11.1 SOURCE TERMS

General Electric has evaluated radioactive material sources (activation products and fission product release from fuel) in boiling water reactors (BWRs) based upon early operating experience. Release of radioactive material from operating BWRs has generally resulted in doses to offsite persons which have been only a small fraction of permissible doses, or of natural background dose.

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

(1) plant equipment design,
(2) shielding design,
(3) understanding system operation and performance,
(4) measurement practicability, and
(5) evaluating radioactive material releases to the environment.

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

Design basis radiation source data applicable to SSES power uprate and Hydrogen Water Chemistry are presented in References 11.1-8 and 11.1-14, respectively. The original SSES design basis source terms in the reactor coolant (liquid and steam) are based primarily on the methodology contained in NEDO-10871, Reference 11.1-9 and were normally used by GE for evaluation of 12 and 18 month fuel cycles. Operation of the reactor, with thermal power increases from 5 to 25 percent, would be expected to have some effect on observed coolant radionuclide concentrations. However, the design basis data utilized in the development of the radiation source terms is considered to contain sufficient margin that application of the data for power levels in the range given is conservative and reasonable. In addition, it was determined (Reference 11.1-10) that the power uprate source term is applicable for use in evaluating 24 month cycles and the current fuel configuration per Section 4.2. As such, continued use of the original design basis concentrations, without change, is being maintained for plant operation under power uprate conditions with 24 month fuel cycle.

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 whose sources vary from minuscule defects in cladding to "tramp" uranium on external cladding surfaces. The relative concentrations or amounts of noble radiogas isotopes can be described as follows:

Equilibrium: $R_g–K_1Y$ (11.1-1)

Recoil: $R_g–K_2Y\lambda$ (11.1-2)

The nomenclature in Subsection 11.1.1.4 defines the terms in these and succeeding equations. The constants K_1 and K_2 describe the fractions of the total fissions that are involved in each of the releases. The equilibrium and recoil mixtures are the two extremes of the mixture spectrum that are physically possible. When a sufficient time delay occurs between the fission event and the time of release of the radiogases from the fuel to the coolant, the radiogases approach equilibrium levels in the fuel and the equilibrium mixture results. When there is no delay or impedance between the fission event and the release of the radiogases, the recoil mixture is observed.

Prior to Vallecitos Boiling Water Reactor (VBWR) and Dresden 1 experience, it was assumed that noble radiogas leakage from the fuel would be the equilibrium mixture of the noble radiogases present in the fuel.

VBWR and early Dresden 1 experience indicated that the actual mixture most often observed approached a distribution which was intermediate in character to the two extremes (Reference 11.1-1). 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: $R_g–K_3Y\lambda^{0.5}$ (11.1-3)

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

Although the previously described "diffusion" mixture was used by GE as a basis for design since 1963, the design basis release magnitude used has varied from 0.5 Ci/sec to 0.1 Ci/sec as measured after 30-min decay (t = 30 min). The noble radiogas source-term rate after 30-min decay has been used as a conventional measure of the design basis fuel leakage rate since it is conveniently measurable and was consistent with the nominal design basis 30-min off-gas holdup system used on a number of plants. Since about 1967, the design basis release magnitude used (including the 1971 source terms) was established at an annual average of 0.1 Ci/sec (t = 30 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 turbine and other component contamination affecting maintenance, have been considered in establishing this level.

Noble radiogas source terms from fuel above 0.1 Ci/sec (t = 30 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 (References 11.1-2 and 11.1-3).

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 was 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:

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

With the exception of Kr-85 with a half-life of 10.74 yr, the noble radiogas fission products in the fuel are essentially at an equilibrium condition after an irradiation period of several months (rate of formation is equal to 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 steady-state for long periods of time. The term ($e^{-\lambda t}$) is used to adjust the releases from the fuel (t = 0) to the decay time for which values are needed. Historically t = 30 min has been used. When discussing long steady-state operation and leakage from the fuel (t = 0), the following simplified form of Equation 11.1-4 can be used to describe the leakage of each noble radiogas:

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

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

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

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

Equation 11.1-6 represents a straight line when log (R_g/Y) is plotted versus log (λ); where 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-yr 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 $\pm$0.07. This is illustrated in Figure 11.1-1 as a frequency histogram. As can be seen from this figure, variations in m were observed in the range m = 0.1 to m = 0.6. After establishing the value of m = 0.4, the value of K_g can be calculated by selecting a value for R_g, or as has been done historically, the design basis is set by the total design basis source-term magnitude at t = 30 min. With R_g at 30 min = 100,000 Ci/sec, K_g can be calculated as being 2.6 x 10^7 and Equation 11.1-4 becomes:

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

This 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 min 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.

11.1.1.2 Radiohalogen Fission Products

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

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

The constant, K_h, describes the magnitude of leakage from fuel. The relative rates of halogen radioisotope leakage is expressed in terms of n, the exponent of the decay constant, λ. As was done with the noble radiogases, the average value was determined for n. The value for $\bar{n}$ is 0.5 with a standard deviation of $\pm$0.19. This is illustrated in Figure 11.1-2 as a frequency histogram. As can be seen from this figure, variations in n were observed in the range of n = 0.1 to n = 0.9.

It appeared that the use of the previous method of calculating radio-halogen leakage from fuel was overly conservative. Figure 11.1-3 relates KRB and Dresden 2 noble radiogas versus I-131 leakage. While it can be seen from Dresden 2 data during the period August 1970 to January 1971 that there is a relationship between noble radiogas and I-131 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 505μCi/sec I-131. Even at Dresden 1 in March 1965, when severe defects were experienced in stainless-steel-clad fuel, I-131 leakages greater than 500μCi/sec I-131 were not experienced. Figure 11.1-3 shows that these higher radioiodine leakages from the fuel were related to noble radiogas source terms of less than the design basis value of 0.1 Ci/sec (t = 30 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 700μCi/sec. When high radiohalogen leakages are observed, other fission products will be present in greater amounts. This may increase potential radiation exposure to operating and maintenance personnel during plant outages following such operation.

Using these judgement factors and above experience, the design basis radiohalogen source terms from fuel were established based on I-131 leakage of 700μCi/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:

$$R^h = 2.4 \times 10^7\ Y\lambda^{0\ 5}(1-e^{\lambda T})(e^{-\lambda t}) \qquad (11.1\text{-}9)$$

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

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

The concentration of radiohalogens in reactor steam can be determined by multiplying the reactor water concentration by the carryover fraction. The carryover fraction for radiohalogens is influenced by reactor water chemistry parameters such as the copper concentration and the concentration of oxidizing agents. For BWR's operating without hydrogen injection (Normal Water Chemistry), the observed "carryover" for radiohalogens has varied from about 0.1% when copper concentrations are high to about 2% when concentrations are low. The average of observed radiohalogen carryover measurements has been 1.2% by weight of reactor water in steam with a standard deviation of ±0.9. Since Susquehanna is designed with no copper in the steam cycle, a radiohalogen carryover of 2% (0.02 fraction) was used for Normal Water Chemistry operation.

Hydrogen Water Chemistry has been implemented to control the potential for stress corrosion cracking of vessel internals. Hydrogen is injected into the feedwater to reduce the radiolytic production of oxygen and hydrogen peroxide and to promote recombination of residual oxidants. Under Hydrogen Water Chemistry and conditions of low copper, the carryover fraction for halogen isotopes can be enhanced as reported in Reference 11.1-12. For a given release rate from the fuel, this will reduce the concentrations of radiohalogen isotopes in reactor water and increase their concentrations in reactor steam.

To bound operation under both Normal and Hydrogen Water Chemistry, the concentrations of radiohalogens in reactor water and steam are calculated using assumptions that maximize the concentrations. Reactor water concentrations are calculated assuming Normal Water Chemistry with 2% carryover. Reactor steam concentrations are calculated by assuming Hydrogen Water Chemistry with 8% carryover. The steam concentration was calculated by multiplying the reactor water concentration under Hydrogen Water Chemistry by 0.08 (8% carryover). The resultant concentrations are presented in Table 11.1-2.

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 11.1-3. Carryover of these radioisotopes from the reactor water to the steam is estimated to be < 0.1% (< 0.001 fraction). There is no measurable change in carryover for these other fission products under HWC. In addition to carryover, however, decay of noble radiogases in the steam leaving the reactor will result in production of noble gas daughter radioisotopes in the steam and condensate systems.

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

Except for Np-239, trace concentrations of transuranic isotopes have been observed in only a few samples where extensive and complex analyses were carried out. The predominant alpha emitter present in reactor water is Cm-242 at an estimated concentration of $10^{-6}\mu$Ci/g or less, which is below the maximum permissible concentration in drinking water application to continuous use by the general public. The concentration of alpha-emitting plutonium radioisotopes is more than one order of magnitude lower than that of Cm-242.

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

11.1.1.4 Nomenclature

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

R_g = leakage rate of a noble gas radioisotope (μCi/sec)

R_h = leakage rate of a halogen radioisotope (μCi/sec)

Y = fission yield of a radioisotope (atoms/fission)

λ = decay constant of a radioisotope (sec^{-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

C_h = concentration of a halogen radioisotope in reactor water (μCi/grams)

M = mass of water in the operating reactor (grams)

ß = cleanup system removal constant (sec^{-1}) = $\frac{\text{cleanup system flowrate(grams/sec)}}{M}$

ν = halogen steam carryover removal constant (sec^{-1})

$$= \frac{\text{steam flowrate (g ram/sec)} \times (\text{halogen carryover fraction})}{M}$$

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 11.1-4.

Under conditions of Hydrogen Water Chemistry (HWC), enhanced evolution of nitrogen to steam is experienced in a non-linear fashion as reported in EPRI Report TR-103515 (Reference 11.1-13). At conditions of moderate hydrogen injection (≤2.0 ppm in feedwater), radiation rates and equivalently nitrogen concentrations in the steam increase up to a factor of five above normal rates. Table 11.1-4 shows steam concentrations both with and without moderate HWC. In Table 11.1-4, other isotopes than nitrogen are shown but the affect of HWC upon these isotopes is not characterized because of (1) experimental difficulties in making meaningful measurements and (2) the overwhelming dominance of nitrogen in the radiation signature of the steam. Therefore, the concentrations of these isotopes are not assumed to be changed in any significant manner under HWC. In addition, though the water concentrations of nitrogen will decrease under HWC, the decrease is ignored for purposes of conservatively bounding the majority of normal operating conditions.

11.1.2.2 Noncoolant Activation Products

The activation products formed by activation of impurities in the coolant or by corrosion of irradiated system materials are not adequately correlated by simple equations. The design basis source terms of noncoolant activation products have been estimated conservatively from experience data. The resultant concentrations are presented in Table 11.1-5. Carryover of these isotopes from the reactor water to the steam is estimated to be < 0.1% (< 0.001 fraction).

11.1.3 TRITIUM

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

(1) activation of naturally occurring deuterium in the primary coolant,

(2) nuclear fission of $U0_2$ fuel, and

(3) neutron reactions with boron used in reactivity control rods.

The tritium, formed in control rods may be released from a BWR in liquid or gaseous effluents. 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 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 (μ Ci/sec/MWt)

Σ = macroscopic thermal neutron cross section (cm^{-1})

ϕ = thermal neutron flux (neutrons/(cm5) (sec))

V coolant volume in core (cm;)

λ = tritium radioactive decay constant (1.78×10^{-9} sec^{-1})

P = reactor power level (MWt)

For recent BWR designs, R_{act} is calculated to be $1.3 \pm 0.4 \times 10^4$ μCi/sec/Mwt. The uncertainty indicated is derived from the estimated errors in selecting values for the coolant volume in the core, coolant density in the core, abundance of deuterium in light water (some additional deuterium will be present because of the H (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 will then be available for release in liquid and gaseous effluents) is much more difficult to estimate. However, since zircaloy-clad fuel rods are used in BWRs, essentially all fission product tritium will remain in the fuel rods unless defects are present in the cladding material (Reference 11.1-4).

The study made at Dresden 1 in 1968 by the U.S. Public Health Service suggests that essentially all of the tritium released from the plant could be accounted for by the deuterium activation source (Reference 11.1-3). For purposes of estimating the leakage of tritium from defected fuel, it can be assumed that it leaks in a manner similar to the leakage of noble radiogases. Thus, use can be made of the empirical relationship described as the "diffusion mixture" used for predicting the source term of individual noble gas radioisotopes as a function of the total noble gas source term. The equation which describes this relationship is:

$$R_{dif} = KY\lambda \qquad (11.1\text{-}12)$$

where,

R_{dif} = leakage rate of tritium from fuel (μ Ci/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 $10^5 \mu Ci/sec$ after 30-min decay, leakage from fuel can be calculated to be about 0.24μCi/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 0.10μCi/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 $0.12 \pm 0.12 \mu Ci/sec$, or 0.0 to 0.24μCi/sec.

For this reactor the estimated total tritium appearance rate in reactor coolant and release rate in the effluent is about 150 Ci/year/unit.

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

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

Recombination of radiolysis gases in the air ejector off-gas system will form water, which is condensed and returned to the main condenser. This tends to reduce the amount of tritium leaving in gaseous effluents. Reducing the gaseous tritium release will result in a slightly higher tritium concentration in the plant process water. Reducing the amount of liquid effluent discharged will also result in a higher process coolant equilibrium tritium concentration.

Essentially, all tritium entering the primary coolant will eventually be 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-yr. half-life of tritium.

The USPHS study at Dresden 1 estimated that approximately 90% of the tritium release was observed in liquid effluent, with the remaining 10% leaving as gaseous effluent (Reference 11.1-5). Efforts to reduce the volume of liquid effluent discharges may change this distribution so that a greater amount of tritium will leave as gaseous effluent. From a practical standpoint, the fraction of tritium leaving as liquid effluent may vary between 60 and 90% with the remainder leaving in gaseous effluent.

11.1.4 CORE INVENTORY, FUEL EXPERIENCE AND DEPRESSURIZATION SPIKING

11.1.4.1 Core Inventory

Core fission product inventory information is used in establishing fission product source terms for accident analysis and is, therefore, discussed in Chapter 15 and Chapter 18.

11.1.4.2 Fuel Experience

A discussion of fuel experience gained for BWR fuel including failure experience, burnup experience, and thermal conditions under which the experience was gained is available in three GE topical reports (references 11.1-2, 11.1-3 and 11.1-6). The basis for this experience is fuel produced and used during the late 1960's and early 1970's, generally with 7x7 fuel lattices. In the two decades since these reports were produced, fuel manufacturing and design has advanced through 8x8, 9x9 and 10x10 lattice designs using barrier, partial length, and formed (variable loading) fuel rods. With these advances, fuel performance has gradually improved to the point that fuel rod leaks are the exception in reactor performance. Along with performance improvement, the operating cycle lengths have also increased to the degree that 24 month operating cycles and plant availabilities in the range of 80% are becoming the norm. Increases in operating cycle length involve higher fuel exposures, but do not result in significant increase in radiologically significant fission products with the exception of isotopes of Cs-134, Cs-137, and Sr-90. Gap or plenum inventories are expected to increase with extended exposure which result in higher core sources to the reactor coolant. Manufacturing improvements and quality control have reduced leakage source terms so that no increase in coolant concentrations have been observed that vary significantly over current values. The existing design basis concentrations as given in Tables 11.1-1, 11.1-2, 11.1-3, 11.1-4 and 11.1-5 still effectively bound operating experiences (often by orders of magnitude). Reference 11.1-11 discusses the basis for these conclusions.

11.1.4.3 Depressurization Spiking

The data presented in Table 11.1-6 provides a conservative representation of "spiking" source terms for iodines and noble gases. The tabulated values represent the total activity per bundle which is available for release during a complete depressurization of the reactor vessel. These source terms may be applied in analysis of events in which reactor coolant is released while the vessel is being depressurized.

The data were developed from early observations of spiking releases at plants with 7x7 fuel assemblies and projected to estimate the activities associated with a 95th percentile cumulative probability spiking event with 8x8 fuel. Given the greatly improved performance of current fuel designs, it is expected that the data will be conservative for power uprate. Further, Reference 11.1-10 concluded that the use of the power uprate data can be continued for the extended 24 month fuel cycle and is applicable to other fuel types of different mechanical design.

11.1.5 PROCESS LEAKAGE SOURCES

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

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

Experience with the airborne radiological releases from BWR building heating, ventilating, and air conditioning and the main condenser mechanical vacuum pump have been compiled and evaluated in NEDO-21159, "Airborne Releases from BWRs for Environmental Impact Evaluations", March 1976, Licensing Topical Report (Reference 11.1-7). This report is periodically updated to incorporate the most recent data on airborne emission. The results of these evaluations are based on data obtained by utility personnel and special in-plant studies of operating BWR plants by independent organizations and the General Electric Company. An evaluation of the radioactive releases from ventilation systems, for compliance with Appendix I to 10CFR50, is given in Section 11.3. An evaluation of important exposure to airborne activity is given in Subsection 12.2.2.

11.1.6 OTHER RELEASES

All other releases are covered in Section 11.3.

11.1.7 REFERENCES

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).

11.1-2 Williamson, H. E., Ditmore, D.C., "Experience with BWR Fuel Through September 1971" NEDO-10505, May 1972. (Update).

11.1-3 Elkins, R. B., "Experience with BWR Fuel Through September 1974," NEDO-20922, June 1975.

11.1-4 Ray, J. W., "Tritium in Power Reactors," Reactor and Fuel Processing Technology, 12 (1), pp. 19-26, Winter 1968-1969.

11.1-5 Kahn, B., et al, "Radiological Surveillance Studies at a Boiling Water Nuclear Power Reactor," BRH/DER 70-1, March 1970.

11.1-6 Williamson, H. E., Ditmore, D. C., "Current State of Knowledge of High Performance BWR Zircaloy Clad U02 Fuel," NEDO-10173, May 1970.

11.1-7 Marrero, T. R., "Airborne Releases From BWRs for Environmental Impact Evaluations," NEDO-21159, March 1976.

11.1-8 Pankratz, D. R., Faynshtein, K, "Power Uprate Engineering Report For Susquehanna Steam Electric Station Units 1 and 2," NEDC-32161P, December 1993.

11.1-9 "Technical Derivation of BWR 1971 Design Basis Radioactive Material Source Terms", NEDO-10871, March 1973.

11.1-10 GE Document B13-01869-095, Susquehanna Steam Electric Station Extended Fuel Cycle, Radiological Source Terms, S.S. Wang, August 1997. (PP&L Calculation EC-FUEL-1333 Rev. 0, “GE Source Term Study for 24 Month Cycles”).

11.1-11 Baker, D.A., et al, “Assessment of the Use of Extended Burnup Fuel in Light Water Power Reactors”, NUREG/CR-5009 (PNL-6258), U.S. NRC, February 1988

11.1-12 Lin, C.C. “Chemical Behavior of Radioiodine in Boiling Water Reactor Systems II: Effects of Hydrogen Water Chemistry”, Nuclear Technology, Vol. 97, Jan. 1992, pg. 71-78.

11.1-13 “BWR Water Chemistry Guidelines – 1996 Revision, Normal and Hydrogen Water Chemistry”, EPRI TR-103515, R1, BWRUIP-29, December, 1996.

11.1-14 Careway, H.A., and E. M. Skeehan, “Susquehanna Steam Electric Station, Units 1 and 2, Radioactive Material Source Term Evaluation for Normal Conditions with Hydrogen Water Chemistry.” GE Document GE-NEA41-00074-07, Revision 1. April 1997.

TABLE 11.1-1

NOBLE RADIOGAS SOURCE TERMS

Isotope	Half-life	Source Term @ t = 0 (μ Ci/sec)	Source Term @ t = 30 min (μ Ci/sec)
Kr-83m	1.86 hr	3.4×10^3	2.9×10^3
Kr-85m	4.4 hr	6.1×10^3	5.6×10^3
Kr-85	10.74 yr	10 to 20*	10 to 20*
Kr-87	76 min	2.0×10^4	1.5×10^4
Kr-88	2.79 hr	2.0×10^4	1.8×10^4
Kr-89	3.18 min	1.3×10^5	1.8×10^2
Kr-90	32.3 sec	2.8×10^5	------
Kr-91	8.6 sec	3.3×10^5	------
Kr-92	1.84 sec	3.3×10^5	------
Kr-93	1.29 sec	9.9×10^4	------
Kr-94	1.0 sec	2.3×10^4	------
Kr-95	0.5 sec	2.1×10^3	------
Kr-97	1.0 sec	1.4×10^1	------
Xe-131m	11.96 day	1.5×10^1	1.5×10^1
Xe-133m	2.26 day	2.9×10^2	2.8×10^2
Xe-133	5.27 day	8.2×10^3	8.2×10^3
Xe-135m	15.7 min	2.6×10^4	6.9×10^3
Xe-135	9.16 hr	2.2×10^4	2.2×10^4
Xe-137	3.82 min	1.5×10^5	6.7×10^2
Xe-138	14.2 min	8.9×10^4	2.1×10^4
Xe-139	40 sec	2.8×10^5	------
Xe-140	13.6 sec	3.0×10^5	------
Xe-141	1.72 sec	2.4×10^5	------
Xe-142	1.22 sec	7.3×10^4	------
Xe-143	0.96 sec	1.2×10^4	------
Xe-144	9.0 sec	5.6×10^2	------
TOTALS		$\sim 2.5 \times 10^6$	$\sim 1.0 \times 10^5$

* Estimated from experimental observations.

TABLE 11.1-2

HALOGEN RADIOISOTOPES IN REACTOR WATER AND STEAM

Isotope	Half-Life	Water Concentration ($\mu Ci/g$)	Steam Concentration ($\mu Ci/g$)
Br-83	2.40 hr	1.5×10^{-2}	4.9×10^{-4}
Br-84	31.8 min	2.7×10^{-2}	1.2×10^{-3}
Br-85	3.0 min	1.7×10^{-2}	1.2×10^{-3}
I-131	8.065 day	1.3×10^{-2}	3.4×10^{-4}
I-132	2.284 hr	1.2×10^{-1}	3.9×10^{-3}
I-133	20.8 hr	8.9×10^{-2}	2.4×10^{-3}
I-134	52.3 min	2.4×10^{-1}	9.7×10^{-3}
I-135	6.7 hr	1.3×10^{-1}	3.7×10^{-3}

Water concentrations maximized by use of 2% steam carryover, while steam concentrations maximized by use of 8% steam carryover commensurate with operations with hydrogen water chemistry.

TABLE 11.1-3

OTHER FISSION PRODUCT RADIOISOTOPES IN REACTOR WATER

Isotope	Half-Life	Concentration (μCi/g)
Sr-89	50.8 day	3.1×10^{-3}
Sr-90	28.9 yr	2.3×10^{-4}
Sr-91	9.67 hr	6.9×10^{-2}
Sr-92	2.69 hr	1.1×10^{-1}
Zr-95	65.5 day	4.0×10^{-5}
Zr-97	16.8 hr	3.2×10^{-5}
Nb-95	15.1 day	4.2×10^{-5}
Mo-99	66.6 hr	2.2×10^{-2}
Tc-99m	6.007 hr	2.8×10^{-1}
Tc-101	14.2 min	1.4×10^{-1}
Ru-103	39.8 day	1.9×10^{-5}
Ru-106	368 day	2.6×10^{-6}
Te-129m	34.1 day	4.0×10^{-5}
Te-132	78.0 hr	4.9×10^{-2}
Cs-134	2.06 yr	1.6×10^{-4}
Cs-136	13.0 day	1.1×10^{-4}
Cs-137	30.2 yr	2.4×10^{-4}
Cs-138	32.3 min	1.9×10^{-1}
Ba-139	83.2 min	1.6×10^{-1}
Ba-140	12.8 day	9.0×10^{-3}
Ba-141	18.3 min	1.7×10^{-1}
Ba-142	10.7 min	1.7×10^{-1}
Ce-141	32.53 day	3.9×10^{-5}
Ce-143	33.0 hr	3.5×10^{-5}
Ce-144	284.4 day	3.5×10^{-5}
Pr-143	13.58 day	3.8×10^{-5}
Nd-147	11.06 day	1.4×10^{-5}
Np-239	2.35 day	2.4×10^{-1}

TABLE 11.1-4

COOLANT ACTIVATION PRODUCTS IN REACTOR WATER AND STEAM

Isotope	Half-life	Steam Concentration (μCi/g)		Reactor Water Concentration (μCi/g)
		w/o HWC	w/HWC	
N-13	9.99 min	7×10^{-3}	3.5×10^{-2}	4×10^{-2}
N-16	7.13 sec	$5 \times 10^{+1}$	$2.5 \times 10^{+2}$	$5 \times 10^{+1}$
N-17	4.14 sec	2×10^{-2}	1×10^{-1}	6×10^{-3}
O-19	26.8 sec	8×10^{-1}	8×10^{-1}	7×10^{-1}
F-18	109.8 min	4×10^{-3}	4×10^{-3}	4×10^{-3}

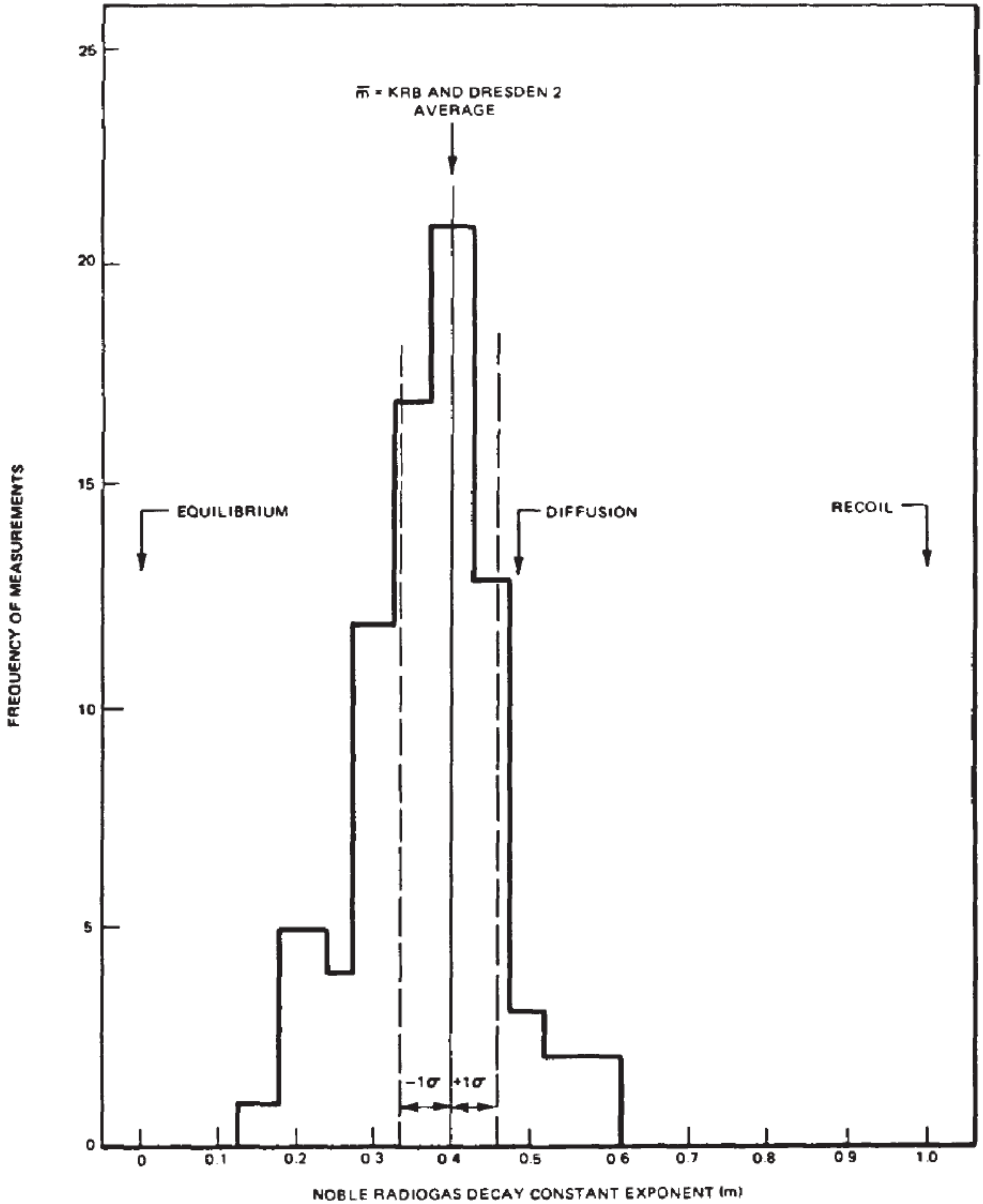
(w/o HWC – without Hydrogen Water Chemistry; w/HWC – with HWC)

TABLE 11.1-5

NONCOOLANT ACTIVATION PRODUCTS IN REACTOR WATER

Isotope	Half-life	Concentration (μ Ci/g)
Na-24	15.0 hr	2.0×10^{-3}
P-32	14.31 day	2.0×10^{-5}
Cr-51	27.8 day	5.0×10^{-4}
Mn-54	313.0 day	4.0×10^{-5}
Mn-56	2.582 hr	5.0×10^{-2}
Co-58	71.4 day	5.0×10^{-3}
Co-60	5.258 yr	5.0×10^{-4}
Fe-59	45.0 day	8.0×10^{-5}
Ni-65	2.55 hr	3.0×10^{-4}
Zn-65	243.7 day	2.0×10^{-6}
Zn-69m	13.7 hr	3.0×10^{-5}
Ag-110m	253.0 day	6.0×10^{-5}
W-187	23.9 hr	3.0×10^{-3}

TABLE 11.1-6 DEPRESSURIZATION SPIKING ACTIVITY 95th PERCENTILE CUMULATIVE PROBABILITY	
Isotope	**Activity** (curie/bundle)
I-131	2.14
I-132	3.21
I-133	5.03
I-134	5.44
I-135	4.79
Kr-83m	0.90
Kr-85m	2.23
Kr-85	0.49
Kr-87	4.33
Kr-88	6.12
Kr-89	7.96
Xe-131m	0.066
Xe-133m	0.33
Xe-133	11.6
Xe-135m	1.80
Xe-135	11.0
Xe-137	10.5
Xe-138	10.6



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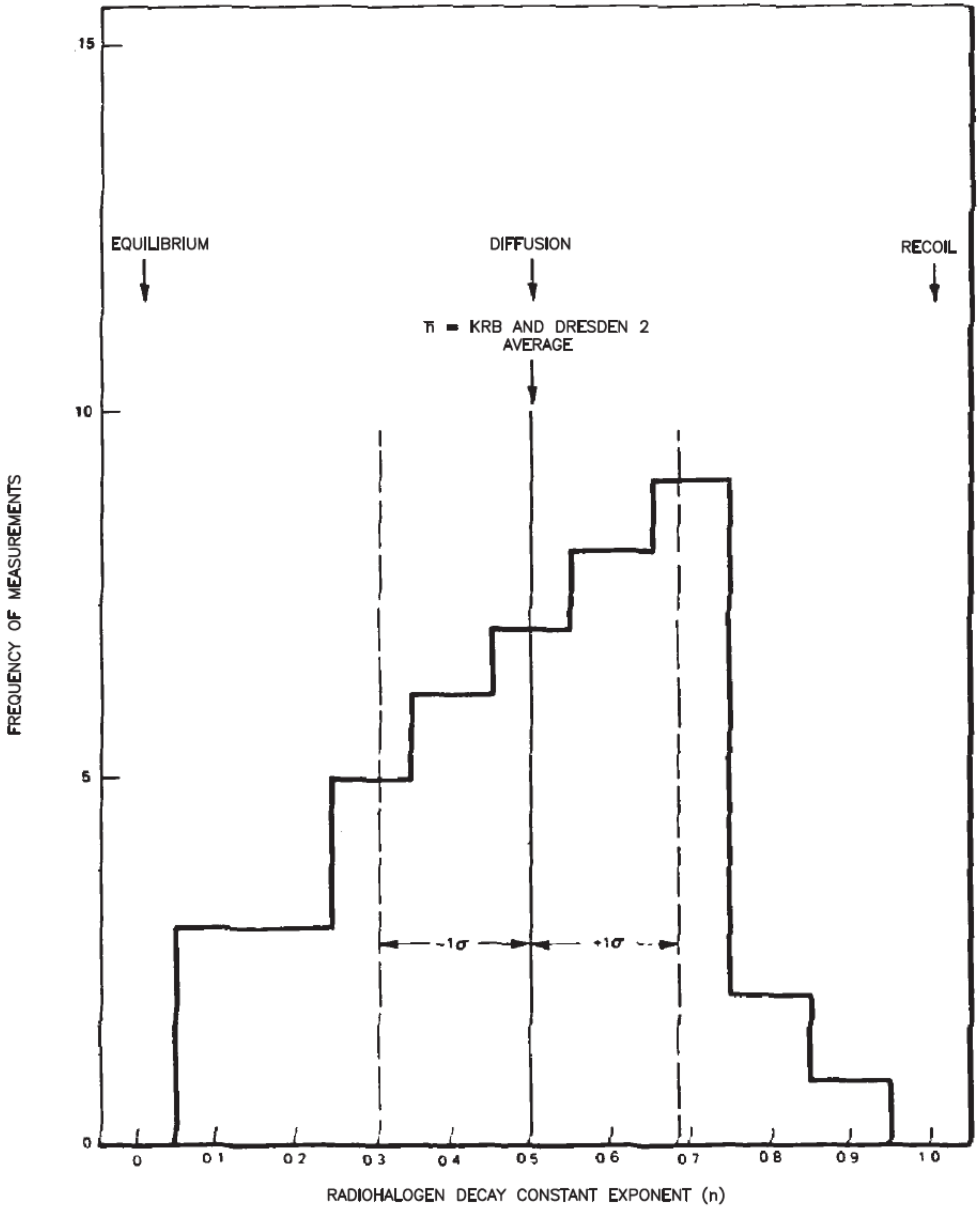
SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

NOBLE RADIOGAS DECAY
CONSTANT EXPONENT
FREQUENCY HISTOGRAM

FIGURE 11.1-1, Rev 47

AutoCAD: Figure Fsar 11_1_1.dwg



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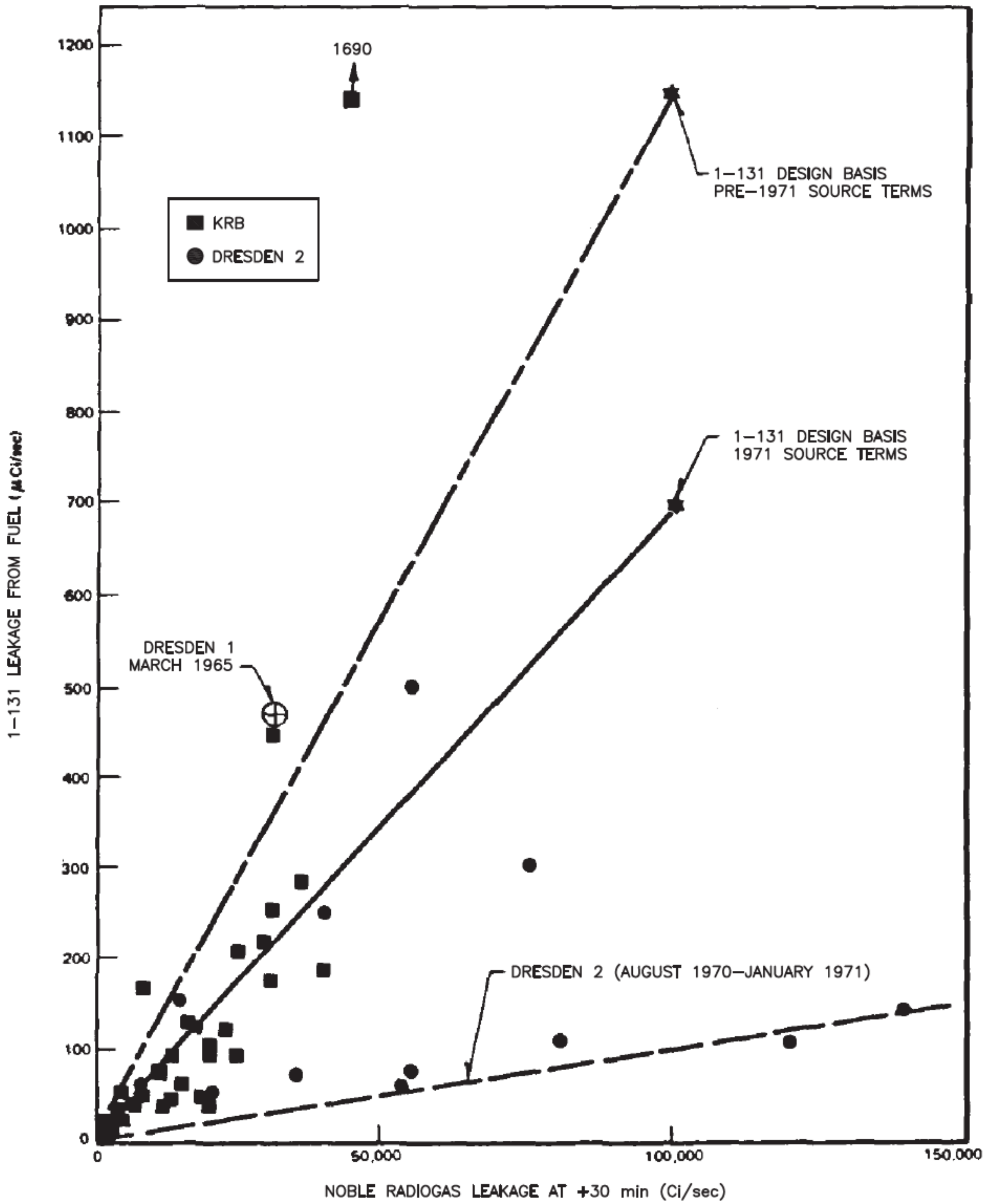
RADIOHALOGEN DECAY
CONSTANT EXPONENT
FREQUENCY HISTOGRAM

FIGURE 11.1-2, Rev 47

AutoCAD: Figure Fsar 11_1_2.dwg



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NOBLE RADIOGAS LEAKAGE
VS.
1-131 LEAKAGE

FIGURE 11.1-3, Rev 47

AutoCAD: Figure Fsar 11_1_3.dwg

11.2 LIQUID WASTE MANAGEMENT SYSTEMS

The Liquid Waste Management System (LWMS) collects, processes, stores, and monitors for reuse or disposal the radioactive liquid wastes generated as a result of Susquehanna SES operation. The LWMS consists of liquid radwaste, (high and low purity wastewater principally from equipment and floor drains), chemical waste, and laundry drain subsystems. Equipment location drawings are shown on Dwgs. M-270, Sh. 1, M-271, Sh. 1, M-272, Sh. 1, M-273, Sh. 1, M-274, Sh. 1, M-220, Sh. 1, and M-230, Sh. 1. A flow diagram for the LWMS subsystems is given on Figure 11.2-8 and Process and Instrumentation Diagrams (P&IDs) are presented on Dwgs. M-162, Sh. 1, M-162, Sh. 2, M-162, Sh. 3, and Dwg. M-164, Sh. 1.

11.2.1 DESIGN BASES

The objectives and criteria which form the bases for the design of the LWMS are as follows:

a. The LWMS is capable of recycling the majority of potentially radioactive wastes to condensate water quality requirements. Sufficient treatment equipment is provided to process the liquid waste from both nuclear units without impairing the operation or availability of the plant during normal operations and anticipated operational occurrences to satisfy the radiation protection requirements and design objectives of 10CFR20 and 10CFR50.

b. The LWMS has no nuclear safety-related function as a design basis.

c. Connections are provided for processing flexibility to permit the installation of vendor supplied mobile liquid processing systems. The mobile liquid processing systems are subjected to the same performance objectives as the permanently installed systems and are approved for use in accordance with applicable SSES programs and procedures.

d. Excess water or liquid wastes that cannot be processed to meet the quality requirements for recycling are sampled and discharged.

e. The LWMS is designed so that no potentially radioactive liquids can be directly discharged to the environment unless they have been monitored and diluted with the cooling tower blowdown. This results in offsite radionuclide releases, activity concentrations, and radiation exposures to individuals and the general population within the limits of 10CFR20 and 10CFR50.

f. The LWMS is designed to keep the exposure to the general population and plant personnel during normal operation and maintenance as low as reasonably achievable (ALARA).

g. The expected radionuclide activity concentrations in the LWMS process equipment are based on reactor water radioactivity concentrations corresponding to fuel defects that result in 50,000 μCi/sec noble gas release rate for one reactor unit after a 30 minute delay. The design basis radionuclide activity concentrations in the LWMS process equipment are based on reactor water radioactivity concentrations corresponding to fuel defects that result in 100,000 μCi/sec noble gas release rate for one reactor unit after a 30 minute delay.

h. The seismic and quality group classifications of the LWMS components and piping and the radwaste building are listed in Section 3.2.

i. Redundant and backup equipment, alternate process routes, interconnections, and spare volumes are designed into the system to provide for operational and unanticipated surge waste volumes due to refueling, abnormal leakage rates, decontamination activities, equipment down time, maintenance, and repair.

j. The expected daily inputs and activities to each of the three subsystems are shown in Tables 11.2-1 and 11.2-2. An evaluation of the causes for the maximum expected inputs for each subsystem shows that operational modes exclude, and the unlikely occurrence of the same failure in both units minimizes, the potential for coincidental maximum input from both units into the same subsystem.

k. Table 11.2-3 shows the design parameters for the LWMS equipment. The usage factors for pumps and processing equipment provided in Table 11.2-3 show sufficient reserve capacities for the maximum expected inputs.

l. Expected flow rates for streams shown on Figure 11.2-8, are as given in Tables 11.2-4 and 11.2-10.

m. Concurrent refuelings or cold startups are not design bases for the Station.

n. The expected and design basis radionuclide activity inventories of major LWMS components are shown in Table 11.2-5 and 11.2-6 and are based upon the following assumptions:

    1) Reactor water radionuclide activity concentrations are listed in Tables 11.1-2, 11.1-3, 11.1-4 and 11.1-5 for design conditions, and Table 11.2-9 for expected conditions.

2) Radwaste inputs, isotopic activities, and component parameters are based on data from operating plants (NUREG 0016), data collected by GE, and design and operating data for Susquehanna SES as shown in Tables 11.2-1, 11.2-2, 11.2-3, 11.2-4 and 11.2-10.

3) Decontamination factors used for determining activity retention by cleanup equipment are as follows:

<u>Filtration</u>:	Expected/Design Basis
Activation/Corrosion Products	10 / 100
<u>Demineralization</u>:	
Cesium and Rubidium	10 / 100
Anions and Other Fission Products	100 / 100

4) While a process stream is collecting in a collection or sample tank, the isotopes already in the tank are undergoing radioactive decay (see Table 11.2-11 for expected holdup times).

o. Major LWMS components are located in separate shielded compartments based on anticipated radiation levels. Accessibility for maintenance and repair while operating redundant components of the system was considered.

p. Instrumentation and controls are designed and located to minimize exposure to the operating personnel.

q. Floor drains and sloped floors are provided in equipment rooms to control the spread of contamination from leakage. Except for indoor tanks containing processed liquids (i.e. sample tanks), equipment rooms containing liquid radwaste are provided with curbs or elevated door thresholds, with drains routed to the appropriate LWMS subsystem, to minimize the potential spread of contamination from leaks or spills. The Equipment and Floor Drainage System, which includes provisions for collecting potentially radioactive liquid, chemical and detergent wastes, is discussed in Section 9.3.3.

r. Table 11.2-16 lists tanks outside reactor containment which contain potentially radioactive liquids and the provisions for high level monitoring and alarm and for collecting and processing overflow.

Atmospheric liquid radwaste tanks are provided with an overflow connection of at least the size of the largest inlet connection. (common overflow from laundry drain tanks (OT-311A&B) to chemical radwaste funnel is exempt from this requirement. Refer to Section 11.2.2.4.) The overflow is connected below the tank vent, at least one inch above the high level alarm trip point. Overflow liquid is routed to a redundant tank or to the nearest atmospheric drainage point.

Tanks located outside the reactor containment and containing radioactive materials in liquids are designed to prevent uncontrolled releases of radioactive materials due to spillage in buildings or from outdoor dikes and storage tanks. The following design features are included for tanks that may contain radioactive materials:

1) Except as noted on Table 11.2-16, tanks have provisions to monitor liquid levels, alarm potential overflow conditions, and have their overflows, drains and sample lines routed to the LWMS. Retention by an intermediate sump or drain tank, designed for handling radioactive materials and having provisions for routing to the LWMS, is employed as shown in Table 11.2-16.

2) Indoor tanks have floor drains routed to the LWMS. Retention by an intermediate sump or drain tank designed for handling radioactive materials and having provisions for routing to the LWMS is employed as shown in Table 11.2-16.

3) Outdoor tanks have a dike or retention pond capable of containing the tank contents in the event of a rupture, preventing runoff in the event of a tank overflow and providing for sampling collected liquids and routing them to the LWMS.

s. Design features provided to reduce maintenance, equipment down time, liquid leakage, radioactive gaseous releases to the building atmosphere, and to facilitate cleaning, or otherwise improve radwaste operations include the following, where practicable:

1) Automatically and manually controlled valves and instrumentation are located outside equipment rooms that contain large volumes of radioactive materials, unless required by the process.

2) Sequencer controlled valve positioning and pump operations upon manual initiation of main process steps.

3) Automatic or manual flushing of subsystems after process termination.

4) Manual override provisions for all sequencer operated and interlocked components.

5) Manholes and access ladders on storage tanks.

6) Remote manual drain valves on storage tanks.

7) Low point piping and equipment drains in isolable portions of systems.

8) Condensate flushing connections on all major piping routes.

9) Vents of LWMS tanks, filters, and demineralizer are routed to the building ventilation system filters. A slight negative pressure against atmosphere is maintained in these components when vented.

10) Welded piping connections, where practical. Line sizes over two inches are butt welded to avoid crud traps.

11) Pumps provided with mechanical seals with flush connections.

12) Pump baseplates with drip lips.

t. Processed wastes are collected in sample tanks prior to their reuse as condensate or are monitored and discharged into the cooling tower blowdown pipe for dilution before entering the Susquehanna River.

u. Control and monitoring of radioactive releases in accordance with General Design Criteria 60 and 64 of Appendix A to 10CFR50 is discussed in Subsection 11.2.3 and Section 11.5.

11.2.2 SYSTEM DESCRIPTIONS

11.2.2.1 General

The Liquid Waste Management System serves both reactor units and consists of three processing subsystems, each for collecting, processing, storing, monitoring, and dispositioning of specific types of liquid wastes according to their conductivity, chemical composition, and radioactivity. These subsystems are:

a) Liquid Radwaste Processing

b) Liquid Radwaste Chemical Processing

c) Liquid Radwaste Laundry Drain Processing

Waste influent to each of the subsystems is collected in batch tanks to allow for quality and volume monitoring before processing.

Recirculation of the collection and sample tank contents while isolated or being pumped out minimizes settling of suspended solids and provides representative grab samples. The recirculation lines with stroke limited valves guarantee a minimum pump flow for cooling in case a pump discharge valve is closed.

Recirculation and pump-out of all process tanks is remote manually initiated and ceases upon a low level signal. This protects the pumps from cavitation. Manual fill selection of multiple tanks is provided. In the automatic mode, the tanks are filled sequentially to their high level. High level alarms and level indication over the live volume range are provided in the radwaste control room.

Simultaneous filling of one tank and mixing, sampling, or processing of another is possible through separate suction and recirculation lines and pumps for each tank.

A local pressure gauge is provided in each pump discharge line. A more detailed description of the instrumentation and controls of the LWMS is contained in Section 7.7.

Suction lines of multiple pump and tank arrangements are cross-connected to provide backup capability. Manual valves and individual controls for all automatic valves and pumps also allow transfer of waste between tanks, complete pump-out of tanks for maintenance or repair, system flushing with condensate, and bypassing of process equipment.

The following subsections describe additional features of each of the three subsystems.

11.2.2.2 Liquid Radwaste Processing Subsystem

The liquid radwaste processing subsystem is used to process radioactive waste water from equipment leakage, floor drains and other sources throughout the Station. (see Table 11.2-1). A schematic flow diagram and P&IDs for the liquid radwaste processing subsystem are presented in Figure 11.2-8 and Dwgs. M-162, Sh. 1, M-162, Sh. 2 and M-162, Sh. 3.

During normal plant operation, waste water is routed directly or through local sumps to collection tanks located in the radwaste building. Three sets of twin tanks, which are not individually isolable, collect the low conductivity waste in batches. When in the Automatic Fill Mode, a radwaste building control room alarm annunciates when two out of three tank sets are unavailable and a main control room alarm annunciates when all three collection tank sets are unavailable. When all three radwaste collection tank sets are unavailable, the wastewater is routed into the liquid radwaste surge tanks. These additional two twin sets of tanks provide surge capacity for unanticipated high waste volumes. These tanks are associated with one common pump. High conductivity waste inadvertently collected in the liquid radwaste tanks can be pumped directly to the chemical waste tank.

The liquid radwaste subsystem process stream normally consists of separate filtration and demineralization. The resultant condensate quality water is collected in sample tanks and subsequently transferred for reuse in the plant or to the condensate storage tank. Excess water, or off quality water, is discharged through the monitored discharge pipe into the cooling tower blowdown pipe.

Branch lines from the liquid radwaste processing subsystem are provided for hookup to a mobile radwaste processing system. Connections from and to the liquid radwaste collection and sample tanks are provided.

Radwaste Filters

Two vertical centrifugal filters are provided for filtering low conductivity liquids. The two filters may be operated in either parallel or in series. Normally both filters are used in series. One filter may be used for filtering of liquid waste with the second used as a backup or out of service.

Normally, the filters are operated with a powdered resin precoat. Normal filtering flow is in the range of 40 to 160 gpm (nominally 100 gpm) which is 1/7 to 1/2 gpm per sq. ft. Based on operating experience, an adjustable amount of filter aid may be injected into the waste inlet stream to extend the filter run length over the full allowable differential pressure range up to 90 psi.

The precoat and filter aid pumps and tanks are supplied with the filters and are located in a normally accessible area. When used to either supplement or back up the ion exchange function of the radwaste demineralizer, the filter plates are precoated with powdered ion exchange resin.

The filtering process is terminated upon a high differential pressure alarm across the filter or when the maximum allowable cake thickness between the filter plates occurs. The latter, although not normally the cause, can be observed through an illuminated sight glass in the filter vessel shell and a filter run timer set accordingly. Experience has shown that a filter will alarm on differential pressure prior to reaching maximum cake thickness. Flow controllers keep the flow rate independent of the increasing pressure drop over a filter run length.

Upon termination of a filter run, the filter vessel is drained to the waste sludge phase separator and then centrifugally discharged to the waste mixing tanks. The filter cake is spun off the filter plates by motorized rotation of the vertical stacking shaft, and a scraper at the vessel bottom discharges it through a vertical chute into the waste sludge phase separator of the solid waste management system described in Section 11.4. After backflushing into the waste sludge phase separator, the filter is filled and ready for a fresh precoat. Normally a filter is back in service in approximately two hours.

Radwaste Demineralizer

The filtered liquid waste is processed at a nominal flow rate of 100 gpm through one nonregenerated deep bed demineralizer before entering the liquid radwaste sample tanks. The differential pressure between the vessel inlet and outlet is indicated and alarmed over an adjustable range up to 25 psi. The differential pressure and a level indication is provided on a local instrument rack and in the radwaste control room.

The effluent conductivity instrumentation is designed to indicate, record, and alarm at a high and high-high value of conductivity as specified in the Chemistry Control Program. The demineralizer inlet valve is designed to close automatically upon high-high conductivity in the effluent, high differential pressure or loss of control air or power. Experience has shown the effluent conductivity instrumentation is inaccurate and unreliable therefore the instrumentation is not operated and the alarm and isolation functions are not available. The alternate more conservative method of controlling the quality of the demineralizer effluent to the LRW sample tanks involves sampling and analysis of the tanks on a batch basis prior to return to the CST. Sample tank water within more detailed and conservative specifications than conductivity will be returned to the CST; out of specification water will be discharged to the river or reprocessed. Exhausted or fouled ion exchange resins are sluiced to the spent resin tank for subsequent dewatering and disposal.

Fresh resin beads are manually loaded through the resin addition funnel located above the demineralizer vessel and mixed inside using low pressure compressed air. The air is vented through the radwaste mist eliminator to the tank exhaust system filter described in Subsection 9.4.3. Total outage time for removal and replacement of the resin bed is approximately six hours.

11.2.2.3 Liquid Radwaste Chemical Processing Subsystem

The chemical processing subsystem generally treats high conductivity wastes from potentially radioactive sources throughout the plant as listed in Table 11.2-1. A schematic flow diagram and the P&IDs for this subsystem are provided in Figure 11.2-8 and Dwgs. M-163, Sh. 1, M-163, Sh. 2 and M-163, Sh. 3.

The high conductivity wastes are routed directly or via local sumps to the chemical processing subsystem collection tanks. Except for the chemical waste neutralizer tanks, which are located in the turbine building, all components of this subsystem are located in the radwaste building.

Two chemical waste neutralizer tanks are associated with each reactor unit. Due to their large volume relative to the chemical waste tank and with the cessation of chemical regeneration of the condensate demineralizers, these tanks are used to store waste originally collected in the chemical waste tank, as described below, prior to processing.

Two chemical waste neutralizer tank pumps recirculate the chemical waste in each tank while local grab samples are taken. In order to bring the pH value of the chemical waste in the neutralizer tanks within the required range for processing, small amounts of sulfuric acid and sodium hydroxide are injected into the system.

Various chemical solutions originating from laboratory, equipment, and sample rack drains and decontamination stations throughout the plant are collected in the chemical waste tank located in the radwaste building. Auxiliary boiler blowdown waste is also collected in the chemical waste tank due to the possibility of radioactive contamination. The chemical waste tank contents are recirculated by one of the two redundant chemical waste tank pumps while remote grab samples may be taken on the radwaste building sample rack.

The pH value of the chemical waste tank contents can be adjusted in the same manner as that of the chemical waste neutralizer tanks. Pumps for chemical wastes are provided with automatic gland seal flushing with condensate.

Liquid waste from the chemical waste tank and from the chemical waste neutralizing tanks is processed through a mobile radwaste processing system. Branch lines from the chemical processing subsystem are provided to the monitoring room near the truck loading area in the radwaste building for hookup of mobile radwaste processing systems. The chemical waste processing subsystem has been modified to allow the mobile radwaste processing equipment to discharge directly to the distillate sample tank where the processed fluid can be sampled and subsequently discharged to the environment.

The controls and instrumentation of the liquid radwaste chemical processing subsystem are as described in Subsection 11.2.2.1 except that the chemical waste neutralizer, chemical waste, and evaporator distillate sample tanks are equipped with level recording instrumentation in place of indicating instruments to provide performance records.

Mobile Radwaste Processing

The waste water collected in the chemical waste tank and the chemical waste neutralizer tanks is processed through a mobile processing system containing both filtration and selective ion removal/demineralization capabilities. A typical schematic for this system is provided on Figure 11.2-14. The system is located in the solid radwaste storage area on the 676’0” elevation of the radwaste building.

With either the chemical waste pump or a chemical waste neutralizer tank pump recirculating its respective tank contents, the waste is diverted to the mobile radwaste processing system at a flowrate of approximately 30 gpm. Effluent from the system is collected in the evaporator distillate sample tank. Bag filters may be used prior to and following the main system vessels. The main system vessels are capable of using either ion exchange media or activated carbon filtration media in order to obtain the appropriate treatment to satisfy effluent water quality requirements for discharge to the environment. The activated carbon provides mechanical filtration capability to remove the suspended solids activity while the ion exchange media removes soluble activities prior to discharge.

Process Vessels - The process vessels are fabricated of stainless steel with a design pressure of 150 psig at 200°F. The vessels are ASME code stamped and have a useful capacity of 30 ft^3. The vessels are provided with a top-mounted inlet, a top-mounted sluice line, outlet, fill, and vent connections. The inlet and outlet connections may be reduced to match the size of the process hoses. A pressure relief valve is provided on each vessel inlet line to prevent overpressure conditions when a vessel is isolated.

Bag Filter Vessel - The bag filter vessel is a standard, top-opening, stainless steel housing. The top has an O-ring seal and quick release eyebolts for rapid filter changeout. Drain and vent valves are provided.

Sluice Manifold - Charcoal/Resin sluice connections are designed to provide double isolation. Sluicing operations may be performed through a common sluice manifold or by individual sluice connections.

Process Hoses - Inlet, outlet, and sluice hoses are designed to safely operate at up to 250 psig at 250°F. They are reinforced with two or more steel braid. End connections are Camlock style quick disconnects with bolted clamps.

Instrumentation - The valve manifolds are provided with sampling vent connections, and pressure gauges. Pressure gauges are provided at the inlet, bagfilter, process vessel inlet and outlet headers, and outlet bag filter.

Charcoal/Resin Transfer/Dewatering Pump - A double diaphragm pump designed to pump a bead resin/liquid slurry from resin drums to refill the process vessels. This pump is also used to drain down the system manifold to allow hose changes without spills and may be used to dewater the filter and radwaste liners as necessary.

Radwaste Evaporators

Radwaste evaporators are physically connected and capable of being used for radwaste processing. However, with the elimination of chemical regeneration of condensate demineralizer resins, using this equipment for processing waste inputs to the chemical processing subsystem is considered to be impractical. The current plant operating practice is to utilize filtration and demineralization via the mobile processing system described above. The radwaste evaporators have been administratively removed from service. The following is a description of the as-installed evaporator equipment.

Two radwaste evaporators are piped in parallel for simultaneous operation and as backup to each other. Depending on the concentration in the shell, each radwaste evaporator can process 15 to 30 gpm of radioactive waste. Concentration is limited by precipitation of solids out of the solution and increased carry-over of iodine and other volatile activity into the distillate to approximately 25 w/o. The contents of one neutralizer tank or the chemical waste tank can be processed through one or both evaporators at the same time.

Each evaporator can separately process the contents of one tank provided the suction streams are not mixed in cross-over lines.

The radwaste evaporators are of the forced circulation design with bowed titanium tubes for chill-shock descaling. A manhole permits access to the shell for clean-out.

Heating steam is provided from the two auxiliary boilers in the turbine building, allowing both evaporators to operate during normal plant operation and reactor shutdowns.

An electric heater is provided in the evaporator shell to keep the concentrate in solution during steam interruptions and startups. The evaporators are designed for automatic unattended process operation until the desired bottom concentration, as determined by on-line indication or local grab sampling is obtained. Pump-out of the cooled concentrate as a batch, startup, and blowdown require attendance of an operator at the radwaste control room panel.

Influent to the evaporators is controlled by the level in the shell to keep the tubes submerged.

Through-put (distillate produced and feed rate) is manually set and automatically controlled by the flow of cooling water to the distillate condenser. The evaporators operate at 0-3 psig and are of fail safe design, recirculating the process streams internally when isolated.

The heating steam of the evaporators is collected and cooled in condensate return tanks for reuse in the auxiliary boilers.

A pump for recycling and returning of this condensate to the auxiliary boiler deaerator is provided with each tank. The discharge stream is monitored and, upon high conductivity that indicates an evaporator tube leak, it is diverted to the liquid radwaste collection tanks.

Service water (cooling tower water quality) is used to cool the evaporator distillate and the auxiliary steam condensate.

Instrumentation and controls of the evaporator assemblies are located in the radwaste control room and include: evaporator shell (concentrate) level indication with high and low alarms, concentrate temperature indication with low alarm, concentrate recirculation flow low alarm, shell pressure indication with high and low alarm, distillate conductivity indication with high alarm, distillate temperature indication with high and low alarms, evaporator condenser level indication with high and low alarms, condensate return tank level indication with high and low alarms, evaporator condenser cooling water inlet and outlet temperature indication, inlet flow and pH to each evaporator with a high and low pH alarm.

The evaporator shell is shielded by a concrete block wall to reduce operator exposure during maintenance and repair of the evaporator condenser, the concentrate or distillate pumps and instrumentation located in a local rack on the evaporator assembly skid.

11.2.2.4 Liquid Radwaste Laundry Drain Processing Subsystem

The laundry drain processing subsystem is located in the radwaste building. This equipment is no longer used for onsite processing of contaminated laundry wastewater. Plant laundry is shipped offsite for processing. The subsystem is used to treat the wastewater from regulated shop and cask cleaning drains as well as detergent-containing wastewater from various equipment washdown stations and personnel decontamination facilities throughout the plant. The bulk of the input to this subsystem originates from decontamination activities (floor decontamination waste). A schematic flow diagram and the P&IDs for this subsystem are provided in Figure 11.2-8 and Dwg. M-164, Sh. 1.

Influent to one of the two laundry drain tanks is selected from the radwaste control room. The two tanks are interconnected by a 4" overflow line below the overflow connection piped to the chemical radwaste sump. Each tank is associated with a pump for recirculation through an internal mixing eductor or processing of the contents through the laundry drain filters. The pumps are protected by coarse strainers in the suction lines. Both pumps and filters can be operated simultaneously. Cross-connections are provided to serve either or both filters by one pump. An internal mixing eductor in the laundry drain sample tank ensures a representative grab sample on the radwaste sample rack.

Effluent from the sample tank is discharged by one or both laundry drain sample tank pumps through the monitored discharge pipe into the cooling tower blowdown pipe. Filtrate with high conductivity can be transferred to the chemical waste tank. A return line allows recycling of sampled water back to the laundry drain tanks.

The controls and instrumentation of the liquid radwaste laundry processing subsystem are as described in Subsection 11.2.2.1, except that the laundry drain and laundry drain sample tanks are equipped with level recording instrumentation instead of indicating instruments to provide performance records of the laundry drain filters. High differential pressure through the strainers in the laundry drain tank pump suction lines is alarmed in the radwaste control room.

Laundry Drain Filters

Two banks of triplex filters are piped in parallel for simultaneous operation. Each filter bank consists of 3 individual filters arranged in parallel. The filter housings are capable of utilizing cartridge or bag filter elements, of various micron ratings, to provide flexibility to meet changing suspended solids concentration and/or particle size distribution. The purpose of these filters is to remove particulate contamination at a normal flow rate of 25 gpm per filter. The maximum flow rate per filter is 50 gpm. The filter elements are replaced when the pressure differential alarms in the radwaste control room trip at a maximum set point of 25 psid. Replacement of the filter elements is done manually because of the low expected radioactivity. Swing bolted housing closures and lift rings facilitate replacement of the cartridges. Depending on the activity level, the spent filter elements are disposed of in either the compacted solid waste or the dewatered radwaste described in Section 11.4.

11.2.3 RADIOACTIVE RELEASES

During liquid processing by the LWMS, radioactive contaminants are removed so that the bulk of the liquid can be either recycled in the plant or discharged to the environment. The radioactivity removed from the liquids is concentrated in filters and ion exchange media. These wastes are sent to the Solid Waste Management System for dewatering, packaging, and eventual shipment to a licensed burial ground. If the liquid is to be recycled back to the plant, it must meet the quality requirements for condensate makeup established by the Chemistry Control Program. If the liquid is to be discharged, the activity concentration must be consistent with the discharge criteria of 10CFR20.

Normally, most of the liquid passing through the liquid radwaste processing subsystem is recycled in the plant. However, the treatment in this subsystem is such that this liquid can be discharged from the plant, after monitoring, if required by plant water balance considerations. Normally most of the liquid passing through the chemical and laundry drain processing subsystems is discharged from the plant. Liquid processed through these subsystems may also be recycled back to the liquid radwaste subsystem for reprocessing and reuse in the plant.

The resulting doses from radioactive effluents will be within the guideline values of Appendix I to 10CFR50. In addition to the radioactivity limitations on releases, water quality standards for discharge may necessitate recycling of the water, rather than discharging.

Although the plant discharges vary as stated above, this analysis assumes the following:

a) Discharge of 2 percent of the liquid radwaste processing stream

b) Discharge of 100 percent of the chemical processing stream

c) Discharge of 100 percent of the laundry drain processing stream.

The assumptions and parameters used to calculate the yearly activity releases and their bases are given in Table 11.2-8. The yearly activity releases for each waste stream and the totals are given in Table 11.2-13.

Design and administrative controls are incorporated into the LWMS to prevent inadvertent releases to the environment. Controls include administrative procedures, operator training, redundant discharge valves, a discharge radiation monitor that alarms and initiates automatic discharge valve closure (see Section 11.5). Prior to any discharging, activity concentrations are measured in samples taken from the various sample tanks. The discharge header receives effluents from the discharge points in the LWMS shown on Figure 11.2-13. A single line is provided for radioactive plant discharges to minimize the potential for operator error.

The processed liquid radwaste that is not recycled in the plant is discharged into the cooling tower blowdown pipe on a batch basis. The flow rate is variable and controlled by a flow control valve. The discharges are mixed with the cooling tower blowdown (minimum dilution flow of 5000 gpm) to maintain the concentrations of radionuclides at the release point below the limits of 10CFR20. Expected average annual radionuclide concentrations in the discharge are compared to 10CFR20 limits in Table 11.2-14.

11.2.4 ESTIMATED DOSES

Dose calculations to assure compliance with Appendix I to 10CFR50, based on the liquid source term described above, were performed in accordance with USNRC Regulatory Guide 1.109 by use of the USNRC computer code "LADTAP." Doses were calculated to a maximum individual consuming aquatic biota, receiving shoreline exposure at the edge of the initial mixing zone, and drinking water from the nearest downstream supply (Danville). Input data for these calculations are given in Table 11.2-15.

The calculated maximum individual doses from liquid effluents are 0.34 mrem/yr/site to the total body of a child and 7.7 mrem/yr/site to the bone of a child. These doses are within the Appendix I design objectives of 6 and 20 mrem/yr/site to the total body and any organ, respectively and are a small fraction of the 10CFR20 dose limit for unrestricted access.

TABLE 11.2-1
EXPECTED DAILY INPUTS AND ACTIVITIES TO THE LIQUID WASTE MANAGEMENT SYSTEM FROM TWO UNITS

SOURCE	EXPECTED AVERAGE FROM TWO UNITS IN NORMAL OPERATION (gpd)(1)	PRIMARY COOLANT ACTIVITY FRACTION(2) PCA	MAXIMUM EXPECTED WITH ONE UNIT COLD STARTUP, ONE UNIT NORMAL OPERATION (gpd)	NOTES
LIQUID RADWASTE				
Drywell Equipment Drains	4400	1.0	4400	About 3 gpm for the station. Recirc pump seal leak is only expected source
Drywell Floor Drains	4400		4400	About 3 gpm for the station.
Unident. floor drains	(1000)	0.001	(1000)	
Drywell cooler drains	(1400)	0.001	(1400)	
Steam valve seal leaks	(800)	1.0	(800)	
Recirc valve seal leaks	(1200)	1.0	(1200)	
Reactor Building Drains	17,248		17,248	About 12 gpm for the station.
Unident. floor drains	(2600)	0.001	(2600)	
Scram valve seal leaks	(540)	0.1	(540)	
Scram valve intern. leaks	(3800)	0.1	(3800)	
Steam valve seal leaks	(800)	0.1	(800)	
Sample drains	(9,504)	0.1	(9,504)	3.3 gpm per unit – 11 sample locations at 1200 cc/min
RCIC & HPCI line drains	(4)	0.1	(4)	
Turbine Building Central Area Drains	10,020		10,020	About 7.0 gpm for the station.
Unident. floor drains	(8000)	0.001	(8000)	
Cond. pump seal leaks	(1920)	0.001	(1920)	
CRD pump seal leaks	(100)	0.001	(100)	

TABLE 11.2-1 (Continued)
EXPECTED DAILY INPUTS AND ACTIVITIES TO THE
LIQUID WASTE MANAGEMENT SYSTEM FROM TWO UNITS

SOURCE	EXPECTED AVERAGE FROM TWO UNITS IN NORMAL OPERATION (gpd)(1)	PRIMARY COOLANT ACTIVITY FRACTION(2) PCA	MAXIMUM EXPECTED WITH ONE UNIT COLD STARTUP, ONE UNIT NORMAL OPERATION (gpd)	NOTES
Turbine Building Outer Area Drains	2,000		2,000	About 2 gpm for the station not including condensate demineralizer operation.
Unident. floor drains	(2000)	0.001	(2000)	
Radwaste Building Drains	2080		2080	About 1.0 gpm for the station.
Unident. floor drains	(2000)	0.001	(2000)	
Off-gas system drains	(80)	0.1	(80)	
Reactor or Cavity Letdown to Radwaste	0		86,400	60 gpm during an outage or startup.
Suppression Pool Transfers to Radwaste	6,000	0.1	8,000	Normal 1-1/2 collection tank sets per week. Maximum 2 collection tank sets per week.
Inputs to Liquid Radwaste from Solid Radwaste				
RWCU Phase Separator Decant	340	0.002	340	
Waste Sludge Phase Separator Decant	3490	0.05	5,576	Maximum based on condensate filter operation

TABLE 11.2-1 (Continued)
EXPECTED DAILY INPUTS AND ACTIVITIES TO THE
LIQUID WASTE MANAGEMENT SYSTEM FROM TWO UNITS

SOURCE	EXPECTED AVERAGE FROM TWO UNITS IN NORMAL OPERATION (gpd)(1)	PRIMARY COOLANT ACTIVITY FRACTION(2) PCA	MAXIMUM EXPECTED WITH ONE UNIT COLD STARTUP, ONE UNIT NORMAL OPERATION (gpd)	NOTES
Spent Resin Tank Decant	80	0.05	80	
TOTAL	50,058		140,544	Maximum based on Reactor Cavity letdown of 60 gpm..
CHEMICAL RADWASTE				
Lab and chemical drains	1000	0.02	1000	
Aux. boiler blowdown	0		1786	Max. for two boilers blowdown of 1% over 24 hr
TOTAL	1,000		2,786	
LAUNDRY RADWASTE				
Decontamination Drains	200	0.02(3)	200	Primarily floor washdown/mop water
TOTAL	200		200	

(1) These inputs are averaged. The expected batch sizes and frequencies are shown in Table 11.2-2.
(2) See Tables 11.1-through 5 and 11.2-9 (Reactor Coolant).

TABLE 11.2-2

EXPECTED BATCHED INPUTS TO THE LIQUID RADWASTE SYSTEM FROM THE SOLID RADWASTE SYSTEM FOR NORMAL OPERATION OF TWO UNITS

Source	First Intermediate Collectors Input, Batch Size Each/Time	Second Intermediate Collectors Input, Batch Size Each/Time	Liquid Radwaste Collection Tank Input From Each Second Intermediate Collector, Batch Size Each/Time
Four RWCU F/D's	Two RWCU backwash receiving tanks, 1,470 gallon/9 days	Alternating at 365 days' interval for one of two RWCU phase separators, 2,940 gallon/9days	3,049 gallon/9 day
Three Fuel Pool F/D's	One fuel pool backwash receiving tank, 1,450 gallon/120 days	Alternating at 365 days' interval for one of two RWCU phase separators, 1,450 gallon/120 days	
Sixteen Cond Demineralizers	Two regen waste surge tanks, 0 gallon/day	One waste sludge phase separator, 0 gallon/day	24,428 gallon/7 day
Fourteen Cond Filters	Two cond filter backwash receiving tanks, 35,392 gallon/21 days	One waste sludge phase separator, 70,784 gallon/21 days	
Two Radwaste Filters		One waste sludge phase separator, 500 gallon/4.2 day	
Sixteen Cond Demineralizers		One spent resin tank, 3,800 gallon/77.5 days	5,383 gallon/69 days
One Radwaste Demineralizer		One spent resin tank, 2,000 gallon/69 days	
Averaged Total Input to Radwaste Collection Tanks From Solid Radwaste System			3907 gallon/day

TABLE 11.2-3

LIQUID WASTE MANAGEMENT SYSTEM COMPONENT DESCRIPTION

A. PUMPS	EQUIPMENT NOS.	TYPE	QUANTITY	MATERIAL	DESIGN CAPACITY, EACH, GPM	DESIGN TDH FT.	USAGE FACTOR NORMAL	DRIVER HP	DESIGN PRESSURE/TEMP. PSIG/°F
Liquid Radwaste Processing									
Collection Tank	0P-301A,B,C	Horiz. Centr.	3	SS	272	238	0.16	50	150/155
Surge Tank	0P-302	Horiz. Centr.	1	SS	272	238	0	50	150/155
Sample Tank	0P-305A,B,C	Horiz. Centr.	3	SS	280	170	0.16	30	150/155
Filter Precoat	0P-324A,B	Horiz. Centr.	2	SS	357	97	0.005	25	150/108
Filter Aid Proport.	0P-303,311	Reciprocating	2	SS/Hypalon	1.25	230	A 0.35 B 0.025	1	175/120
Liquid Radwaste Chemical Processing									
Neutralizing Tank	1P-130A,B	Horiz. Centr.	2	SS	50	175	0.012	10	150/155
Neutralizing Tank	2P-130A,B	Horiz. Centr.	2	SS	50	175	0.012	10	150/155
Chem. Waste Tank	0P-326A,B	Horiz. Centr.	2	SS	100	170	0.007	10	150/155
Conc. Storage Tank	0P-328	Horiz. Centr.	1	SS	20	180	0	10	150/155
Evap. Dist. Sample Tank	0P-327A,B	Horiz. Centr.	2	SS	50	180	0.021	10	150/155
Evap. Concentrate	0P-329A,B	Horiz. Centr.	2	SS	52	68	0	5	150/228
Evap. Distillate	OP-330A,B	Horiz. Centr.	2	SS	36	217	0	8	150/125
Evap. Condensate Return	0P-333A,B	Horiz. Centr.	2	SS	50	155	0	7.5	150/212
Liquid Radwaste Laundry Drain Processing									
Collection Tank	0P-318A,B	Horiz. Centr.	2	SS	25	220	0.006	15	150/155
Sample Tank	0P-319A,B	Horiz. Centr.	2	SS	10	120	0.012	5	150/155

TABLE 11.2-3

LIQUID WASTE MANAGEMENT SYSTEM COMPONENT DESCRIPTION

B. TANKS	EQUIPMENT NOS.	TYPE	QUANTITY	MATERIAL	LIVE/NOMINAL CAPACITY, EACH, GAL	DESIGN PRESSURE/TEMP. PSIG/°F
Liquid Radwaste Processing						
Collection	0T-302A thru F	Vert. Cyl.	6	SS	14000/15000	Atmos./200
Surge	0T-304A thru D	Vert. Cyl.	4	SS	14000/15000	Atmos./200
Sample	0T-303A thru F	Vert. Cyl.	6	SS	14050/15000	Atmos./200
Filter Precoat	0T-305	Vert. Cyl.	1	SS	860/1280	Atmos./200
Filter Aid	0T-310	Vert. Cyl.	1	SS	540/800	Atmos./200
Liquid Radwaste Chemical Processing						
Neutralizing	1T-130A,B	Horiz. Cyl.	2	SS	15850/16000	Atmos./200
Neutralizing	2T-130A,B	Horiz. Cyl.	2	SS	15850/16000	Atmos./200
Collection	0T-314	Vert. Cyl.	1	SS	12000/15000	Atmos./200
Evap. Dist. Sample	0T-321	Vert. Cyl.	1	SS	6270/7500	Atmos./200
Evap. Conc.	0T-322	Vert. Cyl.	1	SS	4140/5000	Atmos./200
Liquid Radwaste Laundry Drain Processing						
Collection	0T-311A,B	Vert. Cyl.	2	SS	770/1000	Atmos./200
Sample	0T-312	Vert. Cyl.	1	SS	1680/2000	Atmos./200

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TABLE 11.2-3

LIQUID WASTE MANAGEMENT SYSTEM COMPONENT DESCRIPTION

C. PROCESSING EQUIPMENT	EQUIPMENT NO.	TYPE	QUANTITY	SIZE EACH	MATERIAL	DESIGN CAPACITY EACH, GPM	USAGE FACTOR NORMAL	DESIGN PRESSURE/TEMP.
Liquid Radwaste Processing								
Liquid Radwaste Filter	0F-302A,B	Vert., Centr. Drycake Discharge	2	300 ft^2	SS	200	0.37	150/150
Liquid Radwaste Demin.	0F-301	Mixed Bed, Non-regen.	1	140 ft^3	SS	200	0.35	150/140
Liquid Radwaste Chemical Processing								
Mobile Processing System	0F323A, B, & C	Resin/Charcoal F/Ds	3	30 ft^3	SS	30 gpm	0.027	150/200
Radwaste Evaporator	0E-302A,B	Horiz., Bowed Tubes. Forced Circulation	2	1200 ft^2 1500 gal	Shell: SS Channel: CS Tubes & Sheets: Ti	30/15	0	Shell: 50/300 Tubes: 65/350
Evap. Absorption Tower	0E-304A,B	Wire Mesh Trays	2	5'diam	SS	30/15	0	50/300
Evap. Condenser	0E-303A,B	Horiz., U-tubes	2	620 ft^2	SS	Shell: 16,400 lb/hr Tubes: 620,000 lb/hr	0	Shell: 50/300 Tubes: 150/200
Evap. Heating Steam Cond. Return Tank	0T-333A,B	Horiz., U-tubes	2	40 ft^2	Shell; Tubes & Sheet: SS; Channel: CS	Shell: 21,000 lb/hr Tubes: 68,500 lb/hr	0	Shell: 65/350 Tubes: 150/200

TABLE 11.2-3

LIQUID WASTE MANAGEMENT SYSTEM COMPONENT DESCRIPTION

C. PROCESSING EQUIPMENT	EQUIPMENT NO.	TYPE	QUANTITY	SIZE EACH	MATERIAL	DESIGN CAPACITY EACH, GPM	USAGE FACTOR NORMAL	DESIGN PRESSURE/TEMP.
Liquid Radwaste Laundry Drain Processing								
Laundry Drain Filter	0F-313 A1,A2,A3 B1,B2 & B3	Triplex Vert., Cyl., Disposable Cartridge/Bag	2 Trains (3 Filters per Train)	Filter Element 7" Dia. x 28" Long	Shell: SS Cartridge/Bag: Disposable	50 maximum 25 Normal Operation	0.006	35/150 (Cartridge) 150/250 (Housing)

TABLE 11.2-4

LIQUID RADWASTE SYSTEM FLOWS (Refer to Figure 11.2-8)

Stream No.	Normal Operation of Both Units Average Number and Batch Frequency	Volume/ Batch (gal)	Nominal Flow Rate (gpm)	Normal Operation of Both Units Ave Volume/ day (gpd)	Maximum Number and Batch Frequency	Maximum Volume/ Day (gpd)	Comment
1. To Liquid Radwaste Filter & Demineralizer (From Coll. and Surge Tanks)	1/0.56 days	28,000	100	50,058	1/.14 days	140,544	Maximum includes cavity letdown, max suppression pool transfer and baseline leakage from both units
2. To Liquid Radwaste Sample Tanks (From Radwaste Demineralizer)	1/0.56 days	28,000	100	50,058	1/.14 days	140,544	Maximum includes cavity letdown, max suppression pool transfer and baseline leakage from both units
3. To Condensate Storage Storage Tanks (From Sample Tanks)	1/0.56 days	28,000	200	50,058	1/.14 days	140,544	Maximum includes cavity letdown, max suppression pool transfer and baseline leakage from both units
4. To Plant Discharge Pipe (From Sample Tanks)	1/17 days	28,000	100	1,640	1/1 days	28,000	Maximum during startup of one unit on aux. steam with additional discharge for inventory control.
5. To Mobile Processing System (From Chem Waste Neutralizer Tanks)	1/28 days	28,000	30	1,000	1/28 days	1,000	Maximum equal to normal operation since chemical input is not operation dependent without regeneration of condensate demineralizer resin.

Stream No.	Normal Operation of Both Units Average Number and Batch Frequency	Volume/ Batch (gal)	Nominal Flow Rate (gpm)	Normal Operation of Both Units Ave Volume/ day (gpd)	Maximum Number and Batch Frequency	Maximum Volume/ Day (gpd)	Comment
6. To Distillate Sample Tank (From Mobile Processing System)	1/5.7 days	5,700	30	1,000	1/5.7 days	1,000	Maximum equal to normal operation since chemical input is not operation dependent without regeneration of condensate demineralizer resin.
7. To Plant Discharge Pipe (From Distillate Sample Tank)	1/5.7 days	5,700	100	1,000	1/5.7 days	1,000	Same comment as line 6. All chemical waste is discharged.
8. To Laundry Drain Filters And Sample Tanks (From Laundry Drain Coll. Tanks)	1/4.1 days	820	25	200	1/4.1 days	200	Laundry input is not operation dependent since laundering is not done onsite, therefore maximum is equal to normal.
9. To Plant Discharge Pipe (From Laundry Drain Sample Tank)	1/7.1 days	1,420	52 (1 pump) 60 (2 pump)	200	1/7.1 days	200	Same comment as line 8. All laundry waste is discharged to the river.
10. To Cooling Tower Blowdown Line into River — LRW Chem Laund	1/17 days 1/5.7 days 1/7.1 days	28,000 5,700 1,420	100 100 60	1,640 1,000 <u>200</u> 2,840	1/1 days 1/5.7 days 1/7.1 days	28,000 1,000 <u>200</u> 29,200 144,000 (perm max)	Normal includes 2% discharge from LRW and 100% discharge from the laundry and chemical systems. The permissible maximum based on pump and piping capacities is continuous discharge at maximum rate of 100 gpm.

TABLE 11.2-5

EXPECTED RADIONUCLIDE ACTIVITY INVENTORIES OF LIQUID RADWASTE SYSTEM COMPONENTS
(Curies per Component)(2)(3)

	Liquid Radwaste Collection Tank	Liquid Radwaste Sample Tank	Chemical Waste Neutralizing Tank	Chemical Waste Tank	Evaporator Distillate Sample Tank	Mobile Processing System	Laundry Drain Collection Tank	Laundry Drain Sample Tank	Liquid Radwaste Filter	Liquid Radwaste Demineralizer	Laundry Drain Filter
TRITIUM											
H-3	4.52E+00	4.53E+00	-	-	-	-	-	-	-	-	-
CORROSION PRODUCTS											
Na-24	6.72E-02	6.39E-04	6.72E-04	6.67E-04	3.22E-06	6.60E-04	1.32E-04	2.82E-05	7.54E-01	8.29E-01	1.20E-04
P-32	1.74E-03	1.74E-05	1.61E-04	1.32E-04	6.87E-07	2.95E-04	1.02E-05	2.23E-06	1.81E-01	3.23E-01	9.80E-06
Cr-51	5.24E-02	5.25E-04	5.69E-03	4.51E-03	2.35E-05	1.69E-02	3.20E-04	7.00E-05	6.43E+00	1.41E+01	3.08E-04
Mn-54	6.14E-04	6.17E-06	7.92E-05	5.99E-05	3.13E-07	7.50E-04	3.87E-06	8.46E-07	8.93E-02	2.54E-01	3.75E-06
Mn-56	1.30E-01	9.69E-04	6.18E-04	6.13E-04	2.08E-06	6.06E-04	1.23E-04	2.33E-05	6.92E-01	7.61E-01	1.12E-05
Fe-55	8.80E-03	8.83E-05	1.14E-03	8.63E-04	4.50E-06	1.22E-02	5.56E-05	1.21E-05	1.29E+00	3.76E+00	5.37E-05
Fe-59	2.62E-04	2.63E-06	3.05E-05	2.38E-05	1.24E-07	1.29E-04	1.62E-06	3.54E-07	3.45E-02	8.35E-02	1.57E-06
Co-58	1.75E-03	1.76E-05	2.14E-04	1.64E-04	8.58E-07	1.22E-03	1.10E-05	2.40E-06	2.41E-01	6.24E-01	1.06E-05
Co-60	3.51E-03	3.53E-05	4.58E-04	3.47E-04	1.81E-06	5.05E-03	2.23E-05	4.87E-06	5.18E-01	1.51E+00	2.15E-05
Ni-63	-	-	-	-	-	-	-	-	-	-	-
Ni-65	7.63E-04	5.65E-06	3.61E-06	3.60E-06	1.21E-08	3.55E-06	7.22E-07	1.36E-07	4.05E-03	4.46E-03	6.55E-07
Cu-64	1.92E-01	1.81E-03	1.70E-03	1.70E-03	8.08E-06	1.68E-03	3.40E-04	7.19E-05	1.91E+00	2.10E+00	3.08E-04
Zn-65	1.76E-03	1.76E-05	2.26E-04	1.72E-04	8.93E-07	2.04E-03	1.11E-06	2.43E-06	2.54E-01	7.19E-01	1.07E-06
Zn-69m	1.31E-02	1.24E-04	1.23E-04	1.23E-04	5.88E-07	1.21E-04	2.44E-05	5.19E-06	1.38E-01	1.52E-01	2.22E-05
Zn-69	1.22E-02	1.24E-04	1.23E-04	1.23E-04	6.14E-07	1.21E-04	2.44E-05	5.30E-06	1.38E-01	1.52E-01	2.22E-05
Ag-110m	-	-	-	-	-	-	-	-	-	-	-
Ag-110	-	-	-	-	-	-	-	-	-	-	-
W-187	2.22E-03	2.15E-05	3.17E-05	3.16E-05	1.57E-07	3.12E-05	5.91E-06	1.27E-06	3.55E-02	3.90E-02	5.42E-06

TABLE 11.2-5

EXPECTED RADIONUCLIDE ACTIVITY INVENTORIES OF LIQUID RADWASTE SYSTEM COMPONENTS
(Curies per Component)(2)(3)

	Liquid Radwaste Collection Tank	Liquid Radwaste Sample Tank	Chemical Waste Neutralizing Tank	Chemical Waste Tank	Evaporator Distillate Sample Tank	Mobile Processing System	Laundry Drain Collection Tank	Laundry Drain Sample Tank	Liquid Radwaste Filter	Liquid Radwaste Demineralizer	Laundry Drain Filter
HALOGENS											
Br-83	8.72E-03	6.35E-05	4.11E-05	4.10E-05	1.35E-07	4.04E-05	8.24E-06	1.53E-05	-	5.07E-02	-
Br-84	-	-	-	-	-	-	-	-	-	-	-
Br-85	-	-	-	-	-	-	-	-	-	-	-
I-129	1.02E-14	1.22E-16	3.94E-14	2.32E-14	1.23E-16	2.26E-12	4.95E-16	1.09E-15	-	2.98E-10	-
I-131	1.53E-02	1.53E-04	2.26E-03	1.94E-03	1.01E-05	2.98E-03	1.71E-04	3.72E-04	-	1.82E+00	-
I-132	8.52E-02	6.14E-04	4.03E-04	4.02E-04	1.31E-06	3.97E-04	8.03E-05	1.49E-04	-	4.97E-01	-
I-133	1.79E-01	1.73E-03	2.29E-03	2.28E-03	1.13E-05	2.25E-03	4.37E-04	9.35E-04	-	2.83E+00	-
I-134	6.83E-02	3.12E-04	3.14E-04	3.14E-04	5.57E-07	3.10E-04	6.29E-05	9.16E-05	-	3.89E-01	-
I-135	1.31E-01	1.17E-03	7.98E-04	7.99E-04	3.48E-06	7.88E-04	1.60E-04	3.31E-04	-	9.87E-01	-
CESIUM AND RUBIDIUM											
Rb-89	-	-	-	-	-	-	-	-	-	-	-
Cs-134	2.63E-04	2.64E-05	3.42E-05	2.59E-05	1.35E-06	3.27E-04	1.67E-06	3.64E-06	-	1.02E-01	-
Cs-136	6.95E-04	6.95E-05	6.24E-05	5.18E-05	2.70E-06	9.89E-05	4.04E-06	8.84E-06	-	1.11E-01	-
Cs-137	1.75E-04	1.76E-05	2.29E-05	1.73E-05	9.05E-07	2.35E-04	1.11E-07	2.43E-06	-	6.89E-02	-
Cs-138	-	-	-	-	-	-	-	-	-	-	-
OTHER FISSION PRODUCT RADIOISOTOPES											
Sr-89	8.77E-04	8.79E-06	1.04E-04	8.04E-05	4.19E-07	4.78E-04	5.44E-06	1.19E-05	-	2.89E-01	-
Sr-90	6.15E-05	6.17E-07	8.04E-06	6.08E-06	3.18E-08	9.07E-05	3.90E-07	8.52E-07	-	2.66E-02	-
Sr-91	2.32E-02	2.14E-04	1.72E-04	1.71E-04	7.87E-07	1.69E-04	3.43E-05	7.19E-05	-	2.11E-01	-
Sr-92	2.72E-02	2.05E-04	1.30E-04	1.29E-04	4.46E-07	1.27E-04	2.59E-05	4.92E-05	-	1.60E-01	-
Y-89m	1.31E-07	1.32E-09	1.55E-08	1.20E-08	6.28E-11	7.17E-08	8.18E-10	1.78E-09	-	4.33E-05	-
Y-90	-	-	-	-	-	-	-	-	-	-	-
Y-91m	1.26E-02	1.26E-04	9.84E-05	9.81E-05	4.79E-07	9.68E-05	1.96E-05	4.25E-05	-	1.21E-01	-
Y-91	4.32E-04	4.46E-06	5.40E-05	7.02E-05	2.82E-07	3.62E-04	3.46E-06	7.63E-06	-	2.03E-01	-
Y-92	4.24E-02	3.86E-04	2.28E-04	2.29E-04	1.00E-06	2.26E-04	4.60E-05	9.54E-05	-	2.83E-01	-

TABLE 11.2-5

EXPECTED RADIONUCLIDE ACTIVITY INVENTORIES OF LIQUID RADWASTE SYSTEM COMPONENTS
(Curies per Component)(2)(3)

	Liquid Radwaste Collection Tank	Liquid Radwaste Sample Tank	Chemical Waste Neutralizing Tank	Chemical Waste Tank	Evaporator Distillate Sample Tank	Mobile Processing System	Laundry Drain Collection Tank	Laundry Drain Sample Tank	Liquid Radwaste Filter	Liquid Radwaste Demineralizer	Laundry Drain Filter
Y-93	2.38E-02	2.20E-04	1.82E-04	1.82E-04	8.44E-07	1.79E-04	3.64E-05	7.63E-05	-	2.25E-01	-
Zr-93	-	-	-	-	-	-	-	-	-	-	-
Zr-95	7.03E-05	7.05E-07	8.46E-06	6.54E-06	3.39E-08	4.54E-05	4.37E-07	9.54E-07	-	2.45E-02	-
Zr-97	-	-	-	-	-	-	-	-	-	-	-
Nb-95m	2.84E-08	3.34E-10	4.60E-08	3.15E-08	1.65E-10	3.51E-07	1.00E-09	2.21E-09	-	1.72E-04	-
Nb-95	7.05E-05	7.08E-07	9.12E-06	6.90E-06	3.60E-08	6.63E-05	4.45E-07	9.73E-07	-	2.91E-02	-
Nb-97m	-	-	-	-	-	-	-	-	-	-	-
Nb-97	-	-	-	-	-	-	-	-	-	-	-
Nb-98	3.54E-06	1.76E-11	1.63E-08	1.63E-08	2.84E-14	1.61E-08	3.26E-09	7.44E-12	-	2.01E-05	-
Mo-99	1.65E-02	1.64E-04	5.68E-04	5.49E-04	2.82E-06	5.70E-04	7.19E-05	1.56E-04	-	7.14E-01	-
Tc-99m	1.01E-01	9.09E-04	1.04E-03	1.02E-03	4.83E-06	1.04E-03	1.69E-04	3.55E-04	-	1.30E+00	-
Tc-99	3.11E-10	3.54E-12	1.36E-10	9.81E-11	5.17E-13	1.79E-09	4.66E-12	1.03E-11	-	5.00E-07	-
Tc-101	-	-	-	-	-	-	-	-	-	-	-
Tc-104	-	-	-	-	-	-	-	-	-	-	-
Ru-103	1.75E-04	1.76E-06	2.01E-05	1.57E-05	8.20E-08	7.79E-05	1.08E-06	2.36E-06	-	5.37E-02	-
Ru-105	7.87E-03	6.62E-05	4.14E-05	4.12E-05	1.606E-07	4.07E-05	8.29E-06	1.66E-05	-	5.11E-02	-
Ru-106	2.63E-05	2.64E-07	3.40E-06	2.58E-06	1.35E-08	3.31E-05	1.66E-07	3.64E-07	-	1.10E-02	-
Rh-103m	1.58E-04	1.67E-06	1.57E-05	2.00E-05	8.18E-08	7.79E-05	1.07E-07	2.34E-06	-	5.36E-02	-
Rh-105m	1.65E-03	1.39E-05	8.70E-06	8.67E-06	3.51E-08	8.55E-06	1.74E-06	3.50E-06	-	1.07E-02	-
Rh-105	1.24E-03	1.38E-05	4.13E-05	4.11E-05	2.13E-07	4.07E-05	6.72E-06	1.48E-05	-	5.11E-02	-
Rh-106	2.63E-05	2.64E-07	3.40E-06	2.58E-06	1.34E-08	3.31E-05	1.66E-07	3.64E-07	-	1.10E-02	-
Te-129m	3.49E-04	3.50E-06	3.92E-05	3.08E-05	1.60E-07	1.35E-04	2.14E-06	4.68E-06	-	1.01E-01	-
Te-129	1.92E-04	2.06E-06	2.45E-05	1.92E-05	1.01E-07	8.49E-05	1.32E-06	2.91E-06	-	6.37E-02	-
Te-131m	7.65E-04	7.48E-06	1.33E-05	1.32E-05	6.61E-08	1.30E-05	2.33E-06	5.04E-06	-	1.63E.02	-
Te-131	1.63E-04	1.66E-06	2.94E-06	2.92E-06	1.48E-08	2.89E-06	5.18E-07	1.13E-06	-	3.63E-03	-
Te-132	8.35E-05	8.29E-07	3.32E-06	3.16E-06	1.62E-08	3.38E-06	3.84E-07	8.33E-07	-	4.24E-03	-
Ba-137m	1.64E-04	1.64E-05	2.15E-05	1.63E-05	8.48E-07	2.21E-04	1.04E-06	2.24E-06	-	6.48E-02	-
Ba-139	1.49E-02	8.81E-05	6.86E-05	6.90E-05	1.69E-07	6.76E-05	1.37E-05	2.30E-05	-	8.49E-02	-
Ba-140	3.47E-03	3.47E.05	2.56E-04	3.09E-04	1.33E-06	5.28E-04	2.02E-05	4.41E-05	-	5.96E.01	-
Ba-141	-	-	-	-	-	-	-	-	-	-	-

TABLE 11.2-5

EXPECTED RADIONUCLIDE ACTIVITY INVENTORIES OF LIQUID RADWASTE SYSTEM COMPONENTS
(Curies per Component)(2)(3)

	Liquid Radwaste Collection Tank	Liquid Radwaste Sample Tank	Chemical Waste Neutralizing Tank	Chemical Waste Tank	Evaporator Distillate Sample Tank	Mobile Processing System	Laundry Drain Collection Tank	Laundry Drain Sample Tank	Liquid Radwaste Filter	Liquid Radwaste Demineralizer	Laundry Drain Filter
Ba-142	-	-	2.75E-04	2.15E-04	1.13E-06	5.28E-04-	1.03E-05	2.28E-05	-	-	-
La-140	3.74E-04	4.37E-06					-	-	-	5.86E-01	-
La-141	-	-	-	-	-	-	-	-	-	-	-
La-142	8.45E-03	5.31E-05	3.91E-05	3.90E-05	1.05E-07	3.85E-05	7.835E-06	1.35E-05	-	4.82E-02	-
Ce-141	2.62E-04	2.63E-06	2.92E-05	2.30E-05	1.20E-07	9.81E-05	1.60E-06	3.50E-06	-	7.50E-02	-
Ce-143	2.32E-04	2.28E-06	4.37E-06	4.35E-06	2.19E-08	4.30E-06	7.48E-07	1.62E-06	-	5.39E-03	-
Ce-144	2.63E-05	2.64E-07	3.38E-06	2.57E-06	1.34E-08	3.15E-05	1.66E-07	3.63E-07	-	1.09E-02	-
Pr-143	3.50E-04	3.51E-06	3.37E-05	2.77E-05	1.44E-07	6.01E-05	2.11E-06	4.62E-06	-	6.67E-02	-
Pr-144m	3.72E-07	3.77E-09	4.83E-08	3.67E-08	1.92E-10	4.51E-07	2.37E-09	5.20E-09	-	1.55E-04	-
Pr-144	2.55E-05	2.63E-07	3.38E-06	2.57E-06	1.34E-08	3.15E-05	1.66E-07	3.63E-07	-	1.09E-02	-
Nd-144	-	-	-	-	-	-	-	-	-	-	-
Nd-147	2.59E-05	2.60E-07	2.18E-06	1.83E-06	9.53E-09	3.40E-06	1.49E-07	3.25E-07	-	3.97E-03	-
Pm-147	5.28E-09	6.21E-11	1.45E-08	8.94E-09	4.72E-11	3.87E-07	2.15E-10	4.76E-10	-	8.47E-05	-
Sm-147	-	-	-	-	-	-	-	-	-	-	-
Np-239	5.72E-02	5.66E-04	1.72E-03	1.68E-03	8.58E-06	1.71E-03	2.35E-04	5.11E-05	1.93E+00	2.14E+00	2.21E-04
Pu-239	1.29E-09	1.52E-11	1.70E-09	1.19E-09	6.26E-12	2.41E-08	4.22E-11	9.29E-12	1.93E-06	6.61E-06	4.26E-11
OTHERS	2.22E+00	2.23E-02	2.90E-01	2.20E-01	1.15E-03	3.30E+00	1.41E-02	3.08E-02	3.28E+02	9.63E+02	

(1) Typical: 6.72E-02 = 6.72×10^{-2}
(2) Values are Curies per Component filled to its live capacity.
(3) Noble gases are not included in tank inventories because they are assumed to escape from solution and are continuously vented to the Radwaste Building ventilation system.

TABLE 11.2-6

DESIGN BASIS RADIONUCLIDE ACTIVITY INVENTORIES OF LIQUID RADWASTE SYSTEM COMPONENTS
(Curies per Component)(2)(3)

	Liquid Radwaste Collection Tank	Liquid Radwaste Sample Tank	Chemical Waste Neutralizing Tank	Chemical Waste Tank	Evaporator Distillate Sample Tank	Mobile Processing System	Laundry Drain Collection Tank	Laundry Drain Sample Tank	Liquid Radwaste Filter	Liquid Radwaste Demineralizer	Laundry Drain Filter
TRITIUM											
H-3	4.52E+00	4.53E+00	-	-	-	-	-	-	-	-	-
CORROSION PRODUCTS											
Na-24	1.70E-02	1.65E-04	1.37E-04	1.37E-04	3.56E-07	1.35E-04	2.70E-05	5.83E-05	2.70E-01	2.70E-01	2.70E-05
P-32	1.83E-04	1.83E-06	1.67E-05	1.38E-05	6.94E-08	3.07E-05	1.06E-06	2.32E-06	4.72E-02	6.15E-02	1.12E-06
Cr-51	4.58E-03	4.59E-05	4.95E-04	3.92E-04	2.01E-06	1.47E-03	2.77E-05	6.06E-05	1.57E+00	2.84E+00	2.94E-05
Mn-54	3.67E-04	3.68E-06	4.72E-05	3.58E-05	1.87E-07	4.48E-04	2.32E-06	5.07E-06	1.73E-01	6.36E-01	2.47E-06
Mn-56	3.05E-01	2.53E-03	5.86E-04	5.86E-04	3.27E-07	5.77E-04	1.17E-04	2.38E-04	1.16E+00	1.16E+00	1.17E-04
Fe-55	-	-	-	-	-	-	-	-	-	-	-
Fe-59	7.31E-04	7.34E-06	8.52E-05	6.63E-05	3.41E-07	3.61E-04	4.51E-06	9.88E-06	2.86E-01	6.50E-01	4.80E-06
Co-58	4.58E-02	4.60E-04	5.56E-03	4.29E-03	2.22E-05	3.17E-02	2.86E-04	6.24E-04	1.94E+01	5.31E+01	3.03E-04
Co-60	4.58E-03	4.60E-05	5.98E-04	4.53E-04	2.37E-06	6.59E-03	2.91E-05	6.35E-05	2.23E+00	8.88E+00	3.09E-05
Ni-63	-	-	-	-	-	-	-	-	-	-	-
Ni-65	1.81E-03	1.50E-05	3.44E-06	3.43E-06	1.87E-09	3.38E-06	6.90E-07	1.39E-06	6.79E-03	6.79E-03	6.87E-07
Cu-64	-	-	-	-	-	-	-	-	-	-	-
Zn-65	1.83E-05	1.84E-07	2.35E-06	1.78E-06	9.29E-09	2.13E-05	1.16E-07	2.53E-07	8.59E-03	3.07E-02	1.23E-07
Zn-69m	2.53E-04	2.45E-06	1.88E-06	1.87E-06	4.62E-09	1.85E-06	3.72E-07	8.01E-07	3.70E-03	3.70E-03	3.72E-07
Zn-69	1.62E-04	1.96E-06	1.88E-06	1.87E-06	4.93E-09	1.85E-06	3.72E-07	8.14E-07	3.70E-03	3.70E-03	3.72E-07
Ag-110m	5.51E-04	5.53E-06	7.02E-05	5.36E-05	2.80E-07	6.41E-04	3.46E-06	7.57E-06	2.58E-01	9.26E-01	3.69E-06
Ag-110	7.31E-06	7.34E-08	9.36E-07	7.13E-07	3.72E-09	8.52E-06	4.63E-08	1.01E-07	3.44E-03	1.23E-02	4.91E-08
W-187	2.62E-02	2.58E-04	3.25E-04	3.24E-04	1.06E-06	3.20E-04	6.05E-05	1.32E-04	6.42E-01	6.42E-01	6.12E-05

TABLE 11.2-6

DESIGN BASIS RADIONUCLIDE ACTIVITY INVENTORIES OF LIQUID RADWASTE SYSTEM COMPONENTS
(Curies per Component)[(2)(3)]

	Liquid Radwaste Collection Tank	Liquid Radwaste Sample Tank	Chemical Waste Neutralizing Tank	Chemical Waste Tank	Evaporator Distillate Sample Tank	Mobile Processing System	Laundry Drain Collection Tank	Laundry Drain Sample Tank	Liquid Radwaste Filter	Liquid Radwaste Demineralizer	Laundry Drain Filter
HALOGENS											
Br-83	8.90E-02	7.29E-04	1.63E-04	1.63E-04	8.41E-08	1.60E-04	3.26E-05	6.61E-05	-	3.22E-01	-
Br-84	5.72E-02	2.55E-04	6.54E-05	6.49E-05	7.47E-09	6.41E-05	1.30E-05	2.02E-05	-	1.29E-01	-
Br-85	3.28E-03	1.55E-06	3.70E-06	3.68E-06	3.82E-11	3.64E-06	7.39E-07	1.98E-07	-	7.29E-03	-
I-129	1.93E-15	3.01E-17	4.12E-14	2.42E-14	1.41E-16	2.36E-12	5.15E-16	1.13E-15	-	2.70E-09	-
I-131	2.38E-01	2.38E-03	1.70E-02	1.47E-02	7.20E-05	2.24E-02	1.29E-03	2.82E-03	-	4.50E+01	-
I-132	8.64E-01	7.87E-03	1.81E-02	1.73E-02	7.47E-05	1.84E-02	2.15E-03	4.67E-03	-	3.69E+01	-
I-133	7.74E-01	7.55E-03	8.40E-03	2.33E-03	2.61E-05	8.28E-03	1.61E-03	3.48E-03	-	1.66E-01	-
I-134	7.84E-01	4.66E-03	9.54E-04	9.53E-04	1.81E-07	9.40E-04	1.91E-04	3.38E-04	-	1.89E+00	-
I-135	1.01E+00	9.42E-03	3.91E-03	6.45E-04	5.45E-06	3.85E-03	7.83E-04	1.66E-03	-	7.73E+00	-
CESIUM AND RUBIDIUM											
Rb-89	-	-	-	-	-	-	-	-	-	-	-
Cs-134	1.47E-03	1.47E-04	1.91E-04	1.44E-04	7.54E-06	1.82E-03	9.28E-06	2.03E-05	-	2.75E+00	-
Cs-136	1.00E-03	1.01E-04	8.94E-05	7.40E-05	3.72E-06	1.41E-04	5.79E-06	1.27E-05	-	3.11E-01	-
Cs-137	2.20E-03	2.21E-04	2.88E-04	2.18E-04	1.14E-05	2.95E-03	1.40E-05	3.05E-05	-	4.34E+00	-
Cs-138	4.05E-01	1.83E-02	4.64E-04	4.63E-04	5.38E-07	4.15E-04	9.28E-05	1.44E-04	-	9.16E-01	-
OTHER FISSION PRODUCTS											
Sr-89	2.84E-02	2.85E-04	3.34E-03	2.60E-03	1.34E-05	1.54E-02	1.76E-04	3.84E-04	-	2.73E+01	-
Sr-90	2.11E-03	2.12E-05	2.76E-04	2.09E-04	1.09E-06	3.11E-03	1.34E-05	2.93E-05	-	4.16E+00	-
Sr-91	5.62E-01	5.32E-03	2.98E-03	2.97E-03	5.62E-05	2.93E-03	5.97E-04	1.27E-03	-	5.86E+00	-
Sr-92	6.84E-01	5.69E-03	1.36E-03	1.35E-03	7.94E-07	1.33E-03	2.71E-04	5.52E-04	-	2.67E+00	-
Y-89m	4.25E-06	4.27E-08	5.02E-07	3.89E-07	2.01E-09	2.32E-06	2.64E-08	5.76E-08	-	4.10E-03	-
Y-90	-	-	-	-	-	-	-	-	-	-	-
Y-91m	2.17E-01	2.57E-03	1.71E-03	1.70E-03	3.51E-06	1.68E-03	3.40E-04	7.44E-04	-	3.36E+00	-
Y-91	3.43E-04	5.17E-06	4.92E-04	3.76E-04	2.01E-06	2.59E-03	2.24E-05	4.92E-05	-	4.46E+00	-
Y-92	2.00E-01	2.54E-03	1.36E-03	1.35E-03	1.82E-06	1.33E-03	2.71E-04	5.91E-04	-	2.67E+00	-
Y-93	-	-	-	-	-	-	-	-	-	-	-

TABLE 11.2-6

DESIGN BASIS RADIONUCLIDE ACTIVITY INVENTORIES OF LIQUID RADWASTE SYSTEM COMPONENTS
(Curies per Component)(2)(3)

	Liquid Radwaste Collection Tank	Liquid Radwaste Sample Tank	Chemical Waste Neutralizing Tank	Chemical Waste Tank	Evaporator Distillate Sample Tank	Mobile Processing System	Laundry Drain Collection Tank	Laundry Drain Sample Tank	Liquid Radwaste Filter	Liquid Radwaste Demineralizer	Laundry Drain Filter
Zr-93	-	-	-	-	-	-	-	-	-	-	-
Zr-95	3.67E-04	3.68E-06	4.41E-05	3.41E-05	1.76E-07	2.38E-04	2.28E-06	4.99E-06	-	4.04E-01	-
Zr-97	2.75E-04	2.68E-06	2.46E-06	2.45E-06	6.47E-08	2.42E-06	4.80E-07	1.04E-06	-	4.86E-03	-
Nb-95m	3.72E-08	5.37E-10	2.41E-07	1.64E-07	9.24E-10	1.84E-06	5.24E-09	1.15E-08	-	3.08E-03	-
Nb-95	3.85E-04	3.87E-06	4.97E-05	3.77E-05	6.83E-07	3.54E-04	2.44E-06	5.33E-06	-	5.74E-01	-
Nb-97m	2.59E-04	2.54E-06	2.33E-06	2.32E-06	7.30E-09	2.29E-06	4.57E-07	9.86E-07	-	4.60E-03	-
Nb-97	1.52E-04	1.91E-06	2.46E-06	2.45E-06	2.75E-08	2.42E-06	4.80E-07	1.05E-06	-	4.86E-03	-
Nb-98	-	-	-	-	-	-	-	-	-	-	-
Mo-99	1.99E-01	1.98E-03	6.48E-03	6.27E-03	3.51E-05	6.49E-03	8.21E-04	1.79E-03	-	1.30E+01	-
Tc-99m	2.17E+00	2.02E-02	1.33E-02	1.30E-02	6.87E-05	1.31E-02	2.20E-03	4.70E-03	-	2.64E+01	-
Tc-99	1.41E-09	1.98E-11	1.74E-09	1.26E-09	0.00E+00	2.26E-08	6.17E-11	1.35E-10	-	2.99E-05	-
Tc-101	1.34E-01	3.08E-04	1.51E-04	1.50E-04	7.70E-09	1.48E-04	3.03E-05	3.23E-05	-	2.97E-01	-
Tc-104	-	-	-	-	-	-	-	-	-	-	-
Ru-103	1.74E-04	1.74E-06	1.99E-05	1.55E-05	8.01E-08	7.71E-05	1.07E-06	2.33E-06	-	1.42E-01	-
Ru-105	-	-	-	-	-	-	-	-	-	-	-
Ru-106	2.39E-05	2.39E-07	3.07E-06	2.33E-06	1.22E-08	3.00E-05	1.51E-07	3.29E-07	-	4.22E-02	-
Rh-103m	1.09E-04	1.35E-06	1.99E-05	1.55E-05	8.01E-08	7.71E-05	1.05E-06	2.31E-06	-	1.42E-02	-
Rh-105m	-	-	-	-	-	-	-	-	-	-	-
Rh-105	-	-	-	-	-	-	-	-	-	-	-
Rh-106	2.37E-05	2.39E-07	3.07E-06	2.33E-06	1.22E-08	3.00E-05	1.51E-07	3.29E-07	-	4.22E-02	-
Te-129m	3.66E-04	3.68E-06	4.09E-05	3.21E-05	1.65E-07	1.41E-04	2.24E-06	4.89E-06	-	2.66E-01	-
Te-129	1.29E-04	1.64E-06	2.57E-05	2.01E-05	1.04E-07	8.87E-05	1.38E-06	3.03E-06	-	1.67E-01	-
Te-131m	-	-	-	-	-	-	-	-	-	-	-
Te-131	-	-	-	-	-	-	-	-	-	-	-
Te-132	4.43E-01	4.42E-03	1.69E-02	1.60E-02	7.20E-05	1.72E-02	1.95E-03	4.25E-03	-	3.45E+01	-
Ba-137m	2.01E-03	2.04E-04	2.71E-04	2.05E-04	1.07E-05	2.78E-03	1.31E-05	2.82E-05	-	4.08E+00	-
Ba-139	7.21E-01	5.13E-03	1.01E-03	1.00E-03	3.01E-07	9.92E-04	2.02E-04	3.85E-04	-	1.99E+00	-
Ba-140	8.22E-02	8.25E-04	7.26E-03	5.99E-03	3.01E-05	1.24E-02	4.74E-04	1.03E-03	-	2.48E+01	-
Ba-141	2.09E-01	6.06E-04	2.36E-04	2.35E-04	1.55E-08	0.00E+00	4.71E-05	5.82E-05	-	-	-
Ba-142	1.22E-01	2.15E-04	1.38E-04	1.38E-04	5.31E-09	1.36E-04	2.76E-05	2.45E-05	-	2.72E-01	-
La-140	2.29E-03	3.30E-05	6.48E-03	5.04E-03	2.73E-05	1.24E-02	2.42E-04	5.32E-04	-	2.48E+01	-
La-141	8.16E-02	8.30E-04	2.36E-04	8.35E-05	2.16E-07	2.32E-04	4.71E-05	1.02E-04	-	-	-

Table Rev. 55

TABLE 11.2-6

DESIGN BASIS RADIONUCLIDE ACTIVITY INVENTORIES OF LIQUID RADWASTE SYSTEM COMPONENTS
(Curies per Component)(2)(3)

	Liquid Radwaste Collection Tank	Liquid Radwaste Sample Tank	Chemical Waste Neutralizing Tank	Chemical Waste Tank	Evaporator Distillate Sample Tank	Mobile Processing System	Laundry Drain Collection Tank	Laundry Drain Sample Tank	Liquid Radwaste Filter	Liquid Radwaste Demineralizer	Laundry Drain Filter
La-142	8.90E-02	7.50E-04	1.38E-04	1.38E-04	5.29E-08	1.36E-04	2.76E-05	5.72E-05	-	2.72E-01	-
Ce-141	4.72E-04	5.28E-06	1.06E-04	8.35E-05	4.34E-07	3.61E-04	5.65E-06	1.24E-05	-	6.83E-01	-
Ce-143	3.11E-04	3.07E-06	5.26E-06	5.22E-06	1.94E-08	5.17E-06	9.02E-07	1.96E-06	-	1.04E-02	-
Ce-144	3.21E-04	3.22E-06	4.12E-05	3.13E-05	1.63E-07	3.85E-04	2.03E-06	4.43E-06	-	5.50E-01	-
Pr-143	3.48E-04	3.50E-06	3.39E-05	2.79E-05	1.41E-07	6.05E-05	2.12E-06	4.62E-06	-	1.21E-01	-
Pr-144m	4.35E-06	4.58E-08	5.89E-07	4.48E-07	2.33E-09	5.50E-06	2.90E-08	6.33E-08	-	7.87E-03	-
Pr-144	2.80E-04	3.11E-06	4.12E-05	3.13E-05	1.63E-07	3.85E-04	2.02E-06	4.43E-06	-	5.50E-01	-
Nd-144	-	-	-	-	-	-	-	-	-	-	-
Nd-147	1.28E-04	1.28E-06	1.06E-05	8.90E-06	4.43E-08	1.65E-05	7.25E-07	1.58E-06	-	3.31E-02	-
Pm-147	6.31E-09	9.15E-11	7.08E-08	4.35E-08	2.51E-10	1.88E-06	1.05E-09	2.30E-09	-	2.44E-03	-
Sm-147	-	-	-	-	-	-	-	-	-	-	-
Np-239	2.16E+00	2.14E-02	6.12E-02	5.95E-02	2.54E-04	6.07E-02	8.35E-03	1.82E-02	-	1.22E+02	8.63E-03
Pu-239	1.17E-08	1.69E-10	6.06E-08	4.23E-08	2.36E-10	8.57E-07	1.50E-09	3.29E-09	-	1.13E-03	1.66E-09
OTHERS	-	-	-	-	-	-	-	-	-	-	-

(1) Typical: 1.70E-02 means 1.70×10^{-2}
(2) Values are Curies per Component filled to its live capacity.
(3) Noble gases are not included in tank inventory because they are assumed to escape from solution and are continuously vented to the Radwaste Building ventilation system.

Table 11.2-7
This Table Has Been Deleted

TABLE 11.2-8

ASSUMPTIONS AND PARAMETERS USED FOR EVALUATION OF RADIOACTIVE RELEASES

ITEM	VALUE OF REFERENCE	SOURCE
1. GENERAL		
a) Maximum core thermal power (Mwt) evaluated for safety considerations.	4032	FSAR 15
b) Total quantity of tritium released from one unit (Ci/yr)	59	SSES Data
Liquid Effluents		
Gaseous Effluents		
2. NUCLEAR STEAM SUPPLY SYSTEM	92	
a) Total steam flow (lb/hr) for 100% power	1.65+7(1)	FSAR 10.1
b) Mass of reactor coolant (lb) in vessel at full power.	3.8+5	NUREG16
3. REACTOR WATER CLEANUP SYSTEM		
a) Average flow rate (lb/hr)	1.46+5	FSAR 5.4.8
b) Powdex demineralizer size (lb of dry resin incl. 10 w/o crud)	30	FSAR 5.4.8
c) Replacement frequency (days)	10.5	FSAR 11.2
d) Backwash volume (gal/event)	1470	FSAR 11.2
4. CONDENSATE DEMINERALIZERS		
a) Average flow rate (lb/hr) total for 8 vessels (VWO)	1.65+7	FSAR 10.1
b) Deep bed demineralizer size (ft^3 of resin per vessel)	276	FSAR 10.4.6
c) Number of demineralizers	8 per unit	FSAR 10.4.6
d) Resin discharge frequency	1/3.8 years	FSAR 11.2

TABLE 11.2-8

ASSUMPTIONS AND PARAMETERS USED FOR EVALUATION OF RADIOACTIVE RELEASES

ITEM	VALUE OF REFERENCE	SOURCE
5. CONDENSATE FILTERS		
a) Filter Vessel	7 filter vessels per unit	FSAR 10.4.7
b) Normal Filter Vessel Flow Rate	4,267 gpm per vessel (7 vessels in service)	FSAR 10.4.7
c) Backwash frequency (days)	7	FSAR 11.4.2
d) Backwash volume (gallons)	5,056	FSAR 11.4.2
6. LIQUID WASTE PROCESSING SYSTEM		
a) 1) Sources, flow rates, and expected activities in flow streams 11.2-13	Tables 11.2-4; 11.2-7; 11.2-10;	FSAR 11.2
2) Holdup times for collection, processing, and discharge	Table 11.2-11	FSAR 11.2
3) Capacities of tanks and processing equipment	Table 11.2-3	FSAR 11.2
4) Decontamination factors	Table 11.2-12	FSAR 11.2
5) Fraction from each stress discharged		
Liquid Radwaste Processing System	0.02	FSAR 11.2
Liquid Radwaste Chemical Processing System	1.0	FSAR 11.2
Liquid Radwaste Laundry Drain Proc. Sys.	1.0	FSAR 11.2
6) Radwaste demineralizer regeneration frequency (days)	None(2)	FSAR 11.2
Radwaste demineralizer regeneration volume (gal/event)	None(2)	FSAR 11.2
7) Liquid source terms for normal operation (Ci/yr)	Table 11.2-14	FSAR 11.2
b) P&IDS and process flow drawings for liquid radwaste system.	Figures 11.2-8 through 11.2-13	FSAR 11.2

TABLE 11.2-8

ASSUMPTIONS AND PARAMETERS USED FOR EVALUATION OF RADIOACTIVE RELEASES

ITEM	VALUE OF REFERENCE	SOURCE
7. MAIN CONDENSER AND TURBINE GLAND SEAL AIR REMOVAL SYSTEMS		
a) Holdup time for offgas prior to offgas treatment system (hr)	0.12	FSAR 11.3
b) Description of offgas treatment system	FSAR 11.3	FSAR 11.3
c) Offgas treatment system		
1) Mass of charcoal (lb)	148,000	FSAR 11.3
2) Operating/dew point (°F)	60-65/40	FSAR 11.3
3) Dynamic adsorption coeff. Xe,Kr(cm^3/g)	516/36.0	FSAR 11.3
d) Gland seal steam flow (lb/hr) and source	25,000 (normal) steam from condensate (clean steam)	FSAR 10.4.3
e) Radioactive iodine reduction systems for the gland seal system	N/A	N/A
f) P&IDs and process flow drawings for gaseous waste systems	Figures 11.3-1 through 11.3-3	FSAR 11.3
8. VENTILATION AND EXHAUST SYSTEMS		
a) Provisions to reduce releases in individual buildings	Table 11.3-4	FSAR 9.4, 11.3
b) Decontamination factors in individual buildings	Table 11.3-4	FSAR 11.3
c) Release rates Ci/yr	Table 11.3-1	FSAR 11.3
d) Release points – heights, temperatures, size, and shape of orifices	Figures 11.3-4	FSAR 11.3
e) Containment purge frequency (per year)	4	FSAR 11.3

TABLE 11.2-8

ASSUMPTIONS AND PARAMETERS USED FOR EVALUATION OF RADIOACTIVE RELEASES

ITEM	VALUE OF REFERENCE	SOURCE
9. EXPECTED RADIONUCLIDE ACTIVITY CONCENTRATIONS IN REACTOR COOLANT AND MAIN STEAM USED FOR EVALUATION OF RADIOACTIVE RELEASES	Table 11.2-9	NUREG 16, FSAR 11.2.3

(1) Typical: 1.45+7 means 1.45×10^{7}

(2) Regeneration of condensate demineralizer resin is not performed. Resin is discharged to the radwaste solidification system rather than regenerated. Without regeneration, the radwaste evaporators are not run. Soluble activity removed by the resin is therefore discharged through the solid waste management system. A fraction of the suspended solids from condensate filtration backwash receiving tank is carried over to the LRW collection tanks in the decant from the water sludge phase separator. These suspended solids are removed by the LRW filters and discharged during filter backwash to the waste mixing tanks and waste sludge phase separator.

(3) Spent resins from the radwaste demineralizer are sluiced to the solid waste management system.

TABLE 11.2-9

EXPECTED RADIONUCLIDE ACTIVITY CONCENTRATIONS IN REACTOR COOLANT AND MAIN STEAM USED FOR EVALUATION OF RADIOACTIVE RELEASES[(1)]

ISOTOPE	REACTOR WATER (μCi/gm)	REACTOR STEAM (μCi/gm)
Noble Gases		
Kr-83m	-	9.1-3[(2)]
Kr-85m	-	1.6-3
Kr-85	-	5.0-6
Kr-87	-	5.5-3
Kr-88	-	5.5-3
Kr-89	-	3.4-2
Xe-131m	-	3.9-6
Xe-133m	-	7.5-5
Xe-133	-	2.1-3
Xe-135m	-	7.0-3
Xe-135	-	6.0-3
Xe-137	-	3.9-2
Xe-138	-	2.3-2
Halogens		
Br-83	5.11-3	7.66-5[(3)]
I-131	2.31-3	3.47-5
I-132	5.16-2	7.74-4
I-133	3.27-2	4.91-4
I-134	1.07-1	1.60-3
I-135	3.59-2	5.39-4
Corrosion and Noncoolant Activation Products		
Na-24	1.33-2	1.33-5
P-32	2.59-4	2.59-7
Cr-51	7.77-3	7.77-6
Mn-54	9.05-5	9.05-8
Mn-56	7.11-2	7.11-5
Fe-55	1.29-3	1.29-6
Fe-59	3.88-5	3.88-8
Co-58	2.59-4	2.59-7
Co-60	5.18-4	5.18-7
Ni-65	4.26-4	4.26-7
Cu-64	4.00-2	4.00-5
Zn-65	2.59-4	2.59-7
Zn-69m	2.66-3	2.66-6
W-187	3.95-4	3.95-7
Np-239	9.12-3	9.12-6

TABLE 11.2-9

EXPECTED RADIONUCLIDE ACTIVITY CONCENTRATIONS IN REACTOR COOLANT AND MAIN STEAM USED FOR EVALUATION OF RADIOACTIVE RELEASES[(1)]

ISOTOPE	REACTOR WATER (μCi/gm)	REACTOR STEAM (μCi/gm)
Fission Products		
Sr-89	1.29-4	1.29-7
Sr-90	9.05-6	7.99-9
Sr-91	5.36-3	5.36-6
Y-91	5.18-5	5.18-8
Sr-92	1.42-2	1.42-5
Y-92	8.39-3	8.39-6
Y-93	5.36-3	5.36-6
Zr-95	1.04-5	1.04-8
Nb-95	1.04-5	1.04-8
Nb-98	6.01-3	6.01-6
Mo-99	2.61-3	2.61-6
Tc-99m	2.73-2	2.73-5
Ru-103	2.59-5	2.59-8
Ru105	2.77-3	2.77-6
Ru106	3.88-6	3.88-9
Te129m	5.18-5	5.18-8
Te-131m	1.31-4	1.31-7
Te-132	1.30-5	1.30-8
Cs-134	3.88-5	3.88-8
Cs-136	1.04-4	1.04-7
Cs-137	2.58-5	2.58-8
Ba-139	1.47-2	1.47-5
Ba-140	5.19-4	5.19-7
Ce-141	3.88-5	3.88-8
La-142	7.03-3	7.03-6
Ce-143	3.93-5	3.93-8
Pr-143	5.18-5	5.18-8
Ce-144	3.88-6	3.88-9
Nd-147	3.89-6	3.89-9
Coolant Activation Products[(4)]		
N-13	4.00-2	3.50-2
N-16	5.00+1	2.50+2
N-17	6.00-3	1.00-1
O-19	7.00-1	8.00-1
F-18	4.00-3	4.00-3
Tritium[(5)]		
H-3	1.0-2	1.0-2`

(1) The values in this table are calculated based on the GALE code in NUREG-0016 Rev. 1 (November, I979) except as noted.

(2) Typical: 9.1-3 = 9.1×10^{-3}

(3) The halogen concentrations listed in reactor steam are based on a carryover of 0.015, which was used to calculate the maximum expected liquid activity release using the NUREG-0016. For adjustment in halogen steam concentrations due to HWC implementation refer to Table 11.1-2.

(4) Coolant activation products are as presented in Table 11.1-4 for operation with hydrogen water chemistry.

(5) The tritium concentration in the reactor coolant and steam are controlled by the loss of water from the reactor coolant system by evaporation or leakage. The estimated total appearance rate in effluents is estimated per NUREG-0016 at 0.03 Ci/MWt (or ~120 Ci/yr). Reactor coolant concentrations are shown based on NUREG-0016 measured values increased to account for liquid recycle. The reactor coolant tritium concentrations shown are consistent with SSES measured values. However, historically the production rate of tritium in the reactor coolant and the release rate in effluents at SSES have been higher (~ 150 curies) than the NUREG-0016 estimates. Therefore estimates of radioactive tritium releases in liquid and gaseous effluent have been conservative based upon measured data.

TABLE 11.2-10

RADWASTE SYSTEM FLOW RATES AND STREAM ACTIVITIES
USED FOR EVALUATION OF RADIOACTIVE RELEASES

Source	Expected Daily Input Flow Rate per Unit (gal/day)	Fraction of Primary Coolant Activity (PCA)
Liquid Radwaste Processing System		
Drywell Drains	2200	0.73
Reactor Building Drains	8624	0.085
Turbine Building Drains	6010	0.001
Turbine Building - Condensate Demineralizer Operation	0	0
Radwaste Building Drains	1040	0.005
Suppression Pool Transfers	3000	0.10
Solid Radwaste Processing Input to Liquid Radwaste Processing System		
RWCU Phase Separator	170	0.002
Waste Sludge Phase Separator	1745	0.05
Spent Resin Tank	40	0.05
TOTAL	25029	0.173
Liquid Radwaste Chemical Waste Processing System		
Lab and chemical drains	500	0.02
Laundry Radwaste Laumdry Drain Processing System		
Decontamination Drains	100	0.02

TABLE 11.2-11

EXPECTED HOLDUP TIMES FOR COLLECTION, PROCESSING, AND DISCHARGE USED FOR EVALUATION OF RADIOACTIVE RELEASES

PROCESS	HOLDUP TIME (DAYS)
1. Liquid Radwaste Processing System	
a) Collection	0.64
b) Processing	0.19
c) Discharge	0.27
d) Total	1.10
2. Liquid Radwaste Chemical Processing System Neutralizing Tanks	
a) Collection	28.00
b) Processing	0.65
c) Discharge	0.61
d) Total	29.30
3. Liquid Radwaste Laundry Drain Processing System	
a) Collection	8.20
b) Processing	0.04
c) Discharge	0.10
d) Total	8.34

TABLE 11.2-12

DECONTAMINATION FACTORS USED FOR EVALUATION OF RADIOACTIVE RELEASES

1. Liquid Radwaste Processing System

	Filter	Demineralizer	Total DF
Halogens	1	100	100
Cs, Rb	1	10	10
Other Nuclides	1	100	100

2. Liquid Radwaste Chemical Processing System

	Mobile Processing System Filter/Demineralizer	Total DF
Halogens	100	100
Cs, Rb	10	10
Other Nuclides	100	100

3. Liquid Radwaste Laundry Drain Processing System

	Filter	Total DF
Halogens	1	1
Cs, Rb	1	1
Other Nuclides	1	1

TABLE 11.2-13

EXPECTED YEARLY ACTIVITY RELEASED FROM LIQUID WASTE MANAGEMENT SYSTEMS USED FOR EVALUATION OF COMPLIANCE WITH APPENDIX I OF 10CFR50 (Ci/yr/site)(1)

NUCLIDE	LIQUID RADWASTE PROCESSING (Curies/Site)	LAUNDRY RADWASTE (Curies/Site)	LRW CHEMICAL PROCESSING (Curies/Site)	TOTAL LWR (Curies/Site)	ADJUSTED TOTAL (Ci/yr/Site)(2)	DETERGENT WASTES (Ci/yr/Site)	TOTAL (Ci/yr/Site)
CORROSION AND ACTIVATION PRODUCTS							
Na-24	1.11E-02(3)	5.22E-03	4.00E-05	1.64E-02	2.72E-02	0.00E+00	2.80E-02
P-32	4.40E-04	8.60E-04	2.00E-05	1.34E-03	2.24E-03	0.00E+00	2.20E-03
Cr-51	1.35E-02	2.87E-02	1.12E-04	4.33E-02	7.20E-02	0.00E+00	7.20E-02
Mn-54	1.60E-04	3.60E-04	2.00E-05	5.40E-04	9.00E-04	0.00E+00	9.00E-04
Mn-56	2.68E-03	2.54E-03	0.00E+00	5.20E-03	8.66E-03	0.00E+00	8.60E-03
Fe-55	2.30E-03	5.28E-03	2.60E-04	7.82E-03	1.30E-02	0.00E+00	1.30E-02
Fe-59	6.00E-05	1.40E-04	0.00E+00	2.20E-04	3.80E-04	0.00E+00	3.80E-04
Co-58	4.60E-04	1.02E-03	4.00E-05	1.52E-03	2.52E-03	0.00E+00	2.60E-03
Co-60	9.20E-04	2.12E-03	1.00E-04	3.14E-03	5.22E-03	0.00E+00	5.20E-03
Ni-65	2.00E-05	2.00E-05	0.00E+00	4.00E-05	6.00E-05	0.00E+00	4.00E-05
Cu-64	2.95E-02	1.31E-02	8.00E-05	4.27E-02	7.11E-02	0.00E+00	7.20E-02
Zn-65	4.60E-04	1.04E-03	6.00E-05	1.56E-03	2.58E-04	0.00E+00	2.60E-03
Zn-69m	2.08E-03	9.60E-04	0.00E+00	3.04E-03	5.06E-03	0.00E+00	5.00E-03
Zn-69	2.24E-03	1.02E-03	0.00E+00	3.26E-03	5.42E-03	0.00E+00	5.40E-03
W-187	4.40E-04	2.60E-04	0.00E+00	7.00E-04	1.16E-03	0.00E+00	1.16E-03
Np-239	1.32E-02	1.36E-02	1.80E-04	2.69E-02	4.48E-02	0.00E+00	4.40E-02
FISSION PRODUCTS							
Br-83	1.60E-04	1.60E-04	0.00E+00	3.20E-04	5.20E-04	0.00E+00	5.20E-04
Sr-89	2.40E-04	5.00E-04	2.00E-05	7.60E-04	1.26E-03	0.00E+00	1.26E-03
Sr-90	2.00E-05	4.00E-05	0.00E+00	6.00E-05	1.00E-04	0.00E+00	1.00E-04
Y-90	0.00E+00	2.00E-05	0.00E+00	2.00E-05	4.00E-05	0.00E+00	4.00E-05
Sr-91	3.02E-03	1.26E-03	0.00E+00	4.30E-03	7.14E-03	0.00E+00	7.20E-03
Y-91m	1.96E-03	8.00E-04	0.00E+00	2.76E-03	4.60E-03	0.00E+00	4.60E-03
Y-91	1.40E-04	3.40E-04	2.00E-05	5.00E-04	8.20E-04	0.00E+00	8.20E-04
Sr-92	6.20E-04	5.60E-04	0.00E+00	1.18E-03	1.96E-03	0.00E+00	1.96E-03

TABLE 11.2-13

EXPECTED YEARLY ACTIVITY RELEASED FROM LIQUID WASTE MANAGEMENT SYSTEMS USED FOR EVALUATION OF COMPLIANCE WITH APPENDIX I OF 10CFR50 (Ci/yr/site)(1)

NUCLIDE	LIQUID RADWASTE PROCESSING (Curies/Site)	LAUNDRY RADWASTE (Curies/Site)	LRW CHEMICAL PROCESSING (Curies/Site)	TOTAL LWR (Curies/Site)	ADJUSTED TOTAL (Ci/yr/Site)(2)	DETERGENT WASTES (Ci/yr/Site)	TOTAL (Ci/yr/Site)
Y-92	3.20E-03	1.50E-03	0.00E+00	4.70E-03	7.82E-03	0.00E+00	7.80E-03
Y-93	3.20E-03	1.34E-03	0.00E+00	4.56E-03	7.58E-03	0.00E+00	7.60E-03
Zr-95	2.00E-05	4.00E-05	0.00E+00	6.00E-05	1.00E-04	0.00E+00	1.00E-04
Nb-95	2.00E-05	4.00E-05	0.00E+00	6.00E-05	1.00E-04	0.00E+00	1.00E-04
Nb-97	0.00+00	2.00E-05	0.00E+00	2.00E-05	2.00E-05	0.00E+00	2.00E-05
Nb-98	3.88E-03	4.42E-03	6.00E-05	8.36E-03	1.39E-02	0.00E+00	1.40E-02
Mo-99	1.10E-02	7.46E-03	6.00E-05	1.85E-02	3.08E-02	0.00E+00	3.00E-02
Tc-99m	4.00E-05	1.00E-04	0.00E+00	1.40E-04	2.40E-04	0.00E+00	2.40E-04
Ru-103	4.00E-05	1.00E-04	0.00E+00	1.40E-04	2.40E-04	0.00E+00	2.40E-04
Rh-103m	4.60E-04	2.40E-04	0.00E+00	7.00E-04	1.14E-03	0.00E+00	1.14E-03
Ru-105	4.60E-04	2.40E-04	0.00E+00	7.00E-04	1.16E-03	0.00E+00	1.16E-03
Rh-105m	4.40E-04	3.40E-04	0.00E+00	7.80E-04	1.32E-03	0.00E+00	1.32E-03
Rh-105	0.00E+00	2.00E-05	0.00E+00	2.00E-05	4.00E-05	0.00E+00	4.00E-05
Ru-106	0.00E+00	2.00E-05	0.00E+00	2.00E-05	4.00E-05	0.00E+00	4.00E-05
Te-129m	1.00E-04	2.00E-04	0.00E+00	3.00E-04	4.80E-04	0.00E+00	4.80E-04
Te-129	6.00E-05	1.20E-04	0.00E+00	1.80E-04	3.20E-04	0.00E+00	3.20E-04
Te-131m	1.60E-04	1.00E-04	0.00E+00	2.60E-04	4.40E-04	0.00E+00	4.40E-04
Te-131	2.00E-05	2.00E-05	0.00E+00	4.00E-05	8.00E-05	0.00E+00	8.00E-05
I-131	3.86E-03	6.74E-03	1.60E-04	1.08E-02	1.79E-02	0.00E+00	1.80E-02
Te-132	2.00E-05	2.00E-05	0.00E+00	4.00E-05	8.00E-05	0.00E+00	8.00E-05
I-132	1.32E-03	1.52E-03	0.00E+00	2.84E-03	4.72E-03	0.00E+00	4.80E-03
I-133	3.37E-02	1.87E-02	1.60E-04	5.26E-02	8.74E-02	0.00E+00	8.80E-02
I-134	2.00E-05	3.00E-04	0.00E+00	3.00E-04	5.00E-04	0.00E+00	5.00E-04
Cs-134	6.80E-04	1.60E-04	8.00E-05	9.20E-04	1.54E-03	0.00E+00	1.54E-03
I-135	1.25E-02	5.36E-03	2.00E-05	1.79E-02	2.98E-02	0.00E+00	3.00E-02
Cs-136	1.78E-03	3.40E-04	1.00E-04	2.22E-03	3.70E-03	0.00E+00	3.60E-03
Cs-137	4.60E-04	1.00E-04	6.00E-05	6.20E-04	1.02E-03	0.00E+00	1.02E-03

TABLE 11.2-13

EXPECTED YEARLY ACTIVITY RELEASED FROM LIQUID WASTE MANAGEMENT SYSTEMS USED FOR EVALUATION OF COMPLIANCE WITH APPENDIX I OF 10CFR50 (Ci/yr/site)[1]

NUCLIDE	LIQUID RADWASTE PROCESSING (Curies/Site)	LAUNDRY RADWASTE (Curies/Site)	LRW CHEMICAL PROCESSING (Curies/Site)	TOTAL LWR (Curies/Site)	ADJUSTED TOTAL (Ci/yr/Site)[2]	DETERGENT WASTES (Ci/yr/Site)	TOTAL (Ci/yr/Site)
Ba-137m	4.20E-04	1.00E-04	4.00E-05	5.80E-04	9.60E-04	0.00E+00	9.60E-04
Ba-139	4.00E-05	1.40E-04	0.00E+00	1.80E-04	3.00E-04	0.00E+00	3.00E-04
Ba-140	8.80E-04	1.70E-03	6.00E-05	2.64E-03	4.38E-03	0.00E+00	4.40E-03
La-140	2.20E-04	1.30E-03	6.00E-05	1.58E-03	2.62E-03	0.00E+00	2.60E-03
La-141	1.60E-04	1.00E-05	0.00E+00	2.60E-04	4.20E-04	0.00E+00	4.20E-04
Ce-141	8.00E-05	1.60E-04	0.00E+00	2.60E-04	4.20E-04	0.00E+00	4.20E-04
La-142	4.00E-05	1.00E-04	0.00E+00	1.40E-04	2.40E-04	0.00E+00	2.40E-04
Ce-143	4.00E-05	4.00E-05	0.00.E+00	8.00E-05	1.40E-04	0.00E+00	1.40E-04
Pr-143	1.00E-05	1.80E-04	0.00E+00	2.80E-04	4.60E-04	0.00E+00	4.60E-04
Ce-144	0.00E+00	2.00E-05	0.00E+00	2.00E-05	4.00E-05	0.00E+00	4.00E-05
Pr-144	0.00E+00	2.00E-05	0.00E+00	2.00E-05	4.00E-05	0.00E+00	4.00E-05
Nd-147	0.00E+00	2.00E-05	0.00E+00	2.00E-05	4.00E-05	0.00E+00	4.00E-05
ALL OTHERS	2.00E-05	2.00E-05	0.00E+00	6.00E-05	1.00E-04	0.00E+00	1.00E-04
TOTAL (EXCEPT H-3)	1.65E-01	1.33E-01	2.96E-03	3.02E-01	5.02E-01	0.00E+00	5.00E-01
TRITIUM RELEASE	118 CURIES PER YEAR						

NOTES:

(1) Liquid radwaste processing systems are common systems. Therefore, system releases are per site rather than per reactor unit.
(2) Per NUREG-0016 Revision 1, Adjusted Total column is based on an increase of 0.1 Ci/yr per unit, using the same isotopic distribution, to account for anticipated operational occurrences such as operator errors that result in unplanned releases.
(3) 9.48E-03 = 9.48x10-3

TABLE 11.2-14

CALCULATED EXPECTED EFFLUENT ACTIVITY CONCENTRATIONS FOR EVALUATION OF RADIOACTIVE RELEASES TO THE SUSQUEHANNA RIVER

Nuclide	Average Annual Concentration After Dilution (μCi/ml)	10CFR20 Table II, Column 2 Effluent Concentration (μCi/ml)	Fraction of Effluent Concentration Limits (dimensionless)
Corrosion and Activation Products			
Na-24	3.52E-09	5.0E-05	7.0E-05
P-32	2.76E-10	9.0E-06	3.1E-05
Cr-51	9.05E-09	5.0E-04	1.8E-05
Mn-54	1.13E-10	3.0E-05	3.8E-06
Mn-56	1.08E-10	7.0E-05	1.5E-05
Fe-55	1.63E-09	1.0E-04	1.6E-05
Fe-59	4.77E-11	1.0E-05	4.8E-06
Co-58	3.27E-10	2.0E-05	1.6E-05
Co-60	6.53E-10	3.0E-06	2.2E-04
Ni-65	5.03E-12	1.0E-04	5.0E-08
Cu-64	9.05E-09	2.0E-04	4.5E-05
Zn-65	3.27E-10	5.0E-06	6.5E-05
Zn-69m	6.28E-10	6.0E-05	1.0E-06
Zn-69	6.79E-10	8.0E-04	8.5E-07
W-187	1.46E-10	3.0E-05	4.9E-06
Np-239	5.53E-09	2.0E-05	2.8E-04
Fission Products			
Br-83	6.53E-11	9.0E-04	7.3E-08
Sr-89	1.58E-10	8.0E-06	2.0E-05
Sr-90	1.26E-11	5.0E-07	2.5E-05
Y-90	5.03E-12	7.0E-06	7.2E-07
Sr-91	9.05E-10	2.0E-05	4.5E-05
Y-91m	5.78E-10	2.0E-03	2.9E-07
Y-91	1.03E-10	8.0E-06	1.3E-05
Sr-92	2.46E-10	4.0E-05	6.2E-06
Y-92	9.80E-10	4.0E-05	2.5E-05
Y-93	9.55E-10	2.0E-05	4.8E-05
Zr-95	1.26E-11	2.0E-05	6.3E-07
Nb-95	1.26E-11	3.0E-05	4.2E-07
Nb-98	2.51E-12	2.0E-04	1.3E-08
Mo-99	1.76E-09	2.0E-05	8.8E-05
Tc-99m	3.77E-09	1.0E-03	3.8E-06
Ru-103	3.02E-11	3.0E-05	1.0E-06
Rh-103m	3.02E-11	6.0E-03	5.0E-09

Nuclide	Average Annual Concentration After Dilution (μCi/ml)	10CFR20 Table II, Column 2 Effluent Concentration (μCi/ml)	Fraction of Effluent Concentration Limits (dimensionless)
Ru-105	1.43E-10	7.0E-05	2.0E-06
Rh-105m	1.46E-10	N/A	-
Rh-105	1.66E-10	5.0E-05	3.3E-06
Ru-106	5.03E-12	3.0E-06	1.7E-06
Rh-106	5.03E-12	N/A	-
Te-129m	6.03E-11	7.0E-06	8.6E-06
Te-129	4.02E-11	4.0E-04	1.0E-07
Te-131m	5.53E-11	8.0E-06	6.9E-06
Te-131	1.01E-11	8.0E-05	1.3E-07
I-131	2.26E-09	1.0E-06	2.3E-03
Te-132	1.01E-11	9.0E-06	1.1E-06
I-132	6.03E-10	1.0E-04	6.0E-06
I-133	1.11E-08	7.0E-06	1.6E-03
I-134	6.28E-11	4.0E-04	1.6E-07
Cs-134	1.94E-10	9.0E-07	2.2E-04
I-135	3.77E-09	3.0E-05	1.3E-04
Cs-136	4.52E-10	6.0E-06	7.5E-05
Cs-137	1.28E-10	1.0E-06	1.3E-04
Ba-137m	1.21E-10	N/A	-
Ba-139	3.77E-11	2.0E-04	1.9E-07
Ba-140	5.53E-10	8.0E-06	6.9E-05
La-140	3.27E-10	9.0E-06	3.6E-05
La-141	5.28E-11	5.0E-05	1.1E-06
Ce-141	5.28E-11	3.0E-05	1.8E-06
La-142	3.02E-11	1.0E-04	3.0E-07
Ce-143	1.76E-11	2.0E-05	8.8E-07
Pr-143	5.78E-11	2.0E-05	2.9E-06
Ce-144	5.03E-12	3.0E-06	1.7E-06
Pr-144	5.03E-12	6.0E-04	8.4E-09
Nd-147	5.03E-12	2.0E-05	2.5E-07
Others	1.26E-11	2.03-09	6.3E-03
Totals	6.31E-08	1.46E-02	1.19E-02
H-3	1.48E-05	1.00E-03	1.48E-02
Total with H-3	1.49E-05	1.56E-02	2.67E-02

NOTES:

(1) Annual average concentration after dilution by the cooling tower blowdown.
(2) N/A = Not Applicable. 10CFR20 Table II, Column 2 does not define ECL limit.

TABLE 11.2-15

INPUT DATA FOR AQUATIC DOSE CALCULATIONS

LOCATION	PATHWAY		
	CONSUMPTION OF AQUATIC BIOTA	SHORELINE EXPOSURE	DRINKING WATER
	Edge of Initial Mixing Zone	Edge of Initial Mixing Zone	Danville
Dilution Factor	15.9*	15.9*	321
Transit Time	25.0 hr	1.0 hr	25.8 hr
Shore Width Factor		0.2	

* Site-specific study indicates that a value of 15.9 is appropriate for the Susquehanna Station. That value is used for determining operational compliance with the SSES Technical Specifications and reporting doses via the Annual Effluent and Waste Disposal Report.

TABLE 11.2-16

Security-Related Information
Text Withheld Under 10 CFR 2.390

Security-Related Information
Text Withheld Under 10 CFR 2.390

TABLE 11.2-16

Security-Related Information
Text Withheld Under 10 CFR 2.390

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-270, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure 11.2-1 replaced by dwg. M-270, Sh. 1 FIGURE 11.2-1, Rev. 55

AutoCAD Figure 11_2_1.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-271, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure 11.2-2 replaced by dwg.
M-271, Sh. 1

FIGURE 11.2-2, Rev. 55

AutoCAD Figure 11_2_2.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-272, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure 11.2-3 replaced by dwg. M-272, Sh. 1 FIGURE 11.2-3, Rev. 55

AutoCAD Figure 11_2_3.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-273, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure 11.2-4 replaced by dwg. M-273, Sh. 1 FIGURE 11.2-4, Rev. 55

AutoCAD Figure 11_2_4.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-274, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure 11.2-5 replaced by dwg. M-274, Sh. 1 FIGURE 11.2-5, Rev. 55

AutoCAD Figure 11_2_5.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-220, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure 11.2-6 replaced by dwg.
M-220, Sh. 1

FIGURE 11.2-6, Rev. 55

AutoCAD Figure 11_2_6.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-230, Sh. 1

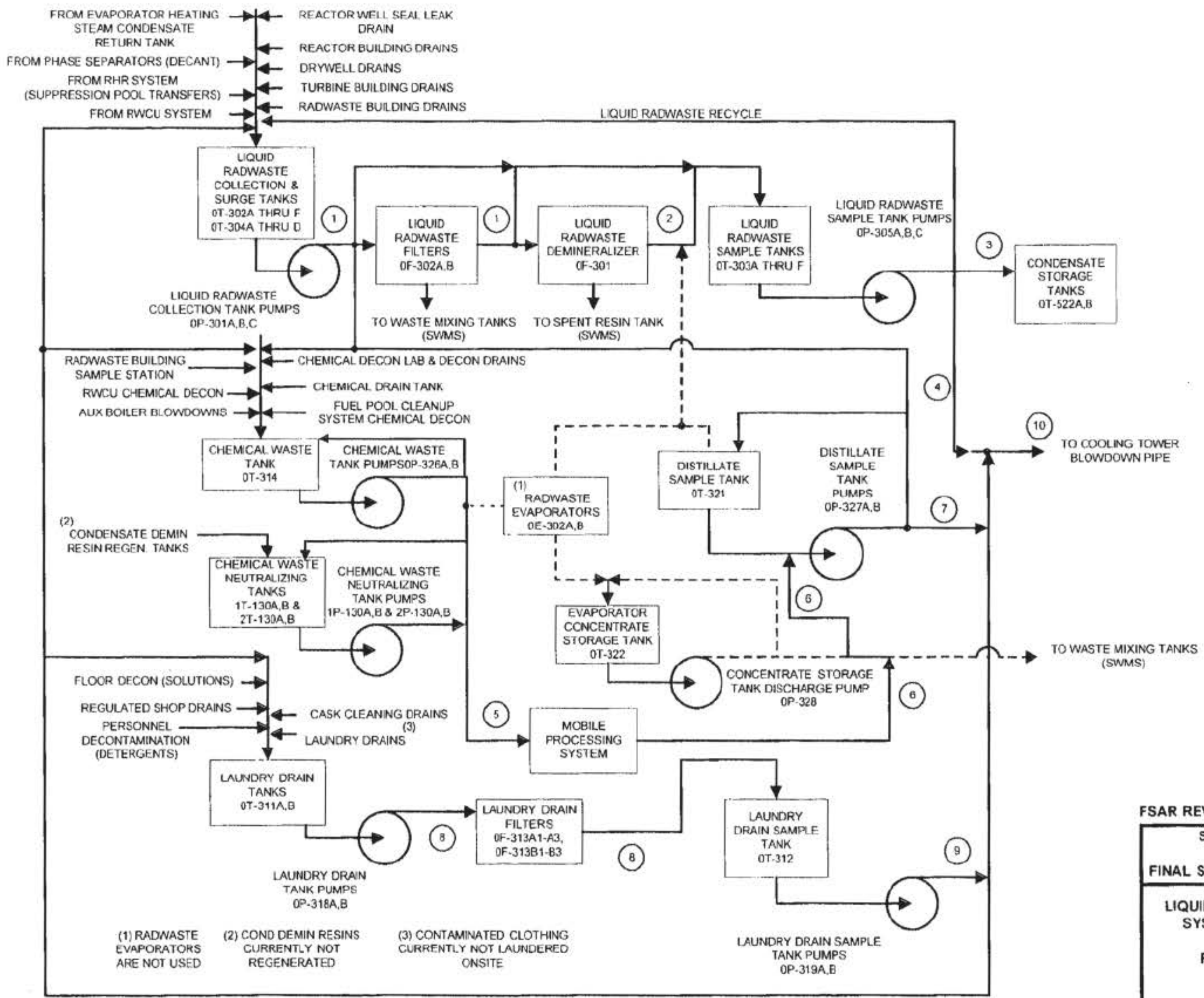
FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure 11.2-7 replaced by dwg.
M-230, Sh. 1

FIGURE 11.2-7, Rev. 55

AutoCAD Figure 11_2_7.doc



FSAR REV 53, 12/98

SUSQUEHANNA SES
UNITS 1 AND 2
FINAL SAFETY ANALYSIS REPORT

LIQUID WASTE MANAGEMENT SYSTEM FLOW DIAGRAM

FSAR FIGURE 11.2-8

PPL DRAWING VISO T11208.VSD

THIS FIGURE HAS BEEN
RENUMBERED TO 11.2-9-1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure renumbered from 11.2-9 to 11.2-9-1 FIGURE 11.2-9, Rev. 54

AutoCAD Figure 11_2_9.doc

THIS FIGURE HAS BEEN
RENUMBERED TO 11.2-9-2

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure renumbered from 11.2-10 to 11.2-9-2

FIGURE 11.2-10, Rev. 54

AutoCAD Figure 11_2_10.doc

THIS FIGURE HAS BEEN
RENUMBERED TO 11.2-11-1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure renumbered from 11.2-11 to 11.2-11-1
FIGURE 11.2-11, Rev. 54

AutoCAD Figure 11_2_11.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-164, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure 11.2-12 replaced by dwg.
M-164, Sh. 1

FIGURE 11.2-12, Rev. 55

AutoCAD Figure 11_2_12.doc

Security-Related Information
Figure Withheld Under 10 CFR 2.390

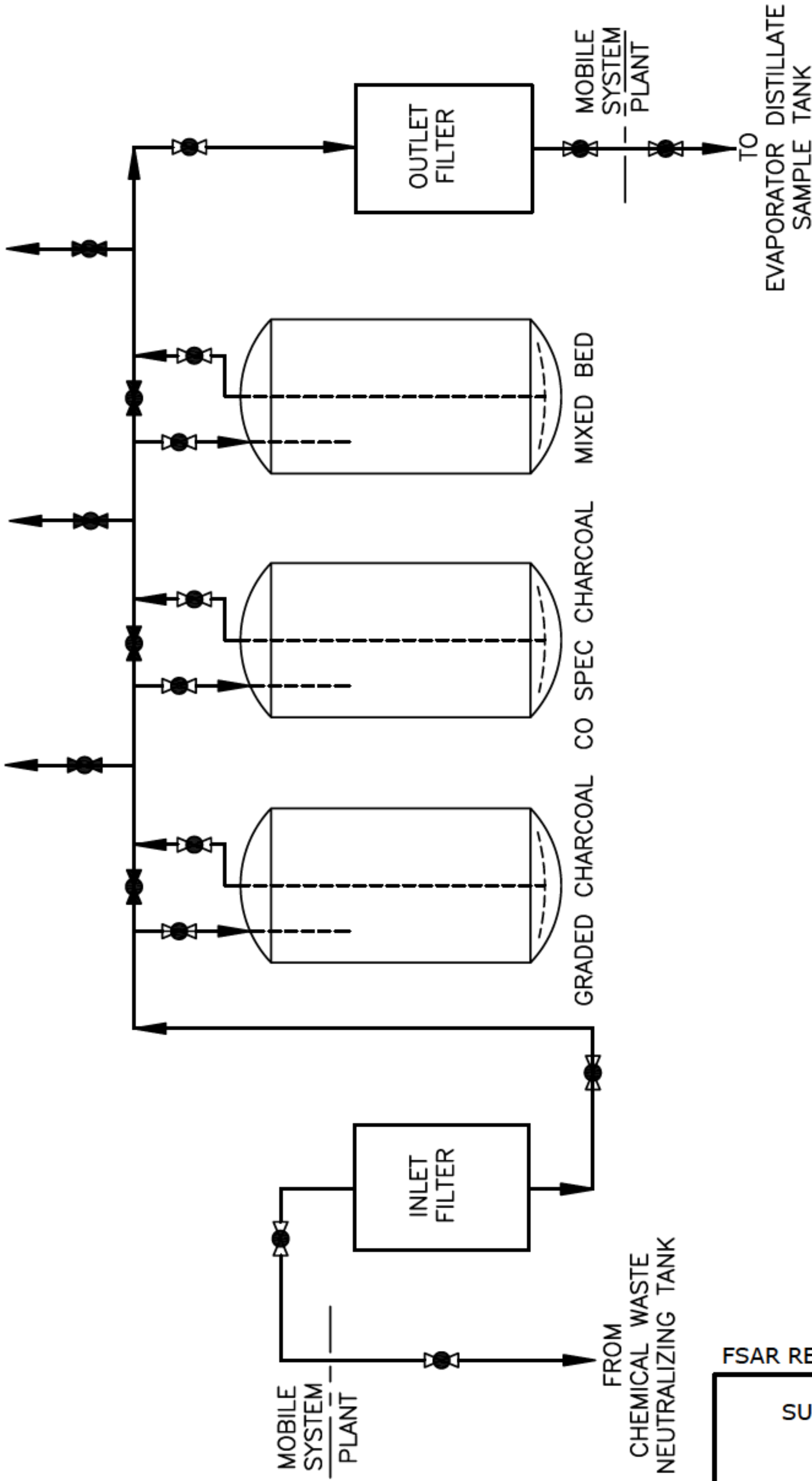
FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

LIQUID WASTE
MANAGEMENT SCHEME

FIGURE 11.2-13, Rev 55

AutoCAD: Figure Fsar 11_2_13.dwg



FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

TYPICAL MOBILE PROCESSING
SYSTEM FLOW DIAGRAM

FIGURE 11.2-14, Rev 54

AutoCAD: Figure Fsar 11_2_14.dwg

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-162, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure 11.2-9-1 replaced by dwg.
M-162, Sh. 1

FIGURE 11.2-9-1, Rev. 55

AutoCAD Figure 11_2_9_1.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-162, Sh. 2

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure 11.2-9-2 replaced by dwg.
M-162, Sh. 2

FIGURE 11.2-9-2, Rev. 56

AutoCAD Figure 11_2_9_2.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-162, Sh. 3

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure 11.2-9-3 replaced by dwg.
M-162, Sh. 3

FIGURE 11.2-9-3, Rev. 55

AutoCAD Figure 11_2_9_3.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-163, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure 11.2-11-1 replaced by dwg. M-163, Sh. 1 FIGURE 11.2-11-1, Rev. 55

AutoCAD Figure 11_2_11_1.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-163, Sh. 2

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure 11.2-11-2 replaced by dwg. M-163, Sh. 2 FIGURE 11.2-11-2, Rev. 55

AutoCAD Figure 11_2_11_2.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-163, Sh. 3

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure 11.2-11-3 replaced by dwg.
M-163, Sh. 3

FIGURE 11.2-11-3, Rev. 55

AutoCAD Figure 11_2_11_3.doc

11.3 GASEOUS WASTE MANAGEMENT SYSTEMS

The Gaseous Waste Management System (GWMS) includes systems that process potential sources of airborne radioactive releases during normal operation and anticipated operational occurrences. The GWMS includes the offgas system and various ventilation systems. These systems reduce radioactive gaseous releases from the plant by filtration or delay, which allows decay of radioactive materials prior to release.

The offgas system collects and delays the release of non-condensable radioactive gases removed from the main condenser by the air ejectors during normal plant operation. Plant ventilation systems process airborne radioactive releases from other plant sources, such as equipment leakage, maintenance activities, and the steam seal system as detailed in Section 9.4.

11.3.1 DESIGN BASES

The objectives and criteria, which form the bases for the design of the GWMS, are as follows:

a. The GWMS is designed to control and monitor the release of radioactive materials in gaseous effluents in accordance with GDC 60 and 64.

b. The GWMS is designed to limit offsite doses from routine station releases to less than the limits specified in 10CFR20, and to operate within the dose objectives established in 10CFR50, Appendix I.

c. The GWMS is designed to keep exposures to plant personnel ALARA while those personnel are conducting normal plant operation and maintenance activities.

d. The design basis and expected source terms correspond to fuel defects that result in a noble gas release rate of 100,000 and 50,000 μCi/sec, respectively, after a 30-minute delay.

e. The assumptions and parameters used for evaluating expected gaseous radioactive releases are based upon NUREG 0016, Revision 1 (Reference 11.3-4) and are listed in Tables 11.2-8, 11.3-2, and 11.3-4.

f. Filtration units in the ventilation systems are designed, operated, and maintained in accordance with the design bases presented in Section 9.4. Table 11.3-4 provides a listing of the filter trains that are used to control gaseous releases.

g. Continuous monitoring is provided for those pathways with significant potential for airborne radioactive releases.

h. A description of the major equipment items in the offgas system is provided in Table 11.3-5. The seismic and quality group classifications of the GWMS components, piping and structures housing them are listed in Section 3.2. The differences in "as-built" configuration of the Unit 2 portion of GWMS piping in regard to quality group "D" classification and stem leak-off connection to valves are shown in Dwgs. M-2169, Sh. 1 and M-2171, Sh. 1.

i. Conservative analyses, similar to those presented in Reference 11.3-1, demonstrate that equipment failure cannot result in doses exceeding acceptable guidelines; thus, neither the offgas system nor the buildings housing the equipment are required to meet Seismic Category I requirements. However, the offgas system is contained in the Turbine and Radwaste buildings, and the offgas vent is routed through the Reactor Building. The Reactor Building is Seismic Category I as described in Section 3.2. Turbine and Radwaste building structural walls are part of the total structural system and were analzyed to withstand a safe shutdown earthquake.

j. The GWMS is designed with sufficient capacity and redundancy to accommodate anticipated processing requirements during normal operation including anticipated operational occurrences.

k. Instrumentation is provided in the offgas system to detect abnormal concentrations of hydrogen and other system malfunctions.

l. The pressure boundary of the offgas system, consisting of piping and major components, is designed to either withstand the effects of multiple hydrogen detonations or to preclude the existence of a detonable gas mixture.

11.3.2 SYSTEM DESCRIPTIONS

11.3.2.1 Offgas System

Non-condensable radioactive offgas is removed from the main condenser by the mechanical vacuum pump (MVP), during startup and hot shutdown or by the steam jet air ejector (SJAE) during plant operation. The offgas consists of activation gases, fission product gases, radiolytic hydrogen and oxygen, and condenser air in-leakage. The SJAE offgas normally contains activation gases, principally N-16, O-19, and N-13. The N-16 and O-19 have short half-lives and are readily decayed. The N-13 isotope, with a 10-minute half-life, is present in small amounts that is further reduced by delay provided in the design of the offgas system. The SJAE offgas will also contain various isotopes of the radioactive noble gases, xenon and krypton, that are precursors 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 and the number and size of fuel cladding leaks.

The common offgas recombiner system receives up to a 5 scfm low flow purge air from the service air system when the common offgas recombiner is in standby. The purpose of the low flow purge air is to dilute hydrogen in the common offgas recombiner and move it into the aligned Unit 1 or Unit 2 ambient temperature charcoal offgas system.

The offgas system is designed to reduce the radioactivity in the offgas to permissible levels for release under all site atmospheric conditions. The system utilizes catalytic recombination for volume reduction and control of hydrogen concentration. Selective adsorption of fission product gases on activated carbon is used to provide time for delay of short-lived radioisotopes before release.

The building layout and equipment location of the offgas system components is shown on Dwgs. M-220, Sh. 1, M-230, Sh. 1, M-272, Sh. 1, M-273, Sh. 1, and M-274, Sh. 1.

11.3.2.1.1 Process Flow Description

Figure 11.3-1 is the process flow diagram for the offgas system. The process data for startup and normal operating conditions are contained in Table 11.3-8. The P&IDs are shown as Dwgs:

M-169, Sh. 1	M-169, Sh. 2	M-169, Sh. 3	M-169, Sh. 4
M-2169, Sh. 1	M-2169, Sh. 2	M-2171, Sh. 1	M-2171, Sh. 2.
M-171, Sh. 1	M-171, Sh. 2		

During startup or shutdown of a unit, the MVP is used to draw or maintain a vacuum on the main condenser as described in Section 10.4.2. The MVP discharge bypasses the offgas recombiner and treatment systems and enters the turbine building exhaust vent downstream of the filter units. Any activity in the MVP discharge mixes with the turbine building vent flow and is monitored by the vent sampling system. The MVP is typically used to maintain vacuum prior to and during reactor heat-up to support condensate de-aeration and flushing of the condensate and feedwater systems. Operation of the vacuum pump is limited to 5% power in order to avoid the formation of explosive gas mixtures in the pump, water separator and discharge piping.

When sufficient steam pressure is available, the offgas system (which requires steam for the recombiner preheater) and the two-stage SJAE train, consisting of 4 parallel primary stage air ejectors and one secondary stage ejector, are placed into service using main steam. Alternatively, clean steam from the auxiliary boilers may be used to drive the SJAE and the recombiner system to minimize the operation of the MVP and the release of fission gases to the turbine building vent.

The non-condensable gases in the main turbine condenser are removed by the SJAE and discharged to the offgas recombiner system. For an in-service recombiner Steam is used to dilute hydrogen in the offgas to less than the flammability limit in air from the discharge of the secondary steam jet through the recombiner to the recombiner condenser. Additional dilution steam is provided by a bypass loop around the ejector nozzle to the discharge. This arrangement allows adjusting the total steam flow for dilution without sacrificing SJAE performance. Piping and components from the dilution steam injection point to the recombiner condenser inlet are electrically heat traced to prevent condensation of the dilution steam, particularly during cold start-up.

For the common offgas recombiner system in standby, low flow purge air is used to dilute hydrogen in the common offgas recombiner system to less than the flammability limit in air. The low flow purge air dilutes hydrogen from the common offgas recombiner skid inlet through to the aligned Unit 1 or Unit 2 ambient temperature charcoal offgas system.

There are three offgas recombiner systems, one for each unit and a common system that can be used by either unit when necessary. The purpose of the recombiner system is to reduce the offgas volume and to eliminate the potential for an explosion. To support the operation of Hydrogen Water Chemistry, oxygen is injected upstream of the recombiner system at a flow rate of approximately one half the hydrogen injection rate (Reference Subsection 9.5.9 for a description of the system). In the recombiner, hydrogen reacts with oxygen in a controlled manner within a catalyst bed. The hydrogen concentration is reduced to less than 1% concentration by volume on a dry basis of 5 scfm air flow and less than 0.5% concentration for an air flow of at least 10 scfm. When the common offgas recombiner is in standby, the hydrogen concentration is reduced to less than 1% concentration by volume by use of the low flow purge air.

The offgas first passes through the recombiner preheater in order to minimize the moisture content prior to entering the catalyst bed. The recombination process takes place inside the recombiner vessel that is electrically preheated during standby by strip heaters on the outside. The reaction temperature is approximately 800°F.

The water vapor in the offgas leaving the recombiner vessel is removed in the recombiner condenser where the offgas is cooled. A motive steam jet then boosts the saturated gas stream pressure from below to slightly above atmospheric pressure.

The reduced pressure main, or auxiliary, motive steam used in the motive jet is removed from the offgas stream in the motive steam jet condenser. The offgas then passes through a delay pipe from the recombiner system in the turbine building to the ambient temperature charcoal offgas system in the radwaste building.

The pressure differential between the condensers in the recombiner systems and the main condenser is sufficient to drain the condensate without additional motive force to the main condenser, while the delay pipe is drained by level controlled valves to the turbine building radwaste sump.

The delay line consists of approximately 689 ft. of 8 in. diameter and 60 ft. of 16 in. diameter piping. At a nominal flow rate of 21.8 scfm, this pipe provides for 12.9 minutes of decay of the radioactive isotopes in the offgas stream prior to entering the adsorption train.

After exiting this line, the gas is cooled to approximately 40°F by a refrigerated chiller unit to condense and remove moisture. The offgas flow is reheated to approximately 65°F to provide a dehumidified (dew point of 40°F) air flow to the activated carbon absorber train. This is necessary to maintain the carbon moisture content ≤ 5%. Moisture and temperature instrumentation measure the process conditions downstream of the chiller to monitor the performance of the water removal assemblies and to guard against degraded activated carbon performance that might result from either an increase in the moisture content or temperature of the gas.

Prior to entering the main activated carbon vessels, the process stream passes through a sacrificial guard bed. The principal function of this guard bed is to protect the main carbon beds against moisture and other contaminants when the dehumidification section is inoperable. Each guard bed has been sized to absorb the moisture that might result from a failure of the chiller over a period of approximately 40 hours. This design feature, in conjunction with the moisture and temperature instrumentation, provides protection against the contamination of the activated carbon adsorber bed. Differential pressure indication is provided.

After passing through the guard bed, the gas enters the main activated carbon adsorption bed. This bed, operating in a controlled temperature vault, selectively adsorbs and delays the xenon and krypton from the bulk carrier gas. This delay on the activated carbon permits the radioactive xenon and krypton isotopes to decay in place. Upon exiting the adsorber beds, the process stream passes through a HEPA outlet filter, where radioactive particulate matter and activated carbon fines are retained. Taps are provided to take effluent samples, if desired, to determine the efficiency of the adsorber system.

The process stream is then directed to the turbine building ventilation exhaust duct where it is diluted, with a minimum of 42,000 scfm of air, prior to being released from the top of the reactor building.

Table 11.3-1 provides the estimated annual expected isotopic activity released from the GWMS based upon assumptions and parameters given in Tables 11.2-8, 11.3-2, 11.3-4 and NUREG 0016, Revision 1.

11.3.2.1.2 Activated Carbon Holdup Time

After passing through the recombiner section, the off gas stream consists primarily of the air in-leakage from the main condenser and the air from the common offgas recombiner low flow purge.

From NUREG 0016, Revision 1, the xenon and krypton holdup times are closely approximated by the following equation:

$$T = 0.26 \frac{MK}{F} \qquad \text{(Equation 11.3-1)}$$

$$= 43.1 \frac{MK}{P^*}$$

*NUREG 0016, Revision 1, recommends the use of 0.0062 scfm/MWt when assessing routine radioactive releases and doses (per Equation 11.3-1).

Where:

T	=	holdup time (hours)
K	=	dynamic adsorption coefficient (cm^3/gm)
M	=	mass of activated carbon adsorber (10^3 lbs)
P	=	thermal power level (MW_t)
F	=	offgas flowrate (cm^3/min)

The Sixth Edition of the Heat Exchange Institute Standards for Steam Surface Condensers (Reference 11.3-3, paragraph 5.16(c)(2)) indicates that with certain conditions of stable operation and suitable construction, non-condensable gases should not exceed 6 scfm per shell for large condensers. The Susquehanna SES has three shells per condenser resulting in an anticipated in-leakage of 18 scfm. Dynamic adsorption coefficients used to determine the holdup times are discussed in NUREG 0016 and are a function of carbon type, temperature, and moisture content. The values used in the analysis are based on the manufacturer demonstrated values of 516 cm^3/gm for xenon and 36 cm^3/gm for krypton. The offgas treatment system contains 74 tons of activated carbon, excluding the guard beds. With the above dynamic adsorption coefficients and a combination condenser air in leakage and low flow purge air rate of approximately 21.8 scfm, the system provides holdup times of 39 days for xenon and 2.7 days for krypton.

11.3.2.1.3 Detonation Resistance

The SSES offgas treatment system is designed to either withstand multiple hydrogen detonations or to preclude the existence of a detonable gas mixture. The system has been analyzed utilizing a conservative design method as a guideline (Reference 11.3-5) for calculating equipment and piping wall thickness capable of withstanding multiple internal hydrogen detonations such that the system pressure boundary will be useable without repair or subsequent inspection.

The basic methodology used in the design of detonation resistant BWR offgas systems is described in American National Standard, ANSI/ANS-55.4, 1979, Appendix C and assumes the absence of simultaneous secondary events such as earthquakes. A refinement of this ANSI methodology was utilized in assessing the detonation resistance of the SSES offgas system.

In addition, gases removed from the SJAE condenser by the second-stage ejector and discharged to the off gas system are mixed with the motive steam to eliminate the possibility of an explosion in the line between the SJAE discharge and the recombiner condenser for the in-service recombiner system. A bypass piping loop around the second stage air ejector provides additional steam to dilute the hydrogen concentration and maintain the recombiner discharge temperature within limits. For the common offgas recombiner system in standby, the low flow purge air, when in service, assists in eliminating the possibility of an explosion in the common offgas recombiner system. Consequently, a detonable mixture of gases will not exist between the dilution steam injection point and the recombiner condenser. Controls close the first stage air ejector suction valves if the second stage SJAE plus bypass steam flow decreases by approximately 15% below the operating set point.

The offgas hydrogen analyzers, pre-treatment radiation monitors and other instrumentation, which is not safety related, may fail following a detonation within the off gas pressure boundary. However, failure of this equipment poses no personnel or public safety hazard. The offgas system detonation resistance was reviewed by the USNRC as documented in License Amendment(s) 179 for Unit 1 and 152 for Unit 2.

11.3.2.2 Component Description

11.3.2.2.1 Recombiner System

Three recombiner assemblies are located in the turbine building in a shielded area below the main condenser steam jet air ejectors. Each recombiner assembly consists of the following major components: a recombiner, preheater, recombiner vessel, recombiner condenser, motive steam jet, motive steam jet condenser, and a condensate cooler.

One recombiner assembly is primarily designated for the service of each nuclear unit and the third assembly is a common standby to both units. Each recombiner assembly is sized to accommodate the design flow from one nuclear unit. The piping and valve manifold upstream of the recombiner assemblies permit the transfer of the offgas stream between a unit designated assembly and the common standby recombiner assembly.

The materials of construction, design pressures and temperatures, and the design codes for the components associated with the recombiner assemblies are listed in Table 11.3-5.

11.3.2.2.2 Activated Carbon Adsorber System

After entering the common inlet header, the gas mixture from each unit can be directed to either of two parallel equipment sub-trains each consisting of a water removal/temperature reduction assembly, and a activated carbon guard bed. The utilized activated carbon adsorption train of each offgas treatment system is primarily designated for the service of the associated nuclear unit. Each adsorption train consists of five activated carbon adsorber beds in series. The trains and sub-trains are isolable at both the inlet and outlet by remotely operated valves. The following subsections describe the various equipment that is associated with each system.

11.3.2.2.3 Inlet HEPA Filter

The activated carbon adsorber system inlet HEPA filter vessels do not contain filter elements.

11.3.2.2.4 Water Removal/Temperature Reduction Assembly

The water removal/temperature reduction assembly is used to cool and dehumidify the offgas to an operating temperature of approximately 65°F and a 40°F dewpoint in order to assure a maximum of 5% moisture is achieved in the activated carbon adsorbers. The offgas flow is first directed through a precooler which was originally designed to use reactor building closed cooling water as the cooling medium. Due to the acceptably low temperature of the offgas, the reactor building closed cooling water has been isolated from the precooler. This heat exchanger is built in accordance with TEMA Standard Class C, Type BEU.

A chiller is used to reduce the offgas stream temperature to approximately 40°F in order to condense and remove moisture. A refrigerant flows in the tube side and the offgas in the shell side. Water cooled refrigeration condensing units are provided for each chiller. This design eliminates the problems generally associated with a system circulating chilled glycol, such as leakage between the sides of the heat exchanger and leakage of glycol solution from pump seals. Also, the direct expansion refrigeration approach eliminates the use of circulation pumps, increasing system reliability. The refrigeration condensing units are located away from the precooler/chiller assembly in a low radiation area.

The condensate from the chiller is collected in a drain pot on the chiller. Since the accumulation rate of condensate is expected to be very small, an on-off type level control has been incorporated into the design. The condensate is directed back to the main condenser. Malfunction of the level control system may result in some of the offgas returning to the main condenser, thereby preventing an uncontrolled release into the radwaste building.

The offgas sides of the precooler and the chiller have been constructed of stainless steel to reduce the amount of corrosion products that might increase maintenance personnel doses or decrease system reliability.

11.3.2.2.5 Guard Beds

The offgas stream leaving the water removal assembly is reheated to approximately 65°F by electric heat tracing prior to entering the guard bed. The moisture content is then measured in order to monitor the performance of the water removal equipment. If the moisture content exceeds a preset level, an alarm is initiated in both the main control room and the local radwaste building control room.

The guard bed is provided to protect the main carbon beds against moisture and other contaminants when the dehumidification section is inoperable. Moisture in the main carbon adsorber beds would reduce the delay time for fission gases. The guard bed contains approximately 1280 lbs. of activated carbon. The guard bed is sized to absorb moisture that could result from a failure of the chiller over a period of approximately 40 hours. A low-pressure air drying/purge system has been provided to dry the guard bed should it become contaminated with water. However, the drying system is no longer used, because of the risk of carbon fires. The carbon in the guard beds is removed and replaced if an unacceptable pressure drop occurs.

The moisture monitor at the discharge of each guard bed will indicate when the guard bed is approaching saturation and corrective measures can be taken prior to any contamination of the main activated carbon adsorber bed.

The carbon steel guard bed vessel is designed to the code requirements of ASME Section VIII, Division 1. The materials of construction, the design pressure and temperature of these vessels are listed in Table 11.3-5.

11.3.2.2.6 Main Activated Carbon Adsorber Bed

Each adsorber train contains five tanks of activated carbon which are connected in series. These tanks provide sufficient delay of the radioactive noble gases, xenon and krypton, to permit releases to the environment that will satisfy the requirements of Appendix I to 10CFR50.

The temperature of the activated carbon is kept below 65°F, which is well below its ignition temperature, thus precluding overheating or fire and the consequential release of radioactive materials.

The adsorbers are located in shielded vaults which are maintained at a temperature below 65°F by one of two 100% capacity air conditioning systems that remove the decay heat generated in the adsorbers, any heat introduced by the process stream and through the vault walls.
The back-up air conditioning unit is activated automatically upon failure of the operating unit. Failure of the operating unit actuates a group alarm in the main control room and at a local control panel. In the unlikely event that both air conditioning units are unable to function, the radioactive emissions from the offgas system would increase slightly; however, the releases to the environs would still be well below acceptable limits for the condenser air in-leakage and purge air normally expected.

Channeling in the activated carbon adsorbers is prevented by maintaining a high bed-to-particle diameter ratio (approximately 750). Underhill (Reference 11.3-2) has stated that channeling or wall effects may reduce efficiency of the holdup bed if this ratio is not greater than 12. During installation of the activated carbon, the adsorber vessels may be vibrated from the outside to minimize voids and to increase the bulk density.

There are no provisions for bypassing the activated carbon adsorbers during any mode of operation, except during operation of the mechanical vacuum pump.

The ability of the activated carbon to delay the noble gases can be evaluated by comparing activities measured in samples taken at the outlet of the motive steam jet condenser and at the exit of the outlet HEPA filters.

The carbon steel activated carbon vessels are designed to the code requirements of ASME Section VIII, Division 1. The physical dimensions, materials of construction, design pressure, and design temperature of these vessels are listed in Table 11.3-5.

11.3.2.2.7 Outlet HEPA Filter

After the offgas stream exits the main activated carbon bed, it passes through a HEPA filter where any entrained particulates or activated carbon dust are collected. The removal efficiency of this HEPA filter is 99.97 percent for particulate sizes 0.3 micron and larger. The outlet HEPA filter is sized to accommodate the full design startup flow rate of 300 scfm.

The offgas stream exiting the outlet HEPA filter can be monitored for radioactivity by grab samples. The offgas stream is then directed to the turbine building exhaust duct, where continuous monitoring occurs, and released through the exhaust vent on top of the reactor building.

11.3.2.2.8 Instrumentation and Control

The offgas system is monitored by means of flow, temperature, pressure, and humidity instrumentation, and by hydrogen analyzers to verify specified operation and control, and to ensure that the hydrogen concentration is maintained below the flammable limit. Dwgs. M-169, Sh. 1, M-169, Sh. 2, M-2169, Sh. 1, M-2171, Sh. 1, and M-2171, Sh. 2 show the process parameters that are monitored to alarm in the main control room and the local radwaste control room, as well as whether the parameters are recorded or just indicated.

Dilution steam is provided in a bypass piping loop around the second stage air ejector in order to ensure that a detonable mixture of gases will not exist between the dilution steam injection point and the recombiner condenser and to keep the recombiner vessel outlet temperature below the design maximum. Controls are provided to close the first stage air ejector suction valves if the second stage SJAE plus bypass steam flow decreases by approximately 15%. Pretreatment radiation monitors continuously record and indicate gaseous radioactivity release from the reactor. These monitors provide information in the main control room on the condition of the fuel cladding and the inlet activity to the recombiner system and the activated carbon adsorbers. These monitors, through the annunciator system, provide redundant high and high-high alarms in the main control room when preset values are exceeded.

Experience with boiling water reactors has shown that the calibration correction factor of the offgas radiation monitors changes with the isotopic content. The isotopic content can change depending on the presence or absence of fuel cladding leaks in the reactor, the nature of the leaks, and the holdup time prior to release. Because of these variations, the monitors are periodically calibrated against grab samples.

Grab samples can be retrieved at the outlet of the motive steam jet condenser, at the test connections of the outlet HEPA filters of the activator carbon adsorber system, and at a connection on the offgas pipe leading to the exhaust vent on the reactor building. The combined second stage SJAE motive and dilution steam flow is measured and recorded by redundant instruments on the local recombiner control panel with low and low-low flow annunciation. Indication and low-low flow annunciation, by a group alarm, is provided on the main control room panel.

The temperature of the recombiner catalyst bed is monitored by three RTDs with each output switchable to one indicator. An alarm is provided to annunciate temperature conditions in excess of the process design value. The inlet temperature to the recombiner is monitored by redundant RTDs and alarms annunciate when the temperature falls below the point where adequate recombination of the radiolytic hydrogen and oxygen would occur. Each recombiner assembly is heat traced. The common standby recombiner assembly is heat traced and monitored to ensure its availability in case the unit designated assembly becomes inoperative.

The recombiner inlet and outlet temperatures are recorded and low, high and high-high alarms annunciate on the local panel while indication and high-high and low temperature alarms are annunciated on the main control room panel. Level controlled valves are used in the drain lines from the recombiner and steam jet condenser shells to the common condensate cooler which, in turn, drains to the main condenser. High condensate level alarms are annunciated at the local control panel, with a group alarm on the main control room panel.

The motive steam jet suction pressure is regulated by a butterfly valve in order to keep the recombiner condenser pressure above the main condenser pressure, thus allowing drainage of the condensate without a motive device.

Two redundant electro-chemical hydrogen analyzers are used to measure the hydrogen content of the offgas process stream at the discharge of each recombiner assembly. The hydrogen concentration from each analyzer is input to a computerized data acquisition system. A high hydrogen concentration alarm annunciates at the local control panel. High-high hydrogen concentration alarms are provided both at the local and main control room panels while indication is provided locally and on the main control room panel. Each hydrogen analyzer can be independently calibrated with the redundant one in operation.

The hydrogen analyzer systems continuously withdraw samples of the offgas, analyze the hydrogen content, and return the sample gas to the recombiner assembly. Hydrogen level setpoints are established in accordance with program and regulatory requirements. Oxygen analyzers are provided in series with the hydrogen analyzers. The oxygen concentration is an input to the Hydrogen Water Chemistry System, described in subsection 9.5.9.

Offgas system flow measurements are made downstream of the water removal assemblies in the charcoal offgas treatment system with indication and high flow alarm at a local and main control room panel and recording on the local panel only.

11.3.2.2.9 Leakage of Gases from Offgas System

Leakage of radioactive gases from the offgas system is limited by the use of welded construction wherever practicable. Leakage is further limited by the use of process valves that are of diaphragm or bellows stem seal design, or by using double stem packed valves with a bleed-off connection that is either pressurized by instrument air to slightly higher than the system pressure or routed to the main condenser.

The offgas system operates at a maximum of 5 psig during startup. During normal operations, the differential pressure between the system and the atmosphere is small thus limiting the potential for leakage of radioactive gases.

All drains from the various heat exchangers associated with the recombiner and activated carbon adsorber system are directed back to the main turbine condenser. Because of the low elevation, the drains from the delay line are routed through a drain pot with two level control valves in series into the radioactive turbine building sump. This minimizes the potential for offgas escape into the building in case of valve malfunctioning. Alarm and level control instrumentation is also provided.

11.3.2.3 Typical Operating Modes

11.3.2.3.1 Standby

During standby mode, the Unit 1 and Unit 2 recombiner systems are isolated from the offgas stream. The assembly steam supply and preheater bleed steam supply valve as well as the condensate cooler drain valve are open. For the common standby recombiner, the recombiner system discharge valve opens automatically to eliminate potential for pressure buildup due to through seal leakage of the common motive steam jet supply valve. Also for the common standby recombiner, the assembly steam supply and preheater bleed steam supply valves are open and the condensate cooler drain valves are aligned to and from either reactor unit. This, in conjunction with the electrical offgas inlet line heat tracing, keeps the system within a temperature range of 240°F to 270°F, thus preventing condensation when switching the offgas stream from an operating recombiner to the standby one.

Depending on the air in-leakage to the main condenser, this transfer is to be performed within approximately 10 minutes in order to keep the condenser pressure below allowable limits.

Cooling water is normally maintained to the standby recombiner assembly and the refrigeration condenser of the activated carbon adsorber system sub-trains not in operation. The refrigeration system for the chiller is placed in the standby mode and will start upon demand.

While in standby, the low flow purge air is aligned to the common offgas recombiner.

11.3.2.3.2 Prestart

In the prestart mode, the Unit 1 and Unit 2 motive steam jet steam supply valve and the recombiner system discharge valve are open in addition to the valves opened during standby. Motive steam may be from the auxiliary or nuclear boiler. For the common standby recombiner only, the common motive steam jet steam supply valve is open in addition to the valves opened during standby. The motive steam is condensed in the motive steam jet condenser while the recombiner system components are evacuated, ready for offgas admission.

The water removal chiller as well as the activated carbon bed vaults of the ambient temperature charcoal adsorber system must be at the required operating temperature and all valves in the normal operation status.

11.3.2.3.3 Normal Operation

Prior to placing the recombiner system from the prestart into the normal operation mode the following permissives must be present:

- Recombiner inlet temperature not low
- Recombiner outlet temperature not high-high

- Recombiner condenser cooling water flow not low
- Motive steam jet condenser cooling water flow not low

Each permissive is incorporated into the controls of the recombiner system by a two out of two logic allowing opening of the offgas inlet valves to the first stage SJAE upon establishing steam flow through the second stage SJAE and the dilution steam bypass.

Closing of the first stage SJAE offgas inlet valves occurs when any of the above permissives trip or there are two out of two trip signals of dilution steam (2nd stage SJAE motive & bypass) flow low-low.

The recombiner system inlet valve closes upon a recombiner condenser outlet temperature high-high which in turn automatically opens a bypass valve recycling the SJAE discharge back to the main condenser. This allows switchover to the standby recombiner without interrupting the SJAE motive steam flow within the period determined by the rise of the main condenser pressure.

11.3.2.3.4 Startup

The offgas system can be started with the main condenser ≤ 8 in HgA with initial vacuum having been drawn by the MVP. The system requires a main condenser vacuum of at least 8 in HgA established to provide a motive force for returning condensate to the main condenser and prevent back flooding the recombiner systems. To prevent hydrogen buildup downstream, recombiner vessel temperature should be 240°F to 270°F before offgas flow is admitted to the vessel. Operation of both activated carbon absorber sub-trains is required for startup or anytime offgas flow is > 150 scfm. After startup, the flow rate of non-condensables exhausted by the SJAE in conjunction with the low flow purge air (if the common recombiner is in standby) should stabilize, primarily as a function of reactor power level and condenser in-leakage.

11.3.2.3.5 Equipment Malfunction

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

11.3.2.4 Other Radioactive Gas Sources

There are three general areas that contain gaseous radioactive sources: the primary and secondary containment, the turbine building, and the radwaste building. The description of the ventilation systems for these buildings is presented in Section 9.4. The building volumes, flow rates, sources, and other information required to calculate the airborne concentrations of radioactive materials and doses are discussed in Subsections 12.2.2, 12.3.3, and 12.4.

11.3.2.4.1 Primary and Secondary Containment

Gaseous radioactive effluents can emanate from several sources. Leakage into the drywell and wetwell of the primary containment will be contained until containment atmosphere is purged in preparation for maintenance. Purged gases are processed through the activated carbon filters of the SGTS prior to release to the plant environs.

As indicated in Section 9.4, the two reactor buildings and the common refueling floor area have been designated as HVAC Zones I, II and III, respectively. Each of these zones has been divided into equipment compartment areas, where radioactive leakage may be expected, and other areas that contain non-radioactive equipment, accessways, and the refueling floor.
The exhaust air from the equipment compartment where radioactive leakage may occur is discharged through exhaust systems containing six-inch deep activated carbon filters. The air from the other areas is usually released unfiltered; however, if high concentrations do occur, the air can be re-circulated and a small fraction discharged through the SGTS until the high concentration condition can be corrected.

In the Appendix I evaluation it was assumed that radioactive releases from the reactor building are processed through the activated carbon and HEPA filters before release to the atmosphere. There may be small quantities of radioactivity released unfiltered from the refueling floor area and the spent fuel pool, especially during the early stages of refueling. However, the quantities of iodine and particulates released from this source are expected to be much less than the releases from equipment leakage, equipment maintenance, drywell purge, and the vessel head lifting operation, all of which are filtered. Considering the uncertainties in the calculation of the reactor building releases and the conservative assumptions use in estimating releases, it is expected that the actual releases from the reactor building to the atmosphere should be lower than the estimates presented in this evaluation.

The main steam relief valves are vented to the suppression pool. The activity released from the actuation of these relief valves will be contained in the primary containment until its atmosphere is purged through the SGTS. Effects of releases to the Suppression Pool are bounded by the effects of the closure of all Main Steam Isolation Valves concurrent with a stuck-open Main Steam Safety Relief Valve. The analyses of these events and their effects are described in Sections 15.1.4 and 15.2.4.

11.3.2.4.2 Radwaste Building

Leakage into the radwaste building atmosphere will be processed through a pre-filter and HEPA filter. In addition, an activated carbon filtration system processes the exhausts from the major radwaste system tanks.

11.3.2.4.3 Turbine Building

As indicated in Subsection 9.4.4, the turbine building ventilation system contains a filtration system with HEPA and six inch deep activated carbon filters. Building air from those areas of the turbine building where equipment leakage and airborne activity could be expected is processed through the filtration system before it is released through the turbine building vent exhaust to the atmosphere. Air from non-contaminated areas is released through the turbine building vent exhaust without filtration.

The process valve stem leakage collection system is used for the collection of stem packing leakage from steam valves in the turbine building. Leakage from these valves is directed to the main turbine condenser and processed through the offgas system prior to being released to the environs.

Valves in the turbine building were originally provided with valve stem packing leakoff connections. Research and testing has shown that improved packing provides an effective seal to prevent leakage into the Turbine Building. As a result, these leakoff connections are in the process of being removed and packing configurations changed, as appropriate, to conform with the new requirements. As part of this effort, leakoff isolation valves and piping will be removed (or abandoned in place) and the leakoff collection header piping will be removed or abandoned in place.

In the past, the steam packing exhaust has presented a source of gaseous radioactive releases in some BWR plants. However, at this station, an auxiliary source of clean steam is provided for gland seal purposes from the steam seal evaporator. Therefore, essentially no activity is released from this system. Subsection 10.4.3 provides a detailed description of the gland seal steam system.

During the startup of each unit, air is removed from the main turbine condenser by a mechanical vacuum pump. This vacuum pump discharges to the turbine building ventilation exhaust system. A radiation detector continuously monitors the effluent from the turbine building exhaust system and an alarm is actuated upon the detection of a high radiation level.

11.3.3 RADIOACTIVE RELEASES

An evaluation of the gaseous radioactive releases was performed to show compliance with the ALARA guidelines. The assumptions used in this evaluation are summarized in Tables 11.2-8, 11.3-2, and 11.3-4 for gaseous releases. Expected radioactive releases from the major buildings, prior to treatment, are presented in Table 12.2-30. The calculated annual expected gaseous radioactive releases per unit are given in Table 11.3-1. Expected average annual radionuclide concentrations are compared to 10CFR20 limits in Table 11.3-11.

The building vent locations, shape, effluent flow rate, and heat input are given on Figure 11.3-4.

Actual plant operations are expected to differ from the assumptions used in the analysis. Air leakage into the condenser and other portions of the steam cycle under vacuum will vary due to aging and degradation of piping, valves and seals. Age and variations in moisture loading may affect the activated carbon dynamic adsorption coefficients. Fission product leakage from the reactor is expected to be much lower than assumed for the great majority of the plant operating time. These variations will be monitored, and Susquehanna will be operated such that the yearly routine releases will be kept ALARA, consistent with the dose guidelines of Appendix I to 10CFR50. The activity released from the various vents will be monitored to ensure that the airborne concentrations at offsite locations will be below the limits of 10CFR20 for unrestricted areas.

11.3.4 ESTIMATED DOSES

Dose calculations to assure compliance with Appendix I to 10 CFR Part 50 based on the expected gaseous source term referenced above were performed in accordance with USNRC Regulatory Guide 1.109 by use of the USNRC computer code "GASPAR". Input data for these calculations are given in Table 11.3-7. The doses resulting from gaseous effluents are a small fraction of the 10CFR20 limits and are within 10CFR50, Appendix I design objectives. A comparison of the estimated releases with the Appendix I design objectives is presented in Table 11.3-10.

11.3.5 REFERENCES

11.3-1 NEDO-10734, "A General Justification for Classification of Effluent Treatment System Equipment as Group D" (February 1973).

11.3-2 "Design of Fission Gas Holdup Systems," Proceedings of the 11th AEC Air Cleaning Conference, D. Underhill et al (1970).

11.3-3 "Standards for Steam Surface Condensers," Sixth Edition, Heat Exchange Institute, N.Y., N.Y. (1970).

11.3-4 NUREG 0016, Revision 1, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Boiling Water Reactors (BWR-Gale Code)," U.S. Nuclear Regulatory Commission (January 1979).

11.3-5 NEDE-11146, "Pressure Integrity Design for New Off-Gas Systems," GE Nuclear Energy, July 1971.

Table 11.3-1

ANNUAL GASEOUS RELEASES PER UNIT

Annual Building or Component Release[1] (Curie per Year)

Nuclide	Turbine Building	Reactor[2] Building	Radwaste Building	Steam Jet Air Ejector	Mechanical Vacuum Pump	Total Annual Release
H-3	6.0E+00	8.6E+01[3]	*[4]	*	*	9.2E+01
C-14	*	*	*	9.5E+00	*	9.5E+00
Ar-41	*	1.5E+01	*	8.7E+01	*	1.0E+02
Kr-83m	*	*	*	*	*	*
Kr-85m	5.00E+00	4.0E+00	*	3.0E+02	*	3.1E+02
Kr-85	*	*	*	2.6E+02	*	2.6E+02
Kr-87	1.2E+01	2.0E+00	*	*	*	1.4E+01
Kr-88	1.8E+01	4.0E+00	*	4.1E+01	*	6.3E+01
Kr-89	1.2E+02	2.0E+00	2.9E+01	*	*	1.5E+02
Xe-131m	*	*	*	5.9E+01	*	5.9E+01
Xe-133m				6.0E+00		6.0E+00
Xe-133	3.0E+01	1.1E+02	2.2E+02	6.7E+03	1.3E+03	8.4E+03
Xe-135m	8.0E+01	6.0E+01	5.3E+02	*	*	6.7E+02
Xe-135	6.6E+01	1.3E+02	2.8E+02	*	5.0E+02	9.7E+02
Xe-137	2.0E+02	1.9E+02	8.3E+01	*	*	4.7E+02
Xe-138	2.0E+02	8.0E+00	2.0E+00	*	*	2.1E+02
I-131	3.4E-04	8.9E-05	3.1E-03	*	1.3E-01	1.3E-01
I-133	5.0E-03	1.3E-03	4.4E-02	*	1.4E+00	1.5E+00
Cr-51	1.8E-06	1.1E-05	7.0E-06	*	1.0E-06	2.1E-05
Mn-54	1.2E-06	1.4E-05	4.0E-05	*	*	5.5E-05
Fe-59	2.0E-07	3.9E-06	3.0E-06	*	*	7.1E-06
Co-58	2.0E-06	3.0E-06	2.0E-06	*	*	7.0E-06
Co-60	2.0E-06	5.0E-05	7.0E-05	*	5.6E-07	1.2E-04
Zn-65	1.2E-05	5.0E-05	3.0E-06	*	3.4E-07	6.5E-05
Sr-89	1.2E-05	5.0E-07	*	*	*	1.3E-05
Sr-90	4.0E-08	1.0E-07	*	*	*	1.4E-07
Zr-95	8.0E-08	1.0E-05	8.0E-06	*	*	1.8E-05
Nb-95	1.2E-08	1.0E-04	4.0E-08	*	*	1.0E-04
Mo-99	4.0E-06	6.6E-04	3.0E-08	*	*	6.6E-04

Table 11.3-1

ANNUAL GASEOUS RELEASES PER UNIT

Annual Building or Component Release(1) (Curie per Year)						
Nuclide	Turbine Building	Reactor(2) Building	Radwaste Building	Steam Jet Air Ejector	Mechanical Vacuum Pump	Total Annual Release
Ru-103	1.0E-07	4.2E-05	1.0E-08	*	*	4.2E-05
Ag-110m	*	2.4E-08	*	*	*	2.4E-08
Sb-124	2.0E-07	5.0E-07	7.0E-07	*	*	1.4E-06
Cs-134	4.0E-07	4.7E-05	2.4E-05	*	3.2E-06	7.5E-05
Cs-136	2.0E-07	5.0E-06	*	*	1.9E-06	7.1E-06
Cs-137	2.0E-06	6.0E-05	4.0E-05	*	8.9E-06	1.1E-04
Ba-140	2.0E-05	2.2E-04	4.0E-08	*	1.1E-05	2.5E-04
Ce-141	2.0E-05	9.0E-06	7.0E-08	*	*	2.9E-05

Notes:

1. Based on NUREG-0016, Revision 1, “Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Boiling Water Reactors”, (BWR - GALE Code), Section 2.2.4.

2. All of Reactor Building tritium is assumed to be released in the Refueling Area.

3. 8.6E+01 = 8.6×10^{1}

4. * indicates a negligible quantity or less than 1 Ci/yr per unit for Noble gases.

TABLE 11.3-2

ASSUMPTIONS AND PARAMETERS USED FOR EVALUATION OF GASEOUS RELEASES

1. Reactor coolant and main steam radionuclide concentrations	Table 11.2-9
2. Radionuclide releases from buildings before treatment (from NUREG 16)	Table 12.2-30
3. Radionuclide input (I-131) into the main condenser offgas treatment (from NUREG 16) (Ci/yr/unit)	6
4. Charcoal Delay System parameters	
a) Mass of charcoal, M, (10^3lb)	148
b) Operating/dew point temperatures (°F)	60-65/40
c) Dynanmic adsorption coefficient, K, (cm^3/gm) for Xe/Kr	516/36
d) Power Level P (MW_t)	4032
e) e) Charcoal holdup time T (hrs)	$43.1\frac{MK}{P}$
5. Gland seal system flow rate lb/hr Activity level	25,000 Clean steam
6. Offgas holdup time before the charcoal delay system (hr)	0.12
7. Filtration systems on building ventilation	Table 11.3-4

Table 11.3-3

This Table Has Been Deleted

Table 11.3-4

Filter Trains Used To Control Gaseous Effluents

Filter	Rated Capacity (CFM)(1)	Filter Components*	Charcoal Thickness (in.)	Iodine Removal Efficiency(2) (%)
Containment/Auxiliary Building				
Unit 1 – Zone I, below refueling floor	16,000	P-H-C-H	6	99
Unit 2 – Zone II, below refueling floor	16,000	P-H-C-H	6	99
Units 1 & 2 – Zone III, refueling floor	4,000	P-H-C-H	6	99
Drywell purge – through SGTS	10,500	M-P-H-C-H	8	99
Turbine Building				
	20,000	P-H-C	6	99
Radwaste Building				
	30,000	P-H	0	0
Tank exhaust filter	1,000	P-H-C-H	2	70
Control Structure				
Sample Room exhaust	1,500	P-H-C	2	70
Radiation chemical lab	1,500	P-H-C	2	70
Radiation chemical lab	1,500	P-H-C	2	70
Decontamination area exhaust	1,500	P-H-C	2	70

* M – Moisture separator
P – Prefilter
H – HEPA
C - Charcoal

NOTES:

1. Flow rates are rated capacities per filter train. For system design basis flow rates, see FSAR Section 9.4.
2. Removal efficiencies for ESF system activated carbon adsorber units are per RG 1.52, Revision 2 and per RG 1.140, Revision 1 for non-ESF systems.

TABLE 11.3-5

OFF GAS SYSTEM MAJOR EQUIPMENT DESCRIPTION
(Design Codes and Standards are provided in Table 3.2-1)

Equipment	Equipment Numbers	Type	Qty.	Material	Capacity	Size	Design Pressure/ Temp. PSIG/°F
				Recombiner System			
Preheater	0E/1E/2E-136	Shell and Straight tubes, BEM	3	Shell, Channel: CS Tubes, Sheet: SS	271,246 Btu/hr (4)	21 0 sq. ft.	Shell Side: 450/450 Tube Side: 300/450
Recombiner Vessel	0S/1S/2S-125	Vertical Cyl.	3	SS; Pourable metal substrate precious metal catalyst	Approximately 39 cf Catalyst	60" Dia., 74" High	300/950
Catalyst Bed	----	GE23A6222	3	Pourable metal substrate precious metal catalyst	Normal Operation: Inlet: 239,340 SCFH total, <4.0 vol.% H_2 Outlet: 1 Vol % H_2 based on 5 SCFM Dry Air	55.25" Dia., 28" Deep	Maximum Operating Outlet: 830°F
Recombiner Condenser	0E/1E/2E-137	Shell and U-tubes, BEU	3	Shell, Tubes Sheet: SS; Channel: CS	$14.6x10^6$ Btu/hr(4)	970 sq. ft.	Shell Side: 300/950 Tube Side: 150/150
Condensate Cooler	0E/1E/2E-152	Shell and U-tubes, BEU	3	Shell, channel: CS; Tubes, Sheet: SS	$580x10^3$ Btu/hr(4)	138 sq. ft.	Shell Side: 300/250 Tube Side: 150/150
Motive Steam Jet	0E/1E/2E-133		3	SS	Startup: Suction: 300 SCFM Air 123 SCFM Steam Disch: 300 SCFM Air 1189 SCFM Steam	Suct: 6", Disch: 4", Motive 2 ½"	Body: 300/200 Steam Chest: 600/500
Motive Steam Jet Condenser	0E/1E/2E-134	Shell and U-tubes, BEU	3	Shell, channel: CS; Tubes, Sheet: SS	$3.53x10^6$ Btu/hr	960 sq. ft.	Shell Side: 300/300 Tube Side: 150/150
Delay Piping	GBC-106, GBC-206		3	CS		8" / 16"	300/850
Inlet HEPA Filter Housing[1]	1F301A,B 2F301A,B	NA (Note 1)	4	CS (Housing)	NA (Note 1)	Housing: 10" Dia x 36" High	450/200
Precooler (Note 3)	1E/2E-302A,B	Shell and U-tubes	4	Shell Side: SS Tube Side: SS	86,930 Btu/hr	41.3 sq. ft.	Shell Side: 150/150 Tube Side: 695/200
Chiller Vessel	1E/2E-303A,B	Shell and Helical Coils	4	Shell Side: SS Tube Side: SS	41,000 Btu/hr	115 Sq. ft.	Shell Side: 520/200 Tube Side: 150/125
Refrigeration Compressor;	1K/2K-321 A,B 1E/2E-301 A,B	Reciprocating Semi Hermetric	4	CI / Freon 12	41,000 Btu/hr.	3 HP	350/150

TABLE 11.3-5

OFF GAS SYSTEM MAJOR EQUIPMENT DESCRIPTION
(Design Codes and Standards are provided in Table 3.2-1)

Equipment	Equipment Numbers	Type	Qty.	Material	Capacity	Size	Design Pressure/ Temp. PSIG/°F
Condenser							
Guard Bed Vessel	1T/2T-305 A,B	Vertical Cyl.	4	CS	0.64/tons of activated charcoal	30” dia. X 122” high	410/150
Charcoal Adsorber Vessel	1T/2T-306 thru 310	Vertical Cyl.	10	CS	14.8 tons	96” dia. X 254” high	375/150
Charcoal Bed	----	PSPEC H1026	10	Activated Carbon	Adsorpt.coeff. Xe: 516 (cc/gm) (Note 2) KR: 36	PSPEC H-1026	PSPEC H-1026
Outlet HEPA Filter	1F/2F-302	Pleated Cartridge	2	Element: Glass Fiber Housing: CS Frame: SS	45 SCFM at rated capacity	Housing: 14” dia., 36” high Cartr.: 10 5/8” dia., 8” high	310/150

Notes:

1. HEPA filter element has been permanently removed from inlet filter housing.
2. Kd at 20°C, 5% moisture in carbon.
3. Precooler is not used.
4. Valves given for the capacity of the pre-heater, recombiner condenser, and condensate cooler represent the component heat load at rated power level operating under normal water chemistry.

TABLE 11.3-6

OFFGAS SYSTEM EQUIPMENT MALFUNCTION ANALYSIS

Equipment Item	Malfunction	Consequences	Design Precautions
1. Second Stage SJAE air-operated Bypass Dilution Steam Flow Control Valves	Fail closed	Dilution of hydrogen in offgas could be insufficient	Redundant flow control valves regulated by redundant flow measuring and controlling instrumentation are provided with local and main control room low flow alarms and system shutdown logics. The second valve would open on failure of the first valve to maintain flow. Low second-stage steam supply flow would trip local alarms, and low-low flow would also isolate the SJAE offgas suction valves and would trip a low-low steam flow alarm in the control room. If dilution flow and SJAE suction cannot be restored, rising main condenser backpressure will trip the turbine, followed by a reactor scram. Offgas would remain in the offgas system and main condenser, and no uncontrolled leakage would result.
		High recombiner temperature due to exothermic recombination of higher hydrogen concentration, possible pressure boundary damage	Control of inlet hydrogen concentration with SJAE bypass dilution steam should limit the available reaction exotherm, and thereby limit the recombiner process and outlet temperatures. Recombiner process or discharge temperature high would alarm locally and by a group trouble alarm in the control room, and discharge temperature is indicated in the control room. Recombiner discharge temperature high-high would alarm both locally and in the control room, and would cause an offgas isolation (closure of SJAE offgas suction isolation valves). Reopening the isolation valves would require the operator to correct the condition or align the common standby recombiner. Offgas would remain within the offgas system and main condenser with no uncontrolled release.

TABLE 11.3-6

OFFGAS SYSTEM EQUIPMENT MALFUNCTION ANALYSIS

Equipment Item	Malfunction	Consequences	Design Precautions
2. Piping between SJAE and Recombiner Assembly	Pipe rupture	Release of offgas mixture to turbine building basement	This event has been analyzed as a Limiting Fault. See FSAR Section 15.7. Radiation monitors would alarm and operators would close the SJAE inlet (offgas isolation) valves. The release would be processed through the turbine building ventilation-filtration system.
3. Recombiner Preheater	Steam leak	Would further dilute process offgas and be condensed in the recombiner condenser. Steam consumption and condensate flow would increase.	Any significant leakage would result in a pressure rise in the offgas stream upstream of the preheater, and would trip a local high pressure alarm and control room group trouble alarm. The excess steam would be condensed out in the recombiner condenser until the leakage was in excess of the condenser drain capacity. Local drain pot high level would then alarm and trip a control room group trouble alarm.
4. Recombiners	Catalyst gradually deactivates	The temperature profile would change through the catalyst. Eventually excess hydrogen would occur in the recombiner discharge.	Temperature probes are installed in the recombiner beds and exit of each recombiner. The hydrogen analyzer would indicate high hydrogen concentrations and provide a recording to compare to previous performance. If catalyst degraded significantly, a high hydrogen concentration alarm might trip, and would trip a control room group trouble alarm. A further increase in hydrogen concentration would trip a control room alarm and the SJAE offgas inlet valves would close.

TABLE 11.3-6

OFFGAS SYSTEM EQUIPMENT MALFUNCTION ANALYSIS

Equipment Item	Malfunction	Consequences	Design Precautions
	Catalyst gets wet	Hydrogen recombination would fall off and higher hydrogen concentrations would occur downstream. Eventually the gas mixture could become combustible.	Temperature probes are installed in the recombiner beds and exit of each recombiner. The hydrogen analyzer would indicate high hydrogen concentrations and provide a recording to compare to previous performance. If catalyst degraded significantly, a high hydrogen concentration alarm might trip, and would trip a control room group trouble alarm. A further increase in hydrogen concentration would trip a control room alarm and the SJAE offgas inlet valves would close.
			The second stage SJAE motive and bypass dilution steam ensures the steam concentration remains above the hydrogen ignition limit between the SJAE and the recombiner condenser inlet, and the offgas system downstream of the recombiner condenser inlet will withstand an internal hydrogen ignition.
5. Recombiner condenser, motive steam jet condenser, Condensate cooler.	Cooling water leak	Cooling water leak to the shell side of a condenser or cooler.	Moderate amounts of coolant leakage would be directed back to the main condenser. If leakage exceeded the capacity of the level control valves and drains, a high level alarm at the local and a group trouble alarm at the main control room panel would indicate this excessive leakage. Long period leakage would produce a low level in the gaseous radwaste recombiner closed cooling water head tank which would annunciate in the main control room. The offgas could then be transferred to the common standby recombiner assembly.
	Drain level control fails closed	Condensate would build up in the affected condenser and pressure drop would increase.	The level control and level alarms are separate instrumentation. The local high level alarm and main control room group trouble alarm would indicate the high level condition.

TABLE 11.3-6

OFFGAS SYSTEM EQUIPMENT MALFUNCTION ANALYSIS

Equipment Item	Malfunction	Consequences	Design Precautions
	Drain level control fails open	Gas would recycle to the main condenser, increase the load on the SJAE, increase main condenser pressure, and if left uncorrected, would eventually cause a turbine trip and reactor scram.	Valves are of the spring loaded fail closed type. Drains are routed to the main condenser thus eliminating any leakage offgas to the building.
	Loss of condenser cooling water	The dilution steam and the recombined hydrogen and oxygen would remain in the system as steam and increase the pressure in the recombiner assembly.	A separate offgas recombiner closed cooling water system is provided for each recombiner assembly. If the loss of cooling water is limited to a single recombiner assembly, common standby recombiner can be started. If there is a total loss of cooling water to all assemblies, the plant would be shut down.
			Low cooling water flow to either condenser would produce alarms and an offgas isolation (closure of SJAE offgas suction isolation valves). Reopening would require the operator to correct the condition or align the common standby recombiner.
			The immediate effect of a cooling water loss would be pressure increase immediately downstream of the recombiner condenser. As the pressure increases, a pressure switch would trip a time delayed local and group alarm in the main control room. A temperature rise in the recombiner condenser outlet would trip a high alarm and upon further increase, to high-high would cause a recombiner shutdown (close the offgas and motive steam inlet valves to the assembly and open the offgas recycle valve to the main condenser), permitting the SJAE's to remain in operation.

TABLE 11.3-6

OFFGAS SYSTEM EQUIPMENT MALFUNCTION ANALYSIS

Equipment Item	Malfunction	Consequences	Design Precautions
			Offgas would remain within the offgas system and main condenser with no uncontrolled release.
		Steam would pass the motive steam jet condenser and partially condense in the delay pipe. The remainder would partially reach the charcoal offgas treatment system.	An alarm would be triggered locally and on a group annunciator in the main control room indicating high motive steam jet condenser outlet temperature. See below for effects of excess moisture reaching the charcoal offgas treatment system.
	Loss of condensate cooler cooling water	Flashing in drain line to main condenser could reduce drain flow below required capacity.	High condensate inlet or outlet temperature would trip a local alarm and control room group trouble alarm. Cooling water is controlled by a single manual valve. Unless the malfunction were a line blockage, any malfunction which caused a loss of cooling water would therefore also include loss of cooling water to the entire recombiner, followed by recombiner isolation and operator action to align the common standby recombiner.
6. Holdup piping between the Recombiner Assembly and Charcoal Adsorber portions of the Offgas System.	Excessive corrosion of line	Leakage of gaseous and liquid radioactive products into the piping tunnel.	Area radiation monitors in the turbine and radwaste buildings. The piping tunnel is provided with floor drains which would direct leakage to the liquid radwaste system.
7. Holdup piping drains	Level valve fails to open.	Accumulation of condensate in pipe.	Level control instrumentation with a high level annunciator in the main control room. The manual bypass valve can be opened around the two redundant level valves.
	Level valve fails to close.	Offgas leakage to turbine building sump	Valves are spring loaded fail-closed types. Two valves in series are provided. The area radiation monitor would alarm.

TABLE 11.3-6

OFFGAS SYSTEM EQUIPMENT MALFUNCTION ANALYSIS

Equipment Item	Malfunction	Consequences	Design Precautions
8.Inlet HEPA filters NOTE: The ambient temperature charcoal offgas system inlet HEPA filter vessels no longer contain filter elements, and the inlet filter differential pressure alarms have been removed.			
9. Pre-cooler NOTE: Cooling water to the precooler has been isolated, the precooler is no longer used, and the precooler cooling water low flow alarm has been disabled.			
	Cooling water leak	None. Cooling water has been isolated.	Cooling water has been isolated.
	Loss of cooling water	None. Operating experience found that the precooler is not required.	None required.
10. Chiller refrigeration machines.	Complete loss of a chiller	The moisture content of the process stream would increase and contaminate the guard bed.	Two chiller-guard bed pretreatment subtrains are provided per unit, with chiller shell local high temperature alarm, control room group trouble alarm, and local and control room temperature indication. The operator would transfer the offgas stream to the other charcoal pretreatment subtrain. Moisture instrumentation is provided for the inlet and outlet of each guard bed, with local indication and alarms and a group trouble alarm in the control room. High moisture content at the inlet of the guard bed is alarmed in the main control room. The guard bed is sized to absorb moisture that would result from failure of a chiller for 48 hours during normal operations.
	Refrigerant tubes leaking	Refrigerant and lubricating oil would contaminate the guard bed and refrigerant might contaminate the charcoal adsorber beds.	Refrigerant would pass through both guard and adsorber beds with only a temporary minor reduction in adsorber capacity and without significant long-term effects. Lubricating oil would be trapped in the guard bed. The total charge of

TABLE 11.3-6

OFFGAS SYSTEM EQUIPMENT MALFUNCTION ANALYSIS

Equipment Item	Malfunction	Consequences	Design Precautions
			lubricating oil per chiller does not exceed the guard bed capture capacity. Two chiller-guard bed pretreatment trains are provided per unit. The operator would transfer the offgas stream to the other pretreatment subtrain on eventual loss of the chiller.
		Loss of cooling capacity would cause increased moisture content of the process stream.	Same as loss of a chiller, above, but might be less rapid.
11. Main Activated Charcoal Adsorbers	Activated carbon gets wet	Adsorption performance would deteriorate gradually as activated carbon gets wet. Holdup times would decrease and plant gaseous releases would increase.	Sacrificial guard bed would absorb moisture. Guard bed outlet local moisture indication and alarms with control room group trouble alarm. The guard bed is sized to absorb moisture that would result from failure of a chiller for 48 hours during normal operations.
12. Charcoal vault air conditioning units (Radwaste Building Chilled Water System, and Offgas System Area Cooling Coils and Fans)	Mechanical failure	If the ambient temperature increases, delay efficiency of the charcoal beds decreases. Increased emission could occur depending on the fuel leakage rate.	A spare air conditioning unit is provided. (Only ventilation is required in winter.)
13. Outlet HEPA filter	Hole in filter element	Filter bypass	Differential pressure instrumentation and filter test connections are provided. The activated carbon media in the guard beds and main adsorber beds should also be a good filter at low air velocity.

TABLE 11.3-7

INPUT DATA FOR ATMOSPHERIC DOSE CALCULATIONS

CRITICAL RECEPTION	SITE BOUNDARY LOCATIONS		VEGATABLE GARDEN	RESIDENCE	DAIRY FARM
Location	Nearest Site Boundary X/Q	Maximum X/Q Site Boundary	Maximum D/Q Garden / Maximum X/Q Residence	Maximum D/Q Residence	Maximum D/Q Dairy
Critical Sector	SSW	WSW	WSW	NE	WSW
Distance from vents-miles(meters)	0.39 (628)	1.22 (1963)	1.3 (2093)	0.9 (1448)	1.7 (2735)
Transit Time (sec)	175	357	381	596	498
X/Q (normal) – (sec/m^3)	1.16E-05	1.25E-05	1.14E-05	2.86.E-06	7.75E-06
X/QD (decayed) (sec/m^3)	1.16E-05	1.24E-05	1.13E-05	2.85.E-06	7.66E-06
X/QDD (decayed) + depleted (sec/m^3)	1.08E-05	1.07E-05	9.70E-06	2.51E-06	6.48E-06
DEP Deposition (1/m^2)	5.03E-08	1.60E-08	1.43E-08	1.81E-08	9.36E-09
Occupancy (hr/yr)	8766	8766	8766	8766	8766

TABLE 11.3-8

OFFGAS SYSTEM FLOWS
(Refer to Fig. 11.3-1)

			Design Normal Operation (40 SCFM Air)				Start-up (300 SCFM Air; Cond. Pressure 5" Hg**					
Stream No.	Press. Psia	Temp. °F	H_2 % Vol.Conc.	PPH/SCFM H_2O	PPH/SCFM Non-Cond.	PPH/SCFM Total	Press. Psia	Temp. °F	H_2 % * Vol.Conc.	PPH/SCFM H_2O	PPH/SCFM Non-Concens.	PPH/SCFM Total
2nd Stage SJAE Outlet	16.7	272	<4.0	10700/3712	635/277	11335/3989	15.7	270	--	10700/3712	1395/300	12095/4012
Recomb. Inlet	14.2	272	<4.0	10700/3712	635/277	11335/3989	13.8	270	--	10700/3712	1395/300	12095/4012
Recomb. Outlet	10.6	830	0.0015	11150/3949	186/40	11335/3989	9.9	720	--	10700/3712	1395/300	12095/4012
Recomb. Cond. Outlet	10.4	140	0.09	40/14	186/40	226/54	9.7	140	--	352/123	1395/300	1747/423
Motive Steam Jet Outlet	15.2	274	0.004	3080/1080	186/40	3226/1120	18.6	239	--	3392/1189	1395/300	4787/1489
Motive Jet Cond. Outlet	15.2	140	0.1	27 / 9	186/40	213/49	18.5	140	--	158/55	1395/300	1553/355
Precooler Inlet	15.1	140	0.1	27/9	186/40	213/49	17.7	140	--	80/28	698/150	778/178
Chiller Inlet	15.1	110	0.11	11/4	186/40	197/44	17.4	114	--	38/13	698/150	736/13
Chiller Outlet	15.0	40	0.125	1/0.3	186/40	187/40	16.7	85	--	10/4	698/150	708/154
Guard Bed Inlet	15.0	65	0.125	1/0.3	186/40	187/40	16.7	85	--	10/4	698/150	708/154
Chiller Inlet	14.9	65	0.125	1/0.3	186/40	187/40	15.8	65	--	10/4	698/150	708/154
Adsorber Chiller	14.8	65	0.125	1/0.3	186/40	187/40	15.0	65	--	10/4	698/150	708/154
Post-Filter Outlet	14.7	65	0.125	1/0.3	186/40	187/40	14.7	65	--	10/4	698/150	708/154

* During cold start-up a negligible amount of hydrogen is introduced into the main condenser.

** During cold start-up both charcoal adsorber trains are used when flow rate exceeds 150 SCFM.

TABLE 11.3-9

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Effects of releases to the Suppression Pool are bounded by effects of closure of all Main Steam Isolation Valves concurrent with a stuck-open Main Steam Safety Relief Valve. The analyses of these events and their effects are described in Sections 15.1.4 and 15.2.4.

Table 11.3-10

COMPARISON OF INDIVIDUAL DOSES FROM EXPECTED GASEOUS EFFLUENT RELEASES WITH 10CFR50, APPENDIX I DESIGN OBJECTIVES

	10CFR50 Appendix I Design Objectives (Per Unit)	**Expected Annual Dose (Per Unit)**
Noble Gas Effluents (at the site boundary)		
Gamma Air Dose(1) (mrad)	≤ 10	3.0
Beta Air Dose(1) (mrad)	≤ 20	5.1
Total Body of Individual (mrem)	≤ 5	1.9
Skin of Individual (mrem/yr)	≤ 15	4.6
Radioiodines and Particulates		
Dose to any Organ Infant Thyroid (Dairy) (mRem)	≤ 15	14.5
Dose to any Organ Child Thyroid (Garden) (mRem)	≤ 15	11.9

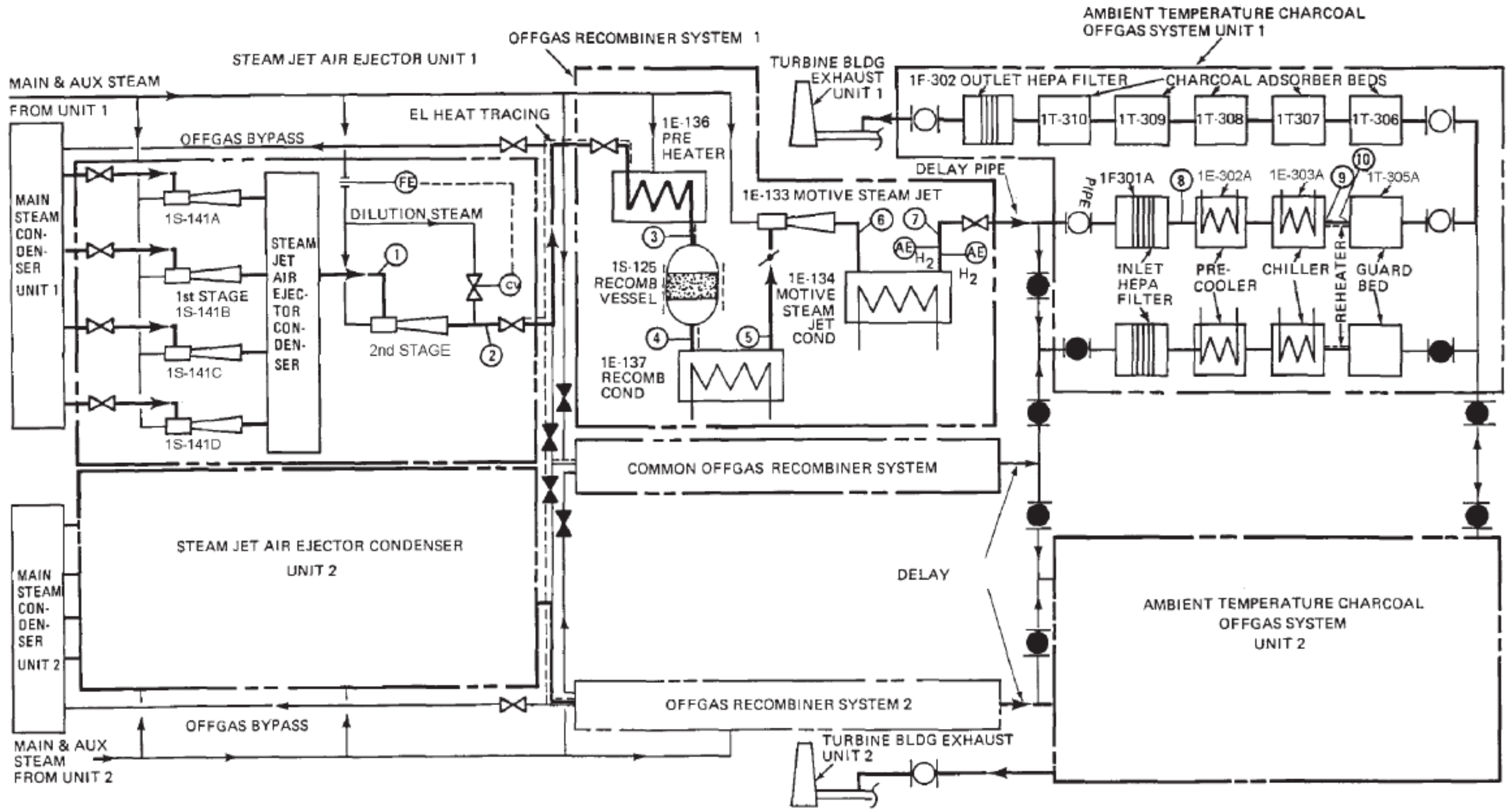
Table 11.3-11

COMPARISON OF EXPECTED RADIONUCLIDE CONCENTRATIONS IN THE ENVIRONMENT FROM ROUTINE ATMOSPHERIC RELEASES TO 10CFR20 LIMITS

LOCATION: WSW SITE BOUNDARY

Isotope	Expected Annual Average Concentration from Two Units (μCi/ml)	10CFR20 Appendix B,Table II, Column 1 Effluent Concentration Limit (μCi/ml)	Fraction of Effluent Concentration Limit
I-131	1.06E-13$^{(-13)}$	2.00E-10	5.30E-04
I-133	1.15E-12	1.00E-09	1.15E-03
Ar-41	8.08E-11	1.00E-08	8.08E-03
Kr-83M	-	5.00E-05	-
Kr-85M	2.44E-10	1.00E-07	2.44E-03
Kr-85	2.06E-10	7.00E-07	2.94E-04
Kr-87	1.13E-11	2.00E-08	5.62E-04
Kr-88	5.02E-11	9.00E-09	5.56E-03
Kr-89	1.17E-10	1.00E-09	1.17E-01
Xe-131M	4.68E-11	2.00E-06	2.34E-05
Xe-133M	4.76E-12	6.00E-07	7.92E-06
Xe-133	6.62E-09	5.00E-07	1.33E-02
Xe-135M	5.32E-10	4.00E-08	1.33E-02
Xe-135	7.74E-10	7.00E-08	1.11E-02
Xe-137	3.72E-10	1.00E-09	3.72E-01
Xe-138	1.66E-10	2.00E-08	8.32E-03
Cr-51	1.65E-17	3.00E-08	5.50E-10
Mn-54	4.38E-17	1.00E-09	4.38E-08
Co-58	5.54E-18	1.00E-09	5.54E-09
Fe-59	5.62E-18	5.00E-10	1.13E-08
Co-60	9.72E-17	5.00E-11	1.94E-06
Zn-65	5.18E-17	4.00E-10	1.29E-07
Sr-89	9.90E-18	2.00E-10	4.96E-08
Sr-90	1.11E-19	6.00E-12	1.85E-08
Nb-95	7.94E-17	2.00E-09	3.96E-08
Zr-95	1.43E-17	4.00E-10	6.58E-08
Mo-99	5.26E-16	2.00E-09	2.64E-07
Ru-103	3.34E-17	9.00E-10	3.70E-08
Ag-110M	1.90E-20	1.00E-10	1.90E-10
Sb-124	1.11E-18	3.00E-10	3.70E-09
Cs-134	5.92E-17	2.00E-10	2.96E-07
Cs-136	5.62E-18	9.00E-10	6.26E-09
Cs-137	8.80E-17	2.00E-10	4.40E-07
Ba-140	1.99E-16	2.00E-09	9.96E-08
Ce-141	2.30E-17	8.00E-10	2.88E-08
H-3	7.30E-11	1.00E-07	7.30E-04
C-14	7.54E-12	3.00E-09	2.52E-04
Site ECL Fraction			5.57E-01

NOTES: (1) 1.19e-13 = 1.19×10^{-13}



FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

OFFGAS SYSTEM
PROCESS FLOW DIAGRAM

FIGURE 11.3-1, Rev 54

AutoCAD: Figure Fsar 11_3_1.dwg

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DELETED

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure Deleted FIGURE 11.3-2, Rev. 48

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THIS FIGURE HAS BEEN
RENUMBERED TO 11.3-3-1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure renumbered from 11.3-3 to 11.3-3-1

FIGURE 11.3-3, Rev. 54

AutoCAD Figure 11_3_3.doc

Security-Related Information
Figure Withheld Under 10 CFR 2.390

FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

RELEASE POINT
LOCATIONS & DETAILS

FIGURE 11.3-4, Rev.56

Auto-Cad Figure Fsar 11_3_4.dwg

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SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure Deleted

FIGURE 11.3-5, Rev. 48

THIS FIGURE HAS BEEN
RENUMBERED TO 11.3-2A-1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure renumbered from 11.3-2A to 11.3-2A-1 FIGURE 11.3-2A, Rev. 55

AutoCAD Figure 11_3_2A.doc

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RENUMBERED TO 11.3-2A-2

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure renumbered from 11.3-2B to 11.3-2A-2 FIGURE 11.3-2B, Rev. 55

AutoCAD Figure 11_3_2B.doc

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RENUMBERED TO 11.3-2C-1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure renumbered from 11.3-2C to 11.3-2C-1 FIGURE 11.3-2C, Rev. 55

AutoCAD Figure 11_3_2C.doc

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RENUMBERED TO 11.3-2D-1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure renumbered from 11.3-2D to 11.3-2D-1

FIGURE 11.3-2D, Rev. 54

AutoCAD Figure 11_3_2D.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-171, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure 11.3-3-1 replaced by dwg. M-171, Sh. 1 FIGURE 11.3-3-1, Rev. 56

AutoCAD Figure 11_3_3_1.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-171, Sh. 2

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure 11.3-3-2 replaced by dwg. M-171, Sh. 2 FIGURE 11.3-3-2, Rev. 55

AutoCAD Figure 11_3_3_2.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-169, Sh. 1

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SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure 11.3-2A-1 replaced by dwg.
M-169, Sh. 1

FIGURE 11.3-2A-1, Rev. 57

AutoCAD Figure 11_3_2A_1.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-169, Sh. 2

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SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure 11.3-2A-2 replaced by dwg.
M-169, Sh. 2

FIGURE 11.3-2A-2, Rev. 57

AutoCAD Figure 11_3_2A_2.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-169, Sh. 3

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SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure 11.3-2A-3 replaced by dwg.
M-169, Sh. 3

FIGURE 11.3-2A-3, Rev. 56

AutoCAD Figure 11_3_2A_3.doc

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REPLACED BY DWG.
M-169, Sh. 4

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure 11.3-2A-4 replaced by dwg. M-169, Sh. 4 FIGURE 11.3-2A-4, Rev. 56

AutoCAD Figure 11_3_2A_4.doc

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REPLACED BY DWG.
M-2169, Sh. 1

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SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure 11.3-2C-1 replaced by dwg. M-2169, Sh. 1 FIGURE 11.3-2C-1, Rev. 57

AutoCAD Figure 11_3_2C_1.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-2169, Sh. 2

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure 11.3-2C-2 replaced by dwg. M-2169, Sh. 2 FIGURE 11.3-2C-2, Rev. 56

AutoCAD Figure 11_3_2C_2.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-2171, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure 11.3-2D-1 replaced by dwg. M-2171, Sh. 1 FIGURE 11.3-2D-1, Rev. 56

AutoCAD Figure 11_3_2D_1.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-2171, Sh. 2

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure 11.3-2D-2 replaced by dwg. M-2171, Sh. 2 FIGURE 11.3-2D-2, Rev. 55

AutoCAD Figure 11_3_2D_2.doc

11.4 SOLID WASTE MANAGEMENT SYSTEM

The Solid Waste Management System (SWMS) is designed to control, collect, handle, process, package, and temporarily store prior to offsite shipping, the wet waste sludges generated by the Liquid Waste Management System, the Reactor Water Cleanup System, Fuel Pool Cleanup System, the Condensate Demineralizer System, and the Condensate Filtration System. Contaminated solids such as HEPA, cartridge filters, rags, paper, clothing, tools, and equipment are also disposed of in the SWMS. The SWMS processes wet and dry solid waste materials. Process and Instrumentation Diagrams for the SWMS are presented on Dwgs. M-166, Sh. 1, M-166, Sh. 2, M-167, Sh. 1 and M-167, Sh. 2. A flow diagram for the SWMS is given on Figure 11.4-3.

The operation of the SWMS is conducted in accordance with the Process Control Program (PCP). The PCP provides administrative control, guidance and records for the processing, packaging, transportation, and disposal of radioactive solid waste. This procedure describes the envelope within which processing and packaging of radioactive waste materials is accomplished to provide reasonable assurance of compliance with low-level radwaste regulations and requirements. The PCP is applicable to Susquehanna SES installed systems and portable systems and equipment provided by vendors for processing, packaging, transportation, and disposal of applicable waste forms.

11.4.1 DESIGN BASES

The objectives and criteria which form the bases for the design of the SWMS are as follows:

a. The SWMS system is capable of receiving, processing, solidifying or dewatering the solid radioactive waste inputs as shown in Tables 11.4-1 and 11.4-2 for permanent offsite disposal. The SWMS is designed to package radioactive solid wastes for offsite shipment and burial in accordance with the requirements of applicable NRC and DOT regulations including 10CFR71 and 49CFR170 through 178. This results in radiation exposures to individuals and the general population within the limits of 10CFR20 and 50.

b. The SWMS has no nuclear safety related functions as a design basis.

c. Connections for mobile radwaste processing systems are available to support additional demands on the SWMS, to provide flexibility in radwaste processing, and to accommodate new technology. Mobile radwaste processing systems are subjected to the same performance objectives as the permanently installed systems and are approved for use in accordance with applicable SSES programs and procedures.

d. Mobile dewatering processing equipment, utilized to treat and package wet wastes, meet the requirements of ETSB 11-3, Revision 2 (Reference 11.4-1).

e. The SWMS is designed to minimize the volume of dewatered, solidified or compacted waste for offsite shipment and burial. There is no liquid plant discharge from the SWMS.

f. Redundant and backup equipment, alternate routes, and interconnections are designed into the system to provide for operational occurrences such as refueling, abnormal leak rates, decontamination activities, SWMS equipment down time, maintenance and repair. Table 11.4-3 shows the design parameters of the SWMS equipment.

g. Equipment locations, room designs, drainage, ventilation, and design features of components are consistent with those shown in Section 11.2 and are provided to reduce maintenance, equipment down time, leakage, gaseous releases of radioactive materials to the building atmosphere, or to otherwise improve the system operations.

h. The seismic and quality group classifications of the SWMS components, piping, and structures are listed in Section 3.2.

i. Remote controls and viewing systems are used to keep exposure to the personnel as low as reasonably achievable (ALARA).

j. Storage space for approximately three weeks' volume of solidified or dewatered waste from each unit is provided in the radwaste building. At expected generation rates, four (or more) additional years of storage capacity are available in the Low Level Radioactive Waste Holding Facility, as described in Section 11.6.

k. Dry Active Waste (DAW) can be stored in the Low Level Radioactive Waste Holding Facility as described in Section 11.6.

l. The expected radionuclide activity concentrations in the SWMS process equipment are based on reactor water radioactivity concentrations corresponding to fuel defects that result in 50,000 μCi/sec noble gas release rate for one reactor unit after a 30 minute delay. The design basis radionuclide activity concentrations in the SWMS process equipment are based on reactor water radioactivity concentrations corresponding to fuel defects that result in 100,000 μCi/sec noble gas release rate for one reactor unit after a 30 minute delay.

m. The expected and design inventories of individual radionuclides in components containing significant amounts of radioactive material are shown in Tables 11.4-5, 11.4-6, and 11.4-7.

11.4.2 SYSTEM DESCRIPTION

11.4.2.1 General

The Solid Waste Management System consists of two processing streams: 1) the wet solid waste process stream is utilized for the collection, processing, and dewatering or solidification of wet solids such as filter material slurries and spent resins, and 2) the dry solid waste process stream collects and packages dry solids such as contaminated filter media, clothing, equipment, tools, paper, and plastic sheeting.

Except for condensate demineralizer regeneration waste surge tanks, condensate filters, and backwash receiving tanks in the turbine building and the RWCU and fuel pool backwash receiving tanks in the reactor buildings, all SWMS equipment serves both reactor units and is located in the radwaste building.

11.4.2.2 Wet Solid Wastes

The wet solid waste processing is shown on Dwgs. M-166, Sh. 1, M-166, Sh. 2, M-167, Sh. 1, and M-167, Sh. 2. Wet solid waste is normally dewatered to meet applicable free standing water requirements. Alternatively, these wastes may be solidified if the need should arise. As shown on the flow diagram (Figure 11.4-3) and detailed in Table 11.4-4 some of the waste inputs are collectively processed due to their expected similar characteristics. Only spent resins from the RWCU system are expected to be of the HSA-type with the remainder of the wet solid waste categorized as low specific activity, as defined in 10CFR71.

Spent Resins

The spent ion exchange resin from the radwaste demineralizer is periodically sluiced to the spent resin tank for radioactive decay, settling, and storage until transferred to the mobile processing system. As an alternate process, the spent resin may be pumped from the spent resin tank to the liquid radwaste filters for dewatering and then transferred to a waste container.

Sufficient capacity is provided in the spent resin tank for several batches of radwaste demineralizer resins or one batch of condensate demineralizer resins of either reactor unit. The vent and overflow nozzles of the spent resin tank are equipped with 30 mesh screens to minimize spread of particulate contamination to the radwaste tank vent system. A spray nozzle with spherical pattern located in the tank center allows remote internal washdown. A manhole and external ladder provide access to the tank interior.

Associated with the spent resin tank is the spent resin transfer pump, which is of the progressing cavity (Moyno) type. This pump is normally used as a decant pump to the spent resin tank and then to transfer the spent resin to the mobile processing system for dewatering and disposal. The decanted water is pumped to the liquid radwaste collection tanks of the LWMS. The spent resin transfer pump may also be used for dewatering of the spent resin directly through the liquid radwaste filters. In this case, liquid radwaste collection tank water is added to the spent resin tank to dilute the spent resin to a pumpable slurry, with condensate transfer water also being available as a backup water source. The spent resin transfer pump is used to mix the tank contents by recirculating tank fluid through internal tank piping and fittings located near the bottom of the tank. After the resins are in suspension, a portion of the spent resin transfer pump discharge is directed to either one of the two radwaste filters for removal and dewatering of the resins from the slurry at a flow rate of 50 to 100 gpm. The spent resin transfer pump is sized to provide continuous recirculation of the tank contents during tank pumpout to keep the resin in the tank in suspension.

The spent resin transfer pump and associated valves are separated from the spent resin tank by a shield wall to permit maintenance access.

When dewatering using the liquid radwaste filters, the amount of spent bead resins being dewatered at one time is expected to be limited by the space between the filter screen plates rather than by the differential pressure across the filter screens. A demineralizer resin bed must therefore be dewatered in several batches.

The resin dewatering and/or discharge cycle of the filter in the dewatering mode is identical to the one described in Subsection 11.2.2.2 for the liquid radwaste filtering mode.

Condensate Demineralizer Waste

A twin set of interconnected conical bottom tanks is located close to each condensate demineralizer resin cleaning system in the turbine building. The corrosion product-containing waste stream is continuously recirculated through tank internal mixing nozzles and a 35 gpm partial flow is discharged to the waste sludge phase separator by one of two redundant in-line pumps. This inlet flow rate to the waste sludge phase separator allows continuous settling of the suspended solids. The supernatant overflows into an internal standpipe and is transferred by the waste sludge phase separator decant pump to the liquid radwaste collection tanks.

The waste sludge phase separator also receives drainage from the radwaste filter vessels prior to and following filter discharge cycles.

A mode selector switch also allows isolation of the waste sludge phase separator for extended settling periods. This mode may be used with resin fines carry-over in the condensate resin cleaning waste inflow. Although a flowpath exists from the fuel pool backwash receiving tank to the waste sludge phase separator, fuel pool demineralizer backwash is normally processed through a RWCU phase separator to maintain the low and high activity wastes segregation. Interlocks are also provided to prevent fuel pool filter demineralizer backwash into the phase separator when in the continuous decant mode in order to minimize the slow settling powdered ion exchange resin from entering the standpipe.

The elevation of the decant nozzle on the phase separator allows collection of approximately 500 gallons of sludge on the slanted bottom.

Before sludge is transferred to either the mobile processing system or the liquid radwaste filter by the waste sludge discharge mixing pump, it is diluted in a phase separator volume of water to a pumpable concentration. Internal mixing is accomplished through recirculated supernatant.

Automatic flushing of slurry carrying lines and mechanical seals of slurry pumps is provided.

Because of the pressure drop and volume limitations of the liquid radwaste filters, several dewatering batches may be necessary for one phase separator sludge load when dewatering with these filters. The resin dewatering and/or discharge cycle of the filter in the dewatering mode is identical to the one described in Subsection 11.2.2.2 for the liquid radwaste filtering mode.

Reactor Water Cleanup Filter Demineralizer and Fuel Pool Demineralizer Backwash Slurries

A RWCU backwash receiving tank is close to the reactor water cleanup filter demineralizer system of each reactor unit in the reactor buildings. One batch of exhausted powdered ion exchange resins from a RWCU filter demineralizer can be collected in each tank.

Exhausted powdered ion exchange resins from the fuel pool filter demineralizers of both reactor units are backwashed into a common fuel pool backwash receiving tank in the reactor building of Unit 1. One batch of exhausted fuel pool filter demineralizer resins can be collected in the backwash receiving tank.

Compressed air at a flow rate of approximately 75 scfm is injected through a diffuser at the tank bottom for approximately 30 minutes to agitate the slurry before and while the tank is gravity drained to one of two reactor water cleanup phase separators in the radwaste building.

The sludge holding capacity of one RWCU phase separator allows collection of one years' backwash sludges from both units RWCU demineralizers (total of four) and from the fuel pool demineralizers (three) at normal frequency. Sufficient settling time for the suspended solids in each backwash slurry batch is allowed before the supernatant is transferred by the reactor water cleanup decant pump to the radwaste collection tanks.

Alternating at approximately one year intervals, each RWCU phase separator is first in the sludge collecting and then in the isolated mode to allow radioactive decay of isotopes with short half-lives.

This provision reduces operator exposures in subsequent processing steps and facilitates handling and shielding for offsite disposal. The sludge holding capacity on the slanted bottom of each RWCU phase separator is 750 gallons. Additional decant nozzles are provided for adjustment of the phase separation height.

Before the sludge is transferred to either the mobile processing system or to the liquid radwaste filter by the reactor water cleanup sludge discharge mixing pump, it is diluted in a phase separator volume of water to a pumpable concentration. Internal mixing is provided by recirculated tank fluids driven through nozzles. Automatic flushing of slurry carrying lines and mechanical seals of slurry pumps is provided.

Condensate Filtration System Waste

Condensate filters are installed directly downstream of the steam packing exhauster (SPE) condenser. The purpose of the condensate filters is to remove iron from the condensate to mitigate dose effects of Hydrogen Water Chemistry and extend the life of the deep bed demineralizers. The filtration subsystem consists of equal size vessels designed to remove the suspended solids (mainly fine iron particles) prior to entering the demineralizer vessel resin beds.

All condensate filter vessels are normally in service at one time except for periodic backwashing. Particles accumulate on the filter elements causing an increasing resistance to flow. At a predetermined flow resistance (filter vessel pressure drop) or radiation level, the vessel will be taken out of service and backwashed. The backwash slurry is drained to the backwash receiving tank and from there pumped to the Waste Sludge Phase Separator (WSPS).

The filter elements in each filter vessel are periodically replaced (every 2 to 4 years) due to accumulated solids and reduced backwash efficiency. Filter element replacement entails removing the filter bundle from the vessel, placing it into a canister, sealing the canister and transporting it to a staging area where the dirty elements are removed from the tube sheets. Prior to placement in the canister, filter bundles are backwashed and drained of water. Dropping of a bundle during transfer could result in localized contamination but will not result in airborne or liquid activity release to the environment.

In order to improve the precipitation and filterability of the iron oxide particles contained in the CFS backwash, two chemical feed systems are installed.

A CPS Polymer Injection System is installed in the Unit 1 Turbine Building on elevation 656’ near the caustic storage tank 1T161. It utilizes one pump for each Unit. Each pumping unit is controlled by its respective CFS PLC to assure proper coordination of polymer addition with the slurry transfers from the BWRT. The injection point is located in the CFS backwash transfer line from the BWRTs to the WSPS. This system is the primary means of coagulating the fine iron oxide particles producing a larger agglomerate of particles (or floc) that can be effectively processed by the WSPS and liquid radwaste filters.

A Chemical Injection System is installed in the Radwaste Building at elevation 646' in the southeast corner of the WSPS room. It utilizes two redundant pumps. Control of the unit is from a local station and is entirely manual except that the injection pump is interlocked with a timer to trip after a preselected operating interval. The chemical injection point is into the WSPS via the Waste Sludge Discharge Mixing pump suction line. This will allow the addition of caustic (for pH control) or a second polyelectrolyte (polymer) directly to the WSPS batch during mixing, if required.

LRW Filter Waste

The dewatered mixture is packaged in disposable radwaste containers for offsite burial. A waste mixing tank beneath each radwaste filter receives, by gravity flow, the dewatered waste discharged from the filtering and dewatering process. While operating the mechanical agitator in the tank, the remainder of the mixing tank volume is then filled with condensate to produce a pumpable slurry for processing.

Two redundant process trains are provided in three separately shielded rooms. Each train consists of a mixing tank with conical bottom, agitator, internal decontamination spray nozzles, heat tracing, level detector, temperature sensor, the associated process feed pump, associated piping, valves, and instrumentation. Process feed pump discharge branch lines permit transfer of wastes between the two mixing tanks.

Wet Radwaste Dewatering, Solidification, and Packaging

Solids discharged from the waste mixing tanks, spent resin tank, RWCU phase separators and waste sludge phase separator are dewatered for offsite shipment. Due to the infrequent need to solidify, solidification equipment is not normally maintained onsite. When solidification is necessary, the process is either performed onsite or the waste is transported to a suitable solidification facility offsite.

Mobile radwaste processing systems may be used to package wet solid radwaste materials for offsite disposal in accordance with applicable burial site requirements (e.g. high integrity containers). Multiple filled waste container storage compartments may be used as waste container processing cubicles in conjunction with a mobile radwaste processing system. To provide radiation shielding for operating personnel, two steel compartment lids with waste container access ports are available as waste container processing shields in substitution for the standard compartment lids. Also, four composite lead/steel compartment lids with waste container access ports are available as waste container processing shields for lower level radiation dose rate waste containers. Branch lines from the spent resin tank, phase separators, and waste mixing tanks, as well as a return line to each phase separator, are provided at the radwaste building monitoring room near the truck loading area for hookup of mobile radwaste processing equipment. Branch lines to/from the Liquid Waste Management System are also provided in this room for the liquid radwaste collection tanks, sample tanks, and chemical waste tank for interfacing with mobile radwaste processing systems. A branch line from the phase separator sludge discharge pumps, a return line to the liquid radwaste collection tanks, and a vent line are provided to the radwaste building truck loading area for hookup of a mobile radwaste dewatering and/or solidification processing system.

Samples of waste from the RWCU phase separator, waste sludge phase separators or the spent resin tank are taken for chemical and radiological analysis to assure compliance with applicable 10CFR61 requirements. Typically, samples of waste from the waste sludge phase separators or the spent resin tank are taken directly from the waste containers after transfer to the mobile processing system, while the RWCU phase separator waste is sampled from a drain line in the tank after recirculating to ensure adequate mixing.

Mobile Radwaste Processing System

Wet waste is dewatered in a mobile radwaste processing system installed in the radwaste processing area located in the radwaste building. Waste can be pumped directly to this system from the spent resin tank, the phase separators, or from the waste mixing tanks. Excess water is returned to a phase separator.

This processing system is designed to handle powdered and bead type ion exchange resins and other filter media by removing excess water utilizing a two step process: filling and dewatering to meet gross or decanted dewatering requirements, wet waste processor requirements, or disposal site criteria. The waste container is filled from the plant's waste tanks using excess water to keep the resin in a slurry and recirculating the waste tank to maintain a homogeneous mixture. As the waste container is being filled, it is dewatered so that the available space in the container is filled with waste to the maximum extent possible. The excess water is pumped out of the container using a positive displacement diaphragm pump. The waste media is then transferred through the fillhead assembly to the dewatering container. During transfer of the waste media, the dewatering pump is energized to return the filtered slurry water back to the client's waste system. At the predetermined setpoint, the level detection system will close the inlet waste valve to prevent overfilling the container and the dewatering pump will continue to run to reduce the level of slurry water in the container. This cycle is repeated until the correct volume of waste is in the container and the slurry water is removed to meet applicable shipping or disposal site criteria.

Waste Transfer System - consists of a high pressure flexible hose, a 1-1/2" air-operated ball valve at the plant radwaste system interface, a manual-operated ball valve on the fillhead, and a portable radiation monitor on the waste transfer piping to provide quantitative radiation levels during transfer and flushing.

Fillhead - provides the connections between the waste transfer system and waste container. Connections to the piping skid provide for the removal of excess water from the container. A connection to the blower skid provides for incoming dry, hot air for the drying process. The electronics enclosure on the fillhead contains a remote video camera, a pressure switch, electrical connections for level indications, and an air supply for cooling.

Piping Skid - contains hose connections for processing either powdered or bead type resins, and an air-driven diaphragm pump to remove the collected water from the processing container

Control Panel - provides a central location to operate equipment and indication of processing parameters (level, temperature, valve position, and fillhead position) and alarms.

Remote Video System - a TV camera mounted in the electronics section of the fillhead provides a secondary level indication during resin transfer operations. A dimmer control for the video light is provided on the control panel.

Radwaste Building Crane

A remotely controlled bridge crane is provided in the solid waste handling area of the radwaste building. It is used for loading of empty waste containers onto the rail dolly, transferring filled containers from the rail dolly into temporary storage compartments and onto a shielded truck for offsite disposal, for disassembling the radwaste filters, and for general use.

The lifting deck on the crane is suspended by four independent cables reeved by two cable drums. This minimizes swaying during lifting and unbalancing of the load should one cable fail. An installed hook on the lifting deck engages the waste containers and storage compartment lids. This allows crane operation from the radio control system, or from the backup stationary control pendant, located in normally accessible areas. A selector switch is provided near the crane that is accessible from the floor to allow the use of either pendant or radio control. The crane uses a positioning system for the bridge and trolley positions provided by positioning lasers and displays on a display screen (mounted near the crane camera monitors), for when the lifting deck center reaches the container loading, pickup, storage compartment, and truck loading positions. A standard 25 ton crane hook can be mounted on the lifting deck for general use of the radwaste building crane.

The crane bridge and trolley travels are interlocked with the lifting deck elevation to prevent interference with shield walls and the container loading and pickup shaft walls.

Closed circuit TV camera on the crane lifting cameras on the crane bridge, the lifting deck, and in the truck bay transmit images of the relative crane position to monitors on the console to assist in remote operation of the crane when handling filled waste containers or compartment lids. Target marks on the container and compartment lids facilitate positioning of the lifting deck. Proper engagement of the grappling device is indicated by a light on the Panel View control console and the backup each pendant. A slack cable indication light is provided on the control console only.

The trolley and bridge speeds are stepped from 5 to 50 ft per minute white the hoist speed is controllable in 10 increments from 1.5 to 15 ft per minute. A speed selector switch is provided on the two radio transmitters and the backup pendant station that, when selected, will reduce the maximum speeds on all crane motions. The hoist motor adjustable frequency drive (AFD) control provides dynamic braking of the hoist. Two additional, independent solenoid operated holding brakes are provided.

<u>Waste Container Storage and Offsite Disposal</u>

Filled waste containers are separately stored in covered concrete compartments for radioactive decay prior to offsite disposal.

The number of compartments in the Radwaste Building allows storage of approximately six weeks anticipated dewatered waste volume for normal operation of one reactor unit considering refueling. The storage capacity consists of twelve shielded compartments for liners up to 200 cubic feet. Each compartment will contain one liner. Shielding of the storage compartments reduces the radiation in the adjacent crane control area to less than 2.5 mR/hr. Sufficient lifting height is provided to place a large waste container into a top entry shield cask on a truck.

Additional storage capacity for packaged dewatered or solidified waste is provided in the Low Level Radioactive Waste Holding Facility as described in Section 11.6.

11.4.2.3 Dry Solid Waste

The dry solid waste consists of contaminated air filter media, miscellaneous paper, rags, plastic sheeting, etc. from contaminated areas; contaminated clothing, tools and equipment parts that cannot be effectively decontaminated, mechanical cartridge filters and solid laboratory wastes and other similar materials. Dry Solid Waste is also called Dry Active Waste (DAW).

Depending upon the activity level, the physical size, and the material, different handling and packaging procedures for dry solid wastes are used. Except for irradiated reactor internals, the dry solid waste is expected to allow temporarily unshielded handling without exceeding the dose limits of 10CFR20. Generally, the dry solid wastes are shipped to an offsite waste processor for volume reduction prior to offsite disposal. If off-site disposal is not practicable, dry solid wastes packaged for burial may be temporarily stored at the Low Level Radioactive Waste Holding Facility (LLRWHF). Dry Solid Wastes may also be packaged for transportation and stored in the LLRWHF until enough is accumulated to permit economical transportation.

Contaminated Protective Clothing and Other Launderable Material

Contaminated laundry consists of multiple use materials such as protective clothing, cloth, or nylon bags, rags, tarps, and mop heads. The volume of dry solid waste is minimized by washing and reusing the contaminated laundry to the extent practical. The contaminated laundry is collected from the areas where it was generated and moved to the on-site laundry handling facility.

The on-site laundry handling facility is a temporary (double wide trailer) structure attached to the Unit 2 Turbine Building. The facility is connected into the Turbine Building wet pipe fire protection system, encompassing all sections of the facility except for any attached transport containers. Ventilation includes a recirculating filtered HVAC system which does not exhaust to the outside. The facility and any attached transport containers are part of a Radiological Controlled Area and are routinely monitored for dose rates and contamination.

The laundry handling facility is used to receive, store, package (into DOT acceptable transport containers) and ship contaminated laundry to a vendor for decontamination. The vendor cleans, monitors for contamination, and returns the laundered items to the laundry handling facility at SSES where it is off loaded into storage areas within the facility. The facility can store up to 5,600 cubic feet of laundry items. Laundry rejected by the vendor as being beyond repair or too contaminated is off loaded from the transport containers and conveyed to a dry solid waste collection area. Approximately 5 to 10 percent of all decontaminated laundry is rejected and becomes a dry solid waste.

Irradiated Reactor Internals

Irradiated reactor internals being replaced are removed from the RPV underwater and stored for radioactive decay in the fuel pool. Subsection 9.1.4.2 describes reactor vessel and in-vessel servicing equipment used for handling reactor components.

An estimated average of 7 percent (14) of the control rod blades are removed from one reactor annually (during the refueling outage - 24 month cycle) and are stored on hangers on the fuel pool walls or in racks interspersed with the spent fuel racks. Offsite shipping is done in suitable containers.

Approximately 50 percent (22) of the power range monitor detectors are replaced in one reactor annually (during the refueling outage - 24 month cycle). The replacement procedure is described in Subsection 9.1.4.2. Spent in-core detectors and dry tubes are transferred on a refueling platform auxiliary hoist or on the LPRM bender underwater to the spent fuel pool.

A pneumatically operated cutting tool supplied with the nuclear steam supply system allows remote cutting of the in-core detectors and dry tubes on the work table in the fuel pool or in the fuel pool to cask pit transfer canal area. The cut in-core monitors and dry tubes and other small sized reactor internals are shipped offsite in suitable containers and/or shielded casks that can be loaded underwater.

A trolley mounted disposal cask with an internal cable drum is supplied with the nuclear steam supply system for spent source and intermediate range neutron monitor detector cables and the traversing in-core probe (TIP) wires.

Offgas System HEPA Filter Elements

The outlet HEPA filter element of the ambient temperature charcoal offgas system is housed in a pressure vessel at the outlet of each unit's system. The annual number of disposed HEPA filter elements from the offgas system is shown in Table 11.4-2. The size of the individual filter elements allows for disposal in approved container with a 55 gallon drum size opening.

Miscellaneous Contaminated Dry Solid Waste

Administrative procedures provide for frequent radiation monitoring and periodic replacement of the ventilation and laundry drain system filter media to limit the dose to maintenance personnel during handling to as low as reasonably achievable (ALARA). Redundant filter trains further allow shutdown of one train for decay of the radioactive isotopes in the filter media before replacement. Portable charcoal removal and loading systems are employed for packaging exhausted charcoal beds in 55 gallon drums.

Pre-filter and HEPA filter elements are manually retrieved from the filter housings and wrapped in dust-tight plastic bags.

Dry solid wastes are collected and processed by various means. They may be packaged in approved containers for direct disposal or for transportation to vendor facilities for volume reduction prior to disposal. Some waste may be compressed into 55-gallon drums by a hydraulic compactor with a vent hood. A fan on the compactor keeps the 55-gallon drum at a slight vacuum with discharge through a HEPA filter to the building ventilation duct.

The averaged annual volumes of unprocessed dry solid waste is shown in Table 11.4-2. The average annual volume of charcoal waste is derived from the bed depth, number of test canisters provided, and required test frequency per NRC Regulatory Guide 1.52. These volumes may be up to six times higher for any given year due to removal of the longest time exposed test canister. The average annual volume of prefilter, HEPA, laundry drain cartridge filter and other miscellaneous waste is estimated from previous plant experience.

Volumes of miscellaneous dry solid wastes may vary widely depending on the housekeeping in the plant, number and type of modifications in progress and other factors. Vendor volume reduction services reduce the waste volume by a factor of three to one hundred, depending on the nature of the waste and the process used. The total generated volume of these wastes is expected to average between 10,000 and 20,000 cubic feet per year. The total disposal volume of this waste is expected to average between 2,000 and 8,000 cubic feet per year.

11.4.3 REFERENCES

11.4-1 USNRC Branch Technical Position ETSB 11-3, Revision 2

11.4-2 Topical Report DW-11118-01-NP

Table Rev. 57

TABLE 11.4-1

INPUTS TO THE SOLID RADWASTE COLLECTION SYSTEM FOR BOTH UNITS

SOURCE(7)	WASTE TYPE(8)	EXPECTED AVERAGE BATCH SIZE AND FREQUENCY(5) (FT^3/DAY)	MAXIMUM EXPECTED YEARLY VOLUME (FT^3)	EXPECTED YEARLY VOLUME (FT^3)(6)	NOTES
Spent Resin Tank					
Radwaste Demineralizer	Dewatered spent ion exchange bead resin	160/69	1010	850	1
Condensate Demineralizers	Dewatered spent ion exchange bead resin	300/77.5	3813	1413	2,3
Waste Mixing Tanks					
LRW Filters Backwash	Dewater powdered resin including crud	8/4.2	750	650	4,6
Waste Sludge Phase Separator					
Condensate Filters	Dewatered corrosion product crud with polyelectrolyte	1.0/7	52	52	10
LRW Filters Drain	Dewatered powdered resin including crud	1.0/4.2	100	85	4,6
RWCU Phase Separators					
RWCU Filter/Demineralizers	Dewatered spent powdered resin including crud	2.7/5.25	240	200	4,6
Fuel Pool Filter/Demineralizers	Dewatered spent powdered resin including crud	5.0/120	30	15	4,6

TABLE 11.4-1

INPUTS TO THE SOLID RADWASTE COLLECTION SYSTEM FOR BOTH UNITS

NOTES:

1) The Maximum Expected Yearly Volume includes an extra demineralizer at about 160 cubic feet above the Expected Yearly Volume.

2) Condensate demineralizer resin is not disposed on a regular frequency. A complete replacement of 16 beds is expected every 3.8 years on a unit operating condensate filtration. One additional bed is expected to be replaced each refueling outage. With a refueling outage every two years for the station, 18.8 beds will be replaced every four years for the station.

3) The Maximum Expected Yearly Volume includes an extra complete condensate demineralizer bed changeout on one unit as a result of a significant condenser tube leak. This includes 8 beds at 300 cubic feet per bed or 2400 ft^3.

4) The Maximum Expected Yearly Volumes are based on the maximum volumes disposed at SSES for the last five years 1992-1996 from the Susquehanna Cumulative Waste Processed Data Log.

5) The Expected Aver Batch Size and Frequency (ft^3/year) is based on the Expected Yearly Volume (ft^3).

6) The Expected Yearly Volume (ft^3) is based on the average disposal volumes for the last five years 1992-1996 from the Susquehanna Cumulative Waste Processed Data Log.

7) There is wet waste volume buried that is not reflected as a source to the solid radwaste collection system, therefore was not included specifically in this table. This waste is from the mobile liquid processing system used for treating chemical liquid waste. The waste is transferred directly from the mobile system to a HIC within the vendor supplied equipment system. In 1996, about 60 ft^3 of charcoal was disposed of from the mobile processing system. Bead resin may also be used in the mobile processing system.

8) The volumes presented in this table are only the wet solids portion of the total input slurry volumes therefore the waste types are all dewatered as stated. Since solidification is no longer performed, the dewatered volumes in the table represent the burial volumes less the packaging efficiency.

9) The Maximum Expected Yearly Volume and Expected Yearly Volume are based on the known iron input. The values are the same since the iron volume removed in year is not dependent on frequency.

TABLE 11.4-2

DRY SOLID WASTE AMOUNT FROM GAS, LAUNDRY DRAIN FILTERS, AND MAINTENANCE ACTIVITIES

SOURCE	EQUIPMENT NOS.	WASTE TYPE	SIZE (INCHES)	EXPECTED NUMBER/ WEIGHT OF FILTER UNITS PER YEAR	EXPECTED ANNUAL VOLUME UNCOMPACTED (FT^3)
Offgas Treatment System					
HEPA (Outlet)	1F-302/2F-302	Glass fiber, steel frame	12 dia. x 12	2	1
Control Structure Emergency O.A. Supply Filters					
Pre	0F-123A,B	Glass fiber, frameless	24x24x12	12	48
HEPA	0F-124A,B	Glass fiber, steel frame	24x24x12	12	48
Adsorption	0F-125A,B	Activ. Charcoal, 30 lb/ft^3	2336 lb	78	
HEPA	0F-126A,B	Glass fiber, steel frame	24x24x12	12	48
Reactor Bldg, Zones I and II Equipment Compartm. Exhaust Filters					
Pre	1F-254A,B/2F-254A,B	Same as above	24x24x12	64	256
HEPA	1F-255A,B/2F-255A,B		24x24x12	64	
Adsorption	1F-257A,B/2F-257A,B			29,336 lb	978
HEPA	1F-258A,B/2F-258A,B		24x24x12	64	256
Reactor Bldg. Zone III Exhaust Filters					
Pre	1F-215A,B/2F-215A,B	Same as above	24x24x12	16	64
HEPA	1F-216A,B/2F-216A,B		24x24x12	16	64
Adsorption	1F-217A,B/2F-217A,B			8180 lb	273
HEPA	1F-218A,B/2F-218A,B		24x24x12	16	64

TABLE 11.4-2

DRY SOLID WASTE AMOUNT FROM GAS, LAUNDRY DRAIN FILTERS, AND MAINTENANCE ACTIVITIES

SOURCE	EQUIPMENT NOS.	WASTE TYPE	SIZE (INCHES)	EXPECTED NUMBER/ WEIGHT OF FILTER UNITS PER YEAR	EXPECTED ANNUAL VOLUME UNCOMPACTED (FT^3)
Turbine Bldg. Exhaust Filters					
Pre	1F-156A,B/2F-156A,B	Same as above	24x24x12	84	336
HEPA	1F-157A,B/2F-257A,B		24x24x12	84	336
Adsorption	1F-158A,B/2F-258A,B			40,888 lb	1370
Control Structure Rad. Chem. Lab., Sample & Decon. Shower Room Filters					
Pre	0F-136/0F-139/0F-133/0F-142	Same as above	24x24x12	8	32
HEPA	0F-140/0F-137/0F-134/0F-143		24x24x12	8	32
Adsorption	0F-141/0F-138/0F-135/0F-144			20 trays 1400 lb	47
Radwaste Bldg. Exhaust Filters					
Pre	0F-354A,B	Glass fiber, frameless	24x24x12	60	240
HEPA	0F-355A,B	Glass fiber, steel frame	24x24x12	60	240
Hydraulic Compactor Exhaust Filter					
Radwaste Tank Vent Filters					
Pre	0F-357	Glass fiber, frameless	24x24x12	1	4
HEPA	0F-358	Glass fiber, steel frame	24x24x12	1	4
Adsorption	0F-359	Activ. charcoal, 30 lb/ft^3		1 tray 210 lb	7

TABLE 11.4-2

DRY SOLID WASTE AMOUNT FROM GAS, LAUNDRY DRAIN FILTERS, AND MAINTENANCE ACTIVITIES

SOURCE	EQUIPMENT NOS.	WASTE TYPE	SIZE (INCHES)	EXPECTED NUMBER/ WEIGHT OF FILTER UNITS PER YEAR	EXPECTED ANNUAL VOLUME UNCOMPACTED (FT^3)
Port. Charcoal Removal and Loading System Filters					
for Vent. Systems					
Pre	0S-512	Glass fiber, steel frame	24x24x12	2	8
HEPA		Glass fiber, steel frame	24x24x12	2	8
for Standby Gas					
Treatment System	0S-110,0S-111				
Pre		Cloth bag	24x24x12	2	8
HEPA		Glass fiber, steel frame	24x24x12	2	8
Standby Gas Treatment System Filters					
Pre	0F-172A,B	Glass fiber, frameless	24x24x12	18	72
HEPA	0F-170A,B	Glass fiber, steel frame	24x24x12	18	72
Adsorption	0F-169A,B	Activ. Charcoal, 28 lb/ft^3	13,840 lb	494	
HEPA	0F-171A,B	Glass fiber, steel frame	24x24x12	18	72
Reactor Bldg. Backwash Tank Exhaust Filters					
for RWCU System	1F-211/2F-211		12x12x12	2	2
for Fuel Pool Clean-up System	0F-211		24x24x12	1	4

TABLE 11.4-2

DRY SOLID WASTE AMOUNT FROM GAS, LAUNDRY DRAIN FILTERS, AND MAINTENANCE ACTIVITIES

SOURCE	EQUIPMENT NOS.	WASTE TYPE	SIZE (INCHES)	EXPECTED NUMBER/ WEIGHT OF FILTER UNITS PER YEAR	EXPECTED ANNUAL VOLUME UNCOMPACTED (FT^3)
Hydraulic Compactor Dust Hood Vent Filter					
Pre	0F-308	Glass fiber, plywood frame	12x12x2	12	2
HEPA		Glass fiber, plywood frame	12x12x12	12	12
Laundry Drain Processing					
Cartridge	0F-313A,B	Epoxy impreg. cellulose fiber steel frame	7 dia.x18-3/16	150	60
Condensate Filters	1F-135A-G 2F-135A-G	Plastic Fiber, plastic frame-polyprophylene or equivalent	2 5/8" ODx1.1Dx10"L cartridges. equivalent effective length = 50"	2028	400
Mechanical/cartridge filters from under-water vacuum operations, CRD's & CRD Liftpumps and other non-permanent liquid filtration systems					900
Metal (debris, piping, supports, etc. from modification and maintenance activities)				---	3000
Disposable/Consumable Material (plastic, tape, cardboard, paper, trash, protective clothing, laydown material, tarps, etc.)				---	8000

TABLE 11.4-3

SOLID WASTE MANAGEMENT SYSTEM COMPONENT DESCRIPTION

A. PUMPS	EQUIPMENT NUMBERS	TYPE	QUANTITY	MATERIAL	CAPACITY EACH, gpm	TDH ft	USAGE FACTOR NORMAL	DRIVER hp	DESIGN PRESSURE/TEMP psig/°F
SOLID RADWASTE COLLECTION									
Spent Resin Transfer	OP-320	Prog. Cavity (Moyno)	1	SS	200	354	0.076	60	150/140
RW Cleanup Phase Sep. Decant	OP-336	Horiz. Centr.	1	SS	50	155	0.0047	10	150/155
RW Cleanup Phase Sep. Sludge Disch.	OP-334	Horiz. Centr.	1	SS	200	272	0.00067	30	150/155
Waste Sludge Phase Sep. Decant	OP-331	Horiz. Centr.	1	SS	50	155	0.49	10	150/155
Waste Sludge Phase Sep. Disch.	OP-332	Horiz. Centr.	1	SS	200	272	0.013	30	150/155
Cond. Demin Regen. Waste Transfer	1P-106A,B	Inline Centr.	2	SS	85	53	0.0	3	150/105
Cond. Demin. Regen. Waste Transfer	2P-106A,B	Inline Centr.	2	SS	85	150	0.0	10	150/105
CFS BWRT Transfer	1P-2P-191	Centrifugal	2	316 SS	120	145	0.0341	7.5	150/125
U1/U2 Polymer Injection	OP-176A/B	Pos Displ	2	Cast Iron	11 gph @100 psi	350 psi max	0.0334	1/3	135/105
WSPS Chemical Injection	OP-347A/B	Pos Displ	2	Cast Iron	11 gph @100 psi	350 psi max	0.012	1/3	135/105
RADWASTE SOLIDIFICATION									
Process Feed	OP-304A,B	Prog. Cavity (Moyno)	2	SS/BUNA N	20	(Setp. 3 psig)	0.00023	2	125/180
Mixing	OP-307A,B	Prog. Cavity (Moyno)	2	SS/BUNA N 32	32	(Setp. 3 psig)	0.0	3	125/180
Sodium Silicate	OP-309	Prog. Cavity (Moyno)	1	CS/BUNA N	2.9	(Setp. 180 psig)	0.0	1/2	125/105
Cement Feed *	OS-305	Rotary Valve	1	C.I.	110 lb/min.		0.0	1	
RWSS Sample Pump	OP-306	Prog. Cavity	1	SS	30	150 psi	0.0065	10	180/140

* COMPONENT ISOLATED FROM THE RWSS AND NO LONGER SERVES ANY RWSS FUNCTIONS.

TABLE 11.4-3

SOLID WASTE MANAGEMENT SYSTEM COMPONENT DESCRIPTION

B. TANKS	EQUIPMENT NUMBERS	TYPE	QUANTITY	MATERIAL	LIVE/NOMINAL CAPACITY, EACH gal.		DESIGN PRESSURE/TEMP. psig/□F
SOLID RADWASTE COLLECTION							
Spent Resin	OT-324	Vert. Cyl.	1	SS	6200/7500		Atmos./200
RW Cleanup Backw. Receiving	1T-225	Vert. Cyl.	1	SS	2450/3000		Atmos./200
RW Cleanup Backw. Receiving	2T-225	Vert. Cyl.	1	SS	2450/3000		Atmos./200
Fuel Pool Backw. Receiving	OT-203	Vert. Cyl.	1	SS	1900/2500		Atmos./200
CFS Backw. Receiving	1T-187	Horiz. Cyl.	1	SS	9243/10920 (Max.)		Atmos./150
CFS Backw. Receiving	2T-187	Horiz. Cyl.	1	SS	9243/10920 (Max.)		Atmos./150
CFS Polymer Injection	0T-137	Vert. Cyl.	1	SS	250		Atmos./100
CFS Chemical Injection	0T-361	Vert. Cyl.	1	SS	100		Atmos./100
Cond. Demin. Regen. Waste Surge	1T-106A,B	Vert. Cyl.	2	CS	2800/3390		Atmos./105
Cond. Demin. regen. Waste Surge	2T-106A,B	Vert. Cyl.	2	CS	2800/3300		Atmos./105
RW Cleanup Phase Separator	OT-318A,B	Vert. Cyl.	2	SS	3100 Supern./898 Sludge/7400		Atmos./200
Waste Sludge Phase Separator	OT-331	Vert. Cyl.	1	SS	7330 Supern./898 Sludge/10780		Atmos./200
RADWASTE SOLIDIFICATION							
Waste Mixing	OT-307A,B	Vert. Cyl.	2	SS	700/750		Atmos./200
Cement Silo	OT-306	Vert. Cyl.	1	CS	3950/4325		Atmos./100
Sodium Silicate	OT-309	Vert. Cyl.	1	CS	1150/1270		Atmos./200
HSA Waste Container			As Required	CS	Up to 100 ft^3		Per DOT 7A
LSA Waste Container			As Required	CS	Up to 200 ft^3		Atmos. 250
C. MISC. EQUIPMENT	**EQUIPMENT NUMBERS**	**TYPE**	**QUANTITY**	**MATERIAL**	**CAPACITY**	**hp**	
Radwaste Building Crane	OH-301	Bridge & Trolley	1	CS	25 tons 48 ft. 2 in. lift 166 ft. 2 in. runway 34 ft. 2 in. span	Hoist Motor: 30 Bridge Motor: 3 Trolley Motor: 1.5 Grab Motor: 0.25 Trans T1: 1500VA Trans T2: 500VA Trans T3: 1500VA	
Waste Compactor	OS-313	Vert. Hydr. Piston	1	CS/SS	10 tons 14-1/2 in. Stroke	5	
Waste Container - Transfer Cart.	OS-315	Rail, Cable	1	CS	15 tons 63.75 ft. Travel	1	
Waste Container Storage Compartments						n/a	
Large	-	(Top Entry)	12	Concr.	For 6 ft. Cyl. Cont.		
Small	-	(With Lid)	4	Concr.	For 4x4 ft. Cyl. Cont. (if required)	n/a	

TABLE 11.4-4

SOLID WASTE MANAGEMENT SYSTEM FLOWS
(Refer to Figure 11.4-3)

No.	Stream (Note 4)	Averaged Number and Frequency of Batch (Notes 1 and 2)	Total Volume/ Batch (gallon)	Solids Volume/ Batch (ft^3)	Flowrate (gpm)	Maximum Expected Number and Frequency of Batch (Note 3)	Notes
1a	To Spent Resin Tank (From Radwaste Demin	1/69 days	2000	160	100	58 days	Total Volume is 1 demin and 40 gpm of sluice water for 20 minutes
1b	To Spent Resin Tank (From Cond Demins)	77.5 days	3800	300	150	29 days	Total Volume is 150 gpm for 25 minute transfer.
2	To Mobile Processing System (From Spent Resin Tank)	48 days	8000	300	10-50	23 days	Total Volume is one tank and 30 gpm sluice water for 60 minutes
3a	To Waste Sludge Phase Separator (CFS Backwash Receiving Tank)	14/21 days	5,056	1.0	35	5/7 day	Total Volume is from Table 11.2-2.
3b	To Waste Sludge Phase Separator (From LRW Filters Drain)	1/4.2 days	500	1.0	By gravity	1/3.7 days	Total Volume is from Table 11.2-2.

TABLE 11.4-4

SOLID WASTE MANAGEMENT SYSTEM FLOWS
(Refer to Figure 11.4-3)

No.	Stream (Note 4)	Averaged Number and Frequency of Batch (Notes 1 and 2)	Total Volume/ Batch (gallon)	Solids Volume/ Batch (ft^3)	Flowrate (gpm)	Maximum Expected Number and Frequency of Batch (Note 3)	Notes
4	To Mobile Processing System(From Waste Sludge Phase Separator)	1/426 days	9550	160	10-50	1/384 days	Total Volume is tank live capacity including 700 gallons of flush water
5	To RWCU Phase Sep. (From Fuel Pool Backwash Tank)	1/120 days	1450	5.0	By gravity	1/60 days	Total Volume is from Table 11.2-2.
6	To RWCU Phase Sep. (From RWCU Backwash Tanks)	1/4.5 days	1470	2.7	By gravity	1/4.1 days	Total Volume is from Table 11.2-2.
7	To Mobile Processing System (From RWCU Phase Separator)	1/340 days	6300	200	10-50	1/270 days	Total Volume is tank live capacity including 440 gallons of flush water.
8	To Waste Mixing Tanks(From Radwaste Filters Backwash)	1/4.2 days	210	8.0	By gravity	1/3.9 days	Total Volume is 30 % of waste mixing tank live capacity

TABLE 11.4-4

SOLID WASTE MANAGEMENT SYSTEM FLOWS
(Refer to Figure 11.4-3)

No.	Stream (Note 4)	Averaged Number and Frequency of Batch (Notes 1 and 2)	Total Volume/ Batch (gallon)	Solids Volume/ Batch (ft^3)	Flowrate (gpm)	Maximum Expected Number and Frequency of Batch (Note 3)	Notes
9	To Mobile Processing System (From Waste Mixing Tanks)	1/8.4 days	560	16	10-50	1/7.8 days	Total Volume is80 % of waste mixing tank live capacity
10	Large Waste Container For Disposal (From Mobile Processing System)	1/28 days	---	175	---	1/13 days	Container disposal volume (size) is about 200 ft3.
11	Medium Waste Container For Disposal (From Mobile Processing System)	1/70 days	---	150	---	1/61 days	Container disposal volume (size) is about 174 ft3.
12	Small Waste Container For Disposal (From Mobile Processing System)	1/170 days	---	100	---	1/135 days	Container disposal volume (size) is about 132 ft3.
13	To Vendor Processing(DAW from Cargo Containers)	---	---	Various cargo container sizes	---	---	About 17,000 ft3/year DAW shipped to vendors per Table 11.4-2

TABLE 11.4-4

SOLID WASTE MANAGEMENT SYSTEM FLOWS
(Refer to Figure 11.4-3)

No.	Stream (Note 4)	Averaged Number and Frequency of Batch (Notes 1 and 2)	Total Volume/ Batch (gallon)	Solids Volume/ Batch (ft^3)	Flowrate (gpm)	Maximum Expected Number and Frequency of Batch (Note 3)	Notes

NOTES:

1) Batch Frequencies are from total number of equipment.

2) The Averaged Number and Frequency of Batch is based on the Expected Yearly Volume (ft3) from Table 11.4-1.

3) The Max. Expected Number and Frequency of Batch is based on the Max Expected Yearly Volume (ft3) from Table 11.4-1.

4) A stream is not included specifically in this table because it is not part of the solid waste management system. This stream is for solid waste disposal from the mobile liquid processing system used for treating chemical liquid waste. The waste is transferred directly from the mobile system to a HIC within the vendor supplied equipment system. The Large Waste Container is typically used for this waste stream. In 1996, about 60 ft3 of charcoal was disposed of from the mobile processing system. Bead resin may also be used in the mobile processing system.

5) The container disposal volumes in the notes and corresponding Solids Volume/Batch (gallon) reflect present processing vendor sizes. These container sizes are considered typical, and may vary some between vendors.

6) The Large Waste Container is typically used for disposal of bead resin from the radwaste demineralizer and condensate demineralizers transferred to the spent resin tank. The Medium Waste Container is typically used for disposal of powdered resin, bead resin fines, and crud from the LRW filters transferred to the waste mixing tanks and WSPS. The Small Waste Container is typically used for disposal of powdered resin and crud from the RWCU Filter/Demineralizers and Fuel Pool Filter/Demineralizers transferred to the RWCU phase separators.

TABLE 11.4-5

EXPECTED INVENTORIES OF RADIOACTIVE MATERIALS IN COMPONENTS OF THE SOLID WASTE MANAGEMENT SYSTEM[2]

ISOTOPE	FP F/D BACKWASH RECEIVING TANK (CURIES)	RWCU F/D BACKWASH RECEIVING TANK (CURIES)	RWCU PHASE SEPARATOR (CURIES)	CFS BACKWASH RECEIVING TANK (CURIES)	LIQUID RADWASTE FILTER (DRAIN) 1/9 ACTIVITY (CURIES)	WASTE SLUDGE PHASE SEPARATOR (CURIES)	LIQUID RADWASTE FILTER 8/9 ACTIVITY (CURIES)	WASTE MIXING TANK (CURIES)	SPENT RESIN TANK (CURIES)	CFS FILTER (CURIES)
Na-24	4.93E-01[1]	4.83E+00	2.03E+01	1.22E+00	9.30E-02	1.31E+00	7.44E-01	7.44E-01	-	1.22E+00
P-32	2.14E-01	1.65E+00	9.05E+00	1.15E+00	4.05E-02	1.19E+00	3.24E-01	1.09E-01	-	1.15E+00
Cr-51	1.07E+01	6.61E+01	5.22E+02	7.59E+01	2.35E+00	7.82E+01	1.88E+01	3.56E+00	-	7.59E+01
Mn-54	2.57E-01	1.06E+00	3.86E+01	7.10E+00	1.90E-01	7.29E+00	1.52E+00	4.57E-02	-	7.10E+00
Mn-56	4.52E-01	4.37E+00	1.84E+01	1.11E+00	8.54E-02	1.20E+00	6.84E-01	6.84E-01	-	1.11E+01
Fe-55	3.89E+00	1.55E+01	7.19E+02	1.38E+02	3.63E+00	1.41E+02	2.90E+01	6.57E-01	-	1.38E+02
Fe-59	6.98E-02	3.75E-01	4.16E+00	6.44E-01	1.90E-02	6.63E-01	1.52E-01	1.86E-02	-	6.44E-01
Co-58	5.61E-01	2.72E+00	4.31E+01	7.02E+00	1.98E-01	7.22E+00	1.58E+00	1.27E-01	-	7.02E+00
Co-60	1.58E+00	6.23E+00	3.07E+02	5.93E+01	1.56E+00	6.08E+01	1.24E+01	2.63E-01	-	5.93E+01
Ni-65	2.65E-03	2.56E-02	1.08E-01	6.51E-03	5.00E-04	7.01E-03	4.00E-03	4.00E-03	-	6.51E-03
Cu-64	1.25E+00	1.22E+01	5.13E+01	3.10E+00	2.37E-01	3.33E+00	1.89E+00	1.89E+00	-	3.10E+00
Zn-65	7.18E-01	3.01E+00	1.00E+02	1.82E+01	4.87E-01	1.86E+01	3.89E+00	1.30E-01	-	1.82E+01
Sr-90	2.80E-02	1.09E-01	5.68E+00	-	-	-	-	-	2.45E-01	-
Y-93	1.33E-01	1.31E+00	5.51E+00	-	-	-	-	-	6.30E-02	-
Nb-98	1.19E-05	0.0	2.40E-05	-	-	-	-	-	1.42E-18	-
I-134	2.31E-01	2.15E+00	9.06E+00	-	-	-	-	-	4.25E+00	-
Cs-134	1.15E-01	4.63E-01	2.07E+01	-	-	-	-	-	6.03E-01	-
Cs-136	7.91E-02	6.31E-01	3.34E+00	-	-	-	-	-	3.82E-02	-
Cs-138	-	-	-	-	-	-	-	-	3.48E-02	
Ba-139	5.03E-02	4.79E-01	2.02E+00	5.79E-02	4.37E-03	6.23E-02	3.49E-02	3.48E-02	2.34E-02	5.79E-02
W-187	2.32E-02	2.28E-01	9.59E-01	2.24E-01	1.71E-02	2.41E-01	1.37E-01	1.37E-01	-	2.24E-01
Zn-69m	9.04E-02	8.85E+01	3.72E+00	2.24E-01	1.71E-02	2.41E-01	1.37E-01	1.37E-01	-	2.24E-01
Zn-69	9.04E-02	8.88E+01	3.73E+00	-	-	-	-	-	-	
Br-83	3.01E-02	2.90E-01	1.22E+00	-	-	-	-	-	4.51E-01	-
Sr-89	2.47E-01	1.28E+00	1.56E+01	-	-	-	-	-	2.34E-01	-
Y-89m	3.70E-05	1.93E-04	2.35E-03	-	-	-	-	-	2.74E-05	-
Sr-92	9.46E-02	9.16E-01	3.85E+00	-	-	-	-	-	4.45E-02	-
Y-92	1.68E-01	1.64E+00	6.90E+00	-	-	-	-	-	7.93E-02	-
Ru-103	4.40E-02	2.44E-01	2.46E+00	-	-	-	-	-	2.86E-02	-

TABLE 11.4-5

EXPECTED INVENTORIES OF RADIOACTIVE MATERIALS IN COMPONENTS OF THE SOLID WASTE MANAGEMENT SYSTEM(2)

ISOTOPE	FP F/D BACKWASH RECEIVING TANK (CURIES)	RWCU F/D BACKWASH RECEIVING TANK (CURIES)	RWCU PHASE SEPARATOR (CURIES)	CFS BACKWASH RECEIVING TANK (CURIES)	LIQUID RADWASTE FILTER (DRAIN) 1/9 ACTIVITY (CURIES)	WASTE SLUDGE PHASE SEPARATOR (CURIES)	LIQUID RADWASTE FILTER 8/9 ACTIVITY (CURIES)	WASTE MIXING TANK (CURIES)	SPENT RESIN TANK (CURIES)	CFS FILTER (CURIES)
Rh-103m	4.40E-02	2.44E-01	2.46E+00	-	-	-	-	-	2.86E-02	-
Ru-106	1.12E-02	4.56E-02	1.75E+00	-	-	-	-	-	3.74E-02	-
Rh-106	1.12E-02	4.56E-02	1.75E+00	-	-	-	-	-	3.74E-02	-
Te-132	2.51E-03	2.46E-02	1.04E-01	-	-	-	-	-	1.19E-03	-
I-132	2.96E-01	2.85E+00	1.20E+01	-	-	-	-	-	4.39E+00	-
Cs-137	7.95E-02	3.12E-01	1.62E+01	-	-	-	-	-	7.04E-01	-
Ba-137m	7.48E-02	2.94E-01	1.53E+01	-	-	-	-	-	6.63E-01	-
Ba-140	3.86E-01	3.10E+00	1.62E+01	-	-	-	-	-	1.86E-01	-
La-140	3.85E-01	2.99E+00	1.62E+01	-	-	-	-	-	1.86E-01	-
La-142	2.87E-02	2.74E-01	1.15E+00	-	-	-	-	-	1.34E-02	-
Ce-143	3.20E-03	3.14E-02	1.32E-01	-	-	-	-	-	1.52E-03	-
Pr-143	4.36E-02	3.44E-01	1.85E+00	-	-	-	-	-	2.12E-02	-
Nd-147	2.51E-03	2.11E-02	1.04E-01	-	-	-	-	-	1.20E-03	-
Pm-147	1.07E-04	2.95E-04	2.38E-02	-	-	-	-	-	7.76E-04	-
Np-239	1.26E+00	1.25E+01	5.25E+01	-	-	-	-	-	6.01E-01	-
Pu-239	7.16E-06	2.62E-05	1.54E-03	-	-	-	-	-	6.89E-05	-
Sr-91	1.25E-01	1.23E+00	5.17E+00	-	-	-	-	-	5.92E-02	-
Y-91m	7.21E-02	7.08E-01	2.98E+00	-	-	-	-	-	3.41E-02	-
Y-91	1.77E-01	8.90E-01	1.22E+01	-	-	-	-	-	1.44E-01	-
Zr-95	2.17E-02	1.07E-01	1.57E+00	-	-	-	-	-	1.86E-02	-
Nb-95m	1.61E-04	7.05E-04	1.22E-02	-	-	-	-	-	1.45E-04	-
Nb-95	2.79E-02	1.22E-01	2.40E+00	-	-	-	-	-	2.88E-02	-
Mo-99	4.24E-01	4.16E+00	1.75E+01	-	-	-	-	-	2.01E-01	-
Tc-99m	7.71E-01	7.55E+00	3.17E+01	-	-	-	-	-	3.64E-01	-
Tc-99	5.37E-07	2.01E-06	1.14E-04	-	-	-	-	-	5.09E-06	-
Ru-105	3.03E-02	2.95E-01	1.24E+00	-	-	-	-	-	1.43E-02	-
Rh-105m	6.36E-03	6.21E-02	2.61E-01	-	-	-	-	-	3.00E-03	-
Rh-105	3.03E-02	2.98E-01	1.25E+00	-	-	-	-	-	1.43E-02	-
Te-129m	8.03E-02	4.67E-01	4.20E+00	-	-	-	-	-	4.87E-02	-

TABLE 11.4-5

EXPECTED INVENTORIES OF RADIOACTIVE MATERIALS IN COMPONENTS OF THE SOLID WASTE MANAGEMENT SYSTEM(2)

ISOTOPE	FP F/D BACKWASH RECEIVING TANK (CURIES)	RWCU F/D BACKWASH RECEIVING TANK (CURIES)	RWCU PHASE SEPARATOR (CURIES)	CFS BACKWASH RECEIVING TANK (CURIES)	LIQUID RADWASTE FILTER (DRAIN) 1/9 ACTIVITY (CURIES)	WASTE SLUDGE PHASE SEPARATOR (CURIES)	LIQUID RADWASTE FILTER 8/9 ACTIVITY (CURIES)	WASTE MIXING TANK (CURIES)	SPENT RESIN TANK (CURIES)	CFS FILTER (CURIES)
Te-129	5.04E-02	2.93E-01	1.27E+00	-	-	-	-	-	3.06E-02	-
I-129	4.64E-10	9.32E-10	6.37E-08	-	-	-	-	-	8.31E-09	-
Te-131m	1.94E-02	1.90E-01	8.01E-01	-	-	-	-	-	9.17E-03	-
Te-131	2.15E-03	2.12E-02	8.91E-02	-	-	-	-	-	1.02E-03	-
I-131	1.11E+00	1.01E+01	4.60E+01	-	-	-	-	-	1.35E+01	-
I-133	1.68E+00	1.66E+01	6.94E+01	-	-	-	-	-	2.06E+01	-
I-135	5.87E-01	5.73E+00	2.41E+01	-	-	-	-	-	7.74E+00	-
Ce-141	5.90E-02	3.47E-01	3.05E+00	-	-	-	-	-	3.53E-02	-
Ce-144	1.08E-02	4.53E-02	1.60E+00	-	-	-	-	-	3.00E-02	-
Pr-144m	1.56E-04	6.47E-04	2.90E-03	-	-	-	-	-	4.29E-04	-
Pr-144	1.08E-02	4.52E-02	2.03E-01	-	-	-	-	-	3.00E-02	-
Rb-87	-	-	-	-	-	-	-	-	4.91E-13	-
Rb-88	-	-	-	-	-	-	-	-	8.33E-03	-
Rb-89	-	-	-	-	-	-	-	-	5.11E-02	-
Cs-135	-	-	-	-	-	-	-	-	1.10E-08	-
TOTAL	2.96E+01	2.02E+02	2.28E+03	3.13E+02	8.92E+00	3.22E+02	7.13E+01	8.55E+00	5.60E+01	3.13E+02

NOTES:

(1) 4.93E-01 = 4.93×10^{-1}

(2) Noble gases are not included in tank inventories because they are assumed to escape from solution and are continuously vented to the Radwaste Building ventilation system.

TABLE 11.4-6

DESIGN INVENTORIES OF RADIOACTIVE MATERIALS IN COMPONENTS OF THE SOLID WASTE MANAGEMENT SYSTEM (Curies) [(2)]

ISOTOPE	FP F/D BACKWASH RECEIVING TANK (CURIES)	RWCU F/D BACKWASH RECEIVING TANK (CURIES)	RWCU PHASE SEPARATOR (CURIES)	CFS BACKWASH RECEIVING TANK (CURIES)	LIQUID RADWASTE FILTER (DRAIN) 1/9 ACTIVITY (CURIES)	WASTE SLUDGE PHASE SEPARATOR (CURIES)	LIQUID RADWASTE FILTER 8/9 ACTIVITY (CURIES)	WASTE MIXING TANK (CURIES)	SPENT RESIN TANK (CURIES)	CFS FILTER (CURIES)
Na-24	2.57E-02[(1)]	1.57E+00	6.33E+00	2.49E-01	3.15E-02	2.80E-01	2.52E-01	2.52E-01	2.73E-01	2.49E-01
P-32	5.69E-03	2.76E-01	1.45E+00	1.20E-01	7.06E-03	1.27E-01	5.65E-02	1.89E-02	6.22E-02	1.20E-01
Cr-51	2.38E-01	9.20E+00	7.01E+01	6.60E+00	3.40E-01	6.94E+00	2.72E+00	5.16E-01	2.92E+00	6.60E+00
Mn-54	3.91E-02	1.01E-00	3.65E+01	4.20E+00	1.88E-01	4.39E+00	1.50E+00	4.53E-02	7.17E-01	4.20E+00
Mn-56	1.10E-01	6.68E+00	2.69E+01	1.06E+00	1.36E-01	1.19E+00	1.08E+00	1.08E+00	1.17E+00	1.06E+00
Fe-59	4.98E-02	1.67E+00	1.80E+01	1.80E+00	8.74E-02	1.88E+00	6.99E-01	8.58E-02	6.83E-01	1.80E+00
Co-58	3.74E+00	1.13E+02	1.75E+03	1.83E+02	8.58E+00	1.91E+02	6.86E+01	5.48E+00	5,72E+01	1.83E+02
Co-60	5.26E-01	1.30E+01	6.36E+02	7.64E+01	3.37E+00	7.97E+01	2.70E+01	5.71E-01	1.02E+01	7.64E+01
Ni-65	6.45E-04	3.92E-02	1.58E-01	6.20E-03	7.92E-04	6.99E-03	6.34E-03	6.34E-03	6.86E-03	6.20E-03
Zn-65	1.91E-03	5.02E-02	1.65E+00	1.88E-01	8.45E-03	1.96E-01	6.76E-02	2.26E-03	3.45E-02	1.88E-01
Br-84	1.22E-02	7.12E-01	2.87E+00	-	-	-	-	-	1.30E-01	-
Sr-90	2.45E-01	6.01E+00	3.09E+02	-	-	-	-	-	4.77E+00	-
Tc-101	2.83E-02	1.54E+00	6.22E+00	-	-	-	-	-	3.00E-01	-
I-134	1.79E-01	1.07E+01	4.32E+01	-	-	-	-	-	1.90E+00	-
Cs-134	1.65E-01	4.13E+00	1.83E+02	-	-	-	-	-	3.14E+00	-
Cs-136	2.90E-02	1.45E+00	7.35E+00	-	-	-	-	-	3.14E-01	-
Cs-138	8.70E-02	5.07E+00	2.05E+01	-	-	-	-	-	9.25E-01	-
Ba-139	1.89E-01	1.14E+01	4.60E+01	-	-	-	-	-	2.01E-00	-
Zn-69m	3.52E-04	2.16E-02	8.71E-02	3.41E-03	4.33E-04	3.84E-03	3.46E-03	3.46E-03	3.74E-03	3.41E-03
Zn-69	3.52E-04	2.16E-02	8.71E-02	3.41E-03	4.32E-04	3.85E-03	3.45E-03	3.45E-03	3.74E-03	3.41E-03
Br-83	3.06E-02	1.86E+00	7.50E+00	-	-	-	-	-	3.26E+01	-
Sr-89	2.04E+00	6.65E+01	7.88E+02	-	-	-	-	-	2.89E+01	-
Y-89m	3.05E-04	9.97E-03	1.18E-01	-	-	-	-	-	4.34E-03	-
Sr-92	2.55E-01	1.55E+01	6.25E+01	-	-	-	-	-	2.70E+00	-
Y-92	2.55E-01	1.55E+01	6.29E+01	-	-	-	-	-	2.70E+00	-
Ru-103	1.11E-02	3.86E-01	3.78E+00	-	-	-	-	-	1.48E-01	-
Rh-103m	1.11E-02	3.86E-01	3.78E+00	-	-	-	-	-	1.49E-01	-
Ru-106	2.58E-03	6.61E-02	2.51E+00	-	-	-	-	-	4.77E-02	-
Rh-106	2.58E-03	6.61E-02	2.51E+00	-	-	-	-	-	4.77E-02	-
Ag-110m	5.74E-02	1.51E+00	5.00E+01	5.70E+00	2.56E-01	5.95E+00	2.05E+00	6.79E-02	1.04E+00	5.70E+00
Ag-110	7.64E-04	2.00E-02	6.65E-01	7.57E-02	3.41E-03	7.92E-02	2.73E-02	9.02E-04	1.38E-02	7.57E-02
Te-129m	2.15E-02	7.80E-01	6.80E+00	-	-	-	-	-	2.75E-01	-

TABLE 11.4-6

DESIGN INVENTORIES OF RADIOACTIVE MATERIALS IN COMPONENTS OF THE SOLID WASTE MANAGEMENT SYSTEM (Curies) (2)

ISOTOPE	FP F/D BACKWASH RECEIVING TANK (CURIES)	RWCU F/D BACKWASH RECEIVING TANK (CURIES)	RWCU PHASE SEPARATOR (CURIES)	CFS BACKWASH RECEIVING TANK (CURIES)	LIQUID RADWASTE FILTER (DRAIN) 1/9 ACTIVITY (CURIES)	WASTE SLUDGE PHASE SEPARATOR (CURIES)	LIQUID RADWASTE FILTER 8/9 ACTIVITY (CURIES)	WASTE MIXING TANK (CURIES)	SPENT RESIN TANK (CURIES)	CFS FILTER (CURIES)
Te-129	1.35E-02	4.90E-01	4.27E+00	-	-	-	-	-	1.73E-01	-
Te-132	3.28E+00	2.01E+02	8.12E+02	-	-	-	-	-	3.48E+01	-
I-132	3.51E+00	2.15E+02	8.68E+02	-	-	-	-	-	3.73E+01	-
I-131	2.14E+00	1.22E+02	5.32E+02	-	-	-	-	-	2.27E+01	-
Cs-137	2.55E-01	6.27E+00	3.23E+02	-	-	-	-	-	4.97E+00	-
Ba-137m	2.41E-01	5.93E+00	3.06E+02	-	-	-	-	-	4.71E+00	-
Ba-140	2.31E+00	1.16E+02	5.83E+02	-	-	-	-	-	2.51E+01	-
La-140	2.31E+00	1.12E+02	5.84E+02	-	-	-	-	-	2.51E+01	-
Ba-142	2.59E-02	1.36E+00	5.49E+00	-	-	-	-	-	2.75E-01	-
La-142	2.59E-02	1.59E+00	6.41E+00	-	-	-	-	-	2.75E-01	-
Ce-143	9.83E-04	6.05E-02	2.44E-01	-	-	-	-	-	1.05E-02	-
Pr-143	1.12E-02	5.54E-01	2.86E+00	-	-	-	-	-	1.22E-01	-
Nd-147	3.12E-03	1.64E-01	7.79E-01	-	-	-	-	-	3.34E-02	-
Pm-147	1.32E-04	2.30E-03	1.84E-01	-	-	-	-	-	2.83E-03	-
W-187	6.10E-02	3.74E+00	1.51E+01	5.95E-01	7.46E-02	6.70E-01	5.97E-01	5.95E-01	6.48E-01	5.95E-01
Re-187	1.82E-13	4.33E-12	2.36E-10	2.87E-11	1.26E-12	2.99E-11	1.01E-11	1.64E-13	3.58E-12	2.87E-11
Np-239	1.15E+01	7.10E+02	2.86E+03	-	-	-	-	-	1.23E+02	-
Pu-239	6.52E-05	1.49E-03	8.76E-02	-	-	-	-	-	1.30E-03	-
Br-85	6.93E-04	2.37E-02	9.62E-02	-	-	-	-	-	7.36E-03	-
Sr-91	5.59E-01	3.42E+01	1.38E+02	-	-	-	-	-	5.92E+00	-
Y-91m	3.21E-01	1.97E+01	7.94E+01	-	-	-	-	-	3.40E+00	-
Y-91	3.24E-01	1.01E+01	1.37E+02	-	-	-	-	-	5.17E+00	-
Zr-95	2.89E-02	8.94E-01	1.28E+01	-	-	-	-	-	4.33E-01	-
Nb-95m	2.14E-04	5.89E-03	9.96E-02	-	-	-	-	-	3.32E-03	-
Nb-95	3.83E-02	1.06E+00	2.00E+01	-	-	-	-	-	6.25E-01	-
Zr-97	4.63E-04	2.83E-02	1.14E-01	-	-	-	-	-	4.91E-03	-
Nb-97m	4.38E-04	2.68E-02	1.08E-01	-	-	-	-	-	4.65E-03	-
Nb-97	4.63E-04	2.83E-02	1.14E-01	-	-	-	-	-	4.91E-03	-
Mo-99	1.24E+00	7.60E+01	3.07E+02	-	-	-	-	-	1.32E+01	-
Tc-99m	2.53E+00	1.55E+02	6.25E+02	-	-	-	-	-	2.69E+01	-
Tc-99	1.74E-06	4.07E-05	2.27E-03	-	-	-	-	-	3.44E-05	-
I-133	1.58E+00	9.69E+01	3.91E+02	-	-	-	-	-	1.68E+01	-

TABLE 11.4-6

DESIGN INVENTORIES OF RADIOACTIVE MATERIALS IN COMPONENTS OF THE SOLID WASTE MANAGEMENT SYSTEM (Curies) (2)

ISOTOPE	FP F/D BACKWASH RECEIVING TANK (CURIES)	RWCU F/D BACKWASH RECEIVING TANK (CURIES)	RWCU PHASE SEPARATOR (CURIES)	CFS BACKWASH RECEIVING TANK (CURIES)	LIQUID RADWASTE FILTER (DRAIN) 1/9 ACTIVITY (CURIES)	WASTE SLUDGE PHASE SEPARATOR (CURIES)	LIQUID RADWASTE FILTER 8/9 ACTIVITY (CURIES)	WASTE MIXING TANK (CURIES)	SPENT RESIN TANK (CURIES)	CFS FILTER (CURIES)
I-135	7.34E-01	4.49E+01	1.81E+02	-	-	-	-	-	7.80E+00	-
Ba-141	4.42E-02	2.48E+00	1.00E+01	-	-	-	-	-	4.70E-01	-
La-141	4.42E-02	2.71E+00	1.09E+01	-	-	-	-	-	4.70E-01	-
Ce-141	5.55E-02	2.03E+00	1.74E+01	-	-	-	-	-	7.06E-01	-
Ce-144	3.38E-02	8.83E-01	3.08E+01	-	-	-	-	-	6.20E-01	-
Pr-144m	4.84E-04	1.26E-02	5.14E-02	-	-	-	-	-	4.80E-03	-
Pr-144	3.38E-02	8.83E-01	3.60E+00	-	-	-	-	-	3.36E-01	-
Rb-88	-	-	-	-	-	-	-	-		-
Rb-89	-	-	-	-	-	-	-	-		-
Cs-139	-	-	-	-	-	-	-	-		-
Rb-90	-	-	-	-	-	-	-	-		-
Y-90	-	-	-	-	-	-	-	-		-
Rb-92	-	-	-	-	-	-	-	-		-
Rb-93	-	-	-	-	-	-	-	-		-
Sr-93	-	-	-	-	-	-	-	-		-
Y-93	-	-	-	-	-	-	-	-		-
Rb-94	-	-	-	-	-	-	-	-		-
Sr-94	-	-	-	-	-	-	-	-		-
Y-94	-	-	-	-	-	-	-	-		-
Cs-140	-	-	-	-	-	-	-	-		-
Cs-142	-	-	-	-	-	-	-	-		-
TOTAL	4.17E+01	2.23E+03	1.30E+04	2.80E+02	1.31E+01	2.93E+02	1.05E+02	8.74E+00	4.85E+02	2.80E+02

NOTES: (1) 2.57E-02 = 2.57×10^{-2}

(2) Noble gases are not included in tank inventories because they are assumed to escape from solution and are continuously vented to the Radwaste Building ventilation system.

Table 11.4-7
This Table Has Been Deleted

Table 11.4-8
This Table Has Been Deleted

TABLE 11.4-9

EXPECTED INVENTORIES OF RADIOACTIVE MATERIALS IN WASTE SHIPPING CASKS[1,2]

ISOTOPE	LARGE CONTAINER 200 FT^3 (SHIPPING)/ 175 FT^3 (RESIN)		MEDIUM CONTAINER 174 FT^3 (SHIPPING)/ 150 FT^3 (RESIN)		SMALL CONTAINER 132 FT^3 (SHIPPING)/ 100 FT^3 (RESIN)	VERY SMALL CONTAINER 51 FT^3 (SHIPPING)/ 39 FT^3 (RESIN)
	CONDENSATE DEMIN. RESIN (175/300 FT^3) (CURIES)	RADWASTE DEMIN. RESIN (175/160 FT^3) (CURIES)	WASTE MIXING TANK (150/16 FT^3) (CURIES)	WASTE SLUDGE PHASE SEPARATOR (150/164 FT^3) (CURIES)	RWCU PHASE SEPARATOR (100/155.4 FT^3) (CURIES)	RWCU PHASE SEPARATOR (39/155.4 FT^3) (CURIES)
Na-24	-	9.17E-01[1]	6.97E+00	1.20E+00	1.31E+01	5.09E+00
P-32	-	3.94E-01	1.02E-00	1.09E+00	5.82E+00	2.27E+00
Cr-51	-	1.95E+01	3.34E+01	7.15E+01	3.36E+02	1.31E+02
Mn-54	-	4.43E-01	4.28E-01	6.67E+00	2.48E+01	9.69E+00
Mn-56	-	8.40E-01	6.41E+00	1.10E+00	1.18E+01	4.62E+00
Fe-55	-	6.66E+00	6.16E+00	1.29E+02	4.63E+02	1.80E+02
Fe-59	-	1.26E-01	1.74E-01	6.06E-01	2.68E+00	1.04E+00
Co-58	-	9.93E-01	1.19E+00	6.60E+00	2.77E+01	1.08E+01
Co-60	-	2.69E+00	2.47E+00	5.56E+01	1.98E+02	7.70E+01
Ni-65	-	4.92E-03	3.75E-02	6.41E-03	6.95E-02	2.71E-02
Cu-64	-	2.33E+00	1.77E+01	3.05E+00	3.30E+01	1.29E+01
Zn-65	-	1.24E+00	1.22E+00	1.70E+01	6.44E+01	2.51E+01
Sr-90	1.43E-01	4.76E-02	-	-	3.66E+00	1.43E+00
Y-93	3.67E-02	2.48E-01	-	-	3.55E+00	1.38E+00
Nb-98	8.31E-19	2.22E-05	-	-	1.54E-05	6.02E-06
I-134	2.48E+00	4.30E-01	-	-	5.83E+00	2.27E+00
Cs-134	3.52E-01	1.98E-01	-	-	1.33E+01	5.19E+00
Cs-136	2.23E-02	1.47E-01	-	-	2.15E+00	8.38E-01
Cs-138	2.03E-02	-	-	-	-	-
Ba-139	1.37E-02	9.38E-02	-	-	1.30E+00	5.07E-01
W-187	0.00E+00	4.31E-02	3.27E-01	5.70E-02	6.17E-01	2.41E-01
Zn-69m	0.00E+00	1.68E-01	1.28E+00	2.20E-01	2.39E+00	9.34E-01
Zn-69	0.00E+00	1.68E-01	1.28E+00	2.20E-01	2.40E+00	9.36E-01
Br-83	2.63E-01	5.60E-02	-	-	7.85E-01	3.06E-01
Sr-89	1.37E-01	4.41E-01	-	-	1.00E+01	3.92E+00
Y-89m	1.60E-05	6.62E-05	-	-	1.51E-03	5.90E-04
Sr-92	2.59E-02	1.77E-01	-	-	2.48E+00	9.66E-01
Y-92	4.62E-02	3.13E-01	-	-	4.44E+00	1.73E+00

TABLE 11.4-9

EXPECTED INVENTORIES OF RADIOACTIVE MATERIALS IN WASTE SHIPPING CASKS(1,2)

ISOTOPE	LARGE CONTAINER 200 FT^3 (SHIPPING)/ 175 FT^3 (RESIN)		MEDIUM CONTAINER 174 FT^3 (SHIPPING)/ 150 FT^3 (RESIN)		SMALL CONTAINER 132 FT^3 (SHIPPING)/ 100 FT^3 (RESIN)	VERY SMALL CONTAINER 51 FT^3 (SHIPPING)/ 39 FT^3 (RESIN)
	CONDENSATE DEMIN. RESIN (175/300 FT^3) (CURIES)	RADWASTE DEMIN. RESIN (175/160 FT^3) (CURIES)	WASTE MIXING TANK (150/16 FT^3) (CURIES)	WASTE SLUDGE PHASE SEPARATOR (150/164 FT^3) (CURIES)	RWCU PHASE SEPARATOR (100/155.4 FT^3) (CURIES)	RWCU PHASE SEPARATOR (39/155.4 FT^3) (CURIES)
Ru-103	1.67E-02	7.90E-02	-	-	1.58E+00	6.17E-01
Rh-103m	1.67E-02	7.90E-02	-	-	1.58E+00	6.17E-01
Ru-106	2.18E-02	1.91E-02	-	-	1.13E+00	4.39E-01
Rh-106	2.18E-02	1.91E-02	-	-	1.13E+00	4.39E-01
Te-132	6.93E-04	4.68E-03	-	-	6.69E-02	2.61E-02
I-132	2.56E+00	5.49E-01	-	-	7.72E+00	3.01E+00
Cs-137	4.11E-01	1.36E-01	-	-	1.04E+01	4.07E+00
Ba-137m	3.87E-01	1.28E-01	-	-	9.85E+00	3.84E+00
Ba-140	1.09E-01	7.13E-01	-	-	1.04E+01	4.07E+00
La-140	1.09E-01	7.11E-01	-	-	1.04E+01	4.07E+00
La-142	7.79E-03	5.33E-02	-	-	7.40E-01	2.89E-01
Ce-143	8.84E-04	5.95E-03	-	-	8.49E-02	3.31E-02
Pr-143	1.23E-02	8.07E-02	-	-	1.19E+00	4.64E-01
Nd-147	6.98E-04	4.64E-03	-	-	6.69E-02	2.61E-02
Pm-147	4.53E-04	1.78E-04	-	-	1.53E-02	5.97E-03
Np-239	3.50E-01	2.36E+00	-	-	3.38E+01	1.32E+01
Pu-239	4.02E-05	1.23E-05	-	-	9.91E-04	3.86E-04
Sr-91	3.46E-02	2.34E-01	-	-	3.33E+00	1.30E+00
Y-91m	1.99E-02	1.35E-01	-	-	1.92E+00	7.48E-01
Y-91	8.42E-02	3.16E-01	-	-	7.85E+00	3.06E+00
Zr-95	1.09E-02	3.86E-02	-	-	1.01E+00	3.94E-01
Nb-95m	8.46E-05	2.84E-04	-	-	7.85E-03	3.06E-03
Nb-95	1.68E-02	4.90E-02	-	-	1.54E+00	6.02E-01
Mo-99	1.17E-01	7.90E-01	-	-	1.13E+01	4.39E+00
Tc-99m	2.12E-01	1.43E+00	-	-	2.04E+01	7.96E+00
Tc-99	2.97E-06	9.14E-07	-	-	7.34E-05	2.86E-05
Ru-105	8.32E-03	5.64E-02	-	-	7.98E-01	3.11E-01
Rh-105m	1.75E-03	1.18E-02	-	-	1.68E-01	6.55E-02

TABLE 11.4-9

EXPECTED INVENTORIES OF RADIOACTIVE MATERIALS IN WASTE SHIPPING CASKS[1,2]

ISOTOPE	LARGE CONTAINER 200 FT^3 (SHIPPING)/ 175 FT^3 (RESIN)		MEDIUM CONTAINER 174 FT^3 (SHIPPING)/ 150 FT^3 (RESIN)		SMALL CONTAINER 132 FT^3 (SHIPPING)/ 100 FT^3 (RESIN)	VERY SMALL CONTAINER 51 FT^3 (SHIPPING)/ 39 FT^3 (RESIN)
	CONDENSATE DEMIN. RESIN (175/300 FT^3) (CURIES)	RADWASTE DEMIN. RESIN (175/160 FT^3) (CURIES)	WASTE MIXING TANK (150/16 FT^3) (CURIES)	WASTE SLUDGE PHASE SEPARATOR (150/164 FT^3) (CURIES)	RWCU PHASE SEPARATOR (100/155.4 FT^3) (CURIES)	RWCU PHASE SEPARATOR (39/155.4 FT^3) (CURIES)
Rh-105	8.37E-03	5.64E-02	-	-	8.04E-01	3.14E-01
Te-129m	2.84E-02	1.45E-01	-	-	2.70E+00	1.05E+00
Te-129	1.79E-02	9.14E-02	-	-	8.17E-01	3.19E-01
I-129	4.85E-09	7.45E-10	-	-	4.10E-08	1.60E-08
Te-131m	5.35E-03	3.61E-02	-	-	5.15E-01	2.01E-01
Te-131	5.94E-04	4.00E-03	-	-	5.73E-02	2.24E-02
I-131	7.90E+00	2.06E+00	-	-	2.96E+01	1.15E+01
I-133	1.20E+01	3.12E+00	-	-	4.47E+01	1.74E+01
I-135	4.52E+00	1.09E+00	-	-	1.55E+01	6.05E+00
Ce-141	2.06E-02	1.07E-01	-	-	1.96E+00	7.65E-01
Ce-144	1.75E-02	1.88E-02	-	-	1.03E+00	4.02E-01
Pr-144m	2.50E-04	2.69E-04	-	-	1.87E-03	7.28E-04
Pr-144	1.75E-02	1.88E-02	-	-	1.31E-01	5.09E-02
Rb-87	2.86E-13	-	-	-	-	-
Rb-88	4.86E-03	-	-	-	-	-
Rb-89	2.98E-02	-	-	-	-	-
Cs-135	6.42E-09	-	-	-	-	-
TOTAL	3.27E+01	5.34E+01	8.01E+01	2.94E+02	1.47E+03	5.73E+02

NOTES:

1. Container volumes are based on disposal volumes; the volume in parentheses is the actual radwaste volume.
2. Cask inventories given above, are estimates for immediately after filling. Actual shipping inventories will include additional decay time.
3. 8.41E-01 = $8.41x10^{-1}$

TABLE 11.4-10

DESIGN INVENTORIES OF RADIOACTIVE MATERIALS IN WASTE SHIPPING CASKS[1,2]

ISOTOPE	LARGE CONTAINER 200 FT^3 (SHIPPING) / 175 FT^3 (RESIN)		MEDIUM CONTAINER 74 FT^3 (SHIPPING) / 150 FT^3 (RESIN)		SMALL CONTAINER 132 FT^3 (SHIPPING) / 100 FT^3 (RESIN)	VERY SMALL CONTAINER 51 FT^3 (SHIPPING) / 39 FT^3 (RESIN)
	CONDENSATE DEMINERALIZER RESIN (175/300 FT^3) (CURIES)	RADWASTE DEMINERALIZER RESIN (175/320 FT^3) (CURIES)	WASTE MIXING TANK (150/16 FT^3) (CURIES)	WASTE SLUDGE PHASE SEPARATOR (150/164 FT^3) (CURIES)	RWCU PHASE SEPARATOR (100/155.4 FT^3) (CURIES)	RWCU PHASE SEPARATOR (39/155.4 FT^3) (CURIES)
Na-24	-	1.49E-01[3]	2.36E+00	2.56E-01	4.07E+00	1.59E+00
P-32	-	3.40E-02	1.77E-01	1.16E-01	9.34E-01	3.64E-01
Cr-51	-	1.59E+00	4.84E+00	6.35E+00	4.51E+01	1.76E+01
Mn-54	-	3.92E-01	4.25E-01	4.02E+00	2.35E+01	9.16E+00
Mn-56	-	6.39E-01	1.02E+01	1.09E+00	1.73E+01	6.76E+00
Fe-59	-	3.74E-01	8.04E-01	1.72E+00	1.16E+01	4.52E+00
Co-58	-	3.13E+01	5.14E+01	1.75E+02	1.13E+03	4.40E+02
Co-60	-	5.55E+00	5.35E+00	7.29E+01	4.09E+02	1.60E+02
Ni-65	-	3.75E-03	5.94E-02	6.39E-03	1.02E-01	3.97E-02
Zn-65	-	1.89E-02	2.12E-02	1.79E-01	1.06E+00	4.14E-01
Br-84	5.55E-01	7.10E-02	-	-	1.85E+00	7.21E-01
Sr-90	4.91E+00	2.61E+00	-	-	1.99E+02	7.76E+01
Tc-101	2.76E-02	1.64E-01	-	-	4.00E+00	1.56E+00
I-134	7.53E+00	1.04E+00	-	-	2.78E+01	1.08E+01
Cs-134	1.96E+00	1.71E+00	-	-	1.18E+02	4.60E+01
Cs-136	3.19E-02	1.72E-01	-	-	4.73E+00	1.85E+00
Cs-138	1.32E-01	5.06E-01	-	-	1.32E+01	5.13E+00
Ba-139	3.34E-01	1.10E+00	-	-	2.96E+01	1.15E+01
Zn-69m	-	2.05E-03	3.25E-02	3.51E-03	5.61E-02	2.19E-02
Zn-69	-	2.05E-03	3.24E-02	3.52E-03	5.61E-02	2.19E-02
Br-83	1.04E+00	1.78E-01	-	-	4.83E+00	1.88E+00
Sr-89	3.52E+00	1.58E+01	-	-	5.07E+02	1.98E+02
Y-89m	5.18E-04	2.37E-03	-	-	7.60E-02	2.96E-02
Sr-92	3.47E-01	1.48E+00	-	-	4.02E+01	1.57E+01
Y-92	3.50E-01	1.48E+00	-	-	4.05E+01	1.58E+01
Ru-103	1.65E-02	8.12E-02	-	-	2.43E+00	9.49E-01

TABLE 11.4-10

DESIGN INVENTORIES OF RADIOACTIVE MATERIALS IN WASTE SHIPPING CASKS(1,2)

ISOTOPE	LARGE CONTAINER 200 FT^3 (SHIPPING) / 175 FT^3 (RESIN)		MEDIUM CONTAINER 74 FT^3 (SHIPPING) / 150 FT^3 (RESIN)		SMALL CONTAINER 132 FT^3 (SHIPPING) / 100 FT^3 (RESIN)	VERY SMALL CONTAINER 51 FT^3 (SHIPPING) / 39 FT^3 (RESIN)
	CONDENSATE DEMINERALIZER RESIN (175/300 FT^3) (CURIES)	RADWASTE DEMINERALIZER RESIN (175/320 FT^3) (CURIES)	WASTE MIXING TANK (150/16 FT^3) (CURIES)	WASTE SLUDGE PHASE SEPARATOR (150/164 FT^3) (CURIES)	RWCU PHASE SEPARATOR (100/155.4 FT^3) (CURIES)	RWCU PHASE SEPARATOR (39/155.4 FT^3) (CURIES)
Rh-103m	1.65E-02	8.12E-02	-	-	2.43E+00	9.49E-01
Ru-106	1.98E-02	2.61E-02	-	-	1.62E+00	6.30E-01
Rh-106	1.98E-02	2.61E-02	-	-	1.62E+00	6.30E-01
Ag-110m	-	5.68E-01	6.37E-01	5.44E+00	3.22E+01	1.26E+01
Ag-110	-	7.57E-03	8.46E-03	7.24E-02	4.28E-01	1.67E-01
Te-129m	2.96E-02	1.51E-01	-	-	4.38E+00	1.71E+00
Te-129	1.86E-02	9.47E-02	-	-	2.75E+00	1.07E-00
Te-132	3.53E+00	1.90E+01	-	-	5.22E+02	2.04E+02
I-132	1.37E+01	2.04E+01	-	-	5.59E+02	2.18E+02
I-131	6.00E+01	1.24E+01	-	-	3.42E+02	1.33E+02
Cs-137	5.13E+00	2.72E+00	-	-	2.08E+02	8.11E+01
Ba-137m	4.85E+00	2.57E+00	-	-	1.97E+02	7.67E+01
Ba-140	2.68E+00	1.37E+01	-	-	3.75E+02	1.46E+02
La-140	2.68E+00	1.37E+01	-	-	3.76E+02	1.47E+02
Ba-142	3.55E-02	1.50E-01	-	-	3.53E+00	1.38E+00
La-142	4.04E-02	1.50E-01	-	-	4.13E+00	1.61E+00
Ce-143	1.06E-03	5.74E-03	-	-	1.57E-01	6.12E-02
Pr-143	1.24E-02	6.69E-02	-	-	1.84E+00	7.17E-01
Nd-147	3.39E-03	1.83E-02	-	-	5.01E-01	1.95E-01
Pm-147	2.20E-03	1.54E-03	-	-	1.18E-01	4.61E-02
W-187	-	3.55E-01	5.58E+00	6.13E+01	9.71E+00	3.79E+00
Re-187	-	1.96E-12	1.54E-12	2.73E-11	1.52E-10	5.93E-11
Np-239	1.25E+01	6.72E+01	-	-	1.84E+03	7.19E+02
Pu-239	1.42E-03	7.12E-04	-	-	5.58E-02	2.18E-02
Br-85	3.21E-02	4.03E-03	-	-	6.19E-02	2.41E-02
Sr-91	6.05E-01	3.24E+00	-	-	8.88E+01	3.46E+01

TABLE 11.4-10

DESIGN INVENTORIES OF RADIOACTIVE MATERIALS IN WASTE SHIPPING CASKS(1,2)

ISOTOPE	LARGE CONTAINER 200 FT^3 (SHIPPING) / 175 FT^3 (RESIN)		MEDIUM CONTAINER 74 FT^3 (SHIPPING) / 150 FT^3 (RESIN)		SMALL CONTAINER 132 FT^3 (SHIPPING) / 100 FT^3 (RESIN)	VERY SMALL CONTAINER 51 FT^3 (SHIPPING) / 39 FT^3 (RESIN)
	CONDENSATE DEMINERALIZER RESIN (175/300 FT^3) (CURIES)	RADWASTE DEMINERALIZER RESIN (175/320 FT^3) (CURIES)	WASTE MIXING TANK (150/16 FT^3) (CURIES)	WASTE SLUDGE PHASE SEPARATOR (150/164 FT^3) (CURIES)	RWCU PHASE SEPARATOR (100/155.4 FT^3) (CURIES)	RWCU PHASE SEPARATOR (39/155.4 FT^3) (CURIES)
Y-91m	3.48E+01	1.86E+00	-	-	5.11E+01	1.99E+01
Y-91	6.06E+01	2.83E+00	-	-	8.79E+01	3.43E+01
Zr-95	5.67E-02	2.73E-01	-	-	8.23E+00	3.21E+00
Nb-95m	4.42E-04	1.815E-03	-	-	6.41E-02	2.50E-02
Nb-95	8.91E-02	3.42E-01	-	-	1.29E+01	5.02E+00
Zr-97	4.79E-04	2.68E-03	-	-	7.34E-02	2.86E-02
Nb-97m	4.71E-04	2.54E-03	-	-	6.96E-02	2.71E-02
Nb-97	4.97E-04	2.68E-03	-	-	7.34E-02	2.86E-02
Mo-99	1.33E+00	7.20E+00	-	-	1.97E+02	7.70E+01
Tc-99m	2.73E+00	1.47E+01	-	-	4.02E+02	1.57E+02
Tc-99	3.74E-05	1.88E-05	-	-	1.46E-03	5.71E-04
I-133	4.42E+01	9.18E+00	-	-	2.51E+02	9.81E+01
I-135	2.21E+01	4.27E+00	-	-	1.17E+02	4.54E+01
Ba-141	4.84E-02	2.57E-01	-	-	6.44E+00	2.51E+00
La-141	5.23E-02	2.57E-01	-	-	7.03E+00	2.74E+00
Ce-141	8.03E-02	3.86E-01	-	-	1.12E+01	4.36E+00
Ce-144	2.14E-01	3.39E-01	-	-	1.98E+01	7.73E+00
Pr-144m	3.05E-03	2.62E-03	-	-	3.31E-02	1.29E-02
Pr-144	2.14E-01	1.84E-01	-	-	2.32E+00	9.04E-01
Rb-88	9.01E-03	-	-	-	-	-
Rb-89	5.94E-02	-	-	-	-	-
Cs-139	1.15E-01	-	-	-	-	-
Rb-90	7.02E-02	-	-	-	-	-
Y-90	1.12E-02	-	-	-	-	-
Rb-92	4.06E-10	-	-	-	-	-
Rb-93	6.56E-09	-	-	-	-	-

TABLE 11.4-10

DESIGN INVENTORIES OF RADIOACTIVE MATERIALS IN WASTE SHIPPING CASKS(1,2)

ISOTOPE	LARGE CONTAINER 200 FT^3 (SHIPPING) / 175 FT^3 (RESIN)		MEDIUM CONTAINER 74 FT^3 (SHIPPING) / 150 FT^3 (RESIN)		SMALL CONTAINER 132 FT^3 (SHIPPING) / 100 FT^3 (RESIN)	VERY SMALL CONTAINER 51 FT^3 (SHIPPING) / 39 FT^3 (RESIN)
	CONDENSATE DEMINERALIZER RESIN (175/300 FT^3) (CURIES)	RADWASTE DEMINERALIZER RESIN (175/320 FT^3) (CURIES)	WASTE MIXING TANK (150/16 FT^3) (CURIES)	WASTE SLUDGE PHASE SEPARATOR (150/164 FT^3) (CURIES)	RWCU PHASE SEPARATOR (100/155.4 FT^3) (CURIES)	RWCU PHASE SEPARATOR (39/155.4 FT^3) (CURIES)
Sr-93	1.40E-02	-	-	-	-	-
Y-93	1.68E-02	-	-	-	-	-
Rb-94	3.82E-18	-	-	-	-	-
Sr-94	1.90E-04	-	-	-	-	-
Y-94	5.78E-04	-	-	-	-	-
Cs-140	3.88E-02	-	-	-	-	-
Cs-142	1.71E-22	-	-	-	-	-
TOTAL	1.99E+02	2.65E+02	8.19E+01	2.67E+02	8.38E+03	3.27E+03

NOTES:
1. Container volumes are based on disposal volumes; the volume in parentheses is the actual radwaste volume.
2. Cask inventories given above, are estimates for immediately after filling. Actual shipping inventories will include additional decay time.
3. 1.55E-01 = 1.55×10^{-1}

THIS FIGURE HAS BEEN
RENUMBERED TO 11.4-1-1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure renumbered from 11.4-1 to 11.4-1-1

FIGURE 11.4-1, Rev. 54

AutoCAD Figure 11_4_1.doc

THIS FIGURE HAS BEEN
RENUMBERED TO 11.4-2-1

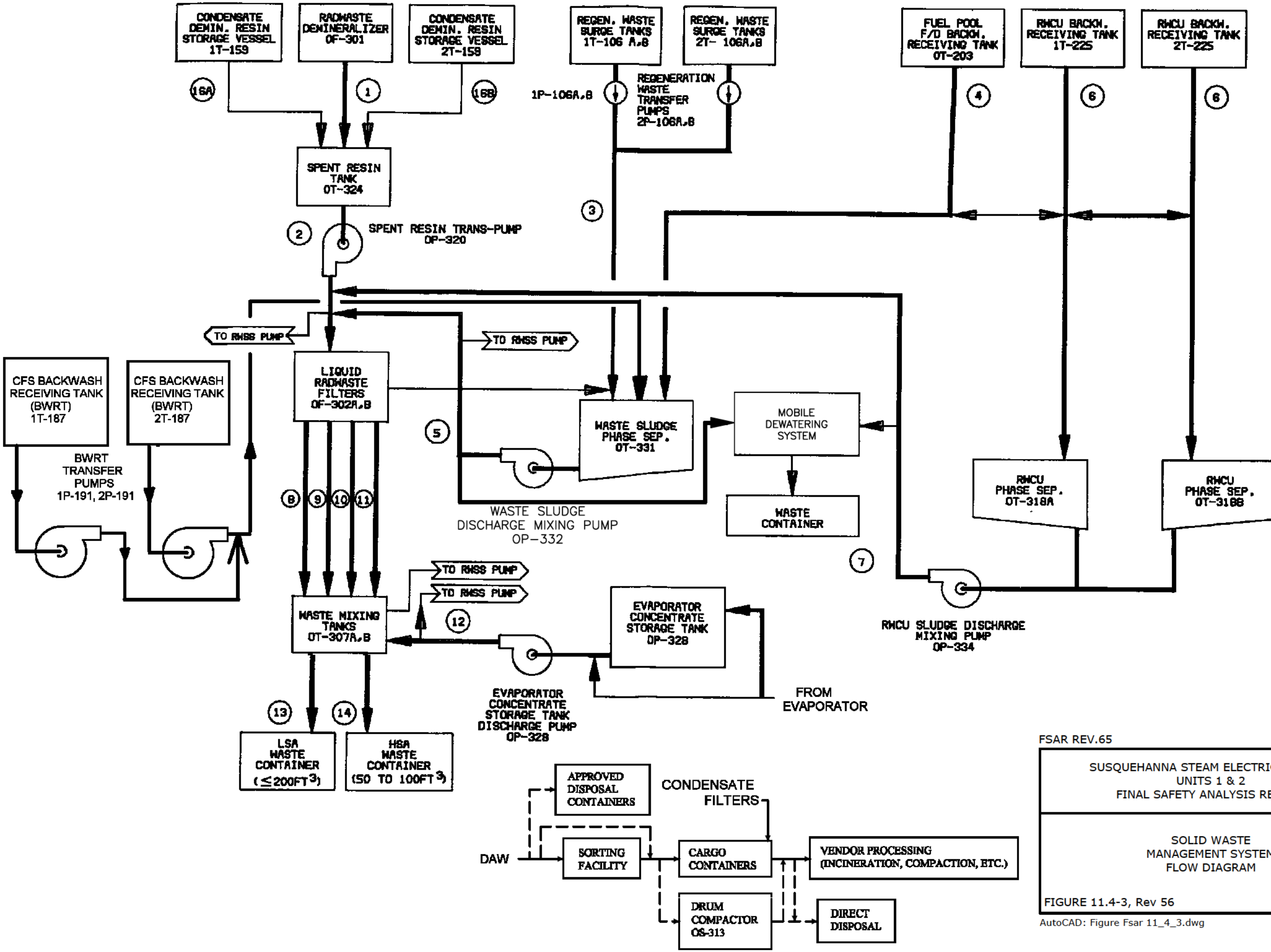
FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure renumbered from 11.4-2 to 11.4-2-1

FIGURE 11.4-2, Rev. 54

AutoCAD Figure 11_4_2.doc



FSAR REV.65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

SOLID WASTE
MANAGEMENT SYSTEM
FLOW DIAGRAM

FIGURE 11.4-3, Rev 56

AutoCAD: Figure Fsar 11_4_3.dwg

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-166, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure 11.4-1-1 replaced by dwg.
M-166, Sh. 1

FIGURE 11.4-1-1, Rev. 56

AutoCAD Figure 11_4_1_1.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-166, Sh. 2

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure 11.4-1-2 replaced by dwg. M-166, Sh. 2 FIGURE 11.4-1-2, Rev. 56

AutoCAD Figure 11_4_1_2.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-167, Sh. 1

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION UNITS 1 & 2 FINAL SAFETY ANALYSIS REPORT
Figure 11.4-2-1 replaced by dwg. M-167, Sh. 1 FIGURE 11.4-2-1, Rev. 55

AutoCAD Figure 11_4_2_1.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
M-167, Sh. 2

FSAR REV. 65

SUSQUEHANNA STEAM ELECTRIC STATION
UNITS 1 & 2
FINAL SAFETY ANALYSIS REPORT

Figure 11.4-2-2 replaced by dwg.
M-167, Sh. 2

FIGURE 11.4-2-2, Rev. 55

AutoCAD Figure 11_4_2_2.doc

11.5 PROCESS AND EFFLUENT RADIOLOGICAL MONITORING AND SAMPLING SYSTEMS

The process and effluent radiological monitoring and sampling systems are provided to monitor releases of radioactive material in the plant gaseous and liquid process and effluent streams in order to control these releases.

11.5.1 DESIGN BASES

11.5.1.1 Design Objectives

The design objectives of the systems described in this section are to generally conform with the requirements of General Design Criteria 60, 63 and 64; and Regulatory Guide 1.21, Rev. 1, 6-74.

Certain of the effluent systems described provide initiating circuits for the Engineered Safety Feature Systems. These systems are designed to be in compliance with IEEE 279-1971 to assure performance of the protective action required. References to Chapter 7 are provided for these systems.

Provision for monitoring postulated accidents is primarily provided by the measuring range of the channel provided. These ranges are noted in Table 11.5-1.

11.5.1.2 Design Criteria

Design Criteria are General Design Criteria 60, 63 and 64, Regulatory Guide 1.21, Rev. 1, 6-74, IEEE 279-1971, and NUREG 0737.

11.5.2 PROCESS AND EFFLUENT RADIOLOGICAL MONITORING SYSTEM DESCRIPTION

11.5.2.1 Continuous Process and Effluent Monitoring Systems

Process and effluent radiological monitoring systems (RMS) are identified below. Systems which provide safety-related functions are noted and appropriate references are made. One influent system is listed since it is the same basic equipment.

(1) Gaseous effluent stream monitoring:

   a) Reactor Building Vent Stack Exhaust Monitoring and Sample RMS

   b) Turbine Building and Radwaste Vent Stack Exhaust Monitor and Sample RMS.

   c) Standby Gas Treatment Vent Stack Exhaust Monitor and Sample RMS

d) Standby Gas Treatment Vent Duct Exhaust RMS: a safety-related system also discussed in Subsections 7.3.1.1b.4, 6.5.1.1 and 9.4.2

e) Refueling Floor Wall Duct Exhaust RMS: a safety-related system also discussed in Subsections 7.3.1.1b.5, 6.5.3 and 9.4.2

f) Refueling Floor High Exhaust Duct RMS: a safety-related system also discussed in Subsections 7.3.1.1b.5, 6.5.3 and 9.4.2

g) Railroad Access Exhaust Duct RMS: a safety-related system also discussed in Subsections 7.3.1.1b.5, 6.5.3 and 9.4.2

h) Outside Air Intake Duct (Influent) RMS: a safety-related system also discussed in Subsections 7.3.1.1b.7, 9.4.2 and Section 6.4.

(2) Liquid effluent monitor systems

a) Plant Radwaste Effluent
b) Service Water Discharge/Supplemental DH Removal

(3) Gaseous process streams monitoring

a) Offgas Pretreatment
b) Main Steamline Radiation Monitoring System (see Subsection 7.3.1)

4) Liquid Process Monitor Systems

a) RHR Service Water RMS
b) Reactor Bldg. Closed Cooling Water RMS

5) Containment Radiation Monitoring System

a) Primary Containment Atmospheric Monitoring System discussed in Subsection 5.2.5.1.2.3.

b) Primary Containment Radiation Monitoring System (High Range) discussed in Subsection 7.6.1b.1.5.

11.5.2.1.1 Reactor Building Vent Stack Exhaust Monitor and Sample RMS

The Reactor Building Vent Effluent Radiation Monitoring System (VERMS) is a General Atomics -ESI Gas Monitor system. A sample representative of the exhaust stream is continuously extracted using a sampling probe array mounted in the exhaust duct. This system utilizes both sampling filters/cartridges and a radiation detector to continually monitor effluent from the reactor building ventilation exhaust zones as noted on Dwgs. VC-175, Sh. 1, VC-175, Sh. 2 and VC-175, Sh. 3. The system is shown on drawing M-165, Sh. 5.

The sample is passed through filters/cartridges located on the Gas Monitor Skid assemblies, 1C216B/2C216B for collection of radioactive particulate and iodine. Radiation detectors mounted on the skids continuously measure noble gas activity in the sample stream.

VERMS uses constant flow rate sampling because flow rate variations in the tubing have a greater affect on sample composition than the corresponding affect due to deviation from true isokinetic flow at the sample nozzle as vent flow varies from nominal flow rate. Sample tubing size is optimized to minimize losses at the sample flow rate. This approach provides the most representative vent sample at the filter holders as required by ANSI N13.1-1969.

The sample then splits into 2 streams. One stream passes through the Bypass Pump Skid Assembly, 1C299C/2C299C. The two bypass pumps on this skid provide the additional flow necessary to maintain a representative sample flow through the sample probe. The second stream, a continuous fixed flow, is passed through the Gas Monitor Skid Assembly, 1C299B/2C299B, for collection of radioactive particulate and iodine samples and continuous monitoring of noble gas activity. Downstream of the skids the two streams are combined and returned to the Reactor Building vent for release. The Gas Monitor Skid Assembly includes connections for collecting a sample volume for laboratory analysis of noble gas activity and calibration of the radiation detector. Provisions are included in the system design for the temporary connection and powering of an alternate sample cart and tritium sample collector.

For the Reactor Building VERMS a single low range radiation detector is used. It has a range which extends from minimum sensitivity to normal operation and up through all operational occurrences and postulated accidents for which the reactor building ventilation system remains operational. For the more severe accidents the reactor building ventilation systems are automatically shutdown and isolated and reactor building ventilation is provided by the Standby Gas Treatment System which has a separate vent and a separate wide range radiation monitoring system (see Section 11.5.2.1.3).

System control is provided by a microprocessor based controller mounted locally on the Gas Monitor Skid Assembly. This controller also provides the radiation monitoring signal processing. The controller provides a local information display, local alarms, system status information and interface with the rest of the plant. This controller communicates system information to a remote control and display assembly located on panel 0C658 in the Control Room. The remote controller provides the same information displays and can provide the same control functions as the local controller. Vent flow and release rate information is continuously recorded by a separate Control Room recorder. An independent communications network provides Reactor Building VERMS information to the Technical Support Center.

Each Reactor Building VERMS provides Inputs to the 3 common plant exhaust vent alarms on Control Room panel 0C653; Vent Monitoring TROUBLE, HIGH RADIATION, HI-HI RADIATION. The plant operator can determine which vent is alarming by observing the remote control and display assemblies on control room panel 0C658.

Representative samples from the Reactor Building exhaust vents are drawn continuously and filters/cartridges replace and analyzed at least weekly to monitor particulate and iodine effluent releases. Radionuclide analyses of samples is also performed when continuous monitoring of noble gases shows an unexplained variance from an established norm which may be indicative of a change in the concentration and composition. The norm is established as a range of readings that may be expected due to normal operating conditions including anticipated operational occurrences. Radionuclide analysis of samples is also performed following each refueling, process change, or other occurrence that could alter the mixture of radionuclides.

11.5.2.1.2 Turbine Building and Radwaste Vent Stack Exhaust Monitor and Sample RMS

The Turbine Building Vent Effluent Radiation Monitoring System (VERMS) is a General Atomics -ESI Wide Range Gas Monitor system. A sample representative of the exhaust stream is continuously extracted using a sampling probe array mounted in the exhaust duct. This system utilizes both sampling filters/cartridges and radiation detectors to continually monitor effluent from the turbine building ventilation exhaust and the processed and filtered reactor gaseous radwaste from the Radwaste Building as shown on drawing VC-174, Sh. 1. The system is shown on drawing M-165, Sh. 4.

VERMS uses constant flow rate sampling because flow rate variations in the tubing have a greater affect on sample composition than the corresponding affect due to deviation from true isokinetic flow at the sample nozzle as vent flow varies from nominal flow rate. Sample tubing size is optimized to minimize sample transport losses. The sample then splits into 2 streams. One stream passes through the Bypass Pump Skid Assembly, 1C1108C/2C1108C. The two bypass pumps on this skid provide the additional flow necessary to maintain a representative sample flow through the sample probe. The second stream, a continuous fixed flow, passes through sample filter/cartridges and radiation detectors for sample collection and radiation monitoring. The two flow streams then recombine and are returned to the Turbine Building vent for release. Provisions are included in the system design for the temporary connection and powering of an alternate sample cart and tritium sample collector.

The sample filters/cartridges and detectors used for monitoring change depending on noble gas activity in the sample stream. During normal power generation operation the sample is drawn through sample cartridges on the Normal Sample Conditioning Skid Assembly, 1C2108/2C2108 located on the refuel floor, and then through the low range radiation detector on Sample Detection Skid Assembly, 1C1108B/2C1108B located on Turbine Building El. 729'. If radiation levels in the sample rise to levels indicating an accident and that Refuel Floor habitability may become a problem motor operated valves in the system automatically transfer the sample flow to sample collection cartridges located on Turbine Building El. 729" in Conditioning Skid Assembly, 1C1108A/2C1108A. At this first stage of accident alignment the flow will continue through the low range radiation detector on Sample Detection Skid Assembly, 1C1108B/2C1108B. If radiation levels in the sample continue to rise the sample flow will be reduced and transferred to shielded sample cartridges on the Sample Conditioning Skid Assembly and pass from there to Mid and High range radiation detectors on the Sample Detection Skid Assembly. The Sample Detection Skid Assembly includes connections for collecting a sample volume for laboratory analysis of noble gas activity and calibration of the radiation detectors.

Three overlapping ranges of radiation detectors are necessary in the Turbine Building VERMS to provide monitoring for the full range of potential radiation, from normal operation to the most severe postulated accident. To reduce maximum radiation exposure to equipment and personnel the sample flow rate is reduced by about a factor of 25 when flow is switched to the shielded cartridges and Mid and High range radiation detectors. This change happens automatically and no operator action is required. Also, system outputs automatically adjust for the changes in flow rate and detector range.

System control and interfaces are essentially identical to those for the Reactor Building VERMS. The only difference are the automatic changes in sample flow alignment, sample flow rate and radiation detector range as radiation level in the sample increases.

11.5.2.1.3 Standby Gas Treatment System Vent Stack Exhaust Monitor and Sample RMS

The Standby Gas Treatment System Vent Effluent Radiation Monitoring System (VERMS) is a General Atomics -ESI Wide Range Gas Monitor system which is almost identical to the Turbine Building VERMS described in 11.5.2.1.2. The only difference is with Bypass Pump Control. Because the Standby Gas Treatment System vent flow is much lower than the Turbine Building Vent flow there are operating conditions where no Bypass Pump flow is needed and both bypass pumps on Bypass Pump Skid Assembly 0C194C are never needed. The automatic Bypass Pump control scheme accounts for these differences. The system is shown on drawing M-165, Sh. 3. The Normal Sample Conditioning Skid on the Refuel Floor is 0C294, the Sample Conditioning Skid on Turbine Building El. 729' is 0C194A and the Sample Detection Skid Assembly is 0C194B.

11.5.2.1.4 Standby Gas Treatment Vent Exhaust Radiation Monitoring System

The system monitors gamma radiation in the exhaust vent of the standby gas treatment system. Two instrument channels constitute this system. Automatic closure of the containment purge isolation valves is actuated upon detection of high radiation in the exhaust vent during containment purge.

(1) The detectors are mounted outside the exhaust vent.

(2) The sensors are Geiger-Muller tubes with converter units. Refer to Table 11.5-1.

(3) Two independent instrument channels are designed each with a detector/converter unit mounted locally and an indicator trip unit in the control structure. The radiation measurement is displayed on the indicator trip unit and both channels are recorded in the main control room on a digital recorder.

The trip circuit detects two upscale trips (high-high and high radiation) and one downscale/inoperative trip.

(4) The following annunciators are located in the control room:

   a) Standby Gas Exhaust Vent High-High Radiation

   b) Standby Gas Exhaust Vent High Radiation

   c) Standby Gas Exhaust Vent Monitor Downscale

5) The power source for instrument channel A is from the reactor protection system 120 V ac supply bus A.

   Instrument Channel B is powered from RPS bus B.

6) The trip allowable value and surveillance requirements are defined in the Technical Specifications.

11.5.2.1.5 Refueling Floor Wall Exhaust Duct Radiation Monitoring System

This system monitors the radiation level in the exhaust duct from the refueling floor prior to its discharge to the atmosphere through the reactor building vent. The refueling floor is included as part of reactor building ventilation Zone III.

Refer to Dwg. VC-175, Sh. 2 for system design.

(1) The detection assemblies are located in the exhaust ducting upstream of the inboard isolation damper. The distance from the inboard isolation damper is defined by the ventilation flow transport time at maximum design flow rate from the detector location to the inboard isolation damper, which is greater than the total time to respond to a trip level radiation for complete closure of the inboard isolation damper.

(2) The detector assembly is a Geiger-Muller tube with a converter unit.

(3) Two redundant independent instrument channels are provided. Each channel consists of a local detector and converter unit, which transmits a signal to the control room indicator and trip unit. Two trip circuits monitor the upscale (high-high)/inoperative condition and the downscale condition. The upscale trip indicates high radiation and the downscale trip indicates instrument trouble. The upscale trip initiates closure of the reactor building Zone III ventilation outboard isolation dampers, starts the SGTS and the reactor building recirculation system (see Section 6.5). Refer to Section 7.3 for isolation logic. The radiation level of each channel is recorded on a digital recorder.

(4) Measurement ranges are defined in Table 11.5-1.

(5) Two alarms for high radiation (high and high-high) and one alarm for downscale trip are located in the control room. Actuation of isolation dampers is discussed in Section 7.3.

(6) Reactor protection system power bus A is the power source for one instrument channel. RPS bus B is the power source for the other channel. Refer to Chapter 8.

(7) The monitors are readily accessible for calibration, inspection and maintenance. A portable gamma source may be used for testing the instrumentation.

(8) Reactor building ventilation Zone III is described in Section 9.4.2 and is a common ventilation zone for Units 1 and 2. Both Unit 1 and Unit 2 Refueling Floor Wall Exhaust Duct Radiation Monitors provide monitoring of this common ventilation zone exhaust. As such, both Unit 1 and Unit 2 Refueling Floor Wall Exhaust Duct Radiation Monitors are required during operation of either Unit 1 or Unit 2.

11.5.2.1.6 Refueling Floor Exhaust Duct High Radiation Monitoring System

This system consists of shielded detectors whose purpose is to monitor the radiation level in the refueling floor ventilation exhaust duct adjacent to the intake register. The refueling floor is included as part of reactor building ventilation Zone III.

The system is identical to the refueling floor wall exhaust radiation monitoring system with the same channel trip logic and protective action initiation. Refer to Dwg. VC-175, Sh. 2 and Table 11.5-1 for system configuration.

Reactor building ventilation Zone III is described in Section 9.4.2 and is a common ventilation Zone for Units 1 and 2. Both Unit 1 and Unit 2 Refueling Floor Exhaust Duct High Radiation Monitors provide monitoring of this common ventilation zone exhaust. As such, both Unit 1 and Unit 2 Refueling Floor Exhaust Duct High Radiation Monitors are required during operation of either Unit 1 or Unit 2.

11.5.2.1.7 Railroad Access Exhaust Duct Radiation Monitoring System (Unit 1 only)

This system monitors the radiation level in the railroad access area air exhaust duct prior to the Unit 1 reactor building vent. The railroad access area is located in and is unique to the Unit 1 reactor building. The system design is identical to the refueling floor wall radiation monitoring system. Table 11.5-1 identifies type of detector, location and instrument ranges. Refer to Section 7.3 for reactor building isolation initiation. Dwg. VC-175, Sh. 2 documents the system configuration.

The railroad access shaft ventilation zone is described in Section 9.4.2. The Railroad Access Exhaust Duct Radiation Monitors are required for monitoring of this ventilation area during the movement of Unit 1 or Unit 2 irradiated fuel within the Railroad Access Shaft, and directly above the Railroad Access Shaft with the Railroad Access Shaft Equipment Hatch open.

11.5.2.1.8 Control Structure Outside Air Intake Radiation Monitoring System

The radioactivity of the outside air intake for the control structure is continuously monitored to detect airborne radioactive material which enters the heating, ventilating, and air conditioning system for the control structure. An increase in gamma radiation is detected by the two redundant instrument channels. The system provides a trip signal to initiate the emergency intake air supply system for the control room habitability engineered safety feature. Refer to Section 7.3 for actuation of this safety system.

(1) The two redundant monitors are located in the outside intake air plenum.

(2) The radiation detection system uses a Geiger-Muller tube with a converter unit.

(3) Two redundant, independent instrument channels are provided, each with an indicator and trip unit in the control structure. The radiation measurement is displayed on a four decade logarithmic scale. A high radiation reading initiates the upscale trip circuit and contacts for alarm and protective action are provided. Power failure or component malfunction causes downscale trip initiation.

    Each channel measurement is recorded on a digital recorder in the control room.

(4) Instrument ranges and scale information are in Table 11.5-1.

(5) High/high radiation and downscale/inoperative alarms are displayed in the control room. The digital recorder initiates a control room high radiation alarm set at a slightly lower level than the high/high alarm.

At high/high radiation level, the trip circuit initiates the emergency outside air intake system as described in Section 7.3.

(6) Each channel is powered by its separate and independent reactor protection system power supply, channel A from RPS bus A, and channel B from RPS bus B.

Refer to Chapter 8 for electrical power distribution.

(7) The trip allowable value and surveillance requirements are per the Technical Specifications.

(8) The instrumentation is readily accessible for calibration, inspection, and maintenance. A portable gamma source unit may be used for testing and calibration.

11.5.2.1.9 Liquid Radwaste Effluent RMS

This system monitors the gross gamma activity in the discharge line from liquid radwaste prior to discharge into the cooling tower blowdown line.

The system monitors a sample of the discharge effluent by use of a side stream of the discharge. The side stream pipe is located downstream of the last point of waste admission to the discharge, and upstream of the first isolation valve. The sample flow is pumped by the Liquid Radwaste Effluent RMS through a shielded sample chamber and back to the discharge line upstream of the flow control valve (Dwg. M-164, Sh. 1).

The Liquid Radwaste Effluent RMS consist of:

1) a flow element to measure total radwaste discharge flow rate. The flow rate is recorded on a recorder on panel 0C301 in the radwaste control room.

2) a gamma sensitive scintillation detector contained in the shielded sample chamber. The count rate is displayed and recorded on panel 0C301 in the radwaste control room.

The monitoring instrumentation provides three trips, all of which terminate radwaste discharge. The trips are high radiation, downscale and low sample flow. The high radiation and downscale trips initiate alarms in the main control room and the low sample flow trip initiates an alarm in the radwaste control room. A check source is provided for periodic testing of the monitoring system. A fourth trip, cooling tower blowdown flow interlock, exists to terminate radwaste discharge which ensures adequate dilution for effluent releases.

All controls, alarms, setpoint adjustment and indicators are located on 0C336 in the Radwaste Building. The Radwaste Control Room has a monitor which communicates with the local skid for operation status, activity status, alarm status and by receiving status inquiries and function control commands. The local skid can be controlled from the Radwaste Control Room. The Main Control Room Panel 1/2C601 each receives two alarms one for high radiation and one for downscale trip. Two isolation valves installed in series are provided. These valves are air operated and fail closed. All controls and permissive interlocks require contact closures to open the valves, thereby providing fail safe valve control. No emergency power is provided.

The sensitivity of the liquid radwaste effluent monitor is 3.07 x 10^8 CPM/μCi/cc (Cs^{137}).

Refer to Section 11.2 for a discussion of concentrations, compositions, flows and measurements.

11.5.2.1.10 Service Water Discharge/Supplemental DH Removal Radiation Monitoring System

The objective of this system is to detect radioactive material inleakage to the service water system and to monitor the service water system discharge to the cooling tower basin. The spent fuel pool heat exchangers are the only potential radioactive release path to the service water system. As such, the radiation monitors are located on the service water discharge line from the spent fuel pool heat exchangers.

(1) Two detectors are provided; one in the service water piping and one in the supplemental decay heat removal piping. During normal operation the service water discharge detector is connected to a radiation monitoring unit (See (4)). During unit outages, when supplemental decay heat removal is placed in service, the supplemental decay heat removal detector is connected to a control structure radiation monitoring unit in place of the service water detector.

(2) The service water detector is located on the downstream side of the fuel pool heat exchangers prior to discharge to the cooling tower. The supplemental decay heat removal detector is located on the downstream side of the fuel pool heat exchangers prior to discharge to the supplemental cooling equipment.

(3) Both detectors are scintillation detectors with the same characteristics. (See Table 11.5-1)

(4) Each detector assembly is mounted in the process flow. Shielding is provided as required to ensure adequate sensitivity to the process radiation. A local pulse preamplifier for each detector transmits the signal from the connected detector to a radiation monitoring unit in the control structure (upper relay room). This radiation monitor provides trip points: one upscale (high radiation) and one downscale (low-low radiation). High radiation and downscale trips are alarmed in the control room. The radiation level can be observed in the control room on a digital recorder.

(5) Table 11.5-1 shows instrument ranges.

(6) No emergency power is provided.

(7) Surveillance requirements including calibration and testing are based on the Technical Requirement Manual.

(8) Surveillance requirements including calibration and testing are based on the Technical Requirement Manual.

11.5.2.1.11 Offgas Pretreatment Radiation Monitoring System

This system monitors radioactivity in the condenser offgas discharge after it has passed through the steam jet air ejector (SJAE), see Dwg. M-107, Sh. 4. 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 stainless steel sample line. It is then passed through a sample chamber and a sample panel before being returned to the condenser. The sample chamber is a steel pipe, which is internally polished to minimize plateout. It can be purged with room air to check detector response to background radiation by using a three-way solenoid operated valve. The valve is controlled by a switch located in the main control room. The sample panel measures and indicates sample line flow.

The sample chamber is monitored by three channels. Each channel has a gamma-sensitive ionization chamber mounted external to the sample chamber. Two channels have logarithmic radiation monitors which provide system alarm output. The third channel has a linear radiation monitor for recording in the control room. The two logarithmic channels are recorded on a separate digital recorder.

Power is supplied from the 125 V nondivisional bus for the logarithmic channels, from 24 V bus A for the linear channel, from the 120 V instrument bus for the recorders, and from a local 120 V bus for the sample and vial sampler panels.

The logarithmic radiation monitors have four trip circuits: two upscale (high-high and high), one downscale (low), and one inoperative.

The trip outputs are used for alarm function only. Each trip actuates a control room annunciator: offgas high-high, offgas high, and offgas downscale/inoperative. High or low sample line flow measured at the sample panel actuates a control room offgas sample high-low flow annunciator.

The radiation level output by the monitor may be correlated to the concentration of the noble gases by using the semiautomatic vial sampler panel to obtain a grab sample. To draw a sample, a serum bottle is inserted into a sample chamber, the sample lines are evacuated from the bottle to a solenoid-operated sample valve. The 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 gases radionuclides.

11.5.2.1.12 Main Steamline 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 the radiation monitor initiates a main control room annunciator and trips the mechanical vacuum pump (MVP).

(1) Four radiation monitors are located near the main steam lines just downstream of the outboard main steam line isolation valves in the space between the primary and

secondary containment walls. The detectors are geometrically arranged so that this system is capable of detecting significant increases in radiation level with any number of main steam lines in operation.

(2) Each monitor has a gamma sensitive ion chamber with 3.7×10^{-10} amp/R/hr sensitivity.

(3) The system consists of four instrument channels. Each ion chamber detector provides a signal to a control room log-radiation monitor with meter and auxiliary trip unit. One four channel recorder powered from the 120 V instrument bus allows the output of all four channels to be recorded simultaneously. The channels A and C are physically and electrically independent of channels B and D.

(4) Table 11.5-1 lists the range of the detectors.

(5) Each radiation monitor has four trip circuits: two upscale (high-high and high), one downscale (low), and one inoperative. Each trip is visually displayed on the affected radiation monitor. A high-high rad count in the radiation monitor initiates a main control room annunciator common to all channels. A high or inoperative trip in the radiation monitor results in a trip of the MVP and isolation of its suction valve. 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 channel has a control room display of the measured radiation level.

(6) Power for two channels (A and C) is supplied from the RPS bus A and for the other two channels (B and D) from the RPS bus B. Refer to Chapter 8 for more detail on emergency power supply.

(7) Alarm and trip setpoints and surveillance requirements including calibration and testing are based on the Technical Requirements Manual.

(8) Testing can be performed during full power operation. Calibration can be performed during shutdown.

(9) During the periodic test of any one radiation monitor channel a control room annunciation from the downscale trip output of the monitor will be provided. In order to confirm proper annunciation of a channel, a simulated input must be introduced into the monitor input and increased in magnitude to initiate the annunciator point. Confirmation of the monitor meter indication relative to the other channels must be performed after the simulated input is replaced by a true detector signal.

11.5.2.1.13 RHR Service Water Radiation Monitoring System

This system is designed to detect primary coolant leakage into the RHR service water during operation of the RHR heat exchanger. The two RHR heat exchangers are each rated for 100 percent of reactor shutdown operation. In the event of a leakage in one heat exchanger, the redundant unit could be placed into service by the control room operator.

Two systems are provided, one for each heat exchanger. Each system operates as follows: A sample is drawn from the service water, downstream from the heat exchanger by means of a sample pump. The sample is returned to the same pipe downstream of the sample point.

The sample is pumped through a shielded sample chamber and monitored by a radiation detector and electronics. All the radiation monitor components are located in panel 2C212A&B which provides local indication and output signals for recording and alarms in the main control room. Radiation monitor pump automatically starts when RHR Heat exchanger is placed in service. Recording is on panel 1C600/2C600. Alarms are on panel 1C601/2C601. A flow transmitter in the sample loop provides a flow signal to trip the monitor on low sample flow.

11.5.2.1.14 Reactor Building Closed Cooling Water Radiation Monitoring System

The radiation monitor system detects leakage from the reactor water cleanup system into reactor building closed cooling water through the nonregenerative heat exchanger. Any increase of the radiation level above background is an indication of leakage.

The detector is located in the suction header piping of the RBCCW pumps.

Table 11.5-1 identifies the provided instrumentation.

11.5.2.2 Routine Sampling

The requirements of the system design bases for routine continuous and discrete sampling of radioactivity are satisfied by a system of liquid, gaseous, and airborne samplers, laboratory equipment for sensitive radio-chemical analyses, and a program of procedures for obtaining and analyzing representative samples when and where appropriate. This subsection provides a description of system hardware and procedures in general, including the type of sampling equipment used, the procedures to obtain representative samples, and analytical procedures. Table 11.5-2 is a tabulation of basic information describing each of the radioactivity sampling locations, including the basis for selecting the location, expected process flows, sampling frequency, analytical procedure, and expected monitor sensitivities. Table 11.5-3 gives the expected composition and concentration of nuclides in routine effluent samples.

Sampling equipment and procedures are provided to assure that representative samples are obtained. Prior to sampling, large tanks of liquid waste are well-mixed in as short a interval as practicable to assure that any sediments or particulate solids are distributed uniformly in the waste mixture. Sample lines are flushed for a sufficient period of time prior to sample extraction in order to remove sediment deposits and air and gas pockets. A sample is taken before discharge to determine the isotopic mixture and concentration of the tank. Composite sampling of the cooling tower blowdown is performed as part of the Radiation Environmental Monitoring Program for estimating doses to the public.

Effluent ventilation vents are sampled continuously and isokinetically for radioactive gases, particulates, and iodines. Particulate and iodine sampler filters are replaced and removed for analysis periodically for all continuous airborne radiation monitors and samplers. A gas sample will be taken monthly or if the gaseous monitor count rate shows a significant change.

11.5.2.2.1 Analytical Procedures

Techniques available in the laboratory for analyzing samples of process and effluent gases and liquids include:

1) Gross alpha counting (Normal vendor analysis)

2) Gamma spectrometry

3) Liquid scintillation counting

4) Radiochemical separations (Normal vendor analysis)

Instrumentation available in the laboratory for the measurement of radioactivity includes:

1) Liquid scintillation counter

2) Gamma spectrometer

    a) Germanium detector

    b) Multichannel analyzer system

Gamma spectrometry is used for isotopic analysis of liquid, gaseous, and airborne particulate and iodine samples. A high-efficiency, high-resolution Germanium detector is available for resolving complex gamma spectra.

Gaseous tritium samples are collected and counted on the liquid scintillation counter.

11.5.3 EFFLUENT MONITORING AND SAMPLING

General Design Criteria 64, "Monitoring Of Radioactivity Releases," is implemented using the equipment and systems described in Subsection 11.5.2. With respect to the specific areas, discharges and environs mentioned in General Design Criteria 64, the following subsections apply.

11.5.3.1 Containment Atmosphere

Monitoring of containment atmosphere for radioactivity is described in Subsection 5.2.5.1.2.3. Description is given of the systems which can detect leakage of radiation from the vessel and piping to the primary containment. Monitoring is continuous and is applicable to normal operations and occurrences.

Monitoring of Containment Atmosphere for H^2 and O^2 gas concentrations is described in Subsection 6.2.5.2. Description is given of redundant systems which monitor gas concentration within the primary containment drywell or the suppression chamber. Monitoring is continuous during normal plant operations. The PASS (Post Accident Sampling System) is used after a LOCA.

11.5.3.2 Reactor Building

The reactor building contains components and piping which are used for the recirculation of LOCA fluids. Radiation monitoring in this space consists of ventilation duct monitors, (refer to Subsections 11.5.2.1.4 through 11.5.2.1.7) and the area radiation monitors, (refer to Subsection 12.3.4 and Table 12.3-7 channels 1 thru 6, 8 through 16, 25, 26, 35, 36 and 41 through 57). The descriptions for this equipment apply for normal, operating, anticipated occurrences and accident conditions.

11.5.3.3 Effluent Discharge Paths

Monitoring of plant effluent discharge paths is described in Subsection 11.5.2.

Normally non-radioactive systems are considered 80-10 systems in accordance with NRC IE Bulletin 80-10 if they have the potential for radioactive contamination and a release pathway to the environment. The systems that have the potential to be contaminated and conditions for their use if they should become radioactive are controlled by the ODCM.

11.5.3.4 Plant Environs Measurement

Pre-operational Environs measurements are discussed in Chapter 6 of the Susquehanna S.E.S. Environmental Reports. Operational Environs Measurements are discussed in the ODCM and Technical Requirement Manual.

11.5.4 PROCESS MONITORING AND SAMPLING

Systems monitoring gaseous process streams are described in Subsections 11.5.2.1.11 and 11.5.2.1.12.

Systems monitoring liquid process streams are described in Subsections 11.5.2.1.10, 11.5.2.1.13 and 11.5.2.1.14.

11.5.4.1 Process Monitoring and Sampling Systems

These systems implement General Design Criteria 60 with respect to automatic closure of isolation valves as described in the following subsections.

11.5.4.1.1 Plant Radwaste Effluent RMS

This monitoring system will initiate isolation of two effluent discharge valves. Description is provided in Subsection 11.5.2.1.9.

11.5.4.1.2 Service Water Discharge/Supplemental DH Removal RMS

This system does not provide initiation of any isolation function. A description is provided in Subsection 11.5.2.1.10.

11.5.4.1.3 Offgas Pretreatment RMS

There is no provision for the offgas pretreatment monitor system to automatically stop this effluent from reaching the turbine building exhaust vent. A description is provided in Subsection 11.5.2.1.11.

11.5.4.1.4 Main Steamline RMS

Detection of high high radiation by this monitoring system initiates isolation of reactor coolant sample valves.

11.5.4.1.5 RHR Service Water RMS

There is no provision for isolation initiation by this System. A description is provided in Subsection 11.5.2.1.13.

11.5.4.1.6 Reactor Building Closed Cooling Water RMS

There is no provision for isolation initiation by this system. A description is provided in Subsection 11.5.2.1.14.

11.5.4.2 Radioactive Waste Process Monitoring and Sampling Systems

These systems implement General Design Criteria 63, "Monitoring Fuel and Waste Storage," with respect to radiation levels in radioactive waste process systems. Radiation levels are measured only at points in the actual discharge lines in the case of liquids and at the discharge vents in the case of gases (Turbine Building vent). Solid wastes are not measured during the solid waste processing operation.

A sample capability is provided in the Radwaste Solidification System. This allows small grab samples to be taken from the waste mixing tanks OT-307 A&B (see 11.4.2.2).

TABLE 11.5-1

PROCESS AND EFFLUENT RADIATION MONITORING SYSTEMS

MONITORED PROCESS	NO. OF CHANNELS	DETECTOR TYPE	DETECTOR LOCATION	CHANNEL RANGE	WARNING ALARM	TRIP	SCALE
A. SAFETY-RELATED SYSTEMS							
Main Steamline	4	Ionization chamber	Immediately downstream of last main steam	1-10^6 mr/hr	Yes	Technical Requirements Manual	6 dec. log
Refuel Floor Wall Exhaust	2	Geiger-Muller tube	Exhaust duct upstream of ventilation isolation damper	0.01 mr/hr to 100 mr/hr	Yes	Technical Specification	4 dec. log
Refuel Floor High Exhaust	2	Geiger-Muller tube	Exhaust duct upstream of ventilation isolation damper	0.01 mr/hr to 100 mr/hr	Yes	Technical Specification	4 dec. log
Emergency Outside Air Intake	2	Geiger-Muller tube	Outside air intake plenum	0.01 mr/hr to 100 mr/hr	Yes	Technical Specification	4 dec. log
Railroad Access Shaft Exhaust	2	Geiger-Muller tube	Exhaust duct upstream of ventilation isolation damper	0.01 mr/hr to 100 mr/hr	Yes	Technical Specification	4 dec. log
Standby Gas Treatment Vent Exhaust	2	Geiger-Muller tube	Outside Exhaust Vent	0.01 mr/hr to 100 mr/hr	Yes	Technical Specification	4 dec. log
B. SYSTEMS REQUIRED FOR PLANT OPERATION							
Service Water Discharge/Supplemental DH Removal	1	Scintillation	Effluent pipe prior to discharge into other systems	10^1 to 10^6 counts/sec	As determined by methodology of the ODCM	Not applicable	7 dec. log
Reactor Building Closed Cooling Water	1	Scintillation	Suction header to closed cooling water pumps	10^1 to 10^6 counts/sec	Above background	Not applicable	7 dec. log
RHR Service Water A/B	2	Scintilla ion	Process pipe downstream of heat exchanger	1 to 10^6 counts/min	As determined by methodology of the ODCM	Not applicable	5 dec. log and Digital Readout
Liquid Waste Effluent (Plant)	1	Scintillation	Effluent pipe prior to discharge	1 to 10^6 counts/min	As determined by methodology of the ODCM	As determined by methodology of the ODCM	7 Digital and Recorder 5 dec. log (10-10^6 CPM)
Offgas Pretreatment	3	Ionization chamber	Sample line	1 to 10^6 mr/hr	As determined by methodology of the ODCM	Not applicable	6 dec. log/ 6 dec. linear
Turbine Building Vent Stack Exhaust Monitor, and Standby Gas Treatment Vent Stack Exhaust	1 1 1 1 1	Particulate Filter Iodine Cartridge Noble Gas* Low Range (Beta Scin illation) Mid Range (cadmium telluride) High Range (cadmium telluride)	TB1-1C1108B TB2-2C1108B SBGT-OC194B	3.4x10^{-7} to 3.4x10^{-1} μCi/cc Xe^{133} 10^{-4} to 10^2 μCi/cc Xe^{133} 10^{-1} to 10^5 μCi/cc Xe^{133}	As determined by methodology of the ODCM	Not applicable	Digital Readout and Digital Printout

TABLE 11.5-1

PROCESS AND EFFLUENT RADIATION MONITORING SYSTEMS

MONITORED PROCESS	NO. OF CHANNELS	DETECTOR TYPE	DETECTOR LOCATION	CHANNEL RANGE	WARNING ALARM	TRIP	SCALE
B. SYSTEMS REQUIRED FOR PLANT OPERATION							
Reactor Building Vent Stack Exhaust Monitor	1 1 1	**Particulate Filter** **Iodine Cartridge** **Noble Gas*** Low Range (Beta Scin illation)	RB1-1C299B RB2-2C299B	$3.4x10^{-7}$ to $3.4x10^{-1}$ μCi/cc Xe^{133}	As determined by methodology of he ODCM	Not applicable	Digital Readout and Digital Printout

TABLE 11.5-2

RADIOLOGICAL ANALYSIS SUMMARY OF ROUTINE EFFLUENT SAMPLING

Sample Location	Basis For Location	Process Flow	Sampling Frequency	Analytical Procedure	Expected Sensitivities
Reactor Building Vents (isokinetic probe)	Determination of identity, concentration & quality of radionuclides released	125,000 cfm (Unit 1) 109,000 cfm (Unit 2)	Continuous	Gross Activity	1 x10^{-6} μCi/cc (Xe-133 equivalent)
			Once per month or when there is a change in monitor count rate from an unknown cause.	Analyzed for noble gas gamma emitting nuclides.	1 x 10^{-4} μCi/cc (for each isotope)
Unit 1 Turbine Building Vent (isokinetic probe)	Determination of identity, concentration & quantity of radionuclides released	174,500 cfm	Continuous	Gross Activity	1 x 10^{-6} μCi/cc (Xe-133 equivalent)
			Once per month or when there is a change in monitor count rate from an unknown cause.	Analyzed for noble gas gamma emitting nuclides.	1 x 10^{-4} μCi/cc (for each nuclide)
Unit 2 Turbine Building Vent (isokinetic probe)	Determination of identity, concentration & quantity of radionuclides released	121,300 cfm	Continuous	Gross Activity	1 x 10^{-6} μCi/cc (Xe-133 equivalent)
			Once per month or when there is a change in monitor count rate from an unknown cause.	Analyzed for noble gas gamma emitting nuclides.	1 x 10^{-4} μCi/cc (for each nuclide)

TABLE 11.5-2

RADIOLOGICAL ANALYSIS SUMMARY OF ROUTINE EFFLUENT SAMPLING

Sample Location	Basis For Location	Process Flow	Sampling Frequency	Analytical Procedure	Expected Sensitivities
Standby Gas Treatment Exhaust Vent (isokinetic probe)	Determination of identity, concentration & quantity of radionuclides released	11,000 cfm	Continuous Once per month or when there is a change in monitor count rate from an unknown cause.	Gross Activity Analyzed for noble gas gamma emitting nuclides.	1×10^{-6} μCi/cc (Xe-133 equivalent) 1×10^{-4} μCi/cc (for each nuclide)
Sample Tanks	Determination of identity, concentration & quantity of radionuclides released	100 gpm	Continuous during discharge	Gross activity monitored during discharge Grab sample prior to discharge to identify gamma emitting nuclides	1×10^{-6} μCi/cc As per R.G. 1.21 Rev. 1 June 1974 for grab samples.

TABLE 11.5-3

COMPOSITION & CONCENTRATION OF NUCLIDES IN
ROUTINE EFFLUENT SAMPLES

A. Liquid Effluents – Common Station Effluent Pipe

Nuclide	Expected Concentration (µCi/ml)
Corrosion and Activation Products	
Na-24	3.52E-09
P-32	2.76E-10
Cr-51	9.05E-09
Mn-54	1.13E-10
Mn-56	1.08E-09
Fe-55	1.63E-09
Fe-59	4.77E-11
Co-58	3.27E-10
Co-60	6.53E-10
Ni-65	5.03E-12
Cu-64	9.05E-09
Zn-65	3.27E-10
Zn-69m	6.28E-10
Zn-69	6.79E-10
W-187	1.46E-10
Np-239	5.53E-09
Fission Products	
Br-83	6.53E-11
Sr-89	1.58E-10
Sr-90	1.26E-11
Y-90	5.03E-12
Sr-91	9.05E-10
Y-91m	5.78E-10
Y-91	1.03E-10
Sr-92	2.46E-10
Y-92	9.80E-10
Y-93	9.55E-10
Zr-95	1.26E-11
Nb-95	1.26E-11
Nb-98	2.51E-12
Mo-99	1.76E-09
Tc-99m	3.77E-09
Ru-103	3.02E-11
Rh-103m	3.02E-11
Ru-105	1.43E-10
Rh-105m	1.46E-10
Rh-105	1.66E-10
Ru-106	5.03E-12
Rh-106	5.03E-12

TABLE 11.5-3

COMPOSITION & CONCENTRATION OF NUCLIDES IN ROUTINE EFFLUENT SAMPLES

A. Liquid Effluents – Common Station Effluent Pipe

Nuclide	Expected Concentration (µCi/ml)
Rh - 106	5.03E-12
Fission Products	
Te-129m	6.03E-11
Te-129	4.02E-11
Te-131m	5.53E-11
Te-131	1.01E-11
I-131	2.26E-09
Te-132	1.01E-11
I-132	6.03E-10
I-133	1.11E-08
I-134	6.28E-11
Cs-134	1.94E-10
I-135	3.77E-09
Cs-136	4.52E-10
Cs-137	1.28E-10
Ba-137m	1.21E-10
Ba-139	3.77E-11
Ba-140	5.53E-10
La-140	3.27E-10
La-141	5.28E-11
Ce-141	5.28E-11
La-142	3.02E-11
Ce-143	1.76E-11
Pr-143	5.78E-11
Ce-144	5.03E-12
Pr-144	5.03E-12
Nd-147	5.03E-12
Others	1.26E-11
Totals	6.31E-08
H-3	1.48E-05
Total w/H3	1.49.1E-05

TABLE 11.5-3

COMPOSITION & CONCENTRATION OF NUCLIDES IN ROUTINE EFFLUENT SAMPLES

B. Gaseous Effluents

		Expected Concentrations (µCi/CC)	
Nuclide	Reactor Bldg. (each unit)	Unit 1 Turbine Bldg.	Unit 2 Turbine Bldg.
I-131	3.72E-14	2.96E-11	3.86E-11
I-133	5.39E-13	3.22E-10	4.16E-10
Ar-41	6.27E-09	4.44E-09	5.92E-09
Kr-83m	0.00E+00	0.00E+00	0.00E+00
Kr-85m	1.67E-09	3.33E-09	4.44E-09
Kr-85	0.00E+00	5.77E-08	7.69E-08
Kr-87	8.36E-10	2.71E-09	3.61E-09
Kr-88	1.67E-09	4.04E-09	5.38E-09
Kr-89	8.36E-10	3.22E-08	3.43E-08
Xe-131m	0.00E+00	6.21E-09	8.28E-09
Xe-133m	0.00E+00	0.00E+00	0.00E+00
Xe-133	4.60E-08	6.32E-07	7.78E-07
Xe-135m	2.51E-08	1.35E-07	2.37E-08
Xe-135	5.31E-08	1.88E-07	1.67E-07
Xe-137	7.73E-08	6.28E-08	5.92E-08
Xe-138	3.34E-09	4.48E-08	5.92E-08
Cr-51	4.60E-15	2.17E-15	8.28E-16
Mn-54	5.85E-15	9.14E-15	3.55E-16
Co-58	1.25E-15	8.88E-16	5.92E-16
Fe-59	1.63E-15	7.10E-16	5.92E-17
Co-60	2.09E-14	1.61E-14	7.57E-16
Zn-65	2.09E-14	3.40E-15	3.65E-15
Sr-89	2.09E-16	2.66E-15	3.55E-15
Sr-90	4.18E-17	8.88E-18	1.18E-17
Nb-95	4.18E-14	1.15E-17	3.55E-18
Zr-95	4.18E-15	1.79E-15	2.37E-17
Mo-99	2.76E-13	8.94E-16	1.18E-15
Ru-103	1.76E-14	2.44E-17	2.96E-17
Ag-110m	1.00E-17	0.00E+00	0.00E+00
Sb-124	2.09E-16	2.00E-16	5.92E-17
Cs-134	1.96E-14	6.12E-15	1.07E-15
Cs-136	2.09E-15	4.66E-16	6.21E-16
Cs-137	2.51E-14	1.13E-14	3.23E-15
Ba-140	9.19E-14	6.89E-15	9.17E-15
Ce-141	3.76E-15	4.45E-15	5.92E-15
H-3	3.59E-08	1.33E-09	1.78E-09
C-14	0.00E+00	2.11E-09	0.00E+00

11.6 LOW LEVEL RADWASTE HOLDING FACILITY (LLRWHF)

11.6.1 Design Basis

The Low Level Radwaste Holding Facility (LLRWHF) has no safety related functions. It is designed to provide temporary on-site storage for low level radioactive waste (LLRW) and radioactive material produced at the Susquehanna Steam Electric Station (SSES). This storage capability is required whenever shipment off-site is temporarily restricted by unavailability of disposal site, unavailability of shipping casks, transportation problems, economic considerations, or other such problems that hamper LLRW disposal.

The original plant design provided very limited storage capacity based on the assumption that the waste would be continuously shipped off-site for disposal. The LLRWHF allows temporary shipping interruptions to occur without impacting plant operations.

This facility also aids in reducing the total man-rem exposure associated with the disposal process by providing the means to allow significant decay before off-site shipment and disposal activities occur.

Codes and standards applicable to the Low Level Radwaste Holding Facility are listed in Table 3.2-1.

11.6.2 LOW LEVEL RADWASTE HOLDING FACILITY DESCRIPTION

The Low Level Radwaste Holding Facility (LLRWHF) is a structural steel frame building with uninsulated metal siding and roofing to provide weather protection. The metal siding and roofing is designed for nominal wind loads and a snow load. The steel frame is designed for the wind and snow loads plus UBC Seismic Zone I loads, a 30 ton bridge crane, 4-10 ton monorails, and dead loads. See Dwg. B1P-005, Sh. 1 for the storage plan general arrangement in the LLRWHF. The building encloses a system of reinforced concrete waste storage vaults. For initial facility operation, two concrete vaults are provided for storage of waste liners (Liner Vault) and dry active waste (DAW) (Trash Vault) and are located in the western half of the building. The eastern half is referred to as the Open Area and is used for storage of radioactive material and shielded liner storage modules (LSMs). An additional concrete vault may be constructed over the eastern half of the building at a later date to accommodate additional storage.

The foundation consists of slab on grade with grade beams running around the perimeter of the building as well as underneath the vault walls. The slab is designed for a 250 psf floor load as well as the steel framing column loads, except in the truck bay which is designed for an AASHO H20-S16 load.

The reinforced concrete vaults consist of concrete walls. There is a wall which divides this area into two separate vaults. This entire area is covered by precast concrete panels with a total of 395 circular plugs which are individually removed while a waste liner is being placed in or retrieved from storage. These precast panels are supported by a structural steel framing system. Both the precast panels and framing system are designed for either a 100 psf uniform load or a 58 kip load from a waste liner and its shielding (shield bell) resting at individual locations, which ever is greater. The walls of this area are designed to withstand a total tornado pressure of 300 psf.

The shielded liner storage modules (LSMs) consist of a reinforced concrete right circular cylinder with removable lid. The concrete thickness on the module top and sides is used to provide radiation shielding for both transport and storage of solidified waste liners in the east open area. The LSMs are capable of being stacked a maximum of 2 modules high to maximize storage capacity in the open area.

For initial operation, waste liners may be stored in the west vault and within LSMs in the open area. DAW is stored in the adjacent east vault. An inspection area is provided in the west vault to allow appropriate waste liner inspections. In the adjacent vault, where DAW will be stored, a labyrinth is provided to allow access by a forklift truck. In addition, a stairway is placed up, over, and back down the vault wall in the south east corner for emergency egress while this vault is used for trash storage. At a later date, if the east vault is converted for use as a waste liner storage vault, an inspection station could be added and the emergency exit closed. High integrity containers containing dewatered waste will be stored in the east open area in LSMs. DAW and radioactive material may also be stored in the east open area.

DAW will be stored in the reinforced concrete Trash Vault or in the Open Area provided it is non-combustible and in accordance with the appropriate fire protection requirements. DAW and radioactive material will be packaged in a manner that will prevent the spread of contamination. Typically, containers that are ‘strong-tight’ or comply with the requirements of 49CFR173 will be used to minimize the need for repackaging prior to shipment. Solidified waste will be stored in reinforced concrete vaults or LSMs located within the LLRWHF. Dewatered waste may be stored in LSMs located in the Open Area. Solidified and dewatered wastes stored in the LLRWHF will typically be packaged in that meet disposal facility requirements in effect at the time of processing to minimize the need for repackaging prior to shipment.

If required, an additional concrete vault could be constructed in the eastern portion of the building to provide additional storage. This area could be enclosed by concrete walls.

A concrete wall is located along the north side of the truck bay. This wall provides shielding during waste liner storage or retrieval in the vaults.

A control room is located at the north east corner of the facility. It has concrete walls on the south and west sides and metal siding with insulated sheetrock walls on the north and east sides. The ceiling is insulated acoustical panels below the metal roofing.

A battery charging station and parking area is located adjacent to the west wall of the control room. It has concrete walls with a roll-up door into the truck bay. The roof is insulated metal roofing.

The Low Level Radwaste Holding Facility (LLRWHF) provides the following:

- interim storage of LLRW generated by five years of operation per unit (current waste generation rates are less than those projected as a result of successful volume reduction efforts)

- interim storage period not to exceed five years (extended storage afforded by reduced waste generation rates shall not be allowed until the LLRWHF has been evaluated with respect to the provisions of Information Notice 90-09)

- operation of the facility as necessary to hold the LLRW when licensed off-site disposal facilities are unavailable or not cost effective.
- the facility may be utilized for temporary projects which have been evaluated on a case by case basis and approved by supervision.

11.6.3 LLRWHF WASTE DESCRIPTION AND CAPACITY

The LLRWHF is designed to store low level DAW, dewatered waste, and solidified waste generated by the Susquehanna SES. The facility may also be used to temporarily store pieces of contaminated plant equipment and radioactive material. The LLRWHF will not be used to store gaseous wastes nor wastes containing free liquids.

The LLRWHF may be used for temporary projected related to the storage of and processing of radioactive material. When used for such projects temporary enclosures, supplemental HEPA ventilation and other containment control practices will be utilized as necessary to control the release and spread of contamination and airborne radioactivity. Each temporary project will be evaluated separately to assure conformance to the FSAR including Section 12.5, Health Physics Program, and design base requirements.

Dry active waste is defined as contaminated material containing sources of radioactivity dispersed in small concentrations throughout large volumes of inert substances, and has no free liquid. It generally consists of paper, high efficiency particulate air (HEPA) and cartridge filters, rags, clothing, small equipment, and other dry materials.

Solidified waste is defined as wet, dewatered waste in the form of resins and sludges which is solidified in a media, meets the free liquid criteria of NUREG 0800, SRP 11.4, BTP ETSB 11-3, and satisfies applicable transportation and disposal site requirements. Dewatered resins and sludges meeting the free liquid criteria of NUREG 0800, SRP 11.4, BTP ETSB 11-3, and satisfies applicable transportation and disposal site requirements are classified as solidified waste.

This definition is intended whenever solidified waste is referenced.

Original LLRWHF storage configuration design estimates of the annual waste generation rates range from 1100 to 1800 m^3 (39,000 to 63,000 ft^3) based on operation of both units. A nominal figure of about 1700 m^3 (60,000 ft^3) was chosen. Table below gives a breakdown of the low-level waste volume by source and waste type. After four years of operation the two Susquehanna SES units will have generated approximately 6800 m^3 (240,000 ft^3) of LLRW that will have required storage; the capacity of the LLRWHF will be about 6800 m^3 (240,000 ft^3).

Original Design Estimated Annual Low-Level Waste
Generation Rate For Operation of Both Units

Source-Waste Type	m^3/yr	ft^3/yr
DAW--compacted	500	18,000
DAW--non-compactible	150	5,300
Evaporator bottoms (25 wt%)	510	18,000
Resins	90	3,200
Waste sludges	450	16,000
TOTAL	1,700	60,500

Current estimated waste generation is described in Section 11.4.

11.6.4 LLRWHF SOURCE TERM

Source term calculations for the LLRWHF were performed for eight (8) radwaste/container combinations: LRW/HIC, Condensate Demin Bead/HIC, Condensate Demin Bead/Steel, Sump Sludge/steel, RWCU/HIC, Cartridge Filter/HIC, Compactible DAW/ strong tight and Non-Compactible DAW/strong tight. Actual radwaste generation volumes for 1990 and 1991, rather than design basis quantities, were used to determine generation rates for the source term calculation. Isotope inventory was determined for each waste type based on information submitted in SSES Semiannual Effluent and Waste Disposal (SAE & WD), Reports from 1990 and 1991. The maximum isotope activity for each period was used to develop shine and release source terms. Shine source terms contain only gamma-emitting isotopes; release source terms contain all isotopes. The total period of generation is five years. Total radwaste generation by type was determined by dividing the average actual generation rate in two years by nominal container volumes supplied. The accumulation rate (number of containers produced in five years) was used to determine whether accumulation is monthly, quarterly or annually. Decay calculations were performed for each type of waste based on the accumulation rate. The following generation rates and container volumes were used as inputs to the calculations:

GENERATION RATE (ft^3/yr) OF WASTE

RADWASTE TYPE	1990	1991
C/D Bead, steel liner	1050	1500
C/D Bead, HIC	5940	5610
Sump sludge, carbon steel	65.6	143.1
LRW HIC	745.6	1739.4
RWCU HIC	55	330
Cartridge Filter, HIC	78	78
Compactible DAW, strong-tight	3457.9	2549.3
Non-compact DAW, strong-tight	207.6	456.74

VOLUME (ft^3) OF CONTAINERS

RADWASTE TYPE	VOLUME
C/D Bead, steel liner	150
CD Bead, HIC	165
Sump sludge, carbon steel	70
LRW HIC	150
RWCU HIC	51
Cartridge Filter, HIC	39
Compactible DAW, strong-tight	64
Non-compact DAW, strong-tight	64

Results of the LLRWHF Source Term Calculation are shown in Reference 11.6-2, Attachment B.

Radioactive material other than that described above (for example contaminated equipment such as scaffolding and large components such as turbine blades) have not been explicitly modeled. Dose profiles based on storage of “maximum activity vans” 8 feet wide by 20 feet long by 8 feet high (10 mR/hr @ 2 meters) and “typical activity vans” (20 mR/hr @ 1 cm) assuming a Co-60 gamma spectrum have been developed to bound offsite dose consequences.

11.6.5 CONTAINERS

The radwaste containers which will be stored in the LLRWHF are designed to preclude or reduce the occurrence of uncontrolled release of radioactive materials due to handling, transportation or storage.

DAW and radioactive material will be packaged in a manner that will prevent the spread of contamination. Typically, containers that are ‘strong-tight’ or comply with the requirements of 49CFR173 will be used to minimize the need for re-packaging prior to shipment. Solidified waste will typically be packaged in containers that meet disposal facility requirements in effect at the time of processing.

At the present time, and for the foreseeable future, there is a wide profusion in sizes and shapes of disposal containers in use in the nuclear industry. Each has its own advantages and applications. It is expected that during the life of the facility it will be required to accommodate several of these different container types manufactured from steel or polyethylene material as deemed acceptable in the nuclear industry.

11.6.6 CONTAINER SHIELDING FOR TRANSPORT AND STORAGE

Transport and storage of high integrity containers (HICs) from the Radwaste Processing Facility to the LLRWHF will utilize LSMs. They serve as a shield for radwaste containers while being transported, moved and stored to minimize radiation exposure to personnel (Reference 11.6-9). LSMs are presently in use throughout the nuclear industry for onsite storage.

Steel liners are typically transported to the LLRWHF in a concrete shielded transfer cask. A shield bell may be used to move liners from the transfer cask to their storage location to minimize radiation exposure to operating personnel. See Figure 11.6-3.

The shield bell incorporates an electrically operated grapple for securing the liner within the shield bell for liner transport. The grapple is operated by an electric motor driven hoist which raises or lowers the grapple into position to engage a waste liner. The shield bell is supported by the bridge crane and receives necessary power and control signals via the crane bridge from a control panel located in the LLRWHF control room.

11.6.7 CRANE LOADING AND UNLOADING SYSTEM

The LLRWHF 30-ton crane with the 10-ton girder mounted monorail hoist is provided with a closed circuit television system for use in operating the crane to handle radwaste liners and vault shield covers plugs remotely. Both loads are imposed on the crane bridge simultaneously. The crane bridge is provided with closed circuit television cameras along with bridge mounted lighting fixtures to optimize visibility of crane loads-liners, shield covers, shield bell--and the path of crane motion. In addition, the crane is provided with a control pendant to facilitate local operation.

Provision is made for crane retrieval from the liner storage vault area back to the truck bay by releasing the crane brakes manually should there be a loss of power.

Provision has also been made to facilitate relocating the crane from the west vault to the east vault. Lifting eyes are provided to facilitate lifting the crane from the runway rails, over the truck bay, using hoist/trolleys provided on monorails over the truck bay. In this manner, the crane may be moved from the west vault to the east vault or vice-versa.

11.6.7.1 LLRWHF 30-Ton Crane Operation

The crane motion speed controls operate in accordance with the following requirements:

The main hoist operates through 5 speed steps and provides a maximum hook speed of 15 feet/minute (fpm). The hoist drive is equipped with a completely independent and separate microdrive system. The microdrive system drives the hoist at 6 inches/minute (maximum) to obviate impact loads and achieve the required load positioning accuracy of ±1/2 inch. An electrical interlock permits lowering the load over the storage vault using the microdrive system only.

The crane main hoist is interlocked with the shield bell grappling device. It may be energized for lifting or lowering the shield bell when the liner is attached to the shield bell, or when the liner is not attached to the shield bell and the liner grapple is withdrawn into the shield bell.

The drive for the trolley operates through 5 speed steps and provides maximum trolley speed of 50 feet per minute. The trolley is also equipped with a completely independent and separate microdrive system. The microdrive system drives the trolley at 6 inches/minute.

The main drive for the crane bridge operates through 5 speed steps and provides a maximum bridge speed 70 feet per minute. The drive is also equipped with a separate and independent microdrive system. The microdrive system drives the crane bridge at 6 inches/minute so as to provide no impact.

The motor-driven monorail hoist operates at a maximum speed at 15 fpm. The hoist is provided with a microdrive for a speed of 6 inches/minute. An electrical interlock permits lowering the load over the storage vault using the microdrive system only. The monorail hoist is provided with an alarm signal at the control panel for loads exceeding 7 tons.

The controls for operating the shield bell hoist motor are located on the shield bell control unit on the crane control panel and interlocked with the crane main hoist.

The crane positioning system accuracy meets the requirements for locating concrete vault shield plugs.

11.6.7.2 Liner Storage in Vault

The 30-ton capacity crane main hoist is used to lift radwaste liners from the transportation vehicle parked in the east end of the Truck Bay of the LLRWHF. The radwaste liner may be contained inside a shield. The assembly--shield bell and liner--is lifted by the crane main hoist and transported into the vault area.

The 10-ton monorail hoist supports the shield plug grapple which is used to engage and lift the vault shield plug. The crane bridge is then moved to position the liner and shield bell, if used, directly over the vault opening from which the shield plug has been removed. The liner is lowered into the vault and released.

When used, the shield bell hoist raises the empty liner grapple up into the shield bell and the crane main hoist lifts the shield bell from the vault roof recess.

The crane bridge is moved to realign the vault shield plug over the opening where the liner was placed and the vault shield plug is lowered into place.

11.6.7.3 Liner Retrieval in Vault

The 30-ton capacity crane is used to remove the liners from the vault. The 10-ton monorail hoist supports the shield plug grapple which is used to lift the vault shield plug.

When the shield bell is used, the crane bridge is then moved to position the shield bell directly over the vault opening from which the shield plug has been removed. The liner grapple is then lowered via the shield bell hoist, into the vault and the grapple is attached to the liner to be removed from storage. The shield bell hoist raises the liner into the shield bell. The crane main hoist lifts the shield bell from the vault opening and the crane bridge is moved to position the vault shield plug directly over the vault opening. The vault shield plug is lowered into place and the grapple is removed from the vault shield plug.

The liner and shield bell, if used, are moved by the crane to the appropriate position required by the next operation, which will be either the inspection station, another vault location or the truck bay for shipping.

11.6.7.4 Mobile Crane Operation

Mobile crane(s) with sufficient load capacity is used in the eastern half open area. This crane is used to lift LSMs containing waste in containers to/from the transportation vehicle parked in the LLRWHF. The entire assembly---LSM and container---is lifted by the crane from the transportation vehicle and placed into the storage area. This sequence is reversed for loading the entire assembly from the storage area to the transportation vehicle. Mobile cranes may also be used to lift radioactive materials and waste packages in the open area.

11.6.8 HEALTH AND SAFETY REQUIREMENTS

The LLRWHF is designed to limit off-site doses from the onsite storage of LLRW to a fraction of the 40CFR190 limits for the Susquehanna SES site, and on-site radiation exposure within the guidelines of 10 CFR 20. In both instances, the facility is designed to maintain dose rates ALARA as outlined in Regulatory Guides 8.8 (NRC 1979, Information Relevant to Ensuring that Occupational Exposures at Nuclear Power Stations will be As Low As is Reasonably Achievable) and 8.10 (NRC 1977, Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As is Reasonably Achievable). Exposure of on-site workers will be minimized by the use of concrete shielding around the stored material, shielded loading equipment, and controlled access to the facility. Since no radioactive materials would be released off-site, dose rates would be minimized through the use of shielding, distance, the self-shielding properties of the storage containers, and temporary shielding.

11.6.9 FLOOR DRAIN SYSTEM

Under normal conditions there will be no free liquids inside the building. Therefore, any free liquids entering the facility would come from sources such as: fire protection water; minute amounts of liquid from a breached container; rainwater or snow melt from roof leakage; cooling system or fuel leakage from equipment inside the facility; and snow brought in on vehicles. All such liquids will be considered contaminated until verified otherwise.

The floor drain and curb system will collect any liquids spilled on the floor of the facility. The system will route all drains to the collection sump located along the periphery of the building. The sump is equipped with liquid-detection devices that signal the facility control room whenever liquid enters the sump.

Sampling of the liquid in the sump is performed from inside the building. These liquids may be pumped to portable containers from inside the building and transferred to the main plant for processing. However, no permanent pumping equipment or piping is connected to the main plant.

11.6.10 LLRWHF HVAC SYSTEM

11.6.10.1 General Functions and Controls

The HVAC systems are designed to provide adequate air flow to remove heat and fumes, and maintain area design temperatures in the facility.

In the event of fire, the Storage Area Ventilating System, the Battery Charging Station Heating and Ventilating System, and the Truck Bay Vehicle Fume Exhaust System are automatically shutdown completely upon a signal from fire detection system. An override switch is provided in the control room to override the fire detection signal and purge smoke after a fire has been suppressed. To prevent outside air infiltration, backdraft dampers are provided for the fans exhausting room air directly to outside of the building and arranged to close automatically when fans are not operated.

11.6.10.2 Storage Area Ventilating System

The system has three separate subsystems and is designed to remove heat and fumes from the storage area and maintain the room design temperature of 100°F using outside air introduced through outside air inlets.

The subsystem equipment consists of five roof mounted exhaust fans for the area above vaults and open area ceiling, and ceiling mounted exhaust fan for the Trash Vault.

The system sequence of operation and controls are as follows:

1. The selected exhaust fans are started manually by individual hand switches located in the control room.
2. With the hand switches in the automatic position, exhaust fans are started automatically by room temperature switches when the temperature exceeds the set point of 100°F.
3. The Trash Vault ceiling fan is interlocked with the roof mounted exhaust fans, so that roof mounted exhaust fans are operated whenever ceiling fans are in operation unless the signals are overridden by the control room switches.

11.6.10.3 Control Room HVAC System

The control room HVAC system is designed to maintain room design temperatures during occupied periods as well as unoccupied periods and maintain positive room pressure to minimize air infiltration from the storage area.

The equipment consists of an air cooled cooling unit which consists of a low efficiency filter band, a supply fan, direct expansion cooling coil, and a multi-step electrical duct heater, with associated ductwork and controls.

The system sequence of operation and controls are as follows:

1. The supply fan on the cooling unit is started manually by a hand switch mounted on unit.
2. The supply fan may also be started automatically by room temperature switches when the hand switch is in the automatic position.
3. The set points of room temperature switches are 78°F and 90°F for summer, 70°F and 45°F for winter, during occupied and unoccupied periods respectively. The supply fan is started with either the cooling coil or the electric duct heater, as required.
4. When the system is started and the cooling or the heating mode is determined by room temperature switches, the system capacity will be adjusted automatically in response to room loads modulating the expansion valve on the cooling coil for cooling mode operation or adjusting the operating step on the electric duct heater for heating mode operation.
5. The supply fan continuously recirculates conditioned air with a fixed quantity of outside air for space pressurization.

11.6.10.4 Battery Charging Station Heating and Ventilating System

The system is designed to remove hydrogen gas produced during battery charging operation and maintain summer and winter room design temperatures. The exhaust fan operates continuous whenever the battery charging operation is required and maintains room pressure slightly below atmospheric to prevent the spread of hydrogen gas.

The equipment consists of a roof mounted exhaust fan, an electric unit heater and associated controls.

The system sequence of operation and controls are as follows:

1. The exhaust fan and electric unit heater are started manually by individual hand switches located on the local control panels.

2. With the hand switch in the automatic position, the exhaust fan is started when room temperature exceeds the set point of 100°F.

3. With the hand switch in the automatic position, room temperature is maintained above 40°F automatically by room temperature switch which starts heater.

11.6.10.5 Sprinkler Valve House Heating Systems

The systems are designed to maintain a minimum room temperature of 40°F for freeze protection. The equipment consists of two 50% capacity electric base board heaters and associated controls for each system.

The system sequence of operation and controls are as follows:

1. Heaters are started manually by individual local hand switches located in the valve houses.

2. The heaters can be started automatically by room temperature switches mounted inside of the valve houses. The set points for room temperature switches is 45°F for the primary heater and 40°F for the secondary heater.

3. If the room temperature drops below 35°F, low temperature switches activate audible low temperature alarms installed near the valve houses and also annunciate to the control room.

4. Normally, room temperature is maintained by the hand switch in the automatic position. The manual controls are required for operating tests.

11.6.10.6 Truck Bay Vehicle Fume Exhaust System

The system is designed to remove vehicle fumes produced during loading and unloading operations. The equipment consists of a wall mounted exhaust fan, with associated ductwork and manual controls.

11.6.10.7 Inspection Station Ventilation

One wall mounted exhaust fan is provided for heat removal from high intensity lighting in the inspection station. The fan exhausts to the area above the vaults and is manually controlled by hand switches on the local panel.

11.6.10.8 Air Inlets

Air inlets are placed to evenly distribute air entering the facility even when it is closed. Air inlets are located such that the accumulation of snow or other substances are prevented from restricting the flow of air and to prevent these substances from being drawn into the facility.

11.6.11 Effluent Monitoring

The LLRWHF was evaluated for potential unmonitored effluent pathways in accordance with the Offsite Dose Calculation Manual (ODCM). The LLRWHF will be used primarily to store solidified wastes and radioactive material. Gases and free liquids will not be stored. The solidified waste, along with radioactive material, will be properly confined and contained to prevent inadvertent airborne release. The LLRWHF has been categorized as an 80-10 System. All material stored in the LLRWHF will contain low levels of fixed or removable contamination. Materials with removable contamination will not be stored in open containers. Contamination on the external surfaces of the containers will be maintained below conditional release limits per Health Physics procedures. Containers which cannot be decontaminated to conditional release standards due to ALARA concerns will be decontaminated to <50,000 dpm/100 cm^2 and stored in LSMs or Vaults. Thus, airborne radioactivity in the LLRWHF would occur only in the event of breach of one or more barriers. LLRWHF general area and local air will be periodically sampled for airborne radioactivity. Any radioactivity detected will be included in determining compliance with site dose limits. Any such occurrences will be reported in the Annual Effluent and Waste Disposal Report. The LLRWHF would be decontaminated or additional safety evaluation would be performed for continued operation of the facility.

The LLRWHF may also be used for temporary projects related to the storage of and processing of radioactive material. When used for such projects temporary enclosures, supplemental HEPA ventilation and other contamination control practices will be utilized as necessary to control the release and spread of contamination and airborne radioactivity. In addition, the general area and local air in the vicinity of the work will be periodically or continuously sampled as necessary to ensure that any radioactivity detected is included in determining compliance with site dose limits.

11.6.12 ELECTRICAL SYSTEMS

11.6.12.1 480 Volt Distribution System

The 480 volt distribution system is a non Class IE, seismic category II design system rated for operation at 277/480 volt, 3 phase, 4 wire, 60 Hz with neutral solidly grounded. Source power for the LLRW holding facility is supplied by PPL Electric Utilities through a 300 KVA, 3 phase transformer. Underground service entrance conductors connect the transformer to the 480 volt motor control center located in the LLRWHF control room. Ground fault protection and overload protection is provided on the main incoming line breaker of the motor control center. All power for the LLRWHF is distributed at appropriate voltage levels from the control room. The motor control center is the main source of power for all LLRWHF loads. The exceptions are: battery packs, for emergency exit lighting; batteries used in the transponder for the fire alarm system; and batteries used for the annunciator system. Batteries are also provided to power the annunciators in the event of loss of power. There is no diesel backup or 125 volt DC station battery power for this facility. Metering, controls, and alarm relaying are provided in the LLRWHF control room. The motor control center has a spare section to allow for additional future loads.

11.6.12.2 Grounding System

The grounding system includes a grounding grid of 250 MCM cables to which every other building column is connected. The grounding grid is connected to the site main grounding grid. The structural steel for liner vaults is bolted together and connected directly to the building ground grid. Electrical and mechanical equipment, stairways, handrails and grating are connected to the building ground grid, either directly, or indirectly through the grounded structural steel. The 480 volt motor control center is connected to the building ground grid to provide a low resistance fault current return path. The system neutral in the MCC, and the lighting transformer neutral are connected to the facility grounding grid by individual grounding conductors.

11.6.12.3 Lighting System

Control of the lighting system is maintained in the LLRWHF control room. Lights on the outside of the LLRWHF over each personnel exit door are controlled by photocell. Roll-up doors have switches at the doors and small rooms are also switched locally.

Long life lamps are installed in low level radiation areas to minimize time spent by maintenance personnel in replacing lamps. General illumination has been provided above the vaults to assist the supplementary high intensity incandescent lights installed on the crane for the closed circuit television operation. High levels of illumination have been provided in the inspection station(s).

Outside area lighting for security purposes is integrated into the total plant system as described in Section 9.5.3 and is not a part of this lighting system description.

Lighting is obtained from the LLRWHF main service via the motor control center and lighting panels. In the event of loss of power, battery operated emergency exit lights have been strategically located to facilitate egress from the LLRWHF. Convenience receptacles are located throughout the LLRWHF.

Provision has been made for future expansion and the addition of extra lights anywhere in the LLRWHF to the extent of 25% of the existing lighting system. This provision is in the form of spare capacity.

11.6.12.4 Communication System

A telephone with a main plant extension is located in the LLRWHF control room. The control room and storage areas are furnished with an extension of the main plant intercommunication system with paging station and speakers. Public address handset stations for conversation and paging are located in the control room, off loading area, and the inspection station. PA speakers are located in the control room, truck bay and storage areas such that paging or alarm can be heard when the facility is at full capacity. The various communication systems are described in Section 9.5.2.

Provision is made to add additional PA speakers within the LLRWHF.

11.6.12.5 Cable and Raceway System

Systems that are an extension of the main plant are: telephone, intercommunications, and alarm. Spare conduits are provided for four systems: telephone, instrumentation, control and low voltage power.

Underground conduits are run in a duct bank between the PP&L 300 KVA transformer, located outside of the LLRWHF battery charging room and the 480 V motor control center in the LLRWHF control room.

11.6.13 CONTROL PANELS

Three control panels are located in the LLRWHF control room that consolidate all functions for the operation of the facility. The panels include; 1) HVAC and Radiation Monitoring, 2) Crane Control and Shield Bell Control and, 3) Fire Protection Panels.

The HVAC and Radiation Monitoring includes the switches that control the roof exhaust fans, the fume exhaust fan and the smoke override. Radiation indicators and annunciation of high radiation reading and high liquid level in the sump for the DAW and vault area are also included in the panel. A common trouble annunciator is also transmitted to the main plant power plant control room.

The crane control panel has all the controls necessary for remote operation of the LLRWHF 30-ton crane and TV monitors for viewing the operation. The shield bell control panel is a part of the crane control panel.

The fire protection panel includes all smoke and fire detection signals, fire alarm signals and the fire sprinkler system for the entire facility. Lighted zone annunciation and alarm bells are provided in the panel to locate trouble areas. All signals received by the panels are retransmitted to the main power plant fire control system as described in Section 9.5.1.

11.6.14 RADIATION ZONING

The facility is divided into radiation zones as described in Section 12.3.1.3.
Dwg. B1N-100, Sh. 1 shows the radiation zones for the proposed storage arrangement.

11.6.15 RADIATION MONITORING SYSTEM

The radiation monitoring system is designed to monitor the general area radiation levels at various locations-inside the DAW vault, the off-loading area, and the LLRWHF control room. The radiation monitor to be used is a gamma measuring device that has a sensor, an indicator, and power supply. The monitors' sensors are strategically located on the walls of the DAW storage areas, control room, and truck bay. There are area radiation monitors in the truck bay; one near the inspection station and one near the catch basin, and one in the control room. In the east vault used to store DAW, an area radiation monitor is near the north entrance and another near the emergency stairs at the south end.

Radiation levels detected by the sensors are sent to indicators located on the LLRWHF control room panel. Channels for additional monitors are provided.

11.6.16 SHIELDING

The LLRWHF contains four types of shielding; 1) fixed shielding for the in-place stored material, 2) transient shielding for waste containers for transport to the facility and for loading and unloading in the waste storage area, 3) Liner Storage Modules (LSMs) for the storage of higher activity items in the Open Area, and 4) temporary shielding that may be used for the radioactive material in the Open Area. Shielding design objectives are described in Section 12.3.2.1.

The fixed shielding consists of concrete storage vaults and modules for the solidified and DAW waste, concrete walls in the truck bay areas and concrete walls for the control room. The storage vault and module walls are reinforced concrete. The storage vaults and modules have precast concrete covers with removable plugs. Reinforced concrete walls are provided for shielding on the north and west sides of the truck bay areas. The control room has reinforced concrete along the south and west walls.

The transient shielding for the solidified wastes consists of portable shielding on the transport vehicle and a portable shield bell.

The shielded liner storage modules (LSMs) consist of a reinforced concrete right circular cylinder with removable lid. These are typically used for storage of the High Integrity Containers, but may also be used for other higher activity radioactive material and radioactive waste.

Temporary shielding such as lead blankets may be used in accordance with the radiation protection program to limit actual dose rate profiles to less than the bounding profiles to meet offsite dose criteria.

Further segregation of the waste containers within the vaults and modules will also be used to take maximum advantage of the self shielding properties of the waste material and to minimize exposure. To the maximal extent practicable, waste stored in the DAW storage vault will be arranged with containers having lower contact dose rates on the top layer of the storage areas and containers with higher contact dose rates stored underneath. Similarly, to the maximal extent practicable, solidified waste stored within the solidified waste storage vaults and modules will be arranged with lower contact dose rate containers stored on the outer perimeter and on the top layer. Containers with higher contact dose rates will be stored inside this perimeter.

11.6.17 SECURITY

The entire LLRWHF is within the restricted area security fence that encloses the Susquehanna SES and is under routine surveillance by plant security patrols. Access to the LLRWHF is administratively controlled through the use of a locked door(s).

11.6.18 FIRE PROTECTION SYSTEM OPERATING DESCRIPTION

11.6.18.1 Dry Pipe Sprinkler Systems

The dry pipe sprinkler systems are hydraulically designed to discharge at a density of 0.25 gpm/sf over the most remote 3000 sf. The design complies with NFPA #13 and provides protection to areas containing combustibles. Sprinkler heads in the storage area are equipped with fusible links rated at 286°F. When the links fuse upon high heat, the air in the piping system is released allowing it to fill with water and discharge at the design rate. A single AC powered air compressor supplies air to the dry pipe systems through the automatic filling system at each valve. Air pressure is maintained in the sprinkler system piping at approximately 50 PSI to maintain the water inlet valve in a closed position and prevent freezing of the protection system.

All radioactive material and radioactive waste shall be stored in an area which is approved in accordance with the fire protection program for the type of material and packaging used. The areas currently approved for storage of “combustible” containers are the DAW vault and the extreme northern section of the east open area, which are protected by the dry pipe sprinkler systems. Temporary projects may use the east open area without dry pipe sprinkler systems for the storage of and processing of radioactive material provided the requirements of the SSES Fire Protection Program are met.

11.6.18.2 Infra Red Smoke Detection Systems

Initially, smoke detectors are provided in the East Vault. Later, if required, additional detectors maybe added where required to assure an adequate detection system. The smoke detection system is comprised of units which project an infra-red beam the length of the vault. Each unit includes a transmitter and a receiver mounted below the vault roof. A smoke emitting fire causes attenuation of the beam at the receiver which generates a smoke alarm when 50% obscuration occurs. Total blockage of the beam is discriminated by the receiver, so that inadvertent blockage (during maintenance) does not result in a smoke alarm. A side benefit of the infra-red beam is that of fire detection when little or no smoke is present. Rising heat waves can cause the beam to shimmer and this pulsing or vibration is interpreted by the receiver as a fire condition. Thus, the infra-red system is in reality a dual mode smoke detection and fire detection system.

11.6.18.3 Photo Electric Smoke Detection System

The photo electric system is comprised of three (3) spot type ceiling mounted detectors of the conventional type operating on the light scattering, photodiode principle. They are located in the LLRWHF control room and will alarm at a smoke density of 1.5% light obscuration per foot.

11.6.18.4 Truck Bay Smoke Detection System

The smoke detection system in the Truck Bay consists of ionization detectors powered, operated, and controlled as a separate sub-system. Alarm transmission is delayed for a pre-determined time-out period to minimize false alarms.

11.6.18.5 Fire Alarm System

The fire alarm system is an extension of the existing plant system as described in Section 9.5.1. The LLRWHF incorporates the use of the Simplex Time Recorder Company basic transponder as the control panel for the local facility. It provides both normal and emergency back up power to the system from its own internal power supply.

All external fire alarm circuits from the transponder are electrically supervised against open and short circuits. In addition, the power supplied for the operation of the infra-red smoke detection system and the spot type smoke detectors is also supervised. Any malfunction is alarmed locally and the signal retransmitted to the plant central control room over supervised multiplex circuits from the Central Processing Unit (CPU).

The fire alarm system incorporates the use of separate alarm zones as follows:

A. The water flow switch of each of the sprinkler systems is connected to a common zone so that the operation of any system provides a common alarm.

B. The alarm contacts of each infra-red beam receiver are connected to a common zone so that the operation of any system provides a common alarm.

C. The alarm contacts of each photo electric smoke -detector are connected to a common zone so that the operation of any detector provides a common alarm.

D. The alarm contacts of the Truck Bay smoke detectors are connected to a common zone so that the operation of any detector provides a common alarm.

E. Several supervisory features for external devices are provided and the operation of any of these units provides a common alarm. Those supervisory features provided are as follows:

1. Low air pressure in a sprinkler system
2. Low air temperature in a valve house
3. Off normal position of any sprinkler system water supply valve.
4. Low air pressure or low temperature are also alarmed on a panel in the valve house and at an external horn (rated 100 DB) near the enclosure.

The control panel (transponder) is located in the LLRWHF control room and is provided with lighted zone annunciation and alarm bell. In addition, a slow whoop horn (rated at 100 DB) is located in the storage area.

11.6.19 EXPOSURE TO OPERATING PERSONNEL

The LLRWHF is designed to and will be operated to minimize the exposure to operating personnel while providing sufficient facility access. This will be accomplished by providing the necessary radiation shielding, by using current design technology and by using appropriate administrative controls to ensure that radiation levels are below applicable limits (10 CFR 20) for all phases of operation. Exposure of on-site workers will be minimized through the use of concrete shielding and shielded loading equipment. The number of operating personnel will be minimized and access to the LLRWHF will be controlled to further eliminate unnecessary radiation exposures.

The technical design and operating procedures of the LLRWHF maintain occupational doses ALARA, in accordance with current plant radiological zoning and control described in Section 12.3.

11.6.20 ENVIRONMENTAL CONSEQUENCES OF OPERATION OF THE LOW-LEVEL RADIOACTIVE WASTE HOLDING FACILITY

During routine operation, and during temporary projects that use the LLRWHF, no significant environmental consequences should occur related to the facility. No gases or liquids will be stored in the facility. Therefore, no releases of radioactive gaseous or liquid effluents should occur during routine operation. Small amounts of solid radioactive waste material could be released but would remain within the confines of the facility until the required decontamination and/or repackaging was completed.

The only expected radiation exposure pathway from operation of the LLRWHF is exposure from penetrating radiation originating within the facility either as direct radiation or as skyshine. Administrative and/or engineering controls assure annual dose remain below the 10CFR20 and 40CFR190 limits consistent with U.S. Nuclear Regulatory Commission (NRC) Branch Technical Position 81-38 (Reference 11.6-8). The dose equivalent from naturally occurring external sources in this geographical area is about 100 mrem per year to which operation of the LLRWHF would make no significant contribution.

11.6.21 LINER INSPECTIONS

The facility is equipped with provisions for liner inspections as required by Generic Letter 81-38 and 49CFR173. Inspections include the following:

1) Visual inspection of the container for deterioration, leakage, or other conditions which might preclude shipment, disposal, or require repackaging.

2) A contact radiation dose reading on the container surface.

3) A radiation dose reading at three feet from the outer surface.

4) An outer surface contamination smear.

5) Weighing of a liner.

The inspection method provides shielding for the person performing the inspections and remote operating capability for these functions to minimize the radiation exposure per ALARA principles (Reference 11.6-10). Inspection method is compatible with the loading system of the facility and is provided in one solidified storage vault bay with the provisions for installing another in the Trash Vault in the future.

The inspection method is equipped with the appropriate lighting to allow inspections. Provisions are made in the facility for electrical power for the system at the location it will occupy. Pre-shipment inspections may also be performed in the Radwaste Building.

In addition to pre-shipment inspections, a container storage inspection and monitoring program shall be implemented on at least a periodic basis. Use of high integrity containers would permit an inspection program of reduced scope. This program shall include the following:

1) Ensure container integrity and detect failures.

2) Capability for remote operation of inspection equipment.

3) Minimize radiation exposure per ALARA principles

In the unlikely event that radioactive contamination is discovered, the container would be transported back to the main plant for decontamination.

11.6.22 SAFETY ANALYSIS

Due to the facility design, the possibility of an equipment failure or serious malfunction is remote. Because strict administrative controls will be exercised during waste transfer operations the possibility of an accident caused by human error is also minimized.

However, an accident analysis has been performed to demonstrate the facility's capability to control or mitigate the consequences of postulated failures or accidents. These accidents are divided into two categories: 1) handling and storage accidents and 2) other accidents.

Radiological consequences for postulated accidents are evaluated at the site boundary and at the western perimeter of the site security fence. The minimum distance from the LLRWHF to the site boundary is provided for each specific accident location. The western perimeter fence is located 75 ft. west of the facility.

11.6.22.1 HANDLING AND STORAGE ACCIDENTS

Handling and storage accidents analyzed include drops, collisions and system failures such as loss of electrical power.

The probability of occurrence or the consequences of an accident or malfunction as a result of storing solidified-dewatered waste, as described in Section 11.6.3, in high integrity containers placed in LSMs were evaluated. Principal concern is from exposure to radiation due to loss of shielding. Radioactivity releases are minimized due to waste in the solidified form and container integrity. These accidents would not create airborne radiation hazards and cause significant releases to the environment.

Airborne radiation hazards are mitigated by the design of the containers used for packaging DAW, radioactive material, dewatered wastes, and solidified wastes, as described in Section 11.6.3. High integrity containers (HICs) are designed, manufactured, tested, and certified to meet disposal facility criteria. LSMs are tested to the applicable standards of 49CFR which include missile penetration and drops as specified in procurement orders. Tests performed ensure container integrity during handling, transportation and storage accidents. LSMs will absorb impacts with minimal damage but the integrity of the High Integrity Container will remain intact.

DAW and radioactive material will be packaged in a manner that will prevent the spread of contamination. Typically, containers that are ‘strong-tight’ or comply with the requirements of 49CFR173 will be used. Solidified waste will typically be packaged in containers that meet disposal facility requirements in effect at the time of processing. These actions minimize the consequences of potential handling and storage accidents.

The potential for drop accidents is minimal due to the operating and design features which are incorporated in the crane(s). Lifting lugs and cables are designed with a minimum factor of safety of 2. Container lifting devices remain engaged until an electrical or mechanical force is applied to release them. Lifting heights, travel speeds and distances are minimized to further reduce the possibility or consequences of a drop accident.

Collisions with transport vehicle, storage vaults and modules are possible. Vehicle speed and facility design minimizes impact on structural integrity and shielding capabilities. Dose rates from unshielded containers due to handling and storage accidents are bounded by Container Drop from a Transport Vehicle (Reference 11.6-7).

System failures such as loss of electrical power are analyzed in Loss of Offsite Power Accident (Reference 11.6-5).

11.6.22.1.1 CONTAINER DROP FROM A TRANSPORT VEHICLE

LLRW storage containers are transferred from the waste processing facility to the LLRWHF on a truck. A container drop from the transport vehicle is assumed to occur during transport as the result of a postulated vehicle collision or vehicle upset and the radiological consequences off site evaluated. Since waste materials inside the containers are in the form of DAW, radioactive material, and solidified wastes, a waste container damaged in a fall to the ground during transport would not create an airborne radiation hazard. However, direct radiation doses as a result of loss of shielding could result.

Solidified waste, as described in Section 11.6.3, containers will have the highest container radiation levels and are shielded with a portable shield while being transferred to the LLRWHF. A container drop accident could cause damage to or loss of the portable shielding resulting in direct radiation doses from the container. It is assumed for this accident that a solidified waste container with a design radiation level of 988 rads/hr is dropped with total loss of container shielding. The worst case drop location for the LLRWHF is immediately outside the southeast truck bay area and is also assumed. The resulting off site doses for this accident are 221 mrem TEDE at the western perimeter fence and 0.274 mrem TEDE at the site boundary. These doses are well within the dose criteria of 10CFR 50.67. The radiation sources and assumptions used for this calculation are given in Table 11.6-1.

11.6.22.1.2 HANDLING AND STORAGE ACCIDENTS INSIDE THE DAW STORAGE VAULT AND OPEN AREA

Handling and storage accidents inside the DAW storage vault and east open area such as container drops or a transport vehicle collision could result in damage to the containers. Since waste materials inside the containers are in the form of DAW, a breached container would not create an airborne radiation hazard. No waste material would leave the facility's confines until the required decontamination and/or repackaging was complete. Also, these accidents would have no impact on the structural integrity or the radiation shielding capabilities of the Trash Vault or the Open Area.

Therefore, there would be no off site radiological consequences due to these accidents.

11.6.22.1.3 DROPPING A HEAVY COMPONENT ONTO THE SOLIDIFIED WASTE STORAGE VAULT

The shield panels covering the solidified waste storage vaults are not designed to withstand the drop impact of a cell plug cover, LLRW container or another shield panel. However, the storage vaults are designed such that the supporting steel frame members are not affected by damage to, or failure of, one or more of the vaults shield panels. Therefore, although the panel and stored waste containers could be damaged by a heavy component drop, the structural integrity of the cell and the remaining facility would not be compromised. Since the waste is solidified, this postulated accident would not create an airborne radiation hazard off site. However, damage to the shield panels could cause a skyshine radiation dose off site.

Assuming two fully loaded storage vaults and no shielding provided by the damaged panels, the skyshine radiation dose rates are 1.9E-1 mrem/hr at the western perimeter fence and 4.8-4 mrem/hr at the site boundary. The sources and assumptions used in this evaluation are given in Table 11.6-2.

If this accident were to occur, the damaged containers would be covered by a spare shield panel and remain in the storage vault until the required decontamination, repair and/or repackaging could be completed. The total dose for the accident duration would be well within the dose criteria of 10CFR 50.67.

11.6.22.1.4 DROPPING A LLRW CONTAINER INTO SOLIDIFIED WASTE STORAGE CELL

During storage cell loading operations, it is possible that a waste container could be dropped into a partially loaded cell and damage the container and the storage cell contents. Since all waste stored in the cells is solidified, there would be no airborne radiation hazard to unrestricted areas. The damaged waste containers would remain shielded in the cell until the required decontamination, repair and/or repackaging could be accomplished. Therefore, no off site radiological consequences would result from this accident.

11.6.22.1.5 COLLISIONS - SOLIDIFIED WASTE CONTAINER HANDLING

Collisions which could occur inside the LLRWHF during handling of solidified waste storage containers would be collisions involving the container transport vehicle, overhead crane or both. Inside the LLRWHF, the transport vehicle moves no faster than 10 miles per hour and the overhead crane no faster than 50 feet per minute.

Collisions occurring at these slow travel speeds would have no impact on the structural integrity or the shielding capabilities of the storage vault and truck bay walls. Also, a collision with a storage container would only result in abrasive damage to the container or shield. The off site radiological consequences of collisions resulting in container drops are bounded by the container drop accident consequences.

11.6.22.1.6 SHIELDED LINER STORAGE MODULE ACCIDENTS

The storage plan requires the use of LSMs for container storage. Loss of shielding would result in radiation dose rates off-site. No radioactivity would be released off-site because container integrity would be maintained. The LSMs are designed, fabricated and tested to 49CFR specifications. The LSMs are fabricated from reinforced concrete which provides sufficient structural integrity. Design includes provisions for safe handling and storage. Testing provisions in 49CFR cover incidents normal to transportation which include penetration, drops and collisions.

Worst case accident would be failure of concrete shielding. Radiological consequences from LSM accidents are bounded by Container Drop from a Transport Vehicle.

If a lid from a LSM became dislodged during an accident, the skyshine radiation doses are 30 mrem TEDE at the western perimeter fence and 0.02 mrem TEDE at the site boundary for an accident duration of 8 hours. The sources and assumptions used in this evaluation are given in Table 11.6-5.

The total dose for the accident duration would be well within the dose criteria of 10CFR50.67.

11.6.22.2 OTHER ACCIDENTS

Other accidents include fires, freezes, tornadoes, floods, earthquakes and sabotage.

11.6.22.2.1 FIRES

A fire in the LLRWHF is extremely unlikely because all combustible materials will be stored inside the concrete Trash Vault or the Open Area in accordance with the fire protection program. Also, the facility is equipped with a fire detection/protection system described in Technical Facility Description. Therefore, a fire inside the facility would not result in significant damage to the LLRWHF or its contents.

The total curie content of DAW to be stored in the LLRWHF is given in Reference 11.6-2. In order to demonstrate that there would be no adverse radiological consequences for a fire in the Trash Vault because of the low activity levels of the DAW, an accidental release due to an unspecified incendiary event involving 100% of the stored DAW was evaluated. No credit was taken for the mitigating effects of the fire protection system. The resulting doses and the radiation sources and assumptions used for this calculation are given in Table 11.6-3. Since a fire in the Trash Vault would result in significantly less damage to the DAW than was assumed for the above analysis, the radiological consequence off site would also be significantly lower and well within the dose criteria of 10CFR 50.67.

11.6.22.2.2 FREEZES

The breech of a container due to water crystallization expansion of its contents is not possible because all stored waste will be solidified and contain less than 1.0 percent free liquid by waste volume in a container.

11.6.22.2.3 TORNADOES

In order to estimate the consequences of a tornado strike on the LLRWHF, a damage analysis was performed for a tornado with a wind velocity of 300 mph. The forces resulting from a tornado wind velocity of 300 mph considered in the analysis were:

Pressure on the windward side	185 psf
Suction on the leeward side	115 psf
Total force on the building	300 psf
Uplift on the roof	140 psf

In addition an atmospheric pressure drop of 3 psi over 3 seconds, remaining at 3 psi for 2 seconds, and then returning to 0 psi in another 3 seconds was considered.

Results of this analysis shows that the damage that would probably result to the LLRWHF from these forces is as follows:

1. Most of the metal roofing and siding will be blown away.
2. The structural steel framings, girts, and purlins will probably deform, however, the connection strength is sufficient to hold the structure together and prevent steel missiles from occurring.
3. Although deformations may result, the concrete walls around the vaults are sufficiently strong to withstand the tornado forces. Any missiles generated by the tornado may cause local damage, but will not affect the structural integrity of the walls.
4. When the 3 psi pressure drop occurs, the equivalent of 2 to 3 plugs will be raised off the Liner and Trash Vaults. This results in an immediate decrease in the pressure differential between the inside and outside of the vaults and as a result no additional damage should occur to the vault roof. If the plugs come off only partially or are tilted, additional plugs may be raised.
5. The walls along column 11 in the east open area are not designed to withstand tornado loads and will probably collapse. Minimal damage would result to LSMs if located in this area.
6. The wall at the north side of the truck bay is not designed to withstand the tornado loads and will probably collapse. However, since it is approximately 23 feet from the vault walls, the truck wall debris should cause no damage to the waste storage vaults.

LSMs would not sustain damage from the consequences of a tornado strike. Results include the following:

1. Although deformations of the LSM may result, the concrete construction is sufficiently strong to withstand the tornado forces. Missiles generated may cause local damage but will not affect the structural integrity.
2. If a lid would become dislodged during this event, the radiological consequences are bounded by Container Drop (LSM) from a Transport Vehicle (Reference 11.6-7) and evaluated by accident dose analysis for RWCU Resin in LSM Without Lid (Reference 11.6-6).

The use of the east open area for the storage of and processing of radioactive materials during temporary projects results in the same consequences from a tornado strike as analyzed above. Missiles which may be generated by the equipment and materials being processed or stored, are bounded by those potential missiles analyzed above. The structural integrity of the LSM's and existing concrete vaults remains intact. Each temporary project will be reviewed separately to assure conformance to these design base tornado requirements.

11.6.22.2.4 FLOODS AND SEISMIC EVENTS

The LLRWHF is designed for the maximum plant design rainfall intensity of 6 inches per hour. The facility does not have other special flood provisions because it is well above the Susquehanna River flood stage for the probable maximum flood. The grade elevation of the LLRWHF is approximately 215 m (700 ft) msl. This elevation exceeds both the probable maximum flood elevation of 167 m (548 ft) msl and the maximum historical flood elevation of 158 m (517 ft) msl.

The LLRWHF is a Non-Category I structure. Failure of this structure during a seismic event would not result in the release of significant radioactivity nor affect safe reactor shutdown.

11.6.22.2.5 Sabotage

Damage to the LLRWHF and its contents due to sabotage is highly improbably because of the inherently safe design. Also, the stored wastes are inert and low in radioactivity making them an unlikely sabotage target. Since the facility is within the site's secured area, is under routine surveillance by plant security patrols, and has access administratively controlled by a locked door, no accidents beyond those already considered are evaluated specifically for sabotage.

11.6.22.2.6 Loss of Off-Site Power

Electric power is provided to the LLRWHF from two sources. The normal power source is Salem 69-12kV substation via West Building tap and LLRW tap no. 2 (two). The alternate source is the Salem 69-12kV substation via Salem 46-01 line and LLRW tap no. 1 (one). The emergency source is the Berwick 66-12 kV substation overhead distribution line. Power is extended to the LLRWHF transformer via an underground duct bank from a point where manual transfer capability between the two power sources is provided.

Loss of off-site power to the LLRWHF could occur as the result of primary source failure, transformer failure or cable failure. Estimated outage duration for these failures are as follows:

Outage	Duration
Primary source failure personnel (transfer to alternate) available	1 Hr - Trained site
Transformer failure (replace transformer)	2 Hrs - Call-out of division trouble man
	12 Hrs - Occurs during normal work hours
	16 Hrs - Occurs off-hours
Cable failure	8 Hrs - Occurs during normal work hours
	12 Hrs - Occurs off-hours

The longest estimated outage duration is 16 hours for a failure of the LLRWHF transformer during off-working hours and is assumed for this accident.

Loss of off-site power to the LLRWHF would not affect the safe storage of waste in the facility. However, if an outage were to occur when a solidified waste container is in the process of being loaded into a vault storage cell with the overhead crane, off site doses could result. Since the overhead crane and auxiliary hoist are equipped with holding brakes to prevent dropping loads if electric power is lost, the shield bell and shield plug would remain suspended above the vault. Sources of radiation exposure would be a direct dose from the shield bell (depending on the location of the storage cell) and a skyshine dose from the waste in the uncovered storage cell. This would result in integrated doses off-site above those that would normally occur from solidified waste handling because of continued exposure from these sources during the outage. It is assumed for this accident that a solidified waste container with a design radiation dose rate of 2.32 rads/hr is in the process of being loaded into a vault storage cell when the power outage occurs. The resulting off-site doses for this accident are 1.78 TEDE mrem at the western perimeter fence and .00026 mrem TEDE at the site boundary. These doses are well within the siting criteria of 10CFR 50.67. The radiation sources and assumptions used for this calculation are given in Table 11.6-4.

It should be noted that it is possible to manually position the shield plug directly over the open cell by using winches and cables to move the overhead crane. This would provide an effective shield against skyshine from the open cell since the cover is only 8 inches above the vault. Also, once the shield plug is positioned above the cell opening, it could be replaced by manually controlling the holding brakes on the auxiliary hoist. However, for this analysis, continuous exposure from the shield bell and the open cell for the duration of the outage is assumed.

Handling LSMs during loss of off-site power would not result in a loss of shielding and subsequent doses as described above. The waste container is within the LSM during handling. It is assumed for this accident that a solidified waste container with a design radiation dose rate of 988 rads/hr is in process of being inspected when an 8 hour power outage occurs. The resulting off-site doses for this accident are 30 mrem TEDE at the western perimeter fence and 0.020 mrem TEDE at the site boundary. The radiation sources and assumptions used in this evaluation are given in Table 11.6-5. These doses are well within the siting criteria of 10CFR 50.67.

In the event that power is lost while the Facility is being used for temporary projects related to the storage and process of radioactive material, activities requiring the use of equipment to prevent spread or minimize the release of radioactivity will be stopped until power can be restored.

11.6.22.3 SUMMARY

The LLRWHF, and its associated storage containers, equipment and operating procedures provide a satisfactory interim storage facility which is capable of controlling and mitigating the radiological consequences of potential accidents and unplanned events. The radiological consequences for all accidents postulated for he LLRWHF are well within a small fraction of the dose guidelines specified in 10CFR 50.67.

11.6.23 REFERENCES

11.6-1 "Environmental Survey of Transportation of Radioactive Materials To and From Nuclear Power Plants," WASH-1238, December, 1972.

11.6-2 PP&L Calculation, EC-012-3149, "LLRWHF Source Terms, Verification and Validation of SOURCE.WK1 Program".

11.6-3 PP&L Calculation, EC-012-3150, “Exposure Rate Profiles for LLRWHF Sources in Liner Storage Modules”.

11.6-4 PP&L Calculation, EC-012-3151 "Exposure Rate Profiles for Maximum (Unshielded) Sources".

11.6-5 PP&L Calculation, EC-012-3152, “Exposure Rate Profiles for Compacted and Non-compacted Dry Active Waste (DAW)”.

11.6-6 PP&L Calculation, EC-012-3153, "Accident Dose Analysis for LLRWHF Dry Active Waste (DAW) Fire".

11.6-7 PP&L Calculation, EC-RADN-0516, "Offsite Dose Consequences Due to Container (LSM) Drop from Transport Vehicle".

11.6-8 PP&L Calculation, EC-RADN-0517, "Accident Dose Analysis for LLRWHF RWCU Resin in LSM Without Lid".

11.6-9 PP&L Calculation, EC-RADN-0518, "SSES Offsite Dose Analysis for Transporting Radwaste from Radwaste Building to LLRWHF."

11.6-10 PP&L Calculation, EC-RADN-0519, "Offsite Dose Analysis for Loss of Power Accident at LLRWHF".

11.6-11 PP&L Calculation EC-RADN-0520, "Annual Dose Analysis From Quarterly Inspections".

11.6-12 PP&L Calculation EC-RADN-0521, "Analysis for Normal Loading and Unloading Radwaste at the LLRWHF".

11.6-13 PP&L Calculation, EC-RADN-0522, "Offsite Dose Analysis for Heavy Component Drop on LLRWHF Vault Shield Panels".

11.6-14 PP&L Calculation, EC-RADN-0523, "Annual Dose and Exposure Rates to Walls for Operations at LLRWHF".

11.6-15 PP&L Calculation, EC-RADN-0524, "LLRWHF Calculation of Direct and Skyshine Dose Rates".

11.6-16 PP&L Calculation, EC-RADN-1033, "SSES Doses Resulting from Storage of Miscellaneous Radioactive Material at the LLRWHF".

11.6-17 PP&L Calculation, EC-ENVR-1025, "SSES Fuel Cycle 40CFR190 Dose Calculation".

11.6-18 PP&L Safety Evaluation NL-92-007, Operation of the LLRWHF at SSES.

TABLE 11.6-1

DESIGN BASIS SOURCES AND ASSUMPTIONS USED TO CALCULATE THE OFF-SITE RADIOLOGICAL CONSEQUENCES DUE TO A CONTAINER DROP FROM A TRANSPORT VEHICLE (REFERENCE 11.6-7)

A. Radiation Source

A design radiation level of 988 rad/hr contact is assumed for the solidified waste container and is based on a solidified-dewatered reactor water clean-up spent resin activity distribution. The source term for the reactor water clean-up spent resin is given in Reference 5.3.

B. Geometry of the Dropped Container

The geometry of the dropped container is based on SEG Radlok-200 Container

Diameter - 51.8"

Height - 60.3"

Volume - 51 cu. ft.

C. Distance from the Source to Receiver Point

The drop point is assumed to be at the south truck bay entrance to the LLRWHF on the southeast side of the building. This is the closest point to the dose receptor locations that the drop could occur. The source to receiver distances are as follows:

Drop point to western perimeter fence	= 287.5 ft.
Drop point to site boundary	= 2026 ft.

D. Duration of Accident

Total time the container is assumed to remain unshielded - 8 hrs.

E. Off-Site Radiological Consequences

Direct Dose Rate at Western Perimeter Fence	= 2.76+1 mrem/hr
Total Integrated Dose at Western Perimeter Fence	= 2.21+2 mrem TEDE
Site Boundary Direct Dose Rate	= 3.42-2 mrem/hr
Total Integrated Site Boundary Dose	= 2.74-1 mrem TEDE

TABLE 11.6-2

DESIGN BASIS SOURCES AND ASSUMPTIONS USED TO CALCULATE THE OFF-SITE RADIOLOGICAL CONSEQUENCES DUE TO DROPPING A HEAVY COMPONENT ONTO THE SOLIDIFIED WASTE STORAGE VAULT (REFERENCE 11.6-1)

A. Skyshine Radiation Source

To maximize accident source term, the following distribution was used:

2 - exposed undecayed solidified sump sludge liners

2 - exposed decayed solidified sump sludge liners

3 - peripheral decayed solidified sump sludge liners

9 - peripheral undecayed condensate demineralizer liners

Source term is given in Reference 5.2.

B. Skyshine Source Geometry

The total activity contained by two vaults assuming maximum capacity is used to determine an equivalent point source. No credit is taken for shielding provided by the damaged panels.

C. Distance from the Source to the Receiver Point

Distance from the facility to the western perimeter fence	= 75 ft.
Distance from the facility to the site boundary	= 1699 ft.

D. Off-Site Radiological Consequences

Direct Dose Rate at Western Perimeter Fence	= 1.9-1 mrem/hr
Site Boundary Direct Dose Rate	= 4.81-4 mrem/hr

TABLE 11.6-3

DESIGN BASIS SOURCES AND ASSUMPTIONS USED TO CALCULATE THE OFF-SITE RADIOLOGICAL CONSEQUENCES DUE TO AN UNSPECIFIED INCENDIARY EVENT INVOLVING 100% OF DAW (REFERENCE 11.6-6)

A. Source Term

The activity source term for this event is based on five years accumulation of DAW. The isotopic inventories are listed as follows.

Isotope	Inventory (Ci)	Isotope	Inventory (Ci)
Am-241	7.18E-03	H-3	6.32E-03
C-14	7.05E-03	I-129	2.77E-02
Cm-242	1.10E-03	Mn-54	1.18E+00
Cm-244	3.96E-03	Ni-63	1.83E-01
Co-58	8.62E-03	Pu-238	1.46E-04
Co-60	7.01E+00	Pu-239	6.04E-05
Cr-51	4.18E-03	Pu-241	1.06E-02
Cs-137	8.68E-05	Sr-89	6.01E-06
Fe-55	2.97E+01	Tc-99	6.13E-02
Fe-59	1.30E-02	Zn-65	1.46E-01

B. Meteorology

Western Perimeter Fence (EAB)	7.3E-03 sec/m^3
Low Population Zone (LPZ)	9.6E-05 sec/m^3
Control Room Habitability Envelope (CRHE)	3.8E-05 sec/m^3

C. Airborne Radiation

Percent of Particulate Assumed Released to Environment	1%

D. Distance from Source to Receiver Point

West Perimeter Fence	23 meters
Low Population Zone (LPZ)	4758 meters
CRHE	504 meters

E. Calculated Radiological Consequences

West Perimeter Fence	1.53E-01 Rem TEDE
Low Population Zone (LPZ)	2.02E-03 Rem TEDE
CRHE	8.03E-04 Rem TEDE

TABLE 11.6-4

DESIGN BASIS SOURCES AND ASSUMPTIONS USED TO CALCULATE THE OFF-SITE RADIOLOGICAL CONSEQUENCES DUE TO LOSS OF OFF-SITE POWER TO THE LLRWHF (REFERENCE 11.6-5)

A. Radiation Sources

A design radiation level of 2.32 rads/hr contact is assumed for the unshielded solidified waste container inside the shield bell and is based on a solidified sump sludge activity distribution. This source is also assumed for the waste inside the open storage cell. The source term is given in Reference 5.3.

The solidified waste container is a SEG LVM-100 and has the following dimensions:

Diameter - 73"
Height - 70.25"
Volume - 170 cu. ft.

B. Waste Handling Assumptions

It is assumed for this accident that a solidified waste container is in the process of being loaded into an open cell in the solidified waste storage vault when loss of off site power to the LLRWHF occurs. As a result, the following conditions based on waste handling procedures for solidified waste exist at the open storage cell, and are assumed for the duration of the accident:

- A solidified waste container is inside the shield bell and is suspended eight inches above the vault by the overhead crane.

- The shield plug from the open storage cell is suspended eight inches above the vault by the auxiliary hoist.

- The shield bell and plug are in a position that results in maximum skyshine exposure from waste in the open cell. In this position, approximately two thirds of the cell opening is shielded from skyshine by the shield bell and plug. Credit is taken for the shielding effects of the shield bell and plug.

- The cell is located in the outer row of the vault at the west end of the building which minimizes distances off site from the sources and provides a direct exposure path from the shield bell.

C. Distances from the Sources to the Receiver Point

Distance from the facility to the western perimeter fence = 75 ft.

Distance from the facility to the site boundary = 1699 ft.

TABLE 11.6-4 (CONTINUED)

DESIGN BASIS SOURCES AND ASSUMPTIONS USED TO CALCULATE THE OFF-SITE RADIOLOGICAL CONSEQUENCES DUE TO LOSS OF OFF-SITE POWER TO THE LLRWHF (REFERENCE 11.6-5)

D. Duration of Accident

The duration of the accident is assumed to be 16 hours. This is based on the longest estimated power outage which occurs for a failure of the LLRWHF transformer during off working hours. Continuous exposure from the shield bell and open storage cell is assumed for the duration of the accident.

E. Off-Site Radiological Consequences

Western Perimeter Fence

Direct Dose Rate from Shield Bell	= 1.09-E-01 mrem/hr
Skyshine Dose Rate from Open Storage Cell	= 2.19-E-03 mrem/hr
Total Dose Rate at Western Perimeter Fence	= 1.11-E-01 mrem/hr
Total Integrated Dose at Western Perimeter Fence	= 1.78 mrem TEDE

Site Boundary

Direct Dose Rate from Shield Bell	= 1.12-E-05 mrem/hr
Skyshine Dose Rate from Open Storage Cell	= 5.43-E-06 mrem/hr
Total Dose Rate at Site Boundary	= 1.66-E-05 mrem/hr
Total Integrated Dose at Site Boundary	= 2.66-E-04mrem TEDE

TABLE 11.6-5

DESIGN BASIS SOURCES AND ASSUMPTIONS USED TO CALCULATE THE OFF-SITE RADIOLOGICAL CONSEQUENCES DUE TO MOBILE SHIELDED STORAGE MODULE WITHOUT LID (REFERENCE 11.6-6)

A. Skyshine Radiation Source

An undecayed reactor water cleanup resin waste container is located within a storage module with lid removed. The source term for the reactor water cleanup resin is given in Reference 5.2.

B. Skyshine Source Geometry

The effects of objects in the line of sight providing additional shielding, such as other LSMs and the vaults, will not be considered.

C. Distance from the source to the receiver point distance from facility to the western perimeter fence = 75 ft.

Distance from facility to the site boundary	= 1960 ft.

D. Duration of Accident

Total time the container is assumed to remain unshielded -8 hrs.

E. Off-Site Radiological Consequences

Skyshine Dose Rate at Western Perimeter Fence	3.7 mrem/hr
Total Integrated Dose at Western Perimeter Fence	30 mrem TEDE
Site Boundary Skyshine Dose Rate	1.90-3 mrem/hr
Total Integrated Site Boundary Dose	2.0-2 mrem TEDE

THIS FIGURE HAS BEEN
REPLACED BY DWG.
B1P-005, Sh. 1

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Figure 11.6-1 replaced by dwg.
B1P-005, Sh. 1

FIGURE 11.6-1, Rev. 49

AutoCAD Figure 11_6_1.doc

THIS FIGURE HAS BEEN
REPLACED BY DWG.
B1N-100, Sh. 1

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Figure 11.6-2 replaced by dwg.
B1N-100, Sh. 1

FIGURE 11.6-2, Rev. 49

AutoCAD Figure 11_6_2.doc

Security-Related Information
Figure Withheld Under 10 CFR 2.390

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SHIELD BELL
ASSEMBLY

FIGURE 11.6-3, Rev 54

AutoCAD: Figure Fsar 11_6_3.dwg

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DELETED

FSAR REV. 65

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Figure Deleted

FIGURE 11.6-4, Rev. 55

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11.7 INDEPENDENT "Dry Fuel Storage"

11.7.1 Design Basis

Independent Dry Fuel Storage is necessary to provide temporary on-site spent fuel dry storage at Susquehanna Steam Electric Station (SSES). This storage capacity is required as a result of the unavailability of an off-site repository. The original plant design provided spent fuel storage capacity in the spent fuel pools based on the assumption the spent fuel would be continuously shipped off-site to a repository for disposal. The Independent Dry Fuel Storage system allows for the temporary storage of spent fuel without impacting plant operations.

The Independent Dry Fuel Storage system is composed of the Independent Spent Fuel Storage Installation (ISFSI) and the NUHOMS® Dry Spent Fuel Storage system. Although many portions of the ISFSI and NUHOMS® system are considered "Important to Safety", only the Lifting Yoke is considered "Safety-Related". No other component of the ISFSI and NUHOMS® system is considered "Safety-Related."

The Dry Fuel Storage System implements the NUHOMS® Horizontal Modular Dry Storage Systems (NUHOMS® System) offered by Transnuclear (formerly Vectra Technologies) and the NUHOMS® ISFSI. The components which make up the Transnuclear NUHOMS® Dry Storage System conform to all requirements of:

1. 10CFR72 as documented in Transnuclear "NUH-003 – Safety Analysis Report for the Standardized NUHOMS® Horizontal Modular System for Irradiated Nuclear Fuel" (known as the Certified Safety Analysis Report (CSAR),
2. USNRC "Safety Evaluation Report of Vectra Technologies Safety Analysis Report for the Standardized NUHOMS® Horizontal Modular Storage System for Irradiated Nuclear Fuel" (SER) and
3. USNRC "Certificate of Compliance for Dry Spent Fuel Storage Casks" Number 1004 (C of C).

The Dry Fuel Spent System is constructed and operated in accordance with general license requirements of 10CFR72. The SSES Spent Fuel Storage Project 10CFR72.212 Evaluation demonstrates that the spent fuel transfer and storage process, equipment and facilities meet the conditions and the requirements of the C of C.

Codes and standards applicable to the ISFSI are listed in both Table 3.2-1 and Transnuclear CSAR (Safety Analysis Report for the Standardized NUHOMS® Horizontal Modular Storage System for Irradiated Nuclear Fuel" [NUH-003])

11.7.2 NUHOMS® Horizontal Modular Dry Storage System Description (NUHOMS® System)

The NUHOMS® Horizontal Modular Dry Storage System (NUHOMS® System) provides for horizontal, dry storage of canisterized spent fuel assemblies in a concrete storage module. Three canister models are utilized at SSES, the 52B, 61BT and 61BTH. Each canister is capable of housing either fifty-two or sixty-one spent fuel assemblies. The 61BT and 61BTH canisters are transportable and the 61BTH canister is capable of housing High Burnup Rate fuel. Each concrete storage module houses one canister. The main components of the

NUHOMS® System are the Independent Spent Fuel Storage Installation (ISFSI), Horizontal Storage Modules (HSM), Dry Shielded Canisters (DSC) and associated transfer/ auxiliary equipment. The ISFSI is comprised of HSMs (105 maximum), basemats, approach slabs, roads, fencing, drainage system, lighting, lightning protection and Temperature Monitoring System. The transfer equipment consists of an On-Site Transfer Cask (TC), Transfer Cask Lifting Yoke, Transfer Trailer and rigging assemblies. The transfer equipment interfaces with the existing SSES Reactor Building Refueling Floor equipment, Unit 1 Reactor Building Crane and overall plant infrastructure. The auxiliary equipment consists of the Vacuum Drying System, the Helium Leakage Testing System and the Automatic Welding System.

11.7.3 ISFSI Source Terms

11.7.3.1 Radiation Source Term

The neutron and gamma radiation sources include the BWR spent fuel, activated portions of the fuel assembly, and secondary gamma radiation. All sources, except secondary gamma radiation, are considered physically bound in the source region. Secondary gamma radiation is produced by radiation passing through shielding regions.

Spent fuel assemblies with various combinations of burnup, enrichment, and cooling times can be stored in the DSCs. The criteria for spent fuel assembly parameters for each DSC model is specified in the C of C. Spent fuel assemblies which meet these criteria are bounded by the source strengths used in the design analysis for the NUHOMS® System and meet the criteria established in 10CFR72.104.

11.7.3.2 Airborne Radioactive Material Source Term

The release of airborne radioactive material is addressed for three phases of NUHOMS® System operation: irradiated fuel handling in the Reactor Building Cask Storage Pit and Spent Fuel Storage Pools, drying and sealing of the DSC, and DSC transfer and storage.

Potential airborne releases from irradiated fuel assemblies in the Reactor Building Cask Storage Pit and Spent Fuel Storage Pools are discussed in Section 12.2.2.

DSC drying and sealing operations are performed using procedures which prohibit airborne leakage. During these operations, all vent lines are routed to the plant's existing radwaste systems. Once the DSC is dried and sealed, there are no design basis accidents which could result in the release of airborne radioactivity.

During transfer of the sealed DSC and subsequent storage in the HSM, the only postulated mechanism for the release of airborne radioactive material is the dispersion of non-fixed surface contamination on the DSC exterior. By filling the TC/DSC annulus with demineralized water, placing an inflatable seal over the annulus, and utilizing procedures which require examination of the annulus surfaces for smearable contamination, the contamination limits on the DSC are below the permissible level for off-site shipments of fuel. Therefore, there is no possibility of significant radionuclide release from the DSC exterior surface during transfer or storage.

11.7.4 Dry Shielded Canister (DSC)

The BWR Dry Shielded Canisters (DSCs) used to store spent fuel in the Horizontal Storage Modules (HSMs) at the ISFSI are designed to preclude or reduce the occurrence of uncontrolled releases of radioactive materials due to handling, transfer or storage of spent fuel.

The CSAR describes the principal design features and design parameters for the DSC. The cylindrical shell, and the top and bottom cover plate assemblies form the pressure retaining containment boundary for the spent fuel. The DSC is equipped with two shield plugs so that occupational dose at the ends is minimized during drying, sealing, and transfer operations.

The DSC has double, redundant seal welds that join the shell and the top and bottom covers to form the containment boundary. The bottom end assembly containment boundary welds are made during fabrication of the DSC. The top end assembly containment boundary welds are made after spent fuel loading. Both top plug penetrations (siphon and vent ports) are redundantly sealed after DSC drying operations are complete. This assures that no single failure of the DSC top or bottom end assemblies will breach the DSC containment boundary.

The internal basket assembly contains a storage position for each of the spent fuel assemblies. Fixed neutron absorbing material is used for criticality control in the DSC. Subcriticality during wet loading, drying, sealing, transfer and storage operations is maintained through the geometric separation of the fuel assemblies by the DSC basket assembly and the neutron absorbing capability of the DSC materials of construction.

11.7.5 Horizontal Storage Module (HSM)

Prefabricated Horizontal Storage Modules (HSMs) are utilized to form an array of HSMs at the ISFSI. Each HSM provides a self contained modular structure for storage of spent fuel in a single DSC. The CSAR describes the principal design features and design parameters for the HSM.

The HSM is constructed from precast reinforced concrete and contains a structural steel support for the DSC and heat shields. Adequate radiation shielding is accomplished by adding shield walls at each end of a row of HSMs and/or adding a shield wall at the rear of each HSM if configured as single row of HSMs.

The HSM provides a means of removing spent fuel decay heat by a combination of radiation, conduction, and convection. Ambient air enters the HSM through ventilation inlet openings in the lower side walls of the HSM and circulates around the DSC and the heat shield. Air exits the HSM through outlet openings in the upper side walls of the HSM. Adjacent modules are spaced to provide a ventilation flow path between modules. The passive cooling system for the HSM is designed to assure that peak cladding temperatures during long term storage remain below acceptable limits to ensure fuel cladding integrity.

The HSMs are constructed on a load bearing foundation, which consists of a reinforced concrete basemat on compacted fill.

11.7.6 Transport Equipment

11.7.6.1 On-Site Transfer Cask

The On-Site Transfer Cask (TC) used in the NUHOMS® System provides shielding and protection from potential hazards while the DSC is loaded with spent fuel. The fully loaded TC has a gross weight of approximately 100 tons and is limited to on-site use under 10CFR72.

The TC is designed to provide sufficient shielding to ensure that dose rates are ALARA. Two lifting trunnions are provided for handling TC in the Reactor Building using a Lifting Yoke and the Unit 1 Reactor Building Crane. Lower support trunnions are provided on the TC for pivoting the TC from/to the vertical and horizontal positions on the Transfer Trailer.

11.7.6.2. Transfer Trailer

The Transfer Trailer consists of a heavy haul industrial trailer, support skid assembly/positioning system and hydraulic ram. The Transfer Trailer is designed to ride as low to the ground as possible to ensure that the TC height during transport is less than the 80 inch drop height used as the accident drop design basis for the TC/DSC. A conventional heavy haul tractor or other suitable mover is used to tow the Transfer Trailer.

11.7.7 NUHOMS® System Operation

The primary operations (in sequence of occurrence) for the NUHOMS® System are described in the following sections:

11.7.7.1 On-Site Transfer Cask (TC) Preparation

TC preparation includes exterior washdown and interior decontamination. These operations are performed in the Steam Dryer and Separator Storage Pit.

11.7.7.2 Placement of DSC in Cask

The empty DSC is inserted into the TC ensuring proper alignment by visual inspection of the alignment match marks on the DSC and TC.

11.7.7.3 Fill TC/DSC Annulus with Water and Seal

The TC/DSC annulus is sealed prior to placement in the Cask Storage Pit to prevent contamination of the DSC outer surface.

11.7.7.4 TC Movement to the Cask Storage Pit

The TC/DSC is moved to the Cask Storage Pit using the Unit 1 Reactor Building Crane. Alternately, the Cask Storage Pit water level can be lowered for this process by installing the

Cask Storage Pit Gates and draining the water from the Pit in accordance with procedures as described in Section 9.1.3.3.

11.7.7.5 DSC Spent Fuel Loading

Prior to transferring spent fuel, the Cask Storage Pit water level is maintained at normal level. Spent fuel assemblies are placed into the DSC using the Refueling Platform. Upon completion of 61BT or 61BTH DSC Spent Fuel loading, a hold down ring is installed. The 52B DSC does not require a hold down ring.

11.7.7.6 DSC Top Shield Plug Placement

This operation consists of placing the DSC top shield plug onto the DSC using the Unit 1 Reactor Building Crane.

11.7.7.7 Lifting TC from Cask Storage Pit

The loaded TC/DSC is moved to the Steam Dryer and Separator Storage Pit. Alternately, the Cask Storage Pit water level can be lowered for this process by installing the Cask Storage Pit Gates and draining the water from the pit in accordance with procedures as described in Section 9.1.3.3.

11.7.7.8 Inner DSC Top Cover Plate

Using a pump, the water level in the DSC in lowered below the inside surface of the DSC top shield plug. The inner top cover plate is put in place and welded.

11.7.7.9 DSC Drying and Backfilling

The initial blow-down of the DSC is accomplished by pressurizing the vent port with nitrogen, helium or shop air. The remaining liquid water in the DSC cavity is forced out the siphon tube and routed back to the pool or to the plant radwaste processing system. For the 52B, the DSC is then evacuated to remove any residual liquid water and water vapor in the DSC. The DSC is backfilled with helium and slightly pressurized. A helium leak test of the inner seal weld is then performed and the drain and fill port penetrations seal welded closed.

11.7.7.10 Outer DSC Top Cover Plate

The DSC outer top cover plate is installed and welded. For the 61BT or 61BTH, the DSC is then backfilled with helium and slightly pressurized. A helium leak test of the inner seal weld is then performed and the drain and fill part penetrations seal welded closed.

11.7.7.11 TC/DSC Annulus Draining and Top Cover Plate Placement

The TC is drained, removing the water from the TC/DSC annulus and flushing the TC/DSC annulus to remove any contamination left on the DSC exterior. The TC top cover plate is then put in place and bolted.

11.7.7.12 Placement of TC on Transfer Trailer

The TC is then moved to the Transfer Trailer using the Unit 1 Reactor Building Crane and downended to a horizontal position. The TC is secured to the Transfer Trailer for the subsequent transport operations.

11.7.7.13 Transport of TC to HSM

The Transfer Trailer is moved to the ISFSI along a predetermined route on plant roads. Upon entering the ISFSI, the Transfer Trailer is positioned and aligned with the HSM in which a DSC is to be stored.

11.7.7.14 TC/HSM Preparation

With the TC positioned in front of the HSM, the TC top cover plate is removed and the HSM door is removed. The Transfer Trailer is then backed into close proximity with the HSM and the skid positioning system is used for the final alignment and docking of the cask with the HSM.

11.7.7.15 Loading DSC into HSM

After final alignment of the Transfer Trailer with the HSM, the DSC is pushed into the HSM by the hydraulic ram.

11.7.7.16 Spent Fuel Storage

After the DSC is inside the HSM and the Transfer Trailer is pulled away, the HSM door is installed and the DSC axial retainer inserted.

11.7.7.17 Spent Fuel Retrieval

For retrieval of the DSC, the TC is positioned at the HSM and the DSC is transferred from the HSM by using the hydraulic ram to pull the DSC into the TC. Once back in the TC, the DSC with spent fuel assemblies is ready for return to the Spent Fuel Storage Pools.

11.7.8 Radiological Assessment

11.7.8.1 Introduction

The ISFSI is designed to limit off-site doses from the on-site storage of dry spent fuel to a fraction of the 40CFR190 limits for SSES, and on-site radiation exposure within the guidelines of 10CFR20 and 10CFR72. In all instances, the facility is designed to maintain dose rates ALARA as outlined in Regulatory Guides 8.8 and 8.10. Exposure of on-site workers is minimized by the use of concrete shielding, shielded transfer equipment, and controlled access to the ISFSI.

11.7.8.2 Dose

Compliance with Subpart K of 10CFR72 requires a written evaluation to demonstrate that the annual whole body, organ, and thyroid dose equivalent limits of 10CFR72.104 for an individual beyond the SSES Controlled Area are not exceeded as the result of the combined exposure to radiation from the storage of spent fuel on-site and all other nuclear fuel cycle contributors during normal operations. Evaluation shows that annual dose equivalents from the ISFSI and SSES operation are below the limits of 10CFR72.104 and that the maximum dose equivalent rates from the ISFSI and SSES operation are less than the 10CFR20.1301 limits for an Unrestricted Area when controlled in accordance with the PPL 72.212 Evaluation.
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11.7.8.3 ISFSI Controlled Area

A design basis accident dose limit to the whole body, or to any organ for any individual beyond the nearest controlled area boundary of the ISFSI is established in 10CFR72.106. The ISFSI design/location ensures that the criteria of 10CFR72.106 are satisfied.

11.7.8.4 Fully Loaded ISFSI Dose Assessment

An analysis was performed to determine the expected annual dose to an individual as a result of operation of a fully loaded ISFSI (105 HSMs). Impacted personnel include operators performing daily surveillance of the HSM bird screens and health physics personnel performing routine weekly surveys of the ISFSI. An analysis was performed to determine the exposure impact for a loading campaign. The analysis which is included in the PPL 72.212 Evaluation determined that the maximum annual dose for routine surveillance of the ISFSI and the estimated dose to perform a loading campaign remain within acceptable limits of the Radiological Protection Program.

11.7.9 Site Specific Evaluations

11.7.9.1 Average Ambient Air Temperature and Temperature Extremes

The average annual air temperature and air temperature extremes for the SSES are within the NUHOMS® System CSAR limits.

11.7.9.2 Earthquake Intensity/Seismic Acceleration

10CFR72 requires an evaluation be performed to establish that the cask storage pads and areas have been designed to adequately support the static load of the stored casks. A seismic evaluation has demonstrated that the design of the SSES ISFSI pad is adequate for all design basis loads for the HSM and DSC loading sequence and is adequate for the site specific loads at SSES due to a seismic event.

11.7.9.3 Flooding

The maximum water level at SSES, under the effects of probable maximum precipitation with coincident wind induced waves produces a flood level in the Susquehanna River that is well below the ISFSI elevation. The SSES ISFSI is not subject to flooding.

11.7.9.4 Tornado Wind Pressure and Missiles

The NUHOMS® System components are designed and analyzed to perform their intended functions under the extreme environmental and natural phenomena specified in 10CFR72. The HSMs are designed to withstand the design basis tornado generated missiles defined in 10CFR72 and the tornado missile loading specified by NUREG0800. Supplemental analysis has demonstrated that the HSM design is adequate to withstand all SSES tornado generated missiles. This evaluation also shows that the TC is structurally adequate to withstand the SSES tornado generated missiles.

11.7.9.5 Snow

The NUHOMS® System snow load capacity envelops the SSES site criteria for snow loads.

11.7.9.6 Lightning

The review of the SSES site for lightning damage was performed in accordance with the criteria of the National Fire Protection Association (NFPA) 780 “Standard for the Installation of Lighting Protection Systems, “formerly NFPA 78 "Lightning Protection Code." Lightning protection system is installed on the HSMs which meets code requirements.

11.7.9.7 Fire and Explosion

10CFR72 requires that the NUHOMS® System ISFSI be designed and located so that it can continue to perform its safety functions effectively under credible fire and explosion exposure conditions. The ISFSI is located away from other plant structures and protected by its own chain link fence.

The ISFSI is located near internal Protected Area roads where minimal, essential traffic is experienced. Some of this traffic involves occasional deliveries of materials such as propane and fuel within the Protected Area. Vehicles transiting the roads are continually attended so that any malfunction would be quickly mitigated.

The closest buildings are the Low Level Radwaste Holding Facility (LLRWHF) and the Security Control Center (SCC). These buildings are approximately 100 feet and 300 feet away respectively. The LLRWHF was designed and built to store low-level radwaste generated by

SSES. Trucks are parked at the LLRWHF door for loading and unloading activities for a minimal time and are continually attended. Due to the distances between the ISFSI and LLRWHF and the minimal time any vehicle is at the LLRWHF, risk of fire or explosion is minimal. The SCC does not present a risk to the ISFSI. There are no hazards (e.g. compressed gases) in the vicinity that would pose an explosion concern to the ISFSI.

The design of the NUHOMS® System does not pose a fire or explosion hazard and provides protection against fire. The concrete and steel construction provides protection against any transient fire that would manage to start. The installed fencing along with the routine inspections of the area will keep any loose combustibles from accumulating in the area of the HSMs.

During the period when a transporter is in use within the storage area, controls will ensure that an attendant is present. Thus the attendant can notify the Control Room should any malfunction occur. The plant fire brigade will respond to the area upon notification of a fire. Manual fire suppression equipment is available for fighting a fire in the area.

The Hydrogen/Oxygen Tank Farm located near the South Gate house is along the transport path of the Transfer Trailer during transit from the Reactor Building to the ISFSI. Evaluation to determine the effects of a postulated accidental hydrogen explosion at the Hydrogen/Oxygen Tank Farm on the TC and Transfer Trailer indicated that the TC/Transfer Trailer are not compromised.

The above review of the NUHOMS® System shows that there is no credible fire or explosion exposure that would prevent the system from performing its function.

11.7.10 Heavy Loads

All lifting of TC or DSCs in the Reactor Building will utilize the Unit 1 Reactor Building Crane (single failure proof) with rigging/lifting mechanisms meeting the requirements of the SSES Heavy Loads Program and NUREG 0612. The Unit 1 Reactor Building Crane single failure proof certification is maintained via the SSES Preventative Maintenance Program.

11.7.11 Auxiliary Systems

11.7.11.1 Electrical Systems

Electrical power is required at the ISFSI for the lighting and for HSM Temperature Monitoring System.

11.7.11.2 Instrumentation

The Independent Spent Fuel Storage Installation contains a Temperature Monitoring System to continuously monitor the temperature of each HSM's concrete roof slab. HSM cooling relies on natural air circulation through the modules, therefore, the roof slab represents the hottest portion of the HSM concrete. Temperature monitoring provides a means to identify abnormal increases in temperature that could threaten proper HSM operation. The normal temperature range of the concrete at the temperature sensor(s) location is predicted in the range of 60° F to 240° F, based on age of contained fuel and ambient conditions.

The Temperature Monitoring System consists of one thermocouple per roof slab to provide a signal to a Programmable Logic Controller (PLC) based system capable of displaying and recording the concrete temperature for each HSM. A local temperature indicator (gauge) is also installed in each HSM to provide a backup in the event of a thermocouple or PLC failure. The Temperature Monitoring System provides the means of monitoring temperature to meet the requirements in the C of C Number 1004.