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11.4-1	Solid Radwaste System Simplified Flow Diagram

DRAWINGS CITED IN THIS CHAPTER*

* The listed drawings are included as "General References" only;, i.e., refer to the drawings to obtain additional detail or to obtain background information. These drawings are not part of the USAR. They are controlled by the Controlled Document Program.

DRAWING*	SUBJECT
105D5099	Process Diagram Off-Gas System - Low Temp
105D5099A	Process Data Off-Gas System - Low Temp
842E420	Diagram of Steam Seal System
A-10649	L*A Water Treatment Drawing - External Regeneration (See Vendor Manual)
A-10793	L*A Water Treatment Drawing - Regeneration Sequence Resin Transfer (See Vendor Manual)
E02-1OG99-313	Schematic Diagram Off-Gas System
E02-1OG99-314	Schematic Diagram Off-Gas System
M01-1502	Radiation Shielding Design - Radwaste Building El. 702'-0"
M01-1508	Radiation Shielding Design - Radwaste Building El. 737'-0"
M01-1514	Radiation Shielding Design - Radwaste Building El. 762'-0"
M01-1531	Radiaiton Shielding Design - Radwaste Building El. 720'-0"
M05-1012	Cycled Condensate Storage
M05-1064	Process Radiation Monitoring System
M05-1079	Reactor Core Isolation Cooling System
M05-1081	Liquid Radwaste Process Diagram
M05-1082	Solid Radwaste Process Diagram
M05-1084	Off-Gas System
M05-1085	Liquid Radwaste Process System
M05-1089	Radwaste Sludge Process Concentration Tank System
M05-1100	Simplified Ventilation and Off-Gas Process Diagram
M05-1102	Control Room HVAC
M05-1104	Fuel Building HVAC System
M05-1110	Drywell Purge
M05-1111-1,2,3 and 5	Containment Building HVAC System

CHAPTER 11 - RADIOACTIVE WASTE MANAGEMENT

11.1 SOURCE TERMS

General Electric has evaluated radioactive material sources (activation products and fission product release from fuel) in operating boiling water reactors (BWR's) over the past decade. These source terms are reviewed and periodically revised to incorporate up-to-date information. Release of radioactive material from operating BWR's has resulted in doses to offsite persons which have been only a small fraction of 10 CFR 20, or of natural background dose.

The information provided in this section defines the design-basis radioactive material levels in the reactor water, steam and offgas. The various radioisotopes listed have been grouped as coolant activation products, noncoolant activation products, and fission products. The fission product levels are based on measurements of BWR reactor water and off-gas at several stations through mid-1971. Emphasis was placed on observations made at KRB and Dresden 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:

a. plant equipment design,

b. shielding design,

c. understanding system operation and performance,

d. measurement practicability, and

e. evaluating radioactive material releases to the environment.

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

11.1.1 Fission Products

11.1.1.1 Noble Radiogas Fission Products

The noble radiogas fission product source terms observed in operating BWR's are generally complex mixtures whose sources vary from miniscule defects in cladding to "tramp" uranium on external cladding surfaces. The relative concentrations or amounts of noble radiogas isotopes can be described as follows:

Equilibrium: $$R_g \approx K_1 y \qquad (11.1\text{-}1)$$

Recoil: $$R_g \approx K_2 Y\lambda \qquad (11.1\text{-}2)$$

The nomenclature in Subsection 11.1.1.4 defines the terms in these and succeeding equations. The constants K_1 and K_2 describe the fractions of the total fissions that are involved in each of the releases. The equilibrium and recoil mixtures are the two extremes of the mixture spectrum that are physically possible. When a sufficient time delay occurs between the fission event and

the time of release of the radiogases from the fuel to the coolant, the radiogases approach equilibrium levels in the fuel and the equilibrium mixture results. When there is no delay or impedance between the fission event and the release of the radiogases, the recoil mixture is observed.

Prior to Vallecitos Boiling Water Reactor (VBWR) and Dresden 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.

VBWR and early Dresden 1 experience indicated that the actual mixture most often observed approached a distribution which was intermediate in character to the two extremes (Reference 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, the "diffusion" distribution pattern which has been described is as follows:

Diffusion: $$R_g \approx K_3 y \lambda^{0.5} \qquad (11.1\text{-}3)$$

The constant K_3 describes the fraction of total fissions that are involved in the release. The value of the exponent of the decay constant, λ, is midway between the values for 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-minutes decay (t = 30 minutes). The noble radiogas source-term rate after 30-minutes decay has been used as a conventional measure of the design-basis fuel leakage rate since it is conveniently measurable and was consistent with the nominal design-basis 30-minutes off-gas holdup system used on a number of plants. Since about 1967, the design-basis release magnitude used (including the 1971 source terms) was established at an annual average of 0.1 Ci/sec (t = 30 minutes). This design-basis is considered as an annual average with some time above and some time below this value. 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.

Noble radiogas source terms from fuel above 0.1 Ci/sec (t = 30 minutes) can be tolerated for reasonable periods of time. Continual assessment of these values is made on the basis of actual operating experience in BWR's (Reference 9).

While the noble radiogas source-term magnitude was established at 0.1 Ci/sec (t = 30 minutes), 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 2 data from operation in 1970 and several months in 1971 to more accurately characterize the noble radiogas mixture pattern for an operating BWR.

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

$$R_g = K_g\ Y\ \lambda\ [1\text{-}e^{-\lambda T}]\ [e^{-\lambda T}] \qquad (11.1\text{-}4)$$

With the exception of Kr-85 with a half-life of 10.74 years, 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 the rate of decay). So for practical purposes the term $[1\text{-}e^{-\lambda T}]$ 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 minutes has been used. When discussing long steady-state operation and leakage from the fuel (t = 0), the following simplified form of Equation 11.1-4 can be used to describe the leakage of each noble radiogas:

$$R_g = K_g y \lambda^m \qquad (11.1\text{-}5)$$

The constant, K_g describes the magnitude of leakage. The relative rates of leakage of the different noble radiogas isotopes is accounted for by the variable, m, the exponent of the decay constant, λ.

Dividing both sides of Equation 11.1-5 by y, the fission yield, and taking the logarithm of both sides results in the following equation:

$$\log (R_g/y) = m \log (\lambda) + \log (K_g) \qquad (11.1\text{-}6)$$

Equation 11.1-6 represents a straight line when log $R_{g/y}$ is plotted versus log (λ); m is the slope of the line. This straight line is obtained by plotting $R_{g/y}$ versus (λ) on logarithmic graph paper.

By fitting actual data from KRB and Dresden 2 (using least squares techniques) to the equation the slope, m, can be obtained. This can be estimated on the plotted graph. With radiogas leakage at KRB over the nearly 5-year period varying from 0.001 to 0.056 Ci/sec (t = 30 minutes) and with radiogas leakage at Dresden 2 varying from 0.001 to 0.169 Ci/sec (t = 30 minutes), the average value of m was determined. The value for m is 0.4 with a standard deviation of ±0.07. This is illustrated in Figure 11.1-1 as a frequency histogram. As can be seen from 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, or as has been done historically, the design basis is set by the total design-basis source-term magnitude at t = 30 minutes. With ΣRg at 30 minutes = 100,000 μCi/sec, K_g can be calculated as being 2.6 x 10^7, and Equation 11.1-4 becomes:

$$R_g = 2.6 \times 10^7\ \ y\lambda\ [1\text{-}e^{-\lambda T}]\ [E^{-\lambda T}] \qquad (11.1\text{-}7)$$

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 11.1-7. The resultant source terms are presented in Table 11.1-1 as leakage from fuel (t = 0) and after 30-minute decay. While Kr-85 can be calculated using Equation 11.1-7, 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 a few actual measurements.

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 minutes, and 100

μCi/sec of I-131. These values can be compared to the design-basis value of 100,000 μCi/sec for the summation of the same 13 noble gases, and 700 μCi/sec for I-131. Table 11.1-2 presents the source terms released to the reactor pressure vessel as a consequence of a Power Isolation Event, which is the only anticipated operational occurrence in which significant activity is expected to be released.

11.1.1.2 Radiohalogen Fission Products

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:

$$R_h = K_h y \lambda^n \qquad (11.1\text{-}8)$$

The constant, K_h, describes the magnitude of leakage from fuel. The relative rates of halogen radioisotope leakage is expressed in terms of n, the exponent of 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. This is illustrated in Figure 11.1-2 as a frequency histogram. As can be seen from this figure, variations in n were observed in the range of n = 0.1 to n = 0.9.

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

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

Using these judgment factors and experience to date, the design basis radiohalogen source terms from fuel were established based on I-131 leakage of 700 μCi/sec. This value, as seen in Figure 11.1-3, accommodates the experience data and the design basis noble radiogas source term of 0.1 Ci/sec (t = 30 minutes). With the I-131 design-basis source term established, K_h can be calculated as being 2.4 x 10^7 and halogen radioisotope release can be expressed by the following equation:

$$R_g = 2.4 \times 10^7\, y\lambda^{(0.5)}\, [1-e^{-\lambda T}]\, [e^{-\lambda T}] \qquad (11.1\text{-}9)$$

Concentrations of radiohalogens in reactor water can be calculated using the following equation:

$$C_h = \frac{R_h}{(\lambda + \beta + \gamma)M} \qquad (11.1\text{-}10)$$

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

The halogen release rate from the fuel can be calculated from Equation 11.1-9. Concentrations in reactor water can be calculated from Equation 11.1-10. The resultant concentrations are presented in Table 11.1-3.

11.1.1.3 Other Fission Products

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

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

Except for Np-239, trace concentrations of transuranic isotopes have been observed in 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} μCi/g or less, which is below the maximum permissible concentration in drinking water applicable to continuous use by the general public. The concentration of alpha-emitting plutonium radioisotopes is more than one order of magnitude lower than that of Cm-242.

Plutonium-241 (a beta emitter) may also be present in concentrations comparable to the Cm-242 level.

11.1.1.4 Nomenclature

The following list of nomenclature defines the terms used in equations for source-term calculations:

R_g = leakage rate of a noble gas radioisotope (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);

T = fuel irradiation time (sec);

t = decay time following leakage from fuel (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 water (μCi/g);

M = mass of water in the operating reactor (g);

β = cleanup system removal constant (sec^{-1}); where

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

γ = halogen steam carryover removal constant (sec^{-1}); where

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

11.1.2 Activation Products

11.1.2.1 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 presented in Section 12.2 (Table 12.2-3 and 12.2-4).

11.1.2.2 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 (Reference 8). The resultant concentrations are presented in Table 11.1-5. Carryover of these isotopes from the reactor water to the steam is estimated to be <0.1% (<0.001 fraction) (Reference 8).

11.1.2.3 Steam and Power Conversion System N-16 Inventory

This information is provided in Section 12.2.

11.1.3 Tritium

In a BWR, tritium is produced by three principal methods:

a. activation of naturally occurring deuterium in the primary coolant,

b. nuclear fission of UO_2 fuel, and

c. neutron reactions with boron used in reactivity control rods.

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

All of the tritium produced by activation of deuterium in the primary coolant is available for release in liquid or gaseous effluents. The tritium formed in a BWR from deuterium activation can be calculated using the equation:

$$R_{act} = \frac{\Sigma \phi V \lambda}{3.7 \times 10^4 P} \qquad (11.1\text{-}11)$$

where:

R_{act} = tritium formation rate by deuterium activation (μCi/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 will be present because of the H(η,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 (which will then be available for release in liquid and gaseous effluents) is more difficult to estimate. However, since Zircaloy-clad fuel rods are used in BWR's, essentially all fission product tritium will remain in the fuel rods unless defects are present in the cladding material (Reference 4).

The study made at Dresden 1 in 1968 by the U.S. Public Health Service suggests that essentially all of the tritium released from the plant could be accounted for by the deuterium

activation source (Reference 5). For purposes of estimating the leakage of tritium from defective fuel, it can be assumed that it leaks in a manner similar to the leakage of noble radiogases. Thus, use can be made of the empirical relationship described as the "diffusion mixture" used for predicting the source term of individual noble gas radioisotopes as a function of the total noble gas source term. The equation which describes this relationship is:

$$R_{dif} = Ky\lambda^{0.5} \qquad (11.1\text{-}12)$$

where:

R_{dif} = leakage rate of tritium from fuel (μCi/sec);

Y = fission yield fraction (atoms/fission);

λ = radioactive decay constant (sec^{1}); and

K = a constant related to total tritium leakage rate.

If the total noble radiogas source term is 10^5 μCi/sec after 30 minute decay, leakage from fuel can be calculated to be about 0.24 μCi/sec of tritium. To place this value in perspective in the USPHS study, the observed rate of Kr-85 (which has a halflife 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 minutes), the annual average tritium release rate from the fission source can be conservatively estimated at 0.12 ± 0.12 μCi/sec, or 0.0 to 0.24 μCi/sec.

Based on this approach, the estimated total tritium appearance rate in reactor coolant and release rate in the effluent is about 20 Ci/yr.

Tritium formed in the reactor is generally present as tritiated oxide (HTO) and to a lesser degree as tritiated gas (HT). Tritium concentration (on a weight basis) in the steam formed in the reactor will be the same as in the reactor water at any given time. This tritium concentration will also be present in condensate and feedwater. Since radioactive effluents generally originate from the reactor and power cycle equipment, radioactive effluents will also have this tritium concentration. The condensate storage tanks receive treated water from the liquid waste management system and reject water from the condensate system. Thus, all plant process water will have a common tritium concentration.

Off-gases released from the plant will contain tritium, which is present as tritiated gas (HT) resulting from reactor water radiolysis as well as tritiated water vapor (HTO). In addition, water vapor from the turbine gland seal steam packing exhauster and a lesser amount present in ventilation air due to process steam leaks or evaporation from sumps and tanks, will also contain tritium. The remainder of the tritium will leave the plant in liquid effluents or with solid wastes.

Recombination of radiolysis gases in the off-gas system will form water, which is condensed and returned to the main condenser. This tends to reduce the amount of tritium leaving in gaseous effluents. Reducing the gaseous tritium release will result in a slightly higher tritium

concentration in the plant process water. Reducing the amount of liquid effluent discharged will also result in a higher process coolant equilibrium tritium concentration.

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

The USPHS study at Dresden 1 estimated that approximately 90% of the tritium release was observed in liquid effluent, with the remaining 10% leaving as gaseous effluent (Reference 5). Efforts to reduce the volume of liquid effluent discharges may change this distribution so that a greater amount of tritium will leave as gaseous effluent. From a practical standpoint, the fraction of tritium leaving as liquid effluent may vary between 60% and 90% with the remainder leaving in gaseous effluent.

11.1.4 Fuel Fission Production Inventory and Fuel Experience

11.1.4.1 Fuel Fission Product Inventory

Fuel fission product inventory information is used in establishing fission product source terms for accident analysis and is, therefore, discussed in Section 15.

11.1.4.2 Fuel Experience

A discussion of BWR fuel experience including fuel failure, burnup, and thermal conditions under which this information was gained is presented in References 2, 3, and 6.

11.1.5 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 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 make the ventilation release relatively significant, General Electric has conducted measurements to identify and qualify these low-level release paths. General Electric has maintained an awareness of other measurements by the Electric Power Research Institute and other organizations; and routine measurements by utilities with operating BWR's.

Leakage of fluids from the process system will result in the release of radionuclides into plant buildings. In general, the noble radiogases will remain airborne and will be released to the atmosphere with little delay via the building ventilation exhaust ducts. The radionuclides will partition between air and water, and airborne radioiodines may "plateout" on metal surfaces, concrete, and paint. A significant amount of radioiodine remains in air or is desorbed from surfaces. Radioiodines are found in ventilation air as methyl iodide and as inorganic iodine which is here defined as particulate, elemental, and hypoiodous acid forms of iodine. Particulates will also be 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 (Reference 7). This report is periodically updated to incorporate

the most recent data on airborne emission. 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 the General Electric Company.

11.1.6 Radwaste System

Radioisotope source inventory in radwaste systems is given in Section 12.2.1.

11.1.7 Radioactive Sources in the Gas Treatment System

The radioactive sources for the gas treatment system are described in Subsection 11.3.2.1.2.

11.1.8 Source Terms for Component Failures

11.1.8.1 Off-Gas System Failure

The source terms for evaluation of the radiological consequences of component failures within the off-gas system are described in Table 12.2-11.

11.1.8.2 Liquid Radwaste System

The liquid radwaste tanks whose rupture can produce the worst off-site dose/contamination consequences are identified as either of the two phase separator tanks located in the basement of the radwaste building, or either of the two concentrate waste tanks, located on the mezzanine floor of the radwaste building. Radioisotope inventory of these tanks is given in Subsection 12.2.1.9.

11.1.8.3 Condensate Storage Tank

Radioisotope inventory of the condensate storage tank is given in Subsection 12.2.1.9.

11.1.9 Other Releases

11.1.9.1 Containment Purge

A listing of the yearly radioisotope releases from the containment building is given in Table 11.3-9. As noted at the bottom of the table, the releases due to drywell purging, which may occur four times per year, are not separated from gaseous releases from the containment building, and represent only a small portion of the total annual releases from the containment building.

11.1.9.2 Mechanical Vacuum Pump Off-Gas

Air is removed from the main condenser by means of a mechanical vacuum pump (MVP), during plant startup. In some operating BWR plants, the MVP has been used during refueling/maintenance periods to minimize airborne activity in the turbine building. The MVP operation is described in Subsection 10.4.2. The MVP exhaust is routed to the station HVAC exhaust stack. The mechanical vacuum pump exhaust air radioactive source term is presented in Table 11.3-9.

11.1.9.3 Turbine Gland Seal System

A listing of the yearly radioisotope releases from the operation of the turbine gland seal system is given in Table 11.3-9.

11.1.10 References

1. F. J. Brutschy, "A Comparison of Fission Product Release Studies in Loops and VBWR," Paper presented at the Tripartite Conference on Transport of Materials in Water Systems, Chalk River, Canada, February 1961.

2. H. E. Williamson and D. C. Ditmore, "Experience with BWR Fuel Through September 1971," NED0-10505, (Update) May 1972.

3. R. B. Elkins, "Experience with BWR Fuel Through September 1974," NED0-20922, June 1975.

4. J. W. Ray, "Tritium in Power Reactors," Reactor and Fuel Processing Technology, 12 (1), pp. 19-26, Winter 1968-1969.

5. B. Kahn, et al, "Radiological Surveillance Studies at a Boiling Water Nuclear Power Reactor," BRH/DER 70-1, March 1970.

6. H. E. Williamson and D. C. Ditmore, "Current State of Knowledge of High Performance BWR Zircaloy Clad UO_2 Fuel," NEDO-10173, May 1970.

7. T. R. Marrero, "Airborne Releases From BWRs for Environmental Impact Evaluations," NED0-21159, March 1976.

8. R. B. Klkins, "Experience With BWR Fuel Through December 1976," NED0-21660, July 1977.

9. J. M. Skarpelos and R. S. Gilbert, "Technical Derivation of BWR 1971 Design Basis Radioactive Material Source Terms," NEDO-10871, March 1975.

TABLE 11.1-1
NOBLE RADIOGAS SOURCE TERMS

ISOTOPE	HALF-LIFE	SOURCE TERM t = 0 (μCi/sec)	SOURCE TERM t = 30 min (μCi/sec)
Kr-83m	1.9 hr	3.4 E3	2.9 E3
Kr-85m	4.4 hr	6.1 E3	5.6 E3
Kr-85	10.8 yr	10 to 20*	10 to 20*
Kr-87	76 min	2.0 E4	1.5 E4
Kr-88	2.8 hr	2.0 E4	1.8 E4
Kr-89	3.2 min	1.3 E5	1.8 E2
Kr-90	33 sec	2.8 E5	--
Kr-91	9.8 sec	3.3 E5	--
Kr-92	3.0 sec	3.3 E5	--
Kr-93	2.0 sec	9.3 E4	--
Kr-94	1.4 sec	2.3 E4	--
Kr-95	0.8 sec	2.1 E3	--
Kr-97	1.0 sec	1.4 EI	--
Xe-131m	11.8 day	1.5 EI	1.5 EI
Xe-133m	2.3 day	2.9 E2	2.8 E2
Xe-133	5.3 day	8.2 E3	8.2 E3
Xe-135	9.1 hr	2.2 E4	2.2 E4
Xe-135m	15.6 min	2.6 E4	6.9 E3
Xe-137	3.9 min	1.5 E5	6.7 E2
Xe-138	17.5 min	8.9 E4	2.1 E4
Xe-139	43 sec	2.8 E5	--
Xe-140	16.0 sec	3.0 E5	--
Xe-141	1.7 sec	2.4 E5	--
Xe-142	1.5 sec	7.3 E4	--
Xe-143	1.0 sec	1.2 E4	--
Xe-144	1.0 sec	5.6 E2	--
TOTALS		2.5 E6	1.0 E5

* Estimated from experimental observations

TABLE 11.1-2
POWER ISOLATION EVENT - ANTICIPATED OCCURRENCE

ISOTOPE	SPIKING ACTIVITY (Ci)
I-131	1310
132	1997
133	3120
134	3370
135	2995
Kr-83m	562
85m	1373
85	312
87	2683
88	3806
89	4992
Xe-131m	62
133m	187
133	7238
135m	1123
135	6864
137	6552
138	6614

TABLE 11.1-3
HALOGEN RADIOISOTOPES IN REACTOR WATER

ISOTOPE	HALF LIFE	CONCENTRATION (μCi/g)
Br-83	2.40 hr	1.7 E-2
Br-84	31.8 min	3.5 E-2
Br-85	3.0 min	2.2 E-2
I-131	8.065 day	1.5 E-2
I-132	2.284 hr	1.5 E-1
I-133	20.8 hr	1.0 E-1
I-134	52.3 min	3.0 E-1
I-135	6.7 hr	1.5 E-1

TABLE 11.1-4
OTHER FISSION PRODUCT RADIOISOTOPES IN REACTOR WATER

ISOTOPE	HALF LIFE	CONCENTRATION (μCi/g)
Sr-89	50.8.day	3.3 E-3
Sr-90	28.9.yr	2.5 E-4
Sr-91	9.67 hr	8.1 E-2
Sr-92	2.69 hr	1.4 E-1
Zr-95	65.5.day	4.3 E-5
Zr-97	16.8.hr	3.6 E-5
Nb-95	35.1.day	4.5 E-5
Mo-99	66.6.hr	2.5 E-2
Tc-99m	6.007 hr	9.4 E-2
Tc-I0I	14.2.min	2.0 E-1
Ru-103	39.8.day	2.1 E-5
Ru-106	368.day	2.8 E-6
Te-129m	34.1.day	3.7 E-4
Te-132	78.0.hr	1.5 E-2
Cs-134	2.06 yr	1.7 E-4
Cs-136	13.0.day	1.1 E-4
Cs-137	30.2.yr	2.6 E-4
Cs-138	32.3.min	2.5 E-1
Ba-139	83.2.min	2.0 E-1
Ba-140	12.8.day	9.5 E-3
Ba-141	18.3.min	2.4 E-1
Ba-142	10.7 min	2.3 E-1
Ce-141	32.53 day	4.3 E-5
Ce-143	33.0 hr	3.9 E-5
Ce-144	284.4 day	3.8 E-5
Pr-143	13.58 day	4.1 E-5
Nd-147	11.06 day	1.5 E-5
Np-239	2.35 day	2.6 E-1

TABLE 11.1-5
NONCOOLANT ACTIVATION PRODUCTS IN REACTOR WATER

ISOTOPE	HALF LIFE	CONCENTRATION (μCi/g)
Na-24	15.0 hr	2.0 E-3
P-32	14.31 day	2.0 E-5
Cr-51	27.8 day	5.0 E-4
Mn-54	313.0 day	4.0 E-5
Mn-56	2.582 hr	5.0 E-2
Co-58	71.4 day	5.0 E-3
Co-60	5.258 yr	5.0 E-4
Fe-59	45.0 day	8.0 E-5
Ni-65	2.55 hr	3.0 E-4
Zn-65	243.7 day	2.0 E-6
Zn-69m	3.7 hr	3.0 E-5
Ag-110m	253.0 day	6.0 E-5
W-187	23.9 hr	3.0 E-3

11.2 LIQUID WASTE MANAGEMENT SYSTEMS

This section describes the capabilities of the station to collect, process, handle, store, and dispose of liquid radioactive waste generated as the result of normal operation, including anticipated operational occurrences.

11.2.1 Design Bases

11.2.1.1 Design Objectives and Criteria

The objectives of the liquid waste management systems are as follows:

a. to collect, monitor, and treat all potentially radioactive liquid wastes produced by the station during normal and anticipated abnormal station operation for reuse within the limits of the station's overall water balance requirements;

b. to minimize, monitor, and control processed liquid waste releases to ensure that quantities and concentrations of radioactive material in such releases are "as low as is reasonably achievable" in accordance with 10 CFR 20 and 10 CFR 50;

c. to minimize the volume of waste material requiring processing by the solid waste management system and subsequent shipment from the plant; and

d. to give proper consideration (through equipment selection, arrangement, and shielding) to keeping radiation exposure of in-plant personnel "as low as is reasonably achievable".

Table 11.2-1 compares expected typical maximum effluent concentrations from the station for normal operations and anticipated operational occurrences, as determined using the BWR-GALE Computer Code (Reference 1). Table 11.2-2 compares the calculated radiological dose due to liquid effluents, based upon Regulatory Guide 1.109 (Reference 2) to the numerical design objectives of Appendix I to 10 CFR 50 and the dose limits of 10 CFR 20. Waste treatment systems similar to the Clinton Power Station (CPS) system have been used effectively in several existing plants. The data in Tables 11.2-1 and 11.2-2 show that the liquid radwaste system is designed to produce effluent concentrations which will satisfy the required limitations and objectives.

11.2.1.2 Component Specifications

In order that the liquid radwaste system does not limit power generation under any conditions, the system is sized to handle normal expected liquid waste inputs on the basis of volume, water quality, and activity. The system size is also sufficient to collect and treat the maximum expected volume and activity of those liquid wastes produced by any anticipated abnormal or transient condition of station operation. Table 11.2-3 gives the design parameters of the various liquid radwaste system components. Subsection 11.2.2.7 discusses system operation during abnormal or transient conditions.

The amount of resin material used in each demineralizer is approximately 90 ft^3 for anion and cation resin. Charcoal may be added to the demineralizer vessel, in addition to the normal load of resin, to remove components from the waste, normally organics. The charcoal will not remove significant activity from the waste.

In the chemical regenerant waste subsystem, the specific gpd can be found in Table 11.2-7, Item 5, Liquid Waste Processing System, Section d.

On revised P&IDs, the conductivity instrument is shown for each demineralizer. (Q&R 460.8)

11.2.1.3 Seismic Design and Quality Group Classification

The liquid radwaste system has no seismic requirements. However, the portion of the Radwaste Building which is below grade is Seismic Category I.

Clinton has committed to comply with Regulatory Guide 1.143, Revision 0, as specified in USAR Section 1.8. However, for Q&R 460.7, the CPS design is compared with Regulatory Guide 1.143, Revision 1 requirements.

a. The structure which houses the liquid and solid radwaste systems is designed as non-Seismic Category I above grade but Seismic Category I below grade (see Chapter 3, Table 3.2-1). This complies with regulatory position 5 of Regulatory Guide 1.143, Revision 1, considering the positions 1.1.3 and 3.1.3 of the Regulatory Guide. The volume of the basement up to grade level is 1,203,300 ft^3. The maximum liquid inventory is 95,500 ft^3. Therefore, there is sufficient volume available in the Seismic Category I basement to contain the maximum liquid inventory even allowing for rubble that may be present from collapse of above grade structures.

b. In the gaseous radwaste system, the components which are subject to the seismic requirements of Regulatory Guide 1.143 are the charcoal delay tanks. These delay tanks are located below grade in a building which has been Seismic Category I designed, and therefore, exceeds the requirements of Regulatory Guide 1.143, Sections 2 and 5. (Q&R 460.7)

The equipment (including tanks, pumps, valves, and piping) in the liquid radwaste system containing radioactive wastes conforms to the codes and standards specified in Regulatory Guide 1.26, as applicable to Quality Group D, and to codes specified in Table 1 of Regulatory Guide 1.143, "Design Guidance for Radioactive Waste Management System, Structure, and Components Installed in Light-Water-Cooled Nuclear Power Reactor Plants" (Reference 3).

No piping or valves of the liquid radwaste system function as part of the containment isolation system.

11.2.1.4 Quality Control

Quality Control Programs were established to assure that the design, construction, and testing requirements for the liquid radwaste system were satisfied. The following areas were covered by the programs:

a. Design and Procurement Document Control - Measures were established to insure that the appropriate guidelines of Regulatory Guide 1.143 (formerly Branch Technical Position, ETSB No. 11-1) were included in design and procurement documents and that deviations therefrom were controlled.

b. Measures were established to assure that purchased material, equipment, and construction services conformed to procurement documents.

c. An inspection program verified conformance with the documented instructions, procedures, and drawings for accomplishing the activity.

d. Procedures were 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.

e. Inspection and testing procedures were used to identify items which satisfactorily passed required inspections and tests.

f. Corrective Action - Measures were established to assure that conditions adverse to quality, such as failures, malfunctions, deficiencies, defective material and equipment, and nonconformances were promptly identified and corrected.

11.2.1.5 Special Design Features

The liquid radwaste system is designed to utilize features and equipment arrangements consistent with the intent of Regulatory Guide 8.8 to minimize personnel exposure. Special design features were also incorporated to improve operation. Discussion of these features as they relate to normal operations and maintenance operations is provided in the following subsections.

11.2.1.5.1 Normal Operations

The following steps were taken to ensure that occupational exposures will be "as low as is reasonably achievable" during normal operating situations:

a. Equipment operation for all treatment systems is from the radwaste operations center. This is in a remote shielded location.

b. Sampling lines that may contain unprocessed wastes are routed to central sampling stations located away from high radiation areas.

c. Wherever practical, valves in processing lines are installed with valve stems vertically upward. This orientation minimizes leakage, maintenance requirements and crud build-up in the stem packing area.

d. Tanks containing unprocessed radioactive waste have vents hard piped to HVAC system exhaust ducts.

e. Tanks and equipment containing radioactive waste are located in shielded compartments with labyrinth entrances.

f. Tanks which receive liquid radwastes that are potentially high in suspended solids are vertical with cone bottoms.

g. Tanks have provision for periodic flushing to minimize the buildup of radioactive sediments.

h. Filter and demineralizer vessels are located in shielded compartments accessible through hatches in the ceiling above the vessels. Penetrations are provided to

allow visual monitoring, by use of boroscopes, of sight glasses supplied with the demineralizer vessels.

i. Piping between the evaporators and the concentrated waste tanks and the solid radwaste handling system is heat traced to minimize crystallization of concentrated fluids.

j. Valves that are normally used are not located inside tank compartments containing unprocessed waste.

k. Butt welding using consumable inserts or open butt welding technique is the preferred welding connection. Backing rings were not used. Socket welds were used for portions of some of the 2 inch and smaller piping systems.

l. Compartments containing tanks or equipment which contain unprocessed waste have floor and wall coatings selected for ease in decontamination in case of spills.

m. Unprocessed waste tank discharge lines have flush and drain connections.

n. Radioactive waste piping 2-1/2" and larger is routed by the designer. The designer routes the 2" and smaller piping considering potential personnel exposure, and issues the drawings for "Construction Reference." The constructor then follows the routine as nearly as is practicable. Most of the radwaste piping actually has been designed to fit within shielded equipment cubicles or pipe tunnels. These three practices, with potential radiation considered throughout, will help to assure that personnel exposures will be kept as low as reasonably achievable.

o. Piping which could enable unprocessed waste to bypass processing equipment and contaminate either the waste sample tanks or the excess water tanks is provided with locked-closed shutoff valves.

Clinton has committed to comply with Regulatory Guide 1.143 (Revision 0) as specified in Section 1.8. However, Clinton also meets the requirements of position 1 of Regulatory Guide 1.143 Revision 1.

a. All Radwaste System tanks have both monitoring and high-level annunciation. The remainder of the tanks listed in Table 11.2-4 are not radwaste tanks and do not have both level indication and high-level alarms as indicated below along with justification.

   <u>Condenser Vacuum Water Separation Tank</u>:

   Supply line will auto-close by makeup float valve at normal level. Overflow is hard piped to the drain system.

   <u>Low Pressure Heater Flash Tank</u>:

   This is a closed system vessel, and level monitoring and annunciation is not required.

Condensate Polisher Demineralizer URC Receiving Tank; Condensate Polisher Demineralizer URC Storage Tank; Condensate Polisher Demineralizer Resin Separation and Cation Regeneration Tank;

Condensate Polisher Demineralizer Anion Regeneration Tank;

Condensate Polisher Demineralizer Resin Mixing and Storage Tank;

These are closed vessels, and level monitoring and annunciation is not required.

Condensate Polisher Demineralizer Hot Water Tank:

This is a closed system and level monitoring and annunciation is not required.

Fuel Pool Skimmer Surge Tank:

Level is monitored by computer and abnormal level conditions are alarmed in computer.

RCIC Storage Tank:

The tank level is monitored and the makeup supply valve on normal fillup will auto-close so high level annunciation is not required.

b. There are no outdoor Radwaste System tanks. Both the cycled condensate and RCIC storage tanks are located within their own retaining basin. The walls of each retaining basin are either earthen dikes, a combination of concrete walls and earthen dikes, or a carbon steel containment tank with the basin sized to retain the full capacity of the enclosed tank. (See Subsection 9.2.6.3.)

The cycled condensate storage tank is equipped with level transmitters which provide high and low level alarms and level indication in the main control room. An additional level switch is mounted in the common standpipe overflow line located indoors. This switch activates a high level alarm in the main control room indicating that the tanks are overflowing. By operating a handswitch located on the main control board, a valve can be opened which discharges excess condensate to radwaste sample tanks. (See Subsection 9.2.6.5.)

The cycled condensate storage tank has provisions for sampling at process sample panel 1PL22J (see Drawing M05-1012, Sheet 1, coordinates B-6). The RCIC storage tank has provisions for sampling at a low point drain in the system piping for analysis (see Drawing M05-1079, Sheet 2, coordinates A-7).

In the event of overflow from any of the above tanks, the water will be collected within the berm, or steel containment tank. Then, based on evaluation of the incident, the water may have to be transferred by portable pump hose to a tank and, if necessary, the contaminated soil removed and shipped to an authorized burial facility. (Q&R 460.10)

11.2.1.5.2 Maintenance Operations

Equipment is arranged to separate sources of high radiation and lower radiation areas. In the evaporator packages, components that may contain highly radioactive fluids, such as concentrate pumps and recirculation pumps, are separated and shielded from each other and from components of lesser radioactivity, such as distillate pumps.

Redundant components are separately shielded, or have space provided for temporary shielding. In general, components which may require maintenance are capable of being flushed prior to maintenance operations.

Air-operated valves are located outside of major equipment compartments to minimize exposure from tanks or components during valve maintenance. Components that may require periodic inspection, maintenance, lubrication, etc. are located so as to be readily accessible and separated from high radiation sources, such as tanks, by shielding. Pumps are provided with mechanical seals to minimize leakage.

11.2.1.6 Uncontrolled Release of Radioactive Materials

Provisions are made to preclude the uncontrolled release of radioactive materials due to tank overflows.

For liquid radwaste tanks, the following design considerations are provided:

a. Tank level instrumentation annunciates tank high-high level condition in the radwaste operations center.

b. At the batch level (i.e., high level) conditions, station operating procedures will direct diversion of wastes from the tank to other tanks.

c. Tank overflows are piped to a drain sump or to an overflow pipe of another tank which is connected to a sump of a different subsystem.

d. Sumps are provided with redundant pumps. Sumps are level instrumented and logic is provided to start and stop the pumps automatically.

e. Tank overflow and sump pump discharge piping is routed in a manner so as to prevent overflow material from being returned to the overflowing tank.

f. Tank compartments have curbs to contain leaks or spills. These compartments are also provided with floor drains.

A list of tanks which contain potentially radioactive fluid and are located outside of the containment is provided in Table 11.2-4. This list indicates the specific types of features utilized on each tank to prevent an uncontrolled release.

The potential for a single operator error to cause an uncontrolled release to the environment is considered remote. Normal releases of radioactive processed liquid waste to the environs can be made at one point in the plant and no uncontrolled release can be made due to any single violation of the discharge procedures. The potential for a single equipment failure to cause an uncontrolled release to the environment is discussed in Subsections 2.4.12 and 2.4.13.

For condensate storage tanks, which are located outdoors, an analysis of storage facility failure is provided in Subsection 9.2.6.

11.2.2 System Description

The liquid radwaste system consists of four major subsystems: (1) the equipment drain subsystem, (2) the floor drain subsystem, (3) the chemical waste subsystem, and (4) the laundry waste subsystem. Figure 11.2-1 is a Liquid Radwaste System Flow Chart. A liquid radwaste process diagram plus piping and instrumentation diagrams for each subsystem are shown in Drawings M05-1081 and M05-1085. The radwaste building equipment arrangement is presented in Drawings M01-1502, M01-1508, M01-1514, and M01-1531. Design-basis activity concentrations of radioactive nuclides in the input waste streams to the various liquid waste subsystems are listed by nuclide in Table 11.2-5. These design-basis activity concentrations and shielding requirements are discussed in Chapter 12. Expected flows into the liquid radwaste system are shown in Table 11.2-6.

11.2.2.1 Equipment Drain Subsystem

This subsystem collects and processes high purity (low conductivity) waste such as equipment drains. This water is treated by settling, filtration, and/or ion exchange demineralization and returned after appropriate sampling to the cycled condensate storage tank for station reuse.

Major input sources to the equipment drain subsystem include:

a. backwash from condensate filters, and radwaste demineralizers, and ultrasonic resin cleaner;

b. decant from waste sludge tank, reactor water cleanup phase separator tank, fuel pool filter/demineralizer sludge tank, and spent resin tank; and

c. flows from equipment drain tanks and sumps in drywell containment, auxiliary, turbine, radwaste, and fuel buildings.

Two waste collector tanks receive regular inputs from the equipment drain system and infrequent inputs of clear decant from the phase separators, spent-resin tank, and sludge tanks. These tanks are sized to hold approximately 2 days' accumulation of wastes. Prior to processing, 500 gallons or less of tank "bottoms" are pumped to a waste sludge tank to remove settled solids and minimize downstream filter and demineralizer loading. Then, unless input is known to have included unusual amounts of suspended solids, tank contents are processed through the waste filters and demineralizers. When suspended solids content is unusually high, tank contents may be transferred to the chemical/floor drain waste subsystem for treatment as "off-standard" equipment drain wastes.

Two waste surge tanks receive infrequent inputs from Unit 1, including occasional unexpected large quantities of equipment drain waste or off-standard batches that may be returned as a result of equipment malfunction or otherwise failing to meet criteria appropriate to their planned disposition. These tanks will regularly receive input of condensate filter backwash water, resin backwash water from the condensate and waste demineralizer regeneration system when resin is being regenerated (refer to Subsection 11.2.2.8), as well as rinse and sluice water from the ultrasonic resin cleaner. These wastes, although low in dissolved solids, may contain appreciable suspended solids, mainly resin fines. These tanks, therefore, serve as settling

tanks for these wastes and are equipped with a bottom discharge port and a decanting port for normal discharge. Otherwise, their operation is similar to that described above for the waste collector tanks.

The three waste filters remove suspended solids prior to demineralization in order to extend the useful ion exchange life of the demineralizer resins. The filters are operated in parallel to ensure that filtration precedes demineralization independent of the configuration of the demineralizers. In normal operation one filter processes the effluent, while the other two filters are available as backups. Filter backwash is periodically discharged to a waste sludge tank. The processed effluent from these filters is normally directed to the waste demineralizers. On occasion, waste from the chemical waste and floor drain subsystems may be processed through these filters prior to being processed through the evaporator.

Three waste demineralizers are provided. They are arranged so that two may be operated in series as "rougher" and "polishing" units, with the third in reserve. Upon exhaustion of the "rougher" unit, it can be taken off-line and replaced by the "polishing" unit, with the reserve unit being put into polishing service. This arrangement provides the maximum decontamination factor and protection against unexpected ion "breakthrough" in the first, more heavily loaded demineralizer. The piping and valving allow alternative demineralizer configurations to be used. Alternate configurations can consist of a demineralizer operating individually or in parallel with one or two others. Nominal operation will be as cascaded pairs with the third demineralizer used as a deep bed charcoal filter. Effluent from the demineralizers is normally routed to one of the waste sample tanks. Conductivity instrumentation is provided in the output line of each demineralizer to allow its performance to be monitored in the radwaste operations center.

An additional conductivity monitor is provided prior to the waste sample tanks that will automatically divert an "off-standard" (high conductivity) effluent back to a waste surge tank for recycle to protect the waste sample tanks from contamination due to demineralizer malfunction. The waste demineralizers also receive input from the floor drain and chemical waste evaporator monitor tanks. This input consists of evaporator distillate that has been sampled and requires only ion exchange polishing to be suitable for reuse or discharge.

Three waste sample tanks are provided, and they serve several functions. The main function is to provide for monitoring and sampling of all processed liquid waste to ensure that the appropriate criteria are met for either reuse or discharge. They also serve to provide "ballast" capacity so that short-term excesses in station water inventory will not unnecessarily require the discharge of treated waste water. Another function of the three waste sample tanks is to allow segregation of waste by source and quality. During normal operation, one of these tanks receives processed waste which will then be sampled and normally returned to condensate storage. A second waste sample tank is available to receive treated waste water while the contents of the first tank are being sampled. The third waste sample tank can receive processed waste which is suitable for discharge but may not meet reuse criteria (i.e., organic content).

When it is necessary to discharge treated wastes from the station, the contents of any of the waste sample tanks is transferred, after sampling, to the excess water tanks for subsequent sampling and release. Waste sample tanks are equipped with recirculation lines and eductors to ensure complete mixing before sampling. Effluent discharged via the bypass valve is still subject to the same controls as a normal discharge. The only item bypassed is an excess water tank.

Two excess water tanks are provided, either of which can receive input from any of the waste sample tanks. They may also receive input from either or both of two channels that are provided to deal with large quantities of low activity water. One of these is from the laundry sample tanks, and the other comes from the waste filters. The main function of these tanks is to store treated waste water that is in excess of the station water balance requirements and to allow for its controlled release to the environment. These tanks are provided with recirculation lines and eductors to allow complete mixing before sampling.

These tanks discharge to the service water discharge pipe through a common line. Alternatively, they may be returned to either the condensate storage tanks or the waste surge tank for reprocessing through the subsystem.

11.2.2.2 Floor Drain Subsystem

The floor drain subsystem collects and processes low purity (high conductivity) waste from the Unit 1 floor drain system. These wastes are normally too high in conductivity for efficient ion exchange treatment. They may also be high in suspended solids. Treatment can be by settling, evaporation, adsorption (normally for organic removal), and ion exchange demineralization for return to the condensate storage tank or for discharge from the station after appropriate sampling.

Major input sources to this subsystem are wastes from the floor drain collector tanks and sumps in the following buildings: containment, turbine, auxiliary, radwaste, control, and fuel.

Floor drain oil separators are used to prevent oil from entering the liquid radwaste processing stream, and thus avoiding potential problems in attaining high-quality effluent for return to condensate storage or for plant discharge. Oil is separated from the water on the basis for the difference in their specific gravities. Oil which is collected on the surface of the water is removed by a skimming process. The effluent from the oil separator overflows, by gravity, to the floor drain sump.

In response to the concern regarding the oil separators, Chapter 9, Section 9.3.3.2.2 gives a description of the oil separators and their location with respect to process flow. Due to the fact that the oil separators are located upstream of the floor drain sumps, they are not a part of the Liquid Radwaste System so have not been included in the Radwaste Process Flow Diagrams.

These oil separators are part of the floor drain system which is described in Chapter 9.

With regard to the disposal of contaminated oil collected by these separators, the oil is packaged for shipment in accordance with approved site procedures. (Q&R 460.9)

These regular wastes are collected in the two floor drain collector tanks. The operation of the floor drain collector tanks is the same as described for the waste collector tanks. These tanks discharge to the floor drain evaporator feed tanks for further processing. Interconnecting piping is provided to allow optional prefiltration through the waste filters.

Two floor drain surge tanks take infrequent inputs from Unit 1, including occasional unexpected or unusual volumes of floor drains and "off-standard" batches returned for recycling through the subsystem due to failure to meet the criteria for their intended disposition. The unit 1 Floor Drain Surge Tank is normally used to collect distillate from Radwaste evaporators. The operation of the floor drain surge tanks is as described for the waste surge tanks.

The evaporator feed tanks receive input from the floor drain collector and/or surge tanks and are used for conditioning these wastes for subsequent treatment by evaporation. This conditioning may include any or all of the following: settling, flocculation, and chemical addition for pH control. A small volume of tank "bottoms" is then pumped to a waste sludge tank to remove precipitated solids prior to directing tank discharge to the appropriate evaporator package for further processing.

The floor drain evaporator packages serve a two-fold purpose. Their first function is to provide a condensed distillate that is substantially free of radioactivity and dissolved solids, suitable for reuse after, at most, ion exchange polishing. The second function of the evaporator packages is to concentrate the residual wastes into as small a volume as is practicable for delivery to the solid radwaste handling system for solidification, packaging, and offsite shipment. The condensed distillate from the evaporator packages is discharged to the floor drain evaporator monitor tanks. The concentrates or "bottoms" from the evaporators are discharged to the concentrated waste tanks and subsequently transferred to the waste solidification system.

The design of these evaporator packages is identical to the chemical waste evaporator package to permit the treatment of either mixed or single streams of floor drain, laundry drain, and/or chemical waste. Crossties are provided to feed any of the three evaporator packages with the floor drain evaporator feed tanks, laundry sample tanks, or chemical waste process tanks. When processing is in mixed stream mode, one evaporator each may process the wastes from the collector or surge tanks, and the third is available as a backup for either stream.

Two floor drain evaporator monitor tanks are provided; one following each of the floor drain evaporator packages. These tanks serve two purposes. The first is to facilitate evaporation batches to be sampled prior by providing holdup volume for typical batches to be sampled prior to further treatment. The second function is to allow for continuous monitoring of evaporator performance to diagnose maintenance requirements.

The contents of these tanks are recirculated while they are filling or after they are filled. The recirculation lines are fitted with conductivity monitors that display in the radwaste operations center. The contents of the floor drain evaporator monitor tanks are normally discharged to the equipment drain system for polishing. Discharge to the waste demineralizers is interlocked to the temperature monitors of the tank content to protect the resin bed in the demineralizers. In "off-standard" conditions, the floor drain evaporator monitor tanks can be discharged back to the floor drain surge tank for recycling through the system. The floor drain evaporator monitor tanks are crosstied with the chemical waste evaporator monitor tank.

11.2.2.3 Chemical Waste Subsystem

This subsystem processes the highest conductivity water in the liquid radwaste system. Major types of waste processed in this subsystem include:

a. radwaste demineralizer regenerants (if resin is regenerated),

b. flows from decontamination drains, and

c. flows from laboratory drains.

These wastes are potentially high in radioactivity, conductivity, and suspended solids including some resin fines. Processing of these wastes is by settling, chemical neutralization,

evaporation, ion exchange demineralization, and whenever practical, holdup for radioactive decay. On occasion, waste may be processed through the waste filters prior to being processed through the evaporator.

Two chemical waste collector tanks take regular inputs from the sources noted above. Wastes are usually held in these tanks for 2 or more days, with the optimum collection, holding, and discharge cycle determined by the requirements of the regeneration cycle for the polishers. These tanks are sized to provide sufficient surge capacity to allow for variations in this cycle.

Since some settling of suspended solids will take place in this holding period, provision is made for pumping small quantities of tank "bottoms" to a sludge tank prior to transfer for further processing. Provision is also made for the recirculation and sampling of the contents of the chemical waste collector tanks, although this may not be done regularly. These tanks normally discharge to the chemical waste processing tanks, or may discharge to the radwaste filters first for optional prefiltration.

Two chemical waste processing tanks serve to assemble chemical waste batches and condition them for subsequent feeding to the appropriate waste evaporator. The operation of these tanks is identical to the floor drain evaporator feed tanks except that an additional provision is made to permit tank discharge to the floor drain evaporator feed tanks via the waste filters.

The function and operation of the chemical waste evaporator package is identical to the floor drain evaporator packages. The condensed distillate from this evaporator package is normally discharged to the chemical waste evaporator monitor tank.

The function of the chemical waste evaporator monitor tank is identical to, and is crosstied with, the floor drain evaporator monitor tanks. Discharge from this tank may be sent to the waste demineralizers, the waste sample tanks, the floor drain surge tanks, or the chemical waste collection tanks.

11.2.2.4 Laundry Waste Subsystem

The laundry waste subsystem receives waste from the station laundry drains, personnel decontamination showers, and any other contaminated sources that may be high in detergent content. These wastes are normally very low in radioactivity content averaging about 1×10^{-4} μCi/cc. The system is designed to allow for complete recycling of all processed waste with return to the condensate storage system.

The laundry wastes are accumulated in the two laundry waste collector tanks. Two tanks are provided to allow the contents of one to be mixed and sampled while the other is filling. These tanks serve to accumulate suitable batches for treatment. A recirculation line with sample point is provided for each tank to assure adequate mixing prior to transfer.

Two laundry drain filters are provided primarily to remove lint and coarse particles from laundry waste before it enters the chemical waste subsystem.

Two laundry sample tanks are provided to receive waste processed through the laundry filters. These tanks are also equipped with a recirculation line and sample point. The recirculation line is equipped with a bypass valve arrangement which allows waste to be transferred to the chemical waste evaporator at either 17 gpm or 30 gpm. These two flow rates allow laundry waste to be transferred directly to the chemical waste evaporator at feed rates which

accommodate the two primary operating levels of the evaporator package. Laundry waste can also be transferred from the sample tank directly to the chemical waste collector tank for storage and/or blending with chemical waste prior to processing tnrouqh the evaporator. This would be done if dilution, neutralization, or settling of laundry waste is required. This subsystem is designed to accommodate the large detergent accumulation rate expected during major station maintenance outages.

Although water from the laundry sample tanks normally is sent to the chemical waste system for evaporation processing for reuse, processed laundry waste maybe routed to the excess water tanks for release from the station. It can also be returned to the laundry collection tank.

11.2.2.5 Control and Instrumentation

The liquid radwaste system is controlled and monitored from the liquid radwaste control panel (LRCP) located in the radwaste operations center (ROC). Radwaste filter, radwaste demineralizer, waste evaporator, BOP sump pump instrumentation, and solid radwaste handling system control panels are also located in the radwaste operations center.

Instrumentation on system tanks includes, as a minimum, high-level detection for LRCP annunciation and low-level detection for pump cutoff.

Other types of parameters monitored in the liquid radwaste system are as follows:

a. demineralizer effluent conductivity,

b. evaporator effluent conductivity,

c. radioactivity level of effluent being discharged,

d. discharged effluent volume totalizing,

e. waste flow rates,

f. evaporator system temperature and pressure, and pump discharge pressure.

The above information allows the operator to evaluate equipment performance and monitor the waste accumulation rate.

The LRCP is approximately 54 feet long and laid out in a logical fashion, with control switches for pump and valve operation and level indicators grouped in subsystem fashion and arranged in a full graphic display. Annunciators and recorders are located on the vertical board above the graphic, with the respective window lights in the vicinity of the monitored device. Each valve hand switch has full open and full closed indicator lights. During the opening or closing of a valve, both lights are illuminated until the open or closed extreme is reached. In the open or closed position, the red or green light will be illuminated, respectively. All pump hand switches are provided with a third light (amber) to indicate automatic trip of the pump.

The vendor supplied control panels for the radwaste filters, demineralizers, and evaporators are located facing the front of the LRCP. These panels contain the necessary controls and instrumentation to safely and efficiently operate the process equipment.

11.2.2.6 Performance Testing

Actual system performance tests (without radioactive materials) for each component are performed prior to plant operation to ensure that the equipment performs as specified. Shop tests were performed on most equipment to ensure that it met the performance requirements prior to shipment.

Once plant operation has begun, a decrease in processing effectiveness will be indicated through a trend analysis program. Then, based upon the nature of the problem, either effectiveness testing or maintenance will be performed.

11.2.2.7 Abnormal Operating Conditions

Certain "anticipated operational occurrences" were considered and flexibility is built into the liquid radwaste system design in order to allow the operator to adjust processing methods to meet the requirements of these conditions.

The following occurrences define the design-basis "anticipated operational occurrences" for the liquid radwaste system:

a. startup,

b. large condenser tube leakage,

c. shutdown,

d. large fuel leak rates - reactor coolant activity above technical specifications,

e. maximum reactor coolant leak rate,

f. refueling and maintenance outages, and

g. unscheduled reactor trips.

These anticipated operational occurrences are discussed in greater detail in the succeeding subsection.

11.2.2.7.1 Initial Startup

Various operations associated with the initial startup can produce flow volumes in both the equipment drain and floor drain processing subsystems of up to 100,000 gallons in a single day, with the highest 3-day average about 50,000 gallons per day at irregular intervals over periods of 2 to 3 months. Most of the liquid radwaste produced during such time is of low radioactivity content. The volumes of waste water involved and the fact that much of it originates from sources outside the primary system makes it difficult to maintain the station's water balance except by discharging larger than normal volumes of processed waste.

The design features provided to meet these demands are discussed by subsystem as follows:

a. Equipment Drain Subsystem - Processing in this subsystem is provided by the waste filters and waste demineralizers. Since in any "shakedown" period such as station startup, some equipment difficulties are almost inevitable, a major

design assumption is that the liquid radwaste processing capacity will be limited to a 25% availability during this period. The design flow capacity of the filters and demineralizers is such that 25% of the daily throughput design capacity is sufficient to process a maximum day's collected liquid waste volume. Since the assumed processing capacity might not be available in a single day, crossties are provided between the waste collector and surge tanks. This provides volume retention capacity for 2 or more days of input so that the system can handle peak surges over any 3 or 4-day period. Similarly, the three waste sample tanks are crosstied, as are the two excess water tanks, to provide storage volume for 1 to 2 full days of processed wastes to help levelize the station water inventory, and to allow for discharge at the minimum design release rate in the event that full circulating water flow is not available for dilution. Under the processing condition described, the minimum design release rate is such that with only service water flowing in the discharge flume, the radioactivity levels in the discharge flume satisfy the requirements of 10 CFR 20 and 10 CFR 50 and are as low as is reasonably achievable.

b. Floor Drain Subsystem - The collector and surge tanks in this system are crosstied in the same manner as described in the preceding subsection on the equipment drain subsystem. Normal design processing in this subsystem is accomplished by settling, evaporation, and demineralization. At initial plant startup, substantial volumes of water of very low radioactivity content may be produced by flushing and washing activities. To serve this case, a special piping arrangement is provided to allow the contents of any one of the floor drain collector or surge tanks to be directed through the waste filter in standby to the excess water tanks. Administrative controls, including the use of locked-closed valves, are utilized to prevent the inadvertent use of this piping arrangement during normal operations. Since the waste filter in standby is crosstied with the other two waste filters, floor drain wastes can be optionally treated in the same manner as equipment drain wastes, subject to the limitation that demineralizer run lengths may be substantially decreased. This possibility is included in the assumption of only 25% availability for the filters and demineralizers to allow for backwash and regeneration time.

   Normal floor drain processing through the floor drain evaporator packages is limited during shutdown and at startup by the availability of steam to heat the evaporators. Sufficient electrode boiler steam is available during shutdown to supply at least two evaporators. During unit startup, sufficient electrode boiler steam is available to supply at least one evaporator. The input to the two floor drain evaporator packages and the chemical waste evaporator package is crosstied to allow full utilization of the minimum available evaporator capacity. To further increase processing throughput, the output of the three evaporator monitor tanks can be directly routed to and processed by two or three of the waste demineralizers operating in parallel. With volume retention capacity for 1 to 2 days of maximum inputs, and the availability of increased throughout via the parallel processing channels, the system can handle an anticipated 3 to 4 day peak load.

c. Chemical Waste Subsystem - The two chemical waste collector tanks are crosstied. Since each of these tanks discharges into a chemical waste processing tank, the total volume retention capacity of all four tanks is available.

This capacity allows sufficient reserve for infrequent batches of high conductivity wastes from other sources, "off-standard" recycle batches from the sample tanks, and detergent waste from the laundry waste subsystem.

d. Laundry Waste Subsystem - This system is sized to meet the volume and radioactivity levels generated during refueling and maintenance outages. Since these outages will create the largest amount of detergent waste over a brief period of time, the laundry waste subsystem will be able to accommodate the waste from the startup.

11.2.2.7.2 Normal Startup Following Any Shutdown

Every startup produces some "one-time" flow volumes which establish the minimum requirement for sizing the tank volume in the equipment drain processing system. The waste surge tanks are sized so that each accommodates one reactor hydrotest volume plus one batch of ultrasonic resin cleaner waste which may be held for settling.

11.2.2.7.3 Condenser Tube Leakage

Condenser tube leakage increases the dissolved solids content of the condensate, forcing frequent bed replacement of the condensate polishers. At some magnitude of leakage, the polishers will no longer be able to maintain feedwater quality, thus forcing a unit shutdown. Recovery from this situation requires the replacement of all the condensate polisher beds.

11.2.2.7.4 Shutdown

Shutdown could result in limited availability of the floor drain and chemical waste evaporator packages due to the availability of only electrode boiler steam. Normally all steam requirements during shutdown, including operation of three evaporators, can be satisfied. Failure of one of the two electrode boilers or short duration steam demand for test purposes, could result in only enough steam for operation of two evaporators. Normal shutdown flows can be collected and held for about 2 days prior to the start of processing.

11.2.2.7.5 Excessive Fuel Leakage

Under a condition of excessive fuel leakage, the liquid radwaste system operates with radioactivity levels at or near design basis. The reactor water cleanup system (RWCU) delivers sludge to the solid radwaste system phase separators at a higher than normal rate. The phase separator tanks normally operate independently but are crosstied to provide storage flexibility. They receive wastes from the RWCU system during typical station operation. Frequent decanting of the phase separator results in the delivery of liquid with higher than normal solids content to the waste collector tank, resulting in short filter and demineralizer runs.

The waste filter and waste demineralizer are sized and arranged so that no more than one filter and/or demineralizer are out of service for backwashing or regeneration at any time during normal operation.

11.2.2.7.6 Maximum Reactor Coolant Leakage

The existence of reactor coolant leaks up to the technical specification limits can result in either the floor drain collector tanks or the equipment drain collector tanks becoming nearly filled with

water at reactor coolant activity. Both the floor drain processing subsystem and the equipment drain processing subsystem are capable of decontaminating this water to levels suitable for return to the condensate storage tanks.

11.2.2.7.7 Refueling and Maintenance Outages

The condition given in Subsection 11.2.2.7.1 covers the major requirements for outages. The laundry processing subsystem has its maximum load during refueling and maintenance outages and is sized to treat these loads to produce an effluent suitable for discharge or return to the condensate storage tank, (via the chemical waste processing subsystem).

11.2.2.7.8 Unscheduled Reactor Trips

Reactor trips can result in substantial additions of water to the suppression pool. This water may have to be removed, treated, and returned to condensate storage. The equipment drain processing subsystem has sufficient reserve capacity in all normal operating modes to accept and treat this volume within 1 day of occurrence.

11.2.2.8 Alternate Mode of Radwaste Operations

In an alternate mode of liquid radwaste operations, resins from condensate polishers and radwaste demineralizers will not be regenerated. After use, these resins will be sent to the spent resin tank. There they will be decanted and then sent to mobile solidification system. The resins are processed in liners for shipment offsite.

11.2.3 Radioactive Releases

11.2.3.1 Discharge Procedures

The processed liquid radwaste stream terminates at three waste sample tanks. Since the liquid radwaste system operates on a batch basis, this arrangement allows each treated batch to be sampled in the respective sample tank to ensure that the treatment was effective. If the sample indicates that the waste is still contaminated beyond acceptable limits, the capability is provided to recycle the waste either through the same treatment or through another subsystem. If the treated waste sample indicates that the water quality is within limits required for recycling, it is sent to the cycled condensate storage tank for reuse. If the plant water balance does not allow for recycling, the treated waste is sent to one of two excess water tanks for short-term storage. If storage capacity does not develop in the cycle condensate storage system in a reasonable period of time, as determined by the waste accumulation rate, a discharge of waste is scheduled.

The excess water tanks discharge to the service water discharge pipe using a common line, where a radiation monitor is installed. The line is connected to the service water discharge line, as shown in Drawings M05-1081-1 and M05-1085-7. The service water discharge line joins the circulating discharge water at the seal well. Circulating water then flows through the 3.4-mile discharge flume to Lake Clinton.

A release rate, low flow (10 to 60 gpm) or high flow (50 to 300 gpm) is selected based upon the dilution flow available and the isotope concentrations in the waste effluent. The liquid radwaste operations control panel has provision for indicating the presence of dilution flow via service water and circulating water pump status. Provision is made to obtain a minimum of 22,000 gpm

of service water flow (one pump running) for any plant operating condition; however, a dilution factor of 702 is considered representative of the maximum release concentration which might be experienced during single unit operation. Administrative controls prohibit the opening of the discharge valves until dilution requirements are met.

Discharge to the service water discharge lines is controlled by two valves for each line. Both valves are locked closed with the keys being administratively controlled. Additionally, the manually operated valve downstream of these two remotely operated valves is locked closed. The key for the manually operated valve is not controlled. The two valves will not be opened until tank contents have been sampled and analyzed; the existence of suitable dilution flow has been verified; and supervisory approval has been obtained via a discharge permit. The Process Radiation Monitor (PRM) is located upstream of the two valves. A high radioactivity signal from this gamma-scintillation detector automatically closes the downstream valves and provides an alarm signal in the Radwaste Operations Center, the Main Control Room, and locally.

Changes in dilution flow, such as the shutting down of a circulating water pump, are controlled in the main control room. The status of service water and circulating water pumps is indicated in the radwaste operation center where the discharge process is controlled. All required information regarding the batch release will be entered on the radioactive discharge permit. The use of the above procedures and equipment will effectively control and monitor the release of radioactive liquid effluent in accordance with General Design Criteria 60 and 64 of Appendix A to 10 CFR 50.

11.2.3.2 Criteria for Recycling

Liquid waste processing results in two streams: a product stream and a reject stream. The reject, or dirty stream, is further processed and solidified with a solidification agent in the solid radwaste system. The product, or clean stream, is normally returned to the primary cycle via the condensate storage tanks.

Certain criteria must be met to recycle water. These water quality requirements are:

a. conductivity, ≤1.0 μmho/cm at 25°C;

b. chlorides (as Cl), ≤0.05 ppm;

c. pH, 5.3 to 7.5 at 25°C; and

d. radioactivity ≤3×10^{-3} μCi/cc;

11.2.3.3 Release Assumptions

Although the liquid radwaste system has sufficient volume retention and processing capability to assure that no treated wastes-need be released under normal operating conditions, substantial additions to the station's water inventory (as from condenser tube leakage) could make it necessary to discharge processed wastes. The system has been designed to assure that sufficient decontamination of the expected inputs, as described in Table 11.2-6, can be achieved and results in releases in compliance with ALARA requirements.

The design-basis tritium concentration value is the equilibrium concentration computed assuming the tritium production rate given in Section 11.1 and a loss rate of tritiated water from

the station equivalent to 2 gpm. This is based on vapor losses only. Expected values of this concentration lie between 1/10 and 1/2 of this design-basis value, which is 3.4×10^{-3} μCi/gm.

11.2.3.4 Total Releases - Comparison with 10 CFR 20 Limits

Calculated annual expected releases of radionuclides in Ci/yr per reactor are given in Table 11.2-1 for normal operations and anticipated operational occurrences. Table 11.2-1A provides the same information for the alternative mode of liquid radwaste operation described in Section 11.2.2.8. Table 11.2-7 indicates the assumptions used to determine the annual expected releases. With a circulating water flow of 609,800 gpm and design-basis discharge rates of either 10 or 300 gpm the release dilution factors will be 60,980 to 2032. Table 11.2-1 lists effluent concentrations prior to discharge for the case of dilution by a factor of 702 as being representative of the maximum concentration which would be experienced during one unit operation. This dilution factor is based on a dilution flow of 210,600 gpm from one of the three circulating water pumps plus one of the two service water pumps and a radwaste discharge rate of 300 gpm. The effluent radioactivity concentrations are found to be within the 10 CFR 20 limits.

A calculation of the annual release of radionuclides for the design-basis fuel leakage condition was not performed. Prolonged station operation under such a condition is not anticipated. The design-basis fuel leakage condition is defined as a condition when the fuel has a total noble gas release rate of 100 μCi/sec per Mwt after a 30-minute decay. For CPS, this would translate to a noble gas release rate of approximately 300,000 μCi/sec after 30 minutes. Experience in the operation of open cycle boiling water reactors has indicated that in-plant contamination and other operating restrictions may limit plant operation even though environmental release limits are not reached. Release rates from the fuel on the order of 100,000 to 200,000 μCi/sec (at 30-minutes decay) can be tolerated for reasonable periods.

Long term operation in this range may be undesirable for operational reasons.

During a design-basis fuel leakage condition, the concentration of radionuclides in waste input streams will increase (see Table 11.2-6). The radioisotope concentrations in the processed waste are also expected to rise if additional processing in the liquid radwaste system is not utilized or available. If processed waste releases are required during a design-basis fuel leakage condition, the radionuclide concentrations shall not exceed 10 CFR 20 limits.

11.2.3.5 Radiological Dose Assessment - Comparison with Appendix I to 10 CFR 50 Guidelines

Expected annual average radiological doses associated with radionuclides released in liquid effluents have been estimated for areas beyond the restricted area boundary. The annual releases given in Table 11.2-1 diluted by 1.048×10^{6} gpm circulating water and discharged at 20 gpm were assumed for calculation of the average annual doses. The resulting annual doses are listed in Table 11.2-2 and are well below the design objectives of 10 CFR 50 Appendix I even for the "maximum" individual cited. Annual dose values will be approximately the same when they are based on releases given in Table 11.2.1A.

11.2.4 References

1. "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Boiling Water Reactors (BWR-GALE Code)," NUREG-0016, Revision 1, January 1979.

2. NRC Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR 50, Appendix I," Revision 1, October 1977.

3. NRC Regulatory Guide 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants."

4. P. P. Stancavage and D. G. Abbott, "Liquid Discharge Doses LIDSR Code," NEDM-20609-01, August 1976.

5. "Final Environmental Statement Concerning Proposed Rule Making Action; Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion 'As Low As Practical' for Radioactive Material In Light-Water-Cooled Power Reactor Effluents," USAEC, WASH-1258, July 1973.

TABLE 11.2-1
LIQUID EFFLUENTS
EXPECTED RELEASE OF RADIONUCLIDES FROM NORMAL OPERATIONS AND
ANTICIPATED OPERATIONAL OCCURRENCES AND COMPARISON WITH 10 CFR 20 LIMITS

NUCLIDE	EQUIPMENT DRAIN (Ci/yr)	FLOOR DRAIN (Ci/yr)	CHEMICAL WASTE (Ci/yr)	TOTAL (Ci/yr)	ADJUSTED TOTAL (Ci/yr)	CONCENTRATION IN A TYPICAL DISCHARGE AFTER DILUTION (μCi/cc)	TABLE 2 COLUMN 2 LIMIT (μCi/cc)	DISCHARGE CONCENTRATION AS A FRACTION OF LIMIT
CORROSION AND ACTIVATION PRODUCTS								
Na 24	1.74-04	2.21-07	1.67-09	1.74-04	5.59-03	2.1-09	5.00E-05	4.20E-05
P 32	5.46-06	1.64-08	3.14-07	5.79-06	1.86-04	7.0-11	9.00E-06	7.78E.06
Cr 51	1.66-04	5.09-07	1.81-05	1.84-04	5.92-03	2.3-09	5.00E-04	4.60E-06
Mn 54	1.95-06	6.13-09	4.53-07	2.41-06	7.75-05	2.3-11	3.00E-05	7.67E-07
Mn 56	1.79-04	1.60-08	3.30-11	1.79-04	5.76-03	2.2-09	7.00E-05	3.14E-05
Fe 55	2.79-05	8.78-08	6.86-06	3.49-05	1.12-03	4.2-10	1.00E-04	4.20E-06
Fe 59	8.32-07	2.58-09	1.22-07	9.57-07	3.07-05	1.2-11	1.00E-05	1.20E-06
Co 58	5.56-06	1.73-08	9.91-07	6.57-06	2.11-04	5.5-11	2.00E-05	2.75E-06
Co 60	1.12-05	3.51-08	2.08-06	1.40-05	4.49-04	1.2-10	3.00E-06	4.00E-05
Ni 65	1.06-06	9.30-11	1.90-13	1.06-06	3.42-05	1.3-11	1.00E-04	1.30E-07
Cu 64	4.83-04	5.38-07	3.43-09	4.84-04	1.55-02	5.8-09	2.00E-04	2.90E-05
Zn 65	5.58-06	1.75-08	1.27-06	6.86-06	2.21-04	8.1-11	5.00E-06	1.62E-05
Zn 69m	3.34-05	3.97-08	2.72-10	3.35-05	1.08-03	4.0-10	6.00E-05	6.67E-06
Zn 69	3.49-05	4.27-08	2.92-10	3.50-05	1.12-03	4.2-10	8.00E-04	5.25E-07
W 187	6.17-06	1.07-08	3.14-10	6.18-06	1.98-04	7.3-11	3.00E-05	2.43E-06
Np239	1.71-04	4.13-07	2.84-07	1.72-04	5.52-03	2.1-09	2.00E-05	1.05E-04
FISSION PRODUCTS								
Br 83	1.26-05	9.31-09	1.97-11	1.26-05	4.05-04	1.5-10	9.00E-04	1.67E-07
Br 84	9.35-07	3.45-14	6.96-19	9.35-07	3.00-05	1.2-11	4.00E-04	3.00E-08
Rb 89	5.54-07	2.80-22	0.00	5.54-07	1.78-05	6.7-12	9.00E-04	7.44E-09
Sr 89	2.80-06	8.68-09	4.39-07	3.24-06	1.04-04	3.9-11	8.00E-06	4.88E-06
Sr 91	5.50-05	4.64-08	2.50-10	5.50-05	1.77-03	6.6-10	2.00E-05	3.30E-05
Y 91m	3.44-05	3.00-08	1.61-10	3.44-05	1.11-03	4.1-10	2.00E-03	2.05E-07
Y 91	1.41-06	5.27-09	2.98-07	1.71-06	5.49-05	2.0-11	8.00E-06	2.50E-06
Sr 92	3.84-05	3.90-09	8.53-12	3.84-05	1.23-03	4.6-10	4.00E-05	1.15E-05
Y 92	7.92-05	2.55-08	8.20-11	7.92-05	2.54-03	9.7-10	4.00E-05	2.43E-05
Y 93	5.68-05	5.08-08	2.81-10	5.69-05	1.83-03	6.9-10	2.00E-05	3.45E-05
Nb 98	2.03-06	8.88-13	1.51-16	2.03-06	6.53-05	2.5-11	2.00E-04	1.25E-07
Mo 99	4.99-05	1.26-07	1.38-07	5.02-05	1.61-03	6.1-10	2.00E-05	3.05E-05

TABLE 11.2-1
LIQUID EFFLUENTS
EXPECTED RELEASE OF RADIONUCLIDES FROM NORMAL OPERATIONS AND
ANTICIPATED OPERATIONAL OCCURRENCES AND COMPARISON WITH 10 CFR 20 LIMITS (Continued)

NUCLIDE	EQUIPMENT DRAIN (Ci/yr)	FLOOR DRAIN (Ci/yr)	CHEMICAL WASTE (Ci/yr)	TOTAL (Ci/yr)	ADJUSTED TOTAL (Ci/yr)	CONCENTRATION IN A TYPICAL DISCHARGE AFTER DILUTION (μCi/cc)	TABLE 2 COLUMN 2 LIMIT (μCi/cc)	DISCHARGE CONCENTRATION AS A FRACTION OF LIMIT
FISSION PRODUCTS (cont'd)								
Tc 99m	2.18-04	1.99-07	1.32-07	2.18-04	7.00-03	2.6-09	1.00E-03	2.60E-06
Tc101	6.40-07	2.36-22	0.00	6.40-07	2.06-05	7.8-12	2.00E-03	3.90E-09
Ru103	5.54-07	1.72-09	7.72-08	6.32-07	2.03-05	7.4-12	3.00E-05	2.47E-07
Rh103m	5.54-07	1.72-09	7.63-08	6.32-07	2.03-05	7.3-12	6.00E-03	1.22E-09
Tc104	1.79-06	3.98-19	7.42-26	1.79-06	5.77-05	2.1-11	4.00E-04	5.25E-08
Ru105	1.42-05	4.08-09	1.41-11	1.42-05	4.55-04	1.7-10	7.00E-05	2.43E-06
Rh105m	1.42-05	4.09-09	1.41-11	1.42-05	4.57-04	1.7-10	--	--
Rh105	3.46-06	1.08-08	1.74-09	3.45-06	1.11-04	4.1-11	5.00E-05	8.20E-07
Ru106	8.37-08	2.63-10	1.97-08	1.04-07	3.33-06	1.3-12	3.00E-06	4.33E-07
Te129m	1.11-06	3.42-09	1.39-07	1.25-06	4.01-05	1.5-11	7.00E-06	2.14E-06
Te129	7.09-07	2.19-09	8.90-08	8.00-07	2.57-05	9.5-12	4.00E-04	2.38E-08
Te131m	2.18-06	4.24-09	3.21-10	2.19-06	7.03-05	2.6-11	8.00E-06	3.25E-06
Te131	3.98-07	7.74-10	5.85-11	3.99-07	1.28-05	4.8-12	8.00R-05	6.00E-08
I 131	4.28-05	1.25-06	2.20-04	2.64-04	8.49-03	3.3-09	1.00E-06	3.30E-03
I 132	1.19-04	7.76-08	1.34-09	1.19-04	3.82-03	1.4-09	1.00E-04	1.40E-05
I 133	4.58-04	7.38-06	1.47-06	4.67-04	1.50-02	5.6-09	7.00E-06	8.00E-04
I 134	4.31-05	2.47-10	5.33-14	4.31-05	1.38-03	5.2-10	4.00E-04	1.30E-06
Cs134	8.37-06	2.63-09	1.14-06	9.51-06	3.06-04	1.2-10	9.00E-07	1.33E-04
I 135	2.79-04	1.51-06	7.52-09	2.81-04	9.02-03	3.4-09	3.00E-05	1.13E-04
Cs136	2.17-05	6.51-09	6.19-07	2.23-05	7.18-04	2.7-10	6.00E-06	4.50E-05
Cs137	5.58-06	1.76-09	7.82-07	6.37-06	2.05-04	7.7-11	1.00E-06	7.70E-05
Ba137m	5.22-06	1.64-09	7.32-07	5.95-06	1.91-04	7.3-11	--	--
Cs138	1.58-05	6.82-15	1.48-18	1.58-05	5.07-04	1.9-10	4.00E-04	4.75E-07
Ba139	1.32-05	1.21-10	8.64-14	1.32-05	4.25-04	1.6-10	2.00E-04	8.00E-07
Ba140	1.09-05	3.26-08	5.47-07	1.15-05	3.69-04	1.4-10	8.00E-06	1.75E-05
La140	1.54-06	1.35-08	6.24-07	2.17-06	6.98-05	2.6-11	9.00E-06	2.89E-06
La141	3.98-06	9.05-10	2.82-12	3.98-06	1.28-04	4.8-11	5.00E-05	9.60E-07
Ce141	8.79-07	2.79-09	1.09-07	9.90-07	3.18-05	1.2-11	3.00E-05	4.00E-07
La142	9.05-06	1.34-10	1.20-13	9.05-06	2.91-04	1.1-10	1.00E-04	1.10E-06
Ce143	6.69-07	1.35-09	1.50-10	6.71-07	2.15-05	7.9-12	2.00E-05	3.95E-07

TABLE 11.2-1
LIQUID EFFLUENTS
EXPECTED RELEASE OF RADIONUCLIDES FROM NORMAL OPERATIONS AND
ANTICIPATED OPERATIONAL OCCURRENCES AND COMPARISON WITH 10 CFR 20 LIMITS (Continued)

NUCLIDE	EQUIPMENT DRAIN (Ci/yr)	FLOOR DRAIN (Ci/yr)	CHEMICAL WASTE (Ci/yr)	TOTAL (Ci/yr)	ADJUSTED TOTAL (Ci/yr)	CONCENTRATION IN A TYPICAL DISCHARGE AFTER DILUTION (μCi/cc)	TABLE 2 COLUMN 2 LIMIT (μCi/cc)	DISCHARGE CONCENTRATION AS A FRACTION OF LIMIT
FISSION PRODUCTS (cont'd)								
Pr143	1.10-06	3.39-09	6.44-08	1.17-06	3.77-05	1.5-11	7.00E-05	2.14E-07
Ce144	8.37-08	2.63-10	1.93-08	1.03-07	3.32-06	1.3-12	3.00E-06	4.33E-07
All Others	1.78-06	4.26-09	2.46-07	2.03-06	6.54-05	2.4-11	--	-
(Total Except Tritium)	2.94-03	1.28-05	2.59-04	3.21-03	1.03-01	3.9-08	--	
Tritium Release					1.21+1	5.6-06	1.00E-03	5.60E-03

*Adjusted to increase the total release by 0.15 Ci/yr, to account for anticipated operational occurrences.

Note: Based on the following assumptions:

1. Annual radwaste discharge volume is 10^6 gallons per reactor.
2. Maximum radwaste discharge rate is 300 gpm.
3. Dilution flow rate is 210,600 gpm

Effluent concentration limits taken from 10 CFR 20, Appendix B, Table 2

TABLE 11.2.1A
LIQUID EFFLUENTS
EXPECTED RELEASE OF RADIONUCLIDES FROM NORMAL OPERATIONS AND ANTICIPATED OPERATIONAL OCCURRENCES
WITHOUT RESIN REGENERATION (SEC. 11.2.2.8), AND COMPARISION WITH 10 CFR 20 LIMITS

NUCLIDE	EQUIPMENT DRAIN (Ci/yr)	FLOOR DRAIN (Ci/yr)	CHEMICAL TOTAL (Ci/yr)	TOTAL	ADJUSTED TOTAL (Ci/yr)	CONCENTRATION IN A TYPICAL DISCHARGE AFTER DILUTION (μCi/cc)	TABLE 2 COLUMN 2 LIMIT (μCi/cc)	DISCHARGE CONCENTRATION AS A FRACTION OF LIMIT
CORROSON AND ACTIVATION PRODUCTS								
Na 24	1.05-004	1.04-007	7.02-010	1.05-004	5.87-003	2.2-09	5.00E-05	4.40E-05
P 32	4.17-006	1.23-008	2.96-010	4.19-006	2.33-004	8.8-11	9.00E-06	9.78E.06
Cr 51	1.27-004	3.87-007	1.03-008	1.28-004	7.12-003	2.7-09	5.00E-04	5.40E-06
Mn 54	1.51-006	4.74-009	1.40-010	1.52-006	8.44-005	3.2-11	3.00E-05	1.07E-06
Mn 56	8.12-005	7.14-009	1.71-011	8.12-005	4.53-003	1.7-09	7.00E-05	2.43E-05
Fe 55	2.16-005	6.79-008	2.02-009	2.17-005	1.21-003	4.6-10	1.00E-04	4.60E-06
Fe 59	6.41-007	1.97-009	5.49-011	6.43-007	3.58-005	1.3-11	1.00E-05	1.30E-06
Co 58	4.29-006	1.33-008	3.80-010	4.31-006	2.40-004	9.0-11	2.00E-05	4.50E-06
Co 60	8.64-006	2.72-008	8.09-010	8.67-006	4.83-004	1.8-10	3.00E-06	6.00E-05
Ni 65	4.82-007	4.15-011	.00	4.82-007	2.69-005	1.0-11	1.00E-04	1.00E-07
Cu 64	2.84-004	2.45-007	1.58-009	2.84-004	1.58-002	5.9-09	2.00E-04	2.95E-05
Zn 65	4.31-006	1.35-008	3.98-010	4.33-006	2.41-004	9.1-11	5.00E-06	1.82E-05
Zn 69m	1.99-005	1.83-008	1.21-010	1.99-005	1.11-003	4.2-10	6.00E-05	7.00E-06
Zn 69	2.08-005	1.97-008	1.30-010	2.08-005	1.16-003	4.4-10	8.00E-04	5.50E-07
W 187	4.05-006	5.61-009	4.45-011	4.05-006	2.26-004	8.5-11	3.00E-05	2.83E-06
Np239	1.23-004	2.62-007	3.13-009	1.23-004	6.86-003	2.6-09	2.00E-05	1.30E-04
FISSON PRODUCTS								
Br 83	6.06-006	4.44-009	1.09-011	6.06-006	3.38-004	1.3-10	9.00E-04	1.44E-07
Br 84	4.56-007	1.69-014	3.95-019	4.56-007	2.54-005	9.6-12	4.00E-04	2.40E-08
Rb 89	2.73-007	1.38-022	.00	2.73-007	1.52-005	5.7-12	9.00E-04	6.33E-09
Sr 89	2.16-006	6.66-009	1.87-010	2.17-006	1.21-004	4.6-11	8.00E-06	5.75E-06
Sr 91	3.03-005	2.03-008	1.20-010	3.04-005	1.69-003	6.4-10	2.00E-05	3.20E-05
Y 91m	1.90-005	1.31-008	7.72-011	1.90-005	1.06-003	4.0-10	2.00E-03	2.00E-07
Y 91	1.21-006	4.25-009	1.24-010	1.21-006	6.74-005	2.5-11	8.00E-06	3.13E-06
Sr 92	1.74-005	1.74-009	4.41-012	1.74-005	9.70-004	3.7-10	4.00E-05	9.25E-06
Y 92	3.86-005	1.13-008	4.22-011	3.86-005	2.15-003	8.1-10	4.00E-05	2.03E-05
Y 93	3.17-005	2.23-008	1.34-010	3.18-005	1.77-003	6.7-10	2.00E-05	3.35E-05
Nb 98	9.59-007	4.21-013	8.32-017	9.59-007	5.35-005	2.0-11	2.00E-04	1.00E-07
Mo 99	3.63-005	8.17-008	1.07-009	3.63-005	2.02-003	7.6-10	2.00E-05	3.80E-05

TABLE 11.2.1A (cont'd)
LIQUID EFFLUENTS
EXPECTED RELEASE OF RADIONUCLIDES FROM NORMAL OPERATIONS AND ANTICIPATED OPERATIONAL OCCURRENCES
WITHOUT RESIN REGENERATION (SEC. 11.2.2.8), AND COMPARISION WITH 10 CFR 20 LIMITS

NUCLIDE	EQUIPMENT DRAIN (Ci/yr)	FLOOR DRAIN (Ci/yr)	CHEMICAL TOTAL (Ci/yr)	TOTAL	ADJUSTED TOTAL (Ci/yr)	CONCENTRATION IN A TYPICAL DISCHARGE AFTER DILUTION (μCi/cc)	TABLE 2 COLUMN 2 LIMIT (μCi/cc)	DISCHARGE CONCENTRATION AS A FRACTION OF LIMIT
Tc 99m	1.20-004	1.14-007	1.19-009	1.20-004	6.68-003	2.5-09	1.00E-03	2.50E-06
Tc101	3.13-007	1.16-022	.00	3.13-007	1.74-005	6.5-12	2.00E-03	3.25E-09
Ru103	4.27-007	1.31-009	3.61-011	4.28-007	2.39-005	9.0-12	3.00E-05	3.00E-07
Rh103m	4.27-007	1.31-009	3.61-011	4.28-007	2.39-005	9.0-12	6.00E-03	1.50E-09
Tc104	8.74-007	1.94-019	4.20-026	8.74-007	4.87-005	1.8-11	4.00E-04	4.50E-08
Ru105	6.68-006	1.76-009	7.06-012	6.68-006	3.72-004	1.4-10	7.00E-05	2.00E-06
Rh105m	6.69-006	1.77-009	7.08-012	6.70-006	3.73-004	1.4-10	--	--
Rh105	3.14-006	6.66-009	6.56-011	3.15-006	1.76-004	6.6-11	5.00E-05	1.32E-06
Ru106	6.48-008	2.03-010	6.00-012	6.50-008	3.62-006	1.4-12	3.00E-06	4.67E-07
Te129m	8.52-007	2.60-009	7.08-011	8.55-007	4.76-005	1.8-11	7.00E-06	2.57E-06
Te129	5.46-007	1.67-009	4.54-011	5.47-007	3.05-005	1.1-11	4.00E-04	2.75E-08
Te131m	1.48-006	2.35-009	2.05-011	1.48-006	8.24-005	3.1-11	8.00E-06	3.88E-06
Te131	2.69-007	4.30-010	3.74-012	2.70-007	1.50-005	5.6-12	8.00E-05	7.00E-08
I 131	3.86-005	1.08-006	2.52-008	3.97-005	2.21-003	8.3-10	1.00E-06	8.30E-04
Te132	1.86-007	4.38-010	6.23-012	1.86-007	1.04-005	3.9-12	9.00E-06	4.30E-07
I 132	5.72-005	3.71-008	9.12-011	5.73-005	3.19-003	1.2-09	1.00E-04	1.20E-05
I 133	3.43-004	4.36-006	3.67-008	3.47-004	1.93-002	7.3-09	7.00E-06	1.04E-03
I 134	2.08-005	1.20-010	3.00-014	2.08-005	1.16-003	4.4-10	4.00E-04	1.10E-06
Cs134	6.48-006	2.04-009	6.75-010	6.48-006	3.61-004	1.4-10	9.00E-07	1.56E-04
I 135	1.58-004	7.22-007	4.10-009	1.59-004	8.86-003	3.3-09	3.00E-05	1.10E-04
Cs136	1.66-005	4.85-009	1.28-009	1.66-005	9.26-004	3.5-10	6.00E-06	5.83E-05
Cs137	4.32-006	1.36-009	4.52-010	4.32-006	2.41-004	9.1-11	1.00E-06	9.10E-05
Ba137m	4.04-006	1.27-009	4.23-010	4.04-006	2.25-004	8.5-11	--	--
Cs138	7.68-006	3.33-015	8.39-019	7.68-006	4.28-004	1.6-10	4.00E-04	4.00E-07
Ba139	6.11-006	5.61-011	4.64-014	6.11-006	3.40-004	1.3-10	2.00E-04	6.50E-07
Ba140	8.31-006	2.42-008	5.72-010	8.34-006	4.65-004	1.7-10	8.00E-06	2.13E-05
La140	2.01-006	1.39-008	5.18-010	2.02-006	1.13-004	4.3-11	9.00E-06	4.78E-06
La141	2.06-006	4.42-010	1.60-012	2.06-006	1.15-004	4.3-11	5.00E-05	8.60E-07
Ce141	6.94-007	2.16-009	5.82-011	6.96-007	3.88-005	1.5-11	3.00E-05	5.00E-07
La142	4.19-006	6.23-011	6.47-014	4.19-006	2.34-004	8.8-11	1.00E-04	8.80E-07
Ce143	4.58-007	7.69-010	6.99-012	4.59-007	2.56-005	9.6-12	2.00E-05	4.80E-07

TABLE 11.2.1A (cont'd)
LIQUID EFFLUENTS
EXPECTED RELEASE OF RADIONUCLIDES FROM NORMAL OPERATIONS AND ANTICIPATED OPERATIONAL OCCURRENCES WITHOUT RESIN REGENERATION (SEC. 11.2.2.8), AND COMPARISION WITH 10 CFR 20 LIMITS

NUCLIDE	EQUIPMENT DRAIN (Ci/yr)	FLOOR DRAIN (Ci/yr)	CHEMICAL TOTAL (Ci/yr)	TOTAL	ADJUSTED TOTAL (Ci/yr)	CONCENTRATION IN A TYPICAL DISCHARGE AFTER DILUTION (μCi/cc)	TABLE 2 COLUMN 2 LIMIT (μCi/cc)	DISCHARGE CONCENTRATION AS A FRACTION OF LIMIT
Pr143	8.51-007	2.56-009	6.25-011	8.54-007	4.76-005	1.8-11	7.00E-05	2.57E-07
Ce144	6.47-008	2.03-010	5.98-012	6.49-008	3.62-006	1.4-12	3.00E-06	4.67E-07
All Others	1.08-006	2.70-009	7.80-011	1.08-006	6.01-005	2.3-11	--	-
Total (Except Tritium)	1.82-003	7.74-006	9.37-008	1.83-003	1.02-001	3.8-08	--	
Tritium Release					1.20+001	4.5-06	1.00E-03	4.50E-03

*Adjusted to increase the total release by 0.15 Ci/yr, to account for anticipated operational occurrences.

Note: Based on the following assumptions:

1. Annual radwaste discharge volume is 10^6 gallons per reactor.
2. Maximum radwaste discharge rate is 300 gpm.
3. Dilution flow rate is 210,600 gpm.

Effluent concentration limits taken from 10 CFR 20, Appendix B, Table 2

TABLE 11.2-2
COMPARISON OF CALCULATED RADIOLOGICAL DOSES FR0M
LIQUID EFFLUENTS TO NUMERICAL DESIGN OBJECTIVES
OF APPENDIX I TO 10 CFR 50 AND THE DOSE LIMITS OF 10 CFR 20*

	DOSES mrem/yr				
	BODY	SKIN	GI-LLI	THYROID	BONE
Appendix I, 10 CFR 50 Design Objective	3	10	10	10	10
10 CFR 20 Limits	100	-	-	-	-
PATHWAY					
Drink	8.8E-04	0.	1.1E-03	1.2E-02	1.0E-04
Eat plants	0.	0.	0.	0.	0.
Eat Inverts	0.	0.	0.	0.	0.
Eat Fish	5.9E-03	0.	8.0E-03	4.6E-03	6.3E-02
Swim	9.3E-06	1.2E-05	0.	0.	0.
Boat	2.4E-06	0.	0.	0.	0.
Sunbathe	7.8E-06	9.1E-06	0.	0.	0.
Fish	3.5E-05	3.8E-05	0.	0.	0.
TOTAL	6.8E-03	5.9E-05	9.1E-03	1.7E-02	6.3E-02
RELEASE, ci/yr		1.21E+01			
CONC. ci/cc		9.96E-09			

* Data used to calculate Table 11.2-2 values are given in NEDM-20609-01, August 1976, except as follows:

1. The sources release are listed in Table 11.2-1.

2. The dilution factor and hold up times are taken from WASH-1258.

Based on the original licensed power level of 2894 MWth

TABLE 11.2-3
LIQUID RADWASTE MANAGEMENT SYSTEM COMPONENTS
AND DESIGN PARAMETERS

TANKS QUANTITY	DESCRIPTION	CAPACITY (gal)	DESIGN PRESSURE	DESIGN TEMPERATURE (°F)	MATERIALS OF CONSTRUCTION
2	Waste Collector	30,000	Full Water	150	304 SS
2	Waste Surge	50,000	Full Water	150	304 SS
3	Waste Sample	30,000	Full Water	150	304 SS
2	Excess Water	25,000	Full Water	150	304 SS
2	Floor Drain Collector	25,000	Full Water	150	304 SS
2	Floor Drain Surge	25,000	Full Water	150	304 SS
2	Floor Drain Evaporator Feed	25,000	Full Water	150	316 SS
2	Floor Drain Evaporator Monitor	15,000	Full Water	212	304 SS
2	Chemical Waste Collector	35,000	Full Water	150	316 SS
2	Chemical Waste Processing	25,000	Full Water	150	316 SS
1	Chemical Waste Evaporator Monitor	15,000	Full Water	212	304 SS
2	Laundry Drain Collector	2,500	Full Water	150	Carb. S+1
2	Laundry Drain Sample	2,500	Full Water	150	Carb. S+1

PUMPS QUANTITY	DESCRIPTION	DESIGN FLOW (gpm)	DESIGN NET DEVELOPED HEAD (ft)
2	Waste Collector	300	313
2	Waste Surge Tank	300	313
3	Waste Sample Tank	300	100
2	Excess Water Tank	300	100
2	Floor Drain Collector Tank	300	263
2	Floor Drain Surge Tank	300	263
2	Floor Drain Evaporator Feed Tank	100	178

TABLE 11.2-3
LIQUID RADWASTE MANAGEMENT SYSTEM COMPONENTS
AND DESIGN PARAMETERS (Continued)

PUMPS QUANTITY	DESCRIPTION	DESIGN FLOW (gpm)	DESIGN NET DEVELOPED HEAD (ft)
2	Floor Drain Evaporator Monitor Tank	150	170
2	Chemical Waste Collector Tank	100	178
2	Chemical Waste Process Tank	100	178
1	Chemical Waste Evaporator Monitor Tank	150	170
2	Laundry Drain Collector Tank	25	97
2	Laundry Drain Sample Tank	100	127

PROCESSING
EQUIPMENT

QUANTITY	DESCRIPTION	FLOW (gpm)	DESIGN TEMPERATURE	STEAM REG'D (lb/hr)	DESIGN PRESSURE
3	Waste Filter	300	125	--	200
3	Waste Demineralizer	300	125	--	150
1	Chemical Waste Evaporator	30		17,600	
2	Floor Drain Evaporator	30		17,600	
	Evaporator Components				
	Vapor Body		270		30 + 11 psi Hydro to full vacuum
	Entrainment Separator		270		30 + 8.5 psi Hydro to full vacuum
	Heater (Shell)		300		75 + 6 psi Hydro
	(Tube)		300		30 + 29 psi Hydro
	Condenser (Shell)		275		30 + 5 psi Hydro
	(Tube)		150		150 psig
	Subcooler (Shell)		275		100 psig
	(Tube)		150		150 psig
	Level Pot		300		30 psi + fullvacuum

TABLE 11.2-4
TANKS LOCATED OUTSIDE THE CONTAINMENT WHICH CONTAIN POTENTIALLY RADIOACTIVE FLUID

TANK	QUANTITY	LOCATION	TANK LEVEL MONITORING	ANNUNCIATION	OVERFLOW CONTROL
Waste collector tank	2	Radwaste Building, El. 702 ft 0 in.	Yes	Yes	Yes
Waste surge tank	2	Radwaste Building. El. 702 ft 0 in.	Yes	Yes	Yes
Waste sample tank	3	Radwaste Building, El. 762 ft 0 in.	Yes	Yes	Yes
Excess water tank	2	Radwaste Building, El. 762 ft 0 in.	Yes	Yes	Yes
Floor drain collector tank	2	Radwaste Building, El. 702 ft 0 in.	Yes	Yes	Yes
Floor drain surge tank	2	Radwaste Building, El. 702 ft 0 in.	Yes	Yes	Yes
Laundry drain sample tank	2	Radwaste Building, El. 702 ft 0 in.	Yes	Yes	Yes
Spent resin tank	1	Radwaste Building, El. 702 ft 0 in.	Yes	Yes	Yes
Phase separator	2	Radwaste Building, El. 702 ft 0 in.	Yes	Yes	Yes
Fuel pool F/D sludge tank	2	Radwaste Building, El. 702 ft 0 in.	Yes	Yes	Yes
Waste sludge tank	2	Radwaste Building, El. 702 ft 0 in.	Yes	Yes	Yes
Concentrated waste tank	2	Radwaste Building, El. 762 ft 0 in.	Yes	Yes	Yes

TABLE 11.2-4
TANKS LOCATED OUTSIDE THE CONTAINMENT WHICH CONTAIN POTENTIALLY
RADIOACTIVE FLUID (Continued)

TANK	QUANTITY	LOCATION	TANK LEVEL MONITORING	ANNUNCIATION	OVERFLOW CONTROL
Floor drain evaporator monitor tank	2	Radwaste Building, El. 762 ft 0 in.	Yes	Yes	Yes
Chemical waste collector tank	2	Radwaste Building El. 702 ft 0 in.	Yes	Yes	Yes
Chemical waste process tank	2	Radwaste Building, El. 702 ft 0 in.	Yes	Yes	Yes
Chemical waste evaporator monitor tank	1	Radwaste Building, El. 762 ft 0 in.	Yes	Yes	Yes
Laundry drain waste collector tank	2	Radwaste Building, El. 702 ft 0 in.	Yes	Yes	Yes
Waste mixing (abandoned in place)	2	Radwaste Building, El. 750 ft 0 in.	Yes	Yes	Yes
Evaporator condensate drain tank	3	Radwaste Building, El. 720 ft 6 in.	Yes	Yes	Yes
Condenser vacuum water separation tank	2	Radwaste Building, El. 762 ft 0 in.	No	No	
LP heater flash tank	2	Turbine Building, El. 762 ft 0 in.	Yes	Yes	Yes
Condensate polisher demineralizer URC receiving tank	1	Radwaste Building, El. 720 ft 6 in.	No	No	No
Condensate polisher demineralizer URC storage	1	Radwaste Building, El. 720 ft 6 in.	No	No	No
Condensate polisher demineralizer URC waste tank	1	Radwaste Building, El. 702 ft 0 in.	Yes	Yes	Yes

TABLE 11.2-4
TANKS LOCATED OUTSIDE THE CONTAINMENT WHICH CONTAIN POTENTIALLY RADIOACTIVE FLUID (Continued)

TANK	QUANTITY	LOCATION	TANK LEVEL MONITORING	ANNUNCIATION	OVERFLOW CONTROL
Condensate polisher demineralizer resin separation and cation re-generation tank	1	Turbine Building, El. 712 ft 0 in.	No	No	No
Condensate polisher demineralizer anion regeneration tank	1	Turbine Building, El. 712 ft 0 in.	No	No	No
Condensate polisher demineralizer resin mixing and storage tank	1	Turbine Building, El. 712 ft 0 in.	No	No	No
Condensate polisher demineralizer hot water tank	1	Turbine Building, El. 712 ft 0 in.	No	No	No
Condensate polisher acid reclaim tank	1	Turbine Building, El. 712 ft 0 in.	Yes	Yes	Yes
Condensate polisher caustic reclaim tank	1	Turbine Building, El. 712 ft 0 in.	Yes	Yes	Yes
Condensate polisher LCLC tank	1	Turbine Building, El. 712 ft 0 in.	Yes	Yes	Yes
Cycled condensate storage tank	1	South of the Diesel Generator Building at grade level	Yes	Yes	Yes
Fuel pool skimmer surge tank	2	Fuel Building, El. 755 ft 0 in.	Yes	No	No
Moisture separator drain tank	2	Turbine Building, El. 762 ft 0 in.	Yes	Yes	Yes
Reheater drain tank	2	Turbine Building El. 762 ft 0 in.	Yes	Yes	Yes

TABLE 11.2-4
TANKS LOCATED OUTSIDE THE CONTAINMENT WHICH CONTAIN POTENTIALLY RADIOACTIVE FLUID (Continued)

TANK	QUANTITY	LOCATION	TANK LEVEL MONITORING	ANNUNCIATION	OVERFLOW CONTROL
Fuel building equipment drain tank	1	Fuel Building, El. 712 ft 0 in.	Yes	Yes	Yes
Fuel building floor drain tank	1	Auxiliary Fuel Building, El. 707 ft 6 in.	Yes	Yes	Yes
Auxiliary building floor drain tank	1	Auxiliary Fuel Building, El. 707 ft 6 in.	Yes	Yes	Yes
Radwaste building equipment drain tank	1	Radwaste Building, El. 702 ft 0 in.	Yes	Yes	Yes
Turbine building equipment drain tank	1	Turbine Building, El. 712 ft 0 in.	Yes	Yes	Yes
Radwaste building floor drain tank	1	Radwaste Building, El. 702 ft 0 in.	Yes	Yes	Yes
Turbine building floor drain tank	2	Turbine Building, El. 712 ft 0 in.	Yes	Yes	Yes
Waste filter body feed tank	1	Radwaste Building, El. 725 ft 0 in.	Yes	Yes	Yes
Floor drain evaporator feed tank	2	Radwaste Building, El. 702 ft 0 in.	Yes	Yes	Yes
RCIC storage tank	1	Outdoors at grade level	Yes	No	Yes

TABLE 11.2-5
DESIGN-BASIS ACTIVITY CONCENTRATIONS (in μCi/cc) IN
MAJOR LIQUID RADWASTE INPUT STREAMS

NUCLIDE	CHEMICAL WASTE SUBSYSTEM	EQUIPMENT DRAIN SUBSYSTEM	FLOOR DRAIN SUBSYSTEM	LAUNDRY DRAIN SUBSYSTEM
F-18	2.0 - 4	4.3 - 3	1.2 - 3	2.1 - 6
Na-24	8.4 - 4	2.1 - 3	6.2 - 4	1.1 - 6
P-32	1.8 - 4	2.1 - 5	6.2 - 6	1.1 - 8
Cr-51	7.2 - 3	5.3 - 4	1.5 - 4	2.7 - 7
Mn-54	1.0 - 3	4.3 - 5	1.2 - 5	2.1 - 8
Mn-56	3.6 - 3	5.3 - 2	1.5 - 2	2.7 - 5
Co-58	1.1 - 1	5.3 - 3	1.5 - 3	2.7 - 6
Fe-59	1.4 - 3	8.5 - 5	2.5 - 5	4.3 - 8
Co-60	1.4 - 2	5.3 - 4	1.5 - 4	2.7 - 7
Ni-65	2.1 - 5	3.2 - 4	9.2 - 5	1.6 - 7
Zn-65	5.1 - 5	2.1 - 6	6.2 - 7	1.1 - 9
Zn-69m	1.1 - 5	3.2 - 4	9.2 - 6	1.6 - 8
Zn-69	1.1 - 5	0.0	0.0	0.0
Br-83	2.4 - 2	1.9 - 2	5.5 - 3	9.7 - 6
Br-84	1.1 - 2	3.9 - 2	1.1 - 2	2.0 - 5
Br-85	7.0 - 4	2.7 - 2	7.7 - 3	1.3 - 5
Sr-89	6.3 - 2	3.5 - 3	1.0 - 3	1.8 - 6
Y-89m	6.3 - 6	0.0	0.0	0.0
Sr-90	6.9 - 3	2.7 - 4	1.1 - 4	1.3 - 7
Y-90	6.5 - 3	0.0	0.0	0.0
Sr-91	2.2 - 2	8.6 - 2	2.5 - 2	4.3 - 5
Y-91m	1.3 - 2	0.0	0.0	0.0
Y-91	1.1 - 2	0.0	0.0	0.0
Sr-92	1.1 - 2	1.5 - 1	4.3 - 2	7.5 - 5
Y-92	1.1 - 2	0.0	0.0	0.0
Zr-95	8.8 - 4	4.6 - 5	1.3 - 5	2.3 - 8
Nb-95m	1.6 - 5	0.0	0.0	0.0
Nb-95	1.1 - 3	4.8 - 5	1.4 - 5	2.4 - 7
Zr-97	1.7 - 5	3.8 - 5	1.1 - 5	1.9 - 8
Nb-97m	1.7 - 5	0.0	0.0	0.0
Nb-97	1.7.- 5	0.0	0.0	0.0
Mo-99	4.5 - 2	2.6 - 2	7.4 - 3	1.3 - 5
Tc-99m	5.5 - 2	1.0 - 1	3.0 - 2	5.2 - 5
Tc-99	3.1 - 8	0.0	0.0	0.0
Tc-101	1.4 - 3	2.3 - 1	6.8 - 2	1.2 - 4
Ru-103	3.6 - 4	2.2 - 5	6.5 - 6	1.1 - 8
Rh-103m	3.6 - 4	0.0	0.0	0.0
Ru-106	7.3 - 5	3.0 - 6	8.6 - 7	1.5 - 9
Rh-106	7.3 - 5	0.0	0.0	0.0
Ag-110m	1.5 - 4	6.4 - 5	1.8 - 5	3.2 - 8
Ag-110	2.0 - 6	0.0	0.0	0.0
Te-129m	5.9 - 3	3.9 - 4	1.1 - 4	2.0 - 7
Te-129	3.8 - 3	0.0	0.0	0.0

TABLE 11.2-5
DESIGN-BASIS ACTIVITY CONCENTRATIONS (in μCi/cc) IN
MAJOR LIQUID RADWASTE INPUT STREAMS (Cont'd)

NUCLIDE	CHEMICAL WASTE SUBSYSTEM	EQUIPMENT DRAIN SUBSYSTEM	FLOOR DRAIN SUBSYSTEM	LAUNDRY DRAIN SUBSYSTEM
I-129	2.4 - 11	0.0	0.0	0.0
I-131	1.6 + 0	1.6 - 2	4.6 - 3	8.0 - 6
Te-132	3.5 - 2	1.7 - 2	4.9 - 3	8.6 - 6
I-132	2.3 - 1	1.6 - 1	4.6 - 2	8.0 - 5
I-133	1.2 - 1	1.1 - 2	3.1 - 3	5.4 - 6
I-134	1.5 - 1	3.4 - 1	9.8 - 2	1.7 - 4
Cs-134	4.6 - 3	1.8 - 4	5.2 - 5	9.1 - 8
I-135	5.6 - 1	1.6 - 1	4.6 - 2	8.0 - 5
Cs-135	2.1 - 8	0.0	0.0	0.0
Cs-136	9.1 - 4	1.1 - 4	3.4 - 5	5.4 - 8
Cs-137	7.2 - 3	2.8 - 4	8.0 - 5	1.4 - 7
Ba-137m	6.7 - 3	0.0	0.0	0.0
Cs-137	4.0 - 3	2.9 - 1	8.3 - 2	1.4 - 4
Ba-139	8.4 - 3	2.3 - 1	6.8 - 2	1.2 - 4
Ba-140	7.9 - 2	1.0 - 2	3.0 - 3	5.2 - 6
La-140	7.8 - 2	0.0	0.0	0.0
Ba-141	2.2 - 3	2.8 - 1	8.0 - 2	1.4 - 4
La-141	2.2 - 3	0.0	0.0	0.0
Ce-141	2.2 - 3	4.5 - 5	1.3 - 5	2.3 - 8
Ba-142	1.2 - 3	2.6 - 1	7.4 - 2	1.3 - 4
La-142	1.2 - 3	0.0	0.0	0.0
Ce-143	3.6 - 5	4.1 - 5	1.2 - 5	2.1 - 8
Pr-143	3.9 - 4	4.4 - 5	1.3 - 5	2.2 - 8
Ce-144	9.8 - 4	4.0 - 5	1.2 - 5	2.0 - 8
Pr-144	9.8 - 4	0.0	0.0	0.0
Nd-147	1.1 - 4	1.6 - 5	4.6 - 6	8.0 - 9
Pm-147	3.5 - 6	0.0	0.0	0.0
W-187	2.0 - 3	3.2 - 3	9.2 - 4	1.6 - 6
Np-239	4.2 - 1	2.9 - 1	8.3 - 2	1.4 - 4
TOTAL	4.6 + 0	2.9 + 0	8.3 - 1	1.4 - 3

TABLE 11.2-5
DESIGN-BASIS ACTIVITY CONCENTRATIONS (in μCi/cc) IN
MAJOR LIQUID RADWASTE INPUT STREAMS (Cont'd)

ASSUMPTIONS USED TO DETERMINE THE DESIGN-BASIS
ACTIVITY CONCENTRATIONS IN MAJOR LIQUID WASTE INPUT STREAMS

a. Maximum anticipated concentrations or radionuclides are present in reactor water and steam (see Section 12.2).

b. Drain systems receive liquids with the highest concentrations (e.g., the containment building drain system receives 100% concentration of reactor water quality).

c. Processing equipment operates at maximum anticipated rates of flow.

d. Where more than one stream flows through a processing system, the most radioactive stream flows at its maximum anticipated rate. The remaining capacity of the system is composed of the other streams in order of decreasing radionuclide concentration.

TABLE 11.2-5A
DESIGN-BASIS ACTIVITY CONCENTRATIONS (in μCi/cc) IN
MAJOR LIQUID RADWASTE INPUT STREAMS
FOR THE ALTERNATE MODE OF RADWASTE OPERATIONS (SEE SEC. 11.2.2.8)

NUCLIDE	CHEMICAL WASTE SUBSYSTEM	EQUIPMENT DRAIN SUBSYSTEM	FLOOR DRAIN SUBSYSTEM	LAUNDRY DRAIN SUBSYSTEM
F-18	7.8 - 6	4.3 - 3	1.2 - 3	2.1 - 6
Na-24	3.2 - 5	2.1 - 3	6.2 - 4	1.1 - 6
P-32	3.8 - 6	2.1 - 5	6.2 - 6	1.1 - 8
Cr-51	1.1 - 4	5.3 - 4	1.5 - 4	2.7 - 7
Mn-54	1.0 - 5	4.3 - 5	1.2 - 5	2.1 - 8
Mn-56	1.4 - 4	5.3 - 2	1.5 - 2	2.7 - 5
Co-58	1.2 - 3	5.3 - 3	1.5 - 3	2.7 - 6
Fe-59	1.9 - 5	8.5 - 5	2.5 - 5	4.3 - 8
Co-60	1.3 - 4	5.3 - 4	1.5 - 4	2.7 - 7
Ni-65	8.2 - 7	3.2 - 4	9.2 - 5	1.6 - 7
Zn-65	5.2 - 7	2.1 - 6	6.2 - 7	1.1 - 9
Zn-69m	4.4 - 7	3.2 - 4	9.2 - 6	1.6 - 8
Zn-69	4.4 - 7	0.0	0.0	0.0
Br-83	4.4 - 5	1.9 - 2	5.5 - 3	9.7 - 6
Br-84	2.0 - 5	3.9 - 2	1.1 - 2	2.0 - 5
Br-85	1.2 - 6	2.7 - 2	7.7 - 3	1.3 - 5
Sr-89	7.9 - 4	3.5 - 3	1.0 - 3	1.8 - 6
Y-89m	7.9 - 8	0.0	0.0	0.0
Sr-90	6.6 - 5	2.7 - 4	1.1 - 4	1.3 - 7
Y-90	4.9 - 5	0.0	0.0	0.0
Sr-91	8.3 - 4	8.6 - 2	2.5 - 2	4.3 - 5
Y-91m	4.9 - 4	0.0	0.0	0.0
Y-91	1.3 - 4	0.0	0.0	0.0
Sr-92	4.0 - 4	1.5 - 1	4.3 - 2	7.5 - 5
Y-92	4.0 - 4	0.0	0.0	0.0
Zr-95	1.0 - 5	4.6 - 5	1.3 - 5	2.3 - 8
Nb-95m	1.4 - 7	0.0	0.0	0.0
Nb-95	1.2 - 5	4.8 - 5	1.4 - 5	2.4 - 7
Zr-97	6.5 - 7	3.8 - 5	1.1 - 5	1.9 - 8
Nb-97m	6.5 - 7	0.0	0.0	0.0
Nb-97	6.5.- 7	0.0	0.0	0.0
Mo-99	1.7 - 3	2.6 - 2	7.4 - 3	1.3 - 5
Tc-99m	2.1 - 3	1.0 - 1	3.0 - 2	5.2 - 5
Tc-99	2.5 - 10	0.0	0.0	0.0
Tc-101	5.0 - 5	2.3 - 1	6.8 - 2	1.2 - 4
Ru-103	4.9 - 6	2.2 - 5	6.5 - 6	1.1 - 8
Rh-103m	4.9 - 6	0.0	0.0	0.0
Ru-106	7.3 - 7	3.0 - 6	8.6 - 7	1.5 - 9
Rh-106	7.3 - 7	0.0	0.0	0.0
Ag-110m	1.5 - 5	6.4 - 5	1.8 - 5	3.2 - 8
Ag-110	2.0 - 7	0.0	0.0	0.0
Te-129m	8.4 - 5	3.9 - 4	1.1 - 4	2.0 - 7

TABLE 11.2-5A
DESIGN-BASIS ACTIVITY CONCENTRATIONS (in μCi/cc) IN
MAJOR LIQUID RADWASTE INPUT STREAMS
FOR THE ALTERNATE MODE OF RADWASTE OPERATIONS (SEE SEC. 11.2.2.8)
(Continued)

NUCLIDE	CHEMICAL WASTE SUBSYSTEM	EQUIPMENT DRAIN SUBSYSTEM	FLOOR DRAIN SUBSYSTEM	LAUNDRY DRAIN SUBSYSTEM
Te-129	5.4 - 5	0.0	0.0	0.0
I-129	7.3 - 14	0.0	0.0	0.0
I-131	2.2 - 3	1.6 - 2	4.6 - 3	8.0 - 6
Te-132	1.2 - 3	1.7 - 2	4.9 - 3	8.6 - 6
I-132	1.6 - 3	1.6 - 1	4.6 - 2	8.0 - 5
I-133	2.2 - 4	1.1 - 2	3.1 - 3	5.4 - 6
I-134	2.8 - 4	3.4 - 1	9.8 - 2	1.7 - 4
Cs-134	4.5 - 5	1.8 - 4	5.2 - 5	9.1 - 8
I-135	1.1 - 3	1.6 - 1	4.6 - 2	8.0 - 5
Cs-135	9.4 - 12	0.0	0.0	0.0
Cs-136	2.0 - 5	1.1 - 4	3.4 - 5	5.4 - 8
Cs-137	6.8 - 5	2.8 - 4	8.0 - 5	1.4 - 7
Ba-137m	6.4 - 5	0.0	0.0	0.0
Cs-137	1.4 - 4	2.9 - 1	8.3 - 2	1.4 - 4
Ba-139	2.9 - 4	2.3 - 1	6.8 - 2	1.2 - 4
Ba-140	1.7 - 3	1.0 - 2	3.0 - 3	5.2 - 6
La-140	1.5 - 3	0.0	0.0	0.0
Ba-141	7.7 - 5	2.8 - 1	8.0 - 2	1.4 - 4
La-141	7.7 - 5	0.0	0.0	0.0
Ce-141	3.0 - 5	4.5 - 5	1.3 - 5	2.3 - 8
Ba-142	4.5 - 5	2.6 - 1	7.4 - 2	1.3 - 4
La-142	4.5 - 5	0.0	0.0	0.0
Ce-143	1.4 - 6	4.1 - 5	1.2 - 5	2.1 - 8
Pr-143	8.2 - 6	4.4 - 5	1.3 - 5	2.2 - 8
Ce-144	9.8 - 6	4.0 - 5	1.2 - 5	2.0 - 8
Pr-144	9.8 - 6	0.0	0.0	0.0
Nd-147	2.6 - 6	1.6 - 5	4.6 - 6	8.0 - 9
Pm-147	1.6 - 8	0.0	0.0	0.0
W-187	7.6 - 5	3.2 - 3	9.2 - 4	1.6 - 6
Np-239	1.5 - 2	2.9 - 1	8.3 - 2	1.4 - 4
TOTAL	3.5 - 2	2.9 + 0	8.3 - 1	1.4 - 3

TABLE 11.2-5A
DESIGN-BASIS ACTIVITY CONCENTRATIONS (in μCi/cc) IN
MAJOR LIQUID RADWASTE INPUT STREAMS
FOR THE ALTERNATE MODE OF RADWASTE OPERATIONS (SEE SEC. 11.2.2.8)
(Continued)

ASSUMPTIONS USED TO DETERMINE THE DESIGN-BASIS
ACTIVITY CONCENTRATIONS IN MAJOR LIQUID WASTE INPUT STREAMS

a. Maximum anticipated concentrations or radionuclides are present in reactor water and steam (see Section 12.2).

b. Drain systems receive liquids with the highest concentrations (e.g., the containment building drain system receives 100% concentration of reactor water quality).

c. Processing equipment operates at maximum anticipated rates of flow.

d. Where more than one stream flows through a processing system, the most radioactive stream flows at its maximum anticipated rate. The remaining capacity of the system is composed of the other streams in order of decreasing radionuclide concentration.

TABLE 11.2-6
EXPECTED FLOWS INTO THE LIQUID
RADWASTE SYSTEM DURING NORMAL
AND ANTICIPATED OPERATING CONDITIONS

1.0 NORMAL OPERATIONS[(1)] [(2)]

1.1 Expected normal volumes to Equipment Drain Subsystem

Equipment drain sumps and tanks plus drywell floor sumps	7,400 gal/day
Demineralizer Polisher resin cleaning per vessel	58,286 gal/14 days
Condensate filter backwashes per vessel	3,607 gal/5 days
Decant from sludge tanks	
Phase separators	4,455 gal/6 days
Fuel pool F/D sludge	4,000 gal/14 days
Waste sludge	3,488 gal/2 days
Spent resin	500 gal (infrequent)
Average daily total	26,990 gallons
Maximum single day total due to coincidence of normal loads (consisting of normal equipment drain plus one condensate backwash, one radwaste backwash, two NRC rinses, all decants, plus one additional waste sludge decant due to a filter backwash)	49,025 gallons
Minimum single day (following above maximum coincidence; no backwashes, no decants, two URC rinses)	15,200 gallons

1.2 Expected normal volumes to Floor Drain Subsystem

Floor drains	5,000 gal/day
Decants	1,000 gal/day
Average daily total	6,000 gal/day
Maximum coincidence (add 1,500 gallons decant)	7,500 gal/day
Minimum	5,000 gal/day

TABLE 11.2-6
EXPECTED FLOWS INTO THE LIQUID
RADWASTE SYSTEM DURING NORMAL
AND ANTICIPATED OPERATING CONDITIONS (Continued)

1.3 Expected normal volumes to Chemical Waste Subsystem

Regeneration solution	
Radwaste demineralizer	5,000 gal/14 days
Sample and decontamination drains	500 gal/day
Laundry detergent waste	1,000 gal/day
Average daily total	1,857 gal/day
Maximum due to coincidence	6,500 gallons
Minimum due to coincidence	1,500 gallons

1.4 In normal operation, the Laundry Waste Subsystem is expected to receive 1,000 gallons per day of detergent water. The maximum normal volume is expected to be 4000 gallons per day for a 30-day period of time.

2.0 ABNORMAL OCCURRENCES

2.1 Startup (maximum flows)

Equipment drains	
Maximum day	100,000 gallons
Maximum 3-day average	50,000 gallons
Floor drains	
Maximum day	100,000 gallons
Maximum 3-day average	50,000 gallons
Condensate polisher	
Maximum backwash	36,100 gallons
Maximum ultrasonic cleaning four/day	31,200 gallons
Maximum RWCU backwash four/day	7,200 gallons

2.2 Outages:

Reactor hydrotest	38,000 gallons

TABLE 11.2-6
EXPECTED FLOWS INTO THE LIQUID
RADWASTE SYSTEM DURING NORMAL
AND ANTICIPATED OPERATING CONDITIONS (Continued)

	Reactor expansion water	26,000 gallons
	CRD testing	18,000 gallons
	Various system vents and drains	10,000 gallons
	Laundry	4,000 gal/day/30 days
2.3	Other occurrences, maxima:	
	Condensate polisher backwash four/day	36,100 gallons
	RWCU backwash, four/day	7,200 gallons
	Radwaste filter backwash four/day	5,000 gallons
	Maximum leak rate	
	Equipment drains, 20 gpm	28,800 gal/day
	Floor drains, 20 gpm	28,800 gal/day

TABLE 11.2-6
EXPECTED FLOWS INTO THE LIQUID
RADWASTE SYSTEM DURING NORMAL
AND ANTICIPATED OPERATING CONDITIONS (Continued)

Notes:

(1) Per GE Plant Requirement Documents No. 22A2739A, "Radioactive Waste," and 22A2707, "Water Quality"; Environmental Protection Branch, Directorate of Regulatory Operations, USAEC, 1973, "Results of Independent Measurements of Radioactivity in Process Streams and Effluents at BWR's"; Directorate of Regulatory Standards, USAEC, 1973, WASH 1258, "Final Environmental Statement Concerning Proposed Rulemaking Action: Numerical Guides for Operation to meet the Criterion"; "As Low as is Practicable" for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents draft; "As low as is Reasonably Achievable" Regulatory Guides; vendor design data; and operating experience data.

(2) The maximum and minimum values due to coincidence represent volumes which can occur during normal operation as a result of variations in scheduling of operations outside of the radwaste systems. The probability of these events is difficult to estimate, but they can happen so they must be anticipated. It should be noted that if the maximum coincidence volume occurs in a single day, the minimum must occur on the following day. Therefore, the average daily volume over any 3 to 4 day period of normal operation should be very close to the stated average.

The uncertainty in the stated expected volumes is estimated at ± 20%.

(3) Not applicable for the alternate mode of operation in which resin are not regenerated. (See Section 11.2.2.8).

Table 11.2-7
DATA AND THEIR REFERENCES
FOR CALCULATING EXPECTED RELEASES OF RADIONUCLIDES IN LIQUID EFFLUENTS
FOR NORMAL OPERATION AND ANTICIPATED OPERATIONAL OCCURRENCES

	ITEM DESCRIPTION	INPUT	REFERENCE
1.	GENERAL		
	Maximum core thermal power level (MWt)	3473	NEDC-32989
	Tritium released in liquid effluent (Ci/yr/reactor)	12	USAR Table 11.2-1
	Tritium released in gaseous effluent (Ci/yr/reactor)	44	USAR Table 11.3-9
	Annual average dilution flow rate for liquid waste discharge (gpm)	610,000	Calculation 1SX01
2.	NUCLEAR STEAM SUPPLY SYSTEM		
	Total steam flow rate (lb/hr)	15.153×10^6	NEDC-32989
	Mass of reactor coolant in reactor vessel at full power (lb)	515,558	General Electric Company/ Sargent & Lundy
	Mass of steam in reactor vessel at full power (lb)	10,740	General Electric Company/ Sargent & Lundy
3.	REACTOR WATER CLEANUP		
	Average flow rate (lb/hr)	124,000	General Electric Company
	Number of demineralizers	2	General Electric Company
	Type of demineralizer	Powdered resin	General Electric Company
	Replacement frequency (batch/day): Normal Startup	1/7 days 1	General Electric Company General Electric Company
	(1 batch includes 2 filter/demineralizer backwashes)		
	Regenerant volume and activity, if applicable (gal/event)	1075 gal/event 1200 curies	General Electric Company General Electric Company

Table 11.2-7
DATA AND THEIR REFERENCES
FOR CALCULATING EXPECTED RELEASES OF RADIONUCLIDES IN LIQUID EFFLUENTS
FOR NORMAL OPERATION AND ANTICIPATED OPERATIONAL OCCURRENCES (Continued)

ITEM DESCRIPTION	INPUT	REFERENCE
4. CONDENSATE DEMINERALIZERS		
Average flow rate (lb/hr)	1.512×10^7	NEDC-32989
Demineralizer type	Mixed Deep Bed	Contract K-2844
Number of demineralizer (Note: There are 9 demineralizers in parallel)	9	Contract K-2844
Size of one demineralizer (ft^3)	195	Contract K-2844
Bed replacement, days for each bed	216	USAR Table 11.4-3
Is ultrasonic resin cleaning used?	Yes	Contract K-2884
Waste liquid volume due to URC (gal/day)	16,500	L*A Water Treatment Drawings A-10649, A-10793
Regenerate backwash chemical volume (gal/event)[1]	13,505	L*A Water Treatment Drawings A-10649, A-10793
Activity of regenerant backwash chemical volume [1]	1.72×10^2 curies	USAR Table 12.2-8

5. LIQUID WASTE PROCESSING SYSTEMS

	SOURCE	FLOW RATE	FRACTION OF PCA	REFERENCE
a.	High Purity (Equipment) Waste	7400 gal/day		USAR Table 11.2-6
	Drywell equipment drain	7400 gal/day		USAR Table 11.2-6
	Containment equipment drain	7400 gal/day		USAR Table 11.2-6
	Auxiliary building equipment drain	7400 gal/day		USAR Table 11.2-6

[1] When resins are regenerated

TABLE 11.2-7
DATA AND THEIR REFERENCES
FOR CALCULATING EXPECTED RELEASES OF RADIONUCLIDES IN LIQUID EFFLUENTS
FOR NORMAL OPERATION AND ANTICIPATED OPERATIONAL OCCURRENCES (Continued)

SOURCE	FLOW RATE	FRACTION OF PCA	REFERENCE
Radwaste building equipment drain	7400 gal/day		USAR Table 11.2-6
Turbine building equipment drain	7400 gal/day		USAR Table 11.2-6
RWCU phase separator decant	742 gal/day		USAR Table 11.2-6
Fuel pool filter – demineralizer sludge decant	285 gal/day		USAR Table 11.2-6
Waste sludge decant	1744 gal/day		USAR Table 11.2-6
Condensate filter backwash (6 vessels)	4328 gal/day		USAR Table 11.2-6
Demineralizer polisher resin cleaning	12,490 gal/day		USAR Table 11.2-6
Total	26,990 gal (avg. daily total) 49,025 gal (max, single day) 15,200 gal (min. single day)	0.1	USAR Table 11.2-6

	ITEM DESCRIPTION	INPUT	FRACTION OF PCA	REFERENCE
b.	Low Purity (floor drain) Waste			
	Radwaste building floor drain			
	Drywell floor drain			
	Containment floor drain	5000 gal/day		USAR Table 11.2-6
	Auxiliary building floor drain			
	Turbine building floor drain and control building hot machine shop			
	Decants	10000 gal/day		USAR Table 11.2-6
	Totals	6000 gal (avg. daily total) 7500 gal (max. single day) 5000 gal (min. singe day)	0.0001	USAR Table 11.2-6
c.	Chemical (Nonregenerant) Waste			
	Turbine building chemical waste sumps			

TABLE 11.2-7
DATA AND THEIR REFERENCES
FOR CALCULATING EXPECTED RELEASES OF RADIONUCLIDES IN LIQUID EFFLUENTS
FOR NORMAL OPERATION AND ANTICIPATED OPERATIONAL OCCURRENCES (Continued)

	ITEM DESCRIPTION	INPUT	FRACTION OF PCA	REFERENCE
	Auxiliary building chemical waste sumps	500 gal/day		USAR Table 11.2-6
	Radwaste building chemical waste sump			
	Total	500 gal (avg. daily total)	0.02	USAR Table 11.2-6
d.	Regenerant Solution Waste [1]	360 gal	0.058	USAR Table 11.2-6
e.	Detergent Waste	1000 gal (avg. daily total) 4000 gal (max. single day)		USAR Table 11.2-6
f.	Holdup Times Associated with Collection, Processing, and Discharge of All Liquid Streams			

WASTE STREAM	COLLECTION TIME (days)	PROCESS TIME (T_p) (days)	DISCHARGE TIME (Td) (days)	EFFECTIVE DECAY TIME (T_p + (Td/2)	REFERENCE
High purity waste	0.63	0.056	0.046	0.079	S&L Calc. No. R-O
Low purity waste	1.67	0.46	0.046	0.483	S&L Calc. No. R-O
Chemical waste*	7.09	0.65	0.046	0.763	S&L Calc. No. R-O
Regenerant solution waste* [1]	7.09	0.65	0.046	0.763	S&L Calc. No. R-O

g. Capacities of All Tanks and Processing Equipment Considered in Calculation Holdup Times

WASTE STEAM	COLLECTOR TANK NUMBER AND CAPACITY (gal)	COLLECTOR TANK PUMP NUMBER AN FLOW RATE (gpd)	PROCESSING EQUIPMENT TYPE, NUMBER, AND FLOW RATE (gpd)	SAMPLE TANK NUMBER AND CAPACITY (gal)	SAMPLE TANK PUMP NUMBER AND DISCHARGE FLOW RATE (gpd)	REFERENCE
High Purity	1WE01T	1WE01P	Waste Filter	Waste Sample	Waste Sample	USAR Table 11.2-3
Waste	30,000 gal	432,000 gpd	OWE01FA/B/C 432,000 gpd	OWE02TA/B/C 30,000 gal	OWE02PS/B/C 432,000 gpd	Drawings M05-1081 and M05-1085

[1] When resins are regenerated

* These waste streams are combined

TABLE 11.2-7
DATA AND THEIR REFERENCES
FOR CALCULATING EXPECTED RELEASES OF RADIONUCLIDES IN LIQUID EFFLUENTS
FOR NORMAL OPERATION AND ANTICIPATED OPERATIONAL OCCURRENCES (Continued)

WASTE STEAM	COLLECTOR TANK NUMBER AND CAPACITY (gal)	COLLECTOR TANK PUMP NUMBER AN FLOW RATE (gpd)	PROCESSING EQUIPMENT TYPE, NUMBER, AND FLOW RATE (gpd)	SAMPLE TANK NUMBER AND CAPACITY (gal)	SAMPLE TANK PUMP NUMBER AND DISCHARGE FLOW RATE (gpd)	REFERENCE
			Waste Demineralizer OWE1DA/B/C 432,000 gpd			
Low Purity Waste	1WF01T 25,000 gal	1WF01P 432,000 gpd	Evaporator 1WF01S 43,200 gpd	Evaporator Monitor 1WF04T 15,000 gal	Evaporator Monitor 1WF04P 216,000 gpd	USAR Table 11.2-3 Drawings M05-1081 and M05-1085
Chemical Waste	1WZ01T 35,000 gal	1WZ01P 144,000 gpd	Evaporator OWZ01S 43,200 gpd	Evaporator Monitor OWZ01T 15,000 gal	Evaporator Monitor OWZ01P216,000 gpd	USAR Table 11.2-3 Drawings M05-1081 and M05-1085
Regenerant [1] solution waste	Goes to chemical waste stream; chemical waste stream data apply here.					

h. Decontamination Factors for Each Processing Step

WASTE STREAM	PROCESSING EQUIPMENT	DECONTAMINATION FACTOR IODINE	DECONTAMINATION FACTOR Cs & Rs	DECONTAMINATION FACTOR OTHER	REFERENCE
High Purity waste	Waste Filter Waste Demineralizer	1.0E+03	1.0E+02	1.0E+03	NUREG-0016, Revision 1
Low Purity waste	Evaporator	1.0E+03	1.0E+04	1.0E+04	NUREG-0016, Revision 1
Chemical waste	Evaporator	1.0E+05	1.0E+05	1.0E+06	NUREG-0016, Revision 1
Regenerant solution waste	Chemical Waste Evaporator	1.0E+05	1.0E+05	1.0E+06	NUREG-0016, Revision 1

[1] When resin are regenerated

TABLE 11.2-7
DATA AND THEIR REFERENCES
FOR CALCULATING EXPECTED RELEASES OF RADIONUCLIDES IN LIQUID EFFLUENTS
FOR NORMAL OPERATION AND ANTICIPATED OPERATIONAL OCCURRENCES (Continued)

i. Fraction of Each Processing Stream Expected to be Discharged Over the Life of the Plant

WASTE STREAM	FRACTION DISCHARGED	REFERENCE
High purity waste	0.01	NUREG-0016, Revision 1
Low purity waste	0.10	NUREG-0016, Revision 1
Chemical waste	0.10	NUREG-0016, Revision 1
Regenerant solutions waste	0.10	NUREG-0016, Revision 1

11.3 GASEOUS WASTE MANAGEMENT SYSTEM

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 10 CFR 50, May 5, 1975). This is to be accomplished while maintaining occupational exposure as low as is reasonably achievable and without limiting plant operation or availability.

11.3.1.2 Design Criteria

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 10 CFR 20 and to operate within the emission rate limits established in the technical specification.

As a design basis for this system, an annual average noble radiogas source term (based on 30-minute decay) of 100,000 µCi/sec of the "1971 Mixture" will be used. Table 11.3-1 indicates the design-basis noble radiogas source terms referenced to 30-minute decay.

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 10 CFR 50 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 is reasonably achievable in accordance with Nuclear Regulatory Commission Regulatory Guide 8.8.

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

General Design Criterion 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 avoid any termination of releases or limitation of plant operation due to unfavorable site environmental conditions.

General Design Criterion 64

Continuous monitoring of activity levels in the system upstream of the charcoal beds provides advance notice of any potentially significant increase in releases. Continuous monitoring of the system effluent, with automatic isolation at activity levels corresponding to Technical Specification release limits and annunciation at lower levels, along with continuous monitoring of the-plant vent release, provide assurance that activity releases to the environment will, in all events, be 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, number of units supplied, and the design codes is provided in Table 11.3-2. Equipment and piping will be designed and constructed in accordance with the requirements of the applicable codes as given in Table 3.2-1 and 3.2-2.

The Quality Group Classifications 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 Quality Group Classification D.

Conservative analyses similar to that presented in Reference 1 demonstrate that equipment failure will not result in doses exceeding 0.5 rem.

This analysis is presented in Subsection 15.7.1.1. The related failure of the steam jet air ejector lines and the gland seal off-gas lines are analyzed in Subsections 15.7.1.3 and 15.7.1.2, respectively.

The containment, 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.

The charcoal adsorber tank support design criteria are given in Table 11.3-2. Plastic pipe of any type is prohibited in the gaseous radwaste system.

11.3.2 System Description

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 11.3-2 and in the following paragraphs.

11.3.2.1 Main Condenser Steam Jet Air Ejector Low-Temp RECHAR System

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

The air ejector off-gas will normally contain activation gases, principally, N-16, 0-19, and N-13. N-16 and 0-19 have short half-lives and are readily decayed. N-13 is present in small amounts and has a half-life of ten minutes.

The air ejector off-gas will also contain 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.

11.3.2.1.1 Process Description

A main condenser off-gas treatment system has been incorporated in the plant design to reduce the gaseous radwaste emission 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 further cooled to 45° F for additional water removal. The gas is then passed through a desiccant dryer that reduces the dewpoint to approximately -60°. The gas is then chilled to less than 50°F. Charcoal adsorption beds, operating in a refrigerated vault at less than 50°F, selectively adsorb and delay the xenons and kryptons from the bulk carrier gas (principally dry air). After the delay, the gas is passed through a HEPA filter and discharged to the environment through the plant vent.

11.3.2.1.1 Process Flow Diagram

Drawing 105D5099 is the process flow diagram for the system. The process data for startup and normal operating conditions are submitted as proprietary data under separate cover on Reference 12.

The information supporting the process data is presented in Reference 2. The off-gas system discharge is routed to the plant vent as indicated on Drawings 105D5099 and M05-1084.

11.3.2.1.2 Noble Gas Radionuclide Source Term and Decay

The design-basis isotopic source terms for the annual average activity input of the main condenser off-gas treatment system are given in Table 11.3.1 at t=30 minutes. The system is mechanically capable of processing three times the source terms of Table 11.3-1 without affecting delay time of the noble gases. Also listed is the isotopic distribution in a delay of 46 hours for krypton and 42 days for xenon.

Tables 11.3-1 and 11.3-1a list isotopic activities at the discharge of the system, and the decontamination factor for each noble gas isotope can be determined.

Section 11.1.1.1 presents source terms for normal operational and anticipated occurrence releases to the primary coolant.

11.3.2.1.3 Piping and Instrumentation Diagram (P&ID)

The P&ID is Drawing M05-1084.

11.3.2.1.4 Recombiner Sizing

The basis for sizing the recombiner is to maintain the hydrogen concentration below 4% (including steam) at the inlet and below 1% at the outlet on a dry basis. The exit hydrogen concentration is normally well below the 1% maximum allowed. The hydrogen generation rate of the reactor is based on data from 9 BWR's. The hydrogen generation rate is given in Reference 12.

11.3.2.1.5 Process Design Parameters

The krypton and xenon holdup time is closely approximated by the following equation:

$$T = \frac{K_D M}{V} \qquad (11.3\text{-}1)$$

where:

T	=	holdup time of a given gas,
K_D	=	dynamic adsorption coefficient for the given gas,
M	=	weight of charcoal, and
V	=	flow rate of the carrier gas in consistent units.

Dynamic adsorption coefficient values for xenon and krypton were reported by Browning (Reference 1). General Electric has performed pilot plant tests at their Vallecitos Laboratory and the results were reported at the 12th AEC Air Cleaning Conference (Reference 3). Moisture has a detrimental effect on adsorption coefficients. It is to prevent moisture from reaching the charcoal that the -60°F dewpoint redundant, adsorbent air dryers are supplied. There are redundant moisture analyzers that will alarm on breakthrough of the dryer beds; however, breakthrough is not expected since the dryer beds will be 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 into the main condenser. The air inleakage design basis is conservatively sized at approximately 55 scfm total. The Sixth Edition of Heat Exchange Institute Standards for Steam Surface Condensers (Reference 4) Paragraph SI(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. 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 5 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 to prevent back-seepage of hydrogen from the operating recombiner, in the event of failure of that recombiner.

11.3.2.1.6 Charcoal Adsorbers

11.3.2.1.6.1 Charcoal Temperature

The charcoal adsorbers operate at less than 50°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 will cause an alarm in the control room.

For no-flow conditions, decay heat produced by absorbed radionuclides, give a maximum axial temperature well below the minimum charcoal ignition temperature of 374° F.

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 having 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 (Reference 5). During transfer of the charcoal into the charcoal adsorber vessels radial sizing of the charcoal will be 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 bypass line, isolated with double block and bleed valves, is 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 and when moisture is most likely to enter the charcoal beds. In addition, there is a separate mode for drying the charcoal beds through use of the regenerator skid should there be a high moisture buildup in the charcoal.

The bypass mode is used during plant startup, since off-gas activity will be low and the likelihood of wetting the charcoal beds is increased. The bypass mode is also used when the plant is shut down. This bypass mode would not be used for normal operation unless some unforeseen system malfunction would necessitate shutting down the power plant or operating in the bypass mode and remaining within the Offsite Dose Calculation Manual limits (Sections 3.4 and 3.5). The activity release is monitored by a process radiation monitor upstream of the vent isolation valve that will cause the bypass valve to close on an alert level radiation alarm. This interlock can be defeated only by a keylock switch. In addition, there is a high level alarm on the same monitor that will cause the off-gas system to be isolated from the vent if established release limits are exceeded. The high level alarm setting is addressed by the Offsite Dose Calculation Manual.

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

To limit the recombiner temperature rise, hydrogen concentration of gases from the air ejector is kept below 4% hydrogen 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 is at 2% hydrogen by volume because this concentration indicates "off-normal" operation.

The indication, annunciation, and control for the hydrogen analyzers in the off-gas system (instrument numbers N66-N012A and N66-N012B) exist only in the main control room. (Panel H13-P845). The hydrogen analyzers are nonsparking and are designed to withstand a hydrogen explosion. They are not required to function after such an explosion. The analyzers are currently shown on drawings E02-1OG99, Sheets 313 and 314. (Q&R 460.11)

11.3.2.1.9 Field Run Piping

Piping and tubing 2 inches and under may be field routed. This does not include major process piping but does include drain lines, steamlines, and sample lines which are shown on the P&ID (Drawing M05-1084).

11.3.2.1.10 Liquid Seals

To ensure against leakage of potentially detonable process gas to areas external to the system pressure boundary, process drains upstream of the catalyst section of the recombiner are routed to the main turbine condenser.

11.3.2.1.11 System Performance

Noble gas activity release is about 50-60 µCi/sec from the exit of the steam jet air ejector off-gas system based upon 30 scfm air inleakage and an input of 100,000 µCi/sec of 30-minute-old "1971 Mixture." The isotopic composition is given in Table 11.3-1 in units of µCi/sec and Ci/yr.

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% of the iodine in 2 inches of charcoal, whereas this system has approximately 76 feet of charcoal in the flow path.

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.

The charcoal adsorber trains are capable of being bypassed, thereby decreasing the delay time of the system to a nominal 10 minutes for krypton and 1 hour for xenon, provided by a separate charcoal bed at design basis normal flow. Therefore, it is intended that the charcoal bypass line be closed, except as noted in Subsection 11.3.2.1.6.3.

11.3.2.1.12 Isotopic Inventory

The isotopic inventory of each equipment piece is given in Table 12.2-11 for design bases conditions.

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" (Reference 2). Nonproprietary 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 failures is provided in Table 11.3-5.

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

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

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

c. All discharge paths to the environment are monitored; the normal effluent path by the process radiation monitoring system and the equipment areas by the area radiation monitoring system and continuous air monitoring system.

d. Dilution steam flow to the stream 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 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 design objectives of station ventilation systems with respect to the airborne radioactivity levels in the ventilated areas are discussed in Subsection 12.3.3. The ventilation flow rates are discussed in Section 9.4.

The steam discharges through the safety relief valves will release activity. 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 11.3-6.

11.3.2.1.16 Maintain Ability of Gaseous Radwaste System

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

a. redundant components for all active, in-process equipment pieces;

b. no rotating equipment in the radioactive process stream, but located only where maintenance can be performed while the system is in operation or on nonradioactive streams;

c. block valves with air bleed pressurization for maintenance which is required during plant operation; and

d. shielding of nonradioactive auxiliary systems from the radioactive process stream.

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

a. extremely stringent leak rate requirements placed upon all equipment, piping, and instruments, and enforced by requiring as-installed helium leak tests of the entire process system;

b. use of welded joints wherever practicable;

c. specification of valve types with extremely low leak rate characteristics, i.e., bellows seal, double stem seal, or equal;

d. routing of drains through steam traps to the main condenser; and

e. 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 Main Condenser Steam Jet Air Ejector Off-Gas Low-Temp System

11.3.2.2.1.1 Quality Classification and Construction and Testing Requirements

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

11.3.2.2.1.2 Seismic Design

11.3.2.2.1.2.1 Equipment

Equipment and components used to collect, process, or store gaseous radioactive waste are classified as non-Seismic Category I as indicated in Table 3.2-1. Conservative analyses presented in Subsection 15.7.1 demonstrate that equipment failure will not result in doses exceeding the 0.5 rem guidelines of Regulatory Guide 1.29.

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

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

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

c. The charcoal adsorber tanks including the support elements are designed with a horizontal static coefficient of 0.2 g.

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

These seismic requirements are acceptable alternatives to the requirements of Regulatory Guide 1.143 (formerly ETSB, BTP 11-1 (Revision 1) as discussed in Reference 8).

11.3.2.2.1.2.2 Buildings Housing Off-Gas Processing Systems

The turbine building, which houses portions of the off-gas system, is a non-Seismic Category I building. The radwaste building, which houses portions of the off-gas system including adsorbers complies with the intent of Regulatory Guide 1.143.

11.3.2.2.1.3 Quality Control

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

a. 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.

b. Control of Purchased Material, Equipment, and Services - Procedures are established to ensure that purchased material, equipment, and construction services conform to the procurement documents.

c. Inspection - A program for inspection of activities affecting quality are 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.

d. 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.

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

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

11.3.2.2.1.4 Welding

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

11.3.2.2.1.5 Materials

Materials for pressure retaining components of radioactive process systems will be 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 will not be used. The components will 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 11.3-2.

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

11.3.2.2.1.6 Construction of Process Systems

Pressure retaining components of process systems will utilize welded construction to the maximum practicable extent. Process piping systems include the first root valve on sample and instrument lines. Process lines will not be less than 3/4-inch nominal pipe size. Sample and instrument lines are not considered as portions of the process systems. Flanged joints or suitable rapid disconnect fittings will not be used except where maintenance requirements clearly indicate that such construction is preferable. Screwed connections in which threads provide the only seal will not be used. Screwed connections backed up by seal welding, or mechanical joints, will be used only on lines of 3/4-inch or greater but less than 2-1/2-inch nominal pipe size. Butt welding using consumable inserts or open butt welding technique is the preferred welding connection. For some lines 3/4-inch or greater, but less than 2-1/2-inch nomina1 pipe size, socket type welds shall be used. In lines 2-1/2-inch nominal pipe size and larger, pipe welds will be of the butt joint type.

11.3.2.2.1.7 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 will be held for a minimum of 30 minutes with no leakage indicated. Pneumatic testing may be substituted for hydrostatic testing in accordance with the applicable codes.

11.3.2.2.1.8 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 11.3-4 lists the process parameters that are instrumented to alarm in the control room. It also indicates whether the parameters are recorded or just indicated. Instrument alarms on the standby recombiner train are deactivated in order to eliminate continuous operationally meaningless alarms thereby improving operator response to important process variables on the operating recombiner train. Instrumentation and controls are described in Subsection 7.7.1.10. The operator is in control of the system at all times.

A radiation monitor after the cooler-condenser continuously monitors radioactivity release from the reactor and input to the charcoal absorbers. 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 vent.

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 effluent 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 Subsection 7.6.1.2.

Environmental monitoring will be 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.

11.3.2.2.1.9 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 (Reference 6).

By this procedure 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 will sustain the whole envelope of feasible detonations.

11.3.2.2.1.10 Operator Exposure Criteria and Controls

The 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.

11.3.2.2.1.11 Equipment Malfunction

Malfunction analysis, indicating consequences and design precautions taken to accommodate failure of various components of the system, is given in Table 11.3-5.

11.3.2.2.1.12 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 Fuskushima I plants in Japan have similar recombiners in service. Similar systems (ambient temperature charcoal) are in service at Dresden 2 & 3, Pilgrim, Quad Cities 1 & 2, Nuclenor, Hatch, Browns Ferry 1, 2, and 3, and Duane Arnold.

11.3.2.3 Operating Procedure

11.3.2.3.1 Treated (Delayed) Radioactive Gas Sources

11.3.2.3.1.1 Main Condenser Steam Jet Air Ejector Off-Gas Low-Temp RECHAR System

11.3.2.3.1.1.1 Prestartup Preparations

Prior to starting the main steam jet air ejectors (SJAE), the charcoal vault is cooled, 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 preheat steam supply is turned on.

11.3.2.3.1.1.2 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 by a fixed steam supply to less then 4% hydrogen by volume, and the off-gas condenser outlet is maintained at less than 1% hydrogen by volume.

11.3.2.3.1.1.3 Normal Operation

After startup, the noncondensibles pumped by the SJAE will 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.

11.3.2.3.1.1.4 Previous Experience

Previous experience is reviewed in Subsection 11.3.2.2.1.12.

11.3.2.4 Performance Tests

11.3.2.4.1 Treated (Delayed) Radioactive Gas Sources

11.3.2.4.1.1 Main Condenser Steam Jet Air Ejector Off-Gas Low-Temp RECHAR System

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

11.3.2.4.1.1.1 Recombiner

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

11.3.2.4.1.1.2 Desiccant Gas Drier

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

11.3.2.4.1.1.3 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 cooler-condenser and at the exit of the charcoal adsorbers.

Experience with boiling water reactors has shown that the calibration of the off-gas and vent 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.

11.3.2.4.1.1.4 Post Filter

On installation, replacements, and at periodic intervals during operation, the particulate filter assembly is tested using a DOP (dioctylphthalate) aerosol to verify whether the installed filter configuration meets the minimum in place efficiency of 99.95% retention.

11.3.2.4.1.1.5 Previous Experience

Previous experience is reviewed in Subsection 11.3.2.2.1.12.

11.3.3 Radioactive Releases

11.3.3.1 Release Points

Airborne radioactive releases to the environs from the building ventilation and the off-gas system are from a single common plant vent. The plant vent release point is above the containment building dome which is the tallest structure in the power block. The plant vent releases ventilation air from the containment, fuel, turbine, portions of the radwaste building, control (except for the control room), auxiliary buildings and discharges from the mechanical vacuum pumps and process off-gas. The standby gas treatment system has its own vent with the release point adjacent to the plant vent. Process off-gas is also released via the plant vent. The plant vent is approximately 200 feet in height and has the inside dimensions of 10 x 12 feet at the point of release. The plant vent and SGTS release points are shown in Figure 6.4-3. Plant vent effluents have an exit velocity of less than 2000 ft/min and a temperature of 65° F to 122° F.

11.3.3.2 Calculated Releases

Expected releases of gaseous radioactivity to the environment are based upon BWR operating experience. This experience has shown that for BWRs operating with predominately 8x8 fuel,

which was the fuel type incorporated initially into the Clinton Nuclear Power Station, that annual release rates equivalent to less than 25,000 μCi/sec as evaluated at t=30 minutes are typical. For this reason the noble gas release rates noted in Table 11.3-9, which are based upon an equivalent 30 minute old mixture of 50,000 μCi/sec, as calculated by the BWR GALE code (Reference 9), have been reduced by a factor of 2. Other fission products released to the environment are as noted in Table 11.3-9 at charcoal temperatures of 0, 20, and 40°F.

11.3.3.3 Dilution Factors

Based on meteorological data collected at the Clinton site (May 1972 - April 1977), annual average dilution factors were estimated. The X/Q and D/Q values for several locations of interest are presented in Table 11.3-10.

11.3.3.4 Estimated Doses

As noted in Section 11.3.3.2, the radiological consequences resulting from exposure to the released noble gases are based upon one half of the noble gas release rates specified in Table 11.3-9 while the inhalation and ingestion doses resulting from exposure to the released iodines and particulates are based upon the values in Table 11.3-9. The calculated radiological exposures resulting from normal operation are based on the models presented in References 10 and 11.

11.3.3.4.1 Gamma Dose

The calculated off site gamma doses resulting from normal operation are presented in Table 11.3-11.

11.3.3.4.2 Annual Average Concentrations

The annual average concentrations of noble gases released to the environment are a small fraction of those limits presented in 10 CFR 20, Appendix B, Table 2, Column 1.

11.3.3.4.3 Milk Ingestion Dose

The annual airborne release for iodine-131 is presented in Table 11.3-9. Based upon 7 months of grazing per year and with the nearest milk cow or goat located 5 miles from the station, the milk ingestion dose for various age groups are presented in Table 11.3-11 for the worst sector (ESE). The controlling location may change based on information obtained during the performance of the periodic land use census.

11.3.3.5 Treated (Delayed) Radioactive Gas Sources

11.3.3.5.1 Main Condenser Steam Jet Air Ejector Low-Temperature RECHAR System

11.3.3.5.1.1 Normal Operation

Based upon an equivalent design basis input rate to the offgas system of 100,000 μCi/sec based upon a 30-minute decay, the 1 estimated release rate of noble gases during normal operation is given in Table 11.3-1. It should be noted that for the purpose of evaluating potential offsite radiological consequences as a result of normal operation, and consistent with 10 CFR 50 Appendix I philosophy and actual BWR operating experience, the design basis values have

been reduced by a factor of 4 to provide "expected" annual average release rates. As discussed in Subsection 11.3.2.1.11, iodine and particulate releases from the RECHAR System are essentially zero.

11.3.4 References

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2. C. W. Miller, "Experimental and Operational Confirmation of Off-Gas System Design Parameters," NED0-10751, (Proprietary), January 1973.

3. D. P. Siegwarth, "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, N.Y., 1970.

5. D. Underhill, et al., "Design of Fission Gas Holdup Systems," Proceedings of the 11th AEC Air Cleaning Conference, p. 217, 1970.

6. L. B. Nesbitt, "Design Basis for New Gas Systems," NEDE11146, (Proprietary), July 1971.

7. "N66 SJAE Off-Gas Treatment System - Amendment No. 1," (Supplements Licensing Topical Report), NEDE-21056-1-P, (Proprietary), August 1978.

8. "Safety Evaluation Report Related to the Preliminary Design of the Gessar-238 Nuclear Island Standard Design," NUREG-0124 (Supplement 1 to NUREG-75/110), U.S. Nuclear Regulatory Commission, pp. 3-1 and 3-2, September 1976.

9. "Calculation of Releases of Materials in Gaseous and Liquid Effluents from Boiling Water Reactors (BWR-Gale Code)," U.S. Nuclear Regulatory Commission, NUREG-0016, January 1979.

10. E. W. Bradley, V. D. Nguyen, "Radiation Exposure From Airborne Effluents - The REFAE Code," NEDO-25257, July 1980.

11. E. W. Bradley, "Gamma And Beta Dose To Man From Noble Gas Release To The Atmosphere GEMAN Program," NEDO-25132A, April 1980.

12. Drawing 105D5099A, GE Process Data Off Gas System - Low Temp. (Proprietary).

TABLE 11.3-1
DESIGN-BASIS AIR EJECTION OFF-GAS RELEASE RATES

(30 scfm Inleakage)
(Release Rates are based on the 1971 Mixture at 30 min)

			T=30 MINUTES	NORMAL DISCHARGE FROM CHARCOAL ADSORBERS	
ISOTOPE	HALF-LIFE	T=0 µCi/sec	µCi/sec	µCi/sec	Ci/yr*
Kr-83m	1.86 hr	3.4E+03	2.9E+03		
Kr-85m	4.4 hr	6.1E+03	5.6E+03	4.3	1.2E+02
Kr-85**	10.74 yr	10-20	10-20	10-20	280-560
Kr-87	76.0 min	2.0E+04	1.5E+04	--	
Kr-88	2.79 hr	2.0E+04	1.8E+04	2.1E-0l	6.0
Kr-89	3.18 min	1.3E+05	1.8E+02	--	
Kr-90	32.3 sec	2.8E+05	--	--	
Kr-91	8.6 sec	3.3E+05	--	--	
Kr-92	1.84 sec	3.3E+05	--	--	
Kr-93	1.29 sec	9.3E+04	--	--	
Kr-94	1.0 sec	2.3E+04	--	--	
Kr-95	0.5 sec	2.1E+03	--	--	
Kr-97	1.0 sec	1.4E+0l	--	--	
Xe-131m	11.96 day	1.5E+0l	1.5E+01	1.3	3.7E+01
Xe-133m	2.26 day	2.9E+02	2.8E+02	--	
Xe-133	5.27 day	8.2E+03	8.2E+03	3.3E+01	9.4E+02
Xe-135m	15.7 min	2.6E+04	6.9E+03	--	
Xe-135	9.16 hr	2.2E+04	2.2E+04	--	
Xe-137	3.82 min	1.5E+05	6.7E+02	--	
Xe-138	14.2 min	8.9E+04	2.1E+04	--	
Xe-139	40.0 sec	2.8E+05	--	--	
Xe-140	13.6 sec	3.0E+05	--	--	
Xe-141	1.72 sec	2.4E+05	--	--	
Xe-142	1.22 sec	7.3E+04	--	--	
Xe-143	0.96 sec	1.2E+04	--	--	
Xe-144	9.0 sec	5.6E+02	--	--	
TOTALS		2.5E+06	1.0E+05	49-59	1383-1663

* This is based on curies present at time of release. No decay in environment is included.

** Estimated from experimental observations.

TABLE 11.3-1a
ESTIMATED AIR EJECTOR OFF-GAS RATES (1)

(Charcoal Temperature = 40 deg F)

ISOTOPE	HALF-LIFE	T=0 μCi/sec	T=30 MINUTES μCi/sec	NORMAL DISCHARGE FROM CHARCOAL ADSORBERS(2) μCi/sec	Ci/yr(3)
Kr-83m	1.86 hr	3.4E+03	2.9E+03	-	-
Kr-85m	4.4 hr	6.1E+03	5.6E+03	4.1E+01	1.2E+03
Kr-85 (4)	10.74 yr	2.0E+01	2.0E+01	2.0E+01	5.7E+02
Kr-87	76.0 min	2.0E+04	1.5E+04	-	-
Kr-88	2.79 hr	2.0E+04	1.8E+04	7.4E+00	2.1E+02
Kr-89	3.18 min	1.3E+05	1.8E+02	-	-
Kr-90	32.3 sec	2.8E+05	-	-	-
Kr-91	8.6 sec	3.3E+05	-	-	-
Kr-92	1.84 sec	3.3E+05	-	-	-
Kr-93	1.29 sec	9.3E+04	-	-	-
Kr-94	1.0 sec	2.3E+04	-	-	-
Kr-95	0.5 sec	2.1E+03	-	-	-
Kr-97	1.0 sec	1.4E+01	-	-	-
Xe-131m	11.96 day	1.5E+01	1.5E+01	2.9E+00	8.2E+01
Xe-133m	2.26 day	2.9E+02	2.8E+02	4.8E-02	1.4E+00
Xe-133	5.27 day	8.2E+03	8.2E+03	2.0E+02	5.6E+03
Xe-135m	15.7 min	2.6E+04	6.9E+03	-	-
Xe-135	9.16 hr	2.2E+04	2.2E+04	-	-
Xe-137	3.82 min	1.5E+05	6.7E+02	-	-
Xe-138	14.2 min	8.9E+04	2.1E+04	-	-
Xe-139	40.0 sec	2.8E+05	-	-	-
Xe-140	13.6 sec	3.0E+05	-	-	-
Xe-141	1.72 sec	2.4E+05	-	-	-
Xe-142	1.22 sec	7.3E+04	-	-	-
Xe-143	0.96 sec	1.2E+04	-	-	-
Xe-144	9.0 sec	5.6E+02	-	-	-
TOTALS		2.4E+06	1.0E+05	2.7E+02	7.7E+03

NOTES:

1. Release rates are based on the 1971 mixture.

2. 30 scfm in-leakage.

3. Plant Capacity Factor = 0.9.

4. 10 to 20 μCi/sec estimated from experimental observations.

5. Charcoal Delay Time for Krypton is 31.8 hours. Charcoal Delay Time for Xenon is 28.4 days.

TABLE 11.3-2
OFF-GAS SYSTEM MAJOR EQUIPMENT ITEMS

Recombiner (2 required, contains preheater, catalyst, and Condenser sections) Carbon steel shell:

Shell length - approximately 23 ft
Shell o.d. - approximately 50 in.
Total unit height - approximately 116 in.
Design pressure: 350 psig
Design temperature: 450° F
Code of Construction: ASME Section VIII, Division 1

Preheater Section

Shell and tube heat exchanger
Tubes - stainless steel, rolled into stainless steel tube sheet
Tube-side design pressure: 350 psig
Design temperature: 450° F

Catalyst Section

Catalyst support: stainless steel
Design temperature: 900° F
Catalyst: Precious metal on metal base

Off-Gas Condenser Section

Shell and tube heat exchanger
Tubes - stainless steel, rolled into stainless steel tube sheet
Tube-side design pressure: 350 psig
Design temperature: 900° F

Cooler Condenser (2 required)

Shell and tube heat exchanger, carbon steel vessel
Shell length - 10 ft maximum
Shell o.d. - 2 to 6 in. maximum
Shell-side design pressure - 350 psig
Shell-side design temperature - 32° F to 250° F
Tubes - stainless steel, welded into stainless tube sheet
Tube-side design pressure - 100 psig
Tube-side design temperature - 32° F to 150° F
Code of Construction: TEMA Class C

Desiccant Dryer Skid (2 required)

Approximate skid size: length 12.5 ft, width 6 ft, height 9ft. Each skid consists of one Desiccant Vessel and associated valves, piping, and instruments. The Desiccant Vessel is of carbon steel approximately 3.5 ft o.d. by 4 ft high (straight side), and contains approximately 30 ft. of molecular sieve desiccant.
Vessel design pressure - 350 psig
Piping and valving design pressure - 1050 psig
Design temperature - 32° F to 600° F
Code of Construction: ASME Section VIII and ANSI B31.1.0

TABLE 11.3-2
OFF-GAS SYSTEM MAJOR EQUIPMENT ITEMS (Continued)

Gas Cooler (1 required)
Stainless steel and carbon steel exposed-tube heat exchanger Maximum dimensions: length 12 ft, width 3 ft, height 6 ft Design pressure - 1050 psig
Design temperature - -50° F to 750° F
Code of Construction - ASME Section VIII, Division 1

Gas Cooler Heaters (Note: Not used, see section 9.4.8.2.)
2 - 25 kW in parallel
480-V - 3 phase - 60 Hz

Charcoal Adsorbers (1 each required)
Carbon steel vessels, filled with 24.6 tons of activated charcoal
Height - approximately 24 ft
Outside diameter - approximately 8 ft
Design pressure - 350 psig
Design temperature - -50° F to 650° F
Code of Construction - ASME Section VIII, Division 1

Seismically designed to a static seismic coefficient of 0.2g horizontal and 0 g vertical. The support elements are designed to AISC Manual of Steel Construction, 7th Edition with the provisions of R.G. 1.143 for allowable stress.

Filter (1 required)
Carbon steel vessel, with removable HEPA filter
Height (includes legs) - approximately 6 ft
Outside diameter - approximately 2 ft
Flow - 250 scfm at 1.0 in H_2O gauge
Design pressure - 350 psig
Design temperature - -50° F to 250° F
Code of Construction - ASME Section VIII, Division 1

Regenerator (1 required)
The regenerator skid approximate dimensions are:
Length - 15 ft
Width - 7 ft 6 in.
Height - 11 ft
The regenerator skid consists of the following equipment pieces, plus associated piping, valves, and instrumentation:

Regenerator Chiller (2 required)
Carbon steel shell, stainless steel tubes welded into stainless steel tube sheet
Shell-side design pressure - 100 psig
Shell-side design temperature - 32° F to 500° F
Tube-side design pressure - 100 psig
Tube-side design temperature - 32° F to 500° F
Code of Construction: TEMA Class C

TABLE 11.3-2
OFF-GAS SYSTEM MAJOR EQUIPMENT ITEMS (Continued)

Regeneration Blower (2 required)
Mechanical blower - 10 hp motor, 480-V - 3 phase - 60 Hz Design pressure - 50 psig
Design temperature - 32° F to 150° F
Code of Construction: Seller's standard
Material: Seller's standard

Regenerator Heater (2 required)
Enclosed element, Calrod-type
Each heater is 34.2 kW - 480-V - 3 phase - 60 Hz
Design pressure - 50 psig
Design temperature - 32° F to 1000° F
Code of Construction: Seller's standard

Glycol Cooler (1 required)
Skid-mounted assembly
Approximate size as follows:
Length - 27 ft
Width - 8 ft
Height - 12 ft

The Glycol Cooler consists of the following equipment pieces, plus associated piping, valves, and instrumentation:

Glycol Tank (1 required)
Carbon steel tank - approximately 1000 gallons
Height - approximately 7.5 ft
Inside diameter - approximately 5 ft
Design pressure - Glycol filled and open to atmosphere
Design temperature - 32°F
Code of Construction - API 650

Refrigeration Machines (3 required)
Freon cycle refrigeration machine
Duty - approximately 9 x 10^4 Btu/hr (each)
Type - Dunham-Bush Model PCB-020DQ or equivalent
Motor - Open, drip-proof, 20 hp - 480-V - 3 phase - 60 Hz
Codes of Construction: Refrigerant piping - ANSI B31.5
Glycol and cooling water piping - ANSI B31.1.0
Heat Exchanger, refrigerant side - ASME Section VIII, Division 1
Condenser, cooling water side - Seller's standard
Compressor and motor - Seller's standard

Glycol Pump and Motor Drive (3 required)
Duty 65 gpm at a nominal 110 ft. developed head
Design temperature - 32°F
Motor - open, drip-proof, 5 hp, 480-V - 3 phase - 60 Hz
Code of Construction - Seller's standard
Material: cast iron

TABLE 11.3-2
OFF-GAS SYSTEM MAJOR EQUIPMENT ITEMS (Continued)

Local Panel (1 required)
Contains control functions for the Glycol Cooler

Local Panels (1 each required)
Contains control functions for, respectively, the upstream and downstream portions of the main process stream.

Local Panel (1 required)
Contains control functions for regenerator skid

TABLE 11.3-3 has been deleted.

TABLE 11.3-4
OFF-GAS SYSTEM ALARMED PROCESS PARAMETERS

PARAMETERS	CONTROL ROOM INDICATED	CONTROL ROOM RECORDED
Air ejector discharge pressure - high	X	
Preheater discharge and Recombiner Inlet temperature - low		X
Recombiner catalyst temperature - high/low		X
Off-gas condenser water level (dual) high/low (Local)		
Off-gas condenser gas discharge temperature - high (LOCAL)		
H_2 analysis (desiccant drier discharge) dual - high		X
Cooler - condenser discharge radiation - high	1	1
Recombiner train pressure drop-high	X	
Cooler - condenser discharge temperature - high/low		X
Glycol solution temperature - high/low		X
Glycol solution level - low (Local)		
Desiccant drier discharge humidity - high		X
Gas cooler dP - high	X	
Charcoal Adsorber temperature - high		X
Carbon vault temperature - high/low		X
Carbon train flow - high/low		X
Carbon train dP - high	X	
After filter dP - high	X	
Off-gas (carbon bed discharge) radiation - high	1	1
Dilution Steam flow - high/low		X

TABLE 11.3-4
OFF-GAS SYSTEM ALARMED PROCESS PARAMETERS (Continued)

PARAMETERS	CONTROL ROOM INDICATED	CONTROL ROOM RECORDED
Regenerator Heater Temperature - high (LOCAL)		
Regenerator Heater Gas Exit Temp - high	X	
Regenerator Chiller Gas Exit Temp - high	X	
Gas Dryer Desiccant Bed Exhaust Temperature - high/low		X
Recombiner Preheater Drain Pot Levels - high/low		

(1) Alarm, current data and historical data are provided by the Radiation Monitoring system at the AR/PR control terminal.

TABLE 11.3-5
EQUIPMENT MALFUNCTION ANALYSIS

EQUIPMENT ITEM	MALFUNCTION	CONSEQUENCES	DESIGN PRECAUTIONS
Steam Jet Air Ejectors	Low flow of motive high-pressure steam	When the hydrogen and oxygen concentrations exceed 4 and 5 volume %, respectively, in air the process gas becomes flammable.	Alarms provided on steam for low steam flow. System automatic isolation on low steam flow.
		Inadequate steam flow will cause overheating and deterioration of the catalyst.	Steam flow to be held at constant maximum flow regardless of plant power level.
	Wear of steam supply nozzle of ejector	Increased steam flow to recombiner. This could reduce degree of recombination at low power levels and high discharge temperature from recombiner condenser could result due to inadequate condenser capacity.	Low temperature alarms on preheater exit (catalyst inlet). Downstream H_2 analyzer.
Recombiner Preheater	Steam leak	Would further dilute process off-gas. Steam consumption would increase.	Spare recombiner.
	Low-pressure steam supply	Recombiner performance would fall off at low power level, and hydrogen content of recombiner gas discharge would increase, eventually to a combustible mixture.	Low-temperature alarms on preheater exit (catalyst inlet). Downsteam H_2 analyzer.
Recombiner Catalyst	Catalyst gradually deactivates	Temperature profile changes through catalyst. Eventually excess H_2 would be detected by H_2 analyzer or by gas flowmeter. Eventually the gas could become combustible.	Temperature probes in catalyst bed and H_2 analyzer provided. Spare recombiner.
	Catalyst gets wet at start	H_2 conversion falls. Eventually the gas could become combustible.	Condensate drains, temperature probes in recombiner. Air bleed system at startup. Spare recombiner. Hydrogen analyzer.

TABLE 11.3-5 (Cont'd)

EQUIPMENT MALFUNCTION ANALYSIS

EQUIPMENT ITEM	MALFUNCTION	CONSEQUENCES	DESIGN PRECAUTIONS
Recombiner Condenser	Cooling water leak	The coolant (reactor condensate) would leak to the process gas (shell) side. This would be detected if drain well liquid level increases. Moderate leakage would be of no concern from a process standpoint. (The process condensate drains to the hotwell.)	Drain well high level alarm. Redundant recombiner condenser.
	Liquid level instruments fail	If drain valve fails to open, water will build up in the condenser and pressure drop will increase. The high DP, if not detected by instrumentation, could cause pressure buildup in the main condenser and eventually initiate a reactor scram. If a drain valve fails to close, gas will recycle to the main condenser, increase the load on the SJAE, and cause a slight back pressure on the main condenser.	High- and low-level alarms on drain well level. Redundant recombiner.
Cooler-Condenser	Icing-up of the tubes	Shell side of cooler could plug up with ice, gradually building up pressure drop. If this happens, the spare unit could be activated. Complete blockage of both units would increase DP and lead to a reactor scram.	Design glycol- H_2O solution temperature of 35° F to 40° F. Spare unit provided. Temperature indication and low alarms on glycol temperature and process gas temperature.
	Corrosion of tubes	Glycol-water solution would leak into process (shell) side and be discharged to clean radwaste. If not detected at radwaste, the glycol solution would discharge to the reactor condensate system.	Stainless-steel tubes specified. Low level alarm on glycol tank level. Spare cooler provided.
Moisture Separator in Cooler Condenser	Corrosion of wire mesh element	Increased moisture would be retained in process gas routed to gas dryers. Over a long period, the desiccant dryer cycle period would deteriorate as a result of moisture pickup.	Stainless steel mesh specified. Spare cooler condenser provided. Moisture detector provided downstream of gas dryer.

TABLE 11.3-5 (Cont'd)

EQUIPMENT MALFUNCTION ANALYSIS

EQUIPMENT ITEM	MALFUNCTION	CONSEQUENCES	DESIGN PRECAUTIONS
Glycol Refrigeration Machines	Mechanical failure	If both spare units fail to operate, the glycol solution temperature will rise and the dehumidification system performance will deteriorate. This will require rapid regeneration cycles for the desiccant beds and may raise the gas dew point as it is discharged from the dryer.	Two spare refrigerators during normal operation are provided. Glycol solution temperature alarms provided. Gas moisture detectors provided downstream of gas dryers.
Desiccant Dryer	Moisture breakthrough	Moisture would freeze out in gas cooler and would result in increased system pressure drop. 0° F dew point gas would reach charcoal bed.	Dryer cycled on timer. Gas humidity analyzer and alarm supplied. Alternate desiccant dryer is normally maintained in a regenerated condition. Gas cooler can be derimed while on stream, with the charcoal adsorbers remaining near the selected operating temperature.
Desiccant Regeneration Equipment	Mechanical failure	Inability to regenerate desiccant.	Redundant regenerator equipment is supplied.
Gas Cooler	Moisture freezeout	System pressure drop will increase ultimately increasing main condenser pressure.	Pressure drop across gas cooler is alarmed and cooler can be derimed while on stream.
Charcoal Adsorbers	Charcoal gets wet (ice)	Charcoal performance will deteriorate gradually as moisture deposits. Holdup times for krypton and xenon would decrease, and plant emissions would increase. Provisions made for drying charcoal as required during annual outage.	Highly instrumented, mechanically simple melt, drain, and dehumidification system.
Vault Refrigeration Units	Mechanical failure	If temperature exceeds Design Limit, increased emission may occur.	Spare refrigeration unit provided. Vault temperature alarms provided.

TABLE 11.3-5 (Cont'd)

EQUIPMENT MALFUNCTION ANALYSIS

EQUIPMENT ITEM	MALFUNCTION	CONSEQUENCES	DESIGN PRECAUTIONS
After Filter	Hole in filter media	Probably of no real consequence. The charcoal itself will retain virtually all solid daughters. Particulates released would be the negligible amount formed in the pipe run after the exit of the last charcoal adsorber.	DP instrumentation provided.
System	Internal detonation	Release of radioactivity if pressure boundary fails.	Main process equipment and piping are designed to contain a detonation.
		Internal damage to the recombiner and its heat exchanger.	Redundant recombiner, damaged internals can be repaired.
		Damage to instrumentation sensors.	Redundant, damaged sensors can be replaced.
		Filter media blown out.	The inventory of this filter is trivial. The filter can be replaced at first outage.
System	Earthquake damage	Release of radioactivity	Dose consequences are within the Design Objectives of Regulatory Guides 1.26 and 1.29.

TABLE 11.3-6
FREQUENCY AND QUANTITY OF STEAM DISCHARGED TO SUPPRESSION POOL

	EVENT	FREQUENCY CATEGORY	QUANTITY OF STEAM LB/EVENT
1.	RCIC Test (Monthly)	Moderate	25,200
2.	Inadvertent RCIC Injection	Moderate	4,200
3.	SRV Test (each valve)	Moderate	3,900
4.	SRV Flow Capacity Test (each valve)	Infrequent	15,300
5.	Total SRV Leakage (16 valve maximum)	Continuous	320/hr
6.	Trip of Both Recirculation Pump Motors	Moderate	30,000
7.	Turbine Trip	Moderate	30,000
8.	Generator Load Rejection	Moderate	30,000
9.	Pressure Regulator Failure, Open	Moderate	874,000(1)
10.	Recirculation Controller Failure	Moderate	0,000
11.	Loss of All Feedwater Flow	Moderate	30,000
12.	Inadvertent MSIV Closure	Moderate	874,000(1)
13.	Loss of Condenser Vacuum	Moderate	874,000(1)
14.	Feedwater Control Failure, Maximum Demand	Moderate	30,000
15.	Loss of Auxiliary Transformer	Moderate	457,000(1)
16.	Loss of All Grid Connections	Moderate	457,000(1)
17.	Turbine Trip w/o Bypass	Infrequent	874,000(1)
18.	Generator Load Rejection w/o Bypass	Infrequent	874,000(1)
19.	Stuck Open SRV	Moderate	457,000

Notes:

Events 1 and 2 based on steam flow rate during test mode per RCIC System Process Diagram, 762E421A, for 60 and 10 minutes, respectively.

Event 3 assumes tested SRV opened 30 seconds maximum at 300-500 psig vessel pressure.

Event 4 assumes tested SRV opened 30-60 seconds at 1000 psig vessel pressure.

Event 5 based on maximum average SRV leakage rate of 20 lb/hr/valve.

Event 6 through 18 based on event description from Clinton USAR Chapter 15.

Event 19 based on vessel depressurized to 100 psia with two additional SRV's opened 10 minutes following scram.

TABLE 11.3-6 (Cont'd)
FREQUENCY AND QUANTITY OF STEAM DISCHARGED TO SUPPRESSION POOL

(1) Isolation event based on initiation of manual depressurization with SRV's at 30 minutes at an average cooldown rate of 50 to 100°F/hr until the vessel reaches 100 psia. Except for events 15 and 16 (for which the motor driven feedwater pump is unavailable) makeup is conservatively assumed to be principally from the feedwater system. (Ref. Calculation M/NSED IP-M-0313).

These values are based on the SRV Mass Blowdown rate at 2894 MWt per USAR Table 15.2.4-2.

TABLE 11.3-7
OFFGAS EQUIPMENT DESIGN REQUIREMENTS

EQUIPMENT	CODES			
	DESIGN AND FABRICATION	MATERIALS(2)	WELDER QUALIFICATION AND PROCEDURE	INSPECTION AND TESTING
Pressure Vessels	ASME Code Section VIII Div 1	ASME Code Section II	ASME Code Section IX	ASME Code Section VIII Div 1
Atmospheric or 0-15 psig Tanks	ASME Code(3) Section III Class 3, API 620;650,AWWA D-100	ASME Code Section II	ASME Code Section IX	ASME Code(3) Section 111 CLASS 3, Apl 620;650,AWWA D-100
Heat Exchangers	ASME Code Section VIII Div 1;and TEMA	ASME Code Section II	ASME Code Section IX	ASME Code Section VIII Div 1
Piping and Valves	ANSI B 31.1	ASTM or ASME Code Section II	ASME Code Section IX	ANSI B 31.1
Pumps	Manufacturers(1) Standards	ASME Code Section II or Manufacturer's Standard	ASME Code Section IX (as required)	ASME Code(3) Section III Class 3; and Hydraulic Institute

NOTES:

(1) Manufacturer's standard for the intended service. Hydrotesting should be 1.5 times the design pressure.

(2) Material Manufacturer's certified test reports should be obtained whenever possible.

(3) ASME Code stamp and material traceability not required.

TABLE 11.3-8
DATA AND THEIR REFERENCES FOR
CALCULATING EXPECTED GASEOUS RELEASE RATES

	ITEM DESCRIPTION	INPUT	REFERENCE
1.	GENERAL		
	Maximum core thermal power level (MWt)	3473	NEDC-32989
	Tritium released in liquid effluent (Ci/yr/reactor)	12	USAR Table 11.2-1
	Tritium released in gaseous effluent (Ci/yr/reactor)	44	USAR Table 11.3-9
	Annual average dilution flow rate for liquid waste discharge (gpm)	610,000	Calculation 1SX01
2.	NUCLEAR STEAM SUPPLY SYSTEM		
	Total steam flow rate (lb/hr)	15.153×10^6	NEDC-32989
	Mass of reactor coolant in reactor vessel at full power (lb)	515,558	General Electric Company
	Mass of steam in reactor vessel at full power (lb)	10,740	General Electric Company
3.	REACTOR WATER CLEANUP SYSTEM		
	Average flow rate (lb/hr)	124,000	General Electric Company
	Number of demineralizers	2	General Electric Company
	Type of demineralizer	Powered resin	General Electric Company
	Replacement frequency (batch/day):		
	Normal	1/7 days	General Electric Company
	Startup	1	General Electric Company
	(1 batch includes 2 filter/demineralizer backwashes)		
	Regenerate volume and activity, if applicable (gal/event)	1075 gal/event 1200 curies	General Electric Company General Electric Company

TABLE 11.3-8
DATA AND THEIR REFERENCES FOR
CALCULATING EXPECTED GASEOUS RELEASE RATES (Continued)

	ITEM DESCRIPTION	INPUT	REFERENCE
4.	MAIN CONDENSER AND TURBINE GLAND SEAL AIR REMOVAL SYSTEMS		
	Holdup time for off-gases from the main condenser air ejector prior to processing by the off gas treatment system (hr)	0.033	General Electric Company
	Treatment system for off-gases from condenser air ejector	Recombination of H_2 and O_2 Condensing to Remove H_2O Drying Charcoal Adsorption Filtering	General Electric Company
	Off-gases from the mechanical vacuum pump	Not processed by off-gas system	General Electric Company
	Air inleakage per condenser shell (cfm)	30 scfm	General Electric Company
	Number of condenser shells	1	General Electric Company
	Mass of charcoal in the charcoal delay system (tons)	24.6	General Electric Company
	Operating temperature of the delay system	0°F	General Electric Company
	Dew point temperature of the delay system	-90°F	General Electric Company
	Dynamic adsorption coefficient for xenon	2032 cm^3/gm @ 70° F and 1 atmosphere	General Electric Company
	Dynamic adsorption coefficient for krypton	92.7 cm^3/gm @ 70° F and 1 atmosphere	General Electric Company
	Cryogenic distillation system	Not applicable	General Electric Company
	Steam flow to turbine gland seal (lb/hr)	17,000 to 30,000	GE Drawing 842E420
	Source of steam to turbine gland seal	Evaporating reactor condensate taken from downstream of the condensate demineralizer	General Electric Company
5.	VENTILATION AND EXHAUST SYSTEMS		

TABLE 11.3-8
DATA AND THEIR REFERENCES FOR
CALCULATING EXPECTED GASEOUS RELEASE RATES (Continued)

ITEM DESCRIPTION		INPUT	REFERENCE
a.	Provisions incorporated to reduce radioactivity releases through ventilation exhaust systems:		
1.	Containment building	Exhaust air may be filtered as required through Drywell Purge Filter Units	Design Criteria DC-VR-01-CP
2.	Drywell purge	3 50% of full capacity filter units, each with 1 medium efficiency filter, 1 upstream HEPA filter, 1 charcoal adsorber, and 1 downstream HEPA filter	DC-VQ-01-CP
3.	Auxiliary building	Not applicable	DC-VA-01-CP
4.	Turbine building	Not applicable	DC-VT-01-CP
5.	Radwaste building	2 50% of full capacity filter units, each with 1 medium efficiency filter and 1 HEPA filter except for Store Room - no filter	DC-VW-01-CP
6.	Fuel building	Not applicable	DC-VF-01-CP
7.	Control Building	2 50% of full capacity filter units, each with 1 medium efficiency filter and 1 HEPA filter; Laboratory only, remainder has no filter	DC-VL-01-CP

TABLE 11.3-8
DATA AND THEIR REFERENCES FOR
CALCULATING EXPECTED GASEOUS RELEASE RATES (Continued)

		ITEM DESCRIPTION	INPUT		REFERENCE
b.		Decontamination factors for iodine and particulates:	HEPA	CHARCOAL	
	1.	Containment building and drywell purge	100	10	NUREG-0016, Rev. 1
	2.	Auxiliary building	0.0	0.0	NUREG-0016, Rev. 1
	3.	Turbine building	0.0	0.0	NUREG-0016, Rev. 1
	4.	Radwaste building	100	0.0	NUREG-0016, Rev. 1
	5.	Fuel building	0.0	0.0	NUREG-0016, Rev. 1
c.		Release rates for radio-iodines, noble gases, and radioactive particulates	HVAC vent - 238,225 scfm and 3,033 fpm		Drawing M05-1111 Sheets 1-3, -5
d.		Description of release points, etc.	Common station HVAC vent		Drawing M05-1111 Sheets 1-3, -5
e.		Continuous containment purge rate cfm (maximum)	30,000 scfm continuously		Drawing M05-1111 Sheets 1-3, -5
6.		SOLID RADWASTE PROCESSING SYSTEM (see Table 11.4-2)			
7.		FUEL POOL FILTER DEMINERALIZER SYSTEM	INPUT		REFERENCE
a.		Volume of fuel pool and canal	138,000 ft^3		Design Criteria DC-FC-01-CP
b.		Source of pool makeup	Cycled condensate		DC-FC-01-CP
c.		Number of filter/demineralizers	4		DC-FC-01-CP
d.		Type of filter/demineralizers	Powdered resin		DC-FC-01-CP
e.		Size of filter/demineralizers	Element area of 555 ft^2		K-2845

TABLE 11.3-8
DATA AND THEIR REFERENCES FOR
CALCULATING EXPECTED GASEOUS RELEASE RATES (Continued)

	ITEM DESCRIPTION	INPUT	REFERENCE
f.	Replacement frequency, batch/day:		
	Normal	1 batch/30 days	Personal communication from vendor
	Refueling	1 batch/8 hours	
g.	Total Water to waste volume per replacement	1950 gal	K-2845
h.	Activity of regenerant backwash	129 Ci	USAR Table 12.2-8
i.	Activity of fuel pool water:		
	during normal operation	8.0×10^{-4} μCi/cm^3	USAR Table 12.2-6
	during refueling	10×10^{-2} μCi/cm^3	S&L Calculation No. FC-1

TABLE 11.3-9
EXPECTED GASEOUS RELEASE RATE

(Curies Per Year)

	NUCLIDE	COOLANT CONCENTRATION (µCi/g)	CONTAINMENT BUILDING (a.)	TURBINE BUILDIN G	AUXILIARY BUILDING	RADWASTE BUILDING	GLAND SEAL	AIR EJECTOR	MECHANICAL VACUUM PUMP	TOTAL
1.	Halogens									
	I-131		1.1-003	1.2-001	2.2-002	1.1-002	0.0	0.0	8.6-002	2.4-001
	I-133		1.5-002	1.6+000	3.0-001	1.5-001	0.0	0.0	9.4-001	3.0+000
2.a	NOBLE GASES									
	Charcoal temperature = 0°F									
	Ar-41	0.000	1.5+001	0.0	0.0	0.0	0.0	2.8+001	0.0	4.3+001
	Kr-83M	9.100-003	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
	Kr-85M	1.600-003	1.0+000	2.5+001	3.0+000	0.0	0.0	0.0	0.0	2.9+001
	Kr-85	5.00-006	0.0	0.0	0.0	0.0	0.0	2.1+002	0.0	2.1+002
	Kr-87	5.500-003	0.0	6.1+001	2.0+000	0.0	0.0	0.0	0.0	6.3+001
	Kr-88	5.500-003	1.0+000	9.1+001	3.0+000	0.0	0.0	0.0	0.0	9.5+001
	Kr-89	3.400-002	0.0	5.8+002	2.0+000	2.9+001	0.0	0.0	0.0	6.1+002
	Xe-131M	3.900-006	0.0	0.0	0.0	0.0	0.0	2.0+000	0.0	2.5+000
	Xe-133M	7.500-005	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
	Xe-133	2.100-003	2.7+001	1.5+002	8.3+001	2.2+002	0.0	8.0+000	1.3+003	1.8+003
	Xe-135M	7.00-003	1.5+001	4.0+002	4.5+001	5.3+002	0.0	0.0	0.0	9.9+002
	Xe-135	6.000-003	3.3+001	3.3+002	9.4+001	2.8+002	0.0	0.0	5.0+002	1.2+003
	Xe-137	3.900-002	4.5+001	1.0+003	1.3+002	8.3+001	0.0	0.0	0.0	1.3+003
	Xe-138	2.300-002	2.0+000	1.0+003	6.0+000	2.0+000	0.0	0.0	0.0	1.0+003
	Total Noble Gases									7.3+003

TABLE 11.3-9
EXPECTED GASEOUS RELEASE RATE (Continued)

	NUCLIDE	COOLANT CONCENTRATION (µCi/g)	CONTAINMENT BUILDING (a.)	TURBINE BUILDIN G	AUXILIARY BUILDING	RADWASTE BUILDING	GLAND SEAL	AIR EJECTOR	MECHANICAL VACUUM PUMP	TOTAL
2.b	Charcoal temperature = 20°F									
	Ar-41	0.000	1.5+001	0.0	0.0	0.0	0.0	3.9+001	0.0	5.4+001
	Kr-83M	9.100-003	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
	Kr-85M	1.600-003	1.0+000	2.5+001	3.0+000	0.0	0.0	9.5+001	0.0	1.2+002
	Kr-85	5.00-006	0.0	0.0	0.0	0.0	0.0	2.5+002	0.0	2.5+002
	Kr-87	5.500-003	0.0	6.1+001	2.0+000	0.0	0.0	0.0	0.0	6.3+001
	Kr-88	5.500-003	1.0+000	9.1+001	3.0+000	0.0	0.0	6.6+000	0.0	1.0+002
	Kr-89	3.400-002	0.0	5.8+002	2.0+000	2.9+001	0.0	0.0	0.0	6.1+002
	Xe-131M	3.900-006	0.0	0.0	0.0	0.0	0.0	1.9+001	0.0	1.9+001
	Xe-133M	7.500-005	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
	Xe-133	2.100-003	2.7+001	1.5+002	8.3+001	2.2+002	0.0	5.8+002	1.3+003	2.4+003
	Xe-135M	7.00-003	1.5+001	4.0+002	4.5+001	5.3+002	0.0	0.0	0.0	9.9+002
	Xe-135	6.000-003	3.3+001	3.3+002	9.4+001	2.8+002	0.0	0.0	5.0+002	1.2+003
	Xe-137	3.900-002	4.5+001	1.0+003	1.3+002	8.3+001	0.0	0.0	0.0	1.3+003
	Xe-138	2.300-002	2.0+000	1.0+003	6.0+000	2.0+000	0.0	0.0	0.0	1.0+003
	Total Noble Gases									8.0+003
2.c	Charcoal temperature = 40°F									
	Ar-41	0.000	1.5+001	0.0	0.0	0.0	0.0	7.9+001	0.0	9.4+001
	Kr-83M	9.100-003	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
	Kr-85M	1.600-003	1.0+000	2.5+001	3.0+000	0.0	0.0	5.1+002	0.0	5.4+002
	Kr-85	5.00-006	0.0	0.0	0.0	0.0	0.0	2.5+002	0.0	2.5+002
	Kr-87	5.500-003	0.0	6.1+001	2.0+000	0.0	0.0	0.0	0.0	6.3+001
	Kr-88	5.500-003	1.0+000	9.1+001	3.0+000	0.0	0.0	9.3+001	0.0	1.9+002
	Kr-89	3.400-002	0.0	5.8+002	2.0+000	2.9+001	0.0	0.0	0.0	6.1+002
	Xe-131M	3.900-006	0.0	0.0	0.0	0.0	0.0	3.6+001	0.0	3.6+001
	Xe-133M	7.500-005	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
	Xe-133	2.100-003	2.7+001	1.5+002	8.3+001	2.2+002	0.0	2.5+003	1.3+003	4.3+003
	Xe-135M	7.00-003	1.5+001	4.0+002	4.5+001	5.3+002	0.0	0.0	0.0	9.9+002
	Xe-135	6.000-003	3.3+001	3.3+002	9.4+001	2.8+002	0.0	0.0	5.0+002	1.2+003
	Xe-137	3.900-002	4.5+001	1.0+003	1.3+002	8.3+001	0.0	0.0	0.0	1.3+003
	Xe-138	2.300-002	2.0+000	1.0+003	6.0+000	2.0+000	0.0	0.0	0.0	1.0+003
	Total Noble Gases									1.0+004

TABLE 11.3-9
EXPECTED GASEOUS RELEASE RATE (Continued)

NUCLIDE	COOLANT CONCENTRATION (µCi/g)	CONTAINMENT BUILDING (a.)	TURBINE BUILDIN G	AUXILIARY BUILDING	RADWASTE BUILDING	GLAND SEAL	AIR EJECTOR	MECHANICAL VACUUM PUMP	TOTAL
3. Particulates									
Cr-51		2.0-006	9.0-004	9.0-004	7.0-006	0.0	0.0	1.0-006	1.8-003
Mn-54		4.0-006	6.0-004	1.0-003	4.0-005	0.0	0.0	0.0	1.6.-003
Co-58		1.0-006	1.0-003	2.0-004	2.0-006	0.0	0.0	0.0	1.2-003
Fe-59		9.0-007	1.0-004	3.0-004	3.0-006	0.0	0.0	0.0	4.0-004
Co-60		1.0-005	1.0-003	4.0-003	7.0-005	0.0	0.0	5.6-007	5.1-003
Zn-65		1.0-005	6.0-003	4.0-003	3.0-006	0.0	0.0	3.4-007	1.0-002
Sr-89		3.0-007	6.0-003	2.0-005	0.0	0.0	0.0	0.0	6.0-003
Sr-90		3.0-008	2.0-005	7.0-006	0.0	0.0	0.0	0.0	2.7-005
Zr-95		3.0-006	4.0-005	7.0-004	8.0-006	0.0	0.0	0.0	7.5-004
Nb-95		1.0-005	6.0-006	9.0-003	4.0-008	0.0	0.0	0.0	9.0-003
Mo-99		6.0-005	2.0-003	6.0-002	3.0-008	0.0	0.0	0.0	6.2-002
Ru-103		2.0-006	5.0-005	4.0-003	1.0-008	0.0	0.0	0.0	4.1-003
Ag-110M		4.0-009	0.0	2.0-006	0.0	0.0	0.0	0.0	2.0-006
Sb-124		2.0-007	1.0-004	3.0-005	7.0-007	0.0	0.0	0.0	1.3-004
Cs-134		7.0-006	2.0-004	4.0-003	2.4-005	0.0	0.0	3.2-006	4.2-003
Cs-136		1.0-006	1.0-004	4.0-004	0.0	0.0	0.0	1.9-006	5.0-004
Cs-137		1.0-005	1.0-003	4.0-005	4.0-005	0.0	0.0	8.9-006	6.1-003
Ba-140		2.0-005	1.0-002	5.0-003	4.0-008	0.0	0.0	1.1-005	3.0-002
Ce-141		2.0-006	1.0-002	7.0-004	7.0-008	0.0	0.0	0.0	1.1-002
Total Particulates									1.5-001

H-3 Released from Turbine Bldg. ventilation system = 2.2+001

H-3 Released from Containment Bldg. ventilation system = 2.2+001

Total H-3 Released via gaseous pathway = 4.4+001

C-14 Released via main condenser offgas system = 9.5 CI/YR

0.0 Appearing in the table indicates release is less then 1.0 CI/YR for Noble Gases

a. Release includes releases from drywell purging approximately four times a year.

TABLE 11.3-10
ATMOSPHERIC DILUTION FACTORS USED IN DETERMINING
OFFSITE DOSES AT SEVERAL LOCATIONS OF INTEREST

LOCATION	X/Q (sec/m^3)	D/Q (1/m_2)
Nearest Site Boundary (3600 ft NE)	9.2-6	2.9-9
Nearest Residence (0.7 mi NW)	5.4-6	1.4-9
Nearest Garden (0.7 mi NW)	5.4-6	1.4-9
Nearest Meat Animal (Approximately 1 mi N)	4.5-6	1.9-9
Nearest Milk Cow (Approximately 5 mi ESE)	1.7-7	1.9-10
Nearest Milk Goat (5 mi ESE)	1.7-7	1.9-10

TABLE 11.3-11
EXPECTED INDIVIDUAL DOSES FROM GASEOUS EFFLUENTS*

LOCATION	PATHWAY	DOSE RATE (mrem/yr/unit)		
		GAMMA AIR	BETA AIR	
Nearest Site Boundary (3600 ft NE)	Plume	2.9	3.2	
		TOTAL BODY	SKIN	
Nearest Residence (0.7 mile NW)	Plume (whole body)	7.0-1	2.8	
	Ground Deposition	5.4-2	6.4-2	
		TOTAL BODY	KIDNEY	THYROID
	Inhalation	4.4-3	1.6-2	1.6 + 0
	Adult	5.4-3	2.3-2	2.1 + 0
	Teen	5.8-3	2.1-2	2.6 + 0
	Child	4.0-3	1.4-2	2.4 + 0
	Infant			
Nearest Garden (0.7 mile NW)	Leafy Vegetables			
	Adult	1.1-3	1.7-3	2.2-1
	Teen	7.4-4	1.5-3	1.8-1
	Child	0.9-4	1.9-3	2.8-1
	Produce			
	Adult	3.8-3	1.6-3	7.1-3
	Teen	4.2-3	2.5-3	1.1-2
	Child	5.4-3	4.0-3	2.1-2

* Based on the original licensed power level of 2894 MWth.

TABLE 11.3-11
EXPECTED INDIVIDUAL DOSES FROM GASEOUS EFFLUENTS (Continued)

		DOSE RATE (mrem/year)		
LOCATION	PATHWAY	TOTAL BODY	KIDNEY	THYROID
Nearest Meat Animal	Meat			
(1.00 mile N)	Adult	1.2-3	8.7-4	3.4-2
	Teen	7.9-4	6.7-4	2.5-2
	Child	1.0-3	8.0-4	3.8-2
Nearest Milk Cow	Milk			
(5 miles ESE)	Adult	6.0-4	6.8-4	6.0-2
	Teen	7.3-4	1.2-3	9.6-2
	Child	9.8-4	1.9-3	1.9-1
	Infant	1.4-3	3.1-3	4.7-1
Nearest Milk Goat	Milk			
(5 miles ESE)	Adult	7.2-4	8.1-4	7.2-2
	Teen	8.8-4	1.4-3	1.1-1
	Child	1.2-3	2.3-3	2.3-1
	Infant	1.6-3	3.7-3	5.6-1

11.4 SOLID WASTE MANAGEMENT SYSTEM

This section describes the capabilities of the station to control, collect, process, handle, store, and dispose of solid radioactive waste generated as the result of normal operation, including anticipated operational occurrences.

11.4.1 Design Bases

11.4.1.1 Design Objectives and Criteria

The objectives of the solid waste management system are as follows:

a. to collect, hold for decay, monitor, package, and temporarily store, prior to offsite shipment, all wet and dry solid radioactive wastes produced by the station during operation and maintenance;

b. to prevent the release of solid radioactive waste materials that could conceivably be hazardous to either operating personnel or the public in accordance with 10 CFR 20 and 10 CFR 50;

c. to minimize the volume of solidified waste requiring shipment offsite; and

d. to take due account (through equipment selection, arrangement, remote handling, and shielding) of the necessity to keep radiation exposure of in-plant personnel "as low as is reasonably achievable."

11.4.1.2 Component Specifications

Equipment capacities are selected to meet the station's solid waste processing requirements in all operational modes of the station, including anticipated operational occurrences, without impairing the power generation availability of the station. Table 11.4-1 provides the design parameters of the various solid radwaste system components.

11.4.1.3 Seismic Design and Quality Group Classification

The solid radwaste system has no seismic requirements. However, the portion of the radwaste building which is below grade is Seismic Category I. All equipment (including tanks, pumps, valves, and piping) of the solid radwaste system containing radioactive wastes conforms to the codes and standards specified in NRC Regulatory Guide 1.26, as applicable to Quality Group D, and to codes specified in Table 1 of Regulatory Guide 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Reactor Plants". Non-pressure retaining attachments and appurtenance are excepted from these requirements. Examples of this type of equipment are nozzles, piping and welds in atmospheric tanks which do not carry process fluid and whose elevation is above the tank overflow level. See USAR Section 1.8, Reg Guide 1.143, item number 8.

Piping that penetrates the containment and the associated isolation valves are classified as Quality Group B and Seismic Category I.

11.4.1.4 Quality Control

The quality control program for the solid radwaste system is the same as described in Subsection 11.2.1.4. This program is in accordance with Regulatory Guide 1.143, with the additional requirements of 10CFR71, Subpart h.

11.4.1.5 Special Design Features

The special design features to minimize radiation exposure to operating personnel during normal and maintenance operations, and to preclude the uncontrolled release of radioactivity as noted in Subsections 11.2.1.5 and 11.2.1.6 for the liquid waste management systems, also apply to the solid radwaste system where appropriate except not all tanks have provisions for flushing using spray nozzles. They can be flushed by filling them with clean water and recirculating it through the system. The following additional features are provided exclusively for the solid radwaste system:

a. System flush capabilities are provided to minimize crud accumulation.

b. The use of compressed gases to dry or convey radioactive waste is avoided in the system design.

c. Waste processing area floor drains are routed to waste sludge Tank A to provide a collection tank that is compatible with the composition of the waste in these drains.

d. The concentrated waste tanks have specifically designed sloping bottoms to preclude crud accumulation on the tank bottom.

The potential for a single operator error to cause an uncontrolled release to the environment is considered remote. The potential for a-single equipment failure to cause an uncontrolled release to the environment is discussed in Subsections 2.4.12 and 2.4.13.

11.4.1.6 Design-Basis and Maximum Waste Volume

Table 11.4-2 provides the maximum and design-basis inputs to the solid radwaste system in terms of weight or volume plus radioactivity. Table 11.4-3 provides the assumptions and basis used to develop Table 11.4-2. Table 11.4-6 provides an isotopic listing of the total activity in the wet solid waste produced annually.

11.4.2 System Description

The solid radwaste system processes wet solid effluents from the liquid radwaste system and dry solid waste. A simplified flow diagram of the system is provided in Figure 11.4-1. A process diagram for the system is provided in Drawing M05-1082. The system is divided into four subsystems; wet solid waste, wet solid waste packaging and handling, mobile solidification station, and dry solid waste packaging. Piping and instrumentation diagrams for the first two subsystems are provided in Drawing M05-1089. The radwaste building equipment arrangement is presented in Drawings M01-1502, M01-1508, M01-1514, and M01-1531. The solid radwaste system is designed in accordance with the guidance provided by Regulatory Guide 1.143, Revision 0.

11.4.2.1 Types of Waste

The major types of wet and dry solid waste handled by the solid radwaste system include:

a. Expended deep bed demineralizer bead resins from condensate polishers and radwaste demineralizers.

b. Filter sludges consisting primarily of fibrous media with small amounts of powdered resins, and/or powdered charcoal from radwaste filters.

c. Tank bottom sludges consisting primarily of piping corrosion products, resin fines from the ultrasonic resin cleaner, and floor drain suspended solids.

d. Cleanup and fuel pool filter-demineralizer sludge consisting primarily of powdered ion exchange resins.

e. Three types of evaporator concentrates can be generated: (1) floor drain wastes, primarily suspended solids with some dissolved solids; and (2) laundry wastes consisting primarily of detergent solutions.

f. Evaporator boil-out consisting primarily of decontamination solution and suspended solids. (Only if evaporator decontamination is ever required).

g. Low-Level dry radioactive wastes consisting of air filters, miscellaneous paper, plastics, rags, etc. from contaminated area; contaminated clothing, tools, and equipment parts which cannot be effectively decontaminated; and solid laboratory wastes. During unusual circumstances, high-level dry radioactive wastes consisting mostly of equipment that has been activated during reactor operation (e.g., core components) may be processed.

The response to Q&R 460.12 has been superseded with the modification to use a contracted solidification system. The important physical and chemical parameters for this system are weight percentage of solids, and pH. The important physical and chemical parameters for each vendor system varies with the particular process. Each process is evaluated per the Process Control Program (PCP).

11.4.2.2 Wet Solid Waste Subsystem

The wet solid waste subsystem consists of a group of tanks and associated pumps which serve as an interface between the liquid radwaste system and the mobile solidification station. These tanks provide intermediate storage for slurries produced by radwaste processing equipment or other radioactive water cleanup systems.

A plug-type divert valve is provided in the recirculation line near the mobile solidification station to enable the waste to be metered to the mobile solidification equipment or recirculated to the tank of origin. Automatic flushing sequences are provided for the recirculation line and the interfacing transfer line.

11.4.2.2.1 Spent Resin Tank

The spent resin tank serves as a receiving tank for exhausted ion exchange bead resins discarded from the condensate polishing demineralizers and the waste demineralizers. Resins are allowed to settle in this tank. Excess sluicing water is drawn off from a decant port with the spent resin tank decant pump and discharged to a waste collector tank. After accumulation of a batch of settled resins, and possibly some additional holdup time to allow for radioactive decay, the resin is fluidized using the tank's sparger system. The resin slurry is then discharged, through the tank bottom discharge port, to the solidification station using the spent resin sludge pump.

The spent resin sludge pump is a progressing cavity pump. Valves in the spent resin handling lines are plug type so that valve closure is not impaired by the presence of solids in the fluid stream.

11.4.2.2.2 Phase Separators

Two phase separators serve to collect, settle, and hold for decay the sludge from the reactor water cleanup filter demineralizer backwash receiving tanks in the containment building. These wastes are periodically discharged to the radwaste system. The solid phase of this sludge is potentially the highest in radioactivity level of the wastes regularly produced in the station. The liquid phase, after separation by settling, is expected to be of moderate activity and low in dissolved solids. The two phase separators are arranged so that one phase separator normally collects inputs for approximately 60 days, while the other phase separator is quiescent, allowing the accumulated sludge to decay prior to its transfer to the mobile solidification station. The radioactive decay during this holdup period plus the decay during the accumulation period reduces the radioactivity of the sludge to levels suitable for processing, packaging, and off-site shipment.

Excess liquid is decanted regularly from the phase separator through the phase separator decant pump as sludge is accumulated. Decant outlets from the phase separators are located sufficiently above the maximum sludge level to provide for an adequate settling volume and to allow decanting without disturbing the settled sludge. Decant liquid is discharged to a waste collector tank. Accumulated sludges are discharged through the phase separator sludge pumps to the mobile solidification station. The phase separator sludge pumps are centrifugal pumps.

Provisions are made for mixing of the sludge by recirculation prior to discharge and continued recirculation during discharge. Flushing capabilities are provided upon termination of the sludge transfer. Valves in the sludge handling lines are plug type so that valve closure is not impaired by the presence of solids in the fluid stream.

11.4.2.2.3 Fuel Pool Filter/Demineralizer Sludge Tanks

The fuel pool filter/demineralizer sludge tanks are located below the fuel pool cleanup filter/demineralizers and receive backwash discharge directly from these filter/demineralizers. These backwashes occur intermittently and are of moderate-to-low radioactivity levels not normally requiring extended hold-up for decay.

Excess water is decanted to a waste collector tank through the fuel pool filter/demineralizers sludge tank decant pumps. Sludges are discharged to the mobile solidification station through

fuel pool filter/demineralizer sludge tank sludge pumps. Sludge and decant transfers are subject to the same requirements as stated for the phase separators.

11.4.2.2.4 Waste Sludge Tanks

The waste sludge tanks are located below the waste filters and receive backwash discharge directly from these filters. These backwashes occur irregularly and are of variable radioactivity. These tanks also receive input from the various collector, surge, processing, or feed tank bottoms as described in Section 11.2. Waste sludge tank decant pumps discharge to the waste collector tanks, floor drain surge tanks, or chemical waste collector tanks. Waste sludge pumps transfer sludges to the mobile solidification station.

11.4.2.2.5 Waste Sludges

Sludges from the phase separator tanks, the fuel pool filter/demineralizer sludge tanks, and the waste sludge tanks can be metered to the mobile solidification station through one of two sludge transfer lines. Plug-type divert valves are provided at the mobile solidification station to enable the wastes to be metered to the mobile solidification equipment or recirculated to the appropriate sludge tank.

When a sludge transfer is completed, the divert valves routes the flow back to appropriate sludge tank. Flushing capability is provided upon termination of the sludge transfer.

11.4.2.2.6 Concentrate Waste Tanks

The concentrate waste tanks receive the concentrated bottoms from the floor drain and chemical waste evaporator packages. Depending upon the composition of the bottoms (regenerant solutions) concentrated slurry solution could solidify at ambient temperatures, the tanks and associated piping are heat traced.

The concentrate waste subsystem is designed to minimize locally quiescent areas during and after the tank contents transfer. Plunger-type valves are used at the tank outlet nozzle. Each tank is furnished with a recirculation line and eductor, and also a mechanical mixer to ensure continuous agitation of tank contents to minimize the chances for crystal accumulation at the tank bottom.

Tank contents are discharged to the mobile solidification station through the concentrated waste pumps. The radioactivity in these tanks may become moderately high, but due to the properties of the contained slurries, it is undesirable to hold wastes any longer than required by the availability of the mobile solidification station equipment.

Two heat traced transfer lines are provided to the mobile solidification station.

Flushing capability is provided upon completion of the transfer to the processing vendor.

11.4.2.3 Mobile Solidification Station

The mobile solidification station consists of concentrated waste, sludge and resin waste transfer lines with isolation valves upstream of flat flanges. These flanges serve as interface connections to mobile solidification or dewatering / drying processing equipment via a manifold and flexible hoses.

The valves in the transfer lines are plug type so that valve closure is not impaired by the presence of solids in the fluid streams.

11.4.2.4 Control of Solidification or Dewatering / Drying of Waste Streams

The mobile solidification or dewatering / drying system is designed to package radioactive solid waste for offsite shipment and subsequent burial in accordance with applicable NRC (10 CFR 61 and 71) and DOT (49 CFR 170-178) regulations.

Packing meeting the applicable requirements for the shipping class are selected. Packaging requirements are detailed in 49CFR. Acceptable waste form requirements are achieved by establishing a suitable Process Control Program (PCP). Elements of the process control program may include, but not limited to:

- Process Control Parameters
- Boundary conditions for the process
- Proper waste form properties and testing
- Specific instructions

11.4.2.5 Dry Solid Waste Packaging Equipment

Dry Active Waste (DAW) is sorted and loaded in packages. Offsite processing services are used to support the disposal of DAW. Materials are prepared according to regulations for transportation of radioactive material 49 CFR and per acceptance criteria referenced in the contracts with the processing vendors.

The packages are normally filled and then moved to temporary storage outside the plant within the Protected Area either at the northwest corner of the Turbine Building in the gravel and asphalt area or the Southeast corner of the Protected Area.

The hydraulic baler is located in the baler room which is adjacent to the waste shipping area. The exhaust of the baler ventilation system is ducted away from the vicinity of the baler to reduce the potential for creating airborne radioactivity at the operator's station.

Handling and packaging of large waste materials will be considered on a case-by-case basis depending on the specific waste material characteristics. Decontamination, packaging, and storage will be accomplished as required. Waste processing alternatives which surpass the capabilities currently on site will be used as necessary to supplement the site capabilities.

11.4.2.6 Solid Waste Handling

The movement of filled waste containers from the mobile solidification or dewatering / drying system to storage or the shipping area is performed by the radwaste overhead bridge crane or the turbine bridge crane. The radwaste overhead crane is remotely operated using television cameras; one mounted on the crane trolley, one in the storage area, and one in the truck loading area. This crane has several safety systems incorporated into its design. A trail cable enables moving the bridge crane to a low radiation area for repair and container removal should the trolley or bridge power system fail. Should the hoist fail, an automatic brake would prevent the container from falling. Brakes are applied when the crane is not in service.

11.4.2.7 Control and Instrumentation

Monitoring and control of the radwaste crane is accomplished in the Radwaste Operations Center (ROC).

An auxiliary control panel is located in the general shipping area. This panel is used primarily for crane maintenance and testing. This panel may also be used for crane operations when the dose rate of the load permits. The direct observation of the load during crane operations from this auxiliary control panel allows for more accurate and, therefore, safer load movements.

In addition to alarms, pump controls, valve switches, and level indication, the solid radwaste control panel has bridge crane controls and closed circuit television controls.

11.4.3 System Malfunction Analysis

All tanks containing any radioactive waste in the solid waste system are located in a shielded cubicle. The cubicles are drained to a sump in the event of a tank failure and have a curb to retain any excess water in the cubicle. Ventilation air will always be from nonradioactive areas to low-level areas to high-level areas, thus precluding airborne contaminants from leaving the cubicle area to an area where station personnel might be located. The solid waste system tanks have means for introducing condensate to flush or rinse down the tanks prior to any required maintenance.

Pumps are located outside the tank cubicles, and any pump associated with high-level wastes in the solid system is located in its own cubicle with a drain.

Highly radioactive piping is routed through piping tunnels or behind shield walls, to segregate it from station personnel during operation and in case of failure. It is normal to flush lines that

carry solid wastes after an operation cycle to prevent any buildup or plugging of the solid-waste piping system.

If a pump malfunction occurs, the pump would be isolated with valves operated remotely, and the process streams transferred to the redundant pump. This would allow the pump to be removed or repaired after any radioactive contamination of the pump is allowed to decay to reduce the occupational dose rate.

Leaking Valves: Depending on the particular valve, it would be flushed, isolated, and repaired. Valves are located outside tank cubicles, to reduce the background radiation level near any valve requiring maintenance.

The floors and the walls near equipment are located to keep contamination buildup to a minimum and to ease decontamination.

Area radiation monitors are provided to detect any unsafe radiation conditions near the waste handling equipment.

The system is designed to permit operation, maintenance, and correction of component malfunction with minimal personnel exposure.

Several modes of failure can be hypothesized for both the Radwaste and Turbine Building cranes that could result in a release of radioactivity due to a container dropping from the crane. Analysis of a container rupture due to being dropped in either the Radwaste or Turbine Building demonstrates that the limits of 10CFR20 are not exceeded.

11.4.4 Packaged Waste Storage Areas

Shielded areas are provided for storage of packaged waste including compacted dry waste drums as indicated in Drawing M01-1508. Area capacities are noted in Table 11.4-5. Visual surveillance is provided by television cameras in the storage area and on the bridge crane trolley. Storage space is available to accommodate liners of solidified or dewatered or dryed waste. Periodic surveillance will be performed by personnel.

11.4.5 Performance Testing

Shop tests were performed on most solid radwaste system equipment prior to shipment to ensure that performance requirements were satisfied. The system is proven operable by its use during normal station operation.

11.4.6 Shipment

Containers normally can be shipped as soon as solidification or dewatered / dryed is complete, provided the proper shielding is available, without exceeding Department of Transportation radiation limits. If 49 CFR 173 dose limitations cannot be met with the available shielding, the containers are stored until the appropriate shielding is available.

The shipment of waste will be a planned and scheduled activity. The monitoring of the solidified or dewatered / dryed waste container's radiation level prior to storage will be used to determine the number of containers to be shipped per truck and the shielding requirement. The monitoring of the solid or dewatered / dryed waste will allow for the accurate recording of surface radiation level for record purposes.

Shipment of solid or dewatered / dryed waste from the station will only be by licensed carriers and will be limited to shipment to licensed commercial or federal waste repositories or licensed secondary processors for additional processing.

TABLE 11.4-1
SOLID RADWASTE MANAGEMENT SYSTEM COMPONENTS AND DESIGN PARAMETERS

TANKS

QUANTITY	DESCRIPTION	CAPACITY (GAL)	DESIGN PRESSURE	DESIGN TEMPERATURE (°F)	MATERIALS OF CONSTRUCTION
2	Phase separator	10,000	Full Water	150	304 SS
2	Concentrated Waste	5,000	Full Water	230	Incoloy 825
1	Spent resin	8,500	Full Water	150	304 SS
2	Waste sludge	10,000	Full Water	150	304 SS
2	Fuel pool F/D sludge	10,000	Full Water	150	304 SS

PUMPS

QUANTITY	DESCRIPTION	DESIGN FLOW (gpm)	DESIGN HEAD (ft)
2	Phase separator sludge tank sludge	300	200
2	Fuel pool F/D sludge tank sludge	300	200
2	Waste sludge tank sludge	300	200
2	RWCU F/D backwash tank	75	62
1	Spent resin tank decant	75	78
2	Phase separator tank decant	75	70
2	Fuel pool F/D sludge tank decant	75	55
2	Concentrated waste tank	100	20
2	Waste sludge tank decant	75	90
1	Spent resin tank sludge	100	173

SORTING AREA

QUANTITY	DESCRIPTION	CAPACITY
1	Dry waste compactor	-----

CONTAINER HANDLING EQUIPMENT

QUANTITY	DESCRIPTION	CAPACITY
1	Bridge crane	10 tons

OTHER EQUIPMENT

QUANTITY	DESCRIPTION	CAPACITY
	Liners and Boxes	Various

TABLE 11.4-2
DESIGN BASIS AND MAXIMUM ANNUAL OUTPUT FOR VARIOUS PLANT CONDITIONS

TYPE OF WASTE	NORMAL OPERATIONS DESIGN-BASIS PARAMETERS: AMOUNT GENERATED	NORMAL OPERATIONS: NUMBER OF CONTAINERS: CEMENT	NORMAL OPERATIONS: NUMBER OF CONTAINERS: DEWATER	EFFECTS OF OPERATIONAL OCCURRENCES (CONDENSER, INLEAKAGE,) REACTOR SHUTDOWN): NUMBER OF CONTAINERS: CEMENT	EFFECTS OF OPERATIONAL OCCURRENCES: NUMBER OF CONTAINERS: DEWATER	TOTAL RADIO-ACTIVITY CONTENTS* IN mci
Deep Bed Resins	1,900 ft^3		11-183.3 ft^3 Liners		15-183.3 ft^3 Liners	6.6E6
Filter Sludges (Equipment Drains)	1,600 ft^3	11-177.9 ft^3 Liners	1-177.9 ft^3 Liners	11-177.9 ft^3 Liners	3-177.9 ft^3 Liners	9.2E6
Evaporator Bottoms	1,100 ft^3	8-177.9 ft^3 Liners		10-177.9 ft^3 Liners		1.5E6
RWCU Sludge	350 ft^3		3-135.5 ft^3 Liners		3-135.5 ft^3 Liners	2.6E7
Fuel Pool F/D Sludge	350 ft^3		3-135.5 ft^3 Liners		3-135.5 ft^3 Liners	3.8E6
Misc Waste (Oil, Sump Sludge & Condenser Silt)	300. ft^3	60-55 gal Drums		70-55 gal Drums		Negligible
TOTAL	5,600 ft^3 (8,400 ft^3)**					
Dry Active Waste After Volume Reduction						9.2E4

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* Based On CPS USAR Table 12.2-12
DAW Activity Assumed 1% Of Filter Sludge Activity

** Actual Amount Shipped Is Based On Radwaste Processing

TABLE 11.4-3
ASSUMPTION USED TO GENERATE THE DESIGN
BASIS AND MAXIMUM ANNUAL OUTPUT
QUANTITIES IN TABLE 11.4-2

a. 30,000 gal/day at 50 ppm solids in equipment drain stream, 50,000 gal/day with URC operations.

b. Crud loading 0.45 lb/lb filter media.

c. 7,500 gal/day at 200 ppm dissolved solids and 250 ppm suspended solids in floor drain stream.

d. Evaporator bottom concentrate 50% solids by weight for the mixture of floor drain and chemical wastes, and 25% solids by weight for laundry wastes.

e. Ultrasonic resin cleaner produces 25,000 gallons wash water at 450 ppm solids per washing.

f. Eight or nine condensate polishers in service, one polisher bed averages 216 days' service.

g. 1000 gal/day laundry waste at 100 ppm dissolved solids and 500 gal/day miscellaneous chemical wastes at 200 ppm dissolved solids when laundry is processed on site.

h. Two reactor water cleanup filter demineralizers.

i. 1.6 reactor water cleanup filter demineralizers backwashed every week.

j. RWCU backwash 1800 gallons plus 42 lb. solids per vessel.

k. Phase separators sized to hold two backwash volumes plus two months' accumulated sludge.

l. Bead resins from 1.5 radwaste demineralizer beds discarded annually when resins are regenerated, 1 RW demineralizer discarded every 2 weeks when resins are not regenerated.

m. One radwaste demineralizer regenerated every 14 days producing 5000 gallons of regenerants with 620 lb solids. This is not applicable when resins are discarded after use rather than being regenerated.

n. One fuel pool cleanup filter demineralizer backwashed every two weeks per unit.

o. Sludges and slurries will generally be dewatered.

p. 130 ft^3 of waste slurry at 22% solids by weight for concentrates.

q. Deleted.

TABLE 11.4-3 (Continued)
ASSUMPTION USED TO GENERATE THE DESIGN BASIS AND MAXIMUM ANNUAL OUTPUT QUANTITIES IN TABLE 11.4-2

r. Eighty percent of floor drain and URC waste solids settle out and go to waste sludge tanks.

s. Compaction ration is 3.08:

Avg. weight is 225 lbs. - 40 lbs. (for drum) = 185 lbs.

$$\frac{185\,\text{lbs}}{7.5\text{ft}^3 \times 81\,\text{lbs}/\text{ft}^3} = 3.08$$

27,142 ft^3 of DAW generated in 1989 before processing;

3,831.14 ft^3 was disposed of in final form in 1989.

TABLE 11.4-4 has been deleted intentionally.

TABLE 11.4-5
SOLID WASTE MANAGEMENT SYSTEM
STORAGE AREA DESIGN CAPACITIES

Material	Location	Number Of Areas	Radioactive Waste Capacity
Solid or Dewatered / Dryed Radioactive Waste (5)	122-126.1/E-F 737' Radwaste	1	6,400 ft^3 (1)
Dry Active Waste	129.7-200/A-F 702' Radwaste	1	12,500 ft^3
Dry Active Waste	129.7-132.4/A-D 720' Radwaste	1	5,700 ft^3
Storage Void	122/M Filter Mezzanine Radwaste	1	1,500 ft^3
Bulk Uncompacted Waste	104-108/A-B 737' Turbine	As Required	-------
Package Dry Active Waste	Outdoor Areas 1. SE corner of Protected Area 2. Outside Protected Area, Inside OCA, Old Electrical Cable Storage Area 3. Resin Storage Building	3	70,200 ft^3 (2)
Turbine Rotor Storage (5)	Indoors, Outside Protected Area, Inside OCA-Old Electrical Cable Storage Area	1	(3)
Turbine Diaphragm and Misc Tool Storage (5)	Indoors, Outside Protected Area, Inside OCA-Old Large Fab Shop	1	43,290 ft^3 (4)

(1) Storage capacity limited by floor loading.

(2) Storage capacity is limited to a total of 60 containers. These areas are used as temporary storage areas for packaged dry active waste to be shipped from the facility or moved to storage areas outside the protected area.

(3) 2 LP and 1 HP Turbine Rotors stored in a 25'W X 20'H X 150'L Quonset Hut. Contamination levels limited to ≤ 50,000 dpm/100 cm^2 smearable and ≤ 50 mrad fixed. No other material storage is allowed in this building.

(4) Storage in this area is limited to up to 22 sealands with turbine diaphragms and up to 15 sealands with contaminated tools. Contamination levels limited to ≤ 50,000 dpm/100 cm^2 smearable and ≤ 50 mrad fixed. No flammable liquids are allowed in this area.

(5) Long Term Storage Areas

TABLE 11.4-6
INVENTORY OF ISOTOPES IN THE WET SOLID WASTE ANNUAL OUTPUT*

Isotope	Activity, Ci/yr.
Sr-89	5.28+3
Sr-90	6.24+2
Y-91	9.12+2
Zr-95	7.68+1
Nb-95	9.60+1
Ru-103	2.88+1
Ru-106	6.72+0
Te-129m	9.60+1
Cs-134	4.08+2
Cs-137	6.48+2
Ba-140	5.52+3
Ce-141	1.78+2
Ce-144	8.40+1
Pr-143	2.88+1
Pm-147	3.12-1
P-32	1.32+1
Cr-51	8.16+4
Mn-54	1.15+5
Co-58	2.64+4
Co-60	2.26+5
Fe-59	2.64+3
Zn-65	6.24+3
Ag-110m	6.24+2
TOTAL	4.73+5

* Based on CPS USAR Table 12.2-12 and 11.4-2

11.5 PROCESS AND EFFLUENT RADIOLOGICAL MONITORING AND SAMPLING SYSTEMS

This section describes the systems that monitor and sample the process and effluent streams in order to control releases of radioactive materials produced as a result of normal operations, anticipated operational occurrences, and during postulated accidents. A schematic flow diagram for all normal ventilation exhaust systems is shown in Drawing M05-1100.

11.5.1 Design Bases

Design bases are described in terms of design objectives and design criteria in the following subsections.

11.5.1.1 Design Objectives

11.5.1.1.1 General Objectives

The design objectives of the process/effluent radiation monitors are:

a. To measure, indicate, and record the levels of radioactivity associated with the process streams and in the effluents released from the plant.

b. To activate alarms and initiate appropriate control actions when preset radioactivity levels are exceeded.

c. To provide data to assist the licensee in complying with applicable federal, state, and local requirements associated with the monitoring and control of radioactivity releases.

The design objectives of the liquid and effluent sampling system are:

a. To provide the means for identifying nuclear and chemical properties of liquid and effluent streams.

b. To complement the measurements made by the liquid/effluent radiation monitors (e.g., enable isotopic analyses of fluids which are monitored only for gross gamma).

c. To monitor performance of plant equipment and processes.

d. To provide data to assist the licensee in complying with applicable federal, state, and local requirements associated with monitoring and control of radioactivity releases.

11.5.1.1.2 Specific Objectives

11.5.1.1.2.1 Normal Conditions

Specific objectives of individual systems under normal and anticipated operational occurrences are as follows:

a. Monitor the common station HVAC vent to provide a record of releases and to detect and alarm noble gas release rates which correspond to dose rates at or beyond the site boundary in excess of 500 mrem/yr to the total body or 3000 mrem/yr to the skin. Detection limits for the monitoring system are given in Table 11.5-1.

b. Monitor the standby gas treatment system effluent under conditions associated with normal and anticipated operational occurrences with the same objective as Item a.

c. Monitor the plant service water effluent to detect inter-system leakage, and measure gross radioactivity prior to discharge to the ultimate heat sink. Detection limits and alarm setpoints for the monitor are established to assure an alarm should a significant inadvertent release occur, yet prevent spurious false alarms due to fluctuations in ambient radiation background levels.

   Clinton continuous effluent monitoring systems will comply with ANSI 13.10-1974 which list the same requirements as in Regulatory Guide 4.15, section C.7.

   Clinton radwaste management systems comply with the provisions of Regulatory Guide 1.143 (Revision 0). (Q&R 460.14)

d. Monitor the shutdown service water effluent with the same objective as paragraph c.

e. Monitor the fuel pool heat exchanger service water effluent to detect and measure inter-system leakage. Detection objectives for the monitor are the same as in paragraph c above.

f. Monitor the radwaste discharge to detect and alarm concentrations in the plant effluent which are in danger of exceeding the limits in 10 CFR 20 Appendix B, Table 2, Column 2 at the liquid discharge point to the unrestricted area. The minimum detectable concentration is the same as in paragraph c above. This monitor is interlocked with the radwaste discharge valve to terminate releases if high radioactivity levels are detected.

g. Monitor the condenser air ejector off-gas process steam and alarm when the Offsite Dose Calculation Manual limit is approached on the total noble gas release from the fuel, and/or when the off-gas release rate from the treatment system approaches a value that corresponds to the Offsite Dose Calculation Manual limits on off-site dose rates.

The process radiation monitoring system, in conjunction with associated process sampling and laboratory analysis, enables qualitative and quantitative determination per Regulatory Guide 1.21 of the radioisotopes and concentrations that are present in each system and assessment of compliance with requirements of 10 CFR 50, Appendix I.

11.5.1.1.2.2 Accident Conditions

The objectives of the process radiation monitoring system under postulated accidents are:

a. To monitor the following in-plant ducts and limit the release of high airborne radioactivity by initiating isolation of the duct and initiating the standby gas treatment system (SGTS);

   1. containment building exhaust;

   2. containment building fuel transfer vent plenum;

   3. fuel building ventilation exhaust; and

   4. continuous containment purge exhaust.

b. To monitor the standby gas treatment system exhaust downstream from the filter trains, and provide operating personnel with quantitative knowledge of particulate, iodine, and noble gas radioactivity levels being released through continuous monitoring of noble gases and continuous sampling of iodines and particulates.

c. To monitor the air in the vicinity of the control room air intake ducts; and initiate the makeup air filter train upon high radioactivity level. (After this initial shutdown action the operator selects which intake will be used, based partially on the readings of these monitors.)

d. To monitor the common station HVAC vent downstream of all ventilation inputs, and provide operating personnel with quantitative knowledge of noble gas, radioactivity levels for postaccident releases.

The process sampling system can be used to complement the measurements made by the PRM's as outlined in Subsection 9.3.2.

All the previously described means of radioactivity detection and determination are provided to satisfy the requirements of 10 CFR 20 and 10 CFR 50, Appendix A and Appendix I, and, in particular, General Design Criterion 64 which requires monitoring or sampling of all potential release pathways. The SGTS HVAC high range radiation monitors are provided to satisfy the requirements of NUREG-0737 II.F.1, Attachments 1 and 2 and Regulatory Guide 1.97.

11.5.1.2 <u>Design Criteria of the Process Radiation Monitoring System and Process Sampling System</u>

The design criteria that have been used for the process and effluent radiation monitoring and sampling system are:

a. To enable the station to comply fully with the measuring, evaluating, and reporting requirements of Regulatory Guide 1.21. This compliance is accomplished by continuously monitoring the processes and effluents which could be contaminated. These potentially contaminated streams (see Subsection 11.5.2 for more details) are sampled in accordance with Regulatory Guide 1.21 requirements (as a minimum) and laboratory analysis is done on the samples. This analysis enables isotopic identification and quantification of the radioactivity in the process streams and effluents.

b. To enable compliance with 10 CFR 20 and 10 CFR 50 requirements, both as to streams that require monitoring and as to required ranges and sensitivities. Sensitivities (minimum detectable levels) have been chosen to be as far below 10 CFR 20, Appendix B, Table 2 levels as commercial state-of-the-art reliability allows. The ranges have been chosen to encompass the minimum detectable level and the level at which alarm and/or control action should take place. Duct monitors have been provided with ranges suitable for postulated accidents.

c. To comply with all other applicable federal regulations and guidance, and with industry codes. This compliance includes:

   1. Regulatory Guide 1.21,
   2. Regulatory Guide 1.22*,
   3. Regulatory Guide 1.29*,
   4. Regulatory Guide 1.53*,
   5. Regulatory Guide 1.97,
   6. Regulatory Guide 8.8,
   7. ANSI N13.1-1969,
   8. ANSI N13.2-1969,
   9. ANSI N13.10-1974,
   10. IEEE 279*,
   11. IEEE 323*,
   12. IEEE 338*,
   13. IEEE 344*,
   14. IEEE 379*, and
   15. NUREG-0737, Item II.F.1.

   NOTE:* For nuclear safety-related monitors only.

d. To be reliable. This reliability is accomplished by providing state-of-the-art detectors and solid-state circuitry.

e. To assist in assuring safety of plant personnel. The system is designed to detect increasing contamination in plant streams as early as possible so as to enable early correction. This early correction in turn will assist in making radiation doses to personnel and members of the public ALARA. In addition, the system is designed for safe and low maintenance operation (by design) for sample stream conditions, by its panel design, and by choice of components.

f. To have a suitable time response to radiological events. This is accomplished by providing a system with adequate time response.

g. To provide local alarms and/or indications at key points when a substantial increase in radiation/radioactivity would be of immediate concern to personnel in the area. This feature is provided by the process radiation monitoring system design (see Subsection 11.5.2 for details).

h. To provide records of radioactivity levels at a central location. Sample analysis results will be recorded.

11.5.2 System Description

11.5.2.1 Systems Required for Safety

Information on the subsystems described in Subsections 11.5.2.1.1 through 11.5.2.1.5 is presented in Table 11.5-1 and 11.5-2. Information on the subsystems described in Subsections 11.5.2.1.6 and 11.5.2.1.7 is presented in Tables 11.5-1 and 11.5-2. Also refer to Drawing M05-1064.

11.5.2.1.1 Main Steamline Radiation Monitoring System (MSLRMS)

This system monitors the gamma radiation level external to the main steamlines. 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, the monitoring system alarms in the main control room and provides channel trip signals to the mechanical vacuum pump, if running.

The system consists of redundant instrument channels as shown on Drawing M05-1064. Each channel consists of a local detector (gamma-sensitive ion chamber) and a control room radiation monitor with an auxiliary trip unit. Power for channels A and D is supplied from NSPS buses A and D, respectively. Each channel is physically and electrically independent of the other channel.

The detectors are physically located near the main steamlines just downstream of the outboard main steamline isolation valves in the space between the primary containment and secondary containment walls. The detectors are geometrically arranged so that this system is capable of detecting significant increases in radiation level with any number of main steamlines in operation.

Each channel provides HI-HI and INOP logic signals and actuation. If both of the channels give either a HI-HI or INOP trip signal, the mechanical vacuum pump will trip if running.

11.5.2.1.2 Containment Building Fuel Transfer Ventilation Plenum Radiation Monitors

Four redundant gross gamma channels are provided to monitor the radiation in the exhaust air from the refueling pool as measured at the containment building fuel transfer ventilation plenum located in the primary containment. Channel sensitivity is indicated in Table 11.5-2. The location of these monitors is shown on Drawing M05-1111, Sheet 3.

Each channel consists of a GM tube detector and local monitor. The detector is mounted external to and adjacent to the plenum. Directional shielding for each detector is provided and the detectors are oriented such that they monitor only the radiation originating from within the plenum.

Whenever the level of radioactivity exceeds a preset level, as indicated in Table 11.5-1, the monitoring system actuates an annunciator in the main control room, isolates the primary containment ventilation, isolates the secondary containment ventilation (fuel building), and starts the standby gas treatment system.

Power for two channels (A and C) is supplied from ESF bus A and for the other two channels (B and D) from ESF bus B. Channels A and C are physically and electrically independent of channels B and D. A one-out-of-two-twice logic is provided to actuate control functions.

A monitor failure provides annunciation in the main control room. Operations will then take actions as prescribed by the Technical Specifications.

11.5.2.1.3 Containment Building Exhaust Radiation Monitors

The containment building exhaust radiation monitors are identical in construction and function to those described in Subsection 11.5.2.1.2 except for the monitoring location. The monitors are located on the containment exhaust duct downstream of the outboard containment isolation valve. The location of these monitors is shown on Drawing M05-1110, Sheet 2 and Drawing M05-1100, Sheet 1.

11.5.2.1.4 Containment Building Continuous Containment Purge (CCP) Exhaust Radiation Monitors

The containment building continuous containment purge (CCP) exhaust monitors are identical in construction and function to those described in Subsection 11.5.2.1.2 except for the monitoring locations. The monitors are located on the CCP exhaust duct downstream of the outboard containment isolation valve. The location of these monitors is shown on Drawing M05-1100, Sheet 1 and Drawing M05-1111, Sheet 5.

11.5.2.1.5 Fuel Building Ventilation Exhaust Radiation Monitors

The fuel building ventilation exhaust radiation monitors are identical in construction and function to those described in Subsection 11.5.2.1.2 with two exceptions; the monitoring location is on the fuel building exhaust duct upstream of the fuel building isolation damper and the monitors do not isolate the primary containment ventilation. The location of these monitors is shown on Drawing M05-1104, Sheet 2.

The monitoring location is sufficiently upstream to assure that the fuel building exhaust will be isolated prior to any significant release of radioactivity to the outside atmosphere.

11.5.2.1.6 Control Room Air Intake Radiation Monitors

Four redundant gross gamma channels are provided to monitor the control room air intakes and control the ventilation system to limit the radiation dose to personnel in the control room under normal and accident conditions. The location of these monitors in the Control Room HVAC system is shown on Drawing M05-1102, Sheet 1, and Drawing M05-1100, Sheet 3.

Each channel consists of a GM tube detector and local monitor. Two detectors are mounted near each of the two separated intakes to monitor the radiation from the cloud.

Whenever the level of radioactivity exceeds a preset level, as indicated in Table 11.5-1, the monitoring system starts one of the two standby makeup air filter trains.

Power for two channels (A and C) is supplied from ESF bus A and for the other two channels (B and D) from ESF bus B. Channels A and C are physically and electrically independent of Channels B and D. One-out-of-two taken twice logic is provided. A monitor failure provides the same input to the trip logic as high radiation.

11.5.2.1.7 Standby Gas Treatment System (SGTS) Exhaust High Range Radiation Monitoring System

One accident range noble gas monitor is provided for continuous monitoring of SGTS effluent for noble gases and continuous sampling of iodines and particulates. The system includes two non-redundant noble gas channels as shown in Tables 11.5-1 and 11.5-2. The operational monitor extracts samples through an isokinetic probe mounted in the exhaust stack. The probe is designed and installed to meet the requirements of ANSI N13.1.

The monitor is an offline sampling type (consisting of four separate assemblies - two skid mounted) and is furnished with particulate and iodine filters to collect radioactivity for laboratory analysis and detectors to monitor for post-accident releases of noble gas radioactivity.

Annunciators in the main control room and local visual and audible alarms are actuated for monitor failure.

The system and sample return line to the SGTS exhaust stack may be purged by local controls or controls in the main control room. Instrument air is drawn through the system by the sample pump and returned to the SGTS exhaust stack.

Operational checks and calibration are as described in Subsection 11.5.2.2.1. Local grab sample capability is also provided with the system. For a more detailed description of this system, refer to Subsection 7.6.1.2.6.

11.5.2.1.8 Common Station HVAC Exhaust High Range Radiation Monitoring System

One accident range noble gas monitor is provided for continuous monitoring of HVAC stack effluent noble gases and continuous sampling of iodines and particulates. The system includes two non-redundant noble gas channels as shown in Tables 11.5-1 and 11.5-2. The operational monitor extracts samples through an isokinetic probe (sampling array configured to the rectangular duct) mounted in the stack downstream of all ventilation inputs to the stack. The probe is designed and installed to meet the requirements of ANSI N13.1.

The monitor is an offline sampling type (consisting of three separate assemblies - two skid mounted) and is furnished with particulate and iodine filters to collect radioactivity for laboratory analysis and with detectors to monitor for post accident releases of noble gas radioactivity.

Annunciators in the main control room and local visual and audible alarms can be actuated on noble gas channel ALERT or HIGH radioactivity or on monitor failure.

Purging of the system is as described in Subsection 11.5.2.1.7 except the purging medium is returned to the HVAC vent stack.

Operational checks and calibration are as described in Subsection 11.5.2.2.1.

Local grab sample capability is also provided with the system.

For a more detailed description of this system, refer to Subsection 7.6.1.2.7.

11.5.2.2 Systems Required for Plant Operation

Information on these subsystems is presented in Tables 11.5-1 through 11.5-3 and shown on Drawing M05-1064.

Power for these subsystems is from 120-Vac nondivisional instrumentation buses, except that the monitors described in Subsection 11.5.2.2.10 are powered from 120-Vac convenience outlets.

Sample lines and flow rates are chosen such that a high Reynolds number is obtained to reduce plateout. Plateout is further reduced by using long radius bends upstream from particulate and mud filters.

11.5.2.2.1 Pretreatment Air Ejector Off-Gas Radiation Monitor

The main condenser pretreatment air ejector off-gas radiation monitor monitors the discharge of the steam jet air ejectors (SJAE) downstream from the off-gas recombiner and cooler condenser (prior to the charcoal adsorbers). This monitor measures the radioactivity in the noncondensible gases drawn from the condenser. The monitor can be used to evaluate the extent of failed fuel rods(s) and combined with the off-gas post treatment monitor, (Subsection 11.5.2.2.2), the effectiveness of the charcoal adsorber(s) in the off-gas system may be determined.

The monitor is an offline sampling type (skid mounted) with a GM tube detector provided for monitoring gross gamma radioactivity. The monitor is calibrated to determine noble gas concentration.

There are two sample paths provided on the skid; one has a GM tube detector, encased in a lead shield for continuous monitoring and the other is to provide grab sample capabilities. Valves on and external to the skid may be aligned to provide the following capabilities:

a. Obtain grab sample taken from the discharge header of the recombiners.

b. Provide continuous sample flow to the detector from the discharge header of the recombiners.

c. Return discharge of either sample to the suction of either SJAE.

Annunciators in the main control room and a local visual and audible alarm actuate on HIGH and ALERT radioactivity or on monitor failure.

The sample chamber and return lines may be purged by local controls or controls in the main control room. Inlet sample line purging may be done locally.

Channel operational checks may be done by actuation of an integrally mounted check source controlled locally or from the main control room. (For further details on channel checks refer to Section 7.7). Provisions are made on the monitor for calibration with a National Bureau of Standards (NBS) traceable source. Special calibration factors are determined from laboratory analysis of grab samples.

11.5.2.2.2 Post-Treatment Air Ejector Off-Gas Radiation Monitors

Two post treatment air ejector off-gas radiation monitors are provided. One monitor is operational and the other is an installed spare. Sample points located upstream from the first charcoal adsorber bed, between the two charcoal adsorber beds, and downstream from the second charcoal adsorber can be lined up to either monitor. These monitors provide:

a. verification of the performance of the charcoal adsorbers,

b. verification that high stack activity is or is not coming from the off-gas system,

c. automatic off-gas system discharge valve closure upon HIGH alarm to limit the discharge or radioactivity from the turbine cycle to the environment, and

d. automatic transfer from bypass to treatment mode upon ALERT alarm.

The monitors are offline sampling type (skid mounted) with detectors to measure particulate, iodine, and noble gas radioactivity.

An ALERT noble gas channel radiation trip will close the off-gas system charcoal absorber bypass valve (a fail closed valve) and a HIGH noble gas channel radiation trip will close the off-gas system discharge valve (a fail open valve). If both of the monitors fail, the discharge valve will automatically close.

Annunciators in the main control and local visual alarms can be actuated on iodine, particulate, or noble gas channel ALERT or HIGH radioactivity or on monitor failure.

Purging, monitor operational checks, and calibration, are as described in Subsection 11.5.2.2.1. Grab sample capability is provided.

11.5.2.2.3 Common Station HVAC Vent Stack Radiation Monitors

Two common station HVAC vent radiation monitors are provided. One monitor is operational and the other is an installed spare. The operational monitor extracts samples via a flow splitter through an isokinetic probe (different from the probe described in Subsection 11.5.2.1.8, but at approximately the same elevation) mounted in the stack downstream of all ventilation inputs to the stack. The probe is designed and installed to meet the requirements of ANSI N13.1.

The monitors are offline sampling type (skid mounted) with detectors are measure particulate, iodine, and noble gas radioactivity. The low range noble gas channel is used in conjunction with the HVAC Accident Range Radiation Monitor (AXM) to monitor accident effluent releases.

Annunciator in the main control and local visual alarms can be actuated on iodine, particulate, or noble gas channel ALERT or HIGH radioactivity or on monitor failure.

Purging, operational checks, and calibration are as described in Subsection 11.5.2.2.1. Grab sample capability is provided.

11.5.2.2.4 Standby Gas Treatment System (SGTS) Exhaust Radiation Monitors

Two SGTS exhaust radiation monitors are provided. One monitor is operational and the other is an installed spare. The operational monitor extracts samples via a flow splitter through an isokinetic probe (different from the probe described in Subsection 11.5.2.1.7, but at approximately the same elevation) mounted in the exhaust stack. The probe is designed and installed to meet the requirements of ANSI N13.1.

The monitors are offline sampling type (skid mounted) with detectors to measure particulate, iodine, and noble gas radioactivity. The low range noble gas channel is used in conjunction with the SGTS Accident Range Radiation Monitor (AXM) to monitor accident effluent releases.

Annunciators in the main control room and local visual alarms can be actuated on iodine, particulate, or noble gas channel ALERT or HIGH radioactivity or on monitor failure.

Purging, operational checks, and calibration are as described in Subsection 11.5.2.2.2. Grab sample capability is provided.

11.5.2.2.5 Plant Service Water Effluent Radiation Monitor

The plant service water effluent radiation monitor, located in the seal well enclosure, measures the radioactivity concentration in the effluent of the plant service water (WS) system to the seal well. The sample extraction point is located in the WS discharge header downstream of all inputs to the header which may possibly contain radioactivity.

The monitor is an offline sampling type (skid mounted) with a gamma scintillator detector to measure gross radioactivity. Grab sample capability is provided.

Annunciators in the main control room and local visual alarms will actuate on HIGH and ALERT radioactivity or on monitor failure.

Sample chamber and sampling return line flushing may be initiated by controls locally or from the main control room. Inlet sample line flushing may be done locally. Channel operational checks may be performed by actuation of an integrally mounted check source controlled locally or from the main control room. Provisions are made on the monitor for calibration with a NBS traceable source.

11.5.2.2.6 Liquid Radwaste Discharge Radiation Monitor

The liquid radwaste discharge radiation monitor measures the radioactivity concentration of the liquid radwaste discharge into the plant service water (WS) discharge header. The sample extraction point is located upstream of the discharge isolation valve prior to the connection to the WS system. The sample point is located sufficiently upstream of the isolation valve to ensure that the monitor response will shut off flow before excess radioactivity passes the valve.

This monitor has the same features and capabilities as the monitor described in Subsection 11.5.2.2.5 with the following additions:

a. Annunciators for ALERT and HIGH radioactivity and monitor failure are provided on the liquid radwaste control panel.

b. Flushing, operational check, and sampling pump may be controlled from the liquid radwaste control panel.

c. Monitor discharge return will be directed back to the liquid radwaste discharge line or to a drain sump which is pumped to the liquid radwaste system. Controls for discharge path selection are on the skid and the liquid radwaste control panel.

The liquid radwaste discharge monitor has two setpoints: ALERT RADIATION LEVEL ALARM and HIGH RADIATION LEVEL ALARM. ALERT provides no automatic isolation function. HIGH ALARM isolates the radwaste discharge to WS isolation valve, regardless of whether WS or CW is providing dilution flow. An isolation signal will also be generated from a separate analog monitor upon low WS flow, low CW flow, or high radwaste effluent flow.

The monitor also provides a signal proportional to the radioactivity concentration to a recorder on the liquid radwaste control panel. For the operator's use in determining dilution factors, a flow indicator for the WS flow and status lights for the circulating water pumps are.provided on the liquid radwaste control panel.

11.5.2.2.7 Shutdown Service Water (SX) Effluent Radiation Monitors

Two monitors are provided on the SX system to 1) detect inter-system leakage in the residual heat removal (RHR) heat exchangers (HX), or from the seal water coolers; and 2) measure the radioactivity concentrations of discharges to Lake Clinton should the SX system be used to cool the RHR HX. One monitor is provided for SX division 1 (System A) and one monitor is provided for SX division 2 (System B). These two divisions provide cooling water to the RHR Systems A and B.

The sample point for each monitor is located downstream from the SX discharge from the RHR HX and the RHR pumps seal water coolers. A significant leak in the HX from the shell side to the tube side, or from the seal water coolers, will be detected by the monitor and alarmed for the operator to correct the condition.

These monitors have the same features and capabilities as the monitor described in Subsection 11.5.2.2.5.

11.5.2.2.8 Fuel Pool Heat Exchanger Service Water Radiation Monitor

Two monitors are provided for fuel pool heat exchanger (FC HX) service water effluent to detect and measure the inter-system leakage. One monitor is provided on the discharge of each of the two FC HX's. The sample extraction point is located upstream from the component cooling water (CC) and shutdown service water (SX) branch connection, such that discharge to either system is monitored.

A significant leak in the HX from the shell side to tube side will be detected by the monitor, which will actuate an alarm to alert the operator to correct the condition.

The CC system, (which is the normal source of FC HX service water) is a closed loop so that continued radioactive leakage will increase the CC system radioactivity concentration but no discharge to the environment will occur.

If the SX system is providing the FC HX service water and leakage occurs, the radioactivity will be diluted by the SX system and discharged to the ultimate heat sink. Operator action would be required to correct the condition.

These monitors have the same features and capabilities as the monitor described in Subsection 11.5.2.2.5.

11.5.2.2.9 Component Cooling Water (CC) Radiation Monitor

A monitor on the CC system is provided to detect radioactive contamination in the CC system. The sample extraction point is located upstream from the CC pump suction header beyond the return point for potentially radioactive sources.

This monitor has the same features and capabilities as the monitor described in Subsection 11.5.2.2.5.

11.5.2.2.10 Miscellaneous Sampling Points

Sample extraction points are provided on the exhausts of the condenser vacuum and gland seal system prior to their connections to the common station HVAC vent. Sample extraction points are also located on the intake to each of the SGTS trains and other sample extraction points located throughout the plant on HVAC ducting to assist locating areas of high radioactivity. Portable three-channel constant air monitors (CAM) may be connected to the points, when required by station operating procedures and conditions. Portable CAM's and other sample extraction points located throughout the plant on HVAC ducting to assist locating areas of high radioactivity are described in Subsection 12.3.4.

Refer to Table 12.3-4 for a listing of CAMs that are

a. Stand-alone operation only

b. Normally are operated in stand-alone mode but can communicate with central control terminal.

11.5.2.3 Radiation Monitor Calibration, Maintenance, Inspection, Decontamination, and Replacement

11.5.2.3.1 Calibration

All process and effluent radiation monitors are initially and periodically calibrated to an NBS traceable source. Operational checks are performed by check source actuation and/or substitution of an electronic pulse generator for the detector. Special calibration factors are determined by comparison of grab sample laboratory analyses data to the monitor data.

11.5.2.3.2 Maintenance

Periodic preventive maintenance is performed in accordance with the equipment manufacturer's recommendations. All maintenance is performed in accordance with the manufacturer's requirements. Monitors are recalibrated whenever maintenance which can affect their calibration is performed.

11.5.2.3.3 Inspection

Process and effluent radiation monitors are located in accessible areas. Periodic inspection and routine surveillance of the monitors is thus readily accomplished.

11.5.2.3.4 Decontamination and Replacement

All radiation monitors described.in Subsections 11.5.2.1.7, 11.5.2.1.8 and 11.5.2.2.1 through 11.5.2.2.9 are continuous offline sampling types with isolation valves provided at the sample and return point and on each monitor. The monitors may thus be decontaminated or replaced without opening the process system. Flushing or purging capability is also provided. Detectors, filters, sample chambers, and valves on the monitors may similarly be replaced.

11.5.3 Effluent Monitoring and Sampling

General Design Criterion 64 requirements for monitoring all effluent discharge paths for radioactivity that may be released from normal operations, anticipated operational occurrences, and from postulated accidents are met.

There are two gaseous effluent discharge paths: the common station HVAC exhaust vent and the standby gas treatment vent. Monitors are provided for each path as described in Subsections 11.5.2.2.3 and 11.5.2.1.8 for the HVAC exhaust vent and Subsections 11.5.2.2.4 and 11.5.2.1.7 for the standby gas treatment vent.

There are two liquid effluent discharge paths: seal well discharges to the discharge flume and shutdown service water system discharges to the ultimate heat sink. The plant service water discharges to the seal well are monitored as described in Subsection 11.5.2.2.5. The shutdown service water discharges are monitored as described in Subsections 11.5.2.2.7 and 11.5.2.2.8.

Sampling frequencies are described in the Offsite Dose Calculation Manual (ODCM).

11.5.4 Process Monitoring and Sampling

The requirements of General Design Criteria 60 for control of radioactive releases to the environment are met. The following sources of radioactive releases have automatic isolation valves as described in the referenced sections:

a. containment ventilation exhaust (Subsections 11.5.2.1.2, 11.5.2.1.3, and 11.5.2.1.4),

b. fuel building exhaust (Subsection 11.5.2.1.5),

c. post treatment air ejector off-gas (Subsection 11.5.2.2.2),

d. liquid radwaste discharge (Subsection 11.5.2.2.6), and

e. main steamline (Subsection 11.5.2.1.1).

General Design Criterion 63 requirements for monitoring radioactive waste process systems and associated handling areas are met.

Process and effluent monitors for the gaseous systems are described in Subsections 11.5.2.2.1 through 11.5.2.2.3.

The process and effluent monitor for the liquid systems are described in Subsection 11.5.2.2.5 through 11.5.2.2.9. Sampling of the liquid system is described in Subsection 9.3.2.

Area radiation monitors and continuous airborne radioactivity monitors, used in areas where radwaste is handled, are described in Subsection 12.3.4.

TABLE 11.5-1
PROCESS RADIATION MONITORING SYSTEMS

	MONITORED PROCESS	NUMBER OF SYSTEMS**	NUMBER OF CHANNELS	DETECTOR TYPE	DETECTOR LOCATION	UPSCALE SETPOINT: ALERT ALARM	UPSCALE SETPOINT: HIGH ALARM/ TRIP	UPSCALE SETPOINT: SCALE
A.	SAFETY RELATED SYSTEMS							
	Main Steamline	2	1	Gamma Sensitive Ionization Chamber	Immediately downstream of outboardmain steamline isolation valve	Above full power background, below trip	ORM	6 decade log
	Containment Building Fuel Transfer Ventilation Plenum	4	1	G-M Tube	Exhaust duct upstream of connection to containment exhaust	Above background, below trip	ORM	Digital
	Containment Building Vent Exhaust	4	1	G-M Tube	Exhaust duct downstream of outboard containment isolation valve	Above background, below trip	ORM	Digital
	Continuous Containment Purge Exhaust	4	1	G-M Tube	Exhaust duct downstream of outboard containment isolation valve	Above background, below trip	ORM	Digital
	Fuel Building Vent Exhaust	4	1	G-M Tube	Exhaust duct upstream of exhaust ventilation isolation valve	Above background below trip	ORM	Digital
	Main Control Room Air Intake	4	1	G-M Tube	Vicinity of the control room minimum outside air intakes	Above background, below trip	Technical specification	Digital

TABLE 11.5-1 (Cont'd)

MONITORED PROCESS	NUMBER OF SYSTEMS**	NUMBER OF CHANNELS	DETECTOR TYPE	DETECTOR LOCATION	UPSCALE SETPOINT ALERT ALARM	UPSCALE SETPOINT HIGH ALARM/ TRIP	SCALE
SGTS Exhaust High Range Monitor*	2	1	G-M Tube	Sample Line	Not Applicable	Not Applicable	Digital
Common Station HVAC Vent High Range Monitor*	2	1	G-M Tube	Sample Line	Not Applicable	Not Applicable	Digital
B. <u>SYSTEM REQUIRED FOR PLANT OPERATION</u>							
Pre-Treatment Offgas	1	1 Gross gamma	G-M Tube	Sample Line	Above design basis	ODCM	Digital
Post-Treatment Offgas	1/ 1 spare	2 Noble gas	Beta Scintillator G-M Tube	Sample Line	Above design basis Below HIGH alarm	ODCM	Digital
		1 Particulate	Beta Scintillator		Variable	Variable	
		1 Iodine	Gamma Scintillator		Variable	Variable	
Common Station HVAC Exhaust	1/ 1 spare	2 Noble gas	Beta Scintillator G-M Tube	Sample Line	10CFR50, App1	ODCM	Digital
		1 Particulate	Beta Scintillator		Variable	Variable	
		1 Iodine	Gamma Scintillator		Variable	Variable	
SGTS Exhaust	1/ 1 spare	2 Noble gas	Beta Scintillator G-M Tube	Sample Line	10CFR50, App 1	ODCM	Digital
		1 Particulate	Beta Scintillator		Variable	Variable	

TABLE 11.5-1 (Cont'd)

MONITORED PROCESS	NUMBER OF SYSTEMS**	NUMBER OF CHANNELS	DETECTOR TYPE	DETECTOR LOCATION	UPSCALE SETPOINT ALERT ALARM	UPSCALE SETPOINT HIGH ALARM/ TRIP	SCALE
		1 Iodine	Gamma Scintillator		Variable	Variable	
Plant Service Water Effluent	1	1 Liquid	Gamma Scintillator	Sample Line	Above bkgd. Below HIGH alarm	ODCM	Digital
Liquid Radwaste Discharge	1	1 Liquid	Gamma Scintillator	Sample Line	Above bkgd. Below HIGH alarm	ODCM	Digital
Shutdown Service Water Effluent	2	1 Liquid	Gamma Scintillator	Sample Line	Above bkgd. Below HIGH alarm	ODCM	Digital
Fuel Pool HX Service Water	2	1 Liquid	Gamma Scintillator	Sample Line	Above bkgd. Below HIGH alarm	ODCM	Digital
Component Cooling Water	1	1 Liquid	Gamma Scintillator	Sample Line	Above bkgd. Below HIGH alarm	ODCM	Digital

* This system was procured and designed as a safety-related system in order to provide a monitor which meets and/or exceeds Category 2 requirements of Reg. Guide 1.97. This system is not redundant and is not designed to mitigate the consequences of an accident.

** Channels provided for background measurement and subtraction with some of the monitors not listed here.

TABLE 11.5-2
GASEOUS AND AIRBORNE PROCESS RADIATION MONITORING SYSTEM

MONITOR	CONFIGURATION	TYPE	NOMINAL RANGE	PRINCIPAL RADIOACTIVITY MEASUREMENT	BASES FOR RANGE	ALARMS AND TRIP
Main Stream line Radiation Monitor	Adjacent to steamlines	γ-ion Chamber	10^0-10^6 mR/hr	Coolant activation gases and fission products	Steamline acti-vity defined in Table 12.2-4	INOP, High, High-High
Containment Building Fuel Transfer Vent Plenum	On-Line	G-M Tube	0.1 to $2.2x10^3$ mR/hr	Noble gases	On scale reading for normal; adequate margin above high setpoint	Fail, Alert, High, Trend
Containment Building Exhaust	On-Line	G-M Tube	0.1 to $2.2x10^3$ mR/hr	Noble gases	On Scale reading for normal; ade-quate margin above high setpoint	Fail, Alert, High, Trend
Continuous Containment Purge Exhaust	On-Line	G-M Tube	0.1 to $2.2x10^3$ mR/hr	Noble gases	On Scale reading for normal; ade-quate margin above high setpoint	Fail, Alert, High, Trend
Fuel Building Vent Exhaust	On-Line	G-M Tube	0.1 to $2.2x10^3$ mR/hr	Noble gases	On Scale reading for normal; ade-quate margin above high setpoint	Fail, Alert, High, Trend
Control Room Air Intake	On-Line	G-M Tube	0.1 to $2.2x10^3$ mR/hr	Noble gases	On Scale reading for normal; ade-quate margin above high setpoint	Fail, Alert, High, Trend
Pretreatment Air Ejector Off-Gas	Off-Line	G-M Tube	5.5 x 10^{-2} to 5.2 x 10^2 µCi/cc (Note 1)	Noble gases	Off-Gas activity defined in Chapter 11	Fail, Alert, High, Trend
Post Treatment Air Ejector Off-Gas	Off-Line					
Particulate		Beta Scintilation	$2.6x10^{-11}$ to $1.2x10^{-6}$ Sr-90/Y-90 µCi/cc (Note 5)	Gross Beta	Low end below DAC adequate margin above high setpoint	Fail, Alert, High, Trend

TABLE 11.5-2
GASEOUS AND AIRBORNE PROCESS RADIATION MONITORING SYSTEM (Continued)

MONITOR	CONFIGURATION	TYPE	NOMINAL RANGE	PRINCIPAL RADIOACTIVITY MEASUREMENT	BASES FOR RANGE	ALARMS AND TRIP
Iodine		Scintillation (NaI)	7.33×10^{-11} to 2.97×10^{-6} I-131 µCi/cc (Note 5)	I131	Low end below DAC adequate margin above high setpoint	Fail, Alert, High, Trend
Low Range Noble Gas		Beta Scintillation	5.8×10^{-7} to 2.6×10^{-2} µCi/cc (Note 2)	Gross Beta	Low end below 2 x DAC, adequate margin above high setpoint	Fail, Alert, High, Trend
High Range Noble Gas		G-M Tube	5.6×10^{-3} to 3.0×10^{2} µCi/cc (Note 2)	Gross Gamma	Range covers carbon bed failure with adequate margin	Fail, Alert, High, Trend
Common Station HVAC Vent	Off-Line					
Particulate		Beta Scintillation	8.1×10^{-11} to 1.2×10^{-6} Sr-90 µCi/cc (Note 5)	Gross Beta	On scale reading for normal; adequate margin above high setpoint	Fail, Alert, High, Trend
Iodine		Scintillation (NaI)	9.19×10^{-11} to 3.61×10^{-6} I-131 µCi/cc (Note 5)	I-131	On scale reading for normal; adequate margin above high setpoint	Fail, Alert, High, Trend
Low Range Noble Gas		Beta Scintillation	6.4×10^{-7} to 2.8×10^{-2} µCi/cc (Note 3)	Gross Beta	On scale reading for normal; adequate margin above high setpoint	Fail, Alert, High, Trend
High Range Noble Gas		G-M Tube	5.0×10^{-4} to 2.6×10^{1} µCi/cc (Note 3)	Gross Gamma	Abnormal releases	Fail, Alert, High, Trend
Standby Gas Treatment Exhaust	Off-Line					

TABLE 11.5-2
GASEOUS AND AIRBORNE PROCESS RADIATION MONITORING SYSTEM (Continued)

MONITOR	CONFIGURATION	TYPE	NOMINAL RANGE	PRINCIPAL RADIOACTIVITY MEASUREMENT	BASES FOR RANGE	ALARMS AND TRIP
Particulate		Beta Scintillation	8.1×10^{-11} to 1.2×10^{-6} Sr-90 µCi/cc (Note 5)	Gross Beta	Including CPS ODCM limit	Fail Alert, High, Trend,
Iodine		Scintillation (NaI)	9.19×10^{-11} to 3.61×10^{-6} I-131 µCi/cc (Note 5)	I131	On scale reading for normal; adequate margin above high setpoint	Fail, Alert, High, Trend
Low Range Noble Gas		Beta Scintillation	6.3×10^{-7} to 2.8×10^{-2} µCi/cc (Note 3)	Gross Beta	On scale reading for normal; adequate margin above high setpoint	Fail, Alert, High, Trend
High Range Noble Gas		G-M Tube	4.5×10^{-4} to 2.3×10^{1} µCi/cc (Note 3)	Gross Gamma	Abnormal releases	Fail, Alert, High, Trend
SGTS Exhaust High Range Monitor	Off-Line					
Intermediate Range Noble Gas		G-M Tube	1.3×10^{-4} to 1.4×10^{1} Xe-133-equivalent µCi/cc (Notes 3,4)	Gross Gamma	RG 1.97, NUREG-0737	Fail, Alert, High, Trend
High Range Noble Gas		G-M Tube	2.4×10^{-1} to 1.0×10^{5} Xe-133-equivalent µCi/cc (Notes 3,4)	Gross Gamma	RG 1.97, NUREG-0737	Fail, Alert, High, Trend
Common Station HVAC Vent High Range Monitor	Off-Line					

TABLE 11.5-2
GASEOUS AND AIRBORNE PROCESS RADIATION MONITORING SYSTEM (Continued)

MONITOR	CONFIGURATION	TYPE	NOMINAL RANGE	PRINCIPAL RADIOACTIVITY MEASUREMENT	BASES FOR RANGE	ALARMS AND TRIP
Intermediate Range Noble Gas		G-M Tube	$1.3x10^{-4}$ to $1.4x10^{1}$ Xe-133-equivalent µCi/cc (Notes 3,4)	Gross Gamma	RG 1.97, NUREG-0737	Fail, Alert, High, Trend
High Range Noble Gas		G-M Tube	$2.4x10^{-1}$ to $1.0x10^{5}$ Xe-133-equivalent µCi/c (Notes 3,4)	Gross Gamma	RG 1.97, NUREG-0737	Fail, Alert, High, Trend

NOTES:

1. Based on the off-gas mix with 2.91 minute decay, which is applicable to the monitor location. The pre-treatment monitor setpoint is also based on this mix, however, the setpoint value is based on an equivalent 30 minute old off-gas activity concentration of 289 mCi/sec per the CPS Technical Specifications.

2. Based on the design basis off-gas mix released from the treatment system, Table 11.3-1.

3. Based on the composition of total annual expected gaseous releases given in Table 11.3-9. A background of 0.5 mR/hr is assumed in determining the lower limit of the range.

4. For additional information, see Note 17 of Table 7.1-13.

5. The range is based on a one-hour deposition on filter.

TABLE 11.5-3
LIQUID PROCESS RADIATION MONITORING SYSTEM

MONITOR	CONFIGURATION	TYPE	RANGE	PRINCIPAL RADIOACTIVITY MEASUREMENT	BASES FOR RANGE	ALARMS
Shutdown Service Water	Off-Line	Gamma Scintillation (NaI)	2.231×10^{-7} to 3.277×10^{-3} I-131* µCi/cc	Gross Gamma	On scale reading for normal; adequate margin above high setpoint	Fail, Alert, High, Trend
Fuel Pool Hx Service Water	Off-Line	Gamma Scintillation (NaI)	2.231×10^{-7} to 3.277×10^{-3} I-131* µCi/cc	Gross Gamma	On scale reading for normal; adequate margin above high setpoint	Fail, Alert, High, Trend
Component Cooling Water	Off-Line	Gamma Scintillation (NaI)	2.231×10^{-7} to 3.277×10^{-3} I-131* µCi/cc	Gross Gamma	On scale reading for normal; adequate margin above high setpoint	Fail, Alert, High, Trend
Liquid Radwaste Discharge	Off-Line	Gamma Scintillation (NaI)	2.231×10^{-7} to 3.277×10^{-3} I-131* µCi/cc	Gross Gamma	On scale reading for normal; adequate margin above high setpoint.	Fail Alert, High, Trend
Plant Service Water Effluent	Off-Line	Gamma Scintillation (NaI)	2.231×10^{-7} to 3.277×10^{-3} I-131* µCi/cc	Gross Gamma	On scale reading for normal; adequate margin above high setpoint.	Fail, Alert, High, Trend

* Range based upon this radionuclide; it varies depending on specific radionuclide.

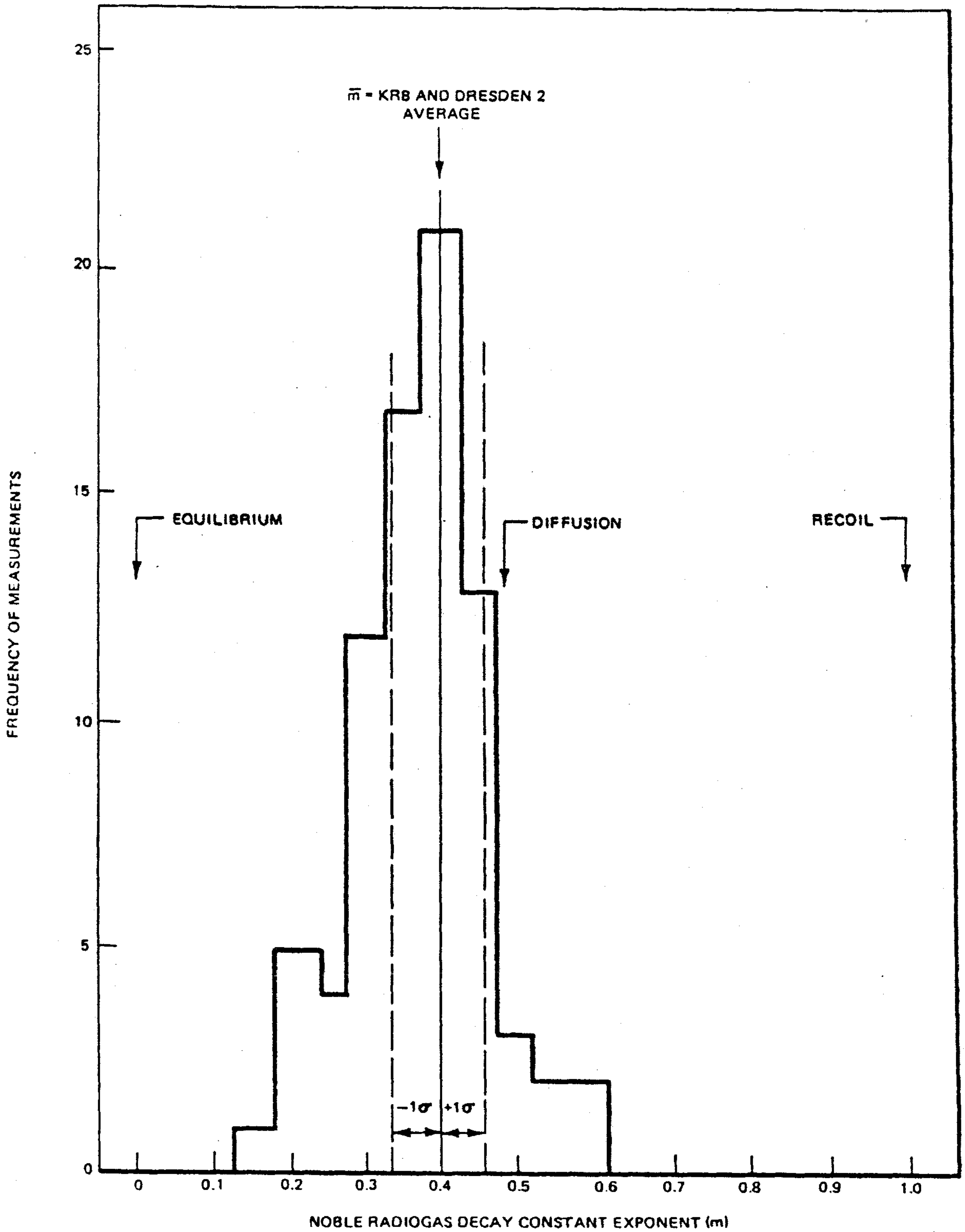

Figure 11.1-1. Noble Radiogas Decay Constant Exponent Frequency Histogram

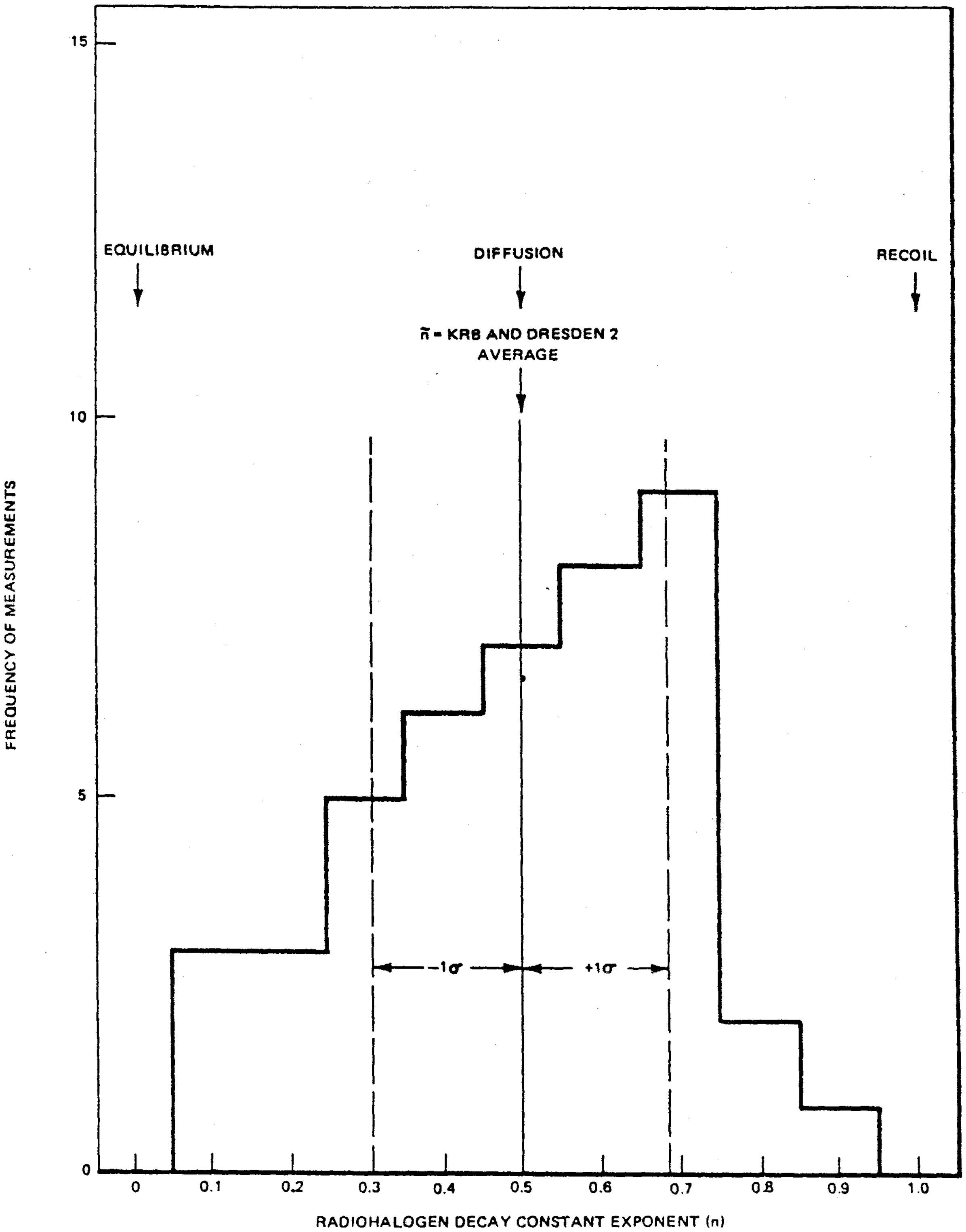

Figure 11.1-2. Radiohalogen Decay Constant Exponent Frequency Histogram

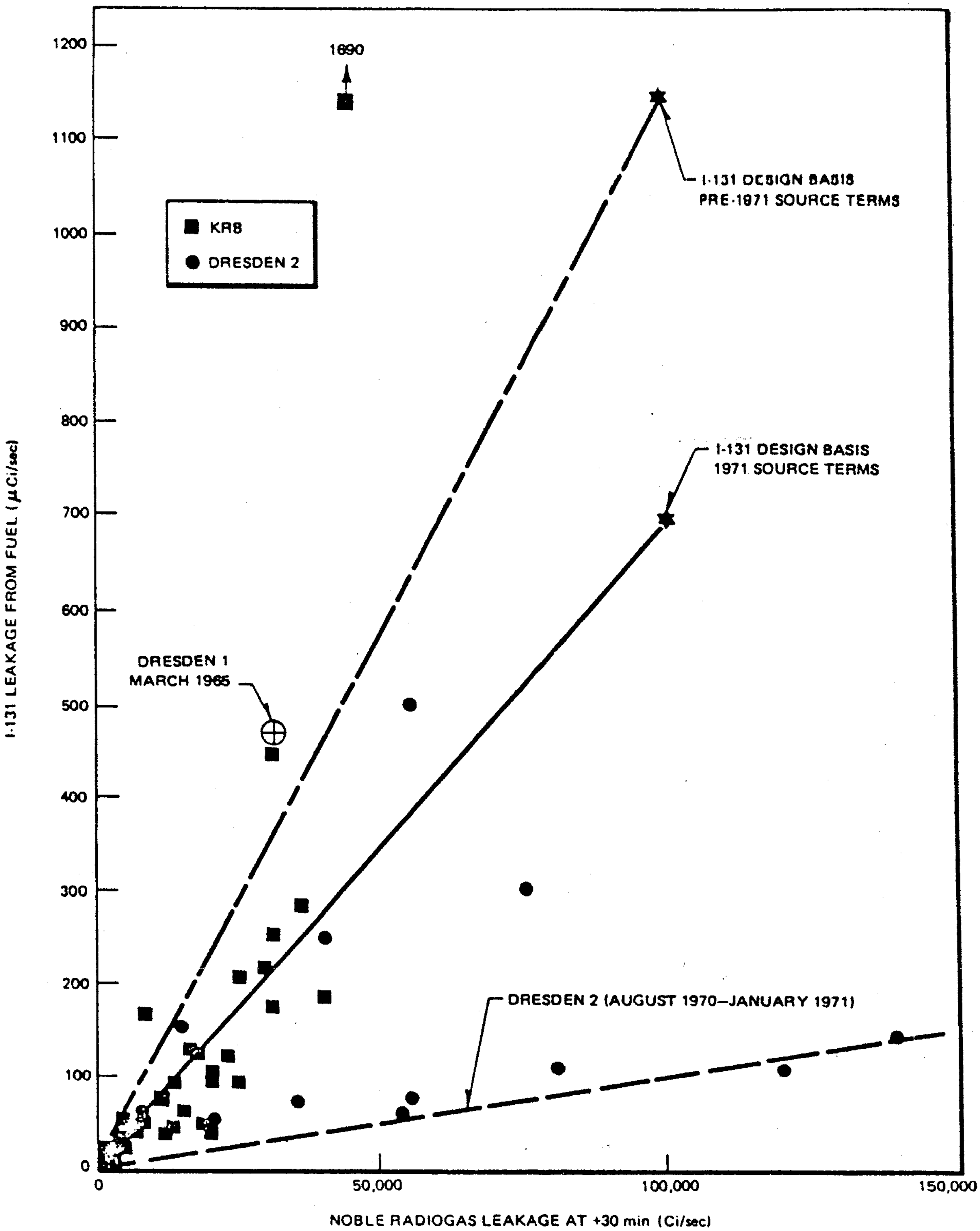

Figure 11.1-3. Noble Radiogas Leakage versus I-131 Leakage

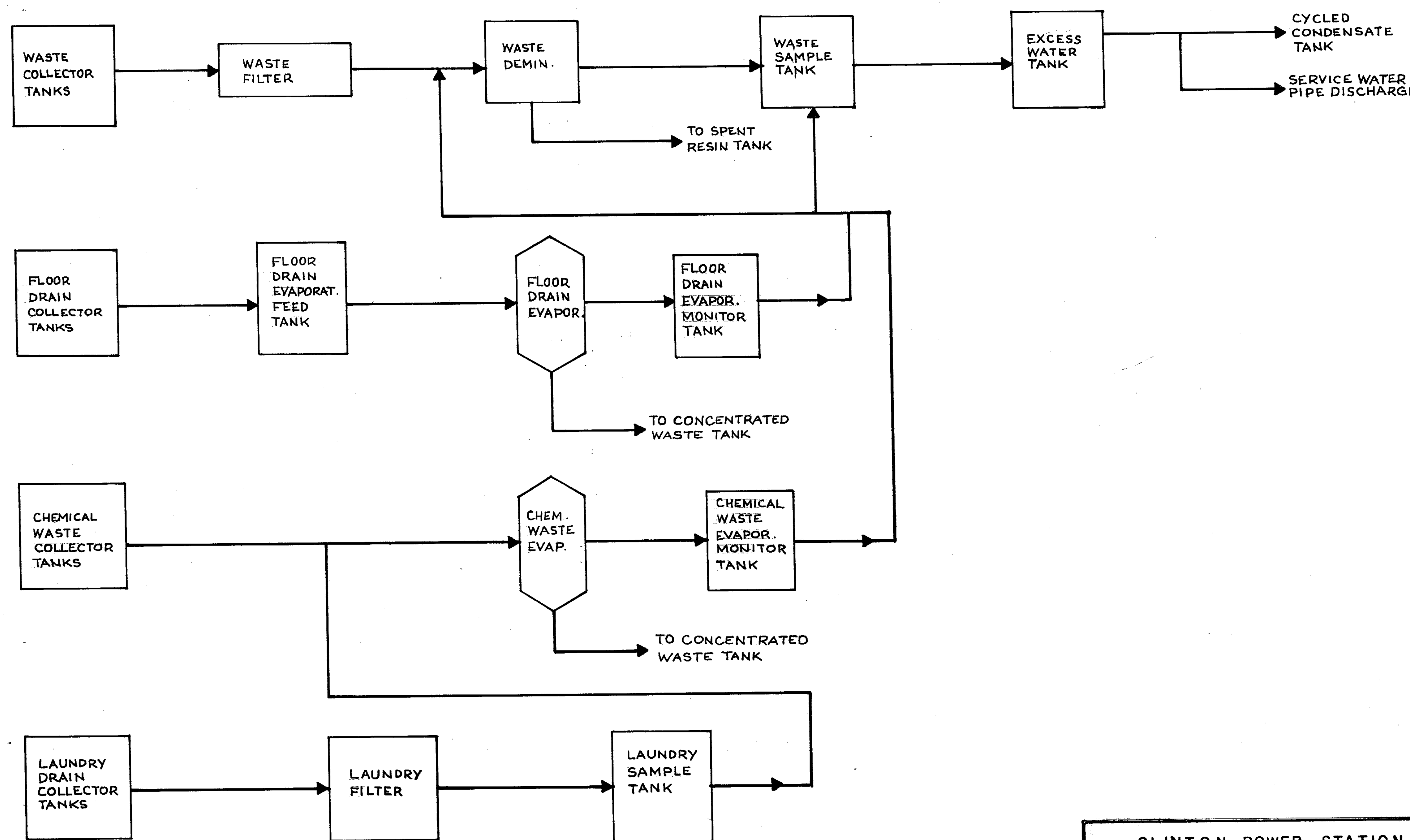

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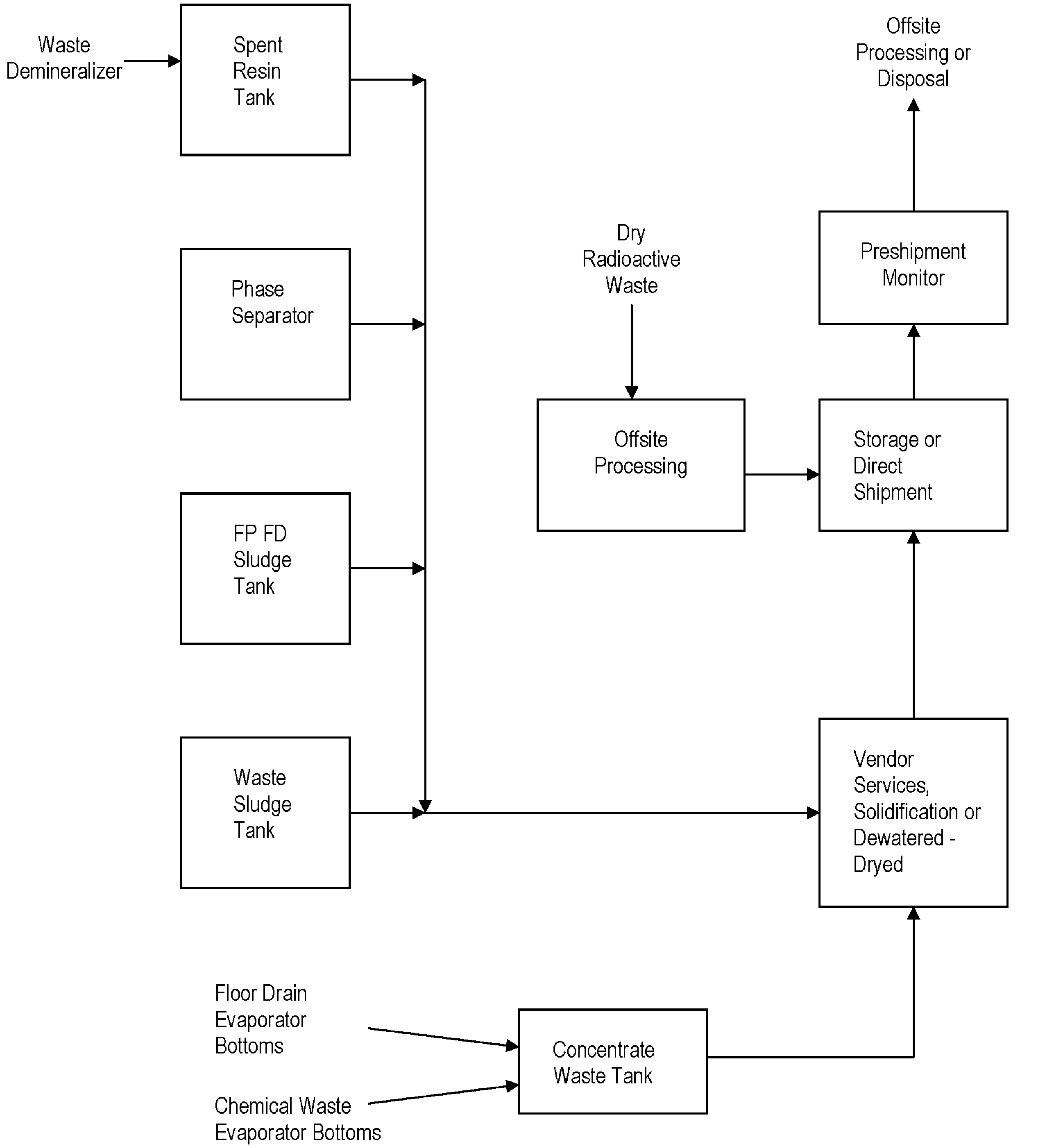
FIGURE 11.2-1

LIQUID RADWASTE SYSTEM
SIMPLIFIED FLOW DIAGRAM



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SOLID RADWASTE SYSTEM
SIMPLIFIED FLOW DIAGRAM

FIGURE 11.4-1