TABLE OF CONTENTS

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

TABLE OF CONTENTS (Continued)

LIST OF TABLES

LIST OF TABLES (Continued)

LIST OF TABLES (Continued)

LIST OF TABLES (Continued)

11.0 RADIOACTIVE WASTE MANAGEMENT

11.1 SOURCE TERMS

General Electric has evaluated radioactive material sources (activation and fission product releases from fuel) in operating boiling water reactors (BWRs) over the past decade. These source terms are reviewed and periodically revised to incorporate up-to-date information. Release of radioactive material from operating BWRs has resulted in doses to offsite persons which have been only a small fraction of <10 CFR 20>, or of natural background dose. (Radiological assessments performed prior to October 4, 1993 that were used for the plant design bases as discussed in this USAR were evaluated against the <10 CFR 20> regulations prior to October 4, 1993. Radiological assessments for plant design bases modifications that are performed after October 4, 1993 will be evaluated using the revised <10 CFR 20> dated October 4, 1993.)

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, non-coolant activation products and fission products. The fission product levels are based on measurements of BWR reactor water and offgas at several stations through mid-1971. Emphasis was placed on observations made at KRB (in the Republic of Germany) 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.

e. Evaluation of radioactive material releases to the environment.

For halogens, radioisotopes with half-lives of less than three minutes were omitted. For other fission product radioisotopes in reactor water, radioisotopes with half-lives of 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 BWRs are generally complex mixtures. Their 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:

$$\text{Equilibrium:} \quad R_g \sim k_1 y \qquad (11.1\text{-}1)$$

$$\text{Recoil:} \quad R_g \sim k_2 y \lambda \qquad (11.1\text{-}2)$$

The nomenclature in <Section 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: $$R_g \sim k_3 y \lambda^{0.5} \qquad (11.1\text{-}3)$$

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 has been used by GE as a basis for design since 1963, the design basis release magnitude used has varied from 0.5 Ci/sec to 0.1 Ci/sec as measured after 30 minute decay (t = 30 min). The noble radiogas source term rate after 30 minute 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 minute offgas holdup system used on a number of plants. Since approximately 1967, the design basis release magnitude used (including the 1971 source terms) has been established at an annual average of 0.1 Ci/sec (t = 30 min). 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 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 min) can be tolerated for reasonable periods of time. Continual assessment of these values is made on the basis of actual operating experience in BWRs (Reference 2).

While the noble radiogas source term magnitude was established at 0.1 Ci/sec (t = 30 min), it was recognized that there may be a more statistically applicable distribution for the noble radiogas mixture. Sufficient data were available from KRB operations from 1967 to mid-1971 along with Dresden 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:

$$R_g = K_g y \lambda^m \ (1 - e^{-\lambda t}) \ (e^{-\lambda t}) \qquad (11.1\text{-}4)$$

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 $(1 - e^{-\lambda T})$ approaches 1 and can be neglected when the reactor has been operating at a 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 equal to 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 are 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 min) and with radiogas leakage at Dresden 2 varying from 0.001 to 0.169 Ci/sec (t = 30 min), 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 equal to 0.1 to m equal to 0.6. After establishing the value of m equal to 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 equal to 30 minutes. With ΣR_g at 30 minutes equal to 100,000 µCi/sec, K_g can be calculated as being 2.6×10^7 and Equation 11.1-4 becomes:

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

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.

Out of the thirteen commonly considered noble gases, normal operational releases to the primary coolant are expected to be approximately 25,000 µCi/sec as evaluated at 30 minutes, and 100 µCi/sec of I-131. These values can be compared to the design base value of 100,000 µCi/sec for the summation of the same thirteen 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 similar fashion, 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 are 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 value for $\bar{n}$ 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 equal to 0.1 to n equal to 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 of August 1970 to January 1971 that there is a relationship between noble radiogas and I-131 leakage 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 were not experienced. <Figure 11.1-3> shows that these higher radioiodine leakages from the fuel were related to noble radiogas source terms of less than the design basis value of

0.1 Ci/sec (t = 30 min). This may be partially explained by inherent limitations due to internal plant operational problems that caused plant derating.

In general, 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 min). 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:

$$R_h = 2.4 \times 10^7 \quad y\lambda^{0.5} \quad (1 - e^{-\lambda T}) \quad (e^{-\lambda t}) \qquad (11.1\text{-}9)$$

The concentrations of the radiohalogens in the reactor water are calculated by modeling the plant's piping network as "line sections" or streams which connect flow junctions. Each line section is described as to the mass flow rate, steam quality, gas and liquid specific volumes, mass inventory, type of unit operations occurring, and the nature of flow for radioactive decay calculations. With this information, along with the plant nuclear and chemical data, the radioactive material transport performance of each line can be expressed mathematically. The resulting system of linear equations is then solved to obtain the required concentrations.

Although carryover of most soluble radioisotopes from reactor water to steam is observed to be less than 0.1 percent (<0.001 fraction), the

observed “carryover” for radiohalogens has varied from 0.1 percent to about 2 percent on newer plants. The average of observed radiohalogen carryover measurements has been 1.2 percent by weight of reactor water in steam, with a standard deviation of ±0.9. In the present source term definition, a radiohalogen carryover of 2 percent (0.02 fraction) was used.

The halogen release rate from the fuel can be calculated from Equation 11.1-9. 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 BWRs 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 less than 0.1 percent (<0.001 fraction). In addition to carryover, however, decay of noble radiogases in the steam leaving the reactor results in production of noble gas daughter radioisotopes in the steam and condensate systems.

Some daughter radioisotopes (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. 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^{-1}).
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.

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 <Table 11.1-5>.

11.1.2.2 Non-coolant 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 non-coolant activation products have been estimated conservatively from experience data. The resultant concentrations are presented in <Table 11.1-6>. Carry-over of these isotopes from the reactor water to the steam is estimated to be less than 0.1 percent (<0.001 fraction).

The effect of operating above the design basis zinc source term due to zinc injection was evaluated. The evaluation determined the effects to be negligible; therefore, the references and tables concerning the zinc concentrations, calculated MPC levels, and doses were not updated as a result of implementing the zinc injection process.

11.1.2.3 Steam and Power Conversion System N-16 Inventory

Steam and power conversion system N-16 inventories are given in <Section 12.2.1>.

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.

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 is present due to the H(n,γ) D reaction), thermal neutron flux, and microscopic cross section for deuterium.

The fraction of tritium produced by fission which may transfer from fuel to the coolant (which is then available for release in liquid and gaseous effluents) is more difficult to estimate. However, since

zircaloy-clad fuel rods are used in BWRs, essentially all fission product tritium remains in the fuel rods unless defects are present in the cladding material (Reference 3).

The study made at Dresden 1 in 1968 by the U.S. Public Health Service (USPHS) suggests that essentially all of the tritium released from the plant could be accounted for by the deuterium activation source (Reference 4). 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:

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

Where:

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

y = Fission yield fraction (atoms/fission)

λ = Radioactive decay constant (sec^{-1})

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 half-life similar to that of tritium) was 0.06 to 0.4 times that calculated using the "diffusion mixture" relationship. This would suggest that the actual tritium leakage rate might range from 0.015 to 0.10 μCi/sec. Since the annual average noble radiogas leakage from a BWR is expected to be less than

0.1 Ci/sec (t = 30 min), the annual average tritium release rate from the fission source can be conservatively estimated at 0.12±0.12 μCi/sec, 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 is the same as in the reactor water at any given time. This tritium concentration is also present in condensate and feedwater. Since radioactive effluents generally originate from the reactor and power cycle equipment, radioactive effluents also have this tritium concentration. 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 has a common tritium concentration.

Offgases released from the plant contain tritium, which is present as tritiated gas (HT) resulting from reactor water radiolysis, as well as 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, tanks and spills on floors also contain tritium. The remainder of the tritium leaves the plant in liquid effluents or with solid wastes.

Recombination of radiolytic gases in the air ejector offgas system forms 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 results in a slightly higher tritium concentration in the plant process water. Reducing the amount of liquid effluent discharged also results in a higher process coolant equilibrium tritium concentration.

Essentially, all tritium in the primary coolant is eventually released to the environs, either as water vapor and gas to the atmosphere, or as liquid effluent to the plant discharge 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 percent of the tritium release was observed in liquid effluent, with the remaining 10 percent leaving as gaseous effluent (Reference 4). Efforts to reduce the volume of liquid effluent discharges may change this distribution so that a greater amount of tritium leaves as gaseous effluent. From a practical standpoint, the fraction of tritium leaving as liquid effluent may vary between 60 and 90 percent, with the remainder leaving in gaseous effluent.

11.1.4 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 <Chapter 15>.

11.1.4.2 Fuel Experience

A discussion of BWR fuel experience, including fuel failure experience, burnup experience and thermal conditions under which the experience was gained, is presented in (Reference 5), (Reference 6), (Reference 7), and (Reference 8).

11.1.5 PROCESS LEAKAGE SOURCES

Process leakage results in potential release paths for noble gases and other volatile fission products through ventilation systems. Liquid from process leaks is collected and routed to the liquid-solid radwaste system. Radionuclide releases through ventilation paths are at extremely low levels and have been insignificant compared to process offgas from operating BWR plants. However, because the implementation of improved process offgas treatment systems makes the ventilation release relatively significant, GE has conducted measurements to identify and qualify these low level release paths. GE has maintained an awareness of other measurements by the Electric Power Research Institute and other organizations and routine measurements by utilities with operating BWRs.

Leakage of fluids from the process system results in the release of radionuclides into plant buildings. In general, the noble radiogases remain airborne and are released to the atmosphere with little delay through the building ventilation exhaust ducts. The radionuclides partition between air and water, and airborne radioiodines may "plate out" 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 are also present in the ventilation exhaust air.

The estimated release rate of radioactive materials in gaseous effluents is presented in <Section 11.3.3>.

11.1.6 LIQUID RADWASTE SYSTEM

Radioactive sources for the liquid radwaste system are described in <Section 11.2.3> and are based on information contained in <NUREG-0016> (Reference 9).

11.1.7 RADIOACTIVE SOURCES IN THE GAS TREATMENT SYSTEM

Radioactive sources for the gas treatment system are described in <Section 11.3.2.1.2>.

11.1.8 SOURCE TERMS FOR COMPONENT FAILURES

11.1.8.1 Offgas System Failure

Source terms for evaluation of the radiological consequences of component failures within the offgas system are contained in <Table 15.7-3A> for the Design Basis and Normal (realistic) operating conditions.

11.1.8.2 Liquid Radwaste System

Radiation sources used for component failures are consistent with an offgas release rate of 100,000 μCi/sec after 30 minutes decay. This results in maximum inventories of radioisotopes in the system and is not anticipated to occur during operation of the plant. The isotopic breakdown of the inventory in each significant component of the liquid radwaste system is presented in <Table 15.7-12>.

11.1.9 REFERENCES FOR SECTION 11.1

1. Brutschy, F. J., "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. Skarpelos, J. M. and R. S. Gilbert, "Technical Derivation of BWR 1971 Design Basis Radioactive Material Source Terms," General Electric Company, NEDO-10871, March 1975.

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

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

5. Williamson, H. E., Ditmore, D. C., "Experience with BWR Fuel Through September 1971," General Electric Company, NEDO-10505, May 1972. (Update)

6. Elkins, R. B., "Experience with BWR Fuel Through September 1974," General Electric Company, NEDO-20922, June 1975.

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

8. Elkins, R. B., "Experience with BWR Fuel Through December 1976," General Electric Company, NEDO-21660, July 1977.

9. U.S. Nuclear Regulatory Commission, "Calculation of Releases of Radioactive Material in Gaseous and Liquid Effluents from Boiling Water Reactors (BWR-GALE Code)," <NUREG-0016>, April 1976.

TABLE 11.1-1

NOBLE RADIOGAS SOURCE TERMS

Isotope	Half-Life	Source Term t=0 (μCi/sec)	Source Term t=30 (μCi/sec)
Kr-83m	1.86 Hr	3.4 +3	2.9 +3
85m	4.4 Hr	6.1 +3	5.6 +3
85	10.74 Yr	10 to 20[1]	10 to 20[1]
87	76 Min	2.0 +4	1.5 +4
88	2.79 Hr	2.0 +4	1.8 +4
89	3.18 Min	1.3 +5	1.8 +2
90	32.3 Sec	2.8 +5	-
91	8.6 Sec	3.3 +5	-
92	1.84 Sec	3.3 +5	-
93	1.29 Sec	9.3 +4	-
94	1.0 Sec	2.3 +4	-
95	0.5 Sec	2.1 +3	-
97	1.0 Sec	1.4 +1	-
Xe-131m	11.96 Day	1.5 +1	1.5 +1
133m	2.26 Day	2.9 +2	2.8 +2
133	5.27 Day	8.2 +3	8.2 +3
135	9.16 Hr	2.2 +4	2.2 +4
135m	15.7 Min	2.6 +4	6.9 +3
137	3.82 Min	1.5 +5	6.7 +2
138	14.2 Min	8.9 +4	2.1 +4
139	40 Sec	2.8 +5	-

TABLE 11.1-1 (Continued)

Isotope	Half-Life	Source Term t=0 (μCi/sec)	Source Term t=30 (μCi/sec)
Xe-140	13.6 Sec	3.0 +5	-
141	1.72 Sec	2.4 +5	-
142	1.22 Sec	7.3 +4	-
143	0.96 Sec	1.2 +4	-
144	9.0 Sec	5.6 +2	-
Total		Approx. 2.5 +6	Approx. 1.0 +5

NOTE:

(1) Estimated from experimental observations.

TABLE 11.1-2

POWER ISOLATION EVENT - ANTICIPATED OCCURRENCE

Isotope	Isotopic Spiking Activity (Ci)/Bundle
I-131	2.1
132	3.3
133	5.1
134	5.5
135	4.9
Kr-83m	0.9
85m	2.2
85	0.5
87	4.4
88	5.2
89	8.2
Xe-131m	0.1
133m	0.3
133	11.8
135m	1.8
135	11.2
137	10.7
138	10.8

TABLE 11.1-3

HALOGEN RADIOISOTOPES IN REACTOR WATER

Isotope	Half-Life	Concentration (μCi/g)
Br-83	2.40 hr	1.4×10^{-2}
84	31.8 min	3.0×10^{-2}
85	3.0 min	1.9×10^{-2}
I-131	8.065 day	1.2×10^{-2}
132	2.284 hr	1.2×10^{-1}
133	20.8 hr	8.3×10^{-2}
134	52.3 min	2.6×10^{-1}
135	6.7 hr	1.2×10^{-1}

TABLE 11.1-4

OTHER FISSION PRODUCT RADIOISOTOPES IN REACTOR WATER

Isotope	Half-Life	Concentration (μCi/g)
Sr-89	50.8 day	2.82×10^{-3}
Sr-90	28.9 yr	2.09×10^{-4}
Sr-91	9.67 hr	6.90×10^{-2}
Sr-92	2.69 hr	1.15×10^{-1}
Zr-95	65.5 day	3.13×10^{-5}
Zr-97	16.8 hr	3.03×10^{-5}
Nb-95	35.1 day	3.76×10^{-5}
Mo-99	66.6 hr	2.09×10^{-2}
Tc-99m	6.007 hr	7.95×10^{-2}
Tc-101	14.2 min	1.11×10^{-1}
Ru-103	39.8 day	1.78×10^{-5}
Ru-106	368 day	2.40×10^{-6}
Te-129m	34.1 day	6.27×10^{-5}
Te-132	78.0 hr	1.25×10^{-2}
Cs-134	2.06 yr	1.46×10^{-4}
Cs-136	13.0 day	9.61×10^{-5}
Cs-137	30.2 yr	2.19×10^{-4}
Cs-138	32.3 min	2.20×10^{-1}
Ba-139	83.2 min	1.78×10^{-1}
Ba-140	12.8 day	8.04×10^{-3}

TABLE 11.1-4 (Continued)

Isotope	Half-Life	Concentration (μCi/g)
Ba-141	18.3 min	2.10×10^{-1}
Ba-142	10.7 min	1.99×10^{-1}
Ce-141	32.53 day	3.55×10^{-5}
Ce-143	33.0 hr	3.24×10^{-5}
Ce-144	284.4 day	3.13×10^{-5}
Pr-143	13.58 day	3.45×10^{-5}
Nd-147	11.06 day	1.25×10^{-5}
Np-239	2.35 day	2.19×10^{-1}

TABLE 11.1-5

COOLANT ACTIVATION PRODUCTS IN REACTOR WATER AND STEAM

Isotope	Half-Life	Steam Concentration (μCi/g)	Reactor Water Concentration (μCi/g)
N-13	9.99 Min	1.5 -3	1.0 -1
N-16	7.13 Sec	5.0 +1	4.0 +1
N-17	4.14 Sec	4.0 -2	2.0 -2
O-19	26.8 Sec	7.7 -1	1.8 +0
F-18	109.8 Min	4.4 -4	4.2 -2

TABLE 11.1-6

NON-COOLANT ACTIVATION PRODUCTS IN REACTOR WATER

Isotope	Half-Life	Concentration (μCi/g)
Na-24	15.0 Hr	2.1×10^{-3}
P-32	14.31 Day	2.1×10^{-5}
Cr-51	27.8 Day	5.2×10^{-4}
Mn-54	313.0 Day	4.2×10^{-5}
Mn-56	2.582 Hr	5.2×10^{-2}
Co-58	71.4 Day	5.2×10^{-3}
Co-60	5.258 Yr	5.2×10^{-4}
Fe-59	45.0 Day	8.4×10^{-5}
Ni-65	2.55 Hr	3.1×10^{-4}
Zn-65	243.7 Day	2.1×10^{-5}
Zn-69m	13.7 Hr	3.1×10^{-5}
Ag-110m	253.0 Day	6.3×10^{-5}
W-187	23.9 Hr	3.1×10^{-3}

11.2 LIQUID WASTE MANAGEMENT SYSTEMS

11.2.1 DESIGN BASES

11.2.1.1 Power Generation Design Objectives

The liquid radioactive waste (LRW) system is designed to collect and treat, for reuse or disposal, all radioactive (or potentially radioactive) liquid wastes produced in the plant. This is done in such a manner that, for all anticipated quantities of waste produced, the availability of the plant for power generation is not adversely affected.

11.2.1.2 Radiological Design Objectives

The LRW system is designed to restrict releases of radioactive material to the environment and exposures to both operating personnel and the general public to "as low as reasonably achievable" (ALARA) in accordance with the guidelines given in <10 CFR 50, Appendix I>.

11.2.1.3 Design Criteria

The original LRW system is designed in accordance with the following design criteria:

a. For each reactor at the site, the estimated annual total quantity of radioactive material (excluding tritium) above background in the liquid effluents released to unrestricted areas is less than 5 curies.

b. For the total radioactive liquid effluents, the resultant whole body dose to any individual offsite is less than 5 mrem/yr.

c. Design and construction of all LRW system components satisfies or exceeds the intent of all applicable criteria set forth in <Regulatory Guide 1.143> (as detailed in <Section 1.8>) and ANSI N197-1976.

d. All LRW system components and the structure in which they are housed are designed and constructed in accordance with the codes, standards, seismic classifications, and safety classifications listed in <Table 3.2-1>.

e. LRW components are located in areas of sufficient size and accessibility to facilitate efficient maintenance.

11.2.1.4 Cost-Benefit Analysis

<Section 11.2.3> includes an analysis that shows that the LRW system, as designed, is capable of controlling releases of radioactive material within the numerical design objectives of <10 CFR 50, Appendix I>. Under the rules of Section II, Paragraph D of <10 CFR 50, Appendix I> a cost-benefit analysis is not required for this system because the design satisfies the "Guides on Design Objectives for Light-Water-Cooled Nuclear Power Reactors," proposed in the "Concluding Statement of Position of the Regulatory Staff," in Docket-RM-50-2.

11.2.1.5 Accident Analysis

An analysis is included in <Chapter 15> to determine the radiological consequences for case situations in which equipment malfunctions and/or operator errors are hypothesized during periods of operation at design basis fuel leakage. Design provisions are included to prevent the uncontrolled release of radioactive material to the environment as a

result of any single equipment malfunction or operator error. An evaluation of failures of single pieces of equipment is provided by <Table 11.2-1>.

11.2.1.6 Component Design Parameters

With the exception of normal wearing parts, such as seals and bearings, all pumps, valves, piping, tanks, pressure vessels, and other components in the LRW system are fabricated from materials which are intended to provide a minimum service life of 40 years without replacement. In selecting materials to satisfy this criterion, due consideration is given to the following:

a. The corrosive nature of both the process fluid and the external environment.

b. Ease with which the material may be decontaminated.

c. Wall thickness requirements dictated by design pressures, temperatures, flow rates, and corrosion rates.

Tabulations of LRW system components and design parameters are presented by <Table 11.2-2>, <Table 11.2-3>, <Table 11.2-4>, <Table 11.2-5>, <Table 11.2-6>, and <Table 11.2-7>.

11.2.1.7 Surge Input Collection Capabilities

Redundancy of equipment for collecting and processing inputs to the LRW system is described in detail in <Section 11.2.2> and summarized by <Table 11.2-8>. Considerable excess capacity is built into the collecting and processing equipment of each subsystem to handle all anticipated normal and maximum input quantities. An evaluation of this capability is presented by <Table 11.2-9> which shows that only when the suppression pool is drained for maintenance does the LRW system fall short of needed capacity. This occurrence is satisfactorily handled by reducing the rate at which the suppression pool is drained. This method adds, at most, three days to the outage and results in no increase in radiation exposures to operating personnel or the general public.

11.2.1.8 Control of Tank Leakage and Overflows

With the exception of the condensate storage tank, all tanks containing radioactive material are housed inside reinforced, concrete structures with floor drains for routing tank leakage to the LRW system. A seismically qualified retaining structure (dike) surrounds the condensate storage tank to contain any leakage from this source. Further information on these dikes is provided in <Section 9.2.6>.

All tanks in the LRW and solid radioactive waste (SRW) systems have overflow lines piped up solid to embedded drain piping that is routed to sumps. Any water collected in these sumps is pumped back to the LRW collection tanks. Any vents and manways for the tanks are located above these overflow lines. Should an operator error result in an overflow through a vent or manway, the release would also be pumped back to the LRW collection tanks, since, as noted above, the tanks are housed in structures with floor drains that route tank leakage to the LRW system.

For each tank in the LRW and SRW systems, level indication and high level alarms (with the exception of those associated with the radwaste evaporators and concentrated waste tanks) are provided in the radwaste building control room. <Table 11.2-16> lists tanks outside containment which may contain potentially radioactive fluids.

11.2.1.9 ALARA Design Features

Numerous features have been incorporated into the design of both the LRW system and the building housing this system to ensure that exposures of operating personnel to radiation will be kept within ALARA guidelines. The following is a listing of the most significant ALARA design features:

a. All floors and wall areas subject to contamination with radioactive material are coated with nuclear grade epoxy coatings to aid in decontamination.

b. With the exception of the detergent drains tanks and chemical waste distillate tanks, which are low in activity content, all redundant tanks are located in separate, shielded cubicles. This allows one tank to be repaired or inspected with minimal personnel exposure while the second tank is being used to process waste.

c. Most redundant pumps and process equipment are located in separate, shielded cubicles similar to those for the tanks as described above.

d. All normal operations are performed remotely from the centralized radwaste building control room. This eliminates exposures to operating personnel during normal operation and minimizes operating errors that could indirectly result in greater exposures to both operating and maintenance personnel.

e. Pipe lines containing radioactive fluids are routed through shielded chases. There is no instrumentation, valves or other equipment located in these chases, eliminating the need to enter them for any reason other than to maintain the piping itself.

f. As much as possible, pipes containing filter backwash slurries or spent resins make use of bends rather than standard elbow fittings to reduce the chances of plugging. As another precaution against sludge buildup, these lines are backflushed after every use.

g. Backflush connections are provided on all process piping in pump cubicles to permit this piping to be decontaminated before entering the cubicles for maintenance purposes.

h. All pump seals are mechanical type to minimize seal leakage and to eliminate the need for periodic adjustment of the seals. Rotating element facing material is chosen to maximize seal operating life.

i. The majority of valves are top-entry, diaphragm type with ethylene propylene terpolymer (EPT) elastomer diaphragms. This type of valve has the following advantages: a) no crud traps; b) no leakage unless the diaphragm fails; and c) quick, simple procedures for replacement of worn seals (diaphragm).

j. Materials of construction for pumps, valves, piping, tanks, and process equipment are selected to provide long-term corrosion resistance and improved decontamination capability. Most pumps, tanks and other process equipment are constructed of austenitic stainless steel. Where protection against chloride stress corrosion is needed, materials such as Alloy 20 stainless steel, and Incoloy are used. Piping and valves are constructed of materials such as austenitic stainless steel, Alloy 20 stainless steel, Incoloy, Yoloy or suitably lined with materials such as polypropylene or nuclear grade epoxy coatings. Other corrosion resistant materials may also be used.

11.2.1.10 Control of Inadvertent Releases

Releases as a result of equipment failures or malfunctions are discussed in <Section 11.2.1.5>. Another way in which unintentional releases could occur would be as a result of operator errors either allowing a tank to overflow or pumping the contents of the wrong tank to the discharge tunnel entrance structure. Provisions for control of tank overflows are discussed in <Section 11.2.1.8>. Provisions for preventing the contents of the wrong tank from being discharged are discussed below.

All LRW system discharges are directed to the Unit 1 emergency service water discharge pipe. All pipe lines going to the discharge point are routed through one central discharge flow control station, where the liquid can be directed through a flow control station. The flow control valve is remote-manually adjusted from the radwaste building control room. Between the control valve station and each sample tank that can be discharged is a power operated shutoff valve that must be opened before a tank can actually be drained to the discharge point.

As protection against inadvertent discharges, an administratively controlled, manual, normally locked closed valve with position indicating limit switches is provided in series with each discharge isolation valve.

11.2.2 SYSTEM DESCRIPTION

11.2.2.1 Input Streams

The LRW system is designed and sized to handle all radioactive liquid wastes for the Perry Nuclear Power Plant, based on having a condensate polishing treatment system as discussed in <Section 10.4.6>.

The input streams for the system are shown on the detailed process flow diagram in <Figure 11.2-1 (1)>, <Figure 11.2-1 (2)>, <Figure 11.2-1(3)>, and <Figure 11.2-1 (4)>. For these streams, normal and expected maximum quantities of significant radioactive nuclides and total flow quantities are given in <Table 11.2-10>.

11.2.2.2 Separation of Inputs

Incoming streams of liquid waste are collected and treated in one of four separate process streams according to their composition. These four subdivisions are high purity/low conductivity wastes (primarily

equipment drains), medium-to-low purity/medium conductivity wastes (primarily floor drains), high conductivity chemical wastes, and detergent drains.

In addition to handling these four categories of liquid waste, the LRW system collects spent resin slurries and filter backwash slurries prior to being sent to the SRW disposal system.

11.2.2.3 Previous Experience

The type of process equipment used in the system described herein has been used effectively in many previous BWR units, including Dresden Units 1, 2 and 3, Quad Cities Units 1 and 2, Oyster Creek, and Nine Mile Point. Justification for the decontamination factors used for this equipment is based on available data from several operating units, equipment manufacturer's data, topical reports, and standards given in (Reference 1), (Reference 2), (Reference 3), (Reference 4), (Reference 5), (Reference 6), (Reference 7), (Reference 8), (Reference 9), (Reference 10), (Reference 11), (Reference 12), (Reference 13), (Reference 14), (Reference 15), (Reference 16), (Reference 17), (Reference 18), (Reference 19), (Reference 20), (Reference 21), (Reference 22), (Reference 23), (Reference 24), and (Reference 25).

11.2.2.4 Treatment of High Purity/Low Conductivity Wastes

Input streams to this subsystem consist of equipment drains, cask pit drawdown, suppression pool water (normally diverted to suppression pool cleanup system), blowdown of reactor water (normally directed to hotwell), rinse water from condensate demineralizers, and residual heat removal system flush/test. These inputs are collected in one of two waste collector tanks, each sized to hold one day's maximum normal input. With the exception of equipment drains, these waste streams can be diverted to the floor drain collector tanks if water quality or flow

conditions warrant. After a batch of waste is collected, it is sent through a traveling belt filter to remove suspended solids, and then a mixed-bed demineralizer to remove dissolved solids. Alternate flow paths for treatment of these wastes are discussed in <Section 11.2.2.13>. Two waste sample tanks, each sized to hold one batch of waste, are provided for sampling, mixing and temporary storage of the treated effluent. After a batch is sampled, it may be recycled to the waste collector tank for further treatment, sent to the condenser Hotwell (normal path), the condensate storage system or discharged. The system is completely redundant, either through backup equipment or cross-ties with identical equipment in one of the other subsystems.

The major inputs to the high purity subsystem are equipment drains. The embedded drainage piping system for collecting this waste water is described in <Section 9.3.3>. The equipment drain piping in each structure housing radioactive (or potentially radioactive) fluid systems is routed to a sump located at the lowest elevation of the building. After one of these sumps is filled, one of two redundant, vertical sump pumps automatically pumps the contents to the waste collector tanks in the LRW system.

11.2.2.5 Treatment of Medium-to-Low Purity/Medium Conductivity Wastes

Input streams to this subsystem consist of floor drains, decantate from the backwash settling tanks, decantate from the solid radwaste disposal system and backwash from the radwaste and condensate demineralizers. These inputs are collected in one of two floor drain collector tanks, each sized to hold approximately three days' maximum normal input. With the exception of floor drains, these waste streams can be diverted to the waste collector tanks if water quality or flow conditions warrant. After a batch is collected, it is normally filtered, demineralized and re-used. Alternate flow paths for treatment of these wastes are discussed in <Section 11.2.2.13>. Two floor drain sample tanks, each

sized to hold one batch of waste, are provided for sampling and temporary storage of treated effluent. After sampling, a batch is either recycled for further treatment, sent to condenser hotwell (normal path), the condensate storage system or discharged. The system is completely redundant, either through backup equipment or cross-ties with identical equipment in one of the other subsystems.

The major inputs to the medium-to-low purity subsystem are floor drains, which consist of miscellaneous unidentified equipment leakage and floor washdown. The embedded drainage piping system for collecting this waste water is described in <Section 9.3.3>. The floor drain piping in each structure housing radioactive (or potentially radioactive) fluid systems is routed to a sump located at the lowest elevation of the building. After one of these sumps is filled, one of two redundant, vertical sump pumps automatically sends the contents to the floor drain collector tanks in the LRW system.

11.2.2.6 (Deleted)

Flow paths for treatment of these wastes are discussed in <Section 11.2.2.13>.

11.2.2.7 Treatment of Detergent Drains

Inputs to this subsystem consist of personnel decontamination solutions and floor drains from nonradioactive areas of the control complex. Cleaning of protective clothing will be performed onsite and/or contracted offsite. All waste inputs are collected in the laundry and floor drains sump located at the lowest elevation of the control complex. When this sump is filled, one of two redundant sump pumps automatically transfers the contents to the LRW system detergent drains tanks. This waste is then collected and manually drained to the Radwaste Floor Drain so that it can be processed.

11.2.2.8 Treatment of Spent Resins

Spent resins from the mixed-bed condensate demineralizers, waste demineralizer, floor drains demineralizer, and suppression pool

demineralizer are collected in two spent resin storage tanks. Each tank is sized to hold the resins for six months. The spent resins are transferred to the SRW disposal system as a water slurry.

11.2.2.9 Treatment of Filter/Demineralizer Backwash

Backwash slurries from the condensate filter, fuel pool filter/demineralizer and RWCU filter/demineralizer backwash receiving tanks are pumped to settling tanks located in the radwaste building. The sludge is allowed to settle to the bottom of these tanks while relatively clean water is drawn off the top and pumped to the floor drain collector tanks or waste collector tanks for further treatment. Periodically, the sludge is transferred to the SRW disposal system as a water slurry.

11.2.2.10 Detailed Component Design

Piping and instrumentation for the LRW system are shown in <Figure 11.2-1>. For a definition of symbols used on this system diagram, see <Figure 1.2-22>. Design data for all LRW system components is given in <Table 11.2-2>, <Table 11.2-3>, <Table 11.2-4>, <Table 11.2-5>, <Table 11.2-6>, and <Table 11.2-7>. The safety class for equipment and piping in the system is given in <Table 3.2-1>. Also shown in this table are the seismic classifications and principal construction codes for LRW system components and for the radwaste building.

a. Collection Tank Design

All collection tanks are atmospheric, cylindrical, stainless steel tanks and are either horizontal or vertical. Vertical tanks have closed tops and dished bottoms for easy drainage. Vent, overflow, recycle, and drain lines are provided for each tank. A level

sensor is provided on each tank for remote level indication, level recording and alarm/control functions.

b. Pump Design

LRW pumps other than sump pumps are horizontal, centrifugal type, driven by 460 volt drip proof motors. Each pump is provided with inlet and outlet shutoff valves for maintenance and a discharge pressure sensor with readout in the radwaste building control room (RWBCR). All pump seals are single or double mechanical type. In addition, filter aid pumps (waste collector and floor drain collector) are positive displacement.

c. Waste Collector Filter/Floor Drains Filter

Each filter is a flatbed, continuous belt type, precoat filter unit rated at 100 to 150 gpm when used to filter waste water. Each unit can also be used to dewater resin or filter backwash slurries, for which case the process rate is 50 gpm.

For improved filtration efficiency, provisions are made for body feed of precoat material to the filter influent. During periods of non-use, water is continuously recirculated through the filter to prevent deterioration of the filter precoat.

Upon completion of a filtration run, the precoat material and accumulated crud is partially dried by air and the filter belt is indexed, causing the semi-dry cake to fall off the end of the belt and down a stainless steel chute into a waste mixing/dewatering tank in the SRW disposal system.

Each filter has a filtration surface area of 68 square feet. Operating differential pressure varies from 2 to 13 psi. Design differential pressure is 15 psi.

d. Waste Demineralizer/Floor Drains Demineralizer

These demineralizers are identical 200 gpm mixed bed units, using a mixture of cation and anion resins. Each demineralizer is designed for a process flow rate of 6 to 8 gpm per square foot. They are cross-tied by manual valves to achieve redundancy in both subsystems.

Maximum pressure differential at rated flow is 22.5 psi. A demineralizer run may be terminated on a high differential pressure or a high conductivity signal. The spent resin is then transferred to one of the spent resin tanks.

e. (Deleted)

f. (Deleted)

g. Settling Tanks

All settling tanks are vertical, atmospheric, cylindrical, stainless steel tanks with closed tops and dished bottoms. Each tank is provided with vent, overflow, drain, recycle, and decant lines. Manways are provided on the settling tanks, which are located above the overflow line. The manways may be left open to support operational practices. Connections for flushwater and sparging air or condensate are also provided.

Four ultrasonic level indicators are provided on each tank to indicate in the RWBCR when the sludge level is at 25, 50, 75, or 100 percent of the maximum permissible level. The tanks are designed so that this maximum level is below the elevation of the decant lines. Each tank is also provided with a liquid level sensor for remote level indication and alarm/control functions.

h. Spent Resin Tanks

Two vertical, atmospheric, cylindrical, stainless steel spent resin tanks are provided. Each tank has a closed top and dished bottom and is provided with vent, drain, overflow, recycle, and flush lines. The entrance to the overflow line is provided with a wire mesh screen to prevent resins from entering the overflow.

Each tank is provided with a liquid level sensor for remote level indication and alarm/control functions.

i. Concentrated Waste Tanks

Two vertical, atmospheric, cylindrical, Incoloy concentrated waste tanks are provided. Each tank has a closed top, dished bottom and vent, overflow, drain, and recycle lines. All lines normally containing concentrated waste are heat traced and insulated to prevent solidification of the concentrate. Each tank is provided with a heating element to maintain the tank temperature between 120°F and 150°F. A level sensor is provided for remote indication and control functions. Temperature elements are provided to monitor and record temperature in the tanks.

j. Sample Tanks

The waste sample tanks and floor drains sample tanks are vertical, atmospheric, cylindrical, stainless steel tanks. The chemical waste distillate tanks are horizontal, atmospheric, cylindrical, stainless steel tanks. All tanks have vent, overflow, drain, and recycle lines. Each tank has a level sensor for remote level indication, level recording and alarm/control functions.

k. Sumps

Radioactive floor and equipment drains are collected in sumps located in the basement of all structures housing radioactive fluid systems. These sumps range in size from 50 to 1,000 gallons. With the exception of those sumps that are normally nonradioactive, all sumps are lined with stainless steel for leakage control and to facilitate decontamination.

Many sumps are provided with a small recessed “boot” in the area of the bottom from which the sump pump takes suction. This ensures that the pump suction is submerged at all times while allowing most of the sump to be drained completely to minimize buildup of radioactive sludge and to facilitate decontamination.

All sumps are covered with grating or solid plates. Solid plates are used where shielding is needed, or if the sump is in an open area where litter could end up in the sump.

Each sump is provided with level switches for alarm and control functions. Sumps inside containment have additional instrumentation for leak rate detection as discussed in <Section 7.6>. The quantity of waste water sent to the LRW system from each sump is monitored in the RWBCR.

l. Sump Pumps

Except for the annulus sump, which is expected to be used very infrequently, all sumps have redundant, duplex sump pumps. Vertical turbine pumps are used in sumps containing relatively clean water. Standard vertical, open impeller, centrifugal sump pumps are used in sumps where trash could accumulate. All sump pumps are provided with suction strainers to prevent refuse from clogging or damaging the pump impeller.

Pump motors are totally enclosed and fan cooled to prevent contamination of the motor internals.

11.2.2.11 Field Routed Pipe

Routing of piping and tubing in the LRW system that normally carry radioactive fluids is shown on piping drawings to ensure proper

protection of operating personnel against exposure to radiation. Therefore, there will be no field routed radioactive piping or tubing for which shielding design criteria or controls will be necessary.

11.2.2.12 System Control and Operating Procedures

a. General

Control and indication of equipment associated with the radwaste evaporators is on control panel H51-P031 in the RWBCR. Control and indication of all other radwaste sub-systems is via the distributed control system (DCS) in the RWBCR. The DCS operator's console (1H51P0510) consists of operator workstations used for control and indication of the LRW system and a maintenance/engineering workstation. Various display screens shown on monitors at the operator's console provide the operating status (off/on) of system pumps and the position (open/closed) of power operated valves. Important system parameters such as tank levels, pump discharge pressures, etc. are also indicated and/or recorded on this operator's console. An alarm sounds at the console if abnormal conditions such as high tank level or high discharge activity should occur.

Additional control panels are located near the radwaste filters and demineralizers for use when reconditioning this equipment. After a filter has received a fresh precoat or a demineralizer has been refilled with new resin, control of this equipment is returned to the radwaste building control room.

b. Distributed Control System

The control logic for the LRW system is controlled by a distributed control system. Redundant power sources are provided to the DCS. Separate logic is provided for train A and train B components to the extent possible. The DCS has two pairs of redundant processors.

The DCS features an automatic and a manual mode, either of which can be used to control the LRW system.

c. Normal Control of Discharges

Except for detergent wastes, all liquid effluents from the LRW system are normally routed to the condensate storage system or main condenser for reuse in the plant. This is done on a batch basis after a sample of the effluent is taken to determine if it is suitable for reuse. If the sample does not meet the water quality standards for condensate makeup given in <Table 11.2-11> the batch is either recycled for further treatment or discharged through the discharge tunnel entrance structure, depending on the chemical content and activity level.

All streams to be discharged, with the exception of the atmospheric drain line from the Turbine Building supply plenums are routed through one central flow control station, where a flow control valve is used. These valves are modulated remote-manually from the RWBCR to achieve the desired flow rate. The stream is then monitored for gross gamma activity and routed to the discharge tunnel entrance structure, which discharges to the environment at the point shown in <Figure 1.2-18>.

For each batch discharged, the activity monitor is set to actuate an alarm in the RWBCR if the activity level exceeds a preselected value. This value is calculated for each batch based on the activity level of a sample taken from the batch and on the flow rate of the dilution flow at the time that it is desired to discharge the batch. The value is set so that after dilution, the concentration will be substantially below the limits as defined by <10 CFR 20>. (Radiological assessments performed prior to October 4, 1993 that were used for the plant design bases as discussed in this USAR were evaluated against the <10 CFR 20> regulations prior to October 4, 1993. Radiological assessments for plant design bases modifications that are performed after October 4, 1993 will be evaluated using the revised <10 CFR 20> dated October 4, 1993.)

11.2.2.13 Selection of Normal and Alternate Flow Paths

Normal flow paths for all input streams to the liquid radwaste system are described in <Section 11.2.2.4>, <Section 11.2.2.5>, <Section 11.2.2.6>, <Section 11.2.2.7>, <Section 11.2.2.8>, and <Section 11.2.2.9>. However, because of the variable nature of these input streams, alternate flow paths for their treatment may sometimes be necessary. In <Figure 11.2-2>, the normal and alternate flow paths for each input stream are summarized. For each flow path, the percentages of total flow are given for expected normal operation, design and sizing of equipment and calculation of quantities of radioactivity discharged. Explanation of each flow path used is given in <Table 11.2-12>.

Water discharged from the atmospheric drain line from the Turbine Building supply plenums will be periodically monitored with grab samples. This source is from a radiologically clean area but does have the potential to condense radioactive tritium that has been recycled back into this plenum from the plant gaseous vents. Detectable tritium in this pathway can also be from naturally occurring tritium production from cosmic radiation. The fluids condensed in this pathway have already been evaluated for compliance with limits defined by <10 CFR 20> via the pathway analysis for gaseous vents.

11.2.2.14 Performance Tests

Prior to plant startup, all equipment in the radwaste system will be tested for operability.

Reports in the literature on performance tests for this equipment are given in (Reference 1), (Reference 2), (Reference 3), (Reference 4), (Reference 5), (Reference 6), (Reference 7), (Reference 8), (Reference 9), (Reference 10), (Reference 11), (Reference 12), (Reference 13), (Reference 14), (Reference 15), (Reference 16), (Reference 17), (Reference 18), (Reference 19), (Reference 20),

(Reference 21), (Reference 22), (Reference 23), (Reference 24), and (Reference 25).

11.2.3 RADIOACTIVE RELEASES

11.2.3.1 Description

The criteria for recycle, treatment and discharge of radioactive wastes is discussed in <Section 11.2.2>. In calculating the radioactive releases to the environment it was assumed that 10 percent of the high purity, chemical waste streams and 25 percent of the low purity waste stream are discharged.

11.2.3.2 Dilution Factors

The liquid waste discharged to the environment from LRW systems is diluted by the service water and/or ESW of Unit 1.

Values in <Table 11.2-13> were calculated using the normal minimum dilution flow of 30,000 gpm. During certain operating conditions or certain seasons, flows may be less than the normal minimum flows indicated. However, flows will exceed the normal minimum flow a substantial portion of the year, thus the values in <Table 11.2-13> are a valid conservative estimate of annual discharges.

11.2.3.3 Release Points

Releases to the environment are by way of the discharge tunnel entrance structure or to the plant storm drains for the discharge from the atmospheric drain line on the Turbine Building supply plenums. The discharge tunnel entrance structure is shown on the process flow diagram in <Figure 11.2-2> and the site plot plan in <Figure 1.2-18>.

11.2.3.4 Estimated Releases

The release rate of radioactive materials in liquid effluents is presented in <Table 11.2-13> and <Table 11.2-14>. These values were calculated with the GALE Code and are based on the assumptions and parameters provided in <NUREG-0016> (BWR-GALE Code) and <Table 11.2-15>. As shown in <Table 11.2-13>, the estimated releases are a small fraction of the limits as defined by <10 CFR 20>, and are considered as low as reasonably achievable. (Radiological assessments performed prior to October 4, 1993 that were used for the plant design bases as discussed in this USAR were evaluated against the <10 CFR 20> regulations prior to October 4, 1993. Radiological assessments for plant design bases modifications that are performed after October 4, 1993 will be evaluated using the revised <10 CFR 20> dated October 4, 1993.) The estimated offsite doses for the Perry site and a comparison with the design objectives of <10 CFR 50, Appendix I> and the dose limits of <40 CFR 190> are presented in <Section 5.2.4> of the PNPP Environmental Report.

Subsequent to the original evaluation discussed above, modifications have been made to the condensate cleanup and liquid radwaste systems which could result in liquid effluents which are different in quantity or activity concentration than those predicted in the original analysis. The control of the liquid radwaste system effluents, as described in <Section 11.2.2.12>, remains unchanged and ensures that liquid effluent releases remain within the design objectives of <10 CFR 50, Appendix I> and substantially below the limits as defined by <10 CFR 20>.

The release of condensate from the atmospheric drain line from the Turbine Building supply plenums may also contain radioactive tritium due to the recycle of released air from the gaseous effluent vents. The release of radioactive tritium in the gaseous vents is analyzed in <Section 11.3>.

11.2.4 REFERENCES FOR SECTION 11.2

1. "BWR Radwaste System Performance," H. L. Loy and W. F. Dietrich, Symposium on Waste Management at Nuclear Reactors, AiChE National Meeting, Cincinnati, Ohio, May 16, 1971, NEDM-10346.

2. "Design and Project of Radioactive Waste Treatment and Disposal System at Large BWR Nuclear Power Stations," A. Shimozoto, M. Takeshima, and K. Osano, Hitachi Review, Volume 19, No. 12, pp 426-434, UDC 621.039.7.

3. "Management of Radioactive Wastes at Nuclear Power Stations:" J. O. Blomeke, et. al., ORNL-4070.

4. “Experience With Precoat Filters in Nuclear Power Plants,” C. H. Becker and J. S. Brown, presented at 34th Annual American Power Conference, Chicago, Illinois, April 18 - 20, 1972.

5. Test of AMF 2 gpm Radioactive Waste Evaporator at Rochester Ginna Power Station - AMF-BEAIRD Engineering Calculation, Dated October 8, 1971.

6. “Removal of Radionuclides from Water by Water Treatment Processes,” Ray J. Morton and Conrad P. Stroub, American Water Work Association Journal, Volume 48, May 1956, p. 545.

7. “Disposal of Radioactive Wastes”, Abel Walman, Ibid., Volume 49, May 1957, p. 505.

8. “Mixed-Bed Ion Exchange for the Removal of Radioactivity,” H. Gladys Swope, Op. Cit., Volume 49, August 1957, p. 1085.

9. “Chemical Coagulation Studies on Removal of Radioactivity in Waters,” Lloyd R. Setter and Helen H. Russell, Op. Cit., Volume 50, May 1958, p. 590.

10. “Decontamination of Radioactive Sea Water,” Minorce Honma and Allen E. Greendale, Op. Cit., Volume 50, November 1958, p. 1490.

11. “Removal of Radionuclides from the Pasco Supply by Conventional Treatment,” Robert L. Junkins, Op. Cit., Volume 52, July 1960, p. 834.

12. “Behavior of Radionuclides on Ion Exchange Resins,” Dade W. Moellor, George W. Leddicatte and Sam A. Reynolds, Op. Cit., Volume 53, July 1961, p. 862.

13. “Effect of Detergents on the Decontamination of Radioactive Waste Water by Precipitation and Coagulation,” Colin R. Frost, Op. Cit., Volume 54, September 1962, p. 1082.

14. “Reducing Radioisotope Concentrations in Reactor Effluent by High Coagulant Feed,” Wyatt B. Silker, Op. Cit., Volume 55, March 1963, p. 355.

15. “Decontamination of Radioactive Aqueous Effluent by the Calcium-Ferric Phosphate Process,” Colin R. Frost, Op. Cit., Volume 55, October 1963, p. 1237.

16. “Studies on Radioisotope Removal by Water Treatment Processes,” Rolf Eliassen, Warren J. Kaufman, John B. Nesbitt and Morton I. Goldman, Op. Cit., Volume 43, August 1951, p. 615.

17. “Studies on Radioisotope Removal by Water Treatment Processes;” “Discussion,” Arthur E. Gorman, Op. Cit., Volume 43, August 1951, p. 630.

18. “Studies on Radioisotope Removal by Water Treatment Processes;” “Discussion,” Arthur E. Gorman, Op. Cit., Volume 43, August 1951, p. 633.

19. “Studies on Radioisotope Removal by Water Treatment Processes;” “Authors’ Closure;” Op. Cit., Volume 43, August 1951, p. 635.

20. “Removal of Radioactive Material from Water By Slurrying With Powdered Metal,” William J. Lacy, Op. Cit., Volume 44, September 1952, p. 824.

21. “Reduction of Radioactivity in Water,” Sol Goodgal, Earnest F. Gloyna and Dayton E. Carritt, Op. Cit., Volume 46, January 1954, p. 66.

22. "Accelerating Calcium Carbonate Precipitation of Softening Plants," Robert F. McCauley and Rolf Eliassen, Op. Cit., Volume 47, May 1955, p. 487.

23. Radiological Surveillance Studies at a Boiling Water Nuclear Power Reactor, U.S. Department of Health, Education and Welfare, BRH/DER 70-1, March 1970.

24. Ibid., BRH/DER 70-2, March 1970.

25. Management of Radioactive Wastes at Nuclear Power Plant, IAEA, Vienna, 1968, STI/PUB/208.

TABLE 11.2-1

SINGLE EQUIPMENT ITEM MALFUNCTION EVALUATION

Equipment Item	Malfunction	Consequences	Design Precautions
Discharge flow control valve	Does not respond to signal to throttle flow	Radioactive isotope concentration in discharge could exceed limits of <10 CFR 20>.	Radiation monitor downstream of the flow control valve signals flow control valve to close. Monitor has high radiation alarm to alert the control room operator. Remote manual isolation of discharge can be initiated using redundant valves.
Discharge radiation monitor	Improperly calibrated or power failure	Activity of liquid discharged to environment is not monitored or recorded.	Monitor has down scale alarm to warn operator in control room of of loss of power. Recycle line is provided on sample tank to permit each batch of waste being discharged to be sampled for isotopic content prior to release.
Discharge flow monitor	Improperly calibrated or power failure	Quantity of liquid being discharged to environment is not monitored or recorded.	Level recorder on sample tank provides indirect record of quantity of liquid released.

TABLE 11.2-1 (Continued)

Equipment Item	Malfunction	Consequences	Design Precautions
Service water discharge header flow	Flow through line is blocked or lost.	Radioactive isotope concentration in discharge could exceed limits of <10 CFR 20>.	Flow sensor on weir of discharge structure signals power operated valve in LRW discharge line to close on loss of dilution water flow.

TABLE 11.2-2

DESIGN DATA FOR LIQUID RADWASTE SYSTEM SUMPS

Sump Description	Number	Operating Capacity (gal)	Type of Cover Plate	Stainless Steel Liner Provided
Drywell Equipment Drains	1	500	Shielding Plate	Yes
Containment Equipment Drains	1	440	Checkered Plate	Yes
Radwaste Building Equipment Drains	1	500	Checkered Plate	Yes
Intermediate Building Equipment Drains	1	500	Grating	Yes
Turbine Power Complex Equipment Drains	1	1,000	Checkered Plate	Yes
Control Complex Equipment Drains	1	500	Checkered Plate	Yes
Drywell Floor Drains	1	125	Shielding Plate	Yes
Containment Floor Drains	1	390	Checkered Plate	Yes
Annulus Floor Drains	1	50	Checkered Plate	No
Intermediate Building Floor Drains	1	500	Grating	Yes
Auxiliary Building Floor/Equipment Drains	1	345/675	Checkered Plate	Yes
Turbine Power Complex Floor Drains	1	1,000	Checkered Plate	Yes

TABLE 11.2-2 (Continued)

Sump Description	Number	Operating Capacity (gal)	Type of Cover Plate	Stainless Steel Liner Provided
Turbine Laydown Area Floor Drains	1	450	Checkered Plate	No
Radwaste Building Floor Drains	1	500	Checkered Plate	Yes
Turbine Lube Oil Area Floor Drains	1	750	Checkered Plate	No
Heater Bay Floor Drains	1	270	Checkered Plate	Yes
Turbine Power Complex Chemical Drains	1	270	Grating	Yes
Control Complex Laundry and Floor Drains	1	500	Checkered Plate	Yes

NOTE:

(Deleted)

TABLE 11.2-3

DESIGN DATA FOR LIQUID RADWASTE SYSTEM SUMP PUMPS

Sump Pump	Type	Quantity	Design Press. (psig)	Design Temp. (°F)	Design Flow Rate (gpm)	TDH @ Design Pt. (ft)	Shutoff Head (ft)	Max. Fluid Temp. (°F)	Material	Tag Number
Drywell Equipment Drain Sump Pumps	Duplex V.T.[1]	2	100	200	50	60	80	150	CI	1G61C001A 1G61C001B
Containment Equipment Drain Sump Pumps	Duplex V.T.[1]	2	100	200	50	60	80	150	CI	1G61C002A 1G61C002B
Radwaste Bldg. Equipment Drain Sump Pumps	Duplex V.T.[1]	2	100	150	50	65	82	100	CI	G61C003A G61C003B
Intermediate Bldg. Equipment Drain Sump Pumps	Duplex V.T.[1]	2	100	150	50	65	82	100	CI	G61C004A G61C004B
Turbine Power Complex Equipment Drain Sump Pumps	Duplex V.T.[1]	2	100	150	100	110	140	100	CI	1G61C007A 1G61C007B
Drywell Floor Drain Sump Pumps	Duplex V.T.[1]	2	100	150	50	65	82	100	CI	1G61C008A 1G61C008B
Containment Floor Drain Sump Pumps	Duplex	2	100	150	50	65	88	100	CI	1G61C009A 1G61C009B
Annulus Floor Drain Sump Pumps	Single	2	100	150	25	55	58	100	CI	1G61C010 2G61C010
Intermediate Bldg. Floor Drain Sump Pumps	Duplex	2	100	150	50	60	80	100	CI	G61C011A G61C011B
Auxiliary Bldg. Floor Drain Sump Pumps	Duplex	2	100	150	100	70	82	100	CI	1G61C012A 1G61C012B
Turbine Power Complex Floor Drain Sump Pumps	Duplex	2	100	150	100	85	109	100	CI	1G61C014A 1G61C014B
Turbine Power Complex Floor Drain Sump Pumps	Single V.T.[1]	1	100	150	750	130	170	100	CI	1G61C014C

TABLE 11.2-3 (Continued)

Sump Pump	Type	Quantity	Design Press. (psig)	Design Temp. (°F)	Design Flow Rate (gpm)	TDH @ Design Pt. (ft)	Shutoff Head (ft)	Max. Fluid Temp. (°F)	Material	Tag Number
Turbine Laydown Area Floor Drain Sump Pumps	Duplex	2	100	150	50	25	32	100	CI	1G61C015A 1G61C015B
Radwaste Bldg. Floor Drain Sump Pumps	Duplex	1	100	150	50	60	82	100	CI	0G61C016A
	Duplex	1	100	150	50	65	88	100	CI	0G61C016B
Turbine Lube Oil Area Floor Drain Sump Pumps	Duplex	2	100	150	50	50	75	100	CI	1G61C005A 1G61C005B
Heater Bay Floor Drain Sump Pumps	Duplex	2	100	150	25	95	98	100	CI	1G61C019A 1G61C019B
Control Complex Equipment Drain Sump Pumps	Duplex V.T.[1]	2	100	150	50	65	82	100	CI	G61C013A G61C013B
Turbine Power Complex Chemical Drain Sump Pumps	Duplex	2	100	150	25	60	64	100	SS	1G61C017A 1G61C017B
Control Complex Laundry and Floor Drain Sump Pumps	Duplex	2	100	175	50	75	96	150	CI	G61C018A G61C018B
Auxiliary Bldg. Equipment Drain Sump Pumps	Duplex V.T.[1]	2	100	150	250	70	104	100	CI	1G61C020A 1G61C020B

NOTE:

[1] V.T. - Abbreviation for "vertical turbine."

TABLE 11.2-4

DESIGN DATA FOR LIQUID RADWASTE SYSTEM TANKS

Tank	Quantity (2 Units)	Type	Head Design	Design Press. (Psig)	Design Temp. (°F)	Mat'l	Operating Capacity(1) (Gal.)	Tag Number
Waste Collector Tanks	2	Vertical	Flat Top, Dish. Bot.	Atmos.	150	304 SS	36,500	G50-A001A G50-A001B
Waste Sample Tanks	2	Vertical	Flat Top, Dish. Bot.	Atmos.	150	304 SS	34,000	G50-A002A G50-A002B
Floor Drain Collector Tanks	2	Vertical	Flat Top, Dish. Bot.	Atmos.	150	304 SS	36,500	G50-A003A G50-A003B
Floor Drain Sample Tanks	2	Vertical	Flat Top, Dish. Bot.	Atmos.	150	304 SS	34,000	G50-A004A G50-A004B
Chemical Waste Tanks	2	Vertical	Flat Top, Dish. Bot.	Atmos.	150	316 SS	19,650	G50-A005A G50-A005B
Concentrated Waste Tanks	2	Vertical	Flat Top, Dish. Bot.	Atmos.	200	Incoloy 825	4,900	G50-A006A G50-A006B
Chemical Waste Distillate Tank	2	Horiz.	Shallow Dished	Atmos.	150	304 SS	19,100	G50-A007A G50-A007B
Detergent Drains Tanks	2	Horiz.	Shallow Dished	Atmos.	150	304 SS	1,550	G50-A008A G50-A008B
Spent Resin Tanks	2	Vertical	Flat Top, Elip. Bot.	Atmos.	150	304 SS	9,500	G50-A009A G50-A009B
Condensate Filter Backwash Receiving Tank	2	Horiz.	Shallow Dished	Atmos.	150	304 SS	9,900	G50-A010 G50-A010
Condensate Filter Backwash Settling Tanks	2	Vertical	Flat Top, Dish. Bot.	Atmos.	150	304 SS	17,600	G50-A011A G50-A011B
RWCU F/D Backwash Settling Tanks	2	Vertical	Flat Top, Dish. Bot.	Atmos.	150	304 SS	4,400	G50-A013A G50-A013B

TABLE 11.2-4 (Continued)

Tank	Quantity (2 Units)	Type	Head Design	Design Press. (Psig)	Design Temp. (°F)	Mat'l	Operating Capacity(1) (Gal.)	Tag Number
Fuel Pool F/D Backwash Settling Tanks	2	Vertical	Flat Top, Dish. Bot.	Atmos.	150	304 SS	17,600	G50-A014A G50-A014B
LRW Filter Precoat Tank	2	Vertical	Flat Top, Dish. Bot.	Atmos.	150	304 SS (Plasite lined)	1,475	G50-A015
LRW Demineralizer Resin Feed Tank	1	Vertical	Flat Top, Cone Bot.	Atmos.	150	CS (Koroseal lined)	825	G50-A016
LRW Filter Aid Tank	1	Vertical	Flat Top, Dish. Bot.	Atmos.	150	304 SS	1,000	G50-A017
Fuel Pool F/D Backwash Receiving Tank	1	Horiz.	Shallow Dished	Atmos.	150	304 SS	9,400	G50-A022
LRW Phosphate Tank	1	Vertical	Flat Top, Flat Bot.	Atmos.	100	304 SS	250	G50-A023
LRW Hot Water Heater	1	Vertical	ASME Dished	130	200	CS (Phenolic Lining)	500	G50-B003
Waste Collector Filtrate Tank	1	Vertical	Flat Top, Dish Bot.	Atmos.	150	304 SS	400	G50-A024
Floor Drains Filtrate Tank	1	Vertical	Flat Top, Dish Bot.	Atmos.	150	304 SS	400	G50-A025

NOTE:

(1) Operating capacity is arbitrarily defined herein to be the volume of that portion of the tank up to 6 inches below the lowest point in the overflow line.

TABLE 11.2-5

DESIGN DATA FOR LIQUID RADWASTE SYSTEM PUMPS

Pump	Type	Quantity	Design Press. (psig)	Design Temp. (°F)	Design Flow Rate (gpm)	TDH @ Design Pt. (ft)	Shutoff Head (ft)	Max. Fluid Temp. (°F)	Material	Tag Number
Waste Collector Transfer Pumps	Horz. Cent.	2	125	150	150	175	205	140	316 SS	G50-C001A G50-C001B
Waste Sample Pumps	Horz. Cent.	2	125	150	200	110	122	100	316 SS	G50-C002A G50-C002B
Floor Drains Collector Transfer Pumps	Horz. Cent.	2	125	150	150	175	205	100	316 SS	G50-C003A G50-C003B
Floor Drains Sample Pumps	Horz. Cent.	2	125	150	200	110	122	100	316 SS	G50-C004A G50-C004B
Chemical Waste Pumps	Horz. Cent.	2	125	150	120	110	125	100	Alloy 20 SS	G50-C005A G50-C005B
Chemical Waste Distillate Pumps	Horz. Cent.	2	125	150	200	110	122	120	316 SS	G50-C006A G50-C006B
Detergent Drains Pumps	Horz. Cent.	2	125	150	50	110	114	140	316 SS	G50-C007A G50-C007B
Spent Resin Pumps	Horz. Cent.	2	125	150	400	135	160	100	316 SS	G50-C008A G50-C008B
Condensate Backwash Transfer Pumps	Horz. Cent.	2	125	150	450	80	90	100	316 SS	1G50-C009A 1G50-C009B
Cond. Sludge Discharge Mixing Pumps	Horz. Cent.	2	125	150	400	180	208	100	316 SS	G50-C010A G50-C010B
Condensate Sludge Decant Pumps	Horz. Cent.	2	125	150	450	75	81	100	316 SS	G50-C011A G50-C011B
RWCU Backwash Transfer Pumps	Horz. Cent.	1	125	150	350	95	105	120	316 SS	1G50-C012

TABLE 11.2-5 (Continued)

Pump	Type	Quantity	Design Press. (psig)	Design Temp. (°F)	Design Flow Rate (gpm)	TDH @ Design Pt. (ft)	Shutoff Head (ft)	Max. Fluid Temp. (°F)	Material	Tag Number
RWCU Sludge Discharge Mixing Pumps	Horz. Cent.	2	125	150	200	175	182	120	316 SS	G50-C013A G50-C013B
RWCU Sludge Decant Pumps	Horz. Cent.	2	125	150	50	55	60	120	316 SS	G50-C014A G50-C014B
Fuel Pool Sludge Discharge Mixing Pumps	Horz. Cent.	2	125	150	400	180	208	100	316 SS	G50-C015A G50-C015B
Fuel Pool Sludge Decant Pumps	Horz. Cent.	2	125	150	450	75	86	100	316 SS	G50-C016A G50-C016B
Waste Collector Filtrate Pump	Horz. Cent.	1	125	150	150	100	120	140	CI	G50-C017
Floor Drains Filtrate Pump	Horz. Cent.	1	125	150	150	100	120	100	CI	G50-C018
Waste Collector Filter Aid Pump	Positive Displnt.	1	125	150	0.12 to 1.2	134	-	100	316 SS	G50-C019
Floor Drains Filter Aid Pump	Positive Displnt.	1	125	150	0.12 to 1.2	134	-	100	316 SS	G50-C020
Radwaste Precoat Pumps	Horz. Cent.	2	150	150	350	55	65	100	CI	G50-C021A G50-C021B
Spent Resin Transfer Pumps	Horz. Cent.	1	125	150	200	175	182	100	316 SS	1G50-C022
WEC Concentrate Pumps	Horz. Cent.	4	125	250	45			220	Alloy 20 SS	G50-C023A G50-C023B G50-C023C G50-C023D
WEC Distillate Pumps	Canned	2	125	150	31.2			120	316 SS	G50-C024A G50-C024B

TABLE 11.2-5 (Continued)

Pump	Type	Quantity	Design Press. (psig)	Design Temp. (°F)	Design Flow Rate (gpm)	TDH @ Design Pt. (ft)	Shutoff Head (ft)	Max. Fluid Temp. (°F)	Material	Tag Number
Concentrated Waste Transfer Pumps	Horz. Cent.	2	125	200	150	150	180	150	Alloy 20 SS	G50-C026A G50-C026B
Fuel Pool F/D Backwash Transfer Pump	Horz. Cent.	1	125	150	450	120	135	100	316 SS	G50-C027

TABLE 11.2-6

DESIGN DATA FOR LIQUID RADWASTE SYSTEM PIPING

Pipe Line Specification	Design Code	Material	Schedule or Wall
G18-4	ANSI B31.1	Stainless steel tubing; ASTM A213, Gr TP316	0.065″ wall
L1-3	ASME Code, Section III, Class 3	Carbon steel; ASME SA 160 Gr B	Sch. 80 (2″ and less) Sch. 40 (2-1/2″ thru 10″)
L1-4	ANSI B31.1	Carbon steel; ASTM A106 Gr B	Sch. 80 (2″ and less) Sch. 40 (2-1/2″ thru 10″)
L2-3	ASME Code Section III, Class 3	Stainless steel; ASME SA 312 or SA 376 TP 304	Sch. 40S
L2-4	ANSI B31.1	Stainless steel; ASTM A 312 or A 376 TP 304	Sch. 40S
L6-4	ANSI B31.1	Yoloy (nickel/ copper alloy steel); ASTM A53	Sch. 80 (2″ and less) Sch. 40 (2-1/2″ to 20″)
L7-3	ASME Code, Section III, Class 3	Alloy 20 stain-less steel, ASME SB 464	Sch. 40S
L7-4	ANSI B31.1	Alloy 20 stain-less steel; ASTM B-464	Sch. 40S
N13-4	ANSI B31.1	Polypropylene lined carbon steel pipe per ASTM A53	Std. wall

TABLE 11.2-7

DESIGN DATA FOR LIQUID RADWASTE SYSTEM PROCESS EQUIPMENT

Equipment	Type	Quantity (2 Units)	Design Press. (psig)	Design Temp. (°F)	Design Flow Rate (gpm)	Material	Tag Number
Waste Demineralizer	Mixed bed	1	125	150	150 to 200	304 and 316 SS	G50-D003
Floor Drains Demineralizer	Mixed bed	1	125	150	150 to 200	304 and 316 SS	G50-D004
Waste Collector Filter	Precoat; Flat bed	1	See Note[2]	150	100 to 150	304 SS	G50-D001
Floor Drains Filter	Precoat	1	See Note[2]	150	100 to 150	304 SS	G50-D002
Waste Evaporator/ Condensers[3]	Horizontal; bowed tube; waste on shell side; forced recirculation	2	50	300	30	Incoloy 825 (evaporator section); 304 SS condenser section)	G50-B001A G50-B001B (evaporator) G50-B002A G50-B002B (condenser)
Detergent[1] Drains Filters	Cartridge	2	125	175	50	304 SS (vessel) Epoxy impregnated cellulose (element)	G50-D005A G50-D005B

NOTES:

(1) Detergent drain filters are abandoned in place.
(2) Enclosure designed to operate at atmospheric pressure; filter shells are designed to cycle up to 15 psig during periodic blowdown/cleaning.
(3) Evaporators steam supply has been cut and capped, rendering the evaporators inoperable for the waste processing function. Condensers in service.

TABLE 11.2-8

SUMMARY OF LIQUID RADWASTE SYSTEM
EQUIPMENT REDUNDANCY

Equipment	Degree of Redundancy	Normal Collecting or Processing Capacity	Maximum Collecting or Processing Capacity
Equipment Drain Sump	None	Varies (440 to 1,000 gal.)	Varies (440 to 1,000 gal.)
Equipment Drain Sump Pump	100%	Varies (50 to 250 gpm)	Varies (100 to 500 gpm)
Waste Collector Tank	100%	36,500 gal.	73,000 gal.
Waste Sample Tank	100%	34,000 gal.	68,000 gal.
Waste Collector Transfer Pump	100%	150 gpm	150 gpm
Waste Sample Pump	100%	200 gpm	200 gpm
Waste Demineralizer	100%[(1)]	Varies (150 to 200 gpm)	Varies (150 to 200 gpm)
Waste Collector Filter	100%[(2)]	150 gpm	150 gpm
Floor Drains Sump	None	Varies (50 to 1,000 gal.)	Varies (50 to 1,000 gal.)
Floor Drains Sump Pump	100%[(3)]	Varies (25 to 750 gpm)	Varies (50 to 750 gpm)
Floor Drains Collector Tank	100%	36,500 gal.	73,000 gal.
Floor Drains Sample Tank	100%	34,000 gal.	68,000 gal.
Floor Drain Collector Transfer Pump	100%	150 gpm	150 gpm

TABLE 11.2-8 (Continued)

Equipment	Degree of Redundancy	Normal Collecting or Processing Capacity	Maximum Collecting or Processing Capacity
Floor Drains Sample Pump	100%	200 gpm	200 gpm
Floor Drains Demineralizer	100%[1]	Varies (150 to 200 gpm)	Varies (150 to 200 gpm)
Floor Drains Filter	100%[2]	150 gpm	150 gpm
Chemical Drains Sump	None	270 gal.	270 gal.
Chemical Drains Sump Pump	100%	25 gpm	50 gpm
Chemical Waste Tank	100%	19,650 gal.	39,300 gal.
Waste Evaporator[5]			
Chemical Waste Distillate Tank	100%	19,100 gal.	38,200 gal.
Chemical Waste Distillate Pump	100%	200 gpm	400 gpm
Laundry and Floor Drains Sump	None	500 gal.	500 gal.
Laundry and Floor Drains Sump Pump	100%	50 gpm	100 gpm
Detergent Drains Tank	100%	1,550 gal.	3,100 gal.
Detergent Drains[4] Pump	100%	50 gpm	100 gpm

TABLE 11.2-8 (Continued)

Equipment	Degree of Redundancy	Normal Collecting or Processing Capacity	Maximum Collecting or Processing Capacity
Detergent Drains[4] Filter	100%	50 gpm	100 gpm
Chemical Waste Pump	100%	120 gpm	240 gpm

NOTES:

(1) Waste demineralizer and floor drains demineralizer can be cross tied.
(2) Waste collector filter and floor drains filter can be cross tied.
(3) Except for annulus floor drain sump pump.
(4) Detergent drain filters are abandoned in place. Detergent drain pump can only be used to recycle or transfer waste between detergent drain tanks. Detergent drain tank waste is drained and processed via radwaste floor drain system.
(5) Steam supply to the Waste Evaporators has been permanently severed.

TABLE 11.2-9

EVALUATION OF LIQUID RADWASTE SYSTEM CAPACITY FOR HANDLING LARGE WASTE INPUT VOLUMES[5]

Waste Input Description	Input Rate, (gpm)	Input Duration, (Minutes)	Total per Occurrence (Gal./day)	Frequency of Occurrence	Disposition of Waste Input
1. High purity (equipment drains) subsystem:					
a. Max. normal quantity of miscellaneous equipment leakage	50 to 500	Inter-mittent	33,000	158 days per year	Collect in one waste collector tank and process. Total process time is approximately 8 hours per occurrence.
b. Maximum quantity of miscellaneous equipment leakage (estimated quantity taken from ANSI N197-1976)[1]	50 to 500	Inter-mittent	73,100	14 days per year	Collect in both waste collector tanks and process. Total process time is approximately 16 hours per occurrence.
c. Condensate polishing demineralizer rinse during condenser tube leak period or plant startup.[3]	200	207	41,500	60 days per year	Collect in two waste collector tanks (or FDCT) and process. Total process time is approximately 10 hours per occurrence.
d. Reactor blowdown via reactor water cleanup system during startup (normally directed to main condenser)[3]	300	180	54,000	Rare	Collect in both waste collector tanks and process. Total process time is approximately 12 hours per occurrence.
e. Suppression pool drain (for decontamination, inspection and maintenance of pool)[2][3]	1,000	1,000 (Inter-mittently)	1,000,000	Once every 10 years	Collect in one waste collector tank and one floor drain collector tank in 34,000 gallon batches as these tanks become available. Total process time is approximately 150 hours per occurrence.
f. Spent fuel shipping cask pit drawdown[3]	200	235	47,000	18 days per year	Collect in two waste collector tanks and process. Total process time is approximately 10 hours.
g. RHR flush/test[3]	2,000	20.4	40,800	24 days per year	Collect in two waste collector tanks and process. Total process time is approximately 10 hours.

TABLE 11.2-9 (Continued)

Waste Input Description	Input Rate, (gpm)	Input Duration, (Minutes)	Total per Occurrence (Gal./day)	Frequency of Occurrence	Disposition of Waste Input
2. Low purity (floor drains) subsystem:					
a. Max. normal quantity of miscellaneous floor drainage	25 to 750	Inter-mittent	13,000	158 days per year	Collect in one floor drain collector tank and process. Total process time is approximately 3 hours per occurrence.
b. Maximum quantity of miscellaneous drainage (Estimated quantity taken from ANSI N197-1976)[(1)]	25 to 750	Inter-mittent	67,600	14 days per year	Collect in both floor drain collector tanks and process. Total process time is approximately 15 hours.
c. Decant from backwash settling tanks for reactor water cleanup filter/ demineralizers and condensate polishing filters during startup or condenser tube leak period.	450	90 (Inter-mittently)	39,200	60 days per year	Collect in both floor drains collector tanks in 8,000 gallon batches and process. Total process time is approximately 9 hours.
3. Chemical waste subsystem:					
a. Condensate polishing demineralizer regeneration solutions during startup[(4)] or condenser tube leak period.	200	65	13,000	60 days per year	Collect in one chemical waste tank and process. Total process time is approximately 8 hours.

NOTES:

(1) The maximum leak rate used here is for the drywell. It is assumed to occur in both drywells simultaneously, even though the probability of this happening is very low. This maximum leak rate could also occur in the containment, turbine power complex, auxiliary bldg., radwaste bldg., control complex, or intermediate bldg. However, the probability of simultaneous leakage in these areas while the maximum leakage rate is assumed in both drywells is extremely low. Since these areas are accessible, it is assumed that repairs could be made quickly enough to avoid such multiple failures.

TABLE 11.2-9 (Continued)

NOTES: (Continued)

(2) The total volume of water in the suppression pool is approximately 1,000,000 gallons. Since the reactor will be completely shut down while the pool is being inspected, the condenser hotwell can be used to store a portion of this volume (approximately 500,000 gallons). The remaining portion of the pool inventory will be pumped to the LRW system as tankage in this system becomes available to collect and process this waste. (It is assumed that one reactor will still be operating, requiring half of the LRW system processing capacity to be available for handling waste from the operating unit.)

(3) These inputs can be diverted to the low purity (floor drains) subsystem if processing conditions warrant.

(4) Condensate demineralizers are no longer regenerated.

(5) Table 11.2-9, Evaluation of Liquid Radwaste System Capacity for Handling Large Waste Input Volumes, was developed with the expectation that Unit 2 would achieve commercial operation. A later decision was made not to complete Unit 2 and subsequently, the unit was abandoned. However, this historical analysis remains bounding based on including Unit 2 operation because single unit operation will generate less waste input volume than that attained by dual unit operation.

TABLE 11.2-10

PROCESS FLOW DATA FOR LIQUID RADWASTE SYSTEM[10]

Stream Number	Stream Description	Normal Batches/Day	Maximum Batches/Day	Gallons/ Batch	Solids/ Batch (Pounds)	Normal Gal./Yr (Both Units)	Isotopic Activity[1]
1-a	Drywell Floor Drains (each unit)	2.82	113.3	255	N/A	525,000	M
1-b	Containment Floor Drains (each unit)	2.56	38.5	390	N/A	729,000	M
1-c	Turbine Building Floor Drains (each unit)	2.0	2.0	1,000	N/A	1,460,000	S
1-d	Radwaste Building Floor Drains (common)	2.0	2.0	500	N/A	365,000	R
1-e	Auxiliary Building Floor Drains (each unit)	1/1.72	43.5	345	N/A	146,000	M
1-f	Heater Bay Floor Drains (each unit)	-	1.0	270	N/A	-	S
1-g	Annulus Floor Drains (each unit)	-	1/5.0	50	N/A	-	(S+M)/2
1-h	Intermediate Building Floor Drains (common)	3.2	20.0	500	N/A	584,000	M
2	Decantate from SRW Disposal System	1.0	-	250	N/A	91,000	S/4
3	RHR Flush/Test (each unit)	1/30	2.0	40,800	N/A	993,000	Negligible
7	Floor Drains Effluent to Condenser	1/2.65	-	35,000	N/A	4,821,000	
8	Floor Drains Effluent Design Discharge	-	1/5.3	35,000	N/A	2,410,000 (max.)	<Table 11.2-15>
9-a	Recirc. Pumps & Valves in Drywell (each unit)	8.6	57.8	360	N/A	2,260,000	M

TABLE 11.2-10 (Continued)

Stream Number	Stream Description	Normal Batches/Day	Maximum Batches/Day	Gallons/ Batch	Solids/ Batch (Pounds)	Normal Gal./Yr (Both Units)	Isotopic Activity[(1)]
9-b	Drywell Steam Valves and Coolers in Drywell (each unit)	8.6	57.8	140	N/A	879,000	S
9-c	Misc. Pumps, Valves and RCIC Equip. in Containment (each unit)	9.64	35.86	265	N/A	1,865,000	M
9-d	Steam Valves in Containment (each unit)	9.64	35.86	50	N/A	352,000	S
9-e	RWCU Sample Drains in Containment (each unit)	9.64	35.86	125	N/A	880,000	M
9-f	Radwaste Building Equipment Drains (common)	1.0	1.0	500	N/A	182,000	R
9-g	Turbine Building Equipment Drains (each unit)	5.76	5.76	1,000	N/A	4,205,000	S
9-h	Auxiliary Building Equipment Drains (each unit)	1/11.2	1/3.0	675	N/A	44,000	M x 10^2
9-j	Intermediate Building Equipment Drains (common)	1/10.0	1/5.0	500	N/A	18,000	R
9-k	Control Complex Equipment Drains (common)	1/5.0	1.0	500	N/A	37,000	Negligible
10	Cond. Demin. Rinse (each unit)	1/14.6	1/2.0	41,500	N/A	2,075,000	S/4
11	Reactor Blowdown via RWCU (each unit)	-	Rare	-	-	-	-
13-g	W.D. Effluent Design Discharge	-	1/4.5	35,000	N/A	2,839,000 (max.)	<Table 11.2-15>
13-h	W.D. Effluent to Condenser	1.12	-	35,000	N/A	14,308,000	

TABLE 11.2-10 (Continued)

Stream Number	Stream Description	Normal Batches/Day	Maximum Batches/Day	Gallons/ Batch	Solids/ Batch (Pounds)	Normal Gal./Yr (Both Units)	Isotopic Activity[1]
14	Cond. Mixed Bed Demin. Regeneration Solutions	1/14.6	1/2.0	13,000	N/A	650,000	See Note[3]
15	Chemical Drains (each unit)	1.1	1.1	500	N/A	401,000	M/4
24	Radioactive Chemical Waste Effluent Design Discharge	-	1/36.5	15,360	N/A	154,000 (max.)	<Table 11.2-14>
25	Hot Shower and Detergent Drains	3.0	3.0	500	-	547,000	Negligible
27	Detergent Waste Effluent	1.48	2.65	1,600	-	864,000	Negligible
30	Floor Drains Demin. Spent Resins Transfer	1/30.5	1/22.35	1,455	1,970	17,000	Buildup on F.D. Demin.
31	Waste Demineralizer Spent Resins Transfer	1/35	1/15.35	1,455	1,970	15,000	Buildup on Waste Demineralizer
34-a	Cond. Demin. Spent Resins to SRW Disposal (both units)	6/3.6 yrs	6/3.6 yrs	9,970	12,090	17,000	<Table 11.4-4>

TABLE 11.2-10 (Continued)

Stream Number	Stream Description	Normal Batches/Day	Maximum Batches/Day	Gallons/ Batch	Solids/ Batch (Pounds)	Normal Gal./Yr (Both Units)	Isotopic Activity[1]
34-b	W.D., F.D. and S.P.D. Spent Resins to SRW Disposal	1/83.5	1/31.6	9,980	11,700	44,000	<Table 11.4-4>
35	Condensate Filter Backwash	1/3.0	8.0	5,200	360	1,265,000	See Note[5]
40	Cond. Filter Sludge to SRW Disposal	2/36.0	1/2	7,000	4,350	142,000	<Table 11.4-2[6]>
43	Avg. CBST Decantate	1/1.5	8.0	4,610	N/A	1,122,000	S/6
45	RWCU F/D Backwash (each unit)	1/6.5	1.0	2,400	70	270,000	See Note[7]
48	RWCU F/D Sludge to SRW Disposal	2/97.5	5/30	2,150	1,040	16,000	<Table 11.4-4[8]>
51	Avg. RBST Decantate	1/3.25	1.0	2,300	N/A	258,000	M/4
53	Fuel Pool F/D Backwash	1/5.2	1/5.2	2,160	65	152,000	See Note[2]
62	Fuel Pool F/D Sludge to SRW Disposal	1/348	1/30	7,000	4,350	7,000	<Table 11.4-2[9]>
65	Decantate from Fuel Pool Filter Backwash	1/5.2	1/5.2	2,055	N/A	144,000	Negligible
81	Cask Pit Drawdown	1/20.3	-	47,000	N/A	845,000	Negligible
84	Suppression Pool Maintenance Drain	-	1/10 yrs	1,000,000	N/A	-	Negligible
86	Cond. Mixed Bed Demin. Spent Resins Transfer	6/3.6 yrs	6/3.6 yrs	4,950	5,750	16,000	See Note[4]
87	Suppression Pool Cleanup Demin. Spent Resins Transfer	1/30	1/15	1,750	2,365	21,000	Buildup on Suppression Pool Cleanup Demin.

TABLE 11.2-10 (Continued)

NOTES:

(1) M = maximum concentration in reactor water.

S = maximum concentration in condensate.

R = maximum concentration in radwaste sump.

(2) Activity for this stream is 1.0 Curie/year, based on operating data from Nine Mile Point Nuclear Station, Unit No. 1.

(3) Activity is calculated on basis of 1.38×10^8 gallons of water treated by the condensate demineralizers every 90 days.

(4) Activity is calculated on basis of 1.38×10^8 gallons of water treated by the condensate demineralizers every 90 days plus M/4 times the demineralizer backwash volume.

(5) Activity is calculated on the basis of the filter buildup per batch (0.98 curies) plus S/6 times the backwash volume.

(6) Activity is calculated on the basis of the filter buildup for 8 days at the normal condensate flow rate (6.5 curies), a fill time of 4 days, and a decay time of 2 days.

(7) Activity is calculated on the basis of the filter buildup per batch (355 curies) plus M/4 times the backwash volume.

(8) Activity is calculated on the basis of the RWCU buildup per batch (355 curies) plus M/4 times the backwash volume, a fill time of 60 days, and a decay time of 60 days.

(9) Activity is calculated on the basis of the filter buildup per batch, a fill time of 100 days and a decay time of 100 days.

(10) Table 11.2-10, Process Flow Data for Liquid Radwaste System, was calculated with the expectation that Unit 2 would achieve commercial operation. A later decision was made not to complete Unit 2 and subsequently, the unit was abandoned. However, this historical analysis remains bounding based on including Unit 2 operation because single unit operation will generate less waste input volume than that attained by dual unit operation.

TABLE 11.2-11

QUALITY REQUIREMENTS FOR CONDENSATE MAKEUP

a.	Specific Conductivity at 25°C	<1.0 µmho/cm
b.	pH at 25°C	5.3 to 7.5
c.	Chloride (as Cl^-)	<0.05 ppm

TABLE 11.2-12

CRITERIA FOR SELECTION OF PROCESS FLOW PATH FOR LIQUID RADWASTE SYSTEM INPUTS

Subsystem		Description Process Flow Path[1]		Selecting Process Flow Path[1]
High purity/ low conductivity	a.	Collect, Sample and Reuse.	a.	Batch is within the limit for condensate makeup given in <Table 11.2-11>.
	b.	Collect, Sample, Body Feed, Filter, Demineralize, and Reuse.	b.	Batch is above any limit given in <Table 11.2-11>. Conductivity <100 μmho/cm
	c.	Collect, Sample, Body Feed, Filter, Demineralize, and Discharge or Reuse.	c.	Batch is above any limit given in <Table 11.2-11>. Conductivity >100 μmho/cm
Medium-to-low purity/medium conductivity	a.	Collect, Sample, Body Feed, Filter, Demineralize, and Reuse.	a.	Conductivity <100 μmho/cm
	b.	Collect, Sample, Body Feed, Filter, Demineralize, and Reuse or Discharge.	b.	Conductivity >100 μmho/cm
Detergent Drains	a.	Collect drain to floor drains.	a.	Always

NOTE:

[1] Depending on actual processing conditions, these flow paths and processing criteria may change.

TABLE 11.2-13

SIGNIFICANT NUCLIDE
ANNUAL RELEASE TO DISCHARGE TUNNEL

Nuclide	Annual Release to Discharge Tunnel (Ci/yr/unit)	Concentration In Plant Discharge (μCi/cc)	MPC (μCi/cc)	Fraction of MPC	<10 CFR 20, Appendix B> Effluent Concentration (μCi/cc)	Fraction of Effluent Conc.
Na-24	.023	3.9-10	3-5	1.3-5	5E-5	7.8E-6
P-32	.0022	3.7-11	2-5	1.9-6	9E-6	4.1E-6
Cr-51	.057	9.5-10	2-3	4.7-7	5E-4	1.9E-6
Mn-54	.00071	1.2-11	1-4	1.2-7	3E-5	4.0E-7
Mn-56	.00075	1.3-11	1-4	1.3-7	7E-5	1.9E-7
Co-58	.0023	3.9-11	9-5	4.3-7	2E-5	1.9E-6
Co-60	.0047	7.9-11	3-5	2.6-6	3E-6	2.6E-5
Fe-55	.012	2.1-10	2-3	1.0-7	1E-4	2.1E-6
Fe-59	.00035	5.9-12	5-5	1.2-7	1E-5	5.9E-7
Ni-63	.00001	1.6-13	7-4	2.3-10	1E-4	1.6E-9
Cu-64	.06	1.0-9	2-4	5.0-6	2E-4	5.0E-6
Zn-65	.01221	2.0-10	1-4	2.0-6	5E-6	4.0E-5
Zn-69m	.0045	7.6-11	6-5	1.3-6	6E-5	1.3E-6
Zn-69	.0048	8.0-11	2-3	4.0-8	8E-4	1.0E-7
Np-239	.053	8.9-10	1-4	8.9-6	2E-5	4.4E-5
Br-83	.00003	5.0-13	3-6	1.7-7	9E-4	5.5E-10
Sr-89	.0012	2.1-11	3-6	6.9-6	8E-6	2.6E-6
Sr-90	.00007	1.2-12	3-7	3.9-6	5E-7	2.4E-6
Y-90	.00002	3.4-13	2-5	1.7-8	7E-6	4.9E-8
Sr-91	.0048	8.0-11	5-5	1.6-6	2E-5	4.0E-6
Y-91m	.0031	5.2-11	3-3	1.7-8	2E-3	2.6E-8
Y-91	.00074	1.2-11	3-5	4.2-7	8E-6	1.5E-6
Y-92	.0019	3.1-11	6-5	5.2-7	4E-5	7.7E-7

TABLE 11.2-13 (Continued)

Nuclide	Annual Release to Discharge Tunnel (Ci/yr/unit)	Concentration In Plant Discharge (μCi/cc)	MPC (μCi/cc)	Fraction of MPC	<10 CFR 20, Appendix B> Effluent Concentration (μCi/cc)	Fraction of Effluent Conc.
Y-93	.0054	9.0-11	3-5	3.0-6	2E-5	4.5E-6
Zr-95	.00008	1.3-12	6-5	2.2-8	2E-5	6.5E-8
Nb-95	.00008	1.3-12	1-4	1.3-8	3E-5	4.3E-8
Zr-97	.00001	1.6-13	2-5	8.2-9	9E-6	1.8E-8
Nb-97	.00001	1.6-13	9-4	1.8-10	3E-4	5.3E-10
Mo-99	.016	2.7-10	4-5	6.7-6	2E-5	1.3E-5
Tc-99m	.023	3.9-10	3-3	1.3-7	1E-3	3.9E-7
Ru-103	.00023	3.9-12	8-5	4.8-8	3E-5	1.3E-7
Rh-103m	.00023	3.9-12	1-2	3.9-10	6E-3	6.5E-10
Ru-105	.00031	5.2-12	1-4	5.2-8	7E-5	7.4E-8
Rh-105	.0016	2.7-11	1-4	2.7-7	5E-5	5.4E-7
Ru-106	.00003	5.0-13	1-5	5.0-8	3E-6	1.7E-7
Rh-106	.00003	5.0-13	3-6	1.7-7	1E-4	5.0E-9
Ag-110m	.00001	1.6-13	3-5	5.4-9	6E-6	2.7E-8
W-187	.0013	2.2-11	6-5	3.7-7	3E-5	7.3E-7
Te-129m	.00046	7.7-12	2-5	3.9-7	7E-6	1.1E-6
Te-129	.00029	4.9-12	8-4	6.1-9	4E-4	1.2E-8
Te-131m	.00053	8.9-12	4-5	2.2-7	8E-6	1.1E-6
I-131	.06	1.0-9	3-7	3.4-3	1E-6	1.0E-3
Te-132	.00009	1.5-12	2-5	7.4-8	9E-6	1.7E-7
I-132	.00035	5.9-12	8-6	7.4-7	1E-4	5.9E-8
I-133	.083	1.4-9	1-6	1.4-3	7E-6	2.0E-4
Cs-134	.0077	1.3-10	9-6	1.4-5	9E-7	1.4E-4
I-135	.012	2.1-10	4-6	5.2-5	3E-5	7.0E-6
Cs-136	.0047	7.9-11	6-5	1.3-6	6E-6	1.3E-5
Cs-137	.018	3.0-10	2-5	1.5-5	1E-6	3.0E-4
Ba-137m	.017	2.8-10	3-6	9.4-5	NA	NA
Ba-140	.0043	7.3-11	2-5	3.6-6	8E-6	9.1E-6

TABLE 11.2-13 (Continued)

Nuclide	Annual Release to Discharge Tunnel (Ci/yr/unit)	Concentration In Plant Discharge (µCi/cc)	MPC (µCi/cc)	Fraction of MPC	<10 CFR 20, Appendix B> Effluent Concentration (µCi/cc)	Fraction of Effluent Conc.
La-140	.002	3.4-11	2-5	1.7-6	9E-6	3.8E-6
La-141	.00007	1.2-12	3-6	3.9-7	5E-5	2.4E-8
Ce-141	.00039	6.5-12	9-5	7.3-8	3E-5	2.2E-7
Ce-143	.00017	2.8-12	4-5	7.0-8	2E-5	1.4E-7
Pr-143	.00045	7.6-12	5-5	1.5-7	2E-5	3.8E-7
Ce-144	.00003	5.0-13	1-5	5.0-8	3E-6	1.7E-7
Pr-144	.00003	5.0-13	3-6	1.7-7	6E-4	8.3E-10
Nd-147	.00003	5.0-13	6-5	8.4-9	2E-5	2.5E-8
All others (except H-3)	.00001	1.6-13	-	-	-	-
Total (except H-3)	.5098	8.3-9	-	5.0-3	-	1.8E-3
H-3	47	7.9-7	3-3	2.6-4	1E-3	7.9E-4

NA - Not Applicable

TABLE 11.2-14

ANNUAL RELEASE BY STREAM TO DISCHARGE TUNNEL[(1)]

Nuclide	Releases to Discharge Tunnel High Purity (Curies)	Low Purity (Curies)	Chemical (Curies)	Total LWS (Curies)	Adjusted Total (Ci/yr)	Total (Ci/yr)
Na24	0.01122	0.00468	See Note[(1)]	0.01589	0.02273	0.02300
P32	0.00108	0.00045	See Note[(1)]	0.00153	0.00219	0.00220
Cr51	0.02804	0.01169	0.00002	0.03975	0.05684	0.05700
Mn54	0.00035	0.00015	See Note[(1)]	0.00049	0.00071	0.00071
Mn56	0.00037	0.00015	See Note[(1)]	0.00053	0.00075	0.00075
Fe55	0.00582	0.00243	0.00001	0.00825	0.01180	0.01200
Fe59	0.00017	0.00007	See Note[(1)]	0.00024	0.00035	0.00035
Co58	0.00115	0.00048	See Note[(1)]	0.00163	0.00233	0.00230
Co60	0.00233	0.00097	See Note[(1)]	0.00330	0.00472	0.00470
Ni63	0.00001	See Note[(1)]	See Note[(1)]	0.00001	0.00001	0.00001
Cu64	0.02949	0.01230	See Note[(1)]	0.04178	0.05975	0.06000
Zn65	0.00605	0.00262	See Note[(1)]	0.00867	0.01221	0.01221
Zn69m	0.00221	0.00092	See Note[(1)]	0.00313	0.00447	0.00450
Zn69	0.00237	0.00099	See Note[(1)]	0.00336	0.00481	0.00480
W187	0.00064	0.00027	See Note[(1)]	0.00091	0.00130	0.00130
Np239	0.02630	0.01096	See Note[(1)]	0.03726	0.05328	0.05300
Br83	0.00002	0.00001	See Note[(1)]	0.00002	0.00003	0.00003
Sr89	0.00058	0.00024	See Note[(1)]	0.00082	0.00117	0.00120
Sr90	0.00003	0.00001	See Note[(1)]	0.00005	0.00007	0.00007
Y90	0.00001	See Note[(1)]	See Note[(1)]	0.00002	0.00002	0.00002
Sr91	0.00239	0.00099	See Note[(1)]	0.00338	0.00483	0.00480
Y91m	0.00154	0.00064	See Note[(1)]	0.00218	0.00312	0.00310
Y91	0.00037	0.00015	See Note[(1)]	0.00052	0.00074	0.00074
Sr92	0.00010	0.00004	See Note[(1)]	0.00014	0.00019	0.00019
Y92	0.00091	0.00038	See Note[(1)]	0.00130	0.00185	0.00190
Y93	0.00265	0.00110	See Note[(1)]	0.00375	0.00536	0.00540

TABLE 11.2-14 (Continued)

	Releases to Discharge Tunnel				Adjusted	
Nuclide	High Purity (Curies)	Low Purity (Curies)	Chemical (Curies)	Total LWS (Curies)	Total (Ci/yr)	Total (Ci/yr)
Zr95	0.00004	0.00002	See Note(1)	0.00006	0.00008	0.00008
Nb95	0.00004	0.00002	See Note(1)	0.00006	0.00008	0.00008
Zr97	0.00001	See Note(1)	See Note(1)	0.00001	0.00001	0.00001
Nb97m	0.00001	See Note(1)	See Note(1)	0.00001	0.00001	0.00001
Nb97	0.00001	See Note(1)	See Note(1)	0.00001	0.00001	0.00001
Mo99	0.00804	0.00335	See Note(1)	0.01140	0.01629	0.01600
Tc99m	0.01116	0.00465	See Note(1)	0.01581	0.02261	0.02300
Ru103	0.00011	0.00005	See Note(1)	0.00016	0.00023	0.00023
Rh103m	0.00011	0.00005	See Note(1)	0.00016	0.00023	0.00023
Ru105	0.00015	0.00006	See Note(1)	0.00021	0.00031	0.00031
Rh105m	0.00015	0.00006	See Note(1)	0.00021	0.00031	0.00031
Rh105	0.00077	0.00032	See Note(1)	0.00110	0.00157	0.00160
Ru106	0.00002	0.00001	See Note(1)	0.00002	0.00004	0.00003
Rh106	0.00002	0.00001	See Note(1)	0.00002	0.00004	0.00003
Ag110m	0.00001	See Note(1)	See Note(1)	0.00001	0.00001	0.00001
Te129m	0.00023	0.00009	See Note(1)	0.00032	0.00046	0.00046
Te129	0.00014	0.00006	See Note(1)	0.00021	0.00029	0.00029
Te131m	0.00026	0.00011	See Note(1)	0.00037	0.00053	0.00053
Te131	0.00005	0.00002	See Note(1)	0.00007	0.00010	0.00010
I131	0.02786	0.01162	0.00214	0.04162	0.05951	0.06000
Te132	0.00004	0.00002	See Note(1)	0.00006	0.00009	0.00009
I132	0.00018	0.00007	See Note(1)	0.00025	0.00035	0.00035
I133	0.04119	0.01717	0.00001	0.05838	0.08348	0.08300
Cs134	0.00174	0.00364	0.00001	0.00539	0.00771	0.00770
I135	0.00573	0.00239	See Note(1)	0.00812	0.01161	0.01200
Cs136	0.00107	0.00224	See Note(1)	0.00331	0.00474	0.00470
Cs137	0.00408	0.00850	0.00001	0.01259	0.01800	0.01800

TABLE 11.2-14 (Continued)

Nuclide	Releases to Discharge Tunnel High Purity (Curies)	Low Purity (Curies)	Chemical (Curies)	Total LWS (Curies)	Adjusted Total (Ci/yr)	Total (Ci/yr)
Ba137m	0.00381	0.00795	0.00001	0.01177	0.01683	0.01700
Ba140	0.00215	0.00089	See Note[1]	0.00304	0.00435	0.00430
La140	0.00100	0.00042	See Note[1]	0.00142	0.00203	0.00200
La141	0.00004	0.00001	See Note[1]	0.00005	0.00007	0.00007
Ce141	0.00019	0.00008	See Note[1]	0.00027	0.00039	0.00039
Ce143	0.00008	0.00003	See Note[1]	0.00012	0.00017	0.00017
Pr143	0.00022	0.00009	See Note[1]	0.00032	0.00045	0.00045
Ce144	0.00002	0.00001	See Note[1]	0.00002	0.00004	0.00003
Pr144	0.00002	0.00001	See Note[1]	0.00002	0.00004	0.00003
Nd147	0.00002	0.00001	See Note[1]	0.00002	0.00003	0.00003
All Others	0.00001	See Note[1]	See Note[1]	0.00001	0.00001	0.00001
Total (Except Tritium)	0.23694	0.11675	0.00224	0.35593	0.50876	0.50981
Tritium Release	47 Curies per year					

NOTE:

[1] Less than .00001 Ci.

TABLE 11.2-15

INPUT PARAMETERS FOR CALCULATING LIQUID RELEASES (GALE)

MAXIMUM CORE THERMAL POWER - 3758 MWt

REACTOR COOLANT CLEANUP SYSTEM

Average flow rate - 1.54×10^5 lb/hr

Demineralizer type - powdered resin

CONDENSATE DEMINERALIZERS

Average flow rate - 10.5×10^6 lb/hr

Demineralizer type - deep bed

Number and size (ft^3) of demineralizers - six condensate demineralizers each containing 260 cubic feet of mixed resin

Regeneration frequency - 3.5 days per demineralizer for a total regeneration time of 21 days[(5)]

Regenerant volume - 12,000 gallons/batch[(5)]

TABLE 11.2-15 (Continued)

LIQUID WASTE PROCESSING SYSTEMS

Name	Sources	Flow[1] Rates (gpd)	Fraction of Primary Coolant Activity	Holdup Times Collection/ Discharge (days)	Fraction Assumed Discharge
High Purity Waste	Equipment Drains				
	Drywell	4300	1.0	1.7/.65	0.1
	Containment	2550	.01	1.7/.65	0.1
	Radwaste Building	500	.01	1.7/.65	0.1
	Turbine Building	5760	.01	1.7/.65	0.1
	Auxiliary Building	60	.01	1.7/.65	0.1
	Intermediate Building	25	.01	1.7/.65	0.1
	Control Complex	50	Negligible	1.7/.65	0.1
	Drywell and Containment Steam Valves	1685	.01	1.7/.65	0.1
	Cond. Demin. Rinse	1230	.002	1.7/.65	0.1
	RHR Flush/Test	340	Negligible	1.7/.65	0.1
Low Purity Waste	Floor Drains				
	Drywell	720	1.0	1.7/.65	0.25
	Containment	1000	.01	1.7/.65	0.25
	Turbine Building	2000	.01	1.7/.65	0.25
	Radwaste Building	500	.01	1.7/.65	0.25
	Auxiliary Building	200	.01	1.7/.65	0.25
	Intermediate Building	800	.01	1.7/.65	0.25
	Decantate	2210	.002	1.7/.65	0.25
Chemical Waste	Chemical Drains	275	.02	6.1/.37[2]	0.1
Regene-rant Waste[5]	Cond. Mixed Bed Demin. Reg. Sol.	820	(3)	6.1/.37[2]	0.1

TABLE 11.2-15 (Continued)

Name	Component	Capacity	Decontamination Factors Halogens/Cs, Rb/Other Nuclides
High Purity Waste	Waste Collector Tank	35,000 gallons	N/A
	Waste Sample Tank	35,000 gallons	N/A
	Waste Collector Filter	144,000 gpd	1/1/1
	Waste Demineralizer	288,000 gpd	$10^2/10/10^2$
Low Purity Waste	Floor Drains Collector Tank	35,000 gallons	N/A
	Floor Drains Sample Tank	35,000 gallons	N/A
	Floor Drains Filter	144,000 gpd	1/1/1
	Floor Drains Demineralizer	288,000 gpd	$10^2/2/10^2$
Chemical Waste	Chemical Waste Tank	20,000 gallons	N/A
	Chemical Waste Distillate Tank	20,000 gallons	N/A
	Waste or Floor Drains Demineralizer	288,000 gpd	$10^2/2/10^2$
Regenerant(4)(5) Waste			

NOTES:

(1) Values based on one-half of the total flow for two units.
(2) Collection time is based on total flow for chemical waste and regenerant waste since they utilize a common tank.
(3) Value calculated internally in BWR-GALE Code.
(4) Part of chemical waste system.
(5) Waste no longer generated.

TABLE 11.2-16

TANKS LOCATED OUTSIDE THE CONTAINMENT WHICH CONTAIN POTENTIALLY RADIOACTIVE FLUID

Tank	Quantity	Location	Tank Level[2] Monitoring	High Level[2] Annunication	Overflow Control[3]
Waste Mixing Dewatering tanks (G51-A001 A+B)	2	Radwaste Bldg.	SRW control panel	SWR and LRW Local control panels	Level switch discontinues flow into tanks on high level
Fuel Pool Surge Tank	2	Intermediate Bldg.	Control Room	Control Room	CRW
RWCU Filter/Demin Backwash settling tank	2	Radwaste Bldg.	RWBCR	RWBCR	DRW
Floor Drain Collector Tanks	2	Radwaste Bldg.	RWBCR	RWBCR	DRW
Waste Collector Tanks	2	Radwaste Bldg.	RWBCR	RWBCR	CRW
LRW Filter Aid Tank	1	Radwaste Bldg.	RWBCR TBF Control Panel	LRW Control Panel[4] TBF Control Panel	DRW
LRW Precoat Tank	1	Radwaste Bldg.	TBF Control Panel	LRW Control Panel[4] TBF Control Panel	DRW
Floor Drains Filtrate Tank	1	Radwaste Bldg.	TBF Control Panel	LRW Control Panel[4] TBF Control Panel	DRW
Waste Collector Filtrate Tank	1	Radwaste Bldg.	TBF Control Panel	LWR Control Panel[4] TBF Control Panel (CRW)	Waste Collector Tank
Backwash Rinse Receiving Tank	1	Turbine Bldg.	Local Panel	Control Room	DRW
Regen. Chemical Receiving Tank	1	Turbine Bldg.	Local Panel	Control room	DRW
Cond. Demin. Hot Water Tank	1	Turbine Bldg.	None	None	Closed System

TABLE 11.2-16 (Continued)

Tank	Quantity	Location	Tank Level[(2)] Monitoring	High Level[(2)] Annunciation	Overflow Control[(3)]
Moisture Separator Drain Tank	4	Turbine Bldg.	Local Panel	display monitor	Bypass to H.P. Condenser
1st Stage Drain Tank	4	Turbine Bldg.	None	Control Room & display monitor	Bypass to H.P. Condenser
2nd Stage Drain Tank	4	Turbine Bldg.	None	Control Room & display monitor	Bypass to H.P. Condenser
Condensate Storage Tank	1	Yard	Control Room	display monitor	Catch Basin
Fuel Transfer Tube Drain Tank	1	Intermediate Bldg.	None	Control Room	Contents Pumped to Surge Tank on Hi Level
Floor Drain Sample Tanks	2	Radwaste Bldg.	RWBCR	RWBCR	DRW
Waste Sample Tanks	2	Radwaste Bldg.	RWBCR	RWBCR	CRW
Chemical Waste Distillate Tanks	2	Radwaste Bldg.	RWBCR	RWBCR	display monitor
Detergent Drain Tanks	2	Radwaste Bldg.	RWBCR	RWBCR	CWT B
Chemical Waste Tanks	2	Radwaste Bldg.	RWBCR	RWBCR	DRW
Concentrated Waste Tanks	2	Radwaste Bldg.	RWBCR	None	CWT B
Spent Resin Tanks	2	Radwaste Bldg.	RWBCR	RWBCR	DRW
Condenstate Filter Backwash Receiving Tanks	1	Turbine Bldg.	RWBCR	RWBCR	DRW
Condensate Filter Backwash Settling Tanks	2	Radwaste Bldg.	RWBCR	RWBCR	DRW

TABLE 11.2-16 (Continued)

Tank	Quantity	Location	Tank Level[2] Monitoring	High Level[2] Annunication	Overflow Control[3]
Fuel Pool Filter Demineralizer Backwash Receiving Tank	1	Intermediate Bldg.	Local Panel RWBCR	RWBCR	DRW
Fuel Pool F/D Backwash Settling Tanks	2	Radwaste Bldg.	RWBCR	RWBCR	DRW
Condensate Return Tanks	2	Radwaste Bldg.	On Tanks	None	Start 2nd drain pump
Blowdown Tank	1	Auxiliary Bldg.	None	None	DRW
Deaerator	1	Auxiliary Bldg.	On Tank - Local Panel	Local Panel	Directed to Blowdown Tank
RF Pumps Seal Leakoff Drain Tanks	1	Heater Bay	Local Panel	Control Room	CRW
Precoat Slurry Tank	1	Turbine Bldg.	None	None	Close inlet valve on Hi Signal (DRW)
Mix and Hold Tank	1	Turbine Bldg.	None	None	Closed System
Anion Regeneration Tank	1	Turbine Bldg.	None	None	Closed System
Cation Regen. Tank	1	Turbine Bldg.	None	None	Closed System

NOTES:

(1) (Deleted)
(2) Local panels are located in the same area (building) as the tank unless specified otherwise.
(3) CRW and DRW indicate clean and dirty radwaste collection systems.
(4) High Level Alarm located on LRW Traveling Belt Filter Control Panel 0H51P0133. General trouble alarm for the TBF Panel can be found in the LRW Control Room.

11.3 GASEOUS WASTE MANAGEMENT SYSTEMS

11.3.1 DESIGN BASES

11.3.1.1 Design Objective

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

11.3.1.2 Design Criteria

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 Offsite Dose Calculation Manual.

In addition, the Offgas Treatment System limits the dose to offsite persons from a Control Rod Drop Accident <Section 15.4.9> to significantly less than the limits specified in <10 CFR 50.67>.

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 11.3-1a> indicates the design basis noble radiogas source terms referenced to 30 minute decay with the charcoal temperature at 0°F. <Table 11.3-1b>, <Table 11.3-1c>, and <Table 11.3-1d> indicate source terms referenced to 30 minute decay with the charcoal temperature at temperatures, 20°F, 40°F and 70°F, respectively.

The annual average exposure at the site boundary during normal operation from gaseous sources is not expected to exceed the dose objectives of to <10 CFR 50, Appendix I> in terms of actual doses to actual persons. The radiation dose design basis for the treated offgas is to delay the gas until the required fraction of the radionuclides has decayed and the daughter products are retained by the charcoal and the HEPA filters.

The gaseous radwaste equipment is selected, arranged and shielded to maintain occupational exposure as low as reasonably achievable in accordance with <Regulatory Guide 8.8>, and <10 CFR 20>.

The gaseous effluent treatment system is designed to the requirements of the General Design Criteria that follow.

General Design Criterion 60

The system has sufficient capacity to reduce the offgas 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 delay line provides advance notice of any potentially significant increase in releases. Continuous monitoring of the system effluent, with automatic isolation at activity levels corresponding to administrative release limits and annunciation at lower levels, along with continuous monitoring of the plant 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 offgas 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>. These equipment items are designed such that they provide no ignition source. Equipment and piping are also designed and constructed in accordance with the requirements of the applicable codes as given in <Table 3.2-1> and <Table 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, with additional quality requirements as recommended in <Regulatory Guide 1.143>.

The failure of the offgas system is analyzed in <Section 15.7.1>. The related failure of the steam jet air ejector lines and the gland seal offgas lines are also analyzed in <Section 15.7.1>.

The reactor building, turbine building and radwaste building contain radioactive sources. The design bases for the ventilation systems for these buildings are discussed in <Section 9.4>.

11.3.2 SYSTEM DESCRIPTION

The offgas from the main condenser steam jet air ejector is treated by a system using 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 sections that follow.

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

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

The air ejector offgas will normally contain activation gases, principally N-16, O-19 and N-13. The N-16 and O-19 have short half-lives and are readily decayed. The 10 minute half-life N-13 is present in small amounts that are further reduced by decay.

The air ejector offgas 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 offgas treatment system has been incorporated in the plant design to reduce the gaseous radwaste emission from the station. The offgas 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) will be delayed in the nominal 10 minute holdup system. The gas is cooled to 45°F and filtered through a HEPA filter. The gas is then passed through a desiccant dryer that reduces the dewpoint between 0°F and -40°F and is then further chilled prior to entering the charcoal adsorption beds.

Charcoal adsorption beds, normally operating in a refrigerated vault between 40°F and 0°F, selectively adsorb and delay the xenons and kryptons from the bulk carrier gas (principally dry air). After the delay, the gas is again passed through a HEPA filter and discharged to the environment through the offgas building vent.

11.3.2.1.1.1 Process Flow Diagram

<Figure 11.3-1> is the process flow diagram for the system. The process data for startup and normal operating conditions are on <Figure 11.3-1 (2)>.

Information supporting the process data is presented in (Reference 1). The vent is the single release point for this system and is located on the offgas building. The vent is indicated on <Figure 11.3-1> and <Figure 11.3-2>.

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 offgas 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.

11.3.2.1.3 Piping and Instrumentation Diagram (P&ID)

The P&ID is shown in <Figure 11.3-2>. The main process routing is indicated by a heavy line.

11.3.2.1.4 Recombiner Sizing

The basis for sizing the recombiner is to maintain the hydrogen concentration by volume, below 4 percent (including steam) at the inlet and below 1 percent at the outlet on a dry basis. The exit hydrogen concentration is normally well below the 1 percent maximum allowed. The hydrogen generation rate of the reactor is based on data from nine BWRs. The hydrogen generation rate is given in the process flow diagram, <Figure 11.3-1 (2)>.

11.3.2.1.5 Process Design Parameters

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

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

where:

T = holdup time of a given gas

K_D = dynamic adsorption coefficient for the given gas

M = weight of charcoal

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

Dynamic adsorption coefficient values for xenon and krypton were reported by Browning (Reference 2). 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. The fully redundant -90°F dewpoint, adsorbent air dryers are supplied to prevent moisture from reaching the charcoal. There are redundant moisture analyzers that will alarm on breakthrough of the drier beds; however, breakthrough is not expected since the drier 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 from the main condenser after the radiolytic hydrogen and oxygen are removed by the recombiner. The air inleakage design basis is conservatively sized at 30 scfm total. The Sixth Edition of Heat Exchange Institute Standards for Steam Surface Condensers (Reference 4), Par. Sl(c) (2), indicates that with certain conditions of stable operation and suitable construction,

noncondensibles (not including radiological decomposition products) should not exceed 6 scfm for large condensers. Dresden 2, Monticello, Fukushima 1, Tsuruga, and KRB have all operated at 6 scfm or below after initial startup. Dilution air is not added to the system unless the air inleakage is less than 6 scfm. In that event, 6 scfm is added to provide for dilution of residual hydrogen from the recombiner. An initial bleed of oil-free air is added on startup until the recombiner comes up to temperature.

11.3.2.1.6 Charcoal Adsorbers

11.3.2.1.6.1 Charcoal Temperature

The charcoal adsorbers normally operate between 40°F and 0°F. 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 are maintained at a constant temperature by a redundant vault refrigeration system. Failure of the refrigeration system will actuate an alarm in the control room. In addition, a radiation monitor is provided to monitor the radiation level in the charcoal bed vault. High radiation will actuate an alarm in the control room.

Limited operations of the charcoal adsorbers above 40°F may occur during maintenance activities on the refrigeration system, provided gaseous effluents are verified to be within the limits of the Offsite Dose Calculation Manual.

The following analysis was calculated with the expectation that Unit 2 would achieve commercial operation. A later decision was made not to complete Unit 2 and subsequently, the unit was abandoned. However, this historical analysis remains bounding based on including Unit 2 operation because single unit data is less than that attained by dual unit operation.

When Unit 1 and Unit 2 are operating simultaneously, temperatures of the refrigeration system must be reduced to +20°F, which will comply with the concluding statement of the regulatory staff to <10 CFR 50, Appendix I>. The calculated annual air dose at the reactor's site, will not exceed 10 mrads for gamma, and 20 mrads for beta.

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.

It may be desirable to use the bypass for short periods during startup or normal operations. This bypass mode would not be used for normal operation unless some unforeseen system malfunction would necessitate shutting down the power plant or operating in the bypass mode and remaining within limits. The activity release is controlled by a process monitor upstream of the vent isolation valve that will cause the bypass valve to close on a high radiation alarm. This interlock can be defeated only by a keylock switch. The alarm setting is set below the Offsite Dose Calculation Manual limit.

In addition, there is a high high-high alarm on the same monitor that will cause the offgas system to be isolated from the vent if established release limits are exceeded.

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

Hydrogen concentration or gases from the air ejector is kept below the flammable limit by maintaining adequate process steam flow for dilution at all times. This steam flow rate is monitored and alarmed. Furthermore, hydrogen concentration is monitored by the redundant hydrogen analyzers prior to the holdup process. If the hydrogen concentration exceeds 2 percent, an alarm is set to annunciate in the control room.

11.3.2.1.9 Field Run Piping

No piping in this system is field routed. This includes major process piping, drain lines, steam lines, and sample lines which are shown on <Figure 11.3-2>.

11.3.2.1.10 Liquid Seals

There are several liquid loop seals to prevent gas escape through drains shown on <Figure 11.3-2>. These seals are protected against loss of liquid by automatic shutoff valves downstream of the loop seals. If liquid level drops due to leakage in the cooler condenser, prefilter, and holdup line loop seals, the level sensors will initiate closure of

these automatic shutoff valves which will mitigate the consequences of the loss of loop seals, and provide a Control Room low level alarm.

Each seal has a manual valve that can be used to fill the loop.

Seals are also equipped with solenoid valves that close if radioactive release from this system exceeds established limits.

11.3.2.1.11 System Performance

Noble gas activity release is about 28-510 μCi/sec from the exit of the steam jet air ejector offgas system based upon 30 scfm air inleakage and an input of 100,000 μCi of 30 minute old "1971 Mixture." The isotopic composition is given in <Table 11.3-1a>, <Table 11.3-1b>, <Table 11.3-1c>, and <Table 11.3-1d> in units of μCi/sec and Ci/yr for charcoal temperatures of 0°F, 20°F, 40°F, and 70°F.

Iodine input into the offgas system is small by virtue of its retention in reactor water and condensate. The iodine remaining is essentially removed by adsorption in the charcoal. This is supported by the fact that charcoal filters remove 99.9 percent of the iodine in 2 inches of charcoal, whereas this system has approximately 74 feet of charcoal in the flow path.

Particulates are removed with a 99.95 percent efficiency by a HEPA filter as gas exits the nominal 10 minute holdup. The noble gas decays within the interstices of the activated charcoal and daughters are entrapped there. The charcoal serves as an excellent filter for other particulates and essentially no particulates exit from the charcoal.

The charcoal is followed with a HEPA filter which is a safeguard against escape of charcoal dust. Particulate activity discharged from this system is essentially zero.

11.3.2.1.12 Isotopic Inventory

The isotopic inventory of each equipment piece is given in <Table 15.7-3A> for the Design Basis and Normal (Realistic) operating 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 General Electric have been submitted in the General Electric Company proprietary topical report, (Reference 1). Non-proprietary portions of this information are reported in (Reference 3).

11.3.2.1.14 Single Failures and Operator Errors

Design provisions are incorporated which preclude the uncontrolled release of radioactivity to the environment as a result of any single operator error or of any single equipment malfunction short of the catastrophic equipment failures described in <Chapter 15>. An analysis of single equipment piece malfunctions is provided in <Table 11.3-4>.

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. Even though measures are taken to avoid a possible detonation, the system pressure boundary is designed to be detonation-resistant.

c. All discharge paths to the environment are monitored.

d. Dilution steam flow to the steam jet air ejector is monitored and alarmed, and valving is 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 is the source of the radioactive gases in the air discharged through the ventilation system. The design of the ventilation system is discussed in <Section 9.4>. The building volumes and ventilation flow rates are discussed in <Section 12.2.2>.

The activity released to the suppression pool from steam discharges is discussed in <Section 12.2.2>. A tabulation of the expected frequency and the quantity of steam discharged to the suppression pool is provided in <Table 11.3-5>.

11.3.2.1.16 Cost-Benefit Ratio

In accordance with <10 CFR 50, Appendix I>, Section II, Paragraph D, a cost-benefit analysis is not required for this system because it satisfies the Guides on Design Objectives for Light-Water-Cooled Nuclear Power Reactors proposed in the Concluding Statement of Position of the Regulatory Staff in Docket RM-50-2.

11.3.2.1.17 Maintainability of Gaseous Radwaste System

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

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

b. No rotating equipment in the process stream.

c. Rotating equipment is located in the system only where maintenance can be performed while the system is in operation.

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

a. Extremely stringent leak rate requirements placed upon all equipment, piping and instruments, and enforced by system integrity testing as discussed in <Section 11.3.2.2.1.7>.

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. Use of loop seals with automatic shutoff valves to prevent loss of liquid due to siphoning.

e. Specification of stringent seat leakage characteristics for valves and lines discharging to the environment through other systems.

11.3.2.2 System Design Description

11.3.2.2.1 Main Condenser Steam Jet Air Ejector Offgas Low-Temp System

11.3.2.2.1.1 Quality Classification and Construction and Testing Requirements

Equipment and piping are designed and constructed in accordance with the requirements of the applicable codes as given in <Table 11.3-6> 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 not designed as Seismic Category I.

11.3.2.2.1.2.2 Buildings Housing Gaseous Radioactive Waste Processing Systems

That portion of the offgas system upstream of the gas dryer prefilters is housed in the turbine building, which is a non-seismic, nonsafety class, reinforced concrete structure. The remaining portion of this system is located in the offgas building, which is a Safety Class 3, Seismic Category I, reinforced concrete structure. A detailed discussion of the seismic design for this building is given in <Section 3.7>.

11.3.2.2.1.3 Quality Control

A program is 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 from these documents 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 is established and executed by or for the organization performing the activity to verify conformance with the applicable documented instructions, procedures and drawings.

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 is performed by qualified welders employing qualified welding procedures per <Table 11.3-6>.

11.3.2.2.1.5 Materials

Materials for pressure retaining components of process systems were 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 are not used. The components satisfy 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 equipment, piping and valves operating at cold temperatures is in accordance with Paragraph UG84, Section VIII, of ASME Boiler and Pressure Vessel Code, Division 1. However, Code exceptions are not permitted for equipment.

11.3.2.2.1.6 Construction of Process Systems

Pressure retaining components of process systems utilize welded construction to the maximum practicable extent. Process piping systems include the first root valve on sample and instrument lines. Process lines are not less than 3/4 inch nominal pipe size. Sample and instrument lines are not considered as portions of the process systems. Flanged joints or suitable rapid disconnect fittings are not used, except where maintenance requirements clearly indicate that such

construction is preferable. Screwed connections in which threads provide the only seal are not used. Screwed connections backed up by seal welding or mechanical joints are used only on lines of 3/4 inch nominal pipe size. However, seal welding is not possible on the 3/4 inch connections on the level sensors for the hold-up line, prefilter, and cooler condenser moisture separator loop seals. Thread sealant is used on these joints. In lines 3/4 inch or greater, but less than two and one-half inch nominal pipe size, socket type welds are used. In lines of two and one-half inch nominal pipe size and larger, pipe welds are of the butt joint type. However, the Dryer Chiller drain line strainers are 3 inch nominal size to maximize strainer screen surface area resulting in socket weld connections larger than two and one half inches.

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 during the construction phase in their entirety, using available valves or temporary plugs at atmospheric tank connections. Hydrostatic testing of piping systems is performed at a pressure 1.5 times the design pressure, but in no case at less than 75 psig. The test pressure is held for a minimum of 30 minutes with no leakage indicated. Pneumatic testing may be substituted for hydrostatic testing in accordance with the applicable codes. A helium leak test is performed on the entire, as-installed after construction, gaseous radwaste process system.

After the initial pressure test and helium leak test (i.e., during the Operations phase) helium leak tests shall be performed on modifications/repairs whenever practicable. All welds performed during such modifications/repairs shall be subject to non-destructive examination (e.g., radiography or liquid penetrant exam).

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 11.3-7> lists the process parameters that are instrumented to alarm in the control room. It also indicates whether the parameters are recorded or indicated. The operator is in control of the system at all times.

This system has redundant hydrogen analyzers to monitor the hydrogen concentrations in the offgas system prior to the holdup process. If the hydrogen concentration exceeds 2 percent, an alarm will be annunciated in the control room. These hydrogen analyzers also support operation of the Hydrogen Water Chemistry (HWC) System. During operation of the Hydrogen Water Chemistry (HWC) System, a stoichiometric amount of oxygen is added upstream of the recombiner to recombine the hydrogen in the offgas system. These redundant hydrogen analyzers monitor the hydrogen concentration to allow HWC operation and shutdown HWC if % hydrogen gets too high.

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

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 offgas 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 offgas 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.

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

11.3.2.2.1.9 Detonation Resistance

Even though the system is designed to be free of ignition sources, the process pressure boundary of the system is 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 (Reference 6).

By the procedure described in (Reference 6), a designer can obtain the required wall thickness of a specific equipment design, which normally or possibly contains a detonable mixture of hydrogen and oxygen. This wall thickness is then translated to the corresponding detonation-containing, static equipment pressure rating by using an appropriate code calculation.

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 11.3-4>.

11.3.2.2.1.11.1 Previous Experience

A system with similar equipment is in service at the KRB plant in Germany. Its performance is reviewed in (Reference 1). The Tsuruga and Fukushima I plants in Japan have similar recombiners in service.

Similar systems (ambient temperature charcoal) are in service at Dresden 2 and 3, Pilgrim, Quad Cities 1 and 2, Nuclenor, Hatch, Browns Ferry 1, 2 and 3, and Duane Arnold.

11.3.2.3 Operating Procedure

11.3.2.3.1 Treated (Delayed) Radioactive Gas Sources

11.3.2.3.1.1 Main Condenser Steam Jet Air Ejector Offgas 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 between 40°F and 0°F, the glycol cooler is chilled to near 35°F and glycol is circulated through the cooler condenser, a desiccant dryer is regenerated and valved in, the offgas condenser cooling water is valved in, and the recombiner heaters are 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 stage of the 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 with a fixed steam supply to less than four percent hydrogen by volume and the offgas condenser outlet is maintained at less than one percent 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 <Section 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 Offgas 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.

During operation of the Hydrogen Water Chemistry (HWC) System, a stoichiometric amount of oxygen must be added upstream of the recombiner to recombine the hydrogen in the offgas system. Redundant hydrogen analyzers monitor hydrogen concentration and allow HWC operation or cause a HWC shutdown if % hydrogen gets too high.

11.3.2.4.1.1.2 Prefilter

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

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

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

11.3.2.4.1.1.3 Desiccant Gas Drier

Desiccant gas drier performance is continuously monitored by an onstream humidity analyzer.

11.3.2.4.1.1.4 Charcoal Performance

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

Experience with boiling water reactors has shown that the calibration of the offgas 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 periodically calibrated against grab samples, and whenever the radiation monitor after the offgas 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. They can be used for periodic sampling if the monitoring equipment indicates degradation of system delay performance.

11.3.2.4.1.1.5 Post Filter

On installation and replacements, these particulate filters will be tested using a DOP smoke test or equivalent.

11.3.2.4.1.1.6 Previous Experience

Previous experience is reviewed in <Section 11.3.2.2.1.11.1>.

11.3.3 RADIOACTIVE RELEASES

11.3.3.1 Release Points

A simplified flow diagram of the radioactive gas flow and treatment for the containment, the control complex, the auxiliary building, the fuel handling building, the radwaste building, the intermediate building, the turbine building, and the offgas building is presented in <Figure 11.3-3>. The physical location and elevation of the release points are shown on <Figure 1.2-18>.

The discharge from the condenser steam jet air ejector is processed by the low TEMP RECHAR System prior to release through the offgas building vent. <Section 11.3.2> discusses the low TEMP RECHAR System.

<Table 11.3-8a> provides the vent dimensions, effluent velocity and effluent gas temperature for each of the release points. <Table 11.3-8b>, <Table 11.3-8c>, and <Table 11.3-8d> provide parameters for charcoal temperatures at 20°F, 40°F and 70°F.

11.3.3.2 Dilution Factors

The atomospheric dilution factors associated with normal plant releases are based upon the average annual meteorological conditions applicable to the site as well as the effective release height of the effluent discharge pathway. The site meteorological conditions are defined in <Section 2.3.5>. Also included in <Table 2.3-27> are the average annual long term dilution factors (x/Q).

11.3.3.3 Estimated Releases and Dose Rates

The release rates of radioactive materials in gaseous effluents are presented in <Table 11.3-9a>, <Table 11.3-9b>, <Table 11.3-9c>, <Table 11.3-9d>, <Table 11.3-10a>, <Table 11.3-10b>, <Table 11.3-10c>, <Table 11.3-10d>, <Table 11.3-11a>, <Table 11.3-11b>, <Table 11.3-11c> and <Table 11.3-11d>. These values were calculated with the GALE computer code and are based on the assumptions and parameters provided in <NUREG-0016> and <Table 11.3-8a>, <Table 11.3-8b>, <Table 11.3-8c>, and <Table 11.3-8d>. As shown in <Table 11.3-11a>, <Table 11.3-11b>, <Table 11.3-11c>, and <Table 11.3-11d>, the estimated releases are a small fraction of the limits of <10 CFR 20>, and are as low as reasonably achievable. (Radiological assessments performed prior to October 4, 1993 that were used for the plant design bases as discussed in this USAR were evaluated against the <10 CFR 20> regulations prior to October 4, 1993. Radiological assessments for plant design bases

modifications that are performed after October 4, 1993 will be evaluated using the revised <10 CFR 20> dated October 4, 1993). The estimated offsite doses for the Perry site and a comparison with the design objectives of <10 CFR 50, Appendix I> and the dose limits of <40 CFR 190> are presented in <Section 5.2.4> and <Section 5.2.5> of the PNPP Environmental Report.

11.3.4 REFERENCES FOR SECTION 11.3

1. Miller, C. W., "Experimental and Operational Confirmation of Offgas System Design Parameters," NEDO-10751, January 1973. (Proprietary)

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

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

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

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

6. Head, R. A., et al., "Releases From BWR Radwaste Management Systems," NEDO-10951, July 1973.

TABLE 11.3-1a

ESTIMATED AIR EJECTOR OFFGAS RELEASE RATES[1]
(Charcoal Temperature = 0°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	5.3E-01	1.5E+01
Kr-85[4]	10.74 yr	2.0E+01	2.0E+01	2.0E+01	5.7E+02
Kr-87	76 min	2.0E+04	1.5E+04	-	-
Kr-88	2.79 hr	2.0E+04	1.8E+04	-	-
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 sec	1.4E+01	-	-	-
Xe-131m	11.96 day	1.5E+01	1.5E+01	6.5E-01	1.8E+01
Xe-133m	2.26 day	2.9E+02	2.8E+02	-	-
Xe-133	5.27 day	8.2E+03	8.2E+03	6.6E+00	1.9E+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 sec	2.8E+05	-	-	-
Xe-140	13.6 sec	3.0E+05	-	-	-

TABLE 11.3-1a (Continued)

Isotope	Half-Life	T=0 μCi/Sec	T=30 Minutes μCi/sec	Normal Discharge from Charcoal Adsorbers[2] μCi/sec	Ci/yr[3]
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 sec	5.6E+02	-	-	-
	TOTALS	2.4E+06	1.0E+05	2.8E+01	7.9E+02

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.

TABLE 11.3-1b

ESTIMATED AIR EJECTOR OFFGAS RELEASE RATES[1]
(Charcoal Temperature = 20°F)

Isotope	Half-Life	T=0 μCi/sec	T=30 min μCi/sec	Normal Discharge from Charcoal Adsorbers[2] μCi/sec	Normal Discharge from Charcoal Adsorbers[2] 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	1.0E+00	2.9E+01
Kr-85[4]	10.74 yr	2.0E+01	2.0E+01	2.0E+01	5.7E+02
Kr-87	76 min	2.0E+04	1.5E+04	-	-
Kr-88	2.79 hr	2.0E+04	1.8E+04	-	-
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 sec	1.4E+01	-	-	-
Xe-131m	11.96 day	1.5E+01	1.5E+01	7.8E-01	2.2E+01
Xe-133m	2.26 day	2.9E+02	2.8E+02	-	-
Xe-133	5.27 day	8.2E+03	8.2E+03	9.9E+00	2.8E+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 sec	2.8E+05	-	-	-
Xe-140	13.6 sec	3.0E+05	-	-	-

TABLE 11.3-1b (Continued)

Isotope	Half-Life	T=0 µCi/sec	T=30 min µCi/sec	Normal Discharge from Charcoal Adsorbers[2] µCi/sec	Ci/yr[3]
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 sec	5.6E+02	-	-	-
	TOTALS	2.4E+06	1.0E+05	3.2E+01	9.0E+02

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.

TABLE 11.3-1c

ESTIMATED AIR EJECTOR OFFGAS RELEASE RATES[1]
(Charcoal Temperature = 40°F)

Isotope	Half-Life	T=0 μCi/sec	T=30 min μCi/sec	Normal Discharge from Charcoal Adsorbers[2] μCi/sec	Normal Discharge from Charcoal Adsorbers[2] 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	9.3E+00	2.6E+02
Kr-85[4]	10.74 yr	2.0E+01	2.0E+01	2.0E+01	5.7E+02
Kr-87	76 min	2.0E+04	1.5E+04	-	-
Kr-88	2.79 hr	2.0E+04	1.8E+04	7.2E-01	2.0E+01
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 sec	1.4E+01	-	-	-
Xe-131m	11.96 day	1.5E+01	1.5E+01	1.8E+00	5.0E+01
Xe-133m	2.26 day	2.9E+02	2.8E+02	-	-
Xe-133	5.27 day	8.2E+03	8.2E+03	6.5E+01	1.8E+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 sec	2.8E+05	-	-	-
Xe-140	13.6 sec	3.0E+05	-	-	-

TABLE 11.3-1c (Continued)

Isotope	Half-Life	T=0 μCi/sec	T=30 min μCi/sec	Normal Discharge from Charcoal Adsorbers[2] μCi/sec	Ci/yr[3]
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 sec	5.6E+02			
	TOTALS	2.4E+06	1.0E+05	9.7E+01	2.7E+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.

TABLE 11.3-1d

ESTIMATED AIR EJECTOR OFFGAS RELEASE RATES[1]
(Charcoal Temperature = 70°F)

Isotope	Half-Life	T=0 µCi/sec	T=30 min µCi/sec	Normal Discharge from Charcoal Adsorbers[2] µCi/sec	Normal Discharge from Charcoal Adsorbers[2] Ci/yr[3]
Kr-83m	1.86 hr	3.4E+03	2.9E+03	1.1E-01	3.1E+00
Kr-85m	4.4 hr	6.1E+03	5.6E+03	7.6E+01	2.2E+03
Kr-85[4]	10.74 yr	2.0E+01	2.0E+01	2.0E+01	5.7E+02
Kr-87	76 min	2.0E+04	1.5E+04	-	-
Kr-88	2.79 hr	2.0E+04	1.8E+04	2.0E+01	5.7E+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 sec	1.4E+01	-	-	-
Xe-131m	11.96 day	1.5E+01	1.5E+01	3.9E+00	1.1E+02
Xe-133m	2.26 day	2.9E+02	2.8E+02	2.4E-01	6.9E+00
Xe-133	5.27 day	8.2E+03	8.2E+03	3.9E+02	1.1E+04
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 sec	2.8E+05	-	-	-
Xe-140	13.6 sec	3.0E+05	-	-	-

TABLE 11.3-1d (Continued)

Isotope	Half-Life	T=0 μCi/sec	T=30 min μCi/sec	Normal Discharge from Charcoal Adsorbers[(2)] μCi/sec	Ci/yr[(3)]
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 sec	5.6E+02			
	TOTALS	2.4E+06	1.0E+05	5.1E+02	1.4E+04

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.

TABLE 11.3-2

OFFGAS SYSTEM MAJOR EQUIPMENT ITEMS

Offgas Preheaters - 2 required.

Construction: Stainless steel tubes and carbon steel shell. 350 psig design pressure, 1,000 psig tube design pressure 40°F/450°F shell design temperature, 40°F/575°F tube design temperature.

Catalytic Recombiners - 2 required.

Construction: Carbon steel cartridge, carbon steel shell. Catalyst cartridge containing a precious metal catalyst on metal base or porous non-dusting ceramic. Catalyst cartridge to be replaceable without removing vessel. 350 psig design pressure. 900°F design temperature.

Offgas Condenser - 1 required.

Construction: Low alloy steel shell. Stainless steel tubes. 350 psig shell design pressure. 250 psig tube design pressure. 900°F shell design temperature. 150°F tube design temperature.

Water Separator - 1 required.

Construction: Carbon steel shell, stainless steel wire mesh. 350 psig design pressure. 250°F design temperature.

Cooler-Condenser - 2 required.

Construction: Carbon or stainless steel shell. Stainless steel tubes. 100 psig tube design pressure. 350 psig shell design pressure. 150°F tube design temperature 32°F/150°F shell design temperature.

Moisture Separators (Downstream of cooler-condenser) - 2 required.

Construction: Carbon steel shell, stainless steel wire mesh. 350 psig design pressure 32°F/150°F design temperature.

Desiccant Dryer - 4 required.

Construction: Carbon steel shell packed with Linde Mol Sieve or equivalent. 350 psig design pressure, 32°F/500°F design temperature.

Desiccant Regeneration Skid - 2 required.(1)

Dryer Chiller - 2 required.(1)

Construction: Carbon steel shell, stainless steel tubes, design temperature 32°F/500°F. Design pressure 50 psig.

TABLE 11.3-2 (Continued)

Regenerator Blower - 2 required.[1]

Construction: Cast iron, design pressure 50 psig, design temperature 32°F/150°F. Seller's standard.

Dryer Heater - 2 required.

Construction: Carbon steel, design temperature 32°F/500°F, design pressure 50 psig.

Gas Cooler - 2 required.

Construction: Carbon or stainless steel material. 1,050 psig tube design temperature. -50°F/150°F design temperature.

Glycol Cooler Skid - 1 required.[1]

Glycol Storage Tank - 1 required.[1]

Construction: Carbon steel 3,000 gallon. Water-filled hydrostat static design pressure. 32°F design temperature. API-650.

Glycol Solution Refrigerators and Motor Drives - 3 required.[1]

Construction: Conventional refrigeration units. Glycol solution exit temperature 35°F. Seller's standard.

Glycol Pumps and Motor Drives - 3 required.[1]

Construction: Cast iron, 3 inch connections 0°F design temperature. Seller's standard.

Prefilters and After Filters - 2 required of each type.

Construction: Carbon steel shell. High-efficiency, moisture-resistant filter element. Flanged shell. 350 psig design pressure. -50°F/150°F design temperature.

Charcoal Adsorbers - 8 beds.

Construction: Carbon steel. Approximately 4-ft od x 21-ft vessels each containing approximately 4 tons of activated carbon. Design pressure 350 psig. Design temperature -50°F/250°F.

<u>NOTE</u>:

(1) Not located within the boundary of the portion of the N64 system that actively processes radioactive materials and which is required to be detonation resistant.

<TABLE 11.3-3>

(DELETED)

TABLE 11.3-4

EQUIPMENT MALFUNCTION ANALYSIS

Equipment Item	Malfunction	Consequences	Design Precaution
Steam jet air ejector	Low flow of motive high pressure steam	When the hydrogen and oxygen concentration exceed 4 and 5 vol %, respectively, the process gas may become flammable, if insufficient steam is supplied.	Alarm provided on steam for low steam flow. Recombiner temperature alarm.
		Inadequate steam flow will cause overheating and deterioration of the catalyst.	Steam flow to be held at constant maximum flow regardless to plant level. Recombiner temperature alarm.
	Wear of supply steam nozzle of ejector	Increased steam flow to recombiner. This would reduce degree of recombination at low power levels.	Low temperature alarms on preheater exit (recombiner inlet). Recombiner H_2 analyzers.
Preheaters	Steam leak	Would further dilute process offgas. Steam consumption would increase.	Spare preheater.
	Low pressure steam supply	Recombiner performance would fall off at low power level, and hydrogen content of recombiner gas discharge may increase, eventually to a combustible mixture.	Low-temperature alarms on preheater exit (recombiner inlet). Recombiner outlet H_2 analyzers.

TABLE 11.3-4 (Continued)

Equipment Item	Malfunction	Consequences	Design Precaution
Recombiners	Catalyst gradually deactivates	Temperature profile changes through catalyst. Eventually excess H_2 would be detected by H_2 analyzer or by a flowmeter. Eventually the stripped gas could become combustible.	Temperature probes in recombiner H_2 analyzer provided. Spare recombiner.
	Catalyst gets wet at start	H_2 conversion falls off and H_2 is detected by downstream analyzers. Eventually the gas could become combustible.	Condensate drains, temperature probes in recombiner. Air bleed system at startup. Recombiner thermal blanket, spare recombiner and heater, hydrogen analyzer.
Offgas 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.)	None.

TABLE 11.3-4 (Continued)

Equipment Item	Malfunction	Consequences	Design Precaution
Offgas condenser (Cont.)	Liquid level instruments fail	If both drain valves fail to open, water will build up in the condenser and pressure drop will increase. The high delta P, 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 increase operating pressure of the main condenser.	Two independent drain systems, each provided with high and low level alarms.
Water separator	Corrosion of wire mesh element	Higher quantity of water collected in holdup line and routed to radwaste.	Stainless steel mesh specified.

TABLE 11.3-4 (Continued)

Equipment Item	Malfunction	Consequences	Design Precaution
Cooler condensers	Corrosion of tubes	Glycol-water solution would leak into process (shell) side and be discharged to clean radwaste. If not detected as radwaste, the glycol solution would discharge to reactor condensate system.	Stainless-steel tubes specified. Low level alarm glycol tank level. Spare cooler condenser provided.
	Icing up of 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.	Design glycol-H_2 solution temperature well above freezing point. Spare unit provided. Temperature indication and low alarms on glycol temperature and process gas temperature.
Glycol refrigeration machines	Mechanical	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 dewpoint as it is discharged from the drier.	Two spare refrigerators during normal operation are provided. Glycol solution temperature alarms provided. Gas moisture detectors provided downstream of gas driers.

TABLE 11.3-4 (Continued)

Equipment Item	Malfunction	Consequences	Design Precaution
Moisture separators	Corrosion wire mesh element	Increased moisture would be retained in process gas routed to gas driers. Over a long period, the desiccant drier cycle period would deteriorate as result of moisture pickup. Pressure drop across prefilter may increase if filter media is wetted.	Stainless steel mesh specified. Spare unit provided. High delta P alarm on prefilter.
Prefilters	Loss of integrity of filter	More radioactivity would deposite in the drier desiccant. This would increase radiation level in the drier vault and make maintenance more difficult, but would not affect releases to the environment.	Spare unit provided in separate vault. Delta P instrumentation provided.
Desiccant drier	Moisture breakthrough	Moisture would freezeout in gas cooler and would result in increased system pressure drop. Gas with a high dewpoint temperature would reach charcoal bed.	Drier cycles on time. Redundant gas humidity analyzers and alarms supplied. Redundant drier system supplied gas drier and first charcoal bed can be bypassed through alternate drier to second charcoal bed.

TABLE 11.3-4 (Continued)

Equipment Item	Malfunction	Consequences	Design Precaution
Desiccant regeneration equipment	Mechanical failure	Inability to regenerate desiccant.	Redundant, shielded desiccant beds and drier equipment supplied.
Charcoal absorbers	Charcoal accumulates moisture	Charcoal performance will deteriorate gradually as moisture deposits. Holdup times for krypton and xenon would decrease, and plant emissions would increase.	Highly instrumented, mechanically simple gas dehumidification system with redundant equipment. Provisions made for drying charcoal.
Vault refrigeration units	Mechanical failure	If temperature exceeds approximately 40°F, increased emission could occur.	Spare refrigeration unit provided. Charcoal adsorber vaults A and B temperature instrumentation provided.
After filter	Loss of integrity of filter media	Probably of no real consequence, the charcoal media itself should be a good filter at the low air velocity.	Delta P instrumentation provided. Spare unit provided.

TABLE 11.3-4 (Continued)

Equipment Item	Malfunction	Consequences	Design Precaution
System	Internal detonation	Release of radioactivity if pressure boundary fails.	Main process equipment and piping are designed to contain a detonation.
System	Earthquake damage	Release of radioactivity.	Dose consequences are within <10 CFR 20> limits. Analysis is included in (Reference 6).

TABLE 11.3-5

FREQUENCY AND QUANTITY OF STEAM DISCHARGED TO SUPPRESSION POOL

	Event[1]	Frequency Category	Quantity of Steam lbs/event)
1.	RCIC Test (Monthly)	Moderate	27,600
2.	Inadvertent RCIC Injection	Moderate	4,600
3.	SRV Test (each valve)	Moderate	3,900
4.	SRV Flow Capacity Test (each valve)	Infrequent	15,300
5.	Total SRV Leakage (19 valve max.)	Continuous	380/Hr
6.	Trip of Both Recirc. Pump Motor	Moderate	30,000
7.	Turbine Trip	Moderate	30,000
8.	Generator Load Rejection	Moderate	30,000
9.	Pressure Regulator Failure, Open	Moderate	374,000[2]
10.	Recirc. Controller Failure	Moderate	30,000
11.	Loss of All Feedwater Flow	Moderate	30,000
12.	Inadvertent MSIV Closure	Moderate	374,000[2]
13.	Loss of Condenser Vacuum	Moderate	374,000[2]
14.	Feedwater Control Failure, Max. Demand	Moderate	30,000
15.	Loss of Auxiliary Transformer	Moderate	934,000[2]
16.	Loss of All Grid Connections	Moderate	934,000[2]
17.	Turbine Trip w/o Bypass	Infrequent	374,000[2]
18.	Generator Load Rejection w/o Bypass	Infrequent	374,000[2]
19.	Stuck Open SRV	Moderate	641,000

NOTES:

[1] Bases and assumptions for the listed events are as follows:

(a) Events 1 and 2 are based on steam flow rate during test mode per RCIC System Process Diagram 762E421C, for 60 and 10 minutes, respectively.

(b) Event 3 assumes test SRV opened 30 seconds maximum at 300-500 psig vessel pressure.

(c) Event 4 assumes tested SRV opened 30-60 seconds at 1,000 psig vessel pressure.

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

TABLE 11.3-5 (Continued)

NOTES: (Continued)

(e) Events 6 through 18 are based on event descriptions from <Chapter 15>.

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

(2) Isolation event. Except for Events 15 and 16, it is assumed that SRV actuation is terminated 30 minutes into the event whereupon the reactor is depressurized at 100°F/hr via RHR steam condensing mode. For Events 15 and 16, it is assumed that loss of plant air prevents availability of RHR steam condensing mode and normal SRV opening, vessel depressurized via ADS SRV's. RHR steam condensing mode is not used at PNPP.

TABLE 11.3-6

GASEOUS RADWASTE EQUIPMENT DESIGN REQUIREMENTS

	Design and Fabrication	Materials[1]	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
Atmos-pheric or 0-15 psig Tanks	ASME Code[2] Section III Class 3, API 620;650, AWWA D-100	ASME Code Section II	ASME Code Section IX	ASME Code[2] Section III Class 3, API 620;650, AWWA D-100
Heat Ex-changers	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	ASME Code[2]
Pumps	Manufacturers[3] Standards	ASME Code Section II or Manufac-turer's Standard	ASME Code Section IX (as required)	ASME Code[2] Section III Class 3; and Hydraulic Institute

NOTES:

(1) Material manufacturer's certified test reports should be obtained whenever possible.
(2) ASME Code stamp and material traceability not required.
(3) Manufacturer's standard for the intended service. Hydrotesting should be 1.5 times the design pressure.

TABLE 11.3-7

OFFGAS SYSTEM ALARMED PROCESS PARAMETERS

Parameters	Control Room Indicated	Control Room Recorded
Air ejector discharge pressure - high	X	
Preheater discharge temperature - low	X	
Recombiner catalyst temperature - high/low		X
Offgas condenser water level (dual) - high/low (LOCAL)		
Offgas condenser gas discharge temperature - high (LOCAL)		
H_2 analysis (offgas condenser discharge) - dual - high		X
Offgas condenser discharge radiation - high	X	X
Gas flow - high/low		X
Moisture separator discharge temperature - high/low		X
Glycol solution temperature - high/low		X
Glycol solution level - low		
Gas drier discharge humidity - high (LOCAL)		
Prefilter dP - high	X	
Charcoal adsorber temperature - high		X
Carbon vault A & B temperature - high/low	X	X
Carbon train flow - high/low		X
After filter dP - high	X	
Offgas (carbon bed discharge) radiation - high	X	X
Steam flow - low		
Carbon train dP - high	X	

TABLE 11.3-8a

INPUT PARAMETERS USED FOR CALCULATING GASEOUS RELEASES[5]
(charcoal temperature = 0°F)

Maximum core thermal power - 3,758 MWt

Total main steam flow rate - 16.3×10^6 lb/hr

Mass of reactor coolant in the reactor vessel - 5.28×10^5 lb

Mass of steam in the reactor vessel - 1.93×10^4 lb

Holdup Times
- Charcoal delay (krypton) - 2.47 days
- Charcoal delay (xenon) - 54.2 days

Mass of charcoal in the offgas system - approximately 32 tons

Operating and dew point temperatures of offgas system - 0°F and -20°F, respectively

Dynamic adsorption coefficient for Xe and Kr is proprietary (General Electric)

Ventilation and Exhaust Systems

Building	Decontamination Factors (DF)	DF Bases	Purge Rate and Frequency (Reactor Building Only)
Reactor[1]	100	HEPA	5,000 cfm, continuous[4]
	10	Charcoal	approximately 25,000 cfm, refueling
Auxiliary[1]	100	HEPA	
	10	Charcoal	
Radwaste[1]	100	HEPA	
	10	Charcoal	
Turbine[2]	1	-	
	1	-	
Offgas[3]	100	HEPA	
	10	Charcoal	

TABLE 11.3-8a (Continued)

Release Points	Effluent Velocity (ft/min)	Vent Dimensions	Effluent Gas Temperature (maximum)
Unit 1			
Plant Vent	4,100	48″x90″	105°F
Turbine Building/ Heater Bay Vent	4,000	120″x120″	115°F
Offgas Vent	2,100	34″x34″	105°F
Unit 2			
Plant Vent	3,500	48″x90″	105°F
Turbine Building/ Heater Bay Vent	4,000	120″x120″	115°F
Offgas Vent	1,900	34″x34″	105°F

NOTES:

(1) The reactor building, auxiliary building and radwaste building releases are through the plant vent.
(2) The turbine building releases are through the turbine building/heater bay vent.
(3) The offgas building releases are through the offgas vent.
(4) Assuming a continuous reactor building purge provides an enveloping dose estimate for routine gaseous releases.
(5) The information in this table was used with the expectation that Unit 2 would achieve commercial operation. A later decision was made not to complete Unit 2 and subsequently, the unit was abandoned. However, this historical analysis remains bounding based on including Unit 2 operation because input parameters used for calculating gaseous releases for a single unit are less than that attained by dual unit operation.

TABLE 11.3-8b

INPUT PARAMETERS USED FOR CALCULATING GASEOUS RELEASES[5]
(charcoal temperature = 20°F)

Maximum core thermal power - 3,758 MWt

Total main steam flow rate - 16.3×10^6 lb/hr

Mass of reactor coolant in the reactor vessel - 5.28×10^5 lb

Mass of steam in the reactor vessel - 1.93×10^4 lb

Holdup Times
Charcoal delay (krypton) - 2.30 days
Charcoal delay (xenon) - 51.1 days

Mass of charcoal in the offgas system - approximately 32 tons

Operating and dew point temperatures of offgas system +20°F and -20°F, respectively

Dynamic adsorption coefficient for Xe and Kr is proprietary (General Electric)

Ventilation and Exhaust Systems

Building	Decontamination Factors (DF)	DF Bases	Purge Rate and Frequency (Reactor Building Only)
Reactor[1]	100	HEPA	5,000 cfm, continuous[4]
	10	Charcoal	30,000 cfm, refueling
Auxiliary[1]	100	HEPA	
	10	Charcoal	
Radwaste[1]	100	HEPA	
	10	Charcoal	
Turbine[2]	1	-	
	1	-	
Offgas[3]	100	HEPA	
	10	Charcoal	

TABLE 11.3-8b (Continued)

NOTES:

(1) The reactor building, auxiliary building and radwaste building releases are through the plant vent.

(2) The turbine building releases are through the turbine building/heater bay vent.

(3) The offgas building releases are through the offgas vent.

(4) Assuming a continuous reactor building purge provides an enveloping dose estimate for routine gaseous releases.

(5) This historical analysis remains bounding based on including Unit 2 operation because input parameters used for calculating gaseous releases for a single unit are less than that attained by dual unit operation.

TABLE 11.3-8c

INPUT PARAMETERS USED FOR CALCULATING GASEOUS RELEASES[5]
(charcoal temperature = 40°F)

Maximum core thermal power - 3,758 MWt

Total main steam flow rate - 16.3×10^6 lb/hr

Mass of reactor coolant in the reactor vessel - 5.28×10^5 lb

Mass of steam in the reactor vessel - 1.93×10^4 lb

Holdup Times
Charcoal delay (krypton) - 1.72 days
Charcoal delay (xenon) - 36.8 days

Mass of charcoal in the offgas system - approximately 32 tons

Operating and dew point temperatures of offgas system +40°F and -20°F, respectively

Dynamic adsorption coefficient for Xe and Kr is proprietary (General Electric)

Ventilation and Exhaust Systems

Building	Decontamination Factors (DF)	DF Bases	Purge Rate and Frequency (Reactor Building Only)
Reactor[1]	100	HEPA	5,000 cfm, continuous[4]
	10	Charcoal	30,000 cfm, refueling
Auxiliary[1]	100	HEPA	
	10	Charcoal	
Radwaste[1]	100	HEPA	
	10	Charcoal	
Turbine[2]	1	-	
	1	-	
Offgas[3]	100	HEPA	
	10	Charcoal	

TABLE 11.3-8c (Continued)

NOTES:

(1) The reactor building, auxiliary building and radwaste building releases are through the plant vent.
(2) The turbine building releases are through the turbine building/heater bay vent.
(3) The offgas building releases are through the offgas vent.
(4) Assuming a continuous reactor building purge provides an enveloping dose estimate for routine gaseous releases.
(5) This historical analysis remains bounding based on including Unit 2 operation because input parameters used for calculating gaseous releases for a single unit are less than that attained by dual unit operation.

TABLE 11.3-8d

INPUT PARAMETERS USED FOR CALCULATING GASEOUS RELEASES[5]
(charcoal temperature = 70°F)

Maximum core thermal power - 3,758 MWt

Total main steam flow rate - 16.3×10^6 lb/hr

Mass of reactor coolant in the reactor vessel - 5.28×10^5 lb

Mass of steam in the reactor vessel - 1.93×10^4 lb

Holdup Times
Charcoal delay (krypton) - 1.16 days
Charcoal delay (xenon) - 23.1 days

Mass of charcoal in the offgas system - approximately 32 tons

Operating and dew point temperatures of offgas system +70°F and -20°F, respectively

Dynamic adsorption coefficient for Xe and Kr is proprietary (General Electric)

Ventilation and Exhaust Systems

Building	Decontamination Factors (DF)	DF Bases	Purge Rate and Frequency (Reactor Building Only)
Reactor[1]	100	HEPA	5,000 cfm, continuous[4]
	10	Charcoal	30,000 cfm, refueling
Auxiliary[1]	100	HEPA	
	10	Charcoal	
Radwaste[1]	100	HEPA	
	10	Charcoal	
Turbine[2]	1	-	
	1	-	
Offgas[3]	100	HEPA	
	10	Charcoal	

TABLE 11.3-8d (Continued)

NOTES:

(1) The reactor building, auxiliary building and radwaste building releases are through the plant vent.

(2) The turbine building releases are through the turbine building/heater bay vent.

(3) The offgas building releases are through the offgas vent.

(4) Assuming a continuous reactor building purge provides an enveloping dose estimate for routine gaseous releases.

(5) This historical analysis remains bounding based on including Unit 2 operation because input parameters used for calculating gaseous releases for a single unit are less than that attained by dual unit operation.

TABLE 11.3-9a

CALCULATED RELEASE OF RADIOACTIVE MATERIALS IN GASEOUS EFFLUENTS - UNIT 1[4]
(Ci/year)

Nuclide	Unit 1 Plant Vent	Unit 1 Turbine Bldg.	Unit 1 Offgas Bldg. Vent[2][3]	Unit 1 Mech. Vac. Pump Discharge
Kr-83m	See Note[1]	See Note[1]	See Note[1]	See Note[1]
Kr-85m	6	68	7	See Note[1]
Kr-85	See Note[1]	See Note[1]	260	See Note[1]
Kr-87	6	130	See Note[1]	See Note[1]
Kr-88	6	230	See Note[1]	See Note[1]
Kr-89	See Note[1]	See Note[1]	See Note[1]	See Note[1]
Xe-131m	See Note[1]	See Note[1]	8	See Note[1]
Xe-133m	See Note[1]	See Note[1]	See Note[1]	See Note[1]
Xe-133	142	250	85	2,300
Xe-135m	92	650	See Note[1]	See Note[1]
Xe-135	113	630	See Note[1]	350
Xe-137	See Note[1]	See Note[1]	See Note[1]	See Note[1]
Xe-138	14	1,400	See Note[1]	See Note[1]
I-131	3.9-2	1.9-1	See Note[1]	3.0-2
I-133	1.5-1	7.6-1	See Note[1]	See Note[1]
Cr-51	9.6-5	1.3-2	-	See Note[1]
Mn-54	3.6-4	6.0-4	-	See Note[1]
Fe-59	1.6-4	5.0-4	-	See Note[1]
Co-58	5.7-5	6.0-4	-	See Note[1]
Co-60	1.1-3	2.0-3	-	See Note[1]
Zn-65	5.5-5	2.0-4	-	See Note[1]
Sr-89	6.3-6	6.0-3	-	See Note[1]
Sr-90	3.1-6	2.0-5	-	See Note[1]
Zr-95	8.5-6	1.0-4	-	See Note[1]
Sb-124	4.7-6	3.0-4	-	See Note[1]
Cs-134	1.3-4	3.0-4	-	3.0-6
Cs-136	1.1-5	5.0-5	-	2.0-6
Cs-137	2.0-4	6.0-4	-	1.0-5
Ba-140	9.0-6	1.1-2	-	1.1-5
Ce-141	2.8-5	6.0-4	-	See Note[1]
C-14	-	-	9.5	-
Ar-41	25	-	13	-
H-3	47	-	-	-

NOTES:

[1] Less than 1 Ci/yr noble gas, less than 10^{-4} Ci/yr iodine.

[2] Charcoal temperature = 0°F.

[3] Design based on 51,300 μCi/sec at T = 30 min, plant capacity factor, 80% per <NUREG-0016>.

[4] This historical analysis remains bounding based on including Unit 2 operation because input parameters used for calculating gaseous releases for a single unit are less than that attained by dual unit operation.

TABLE 11.3-9b

CALCULATED RELEASE OF RADIOACTIVE MATERIALS IN GASEOUS EFFLUENTS - UNIT 1[4]

(Ci/year)

Nuclide	Unit 1 Plant Vent	Unit 1 Turbine Bldg.	Unit 1 Offgas Bldg. Vent[2][3]	Unit 1 Mech. Vac. Pump Discharge
Kr-83m	See Note[1]	See Note[1]	See Note[1]	See Note[1]
Kr-85m	6	68	13	See Note[1]
Kr-85	See Note[1]	See Note[1]	260	See Note[1]
Kr-87	6	130	See Note[1]	See Note[1]
Kr-88	6	230	See Note[1]	See Note[1]
Kr-89	See Note[1]	See Note[1]	See Note[1]	See Note[1]
Xe-131m	See Note[1]	See Note[1]	10	See Note[1]
Xe-133m	See Note[1]	See Note[1]	See Note[1]	See Note[1]
Xe-133	142	250	130	2,300
Xe-135m	92	650	See Note[1]	See Note[1]
Xe-135	113	630	See Note[1]	350
Xe-137	See Note[1]	See Note[1]	See Note[1]	See Note[1]
Xe-138	14	1,400	See Note[1]	See Note[1]
I-131	3.9-2	1.9-1	See Note[1]	3.0-2
I-133	1.5-1	7.6-1	See Note[1]	See Note[1]
Cr-51	9.6-5	1.3-2	-	See Note[1]
Mn-54	3.6-4	6.0-4	-	See Note[1]
Fe-59	1.6-4	5.0-4	-	See Note[1]
Co-58	5.7-5	6.0-4	-	See Note[1]
Co-60	1.1-3	2.0-3	-	See Note[1]
Zn-65	5.5-5	2.0-4	-	See Note[1]
Sr-89	6.3-6	6.0-3	-	See Note[1]
Sr-90	3.1-6	2.0-5	-	See Note[1]
Zr-95	8.5-6	1.0-4	-	See Note[1]
Sb-124	4.7-6	3.0-4	-	See Note[1]
Cs-134	1.3-4	3.0-4	-	3.0-6
Cs-136	1.1-5	5.0-5	-	2.0-6
Cs-137	2.0-4	6.0-4	-	1.0-5
Ba-140	9.0-6	1.1-2	-	1.1-5
Ce-141	2.8-5	6.0-4	-	See Note[1]
C-14	-	-	9.5	-
Ar-41	25	-	43	-
H-3	47	-	-	-

NOTES:

(1) Less than 1 Ci/yr noble gas, less than 10^{-4} Ci/yr iodine.

(2) Charcoal temperature = 20°F.

(3) Design based on 51,300 μCi/sec at T = 30 min, plant capacity factor, 80% per <NUREG-0016>.

(4) This historical analysis remains bounding based on including Unit 2 operation because input parameters used for calculating gaseous releases for a single unit are less than that attained by dual unit operation.

TABLE 11.3-9c

CALCULATED RELEASE OF RADIOACTIVE MATERIALS IN GASEOUS EFFLUENTS - UNIT 1[4]

(Ci/year)

Nuclide	Unit 1 Plant Vent	Unit 1 Turbine Bldg.	Unit 1 Offgas Bldg. Vent[2][3]	Unit 1 Mech. Vac. Pump Discharge
Kr-83m	See Note[1]	See Note[1]	See Note[1]	See Note[1]
Kr-85m	6	68	120	See Note[1]
Kr-85	See Note[1]	See Note[1]	260	See Note[1]
Kr-87	6	130	See Note[1]	See Note[1]
Kr-88	6	230	9	See Note[1]
Kr-89	See Note[1]	See Note[1]	See Note[1]	See Note[1]
Xe-131m	See Note[1]	See Note[1]	23	See Note[1]
Xe-133m	See Note[1]	See Note[1]	See Note[1]	See Note[1]
Xe-133	142	250	840	2,300
Xe-135m	92	650	See Note[1]	See Note[1]
Xe-135	113	630	See Note[1]	350
Xe-137	See Note[1]	See Note[1]	See Note[1]	See Note[1]
Xe-138	14	1,400	See Note[1]	See Note[1]
I-131	3.9-2	1.9-1	See Note[1]	3.0-2
I-133	1.5-1	7.6-1	See Note[1]	See Note[1]
Cr-51	9.6-5	1.3-2	-	See Note[1]
Mn-54	3.6-4	6.0-4	-	See Note[1]
Fe-59	1.6-4	5.0-4	-	See Note[1]
Co-58	5.7-5	6.0-4	-	See Note[1]
Co-60	1.1-3	2.0-3	-	See Note[1]
Zn-65	5.5-5	2.0-4	-	See Note[1]
Sr-89	6.3-6	6.0-3	-	See Note[1]
Sr-90	3.1-6	2.0-5	-	See Note[1]
Zr-95	8.5-6	1.0-4	-	See Note[1]
Sb-124	4.7-6	3.0-4	-	See Note[1]
Cs-134	1.3-4	3.0-4	-	3.0-6
Cs-136	1.1-5	5.0-5	-	2.0-6
Cs-137	2.0-4	6.0-4	-	1.0-5
Ba-140	9.0-6	1.1-2	-	1.1-5
Ce-141	2.8-5	6.0-4	-	See Note[1]
C-14	-	-	9.5	-
Ar-41	25	-	83	-
H-3	47	-	-	-

NOTES:

(1) Less than 1 Ci/yr noble gas, less than 10^{-4} Ci/yr iodine.

(2) Charcoal temperature = 40°F.

(3) Design based on 51,300 μCi/sec at T = 30 min, plant capacity factor, 80% per <NUREG-0016>.

(4) This historical analysis remains bounding based on including Unit 2 operation because input parameters used for calculating gaseous releases for a single unit are less than that attained by dual unit operation.

TABLE 11.3-9d

CALCULATED RELEASE OF RADIOACTIVE MATERIALS IN GASEOUS EFFLUENTS - UNIT 1[4]
(Ci/year)

Nuclide	Unit 1 Plant Vent	Unit 1 Turbine Bldg.	Unit 1 Offgas Bldg. Vent[2][3]	Unit 1 Mech. Vac. Pump Discharge
Kr-83m	See Note[1]	See Note[1]	See Note[1]	See Note[1]
Kr-85m	6	68	990	See Note[1]
Kr-85	See Note[1]	See Note[1]	260	See Note[1]
Kr-87	6	130	See Note[1]	See Note[1]
Kr-88	6	230	260	See Note[1]
Kr-89	See Note[1]	See Note[1]	See Note[1]	See Note[1]
Xe-131m	See Note[1]	See Note[1]	51	See Note[1]
Xe-133m	See Note[1]	See Note[1]	3	See Note[1]
Xe-133	142	250	5,100	2,300
Xe-135m	92	650	See Note[1]	See Note[1]
Xe-135	113	630	See Note[1]	350
Xe-137	See Note[1]	See Note[1]	See Note[1]	See Note[1]
Xe-138	14	1,400	See Note[1]	See Note[1]
I-131	3.9-2	1.9-1	See Note[1]	3.0-2
I-133	1.5-1	7.6-1	See Note[1]	See Note[1]
Cr-51	9.6-5	1.3-2	-	See Note[1]
Mn-54	3.6-4	6.0-4	-	See Note[1]
Fe-59	1.6-4	5.0-4	-	See Note[1]
Co-58	5.7-5	6.0-4	-	See Note[1]
Co-60	1.1-3	2.0-3	-	See Note[1]
Zn-65	5.5-5	2.0-4	-	See Note[1]
Sr-89	6.3-6	6.0-3	-	See Note[1]
Sr-90	3.1-6	2.0-5	-	See Note[1]
Zr-95	8.5-6	1.0-4	-	See Note[1]
Sb-124	4.7-6	3.0-4	-	See Note[1]
Cs-134	1.3-4	3.0-4	-	3.0-6
Cs-136	1.1-5	5.0-5	-	2.0-6
Cs-137	2.0-4	6.0-4	-	1.0-5
Ba-140	9.0-6	1.1-2	-	1.1-5
Ce-141	2.8-5	6.0-4	-	See Note[1]
C-14	-	-	9.5	-
Ar-41	25	-	160	-
H-3	47	-	-	-

NOTES:

(1) Less than 1 Ci/yr noble gas, less than 10^{-4} Ci/yr iodine.

(2) Charcoal temperature = 70°F.

(3) Design based on 51,300 µCi/sec at T = 30 min, plant capacity factor, 80% per <NUREG-0016>.

(4) This historical analysis remains bounding based on including Unit 2 operation because input parameters used for calculating gaseous releases for a single unit are less than that attained by dual unit operation.

TABLE 11.3-10a

CALCULATED RELEASE OF RADIOACTIVE MATERIALS IN GASEOUS EFFLUENTS - UNIT 2[4]
(Ci/year)

Nuclide	Unit 2[5] Plant Vent	Unit 2 Turbine Bldg.	Unit 2 Offgas Bldg. Vent[2][3]	Unit 2 Mech. Vac. Pump Discharge
Kr-83m	See Note[1]	See Note[1]	See Note[1]	See Note[1]
Kr-85m	6	68	7	See Note[1]
Kr-85	See Note[1]	See Note[1]	260	See Note[1]
Kr-87	6	130	See Note[1]	See Note[1]
Kr-88	6	230	See Note[1]	See Note[1]
Kr-89	See Note[1]	See Note[1]	See Note[1]	See Note[1]
Xe-131m	See Note[1]	See Note[1]	8	See Note[1]
Xe-133m	See Note[1]	See Note[1]	See Note[1]	See Note[1]
Xe-133	132	250	85	2,300
Xe-135m	92	650	See Note[1]	See Note[1]
Xe-135	68	630	See Note[1]	350
Xe-137	See Note[1]	See Note[1]	See Note[1]	See Note[1]
Xe-138	14	1,400	See Note[1]	See Note[1]
I-131	3.4-2	1.9-1	See Note[1]	3.0-2
I-133	1.4-1	7.6-1	See Note[1]	See Note[1]
Cr-51	6.0-6	1.3-2	-	See Note[1]
Mn-54	6.0-5	6.0-4	-	See Note[1]
Fe-59	8.0-6	5.0-4	-	See Note[1]
Co-58	1.2-5	6.0-4	-	See Note[1]
Co-60	2.0-4	2.0-3	-	See Note[1]
Zn-65	4.0-5	2.0-4	-	See Note[1]
Sr-89	1.8-6	6.0-3	-	See Note[1]
Sr-90	1.0-7	2.0-5	-	See Note[1]
Zr-95	8.0-6	1.0-4	-	See Note[1]
Sb-124	4.0-6	3.0-4	-	See Note[1]
Cs-134	8.0-5	3.0-4	-	3.0-6
Cs-136	6.0-6	5.0-5	-	2.0-6
Cs-137	1.1-4	6.0-4	-	1.0-5
Ba-140	8.0-6	1.1-2	-	1.1-5
Ce-141	2.0-6	6.0-4	-	See Note[1]
C-14	-	-	9.5	-
Ar-41	25	-	13	-
H-3	47	-	-	-

TABLE 11.3-10a (Continued)

NOTES:

(1) Less than 1 Ci/yr noble gas, less than 10^{-4} Ci/yr iodine.

(2) Charcoal temperature = 0°F.

(3) Design based on 51,300 μCi/sec at T = 30 min, plant capacity factor, 80% <NUREG-0016>.

(4) The information in this table was calculated with the expectation that Unit 2 would achieve commercial operation. A later decision was made not to complete Unit 2 and subsequently, the unit was abandoned. However, this historical analysis remains valid because input parameters used for calculating gaseous releases for a single unit are less than that attained by dual unit operation.

(5) Unit 2 plant vent has active inputs from Unit 1.

TABLE 11.3-10b

CALCULATED RELEASE OF RADIOACTIVE MATERIALS IN GASEOUS EFFLUENTS - UNIT 2[4]
(Ci/year)

Nuclide	Unit 2[5] Plant Vent	Unit 2 Turbine Bldg.	Unit 2 Offgas Bldg. Vent[2][3]	Unit 2 Mech. Vac. Pump Discharge
Kr-83m	See Note[1]	See Note[1]	See Note[1]	See Note[1]
Kr-85m	6	68	13	See Note[1]
Kr-85	See Note[1]	See Note[1]	260	See Note[1]
Kr-87	6	130	See Note[1]	See Note[1]
Kr-88	6	230	See Note[1]	See Note[1]
Kr-89	See Note[1]	See Note[1]	See Note[1]	See Note[1]
Xe-131m	See Note[1]	See Note[1]	10	See Note[1]
Xe-133m	See Note[1]	See Note[1]	See Note[1]	See Note[1]
Xe-133	132	250	130	2,300
Xe-135m	92	650	See Note[1]	See Note[1]
Xe-135	68	630	See Note[1]	350
Xe-137	See Note[1]	See Note[1]	See Note[1]	See Note[1]
Xe-138	14	1,400	See Note[1]	See Note[1]
I-131	3.4-2	1.9-1	See Note[1]	3.0-2
I-133	1.4-1	7.6-1	See Note[1]	See Note[1]
Cr-51	6.0-6	1.3-2	-	See Note[1]
Mn-54	6.0-5	6.0-4	-	See Note[1]
Fe-59	8.0-6	5.0-4	-	See Note[1]
Co-58	1.2-5	6.0-4	-	See Note[1]
Co-60	2.0-4	2.0-3	-	See Note[1]
Zn-65	4.0-5	2.0-4	-	See Note[1]
Sr-89	1.8-6	6.0-3	-	See Note[1]
Sr-90	1.0-7	2.0-5	-	See Note[1]
Zr-95	8.0-6	1.0-4	-	See Note[1]
Sb-124	4.0-6	3.0-4	-	See Note[1]
Cs-134	8.0-5	3.0-4	-	3.0-6
Cs-136	6.0-6	5.0-5	-	2.0-6
Cs-137	1.1-4	6.0-4	-	1.0-5
Ba-140	8.0-6	1.1-2	-	1.1-5
Ce-141	2.0-6	6.0-4	-	See Note[1]
C-14	-	-	9.5	-
Ar-41	25	-	43	-
H-3	47	-	-	-

TABLE 11.3-10b (Continued)

NOTES:

(1) Less than 1 Ci/yr noble gas, less than 10^{-4} Ci/yr iodine.
(2) Charcoal temperature = 20°F.
(3) Design based on 51,300 μCi/sec at T = 30 min, plant capacity factor, 80% <NUREG-0016>.
(4) The information in this table was calculated with the expectation that Unit 2 would achieve commercial operation. A later decision was made not to complete Unit 2 and subsequently, the unit was abandoned. However, this historical analysis remains valid because input parameters used for calculating gaseous releases for a single unit are less than that attained by dual unit operation.
(5) Unit 2 plant vent has active inputs from Unit 1.

TABLE 11.3-10c

CALCULATED RELEASE OF RADIOACTIVE MATERIALS IN GASEOUS EFFLUENTS - UNIT 2[4]

(Ci/year)

Nuclide	Unit 2[5] Plant Vent	Unit 2 Turbine Bldg.	Unit 2 Offgas Bldg. Vent[2][3]	Unit 2 Mech. Vac. Pump Discharge
Kr-83m	See Note[1]	See Note[1]	See Note[1]	See Note[1]
Kr-85m	6	68	120	See Note[1]
Kr-85	See Note[1]	See Note[1]	260	See Note[1]
Kr-87	6	130	See Note[1]	See Note[1]
Kr-88	6	230	9	See Note[1]
Kr-89	See Note[1]	See Note[1]	See Note[1]	See Note[1]
Xe-131m	See Note[1]	See Note[1]	23	See Note[1]
Xe-133m	See Note[1]	See Note[1]	See Note[1]	See Note[1]
Xe-133	132	250	840	2,300
Xe-135m	92	650	See Note[1]	See Note[1]
Xe-135	68	630	See Note[1]	350
Xe-137	See Note[1]	See Note[1]	See Note[1]	See Note[1]
Xe-138	14	1,400	See Note[1]	See Note[1]
I-131	3.4-2	1.9-1	See Note[1]	3.0-2
I-133	1.4-1	7.6-1	See Note[1]	See Note[1]
Cr-51	6.0-6	1.3-2	-	See Note[1]
Mn-54	6.0-5	6.0-4	-	See Note[1]
Fe-59	8.0-6	5.0-4	-	See Note[1]
Co-58	1.2-5	6.0-4	-	See Note[1]
Co-60	2.0-4	2.0-3	-	See Note[1]
Zn-65	4.0-5	2.0-4	-	See Note[1]
Sr-89	1.8-6	6.0-3	-	See Note[1]
Sr-90	1.0-7	2.0-5	-	See Note[1]
Zr-95	8.0-6	1.0-4	-	See Note[1]
Sb-124	4.0-6	3.0-4	-	See Note[1]
Cs-134	8.0-5	3.0-4	-	3.0-6
Cs-136	6.0-6	5.0-5	-	2.0-6
Cs-137	1.1-4	6.0-4	-	1.0-5
Ba-140	8.0-6	1.1-2	-	1.1-5
Ce-141	2.0-6	6.0-4	-	See Note[1]
C-14	-	-	9.5	-
Ar-41	25	-	83	-
H-3	47	-	-	-

TABLE 11.3-10c (Continued)

NOTES:

(1) Less than 1 Ci/yr noble gas, less than 10^{-4} Ci/yr iodine.
(2) Charcoal temperature = 40°F.
(3) Design based on 51,300 μCi/sec at T = 30 min, plant capacity factor, 80% <NUREG-0016>.
(4) The information in this table was calculated with the expectation that Unit 2 would achieve commercial operation. A later decision was made not to complete Unit 2 and subsequently, the unit was abandoned. However, this historical analysis remains valid because input parameters used for calculating gaseous releases for a single unit are less than that attained by dual unit operation.
(5) Unit 2 plant vent has active inputs from Unit 1.

TABLE 11.3-10d

CALCULATED RELEASE OF RADIOACTIVE MATERIALS IN GASEOUS EFFLUENTS - UNIT 2[4]
(Ci/year)

Nuclide	Unit 2[5] Plant Vent	Unit 2 Turbine Bldg.	Unit 2 Offgas Bldg. Vent[2][3]	Unit 2 Mech. Vac. Pump Discharge
Kr-83m	See Note[1]	See Note[1]	See Note[1]	See Note[1]
Kr-85m	6	68	990	See Note[1]
Kr-85	See Note[1]	See Note[1]	260	See Note[1]
Kr-87	6	130	See Note[1]	See Note[1]
Kr-88	6	230	260	See Note[1]
Kr-89	See Note[1]	See Note[1]	See Note[1]	See Note[1]
Xe-131m	See Note[1]	See Note[1]	51	See Note[1]
Xe-133m	See Note[1]	See Note[1]	3	See Note[1]
Xe-133	132	250	5,100	2,300
Xe-135m	92	650	See Note[1]	See Note[1]
Xe-135	68	630	See Note[1]	350
Xe-137	See Note[1]	See Note[1]	See Note[1]	See Note[1]
Xe-138	14	1,400	See Note[1]	See Note[1]
I-131	3.4-2	1.9-1	See Note[1]	3.0-2
I-133	1.4-1	7.6-1	See Note[1]	See Note[1]
Cr-51	6.0-6	1.3-2	-	See Note[1]
Mn-54	6.0-5	6.0-4	-	See Note[1]
Fe-59	8.0-6	5.0-4	-	See Note[1]
Co-58	1.2-5	6.0-4	-	See Note[1]
Co-60	2.0-4	2.0-3	-	See Note[1]
Zn-65	4.0-5	2.0-4	-	See Note[1]
Sr-89	1.8-6	6.0-3	-	See Note[1]
Sr-90	1.0-7	2.0-5	-	See Note[1]
Zr-95	8.0-6	1.0-4	-	See Note[1]
Sb-124	4.0-6	3.0-4	-	See Note[1]
Cs-134	8.0-5	3.0-4	-	3.0-6
Cs-136	6.0-6	5.0-5	-	2.0-6
Cs-137	1.1-4	6.0-4	-	1.0-5
Ba-140	8.0-6	1.1-2	-	1.1-5
Ce-141	2.0-6	6.0-4	-	See Note[1]
C-14	-	-	9.5	-
Ar-41	25	-	160	-
H-3	47	-	-	-

TABLE 11.3-10d (Continued)

NOTES:

(1) Less than 1 Ci/yr noble gas, less than 10^{-4} Ci/yr iodine.

(2) Charcoal temperature = 70°F.

(3) Design based on 51,300 μCi/sec at T = 30 min, plant capacity factor, 80% <NUREG-0016>.

(4) The information in this table was calculated with the expectation that Unit 2 would achieve commercial operation. A later decision was made not to complete Unit 2 and subsequently, the unit was abandoned. However, this historical analysis remains valid because input parameters used for calculating gaseous releases for a single unit are less than that attained by dual unit operation.

(5) Unit 2 plant vent has active inputs from Unit 1.

TABLE 11.3-11a

AVERAGE ANNUAL CONCENTRATIONS OF GASEOUS EFFLUENTS AT EXCLUSION BOUNDARY[5]

Nuclide	Annual Release (Ci/yr - two units)[3][4]	MPC (μCi/cc)	Fraction of MPC[1]	<10 CFR 20, Appendix B> Effluent Concentrations (μCi/cc)	Fraction of Effluent Concentrations[1]
Kr-83m	See Note[2]	3.-8	-	5E-5	-
Kr-85m	162	1.-7	1.4-4	1.-7	1.4-4
Kr-85	520	3.-7	1.5-4	7E-7	6.4E-5
Kr-87	272	2.-8	1.2-3	2.-8	1.2-3
Kr-88	472	2.-8	2.0-3	9E-9	4.4E-3
Kr-89	See Note[2]	3.-8	-	1E-9	-
Xe-131m	16	4.-7	3.4-6	2E-6	6.8E-7
Xe-133m	See Note[2]	3.-7	-	6E-7	-
Xe-133	5,544	3.-7	1.6-3	5E-7	9.6E-4
Xe-135m	1,484	3.-8	4.2-3	4E-8	3.1E-3
Xe-135	2,141	1.-7	1.8-3	7E-8	2.6E-3
Xe-137	See Note[2]	3.-8	-	1E-9	-
Xe-138	2,828	3.-8	8.1-3	2E-8	1.2E-2
I-131	.51	1.-8	4.4-6	2E-10	2.2E-4
I-133	1.8	7.-9	2.2-5	1E-9	1.5E-4
Cr-51	1.3-2	8.-8	1.4-8	3E-8	3.7E-8
Mn-54	1.6-3	1.-9	1.4-7	1.-9	1.4-7
Fe-59	1.2-3	2.-9	5.1-8	5E-10	2.0E-7
Co-58	1.3-3	2.-9	5.6-8	1E-9	1.1E-7
Co-60	5.3-3	3.-10	1.5-6	5E-11	9.0E-6
Zn-65	5.0-4	2.-9	2.1-8	4E-10	1.0E-7
Sr-89	1.2-2	1.-9	1.0-6	2E-10	5.0E-6
Sr-90	4.3-5	2.-10	1.8-8	6E-12	6.0E-7
Zr-95	2.2-4	1.-9	1.9-8	4E-10	4.7E-8
Sb-124	6.1-4	3.-8	1.7-9	3E-10	1.7E-7
Cs-134	8.1-4	4.-10	1.7-7	2E-10	3.4E-7
Cs-136	1.2-4	6.-9	1.7-9	9E-10	1.1E-8
Cs-137	1.5-3	5.-10	2.6-7	2E-10	6.5E-7
Ba-140	2.2-2	1.-9	1.9-6	2E-9	9.5E-7
Ce-141	1.2-3	5.-9	2.1-8	8E-10	1.3E-7
C-14	19	1.-7	1.6-5	3E-9	5.3E-4
Ar-41	76	4.-8	1.6-4	1E-8	6.4E-4
H-3	94	2.-7	4.0-5	1E-7	8.0E-5

NOTES:

(1) Based on an average annual χ/Q of 2.7-6 sec/m^3.

(2) Less than 1 Ci/yr noble gas, less than 10^{-4} Ci/yr iodine.

(3) Offgas system charcoal temperature = 0°F.

(4) Design based on 51,300 μCi/sec at T = 30 min, plant capacity factor, 80% per <NUREG-0016>.

(5) The "Average Annual Concentrations of Gaseous Effluents at Exclusion Boundary" data provided in Table 11.3-11a through 11.3-11d were calculated with the expectation that Unit 2 would achieve commercial operation. A later decision was made to not complete Unit 2 and subsequently, the unit was abandoned. However, this historical analysis remains bounding based on including Unit 2 operation because single unit annual releases are less than those attained by dual unit operation.

TABLE 11.3-11b

AVERAGE ANNUAL CONCENTRATIONS OF GASEOUS EFFLUENTS AT EXCLUSION BOUNDARY[5]

Nuclide	Annual Release (Ci/yr - two units)[3][4]	MPC (μCi/cc)	Fraction of MPC[1]	<10 CFR 20, Appendix B> Effluent Concentrations (μCi/cc)	Fraction of Effluent Concentrations[1]
Kr-83m	See Note[2]	3.-8	-	5E-5	-
Kr-85m	174	1.-7	1.5-4	1.-7	1.5-4
Kr-85	520	3.-7	1.5-4	7E-7	6.4E-5
Kr-87	272	2.-8	1.2-3	2.-8	1.2-3
Kr-88	472	2.-8	2.0-3	9E-9	4.4E-3
Kr-89	See Note[2]	3.-8	-	1E-9	-
Xe-131m	20	4.-7	4.3-6	2E-6	8.6E-7
Xe-133m	See Note[2]	3.-7	-	6E-7	-
Xe-133	5,634	3.-7	1.6-3	5E-7	9.6E-4
Xe-135m	1,484	3.-8	4.2-3	4E-8	3.1E-3
Xe-135	2,141	1.-7	1.8-3	7E-8	2.6E-3
Xe-137	See Note[2]	3.-8	-	1E-9	-
Xe-138	2,828	3.-8	8.1-3	2E-8	1.2E-2
I-131	.51	1.-8	4.4-6	2E-10	2.2E-4
I-133	1.8	7.-9	2.2-5	1E-9	1.5E-4
Cr-51	1.3-2	8.-8	1.4-8	3E-8	3.7E-8
Mn-54	1.6-3	1.-9	1.4-7	1.-9	1.4-7
Fe-59	1.2-3	2.-9	5.1-8	5E-10	2.0E-7
Co-58	1.3-3	2.-9	5.6-8	1E-9	1.1E-7
Co-60	5.3-3	3.-10	1.5-6	5E-11	9.0E-6
Zn-65	5.0-4	2.-9	2.1-8	4E-10	1.0E-7
Sr-89	1.2-2	1.-9	1.0-6	2E-10	5.0E-6
Sr-90	4.3-5	2.-10	1.8-8	6E-12	6.0E-7
Zr-95	2.2-4	1.-9	1.9-8	4E-10	4.7E-8
Sb-124	6.1-4	3.-8	1.7-9	3E-10	1.7E-7
Cs-134	8.1-4	4.-10	1.7-7	2E-10	3.4E-7
Cs-136	1.2-4	6.-9	1.7-9	9E-10	1.1E-8
Cs-137	1.5-3	5.-10	2.6-7	2E-9	6.5E-7
Ba-140	2.2-2	1.-9	1.9-6	8E-10	9.5E-7
Ce-141	1.2-3	5.-9	2.1-8	3E-9	1.3E-7
C-14	19	1.-7	1.6-5	1E-8	5.3E-4
Ar-41	136	4.-8	2.9-4	1E-8	1.2E-3
H-3	94	2.-7	4.0-5	1E-7	8.0E-5

NOTES:

(1) Based on an average annual χ/Q of 2.7-6 sec/m^3.

(2) Less than 1 Ci/yr noble gas, less than 10^{-4} Ci/yr iodine.

(3) Offgas system charcoal temperature = 20°F.

(4) Design based on 51,300 μCi/sec at T = 30 min, plant capacity factor, 80% per <NUREG-0016>.

(5) The "Average Annual Concentrations of Gaseous Effluents at Exclusion Boundary" data provided in Table 11.3-11a through 11.3-11d were calculated with the expectation that Unit 2 would achieve commercial operation. A later decision was made to not complete Unit 2 and subsequently, the unit was abandoned. However, this historical analysis remains bounding based on including Unit 2 operation because single unit annual releases are less than those attained by dual unit operation.

TABLE 11.3-11c

AVERAGE ANNUAL CONCENTRATIONS OF GASEOUS EFFLUENTS AT EXCLUSION BOUNDARY[5]

Nuclide	Annual Release (Ci/yr - two units)[3][4]	MPC (μCi/cc)	Fraction of MPC[1]	<10 CFR 20, Appendix B> Effluent Concentrations (μCi/cc)	Fraction of Effluent Concentrations[1]
Kr-83m	See Note[2]	3.-8	-	5E-5	-
Kr-85m	388	1.-7	3.3-4	1.-7	3.3-4
Kr-85	520	3.-7	1.5-4	7E-7	6.4E-5
Kr-87	272	2.-8	1.2-3	2.-8	1.2-3
Kr-88	490	2.-8	2.1-3	9E-9	4.7E-3
Kr-89	See Note[2]	3.-8	-	1E-9	-
Xe-131m	46	4.-7	9.8-6	2E-6	2.0E-6
Xe-133m	See Note[2]	3.-7	-	6E-7	-
Xe-133	7,054	3.-7	2.0-3	5E-7	1.2E-3
Xe-135m	1,484	3.-8	4.2-3	4E-8	3.1E-3
Xe-135	2,141	1.-7	1.8-3	7E-8	2.6E-3
Xe-137	See Note[2]	3.-8	-	1E-9	-
Xe-138	2,828	3.-8	8.1-3	2E-8	1.2E-2
I-131	.51	1.-8	4.4-6	2E-10	2.2E-4
I-133	1.8	7.-9	2.2-5	1E-9	1.5E-4
Cr-51	1.3-2	8.-8	1.4-8	3E-8	3.7E-8
Mn-54	1.6-3	1.-9	1.4-7	1.-9	1.4-7
Fe-59	1.2-3	2.-9	5.1-8	5E-10	2.0E-7
Co-58	1.3-3	2.-9	5.6-8	1E-9	1.1E-7
Co-60	5.3-3	3.-10	1.5-6	5E-11	9.0E-6
Zn-65	5.0-4	2.-9	2.1-8	4E-10	1.0E-7
Sr-89	1.2-2	1.-9	1.0-6	2E-10	5.0E-6
Sr-90	4.3-5	2.-10	1.8-8	6E-12	6.0E-7
Zr-95	2.2-4	1.-9	1.9-8	4E-10	4.7E-8
Sb-124	6.1-4	3.-8	1.7-9	3E-10	1.7E-7
Cs-134	8.1-4	4.-10	1.7-7	2E-10	3.4E-7
Cs-136	1.2-4	6.-9	1.7-9	9E-10	1.1E-8
Cs-137	1.5-3	5.-10	2.6-7	2E-10	6.5E-7
Ba-140	2.2-2	1.-9	1.9-6	2E-9	9.5E-7
Ce-141	1.2-3	5.-9	2.1-8	8E-10	1.3E-7
C-14	19	1.-7	1.6-5	3E-9	5.3E-4
Ar-41	216	4.-8	4.6-4	1E-8	1.8E-3
H-3	94	2.-7	4.0-5	1E-7	8.0E-5

NOTES:

(1) Based on an average annual χ/Q of 2.7-6 sec/m^3.

(2) Less than 1 Ci/yr noble gas, less than 10^{-4} Ci/yr iodine.

(3) Offgas system charcoal temperature = 40°F.

(4) Design based on 51,300 μCi/sec at T = 30 min, plant capacity factor, 80% per <NUREG-0016>.

(5) The "Average Annual Concentrations of Gaseous Effluents at Exclusion Boundary" data provided in Table 11.3-11a through 11.3-11d were calculated with the expectation that Unit 2 would achieve commercial operation. A later decision was made to not complete Unit 2 and subsequently, the unit was abandoned. However, this historical analysis remains bounding based on including Unit 2 operation because single unit annual releases are less than those attained by dual unit operation.

TABLE 11.3-11d

AVERAGE ANNUAL CONCENTRATIONS OF GASEOUS EFFLUENTS AT EXCLUSION BOUNDARY[5]

Nuclide	Annual Release (Ci/yr - two units)[3][4]	MPC (μCi/cc)	Fraction of MPC[1]	<10 CFR 20, Appendix B> Effluent Concentrations (μCi/cc)	Fraction of Effluent Concentrations[1]
Kr-83m	See Note[2]	3.-8	5.7-6	5E-5	3.4E-9
Kr-85m	2,128	1.-7	1.8-3	1.-7	1.8-3
Kr-85	520	3.-7	1.5-4	7E-7	6.4E-5
Kr-87	272	2.-8	1.2-3	2.-8	1.2-3
Kr-88	992	2.-8	4.2-3	9E-9	9.3E-3
Kr-89	See Note[2]	3.-8	-	1E-9	-
Xe-131m	102	4.-7	2.2-5	2E-6	4.4E-6
Xe-133m	6	3.-7	1.7-6	6E-7	8.5E-7
Xe-133	15,574	3.-7	4.4-3	5E-7	2.6E-3
Xe-135m	1,484	3.-8	4.2-3	4E-8	3.1E-3
Xe-135	2,141	1.-7	1.8-3	7E-8	2.6E-3
Xe-137	See Note[2]	3.-8	-	1E-9	-
Xe-138	2,828	3.-8	8.1-3	2E-8	1.2E-2
I-131	.51	1.-8	4.4-6	2E-10	2.2E-4
I-133	1.8	7.-9	2.2-5	1E-9	1.5E-4
Cr-51	1.3-2	8.-8	1.4-8	3E-8	3.7E-8
Mn-54	1.6-3	1.-9	1.4-7	1.-9	1.4-7
Fe-59	1.2-3	2.-9	5.1-8	5E-10	2.0E-7
Co-58	1.3-3	2.-9	5.6-8	1E-9	1.1E-7
Co-60	5.3-3	3.-10	1.5-6	5E-11	9.0E-6
Zn-65	5.0-4	2.-9	2.1-8	4E-10	1.0E-7
Sr-89	1.2-2	1.-9	1.0-6	2E-10	5.0E-6
Sr-90	4.3-5	2.-10	1.8-8	6E-12	6.0E-7
Zr-95	2.2-4	1.-9	1.9-8	4E-10	4.7E-8
Sb-124	6.1-4	3.-8	1.7-9	3E-10	1.7E-7
Cs-134	8.1-4	4.-10	1.7-7	2E-10	3.4E-7
Cs-136	1.2-4	6.-9	1.7-9	9E-10	1.1E-8
Cs-137	1.5-3	5.-10	2.6-7	2E-10	6.5E-7
Ba-140	2.2-2	1.-9	1.9-6	2E-9	9.5E-7
Ce-141	1.2-3	5.-9	2.1-8	8E-10	1.3E-7
C-14	19	1.-7	1.6-5	3E-9	5.3E-4
Ar-41	370	4.-8	7.9-4	1E-8	3.2E-3
H-3	94	2.-7	4.0-5	1E-7	8.0E-5

NOTES:

(1) Based on an average annual χ/Q of 2.7-6 sec/m^3.

(2) Less than 1 Ci/yr noble gas, less than 10^{-4} Ci/yr iodine.

(3) Offgas system charcoal temperature = 70°F.

(4) Design based on 51,300 μCi/sec at T = 30 min, plant capacity factor, 80% per <NUREG-0016>.

(5) The "Average Annual Concentrations of Gaseous Effluents at Exclusion Boundary" data provided in Table 11.3-11a through 11.3-11d were calculated with the expectation that Unit 2 would achieve commercial operation. A later decision was made to not complete Unit 2 and subsequently, the unit was abandoned. However, this historical analysis remains bounding based on including Unit 2 operation because single unit annual releases are less than those attained by dual unit operation.

11.4 SOLID RADIOACTIVE WASTE MANAGEMENT SYSTEM

11.4.1 DESIGN BASES

11.4.1.1 Power Generation Design Objectives

The primary design objective of the solid radioactive waste (SRW) system is to control, collect, handle, process, and package all wet and dry solid radioactive waste generated by PNPP as a result of normal operation, and to store these wastes until they are shipped to authorized receiving and storage areas offsite. This will be done in such a manner that, for all anticipated quantities of waste produced, the availability of the power plant for power generation will not be adversely affected.

The types of solid radioactive waste to be processed, anticipated quantities and curie content are given in <Table 11.4-1> for the original design basis of the solid radioactive waste system. Waste quantities and curie content by isotope are given in <Table 11.4-2>, <Table 11.4-3>, and <Table 11.4-4>.

Subsequent to the original design of the SRW system, modifications have been made to the condensate cleanup and liquid radwaste systems which could result in quantities or activities of solid radioactive waste which are different than those in the original design basis. The control, processing and packaging of the solid radioactive waste remains unchanged. The ALARA design features, as discussed in <Section 11.4.1.5>, and the safety precautions, as discussed in <Section 11.4.1.6> are unaffected by the changes to the quantities or activities of the waste to be processed.

11.4.1.2 Radiological Design Objectives

Packaging of solid radioactive material is accomplished in a manner which ensures that no radioactive material will be released to the environment during shipment of the waste to offsite burial or storage facilities. The SRW system is designed to limit exposures to both operating personnel and the general public to as low as reasonably achievable.

11.4.1.3 Design Criteria

a. The SRW system components, piping and the structure that houses the system are designed and fabricated in accordance with the codes, standards, seismic categories, and quality group classifications given in <Table 3.2-1>.

b. The SRW system design is in compliance with the guidance provided by <Regulatory Guide 1.143> and Branch Technical Position ETSB 11-3.

c. All wet radioactive waste (filter backwash slurries and spent resins) are processed per the Process Control Program (Reference 1) prior to shipment offsite. Packaging and transporting of radioactive wastes is performed in conformance with <10 CFR 71> and applicable ICC and DOT regulations.

d. The SRW system design and shielding provisions ensure that (during all phases of processing, handling and shipment of radioactive waste) exposure to operating personnel and the general public is within the applicable limits of <10 CFR 20>, <49 CFR 173> and as low as is reasonably achievable in accordance with <Regulatory Guide 8.8>.

e. The Process Control Program (Reference 1) provides a means to verify the absence of free liquid in the containers in accordance with Branch Technical Position ETSB 11-3.

f. The SRW system design, equipment sizing and equipment redundancy ensure that the maximum expected quantities of all radioactive waste inputs during any 30 day period can be prepared for shipping via the Process Control Program (Reference 1) and temporarily stored onsite without affecting plant availability. Design quantities of radioactive waste inputs to the SRW system are presented in <Table 11.4-1>.

11.4.1.4 Component Design Parameters

With the exception of normal wearing parts, such as seals and bearings, all pumps, valves, piping, tanks, and other components in the SRW system are fabricated from materials which are intended to provide a minimum service life of 40 years without replacement. In selecting materials to meet this criterion, due consideration is given to: a) the corrosive nature of both the process medium and the external environment, b) decontaminability of the material, and c) wall thickness requirements dictated by design pressures, flow rates and corrosion rates. The design classifications of SRW system equipment items are given in <Table 3.2-1>.

11.4.1.5 ALARA Design Features

Numerous features have been incorporated into the design of both the SRW system and the building housing this system to ensure that exposures of operating personnel to radiation will be kept within ALARA guidelines. See <Section 11.2.1.9> for a listing of the most significant ALARA design features.

11.4.1.6 Safety Precautions

All tanks, pumps and other equipment containing radioactive liquids are located in shielded cubicles or pipe chases. All access to these areas is strictly controlled by administrative procedures.

11.4.2 SYSTEM DESCRIPTION

11.4.2.1 Treatment of Wet Solid Radioactive Waste

NOTE: Mobile radioactive waste processing is used in combination with portions of the SRW system described in this Chapter. The details of the mobile SRW package and its interface with the SRW processing described in this chapter are contained in the Perry Process Control Program (Reference 1).

The types, anticipated quantities and expected activity levels of wet solid radioactive waste to be processed are identified in <Table 11.4-1>.

The system diagram is presented in <Figure 11.4-1>. This diagram shows the process flow routes, process flow conditions, equipment capacities, instrumentation, and system design data.

Instrumentation, controls, alarms, and protection devices are discussed under <Section 11.4.2.4>.

The SRW system is designed to process spent resin slurry, precoat-type filter backwash slurry, and traveling belt discharge cake. These waste streams are transferred from the LRW system

collection tanks to a vendors dewatering system. After this transfer, the fill isolation valve is closed and the fill line is backflushed to the tank from which the waste stream originated.

Processing of the waste is controlled from the SRW control panel and vendor system control panel. Using selector switches on this panel, the operator selects which waste storage tank to take waste from filling the waste container is controlled from the vendor system control panel.

The method of waste processing is detailed in the Process Control Program.

Once onsite processing of the waste is complete, the overhead bridge crane picks up the container and takes it either to a short term storage area or to a truck bay where it is loaded for transfer to an authorized receiving, reprocessing, or storage area.

11.4.2.1.1 Component Failure and System Malfunctions

The SRW system is designed to preclude the accidental release of radioactive waste into the solid waste packaging area due to component failure or system malfunctions. Instrumentation and controls monitor each phase of the packaging operation, serving to detect possible system malfunctions and terminate the packaging operation as required to prevent inadvertent releases of radioactive waste into the solid waste packaging area. Full operator surveillance is maintained during the entire packaging operation through CCTV monitors located on and adjacent to the control panel. Means are provided for the operator to terminate the

packaging operation in instances of component failures which may cause the release of radioactive materials from the SRW system. The possibility of component failures is considered very low because of the low pressures at which the packaging operation occurs.

The interface between the vendor's system and permanent plant equipment is evaluated for accidental releases of radioactivity before the mobile system is approved for use.

The air flow patterns in the drumming station are such that any radioactive gases released would pass into the radwaste ventilation system, and be treated by a series of roughing, HEPA and charcoal filters prior to release to the environs <Figure 9.4-7>.

11.4.2.2 Treatment of Dry Solid Radioactive Waste

A dry solid radwaste subsystem is provided for processing dry filter media (ventilation filters), contaminated clothing, equipment, tools and glassware, and miscellaneous radioactive wastes that are not amenable to solidification prior to packaging.

Potentially radioactively contaminated waste and radioactive material such as tooling, components and equipment are collected throughout the RRA and brought to one of two main areas: the Waste Abatement and Reclamation Facility (WARF) or the DAW handling area on the 623'-6" elevation of the radwaste building. Other areas may be established temporarily based on operational needs as determined by the Radiation Protection Manager (RPM).

The types, anticipated quantities and expected activity levels of dry solid radioactive wastes are identified in <Table 11.4-1>. These numbers are based on operating plant data.

11.4.2.2.1 Compressible Dry Solid Radioactive Waste

Radioactive Material, contaminated cloth, paper, glass, floor sweepings, and similar low-level activity wastes are accumulated in designated storage areas. The waste is stored until a sufficient amount accumulates to warrant shipment to an authorized processing and storage area located offsite. All radioactive material and stored DAW are kept in metal containers or areas protected by fire suppression systems while onsite.

11.4.2.2.2 Incompressible Dry Solid Radioactive Waste

Spent filter cartridges, air filter elements, contaminated tools, and similar incompressible solid wastes are packaged in various size shipping containers depending on their size. Shielding is provided around the shipping container as required. Highly radioactive material is centered in the shipping container and solidification agent is added, thus providing additional shielding.

11.4.2.2.3 Segregation of Clean and Contaminated Loose Wastes

Normally segregation of clean and contaminated loose wastes is contracted to a licensed offsite vendor. Potentially contaminated waste will be monitored for radioactivity levels above background before disposal as clean. Material exhibiting any level of radioactivity above background, as demonstrated by the use of this equipment (or other equipment utilizing the same type of sensitive monitors) will either be decontaminated or disposed of as radioactive waste. An aggregate of this sorted clean waste and other clean waste from the RRA will be monitored before disposal as clean. This program is in compliance with <NRC Notice 85-92>.

11.4.2.3 Detailed Component Design

All items under this Section address the permanent plant equipment that will interface with a vendor’s mobile system.

a. Collection Tank Design

These tanks are treated as a part of the LRW system; refer to <Section 11.2.2.10.a>, for this information.

b. General Pump Design

All pumps, whether centrifugal or positive displacement, are designed to the requirements of the Hydraulic Institute Standards for rating, testing, application, and materials. For pumps handling radioactive fluids, shafts are sealed with mechanical seals which are balanced, single (or double if process fluid necessitates) seals with a carbon stationary insert, ceramic rotating seal ring, silicone or “EPR” elastomer O-rings, 316L SS metal parts, flushing connection, vent and drain connection, and throttle bushing (for single mechanical seals only). The vent and drain connections and the throttle bushings are provided to permit installation of a drain for the fluid that leaks from a worn seal. The bearing lubrication that may leak out of the lubrication system will be allowed to accumulate on the pump base separate from the pump shaft seal drain piping. A solenoid operated shutoff valve is provided for control of seal water to each pump with mechanical seals. This valve is designed to open when the pump is started, to close when the pump is stopped, and to fail open on loss of power.

c. Waste Mixing/Dewatering Tanks

Two redundant mixing/dewatering tanks are provided in shielded cubicles at Elevation 630’-0”. Each tank is an atmospheric,

750 gallon, vertical, cone bottomed, 316L stainless steel vessel mounted off the floor on carbon steel support legs. Connections are provided for vent/overflow, concentrate and slurry waste feeds, flushwater, level monitors, traveling belt filter chute discharge, dewatering, and drain.

A tank mixer is mounted on top of the tank and is controlled from the SRW control panel. A manway with hinged cover is also located on top of the tank. Inside the tank are the tank washdown nozzles, mixer blades all constructed of 316 or 316L stainless steel.

d. Waste Dewatering Pumps

The dewatering pump is mounted on a base plate attached to the legs of the mixing/dewatering tank. It is a 10 gpm, motor driven centrifugal pump, controlled from the SRW control panel. The pump has two suction connections. The upper connection is not used. When used to drain the tank, it takes suction from a connection near the bottom of the tank. The dewatering pump is constructed of 316 stainless steel. Pump seals are single mechanical type.

e. Waste Feed Pumps

The waste feed pump is mounted on a skid plate attached to the legs of the mixing/dewatering tank. It is a progressing cavity, positive displacement, metering pump built to food industry standards. It is driven by an SCR variable speed, dc motor and is controlled from the SRW control panel. The SCR controller can be reset to adjust the pump flow rate from 15 to 40 gpm. Portions of the feed pump in contact with radioactive fluids are constructed of 316L stainless steel. Seals are double mechanical type.

f. Overhead Bridge Crane

The bridge crane has a rated capacity of 15 tons and a span of 34’-3”. It is mounted on 60 pound ASCE rails that allow full travel of the crane in the north-south direction between column lines RW-A and RW-D, permitting full access to the truck bay, temporary storage facility and processing gallery.

The unit is controlled entirely from the SRW control panel. A 3-position digital indexing and readout system on the control panel indicates where the bridge, trolley and hoist are at all times. In addition to this system, the operator can view all movements of the crane on a closed circuit TV monitor. For maintenance purposes, a local control station is provided, with controls for bridge, hoist and trolley.

The bridge, trolley and hoist have both high and low speeds; the former is for rough positioning and the latter is for accurate final positioning. High/low speeds for the bridge, trolley and hoist are approximately 58/5.8, 50/5.0 and 22.5/2.25 fpm, respectively. The bridge and trolley drives have full magnetic soft start electric starting controls to minimize drive wear. The

crane travel controls are such that when the load is not fully up, the bridge and trolley are permitted to move only when the hoist override control switch is turned “On”. Bridge rail end stops are provided to limit travel of the bridge so that the load cannot hit the end walls.

All necessary controls, relays, etc., for controlling a power-operated container uprighting mechanism are wired into the bridge crane and control panel for use in the event that one is purchased for future use.

g. Shipping Containers

Normally, large containers will be used as shipping containers for processed waste. The exact size varies from vendor to vendor. Standard DOT 17H steel drums and steel boxes that comply with DOT Industrial Packaging requirements are used as shipping containers for compacted or non-compacted waste.

(INTENTIONALLY BLANK)

h. Solidified Waste Storage Vault

A shielded area located adjacent to the radwaste truck bay on the 620’ elevation of the radwaste building, measuring 50’-6” long by 25’-6” wide by 13’-4” high (usable height) is used to provide temporary onsite storage of packaged waste. This allows for further decay time and lessens the effect on plant operations of such events as a trucker’s strike or temporary shutdown of a burial site.

i. Interim/Temporary Storage of Radioactive Wastes

The interim/temporary storage of radioactive wastes were evaluated for compliance with <Generic Letter 81-38>. Radioactive waste collected onsite awaiting disposal off-site is called temporary storage or staged waste. Any radioactive waste remaining on-site greater than 90 days will have waste form and container selection considered for impact and must comply with <Generic Letter 81-38>.

Radioactive waste may be stored anywhere within the generating facility, under the direction and approval of the Radiation

Protection Manager. A Radiation Protection Instruction (RPI) outlines the methods and protocol for storing and inspecting radioactive waste outside the RRA and/or the generating facility.

11.4.2.4 Instrumentation, Controls, Alarms, and Protective Devices

11.4.2.4.1 Controls

The SRW system is controlled entirely from the SRW panel and an adjacent control panel during all normal operations. The control panel is equipped with the following: a semi-graphic display of the processing system; control switches for all normally used valves; control switches for all motor driven equipment; status lights for all power operated valves, pumps, indexing controls and readout for the bridge crane; readout of certain process parameters (levels); add a CCTV monitoring system of the processing gallery, storage vault and truck bay. The control panel contains a solid state programmable controller to control the processing system. In general terms, the process is controlled as explained in the following paragraphs:

To begin filling a waste container the operator verifies that the vendor system is installed properly and all necessary connections have been made to the permanent plant systems. An empty waste container is placed in the waste process area and the vendor dewatering system fill head is placed on the waste container. Valves are aligned to the LRW system necessary to select one of the following waste streams; spent resin, filter/demineralizer sludge or RWCU sludge. Filling of the waste container is controlled from the vendor system control panel. After filling the waste container the process lines are backflushed to the LRW tank from which the waste container was filled.

a. (Deleted)

b. (Deleted)

c. (Deleted)

11.4.2.4.2 Instrumentation

a. Waste Mixing/Dewatering Tanks

Each of these tanks has an ultrasonic level transmitter for level readout and high/low alarms on the control panel, and for control interlocking functions. A back-up high level probe is provided for control interlocking.

b. Waste Feed Pumps

Each discharge line has a magnetic flowmeter for flow recording on the control panel.

11.4.3 REFERENCES FOR SECTION 11.4.

1. Letter from M. R. Edelmen, CEI, to B. J. Youngblood, NRC, August 28, 1985 (PY-CEI/NRR-0329L).

TABLE 11.4-1

MAXIMUM MONTHLY RADIOACTIVE WASTE INPUTS TO SOLID RADIOACTIVE WASTE SYSTEM

Waste Inputs	Quantity[8] Vol. per Batch (ft^3)	 Max. Batches per Month	 Max. Vol per Month (ft^3)	Maximum Activity Level (Ci/ft^3)	Method of Processing	Container Size to be Used (ft^3)
2. Spent resin slurry from:						
a. Condensate demineralizers	813 (Free Water)[2] 1,333 (Total)	0.14	813 (Free Water) 1,333 (Total)	(per ft^3 resin)	Dewater	50 or 200
b. Radwaste demineralizers	504 (Wet Resin)[1] 830 (Free Water)[2] 1,334 (Total)	1	504 (Wet Resin) 830 (Free Water) 1,334 (Total)	2×10^{-1} (per ft^3 resin)	Same as for condensate demin. resins.	50 or 200
3. Backwash slurry from:						
a. Condensate filters	333 (Sludge) 600 (Free Water)[2] 933 (Total)	15	5,000 (Sludge) 9,000 (Free Water) 14,000 (Total)	5.15×10^{-2} per ft^3 sludge)	Dewater	50 or 200
b. Reactor Water Cleanup F/D (Powdered resins)	80 (Sludge)[3] 207 (Free Water)[2] 287 (Total)	5	400 (Sludge) 1,035 (Free Water) 1,435 (Total)	9.9 (per ft^3 sludge)	Same as for condensate filter sludge.	50
c. Fuel Pool F/D (Powered resins)	333 (Sludge)[3] 600 (Free Water)[2] 933 (Total)	1	333 (Sludge) 600 (Free Water) 933 (Total)	1.55×10^{-2} (per ft^3 sludge)	Same as for condensate filter sludge.	50 or 200

TABLE 11.4-1 (Continued)

	Waste Inputs	Quantity[8]: Vol. per Batch (ft^3)	Quantity[8]: Max. Batches per Month	Quantity[8]: Max. Vol per Month (ft^3)	Maximum Activity Level (Ci/ft^3)	Method of Processing	Container Size to be Used (ft^3)
4.	Traveling belt cake	0.133[4]	65	8.65	1.55×10^{-2}	Dewater	50 or 200
5.	Dry compressible waste (clothing, paper, general trash)	-	-	3,000[5]	50×10^{-6} mCi/ft^3[6]		7.3 (55 gal.)
6.	Dry incompressible waste (tools, pipe, filter elements, trash)	-	-	800[5]	5×10^{-6} mCi/ft^3[6][7]	Packaged into waste drums high activity items are surrounded by solidification agent	7.3 or 50

NOTES:

(1) Density of wet resin mixture is approximately 71.5 lb/ft^3, of which, weight of resin is 23.25 lb and weight of absorbed and interstitial water is 48.25 lb.

(2) The term “free water” is used here to mean the volume of water in excess of the sum of the amount absorbed in the resins sludge and the amount occupying the void spaces in the settled resin sludge volume.

(3) Density of powdered resin sludge is approximately 66 lb/ft^3, of which, weight of powdered resin is 13 lb, and weight or absorbed and interstitial water is 53 lb.

(4) Backwash from TBF is a moist cake, containing approximately 15.3 lb of diatomaceous earth and crud, and up to 35.7 lb of water.

(5) The average expected volumes of dry solid wastes are 1,650 ft^3/mo. (compressible) and 500 ft^3/mo. (noncompressible).

(6) The isotopes present are expected to consist primarily of miscellaneous fission products proportional to those found in the reactor coolant.

(7) Certain filters may attain much higher activity levels depending on the fluid stream being filtered; these will be packaged within the solidified waste.

(8) Volumes are based on the process flow data for the liquid radwaste system.

(9) The Evaporators steam supply has been permanently cut.

TABLE 11.4-2

SOLID RADWASTE SYSTEM INFLUENT NUCLIDE ACTIVITIES

Isotope	Condensate Filter Sludge[1] μCi/cc	Radwaste Filter Sludge[2] μCi/cc
Na-24	2.9-3	Negligible
P-32	4.6-3	9.7-6
Cr-51	1.4-1	4.6-3
Mn-54	1.2-2	8.8-3
Co-58	1.4+0	3.7-1
Co-60	1.6-1	1.3-1
Fe-59	2.2-2	2.7-3
Zn-65	6.2-4	4.0-4
Zn-69m	3.1-5	Negligible
Ag-110m	1.8-2	1.2-2
Ag-110	1.8-2	1.2-2
W-187	2.0-2	Negligible
TOTAL	1.8+0	5.5-1

NOTES:

(1) Activity based on 4 days accumulation of 8 batches followed by a 2 day decay period.

(2) Activity based on 100 days accumulation of 149 batches of filter sludge from the waste collector and floor drain systems followed by a 100 day decay period.

TABLE 11.4-3 (Deleted)

TABLE 11.4-4

SOLID RADWASTE SYSTEM DEMINERALIZER ACTIVITIES

Isotope	RWCU Filter/ Demineralizer Sludge (μCi/cc)	Condensate Demineralizer (μCi/cc)	Radwaste Demineralizer (μCi/cc)
P-32	2.1-2	--	--
Cr-51	3.5+0	--	--
Mn-54	2.1+0	--	--
Co-58	1.4+2	--	--
Co-60	3.1+1	--	--
Fe-59	1.3+0	--	--
Zn-65	1.0-1	--	--
Br-83	--	--	6.5-4
Br-84	--	--	1.2-4
I-131	8.7-1	6.6-7	2.9-1
I-134	--	--	1.8-3
Sr-89	7.0+1	2.2-2	4.1-1
Tc-101	--	--	2.2-4
Cs-134	1.1+1	1.8-2	5.3-2
Cs-136	1.0-1	3.2-7	4.1-3
Cs-138	--	--	7.0-4
Ba-139	--	--	2.5-3
Sr-90	1.7+1	3.2-2	8.1-2
Y-90	1.7+1	3.2-2	6.5-2
Sr-92	--	--	6.0-3
Y-92	--	--	9.2-3

TABLE 11.4-4 (Continued)

Isotope	RWCU Filter/ Demineralizer Sludge (μCi/cc)	Condensate Demineralizer (μCi/cc)	Radwaste Demineralizer (μCi/cc)
Mo-99	2.3-5	--	1.6-1
Tc-99m	2.5-5	--	8.1-2
Ru-103	3.0-1	5.5-5	2.1-3
Rh-103m	2.9-1	5.3-5	1.1-3
Ru-106	1.6-1	2.3-4	7.6-4
Rh-106	1.6-1	2.3-4	6.0-4
Ag-110m	2.9+0	--	--
Ag-110	2.9+0	--	--
Te-132	5.2-4	--	3.9-1
I-132	5.2-4	--	2.3-2
I-135	--	--	3.8-2
Cs-137	1.8+1	3.4-2	8.1-2
Ba-137m	1.8+1	3.4-2	6.5-2
Ba-140	6.6+0	1.2-5	3.2-1
La-140	7.7+0	1.4-5	6.5-2
Ba-142	--	--	2.1-4
La-142	--	--	7.6-4
Ce-143	--	--	1.1-4
Pr-143	3.8-2	--	1.5-3
Ce-144	2.1+0	2.9-3	9.7-3
Pr-144	2.1+0	2.9-3	7.6-3
Nd-147	5.6-3	--	4.2-4

TABLE 11.4-4 (Continued)

Isotope	RWCU Filter/ Demineralizer Sludge (μCi/cc)	Condensate Demineralizer (μCi/cc)	Radwaste Demineralizer (μCi/cc)
Pm-147	9.4-3	3.4-6	2.5-5
Np-239	1.3-5	--	1.4+0
Pu-239	2.2-3	4.5-7	6.5-1
Br-85	--	--	9.2-6
Sr-91	--	--	3.8-2
Y-91m	--	--	3.6-3
Y-91	1.3+0	1.9-4	2.7-1
Zr-95	1.1+0	5.3-4	6.5-3
Nb-95m	2.4-2	1.1-5	8.7-5
Nb-95	1.9+0	9.8-4	8.1-3
Zr-97	--	--	4.2-5
Nb-97m	--	--	3.0-6
Nb-97	--	--	4.9-6
Te-129m	4.9-1	6.1-5	3.8-3
Te-129	--	--	8.7-6
I-129	--	--	4.8-6
I-133	--	--	1.4-1
Ba-141	--	--	3.5-4
La-141	--	--	2.8-3
Ce-141	1.2+0	1.4-4	2.4+0
Totals	3.6+2	1.8-1	7.1+0

11.5 PROCESS AND EFFLUENT RADIOLOGICAL MONITORING AND SAMPLING SYSTEMS

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

a. Instrumentation systems required for safety, or

b. Instrumentation systems required for plant operation.

11.5.1 DESIGN BASES

11.5.1.1 Design Objectives

11.5.1.1.1 Systems Required for Safety

The main objective of radiation monitoring systems required for safety is to initiate appropriate protective action to limit the potential release of radioactive materials from the reactor vessel and primary and secondary containment, if predetermined radiation levels are exceeded in major process/effluent streams. An additional objective is to provide control room personnel with an indication of the radiation levels in the major process/effluent streams, plus alarm annunciation if high radiation levels are detected.

Main steam line and containment ventilation exhaust radiation monitoring is provided to meet these objectives.

11.5.1.1.2 Systems Required for Plant Operation

The main objective of radiation monitoring systems required for plant operation is to provide operating personnel with measurement of the content of radioactive material in all effluent and important process streams. This complies with plant normal operational limits by providing gross radiation level monitoring and collection of halogens and particulates on filters (gaseous effluents) as required by <Regulatory Guide 1.21>. Additional objectives are to initiate discharge valve isolation on the offgas or liquid radwaste systems if predetermined release rates are exceeded and to provide for sampling at certain radiation monitor locations to allow determination of specific radionuclide content.

The radiation monitoring provided to meet these objectives are:

a. For gaseous effluent streams

    1. Unit Vent

    2. Offgas Vent Pipe

    3. Turbine Building/Heater Bay Vent

b. For liquid effluent streams

    1. Radwaste discharge

    2. Emergency service water system (Loops A and B)

    3. ADHR heat exchanger service water outlet

c. For gaseous process streams

   1. Offgas pretreatment

   2. Offgas post-treatment

   3. Carbon bed vault

   4. Annulus exhaust

   5. Steam packing exhauster

d. For liquid process streams

   1. Underdrain

   2. Nuclear closed cooling water

11.5.1.2 <u>Design Criteria</u>

11.5.1.2.1 Systems Required for Safety

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

a. Withstand the effect of natural phenomena (e.g., earthquakes) without loss of capability to perform their functions.

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

c. Meet the reliability, testability, independence, and failure mode requirements of engineered safety features.

d. Provide continuous outputs on control room panels.

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

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

g. Initiate prompt protective action prior to exceeding plant limits.

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

i. Insofar as practical, provide self-monitoring of components to the extent that power failure or component malfunction causes annunciation and channel trip.

j. Maintain full scale output if radiation detection exceeds full scale.

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

The applicable General Design Criteria are 63 and 64. The systems meet the design requirements for Safety Class 2, Seismic Category I systems, along with the quality assurance requirements of <10 CFR 50, Appendix B>.

11.5.1.2.2 Systems Required for Plant Operation

The design criteria for operational radiation monitoring systems are that the systems:

a. Provide continuous indication of radiation levels in the control room.

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

c. Insofar as practical, provide self-monitoring of components to the extent that power failure or component malfunction causes annunciation and, for systems initiating discharge isolation, channel trip.

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

e. Have provisions for calibration, function and instrumentation checks.

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

g. Maintain full scale output if radiation detection exceeds full scale.

The instrument channels monitoring discharges from the gaseous and liquid radwaste treatment systems have provisions to alarm and to initiate automatic closure of the waste discharge valve on the affected treatment system prior to exceeding the normal operation limits as required by <Regulatory Guide 1.21>.

The applicable General Design Criteria are 60, 63 and 64.

11.5.2 SYSTEM DESCRIPTION

11.5.2.1 Systems Required for Safety

Information on the system monitors is presented in <Table 11.5-1> and the arrangements shown in <Figure 11.5-1 (1)>, <Figure 11.5-1 (2)>, <Figure 11.5-1 (3)>, <Figure 11.5-1 (4)>, <Figure 11.5-1 (5)>, <Figure 11.5-1 (6)>, <Figure 11.5-1 (7)>, <Figure 11.5-1 (8)>, <Figure 11.5-1 (9)>, <Figure 11.5-1 (10)>, <Figure 11.5-1 (11)>, and <Figure 11.5-1 (12)>.

11.5.2.1.1 Main Steam Line Radiation Monitoring System

This system monitors the gamma radiation level exterior to the main steam lines. The normal radiation level is produced primarily by coolant activation gases plus smaller quantities of fission gases being transported with the steam. In the event of a gross release of fission products from the core, this monitoring system provides channel trip signals for isolation of the CRVICS reactor water sample valves.

The system consists of four redundant instrument channels. Each channel consists of a local detector (gamma-sensitive ion chamber) and a control room ratemeter with an auxiliary trip unit. Power for the two channels (A and C) is supplied from the reactor protection system (RPS) bus A and for the other two channels (B and D) from RPS bus B. Channels A and C are physically and electrically independent of channels B and D.

The detectors are physically located in separate pipe wells which extend into the steam tunnel near the main steam lines just downstream of the outboard main steam line isolation valves. The detectors are geometrically arranged so that this system is capable of detecting significant increases in radiation level with any number of main steam lines in operation. <Table 11.5-2> lists the range of the detectors.

Each radiation monitor has two upscale (high-high and high), one downscale and one inoperative trip circuits. Each trip is visually displayed on the affected radiation monitor. A high-high or inoperative trip in the radiation monitor results in a channel trip, which is an input to the CRVICS. A logic trip from a one-out-of-two twice MSL channel trip results in initiation of reactor water sample valve closure. A logic trip from one-out-of-two MSL channels "A" or "C" results in initiation of condenser air removal pump shutdown, and closure of the condenser air removal pump isolation valve. A high trip actuates a MSL high radiation control room annunciator. A downscale trip actuates a MSL downscale control room annunciator common to all channels. High and low trips do not result in a channel trip. Each radiation monitor visually displays the measured radiation level.

11.5.2.1.2 Containment Ventilation Exhaust Radiation Monitors

This system monitors the radiation level exterior to the containment ventilation system exhaust duct. A high activity level in the ductwork could be due to fission gases from a leak or an accident.

The system consists of four redundant instrument channels. Each channel consists of a local detection assembly (containing a Geiger-Mueller (GM) tube and electronics) and a control room ratemeter. Power for two channels (A and C) is supplied from RPS bus A and for the other two channels (B and D) from RPS bus B. Channels A and C are physically and electrically independent of channels B and D. One two-pen recorder powered from an inverter on the 125 volt dc non-divisional bus allows the output of two channels to be recorded by the use of selection switches. The detection assemblies are physically located outside and adjacent to the exhaust ducting downstream of the containment discharge isolation valves.

Each radiation monitor provides both an analog output signal and a contact which opens on upscale (high-high) radiation or an inoperative

circuit. Two-out-of-two upscale/inoperative trips in channels A and D initiate closure of the containment ventilation outboard isolation valves and the drywell outboard isolation valves. The same condition for channels B and C initiates closure of the containment inboard valves and drywell inboard valves.

An upscale/inoperative trip is visually displayed on the affected radiation monitor ratemeter and actuates a containment and drywell ventilation exhaust high-high inoperative radiation control room annunciator. A downscale trip is also visually displayed on the radiation monitor ratemeter. Containment and drywell vent high radiation and downscale trip control annunciators common to all channels and are generated from the analog signal. Each radiation monitor ratemeter visually displays the measured radiation level. <Table 11.5-2> lists the range of the detectors.

11.5.2.2 Systems Required for Plant Operation

Information on these systems is presented in <Table 11.5-1>, <Table 11.5-2>, and <Table 11.5-3> and the arrangements are shown in <Figure 11.5-1>.

11.5.2.2.1 Offgas Pretreatment Radiation Monitor

This system monitors radioactivity in the condenser offgas at the inlet to the holdup piping after it has passed through the offgas condenser and moisture separator. The monitor detects the radiation level which is attributable to the fission gases produced in the reactor and transported with steam through the turbine to the condenser.

A continuous sample is extracted from the offgas pipe via a sample line. It is then passed through a sample chamber and a sample panel before being returned to the suction side of the steam jet air ejector (SJAE). The sample chamber is a stainless steel pipe which is internally

polished to minimize plateout. It can be purged with room air to check detector response to background radiation by using a three-way solenoid operated valve. The valve is controlled by a switch located in the control room. The sample panel measures and indicates sample line flow. A detector (GM tube) is positioned adjacent to the vertical sample chamber and is connected to a ratemeter in the control room.

Power is supplied from an inverter on the 125 volt dc non-divisional bus for the radiation monitor, detector and recorder; power is also supplied from a 120 volt ac miscellaneous distribution panel for the sample and vial sampler panels which can be transferred to a diesel bus.

The radiation monitor has three trip circuits: one upscale and one downscale in the radiation monitor itself, and one upscale in the recorder.

The trip outputs are used for alarm function only. The ratemeter trip functions are visually displayed and all trip outputs actuate control room annunciators for each of the following: offgas high, offgas recorder and offgas downscale/inoperative. High or low sample line flow measured at the sample panel actuates a control room offgas pre-treatment sample high-low flow annunciator.

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

11.5.2.2.2 Offgas Post-Treatment Radiation Monitor

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

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

Power is supplied from an inverter on the 125 volt dc non-divisional bus for the radiation monitors and recorders, and from a 120 volt ac miscellaneous distribution panel for the sample panel purge circuit.

Each radiation monitor has three trip circuits: two upscale (high-high-high and high), and one downscale (low/inoperative). Each trip is visually displayed on the radiation monitor. These three trips actuate corresponding control room annunciators: offgas post-treatment high-high-high radiation, offgas post-treatment high radiation and offgas post-treatment downscale/inoperative. A trip circuit on the

recorder actuates an offgas post-treatment high-high radiation annunciator. High or low sample flow measured at the sample panel actuates a control room offgas post-treatment sample panel high-low flow annunciator.

An auxiliary trip unit in the control room takes the high-high-high (HHH) and downscale trip outputs and, if its logic is satisfied, initiates closure of the offgas system discharge and drain valves. The logic is satisfied if two HHH, one HHH and one downscale, or two downscale trips occur. The HHH trip setpoints are determined such that valve closure is initiated prior to exceeding release rate limits. Any one high upscale trip initiates closure of offgas system bypass line valve and permits opening of the treatment line valve.

A vial sampler panel similar to the pretreatment sampler panel is provided for grab sample collection to allow isotopic analysis and gross monitor calibration.

11.5.2.2.3 Carbon Bed Vault Radiation Monitor

Carbon vault A and B are monitored for gross gamma radiation level. Each channel includes detector, a ratemeter and a locally mounted auxiliary unit. The ratemeter is located in the control room. The channel provides for sensing and readout, both local and remote of gamma radiation over a range of six logarithmic decades (1 to 10^6 mR/hr).

The ratemeter has one adjustable upscale trip circuit for alarm and one downscale trip circuit for instrument trouble. The trip circuits are capable of operational verification by means of test signals or through the use of portable gamma sources. Power is supplied from an inverter on the 125 volt dc non-divisional bus.

11.5.2.2.4 Plant Vent Radiation Monitor

This unit monitors a sample of the plant vent effluent discharge <Figure 9.4-18> for particulate, iodine and gas radioactivity and also provides samples of the collected particulate and halogen for laboratory analysis. A representative sample is continuously extracted from the plant vent through an isokinetic probe in accordance with ANSI N13.1-1969 with the additional feature of manually regulating the sample flow in proportion to the vent stack flow. The sample is supplied through a 1 inch sample line which is also used to supply a representative sample to the postaccident effluent radiation monitors. This sample line is heat traced to preclude any condensation. A portion of this representative sample is taken by another isokinetic probe and passed through the shielded particulate, iodine and gas detector assemblies which are provided with scintillation detectors and check sources. The ratemeters in the control room analyze and visually display the measured radiation level for the particulate (gross Beta), gas (gross Beta) and Iodine cartridge (gamma).

Power is supplied from the non-1E 120 volt ac miscellaneous distribution panel for the radiation monitor ratemeters and recorders. The 480 volt ac 3∅/non-1E diesel backed bus supplies power for the sample pumps. Power for the isokinetic sample pump is non-1E diesel backed to ensure system availability.

Each of the ratemeters has two upscale and one downscale trip circuits which are visually displayed on the ratemeter and annunciated in the control room. The noble gas ratemeter has an alarm detection circuit which activates at full scale and shuts off the high voltage and sample pump. High or low differential pressure across the filters at the sample panel are annunciated in the control room.

11.5.2.2.5 Turbine Building/Heater Bay Vent Radiation Monitor

This unit monitors a sample of the turbine building/heater bay vent discharge for particulate, iodine and gas radioactivity and also provides samples of the collected particulate and halogen for laboratory analysis. A representative sample is continuously extracted from the turbine building/heater bay discharge vent downstream of the exhaust fans shown in <Figure 9.4-9>. Sampling and monitoring is as described for the plant vent radiation monitor.

Power is supplied from the non-1E 120 volt ac miscellaneous distribution panel for the ratemeters and recorders. The 480 volt ac 3Ø/non-1E diesel backed bus supplies power for the sample pumps. Power for the isokinetic sample pump is non-1E diesel backed to ensure system availability.

Each of the ratemeters has two upscale and one downscale trip circuits which are displayed on the ratemeters and annunciated in the control room. The noble gas ratemeter has an alarm detection circuit which activates at full scale and shuts off the high voltage and sample pump. High or low differential pressure across the filters at the sample panel are annunciated in the control room.

11.5.2.2.6 Offgas Vent Pipe Monitor

This unit monitors a sample of the offgas vent pipe discharge downstream of the exhaust fans <Figure 9.4-10> for particulate, iodine and gas activity and also provides samples of the collected particulate and halogen for laboratory analysis. A representative sample is continuously extracted from the offgas vent pipe and monitored as described for the plant vent radiation monitor.

Power is supplied from non-1E 120 volt ac miscellaneous distribution panel for the radiation monitor ratemeters and recorders. 480 volt ac

3Ø/non-1E diesel backed bus supplies power for the sample pumps. Power for the isokinetic sample pump is non-1E diesel backed to assure system availability.

Each of the ratemeters has two upscale and one downscale trip circuits which are visually displayed on the ratemeters and annunciated in the control room. The noble gas ratemeter has an alarm detection circuit which activates at full scale and shuts off the high voltage and sample pump. High or low differential pressure across the filters, measured at the sample panel, are annunciated in the control room.

11.5.2.2.7 Annulus Exhaust Radiation Monitor

These units monitor the annulus exhaust for gas activity (gross Beta) and provides samples of collected particulate and halogen for laboratory analysis. The units are identified as Annulus Exhaust - Train A Radiation Monitor and Annulus Exhaust - Train B Radiation Monitor. A sample is continuously extracted from the annulus exhaust duct downstream of the annulus exhaust filter trains A and B through an isokinetic probe <Figure 6.5-1>. The sample is passed through a fixed particulate sample filter, a fixed halogen collection cartridge, and through a shielded scintillation detector with a check source. For units with digital readout modules, remote LED pulsers are provided with the detector assemblies in place of the radioactive check sources. The LED pulsers are used to excite the photomultiplier tubes within the scintillation detectors for functional testing of the instrument channel. The detector monitors the gross Beta gas activity. Ratemeters in the control room analyze and visually display the measured gas activity.

Power is supplied from the non-1E 120 volt ac miscellaneous distribution panel for the ratemeters and recorders while the sample pumps are supplied by the 480 volt ac 3Ø/non-1E diesel backed bus.

The ratemeter has two upscale and one downscale trip circuits which are visually displayed on the ratemeter and annunciated in the control room. High or low differential pressure measured across the filters in the sample panel are annunciated in the control room.

11.5.2.2.8 Steam Packing Exhauster Radiation Monitor

The discharge from the steam packing exhauster is monitored for radioactivity by a shielded inline detector assembly which is provided with a scintillation detector and a check source. To provide check source function, a remote LED pulser is provided with the detector assembly in place of the radioactive check source. The LED pulser is used to excite the photomultiplier tube within the scintillation detector for functional testing of the instrument channel. The detector assembly is located on the steam packing exhauster effluent line which discharges to the offgas vent pipe as shown in <Figure 10.1-10>. A ratemeter in the control room analyzes and visually displays the measured radiation (gross Gamma).

Power is supplied from the non-1E 120 volt ac miscellaneous distribution panel for the ratemeter and recorder.

The ratemeter has two upscale and one downscale trips which are displayed on the ratemeter and are annunciated in the control room.

11.5.2.2.9 Liquid Process and Effluent Monitoring Systems

These systems, listed in <Table 11.5-3>, monitor the gamma radiation levels of liquid process and effluent streams. With the exception of the radwaste system effluent, the streams monitored normally contain only background levels of radioactive materials. Increases in radiation level may be indicative of heat exchanger leakage or equipment malfunction.

Power is supplied from an inverter on the 125 volt dc non-divisional buses for the radiation monitors and recorders, and from a 120 volt ac local bus for the sample panels. The underdrain liquid monitors are powered by a non-1E 120 volt bus.

Each radiation monitor has three trip circuits: two upscale (high-high and high) and one downscale (low). Each trip is visually displayed on the affected radiation monitor. Two of these trips actuate corresponding control room annunciators: one upscale (high-high

radiation) and the downscale for the affected liquid monitoring channel. High or low sample flow measured at the sample panel actuates a control room flow annunciator for the affected liquid channel.

For each liquid monitoring location, except for the underdrain system, a continuous sample is extracted from the liquid process pipe, passed through a liquid sample panel which contains a detection assembly for gross gamma radiation monitoring, and returned to the process pipe. The detection assembly consists of a scintillation detector mounted in a shielded sample chamber equipped with a check source. A ratemeter in the control room displays the measured gross radiation level and the analog signal is recorded.

The sample panel chamber and lines can be drained to allow assessment of radiation background. The panel measures and indicates sample line flow. A solenoid operated check source operated from the control room can be used to check operability of the channel.

11.5.2.2.9.1 Radwaste Effluent Radiation Monitor

This system consists of one channel, the radwaste effluent to ESW discharge pipe, which monitors the radioactivity in the radwaste effluent prior to its discharge.

Liquid waste can be discharged from several radwaste processed water tanks such as the floor drain sample tanks, waste sample tanks, chemical waste distillate tanks, or detergent drain tanks. These tanks contain liquids that have been processed through one or more treatment systems such as evaporation, filtration and ion exchange. Prior to discharge from any tank, the liquid in the appropriate tank is sampled and analyzed. Based upon this analysis, discharge is permitted at a specified release rate and dilution rate.

The upscale trip on the radwaste effluent radiation monitor is used to initiate closure of the radwaste system discharge valve. The trip point is set such that closure is initiated prior to exceeding limits for liquid effluents. When the channel is inoperable, an override switch is used to allow discharging to continue. Prior to the switch being in OVERRIDE, additional sampling is performed. Annunciation will occur when the switch is in OVERRIDE. The upscale trip also actuates an annunciator in the control room.

11.5.2.2.9.2 Emergency Service Water Radiation Monitoring

This system consists of two channels <Figure 9.2-1>: one for monitoring downstream of equipment in emergency service water system Loop A and the other for Loop B. If a high radiation level is detected, the affected emergency service water line can be manually isolated.

11.5.2.2.9.3 Nuclear Closed Cooling System Radiation Monitoring

This system has a channel for monitoring downstream of equipment in the nuclear closed cooling water system <Figure 9.2-4>.

11.5.2.2.9.4 Underdrain System Radiation Monitor

The condition where amounts of radioactive material resulting in radionuclide concentration in the underdrain system approaching significant levels has been analyzed and is considered highly unlikely <Section 2.4.13.3>. However, radiation monitors will be located inside the gravity discharge system manholes, at the point where the lower subsystem liquid effluent discharges into the gravity drain system, to continuously monitor and detect gross amounts of radioactive concentrations in the groundwater of the underdrain system.

These radiation monitors are mounted on the platform inside the manhole <Figure 2.4-71>. One monitor

will be located in the east gravity discharge system manhole and one monitor will be located in the west gravity discharge system manhole.

Each radiation monitor will transmit a preamplified signal to its associated ratemeter located in the control room. When the level of radioactivity at either radiation monitor exceeds a preset value, the associated channel will alarm in the control room, alerting the operator, and automatically stop the service pumps, and the backup pumps in the underdrain system. Radioactive concentrations of the magnitude as postulated by the failure of a waste collector tank <Section 15.7.2> and <Section 15.7.3>, can be detected and alarmed by these radiation monitors.

Continuous monitoring of the liquid effluent discharge from the underdrain system will ensure that the limits of <10 CFR 20, Appendix B> are not exceeded, and that early detection of abnormalities is achieved.

11.5.2.2.9.5 ADHR Heat Exchanger Service Water Radiation Monitor

This system has a channel for monitoring leakage from the ADHR system to the Service Water system <Figure 9.2-14>. If a high radiation level is detected, the ADHR system can be manually isolated.

11.5.2.3 Inspection, Calibration and Maintenance

11.5.2.3.1 Inspection and Tests

During reactor operation, daily checks of system operability are made by observing channel behavior. At periodic intervals during reactor operation, the detector response (of each monitor provided with a remotely positioned check source) will be recorded together with the instrument background count rate to ensure proper functioning of the

monitors. Any detector whose response cannot be verified by observation during normal operation or by using the remotely positioned check source will have its response checked with a portable check source. A record will be maintained showing the background radiation level and the detector response.

Appropriate channels are tested in accordance with plant procedures. Verification of valve operation, ventilation diversion or other trip function will be done at the time of testing if it can be done without jeopardizing the plant safety. The tests will be documented.

11.5.2.3.1.1 Detailed Inspection and Tests

a. The following monitors have alarm trip circuits which can be tested by using test signals or portable gamma sources:

   1. Main steam line

   2. Containment ventilation exhaust

   3. Offgas pretreatment

   4. Carbon bed vault

b. The following monitors include built-in check sources which can be operated from the control room:

   1. Offgas post-treatment

   2. Annulus exhaust

   3. Offgas Vent Pipe

   4. Unit Vent

   5. Turbine Building/Heater Bay Vent

   6. Steam packing exhauster

   7. Radwaste effluent to ESW

8. Emergency service water

9. Nuclear closed cooling water

10. ADHR heat exchanger service water outlet

11.5.2.3.2 Calibration

The radiation monitor’s calibration is traceable to certified National Institute of Standards and Technology (NIST) or commercial radionuclide standards. The source-detector geometry during primary calibration is identical to the sample-detector geometry in actual use. Secondary standards which were counted in reproducible geometry during the primary calibration may be used with each monitor for calibration after installation. A calibration can also be performed by using liquid or gaseous radionuclide standards or by analyzing particulate, iodine or gaseous grab samples with laboratory instruments.

11.5.2.3.3 Maintenance

The detectors, electronics, recorders, and sample pumps are serviced and maintained to ensure reliable operations. Such maintenance includes cleaning, lubrication and assurance of free movement of the recorder in addition to the replacement or adjustment of components required after performing a test or calibration check. If work is performed which would affect the calibration, a recalibration is performed at the completion of the work.

11.5.2.3.4 Audits and Verifications

Independent audits and verifications of test, calibration and maintenance records and procedures are conducted as described in <Section 17.2>.

11.5.3 EFFLUENT MONITORING AND SAMPLING

11.5.3.1 Implementation of General Design Criterion 64

All potentially radioactive effluent discharge paths are continuously monitored for gross radiation level. Liquid releases are monitored for gross gamma with the exception of the Turbine Building atmospheric drain line from the Turbine Building supply plenum. The Turbine Building atmospheric drain line will be monitored with grab samples as established in <Table 11.5-6>. Solid waste shipping containers are monitored with gamma sensitive portable survey instruments. Gaseous releases are monitored for gross gamma, gamma, and gross beta. Gaseous batch releases from low radioactivity areas needed to support maintenance activities, plant surveillances, or recovery from a potentially hazardous chemical atmosphere, where discharge through the normal effluent points is not practical, are controlled and monitored in accordance with the methodology of the ODCM for inoperable gaseous effluent monitors. The following gaseous effluent paths are sampled and monitored:

a. Unit Vent

b. Offgas Vent Pipe

c. Turbine Building/Heater Bay Vent

The following liquid effluent paths are sampled and monitored:

Liquid Radwaste System
Emergency Service Water (Loops A and B)
ADHR heat exchanger service water outlet

The monitors and ranges are listed in <Table 11.5-2>.

For the atmospheric drain line on the Turbine Building supply plenum, periodic samples are taken for isotopic and tritium analysis to monitor this pathway.

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

This effluent monitoring and sampling program is used to provide the information for the effluent measuring and reporting programs required by <10 CFR 50.36a>, <10 CFR 50, Appendix A> (General Design Criterion 64), <10 CFR 50, Appendix I> and <Regulatory Guide 1.21> in annual reports to the NRC. The frequency of the periodic sampling and analysis described herein is a minimum and will be increased if effluent levels approach limits. <Table 11.5-4>, <Table 11.5-5>, <Table 11.5-6>, and <Table 11.5-7> present the sample schedules.

11.5.4 PROCESS MONITORING AND SAMPLING

11.5.4.1 Implementation of General Design Criterion 60

The potentially significant radioactive discharge paths are equipped with a control system to automatically isolate the discharge on indication of a high radiation level. These include:

a. Offgas post-treatment

b. Containment ventilation exhaust

c. Liquid radwaste effluent

The effluent isolation functions for each monitor are given in <Table 11.5-1> and <Table 11.5-3>.

11.5.4.2 Implementation of General Design Criterion 64

Radiation levels in radioactive and potentially radioactive process streams are monitored by the following process monitors:

a. Main steam line

b. Offgas pretreatment

c. Offgas post-treatment

d. Carbon bed vault

e. Nuclear closed cooling water

f. Underdrain

g. Steam packing exhauster

h. Annulus exhaust

Airborne radioactivity in the containment, drywell, fuel handling building, and other areas are monitored as described in <Section 12.3.4> as these are used to monitor in-plant airborne radioactivity.

The area radiation monitors described in <Section 12.3.4> detect abnormal radiation levels in the various process equipment rooms.

Batch releases are sampled and analyzed prior to discharge in addition to the continuous effluent monitoring. The radwaste process monitoring systems are listed in <Table 11.5-2>.

TABLE 11.5-1

GASEOUS AND AIRBORNE PROCESS AND EFFLUENT RADIATION MONITOR

Radiation Monitor	Sample Point	Instrument Channels	Function	Location
1D17K610 A,B,C,D Main Steam Line	Pipewells in steam tunnel downstream of outer isolation valve	Ion chambers - redundant channels	Control Room alarms and indication. Isolates Reactor Water Sample Valves and Condenser Air Removal Lines	Steam Tunnel, Auxiliary Building 630'
1D17K612 Offgas Pretreatment	Sample from the offgas water separator effluent	Geiger-Mueller	Control Room alarms and indication	Turbine Building 577'
1D17K601 A,B Offgas Post-treatment	Sample from carbon vault discharge	Geiger-Mueller Redundant channels	Control Room alarms and indication. Isolates Offgas System	Offgas Building 584'
1D17K611 A,B Carbon Bed Vault	Detectors view Carbon Bed Vault's A and B Refrigeration Ductwork	Geiger-Mueller	Control Room indication and alarms	Offgas Building 603'
1D17K609 A,B,C,D Containment Ventilation Exhaust	Ventilation duct downstream of Containment Isolation Valve	Geiger-Mueller Redundant channels	Control Room indication and alarms. Close Containment and Drywell Purge Ventl. System valves	Intermediate Building, Containment Ventl. Exh. Duct 672'

TABLE 11.5-1 (Continued)

Radiation Monitor	Sample Point	Instrument Channels	Function	Location
1D17K690 A,B Annulus Exhaust Train A and Train B	Isokinetic sample downstream of filter trains	Beta scintillation channel for gases and sample filters for particulate and halogen with sample pump	Local and Control Room alarms and indication	Intermediate Building 620’
1D17K780 2D17K780[2] Unit Vent	Isokinetic sample from Unit Vent Auto-isokinetic sampler	3-Channel, Gas-Halogen-Particulate, scintillation type with sample pump	Local and Control Room alarms and indication	Intermediate Building 682’
1D17K850 Turbine Building/ Heater Bay Vent	Isokinetic sample from HB/TB Vent Auto-isokinetic sampler	3-Channel, Gas-Halogen-Particulate, scintillation type with sample pump	Local and Control Room alarms and indication	Heater Bay Equipment House 667’
1D17K830 Offgas Vent Pipe	Isokinetic sample from Offgas Vent Pipe Auto-isokinetic sampler	3-Channel, Gas-Halogen-Particulate, scintillation type with sample pump	Local and Control alarms and indication	Turbine Building 620’
1D17K840 Steam Packing Exhauster	Steam packing exhauster effluent line	Inline gas scintillation channel	Control Room alarms and indication	Turbine Building 624’

NOTE:

(1) (Deleted)
(2) The Unit 2 plant vent receives input from the active Unit 1.

TABLE 11.5-2

PROCESS AND EFFLUENT RADIATION MONITORING SYSTEM CHARACTERISTICS

Monitoring Systems	Number of Units[1]	Detector Sensitivity	Instrument Range (Scale)	No. of Trips Upscale - Downscale	High (Trip) Setpoint[2]	Prealarm Setpoint[2]
Main Steam Lines	4-IC	3.7×10^{-10} amp/R/hr	1 to 10^6 mr/hr	2-1	ORM	Above Full Power Background
Offgas Pretreatment	1-GM	-	1 to 10^6 mr/hr	2-1	ORM	Variable
Offgas Post-Treatment	2-GM	1.2×10^{-4} μCi/cc (Kr-85)	10 to 10^6 counts/min	2-1	ORM	Variable
Carbon Bed Vault	2-GM	-	1 to 10^6 mr/hr	1-1	NA	Above Background
Containment Ventl. Exhaust	4-GM	-	.01 to 100 mr/hr (each channel)	2-1	Technical Specification	Above Background
Annulus Exhaust	2-BSD-G	1×10^{-6} μCi/cc (Xe-133)	10 to 10^6 counts/min 10 to 10^7 counts/min (for digital readout modules)	2-1	Variable	Above Background
Unit Vent Exhaust	1-BSD-G	4.7×10^{-7} μCi/cc (Kr-85)		2-1		
	1-BSD-P	2.7×10^{-11} μCi/cc (Cs-137)			See Note[3]	See Note[3]
	1-GSD-H	1.6×10^{-11} μCi/cc (I-131)	10 to 10^7 counts/min (for digital readout modules)			
Turbine Bldg./ Heater Bay Vent	1-BSD-G	4.7×10^{-7} μCi/cc (Kr-85)	10 to 10^6 counts/min	2-1		
	1-BSD-P	2.7×10^{-11} μCi/cc (Cs-137)	(each channel)		See Note[3]	See Note[3]
	1-GSD-H	1.6×10^{-11} μCi/cc (I-131)	10 to 10^7 counts/min (for digital readout modules)			
Offgas Vent Pipe	1-BSD-G	4.7×10^{-7} μCi/cc (Kr-85)	10 to 10^7 counts/min	2-1		
	1-BSD-P	2.7×10^{-11} μCi/cc (Cs-137)	for digital readout modules		See Note[3]	See Note[3]
	1-GSD-H	1.6×10^{-11} μCi/cc (I-131)				

TABLE 11.5-2 (Continued)

Monitoring Systems	Number of Units[1]	Detector Sensitivity	Instrument Range (Scale)		No. of Trips Upscale – Downscale	High (Trip) Setpoint[2]	Prealarm Setpoint[2]
Steam Packing Exhauster	1-BSD-G	3.8 x 10^{-6} µCi/cc (Xe-133)	10 to 10^6 counts/min.	10 to 10^7 counts/min. (for digital readout modules)	2-1	Variable	Variable
Emergency Service Water Loop A	1-GSD-L	1 x 10^{-6} µCi/cc (Cs-137)	10 to 10^6 counts/min		1-1	Variable	Variable
Emergency Service Water Loop B	1-GSD-L	1 x 10^{-6} µCi/cc (Cs-137)	10 to 10^6 counts/min.		1-1	Variable	Variable
Nuclear Closed Cooling Water	1-GSD-L	1 x 10^{-6} µCi/cc (Cs-137)	10 to 10^6 counts/min.		1-1	Variable	Variable
Plant Radwaste Discharge - ESW Discharge	1-GSD-L	1 x 10^{-6} µCi/cc (Cs-137)	10 to 10^6 counts/min.		1-1	Variable	-
Underdrain	1-GSD-L	8.5 x 10^{-6} µCi/cc (I-GSD-L)	10 to 10^6 counts/min.		2-1	Variable	Variable
ADHR Heat Exchanger Service Water Outlet	1-GSD-L	1 x 10^{-6} µCi/cc (Cs-137)	10 to 10^6 counts/min.		1-1	Variable	Variable

NOTES:

(1) Types of detectors are designated as follows:
- GM – Geiger-Mueller detector
- IC – Ion chamber detector
- BSD-G – Beta scintillation detector – Gas Channel
- BSD-P – Beta scintillation detector – Particulate Filter
- GSD-A – Gamma scintillation detector – Halogen Cartridge
- GSD-L – Gamma scintillation detector – Liquid Channel

(2) Setpoints to be revised as required to be compatible with limits established and current calibrated sensitivity of the applicable channel.

(3) Basis for setpoint calculations:

a. For noble gas channels, setpoints calculated in accordance with the Offsite Dose Calculation Manual based on compliance with <10 CFR 20> limits for unrestricted areas.

b. For particulate and iodine channels, setpoints are variable.

TABLE 11.5-3

LIQUID PROCESS AND EFFLUENT RADIATION MONITORS

Radiation Monitor	Sample Point	Instrument Channels	Function	Location
1D17K604 Emergency Service Water Loop A	ESW - Loop A downstream of RHR Heat Exchanger	Gamma - scint., offline with sample pump	Control Room indication and alarm	Auxiliary Building 568’ - East and West
1D17K605 Emergency Service Water Loop B	ESW - Loop B downstream of RHR Heat Exchanger	Gamma - scint., offline with sample pump	Control Room indication and alarm	Auxiliary Building 568’ - West and East
D17K607 Nuclear Closed Cooling System	Downstream of nuclear closed Cooling Heat Exchangers	Gamma - scint., offline with sample pump	Control Room indication and alarm	Control Complex 599’
D17K606 Radwaste Effluent to ESW - Discharge	Radwaste line downstream of discharge valves G50-F153 and G50-F155	Gamma - scint., offline with sample pump	Control Room and Radwaste Control Room indication and alarm. Close discharge valve on high trip.	Auxiliary Building 620’ - East
D17K820 A Underdrain System	Gravity Drain System discharge lines	Gamma - scint., INLINE	Control Room indication and alarm. Stop underdrain pumps on high trip.	Gravity Drain System Manhole No. 23, 608’

TABLE 11.5-3 (Continued)

Radiation Monitor	Sample Point	Instrument Channels	Function	Location
D17K820 B Underdrain System	Gravity Drain System discharge lines	Gamma - scint., offline with sample pump	Control Room indication and alarm. Stop underdrain pumps on high trip.	Gravity Drain System Manhole No. 20, 608’
1D17K608 ADHR Heat Exchanger Service Water Outlet	Downstream of ADHR Heat Exchanger SW outlet	Gamma - scint., offline with sample pump	Control Room indication and alarm	Auxiliary Building 568’ - East

NOTE:

(1) (Deleted)

TABLE 11.5-4

RADIOLOGICAL ANALYSIS SUMMARY OF LIQUID PROCESS SAMPLES

	Sample Description	Grab Sample Frequency	Analysis	LLD (µCi/ml)	Purpose
1.	Reactor Coolant <Figure 5.1-1>	31 days	Iodine Dose Equivalent	5×10^{-7}	Evaluate fuel cladding integrity
		72 hours	Gamma Spectrum	5×10^{-7}	Determine radionuclides present in system (activity referenced in <Table 11.1-1>, <Table 11.1-2>, <Table 11.1-3>, <Table 11.1-4>, <Table 11.1-5>, and <Table 11.1-6>
2.	Reactor Water Cleanup System <Figure 5.4-16>, <Figure 5.4-17>	Periodically	Gamma Spectrum	1×10^{-6}	Evaluate cleanup efficiency (typical demineralizer activity referenced in <Table 11.4-4>)
3.	Condensate Demineralizer <Figure 10.1-6>				
	Influent	Periodically	Gamma Spectrum	1×10^{-6}	Evaluate demineralizer performance
	Effluent	Periodically	Gamma Spectrum	1×10^{-6}	Evaluate demineralizer performance <Table 11.2-10>
4.	Condensate Storage Tank - Unit 1 (500,000 gal.)	Weekly	Gamma Spectrum	1×10^{-6}	Tank inventory

TABLE 11.5-4 (Continued)

	Sample Description	Grab Sample Frequency	Analysis	LLD (μCi/ml)	Purpose
5.	(Deleted)				
6.	Fuel Pool Filter Demineralizer				
	Inlet and Outlet (~1,000 gpm)	Periodically	Gamma Spectrum	1×10^{-6}	Evaluate system performance
7.	Waste Collector Tank (36,500 gal. nominal cap.)	Periodically	Gamma Spectrum	1×10^{-6}	Evaluate system performance
8.	Floor Drain Collector Tank (36,500 gal. nominal cap.)	Periodically	Gamma Spectrum	1×10^{-6}	Evaluate system performance
9.	Chemical Waste Tank (19,650 gal. capacity)	Periodically	Gamma Spectrum	1×10^{-6}	Evaluate system performance

TABLE 11.5-4 (Continued)

	Sample Description	Grab Sample Frequency	Analysis	LLD (μCi/ml)	Purpose
11.	Evaporator Distillate Tanks (2; 20,000 gal. each)	Periodically	Gamma Spectrum	1×10^{-6}	Evaluate evaporator performance
12.	Nuclear Closed Cooling (<Figure 9.2-4> system flow 23,000 gpm)	Quarterly	Gamma Spectrum	1×10^{-6}	Evaluate system integrity (Normal activity negligible)
13.	ADHR Heat Exchanger Service Water Outlet (<Figure 9.2-14> system flow 3500 gpm)	Periodically	Gamma Spectrum	1×10^{-6}	Monitor leakage from the ADHR system to the Service Water system

TABLE 11.5-5

RADIOLOGICAL ANALYSIS SUMMARY OF GASEOUS PROCESS SAMPLES

Sample Description	Sample Frequency	Analysis	LLD (µCi/ml)	Purpose
1. Offgas Pre-Treatment Sample		Gamma Isotopic[2]	1×10^{-4}	Determine offgas activity[1]
a. Downstream of Offgas Condenser	Monthly			Isotopic composition and fuel performance
2. Offgas Post-treatment Sample		Gamma Isotopic[2]	1×10^{-4}	Determine offgas system cleanup performance[1]
a. Upstream of Charcoal Adsorber	Periodically			
b. Downstream of First Charcoal Adsorber	Periodically			Evaluate Charcoal Noble Gas Delay
c. Downstream of all Charcoal Beds	Monthly			Determine Noble Gas release rate from charcoal beds.

NOTES:

[1] Anticipated process flow is 6-30 cfm; compositions are referenced in <Table 11.3-1>.
[2] Principal emitters for LLD are Kr-85m, Kr-87, Kr-88, Xe-133, Xe-135, Xe-138.

TABLE 11.5-6

RADIOLOGICAL ANALYSIS SUMMARY OF LIQUID EFFLUENT SAMPLES

	Sample Description	Sample Frequency[2]	Analysis	LLD (μCi/ml)	Purpose
1.	Floor Drain Sample Tank (34,000 gal. capacity)	Batch[1]	Gamma Isotopic[5] Dissolved Gas[4] I-131	5×10^{-7} 1×10^{-5} 1×10^{-6}	Effluent discharge record <Table 11.2-15>
2.	Waste Sample Tanks (34,000 gal. capacity)	Batch[1]	Gamma Isotopic[5] Dissolved Gas[4] I-131	5×10^{-7} 1×10^{-5} 1×10^{-6}	Effluent discharge record <Table 11.2-15>
3.	Chemical Waste Distillate Tanks (19,100 gal. capacity)	Batch[1]	Gamma Isotopic[5] Dissolved Gas[4] I-131	5×10^{-7} 1×10^{-5} 1×10^{-6}	Effluent discharge record <Table 11.2-15>
4.	Detergent Drain Tank (1,550 gal. capacity)	Batch[1]	Gamma Isotopic[5] Dissolved Gas[4] I-131	5×10^{-7} 1×10^{-5} 1×10^{-6}	Effluent discharge record (normal activity negligible)
5.	Liquid Radwaste Effluents (<Table 11.2-10> for flow and activities)				
	Composite of all tanks discharged[3]	Monthly	Tritium Gross Alpha	1×10^{-5} 1×10^{-7}	Effluent discharge record
		Quarterly Quarterly	Sr-89/90 Fe-55	5×10^{-8} 1×10^{-6}	

TABLE 11.5-6 (Continued)

	Sample Description	Sample Frequency	Analysis	LLD (μCi/ml)	Purpose
6.	Circulating Water Cooling Tower Blowdown (8,200 gpm)	Quarterly	Gamma Isotopic	5×10^{-7}	Effluent discharge record
7.	Underdrain sumps (~80 gpm)	Quarterly	Gamma Isotopic	5×10^{-7}	Effluent discharge record (negligible normal activity) <Section 15.7.3.5>
8.	Discharge canal (~67,000 gpm)	Weekly	Gamma Isotopic	5×10^{-7}	Effluent discharge record <Table 11.2-13>
9.	Emergency Service Water Loops A and B, outlet of RHR Heat Exchangers	Weekly(6)	Gamma Isotopic(5)	5×10^{-7}	Effluent discharge record
			Dissolved Gas(4)	1×10^{-5}	
			I-131	1×10^{-6}	
	Composite of all Tanks Discharged	Monthly	Tritium	1×10^{-5}	
			Gross Alpha	1×10^{-7}	
		Quarterly	Sr-89/90	5×10^{-8}	
			Fe-55	1×10^{-6}	
10.	Atmospheric drain line from Turbine Building supply plenums	Weekly(7)(10)	Gamma Isotopic(5)	5×10^{-7}	Effluent discharge record
		Monthly(7)	Dissolved Gas(8)	1×10^{-5}	
		Monthly(9)	Tritium	1×10^{-5}	
			Gross Alpha	1×10^{-7}	
		Quarterly(9)	Sr-89/90	5×10^{-8}	
			Fe-55	1×10^{-6}	

TABLE 11.5-6 (Continued)

NOTES:

(1) If tank is to be discharged, analyses will be performed on each batch. If tank is not to be discharged, analyses will be performed periodically to evaluate equipment performance.

(2) If no discharge event occurs during the period frequency shall be so adjusted.

(3) Refer to <Table 11.2-8> for equipment flows.

(4) Analysis is run on one batch release per month.

(5) Principal gamma emitters for which the LLD specification applies: Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141, and Ce-144. (Note: The LLD for Ce-144 is 5×10^{-6})

(6) Samples will be drawn every 12 hours during operation if activity is greater than LLD. Samples are normally drawn daily and composited weekly for analysis.

(7) Sampling is only required when the drains have been directed to storm drains.

(8) Analysis is run on one grab sample per month.

(9) Composite sample from weekly grab samples.

(10) Samples are normally drawn daily when the drains have been directed to storm drains and composited weekly for analysis.

TABLE 11.5-7

RADIOLOGICAL ANALYSIS SUMMARY OF GASEOUS EFFLUENT SAMPLES

	Sample Description	Sample Frequency	Analysis	LLD (μCi/ml)	Purpose
1.	Unit Vents, Heater Bay/ Turbine Building Vents, Offgas Vent Pipe[4]	Weekly	Principal gamma emitters[1] for at least I-131 and Ba-La-140	$1x10^{-11}$	Effluent Record[5]
			I-131[2]	$1x10^{-12}$	
			I-133[2]	$1x10^{-10}$	
		Monthly Grab	Principal gamma emitters[3]	$1x10^{-4}$	
		Monthly Composite	Gross Alpha[1]	$1x10^{-11}$	
		Quarterly Composite	Sr-89, 90	$1x10^{-11}$	
		Monthly	Tritium	$1x10^{-6}$	

NOTES:

(1) On particulate filter - Principal gamma emitters are Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, I-131, Cs-134, Cs-137, Ce-141, and Ce-144.
(2) On charcoal cartridge.
(3) Gas samples - Principal gamma emitters are Kr-87, Kr-88, Xe-133m, Xe-133, Xe-135, and Xe-138.
(4) Effluent flow rates are referenced in <Figure 11.3-3>.
(5) Compositions are listed in <Table 11.3-9> and <Table 11.3-10>.