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RADIOACTIVE WASTE MANAGEMENT

11.1 SOURCE TERMS

Expected Activities

The expected reactor coolant and main steam activities form the basis for estimating the average quantity of radioactive material released to the environment and reflect normal operating conditions, including operational occurrences. These data (Table 11.1-1) are based on methods that are consistent with ANSI/ANS-18.1-1999[12]. Plant parameters used to determine the expected source terms are listed in Table 11.1-2.

Design Basis Activities

The design basis radioactive material levels in the reactor coolant and main steam are also presented in Table 11.1-1. These data conservatively represent the shielding, ventilation, and radwaste liquid release design basis fission product source terms. Design failed fuel conditions correspond to an offgas release rate of 100 uCi/sec/MWt, or 0.3536 Ci/sec, at 30-min delay and are developed by scaling up from expected NUREG-0016 Revision 1 source term data[1]. In addition, the design basis source terms in Table 11.1-1 take into consideration the General Electric Company (GE) design basis data described in Section 11.1.1 and Tables 11.1-3 through 11.1-7. For the design basis concentrations, Table 11.1-1 presents the higher value for a given isotope from either the adjusted NUREG-0016 data or the GE data and represents a conservative data set.

11.1.1 General Electric Reactor Coolant and Main Steam Data

GE has evaluated radioactive material sources (activation products and fission product release from fuel) in operating boiling water reactors (BWRs). These source terms are reviewed and periodically revised to incorporate up-to-date information.

The information provided in this section defines the design basis radioactive material levels in the reactor water, steam, and offgas. The various radioisotopes have been grouped as fission products, coolant activation products, and noncoolant activation products. The fission product activity levels are based on measurements of BWR water and offgas at several stations through mid-1971. Observations made at KRB and Dresden Unit 2 provided input to the design basis values. The design basis radioactive material levels do not necessarily include all the radioisotopes observed or theoretically predicted to be present. The radioisotopes included are considered significant to one or more of the following criteria:

1. Plant equipment design.
2. Shielding design.
3. Understanding of system operation and performance.
4. Measurement practicability.
5. Evaluation of radioactive material releases to the environment.

A combination of GE and NUREG-0016 data was used for the main steam radionuclide concentrations.

11.1.1.1 Fission Products

Noble Radiogas Fission Products

The noble radiogas fission product source terms observed in operating BWRs are generally complex mixtures whose sources vary from minuscule defects in cladding to minute quantities of "tramp" uranium on external cladding surfaces. The relative concentrations or amounts of noble radiogas isotopes relative to the other noble gas isotopes can be described as follows:

$$\text{Equilibrium:} \quad R_g \approx K_1 y$$

$$\text{Recoil:} \quad R_g \approx K_2 y \lambda$$

The terms in these and succeeding equations are defined in the nomenclature section. The constants K_1 and K_2 describe the fractions of the total fissions that are involved in each of the releases. The equilibrium and recoil mixtures are the two extremes of the mixture spectrum that are physically possible. The equilibrium mixture results when a sufficient time delay exists between the fission event and the time of release of the radiogases from the fuel to the coolant for the radiogases to approach equilibrium levels in the fuel. When there is no delay or impedance between the fission event and the release of the radiogases, the recoil mixture is observed.

Prior to Vallecitos Boiling Water Reactor (VBWR) and Dresden Unit 1 experience, it had been assumed that noble radiogas leakage from the fuel would be the equilibrium mixture of the noble radiogases present in the fuel. VBWR and early Dresden Unit 1 experience indicated that the actual mixture most often observed approached a distribution that was intermediate in character to the two extremes[2]. This intermediate decay mixture was termed the diffusion mixture. It must be emphasized that this diffusion mixture is merely one possible point on the mixture spectrum, ranging from the equilibrium to the recoil mixture, and does not have the absolute mathematical and mechanical basis for the calculational methods possible for equilibrium and recoil

mixtures. This diffusion distribution pattern is described as follows[3]:

$$\text{Diffusion:} \qquad R_g \approx K_3 y \lambda^{0.5}$$

The constant K_3 describes the fraction of total fissions that are involved in the release. The exponent of the decay constant, λ, is midway between that of equilibrium, 0, and recoil, 1. The diffusion pattern value of 0.5 was originally derived from diffusion theory.

Although the previously described diffusion mixture has been used by GE as a basis for design since 1963, the design basis release magnitude used has varied from 0.5 Ci/sec to 0.1 Ci/sec as measured after 30-min decay (t=30 min). The noble radiogas source term rate after 30-min decay has been used as a conventional measure of the design basis fuel leakage rate, since it is conveniently measurable at existing BWR plants and is consistent with the nominal design basis 30-min offgas holdup system used in a number of plants. Since about 1967, the design basis release rate from the core was established at an annual average of 0.1 Ci/sec (t=30 min). The expected annual average is significantly below the design basis annual average. This design value was selected on the basis of operating experience rather than predictive assumptions. Several judgment factors, including the significance of environmental release, reactor water radioisotope concentrations, liquid waste handling, and effluent disposal criteria, building air contamination, shielding design, and turbine and other component contamination affecting maintenance, have been considered in establishing this level.

Although noble radiogas source terms from fuel above the design basis can be tolerated for reasonable periods of time, long-term operation at such levels would be undesirable. Continual assessment of this value is made on the basis of actual operating experience in BWRs[4].

While the GE noble radiogas source term magnitude was established at 0.1 Ci/sec (t=30 min) in 1967, it was recognized that there may be a more statistically applicable distribution for the noble radiogas mixture. Sufficient data were available from KRB operations from 1967 to mid-1971, along with Dresden Unit 2 data from operation in 1970 and several months in 1971, to more accurately characterize the noble radiogas mixture pattern from an operating BWR.

The basic equation for each radioisotope used to analyze the collected data is:

$$R_g = K_g \; y \lambda^m \left(1 - e^{-\lambda T}\right)\left(e^{-\lambda t}\right) \qquad (11.1\text{-}1)$$

With the exception of Kr-85 with a half-life of 10.74 yr, the noble radiogas fission products in the fuel are essentially at an equilibrium condition after an irradiation period of several months (rate of formation is equal to rate of decay). Therefore, the term $\left(1-e^{-\lambda T}\right)$ approaches 1 and can be neglected when the reactor has been operating at steady state for long periods of time. The term $\left(e^{-\lambda t}\right)$ is used to adjust the releases from the fuel (t=0) to the decay time for which values are needed. Historically, t=30 min has been used. When discussing long steady state operation and leakage from the fuel (t=0), the following simplified form of Equation 11.1-1 can be used to describe the leakage of each noble radiogas isotope:

$$R_g = K_g y \lambda^m \qquad (11.1\text{-}2)$$

The constant, K_g, describes the magnitude of total leakage. The rate of noble radiogas leakage with respect to each other (composition) is expressed in terms of the decay constant term, λ, and the fission yield fraction, y.

Dividing both sides of Equation 11.1-2 by y and taking the logarithm of both sides results in the following equation:

$$\log\left(R_g / y\right) = m \log\left(\lambda\right) + \log\left(K_g\right) \qquad (11.1\text{-}3)$$

Equation 11.1-3 represents a straight line when log (R_y/y) is plotted versus log (λ); m is the slope of the line. By fitting actual data from KRB and Dresden Unit 2 (using least squares techniques), the value of m can be obtained. With radiogas leakage at KRB over the nearly 5-yr period varying from 0.001 to 0.056 Ci/sec (t=30 min), and with radiogas leakage at Dresden Unit 2 varying from 0.001 to 0.169 Ci/sec (t=30 min), the average value for m of 0.4 with a standard deviation of ±0.07 was determined (Figure 11.1-1). As shown on this figure, variations in m were observed in the range m=0.1 to m=0.6. After establishing the value of m=0.4, the value of K_g can be calculated by selecting a value for R_g. With $\sum R_g$ at 30 min = 0.1 Ci/sec, K_g is 2.6 x 10^7 and Equation 11.1-1 becomes:

$$R_g = 2.6\ x\ 10^7\, y\, \lambda^{\,0.4} \left(1-e^{-\lambda T}\right)\left(e^{-\lambda t}\right) \qquad (11.1\text{-}4)$$

This updated noble radiogas source term mixture has been designated the 1971 Mixture to differentiate it from the diffusion mixture.

The noble gas source term for each radioisotope can be calculated from Equation 11.1-4. The resultant source terms are presented in Table 11.1-3 as leakage from fuel (t=0) and after 30-min decay. Since the NUREG-0016 values were higher than the GE data,

the NUREG-0016 concentrations were used to calculate noble radiogas fission product releases.

Radiohalogen Fission Products

Historically, the radiohalogen design basis source term was established by the same equation as that used for noble radiogases. In a manner similar to that used with gases, a simplified equation can be shown to describe the release of each halogen radioisotope:

$$R_h = K_h y \lambda^n \qquad (11.1\text{-}5)$$

The constant, K_h, describes the magnitude of the total leakage from the fuel. The rate of halogen radioisotope releases with respect to each other (composition) is expressed in terms of the fission yield, y, and the decay constant λ. As was done with the noble radiogases, the average value was determined for n. The average value for n is 0.5, with a standard deviation of ± 0.19 (Figure 11.1-2). As can be seen from this figure, variations in n were observed in the range from n=0.1 to n=0.9.

As mentioned, it appears that the use of the previous method of calculating radiohalogen leakage from fuel is overly conservative. Figure 11.1-3 relates KRB and Dresden Unit 2 noble radiogas releases versus I-131 leakage. While Dresden Unit 2 data during the period August 1970 to January 1971 show a relationship between noble radiogas and I-131 leakage under one fuel condition, there was no simple relationship for all fuel conditions experienced. Also, during this period, high radiogas leakages were not accompanied by high radioiodine leakage from the fuel. Except for one KRB datum point, all steady state I-131 leakages observed at KRB or Dresden Unit 2 were equal to or less than 505 uCi/sec. Even at Dresden Unit 1 in March 1965, when severe defects were experienced in stainless steel clad fuel, I-131 leakages greater than 500 uCi/sec were not experienced. Figure 11.1-3 shows that these higher radioiodine leakages from the fuel were related to noble radiogas source terms of less than the design basis value of 0.1 Ci/sec (t=30 min). This may be partially explained by inherent limitations due to internal plant operational problems that caused plant derating.

In general, one would not anticipate continued operation at full power for any significant time period with fuel cladding defects that would be indicated by I-131 leakage from the fuel in excess of 700 uCi/sec. When high radiohalogen leakages are observed, other fission products will be present in greater amounts.

By using these judgment factors and experience to date, the design basis radiohalogen source terms from fuel were established based on an I-131 leakage of 700 uCi/sec. This value, as seen on Figure 11.1-3, is consistent with the experiential data and the design basis noble radiogas source term of 0.1 Ci/sec (t=30 min).

With the I-131 design basis source term established, K_h equals 2.4×10^7, and the halogen radioisotope release can be expressed by the following equation:

$$R_h = 2.4\ x\ 10^7\ y\ \lambda^{0.5}\ \left(1 - e^{-\lambda T}\right)\left(e^{-\lambda t}\right) \qquad (11.1\text{-}6)$$

Concentrations of radiohalogens in reactor water are listed in Table 11.1-8 and can be calculated using the following equation:

$$C_h = \frac{Rh}{M(\lambda+\beta+\gamma)} \qquad (11.1\text{-}7)$$

Terms used in Equation 11.1-7 are defined in the nomenclature section.

Although carryover of most soluble radioisotopes from reactor water to steam is observed to be <0.1 percent (<0.001 fraction), the observed carryover for radiohalogens has varied from 0.1 percent to about 2 percent on newer plants. The average of observed radiohalogen carryover measurements has been 1.2 percent with a standard deviation of ±0.9. In the present source term definition, a radiohalogen carryover design basis of 2 percent (0.02 fraction) is used.

Other Fission Products

The observations of other fission products (and transuranic nuclides, including Np-239) in operating BWRs are not adequately correlated by simple equations. For these radioisotopes, design basis concentrations in reactor water have been estimated conservatively from experiential data (Table 11.1-4). Carryover of these radioisotopes from the reactor water to the steam is estimated to be <0.1 percent (<0.001 fraction). In addition to carryover, however, decay of noble radiogases in the steam leaving the reactor results in production of noble gas daughter radioisotopes in the steam and condensate systems.

Some daughter radioisotopes (for example, yttrium and lanthanum) were not listed as being in reactor water. Their independent leakage to the coolant is negligible; however, these radioisotopes may be observed in some samples in equilibrium or approaching equilibrium with the parent radioisotope.

Except for Np-239, trace concentrations of transuranic isotopes have been observed on only a few samples where extensive and complex analyses were carried out. The predominant alpha emitter present in reactor water is Cm-242, at an estimated concentration of 10^{-6} uCi/g or less, which is below the maximum concentration allowed in drinking water for continuous use by the general public. The concentration of alpha-emitting plutonium radioisotopes is more than one order of magnitude lower than that of Cm-242. Plutonium-241 (a beta emitter) may also be present in concentrations comparable to the Cm-242 level.

Nomenclature

The terms used in equations for source term calculations are defined as follows:

R_g = Leakage rate of a noble radiogas isotope, uCi/sec

R_h = Leakage rate of a halogen radioisotope, uCi/sec

y = Fission yield of a radioisotope, atoms/fission

λ = Decay constant of a radioisotope, sec^{-1}

T = Fuel irradiation time, sec

t = Decay time following leakage from fuel rod, sec

m = Noble radiogas decay constant exponent, dimensionless

n = Radiohalogen decay constant exponent, dimensionless

K_g = Constant establishing the level of noble radiogas leakage from fuel

K_h = Constant establishing the level of radiohalogen leakage from fuel

C_h = Concentration of a halogen radioisotope in reactor, uCi/g

M = Mass of water in the operating reactor, g

β = Cleanup system removal constant, sec^{-1}

$$\beta = \frac{\text{Cleanup system flowrate (g/sec)}}{\text{M(g)}}$$

γ = Halogen steam carryover removal constant, sec^{-1}

$$\gamma = \frac{\left[\frac{\text{Concentration of halogen radioisotope in steam } (\mu Ci/g)}{C_h (\mu Ci/g)}\right]\left[\text{Steam flow (g/sec)}\right]}{\text{M(g)}}$$

11.1.1.2 Activation Products

Coolant Activation Products

Coolant activation products are not adequately correlated by simple equations. Design basis concentrations in reactor water and steam have been estimated from experiential data and additional conservatism added by using the higher NUREG-0016

Revision 1 values. The resultant concentrations are listed in Table 11.1-5.

Noncoolant Activation Products

The activation products formed by activation of impurities in the coolant or by corrosion of irradiated system materials are not adequately correlated by simple equations. The design basis source terms of noncoolant activation products have been estimated conservatively from experiential data[5] and, in some cases, more conservative values were based on NUREG-0016. The resultant concentrations are listed in Table 11.1-6. Carryover of these isotopes from the reactor water to the steam is estimated to be ≤0.1 percent (<0.001 fraction).

11.1.1.3 Tritium

In a BWR, tritium, which is available for release in liquid and gaseous wastes, is produced by three principal methods:

1. Activation of naturally-occurring deuterium in the primary coolant.

2. Nuclear fission of UO_2 fuel.

3. Neutron capture reactions with boron used in reactivity control rods.

The tritium formed in control rods, which may be released from a BWR in liquid or gaseous effluents, is believed to be negligible. A prime source of tritium is that from activation of deuterium in the primary coolant. A secondary source of tritium may also be transferred from fuel to primary coolant. The following discussion is limited to the uncertainties associated with estimating the amounts of tritium generated in a BWR which are available for release.

Tritium produced by activation of deuterium in the primary coolant is available for release in liquid or gaseous effluents and can be determined by using the equation:

$$R_{act} = \frac{\Sigma \Phi V \lambda}{3.7 \; x \; 10^4 P} \qquad (11.1\text{-}8)$$

Where:

R_{act} = Tritium formation rate by deuterium activation, uCi/sec/MWt

Σ = Macroscopic thermal neutron cross section, cm^{-1}

Φ = Thermal neutron flux, neutrons/(cm^2), sec

V = Coolant volume in core, cm^3

λ = Tritium radioactive decay constant, 1.78 x 10^{-9} sec^{-1}

P = Reactor power level, MWt

For recent BWR designs, R_{act} is calculated to be:

$$(1.3 \pm 0.4)\, x10^{-4}\ uCi/\sec/MWt \qquad (11.1\text{-}9)$$

The uncertainty indicated is derived from the estimated errors in selecting values for the coolant volume in the core, coolant density in the core, abundance of deuterium in light water (some additional deuterium is present because of the H(n,γ) D reaction), thermal neutron flux, and microscopic cross section for deuterium.

The fraction of tritium produced by fission which may transfer from fuel to the coolant is much more difficult to estimate. However, since zircaloy-clad fuel rods are used in BWRs, essentially all fission product tritium remains in the fue1 rods[6].

This is confirmed by the study made at Dresden Unit 1 in 1968 by the U.S. Public Health Service (USPHS), which suggests that essentially all tritium released from the plant could be accounted for by the deuterium activation source[7]. For purposes of estimating the leakage of tritium from defective fuel, it is assumed that the tritium leaks in a manner similar to the leakage of noble radiogases. Thus the empirical relationship described as the diffusion mixture is used for predicting the source term of individual noble gas radioisotopes as a function of total noble gas source term. The equation that describes this relationship is:

$$R_{diff} = Ky\sqrt{\lambda} \qquad (11.1\text{-}10)$$

Where:

R_{diff} = Leakage rate of the radioisotope, uCi/sec

y = Fission yield fraction

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

K = Constant related to total leakage rate

If the total noble radiogas source term is 10^5 uCi/sec after 30-min decay, leakage from the fuel is calculated to be about 0.24 uCi/sec of tritium. To place this value in perspective with the USPHS study, the observed leakage rate of Kr-85 (which has a half-life similar to that of tritium) was 0.06 to 0.4 times that

calculated using the diffusion mixture relationship. This would suggest that the actual tritium leakage rate might range from 0.015 to 0.10 uCi/sec. Since the annual average noble radiogas leakage from a BWR is expected to be less than 0.1 Ci/sec (t=30 min), the annual average tritium release rate from the fission source can be conservatively estimated at 0.12 ±0.12 uCi/sec. Based on this approach, the estimated total tritium appearance rate in reactor coolant and the release rate in effluents can be estimated to be about 16 Ci/yr.

Tritium formed in the reactor is generally present as tritiated oxide (HTO) and, to a lesser degree, as tritiated gas (HT). Tritium concentration in the steam formed in the reactor is the same as in the reactor water at any given time. This tritium concentration is also present in condensate and feedwater. Since radioactive effluents generally originate from the reactor and power cycle equipment, radioactive effluents also have this tritium concentration. Condensate storage receives treated water from the radioactive waste system and rejects water to the condensate system. Thus, all plant process water has a common tritium concentration.

Offgases released from the plant contain tritium which is present as HT resulting from reactor water radiolysis. In addition, water vapor present in ventilation air due to process steam leaks or evaporation from sumps, tanks, and spills on floors also contains tritium. The remainder of tritium leaves the plant in liquid effluents.

Recombination of radiolysis gases in the air ejector offgas system forms water vapor that is condensed and returned to the main condenser. This reduces the amount of tritium leaving in gaseous effluents and results in a slightly higher tritium concentration in the plant process water. Reducing the amount of liquid effluent discharged also results in a higher process coolant equilibrium tritium concentration.

Essentially all tritium entering the primary coolant is eventually released to the environs, either as water vapor and gas to the atmosphere, or as liquid effluent to the plant discharge. Reduction due to radioactive decay is negligible due to the 12-yr half-life of tritium.

The USPHS study at Dresden Unit 1 estimated that approximately 90 percent of the tritium release was observed in liquid effluent, with the remaining 10 percent leaving as gaseous effluent[7]. Efforts to reduce the volume of liquid effluent discharges may change this distribution; however, from a practical standpoint, the fraction of tritium leaving as liquid effluent may vary between 60 and 90 percent, with the remainder leaving in gaseous effluent. The tritium design basis for Unit 2 is taken from conservative values of NUREG-0016.

11.1.2 Fuel Fission Product Inventory and Fuel Experience

11.1.2.1 Fuel Fission Product Inventory

Fuel rod and fuel plenum radioisotopic inventory, along with escape rate coefficients and release fractions, are not used in establishing BWR design basis source term coolant activities. Fuel fission product inventory information is used in establishing fission product source terms for accident analysis and is therefore discussed in Chapter 15.

11.1.2.2 Fuel Experience

A discussion of fuel experience gained for BWR fuel, including failure experience, burnup experience, and thermal conditions under which the experience was gained, is available in NEDO-10505, NEDO-20922, and NEDO-10173[3,8,9].

11.1.3 Process Leakage Sources

Process leakage results in potential release paths for noble gases and other volatile fission products via ventilation systems. All liquid from process leaks is collected and routed to the liquid-solid radwaste system. Radionuclide releases via ventilation paths are at extremely low levels and have been insignificant compared to process offgas from operating BWR plants. However, because the implementation of improved process offgas treatment systems makes the ventilation release comparatively significant, measurements have been conducted to identify and qualify these low-level release paths.

Leakage of fluids from the process system results in the release of radionuclides into plant buildings. In general, the noble radiogases remain airborne and are released to the atmosphere with little delay via the building ventilation exhaust ducts. The radionuclides are partitioned between air and water, and airborne radioiodines may plateout on metal surfaces, concrete, and paint. A significant amount of radioiodine remains in the air or is desorbed from surfaces. Radioiodines are found in ventilation air as methyl and inorganic iodines, which are defined here as particulate, elemental, and hypoiodous acid forms of iodine. Particulates are also present in the ventilation exhaust air.

The airborne radiological releases from BWR building heating, ventilating, and air conditioning (HVAC) and the main condenser mechanical vacuum pump have been compiled and evaluated in NEDO-21159[10]. This report is periodically updated to incorporate the most recent data on airborne emissions. The results of these evaluations (Section 12.2) are based on data obtained by owner and utility personnel and special in-plant studies of operating BWR plants by independent organizations and GE.

11.1.4 Radioactive Sources in the Liquid Radwaste System

The radioactive sources for the liquid radwaste system are described in Section 11.2.

11.1.5 Radioactive Sources in the Offgas System

The radioactive sources for the offgas system are described in Section 11.3.

11.1.6 Radioactive Sources for Component Failures

The radioactive sources considered for evaluating the radiological consequences of component failures are described in Section 15.7.

11.1.7 References

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10. Marrero, T. R. Airborne Releases From BWRs for Environmental Impact Evaluations, NEDO-21159, March 1976.

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12. ANSI/ANS-18.1-1999, "Radioactive Source Term for Normal Operation of Light Water Reactors," American Nuclear Society, September 1999.

TABLE 11.1-1

REACTOR COOLANT AND MAIN STEAM
RADIONUCLIDE CONCENTRATIONS[1]

	Reactor Coolant		Main Steam	
Isotope	Design Basis (uCi/g)	Expected[4] (uCi/g)	Design Basis (uCi/g)	Expected[4] (uCi/g)
Noble Gases[2]				
Kr-83m			6.4-3[3]	5.9E-04
Kr-85m			1.1-2	1.0E-03
Kr-85			3.5-5	4.0E-06
Kr-87			3.9-2	3.3E-03
Kr-88			3.9-2	3.3E-03
Kr-89			2.4-1	2.1E-02
Kr-90			5.3-1	
Kr-91			6.4-1	
Kr-92			6.4-1	
Kr-93			1.7-1	
Kr-94			4.2-2	
Kr-95			3.9-3	
Kr-97			2.5-5	
Xe-131m			2.8-5	3.3E-06
Xe-133m			5.3-4	4.9E-05
Xe-133			1.5-2	1.4E-03
Xe-135m			5.0-2	4.4E-03
Xe-135			4.2-2	3.8E-03
Xe-137			2.8-1	2.6E-02
Xe-138			1.6-1	1.5E-02
Xe-139			5.3-1	
Xe-140			5.7-1	
Xe-141			4.6-1	
Xe-142			1.3-1	
Xe-143			2.3-2	
Xe-144			1.1-3	
Halogens				
Br-83	2.3-2		3.5-4	
Br-84	3.0-2		5.6-4*	
Br-85	1.7-2*		3.4-4*	
I-131	1.3-2*	3.3E-03	2.6-4*	7.1E-05
I-132	2.2-1	2.7E-02	3.3-3	5.9E-04
I-133	1.6-1	2.2E-02	2.6-3	4.7E-04
I-134	4.0-1	4.4E-02	7.8-3	9.9E-04
I-135	1.7-1	3.0E-02	2.8-3	6.6E-04

TABLE 11.1-1 (Cont'd.)

Isotope	Reactor Coolant Design Basis (uCi/g)	Reactor Coolant Expected[4] (uCi/g)	Main Steam Design Basis (uCi/g)	Main Steam Expected[4] (uCi/g)
Cesium and Rubidium				
Rb-89	2.2-2	3.6E-03	2.2-5	1.3E-05
Cs-134	8.5-5	1.7E-05	8.5-8	5.9E-08
Cs-136	5.5-5	1.1E-05	5.5-8	3.9E-08
Cs-137	2.2-4	4.5E-05	2.2-7	1.6E-07
Cs-138	1.6-1*	7.2E-03	1.6-4*	2.5E-05
Water Activation Products				
N-13	5.8-2*		7.0-3	
N-16	1.0+2	6.0+1	1.0+2	5.0+1
N-17	1.8-2*		3.6-2*	
O-19	1.1+0*		5.9-1*	
F-18	4.8-2*		4.0-3	
Tritium				
H-3	1.0-2	1.0-2	1.0-2	1.0-2
Other Nuclides				
Na-24	4.1-3	1.3E-03	4.1-6	4.4E-06
P-32	7.9-5	2.5E-05	7.9-8	8.7E-08
Cr-51	2.4-3	1.9E-03	2.4-6	6.5E-06
Mn-54	4.4-5*	2.2E-05	4.4-8*	7.6E-08
Mn-56	5.5-2*	1.7E-02	5.5-5*	5.9E-05
Fe-55	3.9-4	6.2E-04	3.9-7	2.2E-06
Fe-59	8.8-5*	1.9E-05	8.8-8*	6.5E-08
Co-58	5.5-3*	6.2E-05	5.5-6*	2.2E-07
Co-60	5.5-4*	1.2E-04	5.5-7*	4.4E-07
Ni-63	3.9-7	6.2E-07	3.9-10	2.2E-09
Ni-65	3.3-4*		3.3-7*	
Cu-64	1.3-2	1.9E-03	1.3-5	6.7E-06
Zn-65	7.9-5	6.2E-05	7.9-8	2.2E-07
Zn-69m	1.1-3		1.1-6	
Sr-89	1.2-3*	6.2E-05	1.2-6*	2.2E-07
Sr-90	8.8-5*	4.4E-06	8.8-8*	1.5E-08
Sr-91	3.2-2*	2.6E-03	3.2-5*	9.0E-06
Sr-92	6.6-2*	6.7E-03	6.6-5	2.3E-05
Y-91	1.1-4	2.5E-05	1.1-7	8.7E-08
Y-92	2.0-2	4.0E-03	2.0-5	1.4E-05

TABLE 11.1-1 (Cont'd.)

	Reactor Coolant		Main Steam	
Isotope	Design Basis (uCi/g)	Expected[4] (uCi/g)	Design Basis (uCi/g)	Expected[4] (uCi/g)
Y-93	1.2-2	2.6E-03	1.2-5	9.0E-06
Zr-95	2.3-5	5.0E-06	2.3-8	1.7E-08
Zr-97	1.8-5		1.8-8	
Nb-95	2.3-5	5.0E-06	2.3-8	1.7E-08
Nb-98	1.6-2		1.6-5	
Mo-99	8.9-3*	1.3E-03	8.9-6*	4.4E-06
Tc-99m	6.2-2	1.3E-03	6.2-5	4.4E-06
Tc-101	3.9-1		3.9-4	
Tc-104	3.4-1		3.4-4	
Ru-103	5.6-5	1.2E-05	5.6-8	4.4E-08
Ru-105	6.4-3		6.4-6	
Ru-106	8.5-6	1.9E-06	8.5-9	6.5E-09
Ag-110m	6.6-5*	6.2E-07	6.6-8*	2.2E-09
Te-129m	1.3-4*	2.5E-05	1.3-7*	8.7E-08
Te-131m	2.8-4	6.3E-05	2.8-7	2.2E-07
Te-132	5.6-3*	6.3E-06	5.6-6*	2.2E-08
Ba-139	1.1-1*		1.1-4*	
Ba-140	3.4-3*	2.5E-04	3.4-6*	8.7E-07
Ba-141	1.6-1*		1.6-4*	
Ba-142	1.6-1*		1.6-4*	
La-142	1.8-2		1.8-5	
Ce-141	8.5-5	1.9E-05	8.5-8	6.5E-08
Ce-143	8.5-5		8.5-8	
Ce-144	1.3-5*	1.9E-06	1.3-8*	6.5E-09
Pr-143	1.1-4		1.1-7	
Nd-147	8.5-6		8.5-9	
W-187	3.3-3*	1.9E-04	3.3-6*	6.6E-07
Np-239	9.5-2*	5.0E-03	9.5-5*	1.8E-05

(1) Design basis values marked with an asterisk are based on GE document GE-NE-187-08-1090, Revision 1, July 1992; all other design basis values are based on NUREG-0016, Revision 1.

(2) The design and expected concentrations for noble gases in reactor coolant are negligible.

(3) $6.4\text{-}3 = 6.4 \times 10^{-3}$.

(4) Only isotopes that correspond to the design basis are listed.

TABLE 11.1-2

PARAMETERS USED TO DETERMINE REACTOR COOLANT
AND MAIN STEAM RADIONUCLIDE CONCENTRATIONS

Variable	Unit*
Rated thermal power	3,988 MWt
Rated steam flow rate	17,636,000 lb/hr
Weight of reactor coolant in RPV at rated power	600,000 lb
Reactor coolant cleanup system flow rate	270,000 lb/hr
Ratio of condensate demineralizer flow rate to steam flow rate	0.69
Assumed moisture carryover	0.35%

TABLE 11.1-3

GENERAL ELECTRIC DATA
NOBLE RADIOGAS SOURCE TERMS

Isotope	Half-Life	Source Term at t=0 (uCi/sec)	Source Term at t=30 min (uCi/sec)
Kr-83m	1.86 hr	3.4×10^3	2.8×10^3
Kr-85m	4.48 hr	6.1×10^3	5.7×10^3
Kr-85	10.72 yr	2.0E + 01	2.0E + 01
Kr-87	76.0 min	2.0×10^4	1.5×10^4
Kr-88	2.84 hr	2.0×10^4	1.8×10^4
Kr-89	3.16 min	1.3×10^5	1.8×10^2
Kr-90	32.3 sec	2.8×10^5	---
Kr-91	8.6 sec	3.3×10^5	---
Kr-92	1.84 sec	3.3×10^5	---
Kr-93	1.29 sec	9.3×10^4	---
Kr-94	0.21 sec	2.3×10^4	---
Kr-95	0.78 sec	2.1×10^3	---
Kr-97	0.1 sec	1.4×10^1	---
Xe-131m	11.92 day	1.5×10^1	1.5×10^1
Xe-133m	2.19 day	2.9×10^2	2.9×10^2
Xe-133	5.25 day	8.2×10^3	8.2×10^3
Xe-135m	15.3 min	2.6×10^4	6.7×10^3
Xe-135	9.10 hr	2.2×10^4	2.1×10^4
Xe-137	3.84 min	1.5×10^5	6.7×10^2
Xe-138	14.1 min	8.9×10^4	2.1×10^4
Xe-139	40.0 sec	2.8×10^5	---
Xe-140	13.6 sec	3.0×10^5	---
Xe-141	1.72 sec	2.4×10^5	---
Xe-142	1.20 sec	7.3×10^4	---
Xe-143	0.30 sec	1.2×10^4	---
Xe-144	1.2 sec	5.6×10^2	---
Total		2.4×10^6	1.0×10^5

NOTE: 3.4×10^3 = 3.4E+03

TABLE 11.1-4

GENERAL ELECTRIC DATA
OTHER FISSION PRODUCT RADIOISOTOPES IN REACTOR WATER

Isotope	Half-Life	Concentration* (uCi/g)
Sr-89	50.52 day	1.2×10^{-3}
Sr-90	29 yr	8.8×10^{-5}
Sr-91	9.5 hr	3.2×10^{-2}
Sr-92	2.71 hr	6.6×10^{-2}
Zr-95	64.03 day	1.5×10^{-5}
Zr-97	16.8 hr	1.4×10^{-5}
Nb-95	34.98 day	1.6×10^{-5}
Mo-99	65.94 hr	8.9×10^{-3}
Tc-99m	6.01 hr	4.0×10^{-2}
Tc-101	14.2 min	1.4×10^{-1}
Ru-103	39.24 day	7.5×10^{-6}
Ru-106	372.6 day	1.0×10^{-6}
Te-129m	33.4 day	1.3×10^{-4}
Te-132	78.2 hr	5.6×10^{-3}
Cs-134	2.065 yr	6.1×10^{-5}
Cs-136	13.1 day	4.1×10^{-5}
Cs-137	30.17 yr	9.3×10^{-5}
Cs-138	32.2 min	1.6×10^{-1}
Ba-139	83.1 min	1.1×10^{-1}
Ba-140	12.76 day	3.4×10^{-3}
Ba-141	18.3 min	1.6×10^{-1}
Ba-142	10.7 min	1.6×10^{-1}
Ce-141	32.5 day	1.5×10^{-5}
Ce-143	33.0 hr	1.4×10^{-5}
Ce-144	284.4 day	1.3×10^{-5}
Pr-143	13.58 day	1.5×10^{-5}
Nd-147	10.99 day	5.4×10^{-6}
Np-239	2.35 day	9.5×10^{-2}

* Based on noble gas release rate of 0.1 Ci/sec after 30 min.

TABLE 11.1-5

GENERAL ELECTRIC DATA
COOLANT ACTIVATION PRODUCTS IN REACTOR WATER AND STEAM

Isotope	Half-Life	Steam Concentration (uCi/g)	Reactor Water Concentration (uCi/g)
N-13	9.97 min	1.5×10^{-3}	5.8×10^{-2}
N-16	7.13 sec	5.0×10^{1}	4.1×10^{1}
N-17	4.17 sec	3.6×10^{-2}	1.8×10^{-2}
O-19	26.9 sec	5.9×10^{-1}	1.1×10^{-0}
F-18	109.8 min	4.4×10^{-4}	4.8×10^{-2}

TABLE 11.1-6

GENERAL ELECTRIC DATA
NONCOOLANT ACTIVATION PRODUCTS IN REACTOR WATER

Isotope	Half-Life	Concentration (uCi/g)
Na-24	14.97 hr	2.2×10^{-3}
P-32	14.28 day	2.2×10^{-5}
Cr-51	27.7 day	5.5×10^{-4}
Mn-54	312.2 day	4.4×10^{-5}
Mn-56	2.579 hr	5.5×10^{-2}
Co-58	70.91 day	5.5×10^{-3}
Co-60	5.272 yr	5.5×10^{-4}
Fe-59	44.51 day	8.8×10^{-5}
Ni-65	2.520 hr	3.3×10^{-4}
Zn-65	243.8 day	2.2×10^{-6}
Zn-69m	13.8 hr	3.3×10^{-5}
Ag-110m	249.8 day	6.6×10^{-5}
W-187	23.9 hr	3.3×10^{-3}

TABLE 11.1-7

GENERAL ELECTRIC DATA
POWER ISOLATION EVENT - ANTICIPATED OCCURRENCE

Isotope	Isotopic Spiking Activity (Ci/bundle)
I-131	2.1
I-132	3.2
I-133	5.0
I-134	5.4
I-135	4.8
Kr-83m	0.9
Kr-85m	2.2
Kr-85	0.5
Kr-87	4.3
Kr-88	6.1
Kr-89	8.0
Xe-131m	0.1
Xe-133m	0.3
Xe-133	11.6
Xe-135m	1.8
Xe-135	11.0
Xe-137	10.5
Xe-138	10.6

TABLE 11.1-8

GENERAL ELECTRIC DATA
REACTOR COOLANT FISSION PRODUCT RADIOHALOGENS

Isotope	Half-Life	Concentration* (uCi/g)
Br-83	2.39 hr	1.4×10^{-2}
Br-84	31.8 min	2.8×10^{-2}
Br-85	2.87 min	1.7×10^{-2}
I-131	8.040 day	1.3×10^{-2}
I-132	2.28 hr	1.2×10^{-1}
I-133	20.8 hr	8.4×10^{-2}
I-134	52.5 min	2.4×10^{-1}
I-135	6.585 hr	1.2×10^{-1}

* Based on noble gas release rate of 0.1 Ci/sec after 30 min.

11.2 LIQUID WASTE MANAGEMENT SYSTEMS

11.2.1 Design Bases

11.2.1.1 Power Generation Design Bases

The objective of the liquid waste management system (LWS) is to collect, monitor, and process for reuse or disposal all potentially-radioactive liquid wastes in a controlled manner that meets the objectives of 10CFR20 and 10CFR50 Appendix I, and at the same time does not limit operation of the plant.

The system provides for recycling of water for reuse in the plant. Sufficient treatment and diversity of types of equipment are available to process normal plant wastes to condensate quality. Discharge of processed liquid waste should be necessary only when the plant has a complete water inventory.

The system design also includes all items of reasonably demonstrated technology which, for a favorable cost-benefit ratio, effect reductions in the dose to the population reasonably expected to be within 50 mi of the reactor. The cost-benefit analysis is included in the report entitled Nine Mile Point Nuclear Station Unit 2 Compliance with 10CFR50 Appendix I (Appendix 11A).

The LWS has the capacity and capability of processing the anticipated quantities and activities of liquid wastes resulting from normal operation, maintenance, and anticipated operational occurrences. Cross-connections between the liquid waste subsystems provide additional flexibility for processing of wastes by alternate methods. These cross-connections are explained in Section 11.2.2 and illustrated on Figure 11.2-1.

The equipment and instrumentation selected for the LWS ensure that the radioactive concentrations resulting from liquid discharges from the plant are within the limits set forth in 10CFR20.

11.2.1.2 System Design Basis

The LWS is designed to minimize operational radiation exposure to plant personnel. Equipment has been selected, arranged, and shielded to permit operation, inspection, and maintenance while keeping personnel radiation exposure within the limits of 10CFR20 and as low as reasonably achievable (ALARA) in accordance with Regulatory Guide (RG) 8.8.

Where possible, equipment has been selected that requires minimum maintenance and has been located in accessible areas. Tanks, processing equipment, and associated piping that may contain significant quantities of radioactivity are shielded from personnel access areas and from controls or equipment requiring regular maintenance or operation. There are no outdoor tanks

containing potentially-radioactive liquid. Technical Requirements Manual (TRM) Section 3.7.7 provides requirements for unprotected outdoor tanks, should any be installed.

In some cases, several storage tanks are located in a common shielded area, since no regularly-scheduled maintenance is anticipated within the shielded tank compartments. All regularly-maintained equipment (pumps, valves, etc.) associated with these tanks is located outside the tank compartment but in other shielded and more accessible areas. In all cases, excess tank capacity is provided so that if an operational problem occurs within a compartment, the tank can be isolated without loss of unit operating capability.

To minimize maintenance, operational problems, and personnel exposure, the following items have been incorporated into the system design:

1. Air and condensate connections are provided at various points in all subsystems to facilitate flushing of piping and components.

2. Piping subsystems that can expect a high solids content (i.e., phase separator and evaporator bottoms piping) have been given special consideration in piping arrangement details as follows:

    a. Where possible, laterals and five-diameter bends are used in place of tees and elbows.

    b. Butt-welded fittings are used in place of socket-welded fittings.

    c. All butt welds are made using consumable inserts.

    d. Piping and components have been designed to minimize pockets and crevices that could create crud traps.

    e. Low-point drains have not been used in slurry piping runs.

    f. Pumping capacity and pipe sizes have been selected to ensure adequate flow velocity to prevent sedimentation.

    g. Instrumentation connections have been located in a manner to prevent crud traps. Indicators are in accessible areas.

3. Radiation and corrosion-resistant materials are used to reduce the need for component replacement over the 40-yr plant life.

4. Plug valves are used extensively to minimize valve stem leakage.

5. Equipment is employed which has proven reliability in radwaste service as well as other process industries.

6. Redundancy in components as well as processing flexibility between subsystems is provided to permit continued operation during periods of equipment repair. In addition, this redundancy permits appropriate scheduling of maintenance activities to minimize personnel exposure.

7. Pump seal leakage is minimized by the use of mechanical seals for most of the radwaste pumps. Double mechanical seals are used in pumps that handle slurry. Seal life is extended by the continuous supply of cool, clean seal water provided by the radwaste seal water system, thereby minimizing pump maintenance. Seal water pressure also is kept higher than the stuffing box pressure so that process fluid cannot leak out from the pumps.

The power supply to the system components is provided from non-Class 1E power sources.

Table 11.2-1 lists the equipment design data of each component in the LWS. Surge capacity is provided to cover contingencies such as processing equipment outages, back-to-back refueling outages, or abnormal evolutions resulting in the production of excessive waste volumes. Tank volume provided is in excess of that required by NUREG-0016 Revision 1, Table 1-4. See Section 11.2.2 for a description of holdup volumes and processing capacities.

NUREG-0016 Revision 1 provides guidance in calculational methods associated with liquid radwaste systems. The liquid radioactive waste tank holdup capacity is in excess of that required based on the daily expected average input flows from Table 1-4 of this NUREG (Section 11.2.2.9).

Decontamination factors used in analysis are consistent with Table 1-5 of this NUREG (Section 11.2.2.5).

11.2.1.2.1 Applicable Codes and Standards

The LWS is a nonnuclear safety, non-Category I system. Table 11.2-2 lists the applicable codes and standards for system equipment. The safety class and material selections for equipment in the LWS satisfy the criteria of RG 1.143 (Section 1.8).

The atmospheric storage tanks are filament-wound fiberglass reinforced plastic tanks. They are designed to meet or exceed National Bureau of Standards Voluntary Product Standard PS-15-69

and the American Society of Testing and Materials Specification No. ASTM D3299-74.

11.2.1.2.2 Structural Design

The radwaste equipment is housed in the radwaste building (Figures 1.2-13 and 1.2-14). Section 3.8 describes the building structural design criteria.

11.2.2 System Description

The LWS is divided into the following four subsystems: the waste collector subsystem, the floor drain collector subsystem, the regenerant waste subsystem, and the phase separator subsystem. These subsystems permit wastes from various sources to be combined according to similarity of conductivity and isotopic concentrations for appropriate processing. Flow paths and equipment are shown on Figure 11.2-1. There is no provision for laundry waste processing at Unit 2. Laundry facilities at Nine Mile Point Nuclear Station - Unit 1 (Unit 1) are shared by Nine Mile Point Nuclear Station - Unit 2 (Unit 2) for the decontamination of radiation protection apparel and breathing apparatus. Laundry waste effluent will be treated by the radwaste treatment system at Unit 1. Laundry services may be provided by outside vendors.

11.2.2.1 Waste Collector Subsystem

The waste collector subsystem collects, monitors, and processes for reuse or disposal relatively low-conductivity waste (less than 50 umho/cm) of variable radioactivity. This subsystem removes radioactivity from the liquid via filtration and ion exchange.

The subsystem normally receives input from the following sources:

1. Draindown from the spent fuel pool cooling and cleanup (SFC) system.
2. Draindown from the reactor water cleanup (RWCU) system.
3. Distillate from the waste evaporator.
4. Distillate from the regenerant evaporator.
5. Radwaste demineralizer effluent.
6. Reactor building equipment drains.
7. Decant from the phase separator tanks.
8. Turbine building equipment drains.
9. Draindown from the residual heat removal (RHR) system.

10. Waste collector surge contents.

11. Condensate demineralizer (CND) backwash flush water.

In addition, the subsystems can receive input from the following sources:

1. Floor drain filter effluent.

2. Floor drain collector surge tank.

3. Recovery sample tank recycle.

The waste collector subsystem consists of three waste collector tanks which receive the above inputs, and three waste collector pumps. In addition, a waste collector surge tank and two associated waste collector surge pumps are provided. The surge tank provides a surge capacity in excess of that of the waste collector tanks.

The processing equipment in this subsystem consists of two etched disc-type radwaste filters for removal of insoluble particles, and two mixed bed-type demineralizers for the removal of soluble and colloidal ionic material.

The waste collector subsystem has two recovery sample tanks that receive the effluent from the waste collector process. They are used for holding and testing the processed water before reuse or disposal via the recovery sample pumps.

The waste collector subsystem influents previously listed are collected and directed primarily to the waste collector tanks via a common header. In the event that additional storage capacity is required, influents can be routed to the waste collector surge tank. However, the surge tanks are normally reserved to receive overflows from the main tanks, rather than direct inputs from other systems. Inputs to the waste collector surge and floor drain collector surge tanks are run separately. All tanks are individually isolated from the collection header by air-operated valves (AOV). These valves permit tank isolation for processing and/or influent quality segregation.

Overflows from the three waste collector tanks are tied together and are gravity fed to the waste collector surge tank. This provides for direct collection of tank overflows and minimizes the potential for spillage. Overflow from the waste collector surge tank is directed to the floor of its compartment. The potential for surge tank overflow is extremely remote and would result from failure of high-level alarms in both the normal on-line collector tanks and the surge tanks. Alarms are provided in the radwaste control room where the Operators can identify and isolate inputs to the tanks as necessary. Waste tank level indications are also provided remotely in the main control room

through the emergency response facility (ERF) computer. The tank level detection is not interlocked to shut the waste inflows into the surge tanks. However, an overflow from the surge tank would indicate an excessive influx of liquid into the system which could possibly exceed the collection capacity of the radwaste building floor and equipment drains. Therefore, overflow from the surge tank is directed to the floor of the compartment where it is totally contained. This arrangement prevents the uncontrolled release of radioactive liquid to other sections of the radwaste building and, therefore, meets the intent of RG 1.143. The floor is sloped to a local small sump. A portable, submersible pump and hose would be used to transfer any overflow to the floor drain system and from there back to the liquid waste system. The steel liner prohibits any leakage to the environs.

The contents of the waste collector surge tank can be transferred back to the waste collector tanks, to the radwaste filter, or to the floor drain filter via the waste collector surge pumps. The recirculation lines from the waste collector surge pumps have connections for the addition of acid or caustic to neutralize the contents of the tank as necessary.

The waste collector pumps can take suction from any waste collector tank. The pumps, including the surge pumps, can provide recirculation to each respective tank through their mixing spargers. The spargers mix the tank contents to maintain a uniform volume prior to sampling and processing.

The waste collector pumps can transfer the contents of any of the waste collector tanks to the waste collector surge tank, the etched disc radwaste filters, or the Thermex system (Section 11.2.2.2). In no case can the contents of the waste collector tanks bypass a filtration process.

The etched disc filters are 100-percent capacity units. One is in operation while the other is in standby. The filters can be set to backwash either on high differential pressure or by manual initiation. When backwashing is required the standby filter is put into service and the operating filter is isolated. This isolated filter is then backwashed by high-pressure air from the air receiver tank. The filter backwash is sent to the radwaste filter backwash tank. After several discharges, when the filter backwash tank is full, its contents are transferred, via the radwaste filter backwash pumps, to the spent resin tank (Section 11.2.2.4), the solid waste sludge tank, or the regenerant waste tank. The radwaste filter backwash pump can also provide recirculation to the tank to mix its contents. The effluent from the radwaste filters can be sampled and directed to the waste discharge sample tanks, for discharge to the service water system (SWP) discharge bay, the floor drain collector tank or the waste collector surge tank for reprocessing, or the radwaste demineralizers for further processing, depending on its characteristics.

The radwaste demineralizers are 100-percent capacity, mixed, deep bed types. Each can be operated singly, with the other in standby or in parallel, depending on pressure drop through the system. The demineralizers are charged with resin from the condensate regeneration subsystem (Section 10.4.6). The resins are of identical type, volumes and anion/cation ratio as those used in the CNDs. Therefore, the condensate regeneration subsystem is designed to handle, clean and regenerate the condensate as well as the radwaste demineralizer system. When the resin is depleted or the demineralizers have a high differential pressure, the resin is transferred back to the condensate regeneration subsystem for regeneration, or to the spent resin tank for further processing and disposal.

The radwaste demineralizer effluent can be sampled and then sent through a backwashable strainer. The strainers prevent resin from being introduced to other parts of the system in the event of a failure of the demineralizer resin retention system. Demineralizer effluent can then be sent to one of three places: to the waste discharge sample tank for sampling and disposal, to the waste collector tanks for reprocessing, or to the recovery sample tanks for recycling.

In addition to receiving input from the radwaste demineralizers, the recovery sample tanks can receive input from the waste sample tanks of Unit 1 (intertie piping is cut and capped to isolate units until inter-unit service is required). The inlets of the recovery sample tanks can be individually isolated so that one tank can receive input while the other transfers its contents. Overflow from the recovery sample tank is directed to the waste collector surge tank. The contents of a recovery sample tank are mixed, via the recovery sample pump recirculation line and tank sparger, to provide a uniform tank mixture, and then sampled to determine water quality. Depending on water quality and plant water inventory, the contents of the recovery sample tanks can be sent to one of the following places: the waste collector tanks for reprocessing, the condensate storage tanks (CST) for reuse, the waste sample tanks of Unit 1 (under strict administrative control) for use in that plant's water inventory (intertie piping is cut and capped to isolate units until inter-unit service is required), or through the radwaste discharge radiation monitors for release to the SWP discharge bay.

11.2.2.2 Floor Drain Collector Subsystem

The floor drain collector subsystem collects, monitors, and processes for reuse or disposal potentially-high conductivity (greater than 50 umho/cm) waste of variable radioactivity. This subsystem normally receives input from the following sources:

1. Radwaste building floor drains.

2. Reactor building floor drains (including the drywell floor drain tank).

3. Auxiliary boiler area floor drains.
4. CST area floor drains.
5. Main stack area floor drains.
6. Effluent from the Thermex system.
7. Floor drain collector surge tank.
8. Decant from the spent resin tank.
9. Turbine building floor drains.

The system can receive inputs from the following sources:

1. Distillate from the waste evaporator.
2. Distillate from the regenerant evaporator.
3. Effluent from the radwaste filters.
4. Water from the waste discharge sample tanks.
5. Waste collector surge tank.
6. Floor drain collector tank at Unit 1 (intertie piping is cut and capped to isolate units until inter-unit service is required).
7. CND backwash flush water.

The floor drain collector subsystem includes two floor drain collector tanks which receive the above inputs, and two floor drain collector pumps. In addition, a floor drain collector surge tank and two associated floor drain collector surge pumps are provided. This tank provides a surge capacity in excess of that of the floor drain collector tanks.

The processing equipment in the subsystem consists of the Thermex system and a forced circulation-type evaporator for the concentration of soluble and insoluble waste.

Finally, the floor drain collector subsystem has two waste discharge sample tanks that receive the effluent from the floor drain collector processes. They are used for holding and testing the processed water before reuse or disposal via the waste discharge sample pumps.

The floor drain collector subsystem inputs are collected and directed to the floor drain collector and floor drain collector surge tanks via a common header. The tanks are individually isolated from the collection header by AOVs. These valves permit

tank isolation for processing, influent quality segregation, or maintenance.

Overflows from the floor drain collector tanks are tied together and are gravity fed to the floor drain collector surge tank. Overflow from the floor drain collector surge tank is directed to the floor of its compartment. The floor is sloped to permit the use of a portable, submersible pump to remove any overflow. The arrangement is the same as that for the waste collector surge tank (see Section 11.2.2.1). The steel liner prohibits any leakage to the environs.

The contents of the floor drain collector surge tank can be transferred back to the floor drain collector tanks, the waste or regenerant evaporators, the radwaste filters, or via the floor drain collector surge pumps. The floor drain collection surge pump recirculation lines have connections for the addition of acid or caustic to neutralize the contents of the tank as necessary.

The floor drain collector pumps can take suction from either floor drain collection tank. The pumps (including the surge pumps) provide recirculation to their tanks through their mixing spargers. The spargers mix the tank contents to ensure a uniform volume prior to sampling and processing.

The floor drain collector pumps can transfer the contents of either of the floor drain collector tanks to the following, depending on waste quality, radioactivity level, and equipment availability: the waste or regenerant evaporators, the radwaste filters, or the floor drain surge tank. In no case can the contents of the floor drain collection tanks bypass a filtration or evaporation process.

The Thermex system is a modular-designed wastewater treatment system utilizing reverse osmosis, ion exchange electrodionization and ultraviolet photo degradation methodologies. The modular design allows for simple setup and maintenance in addition to a high degree of operational flexibility. The Thermex system separates the dissolved and suspended solids from recycled wastewater. The dissolved and suspended solids waste products are concentrated, treated, and disposed of by an offsite vendor, or processed through 2LWS-TK7. The purified water from the Thermex effluent is returned to the plant systems via the floor drain filter effluent tank, and is transferred via the floor drain filter effluent pump to one of the following tanks, depending on the effluent quality: the waste discharge sample tanks for disposal, the floor drain collector tanks, the waste collector tanks, or the regenerant waste tanks for further processing.

The waste evaporator is a forced circulation design with a reboiler providing process heat and an overhead entrainment separator and distillation column that minimizes liquid droplets,

particulates, and volatiles in the vapor. The evaporator recirculation line has connections for the addition of acid or caustic to neutralize the contents of the evaporator as necessary. Concentrated evaporator bottoms are sent to the evaporator bottoms tank via the evaporator bottoms pumps. When the evaporator bottoms tank is full, its contents are transferred to the radioactive solid waste system for processing and disposal (Section 11.4). During operation, the evaporator overhead is condensed and subcooled. If the quality of the distillate is satisfactory, it is directed to the waste collector tanks to await demineralization. If the distillate quality is unacceptable, it can be returned to the floor drain collector tanks for reprocessing. If the distillate is to be discharged to the environment, it is directed to the waste discharge sample tanks.

The waste discharge sample tank can receive input from both the waste evaporator and the regenerant evaporator as well as the radwaste filters, the radwaste demineralizers, or the floor drain filter. The inlets to the waste discharge sample tank can be individually isolated so that one tank can receive input while the other transfers its contents. Overflow from these tanks is directed to the floor drain collector surge tank. The contents of the waste discharge sample tanks are mixed via the waste discharge sample pump recirculation line and the tank sparger to provide a uniform tank volume prior to sampling. If the results of sample analysis indicate that further processing or reprocessing is required, the contents of the waste discharge sample tank can be directed to the waste or regenerant evaporators or the floor drain collection tanks. If the results of the sample analysis indicate that no further processing is required, the tank contents can be discharged to the discharge bay. The final discharge is again checked by a radiation monitor. If predetermined activity levels are exceeded, the flow is automatically halted. The radiation monitor is located sufficiently upstream of the discharge valve to prevent any unacceptable release.

11.2.2.3 Regenerant Waste Subsystem

The main purpose of the regenerant waste subsystem is to collect, monitor, and process chemical solutions resulting from the acid/caustic regeneration of CND resins (conductivities greater than 1,000 umho/cm). This subsystem can also collect input from those radwaste building floor drains with a high anticipated chemical content, effluent from the Thermex system, and backwash from the radwaste filters for further processing. The regenerant waste evaporator is identical to the waste evaporator in all respects.

The regenerant waste subsystem includes two regenerant waste tanks, which collect the inputs described above, and two regenerant waste pumps.

The processing equipment of this subsystem consists of a forced circulation-type evaporator for the concentration of radioactive sodium sulfate. This evaporator is identical to the waste evaporator described in Section 11.2.2.2. This includes its sources of influent and the possible discharge paths of its effluent.

This subsystem also shares the evaporator bottoms tank and pumps and the waste discharge sample tanks and pumps of the floor drain collector subsystem. The regenerant waste subsystem inputs are collected and directed to the regenerant waste tanks via a common header. The tanks are individually isolated from the collection header by AOVs. These valves permit tank isolation for processing influent quality segregation or maintenance.

The regenerant waste pumps can take suction from either regenerant waste tank. The pumps also provide recirculation to either tank through their mixing spargers. The spargers mix the tanks' contents to maintain a uniform volume prior to sampling and processing. The pump recirculation lines have acid and caustic addition connections to permit neutralization of the tank contents as necessary.

The regenerant waste pumps can transfer the contents of the regenerant waste tanks to any of the following components, depending on waste quality and equipment availability: the floor drain filter, the waste evaporator, or the regenerant evaporator. The description of the process from this point on through reuse, reprocessing, or disposal is identical to that of the floor drain collector subsystem (Section 11.2.2.2).

11.2.2.4 Phase Separator Subsystem

The phase separator subsystem collects, decants, and holds for radioactive decay, the backwash from the RWCU filter/demineralizers, the SFC filter/demineralizers, the radwaste filters, and spent resins from the condensate and radwaste demineralizers.

The phase separator subsystem consists of three phase separator tanks, three associated phase separator pumps, one spent resin tank, and one associated spent resin pump.

Two of the phase separator tanks receive input from the RWCU filter/demineralizers. The third phase separator tank receives input from the SFC filter/demineralizer and can also receive the contents of the RWCU phase separator tank via the RWCU phase separator pumps. Each tank is capable of being isolated by an AOV. These valves permit tank isolation for holdup or maintenance.

The overflows from the RWCU phase separator tanks are tied together and are gravity fed to the reactor building floor

drains. The overflow from the SFC phase separator is also gravity fed to the reactor building floor drains.

The RWCU phase separator pumps can take suction from either RWCU phase separator tank and can also provide recirculation to either tank through their mixing eductors. The eductors mix the tanks' contents to maintain a homogeneous mixture prior to transferring their contents. The SFC phase separator and the spent resin tank, although separately piped, have the same features.

After the phase separators have received a backwash, the relatively clean water is drawn off through decant screens by the associated pump and transferred to the waste collector tanks. When the phase separator tanks are filled to their solids capacity, the contents are transferred, via their respective pumps, to the spent resin tank for holdup prior to disposal in the solid waste system.

In addition to receiving input from the phase separator tanks, the spent resin tank receives input from the radwaste filters and demineralizers, decant from the solid waste system sludge tank, and CNDs.

The spent resin tank can decant its contents to the floor drain collector tanks. This decant is gravity drained or pumped. Overflow from this tank is directed to the radwaste building floor and equipment drains. Spent resin tank contents are transferred, via the spent resin pump, to the radioactive solid waste system for further processing and disposal.

11.2.2.5 System Operational Analysis

The flow rate and the activity concentrations (fractions of primary coolant concentration) (Table 11.2-3) were developed using a material balance calculation and data from NUREG-0016 Revision 1.

Decontamination factors of the processing units are shown in Table 11.2-4.

11.2.2.6 Instrumentation and Control

The radwaste process computer receives input of system conditions throughout the LWS and provides indication of process operation, equipment performance, system status, and central control of process equipment via computer terminals and video displays. The radwaste control room is located in the turbine building, adjacent to the south side of the radwaste building at el 279 ft.

Fundamentally, the components of the waste collector, floor drain collector, regenerant waste, and phase separator subsystems employ similar basic components such as tanks, pumps, filters, demineralizers, and evaporators (in two of the subsystems). Like

components have similar instrumentation. Instrumentation for the different components is described in the following sections.

11.2.2.6.1 Tanks

In multiple tank subsystems (e.g., waste collector tanks 1A, 1B, and 1C), the inlet valves to each tank are interlocked in such a way that one is always open so that one tank is available to accept influents.

Liquid radwaste tanks have level transmitters which provide a trouble alarm when a high-level setpoint is reached.

11.2.2.6.2 Pumps

The liquid radwaste process pumps (e.g., waste collector pumps 1A, 1B, and 1C) rely on tank level transmitters and the radwaste computer to shut down the pump on low tank levels.

The discharge side of the pumps has the following instrumentation: pressure, temperature, and conductivity. The suction side of the pumps has pH instrumentation.

Pumps with double mechanical seals in the radwaste building are supplied with recirculating, filtered, and cooled seal water. Radwaste pumps in the reactor building are supplied with filtered once-through seal water. Instrumentation has been provided to regulate the seal water flow, to monitor the condition of the seals via pressure retention, and to trip pumps if the seal water flow is interrupted. Flow regulation and monitoring instruments are located in a general access area to minimize radiation exposure to personnel. The seal water system is shown on Figure 11.2-1.

11.2.2.6.3 Radwaste Filters

The radwaste filters have differential pressure transmitters that monitor the pressure drop across the filter and automatically initiate the backwash cycle when a high differential pressure setpoint is reached.

The radwaste filters utilize time delays for the backwash cycle sequence of operation. This permits the filters to be put back into service.

11.2.2.6.4 Radwaste Demineralizers and Resin Traps

Each radwaste demineralizer has a differential pressure transmitter to monitor the pressure drop across the demineralizer, and a conductivity element in the demineralizer outlet line to monitor effluent conductivity. The demineralizers can be backwashed, regenerated, or ultrasonically cleaned due to either high differential pressure or high effluent conductivity.

These operations will be performed utilizing the equipment in the CND system.

Influent to the demineralizers is maintained by a flow element and transmitter that controls a flow control valve (FCV).

The resin traps on the discharge side of the demineralizers also have differential pressure instruments. Radwaste Operators monitor demineralizer strainer differential pressure on a weekly basis for evidence of demineralizer retention screen resin leakage.

11.2.2.6.5 Floor Drain Filter (Retired, Removed) Associated Tanks and Pumps

The tanks associated with the floor drain filter (retired, removed) (body feed, precoat, and effluent) have level switches to shut down their associated pumps on low level and alarm on high level. The pumps have discharge pressure indication.

11.2.2.6.6 Evaporators (Waste and Regenerant)

The evaporators have level transmitters to control the inlet feed flow and to provide trouble alarms.

Pressure, temperature, and liquid level are measured at various points throughout the evaporator system to monitor performance and initiate the various modes of operation.

The conductivity of the effluent distillate is monitored to determine the required flow path (i.e., further processing, disposal, or reuse).

11.2.2.6.7 Diaphragm Seals

Instruments that measure parameters (pressure or level) of liquids with high radioactivity and solids content have diaphragm seals. These seals physically separate the radioactive liquid from a signal transmitting fluid located in a shop-sealed capillary tube including the sensor. This reduces personnel exposure to radiation because it keeps the radioactive liquid within a shielded equipment cubicle while the instrument racks, located in general access areas, contain only nonradioactive transmitting fluid. These diaphragm seals also reduce the possibility of instrument lines plugging with radioactive crud.

11.2.2.7 System Operation

The operating procedures used for all liquid radwaste equipment are based on batch processing through radwaste systems. This type of operation allows time to sample and check the feed and effluent streams before and after each process step to prevent the inadvertent discharge of waste having a radioactivity level above the predetermined limit.

It also allows for maximum reuse of water that has been processed through the radwaste system.

As previously noted, the influents to the LWS can vary widely in activity, conductivity, and volume, depending on the operating conditions of the plant. A description of the conditions that may be used for guidance for the disposition of batches is as follows:

1. High conductivity: evaporate, demineralize, and recover.

2. Low gross activity and high conductivity: filter, evaporate, demineralize, and recover.

3. High gross activity and low conductivity: filter, demineralize, and recover.

4. High activity and high conductivity: filter, evaporate, demineralize, and recover.

11.2.2.8 Performance Tests

Performance tests that monitor the removal of insolubles and organics by filters, reduction of conductivity by demineralizers, and the removal of solubles by the evaporators are conducted on a periodic basis. The radioactivity of the input and output streams of each piece of equipment is checked periodically. Performance tests verify the decontamination factors and other aspects including:

1. Filters:

   a. Insoluble material removal efficiency.

   b. Measurement of influent and effluent activity.

2. Demineralizers:

   a. Conductivity reduction factor (i.e., demineralizer efficiency).

   b. Measurement of influent and effluent activity.

3. Evaporators:

   a. Insoluble material removal efficiency.

   b. Measurement of influent and effluent activity.

   c. Distillate conductivity.

Overall system tests for the waste collector subsystem, the floor drain collector subsystem, and the regenerant waste subsystem establish the overall decontamination factor for the entire subsystem.

11.2.2.9 Summary

Over 225,000 gal of bulk storage capacity is available to collect and store the average liquid radwaste influent volumes of approximately 41,000 gpd. Storage facilities are also available to receive treated liquid. The treated liquid bulk storage capacity is 100,000 gal which allows sufficient time for sampling and disposal of approximately 2,000 gpd to the environs and recycling approximately 39,000 gpd tank.

11.2.3 Radioactive Release and Doses

Table 11.2-5 is a tabulation of the expected annual liquid releases. The release estimates corresponding to EPU conditions from the liquid effluent stream in Ci/yr per nuclide are given in Table 11.2-6, and correspond to operation with design-failed fuel conditions as discussed in Section 11.1. The release is based on effluent discharge to the environs only through the recovery sample tanks and/or waste discharge sample tanks.

Tritium release from liquid pathways is anticipated at 55 Ci/yr.

11.2.3.1 Release Points

All liquid releases from Unit 2 are fed into the SWP discharge bay that is directed to Lake Ontario. Figure 11.5-8 shows all systems that feed the discharge bay. Figure 11.2-1 shows the release point of each system to the discharge line. Figure 1.2-29 shows the physical location of the discharge into the bottom of the lake.

11.2.3.2 Dilution Factors

The dilution factor used in evaluating the release of radioactive liquid effluents is that provided by the cooling tower blowdown and service water discharge.

Treated radioactive effluents are diluted in the discharge bay with a combined plant discharge (cooling tower blowdown and service water discharge) flow of 30,428 gpm (6.05×10^{13} cc/yr).

Prior to discharging any LWS effluents to the lake, the waste discharge sample tank and/or the recovery sample tank contents are recirculated (to provide a homogeneous mixture) and sampled. The analysis of this sample is used to determine the discharge rate and amount of dilution needed for a release within the allowable limits.

To prevent any accidental discharge of effluents greater than the allowable limits, an isolation valve in the LWS discharge will close upon receipt of a high radiation signal. Once the effluent stream has been isolated and sampled, it may be recirculated through the LWS to reduce the activity concentration.

11.2.3.3 Estimated Doses

A summary of the estimated annual radiation doses is presented in Appendix 11A, which shows that the estimated annual doses from liquid releases are below the dose criteria set forth in 10CFR50 Appendix I.

The design liquid release isotopic radioactive concentration estimates (corresponding to EPU conditions) in the unrestricted area are shown in Table 11.2-6, along with the indication that these concentrations are fractions of Maximum Permissible Concentration (MPC) limits given in 10CFR20 Appendix B, Table II, Column 2.

TABLE 11.2-1

EQUIPMENT DESCRIPTIONS
LIQUID WASTE MANAGEMENT SYSTEM

A. Tanks

Item No. 2LWS-TK-	Name	Capacity Each (gal)	Design		Material	Quantity
			Temperature (°F)	Pressure (psig)		
1A,B,C	Waste collector	25,000	150	Atmosphere	Fiberglass	3
2A,B	Floor drain collector	25,000	150	Atmosphere	Fiberglass	2
3A,B	Regenerant waste	25,000	150	Atmosphere	Fiberglass	2
4A,B	Recovery sample	25,000	150	Atmosphere	Fiberglass	2
5A,B	Waste discharge sample	25,000	150	Atmosphere	Fiberglass	2
6A,B	RWCU phase separator	8,200	150	15	Stainless steel	2
7	Spent resin	7,000	150	15	Stainless steel	1
8	Waste evaporator distillate	470	350	100	Stainless steel	1
9	Regenerant evaporator distillate	470	350	100	Stainless steel	1
10	CFS phase separator/Evaporator bottoms	8,000	350	25	Incoloy	1
12	Floor drain filter body feed	150	104	Atmosphere	Carbon steel plastic lined	1
14	Floor drain filter effluent	200	104	Atmosphere	Carbon steel plastic lined	1
15	Air receiver	60	250	350	Stainless steel	1
16	Radwaste filter backwash	1,100	150	150	Stainless steel	1
17	Floor drain collector surge	29,800	150	Atmosphere	Fiberglass	1
18	Waste collector surge	29,800	150	Atmosphere	Fiberglass	1
20	Floor drain filter precoat	1,000	104	Atmosphere	Carbon steel plastic lined	1
30	SFC phase separator	8,200	150	15	Stainless steel	1

TABLE 11.2-1 (Cont'd.)

B. Pumps

Item No. 2LWS-P-	Name	Capacity (gpm)	TDH (ft)	Design Pressure (psig)	Quantity
1A,1B,1C	Waste collector	220	391	275	3
2A,2B	Floor drain collector	110	312	275	2
3A,3B	Regenerant waste	110	330	275	2
4A,4B	Recovery sample	110	211	275	2
5A,5B	Waste discharge sample	165	171	275	2
6A,6B	RWCU phase separator	110	238	275	2
7	Spent resin	66	49	275	1
8	Waste evaporator distillate transfer	30	22	255	1
9	Regenerant evaporator distillate transfer	30	22	225	1
10A,10B	Evaporator bottoms	60	105	225	2
13	Waste evaporator recirculation	4,000	98	80	1
11	Regenerant evaporator recirculation	4,000	98	80	1
15	Floor drain filter effluent	95	140	125	1
16A,16B	Radwaste filter backwash	110	53	275	2
17A,17B	Floor drain collector surge	110	331	275	2
18A,18B	Waste collector surge	220	431	275	2
20	Caustic addition	1	**	110	1
21	Acid addition	1	**	110	1
27	Floor drain filter precoat	200	55	125	1
28	Floor drain filter body feed	61.8	508	165	1
30	SFC phase separator	110	295	275	1

TABLE 11.2-1 (Cont'd.)

C. Miscellaneous Heat Exchangers

Item No. 2LWS-E-	Name	Duty Btu/Hr x 10^6	Design Temperature (°F) Shell/Tube	Design Pressure (psig) Shell/Tube	Material Shell/Tube	Quantity
1	Waste evaporator condenser	12.99	300/150	Full vacuum 100/200	Stainless steel/ stainless steel	1
2	Waste evaporator distillate cooler	1.76	150/300	150/Full vacuum and 100	Stainless steel/ carbon steel	1
4	Waste evaporator reboiler	15.65	350/350	Full vacuum and 100 (both)	Incoloy/ carbon steel	1
5	Regenerant evaporator condenser	12.99	300/150	Full vacuum and 100/200	Stainless steel/ stainless steel	1
6	Regenerant evaporator distillate cooler	1.76	150/300	200/Full vacuum and 100	Stainless steel/ carbon steel	1
7	Regenerant evaporator reboiler	15.65	350/350	Full vacuum and 100 (both)	Incoloy/ carbon steel	1

D. Demineralizers

Item No. 2LWS-DEMN-	Capacity (gpm)	Volume (ft^3)	Design		Type	Material	Quantity
			Temperature (°F)	Pressure (psig)			
4A,B	100	110/110	140	150	Mixed bed	Rubber lined carbon steel	2

TABLE 11.2-1 (Cont'd.)

E. Evaporators

Item No. 2LWS-EV-	Capacity (gpm)	Design		Operating		Batch Volume (gal)	Material	Quantity
		Temperature (°F)	Pressure (psig)	Temperature (°F)	Pressure (psig)			
1,2	30	350	100/full vacuum	250	15	3,450***	Stainless steel and Incoloy	2

F. Filters

Item No. 2LWS-FLT-	Name	Design		Capacity (gpm)	Flow Flux (gpm/ft^2)	Material	Quantity
		Temperature (°F)	Pressure (psig)				
1A,B	Radwaste	150	350	200	6.7	Stainless steel	2

G. Strainers

Item No. 2LWS-STR-	Capacity (gpm)	Design		Operating		Material	Quantity
		Temperature (°F)	Pressure (psig)	Temperature (°F)	Pressure (psig)		
5A,B	100	140	150	120	105	Carbon steel	2

* Progressive cavity-type 80 psig maximum discharge pressure.
** Not applicable (positive displacement-type pump).
*** Includes liquid in reboiler and connecting piping.

TABLE 11.2-2

APPLICABLE CODES AND STANDARDS
FOR LIQUID WASTE MANAGEMENT SYSTEMS

Description	Safety Class	Code	Code Class	Earthquake Criteria	Tornado Criteria	Quality Assurance Category 1
Tanks (steel or alloy)	4	ASME VIII*	-	No	No	No
Tanks (fiberglass)	4	NBS PS15-69	-	No	No	No
Etched disc filter	4	ASME VIII	-	No	No	No
Demineralizers	4	ASME VIII	-	No	No	No
Heat exchangers and reboilers	4	ASME VIII	-	No	No	No
Evaporators	4	ASME VIII	-	No	No	No
Pumps	4	HIS	-	No	No	No

* All except the RWCU phase separator tanks (2LWS-TK6A and 6B), which were not rehydrotested after field modification and, therefore, had their code stamps removed.

TABLE 11.2-3

MATERIAL AND CONCENTRATION BALANCE
FOR LIQUID WASTE MANAGEMENT SYSTEM

Input	Average Flow (gpd)	Fraction of Primary Coolant Activity (PCA)
Floor drains		
Reactor building	4,320	0.001
Radwaste building	1,000	0.001
Turbine building	2,500	0.001
Lab drains	500	0.02
Chemical lab waste	100	0.02
Floor drain subtotal	8,420	
Equipment drains		
Drywell	3,400	1.00
Secondary containment and auxiliary bays	3,700	0.10
Radwaste building	1,100	0.10
Turbine building	3,000	0.001
Equipment drain subtotal	11,200	
Other sources		
Ultrasonic resin cleaner	15,000	0.05
Resin rinse	3,150	0.002
Phase separator decant	1,050	0.002
Regenerants	1,725	-
Total	40,545	

TABLE 11.2-4

DECONTAMINATION FACTORS OF
PROCESSING UNITS

Equipment	DF	Remarks
Radwaste and floor drain collector filters	1	For corrosion/activation products (insolubles)
Demineralizers		
High-purity wastes	10	For Cs and Rb
	100	For anions and other nuclides
Low-purity wastes	2	For Cs and Rb
	100	For anions and other nuclides
Waste and regenerant evaporators	1,000	For anions
	10,000	For all nuclides except anions

TABLE 11.2-5

EXPECTED ANNUAL LIQUID RELEASES*

Isotope	Activity (uCi/ml)	Released (Ci/yr)
Na-24	1.6E-18	9.5E-11
P-32	1.1E-12	6.3E-05
Cr-51	2.5E-10	1.5E-02
Mn-54	9.4E-12	5.5E-04
Mn-56	0.0	0.0
Fe-55	3.0E-10	1.9E-02
Fe-59	4.3E-12	2.5E-04
Co-58	1.8E-11	1.1E-03
Co-60	5.8E-11	3.5E-03
Ni-63	3.0E-13	1.9E-05
Ni-65	0.0	0.0
Cu-64	0.0	0.0
Zn-65	2.6E-11	1.6E-03
Zn-69	0.0	0.0
Sr-89	1.6E-11	9.5E-04
Sr-90	2.2E-12	1.3E-04
Sr-91	0.0	0.0
Sr-92	0.0	0.0
Y-91	8.6E-12	5.2E-04
Y-92	0.0	0.0
Y-93	0.0	0.0
Zr-95	1.4E-12	8.4E-05
Zr-97	0.0	0.0
Nb-95	8.8E-13	5.3E-05
Nb-98	0.0	0.0
Mo-99	2.1E-13	1.2E-05
Tc-99m	0.0	0.0
Tc-101	0.0	0.0
Tc-104	0.0	0.0
Ru-103	2.4E-12	1.4E-04
Ru-105	0.0	0.0
Ru-106	8.6E-13	5.2E-05
Ag-110m	2.7E-13	1.6E-05
Te-129m	4.2E-12	2.7E-04
Te-131m	1.5E-16	9.0E-09
Te-132	1.7E-15	1.0E-07
Ba-139	0.0	0.0
Ba-140	8.0E-12	4.8E-04
Ba-141	0.0	0.0
Ba-142	0.0	0.0
La-142	0.0	0.0
Ce-141	3.2E-12	1.9E-04
Ce-143	0.0	0.0
Ce-144	8.1E-13	4.9E-05

TABLE 11.2-5 (Cont'd.)

Isotope	Activity (uCi/ml)	Released (Ci/yr)
Pr-143	0.0	0.0
Nd-147	0.0	0.0
W-187	6.3E-17	3.8E-09
Np-239	4.2E-13	2.5E-05
Br-83	0.0	0.0
Br-84	0.0	0.0
Br-85	0.0	0.0
I-131	7.2E-11	4.3E-03
I-132	1.0E-15	5.8E-08
I-133	1.8E-14	1.1E-06
I-134	0.0	0.0
I-135	0.0	0.0
Rb-89	0.0	0.0
Cs-134	3.5E-10	2.3E-02
Cs-136	1.7E-11	1.0E-03
Cs-137	1.1E-09	6.5E-02
Cs-138	0.0	0.0
H-3	9.1E-07	5.5E+01

* Anticipated operational occurrences 1.00E-01 Ci/yr added to release. Dilution release rate (gm/yr) 6.05E+13. Total release (excluding tritium) is 1.4E-01. Total release (excluding tritium) is 2.3E-09 uCi/gm.

NOTE: 4.8E-18 = 4.8×10^{-18}

TABLE 11.2-6

PRE-OPERATIONAL DESIGN ANNUAL LIQUID RELEASE ESTIMATES

Isotope	Design Activity Released uCi/ml	Ci/yr	MPC[1] (uCi/ml)	Fraction of MPC[2]
Na-24	5.0E-18	3.0E-10	2.0E-4	2.5E-14
P-32	3.4E-12	2.0E-4	2.0E-5	1.7E-7
Cr-51	3.1E-10	1.9E-2	2.0E-3	1.6E-7
Mn-54	1.9E-11	1.1E-3	1.0E-4	1.9E-7
Mn-56	0.0	0.0	1.0E-4	0.0
Fe-55	1.9E-10	1.2E-2	8.0E-4	2.4E-7
Fe-59	2.0E-11	1.2E-3	6.0E-5	3.3E-7
Co-58	1.6E-9	9.7E-2	1.0E-4	1.6E-5
Co-60	2.6E-10	1.6E-2	5.0E-5	5.3E-6
Ni-63	1.9E-13	1.2E-5	3.0E-5	6.3E-9
Ni-65	0.0	0.0	1.0E-4	0.0
Cu-64	0.0	0.0	3.0E-4	0.0
Zn-65	3.3E-11	2.0E-3	1.0E-4	3.3E-7
Sr-89	3.0E-10	1.8E-2	3.0E-6	1.0E-4
Sr-90	4.4E-11	2.5E-3	3.0E-7	1.5E-4
Sr-91	0.0	0.0	7.0E-5	0.0
Sr-92	0.0	0.0	7.0E-5	0.0
Y-91	3.8E-11	2.3E-3	3.0E-5	1.3E-6
Y-92	0.0	0.0	6.0E-5	0.0
Y-93	0.0	0.0	3.0E-5	0.0
Zr-95	6.3E-12	3.9E-4	6.0E-5	1.1E-7
Zr-97	0.0	0.0	2.0E-5	0.0
Nb-95	4.0E-12	2.4E-4	1.0E-4	4.0E-8
Mo-99	1.4E-12	8.3E-5	2.0E-4	7.2E-9
Tc-99m	0.0	0.0	6.0E-3	0.0
Ru-103	1.1E-11	6.7E-4	8.0E-5	1.4E-7
Ru-105	0.0	0.0	1.0E-4	0.0
Ru-106	3.8E-12	2.3E-4	1.0E-5	3.8E-7
Ag-110m	2.9E-11	1.7E-3	3.0E-5	9.6E-7
Te-129m	2.2E-11	1.4E-3	3.0E-5	7.3E-7
Te-131m	6.5E-16	4.0E-8	6.0E-5	1.1E-11
Te-132	1.5E-12	9.1E-5	3.0E-5	5.1E-8
Ba-140	1.1E-10	6.6E-3	3.0E-5	3.6E-6
Ce-141	1.4E-11	8.5E-4	9.0E-5	1.6E-7
Ce-143	0.0	0.0	4.0E-5	0.0
Ce-144	5.5E-12	3.4E-4	1.0E-5	5.5E-7
Pr-143	0.0	0.0	5.0E-5	0.0
Nd-147	0.0	0.0	6.0E-5	0.0

TABLE 11.2-6 (Cont'd.)

Isotope	Design Activity Released uCi/ml	Ci/yr	MPC[1] (uCi/ml)	Fraction of MPC[2]
W-187	1.1E-15	6.6E-8	7.0E-5	1.6E-11
Np-239	8.0E-12	4.8E-4	1.0E-4	8.0E-8
Br-83	0.0	0.0	3.0E-6	0.0
I-131	2.8E-10	1.7E-2	3.0E-7	9.4E-4
I-132	8.5E-15	4.8E-7	8.0E-6	1.1E-9
I-133	1.3E-13	8.3E-6	1.0E-6	1.3E-7
I-134	0.0	0.0	2.0E-5	0.0
I-135	0.0	0.0	4.0E-6	0.0
Cs-134	1.8E-9	1.1E-1	9.0E-6	2.0E-4
Cs-136	8.5E-11	5.1E-3	9.0E-5	9.4E-7
Cs-137	5.3E-9	3.2E-1	2.0E-5	2.7E-4
H-3[3]	9.1E-7	5.5E+1	3.0E-3	3.0E-4
Total[4]	1.1E-8	6.4E-1	-	1.7E-3

Dilution Release Rate = 6.05 x 10^{13} gm/yr

(1) MPC values are from 10CFR20 Appendix B, Table 11, Column 2.

(2) Fraction of MPC = design activity released (uCi/ml)/MPC (uCi/ml).

(3) Tritium release is in accordance with NUREG-0016 Rev. 1, 1/79, pp 1-8, Section 1.5.1.10.

(4) All totals are excluding tritium.

NOTE: 4.8E-18 = 4.8x10^{-18}

11.3 GASEOUS WASTE MANAGEMENT SYSTEMS

This section describes the capabilities of Unit 2 to collect, control, process, store, and dispose of gaseous radioactive waste generated from normal operation and anticipated operational occurrences. The gaseous waste management systems include the offgas (OFG) system, standby gas treatment system (SGTS), and various building ventilation systems. The SGTS and building ventilation systems are discussed in Sections 6.5.1 and 9.4, respectively. Section 3.1 discusses compliance with the General Design Criteria (GDC).

11.3.1 Design Bases

The design of the gaseous waste management systems meets the following criteria:

1. The systems have the capability to meet the requirements of 10CFR20 and the dose design bases specified in Appendix I to 10CFR50, including provisions to treat gaseous radioactive waste in such a way that:

    a. The calculated total annual quantity of all radioactive material released from Unit 2 to the atmosphere does not result in an estimated annual external dose from gaseous effluents to any individual in unrestricted areas beyond the site boundary in excess of 5 mRem to the whole body or 15 mRem to the skin.

    b. The calculated total annual quantity of all radioactive iodine and radioactive material in particulate form released from Unit 2 in the effluents to the atmosphere does not result in an estimated annual dose for any individual in an unrestricted area beyond the site boundary 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 10CFR20 Appendix B, Table II, Column 1.

2. The OFG 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 and during periods of greater than normal gaseous waste generation.

3. In accordance with RG 1.29, the radioactive gaseous waste systems are classified as nonseismic. An analysis of a postulated waste gas tank failure was

performed according to the guidance provided in RG 1.98. A description of this analysis and the results are given in Section 15.7.1. In accordance with RG 1.26, the OFG system is classified as nonnuclear safety, Quality Group D, and is designed to withstand the effects of a hydrogen detonation where the potential for an explosive mixture exists. Instrumentation with automatic alarm is provided to monitor the concentrations of the hydrogen gas in those portions of the system having the potential for an explosive gas mixture.

a. Each hydrogen analyzer annunciates both on the local panel and in the control room. The control room annunciation is a trouble alarm.

b. The hydrogen analyzers are of the nonsparking thermo-conductivity type. The analysis is off line. Gas is extracted from the main stream through instrument piping, analyzed, and returned to the main stream. Because the analyzers are off line, a hydrogen explosion will not affect the analyzers, although they are not explosion proof.

c. See Figures 11.3-1a, 11.3-1b, and 11.3-1c.

d. See Section 11.3.2.1 for a discussion of protective features.

(The description of hydrogen detonation design criteria is vendor proprietary and was submitted under separate Proprietary cover in response to Nuclear Regulatory Commission (NRC) Question F460.12.)

The design pressure due to hydrogen detonation is as follows:

	psig
All tanks and vessels	350
Piping:	
Air ejector to preheater	575
Preheater to recombiner	260
Recombiner to condenser	260
Condenser to discharge manifold	575
Vacuum pump	200

4. The system design contains provisions to control leakage and to facilitate operation and maintenance in accordance with the guidelines of RG 1.143.

Table 11.3-1 gives the expected annual quantities of radioactive gaseous effluent from all sources (by radionuclide) in Ci/yr. Table 11.3-2 lists the data used in calculating the annual releases of radioactive gaseous effluents. Table 11.3-3 shows the design annual radioactive gaseous releases versus the maximum permissible concentration (MPC). Figure 11.3-1 shows the piping and instrumentation diagram (P&ID) for the OFG system.

Appendix 11A, in compliance with 10CFR50 Appendix I, shows that the Unit 2 systems contain all items of reasonably demonstrated technology that will effect a reduction in dose to the population reasonably expected to be within a 50-mi radius of the plant. The OFG system is located in the turbine building. This portion of the turbine building is designed to meet Category I requirements. Section 15.7.1 describes the analysis of a radioactive gaseous waste system failure.

The following design features are incorporated to minimize maintenance, equipment downtime, leakage, and radioactive gaseous releases. These features facilitate offgas operation and assist in maintaining Operator exposures ALARA.

1. Major components are duplicated to provide two parallel offgas trains.

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

3. The hydrogen concentration in the offgas is kept below the flammable limits by maintaining the required steam flow for dilution at all times.

4. The operating pressure of the OFG system is subatmospheric. Because of the inherent pressure differential between the system and the atmosphere, leakage will be negligible.

5. Leakage of radioactive gas is further reduced by using welded construction wherever possible.

6. The OFG system can be operated from a single remote control panel which is located in a non-radiation area.

7. The system is supplied from non-Class 1E power sources.

Hydrogen concentration will be maintained below the 4 volume percent flammability limit by maintaining the required steam flow for dilution at all power levels. In addition, hydrogen analyzers will be used to confirm these concentrations.

Operability requirements regarding offgas system explosive gas monitoring instrumentation are described in TRM Section 3.3.11.

11.3.2 System Description

11.3.2.1 Offgas System

Noncondensable radioactive process offgas is continuously removed from the main condensers by the steam jet air ejectors (SJAE). The offgas from the main condensers will be the major source of radioactive gas, and is greater than all other sources combined.

The condenser offgas will normally contain activation gases, principally N-16, O-19, and N-13. It will also contain the radioactive noble gas parents of Sr-89, Sr-90, Ba-140, and Cs-137 (Table 11.3-1).

In addition to removing process offgas from the main condensers, the air ejectors also provide adequate mode of force for the gases to flow to the OFG system. The design feed to each offgas train is:

Dry air	40 scfm @ 60°F
H_2	136 scfm
O_2	68 scfm
H_2O	3,740 scfm as dilution steam
Noble gases	Negligible

The offgas mixture enters the system through a preheater. This preheater uses steam to raise the temperature of the gaseous mixture from 260°F to 290°F. The increase in temperature will increase the reaction rate in the recombiner. Hydrogen and oxygen react in the recombiner to form water, thus reducing the potential for hydrogen flammability. Once the system is in normal operation, the steam supply to the preheaters may be secured.

The newly-formed water vapor and dilution steam are removed downstream of the recombiner by a condenser. This also serves to remove the heat of reaction developed in the recombiner. The condensate returns to the main condenser hotwell. Downstream of the OFG system condenser is a 75-sec delay pipe, where the short-lived radioisotopes, such as N-16, N-17, and O-19, decay.

The gas is then passed through dryers which lower the moisture content. This is desirable since moisture reduces the efficiency of the charcoal beds downstream. The charcoal beds are arranged in two parallel trains of four each. Each train is designed to handle 50 percent of the gas flow. The trains can be operated in parallel or in series.

Xenon and krypton isotopes will be adsorbed on the charcoal and delayed from the bulk carrier gas (essentially air) permitting

them to decay, thereby significantly reducing the offsite dose. Design basis holdup times of 20 days for xenon and 26.6 hr for krypton have been selected. Following the charcoal beds is a high-efficiency particulate air (HEPA) filter to collect solid decay products and charcoal fines.

Vacuum pumps are utilized to provide the motive force in drawing the offgas through the system. These pumps discharge the waste gas through the main stack, where it is monitored for radioactivity prior to release to the atmosphere.

The hydrogen water chemistry (HWC) system includes an oxygen injection system to offgas, upstream of the offgas recombiner, to maintain a stoichiometric mixture of hydrogen and oxygen in the recombiner. The system is provided due to an excess ratio of hydrogen to oxygen at the entrance to the offgas system because of hydrogen injection through the condensate system.

The HWC system includes an additional offgas sample system for monitoring of the offgas oxygen concentration from the recombiners to assure that the oxygen addition flows are properly balanced. The sample supply for the oxygen analyzers is taken from the same general location as the existing hydrogen samples.

Control Features of Offgas System

Control Panel The control panel, located in a non-radiation area, provides all remote monitoring and control functions of the OFG system. It contains visual display of various system parameters, annunciator systems (lights, audible alarms, and controls), system status lights and control switches, and a system mimic display mounted on the face of the control panel. The panel will alarm and control all necessary components so the Operator will have complete knowledge of system status. The system has been designed to operate completely unattended during normal operation. The control panel has alarm provisions for the following system parameters:

1. System inlet temperature.
2. System inlet pressure.
3. Recombiner inlet temperature (low).
4. Recombiner inlet temperature (high).
5. Catalytic recombiner outlet temperature.
6. Condenser outlet temperature.
7. Condenser outlet hydrogen concentration.
8. Condenser high level.

9. Condenser low level.
10. Condenser combined outlet hydrogen concentration.
11. Dryer temperature.
12. HEPA filter differential pressure.
13. Vacuum pump inlet pressure.
14. Train shutdown.
15. Vacuum pump motor overload.
16. Refrigerator motor overload.
17. Pretreatment high radiation.
18. Dryer drain tank water level (high).
19. Dryer drain tank water level (low).
20. Dryer differential pressure (high).
21. Discharge valve to stack closed.

The offgas control panel is located in the offgas area in the turbine building. The control room has a trouble alarm which alarms on all the items noted in Section 11.3. In addition, the specified reason for an alarm is printed out on a computer log located in the control room.

Moisture Transmitter The moisture transmitter is located upstream of the charcoal adsorbers and is designed to provide a dew point readout for the process steam. The alarm for high process dew point is obtained from the dryer intermediate stage temperature sensing loop, which is essentially the output temperature of the dryers due to the freeze-out coils being eliminated from the dryer loops. This temperature is an appropriate measure of dew point as the gas in the dryer intermediate stage is at saturation, thus producing a 100-percent humidity level. The temperature alarm serves to guard the charcoal adsorbers from moisture contamination resulting from freeze-out dryer or condenser damage.

Hydrogen Analyzers Three hydrogen analyzers are installed in the process flow line to monitor hydrogen concentration. One hydrogen analyzer is installed downstream of each condenser and analyzes process flow hydrogen concentration and transmits a signal to an annunciator on the control room panel and offgas control panel.

The third analyzer is installed in the common process piping to the dryers and is designed to monitor hydrogen concentration.

This analyzer also transmits a signal to an alarm annunciator on the control panel.

Oxygen Analyzers Two oxygen analyzers are installed in the process flow lines to monitor oxygen concentration in the offgas flow. The analyzers can be configured to monitor the offgas flow at any two of three sample points where the hydrogen analyzers are installed (A train, B train, or common header). The oxygen analyzers will be used to support the operation of the hydrogen water chemistry (HWC) system. A low percent oxygen in the offgas will provide a signal to the HWC system to isolate the hydrogen and oxygen injection lines.

Protective Features All critical system parameters are automatically monitored during automatic operation. Certain abnormal conditions will automatically shut down the process train in which the abnormal condition exists. These conditions are as follows:

1. Condenser high outlet temperature.

2. Recombiner high outlet temperature.

3. Preheater low-low outlet temperature.

5. Manual or automatic vacuum pump shutdown.

6. Pretreatment high radiation level.

Radiation Monitoring Two pretreatment gross gamma radiation monitors mounted in parallel are located downstream of the dryers just upstream of the charcoal adsorber vessels. Their function is to monitor the activity level of the process stream and to alarm when an abnormal condition exists. When this occurs, the system outlet valve automatically closes, thereby terminating discharge to the main stack. Process and effluent radiological monitoring systems are further described in Section 11.5.

11.3.2.2 Ventilation Systems

The following ventilation systems potentially contain radioactive gas.

1. Reactor building HVAC/containment purge (Section 9.4.2).

2. Turbine building HVAC system (Section 9.4.4).

3. Radwaste building HVAC system (Section 9.4.3).

The expected concentrations of airborne radioactivity at various in-plant areas are calculated based on data given in NUREG-0016 and EPRI-495. The concentrations in the reactor building, both for normal operation and anticipated operational occurrences,

e.g., refueling, are presented in Table 12.2-15. Airborne radioactivity concentrations in various turbine and radwaste building areas for full-power operation are also provided in Table 12.2-15.

11.3.2.3 Steam and Power Conversion Systems

During startup of the unit, air is removed from the main condenser by a mechanical vacuum pump. In this mode of operation, when little or no radioactive gas is present in the condenser, vacuum pump discharge is directed to the main stack. During normal operation, the SJAEs provide sufficient force for condenser air removal. This system is further discussed in Section 10.4.2. The turbine gland sealing system utilizes steam to pressurize the turbine seals. This prevents radioactive steam from escaping through the seals to atmosphere. This system is discussed further in Section 10.4.3.

11.3.2.4 Miscellaneous Gaseous Releases

Exposure of the drywell air to neutron leakage fluxes around the reactor vessel results in some activation products. Activity may also be introduced into the drywell atmosphere by steam leaks and by venting of the primary system safety valves into the suppression chamber. Except for leakage of 2 scfm, the drywell essentially forms a closed system. When access is required, the drywell may be purged with normal reactor building air. The drywell can also be vented during startup to accommodate the expansion of air as its temperature increases. The discharge of this air is to the main stack by way of the SGTS (Section 6.5.1).

11.3.3 Radioactive Releases

11.3.3.1 Release Points

The plant airborne radioactive releases to the environs are from two monitored points, the main stack and the combined radwaste/reactor building vent. Figure 1.2-2 shows the release points for each effluent source.

The main stack is 430 ft above grade, which is 2.5 times the height of the reactor building, the tallest structure in the power block. The main stack releases exhaust air from the following plant areas and systems:

1. Turbine building.
2. Containment purge.
3. Turbine generator gland seal and exhaust steam system.
4. OFG system.

5. Mechanical vacuum pump discharge (intermittent release).

6. SGTS.

7. CSTs and sumps.

The combined radwaste/reactor building vent is located 186 ft above ground level and releases exhaust air from the radwaste building equipment and area exhaust, the auxiliary boiler building area exhaust, and the reactor building ventilation exhaust, above and below the refueling floor. Figure 11.3-1 shows the P&ID of the radioactive gaseous waste system.

11.3.3.2 Design and Expected Releases

The parameters used to calculate the design and expected releases are presented in Table 11.3-2. Expected ventilation exhaust, offgas, mechanical vacuum pump, and containment purge releases are presented in Table 11.3-1. The design annual average release concentrations are shown in Table 11.3-3 as activity in uCi/cc, and as a fraction of maximum permissible concentration.

11.3.3.3 Dilution Factors

The atmospheric dilution factor associated with normal plant releases is based on the average annual meteorological conditions applicable to the site as well as the effective release height of the effluent discharge pathway. The site meteorological conditions are given in Section 2.3.

11.3.3.4 Estimated Doses

A summary of the estimated annual radiation doses is presented in Appendix 11A and shows that the estimated annual doses from gaseous effluents are below the dose criteria set forth in 10CFR50 Appendix I and are well below the dose criteria specified in 40CFR190 and 10CFR20.

The maximum hypothetical gamma and beta air doses from noble gas releases occur at the exclusion area boundary (EAB), 1,603 m east of the site. The doses at this location are estimated to be 7.9E-02 mrad/yr gamma and 5.3E-02 mrad/yr beta, as compared with the 10CFR50 Appendix I design objective for gamma and beta air doses of 10.0 mrad/yr and 20.0 mrad/yr, respectively.

TABLE 11.3-1

EXPECTED RADIOACTIVE GASEOUS EFFLUENT
FROM ALL SOURCES (CI/YR)

Isotope	Reactor Bldg	Turbine Bldg	Radwaste Bldg	Mech. Vacuum Pump	Offgas System
I-131	4.4E-02	3.0E-01	2.0E-02	1.1E-01	-
I-132	3.6E-01	2.5E+00	1.6E-01	9.4E-01	-
I-133	2.9E-01	2.0E+00	1.3E-01	7.5E-01	-
I-134	5.8E-01	4.2E+00	2.6E-01	1.6E+00	-
I-135	4.0E-01	2.8E+00	1.8E-01	1.0E-00	-
Kr-83m	-	-	-	-	2.7E+00
Kr-85m	3.0+00	2.5+01	-	-	1.3E+03
Kr-85	-	-	-	-	2.7E+02
Kr-87	2.1+00	6.1+01	-	-	2.2E-01
Kr-88	3.8+00	9.1+01	-	-	4.6E+02
Kr-89	2.4+00	5.8+02	2.9+01	-	-
Xe-131m	-	-	-	-	7.5E+01
Xe-133m	-	-	-	-	9.0E+00
Xe-133	1.1+02	1.5+02	2.2+02	1.3+03	8.3E+03
Xe-135m	6.1+01	4.0+02	5.3+02	-	-
Xe-135	1.3+02	3.3+02	2.8+02	5.0+02	4.2E-10
Xe-137	1.9+02	1.0+03	8.3+01	-	-
Xe-138	7.3+00	1.0+03	2.0+00	-	-
Cr-51	9.7-04	9.0-04	7.0-06	-	-
Mn-54	1.3-03	6.0-04	4.0-05	-	-
Fe-59	3.5-04	1.0-04	3.0-06	-	-
Co-58	2.9-04	1.0-03	2.0-06	-	-
Co-60	5.1-03	1.0-03	7.0-05	-	-
Zn-65	4.1-03	6.0-03	3.0-06	-	-
Sr-89	1.7-04	6.0-03	-	-	-
Sr-90	7.9-06	2.0-05	-	-	-
Zr-95	6.2-04	4.0-05	8.0-06	-	-
Nb-95	1.1-02	6.0-06	4.0-08	--	
Mo-99	6.3-02	2.0-03	3.0-08	-	-
Ru-103	5.9-04	5.0-05	1.0-08	-	-
Ag-110m	2.6-06	-	-	-	-
Sb-124	2.1-05	1.0-04	7.0-07	-	-
Cs-134	4.3-03	2.0-04	2.4-05	-	-
Cs-136	4.9-04	1.0-04	-	-	-
Cs-137	5.7-03	1.0-03	4.0-05	-	-
Ba-140	1.4-02	1.0-02	4.0-08	-	-
Ce-141	7.9-04	1.0-02	7.0-08	-	-
Ar-41	1.5+01	-	-	-	1.06E+02
H-3	3.1E+01	3.1E+01	-	-	-
C-14	-	-	-	-	7.8E+01

NOTE: $3.0+00 = 3.0 \times 10^{0} = 3.0E00$

TABLE 11.3-2

DATA USED IN CALCULATING ANNUAL RELEASES
OF RADIOACTIVE GASEOUS EFFLUENTS

Parameter		Data
Rated thermal power		3,988 (MWt)
Rated steam flow rate		17,633,000 lb/hr
Offgas charcoal bed holdup* times per NUREG-0016	(Kr)	25.4 hr
	(Xe)	18.67 days
Plant capacity factor		95%
Design releases source term failed fuel basis		0.3536 Ci/sec (at 30 min)
Offgas system charcoal mass/train		48,000 lb
Dynamic adsorption coefficients	(Kr)	25 cm^3/gm
	(Xe)	440 cm^3/gm
Charcoal delay system normal operating temperature		70°F
Charcoal delay system dew point temperature		50°F
Ventilation systems		See Section 9.4
Decontamination factors		
Normal operations		
Radwaste building		99% efficient HEPA filter
All other buildings/systems		Unfiltered
Shutdown		
Radwaste building		99% efficient HEPA filter

TABLE 11.3-2 (Cont'd.)

Parameter	Data
Reactor building	10% of effluents - unfiltered 90% of effluents - 90% efficient iodine filter 99% efficient HEPA filter
All other buildings/systems	Unfiltered

* The offgas system holdup times presented are based on NUREG-0016 calculation methods. The holdup time values in this table are used only to calculate annual radioactive gaseous effluent releases from Unit 2. The offgas system holdup times discussed in Section 11.3.2.1 are based on design system parameters.

TABLE 11.3-3
DESIGN ANNUAL AVERAGE GASEOUS RELEASES VS. MPC

Isotope	Main Stack Releases				Radwaste/Reactor Building Vent Releases		MPC* (uCi/cc)	Total Release (fraction of MPC)
	Continuous Annual Average		Intermittent (MVP)		Annual Average			
	Activity at EAB (uCi/cc)	Fraction of MPC	Activity at EAB (uCi/cc)	Fraction of MPC	Activity at EAB (uCi/cc)	Fraction of MPC		
I-131	3.1E-16	3.1E-06	7.2-16	7.2E-06	2.0E-15	2.0E-05	1.0E-10	1.9E-05
I-132	4.1E-15	1.4E-06	9.1-15	3.0E-06	2.1E-14	7.0E-06	3.0E-09	8.2E-06
I-133	3.2E-15	8.0E-06	7.2-15	1.8E-05	1.6E-14	3.9E-05	4.0E-10	4.7E-05
I-134	9.7E-15	1.6E-06	2.1-14	3.5E-06	3.9E-14	6.4E-06	6.0E-09	8.0E-06
I-135	3.5E-15	3.5E-06	7.7-15	7.7E-06	1.7E-14	1.7E-05	1.0E-09	2.0E-05
Kr-83m	1.4E-15	4.6E-08	-	-	-	-	3.0E-08	4.6E-08
Kr-85m	1.7E-12	1.7E-05	-	-	1.0E-13	1.0E-06	1.0E-07	1.8E-05
Kr-85	4.9E-13	1.6E-06	-	-	-	-	3.0E-07	1.6E-06
Kr-87	1.3E-13	6.3E-06	-	-	7.6E-14	3.8E-06	2.0E-08	1.0E-05
Kr-88	6.2E-13	3.1E-05	-	-	1.4E-13	6.8E-06	2.0E-08	3.8E-05
Kr-89	1.2E-12	4.0E-05	-	-	1.1E-12	3.7E-05	3.0E-08	7.7E-05
Xe-131m	1.1E-13	2.8E-07	-	-	-	-	4.0E-07	2.8E-07
Xe-133m	8.4E-15	2.8E-08	-	-	-	-	3.0E-07	2.8E-08
Xe-133	1.4E-11	4.5E-05	1.6-11	5.3E-05	1.2E-11	4.0E-05	3.0E-07	8.5E-05
Xe-135m	8.3E-13	2.8E-05	-	-	2.1E-11	7.1E-04	3.0E-08	7.4E-04
Xe-135	6.7E-13	6.7E-06	6.0-12	6.0E-05	1.5E-11	1.5E-04	1.0E-07	1.6E-04
Xe-137	2.1E-12	7.0E-05	-	-	9.8E-12	3.3E-04	3.0E-08	4.0E-04
Xe-138	2.0E-12	6.8E-05	-	-	3.3E-13	1.1E-05	3.0E-08	7.9E-05
Cr-51	2.6E-19	3.3E-12	-	-	5.0E-18	6.2E-11	8.0E-08	6.6E-11
Mn-54	2.7E-19	2.7E-10	-	-	1.1E-17	1.1E-08	1.0E-09	1.1E-08
Fe-59	2.1E-19	1.0E-10	-	-	1.3E-17	6.6E-09	2.0E-09	6.7E-09
Co-58	2.0E-17	1.0E-08	-	-	1.0E-16	5.1E-08	2.0E-09	6.1E-08
Co-60	1.0E-18	3.4E-09	-	-	9.0E-17	3.0E-07	3.0E-10	3.1E-07
Zn-65	1.7E-18	8.7E-10	-	-	2.1E-17	1.0E-08	2.0E-09	1.1E-08
Sr-89	5.4E-17	1.8E-07	-	-	2.7E-17	8.9E-08	3.0E-10	2.7E-07
Sr-90	1.8E-19	6.1E-09	-	-	1.2E-18	4.2E-08	3.0E-11	4.8E-08
Zr-95	8.6E-20	8.6E-11	-	-	2.3E-17	2.3E-08	1.0E-09	2.3E-08
Nb-95	1.6E-20	5.5E-12	-	-	4.0E-16	1.3E-07	3.0E-09	1.3E-07
Mo-99	6.5E-18	9.3E-10	-	-	3.6E-15	5.1E-07	7.0E-09	5.1E-07
Ru-103	1.1E-19	3.5E-11	-	-	2.1E-17	7.1E-09	3.0E-09	7.1E-09
Ag-110m	3.1E-22	1.0E-12	-	-	2.2E-18	7.4E-09	3.0E-10	7.4E-09
Sb-124	2.0E-19	2.9E-10	-	-	7.9E-19	1.1E-09	7.0E-10	1.4E-09
Cs-134	4.2E-19	1.0E-09	-	-	1.6E-16	3.9E-07	4.0E-10	3.9E-07
Cs-136	2.1E-19	3.5E-11	-	-	1.7E-17	2.9E-09	6.0E-09	2.9E-09
Cs-137	2.1E-18	4.2E-09	-	-	2.0E-16	4.1E-07	5.0E-10	4.1E-07
Ba-140	6.2E-17	6.2E-08	-	-	1.5E-15	1.5E-06	1.0E-09	1.6E-06
Ce-141	2.1E-17	4.1E-09	-	-	2.8E-17	5.6E-09	5.0E-09	9.8E-09
Ar-41	2.5E-14	6.3E-07	-	-	-	-	4.0E-08	6.3E-07
H-3	7.9E-15	3.9E-08	-	-	1.4E-13	6.8E-07	2.0E-07	7.2E-07
C-14	1.9E-14	1.9E-07	-	-	-	-	1.0E-07	1.9E-07
Total		3.3E-04		1.5E-04		1.4E-03		1.7E-03

* In accordance with 10CFR20 Appendix B.

11.4 SOLID WASTE MANAGEMENT SYSTEM

Power plant operation results in various solid radioactive wastes that require disposal. The radioactive solid waste system is designed to collect, hold, monitor, process, package, and provide temporary storage facilities for radioactive materials prior to shipment offsite and ultimate disposal. The solid waste management system is shown on Figure 11.4-1.

11.4.1 Design Basis

The radioactive solid waste system is designed to the following criteria:

1. The system provides for the dewatering/solidification and packaging of wet solid wastes into shipping containers prior to shipment for offsite disposal.

2. All solid waste containers, shipping casks, and methods of packaging meet applicable state and federal regulations. Wastes will be shipped to a licensed burial site in accordance with applicable NRC and Department of Transportation (DOT) regulations (i.e., 10CFR71 and 49CFR171-178).

3. The filling of containers, dewatering/solidification, and storage of radioactive solid waste conforms to 10CFR20 and 10CFR50 requirements and RG 8.8 guidelines in terms of ALARA doses to plant personnel and the general public.

4. Remote automatic and/or manual operation is provided by the radwaste dewatering system control panel and waste handling control panel.

5. System reliability is emphasized through proven design of system components, compartmentalization of equipment layout, shielding, containment of possible spills, accurate process monitoring, and interlocking of process controls.

6. The seismic criteria and analytical procedures for structures housing the solid radwaste system are given in Section 3.2.1 and 3.7. The quality group classification for the system components and piping is given in Table 3.2-1.

11.4.2 System Inputs

Radioactive solid wastes that result from plant operation consist of concentrated liquid wastes from the evaporators in the radioactive LWS, spent resins from all plant demineralizers handling radioactive liquids, filter sludges from LWS filters, phase separators, and miscellaneous solid materials that become

contaminated during plant operation and maintenance. Table 11.4-1 is a conservatively high estimate of the expected volumes of both the wet and dry radioactive solid waste generated by the unit.

11.4.2.1 System Inputs Activity

Expected and design wet solid waste activities are given in Table 11.4-2. The prediction of solid radwaste principal nuclide curie inventories was obtained from a mathematical model of expected and design values of reactor coolant and main steam radionuclide concentrations, as discussed in Section 11.1 and found in Table 11.4-3.

Expected and maximum dry solid radwaste inventories and annual curie content for both compactible and noncompactible wastes are provided in Tables 11.4-5 and 11.4-6. The values in these tables are based on data on radionuclides present in dry solid radwaste from operating plants[1], and are calculated for Unit 2 using the parameters in Table 11.4-7.

11.4.3 System Description

The original plant radioactive waste solidification system (WSS) consisted of an asphaltic-based solidification process described in Topical Reports WPC-VRS-001[2] and WPC-VRS-002[3]. This system primarily consisted of a waste fill station, a monitor and capping station, an extruder/evaporator, control console, and piping, metering pumps, and process equipment required for transfer and solidification of wastes. The asphaltic-based solidification process is no longer used and most of the associated process equipment has permanently been abandoned in place. Table 11.4-4 lists the major equipment of the originally supplied system and identifies the equipment that has been abandoned in place.

In place of the asphaltic-based solidification process, a radwaste dewatering system described in Section 11.4.3.3.1 provides the primary method of waste volume reduction.

The spent resin and filter sludge handling system consists of a waste sludge tank with agitator, redundant transfer pumps, and a decant pump. The system provides for holdup, recirculation, and sampling of waste, and decanting of excess water prior to transfer to the solidification portion of the system.

The evaporator bottoms handling system includes a concentrated waste transfer pump. In conjunction with the LWS evaporator bottoms tank, this system provides for holdup, recirculation, and sampling of evaporator concentrates prior to transfer to dewatering/solidification and phase separation of condensate filtration system (CFS) filter backwash water. Originally, all lines and components in the waste concentrate system were heat traced (to prevent crystallization of the waste salts in the

system). Operating experience and current waste handling practices determined that waste salts are no longer a concern. Thus, some heat trace circuits have been removed or abandoned in place. In addition, waste concentrate lines are equipped with relief valves, where required, that are discharged to the building floor drains.

The various plant systems that interface with the waste dewatering/solidification system are shown on Figures 11.4-1a to 11.4-1h and are as follows:

1. LWS provides the waste feed streams.

2. Radwaste auxiliary steam (ASR) provides hot flush water.

3. Steam released from relief valves is discharged to the auxiliary boiler steam (ABM) relief header.

4. Makeup water system (MWS) supplies boiler feedwater to the WSS boiler.

5. Condensate makeup and drawoff water (CNS) is used for equipment flushing.

6. Instrument air system (IAS) supplies air to all instrument and air-operated devices.

7. Service air system (SAS) provides air for clearing lines via hose connections.

8. All equipment is provided with drains to the radwaste building floor and equipment drains system (DFW). All liquid effluent from the WSS is returned to the liquid waste system.

9. Exhaust air from the equipment is vented to the radwaste building ventilation system (HVW).

10. Turbine building closed loop cooling water (CCS) provides component cooling.

11. Radwaste sampling system (WSS) provides process sampling.

12. Radwaste seal water system (SWR) provides seal water for pumps with double mechanical seals. Details are given in Section 11.2.

A compactor is provided to compress dry wastes such as paper, rags, and plastic for packaging in metal boxes. Incompressible solid wastes are packaged in metal boxes, 55-gal drums, or encapsulated in liners ranging in size from 50 to 200 cu ft.

11.4.3.1 Spent Resin/Filter Sludge Packaging

The waste sludge tank receives spent resin/filter sludge from the LWS spent resin tank and radwaste filter backwash tank. Excess water may be removed from the sludge tank by the decant pump through retention screens to prevent resin carryover.

Contents of the waste sludge tank are fed by one of two redundant waste sludge pumps to the radwaste truck bay for treatment by the radwaste dewatering system or cement solidification by an approved vendor.

11.4.3.2 Evaporator Bottoms Packaging

Evaporator bottoms are processed from the LWS evaporator bottoms tank. Contents are transferred to an approved container in the radwaste truck bay. The waste is then processed to final form in the truck bay or shipped offsite for vendor processing to final form.

11.4.3.3 Radwaste Backup System

The radwaste backup system provides an alternate method for the solidification of spent resin/filter sludge and evaporator bottoms by a mobile, skid-mounted solidification system which would be temporarily located in the radwaste building truck bay. In the event that solidification is required, three pipelines will bypass the extruder and go directly to a hose station in the truck bay area. These three pipelines consist of:

1. A 1 1/2-in diameter, electrically-traced line that carries evaporator bottoms waste directly from the evaporator bottoms transfer pump.

2. A 1 1/2-in diameter line that carries spent bead resins and filter sludge directly from the waste sludge tank pumps.

3. A 2-in diameter line that carries spent powdered resin from the influent header feed line of the spent resin tank.

Flexible hoses will be used to transfer the spent waste from the hose station to a mobile solidification unit.

The truck bay area is classified as Radiation Zone I/IV, restricted during the radwaste loading operation.

This backup facility consists entirely of transfer piping and valves. It is part of and is designed and constructed to the same criteria as the LWS and WSS systems with which it interfaces. All transfer operations are manually controlled from a control panel located in the truck dock area.

11.4.3.3.1 Radwaste Dewatering System

The radwaste dewatering system provides an acceptable method of waste volume reduction by a self-contained, freestanding, portable dewatering system located in the radwaste building truck bay, el 261'. This unit consists of a dewatering skid, a plant connection stand, a control module, a container fillhead, and a waste container and associated interconnecting hoses and cables. Plant services required to properly operate the dewatering system include service air, service water, electrical power, and waste as shown on Figure 11.4-1. The design, operation, and safety evaluation of this dewatering system is described in Chem Nuclear Systems, Inc., Topical Report RDS-25506-01-P/NP[4], which has been reviewed and accepted by Unit 2 management and the NRC as applicable.

11.4.3.4 Dry Waste Packaging

Dry activated waste (DAW) is collected at various locations in the plant and transported to the radwaste building. The DAW, such as paper and rags, is packaged in approved shipping containers prior to shipment to a waste disposal facility, or offsite waste processor. The packaging and shipment is done in accordance with approved Station procedures.

11.4.3.5 Incompressible Waste Packaging

Components of low activity, such as contaminated tools, can be packaged in available or specially-designed shipping containers and stored, if necessary, in the radwaste building prior to shipment.

Spent core components with very high activity levels are handled underwater within the reactor refueling cavity and fuel transfer canal, and stored in the fuel pool until adequate packaging is provided for offsite shipment. Refer to Section 9.1 for additional information on spent fuel handling.

11.4.3.6 Waste Packaging Controls

Complete solidification of the processed waste is ensured by preoperational testing and the implementation of a process control program. Waste sludge and evaporator bottoms tanks are provided with means for obtaining representative samples via a shielded sampling station located in the radwaste building. Waste packaging controls and monitoring are provided from the solidification system control panel. The system is designed to prevent external contamination of the containers by instrumentation interlocks that prevent overfilling.

Specific requirements regarding radioactive waste solidification are described in TRM Section 3.11.1.

11.4.3.7 Waste Handling

Waste handling is provided by a 30-ton, overhead, traveling bridge crane with an 8 1/2-ton auxiliary hoist. Remote handling of processed waste is done by closed-circuit TV monitoring from the waste handling control panel.

11.4.4 Packaging

Filling of containers and storage of radioactive solid wastes conforms with 10CFR20 and 10CFR50 requirements. Packages meet shipping regulations of 49CFR171-178 and 10CFR71, as applicable. The waste packaging procedure is described in Section 11.4.3. Containers used are 50- to 200-cu ft liners, 55-gal drums, and metal boxes with supplementary lead or steel shielding as required for shipment.

11.4.5 Storage Facilities

The radwaste building was designed to process and store Unit 2 low-level radioactive waste (LLRW). El 245 ft of the radwaste building has been modified with the specific intent of providing interim onsite storage of LLRW (primarily compacted DAW and incinerator ash), including storage from both Unit 1 and Unit 2. The need to store LLRW onsite is the result of the federal Low-Level Radioactive Waste Policy Act as amended in 1985, which initiated the process by which the three existing LLRW disposal sites (Barnwell, SC; Beatty, NV; and Hanford, WA) would no longer be required to receive LLRW. El 245 ft of the radwaste building is capable of providing at least 5 yr of interim storage of low activity LLRW produced at both Unit 1 and Unit 2. The storage of Unit 1 LLRW at Unit 2 is considered acceptable based on the following:

1. The isotopic library to be considered is essentially the same for both units;

2. The isotopic distributions for the two units are similar and release pathways will not be affected. The higher Co-60 weighting factor for Unit 1 waste will have no appreciable effect on shielding, release analyses or handling;

3. The material to be stored at Unit 2 is of very low activity level; and

4. The transfer of by-product material between Unit 1 and Unit 2 will be conducted in accordance with approved radiation protection implementing procedures.

The Unit 1 radwaste solidification and storage building (RSSB) provides interim storage of higher activity waste generated at Unit 1 and Unit 2.

Dewatered/solidified waste is stored in a shielded area within the radwaste building in liners ranging in size from 50 to 200 cu ft. The storage capacity of processed waste is approximately a 4-month output of packaged waste at expected generation rates.

11.4.6 Shipment

Shipment of radioactive solid wastes conforms with 10CFR50, 10CFR61, 10CFR71, and 49CFR171 through 49CFR178 requirements. Higher-activity wastes are shipped in shielded casks, as applicable.

Solid waste is transported by a licensed disposal contractor or a common carrier to a licensed burial site.

Tables 11.4-1 and 11.4-2 summarize the annual number of packaged containers, expected number of shipments to be made, and expected and design activities of the wastes.

11.4.7 Process Control Program

The Process Control Program (PCP) was approved by the NRC before implementation. It contains the current formula sampling, analyses, tests, and determinations to be made to ensure that the processing and packaging of radioactive wastes, based on demonstrated processing of actual or simulated wet or liquid wastes, will be accomplished in such a way as to assure compliance with 10CFR20, 10CFR61, 10CFR71, and Federal and State regulations and other requirements governing the transport and disposal of radioactive waste.

Licensee-initiated changes to the PCP shall become effective upon review and acceptance by the PORC and shall be submitted to the NRC in the Semiannual Radioactive Effluent Release Report for the period in which the change(s) was made. This submittal shall contain:

1. Sufficiently detailed information to totally support the rationale for the change without benefit of additional or supplemental information;

2. A determination that the change did not reduce the overall conformance of the solidified waste product to existing criteria for solid wastes; and

3. Documentation of the fact that the change has been reviewed and found acceptable by the PORC.

11.4.8 References

1. Phillips, J., Feizollahi, F., Martineit, R., and Bell, W. Waste Report for Reactor and Fuel-Fabrication Facility Wastes, ONWI-20/NUS-3314, NUS Corporation, March 1979.

2. Radwaste Volume Reduction and Solidification System-Topical Report, Report Number WPC-VRS-001, Revision 1, Werner and Pfleiderer Corporation, May 1978.

3. Topical Report, 10CFR61, Waste Form Conformance Program for Solidified Process Waste Products by a Wastechem Corporation Volume Reduction and Solidification (VRS) System, Report No. VRS-002, Revision 1, August 1987.

4. Topical Report, RDS-1000 Radioactive Waste Dewatering System, Report No. RDS-25506-01-P/NP, Revision 1, Chem Nuclear Systems, Inc.

TABLE 11.4-1

ANNUAL WET AND DRY SOLID WASTE QUANTITIES VOLUME

Source of Wet Wastes	Expected				Design			
	Unsolidified Waste Volume (ft^3)	50 ft^3 Containers	Packaged Volume[3] (ft^3)	Shipments	Unsolidified Waste Volume (ft^3)	50 ft^3 Containers	Packaged Volume[3] (ft^3)	Shipments
Spent resins (radwaste demineralizer, condensate demineralizer)	1,210	31	1,628	19	4,400	113	5,933	68
Filter sludges	4,840	124	6,510	74	9,064	232	12,180	140
Evaporator bottoms								
1. Radwaste	2,487	8	420	35[1]	4,504	14	735	64[1]
2. Regenerative	5,653	39	2,048		10,236	71	3,728	
Subtotal	14,190	202	10,606	128	28,204	430	22,576	272
Source of Dry Wastes	Metal Boxes[2]	55-Gal Drums[4]	Packaged Volume (ft^3)	Shipments	Metal Boxes[2]	55-Gal Drums[4]	Packaged Volume (ft^3)	Shipments
Compacted	-	1,228	9,210	18	-	2,052	15,390	29
Miscellaneous	8	-	1,024	1	13	-	1,664	2
Subtotal	8	1,228	10,234	19	13	2,052	17,054	31
Total			20,840	147			39,630	303

(1) Combined shipments of radwaste and regenerative evaporator bottoms.
(2) Metal boxes contain 128 ft^3 volume.
(3) Packaged volume based on 52.5 ft^3 of external volume.
(4) Packaged volume based on 7.5 ft^3.

TABLE 11.4-2

EXPECTED AND DESIGN WET SOLID WASTE ACTIVITIES

Source of Wet Wastes	Expected		Total Activity (Ci/yr)	Design		Total Activity (ci/yr)
	uCi/cc	Ci/ft^3		uCi/cc	Ci/ft^3	
Spent resin (radwaste demineralizer, condensate demineralizer)	58.62	1.66	$2.01x10^3$	497.96	14.10	$6.20x10^4$
Filter sludges						
Radwaste filter	1.58	0.05	43.42	2.60	0.07	133.47
Evaporator bottoms (radwaste and regenerative)	35.50	1.00	$8.18x10^3$	293.02	8.30	$1.22x10^5$

TABLE 11.4-3

WET SOLID RADWASTE PRINCIPAL NUCLIDE INVENTORIES

Isotope	Activity (uCi/cc)
Radwaste Filter Sludge	
Na-24	0.202+00
P-32	0.725-02
Cr-51	0.217+00
Mn-54	0.334-02
Mn-56	0.411-03
Fe-55	0.374-01
Fe-59	0.759-02
Co-58	0.477+00
Co-60	0.479-01
Ni-65	0.266-02
Cu-64	0.536+00
Zn-65	0.750-02
Zn-69m	0.382-01
Ag-110m	0.576-02
Ag-110	0.115-03
W-187	0.188+00
Nb-98	0.398-02
Waste and Regenerant Evaporator Bottoms	
Co-60	0.655+00
Ni-65	0.459-05
Cu-64	0.365-01
Zn-65	0.951-01
Zn-69m	0.293-02
Ag-110m	0.733-01
Ag-110	0.147-02
W-187	0.362-01
Np-239	0.123+02
Nb-98	0.316-05
Tc-104	0.409-06
Ba-139	0.237-03
Ba-140	0.341+01
Ba-141	0.234-06
Ba-142	0.542-08
La-140	0.362+01
La-141	0.105-02
La-142	0.382-04
Ce-141	0.112+00
Ce-143	0.183-02
Ce-144	0.430-01

TABLE 11.4-3 (Cont'd.)

Isotope	Activity (uCi/cc)
Pr-143	0.478-01
Pr-144	0.430-01
Nd-147	0.461-02
Pm-147	0.157-03
Na-24	0.182-01
P-32	0.329-01
Cr-51	0.157+01
Mn-54	0.494-01
Mn-56	0.807-03
Fe-55	0.504+00
Fe-59	0.692-01
Co-58	0.502+01
Nb-97m	0.175-03
Nb-97	0.197-03
Mo-99	0.141+01
Tc-99m	0.143+01
Tc-101	0.102-06
Ru-103	0.444-01
Ru-105	0.721-03
Ru-106	0.105-01
Rh-103m	0.444-01
Rh-105m	0.723-03
Rh-105	0.631-04
Rh-106	0.105-01
Te-129m	0.649-01
Te-129	0.651-01
Te-131m	0.514-02
Te-131	0.104-02
Te-132	0.385+01
Cs-134	0.206+00
Cs-136	0.447-01
Cs-137	0.317+00
Cs-138	0.528-05
Ba-137m	0.292+00
Br-83	0.415-02
Br-84	0.141-04
I-129	0.452-12
I-131	0.620+02
I-132	0.400+01
I-133	0.236+02
I-134	0.366-01
I-135	0.125+01
Rb-89	0.933-08
Sr-89	0.281+01
Sr-90	0.306+00
Sr-91	0.980-01

TABLE 11.4-3 (Cont'd.)

Isotope	Activity (uCi/cc)
Sr-92	0.206-02
Y-90	0.290+00
Y-91m	0.644-01
Y-91	0.556+00
Y-92	0.124-01
Y-93	0.195-01
Zr-95	0.392-01
Zr-97	0.182-03
Nb-95m	0.753-03
Nb-95	0.503-01
Reactor Water Cleanup Sludge*	
Co-58	0.503+02
Co-60	0.750+01
Ni-65	0.786-03
Cu-64	0.352+00
Zn-65	0.105+01
Zn-69m	0.420-01
Ag-110m	0.809+00
Ag-110	0.162-01
W-187	0.196+00
Np-239	0.560+02
Nb-98	0.472-02
Tc-104	0.123-01
Ba-137m	0.335+01
Ba-139	0.133+00
Ba-140	0.237+02
Ba-141	0.653-02
Ba-142	0.214-02
La-140	0.258+02
La-141	0.773-01
La-142	0.174-01
Ce-141	0.984+00
Ce-143	0.866-02
Ce-144	0.479+00
Pr-143	0.337+00
Pr-144	0.479+00
Nd-147	0.309-01
Pm-147	0.205-02
F-18	0.741-01
Na-24	0.146+00
P-32	0.235+00
Cr-51	0.133+02
Mn-54	0.553+00
Mn-56	0.134+00

TABLE 11.4-3 (Cont'd.)

Isotope	Activity (uCi/cc)
Fe-55	0.576+01
Fe-59	0.644+00
Nb-95	0.290+00
Nb-97m	0.126-02
Nb-97	0.139-02
Mo-99	0.660-01
Tc-99m	0.300+01
Tc-101	0.863-02
Ru-103	0.405+00
Ru-105	0.410-01
Ru-106	0.118+00
Rh-103m	0.405+00
Rh-105m	0.411-01
Rh-105	0.232-03
Rh-106	0.118+00
Te-129m	0.741+00
Te-129	0.742+00
Te-131m	0.252-01
Te-131	0.510-02
Te-132	0.187+02
Cs-134	0.132+01
Cs-136	0.613-02
Cs-137	0.364+01
Cs-138	0.567-05
Br-83	0.541-01
Br-84	0.347-02
Br-85	0.126-04
I-129	0.156-10
I-131	0.191+02
I-132	0.196+02
I-133	0.868+01
I-134	0.127+00
I-135	0.207+01
Rb-89	0.578-03
Sr-89	0.267+02
Sr-90	0.349+01
Sr-91	0.139+01
Sr-92	0.317+00
Y-90	0.343+01
Y-91m	0.898+00
Y-91	0.348+02
Y-92	0.601+00
Y-93	0.213-02
Y-94	0.530-10
Zr-97	0.131-02

TABLE 11.4-3 (Cont'd.)

Isotope	Activity (uCi/cc)
Floor Drain Filter Sludge	
Na-24	0.677+00
P-32	0.263-01
Cr-51	0.786+00
Mn-54	0.139-01
Mn-56	0.930-03
Fe-55	0.136+00
Fe-59	0.275-01
Co-58	0.173+01
Co-60	0.174+00
N-165	0.338-02
Cu-64	0.177+01
Zn-65	0.271-01
Zn-69m	0.127+00
Ag-110m	0.209-01
Ag-110	0.418-03
W-187	0.648+00
Nb-98	0.783-02

* Buildup activity in the spent resin tank corresponding to RWCU F/D backwashes is higher than that corresponding to the spent resin from the condensate demineralizer. Also, this activity is either equal to or less than the SFC F/D sludge.

TABLE 11.4-4

SOLID WASTE MANAGEMENT SYSTEM MAJOR EQUIPMENT LIST

Component	Parameter
Waste sludge tank 2WSS-TK8	
Number	1
Capacity, gal	1,355
Material of construction	Type 316L stainless steel
*Asphalt storage tank 2WSS-TK2	
Number	1
Capacity, gal	10,800
Material of construction	Carbon steel
*Extruder/evaporator 2WSS-EV25	
Number	1
Capacity, gpm	Varies (~1.0)
Material of construction	Mfg standard
*Asphalt metering pump 2WSS-P5A&B	
Number	2
Capacity, gpm	0.3
Material of construction	Cast iron
*Asphalt recirc pump 2WSS-P3A&B	
Number	2
Capacity, gpm	20
Material of construction	Cast iron
Waste sludge transfer pump 2WSS-P50A&B	
Number	2
Capacity, gpm	50
Material of construction	High chrome iron
*Waste sludge metering pump 2WSS-P12A&B	
Number	2
Capacity, gpm	0.2-0.8
Material of construction	Type 316 stainless steel
Decant pump 2WSS-P10	
Number	1
Capacity, gpm	40
Material of construction	Type 316 stainless steel
Waste concentrate transfer pump 2WSS-P6	
Number	1
Capacity, gpm	50
Material of construction	Alloy 20

TABLE 11.4-4 (Cont'd.)

Component	Parameter
*Waste concentrate metering pump 2WSS-P7A&B	
Number	2
Capacity, gpm	0.4-0.8
Material of construction	316TI stainless steel
*Steam dome boilout tank 2WSS-TK46	
Number	1
Capacity, gal	3
Material of construction	Type 304 stainless steel
Overhead crane 2MHN-CRN1	
Number	1
Capacity - main hoist	30 tons
auxiliary hoist	8-1/2 tons

* This equipment is no longer utilized and has permanently been abandoned in place.

TABLE 11.4-5

EXPECTED DRY SOLID RADWASTE ANNUAL NUCLIDE INVENTORIES

Isotope	Compactible (Ci)	Noncompactible (Ci)
Cr-51	5.3-01*	4.0+01
Mn-54	4.3-01	3.3+01
Fe-59	5.1-02	3.9+00
Co-58	1.4-01	1.1+01
Co-60	1.9+00	1.4+02
Zn-65	7.5-01	5.7+01
Zr-95	1.3-02	9.6-01
Nb-95	4.5-02	3.4+00
Cs-134	6.5-01	5.0+01
Cs-137	1.2+00	8.9+01
Total	5.6+00	4.3+02

* 5.3-01 is 5.3×10^{-1}

TABLE 11.4-6

MAXIMUM DRY SOLID RADWASTE ANNUAL NUCLIDE INVENTORIES

Isotope	Compactible (Ci)	Noncompactible (Ci)
Cr-51	8.9-01*	6.7+01
Mn-54	7.2-01	5.5+01
Fe-59	8.6-02	6.6+00
Co-58	2.4-01	1.8+01
Co-60	3.1+00	2.4+02
Zn-65	1.3+00	9.5+01
Zr-95	2.1-02	1.6+00
Nb-95	7.5-02	5.7+00
Cs-134	1.1+00	8.3+01
Cs-137	1.9+00	1.5+02
Total	9.4+00	7.2+02

* 8.9-01 is 8.9×10^{-1}

TABLE 11.4-7 (Historical)

PARAMETERS USED TO CALCULATE DRY SOLID RADWASTE INVENTORIES AND ANNUAL CURIE CONTENT

Average plant waste activity for BWRs (for trash)[1]	Compactible	5.20-03 Ci/MW(e)-yr
	Noncompactible	3.97-01 Ci/MW(e)-yr
Net (expected) electrical output for Unit 2		1,080 MWe
Volume of dry solid radwaste (design) for Unit 2		17,054 ft^3
Volume of dry solid radwaste (expected) for Unit 2		10,234 ft^3

[1] Phillips, J., Feizollahi, F., Martineit, R., and Bell, W. Waste Report for Reactor and Fuel-Fabrication Facility Wastes, ONWI-20/NUS-3314, NUS Corporation, March 1979 (Table 4.2-49 and pp 4-88).

11.5 PROCESS AND EFFLUENT RADIOLOGICAL MONITORING AND SAMPLING SYSTEMS

11.5.1 Design Bases

The process and effluent radiological monitoring and sampling systems are provided to allow determination of the content of radioactive material in various gaseous and liquid process and effluent streams. The design objective and criteria are primarily determined by the system function of either:

1. Monitors/sampling required for safety, or

2. Monitors/sampling required for plant operation.

11.5.1.1 Design Objectives

11.5.1.1.1 Radiation Monitors Required for Safety

The main objective of radiation monitoring systems (RMS) required for safety is to initiate appropriate manual or automatic protective action to limit the potential release of radioactive materials from the reactor vessel, primary and secondary containment, and fuel storage areas if predetermined radiation levels are exceeded in major process/effluent streams, and to provide main control room personnel with radiation level indication throughout the course of an accident. Additional objectives are to have these systems available under all operating conditions including accidents, and to provide main control room personnel with an indication of the radiation levels in the major process/effluent streams, including alarm annunciation if high radiation levels are detected.

Radiation monitors provided to meet these objectives are:

1. Main steam line (MSL).

2. Reactor building ventilation exhaust (above refueling floor).

3. Reactor building ventilation exhaust (below refueling floor).

4. Main control room air intakes.

5. RHR heat exchanger service water discharge.

11.5.1.1.2 Radiation Monitors Required for Plant Operation

The main objective of RMSs required for plant operation is to provide operating personnel with a measurement of the level of radioactivity in potentially-radioactive effluents and process streams. This allows demonstration of compliance with plant normal operational Technical Specifications and the Offsite Dose

Calculation Manual by providing gross radiation level monitoring and off-line isotopic analysis of gaseous effluents including halogens and particulates. Additional objectives are to initiate valve isolation on the offgas system, containment purge, and LWS if predetermined release rates are exceeded. Grab samples can be taken at all process and liquid effluent monitor and both gaseous effluent sample tap locations to allow laboratory determination of isotopic content. In addition to grab samples, provisions have been included for the offgas pretreatment process monitors to allow alternate sampling with local analysis capabilities (Figure 11.3-1b). Connection capability is provided at selected locations for continuous airborne monitors (CAM).

The radiation monitors provided to meet these objectives are:

1. For gaseous effluent streams:

    a. Combined radwaste/reactor building ventilation exhaust.

    b. Plant main stack exhaust.

2. For liquid effluent streams:

    a. Liquid radwaste effluent.

    b. Cooling tower blowdown line (circulating water system [CWS]).

    c. Service water discharge.

3. For gaseous process streams:

    a. Offgas pretreatment.

    b. Standby gas treatment discharge (monitors containment purge system exhaust).

    c. Turbine building ventilation exhaust (CAM only).

    d. Radwaste building area ventilation exhaust (CAM only).

    e. Radwaste building equipment exhaust (CAM only).

    f. Radwaste tank vent exhaust (CAM only).

    g. Radwaste equipment service area (CAM only).

    h. Drywell atmosphere monitoring (CAM).

4. For liquid process streams:

    a. Spent fuel pool cooling and cleanup (SFC).

b. Turbine building closed loop cooling water (TBCLCW).

c. Reactor building closed loop cooling water (RBCLCW).

11.5.1.2 Design Criteria

11.5.1.2.1 Monitors Required for Safety

The radiation monitors are designed to the following criteria:

1. The monitors are designed to Category I criteria to withstand the effects of natural phenomena (e.g., earthquakes) without loss of capability to perform their functions.

2. The monitors perform their intended safety function in the environment resulting from normal, abnormal, and postulated accident conditions.

3. The monitors meet the reliability, testability, independence, and failure mode requirements of engineered safety features (ESF).

4. The monitors provide continuous outputs in the main control room panel.

5. The monitors permit checking of the operational availability of each channel during reactor operation with provision for calibration function and source checks.

6. The monitors assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences.

7. The monitors initiate prompt protective action before exceeding plant Technical Specification and the Offsite Dose Calculation Manual limits.

8. The monitors provide warning of increasing radiation levels indicative of abnormal conditions by alarm annunciation.

9. The monitors provide annunciation to indicate power failure or component malfunction.

10. The monitors register full-scale output if radiation detection exceeds full scale.

11. The monitors have sensitivities and ranges compatible with anticipated radiation levels.

12. The monitors have safety-related power supplies.

11.5.1.2.2 Monitors Required for Plant Operation

The operations design criteria are that the monitors:

1. Provide continuous indication of selected radiation levels in the main control room.

2. Provide warning by alarm annunciation of increasing radiation levels indicative of abnormal conditions.

3. Provide annunciation to indicate power failure or component malfunction.

4. Monitor a sample representative of the bulk stream or volume.

5. Have provisions for calibration, function, and source checks.

6. Have sensitivities and ranges compatible with anticipated radiation levels, and Offsite Dose Calculation Manual limits.

7. Register full-scale output if radiation detection exceeds full scale.

The monitors that detect radioactivity in the discharges from the containment purge system (via SGTS), the liquid radwaste treatment system, and the offgas system pretreatment, have provisions to alarm and to initiate automatic closure of the discharge valve on the affected system prior to exceeding the limits specified in the Offsite Dose Calculation Manual.

11.5.2 System Description

The process and effluent RMS consists of a computer-based digital radiation monitoring system (DRMS), a microprocessor-based gaseous effluent monitoring system (GEMS), and nondigital monitors supplied as part of the reactor protection system (RPS).

11.5.2.1 Computer-Based Radiation Monitoring Systems

Digital Radiation Monitoring System

The function of the DRMS is to measure, evaluate, and report radioactivity in process streams, in liquid effluents, and in selected plant areas (Section 12.3.4), and to annunciate and/or initiate an automatic control function for abnormal system or plant operating conditions. The DRMS process and liquid effluent monitors are shown in Table 11.5-1, and the DRMS area monitors are shown in Table 12.3-1. Each monitoring channel has a

microprocessor located near the detector or sample panel. The DRMS computer system continuously polls the local microprocessors collecting and storing radiation levels, alarms, and status information for these monitoring channels. This information, stored in the computer, is available on demand for analysis of plant conditions, trending of radiation levels, and maintenance purposes.

All monitoring channels have two alarm states: alert radiation (alarm) and high radiation (trip). Alarms are annunciated locally at the detector and in the main control room.

Readouts for all DRMS monitors are provided locally at the monitor electronics panel and at consoles located in the radiation protection (health physics) room, the main control room, the emergency operations facility (EOF), the technical support center (TSC), and the computer room located on el 288'-6" of the control building. An additional control panel for the safety-related monitors also is located in the main control room.

Gaseous Effluent Monitoring System

The function of the GEMS is to measure, evaluate, and report radioactivity in gaseous effluents and to annunciate if release levels approach limits specified in the Offsite Dose Calculation Manual. The GEMS provides real time noble gas activity monitoring and continuous iodine and particulate sample collection for main stack and radwaste/reactor building vent releases. The information from the GEMS is manually combined with information from the meteorological computer (VAX 11/780) to generate the gaseous release calculations for RG 1.21 report generation.

GEMS readouts are provided in the main control room, the EOF, and the TSC.

11.5.2.1.1 Computer-Based Monitor Descriptions

Six basic types of computer-based monitoring or sampling systems are provided, as indicated in Table 11.5-1, for the process and effluent monitoring systems: off-line gas and particulate, off-line gas, off-line liquid, CAM, on-line liquid, and off-line gaseous effluent. Diagrams of these types of monitors are shown on Figures 11.5-1 through 11.5-6. Figures 11.5-7 and 11.5-8 show the gaseous and liquid RMS. Safety and operationally-required monitors are shown on these composite diagrams.

Off-line Gas and Particulate Monitor

The typical off-line gas and particulate monitor consists of a moving particulate filter with detector, iodine filter cartridge, gas sample chamber with detector, and associated pump and valving. Connections are available for taking grab samples of the process stream and taking samples for tritium analysis

downstream of the filter units. Check sources that are remotely operated are provided with each detector to check the function of each channel periodically. Remote purging capability is provided for the gas sample chamber and sample tubing.

Detectors are shielded and designed to obtain the sensitivities and ranges indicated in Table 11.5-1. Process streams in which plateout due to condensation could be a problem have been heat traced so that particulate sampling is representative of the process stream. Plateout is also minimized by using stainless steel for sample tubing and maintaining minimum bend radii in the sample lines in accordance with ANSI N13.1-1969.

Off-line Gas Monitor

The typical off-line gas monitor consists of an isokinetic sampling system, fixed particulate and charcoal filters, a gas sample chamber with detector, and associated pump with valving. Connections are available for taking grab samples of the process streams. All filters are removable for laboratory analysis. Check sources and purging capabilities are provided as described for the off-line gas and particulate monitor. Detector type, ranges, and alarm setpoints are given in Table 11.5-1. Isokinetic sampling systems are used to draw samples for the off-line gas monitor. The isokinetic sampling systems for these monitors are designed in accordance with ANSI N13.1-1969.

On-line Liquid Monitor

The on-line liquid monitor consists of a detector, shielding, and remotely-operated check source. The monitor is mounted to view a solid radwaste system line and shielded to obtain the sensitivities indicated in Table 11.5-1.

Off-line Liquid Monitor

The typical off-line liquid monitor consists of a sample chamber with detector and associated pump, piping, and valving. The detector has a remotely-operated check source and is shielded to obtain the sensitivities indicated in Table 11.5-1. Connections for taking a grab sample from the process stream and purging the liquid sample chamber and sample tubing are provided. Heat exchangers are provided on sampling systems that contain high temperature fluids.

Off-line Gaseous Effluent Monitor

Each off-line GEM consists of a noble gas detector and iodine and particulate filters with associated valving, electronics, and microprocessor. The system automatically determines gaseous effluent noble gas activity, provides the activity level to the PMS and LWS computers, stores the information, and retrieves the data on command. Iodine and particulate filters are manually removed and analyzed in the laboratory. The system automatically

purges itself and also automatically performs daily maintenance routines such as background count, source check, and diagnostics. Isokinetic sample probes are used to draw samples as previously described.

Continuous Airborne Monitor

CAMs are used to monitor process ventilation systems and local area ventilation activity conditions. The monitors contain noble gas and particulate detectors. They are also equipped to accommodate removable charcoal filter cartridges which can be installed, as needed, to obtain grab samples. All CAMs are equipped with a check source and purge capability, and mounted on movable carts to allow greater flexibility in the monitored areas depending on current plant conditions or maintenance schedules.

11.5.2.1.2 Monitors Required for Safety (DRMS)

Reactor Building Ventilation

Redundant off-line gas monitors are provided on the reactor building ventilation exhaust air ductwork above and below the refueling floor. These monitors function to indicate airborne levels of activity in the reactor building (Section 12.3.4). On a high-radiation alarm signal, the reactor building ventilation exhaust air is discontinued and reactor building air is recirculated, with a small fraction being diverted through the SGTS. A CAM connection is provided on the recirculation duct. These monitors are also designed to perform their required functions under all environmental conditions (Section 3.11).

Main Control Room Air Intakes

Redundant off-line gas monitors are provided at the main control room air intake. The main control room ventilation intake monitors divert the intake and recirculated air through HEPA/charcoal filter trains on high radiation. The main control room ventilation intake monitors indicate airborne radiation levels present in the main control room (Section 12.3.4). These monitors are also designed to perform their required functions under all environmental conditions (Section 3.11.1).

RHR Heat Exchanger Service Water Discharge

An off-line liquid monitor is located in the service water effluent on each of the two RHR heat exchangers. These monitors function to detect and alarm on contamination of the service water effluent due to leaks in the heat exchangers following a loss-of-coolant accident (LOCA) or under normal operating conditions. These monitors are also designed to perform their required function under all environmental conditions (Section 3.11.1).

11.5.2.1.3 Monitors Required for Plant Operations

Liquid Effluent Monitors (DRMS)

1. Cooling tower blowdown line (CWS) monitor - The cooling tower blowdown monitor detects and alarms on high levels of radioactivity in the plant CWS effluent to the discharge bay.

2. Liquid radwaste effluent monitor - The liquid radwaste effluent monitor terminates a LWS release if discharge concentrations approach the Offsite Dose Calculation Manual limits.

3. Service water discharge effluent monitors - Two service water discharge monitors are installed in the SWP effluent lines upstream of the discharge bay. These monitors will alert Operators if discharge concentrations approach Offsite Dose Calculation Manual limits.

Gaseous Effluent Monitors

1. Radwaste/Reactor Building Vent

Ventilation exhaust effluents from the reactor and radwaste buildings are combined and released from a single vent between the reactor building and turbine building. This release path is monitored by an off-line noble gas GEM which also continuously samples for iodine and particulate activity. The primary function of this monitor is to collect data for RG 1.21 report generation. Major process streams exhausted through the radwaste/reactor building vent include reactor building ventilation exhaust air, radwaste building area ventilation exhaust air, and radwaste building equipment exhaust air. The radwaste/reactor building vent monitor has extended range to cover post-accident monitoring requirements.

2. Main Plant Stack Exhaust

Effluent from the main stack (consisting of turbine building ventilation system exhaust air, containment purge exhaust, SGTS exhaust, condenser mechanical vacuum pump discharge, gland seal condenser gaseous discharge, and offgas system exhaust) is monitored by an off-line noble gas GEM which also continuously samples for iodine and particulate activity. The primary function of this monitor is to collect data for RG 1.21 report generation. The main stack monitor has the necessary range to cover possible activities released during and following an accident.

Process Gaseous Monitors (DRMS)

1. Standby Gas Treatment Discharge Monitor

An off-line gaseous monitor is installed on the discharge of the SGTS which isolates the normal containment purge system on a high radiation alarm.

2. Drywell Atmosphere Monitors

Redundant off-line gas and particulate monitors are provided to monitor drywell airborne activity levels and detect reactor coolant pressure boundary (RCPB) leakage in accordance with RG 1.45 requirements. The two drywell monitors pull samples through sampling trees located in the drywell and return the samples to the drywell. The sampling trees provide representative samples of drywell air by extracting samples from various elevations.

3. Offgas Pretreatment Monitors

The offgas process flow upstream of the charcoal adsorbers is monitored by off-line gaseous monitors equipped with iodine and particulate sampling capabilities. The offgas pretreatment monitors isolate the offgas effluent upon receipt of a high radiation signal.

4. Turbine Building Ventilation

A connection tap is provided in the system exhaust ductwork for a CAM.

5. Radwaste Building Area Exhaust and Tank Vent Exhaust Monitors

A connection tap for a CAM is provided in the exhaust ductwork of each of the above ventilation subsystems upstream of the filtration units.

Process Liquid Monitors (DRMS)

The following process streams are monitored by off-line liquid monitors for detection of radiation levels:

1. SFC system pumps discharge.

2. TBCLCW.

3. RBCLCW.

The SFC system monitor provides indication of radioactivity levels in the spent fuel pool water and will warn Operators of failures in spent fuel elements stored in the pool. The turbine and RBCLCW monitors detect and alarm following increases in the radioactivity level of the water in these systems.

A process radiation monitor was not installed on the secondary side of the alternate decay heat (ADH) system because the possibility of secondary side contamination has been analyzed and

found not to be a credible event. The basis for this conclusion is that the secondary side operates at a higher pressure than the primary side, a switch shuts down the primary side pumps on low secondary to primary side differential pressure, and plate heat exchanger design results in primary side leakage being to the reactor building rather than into the secondary side. As an added precaution and because the secondary side of ADH interfaces with a radioactive system, the ADH secondary side is routinely sampled for radioactivity.

11.5.2.2 Noncomputer-Based Process Radiation Monitoring System

Main Steam Line Radiation Monitoring System

This system monitors the gamma radiation level exterior to the MSLs. The normal radiation level is produced primarily by coolant activation gases plus smaller quantities of fission gases being transported with the steam. In the event of a gross release of fission products from the core, this monitoring system provides annunciation in the control room.

The system consists of four redundant instrument channels. Each channel consists of a local on-line steam detector (gamma-sensitive ion chamber) and a main control room radiation monitor with an auxiliary trip unit. A diagram of this type of monitor is shown on Figure 11.5-4.

The detectors are physically located near the MSLs just downstream of the outboard main steam isolation valves (MSIV). The detectors are geometrically arranged so that this system is capable of detecting significant increases in radiation level with any number of MSLs in operation. Table 11.5-1 lists the range of the detectors.

Each radiation monitor has four trip circuits: two upscale (high-high and high), one downscale (low), and one inoperative. Each trip is visually displayed locally on the affected radiation monitor panel. A high-high or inoperative trip in the radiation monitor results in a one-out-of-two twice channel trip. This initiates mechanical vacuum pump shutdown, and discharge valve closure. A high-high trip or inoperative trip actuates a MSL high-high radiation main control room annunciator. A high trip actuates a MSL high radiation main control room annunciator. A downscale trip actuates a MSL downscale main control room annunciator common to all channels. High and low trips do not result in a channel trip. Each radiation level is visually displayed locally.

11.5.2.3 Calibration, Maintenance, Inspection, and Tests

Calibration, inspection, and tests of monitors required for safety (except for the RHR heat exchanger service water discharge monitors whose testing is described in Section 7.5.3), effluent monitors, the offgas pretreatment monitor, the SGTS discharge

monitor, and the drywell atmosphere monitors are performed in accordance with the plant Technical Specifications and the Offsite Dose Calculation Manual. Calibration, inspection, and tests of monitors not covered by the Technical Specifications' surveillance program or the Offsite Dose Calculation Manual's surveillance program are conducted in accordance with the manufacturer's recommendations. Maintenance of all process and effluent monitors is performed in accordance with appropriate plant procedures.

The design of all process and effluent radiation monitors meets the applicable criteria cited in RG 4.15. Each monitor is equipped with purge capability for analysis of background radiation and check or keep-alive source(s) for determination of changes in detector counting rates, efficiencies, or energy resolution. Effluent monitoring systems are designed using the guidelines established in ANSI 13.10. Calibration standards and frequency are described in Section 11.5.2.3.2.

The quality assurance (QA) criteria is consistent with the function of the monitor. Safety-related monitors are procured and designed to 10CFR50 Appendix B criteria. Nonsafety-related monitors are designed and procured under standards that meet the criteria established in RG 1.143.

11.5.2.3.1 Inspection and Tests

Inspection and testing of the monitors specified in Section 11.5.2.3 is described in Station procedures and either the Technical Specifications or the Offsite Dose Calculation Manual.

11.5.2.3.2 Calibration

The radiation monitor's calibration is traceable to the National Institute of Standards and Technology (NIST) and is accurate to at least ±15 percent. The source-detector geometry during initial calibration is identical to the sample-detector geometry in actual use. Secondary standards that were counted in a reproducible geometry during the initial calibration are used for calibration after installation. Where applicable, each monitor (except for the RHR heat exchanger service water discharge monitors whose calibration is described in Section 7.5.3) is calibrated in accordance with the frequencies provided in the plant Technical Specifications or the Offsite Dose Calculation Manual during plant operation or during the refueling outage if the detector is not readily accessible.

11.5.2.3.3 Maintenance

The channel detector, electronics, and recorder are serviced and maintained in accordance with manufacturers' recommendations to ensure reliable operations. Such maintenance includes cleaning, lubrication, and assurance of free movement of the recorder in

addition to the replacement or adjustment of any components required after performing a test or calibration check. If any work is performed that would affect the calibration, a recalibration is performed at the completion of the work.

11.5.2.4 Sampling

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

Liquid process and effluent samples are periodically taken and monitored for radioactivity. Those provisions for sampling not covered in Section 9.3.2 are described in the individual system design sections. Sampling of these fluid systems is via local sampling connections. The Technical Specifications and the Offsite Dose Calculation Manual describe various liquid samples and the analysis required, including the sampling frequencies.

Additionally, the process and effluent radiological monitoring system can provide grab samples that are used to locate manually a specific source of high radioactivity when a continuous radiological monitor signals a high radioactivity alarm in the main control room. Tritium in the plant areas is determined on the basis of representative grab samples collected from the effluent points or ventilation exhaust ducts. Grab samples are obtained from locations indicated in Table 11.5-2 and the samples are analyzed. In addition to grab samples, provisions have been included for the offgas pretreatment process monitors to allow alternate sampling with local analysis capabilities (Figure 11.3-1b).

11.5.3 Effluent Monitoring and Sampling

All potentially-radioactive gaseous and liquid effluent discharge paths are either continuously monitored or routinely sampled for radiation level during discharge (Section 11.5.2). Solid waste shipping containers are monitored with gamma-sensitive portable survey instruments. The following gaseous effluent paths are sampled and monitored:

1. Plant main stack exhaust.

2. Combined radwaste/reactor building ventilation exhaust.

The following liquid effluent paths are sampled and monitored:

1. LWS effluent.

2. CWS cooling tower blowdown line.

3. SWP discharge.

All monitor ranges are listed in Table 11.5-1.

An isotopic analysis is performed periodically on samples obtained from each liquid effluent release path to verify the adequacy of effluent processing to meet the discharge limits to unrestricted areas.

This effluent monitoring and sampling program is comprehensive and provides the information for the effluent measuring and reporting programs required by 10CFR50 Section 36a, Appendix A, GDC 64, and Appendix I and RG 1.21 in semiannual reports to the NRC. The frequency of the periodic sampling and analysis described in the Technical Specifications, the Offsite Dose Calculation Manual, and in TRM Section 3.4.8 is a minimum and is increased if effluent levels approach Technical Specification or Offsite Dose Calculation Manual limits. Activity level of noble gas effluents is continuously monitored by off-line monitors. Iodine and particulate in the gaseous effluents are continuously sampled with isotopic content determined by laboratory analysis. All potentially-significant radioactive discharge paths are equipped with a control system to isolate the discharge automatically on indication of a high radiation level. These include:

1. Offgas pretreatment.

2. SGTS discharge (isolates containment purge system).

3. Reactor building ventilation exhaust.

4. Liquid radwaste effluent.

The effluent isolation functions for each monitor are given in Table 11.5-1 and in Section 11.5.2.

Radiation levels in radioactive and potentially-radioactive process streams are monitored by the process and effluent monitors given in Table 11.5-1.

Airborne radioactivity in the fuel-handling area and the radwaste building is detected by CAM and area radiation monitors. Airborne radioactivity in the drywell is detected by the drywell atmosphere monitors and the SGTS discharge monitor which isolates the containment purge on high radioactivity. These monitors are also described in Section 12.3.4 since they are used to monitor in-plant airborne radioactivity to protect plant personnel. The area RMS is also described in Section 12.3.4. A system level/qualitative-type failure modes and effects analysis (FMEA) of the MSL radiation monitoring is provided in Appendix 15A. Originally, the FMEA for other safety-related radiation monitors was contained in the Unit 2 FMEA document, which is historical. FMEAs for plant systems are now performed and controlled by the design process.

TABLE 11.5-1

PROCESS AND EFFLUENT RADIATION MONITORING SYSTEMS

Monitor Location	Monitor Type	Range[4]	Isotope	Trip/High Setpoint	Function
Monitors Required for Safety					
Reactor building ventilation above and below refueling floor (2HVR*CAB14A,B; 2HVR*CAB32A,B)	Offline gaseous	10^{-7} to 10^{-1} uCi/cc	Xe-133, Kr-85	Allowable Value in Tech. Spec.	Monitors radiation levels in the reactor building ventilation system. Isolates reactor building[5]
Main control room intake (2HVC*CAB18A,B,C,D)	Offline gaseous	10^{-7} to 10^{-1} uCi/cc	Xe-133, Kr-85	Allowable Value in Tech. Spec.	Monitors incoming control room air; activates Category I HEPA/charcoal filters[5]
RHR heat exchanger service water (2SWP*CAB23A,B)	Offline liquid	10^{-7} to 10^{-1} uCi/cc	Cs-137	$\leq 3.3 \times 10^{-4}$ uCi/cc	Monitors service water effluent from heat exchangers for contamination[6]
Main steam line[3] (2MSS*RT46A,B,C,D)	Online steam	$1\text{-}10^{6}$ mr/hr	N-16	Allowable Value in Tech. Spec.	Monitors main steam lines for fuel damage and carryover to turbine building, trips mechanical vacuum pumps and associated isolation valve.
Monitors Required for Plant Operation					
Drywell and containment atmosphere[1] (2CMS*CAB10A,B)	Offline gaseous	10^{-7} to 10^{-1} uCi/cc	Xe-133, Kr-85	N2-RTP-129	Monitors drywell for airborne radiation - RCPB leak detection[5]
	Particulate	10^{-11} to 10^{-5} uCi/cc	I-131	N2-RTP-129	
Service water system discharge monitors (2SWP*CAB146A,B)	Offline liquid	10^{-7} to 10^{-1} uCi/cc	Cs-137	ODCM	Monitors service water system discharge[5]
Radwaste/reactor building vent[2] (2RMS-SKD180A&B, 2RMS-PNL180A&B)	Offline gaseous	10^{-6} to 10^{5} uCi/cc	Xe-133, Kr-85[7]	ODCM	Monitors reactor and radwaste building ventilation effluent releases for RG 1.21 report generation[5]

TABLE 11.5-1 (Cont'd.)

Monitor Location	Monitor Type	Range[4]	Isotope	Trip/High Setpoint	Function
Main stack exhaust[2] (2RMS-SKD170A&B, 2RMS-PNL170A&B)	Offline gaseous	10^{-6} to 10^{5} uCi/cc	Xe-133, Kr-85[7]	ODCM	Monitors content of effluent releases for RG 1.21 report generation[5]
Fuel pool cooling pumps discharge (2SFC-CAB142)	Offline liquid	10^{-7} to 10^{-1} uCi/cc	Cs-137	$\leq 2.2 \times 10^{-2}$ uCi/cc	Monitors fuel pool cooling water for fuel damage[6]
Turbine plant closed loop cooling water (2CCS-CAB152)	Offline liquid	10^{-7} to 10^{-1} uCi/cc	Cs-137	$<3.0 \times 10^{-3}$ uCi/cc	Monitors CCS for detection of radioactive in-leakage[6]
Liquid radwaste effluent (2LWS-CAB206)	Offline liquid	10^{-7} to 10^{-1} uCi/cc	Cs-137	ODCM	Monitors radwaste effluent discharge to environment isolates system at trip level[5]
Cooling tower blowdown line (2CWS-CAB157)	Offline liquid	10^{-7} to 10^{-1} uCi/cc	Cs-137	ODCM	Monitors circulating water system discharge to environment[5]
Standby gas treatment discharge (2GTS-CAB105)	Offline gaseous	10^{-6} to 10^{-1} uCi/cc	Xe-133, Kr-85	Allowable Value in Tech. Spec.	Monitors containment purge exhaust; isolates primary containment discharge lines[5]
Reactor plant closed loop cooling water					Monitors each CCP loop for detection of radioactive in-leakage[6]
a. SFC heat exchanger cooling water discharge (2CCP-CAB115)	Offline liquid	10^{-7} to 10^{-1} uCi/cc	Cs-137	$\leq 1.6 \times 10^{-2}$ uCi/cc	
b. RWCU heat exchanger cooling water discharge (2CCP-CAB131)	Offline liquid	10^{-7} to 10^{-1} uCi/cc	Cs-137	$<9.1 \times 10^{-3}$ uCi/cc	
Radwaste building equipment, tank vents and area ventilation exhausts					Monitors radiation levels in radwaste building equipment, tank vents, and area ventilation exhaust prior to filtration and exhaust to the radwaste/reactor building vent[6]
a. Radwaste equipment exhaust (2HVW-CAB195)	CAM	gas: 10^{-7} to 10^{-1} uCi/cc part: 10^{-11} to 10^{-5} uCi/cc	Kr-85, Xe-133 I-131	$\leq 2.3 \times 10^{-2}$ uCi/cc NA	

TABLE 11.5-1 (Cont'd.)

Monitor Location	Monitor Type	Range[4]	Isotope	Trip/High Setpoint	Function
b. Radwaste tank vents (2HVW-CAB196)	CAM	gas: 10^{-7} to 10^{-1} uCi/cc part: 10^{-11} to 10^{-5} uCi/cc	Kr-85, Xe-133 I-131	$\leq 2.3 \times 10^{-2}$ uCi/cc NA	
c. Radwaste building ventilation (2HVW-CAB197)	CAM	gas: 10^{-7} to 10^{-1} uCi/cc part: 10^{-11} to 10^{-5} uCi/cc	Kr-85, Xe-133 I-131	$\leq 2.3 \times 10^{-2}$ uCi/cc NA	
d. Radwaste equipment service area (2HVW-CAB199)	CAM	gas: 10^{-7} to 10^{-1} uCi/cc part: 10^{-11} to 10^{-5} uCi/cc	Kr-85, Xe-133 I-131	$\leq 2.3 \times 10^{-2}$ uCi/cc NA	
Turbine building ventilation exhaust (2HVT-CAB206)	CAM	gas: 10^{-7} to 10^{-1} uCi/cc part: 10^{-11} to 10^{-5} uCi/cc	Kr-85, Xe-133 I-131	$\leq 4.0 \times 10^{-2}$ uCi/cc NA	Monitors radiation levels in turbine building ventilation prior to exhaust to the main stack[6]
Offgas pretreatment (2OFG-CAB13A,B)	Offline gaseous	10^{-3} to 10^{2} uCi/cc	Xe-133, Kr-85	ODCM	Monitors process before charcoal adsorbers; isolates offgas discharge[5]

(1) RCPB leak detection in accordance with RG 1.45.
(2) PAM - Post-accident monitor.
(3) Monitors are not part of computer-based radiation monitoring systems, supplied separately as part of the process system.
(4) Design range is stated for DRMS and main steam monitors. Due to ambient conditions, the low end of the range may not be available or required. For GEMS monitors, the low end of the ranges required by NUREG-0473 and the high end of the ranges required by RG 1.97 are given.
(5) On actuation of high trip, action is required in accordance with the Technical Specifications.
(6) On actuation of high trip, new trip setpoint and/or system corrective actions will be specified.
(7) Kr-85 used to establish calibration curve and subsequently removed from analysis during normal operation to avoid erroneous identification.

TABLE 11.5-2

GRAB SAMPLES FOR RADIOLOGICAL ANALYSIS

Sample Point Location (No.)	Grab Sample at Sample Station	Local Grab Sample	Grab Sample at Radiation Monitor
Reactor Steam Supply System			
Reactor recirculation system pump discharge	X	-	-
Main steam line	X	-	-
Reactor Water Cleanup System			
Common filter/demineralizer influent	X	-	-
Individual filter/demineralizer effluents(4)	X	-	-
Fuel Pool Cooling and Cleanup System			
Pump discharge	-	-	X
Common filter influent	X	-	-
Individual filter effluents(2)	X	-	-
Individual heat exchanger effluents(2)	X	-	-
Reactor Building Closed Loop Cooling Water			
Cooling water sample (outlet of RWCU and SFC heat exchangers)	-	-	X
Turbine Building Closed Loop Cooling Water			
Cooling water sample (common outlet of radwaste system exchangers)	-	-	X
Residual Heat Removal System			
Individual heat exchanger outlet (service water)(2)	-	-	X
Individual heat exchanger outlet (RHR)(2)	X	-	-
Control Rod Drive System			
Common CRD filter effluent	X	-	-
High-Pressure Core Spray System			
Test return line to condensate storage tank	X	-	-

TABLE 11.5-2 (Cont'd.)

Sample Point Location (No.)	Grab Sample at Sample Station	Local Grab Sample	Grab Sample at Radiation Monitor
Radwaste System			
Individual waste collector tank pump effluents(3)	X	-	-
Individual demineralizer effluents(2)	X	-	-
Filtrate pump effluent	X	-	-
Individual filter effluents(2)	X	-	-
Demineralizer (acid influent)	-	X	-
Demineralizer (caustic influent)	-	X	-
Individual recovery sample pump effluents(2)	X	-	-
Individual floor drain collector pump effluents(2)	X	-	-
Floor drain filter effluent pump discharge	X	-	-
Liquid radwaste final discharge	-	-	X
Regenerant waste pump effluents(2)	X	-	-
Regenerant recirculation pump suction and discharge(2)	X	-	-
Phase separator tank pump discharge	-	X	-
Waste evaporator recirculation pump suction and discharge(2)	X	-	-
Waste evaporator distillate	X	X	-
Individual waste sample pumps, discharge(2)	X	-	-
Regenerant evaporator distillate	X	X	-
Common discharge floor drain collector surge pumps	X	-	-
Common discharge waste collector surge pumps	X	-	-
Individual radwaste auxiliary steam cooler effluents(2)	X	-	-
Water Treating System			
Dilute acid effluent	-	X	-
Dilute caustic effluent	-	X	-
Waste water effluent	-	X	-
Condensate Demineralizer System			
Common demineralizer influent	X	-	-
Common demineralizer effluent	X	-	-
Resin hold tank effluent	X	-	-
Individual demineralizers effluent(9)	X	-	-
Resin mix tank effluent	X	-	-
Cation regeneration tank effluent	X	-	-
Anion regeneration tank effluent	X	-	-
Recovered acid tank effluent	-	X	-
Regeneration system effluent	X	-	-
Low conductivity waste tank effluent	X	-	-
Demineralizer waste neutralizing tank effluent	X	-	-
Dilute acid effluent	X	X	-
Recovered caustic tank effluent	-	X	-
Dilute caustic effluent	X	X	-
Recovered water sump effluent	-	X	-

TABLE 11.5-2 (Cont'd.)

Sample Point Location (No.)	Grab Sample at Sample Station	Local Grab Sample	Grab Sample at Radiation Monitor
Condensate Makeup and Drawoff System			
Condensate transfer line	X	-	-
Makeup Water System			
Demineralizer water transfer line	X	-	-
Condensate System			
Condensate pump discharge	X	-	-
Condenser hotwells(6)	X	-	-
LP heater drains(3)	X	-	-
Common effluent fourth-point heaters	X	-	-
LP heater string common effluent	X	-	-
Reactor Feedwater System			
Feedwater (after last heater)	X	-	-
Circulating Water System			
Effluent (blowdown line)	-	X	X
Auxiliary Steam System			
Auxiliary boiler (steam outlet)	X	-	-
Feedwater (pump discharge)	X	-	-
Auxiliary boiler (blowdown)	-	X	-
Auxiliary boiler recirc pump seal heat exchanger outlet and sample cooler discharge (Service Water)(4)	-	X	-
Sealing Steam System			
Individual clean steam reboiler outlets	X	-	-
Reactor Building Ventilation System			
Reactor/radwaste building ventilation exhaust	-	-	X
Main plant stack exhaust	-	-	X
Containment atmosphere	-	-	X
Containment purge	-	-	X
Standby Gas Treatment System			
SGTS effluent	-	-	X
Control Building Ventilation System			
Main control room intakes	-	X	X

TABLE 11.5-2 (Cont'd.)

Sample Point Location (No.)	Grab Sample at Sample Station	Local Grab Sample	Grab Sample at Radiation Monitor
Radwaste Building Ventilation System			
Ventilation exhaust	-	X	X
Radwaste tank vents	-	-	X
Turbine Building Ventilation System			
Ventilation	-	X	-
Mechanical vacuum pump discharge	-	X	-
Offgas pretreatment(2)	-	-	X
Turbine gland seal discharge	-	X	-
Service Water System			
Final discharge	-	X	X
Storm, Underdrain Water, and Site Sewage Systems			
Final discharge	-	X	-
Alternate Decay Heat Removal System			
Discharge to storm drain	-	X	-

NOTE: See Section 9.3.2 for details regarding the reactor, turbine, and radwaste sample systems.

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RADIOLOGICAL DOSES

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APPENDIX 11A

RADIOLOGICAL DOSES

11A.1 SUMMARY OF ANNUAL RADIATION DOSES

The calculated annual radiation doses to maximum individuals from normal operation of Unit 2 are presented in Tables 11A.1-1 through 11A.1-16. Table 11A.1-17 demonstrates that the calculated annual radiation doses are below the design objectives of 10CFR50 Appendix I. In providing guidance for the implementation of Appendix I, the NRC has made use of the maximum exposed individual approach. Maximum individuals are characterized as maximum with regard to food consumption, occupancy, and other usage of the region in the vicinity of the plant site and, as such, represent individuals with reasonable deviations from the average individual considered representative of the population in general.

For gaseous radioactive releases the analyzed pathways included: standing on contaminated ground, ingestion of vegetation, and inhalation of and submersion in gaseous effluents. These pathways were considered for all of the resident locations. Resident locations having cows, goats, and meat animals were also analyzed for ingestion of cow milk, goat milk, and beef meat, respectively. Additionally, the doses associated with ingestion of deer were conservatively added to all resident locations analyzed.

The calculated organ dose due to radioiodines and particulates is 2.8E+00 mRem. This represents the dose to the thyroid of an infant living at the residence location 2,350 m east-southeast of the site. The majority of this dose was due to consumption of cow milk. The highest calculated external annual doses to the total body and skin from immersion in noble gases at an occupied location were 3.9E-02 and 8.3E-02 mRem, respectively. These occurred at the residence location 1,693 m east of the site.

The highest calculated beta and gamma air doses at an unoccupied location from noble gas releases were 5.3E-02 and 7.9E-02 mrad, respectively. These occurred at the EAB 1,603 m east of the site.

For liquid releases, the maximum individual consumed fish whose principal habitat was assumed to be the edge of the initial mixing zone. This location was also conservatively used in calculating doses from swimming and boating.

The calculated annual doses to the population residing within an 80-km radius of the site are presented in Table 11A.1-18. Population doses were calculated for a projected population of 1.2 million residing within 80 km of the site in the year 2010.

Gaseous pathways considered in the population dose analysis included: inhalation, exposure to ground deposits, ingestion of vegetation, cow milk, and beef meat, and submersion in noble gases. The calculated annual dose to the population was 0.6 man-Rem/yr total body[1] and 3.3 man-Rem/yr thyroid[2].

Liquid pathways considered in the population dose analysis included: ingestion of potable water and fish, shoreline recreation, and swimming and boating. The calculated annual dose to the population was 4.3 man-Rem/yr total body and 1.2E-01 man-Rem/yr thyroid.

In addition to the 80-km (50-mi) radius population doses, population doses associated with the export of food crops produced within the 80-km (50-mi) region and the atmospheric and hydrospheric transport of the more mobile effluent species, such as noble gases, tritium, and carbon-14, were calculated.

These calculated annual gaseous and liquid doses to the contiguous U.S. population are presented in Table 11A.1-19. For liquid effluents, the calculated doses to the contiguous U.S. population were 4.3 man-Rem/yr total body and 1.2E-01 man-Rem/yr thyroid. For gaseous effluents, the calculated doses to the contiguous U.S. population were 35.57 man-Rem/yr total body[3] and 43.04 man-Rem/yr thyroid[4].

(1),(2) Refer to the power uprate footnotes (3 and 4) and the revised dose values shown in Table 11A.1-18.

(3),(4) Refer to the power uprate footnote (3) shown in Table 11A.1-19.

TABLE 11A.1-1

ANNUAL DOSES TO MAXIMUM INDIVIDUAL IN THE
ADULT GROUP FROM LIQUID EFFLUENTS

(Annual Dose in mRem/yr)[1]

Pathway	Total Body	Skin	Bone	Liver	Thyroid	Kidney	Lung	GI Tract
Potable water	3.5E-04	0.0	1.7E-04	4.1E-04	2.9E-04	2.6E-04	2.1E-04	2.0E-04
Fish consumption	5.8E-01	0.0	7.0E-01	8.2E-01	3.6E-03	2.7E-01	8.9E-02	4.3E-02
Shoreline recreation	2.3E-05	2.6E-05	2.3E-05	2.3E-05	2.3E-05	2.3E-05	2.3E-05	2.3E-05
Fresh vegetation	5.9E-05	0.0	4.4E-05	7.5E-05	2.6E-05	3.6E-05	2.2E-05	2.0E-05
Stored vegetation	4.1E-04	0.0	3.0E-04	5.2E-04	1.1E-04	2.5E-04	1.6E-04	1.4E-04
Duck consumption	1.3E-04	0.0	2.1E-03	2.2E-04	1.4E-06	4.7E-05	6.4E-06	2.9E-04
Swimming exposure	3.3E-05	4.2E-05	3.3E-05	3.3E-05	3.3E-05	3.3E-05	3.3E-05	3.3E-05
Boating exposure	3.3E-05	4.2E-05	3.3E-05	3.3E-05	3.3E-05	3.3E-05	3.3E-05	3.3E-05
Total dose	5.8E-01	1.1E-04	7.0E-01	8.2E-01	4.1E-03	2.7E-01	8.9E-02	4.4E-02

NOTE: $3.5E-04 = 3.5 \times 10^{-4}$

[1] All annual doses shown in Table 11A.1-1 reflect a power level of 3,323 MWt. Due to an operation at EPU/MELLLA+ at 3,988 MWt, each of the values shown must be multiplied by a factor of 5.35.

TABLE 11A.1-2

ANNUAL DOSES TO MAXIMUM INDIVIDUAL IN THE
TEEN GROUP FROM LIQUID EFFLUENTS

(Annual Dose in mRem/yr)[1]

Pathway	Total Body	Skin	Bone	Liver	Thyroid	Kidney	Lung	GI Tract
Potable water	2.2E-04	0.0	1.6E-04	3.5E-04	2.2E-04	3.2E-04	1.6E-04	1.4E-04
Fish consumption	3.3E-01	0.0	7.6E-01	8.4E-01	3.3E-03	2.8E-01	1.1E-01	3.3E-02
Shoreline recreation	1.3E-04	1.5E-04	1.3E-04	1.3E-04	1.3E-04	1.3E-04	1.3E-04	1.3E-04
Fresh vegetation	3.2E-05	0.0	3.9E-05	6.3E-05	1.8E-05	5.1E-05	1.7E-05	1.3E-05
Stored vegetation	4.4E-04	0.0	5.3E-04	8.7E-04	1.5E-04	5.6E-04	2.4E-04	1.8E-04
Duck consumption	1.0E-04	0.0	1.8E-03	1.8E-04	9.9E-07	3.0E-04	6.0E-06	1.8E-04
Swimming exposure	3.3E-05	4.2E-05	3.3E-05	3.3E-05	3.3E-05	3.3E-05	3.3E-05	3.3E-05
Boating exposure	3.3E-05	4.2E-05	3.3E-05	3.3E-05	3.3E-05	3.3E-05	3.3E-05	3.3E-05
Total dose	3.3E-01	2.3E-04	7.6E-01	8.4E-01	3.9E-03	2.8E-01	1.1E-01	3.4E-02

NOTE: 2.2-04 = 2.2×10^{-4}

[1] All annual doses shown in Table 11A.1-2 reflect a power level of 3,323 MWt. Due to an operation at EPU/MELLLA+ at 3,988 MWt, each of the values shown must be multiplied by a factor of 5.34.

TABLE 11A.1-3

ANNUAL DOSES TO MAXIMUM INDIVIDUAL IN THE
CHILD GROUP FROM LIQUID EFFLUENTS

(Annual Dose in mRem/yr)[1]

Pathway	Total Body	Skin	Bone	Liver	Thyroid	Kidney	Lung	GI Tract
Potable water	3.3E-04	0.0	4.6E-04	6.9E-04	4.7E-04	3.9E-04	3.0E-04	2.5E-04
Fish consumption	1.4E-01	0.0	9.5E-01	7.5E-01	3.3E-03	2.4E-01	8.5E-02	1.4E-02
Shoreline recreation	2.6E-05	3.1E-05	2.6E-05	2.6E-05	2.6E-05	2.6E-05	2.6E-05	2.6E-05
Fresh vegetation	2.5E-05	0.0	6.8E-05	7.9E-05	2.4E-05	3.4E-05	2.0E-05	1.4E-05
Stored vegetation	4.7E-04	0.0	1.3E-03	1.5E-03	2.4E-04	6.3E-04	3.8E-04	2.6E-04
Duck consumption	1.6E-04	0.0	3.4E-03	2.5E-04	1.5E-06	4.2E-05	7.1E-06	1.1E-04
Swimming exposure	1.8E-05	2.4E-05	1.8E-05	1.8E-05	1.8E-05	1.8E-05	1.8E-05	1.8E-05
Boating exposure	1.9E-05	2.4E-05	1.9E-05	1.9E-05	1.9E-05	1.9E-05	1.9E-05	1.9E-05
Total dose	1.4E-01	7.9E-05	9.6E-01	7.5E-01	4.1E-03	2.4E-01	8.6E-02	1.5E-02

NOTE: $3.3E-04 = 3.3 \times 10^{-4}$

[1] All annual doses shown in Table 11A.1-3 reflect a power level of 3,323 MWt. Due to an operation at EPU/MELLLA+ at 3,988 MWt, each of the values shown must be multiplied by a factor of 5.34.

TABLE 11A.1-4

ANNUAL DOSES TO MAXIMUM INDIVIDUAL IN THE
INFANT GROUP FROM LIQUID EFFLUENTS

(Annual Dose in mRem/yr)[1]

Pathway	Total Body	Skin	Bone	Liver	Thyroid	Kidney	Lung	GI Tract
Potable water	2.9E-04	0.0	4.7E-04	7.9E-04	6.0E-04	3.9E-04	3.0E-04	2.5E-04
Total dose	2.9E-04	0.0	4.7E-04	7.9E-04	6.0E-04	3.9E-04	3.0E-04	2.5E-04

NOTE: 2.9E-04 = 2.9×10^{-4}

(1) All annual doses shown in Table 11A.1-4 reflect a power level of 3,323 MWt. Due to an operation at EPU/MELLLA+ at 3,988 MWt, each of the values shown must be multiplied by a factor of 5.32.

TABLE 11A.1-5

ANNUAL DOSES TO MAXIMUM INDIVIDUAL IN THE
ADULT GROUP FROM GASEOUS EFFLUENTS*

At Maximum Residence Location

(Annual Dose in mRem/yr)[1]

Pathway	Total Body	Skin	Bone	Liver	Thyroid	Kidney	Lung	GI Tract
Contaminated ground	7.9-03	9.2-03	7.9-03	7.9-03	7.9-03	7.9-03	7.9-03	7.9-03
Inhalation	1.6-04	0.0	1.6-04	2.4-04	1.3-02	3.0-04	2.8-04	2.1-04
Fresh vegetation	7.5-04	0.0	1.7-03	1.1-03	1.3-01	1.2-03	2.1-04	8.0-04
Stored vegetation	2.7-03	0.0	6.3-03	3.2-03	4.5-03	1.8-03	1.1-03	1.8-03
Deer 1,603 m east	1.4-04	0.0	1.7-04	1.9-04	3.2-04	9.2-05	3.1-05	3.3-04
Total dose	1.2-02	9.2-03	1.6-02	1.3-02	1.6-01	1.1-02	9.5-03	1.1-02

* Analysis performed at maximum residence location is 4,106 m (13,471 ft) east.

NOTE: 7.9-03 = 7.9×10^{-3}

[1] All annual doses shown in Table 11A.1-5 reflect a power level of 3,323 MWt. Due to an operation at EPU/MELLLA+ at 3,988 MWt, each of the values shown must be multiplied by a factor of 1.79; the uprated value represents the maximum possible increase.

TABLE 11A.1-6

ANNUAL DOSES TO MAXIMUM INDIVIDUAL IN THE
TEEN GROUP FROM GASEOUS EFFLUENTS*

At Maximum Residence Location

(Annual Dose in mRem/yr)[1]

Pathway	Total Body	Skin	Bone	Liver	Thyroid	Kidney	Lung	GI Tract
Contaminated ground	7.9-03	9.2-03	7.9-03	7.9-03	7.9-03	7.9-03	7.9-03	7.9-03
Inhalation	1.8-04	0.0	2.2-04	2.9-04	1.7-02	3.8-04	3.8-04	2.3-04
Fresh vegetation	5.4-04	0.0	1.6-03	1.0-03	1.1-01	1.1-02	1.9-04	5.8-04
Stored vegetation	3.5-03	0.0	1.1-02	5.7-03	7.3-03	6.0-02	2.0-03	2.7-03
Deer 1,603 m east	7.8-05	0.0	1.4-04	1.5-04	2.4-04	7.6-03	2.7-05	1.9-04
Total dose	1.2-02	9.2-03	2.1-02	1.5-02	1.4-01	8.7-02	1.0-02	1.2-02

* Analysis performed at maximum residence location is 4,106 m (13,471 ft) east.

NOTE: 7.9-03 = 7.9×10^{-3}

[1] All annual doses shown in Table 11A.1-6 reflect a power level of 3,323 MWt. Due to an operation at EPU/MELLLA+ at 3,988 MWt, each of the values shown must be multiplied by a factor of 1.79; the uprated value represents the maximum possible increase.

TABLE 11A.1-7

ANNUAL DOSES TO MAXIMUM INDIVIDUAL IN THE
CHILD GROUP FROM GASEOUS EFFLUENTS*

At Maximum Residence Location

(Annual Dose in mRem/yr)[1]

Pathway	Total Body	Skin	Bone	Liver	Thyroid	Kidney	Lung	GI Tract
Contaminated ground	7.9-03	9.2-03	7.9-03	7.9-03	7.9-03	7.9-03	7.9-03	7.9-03
Inhalation	1.8-04	0.0	3.0-04	2.8-04	2.1-02	3.6-04	3.3-04	1.9-04
Fresh vegetation	7.1-04	0.0	2.9-03	1.4-03	1.6-01	1.4-03	3.2-04	5.3-04
Stored vegetation	5.7-03	0.0	2.8-02	1.1-02	1.5-02	6.2-03	4.5-03	4.6-03
Deer 1,603 m east	8.6-05	0.0	2.5-04	2.0-04	3.6-04	9.5-05	4.0-05	1.2-04
Total dose	1.5-02	9.2-03	3.9-02	2.1-02	2.0-01	1.6-02	1.3-02	1.3-02

* Analysis performed at maximum residence location is 4,106 m (13,471 ft) east.

NOTE: 7.9-03 = 7.9×10^{-3}

[1] All annual doses shown in Table 11A.1-7 reflect a power level of 3,323 MWt. Due to an operation at EPU/MELLLA+ at 3,988 MWt, each of the values shown must be multiplied by a factor of 1.79; the uprated values represent the maximum possible increase.

TABLE 11A.1-8

ANNUAL DOSES TO MAXIMUM INDIVIDUAL IN THE
INFANT GROUP FROM GASEOUS EFFLUENTS*

At Maximum Residence Location

(Annual Dose in mRem/yr)[1]

Pathway	Total Body	Skin	Bone	Liver	Thyroid	Kidney	Lung	GI Tract
Contaminated ground	7.9-03	9.2-03	7.9-03	7.9-03	7.9-03	7.9-03	7.9-03	7.9-03
Inhalation	1.2-04	0.0	2.2-04	2.2-04	1.9-02	2.3-04	2.4-04	1.1-04
Total dose	8.0-03	9.2-03	8.1-03	8.1-03	2.7-02	8.1-03	8.1-03	8.0-03

* Analysis performed at maximum residence location is 4,106 m (13,471 ft) east.

NOTE: 7.9-03 = 7.9×10^{-3}

[1] All annual doses shown in Table 11A.1-8 reflect a power level of 3,323 MWt. Due to an operation at EPU/MELLLA+ at 3,988 MWt, each of the values shown must be multiplied by a factor of 1.79; the uprated values represent the maximum possible increase.

TABLE 11A.1-9

ANNUAL DOSES TO MAXIMUM INDIVIDUAL IN THE
ADULT GROUP FROM GASEOUS EFFLUENTS*

At Maximum Cow Location

(Annual Dose in mRem/yr)[1]

Pathway	Total Body	Skin	Bone	Liver	Thyroid	Kidney	Lung	GI Tract
Contaminated ground	1.3-02	1.6-02	1.3-02	1.3-02	1.3-02	1.3-02	1.3-02	1.3-02
Inhalation	1.4-04	0.0	1.1-04	2.0-04	9.7-03	2.5-04	2.7-04	1.9-04
Fresh vegetation	7.0-04	0.0	1.3-03	1.1-03	1.3-01	1.1-03	1.3-04	7.3-04
Stored vegetation	2.4-03	0.0	4.3-03	3.0-03	4.1-03	1.4-03	6.9-04	1.4-03
Cow milk	2.4-03	0.0	3.0-03	3.7-03	2.2-01	2.7-03	4.5-04	1.1-03
Deer 1,603 m east	1.4-04	0.0	1.7-04	1.9-04	3.2-04	9.2-05	3.1-05	3.3-04
Total dose	1.9-02	1.6-02	2.2-02	2.1-02	3.8-01	1.9-02	1.5-02	1.7-02

* Analysis performed at maximum cow location is 2,350 m (7,710 ft) east-southeast.

NOTE: 1.3-02 = 1.3×10^{-2}

[1] All annual doses shown in Table 11A.1-9 reflect a power level of 3,323 MWt. Due to an operation at EPU/MELLLA+ at 3,988 MWt, each of the values shown must be multiplied by a factor of 1.79; the uprated values represent the maximum possible increase.

TABLE 11A.1-10

ANNUAL DOSES TO MAXIMUM INDIVIDUAL IN THE
TEEN GROUP FROM GASEOUS EFFLUENTS*

At Maximum Cow Location

(Annual Dose in mRem/yr)[1]

Pathway	Total Body	Skin	Bone	Liver	Thyroid	Kidney	Lung	GI Tract
Contaminated ground	1.3-02	1.6-02	1.3-02	1.3-02	1.3-02	1.3-02	1.3-02	1.3-02
Inhalation	1.5-04	0.0	1.5-04	2.4-04	1.3-02	3.1-04	3.6-04	2.1-04
Fresh vegetation	4.8-04	0.0	1.2-03	9.5-04	1.1-01	1.2-02	1.2-04	5.1-04
Stored vegetation	2.8-03	0.0	7.7-03	5.2-03	6.5-03	6.5-02	1.3-03	1.9-03
Cow milk	3.0-03	0.0	5.3-03	6.4-03	3.5-01	4.7-03	8.5-04	1.5-03
Deer 1,603 m east	7.8-05	0.0	1.4-04	1.5-04	2.4-04	7.6-03	2.7-05	1.9-04
Total dose	2.0-02	1.6-02	2.7-02	2.6-02	4.9-01	1.0-01	1.6-02	1.7-02

* Analysis performed at maximum cow location is 2,350 m (7,710 ft) east-southeast.

NOTE: 1.3-02 = 1.3×10^{-2}

[1] All annual doses shown in Table 11A.1-10 reflect a power level of 3,323 MWt. Due to an operation at EPU/MELLLA+ at 3,988 MWt, each of the values shown must be multiplied by a factor of 1.79; the uprated values represent the maximum possible increase.

TABLE 11A.1-11

ANNUAL DOSES TO MAXIMUM INDIVIDUAL IN THE
CHILD GROUP FROM GASEOUS EFFLUENTS*

At Maximum Cow Location

(Annual Dose in mRem/yr)[1]

Pathway	Total Body	Skin	Bone	Liver	Thyroid	Kidney	Lung	GI Tract
Contaminated ground	1.3-02	1.6-02	1.3-02	1.3-02	1.3-02	1.3-02	1.3-02	1.3-02
Inhalation	1.5-04	0.0	2.1-04	2.3-04	1.6-02	2.9-04	3.1-04	1.7-04
Fresh vegetation	5.8-04	0.0	2.2-03	1.3-03	1.6-01	1.3-03	1.8-04	3.9-04
Stored vegetation	3.8-03	0.0	1.9-02	9.3-03	1.3-02	4.4-03	2.6-03	2.6-03
Cow milk	4.3-03	0.0	1.3-02	1.1-02	7.1-01	7.9-03	1.6-03	1.8-03
Deer 1,603 m east	8.6-05	0.0	2.5-04	2.0-04	3.6-04	9.5-05	4.0-05	1.2-04
Total dose	2.2-02	1.6-02	4.8-02	3.5-02	9.1-01	2.7-02	1.8-02	1.8-02

* Analysis performed at maximum cow location is 2,350 m (7,710 ft) east-southeast.

NOTE: 1.3-02 = 1.3×10^{-2}

[1] All annual doses shown in Table 11A.1-11 reflect a power level of 3,323 MWt. Due to an operation at EPU/MELLLA+ at 3,988 MWt, each of the values shown must be multiplied by a factor of 1.79; the uprated values represent the maximum possible increase.

TABLE 11A.1-12

ANNUAL DOSES TO MAXIMUM INDIVIDUAL IN THE
INFANT GROUP FROM GASEOUS EFFLUENTS*

At Maximum Cow Location

(Annual Dose in mRem/yr)[1]

Pathway	Total Body	Skin	Bone	Liver	Thyroid	Kidney	Lung	GI Tract
Contaminated ground	1.3-02	1.6-02	1.3-02	1.3-02	1.3-02	1.3-02	1.3-02	1.3-02
Inhalation	9.6-05	0.0	1.5-04	1.8-04	1.5-02	1.8-04	2.3-04	9.5-05
Cow milk	6.6-03	0.0	2.3-02	2.2-02	1.7+00	1.3-02	3.2-03	4.7-03
Total dose	2.0-02	1.6-02	3.6-02	3.5-02	1.7+00	2.6-02	1.6-02	1.8-02

* Analysis performed at maximum cow location is 2,350 m (7,710 ft) east-southeast.

NOTE: 1.3-02 = 1.3×10^{-2}

[1] All annual doses shown in Table 11A.1-12 reflect a power level of 3,323 MWt. Due to an operation at EPU/MELLLA+ at 3,988 MWt, each of the values shown must be multiplied by a factor of 1.79; the uprated values represent the maximum possible increase.

TABLE 11A.1-13

ANNUAL DOSES TO MAXIMUM INDIVIDUAL IN THE
ADULT GROUP FROM GASEOUS EFFLUENTS*

At Maximum Beef Animal Location

(Annual Dose in mRem/yr)[1]

Pathway	Total Body	Skin	Bone	Liver	Thyroid	Kidney	Lung	GI Tract
Contaminated ground	2.4-02	2.8-02	2.4-02	2.4-02	2.4-02	2.4-02	2.4-02	2.4-02
Inhalation	2.3-04	0.0	1.3-04	3.1-04	1.2-02	3.6-04	4.9-04	3.2-04
Fresh vegetation	1.6-03	0.0	2.7-03	2.4-03	2.6-01	2.3-03	2.4-04	1.7-03
Stored vegetation	5.7-03	0.0	8.4-03	7.4-03	7.9-03	3.1-03	1.3-03	3.0-03
Beef	9.7-04	0.0	1.7-03	1.4-03	1.5-02	8.1-04	3.2-04	4.4-03
Deer 1,603 m east	1.4-04	0.0	1.7-04	1.9-04	3.2-04	9.2-05	3.1-05	3.3-04
Total dose	3.3-02	2.8-02	3.7-02	3.6-02	3.2-01	3.1-02	2.6-02	3.4-02

* Analysis performed at maximum beef animal location is 1,693 m (5,555 ft) east.

NOTE: 2.4-02 = 2.4×10^{-2}

[1] All annual doses shown in Table 11A.1-13 reflect a power level of 3,323 MWt. Due to an operation at EPU/MELLLA+ at 3,988 MWt, each of the values shown must be multiplied by a factor of 1.79; the uprated values represent the maximum possible increase.

TABLE 11A.1-14

ANNUAL DOSES TO MAXIMUM INDIVIDUAL IN THE
TEEN GROUP FROM GASEOUS EFFLUENTS*

At Maximum Beef Animal Location

(Annual Dose in mRem/yr)[1]

Pathway	Total Body	Skin	Bone	Liver	Thyroid	Kidney	Lung	GI Tract
Contaminated ground	2.4-02	2.8-02	2.4-02	2.4-02	2.4-02	2.4-02	2.4-02	2.4-02
Inhalation	2.4-04	0.0	1.8-04	3.6-04	1.6-02	4.3-04	6.6-04	3.4-04
Fresh vegetation	1.1-03	0.0	2.5-03	2.2-03	2.1-01	3.2-02	2.3-04	1.2-03
Stored vegetation	6.3-03	0.0	1.5-02	1.3-02	1.3-02	1.7-01	2.4-03	4.0-03
Beef	6.1-04	0.0	1.4-03	1.1-03	1.1-02	9.2-02	2.6-04	2.5-03
Deer 1,603 m east	7.8-05	0.0	1.4-04	1.5-04	2.4-04	7.6-03	2.7-05	1.9-04
Total dose	3.2-02	2.8-02	4.3-02	4.1-02	2.7-01	3.3-01	2.8-02	3.2-02

* Analysis performed at maximum beef animal location is 1,693 m (5,555 ft) east.

NOTE: 2.4-02 = 2.4×10^{-2}

[1] All annual doses shown in Table 11A.1-14 reflect a power level of 3,323 MWt. Due to an operation at EPU/MELLLA+ at 3,988 MWt, each of the values shown must be multiplied by a factor of 1.79; the uprated values represent the maximum possible increase.

TABLE 11A.1-15

ANNUAL DOSES TO MAXIMUM INDIVIDUAL IN THE
CHILD GROUP FROM GASEOUS EFFLUENTS*

At Maximum Beef Animal Location

(Annual Dose in mRem/yr)[1]

Pathway	Total Body	Skin	Bone	Liver	Thyroid	Kidney	Lung	GI Tract
Contaminated ground	2.4-02	2.8-02	2.4-02	2.4-02	2.4-02	2.4-02	2.4-02	2.4-02
Inhalation	2.3-04	0.0	2.4-04	3.4-04	2.0-02	4.0-04	5.6-04	2.6-04
Fresh vegetation	1.2-03	0.0	4.5-03	2.8-03	3.3-01	2.6-03	3.1-04	7.8-04
Stored vegetation	7.6-03	0.0	3.6-02	2.2-02	2.6-02	9.2-03	4.4-03	4.4-03
Beef	7.9-04	0.0	2.5-03	1.4-03	1.7-02	8.8-04	4.4-04	1.6-03
Deer 1,603 m east	8.6-05	0.0	2.5-04	2.0-04	3.6-04	9.5-05	4.0-05	1.2-04
Total dose	3.4-02	2.8-02	6.7-02	5.1-02	4.2-01	3.7-02	3.0-02	3.1-02

* Analysis performed at maximum beef animal location is 1,693 m (5,555 ft) east.

NOTE: 2.4-02 = 2.4×10^{-2}

[1] All annual doses shown in Table 11A.1-15 reflect a power level of 3,323 MWt. Due to an operation at EPU/MELLLA+ at 3,988 MWt, each of he values shown must be multiplied by a factor of 1.79; the uprated values represent the maximum possible increase.

TABLE 11A.1-16

ANNUAL DOSES TO MAXIMUM INDIVIDUAL IN THE
INFANT GROUP FROM GASEOUS EFFLUENTS*

At Maximum Beef Animal Location

(Annual Dose in mRem/yr)[1]

Pathway	Total Body	Skin	Bone	Liver	Thyroid	Kidney	Lung	GI Tract
Contaminated ground	2.4-02	2.8-02	2.4-02	2.4-02	2.4-02	2.4-02	2.4-02	2.4-02
Inhalation	1.4-04	0.0	1.8-04	2.5-04	1.8-02	2.5-04	4.1-04	1.4-04
Total dose	2.4-02	2.8-02	2.4-02	2.4-02	4.2-02	2.4-02	2.4-02	2.4-02

* Analysis performed at maximum beef animal location is 1,693 m (5,555 ft) east.

NOTE: 2.4-02 = 2.4×10^{-2}

[1] All annual doses shown in Table 11A.1-16 reflect a power level of 3,323 MWt. Due to an operation at EPU/MELLLA+ at 3,988 MWt, each of the values shown must be multiplied by a factor of 1.79; the uprated values represent the maximum possible increase.

TABLE 11A.1-17

COMPARISON OF MAXIMUM CALCULATED DOSES FROM UNIT 2
WITH APPENDIX I DESIGN OBJECTIVES

Criterion	Appendix I Design Objective[1]	Unit 2 Calculated Dose
Gaseous effluents		
Gamma air dose[2], mRad/yr	10	7.9E-02
Beta air dose[2], mRad/yr	20	5.3E-02
Noble gas - total body[3], mRem/yr	5	3.9E-02
Noble gas - skin[3], mRem/yr	15	8.3E-02
Iodines and particulates[4]		
Any organ (thyroid), mRem/yr	15	2.8E+00
Liquid effluents		
Total body, mRem/yr	3	1.8E-00
Any organ[5], mRem/yr	10	5.1E+00

NOTE: 6.7E-02 = 6.7×10^{-2}

(1) Per reactor.
(2) Calculated at exclusion area boundary 1,603 m (5,259 ft) east.
(3) Calculated at 1,693 m (5,554 ft) east.
(4) Infant thyroid dose from cow milk 2,350 m (7,710 ft) east-southeast.
(5) Child bone dose is calculated to be the highest organ dose.

TABLE 11A.1-18

CALCULATED ANNUAL DOSES FOR
POPULATION WITHIN 80-KM (50-MI) RADIUS

	Total Body (man-Rem)	Thyroid (man-Rem)
Liquid Effluents		
Ingestion of potable water	1.2E-01	6.7E-02
Ingestion of fish	4.1E+00	9.0E-03
Shoreline recreation	6.8E-02	4.3E-02
Swimming	1.4E-04	8.8E-05
Boating	6.8E-05	4.3E-05
Total	4.3E+00	1.2E-01
Gaseous Effluents[3]		
Submersion	3.3-01	3.3-01
Inhalation	1.4-02	9.9-01
Standing on contaminated ground	7.3-02	7.3-02
Ingestion of fruits, grains, and vegetation	1.7-01	1.2+00
Ingestion of cow milk	4.8-02	6.8-01
Ingestion of meat	3.8-03	7.1-03
Total[4]	6.4-01	3.3+00

NOTES:
1. Based upon a projected 80-km (50-mi) population of 1.2+06 for the year 2010.
2. $4.0E-02 = 4.0 \times 10^{-2}$.
3. The values shown for "Gaseous Effluents" are based on a power level of 3,323 MWt. Due to an increase in the power level to 3,988 MWt, the "Submersion" values are to be multiplied by a factor of 1.37 and the rest by a factor of 1.66.
4. The "Total" values shown are based on a 3,323 MWt power level.

TABLE 11A.1-19

CALCULATED POPULATION DOSE COMMITMENT

(Contiguous U.S. Population Dose)

	Annual Dose Per Site	
	Total Body (man-Rem)	Thyroid (man-Rem)
Liquid effluents	4.3E+00	1.2E-01
Noble gas effluents[1]	1.14+00	1.37+00
Radioiodines and particulates*[2]	1.9+01	2.3+01
Total[3]	2.1+01	2.5+01

NOTE: 1.5E+00 = 1.5×10^{0}

(1) The annual dose values shown in Table 11A.1-19 are based on a power level of 3,323 MWt. Due to a power increase to 3,988 MWt, the values must be multiplied by 1.37. The uprated values represent the maximum possible increase.

(2) The annual dose values shown in Table 11A.1-19 are based on a power level of 3,323 MWt. Calculation disposition UE-005-01A and calculation change notice ECP-12-000448-CN-008 UE-005-01.00 provide the required information to adjust for an EPU level of 3,988 MWt.

(3) The "Total" values represent doses at 3,323 MWt. To adjust for 3,988 MWt (except for "Liquid effluents" which has already been incorporated), adjust the values as noted in notes (1) and (2) above.

* Carbon-14 and tritium have been added to this category, applicable for the 3,323 MWt power level values.

11A.2 COST-BENEFIT ANALYSIS

This section presents the results of cost-benefit analyses performed in accordance with Section II.D of 10CFR50 Appendix I.

Augments to the liquid and gaseous effluent systems and respective potential reductions to the annual population exposure are taken from RG 1.110. The beneficial savings of each augment were calculated by multiplying the calculated dose reduction by $1,000/man-Rem or $1,000/man-Rem/thyroid.

Augments to the Liquid Effluent Treatment System

Table 11A.2-1 presents the calculated base case (updated for EPU) annual total body dose (man-Rem) and thyroid dose (man-Rem/thyroid) associated with the operation of the plant LWS 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 $1,400 for reducing man-Rem exposures to zero and $61 for reducing man-Rem/thyroid exposures to zero.

In an analysis of the annualized procurement, installation, operation, and maintenance costs, the least expensive liquid radwaste augment was found to be $20,000/yr 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.

Augments to the Gaseous Effluent Treatment System

Table 11A.2-2 presents the calculated base case (updated for EPU) annual total body dose (man-Rem) and thyroid dose man-Rem/thyroid 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 $640 for reducing man-Rem exposures to zero and $3,300 for reducing man-Rem/thyroid exposures to zero.

In an analysis of the annualized procurement, installation, operation, and maintenance costs, the least expensive gaseous radwaste augment was found to be $10,580/yr 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.

TABLE 11A.2-1

BASE CASE ANNUAL POPULATION DOSES
DUE TO LIQUID EFFLUENTS

Pathway	Total Body Dose (man-Rem)	Thyroid Dose (man-thyroid-Rem)
Ingestion of fish	4.1E+00	9.0E-03
Ingestion of potable water	1.2E-01	6.7E-02
Shoreline recreation	6.8E-02	4.3E-02
Swimming	1.4E-04	8.8E-05
Boating	6.8E-05	4.3E-05
Total	4.3E+00	1.2E-01

NOTE: 1.4E+00 = 1.4×10^{0}

TABLE 11A.2-2

BASE CASE ANNUAL POPULATION DOSES
DUE TO GASEOUS EFFLUENTS

Pathway	Total Body Dose (man-Rem) [1]	Thyroid Dose (man-thyroid-Rem) [1]
Submersion	3.3-01	3.3-01
Inhalation	1.4-02	9.90-01
Standing on contaminated ground	7.30-02	7.30-02
Ingestion of fruits, grains, and vegetation	1.70-01	1.20+00
Ingestion of cow milk	4.8-02	6.80-01
Ingestion of meat	3.8-03	7.10-03
Total [2]	6.40-01	3.30+00

NOTE: 3.3-01 = 3.3×10^{-1}

[1] The values for dose shown in Table 11A.2-2 are based on a power level of 3,323 MWt. Due to an increase in the power level to 3,988 MWt, the "Submersion" values are to be multiplied by a factor of 1.37 and the remaining values by a factor of 1.66.

[2] The "Total" values shown are based on a 3,323 MWt power level.