CHAPTER 11

RADIOACTIVE WASTE MANAGEMENT

11.1 SOURCE TERMS

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 are presented in Table 11.1-1 and are based on methods which are consistent with NUREG-0016, Rev. 1 and represent failed fuel conditions corresponding to an off gas release rate of 50,000 μCi/sec at 30 min delay[10]. Parameters used to determine the expected source terms are listed in Table 11.1-2.

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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 system failure analyses design basis fission product source terms. Design failed fuel conditions correspond to an off gas release rate of 100 μCi/sec/MWt, or 304,000 μCi/sec, at 30 min delay and are developed by scaling up from expected NUREG-0016 source term data. In addition, the design source terms in Table 11.1-1 take into consideration plant operation with hydrogen water chemistry and the GE design basis data described in Section 11.1.1 and Tables 11.1-3, 11.1-5, 11.1-6, 11.1-7, and 11.1-8. Table 11.1-1 presents the higher concentrations for a given isotope from either the adjusted NUREG-0016 data or the GE data, and thus, represents a conservative data set for use in design basis evaluations. The NUREG-0016 data reflects a halogen (iodine and bromine) carry-over fraction of 1.5% for determining the normal water chemistry reactor water halogen concentrations and a conservative application of 4% carry-over in determining the hydrogen water chemistry reactor steam halogen concentrations.
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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) over the past decade. 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 off gas. The various radioisotopes listed 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 off gas at several stations through mid-1971. Emphasis was placed on observations made at KRB and Dresden

Unit 2. The design basis radioactive material levels do not necessarily include all the radioisotopes observed or predicted theoretically 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.

For halogens, radioisotopes with half-lives less than 3 min were omitted. For other fission product radioisotopes in reactor water, radioisotopes with half-lives less than 10 min were not considered.

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 miniscule 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:

Equilibrium: $R_g \approx K_1 y$

Recoil: $R_g \approx K_2 y \lambda$

The terms in these and succeeding equations are defined in the nomenclature section. The constants K and K 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 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 and Dresden Unit 1 experience, it was assumed that noble radiogas leakage from the fuel would be the equilibrium mixture of the noble radiogases present in the fuel.

Vallecitos Boiling Water Reactor and early Dresden Unit 1 experience indicated that the actual mixture most often observed approached a distribution which was intermediate in character to the two extremes[1]. 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 mechanistic basis for the calculational methods possible for equilibrium and recoil mixtures. However, this diffusion distribution pattern is described as follows[2]:

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

The constant K describes the fraction of total fissions that are involved in the release. As can be seen, 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 was 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 was consistent with the nominal design basis 30-min off gas hold up system used on 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 0.1 Ci/sec (t=30 min) 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[9].

While the noble radiogas source term magnitude was established at 0.1 Ci/sec (t=30 min), 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:

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

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 $(1 - e^{-\lambda T})$ approaches 1 and can be neglected when the reactor has been operating at steady state for long periods of time. The term $(e^{-\lambda T})$ 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:

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

Equation 11.1-3 represents a straight line when log (R_g/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 (Fig. 11.1-1). As shown in 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 ΣR_g at 30 min = 100,000 μCi/sec, K_g is 2.6×10^7 and Equation 11.1-1 becomes:

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

This updated noble radiogas source term mixture has been termed 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. While Kr-85 can be calculated using Equation 11.1-4, the number of confirming experimental observations was limited by the difficulty of measuring very low release rates of this isotope. Therefore, the table provides an estimated range for Kr-85 based on actual measurements.

Radiohalogen Fission Products

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Normal operational releases to the primary coolant are expected to be approximately 25,000 μCi/sec of the 13 commonly considered noble gases, as evaluated at 30 min, and 100 μCi/sec of I-131. These values can be compared to the design base value of 100,000 Ci/sec for the summation of the same 13 and 700 μCi/sec for I-131. Table 11.1-4 lists the source terms released to the reactor pressure vessel as a consequence of a Power Isolation Event (MSIV closure at power), which is the only anticipated operational occurrence in which significant activity is expected to be released.

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Historically, the radiohalogen design basis source term was established by the same equation as that used for noble radiogases. In a fashion 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 (Fig. 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. Fig. 11.1-3 relates KRB and Dresden Unit 2 noble radiogas releases versus I-131 leakage. While 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 μ Ci/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 μ Ci/sec were not experienced. Fig. 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 which would be indicated by I-131 leakage from the fuel in excess of 700 μ Ci/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 μ Ci/sec. This value (Fig. 11.1-3) is consistent with the experience 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 x 10^7, and the halogen radioisotope release can be expressed by the following equation:

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

Concentrations of radiohalogens in reactor water are listed in Table 11.1-5 and can be calculated using the following equation:

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

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

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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 by weight of reactor water in steam with a standard deviation of ±0.9. For normal water chemistry, in the present source term definition, a radiohalogen carryover design basis of 2 percent (0.02 fraction) is used.

Hydrogen water chemistry has been implemented to control the potential for stress corrosion cracking of vessel internals. Hydrogen is injected into the feedwater to reduce the radiolytic production of oxygen and hydrogen peroxide and to promote recombination of residual oxidants. Under HWC and conditions of low copper in the feedwater, the carry-over fraction for halogen isotopes can be enhanced as is reported in Reference 11.1-10. This will reduce the activities of iodine isotopes in the reactor water and increase their activities in the steam. To bound operation with and without hydrogen injection, reactor water halogen activities are calculated for normal water chemistry assuming 2% carry-over.

GE has determined the NWC iodine carry-over in the steam at the RBS to be 2.9%. Assuming that RBS will behave like the plant with similar feedwater copper concentrations studied in Reference 11.1-10, the estimated halogen carry-over with hydrogen injection is expected to increase to 3%, representing an operationally anticipated increase of 3.5% from normal to hydrogen water chemistry operating conditions.
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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 experience data (Table 11.1-6). 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.

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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} μ Ci/sec or less which is below the maximum concentration in drinking water applicable to continuous use by the general public. The concentration of alpha emitting plutonium radioisotope 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 Cm-242 level.

Nomenclature

The following nomenclature table defines the terms used in equations for source term calculations:

R_g = Leakage rate of a noble radiogas isotope (μCi/sec)

R_h = Leakage rate of a halogen radioisotope (μCi/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 = A constant establishing the level of noble radiogas leakage from fuel

K_h = A constant establishing the level of radiohalogen leakage from fuel

C_h = Concentration of a halogen radioisotope in reactor (μCi/g)

M = Mass of water in the operating reactor (g)

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

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

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

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

11.1.1.2 Activation Products

Coolant Activation Products

The coolant activation products are not adequately correlated by simple equations. Design basis concentrations in reactor water and steam have been estimated conservatively from experience data. The resultant concentrations are listed in Table 11.1-7.

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Under conditions of hydrogen water chemistry enhanced evolution of nitrogen to steam is experienced in a non-linear fashion as reported in EPRI Report TR-103515 (Reference 11.1-12). Measurement data from hydrogen water chemistry tests at various BWR plants suggests that the dose rate increase is a function of feedwater hydrogen concentration, which in turn is controlled by the hydrogen injection rate. The onset of the nitrogen (N-16) activity increase usually occurs at a feedwater hydrogen concentration above 0.35 ppm. The rise of the main steam activity usually reaches a plateau at around 1.5 ppm. For all BWR plants surveyed, the maximum normalized main steam activity increase over normal water chemistry concentrations conditions is less than a factor of six (6) with feedwater concentrations up to 2 ppm. In Table 11.1-7 other isotopes is not characterized overwhelming dominance of nitrogen in the radiation signature of the steam. Therefore, the concentrations of these isotopes are not assumed to be changed in any significant manner under hydrogen water chemistry. In addition, though the water concentrations of nitrogen will decrease by approximately 10-20% under hydrogen water chemistry, the decrease is ignored for the purposes of conservatively bounding normal and hydrogen water chemistry operating conditions.
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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 experience data[8]. The resultant concentrations are listed in Table 11.1-8. 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

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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:

$$R_{act} = \frac{\Sigma \phi V \lambda}{3.7 \times 10^4 P}$$

Where:

R_{act} = Tritium formation rate by deuterium activation (Ci/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 (1.78×10^{-9} sec^{-1})

P = Reactor power level (MWt)

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

$(1.3 \pm 0.4) \times 10^{-4}$ µCi/sec/MWt.

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 fuel rods[4].

This is confirmed by the study made at Dresden Unit 1 in 1968 by the U.S. Public Health Service[5] (USPHS) which suggests that essentially all of the tritium released from the plant could be accounted for by the deuterium activation source. For purposes of estimating the leakage of tritium from defective fuel, we can make the assumption that it leaks in a manner similar to the leakage of noble radiogases. We can thus use the empirical relationship described as the diffusion mixture, used for predicting the source term of individual noble gas radioisotopes, as a function of total noble gas source term. The equation which describes this relationship is:

$$R_{diff} = Ky\sqrt{\lambda}$$

Where:

R_{diff} = Leakage rate of the radioisotope (μCi/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 μCi/sec after 30-min decay, we would calculate leakage from the fuel to be about 0.24 μCi/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 μCi/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 μCi/sec.

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For a 3091-MWt reactor the total tritium appearance rate in reactor coolant and release rate in effluents can be estimated to be about 17 Ci/yr.
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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.

Off gases 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 off gas system forms water vapor which is condensed and returned to the main condenser. This reduces the amount of tritium leaving in gaseous effluents and result 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[5]. Effort 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.

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 References 2, 3, and 6.

11.1.3 Process Leakage Sources

Process leakage results in potential release paths for noble gases and other volatile fission products via ventilation systems. Liquid from process leaks are all 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 off gas from operating BWR plants. However, because the implementation of improved process off gas treatment systems makes the ventilation release comparatively significant, GE has conducted measurements to identify and qualify these low-level release paths. GE has maintained an awareness of other measurements by the Electric Power Research Institute and other organizations and routine measurements by utilities with operating BWRs. 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 partition 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 here defined 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 and the main condenser mechanical vacuum pump have been compiled and evaluated in NEDO-21159, Airborne Releases from BWRs for Environmental Impact Evaluations, March 1976, Licensing Topical Report[7]. This report is periodically updated to incorporate the most recent data on airborne emissions. The results of these evaluations are based on data obtained by utility personnel and special in-plant studies of operating BWR plants by independent organizations and GE. The results are summarized in Section 12.2.

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 Off Gas System

The radioactive sources for the off gas 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.

References - 11.1

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11.2 LIQUID WASTE MANAGEMENT SYSTEM

11.2.1 Design Bases

11.2.1.1 Power Generation Design Bases

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The objective of the radioactive liquid waste (radwaste) system is to collect, monitor, and process for reuse or disposal all potentially radioactive liquid wastes in a controlled manner so that the operation or availability is not limited. The liquid waste subsystems are grouped as equipment and floor drains, regenerative waste and phase separator/backwash tank. The system provides for maximum recycle of water to the condensate storage tanks. Sufficient treatment and diversity of types of equipment is available to process normal plant waste to condensate quality. Discharge of processed liquid wastes will only be necessary to ensure the radwaste system operability to support plant operating conditions or to control tritium buildup in the condensate storage tank. Discharge of processed liquid waste to the environs is via the cooling tower blowdown line. Maximum recycle of waste water results in a radwaste material release which conforms to the "as low as reasonably achievable" (ALARA) requirements of 10CFR50, Appendix I.
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The radioactive liquid waste system has the capacity and capability of processing the anticipated quantities and activities of liquid wastes resulting from normal operation.

The system includes the proper selection of equipment and instrumentation to ensure that the radioactivity concentrations resulting from liquid discharges from the plant are within the limits set forth in 10CFR20.

Refer to Appendix 11A for discussion of the liquid radwaste system conformance to the design objectives of 10CFR50, Appendix I.

11.2.1.2 System Design Bases

The liquid radwaste system is designed to maintain safe operating conditions by minimizing radiation hazards to plant personnel by applying meaningful design techniques to each subsystem. The system collects, processes, stores, monitors, and disposes of all liquid radioactive wastes received from the reactor coolant system or liquids which are potentially contaminated due to contact with liquids from the reactor coolant system. Equipment is selected, arranged, and shielded to permit operation, inspection, and maintenance with personnel radiation exposure as low as reasonably achievable and within the limits as delineated in 10CFR20.

Equipment is selected requiring minimum maintenance. Sumps, pumps, valves, and instruments are located in accessible areas. Tanks and processing equipment which may contain significant quantities of radioactivity are to the greatest extent possible appropriately shielded from each other and from personnel access areas and controls or equipment requiring regular maintenance. Air-operated valves conveying highly radioactive fluids are located in shielded compartments and are provided with rack-mounted solenoid operators outside the high radiation fields.

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In some cases several storage tanks are located in a common shielded area. Since all regularly maintained equipment (pumps, valves, etc.) associated with these tanks is located outside the shielding in accessible areas, no regular maintenance is anticipated within the shielded tank compartments. In all cases, excess tank capacity is provided for normal operation. If an operational problem occurs within a compartment, the tank can be isolated without loss of unit operating capability. The liquid radwaste system is divided into subsystems that combine various sources of liquid wastes with similar conductivity and isotopic concentrations for appropriate processing. Major flow paths, equipment, leakage and drain liquids are indicated in Fig. 11.2-1a through 11.2-1k.
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Selection of the equipment's type and size for the radwaste facility at the River Bend Station was done on the basis of both previous operating experience and demonstrated reliability in other industrial applications. Table 11.2-1 lists the equipment design data of the liquid radwaste facility. Surge capacity in tankage is provided to cover contingencies such as processing equipment outages, or abnormal evolutions resulting in the production of excessive waste volumes. Tank volume provided is in excess of that required with waste inputs into the liquid radwaste system based on NUREG-0016, Rev. 1, Table 1-4. Surges can be accommodated by extended operation. This is described in detail in Section 11.2.2.

Laundry may be processed in two different ways. It can be either wet washed or dry cleaned, on or offsite.

11.2.1.2.1 Applicable Codes and Standards

Table 11.2-2 lists the applicable codes and standards for equipment in the liquid radwaste systems. The basis for Safety Class NNS and the material selections for equipment in the liquid radwaste systems is the criteria as established by Regulatory Guide 1.143, as discussed in Section 11.2.1.3.2.

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 for Testing and Materials Specification No. ASTM D3299-74.

11.2.1.2.2 Structural Design

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The radwaste equipment arrangement is presented in Fig. 1.2-29 through 1.2-32 and is described in Section 3.8.4.1.8. In accordance with Regulatory Guide 1.143 and Branch Technical Position ETSB 11-1 (Revision 1, 4/75), the building is seismically analyzed as described in Section 3.7.2.16A. The radwaste building layout provides design features consistent with maintaining personnel exposure ALARA.

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The compartment containing the waste collectors and, the floor drain collector tanks, and the regenerative waste tanks is watertight to an elevation above that resultant from a catastrophic failure of all tanks within the compartment. This is accomplished by providing watertight sleeves for all piping penetrating the compartment walls and restricting personnel accesses to levels above the resultant elevation identified previously.

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11.2.1.3 Compliance with Design Criteria and Standards

11.2.1.3.1 General Design Criteria (GDC)

Control and monitoring of radioactive liquid releases in accordance with the requirements of GDC 60 is described in Section 11.2.2.5. Liquid waste holdup capacity, and the basis therefor, is described in Section 11.2.2.

Monitoring of effluent discharged from the liquid radwaste system is described in Section 11.2.2.6. The process and effluent radiation monitoring systems are described in Section 11.5.

11.2.1.3.2 Regulatory Guides

Regulatory Guide 1.143 identifies the quality level, quality group classification (safety class), seismic requirements, and material requirements for equipment and structures containing radioactive waste. The liquid radwaste system is in conformance with the guide in that:

1. The equipment within the system is designed in accordance with requirements identified in Table 1 of the guide.

2. Materials for pressure-retaining components conform to the requirements of the specification for materials in Section II of the ASME Code, except that nickel alloy steel, fiberglass-reinforced plastic (FRP) piping, High Density Polyethylene (HDPE) piping and polypropylene-lined (PPL) steel piping are used. The use of FRP piping is restricted to condensate flush connections to liquid radwaste piping and components. HDPE piping is only used to handle condensate quality water that is being discharged. PPL piping is used in the demineralizer systems because of its improved corrosion resistance.

3. Foundations and walls of the radwaste building, to an elevation above that sufficient to contain the maximum liquid inventory expected in the building, are designed to the seismic criteria described in the guide. This is described in detail in Section 3.8.

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4. Tanks are provided with means to monitor and alarm abnormal liquid levels. The condensate storage tank, which is located outside, is provided with an overflow piped to a sump. The sump is provided with two pumps which discharge to the radwaste system. An analysis has shown (Section 15.7.3.2) that the dose consequences to surrounding or downriver population associated with a tank failure are negligible. Therefore, the condensate storage tank is not provided with dikes.

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5. Indoor tanks are in compartments or areas provided with intermediate sumps designed for the handling of radioactive wastes and for having provision for discharge of the wastes to the liquid radwaste system.

11.2.1.3.3 NUREGs

NUREG-0016, Rev. 1, provides guidance in calculational methods associated with liquid radwaste systems. The liquid radioactive waste tankage provided is in excess of that required based on twice the single unit daily expected average input flows (Table 1-4, p 1-11 of NUREG-0016, Rev. 1). Equipment decontamination factors (DFs) used in analysis are consistent with Table 1-5 of the NUREG.

11.2.2 System Description

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The liquid radwaste system consists of one major subsystem plus one minor subsystem as shown in Fig. 11.2-1a through 11.2-1k. A simplified flow diagram of the liquid radwaste system is shown in Fig. 11.2-2. The collection tanks, process equipment, and sample tank systems are located in the radwaste building.

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Overall material and activity balances for the radwaste system are listed in Table 11.2-3 and the derivation bases are provided in Section 11.2.2.6. The isotopic concentrations for release to the environs are given in Tables 11.2-4 and 11.2-5. All equipment and systems have been designed to process the maximum anticipated waste volume within a 24-hr period.

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Over 172,400 gal of usable storage capacity is available to collect and store the average peak liquid radwaste influent volumes of approximately 40,280 gpd, with in-process inventory at approximately 10,000 gal. The daily average radwaste system inleakage may vary by as much as 7200 gpd when comparing the summer months to winter months. This is due to the various building unit coolers removing condensation from the atmosphere. Radwaste total system inleakage averaged over an entire year is expected to be approximately 36,680 gpd with a low average of 33,080 gpd and high average of 40,280 gpd. Storage facilities are also available to receive the treated liquids. The treated liquid bulk storage capacity is 68,800 gal. The treated liquid is discharged to the environment when necessary to ensure the radwaste system operability to support plant operating conditions and to control tritium buildup in the condensate storage tank to keep offsite releases as low as reasonably achievable, and to control total plant water inventory when necessary.

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Briefly, the waste and floor drain collector subsystem consists of the waste, regenerant and floor drain collector tanks which receive and store the feed to the subsystem; the radwaste filter which removes insoluble particles; a radwaste treatment train which removes the soluble and colloidal ionic material, and the recovery sample tanks which hold the processed water for testing before it is returned to the condensate system, discharged to the cooling tower blowdown, or transferred back to the waste collector tank inlet header for reprocessing.

•→11 •→1

In addition to the processing subsystem, a supplementary system is included in the liquid radwaste facility. This is the phase separator/backwash tank system which provides storage and holdup of filter sludges and spent resins from the reactor water cleanup filter/demineralizers, the condensate polishers, the radwaste treatment vessels, the fuel pool demineralizers, and the radwaste filters. Following a period which allows for sufficient decay of the radionuclides and settling of solids, the contents of the phase separator/backwash tanks may be transferred directly to a solid waste system or to a waste sludge tank for processing and then disposal in the solid waste system (refer to Section 11.4).

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All liquid radwaste equipment is located in the radwaste building. Major equipment included in the liquid radwaste system is listed below (see Table 11.2-1 for equipment design parameters):

2. Waste and Floor Drain Collector Subsystem

   a. Four waste collector tanks and two pumps

   b. Three floor drain collector tanks, two pumps, and one plate-type oil separator

   c. Two radwaste filters

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   d. Two trains of radwaste

   e. Four recovery sample tanks and three pumps

•→3

   f. Two chemical regenerant tanks/chemical waste tanks and one pump.

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2. Phase Separator/Backwash Tank Subsystem

   a. Two phase separator tanks and two pumps

   b. One backwash tank and two pumps

11.2.2.1 Summary

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The input waste streams to the various subsystems of the liquid radwaste system are identified on Fig. 11.2-1a through 11.2-1k. Average flow rates are presented in Table 11.2-3 for the primary flow streams shown in these figures. A composite summary of expected and design values for the waste stream release to the environment is given in Tables 11.2-4 and 11.2-5, respectively.

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11.2.2.2 Waste and Floor Drain Collector Subsystem

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Relatively low conductivity (less than 50 S/cm) and variable activity level wastes are stored in the waste collector tanks. The tank influents include low conductivity drains from piping and equipment that cannot be returned directly to the condenser hotwell, wastes from the reactor coolant, condensate and feedwater systems, and other associated auxiliaries. Influents to the tanks also include decanted liquids from the phase separator tanks, condensate demineralizers resin rinse water, condensate storage tank overflow, decontamination and chemistry laboratory drain and ultrasonic resin cleaner wastes.

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Radioactive materials are removed from the input wastes by filtration (insolubles and organic removal) and ion exchange (soluble and colloidal removal). Radwaste treatment train effluent is then routed to the recovery sample tanks for transfer to the condensate storage tank, the waste collector tank inlet header for reprocessing through the radwaste treatment train, or the cooling tower blowdown line for discharge. Prior to discharge into the cooling tower blowdown line, this waste is checked for activity by a radiation monitor. Liquids with radioactivity levels exceeding specified limits or with unacceptable chemistry may be recycled back to the waste collector tanks for further processing. The radwaste deep bed filters are provided for removal of insolubles. Two identical, cross connecting filters are provided so that maintenance may be performed on one filter or a failure of one would not inhibit daily liquid radwaste processing.
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The radwaste treatment train includes a three pressure (treatment) vessels with effluent retention elements in series. Depleted treatment media is flushed out of the pressure vessel with air and water and directed, via the phase separator/backwash tank subsystem (Section 11.2.2.5), to the radioactive solid waste system (Section 11.4). Four waste collector tanks are provided with a total capacity of 90,520 gallons. This volume is significantly in excess of that required to accommodate 1 day's influent to the waste and floor drain collector subsystem. However, should operational requirements dictate, the influent can be diverted to the floor drain collector tanks.
•→3
Mobile, portable filter/demineralizers may be used whenever necessary for special applications or temporary replacement of an installed process train. Space is available for these portable filter/demineralizers in a spare cubicle, on elevation 117 ft of the radwaste building, provided with curbs, floor drains, and 2-ft thick concrete walls. Valved piping connections are provided in the cubicle for bypassing the filter only, the demineralizer only, or the entire filter/demineralizer train. This arrangement meets the intent of Regulatory Guide 1.143.

Additional connections and interfaces with the radwaste system, drain system, service air system, condensate demineralizer system, and 480 VAC power are provided to support a vendor-supplied filtration system to supplement installed process trains. An area for vendor skid-mounted equipment is provided on 65’ elevation of the Radwaste Building. The area is bermed to contain any spills and radiation shielding is provided where necessary.

Process rates may be varied depending upon the desired results of processing (e.g. treatment of impurities or increased processed water volume). Maximum train flow can be expected to be 100 gpm.
•→11
Potentially high conductivity liquid wastes (50 umho/cm and greater) from the radwaste building sumps, reactor building floor drain sumps, auxiliary building floor drain sump, fuel building floor drain sumps, turbine building floor drain sumps, and shop floor drain sumps are collected in the floor drain collector tanks. Influent to the tanks also includes the waste solidification/dewatering stream. The liquid waste is treated as necessary using either filter with either process train and then recovered, discharged, or returned to the waste collectors subsystem for reprocessing. Section 11.2.2.6 gives the disposition guide for control of waste activity movement.
3←• 10←• 11←•

Three floor drain collector tanks are provided with a total capacity of 64,800 gallons. This volume is significantly in excess of that required to accommodate 1 day's influent to the waste and floor drain subsystem. However, should operational requirements dictate, the influent can be diverted to the waste collector tanks.

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The floor drain collector stream can be processed through the radwaste filter(s) and treatment trains at varied rates depending on media selection and desired results. Maximum train flow can be expected to be 100 gpm.

11.2.2.3 Phase Separator/Backwash Tank Subsystem

•→16

Filter sludges, slurries, and spent resins are collected, decanted, and conveyed to the radioactive solid waste system (Section 11.4) by the phase separator/backwash tank subsystem. The phase separator tank influents include filter sludges (powdered resin and crud) produced during the operation of the reactor water cleanup system filter/demineralizers. The backwash tank influent, discussed below, can also be diverted to either of the phase separator tanks and to the waste sludge tank.

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Normally one phase separator tank is in service and one is isolated to permit the short-lived isotopes to decay prior to processing through the radioactive solid waste system. Sufficient time exists between influent batches entering the inservice phase separator for the batch to settle and the decant to be drawn off. The phase separation tank pump transfers the liquid phase to the waste collector subsystem, and the concentrated sludge and expended resin is pumped directly to the radioactive solid waste system.

The backwash tank accepts filter backwashes from the waste collector and floor drain influent strainers and filters, the spent fuel pool and suppression pool cleanup filters, spent resins from the condensate, radwaste, suppression pool cleanup and fuel pool treatment vessels. These influents can be diverted from the backwash tank to either of the phase separator tanks via three-way valves in the backwash tank inlet header. Operation of the backwash tank is similar to the phase separator tanks in that solids are allowed to settle between influent batches to the tank and the decant transferred to the floor drain and waste collector subsystem. The concentrated sludge and expended media is sent directly to the radioactive solid waste system for processing.

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11.2.2.4 System Operational Analysis

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

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Decontamination factors (DFs) of the processing units can be expected to be as follows (depending on the treatment media selected):

Equipment	DF	Remarks
Radwaste media filter	1	For corrosion/activation products (insolubles)
Radwaste treatment vessels		
1. Waste collector stream	100/1,000	DF for Cs, Y, Nb, Zr/ DF for other isotopes (solubles)
2. Floor drain collector stream		DF for Cs, Y, Nb, Zr/ DF for other isotopes (solubles)

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11.2.2.5 Instrumentation and Control

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The purpose of the instrumentation and control system is to provide indications of process operation, equipment performance system status, and provide central control of process equipment. The radwaste control panel is located in the auxiliary control room.

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Instrumentation for the different subsystems is described separately in the following sections.

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In multiple tank subsystems, e.g., floor drain collector, recovery sample, regenerative waste and phase separator tanks, the inlet valves for each tank system with the exception of the waste collection tanks are interlocked such that at least one valve must be open at all times. For the waste collection tanks, administrative controls are relied upon in lieu of an interlock. This ensures that at least one of each type of tank is available to accept influents.

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11.2.2.5.1 Waste and Floor Drain Collector Subsystem

Control switches are provided on the radwaste control panel for manual operation/automatic shutdown of the waste and floor drain collector tank pumps. The tank levels are indicated and annunciated on the radwaste control panel. Since two waste collector pumps are provided for four tanks, and two floor drain collector pumps are provided for three tanks, automatic shutdown of the operating pump is obtained by interlocking the low tank level switch with the tank outlet valve position and the pump suction valve position, such that either pump can be shut down by any tank low level by the operator establishing a suction flow path. Pump high or low discharge pressure conditions are annunciated on the control panel.

Control switches are provided for either manual or automatic operation of the pump discharge air-operated valves.

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An automatic backflushable duplex strainer assembly is provided immediately upstream of the radwaste filter. This strainer may be backflushed when a predetermined differential pressure is attained. Strainer internals may be removed under administrative controls. A polyelectrolyte (cationic polymer) which tends to agglomerate suspended solids is provided to maximize filter performance. Polyelectrolyte injection into the filter influent can be set manually or operated in automatic. Excessive turbidity or total organic carbon (suspended or dissolved oil) in the filter effluent can result in the effluent being recycled back to the waste collector tanks. A filter flow rate of 100 gpm can be maintained over the range of filter differential pressures by the use of a flow controller. Filter backwash can be automatically initiated by a high differential pressure across the filter vessel when the filter is operated in the automatic mode. When operated manually, the filter may be backwashed manually or automatically when the differential pressure across the filter is observed to be high.

6←•

Conductivity, flow, pressure, and differential pressure indication may be used to allow evaluation of media performance. The treatment vessel effluents are monitored and/or tested for depletion of the treatment media selected.

The recovery sample tanks are provided with high and low level alarms; the high alarm gives the operator time to make an empty tank available for receiving influent, and the low level alarm shuts down the operating discharge pump. The pumps are operated on a remote manual basis. Pump discharge pressures are indicated on the radwaste control panel. The recovery sample tank contents can be transferred to the condensate storage tanks; however, should it become necessary, the water can be reprocessed or discharged via the cooling tower blowdown line, under strict administrative controls.

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A control switch is provided for manual operation of the recovery sample tank discharge valve to the cooling tower blowdown line. Interlocks prevent the valve from being opened if a high radiation level is present.

A control switch is provided for either manual or automatic operation of the recovery sample tank discharge diverting valve. In the automatic mode, the valve is opened when a high radiation level is sensed, diverting the recovery sample tank contents back to the waste collector tank.

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The recovery sample tank discharge flow rate to the cooling tower blowdown is maintained at its set point by a flow control valve operated from the radwaste control panel. During liquid releases the flow rate and temperature are continuously recorded. The activity level information is maintained by the radiation monitor.

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11.2.2.5.2 Phase Separator/Backwash Tank Subsystem

Each phase separator tank has level switches as follows:

Four pump suction paths are available per tank, the upper and lower decant connections and the tank outlet connection. The bubbler-type tank level transmitter has switches associated with each tank connection. Pump operation is interlocked such that a suction flow path must be established and the tank level must be above that suction point prior to establishing a pump start permissive. A fourth suction path, a flush connection for the pump suction piping, is also provided. This is similarly interlocked such that the open suction flush valve results in a start permissive. Since two-phase separator pumps are provided for the two tanks, automatic shutdown is obtained in a similar manner to that used for the other tank/pump subsystems.

An offline turbidity analyzer is located in the pump common discharge header. This analyzer is provided to ensure that the turbidity of the decant from the tanks is acceptable for transfer to the waste collector tanks. If unacceptable, an alarm warns the operator to either terminate the transfer or divert to another flow path.

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The phase separator tank sludge and expended resin can be sent to either the waste sludge tank or directly to a liner for reprocessing.

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The backwash tank has level switches similar to those provided for the phase separator tank, i.e., for the two decant connections and the tank outlet connection. Similarly, the pump suction flush valves are interlocked with the pumps. Two pumps are provided for the backwash tank with the operating pump shut down automatically by tank level.

An offline turbidity analyzer is located in the pump common recirculation line. This analyzer is provided to ensure that the turbidity of the water from the backwash tank, if decanted, is acceptable for transfer to the floor drain collector tanks. If unacceptable, an alarm warns the operator to either terminate the transfer or divert to another flow path.

The backwash tank sludge and expended media is handled in a manner similar to that for the phase separator. Normally, however, the entire contents of the backwash tank are transferred directly to the radioactive solid waste system.

11.2.2.5.3 Power Sources

Instruments are powered from the 120-V instrument bus and receive operating air from the 100 psig instrument air system.

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Motors and control circuits are powered from normal 480-V ac supplies.

11.2.2.6 Operating Procedures

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The operating procedures used for all liquid radwaste equipment are based on batch processing through the radwaste systems. This type of operation allows a 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 control limit.

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The filters in the waste and floor drain processing subsystem are designed for operation at 100 gpm maximum. When the pressure drop across the filter reaches a preselected level, the flow is discontinued. The filter is then backwashed to the backwash tank. During filter maintenance, the other filter can be used with either process train.

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The radwaste process trains are operated at varied rates depending on the desired processing results. Maximum process flow can be expected to be 100 gpm. There are three demineralizers in series composed of a cation, an anion, and a mixed bed unit. The feed is the effluent from the radwaste filter. Each process vessel is monitored for pressure drop across the bed, conductivity of the effluent, and other key chemistry parameters dependent upon treatment media selection. Media is selected to maximize recycle of water back to the Condensate Storage Tank and minimize offsite dose (ALARA) should a discharge of radioactive liquid water be necessary. Media is also evaluated in system compatibility. This will ensure expected corrosion rates are essentially unchanged and system integrity is maintained. When one or more of the key chemistry parameters is exceeded, flow is stopped and the treatment media is replaced. This ensures maximum recycle of water to the Condensate Storage Tank as well as minimizes the dose to the general public (ALARA). Section 11.2.2.2 contains discussions on the use of a portable filter/demineralizer system when necessary for special applications or temporary replacement of an installed process train and available capability for utilizing an ultra-filtration system to supplement the current process.

Normally, all purified liquid from the waste collector tanks is sampled in the recovery sample tank and returned to the condensate system via the condensate storage tank. Periodically, purified water from the waste collector tanks is returned to its sub-system for reprocessing. On rare occasions, this purified water source is discharged to the environment.

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No waste is discharged to the environment from the recovery sample tanks until the requirements of the Technical Requirements, 10CFR20 (for radionuclides other than dissolved and entrained noble gases), and the site NPDES permit are met. This ensures that only water of acceptable quality is discharged from the plant.

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Liquid radwaste streams are combined to make the most effective use of the processing equipment available and to minimize the number of processing cycles received by an individual waste batch prior to final disposition. It is intended that recycling of treated water within the system is used as a tool to minimize radioactive liquid waste discharges.

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Condensate is used as flushing agent when required; the use of demineralized makeup water is held to a minimum and closely controlled.

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The procedural approach to monitor and control liquid waste effluents is contained in Section 11.5. Liquid waste is released into the cooling tower blowdown line from the recovery sample tanks on a batch basis. Each batch is analyzed prior to release for principle gamma emitters. The resultant isotopic analysis is used to determine the batch discharge flow rate and DRMS radioactive liquid effluent monitor alert and alarm setpoints. These actions ensure that the radionuclide activities released to the environment are maintained below the effluent concentration limit (ECL) specified in 10CFR20. Detailed administrative records are maintained for all radioactive liquid releases (e.g., total activity, temperature, flow rate, etc). Table 11.2-5 shows that the design radioactivity concentrations in discharged liquid wastes are significantly below the effluent concentration limit specified in 10CFR20.

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Inadvertent discharges of radioactive liquid wastes are avoided by:

1. The imposition of administrative control to determine the most effective means of processing any given batch

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2. Automatic controls, indicator lights, and alarms at the auxiliary control room situated adjacent to the radioactive waste building

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3. Batch processing controls and procedures to ensure that known quantities of activity and dissolved solids are moved in a controlled manner within the system

4. Adequate sizing of tanks and processing equipment, so that a normal batch collected over a period of one day may be processed while another is being received.

A radiation monitor in the waste discharge pipe provides a high radioactivity alarm and a discharge valve trip signal. The radiation monitor is located far enough upstream of the discharge valve to ensure valve closure prior to the release of radioactive liquids. Liquids with radioactivity exceeding the radiation monitor set point are recycled back to the radwaste system for further processing. Strict administrative controls are imposed on the trip set point of the discharge line radiation monitor. Sampling, monitoring, and analysis of the discharge tunnel water in accordance with Section 11.5 are used to help verify the readjustment of the set point.

11.2.2.7 Performance Tests

Tests for the removal of insolubles by filtration and the reduction of conductivity by demineralization are conducted on a periodic basis. The radioactivity of the input and output streams of each radwaste train is checked periodically.

Overall system tests for the waste and floor drain collector subsystem are run for the activity reduction factor.

11.2.3 Radioactive Releases and Doses

Table 11.2-4 is a tabulation of the expected annual activity released and conforms with the method and parameters given in NUREG-0016, Rev. 1. The design base release from the liquid effluent stream in Ci/yr per nuclide is given in Table 11.2-5 and corresponds to operation with design failed fuel conditions as discussed in Section 11.1.

Tritium release is anticipated at 45.6 Ci/yr.

11.2.3.1 Release Points

All liquid effluent releases from River Bend Station are discharged into the cooling tower water blowdown which is directed to the Mississippi River. Fig.11.2-1a through 11.2-1f show the system relationship of the release point into the discharge line, and Fig.1.2-1 shows the physical locations of the discharge outfall and the river.

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The systems and tanks within the plant which contain radionuclides, but are not specifically designed to withstand the effects of a tornado, probable maximum flood, or a design basis earthquake, are the reactor water cleanup system, service water system, condensate storage tank, and liquid radwaste collector tanks. The reactor water cleanup system is located in a Seismic Category I structure. The liquid radwaste collector tanks are located in a seismically analyzed structure.

All radioactive liquid released by a system or tank failure within a Seismic Category I or seismically analyzed structure except service water is collected by the floor drainage system within that structure, and it is directed to the radioactive liquid waste system for processing. As a result, it is not likely that any radioactive liquid will be released to the environment due to failure of these systems. Some of the service water treatment chemicals may be activated when the treated water passes through the drywell unit coolers. Because the service water system is a closed, chemically treated system, there is no system blowdown. Normal service water losses are the result of system leakage (e.g., pump seal leakage, valve packing leakage,

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•→9
etc.). The bulk of the system leakage from components located within plant structures is directed to the liquid radwaste system. Areas of the plant where service water leakage is not directed to radwaste are the diesel generator building, control building and electrical tunnel. These drains are directed to the sewage treatment plant, then to the blowdown line via a treated waste sump. Treatment plant effluents are monitored via grab samples. Normal service water components, including pumps, valves, and flanged connections, which are located outside of plant buildings are equipped with berms or other structures to contain leakage.
9←•

11.2.3.2 Dilution Factors

The only dilution factor used in evaluating release of radioactive liquid effluents is that provided by the cooling tower water blowdown flow of 2,200 gpm before it is discharged into the river. There, it is diluted with the river water which varies from a minimum of $5.8 \times 10^{(7)}$ gpm to an average of $2.2 \times 10^{(8)}$ gpm. The mode of release is that treated radioactive effluents are collected in the recovery sample tanks. The filled tank is sampled and then discharged. The treated and analyzed effluent is diluted with 2,200 gpm of cooling tower blowdown prior to discharging into the Mississippi River. The concentration (expressed in Ci/cc) indicates the activity levels following dilution with cooling tower blowdown.

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During shutdown when the cooling tower blowdown is unavailable, an alternate source of dilution water will be provided.
←•8 6←•

11.2.3.3 Estimated Doses

A summary of the estimated annual radiation doses from the radwaste system is presented in Appendix 11A. As shown in Appendix 11A, the estimated annual doses from liquid effluents are below the dose criteria set forth in 10CFR50, Appendix I, and hence below the dose criteria specified in 40CFR190 and 10CFR20.

11.3 GASEOUS WASTE MANAGEMENT SYSTEMS

11.3.1 Design Bases

11.3.1.1 Design Objective

The objective of the gaseous waste management system is to process and control the release of gaseous radioactive effluents to the site environs so as to maintain as low as reasonably achievable, the exposure of persons in unrestricted areas, to radioactive gaseous effluents (Appendix I to 10CFR50, May 5, 1975). This is to be accomplished while maintaining occupational exposure as low as reasonably achievable and without limiting plant operation or availability.

11.3.1.2 Design Criteria

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The gaseous effluent treatment systems are designed to limit the dose to offsite persons from routine station releases to significantly less than the limits specified in 10CFR20 and to operate within the emission rate limits established in the technical requirements.

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Design basis source terms are presented and discussed in Section 11.1. Corresponding expected radioactive gaseous effluents from all sources are shown in Table 11.3-1 as calculated using the data shown in Table 11.3-2. These are consistent with NUREG 0016, Rev. 1.

The annual average exposure at the site boundary during normal operation from gaseous sources is not expected to exceed the dose objectives of Appendix I to 10CFR50 in terms of actual doses to actual persons. The radiation dose design basis for the treated off gas is to delay the gas until the required fraction of the radionuclides has decayed and the daughter products are retained by the charcoal and the HEPA filters.

The gaseous radwaste equipment is selected, arranged, and shielded to maintain occupational exposure as low as reasonably achievable.

•→14

An evaluation of the system's capability to meet concentration limits of 10CFR20 when operating at design basis fuel leakage (304,000 μCi/sec at 30-min decay) is included in Table 11.3-8.

14←•

Section 11.3.2.1.14 includes discussion of equipment design where a potential for explosion exists.

The gaseous effluent treatment system is designed to the requirements of General Design Criteria (GDC) as follows:

GDC 60

The system has sufficient capacity to reduce the off-gas activity to permissible levels for release during normal operation, including anticipated operational occurrences, and to make it unnecessary to provide for termination of releases or limitation of plant operation due to unfavorable site environmental conditions.

GDC 64

Continuous monitoring of activity levels in the system upstream of the delay line provides advance notice of any potentially significant increase in releases. Continuous monitoring of the system effluent, with automatic isolation at activity levels corresponding to administrative release limits and annunciation at lower levels, along with continuous monitoring of the plant exhaust duct release, provide assurance that activity releases to the environment are in all events maintained within established limits.

11.3.1.3 Equipment Design Criteria

A list of the off gas system major equipment items which includes materials, rates process conditions, and number of units supplied, is provided in Table 11.3-3. Equipment and piping will be designed and constructed in accordance with the requirements of the applicable codes as given in Table 11.3-7.

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

The reactor building, auxiliary building, fuel building, turbine building, and radwaste building contain radioactive gas sources. The design bases for the ventilation systems for these three buildings are described in Section 9.4.

Conservative analyses of the off gas system are presented in Section 15.7 and demonstrate that equipment failure results in doses that are well within the guidelines of 10CFR100.

11.3.2 Main Condenser Steam Jet Air Ejector Low-Temp (RECHAR) System

The off gas from the main condenser steam jet air ejector is treated by means of a system utilizing catalytic recombination and low-temperature charcoal adsorption (RECHAR system). Descriptions of the major process components including design temperature and pressure are given in Table 11.3-3 and in the following sections.

11.3.2.1 System Process Description

Noncondensible radioactive off gas is continuously removed from the main condenser by the air ejector during plant operation.

The air ejector off gas normally contains activation gases, principally N-16, O-19, and N-13. The N-16 and O-19 have short half-lives and are readily decayed. The 10-min N-13 is present in small amounts that are further reduced by decay.

The air ejector off gas also contains radioactive noble gases, including parents of biologically significant Sr-89, Sr-90, Ba-140, and Cs-137. The concentration of these noble gases depends on the amount of tramp uranium in the coolant and on the cladding surfaces (usually extremely small) and the number and size of fuel cladding leaks.

A main condenser off gas treatment system has been incorporated in the plant design to reduce these gaseous radwaste emissions from the station. The off gas system uses a catalytic recombiner to recombine radiolytically dissociated hydrogen and oxygen. After cooling (to approximately 130°F) to strip the condensibles and reduce the volume, the remaining noncondensibles (principally air with traces of krypton and xenon) are delayed in the nominal 10-min holdup system. The gas is cooled to 45°F and filtered through a HEPA filter. The gas is then passed through a desiccant dryer that reduces the dewpoint to approximately -90°F and is then chilled to about 0°F. Charcoal adsorption beds, operating in a refrigerated vault at about 0°F, selectively adsorb and delay the xenons and kryptons from the bulk carrier gas (principally dry air). After the delay, the gas is again passed through a HEPA filter and discharged to the environment through the plant exhaust duct.

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11.3.2.1.1 Process Flow Data

The information supporting the process data is presented in Reference 2. The plant exhaust duct is the single release point for this system and is located on the reactor building. The plant exhaust duct is indicated in Fig. 11.3-2.

10←•

11.3.2.1.2 Noble Gas Radionuclide Source Term and Decay

Table 11.3-1 lists expected isotopic activities at the discharge of the system.

The expected main steam source term data, from which the annual average noble gas activity input to the main condenser off gas treatment system is derived, is discussed in Section 11.1 and is consistent with the methods given in NUREG-0016. The data used to calculate gaseous releases, including off gas system radionuclide decay, are shown in Table 11.3-2.

11.3.2.1.3 Piping and Instrumentation Diagram (P&ID)

The P&ID is submitted as Fig. 11.3-2. The main process routing is indicated by a heavy line.

11.3.2.1.4 Recombiner Sizing

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The basis for resizing the recombiner is to maintain the hydrogen concentration below 1 percent by volume on a dry basis at the outlet of the recombiner, regardless of the radiolytic hydrogen input (up to its full design basis value). The exit hydrogen concentration is normally well below the 1 percent maximum allowed. The hydrogen generation rate of the reactor is based on data from nine BWRs.

10←• 12←•

11.3.2.1.5 Process Design Parameters

The Kr and Xe holdup time is closely approximated by the following equation:

$$T = \frac{K_D M}{V}$$

where:

T = Holdup time of a given gas

K_D = Dynamic adsorption coefficient for the given gas

M = Weight of charcoal

V = Flow rate of the carrier gas in consistent units.

Dynamic adsorption coefficient values for xenon and krypton were reported by Browning[1]. GE has performed pilot plant tests at their Vallecitos Laboratory and the results were reported at the 12th AEC Air Cleaning Conference[3]. Moisture has a detrimental effect on adsorption coefficients. It is to prevent moisture from reaching the charcoal that the -90°F-dewpoint fully redundant, adsorbent air driers are supplied. There are redundant moisture analyzers that alarm on breakthrough of the drier beds; however, breakthrough is not expected since the drier beds are regenerated on a time basis. The system is slightly pressurized which, together with very stringent leak rate requirements, prevents leakage of moist air into the charcoal.

Carrier gas is the air inleakage from the main condenser after the radiolytic hydrogen and oxygen are removed by the recombiner. The air inleakage design basis is conservatively sized at 40 scfm total. The Sixth Edition of Heat Exchange Institute Standards for Steam Surface Condensers, Paragraph Sl(c)(2) indicates that, with certain conditions of stable operation and suitable construction, noncondensibles (not including radiological decomposition products) should not exceed 6 scfm for large condensers[4]. Dresden 2, Monticello, Fukushima 1, Tsuruga, and KRB have all operated at 6 scfm or below after initial startup. Dilution air is not added to the system unless the air inleakage is less than 6 scfm. In that event, 6 scfm is added to provide for dilution of residual hydrogen from the recombiner. An initial bleed of oil-free air is added on startup until the recombiner comes up to temperature. Another source of air is the constant 1-scfm bleed through the standby recombiner train to prevent back-seepage of hydrogen from the operating recombiner in the event of its failure.

11.3.2.1.6 Charcoal Adsorbers

11.3.2.1.6.1 Charcoal Temperature

The charcoal adsorbers operate at a nominal 0°F temperature. The decay heat is sufficiently small that, even in the no-flow condition, there is no significant loss of adsorbed noble gases due to temperature rise in the adsorbers. The adsorbers are located in a shielded room, and maintained at a constant temperature by a redundant vault refrigeration system. Failure of the refrigeration system causes an alarm in the main control room. Decay heat produces about 60 Btu/h in the adsorber charcoal due to an inventory of 7,300 curies corresponding to residence times of 46 hr for Kr and 42 days for Xe. For no-flow conditions, conservative calculations using this heat source and an assumed initial temperature of 0°F give a maximum axial temperature of about 90°F, which is well below the minimum charcoal ignition temperature of 374°F. If a possible fire is detected in the charcoal vessels, the adsorbers are isolated from off gas flow and the charcoal bypass is used. A nitrogen purge is then injected upstream of the vault entrance, so that further combustion is prevented and the charcoal is cooled below its ignition temperature.

11.3.2.1.6.2 Gas Channeling in the Charcoal Adsorber

Channeling in the charcoal adsorbers is prevented by supplying an effective flow distributor on the inlet, having long columns and a high bed-to-particle diameter ratio of approximately 500. Underhill has stated that channeling or wall effects may reduce efficiency of the holdup bed if this ratio is not greater than 12[5]. During transfer of the charcoal into the charcoal adsorber vessel, radial sizing of the charcoal is minimized by pouring the charcoal (by gravity or pneumatically) over a cone or other instrument to spread the granules over the surface.

11.3.2.1.6.3 Charcoal Bypass Mode

A valve and spectacle flange are provided to bypass the charcoal adsorbers. The main purpose of this bypass is to protect the charcoal during preoperation and startup testing when gas activity is zero or very low.

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It may be desirable to use the bypass for short periods during startup or normal operations. This bypass mode would not be used for normal operation unless some unforeseen system malfunction necessitates shutting down the power plant or operating in the bypass mode to remain within the technical requirement limits. The activity release is controlled by a process monitor upstream of the plant exhaust duct isolation valve that causes the bypass valve to close on a high radiation alarm. This interlock can be defeated only by a keylock switch. In addition, there is a high-high-high alarm on the same monitor that causes the off gas system to be isolated from the plant exhaust duct if established release limits are exceeded.

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11.3.2.1.7 Leakage of Radioactive Gases

Leakage of radioactive gases from the system is limited by welding piping connections where possible and using bellows stem seals or equivalent valving. The system operates at a maximum of 7 psig during startup and less than 2 psig during normal operation so that the differential pressure to cause leakage is small.

11.3.2.1.8 Hydrogen Concentration

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To limit the recombiner temperature delta, hydrogen concentration of gases from the air ejector is kept below 4 percent by maintaining adequate process steam flow for dilution at all times. This steam flow rate is monitored and alarmed, and the steam jet air ejector noncondensible suction line is isolated when there is insufficient steam. The hydrogen concentration itself is monitored by redundant hydrogen analyzers. The high alarm, at 2 percent hydrogen, indicates "off-normal" operation.

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11.3.2.1.9 Field Run Piping

Piping and tubing 2 in and under are field routed. This does not include major process piping but does include drain lines, steam lines, and sample lines which are shown on Fig. 11.3-2.

11.3.2.1.10 Liquid Seals

There are several liquid seals to prevent gas escape through drains shown on Fig. 11.3-2. These seals are protected against permanent loss of liquid by an enlarged section downstream of the seal that can hold the seal volume and drains by gravity back into the loop after the momentary pressure surge has passed. Each seal has a manual valve that can be used to fill the loop. In the event that a loop seal goes dry, level sensors initiate closure of an isolation valve in the loop. Seals are also equipped with solenoid valves that close if radioactive release from this system exceeds established limits.

11.3.2.1.11 System Performance

Noble gas activity release is given in Table 11.3-1 in units of Ci/yr/unit.

Iodine input into the off gas system is small by virtue of its retention in reactor water and condensate. The iodine remaining is essentially removed by adsorption in the charcoal. This is supported by the fact that charcoal filters remove 99.9 percent of the iodine in 2 in of charcoal, whereas this system has approximately 76 ft of charcoal in the flow path.

Particulates are removed with a 99.95 efficiency by a HEPA filter as gas exits the nominal 10-min holdup. The noble gas decays within the interstices of the activated charcoal and daughters are entrapped there. The charcoal serves as an excellent filter for other particulates and essentially no particulates exit from the charcoal. The charcoal is followed with a HEPA filter which is a safeguard against escape of charcoal dust. Particulate activity discharged from this system is essentially zero.

11.3.2.1.12 Isotopic Inventory

The isotopic inventory of each equipment piece is given in Chapter 12.

11.3.2.1.13 Previous Experience

Performance of a similar system operating at ambient temperatures and the results of experimental testing performed by GE have been submitted in the General Electric Company proprietary topical report, "Experimental and Operational Confirmation of Off Gas System Design Parameters[2]." Non-proprietary portions of this information are reported in Reference 3.

11.3.2.1.14 Single Failures and Operator Errors

Design provisions are incorporated which preclude the uncontrolled release of radioactivity to the environment as a result of any single operator error or of any single equipment malfunction short of the catastrophic equipment failures described in Chapter 15. An analysis of single equipment piece malfunctions is provided in Table 11.3-6.

Design precautions taken to prevent uncontrolled releases of activity include the following:

1. The system design seeks to eliminate ignition sources so that a hydrogen detonation is highly unlikely even in the event of a recombiner failure.

2. The system pressure boundary is detonation-resistant, despite the measures taken to avoid a possible detonation.

3. All discharge paths to the environment are monitored: the normal effluent path by the process and effluent radiation monitoring system (Section 11.5); general plant areas by the area radiation monitoring system (Section 12.3.4).

4. Dilution steam flow to the steam jet air ejector is monitored and alarmed, and the valving is required to be such that loss of dilution steam cannot occur without coincident loss of motive steam, so that the process gas is sufficiently diluted if it is flowing at all.

11.3.2.1.15 Maintainability of Gaseous Radwaste System

Design features which reduce or ease required maintenance or which reduce personnel exposure during maintenance include the following:

1. Redundant components for active, in-process equipment pieces, excluding the off gas condenser and water separator.

2. No rotating equipment in the radioactive process stream, but located where maintenance can be performed while the system is in operation, or in nonradioactive streams.

3. Block valves with air bleed pressurization for maintenance of the desiccant dryer which is required during plant operation.

4. Shielding of nonradioactive auxiliary subsystems from the radioactive process stream.

Design features which reduce leakage and releases of radioactive material include the following:

1. Extremely stringent leak rate requirements placed upon all equipment, piping, and instruments are required. These requirements are verified to have been met by performance of initial service leak tests and leak tests of all applicable modifications to the process system. Leak tests are performed by acceptable methods capable of accurate detection of leaks within the specified maximum allowable leak rate for that portion of the system being tested.

2. Use of welded joints wherever practicable.

3. Specification of valve types with extremely low leak rate characteristics, i.e., bellows seal, double stem seal, or equal.

4. Use of loop seals with enlarged discharge section to avoid syphoning and to be self-refilling following a pressure surge.

5. Specification of stringent seat-leak characteristics for valves and lines discharging to the environment via other systems.

11.3.2.2 System Design Description

11.3.2.2.1 Quality Classification and Construction and Testing Requirements

Equipment and piping are designed and constructed in accordance with the requirements of the applicable codes as given in Table 11.3-7 and comply with the welding and material requirements and the system construction and testing requirements as follows.

11.3.2.2.2 Seismic Design

11.3.2.2.2.1 Equipment

Equipment and components used to collect, process, or store gaseous radioactive waste are classified as nonseismic.

The support elements, including the skirts, legs, and anchor bolting, for the charcoal adsorber tanks of the off gas system are designed as follows:

1. The fundamental frequency of the charcoal adsorber tanks, including the support elements, is greater than 33 Hertz.

2. The charcoal adsorber tanks are mounted on the base mat of the building housing the tanks.

3. The charcoal adsorber tanks, including the support elements, are designed with a vertical static coefficient of 0.11 g and a horizontal static coefficient of 0.28 g.

4. The stress levels in the support elements of the charcoal adsorber tanks do not exceed 1.33 times the allowable stress levels permitted by the AISC Manual of Steel Construction, Seventh Edition, 1970.

These seismic criteria are considered to meet the requirements of Regulatory Guide 1.143 (formerly Branch Technical Position ETSB No. 11-1). Further discussion may be found in Reference 7.

11.3.2.2.2.2 Buildings Housing Gaseous Radioactive Waste Processing Systems

The equipment and the components of the off gas system are housed in the off gas area of the turbine building. The turbine building is classified as non-seismic and is designed and analyzed in accordance with seismic criteria described in Sections 3.7.2.17A and 3.8.4.4.9.

11.3.2.2.3 Quality Control

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

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

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

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

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

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

11.3.2.2.4 Welding

All welding constituting the pressure boundary of pressure retaining components is performed by qualified welders employing qualified welding procedures according to Table 11.3-7. Nonconsumable weld inserts are prohibited in process lines unless they are ground out after the weld is completed.

11.3.2.2.5 Materials

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

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

Impact testing of carbon steel components operating at cold temperatures is in accordance with Paragraph UG84, Section VIII, of ASME "Pressure Vessel - Division 1."

11.3.2.2.6 Construction of Process Systems

Pressure retaining components of process systems utilize welded construction to the maximum practicable extent. Process piping systems include the first root valve on sample and instrument lines. Process lines are not less than 3/4-in nominal pipe size. Sample and instrument lines are not considered as portions of the process systems. Flanged joints or suitable rapid disconnect fittings are not used except where maintenance requirements clearly indicate that such construction is preferable. Screwed connections in which threads provide the only seal are not used. Screwed connections backed up by seal welding or mechanical joints are used only on lines of 3/4-in nominal pipe size. In lines 3/4-in or greater, but less than 2 1/2-in nominal pipe

size, socket type welds are used. In lines 2 1/2-in nominal pipe size and larger, pipe welds are of the butt joint type, but backing rings are not used in lines carrying sludges, resins, etc.

11.3.2.2.7 System Integrity Testing

Completed process systems are pressure tested to the maximum practicable extent. Piping systems are hydrostatically tested in their entirety, utilizing available valves or temporary plugs at atmospheric tank connections. Hydrostatic testing of piping systems is performed at a pressure 1.5 times the design pressure, but in no case at less than 75 psig. The test pressure is held for a minimum of 30 min with no leakage indicated. Pneumatic testing may be substituted for hydrostatic testing in accordance with the applicable codes.

11.3.2.2.8 Instrumentation and Control

This system is monitored by flow, temperature, pressure, and humidity instrumentation, and by hydrogen analyzers to ensure correct operation and control. Table 11.3-5 lists the process parameters that are instrumented to alarm in the main control room. It also indicates whether the parameters are recorded or just indicated. Instrumentation alarms on the standby recombiner train are deactivated in order to eliminate continuous, operationally meaningless alarms, and thereby improve operator response to important process variables on the operating recombiner train. The operator is in control of the system at all times.

A radiation monitor after the off gas condenser continuously monitors radioactivity release from the reactor and input to the charcoal adsorbers. This radiation monitor is used to provide an alarm on high radiation in the off gas.

A radiation monitor is also provided at the outlet of the charcoal adsorbers to continuously monitor the rate from the adsorber beds. This radiation monitor is used to isolate the off gas system on high radioactivity to prevent treated gas of unacceptably high activity from entering the plant exhaust duct.

The activity of the gas entering and leaving the off gas treatment system is continuously monitored. Thus, system performance is known to the operator at all times. Provision is made for sampling and periodic analysis of the influent and influent gases for purposes of determining their compositions. This information is used in calibrating the monitors and in relating the release to calculated environs dose. Process radiation instrumentation is described in Section 11.5.2.

Environmental monitoring is used; however, at the estimated low dose levels, it is doubtful that the measurements can distinguish doses from the plant from normal variation in background radiation.

11.3.2.2.9 Detonation Resistance

The pressure boundary of the system is designed to be detonation resistant. The pressure vessels are designed to withstand 350 psig static pressure, and piping and valving are designed to resist dynamic pressures encountered in long runs of piping at the design temperature. This analysis is covered in a proprietary report submitted to the NRC[6].

Using this report[6], a designer can obtain the required wall thickness of a specific equipment design, which normally or possibly contains a detonable mixture of hydrogen and oxygen, which is then translated to the corresponding detonation-containing, static equipment pressure rating by using an appropriate code calculation.

The method assumes the absence of simultaneous secondary events such as earthquakes.

This procedure is the simplest that has been found that does not include a detailed and laborious analysis of the gas dynamics of the system. It results in a design that sustains the whole envelope of feasible detonations.

11.3.2.2.10 Operator Exposure Criteria and Controls

This system is normally operated from the main control room. Equipment and process valves containing radioactive fluid are placed in shielded cells maintained at a pressure less than that of normally occupied areas.

11.3.2.2.11 Equipment Malfunction

Malfunction analysis, indicating consequences and design precautions taken to accommodate failure of various components of the system, is given in Table 11.3-6.

11.3.2.2.12 Previous Experience

A system with similar equipment is in service at the KRB plant in Germany. Its performance is reviewed in Reference 2. The Tsuruga and Fukushima I plants in Japan have similar recombiners in service. Similar systems (ambient temperature charcoal) are in service at Dresden 2 and 3, Pilgrim, Quad Cities 1 and 2, Nuclenor, Hatch, Browns Ferry 1, 2 and 3, and Duane Arnold.

11.3.2.3 Operating Procedure

11.3.2.3.1 Prestartup Preparations

Prior to starting the main steam jet air ejectors (SJAE), the charcoal vault is cooled to near 0°F, the glycol cooler is chilled to near 35°F and glycol is circulated through the cooler condenser, a desiccant dryer is regenerated and valved in, the off gas condenser cooling water is valved in, and the recombiner heaters are turned on.

11.3.2.3.2 Startup

As the reactor is pressurized, preheater steam is supplied and air is bled through the preheater and recombiner. The recombiner is preheated to at least 225°F with this air bleed and/or by admitting steam to the final SJAE. With the recombiners preheated, and the desiccant drier and charcoal adsorbers valved in, the SJAE string is started. The bleed air is terminated. As the condenser is pumped down and the reactor power increases, the recombiner inlet stream is diluted to less than 1 percent hydrogen by volume by a fixed steam supply, and the off gas condenser outlet is maintained at less than 1 percent hydrogen by volume.

11.3.2.3.3 Normal Operation

After startup, the noncondensibles pumped by the SJAE stabilize. Recombiner performance is closely followed by the recorded temperature profile in the recombiner catalyst bed. The hydrogen effluent concentration is measured by a hydrogen analyzer.

Normal operation is terminated following a normal reactor shutdown or a scram by terminating steam to the SJAEs and the preheater.

11.3.2.3.4 Previous Experience

Previous experience is reviewed in Section 11.3.2.2.12.

11.3.2.4 Performance Tests

This system is used on a routine basis and does not require specific testing to assure operability. Monitoring equipment is calibrated and maintained on a specific schedule and on indication of malfunction.

11.3.2.4.1 Recombiner

Recombiner performance is continuously monitored and recorded by catalyst bed thermocouples that monitor the bed temperature profile and by a hydrogen analyzer that measures the hydrogen concentration of the effluent.

11.3.2.4.2 Prefilter

These particulate filters are tested at the time of filter installation or replacement using DOP (dioctylphthalate) aerosol to determine whether an installed filter meets the minimum in-place efficiency of 99.95 percent retention.

The DOP from filter testing is not allowed into the desiccant or the activated charcoal. This equipment is isolated during filter DOP testing and is bypassed until the process lines have been purged clear of test material.

Because the DOP would have a detrimental effect on the desiccant and charcoal, this filter is not periodically tested. This is justified because the main function of this prefilter is to prevent the long-lived daughters of the radioactive xenons generated in the holdup pipe from depositing in the downstream equipment, thereby minimizing contamination. Leakage through the filter has no effect on environmental release.

11.3.2.4.3 Desiccant Gas Drier

Desiccant gas drier performance is continuously monitored by an on-stream humidity analyzer.

11.3.2.4.4 Charcoal Performance

The ability of the charcoal to delay the noble gases can be continuously evaluated by comparing activity measured and recorded by the process activity monitors at the exit of the off gas condenser and at the exit of the charcoal adsorbers.

Experience with boiling water reactors has shown that the calibration of the off gas and plant exhaust duct effluent monitors changes with isotopic content. Isotopic content can change depending on the presence or absence of fuel cladding leaks in the reactor and the nature of the leaks. Because of this possible variation, the monitors are calibrated against grab samples periodically and whenever the radiation monitor after the off gas condenser shows significant variation in noble gas activity indicating a significant change in plant operations.

Grab sample points are located upstream and downstream of the first charcoal bed and downstream of the last charcoal bed and can be used for periodic sampling if the monitoring equipment indicates degradation of system delay performance.

11.3.2.4.5 Post Filter

On installation, replacements, and at periodic intervals during operation, these particulate filters are tested using a DOP smoke test or equivalent.

11.3.2.4.6 Previous Experience

Previous experience is reviewed in Section 11.3.2.2.12.

11.3.2.5 Other Radioactive Gas Sources

Radioactive gases are present in the power plant buildings as a result of process leakage and steam discharges. The process leakage creates the radioactive gases in the air discharged through the ventilation system. The design of the ventilation system is described in Section 9.4. The radiation activity levels from the ventilation system are treated in Chapter 12. The building volumes and ventilation flow rates are shown in Chapter 9.

The steam discharges to the suppression pool release radioactive gases to the primary containment. The activity released to the suppression pool is the product of the activity per unit volume, as shown in Section 11.1, and the quantity of steam discharged. The quantity of activity which becomes airborne depends on many factors, such as water temperature, nuclide specie, evolution rate, etc. A tabulation of the expected frequency and the quantity of steam discharged to the suppression pool is provided in Table 11.3-9.

11.3.3 Radioactive Releases

11.3.3.1 Release Points

The plant airborne radioactive releases to the environs are from three monitored roof-vent locations or points. These points are the plant exhaust duct, the fuel building exhaust duct, and the radwaste building exhaust duct.

The plant exhaust duct is above the reactor building dome which is the tallest structure in the power block. The main plant exhaust duct releases ventilation air from the following plant areas and systems:

1. Reactor building
2. Auxiliary building
3. Turbine building
4. Plant piping and electrical tunnels
5. Backwash receiving tank vent
6. Sample station vents
7. Turbine gland seal exhaust steam system

8. Offgas system

9. Mechanical vacuum pump exhaust.

The fuel and radwaste buildings exhaust ducts release ventilation air from their respective ventilation systems. These ventilation systems include sample station vents, tank vents, spent fuel pool sweep gas system, and building area ventilation exhaust.

11.3.3.2 Gland Seal System

This system is provided with separate clean steam made from demineralized condensate. The effect of clean steam utilization is negligible activity releases from the turbine gland sealing system.

11.3.3.3 Mechanical Vacuum Pump

Activity releases from the mechanical vacuum pump are presented in Table 11.3-1. The mechanical vacuum pump is operated for short periods of time during startups. The effluent from the mechanical vacuum pump is routed to the iodine filtration unit before being discharged to the environs through the plant exhaust duct.

11.3.3.4 Ventilation Releases

Expected ventilation, off gas, and drywell purge releases are presented in Table 11.3-1. The conservative estimated releases from ventilation and mechanical vacuum pump are based on NUREG-0016.

11.3.3.5 Dilution Factors

The atmospheric dilution factor associated with normal plant releases is based upon 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 defined in Section 2.3.

11.3.3.6 Estimated Doses

The maximum hypothetical gamma air dose and beta air dose occur in the west direction at the restricted area boundary. The doses at this location are estimated to be 7.0 mrad/yr gamma and 6.6 mrad/yr beta as compared with the Appendix I design objective for gamma and beta air doses of 10.0 mrad/yr and 20.0 mrad/yr, respectively.

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

11.3.3.7 Main Condenser Steam Jet Air Ejector Low-Temperature (RECHAR) System

The estimated annual release of noble gases during normal operation is given in Table 11.3-1. Values for each nuclide are given in Ci/yr/unit. Assumptions used to determine these releases are listed in Table 11.3-2.

11.3 References

1. Browning, W.E., et al. Removal of Fission Product Gases from Reactor Off Gas Streams by Adsorption. (ORNL) CF59-6-47, June 11, 1959.

2. Miller, C.W. Experimental and Operational Confirmation of Off Gas System Design Parameters. NEDO-10751, January 1973. (Proprietary)

3. Siegwarth, D.P. Measurement of Dynamic Adsorption Coefficients for Noble Gases on Activated Carbon. 12th AEC Air Cleaning Conference.

4. Standards for Steam Surface Condensers, Sixth Edition, Heat Exchange Institute, New York, NY, 1970.

5. Underhill, Dwight, et al. Design of Fission Gas Holdup System. Proceedings of the Eleventh AEC Air Cleaning Conference, 1970, p.217.

6. Nesbitt, L.B. Design Basis for New Gas Systems. NEDE-11146, July 1971. (Proprietary)

7. NUREG-0124 (Supplement 1 to NUREG-75/110 "Safety Evaluation Report Related to the Preliminary Design of the GESSAR-238 Nuclear Island Standard Design," US Nuclear Regulatory Commission, September 1976, p.3-1 and 3-2.

11.4 RADIOACTIVE SOLID WASTE SYSTEM

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The objective of the radioactive solid waste system is to collect, monitor, process, and package solid wastes for shipment offsite. Processing of waste is currently being performed by RBS personnel. Vendors may be used in the future depending on processing needs and vendor availability. Each vendor's specific system components are as described in their approved Topical Report. The solid waste system is located in the radwaste building.

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11.4.1 Design Bases

11.4.1.1 Power Generation Design Bases

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The solid waste system accepts sludges from the phase separator and backwash tanks in the liquid radwaste system. These wastes consist of spent resin beads, resin fines, and sludges in varying proportions as described in Section 11.2. The solid waste system is designed to pump wet wastes to portable skid-mounted equipment for processing. Current methods of processing include dewatering and/or solidification. Alternative processing methods may be utilized at a later date.

The system also provides a means of segregating, compacting and packaging miscellaneous dry radioactive materials, e.g., paper, rags, contaminated clothing, gloves, and shoe covers, and for packaging contaminated metallic materials and incompressible solid objects such as small tools and equipment parts.

The solid waste system components permanently installed within the radwaste building, and the radwaste building itself, are in conformance with Regulatory Guide 1.143, as described in Section 11.2. Portable equipment used in the processing of solid wastes is designed in accordance with its respective approved Topical Report.

Collection, processing, packaging, and storage of radioactive wastes are performed so as to maintain any potential radiation exposure to plant personnel to as low as is reasonably achievable (ALARA) levels, in accordance with Regulatory Guide 8.8, and within the dose limits of 10CFR20. Packaging and transportation of radioactive materials are in accordance with 49CFR170-178, 10CFR30, 10CFR61, and 10CFR71; shipments are in conformance with 49CFR173 dose limits.

3←•

11.4.1.1.1 Applicable Codes and Standards

Table 11.4-4 lists the applicable codes and standards for equipment in the solid waste system. The basis for the nonnuclear safety (NNS) classification and the material selections for equipment in the solid radwaste system are the criteria as established by Regulatory Guide 1.143, as discussed in Section 11.4.1.2.

The atmospheric waste sludge storage tank is a filament-wound, fiberglass-reinforced plastic tank. It is designed to meet or exceed the National Bureau of Standards Voluntary Product Standard PS 15-69 and the American Society for Testing and Materials Specification No. ASTM D3299-74.

11.4.1.1.2 Structural Design

•→14

The solid radwaste equipment arrangement is presented in Fig. 1.2-30 through 1.2-32. The structural design is described in Section 3.8.4.1.8. In accordance with Regulatory Guide 1.143 and Branch Technical Position ETSB 11-1 (Revision 1, April 1975), the building is seismically analyzed as described in Section 3.7.2.16A. The solid radwaste layout provides design features consistent with maintaining personnel exposure ALARA, as required by Regulatory Guide 8.8.

14←•

11.4.1.2 Compliance with Regulatory Guides and Code of Federal Regulations

Regulatory Guide 1.143 identifies the quality level, quality group classification (safety class), seismic requirements, and material requirements for equipment and structures containing radioactive wastes. The solid waste system is in conformance with the guide in that:

1. The equipment within the system is designed in accordance with requirements identified in Table 1 of the guide.

2. Materials for pressure-retaining components conform to the requirements of the specification for materials in Section II of the ASME Code, except that nickel alloy stainless steel and fiberglass- reinforced plastic (FRP) piping are used. The use of FRP piping is restricted to condensate flush connections to solid waste piping and components. Nickel alloy stainless piping is used in the solid waste system due to its corrosion resistance when transferring acidic wastes and evaporator bottoms.

3. Foundations and walls of the radwaste building, to an elevation above that sufficient to contain the maximum liquid inventory expected in the building, are designed to the seismic criteria described in the guide. They are described in Section 3.8.4.

Regulatory Guide 8.8 provides criteria for maintaining potential radiation exposure to plant personnel to ALARA levels. Design features incorporated to maintain ALARA criteria and meet the limits of 10CFR20 include:

•→3

1. Totally remote operation of permanently installed wet solid waste equipment (i.e., waste sludge tank and transfer pump) from the radwaste control panel, located in the auxiliary control room or the local panel, mounted separately from the equipment, in the radwaste building.

3←•

2. Remote flushing of all lines containing radioactive solids.

3. Minimizing lengths of piping runs.

4. Provide drip trays under pumps and control panels.

5. Waste lines and valves utilize butt weld or socket weld end connections to the maximum extent practical.

•→11 •→7

6. Provide operator training.

7←• 11←•

7. Providing adequate shielding of piping and components. To the greatest extent possible, components requiring access or maintenance are located in separate, shielded cubicles or are otherwise provided with features to reduce personnel exposure (i.e., tanks are in separate shielded compartments from pumps, air-operated valves located in high radiation areas have their air sets, regulators and/or solenoid valves in low radiation areas).

8. Providing curbs to contain spills.

9. Compliance with Regulatory Guide 8.10, as indicated in Section 1.8.

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10. Potentially contaminated exhaust air from the vendor processing equipment is routed to the radwaste building ventilation system.

CNSI, or any other vendor used, operates under and is subject to the River Bend Station plant specific Radioactive Waste Process Control Plan.

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All documents, procedures, and drawings used in the fabrication, testing, operation, and maintenance of the solid waste equipment are developed and controlled in accordance with the provisions established in the Code of Federal Regulations, Title 10CFR Part 50, Appendix B, Sections V and VI. All vendors are required to follow RBS Radiation Protection Procedures in processing radwaste in order to maintain their occupational radiation exposure ALARA. A Radiation Work Permit (RWP) is issued to the vendor's operator prior to beginning operations.
3←•
Packaging and transportation of radioactive materials are accomplished in accordance with 49CFR, 10CFR71, Appendix E, and 10CFR61.
•→3
Depending on the specific processing method being used, one or more approved Topical Report(s) describe the specific process. Refer to the applicable Topical Report for details concerning vendor's implementation of the requirements of 10CFR Parts 20, 50, 61, and 71, and Regulatory Guides 1.143, 8.8, and 8.10.
3←•

11.4.2 System Description

11.4.2.1 General Description

The solid radwaste system consists of the following:

1. One waste sludge tank, complete with level detection devices and mixing and flushing equipment.

2. One waste sludge pump with associated controls and instrumentation.

3. One indoor, electric, overhead, single-trolley bridge crane

4. One waste compactor

•→11 •→3 •→1
Wastes consisting of spent resin beads, resin fines, filter sludges, and other processing media from the liquid radwaste system are collected and mixed in the waste sludge tank or may be delivered directly to the portable processing system. If the waste is to be solidified, the solids are mixed for uniform dispersion of activity and analyzed. If the wastes are to be dewatered, a representative sample will be obtained via an Isolok sampler system or manually by the dip sampling method. Refer to the applicable Topical Report for details concerning the specific vendor's subsystem to be used for waste processing. The waste sludge system is presently being bypassed and all solid waste is being pumped to the processing unit where it is sampled and analyzed prior to shipping. This operational configuration will be continued until the waste sludge system is actually needed (i.e., waste evaporators put into service).
1←• 3←• 11←•

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Note: Currently applicable License Topical Reports include CNSI-2(4313-01354-01PA), CNSI (DW-11118-01-P-A) and CNSI (RDS-25506-01-P-A).

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A physical layout drawing illustrating the solid waste packaging, storage, and shipping areas of the radwaste building is presented in Fig. 1.2-30 through 1.2-32. Table 11.4-1 lists annual waste volumes, specific activities, and curie content of solid waste based on the radioactive source terms discussed in Section 11.1 and operating plant data.

11.4.2.2 Component Description

A description of the permanent solid waste system components, including materials of construction, as shown in the process flow diagram, is given in Table 11.4-3.

The following is a functional description of the permanent system components:

•→3 •→11

1. Waste Sludge Tank - This tank provides the capability for mixing various types of wastes prior to processing. An agitator provides a homogeneous waste slurry for feeding to the portable waste solidification/dewatering system. The tank is vented to the radwaste building ventilation system. An overflow from the tank is returned to the liquid radwaste backwash tank for reprocessing.

11←•

2. Waste Sludge Pump - This pump transports the homogeneous waste slurry from the waste sludge tank to the processing equipment.

3←• •→11

3. Bridge Crane - This crane is controlled locally using a hand-held radio controller. It is the primary means of moving waste containers from the fill area to the solid waste storage area and from the waste storage area to the shipping area. The crane is also used for moving empty containers to the fill area.

11←•

4. Waste Compactor - This unit is designed to reduce the volume of compressible dry radioactive wastes. The compactor is vented through a hooded exhaust fan and filter in order to control airborne particles during dry waste compaction.

11.4.2.3 System Operation

11.4.2.3.1 General Operation

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Operation of the solid radwaste system is described in Section 11.4.2.1. The High Integrity Container/liner will be based on activity levels, volume, processing method, or vendor used. Processing takes place in a controlled area. RBS identifies the radionuclides and the curie content for the processed waste prior to shipping the waste.

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11.4.2.3.2 Instrumentation and Controls

Instrumentation and controls are provided for indication of process operation, equipment performance status, and remotely located control of process equipment. The main control panel for the permanent solid waste equipment is located in the auxiliary control room. Level and temperature indicators as well as high/low level alarms are provided for the waste sludge tank. The waste sludge pump automatically shuts off when a low level is indicated in the waste sludge tank. High pressure and low flow indicators and alarms are provided for the waste sludge pump. If high pressure or low flow is indicated, the waste sludge pump automatically recirculates the solid waste back to the waste sludge tank. Control switches for the waste sludge pump and the waste sludge tank agitator are also located on the radwaste control panel in the auxiliary control room.

•→11 •→3

The processing system and liner filling are monitored by closed circuit television (CCTV). Level indication and alarm, and high-high level automatic shutoff features, are incorporated into the process controls to prevent overflows from the liners containing liquids, sludges, and spent resins. All equipment provided within the waste processing system is designed to fail in the safe position. Refer to the applicable topical report for the specific process used for a complete description of the instrumentation and controls system.

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11.4.2.3.3 Dry Waste Disposal

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The solid waste system also disposes of dry wastes consisting of dry filter media, contaminated clothing, small tools, rags, miscellaneous paper, glassware, wood, and equipment and miscellaneous wastes which cannot be effectively decontaminated prior to packaging. The segregation and removal of clean waste is usually performed to minimize the volume of waste to be buried. This may be performed on or off site. Temporary vendor services or Entergy facilities may be used to accomplish this. Compressible waste then may be compacted, using a compactor, into metal drums or boxes on or off site to reduce its volume. Compressible wastes are compacted by a compactor into 52- or 55-gallon drums to reduce their volume. The compactor exhaust is filtered to minimize airborne contamination during compaction. Noncompressible wastes are packaged manually in appropriate containers. The packaging of large waste materials and equipment that has been activated during reactor operation is handled on a case-by-case basis. Storage space for approximately 26,800 cu. ft. Of miscellaneous dry active waste in drums and boxes is provided. This waste is stored in the radwaste building, the low level radwaste storage facility, or approved temporary storage facilities. These facilities are used to store radioactive material, compacted waste, and packaged non-compatible waste. These facilities are used to store radioactive material, compacted waste, and packaged non-compatible waste. Radiation control access barrier are used as required. Dry active wastes which cannot be packaged into drums or boxes may be stored in a temporary dry active waste storage area of the radwaste building until transfer to one of the temporary dry active waste storage facilities. Segregation, packaging, and compacting of loose radwaste is performed prior to transfer of the waste to these facilities. Fig. 1.2-30 through 1.2-32 show the solid waste area general arrangement.

8←• 11←•

11.4.2.3.4 Radiation Monitoring

•→6 •→3

Area radiation detectors and monitors are provided as described in Section 12.3, Table 12.3-1 and Figures 12.3-6 through 12.3-9. Area radiation detectors and monitors supporting the radioactive solid waste system are located near:

1. Portable waste processing equipment.

2. Low level compacted waste drum storage area which is in the general area on the 136' elevation of the radwaste building.

These area detectors and monitors are provided to alert local personnel and the control room operator of increasing or abnormally high radiation levels as described in Section 12.3.4.

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11.4.2.4 Postulated Accident Analysis

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The design, fabrication, and operation of both the solid waste system and the portable waste processing system are in accordance with the appropriate codes and standards to ensure the safe and reliable processing of radioactive waste. Accidents can occur which have the potential for the release of contamination to the surrounding area. In the radwaste building, the release of the contents of the regenerant evaporator is postulated, and is the maximum credible accident due to failure of any of the radwaste equipment. The radiological consequences of any solid radwaste component failure are enveloped by those based on failure of the liquid waste system as described in Section 15.7.3.1.

11←•

Physical features designed to control liquid spills have been incorporated into the layout of the radwaste building. The waste liner fill area is surrounded by walls on four sides and a 6-in curb on two sides to contain spills. The floor and walls to an elevation 1 ft above the floor and the curb are epoxy-coated within the spill area for ease of decontamination. The spilled liquid and decontamination liquids can be returned to the liquid waste system for reprocessing via the floor drain lines.

3←•

11.4.2.5 Annual Volumes

The expected quantities of wet and dry waste and activities are given in Table 11.4-1. The total activity of the wet wastes is directly related to the activity in the liquids from each source and the decontamination factors (DF) for each process component. However, when the DF is 100 or greater, it is assumed that all the activity is transferred to the filter media or resins. It is assumed that both soluble and particulate nuclides are deposited in the demineralizers. Decay, prior to shipment, is not assumed. However, it is anticipated that, in many cases, intervals of at least 30 days will occur prior to shipment.

11.4.2.6 Packaging

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The appropriate High Integrity Container/liner is used for package solid wastes based on activity levels, volume, and processing method. All wastes collected in the solid radwaste system for disposal will be processed as described in Section 11.4.2.3 and shipped in accordance with regulatory and burial site requirements.

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The expected activity levels of solid radwaste volume generated annually are given in Table 11.4-1.

11.4.2.7 Storage Facilities

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Packaged processed radwaste is stored in a shielded storage area in the vicinity of the liner fill area, as shown in Fig. 1.2-30. Approximately 35 filled liners can be stored in this area at one time. Filled liners are not stacked due to the height restrictions of the storage area shield walls. If additional space is required to store liners, the liners can be transferred to the low level radwaste storage facility or the height of the radwaste building storage shield walls can be increased to accommodate the stacking of liners. The waste storage areas in the radwaste building and in the low level radwaste storage facilities can provide a combined storage capability for dewatered resins for approximately five years. Also, the combined capacity of the low level radwaste facilities and available storage in the radwaste building is approximately 26,800 cu ft of dry active waste. This is equivalent to dry active waste storage capacity for approximately five years

11.4.2.7.1 Low Level Radwaste Storage Facility (LLRWSF)

The LLRWSF is located outside the Protected Area at plant site coordinates N15,460 and E17,500, see Figure 1.2-2. The facility is an 80 ft. wide by 200 ft. long by 51 ft. tall steel frame building with metal siding and designed to support a 20 ton traveling crane. The size of the facility was established based upon the total number of radioactive material containers, size of the concrete vault, and requirements for inspection aisles. The LLRWSF has the capacity to store 8 ft. x 20 ft. x 8 ft. high Sea/Land containers and 96 HICs. In general, the inspection aisles are 3 ft. wide. The facility contains twelve concrete cubicles to store HICs. Each cubicle is 16 ft. x 16 ft. x 16 ft. with 2 ft. thick concrete walls and topped with 1 ft. thick removable concrete panels. Each cubicle has the capacity to hold eight HICs, i.e. four stacked two high. The entire facility is surrounded by a 1 ft. thick by 16 ft. high concrete shield wall, except at the openings for the 3 ft. x 7 ft. personnel door and the 14 ft. x 16 ft. roll-up door.

The LLRWSF is equipped with a natural convection ventilation system consisting of a building ridge vent, two supply fans and louvers mounted in the building walls. The system is of sufficient capacity to moderate inside temperatures for worker comfort and safety. There is no freeze protection in the facility.

The LLRWSF is provided with 480 volt electrical power to supply power for the interior and exterior lighting, the overhead traveling bridge crane, supply fans, and receptacles throughout the facility. Lightning protection is also provided and the building grounding is tied into the plant grounding system.

The LLRWSF is considered a storage occupancy with a fire loading classification of low hazard (NFPA-101 Section 4-2.2.2) since it is unoccupied except during the movement of radioactive material and all radioactive material is stored in noncombustible containers or within a concrete vault. In order to reduce the number of openings in the surrounding 16 ft. concrete wall for shielding purposes, only one personnel door is installed on the south end of the west wall and one roll-up door is installed in one south wall. Per NFPA 101 "Life Safety Code", Chapter 29-2.4.1, Exception No. 1, for low hazard storage occupancies only a single means of egress is required from any story or section; therefore, one exit is acceptable. Furthermore, since the LLRWSF will be unoccupied, no potable water will be required.

The LLRWSF is contained within a 6 ft. (approximately) fence to provide security as well as restricting general access to the facility.

Refer to Figure 11.4-3 for a layout of the LLRWSF.

11.4.2.7.2 Independent Spent Fuel Storage Installation (ISFSI)

The ISFSI or dry fuel storage system utilized is Holtec International's HI-STORM system. As stated in Chapter 9.1.2.5, River Bend Station's ISFSI is located approximately 470 feet southwest of the Reactor Building, and 134 feet southwest of the Auxiliary Control Building. The dry fuel storage cask system has been independently reviewed and approved for use by the NRC apart from the site reactor licenses. The full description of this system is in other documents. These documents include the Holtec Certificate of Compliance (CoC), document HI-20443196, Holtec Final Safety Analysis Report (FSAR), NRC Safety Evaluation Report (SER) and the Entergy Nuclear South 10 CFR 72.212 Evaluation Report.

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11←• 3←•

11.4.2.8 Shipment

The packaged solid waste is transported by a licensed disposal contractor or a licensed carrier. All containers shipped are appropriately shielded to meet the Department of Transportation radiation dose limits as described in 49CFR173.

The expected annual volumes of solid radwaste to be shipped offsite are given in Table 11.4-1. The corresponding expected isotopic activity level of the solid waste is provided in Table 11.4-2.

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 designation of either:

1. Instrumentation systems required for safety, or

2. Instrumentation systems required for plant operation.

11.5.1.1 Design Objectives

11.5.1.1.1 Systems Required for Safety

The main objective of radiation monitoring systems 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 handling areas if predetermined radiation levels are exceeded in major process/effluent streams, and to protect main control room personnel throughout the course of an accident. Additional objectives are to have these systems available under all operating conditions including accidents.

The radiation monitoring systems (RMS) provided to meet these objectives are:

1. Main steam line

•→16

2. Fuel building ventilation exhaust

3. Main control room air intakes

4. Containment purge isolation

16←•

11.5.1.1.2 Systems Required for Plant Operation

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The main objective of RMS required for plant operation is to provide operating personnel with measurement of the content of radioactive material in all potentially radioactive effluents and significantly contributing process streams. This allows demonstration of compliance with plant normal operational technical specifications/requirements by providing gross radiation level monitoring and collection of halogens and particulates on filters (gaseous effluents) as required by Regulatory Guide 1.21. Radiation monitoring is also provided for major process/effluent streams to cover abnormal and accident releases consistent with Regulatory Guide 1.97. These

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•→8
monitors are not required to be Category I per Regulatory Guide 1.97, Table 2; however, some have upgraded capabilities in order to be available post-LOCA. Additional objectives are to initiate discharge valve isolation on the offgas or liquid radwaste systems if predetermined release rates are exceeded and to provide for sampling at certain radiation monitor locations to allow determination of specific radionuclide content.
8←•
The RMS provided to meet these objectives are:

1. For gaseous effluent streams

    a. Radwaste building ventilation exhaust

    b. Main plant exhaust duct (normal range and extended range gas monitors)

2. For liquid effluent streams

    a. Liquid radwaste effluent

•→12 12←•

3. For gaseous process streams

    a. Offgas pretreatment

    b. Offgas post-treatment

    c. Auxiliary building ventilation

    d. Containment purge

•→16

    e. Deleted

    f. Turbine building ventilation

    g. Deleted

    h. Deleted

    i. Condensate demineralizer and offgas building ventilation

    j. Deleted

    k. Standby gas treatment system effluent

    l. Containment and drywell atmosphere monitoring

    m. Reactor building annulus ventilation

4. For liquid process streams

    a. Deleted

16←•

b. Turbine plant component cooling water

c. Reactor plant component cooling water

•→12

d. Cooling Tower blowdown line

•→16

e. RHR heat exchanger service water effluent

12←• 16←•

11.5.1.2 Design Criteria

11.5.1.2.1 Systems Required for Safety

The design criteria for the safety-related radioactivity monitoring systems are that the systems:

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

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

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

4. Provide continuous outputs on main control room panels.

5. Permit checking of the operational availability of each channel during reactor operation with provision for calibration function and instrument checks.

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

•→8

7. Initiate prompt protective action before plant technical specification/requirements limits are exceeded.

8←•

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

9. Insofar as practical, provide annunciation to indicate power failure or component malfunction.

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

11. Have sensitivities and ranges compatible with anticipated radiation levels.

The safety-related radioactivity monitoring systems satisfy General Design Criteria (GDC) 60, 63, and 64 of 10CFR50, Appendix A. The systems meet the design requirements for Safety Class 2, Seismic Category I systems, together with the quality assurance requirements of 10CFR50, Appendix B.

11.5.1.2.2 Systems Required for Plant Operation

The design criteria for operational RMS are that the systems:

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

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

3. Insofar as practical, 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 instrumentation checks.

•→8

6. Have sensitivities and ranges compatible with anticipated radiation levels and technical specification/requirements limits.

8←•

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

Additional design criteria for the main plant exhaust duct extended range monitor and drywell and containment atmosphere monitors are discussed in Sections 11.5.2.1.3.2.1 and 11.5.2.1.3.3.

•→8

The monitors installed on the containment purge system, the offgas system, and the liquid radwaste treatment systems have provisions to alarm and to initiate automatic closure of the discharge valves on the affected treatment system before the limits specified in technical specifications/requirements are exceeded, as required by Regulatory Guide 1.21.

8←•

The design bases conform to GDC 60, 63, and 64 of 10CFR50, Appendix A.

11.5.2 System Description

The process and effluent radiation monitoring system consists of a computer based digital radiation monitoring system (DRMS) and nondigital monitors supplied as part of the reactor protection system (RPS) and offgas treatment system.

11.5.2.1 Digital Radiation Monitoring System

The function of the DRMS is to measure, evaluate, and report radioactivity in process streams, in liquid and gaseous effluents, and in selected plant areas, and to annunciate abnormal system conditions. The process and effluent monitors, except as noted in Table 11.5-1, and area monitors in Table 12.3-1 constitute the DRMS. Each monitoring channel has a

•→16
microprocessor associated with the detector or sample station. The DRMS computer system continuously polls the local microprocessors collecting and storing radiation levels, alarms and status information for these monitoring channels. This information is available on demand for analysis of plant conditions, trending of radiation levels, and maintenance purposes. Associated with the DRMS computers is a report processor which collects meteorological tower data. This information is combined with information from the gaseous effluent monitors to generate the gaseous release calculations for Regulatory Guide 1.21 report generation.

Monitors are provided for in the following gaseous release points:

1. Main plant exhaust

2. Fuel building ventilation exhaust

3. Radwaste building ventilation exhaust

•→8
Liquid radwaste effluent data is determined by batch sampling before release in accordance with Regulatory Guide 1.21. The isotopic analysis is used to determine the monitor setpoints. The liquid radwaste effluent monitor terminates the release if the technical requirements limits are exceeded.

8←• 16←•
All monitoring channels have three alarm states: alert radiation, high radiation, and monitor failure. Alarms for all digital-based process monitors monitor microprocessors and are annunciated at each DRMS CRT console. Consoles are located in the main control room, technical support center, emergency operations facility, and radiation protection work area.

System panels are provided in the main control room for safety-related effluent monitors and post-accident monitors. In the unlikely event of failure of the redundant central processors, current radiation levels, alarms, and controls for these monitors are provided on these panels.

For nonsafety-related monitors, communication interface is at the system CRT in the main control room, technical support center, emergency operation facility, and radiation protection work area. All monitors can be communicated with at the local level by using a portable control unit.

11.5.2.1.1 DRMS Monitor Descriptions

Four basic types of monitoring or sampling systems are provided as indicated in Table 11.5-1 for the DRMS process and effluent monitoring systems; offline gas and particulate, offline gas, online steam, and offline liquid.

Offline Gas and Particulate Monitor

•→16
The typical offline gas and particulate monitor consists of an isokinetic sampling system, moving particulate filter with detector, gas sample chamber with detector, and associated pump and valving. Connections are available for taking grab samples of the process stream, and for taking tritium samples downstream of the filter units (effluent monitors only). Check sources that are remotely operated are provided with each detector to check the function of each channel regularly. Remote purging capability is provided for the gas sample chamber. Detectors are designed to obtain the nominal ranges indicated in Table 11.5-1.

The isokinetic sampling systems for these monitors are designed in accordance with ANSI N13.1-1969. Flow straighteners are provided in process streams that do not meet the minimum straight-run duct lengths specified by the standard. Sample lines from 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 lines and for all surfaces of the sampler which are in contact with the sample stream.

Offline Gas Monitor

The typical offline gas monitor consists of a sampling system, fixed particulate and charcoal filters, a gas sample chamber with detector and associated pump with valving. Connections are available to facilitate 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 offline gas and particulate monitor. Detector type, and nominal ranges are given in Table 11.5-1.
16←•
For post-accident monitors, multiple detectors are provided with a minimum overlap of a decade in range to cover the extended ranges indicated on gaseous channels. Sampling and collection capability only is provided to cover the particulate channel range from 10μ to 10 uCi/cc. These monitors are also designed to perform their required function under the appropriate environmental conditions as defined in Section 3.11.

Online Steam Monitor

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

Offline Liquid Monitor

The typical offline liquid monitor consists of a sample chamber with detector and associated pump, piping, and valving.

The detector is provided with a remotely operated check source and 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 for which the process stream would cause detector failure.

11.5.2.1.2 DRMS Monitoring Systems Required for Safety

11.5.2.1.2.1 Fuel Building Ventilation Exhaust

One offline gas monitor and one offline gas and particulate monitor as described in Section 11.5.2.1.1 are provided for monitoring the fuel building ventilation exhaust before discharge to the environment. These monitors function to collect data for Regulatory Guide 1.21 report generation during normal operation, and indicate airborne levels of radiation in the fuel building (Section 12.3.4).

The fuel building monitors are required for safety in the event that high airborne levels of radiation are present in the fuel building. They divert the ventilation exhaust through the fuel building ventilation system safety-related filter trains on a high alarm signal. The fuel building offline gas monitor has an extended range as indicated in Table 11.5-1 to cover releases throughout a design basis accident (DBA). These monitors are designed to perform their required function under all environmental conditions as defined in Section 3.11. Reliable Class 1E safety-related 120-V ac electrical power is provided to these monitors as described in Section 8.3.

11.5.2.1.2.2 Containment Purge Isolation Monitors

Redundant area monitors are provided on the containment purge system. These monitors are intended to meet the requirements of NUREG 0737 Item II.E.4.2 (Containment Isolation Dependability). On receipt of a high radiation signal, the containment purge is isolated. Reliable safety-related Class 1E 120-V ac electrical power is provided to these monitors as described in Section 8.3.

•→16
11.5.2.1.2.3 Deleted

11.5.2.1.2.4 Main Control Room Air Intakes

Redundant offline gas monitors are provided at all main control room air intakes (local and remote). The main control room ventilation local intake monitors divert the intake air through safety grade filter systems on a high radiation alarm. The main control room ventilation intake monitors enable the operator to choose the least contaminated air intake throughout the course of an accident (Section 6.4) and provide an indication of airborne radiation levels present in the main control room intake (Section 12.3.4). These monitors are also designed to perform their required function under all environmental conditions as defined in Section 3.11. Reliable safety-related Class 1E 120-V ac electrical power is provided to these monitors as described in Section 8.3.

11.5.2.1.2.5 Deleted
•→13
13←• 16←•
11.5.2.1.3 DRMS Monitoring Systems Required for Plant Operations

11.5.2.1.3.1 Liquid Effluent Monitors
•→12
One monitor is provided to prohibit unidentified radioactive liquid releases from the plant:

1. Liquid radwaste effluent monitor.

12←•

•→12 •→8
The liquid radwaste effluent monitor terminates a liquid radwaste system release if radiation levels exceed the technical requirement limits. Nonsafety-related electrical power is provided to this monitor as described in Section 8.3.
12←•

11.5.2.1.3.2 Gaseous Effluent Monitors

11.5.2.1.3.2.1 Main Plant Exhaust Duct

Effluent from the main plant exhaust duct is monitored by an extended-range offline gas monitor. The primary function of this monitor is to assure that technical requirements limits for releases are not exceeded, to collect data for Regulatory Guide 1.21 report generation, and to provide extended-range post-accident monitoring. Major process streams exhausted through the main plant exhaust duct include reactor building ventilation, auxiliary building ventilation, turbine building ventilation, piping tunnel ventilation, standby gas treatment system exhaust, and offgas building ventilation exhausts. Consistent with Regulatory Guide 1.97, this monitor is designed to perform its required function under all environmental conditions as defined in Section 3.11. Reliable safety-related Class 1E 120-V ac electrical power is provided to this monitor as described in Section 8.3.

Additionally, one normal range offline gas and particulate monitor is provided to monitor the main plant ventilation exhaust. This monitor functions primarily to assure that technical requirements limits for releases are not exceeded, and to collect data for Regulatory Guide 1.21 report generation. Nonsafety-related electrical power is provided to this monitor as described in Section 8.3.

11.5.2.1.3.2.2 Fuel Building Ventilation Exhaust

One normal range offline gas and particulate monitor is provided to monitor the fuel building exhaust. The monitor functions are described in Section 11.5.2.1.2.1.

11.5.2.1.3.2.3 Radwaste Building Ventilation Exhaust

One offline gas monitor and one offline gas and particulate monitor are provided to monitor the radwaste building ventilation exhaust. These monitors function primarily to assure that technical requirement limits for releases are not exceeded, to monitor airborne levels of radiation in the radwaste building (Section 12.3.4), and to collect data for Regulatory Guide 1.21 report generation. Gases from the radwaste tanks are filtered and discharged through the radwaste
building ventilation exhaust duct. The radwaste building offline gas monitor has an extended range to cover post-accident monitoring requirements for radwaste building effluents. Nonsafety-related electrical power is provided to these monitors as described in Section 8.3.
8←•

11.5.2.1.3.3 Process Ventilation Monitors

Offline gas and particulate monitors are provided on the following process ventilation streams:

1. Auxiliary building ventilation exhaust

2. Containment purge exhaust (gas monitor only)

3. Turbine building ventilation exhaust (including condensate demineralizer area)

4. Offgas building ventilation exhaust

5. Standby gas treatment effluent (gas monitor only)

The function of the preceding monitors is to identify sources of radiation in main plant exhaust duct effluent in accordance with Regulatory Guide 1.21 for monitoring separate streams into a common release point for better resolution. These monitors also function to indicate airborne levels of radiation in the corresponding plant buildings (Section 12.3.4).

The containment purge exhaust monitor isolates the normal containment purge on a high radiation alarm. The standby gas treatment system effluent is monitored to ensure adequate performance of the system and to alarm if release limits are exceeded. Nonsafety-related electrical power is provided to these monitors as described in Section 8.3.

11.5.2.1.3.4 Containment and Drywell Atmosphere

Offline gas and particulate monitors are provided to monitor the containment and drywell airborne levels of activity. The drywell monitor pulls a sample from the drywell through the monitoring system located in the containment and returns the sample exhaust to the drywell. The containment monitor is located near the reactor building ventilation unit coolers at 162 ft 0 in of the reactor building. The unit coolers provide mixed air that is representative of the containment atmosphere.

The containment and drywell atmosphere monitors are provided to aid in detecting reactor coolant pressure boundary (RCPB) leakage in accordance with Regulatory Guide 1.45. They are designed to remain functional during and after the seismic loading conditions as defined in Section 3.7. The containment atmosphere monitor also functions to indicate airborne radiation levels in containment (Section 12.3.4) for maintaining workers' exposure ALARA. Reliable safety-related Class 1E 120-V ac electrical power is provided to these monitors as described in Section 8.3.

11.5.2.1.3.5 Process Liquid Monitors

The following process streams are monitored by offline liquid monitors for detection of radiation levels:

•→16 •→12

1. Deleted

2. Reactor plant component cooling water

3. Cooling tower blowdown line

4. RHR Heat Exchanger Service Water Effluent

The reactor plant component cooling water monitors detect and alarm contamination of the circulating water in these systems. The cooling tower blowdown line monitor detects and alarms on increase in background or contamination in circ water cooling tower blowdown. Nonsafety-related electrical power is provided to these monitors as described in Section 8.3.

An offline liquid monitor is provided to monitor the service water effluent on each of the two RHR heat exchanger trains. These monitors function to detect and alarm contamination of the service water effluent due to leaks in the heat exchangers following a DBA or under normal operating conditions. These monitors are also designed to perform their required function under all environmental conditions as defined in section 3.11 except an HELB or MELC; however, they are not required to function after an HELB or MELC. Reliable safety-related Class 1E 120-V ac electrical power is provided to the monitors as described in Section 8.3.

11.5.2.1.3.6 Process Gaseous Monitors

The following process gaseous streams are monitored for radiation level and alarm if abnormal levels are detected:

1. Deleted

2. Reactor Building Annulus Ventilation

•→8 8←• 12←•

Redundant offline gas monitors are provided on the reactor building annulus ventilation exhaust. The annulus monitors function to indicate airborne levels of activity in the annulus area (Section 12.3.4). On a high radiation alarm signal the containment, auxiliary, and annulus ventilation exhaust is diverted through the SGTS.

16←•

•→16 •→12
These monitors are also designed to perform their required function under all environmental conditions as defined in Section 3.11. Reliable safety-related Class 1E 120-V ac electrical power is provided to these monitors as described in Section 8.3.
12←•

11.5.2.2 Noncomputer Based Process Radiation Monitoring Systems

11.5.2.2.1 Main Steam Line Radiation Monitoring System

This system monitors the gamma radiation level exterior to the main steam lines. 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 channel trip signals to the containment and reactor vessel isolation control systems to initiate protective action.
•→8
The system consists of two redundant instrument channels. Each channel consists of a local detector (gamma-sensitive ion chamber) and a main control room log radiation monitor. Power for two channels (A and C) is supplied from RPS bus A.
8←• 16←•

•→8
The detectors are physically located near the main steam lines 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 main steam lines in operation. Table 11.5-1 lists the range of the detectors.
•→16
Additional description and isolation functions are discussed in USAR section 7.3.1.1.2.3.
•←8
11.5.2.2.2 Off Gas Pretreatment Radiation Monitoring System

This system monitors radioactivity in the condenser off gas before it enters the delay pipe and after it has passed through the off gas condenser and water separator. The monitor detects the radiation level which is attributable to the fission gases produced in the reactor and transported with steam through the turbine to the condenser.

A continuous sample is extracted from the off gas pipe via a sample line. It is then passed through a sample chamber and a sample panel before being returned to the suction side of the steam jet air ejector (SJAE). The sample chamber is a steel pipe which is internally polished to minimize plateout. It can be purged with room air to check detector response to background radiation by using a three-way solenoid operated valve. The valve is controlled by a switch located in the main control room. The sample panel measures and indicates sample line flow. A sensor and converter (GM tube) is positioned adjacent to the vertical sample chamber and is connected to a radiation monitor in the main control room.

Nonsafety-related power is supplied from the 125-V dc bus B for the radiation monitor and recorder, and from a 120-V ac local bus for the sample and vial sampler panels.

The monitor has two trip circuits: one upscale (high) and one downscale (low). The trip outputs are used for alarm function only. Each trip is visually displayed on the monitor and actuates a main control room annunciator: offgas pretreat high radiation and offgas pretreat downscale. -High or low sample line flow measured at the sample panel actuates a main control room off gas sample high-low flow annunciator.
16←•

The radiation level output by the monitor can be directly correlated to the concentration of the noble gases by using the semiautomatic vial sampler panel to obtain a grab sample. To draw a sample, a serum bottle is inserted into a sample chamber, the sample lines are evacuated, and a solenoid-operated sample valve is opened to allow off gas to enter the bottle. The bottle is then removed and the sample is analyzed in the counting room with a multichannel gamma pulse height analyzer to determine the concentration of the various noble gas radionuclides. A correlation between the observed activity and the monitor reading permits calibration of the monitor.

11.5.2.2.3 Off Gas Post-Treatment Radiation Monitoring System

This system monitors radioactivity in the off gas piping downstream of the off gas system charcoal adsorbers and upstream of the off gas system discharge valve. A continuous sample is extracted from the off gas system piping, passed through the off gas post-treatment sample panel for monitoring and sampling, and returned to the off gas system piping. The sample panel has a pair of filters (one for particulate collection and one for halogen collection) in parallel (with respect to flow) with two identical continuous gross radiation detection assemblies. Each gross radiation assembly consists of a shielded chamber, a set of GM tubes, and a check source. Two radiation monitors analyze and visually display in the main control room the measured gross radiation level.

The sample panel shielded chambers can be purged with room air to check detector response to background radiation by using a three-way solenoid valve operated from the main control room. The sample panel measures and indicates sample line flow. A solenoid-operated check source for each detection assembly operated from the main control room can be used to check operability of the gross radiation channel.

Power is supplied from 125-V dc bus A for one radiation monitor, 125-V dc bus B for the other radiation monitors and a common two point recorder, and from a 120-V ac local bus for the sample panel.

•→16
Each radiation monitor has four trip circuits: three upscale (high-high-high\inop, high-high, and high), one downscale (low). Each trip is visually displayed on the radiation monitor. The three trips actuate corresponding main control room annunciators: off gas post-treatment high-high-high radiation\inop, off gas post-treatment high radiation, and off gas post-treatment downscale. The fourth trip circuit actuates an off gas post treatment high-high radiation annunciator. High or low sample flow measured at the sample panel actuates a main control room off gas posttreat sample high-low flow annunciator.
16←•

•→8

11.5.2.3 Calibration, Maintenance, and Inspection

Calibration, maintenance, and inspection is performed in accordance with the plant technical requirements.

8←•

11.5.2.3.1 Inspection and Tests

During reactor operation, daily checks of system operability are made by observing channel behavior. At periodic intervals during reactor operation, the detector response (of each monitor provided with a remotely positioned check source) is recorded together with the instrument background count rate to ensure proper functioning of the monitors. Any detector whose response cannot be verified by observation during normal operation or by using the remotely positioned check source has its response checked with a portable check source. A record is maintained showing the background radiation level and the detector response.

The system has electronic testing and calibrating equipment which permits channel testing without relocating or dismounting channel components. An internal trip test circuit, adjustable over the full range of the readout meter, is used for testing. Each channel is tested at least semiannually prior to performing a calibration check. Verification of valve operation, ventilation diversion, or other trip functions is done at this time if it can be done without jeopardizing the plant safety. The tests are documented.

11.5.2.3.2 Calibration

•→16

The continuous radiation monitor's calibration is traceable to certified National Institute of Standards and Technology and is accurate to at least Æ 15 percent. The source-detector geometry during primary calibration is identical to the sample-detector geometry in actual use. Secondary standards which were counted in reproducible geometry during the primary calibration are supplied with each continuous monitor for calibration after installation. Each continuous monitor is calibrated periodically. A calibration can also be performed by using liquid or gaseous radionuclide standards or by analyzing particulate, iodine, or gaseous grab samples with laboratory instruments.

11.5.2.3.3 Maintenance

The channel detector, electronics, and recorder are serviced and maintained on a periodic basis or in accordance with manufacturers' recommendations to ensure reliable operations. Such maintenance includes cleaning, and any required mechanical maintenance 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 which would affect the calibration, a recalibration is performed at the completion of the work.

16←•

11.5.2.4 Sampling

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

•→8

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 requirements describe various liquid effluent sampling and analyses frequencies.

8←•

Additionally, the process and effluent radiological monitoring system consists of periodic ventilation samples. Section 11.5.2.1 describes the ventilation samples monitored for airborne radioactivity by the ventilation sample particulate and gas monitors. In addition to the programmed sampling of various ventilation systems, the ventilation sample particulate and gas monitors (Section 11.5.2.1) are used to locate manually a specific source of high airborne 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 a representative grab sample collected from the ventilation exhaust ducts. Grab samples are obtained from locations indicated in Table 11.5-2. Samples are analyzed in the Chemistry Laboratory. A discussion of the post-accident sampling system design is provided in Section 9.3.2.6.

11.5.3 Effluent Monitoring and Sampling

•→9

All potentially radioactive gaseous and selected liquid effluent discharge paths are continuously monitored for radiation level as described in 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:

9←•

1. Main plant exhaust duct

2. Radwaste building ventilation exhaust

3. Fuel building ventilation exhaust.

•→12

The following liquid effluent path is sampled and monitored:

1. Liquid radwaste system effluent

12←•

All monitor ranges are listed in Table 11.5-1.

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

•→15
This effluent monitoring and sampling program is of such a comprehensive nature as to provide the information for the effluent measuring and reporting programs required by 10CFR50 Section 36a, Appendix A, General Design Criterion 64, and Appendix I and Regulatory Guide 1.21 in annual reports to the NRC.
15←•

11.5.4 Process Monitoring and Sampling

11.5.4.1 Implementation of General Design Criterion 60

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. Off gas post-treatment
2. Containment normal purge exhaust
3. Annulus building ventilation exhaust
4. Fuel building ventilation exhaust
5. Liquid radwaste effluent.

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

11.5.4.2 Implementation of General Design Criterion 63

•→9
Radiation levels in radioactive and selected potentially radioactive process streams are monitored by the process and effluent monitors given in Table 11.5-1.
9←•
Airborne radioactivity in the fuel handling area is detected by the fuel building ventilation exhaust monitors which initiate the fuel building ventilation system on high radioactivity. Airborne radioactivity in the containment is detected by the containment and drywell atmosphere monitors and the containment purge monitor which isolate the containment normal 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 the workers. The area radiation monitors described in Section 12.3.4 detect abnormal radiation levels in the various plant areas.

Batch releases are sampled and analyzed prior to discharge in addition to the continuous effluent monitoring. The recirculation pumps for liquid waste tanks are capable of recirculating the tank volumes twice in 8 hr.

APPENDIX 11A

SUMMARY OF ANNUAL RADIATION DOSES

APPENDIX 11A

LIST OF TABLES

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APPENDIX 11A

SUMMARY OF ANNUAL RADIATION DOSES

•→14
The values presented in this appendix are based on data current at the time of plant licensing except as noted.

The calculated annual radiation doses to the maximum individual from liquid and gaseous pathways are presented in Tables 11A-1 through 11A-13. Table 11A-14 demonstrates that the calculated annual radiation doses are below the design objectives of 10CFR50, Appendix I.

The maximum calculated organ dose per reactor for an individual from gaseous releases (particulates and radioiodines) is 14.1 mrem/yr to an infant thyroid. This represents a hypothetical situation of an infant who resides at a location of 2.0 km from the site in the northwest direction and who obtains all of their milk from a cow grazing 1.3 km north-northwest of the facility.
14←•
The calculated external exposure to the total body and skin from immersion in noble gases is 1.7 and 4.0 mrem/yr, respectively. These represent an individual residing at a location 1,260 m from the site in the northwest direction. The maximum calculated beta and gamma air doses from noble gas releases are 6.6 and 7.0 mrad/yr, respectively. This was calculated at the maximum X/Q location at the restricted area boundary 976 m from the site in the west direction.

For liquid releases, it was assumed that the maximum individual obtains drinking water from the closest downstream public water supply - Peoples Water Service Company (River Mile 175.5) from the Station. The maximum individual was assumed to consume fish and ducks whose principle habitat is the edge of the initial mixing zone (EIMZ). This location was also conservatively used in calculating doses from swimming and boating. Food products obtained from the individual's garden were assumed to be irrigated with water taken from the nearest public water supply at a rate equal to the average annual rainfall. The calculated doses from shoreline (recreation were performed for the shoreline) location nearest to the EIMZ. The maximum calculated total body dose for an individual from liquid pathways is 0.022 mrem/yr in the adult age group, and the maximum calculated organ dose for an individual from liquid pathways is 0.800 mrem/yr to a child's thyroid. These doses were primarily due to fish consumption.

The calculated annual gaseous and liquid doses for the population residing within a 50-mi radius of the site are presented in Table 11A-15. For the liquid effluents, the calculated population dose commitments within 50 mi for total body and thyroid are 0.44 and 0.068 manrem/yr, respectively. For the gaseous effluents, the calculated population dose commitments within 50 mi from noble gas effluents and radioiodines and particulates are 1.8 manrem/yr total body and 4.1 manrem/yr thyroid. These doses were calculated for a projected population in the year 2010 of 1,163,282 people within 50 mi of the site.

TABLE 11A-1

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

	Maximum Individual Liquid Pathways Annual Dose (mrem/yr)							
Pathway	Total Body	Skin	Bone	Liver	Thyroid	Kidney	Lung	GI-Tract
Potable water	4.2-05	0.0	1.1-05	4.6-05	2.1-04	3.8-05	3.3-05	3.5-05
Fish consumption	1.5-02	0.0	5.7-02	2.2-02	7.2-01	1.2-02	3.5-03	2.4-02
Invrt. consumption	5.5-03	0.0	1.4-02	1.5-02	4.3-02	4.6-03	1.9-03	4.1-02
Shoreline recreation	1.3-03	1.5-03	1.3-03	1.3-03	1.3-03	1.3-03	1.3-03	1.3-03
Fresh vegetation	8.0-06	0.0	5.6-06	9.9-06	2.9-05	5.3-06	3.6-06	3.0-06
Stored vegetation	6.3-05	0.0	4.1-05	7.8-05	2.4-05	4.1-05	2.9-05	2.6-05
Duck consumption	7.7-05	0.0	1.7-03	1.3-04	2.5-06	1.2-05	1.5-06	2.0-04
Swimming exposure	5.3-05	7.2-05	5.3-05	5.3-05	5.3-05	5.3-05	5.3-05	5.3-05
Boating exposure	1.7-04	2.3-04	1.7-04	1.7-04	1.7-04	1.7-04	1.7-04	1.7-04
TOTAL DOSE	2.2-02	1.8-03	7.4-02	3.9-02	7.6-01	1.8-02	7.0-03	6.7-02

NOTE: 4.2-05 = 4.2×10^{-5}

TABLE 11A-2

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

	Maximum Individual Liquid Pathways Annual Dose (mrem/yr)							
Pathway	Total Body	Skin	Bone	Liver	Thyroid	Kidney	Lung	GI-Tract
Potable water	2.8-05	0.0	1.0-05	3.6-05	1.8-04	3.9-05	2.4-05	2.5-05
Fish consumption	1.1-02	0.0	6.1-02	2.2-02	6.8-01	1.5-02	3.4-03	1.9-02
Invrt. consumption	4.5-03	0.0	1.5-02	1.5-02	4.1-02	5.7-03	2.1-03	3.0-02
Shoreline recreation	7.2-03	8.4-03	7.2-03	7.2-03	7.2-03	7.2-03	7.2-03	7.2-03
Fresh vegetation	4.5-06	0.0	5.0-06	8.2-06	2.3-05	8.5-06	2.7-06	2.3-06
Stored vegetation	6.4-05	0.0	6.8-05	1.2-04	3.0-05	9.3-05	4.0-05	3.2-05
Duck consumption	6.2-05	0.0	1.4-03	1.0-04	1.9-06	8.4-05	1.4-06	1.3-04
Swimming exposure	3.0-04	4.1-04	3.0-04	3.0-04	3.0-04	3.0-04	3.0-04	3.0-04
Boating exposure	1.7-04	2.3-04	1.7-04	1.7-04	1.7-04	1.7-04	1.7-04	1.7-04
TOTAL DOSE	2.3-02	9.0-03	8.5-02	4.5-02	7.3-01	2.9-02	1.3-02	5.7-02

NOTE: 2.8-05 = 2.8×10^{-5}

TABLE 11A-3

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

	Maximum Individual Liquid Pathways Annual Dose (mrem/yr)							
Pathway	Total Body	Skin	Bone	Liver	Thyroid	Kidney	Lung	GI-Tract
Potable water	4.9-05	0.0	2.9-05	7.0-05	4.3-04	5.4-05	4.6-05	4.6-05
Fish consumption	7.9-03	0.0	7.8-02	2.0-02	7.5-01	1.1-02	2.8-03	9.1-03
Invrt. consumption	4.2-03	0.0	2.0-02	1.3-02	4.7-02	3.7-03	1.9-03	1.6-02
Shoreline recreation	1.5-03	1.8-03	1.5-03	1.5-03	1.5-03	1.5-03	1.5-03	1.5-03
Fresh vegetation	3.8-06	0.0	8.7-06	1.0-05	3.4-05	5.0-06	3.1-06	2.5-06
Stored vegetation	7.3-05	0.0	1.6-04	2.0-04	4.8-05	9.3-05	6.2-05	4.7-05
Duck consumption	5.8-05	0.0	1.4-03	7.6-05	1.5-06	5.4-06	8.5-07	4.1-05
Swimming exposure	1.9-04	2.5-04	1.9-04	1.9-04	1.9-04	1.9-04	1.9-04	1.9-04
Boating exposure	9.7-05	1.3-04	9.7-05	9.7-05	9.7-05	9.7-05	9.7-05	9.7-05
TOTAL DOSE	1.4-02	2.2-03	1.0-01	3.5-02	8.0-01	1.7-02	6.6-03	2.7-02

NOTE: 4.9-05 = 4.9×10^{-5}

TABLE 11A-4

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

Maximum Individual Liquid Pathways Annual Dose (mrem/yr)

Pathway	Total Body	Skin	Bone	Liver	Thyroid	Kidney	Lung	GI-Tract
Potable water	4.7-05	0.0	3.1-05	7.6-05	6.5-04	5.4-05	4.6-05	4.4-05
TOTAL DOSE	4.7-05	0.0	3.1-05	7.6-05	6.5-04	5.4-05	4.6-05	4.4-05

NOTE: 4.7-05 = 4.7×10^{-5}

TABLE 11A-5

ANNUAL DOSES TO MAXIMUM INDIVIDUAL IN THE ADULT GROUP
FROM GASEOUS RADIOIODINE AND PARTICULATE EFFLUENTS

	Resident 2.0 km NW Cow Pasture 1.3 km NNW Annual Dose (mrem/yr) per unit							
Pathway	Total Body	Skin	Bone	Liver	Thyroid	Kidney	Lung	GI-Tract
Contaminated ground	1.7-02	2.0-02	1.7-02	1.7-02	1.7-02	1.7-02	1.7-02	1.7-02
Inhalation	2.0-03	0.0	4.0-03	2.4-03	9.1-02	2.6-03	3.2-03	2.4-03
Fresh vegetation	6.3-03	0.0	2.6-02	6.9-03	1.8-01	6.7-03	5.2-03	6.3-03
Stored vegetation	3.6-02	0.0	1.6-01	3.8-02	3.7-02	3.4-02	3.2-02	3.4-02
Cow milk	4.9-02	0.0	1.7-01	5.8-02	1.2+00	4.8-02	3.4-02	3.8-02
TOTAL DOSE	1.1-01	2.0-02	3.8-01	1.2-01	1.5+00	1.1-01	9.1-02	9.8-02

NOTE: 1.7-2 = 1.7×10^{-2}

TABLE 11A-6

ANNUAL DOSES TO MAXIMUM INDIVIDUAL IN THE TEEN GROUP
FROM GASEOUS RADIOIODINE AND PARTICULATE EFFLUENTS

Resident 2.0 km NW Cow Pasture 1.3 km NNW Annual Dose (mrem/yr) per unit

Pathway	Total Body	Skin	Bone	Liver	Thyroid	Kidney	Lung	GI-Tract
Contaminated ground	1.7-02	2.0-02	1.7-02	1.7-02	1.7-02	1.7-02	1.7-02	1.7-02
Inhalation	2.3-03	0.0	5.7-03	2.9-03	1.2-01	3.3-03	4.3-03	2.8-03
Fresh vegetation	5.5-03	0.0	2.5-02	6.3-03	1.5-01	3.9-02	4.9-03	5.6-03
Stored vegetation	5.9-02	0.0	2.7-01	6.5-02	6.2-02	2.6-01	5.5-02	5.6-02
Cow milk	7.8-02	0.0	3.2-01	1.9-01	1.9+00	8.7-02	6.3-02	6.6-02
TOTAL DOSE	1.6-01	2.0-02	6.4-01	1.9-01	2.2+00	4.1-01	1.4-01	1.5-01

NOTE: 1.7-2 = 1.7×10^{-2}

TABLE 11A-7

ANNUAL DOSES TO MAXIMUM INDIVIDUAL IN THE CHILD GROUP
FROM GASEOUS RADIOIODINE AND PARTICULATE EFFLUENTS

		Residence 2.0 km NW Cow Pasture 1.3 km NNW Annual Dose (mrem/yr) per Unit						
Pathway	Total Body	Skin	Bone	Liver	Thyroid	Kidney	Lung	GI-Tract
Contaminated ground	1.7-02	2.0-02	1.7-02	1.7-02	1.7-02	1.7-02	1.7-02	1.7-02
Inhalation	2.6-03	0.0	7.8-03	3.1-03	1.5-01	3.5-03	4.1-03	2.6-03
Fresh vegetation	9.4-03	0.0	4.5-02	1.1-02	2.2-01	1.0-02	8.8-03	9.2-03
Stored vegetation	1.4-01	0.0	6.7-01	1.5-01	1.5-01	1.4-01	1.3-01	1.3-01
Cow milk	1.7-01	0.0	7.7-01	2.2-01	3.9+00	1.9-01	1.5-01	1.5-01
TOTAL DOSE	3.4-01	2.0-02	1.5+00	4.0-01	4.4+00	3.6-01	3.1-01	3.1-01

NOTE: 1.7-2 = 1.7×10^{-2}

TABLE 11A-8

ANNUAL DOSES TO MAXIMUM INDIVIDUAL IN THE INFANT GROUP
FROM GASEOUS RADIOIODINE AND PARTICULATE EFFLUENTS

●→14

	Residence 2.0 km NW Cow Pasture 1.3 km NNW Annual Dose (mrem/yr) Per Unit							
Pathway	Total Body	Skin	Bone	Liver	Thyroid(b)	Kidney	Lung	GI-Tract
Contaminated ground	1.7-02(a)	2.0-02	1.7-02	1.7-02	2.5-02	1.7-02	1.7-02	1.7-02
Inhalation	1.8-03	0.0	5.8-03	2.4-03	2.0-01	2.4-03	3.1-03	1.7-03
Cow milk	3.3-01	0.0	1.5+00	4.5-01	1.4+01	3.7-01	3.1-01	3.2-01
TOTAL DOSE	3.5-01	2.0-02	1.5+00	4.7-01	1.4+01	3.9-01	3.3-01	3.4-01

14←●

●→14
NOTES: (a) 1.7-2 = 1.7 x 10^{-2}
(b) Values shown have been conservatively adjusted to account for increased doses that may result from plant operation with hydrogen water chemistry.

14←●

TABLE 11A-9

ANNUAL DOSES TO MAXIMUM INDIVIDUAL IN THE ADULT GROUP
FROM GASEOUS RADIOIODINE AND PARTICULATE EFFLUENTS

Residence 1,260 m NW Cow Pasture 1,260 m NW Annual Dose (mrem/yr) per Unit

Pathway	Total Body	Skin	Bone	Liver	Thyroid	Kidney	Lung	GI-Tract
Contaminated ground	5.2-02	6.1-02	5.2-02	5.2-02	5.2-02	5.2-02	5.2-02	5.2-02
Inhalation	4.9-03	0.0	9.9-03	5.9-03	2.3-01	6.6-03	7.9-03	6.0-03
Fresh vegetation	1.6-02	0.0	6.6-02	1.8-02	5.2-01	1.7-02	1.3-02	1.6-02
Stored vegetation	9.3-02	0.0	3.9-01	9.8-02	9.4-02	8.5-02	8.0-02	8.4-02
Beef	3.6-02	0.0	1.7-01	3.7-02	9.1-02	3.5-02	3.3-02	6.0-02
TOTAL DOSE	2.0-01	6.1-02	6.9-01	2.1-01	9.9-01	2.0-01	1.9-01	2.2-01

NOTE: 5.2-02 = 5.2×10^{-2}

TABLE 11A-10

ANNUAL DOSES TO MAXIMUM INDIVIDUAL IN THE TEEN GROUP
FROM GASEOUS RADIOIODINE AND PARTICULATE EFFLUENTS

Residence 1,260 m NW Cattle Pasture 1,260 m NW Annual Dose (mrem/yr) Per Unit

Pathway	Total Body	Skin	Bone	Liver	Thyroid	Kidney	Lung	GI-Tract
Contaminated ground	5.2-02	6.1-02	5.2-02	5.2-02	5.2-02	5.2-02	5.2-02	5.2-02
Inhalation	5.8-03	0.0	1.4-02	7.2-03	3.0-01	8.3-03	1.1-02	6.9-03
Fresh vegetation	1.4-02	0.0	6.2-02	1.6-02	4.3-01	1.2-01	1.2-02	1.4-02
Stored vegetation	1.5-01	0.0	6.8-01	1.7-01	1.6-01	7.5-01	1.4-01	1.4-01
Beef	2.9-02	0.0	1.4-01	3.1-02	7.0-02	5.6-01	2.8-02	4.3-02
TOTAL DOSE	2.5-01	6.1-02	9.5-01	2.8-01	1.0+00	1.5+00	2.4-01	2.6-01

NOTE: 5.2-02 = 5.2×10^{-2}

TABLE 11A-11

ANNUAL DOSES TO MAXIMUM INDIVIDUAL IN THE CHILD GROUP
FROM GASEOUS RADIOIODINE AND PARTICULATE EFFLUENTS

	Residence 1,260 m NW Cattle Pasture 1,260 m NW Annual Dose (mrem/yr) Per Unit							
Pathway	Total Body	Skin	Bone	Liver	Thyroid	Kidney	Lung	GI-Tract
Contaminated ground	5.2-02	6.1-02	5.2-02	5.2-02	5.2-02	5.2-02	5.2-02	5.2-02
Inhalation	6.4-03	0.0	1.9-02	7.8-03	3.8-01	8.7-03	1.0-02	6.5-03
Fresh vegetation	2.4-02	0.0	1.1-01	2.8-02	6.6-01	2.7-02	2.2-02	2.3-02
Stored vegetation	3.4-01	0.0	1.7+00	3.8-01	3.7-01	3.4-01	3.3-01	3.3-01
Beef	5.3-02	0.0	2.6-01	5.6-02	1.2-01	5.4-02	5.2-02	6.0-02
TOTAL DOSE	4.8-01	6.1-02	2.1+00	5.2-01	1.6+00	4.8-01	4.7-01	4.7-01

NOTE: 5.2-02 = 5.2×10^{-2}

TABLE 11A-12

ANNUAL DOSES TO MAXIMUM INDIVIDUAL IN THE INFANT GROUP
FROM GASEOUS RADIOIODINE AND PARTICULATE EFFLUENTS

	Residence 1,260 m NW Annual Dose (mrem/yr) Per Unit							
Pathway	Total Body	Skin	Bone	Liver	Thyroid	Kidney	Lung	GI-Tract
Contaminated ground	5.2-02	6.1-02	5.2-02	5.2-02	5.2-02	5.2-02	5.2-02	5.2-02
Inhalation	4.5-03	0.0	1.4-02	5.9-03	3.5-01	5.9-03	7.7-03	4.3-03
TOTAL DOSE	5.7-02	6.1-02	6.6-02	5.8-02	4.0-01	5.8-02	6.0-02	5.6-02

NOTE: 5.2-02 = 5.2×10^{-2}

TABLE 11A-13

ANNUAL DOSES
FROM NOBLE GAS RELEASES
PER UNIT

Pathway	Total Body Dose [1] (mrem/yr)	Skin Dose [1] (mrem/yr)	Beta Air Dose [2] (mrad/yr)	Gamma Air Dose [2] (mrad/yr)
Submersion	1.7+00[3]	4.0+00	6.6+00	7.0+00

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[1] Location of analysis is residence 1,260 m northwest of the site.

[2] Location of analysis is 976 m west of the site (maximum X/Q location at restricted area boundary).

[3] 1.7+0 = 1.7×10^0

TABLE 11A-14

COMPARISON OF MAXIMUM INDIVIDUAL DOSE COMMITMENTS WITH APPENDIX I TO 10CFR PART 50

Dose Criterion	Calculated Dose Single Unit Operation	RM-50-2 Design Objectives
Noble Gas Releases		
Beta dose in air	6.6 mrad/yr	20 mrad/yr
Gamma dose in air	7.0 mrad/yr	10 mrad/yr
Total-body dose	1.7 mrem/yr	5 mrem/yr
Skin dose	4.0 mrem/yr	15 mrem/yr
Liquid Releases[1]		
Total-body dose	0.02 mrem/yr	5 mrem/yr
Organ Dose	0.80 mrem/yr	5 mrem/yr
Iodines and Particulate Releases[2]		
•→14 Organ dose 14←•	14.1 mrem/yr	15 mrem/yr

[1] The radiological doses presented in this table for the liquid pathways are for two-unit operation (blowdown flow = 4,400 gpm). For single unit operation (blowdown flow = 2,200 gpm), the discharge velocity would be half the two-unit value (2.1 fps) due to the reduced flow through the outfall pipe. The concentration of radionuclides in the blowdown is identical for one-and two-unit operation. The quantity of radionuclides released to the river for one-unit operation would be half the two-unit value. At the downstream location of the nearest domestic water intake, the radionuclide release is dispersed throughout the river cross section at approximately equal concentration. River flow is the main factor causing dilution at this point, and the radionuclide concentrations for single unit operation would be about half the two-unit values. At the edge of the mixing zone, dilution is affected by both river flow and discharge characteristics. The one-unit concentrations at the edge of the mixing zone will be approximately equal to the two-unit values.

•→14
[2] Carbon-14 and tritium have been added to this category. (b) Values shown have been conservatively adjusted to account for increased doses that may result from plant operation with hydrogen water chemistry.
14←•

THIS TABLE HAS BEEN DELETED

TABLE 11A-15

CALCULATED POPULATION DOSE COMMITMENT

RIVER BEND STATION

UPDATED SAFETY ANALYSIS REPORT