar-41	0.0	4.0E-08	0.0
H-3	0.0	2.0E-07	0.0

NOTES:

(1) Open cycle blowdown.

(2) MPC values are taken from 10 CFR 20, Appendix B, Table II, Column I. When releasing radioactivity in air or water, instantaneous release rate limits are based on the values in Appendix B of the version of 10 CFR 20 prior to January 1, 1994 (Reference 11.3-1.)

(3) This is a continuous release, and the applicable X/Q is 1.01 x 10^{-5} sec/m^3, which is the historical value used in the original estimates of radioactive concentrations at the Site Boundary.

(4) 2.79E-15 = 2.79 x 10^{-15}.

(5) This is an intermittent release, and the applicable X/Q is 4.84 x 10^{-5} sec/m^3, which is the historical value used in the original estimates of radioactive concentrations at the Site Boundary.

Historical, not subject to future updating. This table has been retained to preserve original license basis.

TABLE 11.3-11 DESIGN RADIOACTIVE GASEOUS RELEASES TO ATMOSPHERE FROM MILLSTONE 3 (HISTORICAL)

A. TOTAL RELEASES:

Nuclide	Millstone Stack (Ci/yr)	Ventilation Vent (Ci/yr)	Turbine Building (Ci/yr)	Condensate Polishing Building (1) (Ci/yr)	Total (Ci/yr)
K-83m	8.2E+01 (2)	9.5E+00	8.7E-03	4.3E-02	9.2E+01
Kr-85m	3.2E+02	3.6E+01	3.3E-02	1.7E-01	3.6E+02
K-85	4.0E+03	2.4E+01	6.7E-04	3.3E-03	4.0E+03
Kr-87	2.2E+02	2.6E+01	2.4E-02	1.2E-01	2.5E+02
Kr-88	6.2E+02	7.2E+01	2.4E-02	3.3E-01	6.9E+02
Kr-89	1.9E+01	2.2E+00	2.1E-03	1.0E-02	2.1E+01
Xe-131m	2.5E+00	1.9E+00	2.2E-04	1.1E-03	4.4E+00
Xe-133m	1.1E+02	2.8E+01	1.2E-02	5.9E-02	1.4E+02
Xe-133	4.8E+03	2.3E+03	5.1E-01	2.5E+00	7.1E+03
Xe-135m	2.1E+02	2.4E+01	2.2E-02	1.1E-01	2.3E+02
Xe-135	9.3E+02	1.2E+02	1.0E-01	5.0E-01	1.1E+03
Xe-137	3.0E+01	3.5E+00	3.2E-03	1.6E-02	3.4E+01
Xe-138	1.1E+02	1.3E+01	1.2E-02	5.8E-02	1.2E+02
I-131	2.3E+00	4.2E-01	9.0E-01	2.8E-02	3.6E+00
I-133	3.7E+00	6.6E-01	1.4E+00	4.2E-02	5.8E+00
Co-58	0.0	6.4E-02	0.0	0.0	6.4E-02
Co-60	0.0	2.9E-02	0.0	0.0	2.9E-02
Mn-54	0.0	1.9E-02	0.0	0.0	1.9E-02
Fe-59	0.0	6.4E-03	0.0	0.0	6.4E-03
Sr-89	0.0	1.6E-02	0.0	0.0	1.6E-02
Sr-90	0.0	3.7E-03	0.0	0.0	3.7E-03
Cs-134	0.0	2.3E-01	0.0	0.0	2.3E-01
Cs-137	0.0	2.8E+00	0.0	0.0	2.8E+00
C-14	7.0E+00	1.0E+00	0.0	0.0	8.0E+00
Ar-41	0.0	2.5E+01	0.0	0.0	2.5E+01
H-3	0.0	2.6E+03	0.0	0.0	2.6E+03

TABLE 11.3-11 (CONTINUED) DESIGN RADIOACTIVE GASEOUS RELEASES TO ATMOSPHERE FROM MILLSTONE 3 (HISTORICAL)

B. CONCENTRATIONS AND FRACTION OF MPC (3) AT SITE BOUNDARY DUE TO RELEASES FROM CONDENSATE POLISHING BUILDING (4):

Isotope	Concentration (μCi/cc)	MPC Value (μCi/cc)	Fraction of MPC
K-83m	3.08E-14	3.0E-08	1.03E-06
Kr-85m	1.22E-13	1.0E-07	1.22E-06
Kr-85	2.36E-15	3.0E-07	7.88E-09
Kr-87	8.60E-14	2.0E-08	4.30E-06
Kr-88	2.36E-13	2.0E-08	1.18E-05
K-89	7.17E-15	3.0E-08	2.39E-07
Xe-131m	7.88E-16	4.0E-07	1.97E-09
Xe-133m	4.23E-14	3.0E-07	1.41E-07
Xe-133	1.79E-12	3.0E-07	5.97E-06
Xe-135m	7.88E-14	3.0E-08	2.63E-06
Xe-135	3.58E-13	1.0E-07	3.58E-06
Xe-137	1.15E-14	3.0E-08	3.82E-07
Xe-138	4.16E-14	3.0E-08	1.39E-06
I-131	2.01E-14	1.0E-10	2.01E-04
I-135	3.01E-14	4.0E-10	7.52E-05
Co-58	0.0	2.0E-09	0.0
Co-60	0.0	3.0E-10	0.0
Mn-54	0.0	1.0E-09	0.0
Fe-59	0.0	2.0E-09	0.0
Sr-89	0.0	3.0E-10	0.0
Sr-90	0.0	3.0E-11	0.0
Cs-134	0.0	4.0E-10	0.0
Cs-137	0.0	5.0E-10	0.0
C-14	0.0	1.0E-07	0.0
Ar-41	0.0	4.0E-08	0.0
H-3	0.0	2.0E-07	0.0

TABLE 11.3-11 (CONTINUED) DESIGN RADIOACTIVE GASEOUS RELEASES TO ATMOSPHERE FROM MILLSTONE 3 (HISTORICAL)

C. FRACTION OF MPC AT SITE BOUNDARY DUE TO ALL RELEASES (5):

Nuclide	Millstone Stack	Ventilation Vent	Turbine Building	Condensate Polishing Building	Total Fraction of MPC
Kr-83m	1.02E-06	4.01E-05	9.29E-08	1.03E-06	4.22E-05
Kr-85m	1.20E-06	4.86E-05	1.06E-07	1.22E-06	5.11E-05
Kr-85	4.99E-06	6.55E-05	7.15E-10	7.88E-09	7.05E-05
Kr-87	4.12E-06	1.64E-04	3.84E-07	4.30E-06	1.73E-04
Kr-88	1.16E-05	4.59E-04	1.04E-06	1.18E-05	4.83E-04
Kr-89	2.37E-07	9.28E-06	2.24E-08	2.39E-07	9.78E-06
Xe-131m	2.34E-09	3.69E-06	1.76E-10	1.97E-09	3.69E-06
Xe-133m	1.37E-07	4.80E-05	1.28E-08	1.41E-07	4.83E-05
Xe-133	5.99E-06	5.06E-03	5.44E-07	5.97E-06	5.07E-03
Xe-135m	2.62E-06	1.01E-04	2.35E-07	2.63E-06	1.06E-04
Xe-135	3.48E-06	2.06E-04	3.20E-07	3.58E-06	2.13E-04
Xe-137	3.74E-07	1.48E-05	3.42E-08	3.82E-07	1.56E-05
Xe-138	1.37E-06	5.48E-05	1.28E-07	1.39E-06	5.77E-05
I-131	8.61E-06	5.31E-04	1.34E-02	2.01E-04	1.41E-02
I-133	3.46E-06	2.09E-04	5.03E-03	7.52E-05	5.32E-03
Co-58	0.0	5.41E-06	0.0	0.0	5.41E-06
Co-60	0.0	1.62E-05	0.0	0.0	1.62E-05
Mn-54	0.0	3.22E-06	0.0	0.0	3.22E-06
Fe-59	0.0	5.41E-07	0.0	0.0	5.41E-07
Sr-89	0.0	9.13E-06	0.0	0.0	9.13E-06
Sr-90	0.0	2.17E-05	0.0	0.0	2.17E-05
Cs-134	0.0	9.72E-05	0.0	0.0	9.72E-05
Cs-137	0.0	9.47E-04	0.0	0.0	9.47E-04
C-14	2.62E-08	8.50E-06	0.0	0.0	8.53E-06
Ar-41	0.0	5.31E-04	0.0	0.0	5.31E-04
H-3	0.0	3.28E-03	0.0	0.0	3.28E-03

<u>NOTES</u>:

(1) Releases from turbine gland sealing system exhaust.

(2) 8.2E+01 = 8.2×10^{1}.

(3) MPC values are taken from 10 CFR 20, Appendix B, Table II, Column I. When releasing radioactivity in air or water, instantaneous release rate limits are based on the values in Appendix B of the version of 10 CFR 20 prior to January 1, 1994 (Reference 11.3-1).

(4) This is a continuous release, and the applicable X/Q is 2.26×10^{-5} sec/m^3, which is the historical value used in the original estimates of radioactive concentrations at the Site Boundary.

(5) The values provided in this table are based on historical X/Q values.

Historical, not subject to future updating. This table has been retained to preserve original license basis.

TABLE 11.3-12 ASSUMPTIONS USED FOR THE PROCESS GAS CHARCOAL BED ADSORBER BYPASS ANALYSIS [1]

	Design Release
Letdown flow to degasifier (gpm)	75
Reactor coolant activity	Table 11.2-1
Charcoal bed adsorber holdup time	
Kr (days)	6.1
Xe (days)	142
Fraction of noble gas released from bed	Table 11.3-13
Duration of release (min.)	60

NOTES:

(1) The bypass of the process gas charcoal bed adsorbers releases the gaseous activity discharged from the degasifier for 60 minutes and a fraction of the activity on the beds.

TABLE 11.3-13 RADIOISOTOPE RELEASES FROM THE PROCESS GAS CHARCOAL BED ADSORBER AND ASSOCIATED PIPING

Isotope	Design			
	Discharge from the degasifier (Ci)	Inventory in the Charcoal Bed (Ci)	Fraction Released from Bed	Total Release (Ci)
Kr-83m	7.65E+00 (1)	2.06E+01	1.000	2.83E+01
Kr-85m	2.93E+01	1.86E+01	0.998	2.15E+02
Kr-85	5.88E-01	8.64E+01	0.109	1.00E+01
Kr-87	2.11E+01	3.84E+01	1.000	5.95E+01
Kr-88	5.76E+01	2.32E+02	1.000	2.90E+02
Kr-89	1.80E+00	1.38E-01	1.000	1.94E+00
Xe-131m	1.91E-01	7.91E+01	0.095	7.69E+00
Xe-133m	1.04E+0.1	8.15E+02	0.410	3.44E+02
Xe-133	4.48E+02	8.27E+04	0.202	1.72E+04
Xe-135m	1.96E+01	7.39E+00	1.000	2.70E+01
Xe-135	8.74E+01	1.16E+03	1.000	1.25E+03
Xe-137	2.84E+00	2.62E-01	1.000	3.10E+00
Xe-138	1.02E+01	3.48E+00	1.000	1.37E+01

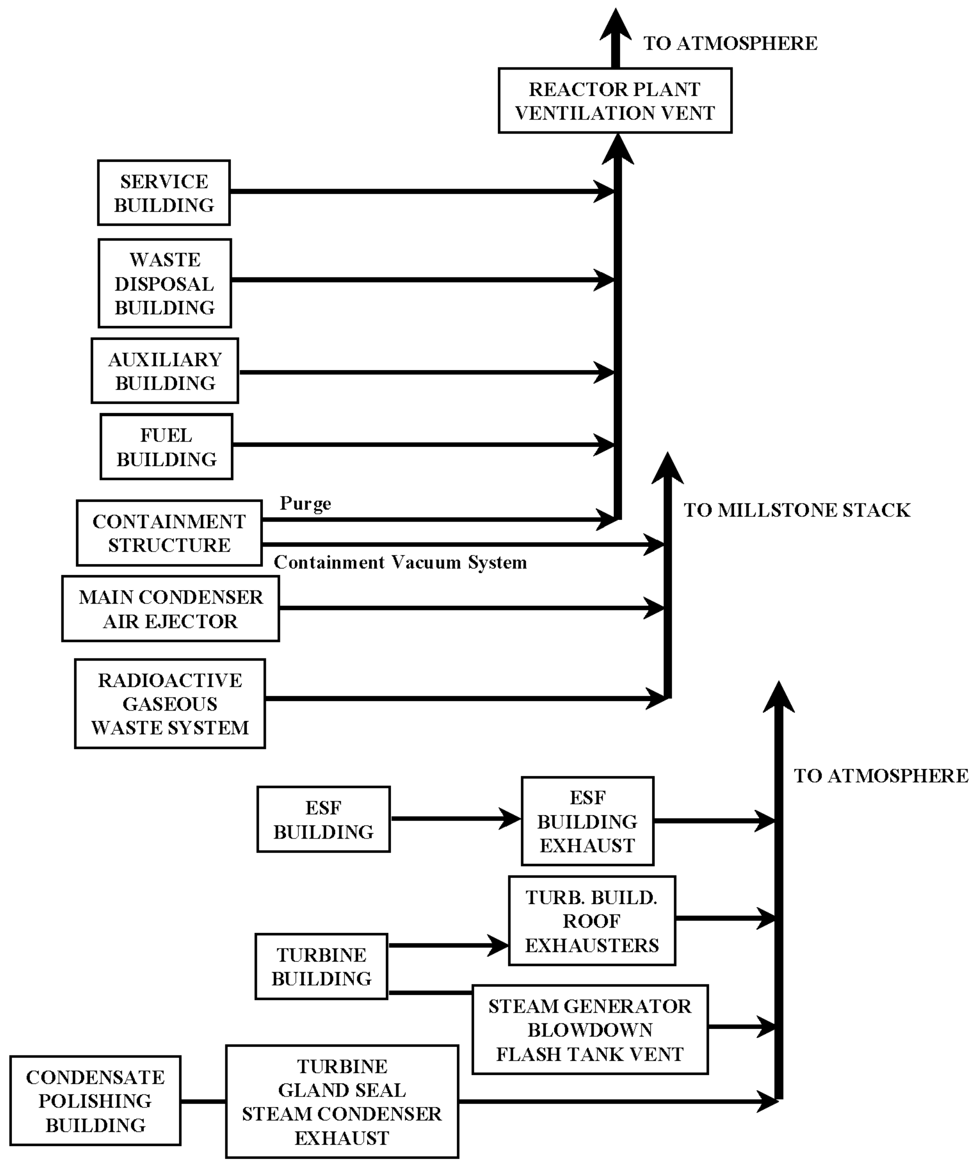
NOTES:

(1) 7.65E+00 = 7.65×10^0

FIGURE 11.3–1 P&ID RADIOACTIVE GASEOUS WASTE SYSTEM (SHEETS 1-2)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-3 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 11.3–2 VENTILATION SYSTEM COMPOSITE DRAWING NORMAL OPERATION



11.4 SOLID WASTE MANAGEMENT

The radioactive solid waste system is designed to collect, hold, process, dewater or solidify, package, handle, and temporarily store radioactive materials prior to their shipment offsite and ultimate disposal.

The radioactivity values provided in this section are the design basis values used for the design of the Solid Waste System. As such, they are considered historical and not subject to future updating. The information is retained to avoid loss of the original design bases. Volumes and radioactivity content, including specific nuclide percentages of actual shipments can be found in the annual radioactive effluent release reports as submitted to the NRC.

11.4.1 DESIGN BASES

The radioactive solid waste system is designed in accordance with the following criteria.

1. The system design parameters are based on radionuclide concentrations and volumes consistent with reactor operating experience for similar designs and with the source terms of Section 11.1.

2. All wet solid wastes including bulk liquids, sludges, or spent resin are dewatered, solidified or otherwise treated (as required). Procedures are used to ensure the absence of free liquid in the containers and control other appropriate waste form characteristics. Plant procedures are also used to ensure that equipment is operated such that federal and state disposal regulations are met. Dewatering equipment is described in FSAR Section 11.4.2.2 and Figure 11.4–1 (2 of 2).

   Processing equipment is sized to handle the design inputs without the need to ship bulk liquids. In-plant waste storage facilities provide sufficient temporary storage capacity to allow time for shipping delays. Tanks accumulating spent resins have the capability of accommodating at least 60 days' waste generation at normal generation rates. Temporary storage capacity exceeds 30 days' waste generation at expected generation rates. Longer term temporary storage is described in FSAR Section 11.4.2.5.

3. Solid waste containers, shipping casks, and methods of packaging meet applicable federal regulations, e.g., 10 CFR Part 71, and wastes are to be shipped to a licensed burial site in accordance with applicable NRC, e.g., 10 CFR Part 61, and Department of Transportation regulations, e.g., 49 CFR 171-178. Solid waste treatment design is in compliance with the relevant requirements of 10 CFR Part 20, sections 105 and 106 (version prior to January 1, 1994) as it relates to radioactivity in effluents to unrestricted areas.

4. Permanent plant components and piping systems are designed in accordance with the provisions of Regulatory Guide 1.143 and Branch Technical Position ETSB 11-3.

5. The permanent plant system contains provisions to reduce leakage and facilitate operations and maintenance in accordance with the provisions of Regulatory Guide 1.143 and Branch Technical Position, ETSB 11-3 (July 1981).

6. The filling of containers, the dewatering, the solidification, and/or the storage of radioactive solid wastes conforms to 10 CFR 20 and 10 CFR 50 requirements and Regulatory Guide 8.8 guidelines in terms of “as low as is reasonably achievable” (ALARA) doses to plant personnel and the general public.

7. The temporary waste storage facilities in the waste storage building are shielded to provide protection of operating personnel in accordance with the radiation protection design considerations in Section 12.1.

8. Radiation protection personnel conducting surveys use portable radiation detectors to determine radiation levels outside and inside shielded areas. These surveys are performed to ensure that area guidelines are met and to ensure adequate personnel protection through area posting and access limitations.

9. This system is not safety-related and is classified as nonnuclear safety (NNS).

10. The portion of the solid waste building’s foundation and adjacent walls up to a height sufficient to contain the liquid inventory in the building is designed to the seismic criteria used for Millstone 3 and to the quality assurance criteria of Section 6 of Regulatory Guide 1.143.

11. Portions of the system that handle radioactive liquid waste meet the applicable design bases of Section 11.2.1.

12. The system is designed to process, handle, and store the waste types, and quantities described in Section 11.4.2 below, which are generated as a result of normal operation.

13. The system is able to reduce the volume of selected input streams to minimize packaged quantities for transport and disposal.

11.4.2 SYSTEM DESCRIPTION

The radioactive solid waste system is shown on Figure 11.4–1.

The layout of the solid waste building, including packaging, storage, and shipping areas, is shown in Figure 3.8–74.

Radioactive solid waste system components are listed in Table 11.4-2, which summarizes design and operating conditions.

11.4.2.1 System Inputs

Materials handled as solid wastes may include any of the following: concentrated waste solutions from the waste evaporator, concentrated boric acid discarded from the boron evaporator in the boron recovery system (Section 9.3.5), spent resin from radioactive process demineralizers and exchangers, spent filter cartridges, and miscellaneous sludges. The waste evaporator has been removed from service.

Figure 11.4–2 is a flow chart of the expected quantities and activity levels of the radioactive solid waste generated by the plant. Gross activities and weights or volumes for radioactive solid waste sources are also given on Figure 11.4–2. Figure 11.4–3 provides the design volumes and activity levels of radioactive solid wastes and waste sources.

11.4.2.1.1 Spent Resins

Figure 11.4–2 provides the estimated volumes of spent demineralizer and ion exchanger resins per year. These resins come from a variety of different services, and the total volume estimated is based on the individual resin bed volumes and the expected frequency of replacement (Table 11.4-1).

11.4.2.1.2 Waste Evaporator Bottoms (The waste evaporator subsystem has been removed from service)

Evaporator bottoms from the waste evaporator in the radioactive liquid waste system may be transferred to the solid waste system. The estimated volume of bottoms solution to be shipped off site is given on Figure 11.4–2. The calculated activity of these bottoms, also shown on Figure 11.4–2, is based on the input of the radioactive liquid waste system (Section 11.2).

11.4.2.1.3 Regenerant Chemical Evaporator Bottoms (Removed From Service)

11.4.2.1.4 Boron Evaporator Bottoms

The boron evaporator in the boron recovery system processes reactor coolant letdown to the boron recovery system for separation of boric acid and water. The estimated volume given on Figure 11.4–2 for the boron evaporator bottoms is a 1 year average of the contents requiring processing for eventual off site disposal. The activity shown for these bottoms, given on Figure 11.4–2, is based on the expected performance of the boron recovery system (Section 9.3.5). Boric acid may be either recycled to the plant or transferred to the solid waste system depending upon operational considerations.

11.4.2.1.5 Miscellaneous Radioactive Solid Wastes

It is estimated that approximately 4,000 cubic feet per year of additional waste requiring disposal is processed as radioactive solid waste. This volume of spent filters, contaminated cloths, and other radioactive material and their activity, given on Figure 11.4–2, was originally estimated from operating experience at other nuclear facilities.

This category includes waste charcoal absorber media from filtration units. Charcoal is removed by an external vacuum system outlined in CVI Topical Report No. CVI-TR-7301 (February 1975).

11.4.2.2 Equipment Description

The radioactive solid waste system equipment is operated on a batch basis. Individual components are designed to support the system's rated capacity.

The permanently installed system consists of a control panel, spent resin dewatering and hold tanks, and the piping, pumps, and process equipment modules required for transfer of wastes to a shipping container. Operation of the solid waste equipment requires minimal manual actions and, in conjunction with the building layout, is designed to minimize occupational radiation exposures. A portion of the system is designed to permit filling and dewatering of the resin granulated activated carbon (GAC) slurries in the shipping container.

Shipping containers are approved for the process for which they are to be used.

When slurries are being processed, the fill head is lifted by the overhead bridge crane and lowered onto the shipping container having the integral dewatering line. When the fill head is lowered onto the shipping container, it is aligned with the internal dewatering line and secured prior to filling.

Boric acid concentrated in the boron evaporator in the boron recovery system is typically recycled. However, the capability exists to transfer waste concentrated from the waste evaporator in the radioactive liquid waste system to the solid waste system. The waste evaporator has been removed from service therefore this capability no longer exists.

Resins sluiced from demineralizers and ion exchangers are stored in the spent resin hold tank, with the exception of any resins generated by a liquid radioactive waster processing system. Resins generated by such sources are slurried directly to a solid radioactive waste processing system. The resins are then slurried to the shipping container, where they are allowed to settle. Excess water is removed by the spent resin dewatering pump within the portable dewatering unit and transferred to the spent resin dewatering tank.

If solidification were desired, it would be performed by an approved vendor whose Process Control Program was reviewed and SORC approved. The solidification will be performed with a mobile solidification system provided by the approved vendor.

Spent filters, contaminated tools, and other incompressible contaminated solid wastes can be inserted into a shipping container prior to filling. If solidification is required, a vendor will be utilized. Containers and shields are handled by overhead bridge crane.

Waste containers are sealed in various ways depending on the system chosen from a number of solid waste system suppliers.

The systems are designed to prevent external contamination of containers by use of reliable container sealing, appropriate system flushing into the container, and necessary mechanical design or instrumentation interlock signals that prevent overfilling of containers.

The sealed disposable container is transported to an interim on-site storage facility or a disposal site. In general, the disposal container may hold spent resin, spent filters, and other incompressible waste in addition to evaporator bottoms. The waste evaporator and Waste Bottoms Holding Tank have been removed from service.

Compressible dry solid waste (e.g., contaminated clothing, wipeup toweling) are collected and forwarded for processing.

11.4.2.2.1 Boron, Waste Evaporator Bottoms

Concentrated liquids from the evaporators may be pumped to the waste bottoms tank for holdup until they are to be solidified. The tank is insulated, heat traced, and is recirculated to prevent crystallization, stratification, settling, and accumulation of undissolved solids. The waste evaporator and Waste Bottoms Holding Tank have been removed from service.

The amount of waste liquid allowed per container is determined by prior analysis of the waste. Radiation levels are also monitored while filling the container to ensure DOT shipping limits are not exceeded. After the container is filled and solidified, it is sealed and stored. Prior to shipment, it is shielded, as necessary.

Shipments are made in accordance with NRC regulations 10 CFR 20, 10 CFR 50, and 10 CFR 71, and Department of Transportation regulations 49 CFR 171 through 178. The shipping containers are stored temporarily in the solidification area (Section 11.4.2.5) or an interim on-site storage facility prior to off-site disposal.

11.4.2.2.2 Spent Resin Handling

Resin in a demineralizer or ion exchanger is considered spent when the decontamination factor falls below a predetermined value, when the demineralizer or ion exchanger surface dose approaches a predetermined limit, or when the resin bed pressure drop becomes excessive. The demineralizer or ion exchanger is then isolated. Water from the spent resin dewatering tank is used to flush the spent resin, generated by plant sources, into the spent resin hold tank utilizing the spent resin transfer pump. The flush water passes out of the spent resin hold tank, through a screened element to prevent any resin carryover, and is recirculated. When the resin, from the spent resin hold tank, is to be packaged for disposal, it is pumped as a resin-water slurry to the shipping container in the processing area. Any resin or granulated activated carbon (GAC) generated by a liquid radioactive waste processing system will be slurried to a solid radioactive processing system for handling and ultimate disposal in a shipping container. This flushing operation will be completed using plant primary grade water and service air. Excess water is dewatered from the shipping container, via a solid radioactive waste processing system, and returned to the spent resin dewatering tank. Radiation levels are monitored to ensure that DOT shipping limits are not exceeded. After dewatering, the container is sealed, labeled, and temporarily stored until shipped off site. Prior to shipment the container is shielded as necessary.

11.4.2.2.3 Filter Handling

Cartridge filter elements are removed from service when the surface dose on the filter housing reaches a predetermined level or when the element pressure drop becomes excessive. The operation is carried out using remote handling equipment and a filter removal shield, when required. Radiation protection personnel conduct surveys during filter changeouts as required, using portable radiation detectors and take action as appropriate in accordance with ALARA guidelines.

11.4.2.2.4 Incompressible Waste Handling

Contaminated metallic materials and solid objects are placed in disposable shipping containers.

11.4.2.2.5 Waste Compaction Operation

Contaminated compressible materials are temporarily stored in suitably labeled containers in different plant locations. These materials are then transported for processing or packaging.

11.4.2.3 Expected Volumes

Table 11.4-1 presents a listing of the expected volumes of spent resins from various sources entering the radioactive solid waste system. (Note that resin from the condensate polishing demineralizers indicated on Table 11.4-1 is processed by Millstone Unit 2 - See Figure 10.4–5.) Figure 11.4–2 presents gross activities and weights or volumes for radioactive solid waste sources.

11.4.2.4 Packaging

Based on the gross activities supplied on Figure 11.4–2 and DOT Low Specific Activity Limits, container activity varies between negligible for most compressible or compacted wastes to less than 300 μCi/cc for reactor water purification demineralizer resins. The specific radionuclide content of the solid wastes is available, as discussed in Section 11.4.

The filling of containers and the storage of radioactive solid wastes conform with 10 CFR 20 and 10 CFR 50 requirements. Packages meet shipping regulations of 49 CFR 171-178 and 10 CFR 71 as applicable.

Complete solidification and absence of free liquid is ensured by the implementation of a process control program and preoperational testing.

11.4.2.5 Temporary On-site Storage Facilities

The processing area is located on the ground floor of the solid waste building. Wastes are packaged to allow for shipment after processing. Therefore, no decay time is assumed in the given estimate of package contents and activity levels.

Storage capacity and storage time are based on anticipated operational factors.

In addition to the above, interim on-site storage facilities accept waste from Millstone Units 1, 2 and 3 in accordance with the following criteria.

1. The Millstone Radwaste Storage Facility and the On-Site Storage Containers (OSSC's) are used to store liners that contain wastes like dewatered resin and filters. Safety Evaluations performed on the Radwaste Storage Facility and OSSC's conclude that these structures meet the criteria set forth in Section 3.5.1.4.

2. In addition to being used for the sorting, processing, loading and shipping of radioactive materials, the MRRF may be used to store dry activated waste. The capacity of the MRRF is dependent upon:

   a. the waste generated from unit/site activities; and

   b. waste volume reduction techniques employed.

3. The Unit 2/Unit 3 Condensate Polishing Facility (CPF) waste processing area may be used to store mixed waste (radioactive and hazardous combined) material.

11.4.2.6 Shipment

The shipment of radioactive solid waste conforms with 10 CFR 20, 10 CFR 50, and 10 CFR 61 requirements and 10 CFR 71 and 49 CFR 171 through 178. Solid waste is transferred either directly to a licensed disposal contractor or to a common carrier for delivery to a licensed burial site, or secondary processor as appropriate.

Table 11.4-4 summarizes the annual number of shipments and shipped containers for the expected and design cases.

11.4.2.7 Protection Against Uncontrolled Releases

Protection against uncontrolled releases of radioactive material from the radioactive solid waste system is achieved through the use of alarms, interlocks, and a retaining structure.

The spent resin dewatering tank, evaporator bottoms tank (removed from service), spent resin transfer pump, spent resin recycle pump, and any liquid and/or solid radioactive waste processing system are located in curbed cubicles where any leakage is retained. The walls and floors are suitably finished to facilitate decontamination. The disposal waste shipping container and the resin fill and dewater head are located on the 24 foot 6 inch elevation of the waste solidification building. In the event of spillage of radioactive liquid in this area, the liquid is collected by a network of floor drains. The floor is pitched toward the floor drains. The drains are piped to the waste disposal building sumps (FSAR Figure 3.8–74) which are collected by the aerated drains system (FSAR Section 9.3.3) and forwarded to the radioactive liquid waste system (FSAR Section 11.2) for processing.

The spent resin dewatering tank and the evaporator bottoms tank (removed from service) are provided with low level alarms at the solid waste panel. For an alarm condition, a visual inspection by the operator is conducted.

Health physics personnel, equipped with portable radiation detectors, are present at the fill area to ensure that the radiation levels are within the design levels.

A solid radioactive waste system receives solid waste from the plant and transfers excess water back to the plant.

The single worst operator error or equipment failure would result in the spillage of spent resin and/or evaporator bottoms tank (removed from service) contents or spent resin and/or granulated activated carbon (GAC). The drainage system in the waste solidification building is designed to handle such an event.

TABLE 11.4-1 VOLUME OF SPENT RESIN GENERATED ANNUALLY

Demineralizers	Number of Beds	Volume ft^3/ bed	Frequency of Replacement (beds/yr)
Fuel pool demineralizer	1	15	1
Cesium removal ion exchangers	2	35	4
Boron demineralizer	2	35	2
Waste demineralizer	2	35	2
Condensate polishing demineralizers (1)	8	196	1
Mixed bed demineralizers	2	30	6
Cation bed demineralizers	1	20	2
Thermal regeneration demineralizers	5	74	(2)

NOTE:

(1) The condensate polishing demineralizer resin will be processed by Millstone 2.

(2) The Thermal generation demineralizers are no longer used.

TABLE 11.4-2 RADIOACTIVE SOLID WASTE SYSTEM COMPONENT DATA

Spent Resin Hold Tank	Parameters
Number	1
Capacity (gal)	2,000
Operating pressure	50
Design temperature (°F)	200
Material of construction	SS
Spent Resin Dewatering Tank	
Number	1
Capacity (gal)	500
Operating pressure	Atmospheric
Design temperature (°F)	200
Material of construction	SS or FRP
Binder Storage Tank (not used)	
Number	1
Capacity (gal)	6,000
Operating pressure	Atmospheric
Design temperature (°F)	100
Material of construction	SS
Spent Resin Recycle Pump	
Number	1
Capacity (gpm)	125
Operating pressure (psig)	75
Design temperature (°F)	100
Material of construction	SS

TABLE 11.4-2 RADIOACTIVE SOLID WASTE SYSTEM COMPONENT DATA

Spent Resin Transfer Pump	
Number	1
Capacity (gpm)	120
Operating pressure (psig)	146
Design temperature (°F)	100
Metal of construction	SS
Binder Pump (not used)	
Number	1
Capacity (gpm)	65
Design temperature (°F)	150
Material of construction	Cast iron
Spent Resin Transfer Pump Filter	Parameters
Number	1
Capacity (gpm)	150
Design pressure (psig)	150
Design temperature (°F)	200
Material of construction	SS

NOTE:

(1) Designed in accordance with ASME VIII, Division I.

TABLE 11.4-3 OMITTED

TABLE 11.4-4 RADIOACTIVE SOLID WASTE ANNUAL SHIPMENTS (1)(2)

Type of Waste and Packaging	Expected	Design
Solidified/Dewatered Wastes (3)		
Containers/shipments	13 containers/13 shipments	33 containers/33 shipments
Miscellaneous Incompressible Solids		
128 cu ft crates/shipments	4 crates/1 shipment	8 crates /1 shipment

NOTES:

(1) Condensate polishing spent resins are designed to be processed by MP2.

(2) Miscellaneous compressible solids for Millstone 3 are processed at the MRRF.

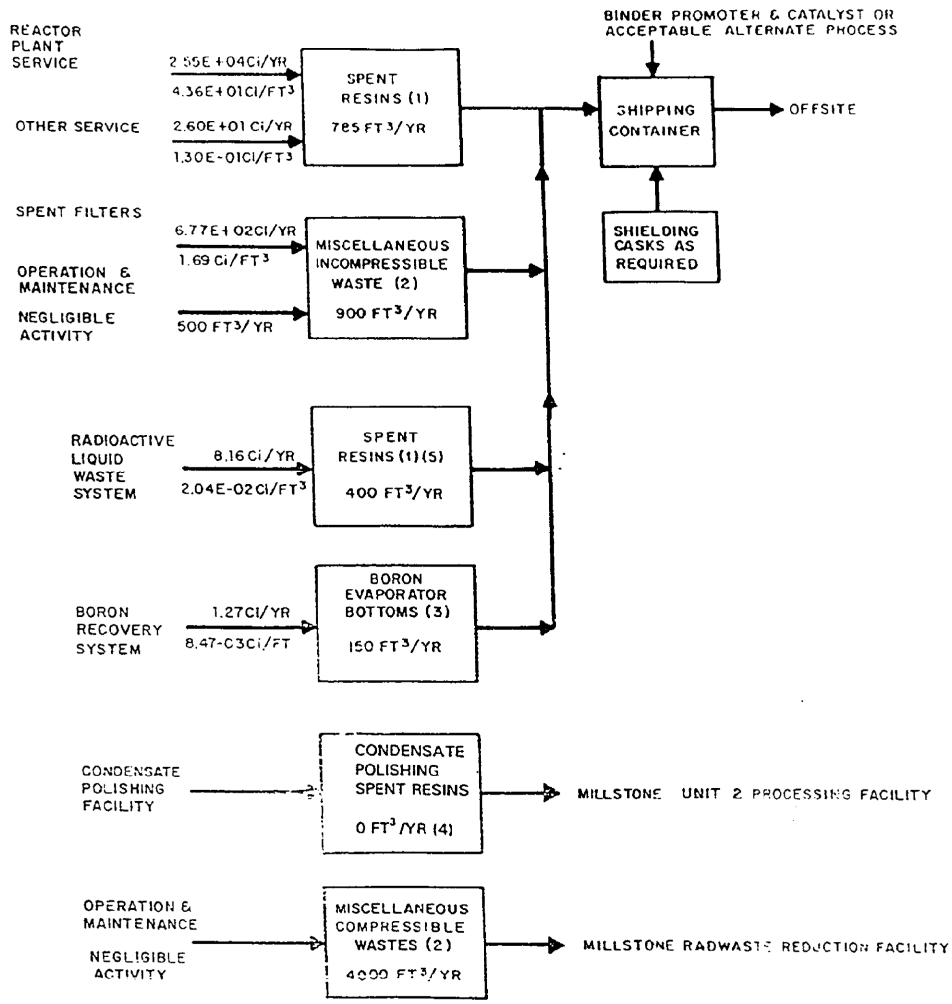
(3) If evaporators are used to process waste instead of resins, then expected shipments will be 56 containers/56 shipments. Evaporation is no longer used for waste processing.

FIGURE 11.4–1 P&ID RADIOACTIVE SOLID WASTE (SHEETS 1-2)

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-3 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

FIGURE 11.4–2 (HISTORICAL) RADIOACTIVE SOLID WASTE SYSTEM EXPECTED QUANTITIES

Historical, not subject to future updating. Has been retained to preserve orignal design basis.

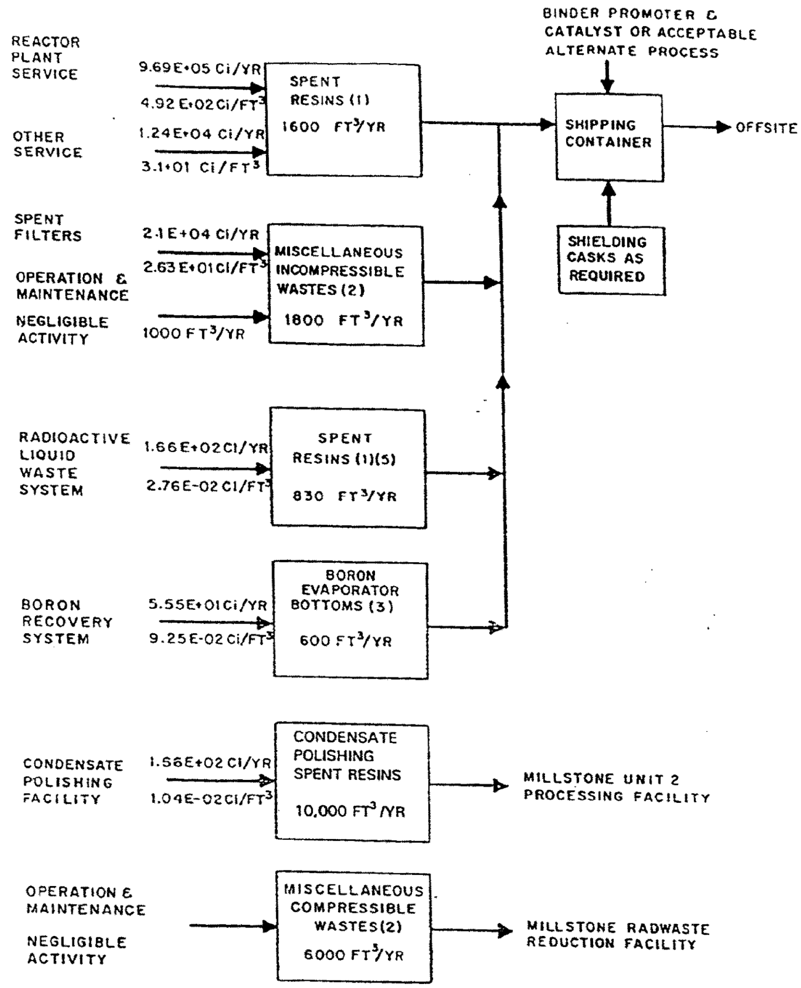


NOTES:

1. Ci FT³/YR VALUES BASED UPON VOLUME OF RAW SPENT RESIN.
2. Ci FT³/YR VALUES BASED UPON VOLUME OF PACKAGED WASTE.
3. Ci FT³/YR VALUES BASED UPON VOLUME OF RAW BOTTOMS.
4. NO CONDENSATE POLISHING SPENT RESINS ARE EXPECTED TO BE GENERATED FOR NORMAL EXPECTED RADIATION LEVELS.
5. ALTERNATE METHOD WOULD PRODUCE APPROXIMATELY 3,000 FT³/YR OF RAW EVAPORATOR BOTTOMS. THIS METHOD WOULD NOT BE THE NORMAL OR PREFERRED METHOD OF DISPOSAL.

FIGURE 11.4–3 (HISTORICAL) RADIOACTIVE SOLID WASTE SYSTEM DESIGN QUANTITIES

Historical, not subject to future updating. Has been retained to preserve orignal design basis.



NOTES:

1. Ci FT³/YR VALUES BASED UPON VOLUME OF RAW SPENT RESIN.
2. Ci FT³/YR VALUES BASED UPON VOLUME OF PACKAGED WASTE.
3. Ci FT³/YR VALUES BASED UPON VOLUME OF RAW BOTTOMS.
4. WHEN RADIATION LEVELS IN THE CONDENSATE REQUIRE THE PROCESSING OF RESIN
5. ALTERNATE METHOD WOULD PRODUCE APPROXIMATELY 6,000 FT³/YR OF RAW EVAPORATOR BOTTOMS. THIS METHOD WOULD NOT BE THE NORMAL OR PREFERRED METHOD OF DISPOSAL.

11.5 PROCESS, EFFLUENT, AND AIRBORNE RADIATION MONITORING SYSTEMS

11.5.1 DESIGN BASES

The process, effluent, and airborne radiation monitoring system (RMS) is designed in accordance with NRC General Design Criterion (GDC) 64 (Section 3.1.2.64) and American National Standards Institute (ANSI) N13.1-1969.

Normal and potential paths for release of radioactive materials, during both normal operation and anticipated operational occurrences, are continuously monitored to ensure compliance with the requirements of 10 CFR 20, 10 CFR 50, and the guidelines of Regulatory Guide 1.21 (Section 1.8). Sections 11.5.2 and 11.5.3 describe the design features provided to ensure agreement with Regulatory Guide 1.21. Potential pathways for release of radioactive materials during accident conditions are continuously monitored to ensure agreement with the guidelines of Regulatory Guide 1.97 (Sections 1.8, 7.5).

Continuous monitoring means that the system operates essentially uninterrupted for extended periods during normal plant operation, but does not preclude periods when individual monitors may be out of service for maintenance, repair, calibration, etc.

The reactor coolant system (RCS) is monitored continuously for gross activity level. Maintaining the RCS activity within acceptable levels ensures that the activity levels in the normally radioactive auxiliary systems are at acceptable levels. Nonradioactive systems which may become contaminated by leaks from radioactive systems are also monitored continuously. This monitoring ensures that no conditions develop that are potentially hazardous to the operating personnel or to the general public.

In the event of an accident releasing radionuclides, the process, effluent, and airborne RMS and the area RMS (Section 12.3.4), provide information on the concentration and dispersion of radioactivity throughout the plant. This enables operating personnel to evaluate the severity and to mitigate the consequences of the accident.

The following automatic actions are initiated by the process and effluent RMS to mitigate both the consequences of postulated accidents and excessive releases during normal operations.

1. Containment purge (Section 9.4.6) is automatically terminated in the event of a high radiation alarm.

2. Liquid effluent discharges from the turbine building drains are diverted to the radioactive liquid waste systems in the event of a high radionuclide concentration alarm (Section 11.2).

3. Waste gas holdup system effluent release is terminated automatically upon a high radionuclide concentration alarm (Section 11.3.3).

4. Liquid waste system discharge to the circulating water discharge tunnel is automatically halted on a high radionuclide concentration alarm (Section 11.2).

5. Control building ventilation intake is terminated on high radiation alarm, and the entire control building ventilation system is isolated (Section 9.4.0).

6. Discharge from the condensate regenerant demineralizer (removed from service) to the environment was designed to be diverted to the regenerant evaporator feed tank (removed from service) upon a high radionuclide concentration alarm (Section 11.5.2.3.8).

7. Following a high radionuclide concentration alarm from the auxiliary condensate monitor, effluent from the auxiliary condensate flash tank is diverted from the auxiliary condensate feed tank to the auxiliary building sump.

8. A high radionuclide concentration alarm from either hydrogen recombiner ventilation monitor automatically shuts down ventilation from its cubicle (Section 9.4.10).

9. Waste neutralization sump effluent discharge to the circulating water discharge tunnel is redirected back to the waste neutralization sump on a high radionuclide concentration alarm (Section 10.4.6.5).

10. Steam generator blowdown is automatically terminated on high radionuclide concentration alarm on the steam generator blowdown monitor.

11.5.2 SYSTEM DESCRIPTION

11.5.2.1 Instrumentation

The process, effluent, and airborne RMS consists of separate and independent monitors, which incorporate the following features:

1. Each liquid monitor has one detector. Each particulate and gas monitor has one gaseous detector and either a fixed paper disc filter for specific radionuclide laboratory analysis, or a moving filter assembly with an internal particulate detector for gross activity measurement.

2. Each monitor is equipped with a dedicated microprocessor monitoring all its functions. The RMS computer system polls each microprocessor every few seconds. Radiation alarms are displayed and annunciated in the main control room. Many monitors also have local alarms in addition to the control room.

3. For specified monitors, a record of radiological events is printed.

Alert/alarm setpoints are established to allow observation of changes in radioactivity and/or to ensure that release rates are within guidelines established in 10 CFR 20 for effluent pathways. The applicable guidelines are those in Appendix B of 10 CFR 20.

The dedicated microprocessor sends an alarm message to the RMS computer system in the event of local power loss, loss of sample flow, filter failure, or other conditions specific to a given monitor.

Means are provided to purge each fixed volume sample chamber in the system with clean fluid to minimize contamination of the chambers. Sample lines and all surfaces of each sampler exposed to the sample are stainless steel and are run to minimize fixed contamination.

Operability of many detectors can be checked using an installed check source remotely activated from the control room. Tests and calibrations of the radiation monitors are performed at specified intervals. For ease in calibration and maintenance, each detector is located in an easily accessible area and is provided with a local readout device or connects to a plug-in portable indication and control (PIC) module.

All monitors and microprocessors are powered by 120 V AC buses. Monitor skids designed in accordance with Regulatory Guide 1.97 and safety-related monitors, such as the control building inlet and ventilation monitors, use the safety-related Class 1E buses (Section 8.3.1.1.2). All other monitors use regular 120 V AC buses (Section 8.3.1.1.1).

Additionally, all monitors with pumps require 480 V AC power. Safety-related monitors with pumps use safety-related Class 1E buses, and all others use regular 480 V AC buses.

N-16 and Fission Product Main Steam Line Monitor alarms are processed by the plant process computer. All other alarms are displayed visually on the RMS workstations in the control room. These include equipment malfunctions, alarm/high radiation levels, conductivity, sample flow, and sample temperature. Local and control room annunciation is provided for high radiation alarms, as well as an interface to the main plant annunciator panel. All alarms can be acknowledged locally or in the control room.

Additionally, those variables designated Class 1E are also displayed on the Class 1E control room panels, as required by Regulatory Guide 1.97. A digital display and control module is provided for each Class 1E monitor, as well as a dedicated two-pen recorder. In order to record this data at the RMS computer along with that from the non-Class 1E devices, these cabinets are connected to the RMS computer via electronic isolators.

Processes Monitored

Table 11.5-1 gives numbers and location of gaseous process and effluent radiation monitors, and Table 11.5-2 gives numbers and locations of liquid process and effluent radiation monitors.

Sensitivities and Ranges

Each detector has sufficient shielding to ensure that the required sensitivity is achieved for the maximum expected background radiation level at the detector location.

All monitors measure gross concentrations. The output is typically measured in microcuries per cubic centimeter (μCi/cc), with a minimum range of five decades.

11.5.2.2 Process and Effluent Monitors

11.5.2.2.1 Ventilation Vent Monitors—Normal Range

Each skid has its own microprocessor. The two microprocessors are linked by a dedicated interface. When the normal range skid senses the upper level of its range, control is automatically passed to the high range skid. When the radionuclide concentration returns to normal, the reverse happens. Both skids use the same isokinetic nozzle. During high range operation, the normal range gas sampler is automatically isolated.

The sample line and all surfaces of the sampler exposed to the sample are stainless steel. A sample pump provides isokinetic sample flow via a flow control valve which modulates proportional to the flow signal from the process flow transmitter. Isokinetic sample flow is maintained through the normal range skid even after the detectors are isolated during high range operation.

The ventilation vent normal range monitor takes a continuous effluent sample from an isokinetic nozzle common to this normal and high range monitor and draws the gas sample through a particulate and charcoal filter and then to a gas sampling assembly where activity is measured by a beta scintillation detector. Lead shielding is provided to reduce the background radiation to a level that minimizes interference with the detector sensitivity.

After leaving the gas sampler, the sample is returned to the duct downstream of the sample point. A purge system for flushing the sample volume is also provided.

The ventilation vent normal range monitor detector output is transmitted via the dedicated microprocessor to the RMS computer system Class 1E panels located in the control room. Here, the activity level is displayed and also recorded on a two-pen strip chart recorder. Activity levels are also digitally displayed locally at the microprocessor location. Alarm conditions, such as high activity or monitor failure, are indicated by audible and visible alarms in the control room and by visible alarm locally.

11.5.2.2.2 Ventilation Vent Monitor-High Range

The vent sample point is common to that of the ventilation vent normal range monitor and is located downstream of the last point where radioactivity is introduced to the flow stream prior to its release via the ventilation vent stack. Sample flow through the high range monitor is maintained constant via a set hand control valve. Isokinetic flow through this monitor depends on continued operation of the normal range monitor pump to maintain isokinetic flow conditions through the portion of sample line which is common to both monitors. The sample flow is drawn through a fixed 0.3 micron filter paper. The activity of the deposited particulate is continuously monitored by a Geiger-Mueller detector. An alarm will automatically direct flow to another filter assembly at a level which still allows analysis of the filter in a laboratory GeLi detector. An adequate amount of lead shielding around the detector assembly reduces the background radiation to a level that minimizes interference with the detector sensitivity. After passing through the filter paper, the sample passes through an inline easily removable charcoal filter cartridge arrangement and then into a fixed and shielded volume where it is monitored by mid-range and high range monitors.

11.5.2.2.3 Hydrogenated Vent Monitor

The hydrogenated vent monitor continuously monitors the effluent from the gaseous waste system downstream of the charcoal decay beds and prior to their release via the ventilation vent stack. The gas detector, a beta scintillator, is located in the well of an inline gas sampler. Four pi lead shielding is provided in order to reduce the background radiation to a level that minimizes interference with the detector sensitivity.

This monitor’s output is transmitted, indicated, recorded, and alarmed in a manner similar to that of all non-Class 1E process monitors. When high activity is present, gas flow to the ventilation vent stack is isolated. The sample line and all surfaces of the sampler exposed to the sample line are stainless steel. Due to the presence of free hydrogen in this effluent, this monitor is purged with nitrogen.

11.5.2.2.4 Containment Fuel Drop Monitors

The gross radionuclide concentration entering the containment purge air vent is monitored by two redundant containment fuel drop monitors. These monitors, each consist of an ion chamber detector measuring dose rates just above the surface of the refueling canal. These are safety-related, Class 1E, monitors. Due to high radiation levels inside the containment, their microprocessors are located in the auxiliary building.

The outputs of these monitors are transmitted, indicated, recorded, and alarmed in a manner similar to that of the ventilation vent monitor. A high activity indication from either of these monitors automatically isolates containment purge (Section 9.4.6.2) based on the assumption that the high dose rates are due to high airborne activity.

11.5.2.2.5 Supplementary Leak Collection and Release System Monitor

The supplementary leak collection and release system (SLCRS) normal range and high range monitors are identical to the ventilation vent monitors. Using an isokinetic nozzle, the monitor withdraws a sample from the SLCRS exhaust vent prior to its discharge to the Millstone stack.

The SLCRS collects, filters, and releases the leakage from the containment enclosure building and contiguous areas to the atmosphere following a design basis accident (DBA) (Section 6.2.3). During normal operation, this detector monitors the discharge point for the containment vacuum pumps, the condenser air removal system, the reactor plant gaseous vents, and the reactor plant aerated vents. The supplementary leak collection monitor is located downstream of the SLCRS filters and prior to the exhaust discharge point. The SLCRS monitor functions as a final effluent monitor.

This monitor's output is transmitted, indicated, recorded, and alarmed in a manner similar to that of the ventilation vent monitors. An alarm from this monitor warns the operator of a potential problem so that he may take appropriate action, which may include the manual transfer of flow to a standby SLCRS filter bank.

11.5.2.2.6 Condenser Air Ejector Monitor

The condenser air ejector monitor continuously analyzes the gaseous effluents from the condenser air ejector discharge (Section 10.4.2). A gamma scintillation detector is inserted into an in-line air well. The detector is shielded with lead to reduce the background radiation to a level which minimizes interference with the detector sensitivity.

The monitor's output is transmitted, indicated, recorded, and alarmed in a manner similar to that of the hydrogenated vent monitor. Activity readings are indicative of primary-to-secondary leakage.

11.5.2.2.7 Control Building Inlet Ventilation Monitors

The two control building inlet monitors are located in the upper level of the control building in the inlet plenum. Here, they continuously analyze the ventilation being supplied to the control building by measuring gross activity. Each monitor consists of a beta scintillation detector. These are Class 1E, safety-related monitors.

The detector outputs are transmitted, indicated, and alarmed in a manner similar to that of the ventilation vent monitor. A high activity condition initiates a control building isolation signal which will isolate the control building atmosphere from the outside atmosphere (Section 6.4).

11.5.2.2.8 Hydrogen Recombiner Ventilation Monitors

Following a loss-of-coolant accident (LOCA), the hydrogen recombiners were originally installed to be used to eliminate the free hydrogen in the containment atmosphere. However, the containment atmosphere following a LOCA also contains large amounts of gaseous radionuclides. These radionuclides are collected and discharged to the atmosphere via the hydrogen recombiner ventilation system (Section 9.4.10).

Therefore, one safety-related, post-accident, beta scintillation radiation detector is placed within the exhaust duct of each ventilation system, and its dedicated microprocessor is located in the hydrogen recombiner control room. Upon a high radiation alarm, a signal from the monitor automatically activates closure of the supply and exhaust duct dampers securing the ventilation.

The output of these monitors is transmitted, indicated, and alarmed in a manner similar to that of the ventilation vent monitor.

11.5.2.2.9 Normal Range Particulate and Gas Monitors

In addition to the ventilation vent and SLCRS monitors, the RMS also contains 11 normal range particulate and gas monitors. They are of a single, basic skid design. The particulate channels employ either a remotely controlled, variable speed moving paper filter with an integral beta detector, or a fixed paper disc filter for specific radionuclide analysis. The gas channel is a beta detector. Each monitor includes a removable charcoal filter. These monitors are as follows.

1. Reactor Plant Heating and Ventilation System (Section 9.4.2)

   Eight of these monitors are located in the reactor plant heating and ventilation system upstream of the ventilation vent monitor such that any effluent sampled by any one of them is also sampled by the ventilation vent monitor before it is released. These monitors allow operating personnel to locate the source of radionuclide leakage into the air of the relevant spaces. The areas sampled are in auxiliary building, the fuel building, and the waste disposal building.

   These monitors are all designated non-safety related, and their outputs are indicated, recorded, and annunciated similarly to that of the condenser air ejector monitor.

2. The Engineered Safety Features Building Heating and Ventilation System (Section 9.4.4)

   The engineered safety features (ESF) building heating and ventilation system normally discharges directly into the atmosphere. This monitor samples this effluent before release and employs the fixed paper filter disc design for a radionuclide analysis in compliance with Regulatory Guide 1.21. This monitor is also designated non-safety related, and its output is indicated, recorded, and annunciated similarly to that of the condenser air ejector monitor. Following an accident, this system is secured and the ESF building is ventilated by the SLCRS system, which has its own extended range radiation monitor (Section 11.5.2.2.5).

3. The Control Building Heating and Ventilation System (Section 9.4.0)

   Since there are no radionuclide sources within the control building, this system is not a release point. However, this monitor provides an indication of radionuclide concentration within the control building. It provides greater sensitivity for detecting airborne activity in the control room than the control building inlet monitors.

   This monitor is also designated non-safety related, and its output is indicated, recorded, and annunciated similarly to that of the condenser air ejector monitor.

4. The Containment Atmosphere Monitoring System

   This monitor continually withdraws a sample of the containment atmosphere, analyzes it, and returns it to the containment, using dedicated sample lines. These lines are heat traced to prevent condensation and slope backwards toward the containment structure. This prevents the sample lines from becoming sources of radiation. The sample lines contain valves to isolate this monitor upon a containment isolation signal. The removable charcoal filter is not used, Iodine samples are collected using a temporary sampler.

11.5.2.2.10 Main Steam Relief Line Monitors

These four ion chamber monitors, located in the main steam valve building, measure the gross radionuclide concentration in the main steam lines in the event of a steam generator tube failure. They are used to meet the intent of Regulatory Guide 1.97 Rev. 2. Lead shielding is provided to reduce the background radiation to a level which does not interfere with the detector sensitivity.

The main steam relief line monitors' output is transmitted via their dedicated microprocessors to the RMS workstations located in the control room where the activity level is digitally displayed.

A high activity level is indicated by audible and visible alarms in the main control room.

11.5.2.2.11 Turbine Driven Auxiliary Feedwater Pump Discharge Monitor

An alternate effluent path for radionuclides entrained in the main steam is monitored by this detector located in the ESF building. This is an ion chamber gross detector used to meet the intent of Regulatory Guide 1.97, Rev. 2. Lead shielding is provided to reduce the background radiation to a level which does not interfere with the detector sensitivity.

The turbine driven auxiliary feedwater pump discharge monitor's output is transmitted via its dedicated microprocessor to the RMS workstations located in the control room where the activity is digitally displayed. A high activity level is indicated by audible and visible alarms in the main control room.

11.5.2.2.12 Main Steam Line Monitor; N-16 and Fission Product

The monitor has four unshielded gamma scintillation detectors, one mounted on each main steam line at the containment penetration in the main steam valve building. Each detector has two channels, one to measure high energy N-16 activity and one to measure lower energy fission product activity. This monitor inputs all eight channels and an equipment failure signal to the plant process computer. The plant process computer records, displays, and provides main control board annunciation for this monitor. This monitoring system is Non-QA, designed to support Millstone's Primary-to-Secondary leak rate program.

11.5.2.3 Liquid Process Monitors

Table 11.5-2 lists the locations of liquid process monitors and the streams being monitored.

11.5.2.3.1 Containment Recirculation Cooler Service Water Outlet Monitors

During the recirculation phase of emergency core cooling, the containment recirculation cooler service water outlet monitors continuously measure the radionuclide concentration in the service water effluent from each pair of containment recirculation coolers (one monitor per pair of coolers). These are Class 1E, on-line, gamma scintillation, gross detectors, placed atop the discharge pipes just outside the ESF building. As the pipes are underground, only the detectors are so mounted, with their dedicated microprocessors located in the fuel building.

The containment recirculation cooler service water outlet monitor output is transmitted via the dedicated microprocessor to the RMS computer system Class 1E cabinets located in the control room. Here, the concentration is digitally displayed and also recorded on a strip chart recorder. The activity level is also digitally displayed locally at the microprocessor location. A high activity level is indicated by audible and visible alarms locally and in the main control room. These monitors warn of a leak into the service water system, within the containment recirculation coolers.

11.5.2.3.2 Liquid Waste Monitor

The liquid waste monitor continuously analyzes the liquid waste effluent discharge pipe (Section 11.2) downstream of the last possible point of radioactive liquid addition. The detector assembly consists of a gamma scintillation detector inserted into the well of a four-pi, lead shielded liquid sampler. All surfaces in contact with the liquid sample are of stainless steel. Lead shielding is provided in order to reduce the background radiation to a level which does not interfere with the detector sensitivity.

The detector output is transmitted, indicated, recorded, and alarmed in a manner similar to that of the condenser air ejector monitor. A high activity situation initiates closure of a discharge valve, thereby preventing the discharge of effluent to the environment in excess of dose limits. The high activity alarm and isolation function are based on a time averaged activity to verify consistently high levels.

11.5.2.3.3 Steam Generator Blowdown Sample Monitor

The steam generator blowdown sample monitor analyzes the steam generator blowdown effluent (Section 10.4.8) for radioactivity which would be indicative of primary-to-secondary leakage. Samples from each of the four steam generator bottoms are mixed in a common header. This common sample is continuously monitored by a gamma scintillation detector inserted into the well of a four-pi, lead-shielded liquid sampler. All surfaces in contact with the liquid sample are of stainless steel. Lead shielding is provided in order to reduce the background radiation to a level which does not interfere with the detector sensitivity.

This monitor's output is transmitted, indicated, recorded, and alarmed in a manner similar to that of the condenser air ejector monitor. If significant activity is detected, the monitor will automatically isolate the steam generator blowdown.

11.5.2.3.4 Auxiliary Condensate Monitor

The auxiliary condensate monitor continuously analyzes samples drawn from the discharge of the auxiliary condensate flash tank (Section 10.4.10). The detector assembly consists of a gamma scintillation detector inserted into the well of a four pi, lead-shielded liquid sampler. All surfaces in contact with the liquid sample are of stainless steel. Lead shielding is provided in order to reduce the background radiation to a level which does not interfere with the detector sensitivity. An inline conductivity element provides sample specific conductivity measurement, indication, and alarm. A sample cooler reduces the sample temperature to 140°F or less, and inlet and outlet solenoid valves isolate the detector assembly and stop the sample pump automatically on high sample temperatures.

The detector output is transmitted, indicated, recorded, and alarmed in a manner similar to that of the condenser air ejector monitor. During normal operation, activities significantly above background are indicative of a leak into the auxiliary steam system (Section 10.4.10) from one of the systems containing radioactive fluids which exchange heat with the auxiliary steam system. A high radiation alarm will automatically divert flow to auxiliary building sumps.

11.5.2.3.5 Turbine Building Floor Drains Monitor

The turbine building floor drains monitor analyzes a sample from the turbine building floor drains discharge line (Section 11.2), downstream of any possible fluid addition to the discharge line. The detector assembly consists of a gamma scintillation detector inserted into the well of a four-pi, lead shielded liquid sampler. All surfaces in contact with the liquid sample are of stainless steel. An adequate amount of lead shielding is provided in order to reduce the background radiation to a level which does not interfere with the detector sensitivity.

This monitor's output is transmitted, indicated, recorded, and alarmed in a manner similar to that of the condenser air ejector monitor. A high activity level initiates valve action to divert the effluent to the liquid waste system, thereby preventing the discharge of effluent to the environment in excess of dose limits.

11.5.2.3.6 Reactor Plant Component Cooling Water System Monitor

The reactor plant component cooling water system monitor continuously analyzes the component cooling water for radioactivity (Section 9.2.2.1). A sample is continuously withdrawn from the reactor plant component cooling water system downstream of the reactor plant component cooling pumps discharge. The sample is monitored by a gamma scintillation detector inserted into the well of a four-pi, lead shielded liquid sampler. All surfaces in contact with the liquid sample are of stainless steel. Lead shielding is provided in order to reduce the background radiation to a level which does not interfere with the detector sensitivity.

This monitor’s output is transmitted, indicated, recorded, and alarmed in a manner similar to that of the condenser air ejector monitor. During normal operation, activities significantly above background are indicative of a leak into the reactor plant component cooling water system from one of the systems containing radioactive fluids which exchange heat with the reactor plant component cooling water subsystem.

11.5.2.3.7 Deleted by FSARCR 05-MP3-015

11.5.2.3.8 Regenerant Evaporator Monitor (Removed from Service)

Located in the condensate demineralizer liquid waste system (removed from service) the regenerant evaporator monitor was designed to measure gross radionuclide concentration in the distillate discharged from the regenerant evaporator. Following a high concentration alarm, the effluent would be automatically isolated.

An evaluation has been performed which has determined that the LWC is not needed. This evaluation is documented by change to the Radiological Effluent Monitoring Offsite Dose Calculation Manual (ref. REMODCM CR# 95-7).

11.5.2.3.9 Waste Neutralization Sump Monitor

Located in the condensate polishing facility, the waste neutralization sump monitor measures the radionuclide concentration in the effluent from a sump located below potentially contaminated condensate polishing equipment.

The sample is monitored by a gamma scintillation detector inserted into the well of a four-pi, lead shielded liquid sampler.

All surfaces in contact with the liquid sample are of stainless steel. Lead shielding is provided in order to reduce the background radiation to a level which does not interfere with the detector sensitivity.

This monitor’s output is transmitted, indicated, recorded, and alarmed in a manner similar to that of the condenser air ejector monitor.

11.5.2.4 Inservice Inspection, Calibration, and Maintenance

Most channels of the process and effluent RMS are checked routinely with an installed check source or manually at the monitor. Periodic tests are also used to verify the operability of alarms and automatic actions.

Calibration of monitors is conducted periodically in accordance with the technical specifications.

Process and effluent monitors were initially isotopically calibrated at the vendor and concurrently cross calibrated to a secondary National Institute of Science and Technology (NIST) traceable source in a fixed field geometry. Field calibration is accomplished using these secondary sources.

Repair of failed electronic components is accomplished in the units' instrument and control shop. Sufficient spare parts, staffing, and training are being provided to support this effort.

Radiochemical analysis takes place in the chemistry laboratory counting room. The facility is equipped with a gamma spectrometer traceable to NIST for the purpose of isotopic analysis.

11.5.2.5 Sampling

Section 9.3.2 discusses the various process and effluent samples taken periodically for chemical and radiochemical analysis.

Table 11.5-3 lists liquid process and effluent samples to be taken periodically and monitored for radioactivity. Those not covered in Section 9.3.2 are included in the individual system designs. Sampling of these fluid systems is via local sampling connections, e.g., the fuel pool cooling and purification system (Section 9.1.3).

Prior to collecting a sample, liquid sample lines are purged of stagnant water and undissolved solids for a sufficient time to ensure that a representative sample is obtained.

Sample taps suitable for connection to a sampling chamber are provided at all off-line process monitors to obtain a sample for laboratory gamma spectrum analysis.

Gas samples are collected in a sample vessel with valves on each end. After adequate purging of the sample vessel, the gas sample is collected by closing valves at both ends of the sample vessel.

The isokinetic nozzles used for obtaining uniform samples (Section 11.5.2.2) are designed according to the guidelines of ANSI N13.1.

A sampling room (Section 9.3.2) is provided for remote sampling. Recirculation loops are separated from the sample taps by shielding. Sample sink drains are collected and sent to various systems, depending on the nature of the sample being taken, for reclaiming or processing as necessary. Sample sinks are provided with ventilation exhaust hoods.

11.5.3 REFERENCES FOR SECTION 11.5

11.5-1 ANSI N13.1 - 1969. Guide to Sampling Airborne Radioactive Materials in Nuclear Facilities, American National Standards Institute (1974).

11.5-2 EPRI TR104788-R2 PWR Primary-To-Secondary Leak Guidelines - Revision 2.

TABLE 11.5-1 GASEOUS MONITORS

Monitor	Mark Number (1)	Number of Channels	Medium	Location	Measurement Range
Ventilation Vent (2)	3HVR*RE10		Air	Aux Bldg 66'-6"	
Normal Range		1			Normal OPS
High Range		1			Post Accident
Hydrogenated Vent (2)	3GWS-RE48	1	Gas	Aux Bldg 43'-6"	Normal OPS
Fuel Drop (3)	3RMS*RE41	2 (4)	Air	Containment 51'-4"	Normal OPS
	3RMS*RE42				
Supplementary Leak Collection (2)	3HVR*RE19		Air	Aux Bldg 66'-6"	
Normal Range		1			Normal OPS
High Range		1			Post Accident
Condenser Air Ejector (5)	3ARC-RE21	1	Air	Turbine Bldg 38'-6"	Normal OPS -- Will Alarm With Tube Failure
Control Building Inlet Ventilation (5)	3HVC*RE16A	2 (4)	Air	Control Bldg 64'-6"	Post Accident
	3HVC*RE16B				
Containment Atmosphere (2)	3CMS*RE22		Air	Aux Bldg 66'-6"	
Particulate		1			Normal OPS
Gas		1			Normal OPS
Auxiliary Building (2)	3HVR-RE11		Air	Aux Bldg 66'-6" & 43'-6"	
Particulate	3HVR-RE12	6			Normal OPS

TABLE 11.5-1 GASEOUS MONITORS (CONTINUED)

Monitor	Mark Number (1)	Number of Channels	Medium	Location	Measurement Range
Gas	3HVR-RE13	6			Normal OPS
	3HVR-RE14				
	3HVR-RE15				
	3HVR-RE16				
Fuel Building (2)	3HVR-RE17		Air	Aux Bldg 66'-6"	
Particulate		1			Normal OPS
Gas		1			Normal OPS
Waste Disposal (2)	3HVR-RE18		Air	Aux Bldg 66'-6"	
Particulate		1			Normal OPS
Gas		1			Normal OPS
Control Building (2)	3HVC-RE91		Air	Control Bldg 64'-6"	
Particulate		1			Normal OPS
Gas		1			Normal OPS
ESF Building (2)	3HVQ-RE49		Air	ESF Bldg 36'-6"	
Particulate		Note 6			Normal OPS
Gas		1			Normal OPS
Hydrogen (5) Recombiner Cubicle Vent	3HVZ*RE09A	2 (4)	Air	HR Bldg 37'-6"	Post Accident
	3HVZ*RE09B				
Main Steam Relief Line (5)	3MSS-RE75 to 78	4	Steam	MSVB 70'-6"	Post Accident

TABLE 11.5-1 GASEOUS MONITORS (CONTINUED)

Monitor	Mark Number (1)	Number of Channels	Medium	Location	Measurement Range
Turbine Driven Auxiliary Feedwater Pump Discharge (5)	3MSS-RE79	1	Steam	ESF 36'-6"	Post Accident
Main Steam Line	3MSS-RE80A				
N-16	3MSS-RE80B	4	Steam	MSVB 70’6”	Normal OPS
Fission Product	3MSS-RE80C	4	Steam	MSVB 70’6”	Normal OPS
	3MSS-RE80D				

NOTES:

(1) A and B used to indicate redundant monitors powered from separate safety trains.

(2) Offline monitors.

(3) The fuel drop monitors are configured as high range area monitors, having a minimum sensitivity of 0.1 R/hr and a range of 6 decades.

(4) Redundant monitors.

(5) Inline monitors.

(6) Offline laboratory radionuclide analysis.

TABLE 11.5-2 LIQUID PROCESS MONITORS

Monitor	Mark Number (1)	Number of Channels	Medium	Location	Measurement Range
Containment Recirculation Cooler Service Water Outlet (2)	3SWP*RE60A	2	Water	Yard	Post Accident
	3SWP*RE60B				
Liquid Waste (3)	3LWS-RE70	1	Water	Aux Bldg 4'-6"	Normal OPS
Steam Generator Blowdown Sample (3)	3SSR-RE08	1	Water	Aux Bldg 43'-6"	Normal OPS
Auxiliary Condensate (3)	3CNA-RE47	1	Water	Aux Bldg 4'-6"	Normal OPS
Turbine Building Floor Drains (3)	3DAS-RE50	1	Water	Turbine 14'-6"	Normal OPS
Reactor Plant Component Cooling Water Subsystem (3)	3CCP-RE31	1	Water	Aux Bldg 43'-6"	Normal OPS
Regenerant Evaporator (3) (Removed from Service)	3LWC-RE65	1	Water	Warehouse 5 4'-6"	Normal OPS
Waste Neutralization Sump (3)	3CND-RE07	1	Water	Condensate Polishing Facility 14'-6"	Normal OPS

(1) A and B used to indicate redundant monitors powered from separate safety trains.

(2) Inline monitors.

(3) Offline monitors.

TABLE 11.5-3 RADIOLOGICAL SAMPLES TAKEN AT REACTOR PLANT SAMPLE SINK

Sample Locations	Number
Reactor Coolant System (Chapter 5)	
Loop No. 1 (Hot Leg)	1
Loop No. 3 (Hot Leg)	1
Pressurizer Vapor Space	1
Residual Heat Removal System (Chapter 5.4.7)	
RHR Heat Exchanger Outlet	2
Reactor Plant Component Cooling Water System (Section 9.2.2.1)	
Pump Discharge	2
Primary Grade Water System (Section 9.2.8)	
Primary Grade Water Tanks	2
Chemical and Volume Control System (Section 9.3.4)	
Letdown Heat Exchanger Outlet	2
Reactor Coolant Filter Inlet	1
Volume Control Tank	1
Boron Thermal Regeneration Outlet	1
Boron Recovery System (Section 9.3.5)	
Boron Recovery Tanks	2
Boron Test Tanks	2
Steam Generator Blowdown System (Section 10.4.8)	
Blowdown Sample	4
Radioactive Liquid Waste System (Section 11.2)	
Waste Test Tank	2
Radioactive Gaseous Waste System (Section 11.3)	
Degasifier Condenser Process Effluent	1

APPENDIX 11A

Part I—Summary of Annual Radiation Doses (Historical)

Part II—Dose Calculation Models and Assumptions (Historical)

Part III—Cost-benefit Analysis (Historical)

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

PART I—SUMMARY OF ANNUAL RADIATION DOSES (HISTORICAL)

The dose evaluation presented in this Appendix was developed in support of the original license and portions were updated during the MPS-3 restart. These dose estimates are considered historical and not subject to future updating. This information is retained to avoid loss of original design basis.

As stated in Section 11.0, the Radiological Effluent Monitoring and Offsite Dose Calculation Manual (REMODCM) provides guidance requirements for system operation, dose calculations, and monitoring requirements to ensure MPS-3 compliance with effluent limits. Actual measured concentrations of radioactivity released and real time dilution and dispersion estimates are utilized to verify compliance with effluent limits. Therefore, MPS-3 operation within the requirements of the REMODCM ensures compliance within effluent limits, rather than operations within the nominal assumptions utilized in the dose evaluation presented in this Chapter.

The calculated annual radiation doses to the maximum individual from liquid and gaseous pathways are presented in Tables 11A.1-1 through 11A.1-8 and 11A.1-13 through 11A.1-15. Table 11A.1-16 demonstrates that the calculated annual radiation doses are below the design objectives of 10 CFR 50, Appendix I.

The maximum calculated organ dose per reactor for an individual from gaseous releases (particulates and radioiodines) is 4.4 mRem/yr to an infant's thyroid. This represents a hypothetical infant living at the residence 2.4 km north-northeast of the site consuming milk from a goat at the same location.

The calculated external exposure to the whole body and skin from immersion in noble gases is 3.8E-02 and 6.9E-02 mRem/yr, respectively. These represent the maximum values which occur at the site boundary in the direction of the maximum overland χ/Q, 650 meters east-northeast of Millstone 3. The maximum calculated beta and gamma air doses from noble gas releases are 6.6E-02 and 8.6E-02 mrad/yr, respectively. These were also calculated 650 meters east-northeast of Millstone 3.

For liquid releases, the maximum individual was assumed to consume aquatic foods whose principal habitat is the edge of the initial mixing zone (EIMZ). This location was also conservatively used in calculating doses from boating. Doses from swimming and shoreline recreation were calculated at the nearest resident's beach, located 1.1 km from the point of discharge. The maximum calculated whole body dose for an individual from liquid pathways is 2.4E-02 mRem/yr in the adult age group. The maximum calculated organ dose for an individual from liquid pathways is 4.4E-01 mRem/yr to an adult's GI-UI and 1.8E-01 mRem/yr to a child's thyroid. These doses were primary due to consumption of aquatic foods.

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

The calculated annual gaseous and liquid doses from the population residing within an 80 km radius of the site are presented in Table 11A.1-17. For liquid effluents, the calculated population dose commitment within 80 km for whole body and thyroid are 1.7E+00 and 1.6E+01 man-Rem/yr, respectively.

For the gaseous effluents, the calculated population doses within 80 km from noble gas effluents and radioiodines and particulates are 4.8 man-Rem/yr whole body and 7.8 man-Rem/yr thyroid.

Population doses were calculated for a projected population of 3.3 million people residing within 80 km of the site in the year 2010.

The calculated annual gaseous and liquid doses to the contiguous U.S. population are also presented in Table 11A.1-17. For liquid effluents, the calculated dose to the contiguous U.S. population is 1.7E+00 man-Rem whole body and 1.6E+01 man-Rem thyroid. For gaseous effluents, the calculated dose to the contiguous U.S. population is 2.2E+01 man-Rem whole body and 2.5E+01 man-Rem thyroid.

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

TABLE 11A.1-1 ANNUAL DOSES TO MAXIMUM INDIVIDUAL IN THE ADULT GROUP FROM GASEOUS EFFLUENTS

(Residence 0.81 km ENE) Annual Dose (mRem/yr)

Pathway	Total Body	Skin	Bone	Liver	Thyroid	Kidney	Lung	GI-tract
Contaminated ground	1.7E+00 (1)	2.0E+00	1.7E+00	1.7E+00	1.7E+00	1.7E+00	1.7E+00	1.7E+00
Inhalation	1.2E-01	0.0	8.6E-03	1.2E-01	2.5E-01	1.1E-01	1.4E-01	1.1E-01
Fresh vegetation	9.7E-02	0.0	8.3E-02	1.2E-01	1.5E+00	6.2E-02	3.5E-02	6.0E-02
Stored vegetation	5.7E-01	0.0	4.7E-01	6.7E-01	2.2E-01	3.3E-01	2.2E-01	3.2E-01
Total dose	2.5E+00	2.0E+00	2.3E+00	2.6E+00	3.7E+00	2.2E+00	2.1E+00	2.2E+00

NOTE:

(1) 1.7E+00 = 1.7×10^0

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

TABLE 11A.1-2 ANNUAL DOSES TO MAXIMUM INDIVIDUAL IN THE TEEN GROUP FROM GASEOUS EFFLUENTS

(Residence 0.81 km ENE) Annual Dose (mRem/yr)

Pathway	Total Body	Skin	Bone	Liver	Thyroid	Kidney	Lung	GI-tract
Contaminated ground	1.7E+00 (1)	2.0E+00	1.7E+00	1.7E+00	1.7E+00	1.7E+00	1.7E+00	1.7E+00
Inhalation	1.2E-01	0.0	1.1E-02	1.2E-01	2.9E-01	1.2E-01	1.6E-01	1.2E-01
Fresh vegetation	5.6E-02	0.0	7.1E-02	9.7E-02	1.2E+00	4.9E-02	2.7E-02	3.8E-02
Stored vegetation	5.9E-01	0.0	4.7E-01	1.0E+00	2.8E-01	4.7E-01	3.0E-01	3.7E-01
Total dose	2.5E+00	2.0E+00	2.5E+00	2.9E+00	3.5E+00	2.3E+00	2.2E+00	2.2E+00

NOTE:

(1) 1.7E+00 = 1.7×10^0

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

TABLE 11A.1-3 ANNUAL DOSES TO MAXIMUM INDIVIDUAL IN THE CHILD GROUP FROM GASEOUS EFFLUENTS

(Residence 0.81 km ENE) Annual Dose (mRem/yr)

Pathway	Total Body	Skin	Bone	Liver	Thyroid	Kidney	Lung	GI-tract
Contaminated ground	1.7E+00 (1)	2.0E+00	1.7E+00	1.7E+00	1.7E+00	1.7E+00	1.7E+00	1.7E+00
Inhalation	1.0E-01	0.0	1.4E-02	1.1E-01	3.1E-01	1.0E-01	1.4E-01	1.0E-01
Fresh vegetation	5.0E-02	0.0	1.2E-01	1.2E-01	1.8E+00	5.9E-02	3.1E-02	3.1E-02
Stored vegetation	6.9E-01	0.0	1.6E+00	1.7E+00	4.8E-01	7.6E-01	4.7E-01	4.3E-01
Total dose	2.5E+00	2.0E+00	3.4E+00	3.6E+00	4.3E+00	2.6E+00	2.3E+00	2.3E+00

NOTE:

(1) 1.7E+00 = 1.7×10^0

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

TABLE 11A.1-4 ANNUAL DOSES TO MAXIMUM INDIVIDUAL IN THE INFANT GROUP FROM GASEOUS EFFLUENTS

(Residence 0.81 km ENE) Annual Dose (mRem/yr)

Pathway	Total Body	Skin	Bone	Liver	Thyroid	Kidney	Lung	GI-tract
Contaminated ground	1.7E+00 (1)	2.0E+00	1.7E+00	1.7E+00	1.7E+00	1.7E+00	1.7E+00	1.7E+00
Inhalation	5.9E-02	0.0	9.1E-03	6.2E-02	2.5E-01	6.0E-02	8.2E-02	5.8E-02
Total dose	1.8E+00	2.0E+00	1.7E+00	1.8E+00	1.9E+00	1.8E+00	1.8E+00	1.8E+00

NOTE:

(1) 1.7E+00 = 1.7×10^0

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

TABLE 11A.1-5 ANNUAL DOSES TO MAXIMUM INDIVIDUAL IN THE ADULT GROUP FROM GASEOUS EFFLUENTS

(Residence 2.4 km NNE; Goat Pasture 2.4 km NNE)

Annual Dose (mRem/yr)

Pathway	Total Body	Skin	Bone	Liver	Thyroid	Kidney	Lung	GI-tract
Contaminated ground	1.8E-01 (1)	2.2E-01	1.8E-01	1.8E-01	1.8E-01	1.8E-01	1.8E-01	1.8E-01
Inhalation	3.6E-02	0.0	3.4E-03	3.7E-02	6.2E-02	3.6E-02	4.4E-02	3.6E-02
Fresh vegetation	2.0E-02	0.0	1.4E-02	2.2E-02	1.9E-01	1.4E-02	1.1E-02	1.4E-02
Stored vegetation	1.2E-01	0.0	8.1E-02	1.3E-01	6.4E-02	8.2E-02	6.6E-02	8.1E-02
Goat milk	1.8E-01	0.0	1.2E-01	2.4E-01	5.8E-01	1.1E-01	6.6E-02	5.0E-02
Total dose	5.4E-01	2.2E-01	4.0E-01	6.1E-01	1.1E+01	4.2E-01	3.7E-01	3.6E-01

NOTE:

(1) 1.8E-01 = 1.8×10^{-1}

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

TABLE 11A.1-6 ANNUAL DOSES TO MAXIMUM INDIVIDUAL IN THE TEEN GROUP FROM GASEOUS EFFLUENTS

(Residence 2.4 km NNE; Goat Pasture 2.4 km NNE)

Annual Dose (mRem/yr)

Pathway	Total Body	Skin	Bone	Liver	Thyroid	Kidney	Lung	GI-tract
Contaminated ground	1.8E-01 (1)	2.2E-01	1.8E-01	1.8E-01	1.8E-01	1.8E-01	1.8E-01	1.8E-01
Inhalation	3.7E-02	0.0	4.6E-03	3.8E-02	6.9E-02	3.7E-02	4.8E-02	3.6E-02
Fresh vegetation	1.2E-02	0.0	1.2E-02	1.8E-02	1.5E-01	1.1E-02	7.7E-03	9.3E-03
Stored vegetation	1.3E-01	0.0	1.3E-01	1.9E-01	8.2E-02	1.1E-01	8.7E-02	9.8E-02
Goat milk	1.9E-01	0.0	2.2E-01	3.9E-01	9.0E-01	1.7E-01	1.0E-01	6.5E-02
Total dose	5.5E-01	2.2E-01	5.5E-01	8.2E-01	1.4E+00	5.1E-01	4.2E-01	3.9E-01

NOTE:

(1) 1.8E-01 = 1.8×10^{-1}

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

TABLE 11A.1-7 ANNUAL DOSES TO MAXIMUM INDIVIDUAL IN THE CHILD GROUP FROM GASEOUS EFFLUENTS

(Residence 2.4 km NNE; Goat Pasture 2.4 km NNE)

Annual Dose (mRem/yr)

Pathway	Total Body	Skin	Bone	Liver	Thyroid	Kidney	Lung	GI-tract
Contaminated ground	1.8E-01 (1)	2.2E-01	1.8E-01	1.8E-01	1.8E-01	1.8E-01	1.8E-01	1.8E-01
Inhalation	3.2E-02	0.0	6.1E-03	3.4E-02	7.0E-02	3.3E-02	4.2E-02	3.2E-02
Fresh vegetation	1.2E-02	0.0	2.1E-02	2.2E-02	2.2E-01	1.3E-02	9.3E-03	9.3E-03
Stored vegetation	1.7E-01	0.0	2.9E-01	3.2E-01	1.4E-01	1.8E-01	1.4E-01	1.4E-01
Goat milk	2.0E-01	0.0	5.2E-01	6.5E-01	1.8E+00	2.8E-01	1.6E-01	1.0E-01
Total dose	5.9E-01	2.2E-01	1.0E+00	1.2E+00	2.4E+00	6.9E-01	5.3E-01	4.6E-01

NOTE:

(1) 1.8E-01 = 1.8×10^{-1}

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

TABLE 11A.1-8 ANNUAL DOSES TO MAXIMUM INDIVIDUAL IN THE INFANT GROUP FROM GASEOUS EFFLUENTS

(Residence 2.4 km NNE; Goat Pasture 2.4 km NNE)

Annual Dose (mRem/yr)

Pathway	Total Body	Skin	Bone	Liver	Thyroid	Kidney	Lung	GI-tract
Contaminated ground	1.8E-01 (1)	2.2E-01	1.8E-01	1.8E-01	1.8E-01	1.8E-01	1.8E-01	1.8E-01
Inhalation	1.9E-02	0.0	4.1E-03	2.0E-02	5.3E-02	1.9E-02	2.5E-02	1.9E-02
Goat milk	2.4E-01	0.0	8.5E-01	1.2E+00	4.2E+00	4.4E-01	2.6E-01	1.5E-01
Total dose	4.4E-01	2.2E-01	1.0E+00	1.4E+00	4.4E+00	6.4E-01	4.6E-01	3.5E-01

NOTE:

(1) 1.8E-01 = 1.8×10^{-1}

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

TABLE 11A.1-9 OMITTED

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

TABLE 11A.1-10 OMITTED

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

TABLE 11A.1-11 OMITTED

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

TABLE 11A.1-12 OMITTED

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

TABLE 11A.1-13 ANNUAL DOSES TO MAXIMUM INDIVIDUAL IN THE ADULT GROUP FROM LIQUID EFFLUENTS

Annual Dose (mRem/yr)

Pathway	Skin	Bone	Liver	Total Body	Thyroid	Kidney	Lung	GI-LLI
Fish	0.00E+00	7.29E-03	1.40E-02	1.04E-02	7.72E-02	5.15E-03	1.90E-03	1.02E-02
Invertebrate	0.00E+00	1.97E-02	1.34E-02	1.05E-02	1.02E-01	1.01E-01	6.33E-04	4.26E-01
Shoreline	3.18E-03	2.73E-03	2.73E-03	2.73E-03	2.73E-03	2.73E-03	2.73E-03	2.73E-03
Swimming	0.00E+00	2.89E-05	2.89E-05	2.89E-05	2.89E-05	2.89E-05	2.89E-05	2.89E-05
Boating	0.00E+00	1.80E-05	1.80E-05	1.80E-05	1.80E-05	1.80E-05	1.80E-05	1.80E-05
Total	3.18E-03	2.98E-02	3.02E-02	2.37E-02	1.82E-01	1.09E-01	5.31E-03	4.39E-01

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

TABLE 11A.1-14 ANNUAL DOSES TO MAXIMUM INDIVIDUAL IN THE TEEN GROUP FROM LIQUID EFFLUENTS

Annual Dose (mRem/yr)

Pathway	Skin	Bone	Liver	Total Body	Thyroid	Kidney	Lung	GI-LLI
Fish	0.00E+00	7.63E-03	1.43E-02	6.28E-03	7.24E-02	5.16E-03	2.11E-03	7.29E-03
Invertebrate	0.00E+00	2.09E-02	1.38E-02	1.02E-02	9.63E-02	1.05E-01	7.12E-04	3.01E-01
Shoreline	3.18E-03	2.73E-03	2.73E-03	2.73E-03	2.73E-03	2.73E-03	2.73E-03	2.73E-03
Swimming	0.00E+00	2.89E-05	2.89E-05	2.89E-05	2.89E-05	2.89E-05	2.89E-05	2.89E-05
Boating	0.00E+00	1.80E-05	1.80E-05	1.80E-05	1.80E-05	1.80E-05	1.80E-05	1.80E-05
Total	3.18E-03	3.13E-02	3.09E-02	1.93E-02	1.71E-01	1.12E-01	5.60E-03	3.11E-01

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

TABLE 11A.1-15 ANNUAL DOSES TO MAXIMUM INDIVIDUAL IN THE CHILD GROUP FROM LIQUID EFFLUENTS

Annual Dose (mRem/yr)

Pathway	Skin	Bone	Liver	Total Body	Thyroid	Kidney	Lung	GI-LLI
Fish	0.00E+00	9.43E-03	1.23E-02	2.95E-03	7.57E-02	4.34E-03	1.67E-03	2.74E-03
Invertebrate	0.00E+00	2.73E-02	1.27E-02	1.14E-02	1.06E-01	9.31E-02	5.94E-04	9.47E-02
Shoreline	1.78E-03	1.53E-03	1.53E-03	1.53E-03	1.53E-03	1.53E-03	1.53E-03	1.53E-03
Swimming	0.00E+00	1.62E-05	1.62E-05	1.62E-05	1.62E-05	1.62E-05	1.62E-05	1.62E-05
Boating	0.00E-00	1.00E-05	1.00E-05	1.00E-05	1.00E-05	1.00E-05	1.00E-05	1.00E-05
Total	1.78E-03	3.83E-02	2.66E-02	1.59E-02	1.84E-01	9.90E-02	3.82E-03	9.90E-02

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

TABLE 11A.1-16 COMPARISON OF MAXIMUM CALCULATED DOSES FROM MILLSTONE 3 NUCLEAR PLANT WITH APPENDIX I DESIGN OBJECTIVES

Criterion	Appendix I Design Objective (1)	Calculated Dose
Gaseous Effluents		
Gamma air dose	10 mrad/yr	8.6E-02 (2) mrad/yr (3)
Beta air dose	20 mrad/yr	6.6E-02 mrad/yr (3)
Noble gas - total body	5 mRem/yr	3.8E-02 mRem/yr (3)
Noble gas - skin	15 mRem/yr	6.9E-02 mRem/yr (3)
Iodines and particulates, any organ	15 mRem/yr	4.4E+00 mRem/yr (4)
Liquid Effluents:		
Total body	3 mRem/yr	2.4E-02 mRem/yr
Any organ	10 mRem/yr	4.4E-01 mRem/yr (5)

NOTES:

(1) Per reactor.

(2) 8.6E-02 = 8.6×10^{-2}.

(3) Site boundary 650 meters ENE of Millstone 3.

(4) Infant thyroid dose at residence with goat, 2.4 km NNE.

(5) Adult GI-UI dose is calculated to be the highest organ dose.

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

TABLE 11A.1-17 CALCULATED POPULATION DOSE

80-KM POPULATION DOSE		
	Annual Dose Per Reactor Unit	
	Total Body (man-Rem)	Thyroid (man-Rem)
Natural radiation background (1)	3.3E+05 (2)	3.3E+05
Liquid effluents	1.7E+00	1.6E+01
Noble gas effluents	1.3E-01	1.3E-01
Radioiodines and particulates (3)	4.7E+00	7.7E+00

CONTIGUOUS U.S. POPULATION DOSE		
	Annual Dose Per Reactor Unit	
	Total Body (man-Rem)	Thyroid (man-Rem)
Liquid effluents	1.7E+00	1.6E+0
Noble gas effluents	1.5E-0	4.0E-0
Radioiodines and particulates (3)	2.2E+0	2.5E+0

NOTES:

(1) Natural Radiation Exposure in the United States, U.S. Environmental Protection Agency, ORP-SID-72-1 (June 1972), using the average state background dose (100 mRem/yr), and year 2010 projected population of 3.3 million.

(2) 3.3E+05 = 3.3×10^5

(3) Carbon-14 and tritium have been added to this category.

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

PART II—DOSE CALCULATION MODELS AND ASSUMPTIONS (HISTORICAL)

Doses to Humans

Calculation of dose rates to the maximum individual and to the population residing within an 80-km radius of the site are based on the methodology and equations of U.S. Nuclear Regulatory Guide 1.109 "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for The Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I, Revision 1, October 1977." Stone & Webster Engineering Corporation Computer Programs IND1109E, POP1109E, NG1109E, DUCKMANE, and NEPA were used in the design case analysis of the maximum individual and population doses. These computer codes have been verified by comparing the Stone & Webster program outputs to results obtained from the U.S. NRC Computer Codes GASPAR and LADTAP and to hand calculations, where appropriate. The U.S. NRC Computer Code, LADTAP II was used for the expected case analysis. The U.S. NRC Computer Code RABFIN was used to analyze the elevated release of noble gases. NRC default values have been used in lieu of site specific data, where site data was unavailable. The site specific data that was used for this analysis is listed in Tables 11A.2-1, 11A.2-2, and 11A.2-3.

The following sections present the equations used in the analysis for each pathway considered.

Doses from Liquid Pathways

The generalized equation for calculating radiation doses to humans via liquid pathways is:

$$R_{aipj} = (C_{ip})(U_{ap})(D_{aipj}) \qquad (11A.2\text{-}1)$$

where:

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

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

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

D_{aipj} = the dose factor, specific age group a, radionuclide i, pathway p, and organ j, in mRem/pCi ingested or mRem per hr/pCi per sq m from exposure to deposited activity in sediment or on the ground

1. Aquatic Foods

$$R_{apj} = 1100\frac{U_{ap}M_p}{F}\sum_i Q_i B_{ip} D_{aipj}\exp(-\lambda_i t_p) \qquad (11A.2\text{-}2)$$

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

where:

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

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

F = the flow rate of the liquid effluent in cu ft/s

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

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

i = the radioactive decay constant of nuclide i, in hr

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

1,100 = the factor to convert from (Ci/yr)/(cu ft/s) to pCi/l

All the other symbols are as previously defined.

2. Doses from Shoreline Deposits Foods

$$R_{apj} = 110,000 \frac{U_{ap} M_p W}{F} \sum_i Q_i T_i D_{aipj} [\exp(-\lambda_i t_p)][1 - \exp(-\lambda_i t_b)] \quad (11A.2\text{-}3)$$

where:

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

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

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

110,000 = the factor to convert from (Ci/yr)/(cu ft/s) to pCi/l and to account for the proportionality constant used in the sediment radioactivity model

All other symbols are as previously defined.

3. Doses from Swimming and Boating

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

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

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

$$R_{apj} = 1100 \frac{U_{ap} M_p}{F K_p} \sum_i Q_i D_{aipj} \exp(-\lambda_i t_p) \qquad (11A.2\text{-}4)$$

where:

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

All other symbols are as previously defined.

<u>Doses from Air Pathways</u>

1. Gamma and Beta Doses from Noble Gas Discharged to the Atmosphere

a. Annual Gamma and Beta Air Doses from Noble Gas Releases (ground level)

$$D^{\gamma}(r, \theta) \text{ or } D^{\beta}(r, \theta) = 3.17 \times 10^4 \sum_i Q_i [\lambda / Q](r, \theta)(DF_i^{\gamma} \text{ or } DF_i^{\beta}) \qquad (11A.2\text{-}5)$$

where:

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

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

$[\chi/Q](r,\theta)$ = the annual average gaseous dispersion factor at the distance r in sector in sec⁄cu m

$DF^{\gamma}{}_i$, $DF^{\beta i}$ = the gamma and beta air dose factors for a uniform semi-infinite cloud of radionuclide i, in mrad-cu m/pCi-yr

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

b. Annual Total Body Dose from Noble Gas Releases (ground level)

$$D^T_{\infty}(r, \theta) = S_F \sum_i \chi_i (r, \theta) \; DFB_i \qquad (11A.2\text{-}6)$$

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

where:

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

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

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

DFB_i = the total body dose factor for a semi-infinite cloud of the radionuclide i which includes the attenuation of 5 g/sq cm of tissue, in mRem-cu m/pCi-yr

c. Annual Skin Dose from Noble Gas Releases (ground level)

$$D^S_\infty(r,\theta) = 1.11\, S_F \sum_i \chi_i(r,\theta)\ DF^\gamma_i + \sum_i \chi_i(r,\theta)\ DFS_i \qquad (11A.2\text{-}7)$$

where:

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

DFS_i = the beta skin dose factor for a semi-infinite cloud of radionuclide i, which includes the attenuation by the outer “dead” layer of the skin, in mRem-cu m/pCi-yr

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

All other parameters are as previously defined.

d. Annual Gamma Air Dose from Noble Gas Releases from Free-Standing Stacks More Than 80 Meters High

$$D^\gamma(r,\theta) = \frac{260}{r(\Delta\theta)} \sum_n \frac{1}{u_n} \sum_s f_{ns} \sum_k \mu_a(E_k)\, E_k\, I(H, u, s, \sigma_z, E_k) \sum_i Q^D_{ni} A_{ki} \qquad (11A.2\text{-}8)$$

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

where:

A_{ki} is the photon yield for gamma-ray photons in energy group k from the decay of radionuclide i, in photons/disintegration

$D^{\gamma}(r,\theta)$ is the annual gamma air dose at a distance r (meters) in the sector at an angle in mrad/yr

Ek is the energy of the kth photon energy group, in MeV/photon

f_{ns} is the joint frequency of occurrence of stability class s and wind speed class n for sector θ, dimensionless

$I(H,u,s,\sigma_z,E_k)$ is the dimensionless numerical integration constant accounting for the distribution of radioactivity according to meteorological conditions of wind speed (u) and atmospheric stability (s) which in part determine the effective stack height (H) and the vertical plume standard deviation (σ_z). See Regulatory Guide 1.109 for deviation

$Q^D{}_{ni}$ is the release rate of radionuclide i, corrected for decay during transit to the distance r under wind speed u_n, in Ci/yr

u_n is the mean wind speed of wind speed class n, in m/sec

Δθ is the sector width over which atmospheric conditions are average, in radians

μ_a (E_k) is the air energy absorption coefficient for the kth photon energy group, in m^{-1}

260 is the conversion factor to obtain D^{γ} (r,θ), in mrad/yr, and has the units of mrad-radians-m^3 - disintegration/sec-MeV-Ci

e. Annual Total Body Dose from Noble Gas Releases from Free-Standing Stacks More Than 80 Meters High

$$D^T(r,\theta) = 1.11 S_F \sum_k D_k^{\gamma}(r,\theta)\exp[-\mu_a^T(E_k)t_d] \qquad (11A.2\text{-}9)$$

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

where:

$D^T(r,\theta)$ is the annual total body dose at the distance r in sector θ, in mrem/yr

$D^{\gamma}_k(r,\theta)$ is the annual gamma air dose associated with the kth photon energy group at the distance r in sector θ, in mrad/yr

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

t_d is the product of tissue density and depth used to determine a total body dose, in g/cm^2

$\mu^T_a(E_k)$ is the tissue energy absorption coefficient, in cm^2/g; and

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

f. Annual Skin Dose from Noble Gas Releases from Free-Standing Stacks More Than 80 Meters High

$$D^s(r,\theta) = 1.11\, S_F\, D^{\gamma}(r,\theta) + 3.17 \times 10^4 \sum_i Q_i[\chi/Q]^D(r,\theta)\, DFS_i \qquad (11A.2\text{-}10)$$

where:

DFS_i is the beta skin dose factor for a semi-infinite cloud of radionuclide i, which includes the attenuator by the outer “dead” layer of the skin, in mrem-m^3/pCi-yr

$D^s(r,\theta)$ is the annual skin dose at the distance r in sector θ, in mrem/yr

All other parameters are as defined in preceding paragraphs.

2. Doses from Radioiodines and Other Radionuclides (not including Noble Gases) Released to the Atmosphere

a. Annual Organ Dose from External Irradiation from Radionuclides Deposited onto the Ground Surface

$$D^G_j(r,\theta) = 8760\ S_F \sum_i C^G_i(r,\theta)\ DFG_{ij} \qquad (11A.2\text{-}11)$$

where:

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

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

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

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

DFG_{ij} = the open field ground plane dose conversion factor for organ j from radionuclide i, in mRem-sq m/pCi-hr

8,760 = the number of hours in a year

b. Annual Organ Dose from Inhalation of Radionuclides in Air

$$D^A_{ja}(r, \theta) = R_a \sum_i \chi_i(r, \theta)\ DFA_{ija} \qquad (11A.2\text{-}12)$$

where:

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

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

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

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

c. Annual Organ Dose from Ingestion of Atmospherically Released Radionuclides in Food

$$D^D_{ja}(r, \theta) = \sum_i DFI_{ija}\ [U^v_a f_g C^v_i(r, \theta) \qquad (11A.2\text{-}13)$$

$$+ U^m_a C^m_i(r, \theta)$$

$$+ U^F_a C^F_i(r, \theta) + U^L_a f_l C^L_i(r, \theta)]$$

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

where:

$C^v_i(r,\theta)$, $C^m_i(r,\theta)$ = the concentrations of radionuclide i in produce

$C^L_i(r,\theta)$, $C^F_i(r,\theta)$ = (non-leafy vegetables, fruits, and grains), milk, leafy vegetables, and meat, respectively, at location (r,θ), in pCi/kg or pCi/l

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

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

f_g, f_1 = the respective fractions of the ingestion rates of produce and leafy vegetables that are produced in the garden of interest

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

<u>General Expression for Population Doses</u>

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

$$D_j^P = 0.001 \sum_d P_d \sum_a D_{jda} f_{da} \tag{11A.2-14}$$

where:

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

P_d = the population associated with subregion d

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

f_{da} = the fraction of the population in subregion d that is in age group a

0.001 = the conversion factor from mRem to Rem

The above equation used in conjunction with the preceding equations and average usage factors for each age group was used to calculate the population doses.

For further refinements on the preceding equations used to calculate the doses to man, see Regulatory Guide 1.109, Revision 1.

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

Doses to Biota Other Than Man

Calculation of dose rates to biota other than man was performed by means of the computer programs ARRRG and CRITER (Soldat et al., 1974), developed at the Pacific Northwest Laboratory of Battelle Memorial Institute under contract to the Atomic Energy Commission (NRC). Bioaccumulation factors used in ARRRG and CRITER have been updated to correspond to the latest published values in Regulatory Guide 1.109, Revision 0 (plants) and Regulatory Guide 1.109, Revision 1 (all others). Site specific data used in this analysis are presented in Table 11A.2-4.

The following sections provide a summary of the dose models used in the analysis for each pathway considered.

Internal Doses to Aquatic Organisms

Aquatic organisms were considered to receive an internal dose rate from uptake and concentration of radiochemicals in the water and from exposure through the food chain. Dose rates to primary organisms were calculated directly from radioisotopic concentrations in discharge water and from bioaccumulation factors. The dose rate through the food chain was estimated for secondary organisms such as muskrats and raccoons feeding on primary organisms whose radionuclide content was estimated in the first calculation.

Equations used by the program CRITER for these calculations are as follows:

$$(DR)_i = AE_ib_i \qquad (11A.2\text{-}15)$$

where:

$(DR)_i$ = dose rate for radionuclide i (mrad/yr)

E_i = effective absorbed energy in organ of interest (MeV/dis)

b_i = specific body burden of nuclide i (pCi/kg)

A = conversion factor = $0.0187\dfrac{\text{dis} - \text{kg} - \text{mrad}}{\text{pCi} - \text{yr} - \text{MeV}}$

and

$b_i = C_{iw} B_i$

where:

C_{iw} = concentration of nuclide i in water (pCi/l)

B_i = bioaccumulation factor for nuclide i (pCi/kg per pCi/l)

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

The concentration in water C_{iw} is calculated from:

$$C_{iw} = 1119\frac{Q_i R_i M_p}{F}\exp(-\lambda_i t_p) \qquad (11A.2\text{-}16)$$

where:

Q_i = release rate of nuclide i (Ci/yr)

R_i = reconcentration factor to estimate recycling of effluent

M_p = mixing ratio at point of exposure (1/dilution factor)

F = flow rate of the liquid effluent (cu ft/s)

λ_i = radiological decay constant of nuclide i (hr^{-1})

t_p = transit time for nuclides to reach point of exposure (hr)

1,119 = constant to convert Ci/yr per cu ft/s to pCi/l

The total body dose rate to secondary organisms was calculated as follows (Soldat et al., 1974):

$$DR'_i = 0.365\ b_i\ P'\ D'_i \qquad (11A.2\text{-}17)$$

where:

DR'_i = total body dose rate to secondary organisms due to nuclide i (mrad/yr)

0.365 = kg-day/g-yr

$$D'_i = 70{,}000\ \frac{D_i(\text{man})}{e_i(\text{man})}\ \frac{e'_i}{m'}$$

D_i (man) = total body dose conversion factor for man for radionuclide in mRem/pCi

e_i (man) = effective absorbed energy for man for radionuclide i (meV/disintegration)

e'_i = effective absorbed energy for secondary organism for radionuclide i (meV/disintegration)

m^i = mass of secondary organism (grams)

P^i = consumption rate of primary organisms by the secondary organism (grams/day)

70,000 = total body mass of adult (grams)

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

The actual equation used by CRITER was of the form:

$$DR' = 2.86 \times 10^7 \frac{M_p P'}{Fm'} \sum_{i=1}^{n} Q_i R_i B_i e'_i \exp(-\lambda_i t_p) [D_i / e_i](man) \quad (11A.2\text{-}18)$$

where:

DR' = total body dose rate to secondary organisms (mrad/yr)

n = 136, number of isotopes

$2.86x10^7$ = (0.365) (1119) (70,000)

All other parameters are as previously defined.

Exposure to Shoreline Deposits

$$(DR)_{pr} = 111,900 \frac{U_p M_p W_f}{F} \sum_{i=1}^{n} Q_i R_i T_i \exp(-\lambda_i t_p)(1 - \exp(\lambda_i t) D_{ipr}) \quad (11A.2\text{-}19)$$

where:

$(DR)_{pr}$ = dose rate to organ r (total body or skin) from pathway P (mrad/yr)

W^f = shore width factor = 0.5 (ocean shoreline)

T_i = radiological half-life of isotope i (days)

n = 136, number of isotopes

111,900 = constant to convert (Ci/yr)/(cu ft/sec) to pCi/liter

Dose for Swimming and Water Surface Exposure

$$DR'_{pr} = 1119 \frac{U_p M_p}{F K_p} \sum_{i=1}^{n} Q_i R_i D_{ipr} \exp(-\lambda_i t_p) \quad (11A.2\text{-}20)$$

where:

K_p = hemispherical correction constant = 1 for swimming and 2 for boating

n = 136, number of isotopes

Dose from Immersion in Gaseous Effluents

These doses were calculated in the same manner as doses to humans with appropriate changes in use factors as shown in Table 11A.2-1.

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

References for Appendix 11A Part II - Dose Calculation Models and Assumptions

Atomic Energy Commission 1973. Final Environmental Statement Concerning Proposed Rule Making Action; Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion (“as low as practicable”) for Radioactive Material in Light Water Cooled Nuclear Power Reactor Effluents. Washington, D.C.

Soldat, S.K.; Robinson, N.M.; and Baker, D.A. 1974. Models and Computer Codes for Evaluating Environmental Radiation Doses. Battelle Pacific Northwest Laboratories BNWL-1754, Richland, Wash.

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

TABLE 11A.2-1 DILUTION FACTORS, TRAVEL TIMES FROM THE SITE, AND POPULATION SERVED

Location of Analysis	Approximate Distance from Site (km)	Dilution Factor	Transit Time to Point of Analysis (hr)	Population Served
Edge of initial mixing zone (1)	0	3	0.0 (assumed)	-
Closest accessible shoreline (2)	1.1	7.2	0.0 (assumed)	-
Edge of initial mixing zone (3)	0	3	0.0 (assumed)	3.3E+06 (4)

NOTES:

(1) Location used to calculate doses to maximum offsite individual from ingestion of aquatic foods and boating.

(2) Location used to calculate doses to maximum offsite individual from shoreline recreation and swimming.

(3) Location used to calculate doses to population from ingestion of aquatic foods, boating, swimming, and shoreline recreation. The travel time and dilution factor for the edge of the initial mixing zone radius is conservatively applied to the entire 80 km radius. It is also assumed that the entire 80 km radius population participates in swimming and boating.

(4) 3.3E+06 = 3.3 x 10^6.

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

TABLE 11A.2-2 PARAMETERS AND ASSUMPTIONS USED IN EQUATIONS FOR ESTIMATING DOSES TO HUMANS

All parameters and assumptions used are recommended values to be used, in lieu of site specific data, from Regulatory Guide 1.109, Revision 1.

The following are site specific parameters or parameters for which there is no recommended value:

F = normal circulation flow rate (for 3-unit operation) = 4,160 cu ft/sec

T_p = transit time = see Table 11A.2-1

Note: T_p used in calculations was increased, where appropriate, by the distribution or holdup time recommended by Regulatory Guide 1.109, Revision 1.

P = fractional equilibrium ratio of C^{14} = 1 (continuous release); = 0.0073 (intermittent containment release); = 0.062 (intermittent steam generator blowdown release)

Q_i = annual release rate of radionuclide i, Ci/yr (Tables 11.2-7 and 11.3-1)

f_p = fraction of year animals graze on pasture = 0.67 (8 months)

f_s = fraction of daily feed which is pasture grass when animal is grazing = 1 (100%)

H = absolute humidity of atmosphere at location of analysis 9.86 g/cu m

U_{ap} = usage factor (hr/year of exposure):

Maximum Individual

	Adult	Teen	Child
Swimming	100	100	56
Boating	52	52	29

80 km Radius

	Adult	Teen	Child
Swimming	3.5	10	12
Boating	29	29	16.5

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

V^p = Total commercial U.S. fish harvest = 1.1E+09 kg/yr [1]

$V^{p\,'}$ = Total commercial U.S. shellfish harvest = 5.2E+08 kg/yr

V_{dp} = 80 km commercial fish harvest = 9.7E+06 kg/yr

$V_{dp}\,'$ = 80 km sports fish harvest = 9.7E+06 kg/yr

$V_{dp}\,''$ = 80 km invertebrate harvest = 2.8E+06 kg/yr

$V_{dp}\,'''$ = 80 km milk production = 5.6E+08 1/yr

$V_{dp}\,''''$ = 80 km meat production = 1.7E+07 kg/yr

V = 80-km vegetation production = 1.8E+08 kg/yr

<u>NOTE</u>:

(1) 1.1E+09 = 1.1 x 10^9

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

TABLE 11A.2-3 METEOROLOGICAL DATA

Radiological Release Points:

1. Millstone stack (continuous release)
2. Ventilation vent (intermittent release)
3. Ventilation vent (continuous release)
4. Turbine building vent (continuous release)
5. Steam generator blowdown vent (intermittent release)
6. Condensate polishing building vent (continuous release)

Meteorological Parameters - c/Q = Sec/m3; D/Q = m-2

Location	Resident		Annual Average	Growing/Grazing Season
810 m ENE	Maximum resident	χ/Q^1 (1)	4.03E-08 (2)	4.59E-08
		D/Q^1	2.16E-09	1.70E-09
		χ/Q^2	5.24E-06	5.70E-06
		D/Q^2	6.22E-08	5.40E-08
		χ/Q^3	3.50E-06	3.98E-06
		D/Q^3	4.15E-08	3.72E-08
		χ/Q^4	1.22E-05	1.54E-05
		D/Q^4	7.57E-08	7.86E-08
		χ/Q^5	1.95E-05	2.45E-05
		D/Q^5	1.21E-07	1.28E-07
		χ/Q^6	1.19E-05	1.51E-05
		D/Q^6	7.39E-08	7.86E-08
2,400 m NNE	Maximum goat	χ/Q^1	6.60E-08	8.00E-08
		D/Q^1	7.04E-10	7.58E-10
		χ/Q^2	3.08E-06	3.56E-06
		D/Q^2	1.48E-08	1.55E-08

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

TABLE 11A.2-3 METEOROLOGICAL DATA (CONTINUED)

Location	Resident		Annual Average	Growing / Grazing Season
		χ/Q^3	7.96E-09	1.04E-06
		D/Q^3	3.94E-09	4.87E-09
		χ/Q^4	9.71E-06	1.28E-06
		D/Q^4	3.81E-09	4.62E-09
		χ/Q^5	2.41E-06	3.02E-06
		D/Q^5	9.84E-09	1.12E-08
		χ/Q^6	9.20E-07	1.23E-06
		D/Q^6	3.76E-09	4.59E-09
650 m ENE	Maximum site boundary	χ/Q^1	1.27E-08	-
		D/Q^1	2.34E-09	-
		χ/Q^2	8.08E-06	-
		D/Q^2	9.70E-08	-
		χ/Q^3	4.81E-06	-
		D/Q^3	6.29E-08	-
		χ/Q^4	1.90E-05	-
		D/Q^4	1.22E-07	-
		χ/D^5	3.06E-05	-
		D/Q^5	1.96E-07	-
		χ/Q^6	1.86E-05	-
		D/Q^6	1.19E-07	-

NOTES:

(1) Numerical superscripts correspond to release point.

(2) 4.03E-08 = 4.03 x 10^{-8}.

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

TABLE 11A.2-4 PARAMETERS AND ASSUMPTIONS USED IN ESTIMATING DOSES TO BIOTA

Parameter	Values Assigned				
	Primary Organisms	Secondary Organisms			
	(Fish, Crustaceans, Mollusks, Algae)	Muskrat	Heron	Duck	Raccoon
R_i (recirculation factor	0	0	0	0	0
F (flow rate, cfs)	4,160	4,160	4,160	4,160	4,160
M_p (mixing ratio) (1)	0.333	0.333	0.333	0.333	0.333
W_f (shore width factor)	-	0.5	0.5	0.5	0.5
K (water immersion)	-	1	-	-	-
(water surface)	-	-	2	2	-
Effective radius (cm)	2	6	11	5	14
M mass (kg)	-	1	4.6	1	12
P food consumption (gpd)					
aquatic plants	-	100	-	100	-
fish	-	-	600	-	-
invertebrates	-	-	-	-	200
U_p usage (hr/yr)					
shoreline	-	2,922	2,922	4,383	2,191
water immersion	-	2,922	-	-	-
water surface	-	-	2,922	4,383	-
t holdup time (hr)	0	0	0	0	0
Residence time (mo)	12	12	12	12	12

NOTE:

(1) Edge of mixing zone and nearest shoreline.

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

PART III—COST-BENEFIT ANALYSIS (HISTORICAL)

This appendix presents the results of cost-benefit analyses performed in accordance with Section II. D of 10 CFR 50, Appendix I.

Augments to the liquid and gaseous effluent systems and respective potential reductions to the annual population exposure are taken from the U.S. NRC Regulatory Guide 1.110, Cost-Benefit Analysis for Radwaste Systems for Light-Water-Cooled Nuclear Power Reactors (Regulatory Guide 1.110, 1976). The beneficial savings of each augment were calculated by multiplying the calculated dose reduction by $1,000 per man-Rem or $1,000 per man-thyroid-Rem. The cost of borrowed money was conservatively assumed to be 10 percent. The equations and site specific data required for the dose calculations are presented in Part I of this appendix.

<u>Augments to the Liquid Effluent Treatment System</u>

Table 11A.3-1 presents the calculated base case annual total body dose (man-Rem) and thyroid dose (man-thyroid-Rem) associated with the operation of the plant liquid radwaste system for the population expected to live within an 80-km radius of the plant for the year 2010. Assuming that each augment is capable of reducing the population doses to zero (an extremely conservative assumption), the maximum benefit to be derived from any augment would be $1,700 for reducing man-Rem exposures to zero and $16,000 for reducing man-thyroid-Rem exposure to zero.

In an analysis of the annualized procurement, installation, operation, and maintenance costs, the least expensive liquid radwaste augment was found to be $19,000 per year for a plant located in the northeastern United States. Since the benefit from this augment would be less than the corresponding total annualized cost, the cost-benefit ratio is greater than 1. The operation of additional equipment for the purpose of reducing the annual population dose would not be cost effective. Therefore, the most cost-beneficial system has been included in the current plant design.

<u>Augments to the Gaseous Effluent Treatment System</u>

Table 11A.3-2 presents the calculated base case annual total body man-Rem and thyroid man-Rem associated with the operation of the gaseous radwaste system for the 80-km radius population.

Assuming that each augment is capable of reducing the population doses to zero, the maximum benefit to be derived from any augment would be $4,800 for reducing man-Rem exposures to zero and $7,800 for reducing man-thyroid-Rem exposures to zero.

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

In an analysis of the annualized procurement, installation, operation, and maintenance costs, the least expensive gaseous radwaste augment was found to be $8,700 per year for a plant located in the northeastern United States. Since the benefit from this augment would be less than the corresponding total annualized cost, the cost-benefit ratio is greater than 1. The operation of additional equipment for the purpose of reducing the annual population dose would not be cost effective. Therefore, the most cost-beneficial system has been included in the current plant design.

Reference for Appendix 11A Part III - Cost-Benefit Analysis

Regulatory Guide 1.110, 1976. Cost-Benefit Analysis for Radwaste Systems for Light Water-Cooled Nuclear Power Reactor. March 1976.

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

TABLE 11A.3-1 BASE CASE ANNUAL POPULATION DOSES DUE TO LIQUID EFFLUENTS (1)

Pathway	Total Body Dose (man-Rem)	Thyroid Dose (man-thyroid-Rem)
Ingestion of fish	1.2E-00 (2)	9.2E+00
Ingestion of other seafood	2.5E-01	6.5E+00
Shoreline recreation	2.4E-01	2.4E-01
Swimming	3.1E-03	3.1E-03
Boating	6.2E-03	6.2E-03
Total	1.7E+00	1.6E+01

NOTES:

(1) Total annual dose from all existing pathways for Millstone 3 operation.

(2) 1.2E-00 = 1.2×10^0

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

TABLE 11A.3-2 BASE CASE ANNUAL POPULATION DOSES DUE TO GASEOUS EFFLUENTS [(1)]

Pathway	Total Body Dose (man-Rem)	Thyroid Dose (man-thyroid-Rem)
Submersion	1.3E-01 [(2)]	1.3E-01
Inhalation	8.8E-01	1.9E+00
Standing on contaminated ground	3.3E+00	3.3E+00
Ingestion of fruits, grains, and vegetation	8.6E-02	2.6E-01
Ingestion of cow milk	3.9E-01	2.2E+00
Ingestion of meat	2.1E-02	2.7E-02
Total	4.8E+00	7.8E+00

NOTES:

(1) Total annual dose from all existing pathways for Millstone 3 operation

(2) 1.3E-01 = 1.3 x 10^{-1}