CHAPTER 11.0 - RADIOACTIVE WASTE MANAGEMENT

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

CHAPTER 11.0 - RADIOACTIVE WASTE MANAGEMENT

This chapter describes the capabilities of the plant to control, collect, handle, process, store, and dispose of liquid, gaseous, and solid wastes that may contain radioactive materials and the instrumentation used to monitor the release of these wastes.

This chapter also describes normal operation and anticipated operational occurrences such as refueling, purging, equipment downtime, maintenance, etc. The radioactive waste (radwaste) treatment systems meet the requirements of 10 CFR 20 and 10 CFR 50 and the intent of Appendix I to 10 CFR 50 (as low as reasonably achievable-ALARA).

The releases of radioactivity and their resulting doses included in this Chapter were calculated during plant licensing from assumed values for many parameters. These included coolant activity, iodine partitioning, amount of failed fuel, filter efficiencies, system flow rates, component leak rates, and associated activity for all potentially radioactive water and steam systems. Estimates were made for many individual contributors and then summed to obtain estimates for total annual dose. These values were then compared to appropriate regulatory limits, such as 10 CFR 20 and Appendix I to 10 CFR 50, to show that the plant could be operated, if granted a license, in compliance with these regulations. The NRC reviewed these estimated values and confirmed in their Safety Evaluation Report that the plant could be operated and meet the regulations.

After receipt of an Operating License, the requirement to meet the applicable radiological dose regulations is demonstrated in the station's annual radiological effluent release report. The values for the report are calculated using the measured total radioactivity releases from all sources and equations and data included in the Offsite Dose Calculation Manual. Only the total values are calculated; there is no requirement to calculate the dose from each of the individual sources listed in this Chapter. The information on release estimates and offsite doses is maintained in the UFSAR for historical reference and is not intended to be used to establish current operating limits.

Specific limits on radioactive effluents, methods for calculating the resulting offsite doses, details of the radiological environmental monitoring program, and reporting requirements for the release of radioactive effluents are given in the Technical Specifications and other documents.

11.1 SOURCE TERMS

General Electric has evaluated radioactive material sources (activation products and fission product release from fuel) in operating boiling water reactors (BWR's) since the early 1960s. These source terms are reviewed and periodically revised to incorporate up-to-date information. Release of radioactive material from operating BWR's 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 off-gas. The various radioisotopes listed have been grouped as coolant activation products, noncoolant activation products, and fission products. The fission product levels are based on measurements of BWR reactor water and off-gas at several operating stations. Emphasis was placed on observations made at KRB and Dresden-2. The design-basis radioactive material levels do not necessarily include all the radioisotopes observed or predicted theoretically to be present. The radioisotopes included are considered significant to one or more of the following criteria:

a. plant equipment design,

b. shielding design,

c. understanding system operation and performance,

d. measurement practicability, and

e. evaluating radioactive material releases to the environment.

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

11.1.1 Fission Products

11.1.1.1 Noble Radiogas Fission Products

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

Equilibrium: $R_g \simeq k_1 y$

Recoil: $Rg \simeq k_2 y$

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. 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: $R_g \simeq k_3 y \lambda^{0.5}$ (Reference 1)

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.

Although the previously described "diffusion" mixture has been described historically by GE since 1963 with a value of 0.5, the recent design-basis release magnitude has varied from 0.5 Ci/sec to 0.1 Ci/sec as measured after 30-minute decay (t = 30 minutes). (The noble radiogas source-term rate after 30-minute decay has been used as a conventional measure of the design-basis fuel leakage rate since it is conveniently measurable and was consistent with the nominal design-basis 30-minute off-gas holdup system used on a number of previous plants.) Since about 1967, the design-basis release magnitude used by GE, including the 1971 source terms used herein, has been established at an annual average of 0.1 Ci/sec (t = 30 minutes). This design basis is considered as an annual average with some time above and some time below this value. This design value was selected on the basis of operating experience rather than predictive assumptions. Several judgment factors, including the significance of environmental release, reactor water radioisotope concentrations, liquid waste handling and effluent disposal criteria, building air contamination, shielding design, and turbine and other component contamination affecting maintenance, have been considered in establishing this level.

Noble radiogas source terms from fuel above 0.1 Ci/sec (t = 30 minutes) can be tolerated for reasonable periods of time. Continual assessment of these values is made on the basis of actual operating experience in BWR's. There is no experimental or operational basis for relating this design-basis value to reactor size or fuel power density, since limiting conditions on plant releases are largely independent of these parameters.

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

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

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

With the exception of Kr-85 with a half-life of 10.74 years, the noble radiogas fission products in the fuel are essentially at an equilibrium condition after an irradiation period of several months (rate of formation is equal to the rate of decay). So for practical purposes the term $(1-e^{-\lambda t})$ approaches 1 and can be neglected when the reactor has been operating at 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 represent the actual release out the stack. Historically t = 30 minutes has been used. When discussing long steady-state operation and leakage from the fuel (t = 0), the following simplified form of Equation 11.1-1 can be used to describe the leakage of each noble radiogas:

$$R_g = K_g y \lambda^m \tag{11.1-2}$$

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

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

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

Equation 11.1-3 represents a straight line when $\log (R_g/y)$ is plotted versus $\log (\lambda)$; 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. With radiogas leakage at KRB over the nearly 5-year period varying from 0.001 to 0.056 Ci/sec (t = 30 minutes) and with radiogas leakage at Dresden-2 varying from 0.001 to 0.169 Ci/sec (t = 30 minutes), the average value of m was determined. The value for m is 0.4 with a standard deviation of ± 0.07. This is illustrated in Figure 11.1-1 as a frequency histogram. As can be seen from this figure, variations in m were observed in the range of m = 0.1 to m = 0.6. After establishing the value of m = 0.4, the value of K_g can be calculated by selecting a value for R_g, or as has been done historically, the design basis is set by the total design-basis source-term magnitude at t = 30 minutes. With ΣR_g at 30 minutes = 100,000 µCi/sec, K_g was calculated as being 2.6×10^7; thus Equation 11.1-1 becomes:

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

This updated noble radiogas source-term mixture has been termed the "1971 mixture" to differentiate it from the "diffusion mixture." The noble gas source term for each radioisotope is calculated from Equation 11.1-4. The resultant source terms are presented in Table 11.1-1 as leakage from fuel (t = 0) and after 30-minute decay. While Kr-85 can be calculated using Equation 11.1-4, the number of confirming experimental observations was limited by the difficulty of measuring very low release rates of this isotope. Therefore, the table provides an estimated range for Kr-85 based on actual measurements.

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:

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

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

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

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

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

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

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

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

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

The halogen release rate from the fuel was calculated from Equation 11.1-6. Concentrations in reactor water were calculated from Equation 11.1-7. The resultant concentrations are presented in Table 11.1-2.

11.1.1.3 <u>Other Fission Products</u>

The observations of other fission products (and transuranic nuclides, including Np-239) in operating BWR's are not adequately correlated by simple equations. For these radioisotopes, design-basis concentrations in reactor water have been estimated conservatively from experience data and are presented in Table 11.1-3. Carry-over of these radioisotopes from the reactor water to the steam is estimated to be <0.001 (fraction). In addition to carry-over, however, decay of noble radiogases in the steam leaving the reactor results in production of noble gas daughter radioisotopes in the steam and condensate systems.

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

Except for Np-239, trace concentrations of transuranic isotopes have been observed in only a few samples where extensive and complex analyses were carried out. The predominant alpha emitter present in reactor water is Cm-242 at an estimated concentration of 10^{-6} μCi/g or less, which is below the maximum permissible concentration in drinking water applicable to continuous use by the general public. The concentration of alpha-emitting plutonium radioisotopes is more than 1 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 (µCi/g)
M	=	mass of water in the operating reactor (g)
β	=	cleanup system removal constant (sec^{-1})
β	=	$\dfrac{\text{cleanup system flow rate (g/sec)}}{M}$
γ	=	halogen steam carry-over removal constant (sec^{-1})
γ	=	$\dfrac{\text{concentration of halogen radioisotope in steam (µCi/g)}}{C_h} \times \left[\dfrac{\text{steam flow (g/sec)}}{M}\right]$

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.

11.1.2.2 Noncoolant Activation Products

The activation products formed from the activation of solid suspension and dissolved 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 conservatively estimated from experience data. The resultant concentrations are presented in Table 11.1-5. Carry-over of these isotopes from reactor water to steam is estimated to be <0.001 (fraction).

11.1.3 Tritium

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

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

b. nuclear fission of UO_2 fuel, and

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

Release of the tritium formed in the control rods is believed to be negligible, since the boron is sealed in tubes. A prime source of tritium available for release from a BWR is from deuterium activation in the primary coolant. Some gaseous fission product tritium may also transfer from fuel to primary coolant. Several uncertainties are involved with estimating the amount of tritium generated in a BWR and 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 gross tritium formation in a BWR can be calculated using the equation:

$$R_{act} = \frac{\Sigma \phi \, v \lambda}{3.7 \times 10^{4} \, P} \qquad (11.1\text{-}8)$$

where:

R_{act} = tritium formation rate by deuterium activation (μCi/sec/MWt)

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

ϕ = thermal neutron flux at power P (neutrons/cm^2-sec),

V = coolant volume in core (cm^3),

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

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 indicated uncertainty is derived from estimation errors in describing values for the core coolant volume, coolant density in the core, abundance of deuterium in light water (some additional deuterium will be present because of the $H(\eta,\gamma)$ D reaction), thermal neutron flux, and the microscopic cross section for deuterium.

The fraction of fission-produced tritium which may transfer from the fuel to the coolant and then be available for release in liquid and gaseous effluents is much more difficult to estimate. However, because Zircaloy-clad fuel rods are used in BWR's, essentially all fission product tritium remains in the fuel rods unless defects are present in the cladding material (Reference 2).

The 1968 study made at Dresden-1 by the U.S. Public Health Service suggests that essentially all of the tritium released from the plant can be accounted for by the deuterium activation source (Reference 3). For estimation purposes, the leakage of tritium from defective fuel can be assumed to be similar to the leakage of noble radiogases. Thus, an empirical relationship described as the "diffusion mixture" can be used to predict this source term (refer to Equation 11.1-2). The equation which describes this relationship is:

$$R_{dif} = Ky\sqrt{\lambda} \qquad (11.1\text{-}9)$$

where:

R_{dif} = leakage rate of the radioisotope (μCi/sec),

y = fission yield fraction,

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

K = a constant related to total leakage rate.

Based on a total noble radiogas source term of 10^5 μCi/sec after 30-minute decay, tritium leakage from fuel can be calculated to be about 0.24 μCi/sec. The conservatism of this value is evident from the USPHS study, where 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 this "diffusion mixture" relationship. This suggests that the actual tritium leakage rate might range from 0.015 to 0.10 μCi/sec. Since the annual average noble radiogas leakage from a BWR is expected to be less than 0.1 Ci/sec (t = 30 minutes), the annual average tritium release rate from the fission source can be conservatively estimated at 0.12 ± 0.12 μCi/sec, or 0.0 to 0.24 μCi/sec.

For a 3293-MWt reactor, the estimated total tritium appearance rate in reactor coolant and release rate in the effluent is about 17.6 Ci/yr.

11.1.4 Fuel Fission Product Inventory and Fuel Experience

11.1.4.1 Fuel Fission Product Inventory

Fuel fission product inventory information used in establishing fission product source terms for accident analysis is discussed in Chapter 15.0.

11.1.4.2 Fuel Experience

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 4, 5, and 6).

11.1.5 Process Leakage Sources

Process leakage results in potential release paths for noble gases and other volatile fission products via ventilation systems. Liquids from process leaks are normally collected and routed to the liquid-solid radwaste system. Radionuclide releases via ventilation paths are at extremely low levels and have been insignificant compared to process off-gas from operating BWR plants. However, the implementation of improved process off-gas treatment systems makes the ventilation release relatively greater.

11.1.6 Power Conversion System Source Terms

The dominant radionuclide in BWR power conversion equipment is nitrogen-16 (N-16). Typical inventories in power conversion equipment are given in Table 2-12 of Reference 6. LSCS specific inventories are given in Tables 12.2-7 and 12.2-8.

11.1.7 References

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

2. J. W. Ray, "Tritium in Power Reactors," Reactor and Fuel-Processing Technology, Vol. 12, No. 1, pp. 19-26, Winter 1968-1969.

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

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

5. H. E. Williamson and D. C. Ditmore, "Experience With BWR Fuel Through September 1971," NEDO-10505, May 1972.

6. D. R. Rogers, "BWR Equipment N-16 Radiation Shielding Studies," NEDO-20206, December 1967.

TABLE 11.1-1

NOBLE RADIOGAS SOURCE TERMS (1971 MIXTURE)

ISOTOPE	HALF-LIFE	SOURCE TERM AT t = 0 (µCi/sec)	SOURCE TERM AT t = 30 min (µCi/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 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 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

ISOTOPE	HALF-LIFE	CONCENTRATION (μCi/g)
Br-83	2.40 hr	1.5×10^{-2}
Br-84	31.8 min	2.7×10^{-2}
Br-85	3.0 min	1.7×10^{-2}
I-131	8.06 day	1.3×10^{-2}
I-132	2.284 hr	1.2×10^{-1}
I-133	20.8 hr	8.9×10^{-2}
I-134	52.3 min	2.4×10^{-1}
I-135	6.7 hr	1.3×10^{-1}

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	35.1 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 hr	4.9×10^{-2}
Cs-134	2.06 yr	1.6×10^{-4}
Cs-136	13 day	1.1×10^{-4}
Cs-137	30.2 yr	2.4×10^{-4}
Cs-138	32.2 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)
N-13	9.99 min	7.0×10^{-3}	4.0×10^{-2}
N-16	7.13 sec	5.0×10^{1}	4.0×10^{1}
N-17	4.14 sec	2.0×10^{-2}	6.0×10^{-3}
O-19	26.8 sec	8.0×10^{-1}	7.0×10^{-1}
F-18	109.8 min	4.0×10^{-3}	4.0×10^{-3}

TABLE 11.1-5

NONCOOLANT ACTIVATION PRODUCTS IN REACTOR WATER

ISOTOPE	HALF-LIFE	CONCENTRATION (μCi/g)
Na-24	15 hr	2×10^{-3}
P-32	14.31 day	2×10^{-5}
Cr-51	27.8 day	5×10^{-4}
Mn-54	313 day	4×10^{-5}
Mn-56	2.582 hr	5×10^{-2}
Co-58	71.4 day	5×10^{-3}
Co-60	5.258 yr	5×10^{-4}
Fe-59	45 day	8×10^{-5}
Ni-65	2.55 hr	3×10^{-4}
Zn-65	243.7 day	2×10^{-6}
Zn-69m	13.7 hr	3×10^{-5}
Ag-110m	253 day	6×10^{-5}
W-187	23.9 hr	3×10^{-3}

11.2 LIQUID WASTE MANAGEMENT SYSTEMS

The liquid radwaste system collects, monitors, and treats all potentially radioactive liquid wastes produced by the station during normal operation and maintenance, as well as transient conditions.

11.2.1 Design Bases

11.2.1.1 Safety Design Basis

The liquid radwaste system is designed so that liquid radwastes discharged from the site will have radioactivity concentrations well within limits specified in 10 CFR 20 and meet the design objectives of 10 CFR 50, Appendix I.

Each liquid radwaste stream terminates in a sample tank. Since the liquid radwaste system operates on a batch basis, this arrangement allows each treated batch to be sampled when appropriate to assure that the treatment was effective. If the sample indicates that the waste needs further processing, it is recycled either through the same treatment or through a subsystem providing a higher degree of treatment. If the treated waste water is not needed for recycle, the water is sent to the discharge tank and held until a discharge batch is accumulated. Each batch is sampled to verify that its activity level is within limits for discharge. The actual discharge to the lake blowdown line requires opening a keylock valve in accordance with written operating procedures, only after sampling.

A system bypass allows connection to portable waste treatment equipment. This enables the efficient processing of liquid radwaste through the use of state of the art radwaste processing technology.

11.2.1.2 Power-Generation Design Basis

The liquid radwaste system is sized to handle expected liquid waste inputs on the basis of both volume and activity as the result of normal operation, including anticipated abnormal occurrences.

Unit 1 and Unit 2 have separate equipment and floor drain collection and process systems. This is to prevent mixing of the liquid wastes between units, if desired. The corresponding subsystems in each independent radwaste system are provided with extensive crossties to provide a high degree of radwaste system availability. The laundry and sludge subsystems are common to both units.

The LaSalle County Station (LSCS) Units 1 and 2 liquid radwaste system is designed to permit maximum recycling of plant water, whose inventory is kept nearly constant at all times. The plant is designed to minimize noncontaminated inputs from leakage of service water, circulating water, and groundwater into the plant floor drain system. Clean demineralized well water is used as makeup to the primary system or for decontamination work. Since it is not feasible to eliminate or separate all noncontaminated inputs to the liquid radwaste system, some processed liquid waste must be discharged to the Illinois River from the plant. The discharge is mixed with the blowdown released to the Illinois River.

11.2.1.3 Radioactive Releases

Table 11.2-1 gives expected annual releases by nuclide. Table 11.2-1 is not intended to contain limits for the radionuclides or imply that all the listed radionuclides must be present in the given proportions. Table 11.2-2 gives expected doses to individuals at or beyond the site boundary.

11.2.1.4 10 CFR 50 Comparison

Table 11.2-2 compares the calculated radiological doses from LSCS to the numerical design objectives of Appendix I to 10 CFR 50. It can be seen from this comparison that releases are significantly below the guideline levels.

11.2.1.5 10 CFR 20 Comparison

Table 11.2-1 compares typical maximum effluent concentrations with 10 CFR 20 limits. It can be seen from this comparison that concentrations are significantly below the specific limits.

11.2.1.6 Component Specifications

Table 11.2-3 gives the design parameters of various radwaste system components.

11.2.1.7 Anticipated Operating Conditions

Certain "anticipated operational occurrences" are considered in this section. The liquid radwaste system is designed to meet the requirements of these conditions. Therefore, in normal operations, only a small fraction of the system design capacity is utilized at any one time.

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

a. initial startup (especially Unit 2 startup with Unit 1 operating),

b. Unit 1 and Unit 2 shutdown,

c. large fuel leak rates,

d. maximum reactor coolant leak rate,

e. refueling and maintenance outages, and

f. unscheduled reactor trips.

11.2.1.7.1 Initial Startup - Unit 1 (This information is historical)

Various operations associated with the initial startup of a nuclear station can produce flow volumes in both the equipment drain and floor drain processing system of up to 100,000 gallons in a single day, with the highest 3-day average about 50,000 gallons per day at irregular intervals over periods of 2 to 3 months. The volumes of waste water involved and the fact that much of it originates from sources outside the primary system makes it nearly impossible to maintain the station's water balance except by discharging larger than normal volumes of processed waste. The design of the liquid radwaste systems allowed segregation of liquid wastes while Unit 1 was in startup and Unit 2 was still being constructed. The design features provided to meet these demands are discussed by subsystem in the following:

a. Equipment drain processing subsystem - Processing in this subsystem is provided by the waste filters (1,2WE01F) and waste demineralizers (1,2WE01D). Since in any "shakedown " period such as station startup, some equipment difficulties are almost inevitable, a major design assumption is that the liquid radwaste processing capacity may be limited to a 25% availability during this period. The design flow capacity of the filters and demineralizers is such that 25% of the daily throughput design capacity is sufficient to process a maximum day's collected liquid waste volume. The three waste sample tanks (1,2WE03T, 0WE01T) and the two discharge tanks (1,2WF05T) are crosstied to provide storage volume for 1 to 2 full days of process wastes. The storage helps to "smooth out" the station water inventory maintenance and allows for discharge at the minimum design release rate in the event that full lake blowdown water flow is not available for dilution. Under the processing condition described, the maximum design release rate is such that with dilution from the lake blowdown in the discharge flume, the radioactivity levels in the discharge flume meet the requirements of 10 CFR 20.

b. Floor drain processing subsystem - Normal design processing in this subsystem is by settling, evaporation, and demineralization. At startup, substantial volumes of water of very low radio activity content may be generated by housekeeping activities. The floor drain concentrator packages(1WF01S, 2WF01S) normally receive steam from the main steam reboilers. If this is not available, the steam would be from the electrode steam reboiler (60,000 lb/hr). This rate is sufficient to operate all three concentrators. The inputs to the two floor drain concentrator packages and the chemical waste concentrator package are crosstied to allow full utilization of the 33 1/3% availability of evaporation capacity. Volume retention capacity for 1 to 2 days' maximum inputs and the availability of increased throughput processing channels allow this system to handle anticipated 3-day to 4-day peak inputs.

c. Chemical waste processing subsystem - Frequent regeneration of condensate polishers occurs when at least one unit is operating. In this case, the steam supply can be drawn from the operating unit if two evaporators are sufficient to process floor drains and chemical wastes (i.e., a nuclear steam reboiler can supply steam for two of the three evaporators; if operation of all three evaporators is necessary, the electrode steam reboiler can be used).

d. Laundry waste processing subsystem - This system is sized to meet the volume and radioactivity levels generated during a refueling outage, 4 gpm. This is greater than is anticipated during the initial startup.

e. Sludge subsystem - Sludge generation during startup can be variable, depending on filter and demineralizer performance as well as on initial water quality. The tanks in this subsystem have ample capacity for several days' accumulation at maximum backwash rates.

11.2.1.7.2 Unit 2 Initial Startup - Unit 1 in Normal Operation (This information is historical)

Meeting the demands of this situation places the most stringent requirements on the design of the liquid radwaste system. The following are the major differences between this case and initial startup:

a. The liquid radwaste system is processing contaminated wastes from Unit 1.

b. Less volume retention capability is available to hold up surge volumes.

c. Full auxiliary steam capacity is available to supply all three evaporators when needed. The electrode boilers act as a backup steam source.

d. Accumulated operating experience and maintenance lead to a somewhat higher availability factor for processing equipment.

e. The design assumption availability factors for processing equipment are 66 2/3% of evaporation capacity and 50% of filter and demineralizer capacity. The discharge rate, with lake blowdown water flow available for dilution, should not place a limitation on liquid radwaste system processing rates. The design collection volume capacity exclusively available for either Unit 1 or Unit 2 wastes is sufficient to hold 1 day's maximum accumulation of liquid wastes when processing at 25% of design flow rates. The display of tank level recorders on the liquid radwaste control panel allows the identification and segregation of waste batches. The scheduling of batch processing should not impair the operation of Unit 1.

11.2.1.7.3 <u>Normal Startup Following Any Shutdown</u>

The waste surge tanks (1WE02T, 2WE02T) are sized so that each accommodates one reactor hydrotest volume. It also accommodates the thermal expansion water when the unit is started up.

11.2.1.7.4 <u>Condenser Tube Leakage</u>

<u>Original Design Basis</u>

Condenser tube leakage increases the dissolved solids content of the condensate, forcing frequent regeneration of the condensate polishers. At some magnitude of leakage, the polishers are no longer able to maintain feedwater quality, thus forcing a unit shutdown. Recovery from this situation requires the regeneration of all the condensate polishers at least once and perhaps twice. The liquid radwaste processing capacity is sufficient to assure that the treatment of the liquid radwaste produced by this situation will not extend the outage time required for repairs. The design capacity of the chemical waste subsystem is sufficient to process the volume of waste from the regeneration of four condensate polisher beds per day. The equipment drain subsystem is able to treat the backwash water from four regenerations per day without impairing the capacity to treat the normal flows from either the affected unit or the normally operating unit. Condenser tube leaks of lesser magnitude can occur without forcing a shutdown but can still cause a significant increase in regeneration frequency. The liquid radwaste system maintains the station water balance by the controlled discharge of treated wastes as previously described in Subsections 11.2.1.2 and 11.2.1.7.

Current Design Basis

Condenser tube leakage increases the dissolved solids content of the condensate, quickly depleting the ion exchange capacity of the condensate polisher resin. At some magnitude of leakage, the polishers are no longer able to maintain feedwater or vessel water quality, thus forcing a unit shutdown. As a worst case, recovery may require the replacement of all the on-line condensate polisher resin beds. The capacity of the spent resin tank is sufficient to contain the volume of waste from the replacement of three condensate polisher beds. The equipment drain subsystem is able to treat the transfer water from resin bed replacements without impairing the capacity to treat the normal flows from either the affected unit or the normally operating unit. The solid radwaste system is capable of processing 25% of the spent resin tank contents per day. Condenser tube leaks of lesser magnitude can occur without forcing a shutdown but can still cause a significant increase in resin replacement frequency. The liquid radwaste system maintains the station water balance by the controlled discharge of treated wastes as previously described in Subsections 11.2.1.2 and 11.2.1.7.

The outlet of the resin bed is sampled to estimate ionic depletion. During the operating cycle, the remaining capacity of the resin is not determined; if needed the capacity can be calculated.

11.2.1.7.5 Unit 1 and Unit 2 Shutdown

When both units are shut down, there is no steam to operate the three evaporators and other liquid radwaste processing methods are used.

11.2.1.7.6 Design Basis Fuel Leakage

Under this condition, the liquid radwaste system operates with radioactivity levels at or near design basis. The reactor water cleanup system delivers sludge to the phase separators at a rate higher than normal. Unit 1 and Unit 2 phase separator tanks operate independently, receiving wastes from the respective Unit 1 and 2 RWCU systems. Frequent decanting of the phase separator serving the affected unit results in the delivery of liquid with higher than normal solids content to the waste collector tank, resulting in short filter and demineralizer runs.

The waste filter and waste demineralizer are sized and arranged so that no more than one filter and one demineralizer are out of service for backwashing or regeneration/resin replacement at any time while Unit 1 or Unit 2 is in normal operation.

11.2.1.7.7 Reactor Coolant Leakage

The existence of reactor coolant leaks can result in the equipment drain tanks becoming nearly filled with water at reactor coolant activity. Table 11.2-4 compares design-basis influent radioactivity concentration to reactor coolant activity.

The equipment drain processing subsystem is capable of decontaminating this activity to levels suitable for return to the cycled condensate storage tanks as specified in Subsection 11.2.3.1.

11.2.1.7.8 Refueling and Maintenance Outages

The condition given in Subsection 11.2.1.7.1 covers the major requirements for outages. The laundry processing subsystem was designed for a maximum load during refueling and maintenance outages and was designed to treat these loads to produce an effluent suitable for discharge as stated previously in Subsections 11.2.1.4 and 11.2.1.5. The site does not launder anti-contamination clothing any longer, significantly reducing the input to the system.

11.2.1.7.9 Unscheduled Reactor Trips

Reactor trips may produce an additional amount of water to the suppression pool. The equipment drain processing subsystem has sufficient capacity in all normal operating modes to process this volume within 1 day.

11.2.1.8 Seismic Design and Quality Group

The portion of the structure housing the liquid radwaste system which is below grade is Seismic Category I for the auxiliary building, and Seismic Category II for the turbine building area. The turbine building walls below grade are part of the total structural shear wall system and are designed to withstand the effects of the OBE and SSE. All equipment (including tanks, pumps, valves, and piping) of the liquid radwaste system containing radioactive wastes is classified as Quality Group D.

11.2.1.9 Facility and Equipment Design

The liquid radwaste system has been designed to minimize radiation exposure to operating personnel. Normal operations, maintenance, and nonroutine operations are discussed in the following from the viewpoint of design for ALARA exposures to personnel.

11.2.1.9.1 Normal Operations

The following steps have been taken to ensure that occupational exposures will be ALARA during normal operating situations:

a. Equipment operation for all liquid waste treatment systems is from the radwaste control room. This is a shielded room located in the solid radwaste building.

b. Tanks and equipment are located in shielded compartments with labyrinth entrances.

c. The sludge subsystem carries waste slurries which could settle into crud traps. Flushing of the sludge system equipment will be performed as tank capacities allow to minimize this buildup.

d. Piping carrying radioactive fluids is routed to avoid inadvertent exposures. Many pipe runs are within shielded pipe tunnels.

e. Tank overflows are hard-piped to adjacent sumps with two exceptions. Reactor water cleanup phase separator overflows are hard-piped to the reactor building equipment drain tanks and the concentrator waste tanks overflow to the URC Sludge Tank.

f. The gases from the RWCU phase separator vents flow to its cubicle in the reactor building. For normal conditions this cubicle air supply discharges to the station vent stack as part of the normal reactor building HVAC flow path. For abnormal conditions of high radioactivity levels in the RWCU cubicle, a normally closed valve is opened to connect this cubicle to the reactor building vent and purge system and then to the station vent stack.

11.2.1.9.2 Maintenance Operations

Equipment is arranged to separate sources of high exposure from sources of lower exposure. The evaporators, for example, have the process equipment separated from the pump skid by a shield wall to minimize exposure during pump maintenance. The instrument rack is located as far from the pump skid as possible to minimize exposure during instrument maintenance.

Air-operated valves are located outside of compartments to minimize exposure from tanks or components during valve maintenance. Pumps are provided with mechanical seals to minimize maintenance.

In general, all components which may require maintenance are capable of being flushed prior to maintenance.

11.2.1.9.3 Nonroutine Operations

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

11.2.1.10 Tank Level Control

Provisions have been made to preclude the uncontrolled spills to the building due to tank overflows. The following conditions pertain to all tanks outside the containment which potentially contain radioactive fluids:

a. Tank level instrumentation is provided on all tanks in the radwaste control room. A high-level condition on any tank will be annunciated. The only exceptions to the above information are the concentrator waste tanks, the boil-out tank, and the filter aid tank since these tanks are no longer in service.

b. Tank overflows are hard piped to an adjacent sump with two exceptions. The Reactor Water Cleanup Phase Separator overflows are hard-piped to the reactor building equipment drain tanks. The concentrator waste tanks overflow to the URC Sludge Tank. All sumps are provided with duplex (redundant) pumps with the exception of the Tendon Tunnel sumps. These sumps have only one pump. Sumps are level controlled and logic is provided to start and stop pumps automatically.

c. Concentrator waste tanks must be maintained at 140° F to 160° F. For this reason, piping is heat traced to prevent crystallization of the waste. If capacity for the concentrates is not available, the inputs are stopped. (The concentrator waste tanks are abandoned-in-place as is the associated heat trace circuits.)

11.2.1.11 Prevention of Uncontrolled Releases

The liquid radwaste system is the only flow path from a licensed unit to the environment for radioactive liquids during normal operations. The discharge tanks are the only flow path from the liquid radwaste system to the environment during normal operations.

After processing of a batch of waste, it is transferred to respective sample tanks (e.g., waste sample tank, floor drain sample tank) before being transferred to the discharge tanks. In the discharge tanks it is again sampled for activity prior to discharge. All discharge must pass through a manual key-locked valve (0WF201 on Sheet 10 of Drawing No. M-103). Limit switches supply status information on valve position to the operator at the radwaste control panel. This valve is intentionally manual to prevent bypassing the key-lock feature from the control room. Processing and sampling of liquids resulting from Unit 2 construction activities will commence at the time of Unit 2 fuel load.

11.2.2 System Description

The liquid radwaste system consists of five major subsystems: (1) the waste processing subsystem, (2) the floor drain subsystem, (3) the chemical waste subsystem, (4) the laundry waste subsystem, and (5) the sludge subsystem. Liquid radwaste diagrams are shown in Drawing No. M-103.

Design-basis concentrations of radioactive nuclides in the input waste streams to the various liquid waste subsystems are listed by nuclide in Table 11.2-4. Expected inventories of radioactive nuclides in major liquid waste system components are tabulated in Table 11.2-5. Maximum and expected inputs to liquid radwaste systems are shown in Table 11.2-6. As noted, input to equipment drains has been assumed to be design basis reactor coolant activity; floor drains have been assumed to be 0.31 times reactor coolant activity. As a result of this conservative assumption, short-lived radionuclides appear in collection tanks but they quickly decay.

Table 11.2-7 lists the design-basis process decontamination factors used in the analysis. The system is assumed to be operating continuously; no credit for radioactive decay is taken for holdup in tanks, for example, while waiting for a demineralizer to be made available for batch processing.

The design-basis input source terms to the various streams are based on the distribution of isotopes in reactor coolant and steam as described in Section 11.1.

The liquid radwaste system is designed to recycle as much processed liquid waste as can be accommodated within the station water balance.

Equipment drains are processed through the waste-processing system by filtration and demineralization. The flow is then normally routed to the cycled condensate storage tanks for reuse in the station. Radioactive floor drains are treated by demineralization. After sampling, it is normally sent to the cycled condensate storage tanks for reuse. The Floor drain system is cross-tied to the Equipment drain system to provide backup processing capability should the vendor processing unit be unavailable. The valve on the cross-tie is administratively controlled to ensure proper routing of floor drain water.

Laundry wastes are treated by filtration and may be discharged after sampling. Optional paths after treatment in addition to the discharge tank are to the chemical waste collector tank, or the laundry collector tank depending on water quality in the laundry sample tank (Drawing No. M-103).

Design-basis quantities of waste to be treated and released from the station are processed through the chemical waste subsystem by demineralization. This waste originally consisted of primarily regenerant chemicals from the regeneration of the condensate and radwaste polishers. Regenerant waste is no longer generated. The operating procedures define the conditions under which water can be transferred from any sample tank to the discharge tanks (Subsection 11.2.2.6).

Equipment redundancy and capacities, together with crossties among subsystems, provide flexibility for the liquid radwaste system to accommodate a wide variety of extreme station operating conditions as discussed in Subsection 11.2.1.7.

The design of the processing systems for liquid wastes assumes that they will be discharged, although normally they will be returned to the station for reuse via the two cycled condensate storage tanks. This assures that the processing systems will always be capable of meeting environmental discharge standards.

11.2.2.1 Waste Processing Subsystem

This subsystem collects and processes high purity (low conductivity) sources such as equipment drains. This water is treated by settling, filtration, and demineralization. After appropriate sampling it is returned for station reuse through the cycled condensate storage tanks.

Input sources to the waste processing subsystem include the following:

a. equipment drain sumps,

b. condensate demineralizer backwash,

c. reactor water cleanup (RWCU) phase separator decantate,

d. fuel pool system decantate,

e. ultrasonic resin cleaner (URC) backwash,

f. radwaste demineralizer backwash, and

g. suppression pool drains.

Two waste-processing streams are provided, one each for Unit 1 and Unit 2, with cross-ties for system redundancy.

Two waste collector tanks, each having a capacity of 30,000 gallons, take regular inputs from the Unit 1 and Unit 2 equipment drain systems. These tanks are sized to hold 1 day's accumulation of wastes. Prior to processing, approximately 500 gallons of tank "bottoms" may be pumped to the URC sludge tank via the waste flocculation tank to remove settled solids and minimize downstream filter loading. The contents of the tank are then recirculated to assure complete mixing; at this point a composite sample can be drawn from the recirculation line. The tank contents can be pumped through the waste filters and demineralizers or transferred to the chemical waste system for treatment.

Two waste surge tanks, each having a capacity of 50,000 gallons, take infrequent inputs from Unit 1 and 2 waste collector tank overflows, from reactor water cleanup, and from cycled condensate storage tanks. Their operation is similar to that described above for the waste collector tanks.

Two waste flocculation tanks are provided to process the ultrasonic resin cleaner (URC) backwashes. Each backwash produces approximately 6,000 gallons of washwater at 200 ppm to 500 ppm suspended solids and each condensate polisher is URC cleaned approximately once a week. Provision is made for the addition of a chemical flocculant to the tanks. This destabilizes the suspension, resulting in the formation of larger particles which settle out. The decantate is processed through the waste filter and demineralizer and the solids are pumped to the URC sludge tank. (Note the chemical flocculant system is abandoned in place.)

The two waste filters remove suspended solids prior to demineralization in order to extend the useful ion exchange life of the demineralizer resins. The Unit 1 filter contains precoatable filter elements. The Unit 2 filter contains either precoatable or non-precoatable filter elements. Precoat material is applied to the precoatable elements, and when depleted, as indicated by effluent conductivity or filter differential pressure, the elements are backwashed. Both the used precoat material and the material filtered out of the water is routed to the waste sludge tank. No precoat material is added onto the non-precoatable elements. As necessary, the elements are backwashed, and only the material filtered out of the water is routed to the waste sludge tank. A handswitch located on a Radwaste Control Room panel controls whether the logic applied to filter manipulations includes Precoat applications. The processed effluent from these filters is directed to the input of the waste demineralizers. Demineralizer influent can be processed separately via a once through single demin line-up, or in series via a once through, two demin line-up. Effluent from the demineralizers is routed to one of the three waste sample tanks, each of which has a capacity of 30,000 gallons. Conductivity instrumentation is provided to allow its performance to be monitored in the radwaste control room. This monitor will automatically divert an off-standard (high conductivity) batch back to the waste collector tanks for reprocessing.

If the water in the sample tanks is found to be of suitable quality for reuse, it is pumped to the cycled condensate storage tanks. When it is necessary to discharge treated wastes from the station, the contents of any of the waste sample tanks may be transferred to the discharge tanks for subsequent sampling and release.

The Unit 2 Floor drain collector pump discharge line is cross-tied to the equipment drain system by way of the Unit 2 Waste Surge tank to provide backup processing capability should the vendor processing unit be unavailable. The valve on the cross-tie line is administratively controlled to ensure proper routing of floor drain water.

11.2.2.2 Floor Drain Processing Subsystem

This subsystem collects and processes low purity (high conductivity) waste from the Unit 1 and Unit 2 floor drain systems. These are normally too high in conductivity for effective ion exchange treatment and may also be high in suspended solids. Note: A system bypass allows connection to portable waste treatment equipment. This enables the efficient processing of liquid radwaste through the use of state of the art radwaste processing technology. This technology may utilize ion exchange resins. They are treated by ion exchange for return to the condensate storage tank or discharge after appropriate sampling.

Input sources to this subsystem include the following:

a. reactor building floor drains (off-gas filter building floor drains),

b. turbine building floor drains, and

c. auxiliary building floor drains.

A provision is made for input from Chemical Waste Processing Subsystem to be transferred to the Floor Processing Subsystem. These wastes are collected in the two floor drain collector tanks, each of which has a capacity of 25,000 gallons. The operation of the floor drain collector tanks is the same as described for the waste collector tanks in Subsection 11.2.2.1.

The Unit 2 Floor drain collector pump discharge line is cross-tied to the equipment drain system by way of the Unit 2 Waste Surge tank to provide backup processing capability should the vendor processing unit be unavailable. The valve on the cross-tie line is administratively controlled to ensure proper routing of floor drain water.

The floor drain concentrator feed tanks (25,000 gallon capacity each) take input from the floor drain collector tanks and serve to condition these wastes for subsequent treatment. This conditioning is primarily caustic addition for pH control.

The floor drain concentrator units are abandoned-in-place. Historically, the evaporators provide a condensed distillate which is substantially free of radioactivity and dissolved solids; they also concentrate the residual wastes for delivery to the concentrator waste tanks. The condensed distillate is pumped to the floor drain head tanks (5,000-gallon capacity) for sampling and verification of concentrator performance.

The evaporator distillate is then pumped to the floor drain sample tanks (25,000 gallon capacity). The floor drain sample tanks are sampled to verify the water quality is suitable for further processing via the equipment waste processing system. The floor drain sample tank is then pumped to the waste collector tank and processed in the equipment drain system as described in Subsection 11.2.2.1.

Although more costly, the evaporator distillate can be processed through the floor drain demineralizer as opposed to being pumped from the evaporator head tank directly to the floor drain sample tank. The floor drain demineralizer effluent is routed to the floor drain concentrator sample tanks (25,000-gallon capacity). The demineralizer resins are not regenerated but are disposed of via the solid radwaste system when exhausted. The floor drain concentrator sample tanks provide for sampling of processed liquid waste to ensure that criteria for reuse or discharge are met.

Conductivity instrumentation is provided with automatic return to the collector tank on high conductivity.

Piping, electrical, and instrumentation & control systems exist to process floor drain water to a vendor skid. The system operation is the same as described for the chemical waste system in Subsection 11.2.2.3.

11.2.2.3 Chemical Waste Subsystem

This subsystem processes the highest conductivity water in the liquid radwaste system. Input sources to this system include the following:

a. radwaste demineralizer regenerants (no longer created),

b. decontamination drains,

c. laboratory drains,

d. radwaste building sump,

e. spent resin tank decantate,

f. waste sludge tank decantate, and

g. URC sludge tank decantate.

A provision is made for input from Radwaste Floor Drain processing subsystem to be transferred to the Chemical Waste System. These wastes are potentially high in radioactivity and conductivity; they may also be high in suspended solids, including some resin fines. Processing of these wastes is by chemical neutralization, and ion exchange demineralization.

<u>Processing through evaporators</u> (Abandoned-in place)

Wastes will usually be held in one of two chemical waste collector tanks (36,000-gallon capacity each). Sufficient surge capacity is provided to allow for variations in this cycle. Each collector tank is provided with a recirculation line and a 400-gpm pump to mix and transfer wastes to the chemical waste process tanks (25,000-gallon capacity each). Here the wastes are chemically neutralized for subsequent feed at 30 gpm to the chemical waste evaporator. One evaporator is provided to process effluent from both chemical waste processing tanks; however, it is crosstied with the floor drain concentrator units to provide additional evaporator capability if required. Chemical waste concentrator operation is identical to that of the floor drain concentrator described in Subsection 11.2.2.2. Evaporator concentrates are discharged to the concentrator waste tanks for transfer to the solid waste handling system. The condensed distillate is pumped to the chemical waste head tank. This tank has a capacity of 5,000 gallons and is fitted with two 75-gpm pumps connected in parallel to provide for recirculation and discharge. After verification of evaporator performance, the distillate is pumped to the chemical waste demineralizer, whose function and operation are identical to those of the floor drain demineralizers described in Subsection 11.2.2.2. This demineralizer is crosstied with the floor drain demineralizers to ensure sufficient demineralizer capability during abnormal surges of waste. Demineralizer effluent is discharged to the waste sample tanks for subsequent transfer to either the cycled condensate storage tanks or discharge tanks.

Capability exists to bypass the abandoned-in-place evaporators and process through a vendor system utilizing state-of-the-art technology. Processed water from the vendor system is routed to the floor drain head tank or collector tank dependent on water quality.

11.2.2.4 Laundry Waste Subsystem

This subsystem was designed to collect, process, and discharge laundry waste resulting from the washing of radioactive laundry, or other soapy, high organic wastes. The anti-contamination clothing used by the site is laundered by a vendor, thereby reducing the inputs to the system and the volume of waste created.

The treatment of this low radioactivity waste is by filtration. Depending on water quality in the laundry sample tank, the water is sent to the discharge tank, the chemical waste collector tank, or the laundry collector tank.

The use of filtration for low radioactivity waste is considered sufficient effort at reducing radioactive discharges to ALARA.

11.2.2.5 Sludge Subsystem

Liquid radwaste processing is designed to remove radioactive contaminants from waste liquids and to affix them on solids or concentrates, hold them for decay of radioactivity, and transfer them to the solid radwaste system for processing, temporary storage at the station, and shipment from the station.

The sludge subsystem, unlike those previously described, does not comprise a processing stream. It consists of a group of tanks and associated pumps which serve as an interface between the liquid radwaste system and the solid radwaste handling system. These tanks provide intermediate storage for slurries produced by radwaste decontamination equipment or other radioactive water cleanup systems. Most of these tanks provide settling capability for separation of liquid and solid wastes. Some also provide holdup for radioactive decay, while others (concentrator waste tanks) only provide batch accumulation and storage before transfer to the solid radwaste handling system. The following are the five major components of the sludge subsystem with their respective capacities:

NUMBER PROVIDED	COMPONENT	CAPACITY EACH (GAL)
1	Spent resin tank	15,000
1	Waste sludge tank	15,000
1	URC sludge tank	15,000
4	RWCU phase separators	6,900
2	Concentrator waste tanks *	5,000

* abandoned-in-place

Each tank in the sludge subsystem is provided with a means of recirculation to assure complete mixing of tank contents prior to and during discharge. Associated pumps and valves are operated from the liquid/solid radwaste interface panel adjacent to the solid radwaste system control panel. Both panels may be attended by one operator.

11.2.2.5.1 Spent Resin Tank

The spent resin tank receives exhausted ion exchange bead resins and charcoal discarded from the condensate polishing demineralizers, waste demineralizers, floor drain demineralizers, and chemical waste demineralizer. Resin/charcoal is allowed to settle in this tank, and excess sluicing water is drawn off from a decant port and pumped to a chemical waste collector tank. Settled resin/charcoal is discharged in batches to the solid radwaste handling system after appropriate recirculation.

The spent resin tank decant pump has a capacity of 76 gpm. Two progressive cavity type spent resin sludge pumps are connected in parallel, each having a capacity of 50 gpm. All valves in slurry-handling lines are plug type to preclude valve closure impairment due to the presence of solids in the fluid streams.

11.2.2.5.2 Waste Sludge Tank

The waste sludge tank, located below the waste collector filters, receives backwash from these filters and the fuel pool filter-demineralizers, as well as inputs from the various collector, surge, processing, or feed tank bottoms previously described.

The waste sludge tank decant pump is capable of discharging decantate at 60 gpm to the chemical waste collector tanks. Recirculation is provided by a mechanical mixer. Backflush capability is provided by the CY system. Two waste sludge pumps connected in parallel provide transportation of sludges to the solid radwaste handling system. The waste sludge tank component design is similar to that of the spent resin tank described in Subsection 11.2.2.5.1.

11.2.2.5.3 URC Sludge Tank

The URC sludge tank receives concentrated solids from the ultrasonic resin cleaner (URC) backwashes via the waste flocculation tanks described in Subsection 11.2.2.1. The URC sludge tank receives condensate prefilter backwash, the radwaste building floor drains (with the exception of the 710'-0" floor drains which go to the solid radwaste building floor drain sump) and the contents of the Concentrator Waste Tanks as described in Subsection 11.2.2.5.5. These are irregular inputs and of variable radioactivity.

The URC sludge decantate pump is capable of discharging decantate at 60 gpm to the chemical waste collector tank. Two URC sludge pumps connected in parallel, each having a capacity of 400 gpm, provide backflushing and recirculation. Backflush capability may also be provided by the CY system. Recirculation is also provided by a mechanical mixer. Two additional URC sludge pumps connected in parallel provide transportation of sludges to the solid radwaste handling system. The design is similar to that of the spent resin tank described in Subsection 11.2.2.5.1. Provision is made for adjusting the chemical properties of the tank.

11.2.2.5.4 RWCU Phase Separator Tanks

The four RWCU phase separator tanks collect, settle, and hold for decay sludge from the reactor water cleanup (RWCU) filter-demineralizer backwash. Backwash is discharged to the RWCU phase separator tanks in batches, about once a week per filter demineralizer. The solid phase of this waste constitutes the highest potential radioactivity level of the wastes regularly produced at the station. Two phase separators are provided for each unit so that one phase separator can collect inputs from that unit for a period of approximately 60 days while the other phase separator is quiescent, allowing the accumulated sludge to decay prior to its transfer to the solid radwaste handling system. As sludge is accumulated, excess liquid is decanted regularly and discharged to a waste collector tank.

Each phase separator is fitted with a phase separator sludge pump through which accumulated sludges are discharged to the solid radwaste handling system. Each of these centrifugal pumps has a capacity of 50 gpm. All valves in the sludge-handling lines are plug type or such other design as to preclude valve closure impairment due to the presence of solids in the fluid stream.

11.2.2.5.5 Concentrator Waste Tanks
(The concentrator waste tanks are abandoned-in-place, and the following information is historical.)
The two concentrator waste tanks were designed to receive concentrates from the floor drain and chemical waste concentrator packages. The tanks and all associated process lines are heat traced to preclude crystallization of these highly concentrated solutions. Each tank is fitted with a 50-gpm concentrator waste pump through which tank contents were designed to be discharged to the solid radwaste handling system. If solids concentration in the tanks is too low, or as needed to facilitate maintenance, the tank contents may be transferred to the URC Sludge Tank.

The air supply to the inlet valves to the concentrator waste tanks has been disconnected, ensuring the valves remain closed. The valves may be opened under administrative control to allow batching of evaporator concentrate provided level monitoring is established.

11.2.2.6 Operating Procedures

All inputs to the major subsystem collector tanks are thoroughly mixed so that appropriate treatment for each batch may be determined. Any batch intended for release from a licensed unit will be sampled for total activity at the discharge tanks.

If the discharge tank sample indicates the contents are suitable for discharge, a pumpout rate will be determined based upon dilution required as well as diluent lake blowdown available. The liquid radwaste system control panel has provision for recording the lake blowdown flow rate. The lake blowdown flow is also monitored locally at intake structure and recorded in the Main Control Room. The discharge flow rate is adjusted according to the blowdown flow rate. Administrative controls are exercised to ensure that the discharge valves are not opened until suitable conditions have been met. The activity in the Radwaste Discharge pipe is monitored continuously during release. Instrumentation is provided to close the discharge valve automatically upon measure of high activity. Further discussion of discharge can be found in Subsection 11.2.1.11.

11.2.2.7 Performance Tests

All batches of liquid wastes may be monitored before and after each processing step. The equipment is therefore subjected to continuous performance testing under station operating conditions.

Since data on specific isotope removal efficiencies are not conclusive, this system is designed to use conservative overall decontamination factors based on currently available equipment specifications now being incorporated into the design of similar power plant systems.

Through system crossties, redundancy of equipment, and excess storage capacity, ample provision has been made for equipment maintenance and for recycling inadequately processed wastes, if required.

11.2.2.8 Control and Instrumentation

The liquid radwaste system is primarily controlled and monitored from the liquid radwaste control panel (LRCP) located in the radwaste control room. Radwaste filter, radwaste demineralizer, waste concentrators, laundry RO, (no longer used) liquid/solid radwaste interface, and solid radwaste handling control panels are also located in the control room.

Operation of the liquid radwaste system is started or stopped manually or stopped automatically. A few subsystem operations are provided with automatic sequencing. (See Subsection 7.7.11 for a further description.) Instrumentation on system tanks includes as a minimum a high level detector for LRCP annunciation, a low level detector for pump cutoff, and LRCP level recording. The only exceptions to the above information are the concentrator waste tanks and the filter aid tank. Since these tanks are no longer in service, the high and low tank level alarms for these tanks have been defeated. Tanks containing waste sludges have an additional level detector to monitor the water/sludge interface for interlocking tank discharge valves and decant pump cutoff.

The LRCP is laid out in a logical fashion, with control switches for pump and valve operation and level recorders grouped in subsystem fashion under the appropriate section of a semigraphic display. Annunciators are located above the semigraphic, with the respective window lights in the vicinity of the monitored device.

Each valve hand switch has full open and full closed indicator lights. During the opening or closing of a valve, both lights are illuminated until the open or closed extreme is reached. In the open or closed position, the red or green light will be illuminated. All pump hand switches are provided with a third light to indicate automatic trip of the pump. Alternate pumps may then be controlled from the board for pumping tanks after correct valve alignment has been obtained by hand-switch operation. Each operating pump and each valve not fully closed is indicated in the semigraphic display by a red light.

The Radwaste Discharge line is continuously monitored for gross radioactivity by a monitor of the process radiation monitoring system (see Subsection 7.7.14.3). In case of radioactivity levels higher than a preset level, the discharge is automatically stopped, and simultaneous annunciation occurs in the main control room and on the LRCP.

Tank level control is discussed in Subsection 11.2.1.10.

11.2.3 Radioactive Releases

11.2.3.1 Criteria for Recycling

Liquid waste processing results in two streams, a product stream and a reject stream. In all cases the reject, or dirty, stream is processed for disposal in the solid radwaste system. The product, or clean, stream is returned to the primary cycle via the cycled condensate return tanks providing the water quality is acceptable and the plant has a makeup water requirement. New or nonradioactive water addition to the system would result in a discharge of radioactive water of equal volume (considering the plant radioactive water inventory as a "control volume").

11.2.3.2 Release Assumptions

The expected annual release of 0.051 Ci/year per reactor was the result of several assumptions. To assure a conservative approach, it was assumed that all chemical waste would be discharged after treatment. This was primarily regenerant chemicals from the condensate and radwaste demineralizers. (Note that bead resin is no longer regenerated.)

The following information is taken from the equipment vendor:

a. condensate demineralizers: (note the resin is no longer regenerated)

   1. 14 condensate polishers (7 per unit);
   2. 60-day regeneration cycle;
   3. 12,400 gallons of chemical regenerants at 2% solids by weight; and
   4. 20,000 gallons of low conductivity backwash (to equipment drains, not to chemical waste).

b. radwaste demineralizers: (note the resin is no longer regenerated)

   1. two demineralizers (1 per unit);
   2. 7-day regeneration cycle;
   3. 12,400 gallons chemical regenerants/cycle; and
   4. 20,000 gallons of low conductivity backwash per cycle (to equipment drains, not to chemical waste).

Evaporator and demineralizer decontamination factors (DF's) are presented in Table 11.2-7.

11.2.3.3 Total Releases - Comparison with 10 CFR 20 Limits

Calculated annual releases of radionuclides are given in Table 11.2-1. Liquid waste is discharged into the cooling lake blowdown line; the resulting mixture of liquid radwaste and cooling lake blowdown is subsequently discharged into the Illinois River as shown in Drawing No. M-2. The cooling lake blowdown flow ranges to a maximum of 81,000 gpm depending on cooling lake conditions. The average lake blowdown on an annual basis is 30,000 gpm. Liquid radwaste discharge rate ranges from 30 gpm to a calculated maximum of 90 gpm. Table 11.2-1 lists effluent concentrations prior to discharge in the river after dilution based on 90 gpm radwaste discharge and 30,000 gpm lake blowdown. These are well below 10 CFR 20 Table II, Column 2 limits also listed in Table 11.2-1.

11.2.3.4 Radiological Dose Assessment - Comparison with Appendix I Guidelines

Expected annual average radiological doses associated with radionuclides released in liquid effluents have been estimated for areas beyond the restricted area boundary. The annual releases given in Table 11.2-1 were assumed for calculation of the average annual doses. Other assumptions are listed in Table 11.2-8. The resulting annual doses are listed in Table 11.2-2 and are well below the design objectives of Appendix I to 10 CFR 50.

An alternate calculation of releases and dose assessment was made to demonstrate compliance to Appendix I and submitted to the NRC for review (Reference 1). This submittal, as requested by the NRC, utilized assumptions of Reference 2 and the BWR-GALE code. There are some differences between the UFSAR and the Reference 1 transmittal due to the following:

a. Differences in initial coolant activities. (The present analysis used design basis coolant activities; Reference 1 utilized Reference 2 and draft ANS standard data.)

b. Differences in analytic modeling of the radwaste system. (The BWR-GALE code utilized in Reference 1 is limited to three process streams. The user must input single "effective" holdup times and total decontamination factors descriptive of the complete system. The present analysis considered all of the separate processing paths and separate equipment. The present modeling is considered to be much more representative of the LSCS liquid radwaste system.)

c. Differences in discharge fraction. (The present analysis assumed 100% discharge of the chemical waste stream and 100% recycle of the remaining streams: Reference 1 assumed 1% discharge of the high purity stream and 10% discharge of all other streams.)

d. Additional release to account for "anticipated operational occurrences." (Reference 1 included an additional 0.15 Ci/year release to account for anticipated operational occurrences as per the guidance of Section 2.2.20.1 of Reference 2. It should be noted that before the additional 0.15 Ci/year is added, the total curie release reported in Table 11.2-1 is similar to that of Reference 2 in spite of the differences in the assumptions.)

The doses calculated using the NRC models and reported in Reference 1 are also below the design objectives of Appendix I to 10 CFR 50.

11.2.4 <u>References</u>

1. "Information Relevant to Keeping Levels of Radioactivity in Effluents to Unrestricted Areas as Low as Reasonably Achievable" (filed in accordance with Section V.B. of 10 CFR 50, Appendix I), LaSalle County Station, Units 1 and 2, June 4, 1976.

2. "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Boiling Water Reactors (BWR-GALE Code)," NUREG-0016, April 1976.

TABLE 11.2-1
(Sheet 1 of 2)

EXPECTED RELEASES OF RADIOACTIVE MATERIALS FROM NORMAL OPERATIONS AND OPERATIONAL OCCURRENCES COMPARED TO 10 CFR 20 LIMITS (f)

NUCLIDE	ANNUAL DISCHARGE NORMAL OPERATIONS (Ci/yr PER REACTOR) (a)*	CONCENTRATION IN BLOWDOWN LINE BASED ON 30,000 gpm LAKE BLOWDOWN (μCi/cm^3)	MAX PERMISSIBLE CONCENTRATION (μCi/cm^3) (b)
NA 24	7.4-07 (c)	1.4-12	3-05
P 32	3.5-07	6.4-13	2-05
CR 51	3.5-04	6.4-10	2-03
MN 54	5.2-05	9.1-11	1-04
MN 56	7.0-06	1.3-11	1-04
CO 58	5.2-03	9.2-09	9-05
FE 59	7.2-05	1.3-10	5-05
CO 60	6.9-04	1.3-09	3-05
NI 65	5.1-08	7.6-14	1-04
ZN 65	1.0-07	1.9-13	1-04
ZN 69M	9.5-09	1.7-14	6-05
ZN 69	1.0-08	1.9-14	2-03
BR 83	1.4-06	2.6-12	3-06
BR 84	2.4-08	4.4-14	(d)
BR 85	4.8-12	8.8-18	(d)
SR 89	1.2-04	2.2-10	3-06
Y 89M	1.2-08	2.2-14	(d)
SR 90	1.3-05	2.4-11	3-07
Y 90	1.2-05	2.2-11	2-05
SR 91	1.1-05	2.0-11	5-05
Y 91	1.9-05	1.3-10	3-05
SR 92	7.1-07	1.3-12	6-05
Y 92	2.6-06	4.8-12	6-05
ZR 95	4.1-05	7.4-11	6-05
NB 95M	7.7-07	1.4-12	3-06
NB 95	5.2-05	9.6-11	1-04
ZR 97	3.7-07	6.8-13	2-05
NB 97M	3.7-07	6.8-13	(e)
NB 97	3.9-07	4.2-13	9-04
MO 99	1.7-03	3.0-09	4-05
TC 99M	1.0-03	1.9-09	3-03
TC 99	7.6-11	1.4-16	2-04
TC 101	4.5-10	8.4-16	(d)
RU 103	1.6-05	3.0-11	8-05
RH 103M	1.6-05	3.0-11	1-03
RU 106	3.4-06	6.2-12	1-05
RH 106	3.4-06	6.2-12	(e)
AG 110M	7.7-05	1.4-10	3-05
AG 110	1.0-06	1.8-12	(d)
TE 129M	1.3-06	2.4-12	2-05
TE 129	8.1-07	1.5-12	8-04
I 129	5.1-14	9.4-20	8-04

- Notes a, b, c, d, and e, and f are found on the last page of this table.

TABLE 11.2-1
(Sheet 2 of 2)

NUCLIDE	ANNUAL DISCHARGE NORMAL OPERATIONS (Ci/yr PER REACTOR) [(a)]	CONCENTRATION IN BLOWDOWN LINE BASED ON 30,000 gpm LAKE BLOWDOWN ($\mu Ci/cm^3$)	MAX PERMISSIBLE CONCENTRATION ($\mu Ci/cm^3$) (b)
I 131	2.6-02	4.8-08	3-07
TE 132	1.8-04	3.2-10	2-05
I 132	3.4-04	6.2-10	8-06
I 133	1.1-02	2.0-08	1-06
I 134	9.9-06	1.8-11	2-05
CS 134	7.8-05	1.4-10	9-06
I 135	1.9-03	3.4-09	4-06
CS 136	1.6-05	2.8-11	6-05
CS 137	1.2-04	2.2-10	2-05
BA 137M	1.1-04	2.0-10	(d)
CS 138	7.9-08	1.5-13	(d)
BA 139	1.4-07	2.6-13	(d)
BA 140	1.4-04	2.6-10	2-05
LA 140	1.5-04	2.6-10	2-05
BA 141	1.2-09	2.2-15	(e)
LA 141	2.6-07	4.6-13	2-05
CE 141	1.3-04	2.4-10	9-05
BA 142	2.5-10	4.4-16	(d)
LA 142	2.7-08	5.0-14	(d)
CE 143	1.1-06	2.0-12	4-05
PR 143	1.7-05	3.2-11	5-05
CE 144	4.5-05	8.2-11	1-05
PR 144	4.5-05	8.2-11	(d)
ND 147	4.8-06	8.8-12	6-05
PM 147	1.7-07	3.0-13	2-04
W 187	5.9-05	1.1-10	6-05
NP 239	6.0-04	1.1-9	1-04
TOTALS	5.1-02	9.2-08	
H-3	8.2-00(e)	1.5-05(e)	3-03

[(a)] Based on assumptions given in Subsection 11.2.3.2.
[(b)] From 10 CFR 20, Table II, Column 2.
[(c)] 7.4-07 means 7.4 x 10-7.
[(d)] Half-life less than 2 hours.
[(e)] At equilibrium.
[(f)] Table 11.2-1 is not intended to contain limits for the radionuclides or imply that all the listed radionuclides must be present in the given proportions.

TABLE 11.2-2

RADIOLOGICAL DOSE TO INDIVIDUAL AT OR

BEYOND THE SITE BOUNDARY*

ANNUAL DOSES, mrem/yr PER UNIT

		ORGAN:			
	TOTAL BODY	SKIN	GI-LLI	THYROID	BONE
APPENDIX I, 10 CFR 50 DESIGN OBJECTIVE	3	10	10	10	10
ANNUAL DOSE AT LSCS DUE TO:					
Drinking water	5.4×10^{-5}	-	6.1×10^{-5}	2.6×10^{-3}	8.3×10^{-6}
Eating fish	6.6×10^{-4}	-	9.3×10^{-4}	1.4×10^{-2}	5.4×10^{-4}
Shoreline activities	1.5×10^{-5}	1.8×10^{-5}	1.5×10^{-5}	1.5×10^{-5}	1.5×10^{-5}
Swimming	1.1×10^{-6}	1.5×10^{-6}	1.1×10^{-6}	1.1×10^{-6}	1.1×10^{-6}
Boating	2.6×10^{-6}	3.5×10^{-6}	2.6×10^{-6}	2.6×10^{-6}	2.6×10^{-6}

- Principal assumptions are listed in Table 11.2-8.

TABLE 11.2-3
(Sheet 1 of 2)

LIQUID RADWASTE SYSTEM COMPONENTS AND DESIGN PARAMETERS

I. TANKS

NUMBER	DESCRIPTION	CAPACITY (gal)
2	Waste collector (1WEO1T, 2WEO1T)	30,000
2	Waste surge (1WEO2T, 2WEO2T)	50,000
3	Waste sample (1, 2WEO3T, OWEO1T)	30,000
2	Discharge (1, 2WFO5T)	25,000
2	Floor drain collector (1WFO1T, 2WFO1T)	25,000
2	Waste flocculation (1WEO4T, 2WEO4T)	25,000
4	Floor drain concentrator feed (1WFO3TA, B; 2WFO3TA, B)	25,000
2	Floor drain head (1WFO4T, 2WFO4T)	5,000
2	Chemical waste collector (1WZO1T, 2WZO1T)	36,000
2	Chemical waste processing (1WZO2T, 2WZO2T)	25,000
1	Chemical waste head (OWZO1T)*	5,000
2	Laundry drain collector (OWYO1TA, B)*	1,000
1	Laundry sample (OWYO2T)*	1,000
4	Phase separator (1, 2WXO1TA, B)	6,900
2	Concentrated waste (OWXO7TA, B)* (abandoned-in-place)	5,000
1	Spent resin (OWXO3T)*	15,000
1	Waste sludge (OWXO1T)*	15,000
1	URC sludge (OWXO2T)*	15,000

II. FILTERS AND DEMINERALIZERS

NUMBER	DESCRIPTION	FLOW (gal/min)
2	Waste filter (1WEO1F, 2WEO1F)	300
2	Waste demineralizers (1WEO1D, 2WEO1D)	300
2	Floor drain demineralizers (1WFO1D, 2WFO1D)	75
1	Chemical waste demineralizer (OWZO1D)*	75

* Asterisk indicates equipment common to Units 1 and 2.

TABLE 11.2-3
(Sheet 2 of 2)

III. CONCENTRATORS

NUMBER	DESCRIPTION	FLOW (gal/min)	STEAM REQUIRED (lb/hr)
1	Chemical waste concentrator (0WZ01S)* (abandoned-in-place)	30	21,000
2	Floor drain concentrator (1WF01S, 2WF01S) (abandoned-in-place)	30	21,000

IV. LAUNDRY

NUMBER	DESCRIPTION	FLOW (gal/min)
1	Laundry (0WY02F)*	4.0

* Asterisk indicates equipment common to Units 1 and 2.

TABLE 11.2-4

DESIGN-BASIS INFLUENT RADIOACTIVITY CONCENTRATIONS IN LIQUID WASTE

PROCESSING STREAMS AND COMPARISON TO REACTOR WATER ACTIVITY

ISOTOPE	CHEMICAL WASTE (μCi/cm³)	EQUIPMENT DRAIN (μCi/cm³)	FLOOR DRAIN (μCi/cm³)	LAUNDRY (μCi/cm³)
N-13	0.0	0.0	0.0	0.0
N-16	0.0	0.0	0.0	0.0
N-17	0.0	0.0	0.0	0.0
0-19	0.0	0.0	0.0	0.0
F-18	0.0	0.0	0.0	0.0
Na-24	1.0-03	2.0-03	6.2-04	0.0
P-32	2.2-04	2.0-05	6.2-06	0.0
Cr-51	4.5-03	5.0-04	1.6-04	0.0
Mn-54	6.3-04	4.0-05	1.2-05	1.5-06
Mn-56	2.2-03	5.0-02	1.6-02	0.0
Co-58	6.4-02	5.0-03	1.6-03	8.4-06
Fe-59	8.9-04	8.0-05	2.5-05	0.0
Co-60	8.4-03	5.0-04	1.6-04	1.3-05
Ni-65	1.3-05	3.0-04	9.3-05	0.0
Zn-65	6.2-05	2.0-06	6.2-07	0.0
Zn-69m	1.4-05	3.0-05	9.3-06	0.0
Br-83	1.2-03	1.5-02	4.6-03	0.0
Br-84	4.9-04	2.7-02	8.3-03	0.0
Br-85	3.2-05	1.7-02	5.2-03	0.0
Sr-89	7.2-02	3.1-03	9.6-04	0.0
Sr-90	7.8-03	2.3-04	7.1-05	0.0
Sr-91	2.3-02	6.9-02	2.1-02	0.0
Sr-92	1.0-02	1.1-01	3.4-02	0.0
Zr-95	1.0-02	4.0-05	1.2-05	2.0-06
Nb-95	6.4-04	4.2-05	1.3-06	2.9-06
Zr-97	9.3-06	3.2-05	9.9-06	0.0
Mo-99	2.5-02	2.2-02	6.8-03	0.0
Tc-99m	7.9-02	2.8-01	8.7-02	0.0
Tc-99	4.3-08	0.0	0.0	0.0
Tc-101	1.1-03	1.4-01	4.3-02	0.0
Ru-103	2.0-04	1.9-05	5.9-06	2.0-07
Ru-106	4.2-05	2.6-06	8.1-07	3.5-06
Ag-110m	9.4-04	6.0-05	1.9-05	6.4-07
Te-129m	7.8-04	4.0-05	1.2-05	0.0
I-131	8.5-01	1.3-02	4.0-03	0.0
Te-132	1.3-01	4.9-02	1.5-02	0.0
I-132	2.2-01	1.2-01	3.7-02	0.0
I-133	6.3-01	8.9-02	2.8-02	0.0
I-134	7.1-02	2.4-01	7.4-02	0.0
Cs-134	4.8-03	1.6-04	5.0-05	1.9-05
I-135	3.0-01	1.3-01	4.0-02	0.0
Cs-136	1.0-03	1.1-04	3.4-05	0.0
Cs-137	7.3-03	2.4-04	7.4-05	3.5-05
Cs-138	3.2-03	1.9-01	5.9-02	0.0
Ba-139	7.5-03	1.6-01	5.0-02	0.0
Ba-140	9.0-02	9.0-03	2.8-03	0.0
Ba-141	1.8-03	1.7-01	5.3-02	0.0
Ce-141	1.6-03	3.9-05	1.2-05	0.0
Ba-142	1.1-03	1.7-01	5.3-02	0.0
Ce-143	2.0-05	3.5-05	1.1-05	0.0
Pr-143	2.2-04	3.8-05	1.2-05	0.0
Ce-144	5.5-04	3.5-05	1.1-05	7.3-06
Nd-147	6.2-05	1.4-05	4.3-06	0.0
W-187	1.2-03	3.0-03	9.3-04	0.0
Np-239	4.6-01	2.4-01	7.4-02	0.0
Equivalent fraction of reactor water	Not a direct fraction. Calculated values based on assumptions stated in Subsection 11.2.3.2.	1.00*	0.31**	Not a direct fraction.

* Basis given in Subsection 11.2.1.7.7.

** Draft ALARA Regulatory Guide gives range of 0.01 to 1.00 for influent streams. NEDO-10951: Releases from BWR Radwaste Management systems (GE) gives 0.00017 to 0.30 as a range. General Electric Standard Safety Analysis Report gives 0.009 to 0.365 as a range. The 0.31 value is a weighted average from these references.

TABLE 11.2-5
(Sheet 1 of 8)

EXPECTED INVENTORIES IN LIQUID RADWASTE SYSTEM COMPONENTS
INVENTORY SUMMARY (CURIES)

ISOTOPE NAME	WASTE COLLECTION TANK	WASTE SURGE TANK	WASTE FLOCCULATION TANK	WASTE DEMINERALIZER	WASTE FILTER	FLOOR DRAIN TANK	FLOOR DRAIN EVAPORATOR *
N 13	2.180-02	3.796-02	4.647-04	9.688-04	0.000	1.506-02	2.228-06
N 16	1.308-01	2.279-01	2.789-03	5.816-03	0.000	9.043-02	4.104-07
N 17	2.272-05	3.957-05	4.844-07	1.010-06	0.000	1.570-05	4.158-11
F 18	2.384-02	4.164-02	5.097-04	1.061-03	0.000	1.651-02	3.749-06
O 19	1.847-02	3.217-02	3.938-04	8.210-04	0.000	1.277-02	2.210-07
NA 24	5.031-02	1.525-01	1.751-03	1.376-01	4.022-03	3.931-02	3.123-04
P 32	6.937-04	3.611-03	3.634-05	2.215-02	1.348-03	5.760-04	4.606-06
CR 51	1.748-02	9.239-02	9.258-04	3.280-02	1.311+00	1.453-02	1.162-04
MN 54	1.408-03	7.560-03	7.544-05	2.980-03	1.332-01	1.173-03	9.383-06
MN 56	4.146-01	7.333-01	8.976-03	9.699-03	1.023-01	2.891-01	2.223-03
CO 58	1.756-01	9.379-01	9.373-03	3.571-01	1.536+01	1.462-01	1.169-03
FE 59	2.805-03	1.492-02	1.493-04	5.534-03	2.315-01	2.334-03	1.867-05
CO 60	1.762-02	9.468-02	9.445-04	3.767-02	1.700+00	1.467-02	1.174-04
NI 65	2.473-03	4.373-03	5.352-05	5.757-05	6.065-04	1.724-03	1.326-05
ZN 65	7.041-05	3.778-04	3.770-06	2.851-03	2.142-04	5.864-05	4.690-07
ZN 69M	7.327-04	2.140-03	2.475-05	1.802-03	5.159-05	5.697-04	4.523-06
ZN 69	6.871-04	2.124-03	2.445-05	1.918-03	5.512-05	5.428-04	4.360-06
BR 83	1.167-01	2.055-01	2.516-03	4.465-02	8.672-04	8.119-02	6.226-04
KR 83M	0.000	0.000	0.000	0.000	0.000	0.000	0.000
BR 84	4.681-02	8.153-02	9.980-04	5.506-03	7.581-05	3.235-02	2.164-04
BR 85	2.783-03	4.847-03	5.932-05	1.396-04	4.254-07	1.923-03	5.005-06
KR 85M	0.000	0.000	0.000	0.000	0.000	0.000	0.000
KR 85	0.000	0.000	0.000	0.000	0.000	0.000	0.000
SR 89	1.088-01	5.794-01	5.795-03	4.163+00	2.967-01	9.052-02	7.240-04
Y 89M	1.087-05	5.794-05	5.794-07	4.163-04	2.967-05	9.049-06	7.240-08
SR 90	8.104-03	4.356-02	4.346-04	3.330-01	2.539-02	6.750-03	5.400-05
Y 90M	0.000	0.000	0.000	0.000	0.000	0.000	0.000
Y 90	6.458-04	9.596-03	8.038-05	2.342-01	2.161-02	6.413-04	5.209-06
SR 91	1.466+00	3.679+00	4.370-02	2.384+00	6.240-02	1.114+00	8.819-03
Y 91M	8.209-01	2.153+00	2.550-02	1.512+00	3.977-02	6.323-01	5.071-03
Y 91	6.120-03	6.255-02	5.752-04	6.284-01	4.591-02	5.855-03	4.747-05
SR 92	9.567-01	1.698+00	2.079-02	4.084-01	8.073-03	6.683-01	5.149-03
Y 92	8.371-01	1.698+00	2.075-02	8.452-01	1.859-02	6.191-01	4.958-03
ZR 95	1.405-03	7.497-03	7.493-05	2.842-03	1.217-01	1.169-03	9.352-06
NB 95M	1.619-06	2.468-05	2.053-07	3.466-05	1.943-03	1.612-06	1.310-08
NB 95	1.479-03	7.945-03	7.927-05	3.139-03	1.400-01	1.232-03	9.856-06
ZR 97	8.355-04	2.664-03	3.026-05	1.402-04	2.459-03	6.571-04	5.225-06
NB 97M	8.321-04	2.655-03	3.016-05	1.399-04	2.454-03	6.546-04	5.213-06
NB 97	7.631-04	2.630-03	2.967-05	1.491-04	2.631-03	6.132-04	4.931-06
MO 99	7.160-01	3.284+00	3.419-02	5.166-01	1.184+01	5.870-01	4.688-03
TC 99M	4.922+00	1.196+01	1.398-01	5.426+00	1.046+01	3.669+00	2.892-02
TC 99	1.816-08	1.650-07	1.528-09	1.516-06	1.799-06	1.686-08	1.366-10
TC 101	1.069-01	1.862-01	2.280-03	8.123-03	7.629-05	7.390-02	4.093-04
RU 103	6.657-04	3.537-03	3.539-05	1.299-03	5.376-02	5.539-04	4.430-06
RH 103H	6.074-04	3.437-03	3.417-05	1.299-03	5.381-02	5.137-04	4.141-06
RU 106	9.156-05	4.916-04	4.905-04	1.942-04	8.694-03	7.626-05	6.100-07
RH 106M	0.000	0.000	0.000	0.000	0.000	0.000	0.000
RH 106	9.148-05	4.914-04	4.904-06	1.942-04	8.694-03	7.621-05	6.100-07
AG 110M	2.112-03	1.133-02	1.131-04	4.459-03	1.988-01	1.759-03	1.407-05

* Abandoned-in-place

TABLE 11.2-5
(Sheet 2 of 8)
INVENTORY SUMMARY (CURIES)

ISOTOPE NAME	WASTE COLLECTION TANK	WASTE SURGE TANK	WASTE FLOCCULATION TANK	WASTE DEMINERALIZER	WASTE FILTER	FLOOR DRAIN TANK	FLOOR DRAIN EVAPORATOR *
AG110	2.744-05	1.473-04	1.470-06	5.797-05	2.584-03	2.286-05	1.829-07
TE129M	1.400-03	7.424-03	7.434-05	5.163-02	3.556-03	1.165-03	9.316-06
TE129	8.010-04	4.591-03	4.559-05	3.307-02	2.279-03	6.799-04	5.482-06
I 129	4.450-14	7.998-13	6.439-15	3.425-11	4.702-12	4.559-14	3.705-16
I 131	4.455-01	2.262+00	2.292-02	1.186+01	6.299-01	3.690-01	2.948-03
XE131M	0.000	0.000	0.000	0.000	0.000	0.000	0.000
TE132	1.613+00	7.558+00	7.823-02	2.526+01	1.020+00	1.325+00	1.058-02
I 132	2.179+00	8.698+00	9.174-02	2.622+01	1.055+00	1.725+00	1.365-02
I 133	2.453+00	8.463+00	9.448-02	9.960+00	3.118-01	1.948+00	1.550-02
XE133M	0.000	0.000	0.000	0.000	0.000	0.000	0.000
XE133	0.000	0.000	0.000	0.000	0.000	0.000	0.000
I 134	6.804-01	1.185+00	1.451-02	1.122-01	1.802-03	4.702-01	3.356-03
CS134	5.636-03	3.028-02	3.021-04	2.096-01	1.750-02	4.694-03	3.755-05
I 135	2.279+00	4.930+00	5.971-02	2.443+00	5.798-02	1.686+00	1.327-02
XE135M	0.000	0.000	0.000	0.000	0.000	0.000	0.000
XE135	0.000	0.000	0.000	0.000	0.000	0.000	0.000
CS135	0.000	0.000	0.000	0.000	0.000	0.000	0.000
CS136	3.810-03	1.976-02	1.991-04	1.079-01	7.083-03	3.162-03	2.529-05
CS137	8.456-03	4.546-02	4.535-04	3.160-01	2.650-02	7.044-03	5.635-05
BA137M	7.877-03	4.245-02	4.234-04	2.955-01	2.477-02	6.565-03	5.264-05
CS138	3.331-01	5.802-01	7.101-03	3.718-02	5.454-04	2.302-01	1.543-03
BA139	7.215-01	1.257+00	1.539-02	1.710-01	3.042-03	4.988-01	3.711-03
BA140	3.116-01	1.616+00	1.628-02	9.673+00	5.754-01	2.586-01	2.068-03
LA140	3.850-02	5.249-01	4.497-03	8.432+00	5.681-01	3.789-02	3.076-04
BA141	1.666-01	2.903-01	3.553-03	1.421-02	1.526-04	1.152-01	6.844-04
LA141	1.553-01	2.902-01	3.549-03	1.011-01	2.135-03	1.107-01	8.805-04
CE141	2.798-03	1.769-02	1.737-04	2.349-02	2.610-01	2.427-03	1.953-05
BA142	1.019-01	1.775-01	2.172-03	7.021-03	5.705-05	7.042-02	3.597-04
LA142	1.018-01	1.775-01	2.172-03	2.895-02	5.344-04	7.041-02	5.503-04
CE143	1.052-03	4.178-03	4.511-05	3.762-04	7.473-03	8.493-04	6.772-06
PR143	1.335-03	7.082-03	7.097-05	2.348-03	8.450-02	1.111-03	8.888-06
CE144	1.232-03	6.614-03	6.600-05	2.606-03	1.163-01	1.026-03	8.209-06
PR144	1.199-03	6.556-03	6.530-05	2.605-03	1.163-01	1.004-03	8.076-06
ND147	4.835-04	2.493-03	2.515-05	7.516-04	2.567-02	4.011-04	3.207-06
PM147	1.143-07	1.844-06	1.511-08	3.538-06	2.556-04	1.145-07	9.303-10
W 187	8.507-02	3.067-01	3.389-03	2.112-02	3.968-01	6.792-02	5.409-04
NP239	7.698+00	3.435+01	3.601-01	9.113+01	3.391+00	6.292+00	5.024-02
TOTAL =	3.040+01	1.009+02	1.106+00	2.037+02	5.097+01	2.354+01	1.842-01

* Abandoned-in-place

INVENTORY SUMMARY (CURIES)

ISOTOPE NAME	FLOOR SAMPLE TANK	FLOOR MIXED-BED DEMINERALIZER	FLOOR CONCENTRATOR SAMPLE TANK	CONDENSATE POLISHER	CHEMICAL WASTE COLLECTION TANK
N 13	1.877-04	6.193-06	1.169-05	0.000	0.000
N 16	1.352-05	4.463-07	1.011-08	0.000	0.000
N 17	1.359-09	4.486-11	5.884-13	0.000	0.000
F 18	1.758-03	5.800-05	9.356-04	0.000	0.000
O 19	7.702-06	2.542-07	2.323-08	0.000	0.000
NA 24	7.411-07	4.148-06	6.986-09	4.647-02	4.153-02
P 32	1.184-08	9.333-07	1.217-10	1.002-02	8.996-03
CR 51	2.993-07	1.466-05	1.541-07	2.000-01	0.000
MN 54	2.420-08	1.471-06	1.248-08	2.883-02	0.000
MN 56	3.660-06	1.882-06	1.159-06	9.996-02	0.000
CO 58	3.014-06	1.703-04	1.554-06	2.920+00	0.000
FE 59	4.810-08	2.575-04	2.478-08	4.027-02	0.000
CO 60	3.028-07	1.876-05	1.562-07	3.815-01	0.000
NI 65	2.177-08	1.113-08	6.872-09	5.961-04	0.000
ZN 65	1.210-09	1.448-07	1.248-11	2.835-03	2.546-03
ZN 69M	1.065-08	5.453-08	9.956-11	6.373-04	5.693-04
ZN 69	1.098-08	5.840-08	1.054-10	6.370-04	5.719-04
BR 83	9.966-07	9.201-07	6.117-09	1.117+00	9.735-01
KR 83M	0.000	0.000	0.000	0.000	0.000
BR 84	1.275-07	2.923-08	2.514-10	4.467-01	3.510-01
BR 85	7.196-10	3.708-11	1.346-13	2.913-02	8.548-03
KR 85M	0.000	0.000	0.000	0.000	0.000
KR 85	0.000	0.000	0.000	0.000	0.000
SR 89	1.866-06	2.016-04	1.923-08	3.289+00	2.953+00
Y 89M	1.866-10	2.016-08	1.923-12	3.289-04	2.953-04
SR 90	1.393-07	1.714-05	1.437-09	3.537-01	3.176-01
Y 90M	0.000	0.000	0.000	0.000	0.000
Y 90	1.605-08	1.424-05	1.941-10	3.311-01	2.975-01
SR 91	2.001-05	7.226-05	1.796-07	1.031+00	9.189-01
Y 91M	1.235-05	4.624-05	1.138-07	6.051-01	5.431-01
Y 91	1.416-07	3.096-05	1.652-09	5.191-01	4.662-01
SR 92	8.651-06	8.964-06	5.599-08	4.615-01	4.035-01
Y 92	1.189-05	2.461-05	1.048-07	4.606-01	4.135-01
ZR 95	2.411-08	1.350-06	1.242-08	2.279-02	2.046-02
NB 95M	4.009-11	2.091-08	2.414-11	4.246-04	3.146-07
NB 95	2.542-08	1.547-06	1.311-08	2.919-02	2.477-06
ZR 97	1.252-08	4.026-08	5.962-09	4.216-04	3.770-04
NB 97M	1.249-08	4.018-08	5.950-09	4.201-04	3.327-04
NB 97	1.272-08	4.315-08	6.266-09	4.214-04	1.755-05
MO 99	1.187-05	1.479-04	5.997-06	1.139+00	0.000
TC 99M	6.275-05	2.701-04	1.566-06	3.591+00	3.186+00
TC 99	3.989-13	9.461-11	5.642-15	1.959-06	1.759-06
TC 101	1.290-07	1.541-08	1.127-10	5.168-02	3.459-02
RU 103	1.141-08	5.987-07	5.879-09	9.064-03	0.000
RH 103M	1.115-08	5.992-07	5.839-09	9.059-03	0.000
RU 106	1.573-09	9.603-08	8.116-10	1.896-03	0.000
RH 106M	0.000	0.000	0.000	0.000	0.000
RH 106	1.573-09	9.603-08	8.116-10	1.896-03	0.000
AG 110M	3.629-08	2.196-06	1.872-08	4.268-02	0.000

TABLE 11.2-5
(Sheet 4 of 8)
INVENTORY SUMMARY (CURIES)

ISOTOPE NAME	FLOOR SAMPLE TANK	FLOOR MIXED-BED DEMINERALIZER	FLOOR CONCENTRATOR SAMPLE TANK	CONDENSATE POLISHER	CHEMICAL WASTE COLLECTION TANK
AG110	4.718-10	2.855-08	2.434-10	5.548-04	0.000
TE129M	2.399-08	2.426-06	2.472-10	3.556-02	3.193-02
TE129	1.486-08	1.554-06	1.564-10	2.275-02	2.043-02
I 129	9.624-18	4.073-15	1.016-19	1.424-10	1.279-10
I 131	7.561-05	4.445-03	7.746-07	7.724+01	6.933+01
XE131M	0.000	0.000	0.000	0.000	0.000
TE132	2.686-05	7.661-04	2.723-07	5.910+00	5.302+00
I 132	2.185-04	9.580-04	1.427-06	1.445+01	1.274+01
I 133	3.768-04	2.942-03	3.643-06	5.778+01	5.171+01
XE133M	0.000	0.000	0.000	0.000	0.000
XE133	0.000	0.000	0.000	0.000	0.000
I 134	2.921-05	1.034-05	9.075-08	6.468+00	5.349+00
CS134	9.686-08	1.074-05	9.993-09	2.161-01	1.940-01
I 135	2.852-04	7.172-04	2.413-06	2.695+01	2.394+01
XE135M	0.000	0.000	0.000	0.000	0.000
XE135	0.000	0.000	0.000	0.000	0.000
CS135	0.000	0.000	0.000	0.000	0.000
CS136	6.497-08	4.469-06	6.674-09	4.574-02	0.000
CS137	1.454-07	1.626-05	1.500-08	3.326-01	0.000
BA137M	1.360-07	1.520-05	1.384-08	3.110-01	0.000
CS138	9.173-07	1.957-07	1.827-08	1.433-01	0.000
BA139	4.449-06	2.418-06	1.984-08	3.423-01	2.918-01
BA140	5.313-06	3.995-04	5.458-08	4.102+00	3.683+00
LA140	9.343-07	3.825-04	1.115-08	4.077+00	3.661+00
BA141	2.581-07	3.715-08	2.892-10	8.014-02	5.708-02
LA141	1.758-06	2.620-06	1.313-08	7.872-02	7.052-02
CE141	5.340-08	3.582-06	2.752-08	7.337-02	6.452-06
BA142	9.650-08	9.730-09	6.612-11	4.955-02	3.077-02
LA142	7.521-07	4.582-07	3.652-09	4.813-02	4.283-02
CE143	1.681-08	1.043-07	8.321-09	8.935-04	0.000
PR143	2.291-08	9.555-07	1.180-08	9.980-03	0.000
CE144	2.117-08	1.285-06	1.092-08	2.511-02	0.000
PR144	2.113-08	1.285-06	1.092-08	2.511-02	0.000
ND147	8.234-09	2.927-07	4.226-09	2.817-03	0.000
PM147	2.860-12	2.712-09	1.731-12	9.112-05	0.000
W 187	1.324-06	5.950-06	6.448-07	5.541-02	0.000
NP239	1.267-04	2.644-03	1.276-06	2.093+01	0.000
TOTAL =	3.261-03	1.446-02	9.693-04	2.371+02	1.874+02

INVENTORY SUMMARY (CURIES)

ISOTOPE NAME	CHEMICAL WASTE EVAPORATOR *	CHEMICAL WASTE SAMPLE TANK	CHEMICAL WASTE DEMINERALIZER	WASTE SAMPLE TANK	RWCU FILTER/ DEMINERALIZER	RWCU PHASE SEPARATOR	FUEL POOL F/D
N 13	0.000	0.000	0.000	1.839-04	0.000	0.000	0.000
N 16	0.000	0.000	0.000	1.325-05	0.000	0.000	0.000
N 17	0.000	0.000	0.000	1.332-09	0.000	0.000	0.000
F 18	0.000	0.000	0.000	2.208-03	0.000	0.000	0.000
O 19	0.000	0.000	0.000	7.547-06	0.000	0.000	0.000
NA 24	6.747-04	1.277-06	7.969-07	3.157-04	1.197+00	2.395+00	6.752-03
P 32	1.497-04	3.705-07	2.858-07	9.527-06	7.869-02	5.469-01	5.918-04
CR 51	0.000	0.000	0.000	1.263-03	2.129+00	2.628+01	1.652-02
MN 54	0.000	0.000	0.000	1.039-04	1.841-01	7.823+00	1.477-03
MN 56	0.000	0.000	0.000	2.714-03	5.098+00	1.020+01	2.891-02
CO 58	0.000	0.000	0.000	1.287-02	2.242+01	5.624+02	1.779-01
FE 59	0.000	0.000	0.000	2.045-04	3.519-01	6.454-00	2.768-03
CO 60	0.000	0.000	0.000	1.302-03	2.317+00	1.155+02	1.864-02
NI 65	0.000	0.000	0.000	1.609-05	3.040-02	6.080-02	1.724-04
ZN 65	4.242-05	1.062-07	8.275-08	1.008-06	9.188-03	3.741-01	7.363-05
ZN 69M	9.228-06	1.704-08	1.043-08	4.329-06	1.641-02	3.284-02	9.260-05
ZN 69	9.443-06	1.814-08	1.118-08	4.603-06	1.644-02	3.288-02	9.260-05
BR 83	1.398-02	9.879-06	2.635-06	1.384-04	1.428+00	2.856-00	8.103-02
KR 83M	0.000	0.000	0.000	0.000	0.000	0.000	0.000
BR 84	3.385-03	7.991-07	6.960-08	1.220-05	5.523-01	1.105-00	3.214-02
BR 85	1.634-05	1.837-09	7.000-11	6.846-08	2.400-02	4.799-02	1.911-03
KR 85M	0.000	0.000	0.000	0.000	0.000	0.000	0.000
KR 85	0.000	0.000	0.000	0.000	0.000	0.000	0.000
SR 89	4.920-02	1.229-04	9.549-05	1.543-03	1.373+01	2.804+02	1.084-00
Y 89M	4.920-06	1.229-08	9.550-09	1.543-07	1.373-03	2.804-02	1.084-04
SR 90	5.293-03	1.327-05	1.034-05	1.164-04	1.067+00	5.455+01	8.585-02
Y 90M	0.000	0.000	0.000	0.000	0.000	0.000	0.000
Y 90	4.960-03	1.249-05	9.766-06	3.124-05	5.759-01	5.339+01	5.529-02
SR 91	1.473-02	2.420-05	1.347-05	6.637-03	2.655+01	5.310+01	1.498+00
Y 91M	8.922-03	1.538-05	8.633-06	4.214-03	1.561+01	3.123+01	8.796-01
Y 91	7.771-03	1.951-05	1.520-05	1.893-04	1.930-00	4.679+01	1.570-01
SR 92	5.886-03	4.567-06	1.329-06	1.281-03	1.181+01	2.362+01	6.695-01
Y 92	6.791-03	9.710-06	4.237-06	2.769-03	1.189+01	2.377+01	6.695-01
ZR 95	3.409-04	8.520-07	3.486-07	1.028-04	1.788-01	4.241+00	1.417-02
NB 95M	3.438-08	9.026-10	6.105-10	4.162-07	1.587-03	8.394-02	1.562-04
NB 95	1.948-07	4.926-09	3.378-09	1.092-04	1.938-01	6.287+00	1.555-02
ZR 97	6.143-06	1.200-08	4.072-09	2.923-05	2.168-02	4.341-02	1.224-03
NB 97M	6.101-06	1.197-08	4.063-09	2.917-05	2.163-02	4.330-02	1.220-03
NB 97	1.715-06	1.092-08	4.195-09	3.105-05	2.170-02	4.345-02	1.224-03
MO 99	0.000	0.000	0.000	4.251-02	4.839+01	1.174+02	3.032+00
TC 99M	4.988-02	6.583-05	3.055-05	4.078-02	1.081+02	2.361+02	6.396+00
TC 99	2.933-08	7.369-11	5.751-11	5.912-10	5.801-06	4.014-04	5.055-07
TC101	2.173-04	3.469-08	1.971-09	1.228-05	1.210+00	2.420+00	7.342-02
RU103	0.000	0.000	0.000	4.843-05	8.296-02	1.375+00	6.507-03
RH103M	0.000	0.000	0.000	4.834-05	8.233-02	1.375+00	6.473-03
RU106	0.000	0.000	0.000	6.759-06	1.198-02	5.260-01	9.617-04
RH106M	0.000	0.000	0.000	0.000	0.000	0.000	0.000
RH106	0.000	0.000	0.000	6.759-06	1.198-02	5.260-01	9.616-04
AG110M	0.000	0.000	0.000	1.558-04	2.757-01	1.130+01	2.210-02

*Abandoned-in-place

TABLE 11.2-5
(Sheet 6 of 8)
INVENTORY SUMMARY (CURIES)

ISOTOPE NAME	CHEMICAL WASTE EVAPORATOR *	CHEMICAL WASTE SAMPLE TANK	CHEMICAL WASTE DEMINERALIZER	WASTE SAMPLE TANK	RWCU FILTER/ DEMINERALIZER	RWCU PHASE SEPARATOR	FUEL POOL F/D
AG110	0.000	0.000	0.000	2.025-06	3.584-03	1.469-01	2.873-04
TE129M	5.318-04	1.326-06	1.029-06	1.973-05	1.729-01	2.534+00	1.351-02
TE129	3.404-04	8.494-07	6.593-07	1.259-05	1.097-01	1.622+00	8.593-03
I 129	2.131-12	5.346-14	4.169-14	2.690-15	6.825-11	3.831-08	7.719-12
I 131	1.152+00	2.824-02	2.162-02	5.914-03	4.532+01	2.004+02	3.256+00
XE131M	0.000	0.000	0.000	0.000	0.000	0.000	0.000
TE132	8.792-02	2.082-04	1.552-04	1.916-02	1.181+02	3.049+02	7.569+00
I 132	1.942-01	1.401-03	4.658-04	2.032-02	1.280+02	3.268+02	8.158+00
I 133	8.454-01	1.729-02	1.147-02	1.879-02	7.418+01	1.490+02	4.197+00
XE133M	0.000	0.000	0.000	0.000	0.000	0.000	0.000
XE133	0.000	0.000	0.000	0.000	0.000	0.000	0.000
I 134	6.162-02	1.991-04	2.418-05	2.899-04	8.129+00	1.626+01	4.672-01
CS134	3.234-03	8.103-06	5.764-06	8.086-04	4.111-01	1.954+01	3.304-02
I 135	3.769-01	5.277-03	2.572-03	7.489-03	3.466+01	6.932+01	1.957+00
XE135M	0.000	0.000	0.000	0.000	0.000	0.000	0.000
XE135	0.000	0.000	0.000	0.000	0.000	0.000	0.000
CS135	0.000	0.000	0.000	0.000	0.000	0.000	0.000
CS136	0.000	0.000	0.000	5.208-04	2.366-01	1.517+00	1.768-02
CS137	0.000	0.000	0.000	1.214-03	6.185-01	3.163+01	4.977-02
BA137M	0.000	0.000	0.000	1.133-03	5.782-01	2.958+01	4.652-02
CS138	0.000	0.000	0.000	8.777-04	2.184+00	4.368+00	1.271-01
BA139	3.799-03	1.727-06	2.989-07	4.894-04	8.689+00	1.738+01	4.957-01
BA140	6.129-02	1.514-04	1.167-04	4.257-03	3.475+01	2.202+02	2.594+00
LA140	6.101-02	1.528-04	1.189-04	1.673-03	2.432+01	2.202+02	2.066+00
BA141	4.157-04	7.348-08	4.672-09	2.456-05	1.917+00	3.835+00	1.144-01
LA141	1.107-03	1.149-06	4.268-07	3.252-04	2.031+00	4.062+00	1.144-01
CE141	6.469-07	9.931-09	6.373-09	2.330-04	4.366-01	6.376+00	3.431-02
BA142	1.650-04	2.422-08	1.252-09	9.180-06	1.130+00	2.260+00	6.997-02
LA142	6.034-04	3.057-07	5.795-08	8.594-05	1.241+00	2.483+00	6.997-02
CE143	0.000	0.000	0.000	5.096-05	4.468-02	9.206-02	2.578-03
PR143	0.000	0.000	0.000	9.683-05	1.586-01	1.080+00	1.198-02
CE144	0.000	0.000	0.000	9.092-05	1.610-01	6.758+00	1.291-02
PR144	0.000	0.000	0.000	9.090-05	1.606-01	6.758+00	1.289-02
ND147	0.000	0.000	0.000	3.376-05	5.263-02	2.975-01	3.889-03
PM147	0.000	0.000	0.000	3.157-08	1.430-04	3.332-02	1.551-05
W 187	0.000	0.000	0.000	3.581-03	2.832+00	5.707+00	1.607-02
NP239	0.000	0.000	0.000	8.539-02	4.711+02	1.080+03	2.882+01
TOTAL =	3.023+00	5.323-02	3.674-02	2.950-01	1.239+03	4.420+03	7.538+01

* Abandoned-in-place

INVENTORY SUMMARY (CURIES)

ISOTOPE NAME	LAUNDRY COLLECTION TANK	LAUNDRY FILTER	LAUNDRY R.O. UNIT *	LAUNDRY SAMPLE TANK	CONCENTRATES WASTE TANK	SPENT RESIN TANK	WASTE SLUDGE TANK
N 13	0.000	0.000	0.000	0.000	1.664-08	3.830-07	0.000
N 16	0.000	0.000	0.000	0.000	2.005-11	2.760-08	0.000
N 17	0.000	0.000	0.000	0.000	1.169-15	2.774-12	0.000
F 18	0.000	0.000	0.000	0.000	5.711-07	4.592-06	0.000
O 19	0.000	0.000	0.000	0.000	4.540-11	1.571-08	0.000
NA 24	0.000	0.000	0.000	0.000	3.074-02	2.335-06	3.637-02
P 32	0.000	0.000	0.000	0.000	1.145-01	1.332-05	5.612-03
CR 51	0.000	0.000	0.000	0.000	1.071-01	8.694-06	3.524-00
MN 54	5.600-06	4.105-04	5.544-08	5.600-09	1.314-02	4.474-06	6.909-01
MN 56	0.000	0.000	0.000	0.000	2.995-03	5.304-08	3.598-01
CO 58	3.171-05	1.712-03	3.136-07	3.159-08	1.416+00	2.222-04	5.241+01
FE 59	0.000	0.000	0.000	0.000	2.035-02	2.282-06	6.911-01
CO 60	4.962-05	3.960-03	4.913-07	4.967-08	1.707-01	7.756-05	1.072+01
NI 65	0.000	0.000	0.000	0.000	1.774-05	3.130-10	2.137-03
ZN 65	0.000	0.000	0.000	0.000	6.596-02	4.146-05	1.346-03
ZN 69M	0.000	0.000	0.000	0.000	3.849-04	2.857-08	4.908-04
ZN 69	0.000	0.000	0.000	0.000	4.121-04	3.067-08	4.990-04
BR 83	0.000	0.000	0.000	0.000	8.784-02	2.773-06	3.262-01
KR 83M	0.000	0.000	0.000	0.000	0.000	0.000	0.000
BR 84	0.000	0.000	0.000	0.000	4.691-03	7.026-08	1.287-01
BR 85	0.000	0.000	0.000	0.000	2.166-06	7.168-11	7.645-03
KR 85M	0.000	0.000	0.000	0.000	0.000	0.000	0.000
KR 85	0.000	0.000	0.000	0.000	0.000	0.000	0.000
SR 89	0.000	0.000	0.000	0.000	6.360+01	1.573-02	5.243-00
Y 89M	0.000	0.000	0.000	0.000	6.360-03	1.573-06	5.243-04
SR 90	0.000	0.000	0.000	0.000	8.659+00	8.188-03	5.079-01
Y 90M	0.000	0.000	0.000	0.000	0.000	0.000	0.000
Y 90	0.000	0.000	0.000	0.000	8.600+00	8.184-03	3.771-01
SR 91	0.000	0.000	0.000	0.000	4.334-01	2.888-05	6.139+00
Y 91M	0.000	0.000	0.000	0.000	2.770-01	1.853-05	3.612+00
Y 91	0.000	0.000	0.000	0.000	1.045+01	2.828-03	7.739-01
SR 92	0.000	0.000	0.000	0.000	4.867-02	1.653-06	2.697+00
Y 92	0.000	0.000	0.000	0.000	1.217-01	5.938-06	2.722+00
ZR 95	7.650-06	3.987-04	7.566-08	7.617-09	4.668-01	6.868-05	4.546-01
NB 95M	1.333-08	7.737-06	1.571-10	2.558-11	8.200-03	1.376-06	7.617-03
NB 95	1.095-05	6.512-04	1.084-07	1.093-08	1.557-01	6.822-05	5.828-01
ZR 97	0.000	0.000	0.000	0.000	3.616-04	1.451-08	1.060-02
NB 97M	0.000	0.000	0.000	0.000	3.608-04	1.449-08	1.058-02
NB 97	0.000	0.000	0.000	0.000	3.737-04	1.543-08	1.100-02
MO 99	0.000	0.000	0.000	0.000	5.573-01	2.612-05	3.902+01
TC 99M	0.000	0.000	0.000	0.000	1.401+00	7.449-05	4.936+01
TC 99	0.000	0.000	0.000	0.000	4.842-05	4.623-08	1.654-05
TC101	0.000	0.000	0.000	0.000	1.602-04	2.410-09	2.939-01
RU103	7.623-07	3.109-05	7.535-09	7.565-10	4.648-03	4.754-07	1.788-01
RH103M	7.192-07	3.112-05	7.384-09	7.548-10	4.652-03	4.758-07	1.789-01
RU106	1.317-05	9.824-04	1.304-07	1.317-08	9.073-04	3.102-07	5.025-02
RH106M	0.000	0.000	0.000	0.000	0.000	0.000	0.000
RH106	1.316-05	9.824-04	1.304-07	1.317-08	9.073-04	3.102-07	5.025-02
AG110M	2.414-06	1.735-04	2.389-08	2.413-09	1.951-02	6.252-06	1.071+00
AG110	3.137-08	2.255-06	3.106-10	3.137-11	2.537-04	8.128-08	1.393-02
TE129M	0.000	0.000	0.000	0.000	6.116-01	1.120-04	6.380-02

* Abandoned-in-place

TABLE 11.2-5
(Sheet 8 of 8)
INVENTORY SUMMARY (CURIES)

ISOTOPE NAME	LAUNDRY COLLECTION TANK	LAUNDRY FILTER	LAUNDRY R.O. UNIT *	LAUNDRY SAMPLE TANK	CONCENTRATES WASTE TANK	SPENT RESIN TANK	WASTE SLUDGE TANK
TE129	0.000	0.000	0.000	0.000	3.920-01	7.176-05	4.063-02
I 129	0.000	0.000	0.000	0.000	4.910-09	3.707-11	1.325-10
I 131	3.158-06	1.071-06	3.010-07	2.940-08	5.556-02	5.594-01	1.448+01
XE131M	0.000	0.000	0.000	0.000	0.000	0.000	0.000
TE132	0.000	0.000	0.000	0.000	1.929-01	1.807-03	3.259+01
I 132	0.000	0.000	0.000	0.000	2.049-01	2.190-03	3.503+01
I 133	0.000	0.000	0.000	0.000	4.636-01	3.792-02	1.751+01
XE133M	0.000	0.000	0.000	0.000	0.000	0.000	0.000
XE133	0.000	0.000	0.000	0.000	0.000	0.000	0.000
I 134	0.000	0.000	0.000	0.000	1.396-01	2.440-05	1.873+00
CS134	7.157-05	1.802-04	6.879-06	6.953-07	5.215+00	3.910-03	2.351-01
I 135	0.000	0.000	0.000	0.000	6.602+00	3.816-03	7.966+00
XE135M	0.000	0.000	0.000	0.000	0.000	0.000	0.000
XE135	0.000	0.000	0.000	0.000	0.000	0.000	0.000
CS135	0.000	0.000	0.000	0.000	0.000	0.000	0.000
CS136	0.000	0.000	0.000	0.000	1.496-02	1.526-06	8.762-02
CS137	1.318-04	3.450-04	1.267-05	1.281-06	8.715-02	7.074-05	3.708-01
BA137M	1.230-04	3.226-04	1.185-05	1.198-06	8.148-02	6.614-05	3.466-01
CS138	0.000	0.000	0.000	0.000	3.224-04	4.508-09	5.097-01
BA139	0.000	0.000	0.000	0.000	1.603-02	3.520-07	1.990+00
BA140	0.000	0.000	0.000	0.000	4.360+01	4.881-03	1.175+01
LA140	0.000	0.000	0.000	0.000	4.905+01	5.516-03	9.627+00
BA141	0.000	0.000	0.000	0.000	3.897-04	5.651-09	4.580-01
LA141	0.000	0.000	0.000	0.000	1.352-02	5.809-07	4.627-01
CE141	0.000	0.000	0.000	0.000	3.078-02	3.058-06	8.409-01
BA142	0.000	0.000	0.000	0.000	9.662-05	1.552-09	2.800-01
LA142	0.000	0.000	0.000	0.000	2.919-03	6.839-08	2.812-01
CE143	0.000	0.000	0.000	0.000	3.141-04	1.385-08	2.742-02
PR143	0.000	0.000	0.000	0.000	5.873-03	3.388-07	2.504-01
CE144	2.743-05	1.997-03	2.715-07	2.743-08	1.154-02	3.822-06	6.415-01
PR144	2.696-05	1.997-03	2.709-07	2.742-08	1.154-02	3.822-06	6.418-01
ND147	0.000	0.000	0.000	0.000	1.678-03	9.094-08	7.603-02
PM147	0.000	0.000	0.000	0.000	3.600-05	2.286-08	2.701-03
W 187	0.000	0.000	0.000	0.000	1.567-02	6.610-07	9.784-01
NP239	0.000	0.000	0.000	0.000	4.808+00	4.386-04	1.230-02
TOTAL =	5.197-04	1.419-02	3.359-05	3.395-06	8.493+02	6.558-01	4.447+02

* Abandoned-in-place

TABLE 11.2-6
(Sheet 1 of 3)

MAXIMUM AND EXPECTED INPUTS TO RADWASTE SYSTEMS

I. Equipment Drain System (all figures are per unit)

Source	Normal GPD/Days Per Year	Maximum GPD/Days Per Year
Reactor equipment drains	2000/365 (1)*	3760/365 (2)
Drywell equipment drains	5800/365 (1)	28000/17 (2)
Drywell floor drains	2900/365 (1)	28000/17 (2)
Turbine building equipment drains	5700/365 (1)	5730/365 (2)
Reactor water cleanup F/D backwash	1200/365 (1)	4200/100 (2)
Condensate demineralizer backwash	4000/365 (3)	80000/2 (4)
Radwaste demineralizer backwash	2850/365 (3)	5700/365 (5)

* Numerically referenced notes are found on the last page of this table.

TABLE 11.2-6
(Sheet 2 of 3)

II. Floor Drain System (all figures are per unit)

Source	Normal GPD/Days Per Year	Maximum GPD/Days Per Year
Reactor building floor drain sump	2000/365 (1)	2000/365 (2)
Radwaste floor drains	1000/365 (1)	1000/365 (2)
Turbine building floor drains	2000/365 (1)	2000/365 (2)

III. Chemical Waste System (all figures are per unit)

Source	Normal GPD/Days Per Year	Maximum GPD/Days Per Year
Condensate demineralizer regenerants (no longer created)	1240/365 (3)	24800/2 (4)
Radwaste demineralizer regenerants (no longer created)	1770/365 (3)	3540/365 (5)

TABLE 11.2-6
(Sheet 3 of 3)

IV. Laundry waste

Source	Normal GPD/Days Per Year	Maximum GPD/Days Per Year
________________	800/365 (6)	4000/30 (6)

Notes:

(1) Draft ALARA Regulatory Guides

(2) GESSAR

(3) Subsection 11.2.3.2

(4) Subsection 11.2.1.7.4

(5) Estimate. In comparison to the design base flow of the condensate demineralizer regenerants (80,000 gpd), this assumption does not affect equipment sizing.

(6) Estimate based on operating experience.

TABLE 11.2-7

DESIGN-BASIS PROCESS DECONTAMINATION FACTORS

EQUIPMENT	DESIGN DECONTAMINATION FACTOR
Waste mixed-bed demineralizers	Cation: 10^2 (except Cs, Rb, Mo, Tc)*
	Anion: 10^2
	Cs, Rb: 10
	Insolubles:** 2
Filters	Insolubles: 10
	Solubles: 1.03
Evaporators (abandoned-in-place)	Halogens: 10^3
	Other nuclides: 10^4
Reverse osmosis (abandoned-in-place)	30

* A plateout removal factor of 10 for Mo, Tc taken in each stream.

** Table 4-2, "Releases from BWR Radwaste Management Systems" R. A. Head, C. W. Miller, J. E. Oesterle, NEDO-10951, July 1973.

TABLE 11.2-8

PRINCIPAL ASSUMPTIONS USED IN "MAXIMUM INDIVIDUAL" RADIOLOGICAL DOSE ASSESSMENT FOR RADIONUCLIDES RELEASED IN LIQUID EFFLUENTS

I. Annual releases	Table 11.2-1	
II. Method, radiological data libraries used:	Reference 1 (Section 11.2)	
III. Usage factors Reference 2 (Section 11.2)	Drinking water Fish consumption Shoreline activity Swimming Boating	440 l/yr 1.2 kg/yr 15 hr/yr 6 hr/yr 29 hr/yr
IV. Dilution flow (cooling lake blowdown) prior to discharge to unrestricted area	30,000 gpm (67 cfs)	
V. Hydrological Dispersion Factors Used		

Pathway	Principal Assumption/Location	Dilution Factor
Fish	Fish live in area discharge	1:1
Shoreline Activities	Near vicinity of discharge canal (shoreline width factor = 0.2)	1:5
Swimming and Boating	Near vicinity of discharge canal	1:5
Drinking Water	Peoria, Illinois (first municipal water supply)	
	Average river flow (10,750 cfs)	1:160
	7-day, 10 year low flow (3228 cfs) (for reference only, not used in analysis)	
	(Illinois Nitrogen Corporation located 3300 feet downstream of the LSCS discharge but on the opposite bank of the river has been identified as using Illinois River for drinking water for its employees. Calculations indicate, though, that the LSCS discharge will not reach the Illinois Nitrogen Corporation water inlet under high or low river conditions.)	

11.3 GASEOUS WASTE MANAGEMENT SYSTEMS

11.3.1 Design Bases

11.3.1.1 Design Objective

The gaseous waste management systems are designed to process and control the release of gaseous radioactive wastes to the site environs so that the total radiation exposure of persons in offsite areas is as low as reasonably achievable and does not exceed applicable guidelines. This is to be accomplished while maintaining the occupational exposure as low as reasonably achievable and without limiting plant operation or availability.

11.3.1.2 Design Criteria

The gaseous waste management systems are designed to limit offsite doses from routine station releases to significantly less than the limits specified in 10 CFR 20, and to operate within the dose objectives established in 10 CFR 50, Appendix I.

A noble gas input equivalent to an annual average off-gas rate (based on a 7-minute decay) of 230,000 muCi/sec of the "1971 mixture" as shown in Table 11.3-1 was used as the design basis. A conservative value of 80 scfm for off-gas flow has been used as a design basis.

The design approach for the off-gas treatment includes delay until the required fraction of the radionuclides has decayed. The daughter products are then retained by the charcoal and HEPA filters.

A list of the major equipment items in the off-gas system and the design codes is provided in Table 11.3-2. Equipment and piping were designed and constructed in accordance with the applicable code requirements as listed. Welding and material requirements are also listed.

Conservative analyses similar to that presented in Reference 1 demonstrate that equipment failure cannot result in doses exceeding acceptable guidelines; thus, neither the off-gas system nor the buildings housing the equipment were designed to meet Seismic Category I requirements. The off-gas structure walls below grade are part of the total structural shear wall system and were designed to withstand the effects of earthquakes.

The failure of the off-gas system is analyzed in Subsection 15.7.1. The related failure of the steam jet air ejector lines and the gland seal off-gas lines are analyzed in Subsections 15.7.1.2 and 15.7.1.3.

11.3.2 System Descriptions

11.3.2.1 Off-Gas (RECHAR) System

Noncondensible radioactive off-gas is continuously removed from the main condenser by the air ejector during plant operation. This is the major source and is larger than all other sources combined. The air ejector off-gas will normally contain activation gases, principally N-16, O-19, and N-13. The N-16 and O-19 have short half-lives and are readily decayed. The 10-minute N-13 is present in small amounts that are further reduced by decay. The air ejector off-gas will also contain the radioactive noble gas 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 (usually extremely small) and the number and size of fuel cladding leaks. An off-gas (RECHAR) system has been provided to treat this source; the system utilizes catalytic recombination and charcoal adsorption as discussed below.

11.3.2.1.1 Process Description

The condenser off-gas (RECHAR) system, shown in Figure 11.3-1, will use a catalytic recombiner to recombine radiolytically dissociated hydrogen and oxygen from the air ejector system. After cooling (to approximately 130° F) to strip the condensables and reduce the volume, the remaining noncondensables (principally krypton, xenons, and air) are delayed in the holdup pipe. The gas is cooled to 45° F and reheated to 74° F for humidity control before reaching the adsorption bed. The charcoal adsorption bed, operating in a constant-temperature vault, selectively adsorbs and delays the xenons and kryptons from the bulk carrier gas (principally air). This delay on the charcoal permits the Xe and Kr to decay in place. This system results in a reduction of the off-gas activity (curies) released by a factor of approximately 9 relative to a 7-minute holdup system and based on a "1971 mixture." Table 11.3-1 shows the estimated annual release rates from the charcoal adsorbers of various isotopes of krypton and xenon compared to a system releasing 230,000 μCi/sec after a 7-minute delay.

The adsorption of noble gases on charcoal depends primarily on gas flow rate, mass of charcoal, and gas-unique coefficients known as the dynamic adsorption coefficients. The parametric interrelationships and governing equations are well proved from 7 years of operation of a similar unit at KRB in Germany.

11.3.2.1.1.1 Process Flow Diagram

Figure 11.3-1 is the process flow diagram for the system. The process data for startup and normal operating conditions were submitted as proprietary data under separate cover as Table 11.3-3 of the FSAR. The information supporting the process data is presented in References 2 and 3. The vent is the single release point for this system and is located on the auxiliary building. The vent is indicated on Figure 11.3-1.

11.3.2.1.1.2 Noble Gas Radionuclide Source Term and Decay

The design basis isotopic source terms for the annual average activity input of the main condenser off-gas treatment system are given in Table 11.3-1 at t = 7 minutes. The system is mechanically capable of processing three times the source terms of Table 11.3-1 without affecting delay time of the noble gases. Also listed is the isotopic distribution at t = 0. With an air inleakage of 80 scfm, this treatment system results in a delay of 2.9 hours for krypton and 2.2 days for xenon. Table 11.3-1 lists isotopic activities at the discharge of the system, and the decontamination factor for each noble gas isotope can be determined.

11.3.2.1.1.3 Process and Instrumentation Diagram (P&ID)

The P&ID is submitted as proprietary data under separate cover. The main process routing is indicated by a heavy line.

11.3.2.1.1.4 Recombiner

The basis for sizing the recombiner is to maintain the hydrogen concentration below 4% (including steam) at the inlet and below 4% at the outlet on a dry basis. The exit hydrogen concentration is normally well below the 4% maximum allowed. The hydrogen generation rate of the reactor is based on data from nine BWR's. The hydrogen generation rate is given in the data referenced in Subsection 11.3.2.1.1.1. Recombiner performance is continuously monitored and recorded by catalyst bed thermocouples that monitor the bed temperature profile and by a hydrogen analyzer that measures the hydrogen concentration of the effluent.

11.3.2.1.1.5 Charcoal Adsorbers

11.3.2.1.1.5.1 Charcoal and Charcoal Vault Temperature

The charcoal adsorbers operate at room temperature essentially so that on system shutdown, radioactive gases in the adsorbers are subject to the same holdup time as during normal operation, even in the presence of continued airflow. The charcoal adsorbers are designed to limit the temperature of the charcoal (normally to about 77° F) to well below the charcoal ignition temperature, thus precluding overheating or fire and consequent escape of radioactive materials. The adsorbers are located in a shielded room and are normally maintained at a constant temperature by one of two air conditioning systems that remove the decay heat generated in the adsorbers. A full capacity air conditioning unit is provided as a spare and can be activated from the main control room following an alarm indicating failure of the normally operating unit. In the unlikely event that both air conditioning units are unable to function, the radioactive emissions from the off-gas system would increase slightly; however, the releases would still be well below acceptable limits.

A radiation monitor is provided in the charcoal adsorber vault to monitor the radiation levels in the room. High radiation in the vault, indicating off-gas system leakage or component failure, is alarmed in the control room for operator action, as required.

11.3.2.1.1.5.2 Gas Channeling in the Charcoal Adsorber

Channeling in the charcoal adsorbers is prevented by supplying an effective flow distributor on the inlet, having long columns, and having a high bed-to-particle diameter ratio (approximately 500). Underhill has stated that channeling or wall effects may reduce efficiency of the holdup bed if this ratio is not greater than 12 (Reference 2). During transfer of the charcoal into the charcoal adsorber vessels, radial sizing of the charcoal is minimized by pouring the charcoal (by gravity or pneumatically) over a cone or other instrument to spread the granules over the surface.

11.3.2.1.1.5.3 Charcoal Bypass Mode

A valve is provided to bypass the charcoal adsorbers. The main purpose of this bypass is to protect the charcoal during preoperation and startup testing when gas activity is zero or very low. It may be desirable to use the bypass for short periods during startup or normal operations. This bypass mode is not used for normal operations unless some unforeseen system malfunction necessitates shutting down the power plant or operating in the bypass mode while remaining within the technical specification limits. The activity release is controlled by a process monitor upstream of the vent isolation valve that causes the bypass valve to close on a high radiation alarm. This alarm can be defeated only by a keylock switch. The alarm setting is covered in Subsection 7.7.14.1. In addition, there is a high-high-high alarm on the same monitor that causes the off-gas system to be isolated from the station vent stack if established release limits are exceeded.

11.3.2.1.1.5.4 Charcoal Performance

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

Experience with boiling water reactors has shown that the calibration correction factor of the off-gas and vent effluent monitors changes with isotopic content. Isotopic content can change depending on the presence or absence of fuel cladding leaks in the reactor, 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 sample points are located upstream and downstream of the first charcoal bed and downstream of the last charcoal bed and can be used for periodic sampling if the monitoring equipment indicates degradation of system delay performance.

11.3.2.1.1.6 HEPA Prefilter

These particulate filters are tested at the time of filter installation or replacement to determine whether an installed filter meets the minimum initial in-place efficiency of 99.95%
retention.

11.3.2.1.1.7 Desiccant Gas

Desiccant gas dryer performance is not monitored.

11.3.2.1.1.8 HEPA Afterfilter

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

11.3.2.1.1.9 Instrumentation and Control

The off-gas system is normally monitored by flow, temperature, pressure, and by hydrogen analyzers to ensure correct operation and control and to ensure that hydrogen concentration is maintained below the flammable limit. Table 11.3-3 lists the process parameters that are instrumented to alarm in the main control room. It also indicates whether the parameters are recorded or just indicated.

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

A radiation monitor is also provided at the outlet of the charcoal adsorbers to continuously monitor the release rate from the adsorber beds. This radiation monitor is used to isolate the off-gas system on high radioactivity to prevent treated gas of unacceptably high activity from entering the station vent stack. Therefore, the activity of the gas entering and leaving the off-gas system is continuously monitored, and system performance is known to the operator at all times. Provision is made for sampling and periodic analysis of the influent and effluent gases for purposes of determining their compositions. This information is used in calibrating the monitors and in relating the release to calculated environs dose. Off-gas system process radiation instrumentation is further discussed in Section 11.5.

The catalytic recombiner vessel temperatures are monitored by thermocouples and then recorded. High or low temperature is annunciated in the main control room. The standby recombiner is temperature-controlled, maintained, monitored, and recorded. Any low temperature is annunciated in the main control room. Inlet process gas is monitored for temperature, and if low temperatures are obtained, they are annunciated in the main control room.

The off-gas condenser condensate level is maintained at a given level within the condenser shell. A level control system is used to provide drainage of condensate from the condenser shell. High and low level and high gas discharge temperature are annunciated locally (see Table 11.3-3).

Differential pressure measurements are made across the prefilters, the charcoal vessel train, and the afterfilters. High differential pressure is annunciated in the control room.

Hydrogen analyzers are used to measure the hydrogen content of the off-gas process stream. The hydrogen concentration percentage output from each analyzer is indicated and recorded in the main control room along with alarm annunciation for high hydrogen concentration percentage in the off-gas process stream.

The hydrogen/oxygen analyzer system continuously withdraws a sample of the process off-gas, analyzes the hydrogen content, and returns the sample gas to the main condenser. During normal plant operation, the main turbine condenser vacuum provides the pumping force to move the sample gas from the off-gas process line and through the hydrogen analyzer system. A hydrogen level of 1.5% will alarm and annunciate in the main control room.

Off-gas system flow measurements are made just downstream of the charcoal adsorbers. Startup range high flow is recorded in the main control room and annunciated locally. A startup range high flow of 265 scfm will alarm and annunciate in the main control room. A normal range high flow of 80 scfm or low flow of 6 scfm will alarm and annunciate in the main control room. The flow recorder is provided to keep a record of all discharge volumes. The flow measurements and recording accuracies are within 5% of indication for the flows measured.

11.3.2.1.1.10 Explosion Resistance

11.3.2.1.1.10.1 Pressure Boundary

The pressure boundary of the system is designed to be explosion-resistant. The pressure vessels are designed to withstand 350 psig static pressure, and the piping and valving are designed to resist dynamic pressures encountered in long runs of piping at the design temperature. This analysis is covered in a proprietary report submitted to the NRC (Reference 4).

An equivalent detonation-containing static pressure is then derived for which the component can be rated, based upon the wall thickness calculated by the above procedure.

11.3.2.1.1.11 Leakage of Radioactive Gases

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

11.3.2.1.1.12 Hydrogen Concentration

Hydrogen concentration of gases from the air ejector is kept below the flammable limit by maintaining adequate process steam flow for dilution at all times. This steam flow rate is monitored and alarmed. Hydrogen analyzers are used to ensure that hydrogen concentration is maintained below the flammable limit.

11.3.2.1.1.13 Drain Seals

Drains from the off-gas system which do not discharge to the main condenser utilize liquid seals to prevent the escape of radioactive gases from the system. Loop seals are provided where possible. However, most of the off-gas drains are piped to the off-gas pressure drain tank. This tank provides a pressure balanced liquid seal to prevent gas leakage. Drainage from the off-gas system is pumped from the off-gas pressure drain tank to the liquid radwaste system. Drain pumps are automatically started by a high level signal from the pressure drain tank.

11.3.2.1.1.14 Process Design Parameters

The krypton and xenon holdup times are closely approximated by the following equation:

$$T = \frac{K_D M}{V} \tag{11.3-1}$$

where:

T = holdup time of a given gas,

K_D = dynamic adsorption coefficient for the given gas,

M = weight of charcoal, and

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

Dynamic absorption coefficient values for xenon and krypton were reported by Browning (Reference 5). General Electric (GE) has performed pilot plant tests at their Vallecitos Laboratory, and the results were reported at the 12th AEC Air Cleaning Conference (Reference 6). Further GE data on a similar system operating at ambient temperatures have been reported in the GE proprietary topical report, "Experimental and Operational Confirmation of Off-Gas System Design Parameters" (Reference 3); nonproprietary portions of this information are reported in Reference 2.

Carrier gas is the air inleakage from the main condenser after the radiolytic hydrogen and oxygen are removed by the recombiner. The air inleakage design basis is conservatively sized at 80 scfm total. The Sixth Edition of the Heat Exchange Institute Standards for Steam Surface Condensers (Reference 7, Par. 5-16 (c)(2)) indicates that with certain conditions of stable operation and suitable construction, noncondensables (not including radiological decomposition products) should not exceed 6 scfm for large condensers. Dresden-2, Fukushima-1, Tsuruga, and KRB have all operated at 6 scfm or below after initial startup (Table 6 of Reference 8). Dilution air is added to the system to maintain off-gas flow, if condenser inleakage is too low.

The system is mechanically capable of processing 3 times the source term quantities of Table 11.3-1 without affecting delay time of the noble gases. With an air inleakage of 80 scfm, this treatment system results in a delay of 2.9 hours for krypton and 2.2 days for xenon.

Hydrogen concentration of gases from the air ejector is kept below the flammable limit by maintaining adequate process steam flow for dilution at all times. This steam flow is monitored and alarmed in the main control room. The basis for sizing the catalytic recombiner is to maintain the hydrogen concentration below 4% (including steam) at the inlet using the dilution flow, and below 4% at the outlet on a dry basis. The exit hydrogen concentration is normally well below the 1% maximum allowed.

Figure 11.3-1 is the flow diagram for the system. The process data for startup and normal operating conditions and the piping and instrument diagram are submitted as proprietary information under separate cover.

11.3.2.1.1.15 System Design

The system design was based on a noble gas activity release to the environs of about 25,000 μCi/sec from the system, based upon 80 scfm air inleakage and an input of

230,000 μCi of 7-minute- old "1971 Mixture." The isotopic composition is given in Table 11.3-1 in units of μCi/sec and Ci/yr.

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

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

11.3.2.1.1.16 Design Basis Inventories of Radionuclides

The isotopic inventory of each piece of equipment is given in Chapter 15.0.

11.3.2.1.1.17 Single Failure and Operator Errors

Design provisions are incorporated to preclude the uncontrolled release of radioactive materials in the gaseous effluents. A single-failure analysis is included in Table 11.3-4.

11.3.2.1.1.18 Typical Operating Sequences

11.3.2.1.1.18.1 Prestartup Preparations

Prior to starting the main steam jet air ejectors (SJAE), the charcoal vault is temperature-controlled to near 77° F, the glycol cooler is chilled to near 35° F, glycol is circulated through the cooler condenser, the off-gas condenser cooling water is valved in, and the recombiner heaters are turned on.

11.3.2.1.1.18.2 Startup

As the reactor is pressurized, preheated steam is supplied and air is bled through the preheater and recombiner. The recombiner is preheated to at least 225° F with the air bleed and/or by admitting steam to the final SJAE. With the recombiners preheated and the charcoal adsorbers valved in, the SJAE string is started. As the condenser is pumped down and the reactor power increases, the recombiner inlet stream is diluted to less than 4% H_2 by volume by a fixed steam supply, and the off-gas condenser outlet is maintained at less than 1% hydrogen by volume.

11.3.2.1.1.18.3 Normal Operation

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

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

11.3.2.1.1.19 Performance Tests

11.3.2.1.1.19.1 Main Condenser Steam Jet Air Ejector Off-Gas (RECHAR) System

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

11.3.2.1.1.19.2 Recombiner

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

11.3.2.1.1.20 Operator Exposure Criteria and Controls

This system is normally operated from the main control room. Equipment and process valves containing radioactive fluids are placed in shielded cells maintained at a pressure negative to normally occupied areas.

11.3.2.1.1.21 Equipment Malfunction

A malfunction analysis, indicating consequences and design precautions taken to accommodate failure of various components of the system, is given in Table 11.3-4.

11.3.2.1.1.22 Equivalent Service Experience

A system with similar equipment is in service at the KRB plant in Germany. Its performance is reviewed in Supplement 1 to Reference 8. Similar systems (ambient temperature charcoal) are in service at Dresden-2 and -3, Quad Cities-1 and -2, Pilgrim, Hatch-1, and Duane Arnold. Non-proprietary portions of Reference 8 are reported in Reference 3.

11.3.2.2 Other Radioactive Gas Sources

There are three general areas that contain gaseous radioactive sources: the primary containment, the secondary containment, and the radwaste building. Ventilation systems for these areas are described in Section 9.4.

11.3.2.2.1 Primary Containment

The primary containment is divided into two sections which are designated as the drywell and wetwell. These are separated by the drywell floor which serves as a pressure barrier between the drywell and wetwell (suppression chamber). The drywell atmosphere is exposed to a small neutron flux which penetrates the reactor vessel. This results in the production of a small amount of N-16. Since N-16 has a very short half-life (7.13 seconds), it will decay before anyone can be exposed to it and is, therefore, not a contributing source.

Radioactive halogens and noble gases can be introduced into the drywell atmosphere from two sources. One source is the steam leakage that occurs from valves and equipment such as main recirculation pumps. This leakage is collected by means of leakoff lines which are directed to the drywell equipment drain sump. The other source of activity results from a pressure transient causing the main steam relief valves to open, with the resulting flow of steam into the suppression pool. When this occurs, there is a small pressure buildup in the wetwell atmosphere. If the pressure differential across the drywell floor is greater than 0.5 psi, then the vacuum breaker valves relieve this pressure into the drywell atmosphere. Any elements that are present in the wetwell atmosphere are thus introduced into the drywell. The drywell atmosphere is purged or transported to the outside atmosphere via the station vent stack. When drywell access is required for a typical entry into the containment, this atmosphere is purged through the drywell purge and vent system. This purge and vent system is described in Section 9.4.

11.3.3 Radioactive Releases

A single station vent stack has been built at LSCS to ensure that radioactive gaseous effluents and most plant ventilation air is exhausted and dispersed in a manner that minimizes effluent concentrations. The effluent streams from typical radioactive sources are all treated prior to entry into the station vent stack. For instance, a multiple redundant off-gas system collects, treats, and filters the normal gaseous products from the main steam condenser. A standby gas treatment system with dual filter trains is provided for the treatment of airborne contaminants in the secondary containment as a result of accident conditions. It is interconnected to accept effluent from either reactor building and exhaust it through the station vent stack. In parallel with this dual SGTS is a pair of primary containment purge and vent filter packages to clean up the drywell atmosphere prior to workers' entry for maintenance and service. It also purges the gaseous volume above the suppression pools. Similarly, these dual filter trains are interconnected to serve either reactor unit and to exhaust through the station vent stack.

Each reactor building ventilation system serves the refueling floor, the interconnected main steam tunnel, and the reactor building. Each ventilation system has a capacity of 110,800 scfm which undergoes industrial filtration prior to exhausting through the station vent stack. Multiple supply fans and exhaust fans are used with interlocking dampers and isolation controls to isolate the reactor building as needed.

The turbine building ventilation system exhaust fans serves each turbine unit with a capacity of 210,000 scfm. This effluent also exhausts through the same station vent stack.

The radwaste building ventilation system is HEPA filtered and monitored for radiation prior to its discharge into the station vent stack. Additionally, the station vent stack employs an isokinetic probe to assure precise effluent monitoring. When stack threshold values are reached, an alarm occurs which alerts operating personnel that if the increase continues, the possibility of exceeding 10 CFR 20 limits exists.

Minor ventilation systems from the machine shop and the laundry areas are also ducted to the station vent stack.

The station vent stack has a physical elevation of 370 feet above plant grade. It is mounted on top of the auxiliary building. It also extends approximately 185 feet above the reactor building structure and was sized to eliminate building wake effects on effluent dispersal into the atmosphere. Exhaust velocities for normal operating conditions preclude deleterious wind interactions with the emergent plume for all except the most severe postulated winds. The standby gas treatment stack within the station vent stack assures plume rise above physical stack height for the postaccident effluent treatment situation. Stack diameters are shown on Figure 11.3-2 along with other gaseous effluent data and treatment equipment.

Releases from the station vent stack are considered to be elevated releases at all times for annual average estimates of downstream atmospheric dispersion because the stack exit velocity exceeds the horizontal wind speed by a factor of five or more on the annual average basis. A summary of typical LSCS atmospheric dispersion factors and deposition factors is shown in Table 11.3-5.

11.3.3.1 Expected Releases

Expected releases of radioactive fission products including noble gases, halogens, and other specific isotopes were computed via assumptions outlined for the Appendix I tabulation of 10 CFR 50. Dose assessments were all within Appendix I guidelines for gaseous effluents as well as for all other effluents. These calculated values and other effluent concentrations compared to Table II, Volume 1 of 10 CFR 20 provide analytical validations that radioactive releases are all within applicable guidelines even though extremely conservative assumptions are arbitrarily assigned for transport, release, dispersion, and intake parameters (Reference 9).

A practical representation of the expected releases of radioactive gaseous effluents exists as a summary report of all available release data from operating BWR plants of similar design but earlier vintage. The most recent compilation of such release data covering the years 1972 to 1975 included more than 16 plant-years of commercial operation. These measurements were accumulated by utility, federal, commercial, and industrial groups including the Atomic Energy Commission. Annual average release values were determined from statistical evaluations of thousands of individual measurements from sixteen domestic BWR stations as reported in Reference 10.

The gaseous effluent treatment equipment in these plants is conceptually similar to that of LSCS; however, the LSCS systems include unique improvements in effluent conditioning and in their single point of release via the station vent stack.

11.3.3.2 Reported Releases

The annual release of inorganic iodine-131 is 0.093 curies per plant per year. This inorganic I-131 includes particulate matter, elemental I_2 , and hypoiodous acid.

The majority of the I-131 is released from the reactor building/containment structure with the principle form defined as methyl iodide (0.37 curies per plant per year) released during operation of the mechanical vacuum pump while refueling/maintenance outages exist. Approximately 80% of the total I-131 annual release occurs under this outage condition. See Table 11.3-6 for experience based annual estimated airborne releases of iodine-131. It is expected that LSCS gaseous effluence releases will be of this same magnitude.

The annual release of noble radiogases is expected to be 6500 curies per plant. See Table 11.3-7 for the identity and quantity of each radioisotope expected to be released from an LSCS type plant per year. Noble radiogases contribute negligible offsite doses in comparison to normal background levels. For LSCS, it is expected that noble radiogas release will be of this same magnitude.

11.3.4 References

1. C. W. Miller et al., "A General Justification for Classification of Effluent Treatment System Equipment as Group D," NEDO-10734, February 1973.

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

3. C. W. Miller, "Experimental and Operational Confirmation of Off-Gas System Design Parameters," NEDO-20116, October, 1973.

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

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

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

7. Standards for Steam Surface Condensers, Sixth Edition, Heat Exchange Institute, New York, N.Y., 1970.

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

9. "Information Relevant to Keeping Levels of Radioactivity in Effluents to Unrestricted Areas As Low As Reasonably Achievable," (filed in accordance with Section V.B. of 10 CFR 50, Appendix I), LaSalle County Station, Units 1 and 2, June 4, 1976.

10. T. R. Marrero, "Airborne Released from BWR's for Environmental Impact Evaluations," NEDO-21159, March 1976.

TABLE 11.3-1

ESTIMATED RELEASE RATES FROM OFF-GAS SYSTEM CHARCOAL BEDS (PER UNIT)

Isotope	Release Rate from Reactor[1] (μCi/sec)	Discharge Rate from Holdup Line[5] (μCi/sec)	Discharge Rate from Charcoal Adsorbers[2] (μCi/sec)	Annual Discharge from Charcoal Adsorbers[3] (Ci/yr)
Kr-83m	3.4E+03	3.3E+03	1.1E+03	3.0E+04
Kr-85m	6.1E+03	6.0E+03	3.8E+03	1.0E+05
Kr-85	10 - 20	10 - 20	10 - 20	270 - 540
Kr-87	2.0E+04	1.9E+04	3.8E+03	1.0E+05
Kr-88	2.0E+04	1.9E+04	9.5E+03	2.5E+05
Kr-89	1.3E+05	2.9E+04	-	-
Kr-90	2.8E+05	3.7E+01	-	-
Kr-91	3.3E+05	-	-	-
Kr-92	3.3E+05	-	-	-
Kr-93	9.3E+04	-	-	-
Kr-94	2.3E+04	-	-	-
Kr-95	2.1E+03	-	-	-
Kr-97	1.4E+01	-	-	-
Xe-131m	1.5E+01	1.5E+01	1.3E+01	3.5E+02
Xe-133m	2.9E+02	2.9E+02	1.5E+02	4.0E+03
Xe-133	8.2E+03	8.2E+03	6.1E+03	1.6E+05
Xe-135m	2.6E+04	1.9E+04	-	-
Xe-135	2.2E+04	2.2E+04	4.0E+02	1.1E+04
Xe-137	1.5E+05	4.3E+04	-	-
Xe-138	8.9E+04	6.3E+04	-	-
Xe-139	2.8E+05	2.1E+02	-	-
Xe-140	3.0E+05	-	-	-
Xe-141	2.4E+05	-	-	-
Xe-142	7.3E+04	-	-	-
Xe-143	1.2E+04	-	-	-
Xe-144	5.6E+02	-	-	-
Total	2.4E+06	2.3E+05	2.5E+04	6.7E+05

NOTES:

1. 1971 mixture with no decay (see UFSAR Table 11.1-1).
2. Condenser air inleakage rate = 80 scfm.
3. Plant Capacity Factor = 0.85.
4. 10 to 20 μCi/sec Kr-85 estimated from experimental observations.
5. Holdup time approximately 7 minutes.

TABLE 11.3-2
(Sheet 1 of 2)

OFF-GAS SYSTEM MAJOR EQUIPMENT ITEMS*

Off-Gas Preheaters - Two

Construction: Stainless steel tubes and carbon steel shell. 350 psig shell design pressure, 1,000 psig tube design pressure. 400°F shell design temperature, 575°F tube design temperature.

Catalytic Recombiners - Two

Construction: Stainless steel cartridge, Cr-Mo steel shell. Catalyst cartridge containing a precious metal catalyst on nichrome strips. Catalyst cartridge to be replaceable without removing vessel. 350 psig design pressure. 900°F design temperature.

Off-Gas Condenser - Two

Construction: Cr-Mo steel shell. Stainless steel tubes. 350 psig shell design pressure. 250 psig tube design pressure. 900°F design temperature.

Water Separator - Two

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

Holdup Piping

Construction: Carbon steel. Radiographed, buried, with the outside wrapped and coated for corrosion protection. Ends and elbows reinforced to 1,000 psig design pressure. 150°F design temperature.

Cooler-Condenser - Two

Construction: Stainless steel shell. Stainless steel tubes. 100 psig tube design pressure. 350° psig shell design pressure. 150°F tube design temperature. 150°F shell design temperature.

Moisture Separators (Downstream of cooler-condenser) - Two

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

Off-Gas Reheater - One

Construction: Carbon steel pipe. Electrical resistance heaters on outside of pipe.

* Design codes and standards are provided in Chapter 3.0, Table 3.2-1.

TABLE 11.3-2
(Sheet 2 of 2)

Glycol Tank - One

Construction: Carbon steel. 3000 gal. Water-filled hydrostatic design pressure. O° F design temperature.

Glycol Refrigeration Machines and Motor Drives - Two

Construction: Conventional refrigeration units. Glycol exit solution temperature 35° F.

Glycol Pumps and Motor Drives - Two

Construction: Carbon steel, 1-1/2-inch suction x 3-inch discharge connections, 85-ft total head, 0° F design temperature.

Prefilters and After Filters - Two of each type

Construction: Carbon steel shell. HEPA, moisture-resistant filter element. Flanged shell. 350 psig design pressure. 130° F design temperature.

Carbon Bed Adsorbers - 8 beds

Construction: Carbon steel. 4 feet ID x 21 feet vessels each with a 16-foot packed section containing ~3 tons of 8-14 mesh carbon, Columbia G or equivalent. Design pressure 350 psig Design temperature 130° F.

Off-Gas Pressurized Drain Tank - One

Construction: Carbon steel. 300 gallon capacity, 350 psig design pressure, 140° F design temperature.

Steam Jet Air Ejectors - Two

Construction: Carbon steel shell. Stainless steel tubes. Shell side design pressure: 250 psig. Ejector side design pressure: 1050 psig. Final discharge pressure: 6 psig. Temperature 228°F

TABLE 11.3-3

OFF-GAS SYSTEM PROCESS INSTRUMENT ALARMS

	MAIN CONTROL ROOM	
PARAMETER	INDICATED ONLY	RECORDED
Preheater discharge temperature - low	X	X
Recombiner catalyst temperature - high/low		X
Off-gas condenser drain well (dual) level - high/low	X (Local)	
Off-gas condenser gas discharge temperature - high	X (Local)	
H_2 analyzer (off-gas condenser discharge) - (dual) - high		X
Refrigeration machine inoperable	X	
Off-gas condenser discharge radiation - high		X
Normal gas flow (off-gas condenser discharge) - high/low		X
Startup gas flow (off-gas condenser discharge) - high		X
Cooler-condenser discharge temperature high/low		X
Glycol solution temperature - high/low		X
Glycol level - low	X	
Charcoal bed temperature - high		X
Charcoal vault temperature - high/low		X
Prefilter differential pressure - high	X	
After filter differential pressure - high	X	
Off-gas (carbon bed discharge) radiation high	X	X
Charcoal Vault Radiation - high	X	
Off-gas pressure drain tank levels	X	
Instrumentation elements:		
Temperature- thermocouple		
Level - differential pressure diaphragm		

TABLE 11.3-4
(Sheet 1 of 4)

OFF-GAS SYSTEM EQUIPMENT MALFUNCTION ANALYSIS

EQUIPMENT ITEM	MALFUNCTION	CONSEQUENCES	DESIGN PRECAUTIONS
Preheaters	Steam leak	Would further dilute process off-gas. Steam consumption would increase.	Spare preheater.
	Low pressure steam supply	Recombiner performance would fall off at low power level, and hydrogen content of recombiner gas discharge would increase, eventually to a combustible mixture.	Low-temperature alarms on preheater exit and recombiner inlet. Recombiner H_2 analyzer.
Recombiners	Catalyst gradually deactivates	Temperature profile changes through catalyst. Eventually excess H_2 would be detected by H_2 analyzer or by gas flowmeter. Eventually the gas could become combustible.	Temperature probes in recombiner and H_2 analyzer provided. Spare recombiner.
	Catalyst gets wet at start	H_2 conversion falls off and H_2 is detected by downstream analyzers. Eventually the gas could become combustible.	Condensate drains, temperature probes in recombiner. Air bleed system at startup. Recombiner thermal blanket, spare recombiner, and heater. Hydrogen analyzer.
Off-gas condenser	Cooling water leak	The coolant (reactor condensate) would leak to the process gas (shell) side. This would be detected if drain well liquid level increases. Moderate leakage would be of no concern from a process stand point. (The process condensate drains to the hotwell.)	Spare condenser provided.

TABLE 11.3-4
(Sheet 2 of 4)

EQUIPMENT ITEM	MALFUNCTION	CONSEQUENCES	DESIGN PRECAUTIONS
Drain well	Liquid level instruments fail	If both drain valves fail to open, water will build up in the condenser and pressure drop will increase.	Two separate drain systems, each, provided with high- and low-level alarms.
		The high ΔP, if not detected by instrumentation, could cause pressure buildup in the main condenser and eventually initiate a reactor scram.	
		If a drain valve fails to close, gas will recycle to the main condenser, increase the load on the SJAE, and cause back pressure on the main condenser, eventually causing a reactor scram.	
Water separator	Corrosion of wire mesh element	Higher quantity of water collected in 30-minute holdup line and routed to radwaste.	Stainless steel mesh specified.
Holdup line	Corrosion of line	Leakage to soil of gaseous and liquid fission products.	Outside of pipe dipped and wrapped.
Cooler-condensers	Corrosion of finned tube	Glycol-water solution would leak into process (shell) side and be discharged to clean radwaste. If not detected at radwaste, the glycol solution would discharge to the reactor condensate system.	Stainless-steel-finned tubes specified. The inventory of glycol-water can be observed in tank. A002 - spare cooler provided.

TABLE 11.3-4
(Sheet 3 of 4)

EQUIPMENT ITEM	MALFUNCTION	CONSEQUENCES	DESIGN PRECAUTIONS
	Icing up of finned tube	Shell side of cooler could plug up with ice, gradually building up pressure drop. If this happens the spare unit could be activated. Complete blockage of both units would increase ΔP and lead to a reactor scram.	Design glycol-H_2O solution temperature of 35° to 40°F. Spare unit provided. Redundant temperature indication and alarm systems.
Moisture Separators	Corrosion of wire mesh element	Increased moisture would be retained in process gas routed to charcoal adsorbers. Over a long period, the charcoal performance would deteriorate as a result of moisture pickup.	Stainless steel mesh specified. Spare unit provided.
Prefilters	Holes in filter media	More radioactivity would deposit on the charcoal in the first adsorber vessel of the train. This would increase the radiation level in the charcoal vault and make maintenance more difficult.	ΔP instrumentation provided. Spare unit provided.
Charcoal adsorbers	Charcoal gets wet	Charcoal performance will deteriorate gradually as charcoal gets wet. Holdup times for krypton and xenon will decrease, and plant emissions will increase.	Highly instrumented mechanically simple gas dehumidification system with redundant equipment.
Vault air conditioning units	Mechanical failure	If ambient temperature exceeds approximately 80°F, increased emission could occur.	Spare air conditioning unit provided.

TABLE 11.3-4
(Sheet 4 of 4)

EQUIPMENT ITEM	MALFUNCTION	CONSEQUENCES	DESIGN PRECAUTIONS
		If ambient temperature is below approximately 60° F, charcoal could pick up additional moisture.	Vault temperature alarms provided.
After filters	Hole in filter media	Probably of no real consequence. The charcoal media itself should be a good filter at the low air velocity.	ΔP instrumentation provided. Spare unit provided.
Glycol refrigeration machines	Mechanical failure	If spare unit fails to operate, the glycol solution temperature will rise and the dehumidification system performance will deteriorate. This will cause gradual buildup of moisture on the charcoal, with increased plant emissions.	Spare refrigerator provided. Glycol solution temperature alarms provided.
Steam jet air ejectors	Low flow of motive high-pressure steam	When the hydrogen and oxygen concentrations exceed 4 and 5 volume percent, respectively, the process gas becomes flammable.	Alarms provided on steam for low steam flow and low steam pressure.
		Inadequate steam flow will cause overheating and deterioration of the catalyst.	Steam flow to be held at constant maximum flow regardless of plant power level.
	Wear of steam supply nozzle of ejector	Increased steam flow to recombiner. This could reduce degree of recombination at low power levels.	

TABLE 11.3-5

SUMMARY OF ATMOSPHERIC DISPERSION FACTORS**
AND DEPOSITION VALUES* FOR SELECTED LOCATIONS**

LOCATION	χ/Q (sec/m^3)	D/Q (1/m^2)
Nearest milk cow (5.00 mi E)	1.7E-08	1.6E-10
Nearest meat animal (0.80 mi SW)	1.4E-08	5.0E-10
Nearest residence (0.60 mi S)	7.5E-09	3.7E-10
Nearest vegetable garden (0.90 mi WSW)	1.2E-08	4.9E-10
Nearest site boundary (0. 31 mi W)	5.7E-09	4.8E-10

* Using one year of meteorological data obtained at LSCS site.
** Calculated using the methodology of Regulatory Guide 1.111, "Methods of Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water Cooled Reactors, " Revision 1, July 1977.

TABLE 11.3-6

ESTIMATED ANNUAL AIRBORNE RELEASES OF IODINE-131
FROM BWR'S DURING POWER GENERATION AND REFUELING/
MAINTENANCE OUTAGE CURIES PER PLANT

CHEMICAL FORM OF IODINE-131

	INORGANIC (PARTICULATE + ELEMENTAL + HYPOIODOUS ACID) OPERATING PERIOD					ORGANIC (CH_3I) OPERATING PERIOD				
Plant Source	Power Generation	+	Refueling Outage		Total Release (Inorganic)	Power Generation	+	Refueling Outage		Total Release (Organic)
Reactor Building	0.0160	+	0.0434	=	0.0594	0.0083	+	0.0233	=	0.0316
Radwaste Building	0.0039	+	0.0011	=	0.0050	0.0146	+	0.0144	=	0.0290
Turbine Building	0.0112	+	0.0044	=	0.0156	0.0015	+	0.0048	=	0.0065
Gland Seal	0	-		=	0	0		-	=	0
Mechanical Vacuum Pump	-		0.0085	=	0.0085	-		0.2741	=	0.2741
		Total*	(Inorganic)	=	0.093		Total*	(Organic)	=	0.37
			rate	≈	0.0029µCi/s			rate	≈	0.012µCi/s

* Total rounded to 2 significant figures.
** The gland seal steam release for LSCS is zero because of its separate steam on the gland seals.

TABLE 11.3-7

MIXED NOBLE RADIOGAS NUCLIDES, CURIES/YEAR

		SOURCE				
NUCLIDE	HALF-LIFE	REACTOR	TURBINE	RADWASTE	MVP	TOTAL[b]
Kr-89	3.2m	1	503	34[c]	ND	538
Xe-137	3.9m	78	386	113[c]	ND	577
Xe-138	14.2m	12	1179	2	ND	1193
Xe-135m	15.7m	111	464	667	ND	1242
Kr-87	76m	6	95	a	ND	101
Kr-88	2.79h	9	102	a	ND	111
Kr-85m	4.4h	6	2	a	ND	8
Xe-135	9.16h	173	672	328	200	1373
Xe-133m	2.3d	ND	ND	60	ND	60
Xe-133	5.27d	103	581	294	300	1278
Total Annual Release		500	4000	1500	500	6500

a = Less than 1 curie/year.
b = Excludes gland seal steam release because it is nonradioactive steam.
c = These results are overestimates because such short-lived radioisotopes are not normally expected in the radwaste building; however, data have been reported for these species at one plant.
ND = Not Detected.

11.4 SOLID WASTE MANAGEMENT SYSTEM

11.4.1 Design Bases

The solid radwaste system has been designed to receive, dewater, solidify, package, handle, and provide temporary storage facilities for all radioactive wet solid wastes generated by Units 1 and 2 prior to offsite shipment and disposal. The solid waste management system also receives, decontaminates and/or compacts, and provides temporary storage facilities for all radioactive dry wastes produced during station operation and maintenance prior to offsite shipment and disposal. In addition, the station utilizes offsite vendor servicesw for Dry Active Waste (DAW) processing. These services may include compaction, incineration, thermal processing, and/or sorting of the radioactive waste (DAW).

A system bypass (Figure 11.4-1) allows connection to portable equipment. This enables the use of state of the art radwaste processing technology necessary to comply with evolving federal, state and burial site criteria.

Process Control Program (PCP) – Shall contain the current formulas, sampling, analysis, test, and determinations to be made to ensure that processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR parts 20, 61, and 71, State regulation, burial ground requirements, and other requirements governing the disposal of solid radioactive waste.

Changes to the PCP shall be documented and records of reviews performed shall be retained. This documentation shall contain:

- Sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s) and,

- A determination that the change will maintain the overall conformance of the solidified waste product to existing requirements of Federal, State, or other applicable regulations.

Changes shall become effective after review and acceptance, including approval by the Station Manager.

11.4.1.1 Power Generation Design Bases

The solid radwaste system is designed to minimize the volume of waste requiring shipment offsite. The system is designed specifically for a 40-year service life, maximum reliability, minimum maintenance, and minimum exposure to operating and maintenance personnel. The system has the flexibility to handle a wide range of radioactive waste products.

Equipment and storage capacities, as noted in Table 11.4-1, are selected to meet the station's solid waste processing needs in all the operational modes of the station, including anticipated operational occurrences, without impairing the power generation availability of the station.

In addition, a separate, interim radwaste storage facility (IRSF) provides contingency storage space for approximately 2 years of abnormal output of processed, solid waste. Any waste stored in the IRSF will have been prepared for shipment and disposal.

An interim storage facility for dry active waste is provided. The facility is designed to hold an approximate two-year inventory of dry active waste which may be stored for a period of up to five years time.

11.4.1.2 Safety Design Bases

The solid radwaste system is designed to package radioactive wastes for offsite shipment and burial or onsite storage in accordance with applicable NRC, DOT, and Burial Site regulations including 49 CFR 170-178 and 10 CFR 71.

System safety is emphasized through redundancy in design of primary components, compartmentalization of equipment layout, remote automatic and/or manual operation, shielding, containment of possible spills and displaced air, remote decontamination, if required, accurate process monitoring, and interlocking of process controls.

Complete solidification of concentrator waste and sump sludge wastes is maintained by testing probable waste constituents and the solidification agent prior to start of operation. A set of process parameters is formulated for each type of waste which establishes operational boundary conditions. Operation of the solid radwaste system adheres to these boundary conditions with conformance recorded for individual batches, thereby reasonably assuring complete solidification.

The solid radwaste system is designed to "fail safe" upon loss of system power, water, or air supply. System controls are designed to avoid a malfunction or spill due to operator error. The system is designed to reduce decontamination and cleaning requirements.

The solid radwaste system is enclosed in a Seismic Category II structure. The below grade walls are part of the total structural shear wall system and as such are designed to withstand the effects of an earthquake. All piping and components of the system are designed and constructed in accordance with requirements for classification of NRC Quality Group D.

Portable equipment should incorporate safety design features as stated for the solid radwaste system where feasible. Regulatory Guide 1.143 seismic criteria to structures housing solid waste systems are not applicable to portable Solid Waste System (ETSB-11-3).

The IRSF is designed in accordance with the guidance provided by the Nuclear Regulatory Commission (NRC) Standard Review Plan (SRP) 11.4, Solid Waste Management Systems, Appendix 11.4-A, Design Guidance for Temporary Onsite Storage of Low Level Radioactive Waste.

The design bases solid radwaste system output is 3500 to 8050 drums per year. The Solid Radwaste System operation is controlled by the station Process Control Program.

11.4.1.3 Types of Waste

Wet solid wastes are transported to the solid radwaste system from the liquid radwaste system via the sludge subsystem described in Subsection 11.2.2.5. The types of waste handled by the solid radwaste system consist of the following:

a. Expended deep bed demineralizer bead resins consisting primarily of a copolymer of styrene and divinylbenzene.

b. Waste filter sludge consisting of spent precoat material, and solid filter cartridges.

c. Cleanup and fuel pool filter-demineralizer sludge consisting primarily of powdered demineralizer resins.

d. Evaporator concentrates: consists of inputs from floor drain and chemical waste concentrators (evaporator bottoms concentrated to approximately 25% total solid by weight). (The evaporators are no longer operated and are abandoned-in-place.)

e. Laundry waste consisting primarily of detergent water with suspended and dissolved solids.

f. Ultrasonic resin cleaner sludge consisting primarily of piping corrosion products and resin fines.

g. Low-level dry active wastes consisting of air filters; miscellaneous paper, rags, etc., from contaminated areas; contaminated clothing, tools, and equipment parts which cannot be effectively decontaminated; and solid laboratory wastes. High- level dry wastes consist mostly of equipment that has been activated during reactor operation (e.g., core components).

11.4.1.4 Expected Volumes

Table 11.4-2 indicates the design-basis solid radwaste system output (maximum and expected annual volumes). The radionuclide content of the various types of waste is indicated in Table 11.2-4.

11.4.2 System Description

Operation of the solid waste management system is indicated by Figure 11.4-1.

The solid radwaste system is comprised of the following components or subsystems.

11.4.2.1 Drum Preparation Station

This subsystem is no longer used and is abandoned-in-place. The design basis information for the system remains the historical reference. This subsystem consists of cement unloading, storing, feeding, weighing, and conveying equipment used to load 55-gallon drums with the correct weight of cement. A mixing weight is added to the drum to ensure uniform mixing when the drum is tumbled.

The drum preparation station area is maintained dry and is enclosed within curbing to prevent mixing of spilled cement with water. The cement storage tank contains a pressure relief valve to prevent pressurization of the tank during filling with cement. This subsystem is designed for dust-free operation with an exhaust air filter assembly attached to the side of the cement storage tank to capture cement dust generated within the tank.

11.4.2.2 Decanting Station

Original Design Basis

This subsystem consists of a stainless steel decanting tank that can receive and mix spent resins, waste filter sludges, filter-demineralizer sludges, or ultrasonic resin cleaner sludges; a progressive cavity decanting pump that dewaters excess liquid; a positive displacement piston-type metering pump that transports accurate quantities of waste from the decanting tank to the drum; and all associated valves and instrumentation.

This equipment is bolted to a thick machined steel shield wall which is attached to a mechanical steel insert fit into an opening in the concrete shield wall. Processing equipment in contact with radioactive materials is located on the radioactive side of the shield wall. Located on the low radiation side of the shield wall are the decanting station drives, limit switches, and instrumentation so that most maintenance can be performed with minimum exposure to personnel. Most accessory valves whose operators and limit switches are located on the decanting tank side of the shield wall can be mutually operated from the lower radiation side.

Operation shafts and plugs extending through the wall are stepped to prevent direct radiation passage from one side of the wall to the other. Service pipes or tubing and electrical wires from the radioactive side to the low radiation side of the shield wall pass over the top rather than through it.

At the upper side of the closed tank a connection for emergency overflow directs any overflow to a sump. A vent nozzle is provided for venting air displaced when filling the tank to the plant radioactive vent system. A movable decanting arm is used to decant water to the liquid and liquid slurry interface level. Two ultrasonic sensors located at the inlet nozzle are used to detect arm position and the liquid and liquid slurry interface level. Mechanical limit switches prevent overtravel of the decanting arm.

The metering pump can also be used to flush radwaste pipe to the radwaste holding tanks. When used in conjunction with the task internal spray nozzles, the metering pump can be used to minimize the decant tank radiation levels for required maintenance in the tank areas.

Current Design Basis

This subsystem consists of a stainless steel decanting tank that receives decant water from vendor waste processing, a positive displacement piston-type metering pump that returns the tank contents to the liquid radwater system, and associated valves and instrumentation.

This equipment is located on the radioactive side of a shield wall. Most valves whose operators and limit switches are located on the radiation side of the shield wall can be operated from the low radiation side. Where practical, valve operators, limit switches, and instrumentation are located on the low radiation side of the shield wall so that most maintenance can be performed with minimum exposure to personnel.

Process piping, hoses, and electrical wires from the radiation side to the low radiation side of the shield wall pass over the shield wall rather than through it.

The decant tank is a closed tank, with an emergency overflow line which directs overflows to a sump. A vent line routes air displaced when filling the tank to the plant radioactive vent system.

Drum processing unit components not used in the Current Design Basis are abandoned-in-place.

11.4.2.3 Drumming Station

The clean condensate line to the 'A' Drum Processing Unit (DPU) (which previously routed water for DPU and drum washing) has been converted to a waste line for routing floor drain and chemical waste to a vendor skid. (See Subsections 11.2.2.2 and 11.2.2.3).

The 'B' DPU is no longer used to process solid waste. Instead, the waste is processed through a vendor skid, as discussed in subsections 11.4.1 and 11.4.2.

Except as described above, this subsystem is no longer used and is abandoned-in-place. The design basis information for the system remains for historical reference. This station consists of a drum processing unit and a heat traced positive displacement metering pump. Both are attached to the radioactive side of a shielding wall identical to the one supporting the decanting station (described in Subsection 11.4.2.2). Also attached to the shield wall is a remotely removable platform with a scale and radiation monitor. This provides weight and activity level readouts on the control console of the drum after removal from the drum processing unit.

The pump fills the drum with accurate quantities of evaporator concentrates. The drum processing unit is essentially a stainless steel box with an air-cylinder actuated hatch in the top.

The following operations occur within the confines of the drum processing unit: cap removal, drum filling, cap reinsertion, tumbling of the drum for mixing, and washing the exterior of the drum, if necessary. These operations are performed remotely. The operating mode can be automatic or manual. For safety reasons, opening and closing of the hatch is not part of the automatic cycle. All motions are interlocked to ensure correct sequencing. Lights at the control console indicate the operating status of the drum processing unit. If malfunction occurs, the operator can quickly diagnose the cause and decide whether to clear the fault or change to manual control. Automatic sequencing is locked out until the malfunction has been corrected.

Air displaced during filling is vented through an annular passage around the fill nozzle and discharged to the drum processing unit which is maintained at a negative pressure with respect to the plant radioactive vent system assuring sufficient air circulation to keep the drums clean. Two separate fill nozzles are provided: one for sludge/slurry waste, one for concentrated waste. A sonic type overflow sensor is located in the filling nozzles to prevent spills or excessive liquid level in the drum due to automatic control failure or operator error. In the drum processing unit the drum is placed in a movable cradle which raises the drum until the fill nozzle is inside the drum cap fitting before filling starts. Limit switches will not permit flow until the drum is in the proper position. A measured amount of clean water flush occurs automatically after the filling cycle to clear radioactive material from the fill nozzle. After each drum filling a pause occurs to permit dribble to enter the drum before it is disengaged from the filling nozzle.

The assembly is equipped with spray nozzles for decontamination of the inside of the drum processing unit as well as washing the exterior of a drum during tumbling, if required.

The system has the capability of refilling a drum with waste after the initial tumbling mixing sequence. Separate nozzles provide the flexibility of filling the second time with a different type of waste.

11.4.2.4 Container Handling Equipment

This equipment includes two remote operated cranes with television cameras for visual surveillance. The cranes are used to remotely transport and position sealed drums or high integrity containers and liners in either high, low, or intermediate level storage, and retrieve and transport them to trucks for offsite disposal. Two electrical circuits are provided for the trolley, bridge, and hoist: one for the high-speed and one for the low-speed motors. This ensures that electrical failure will not prevent remote removal of the crane from a radiation zone or completing the operation in process at time of failure. A crane target grid system combined with television cameras is provided for accurate remote control positioning. For safety, the container must be raised to the full-up position before high speed operation is possible, and the container cannot be released from the grab while the container is suspended. A soft start-stop drive system for the crane is installed for smoother operation and reduced mechanical brake maintenance.

11.4.2.5 Smear Test and Label Station

This subsystem is no longer used and is abandoned in place. The design information for the system remains for historical reference. This portion of the solid waste management system consists of a setdown position for drums behind a small shield wall equipped with access plugs and working tools to accomplish remote labeling, smear testing, and radiation monitoring of all external surfaces of sealed drums prior to offsite disposal.

11.4.2.6 Dry Waste Compactor

The dry waste compactor compresses paper, fabrics, plastics, and light metal into 55-gallon drums. An air filtration assembly is provided to maintain control of contaminated particles during compactor operation. Capture of radioactive dust is accomplished by means of a roughing filter and two HEPA filters operating in parallel. A connection is included for attaching the filtration system outlet to the plant radioactive vent system.

The radioactivity of most of the dry waste is low enough to permit handling by contact. The compactor is designed to meet OSHA standards for safe operation and built to standards for a 40-year expected life of the station.

11.4.2.7 Storage Areas

Shielded areas are provided in the radwaste building for storage of high, low, and intermediate level sealed drums and compacted dry waste drums per requirements noted in Table 11.4-1. The low level and intermediate level storage areas located on the southern portion of the radwaste building are currently used to stage a vendor liquid radwaste processing system. Visual surveillance for the high, low and intermediate level storage area is provided by television cameras. The high level storage area consists of a series of concrete plugs for additional shielding during storage. Storage is provided for incompressible dry solid waste.

The IRSF provides contingency interim storage for processed, solid radwaste. A program of quarterly visual inspection and surveillance of the IRSF will be performed with CCTV monitors. Visual inspection of waste containers will be performed to the greatest extent possible without movement of the containers. Containers will be inspected for signs of degradation and loss of integrity. The placement of containers of radwaste in the IRSF is controlled by administrative procedures to assure that the building exterior dose rate limits are never exceeded.

An interim storage facility for dry active waste is provided. The facility is designed to hold an approximate two-year inventory of dry active waste which may be stored for a period of up to five years time.

Radioactive waste and materials are stored in various locations within the station's outside buildings. The exterior dose rate limits of the storage locations are administratively controlled. Routine surveillances of these storage areas ensure container integrity.

11.4.2.8 Control Station

This area provides the capability of remote visual monitoring and control of the solid radwaste system at one control panel located in the radwaste control room. A liquid/solid interface control panel is provided for transferring waste to the solid radwaste system from the liquid radwaste system sludge subsystem. A record board is mounted on the radwaste control room wall to record the location of all drums within the drumming area. Drum set-down positions are represented by hooks onto which are placed circular discs. The discs are inscribed with the identification number of the drum, its weight, the date it was drummed, and the drum radiation level as of that date.

The IRSF has a control room from which all waste monitoring and handling is controlled, using the IRSF building crane. The system for recording the location of stored waste in the IRSF is the same as that discussed above.

11.4.2.9 Processing Area For Radwaste Shipping Containers

This area consists of six concrete cubicles each designed to hold one Radwaste Shipping Container (up to 200 cubic feet size). The containers are either carbon steel liners or polyethylene High Integrity Containers. The containers are remotely transported using the overhead cranes discussed in Section 11.4.2.4.

Each cubicle will have a removable lead/steel cover to provide shielding for workers. The covers have a removable center plug to allow access to the container for filling.

The containers will be filled, solidified, or dewatered, as required by the Radwaste Process Control Program.

Each cubicle is connected to the floor drain system which is routed to the liquid radwaste system. Additionally this area is provided with curbing to contain potential spills outside of the cubicles.

TABLE 11.4-1
(SHEET 1 of 2)

SOLID WASTE MANAGEMENT SYSTEM EQUIPMENT AND STORAGE DESIGN CAPACITIES RADWASTE BUILDING

PROCESSING EQUIPMENT	QUANTITY	DESIGN CAPACITY
Decanting Tank	2	500 gallons
Decanting Pump	2	18 gpm
Metering Pump	4	15 gpm
Cement Storage Tank	1	1000 ft^3 (No longer used)
Drum Processing Unit	2	---- (No longer used)
Packaging Container	-	55 gallons (No longer used)
Dry Waste Compactor	1	----
Bridge Crane*	2	9.3 tons

STORAGE AREA	NUMBER OF STORAGE AREAS	DESIGN CAPACITY PER STORAGE AREA
Low level (<1.5 R/hr) ***	1	435 drums (Historical)
Intermediate level (1.5 R/hr - 5 R/hr) ***	1	395 drums (Historical)
High level (>5 R/hr)	2	208 drums
Dry compacted waste	1	75 drums
Dry uncompacted waste	1	90 ft^3
Empty drum	4	200 drums (total)
DAW**	1	16,200 ft^3

*Overhead Crane Operating Speeds
** Not in Radwaste Building
***Currently used to stage Vendor Liquid Radwaste Processing System

	High-Speed	Low-Speed
Drum Grab Hoist	30 fpm	7 5 fpm
	LOAD < 1 TON	LOAD > 1 TON
BRIDGE **	0-125 fpm	0-40 fpm
TROLLEY **	0-125 fpm	0-40 fpm

** Backup circuit speed - 2 fpm

TABLE 11.4-1
(SHEET 2 of 2)

SOLID WASTE MANAGEMENT SYSTEM EQUIPMENT
AND STORAGE DESIGN CAPACITIES

Interim Radwaste Storage Facility

STORAGE AREA	NUMBER OF STORAGE AREAS	DESIGN CAPACITY PER STORAGE AREA
15 R/hr*	1	8875 drums or 540 liners/HICs

20-Ton Overhead Crane Operating Speeds

	High-Speed (fpm)	Low-Speed (fpm)
Bridge	94	5
Trolley	70	5
Drum Grab Hoist	17	3.3

*Dose rate at inside surface of storage area wall.

TABLE 11.4-2

EXPECTED AND DESIGN ANNUAL VOLUMES OF (UNITS 1 & 2)
SOLID WASTE MANAGEMENT SYSTEM OUTPUT

TYPE OF WASTE	VOLUME	DRUMS*
a. Deep Bed Resin	2,968 ft^3	740
b. Waste Filter Sludge	26,242 - 79,735 gal	875 – 2,660
c. Filter-Demineralizer Sludge	2,019 - 7,065 gal	70 - 235
d. Evaporator Concentrates (Note 4)		
Floor Drain Waste	8,000 - 12,000 gal	270 - 400
Regenerant Solutions (Note 1)	50,000 - 120,000 gal	1670 - 4000
e. Laundry (Note 2)	3,000 - 4,000 gal	100 - 140
f. URC Sludge	5,300 - 7,420 gal	175 - 250
TOTAL SOLIDIFIED	116,760 - 252,420 gal	3,900 – 8,425
g. Dry Active Waste	8,000 ft^3	
h. Vendor System Waste (Note 3)	2,120 ft^3	

* 30-gallon drums assumed.

Note 1: Resin is no longer regenerated

Note 2: The Reverse Osmosis Unit is no longer used.

Note 3: The vendor system waste is waste generated from a vendor liquid radwaste system. The waste may include ion exchange resin and/or charcoal.

Note 4: The evaporators are abandoned-in-place and no longer produce waste volumes.

11.5 PROCESS AND EFFLUENT RADIOLOGICAL MONITORING AND SAMPLING SYSTEMS

11.5.1 Design Bases

11.5.1.1 Design Objectives

The process radiological monitoring and sampling system provides measurement, indication, and/or control of radioactivity in those streams which could conceivably be contaminated by radioactive materials.

The effluent radiological monitoring and sampling system provides measurement, indication, and control of radioactivity in those streams which discharge to the environs outside the plant boundaries.

The systems are subdivided into gaseous (airborne) systems and liquids systems. Both continuous monitoring and sampling plus laboratory analysis are used for all systems.

The process monitor systems provide operating personnel with radiological measurements within the plant process systems. The continuous monitors provide a continuous readout of the radiation levels, and they annunciate or generate automatic control of the process stream when a significant increase occurs. By sampling and laboratory analysis, the type of radioactive material and the specific radionuclide present can be determined qualitatively and/or quantitatively.

The effluent monitoring systems provide operating personnel with a measurement of the radioactivity levels present in each of the plant's air exhaust and liquid discharge streams.

The objective of the effluent radiological monitoring system is to sample and monitor each plant effluent discharge path for radioactivity prior to discharge. This is satisfied by the installation of sampling monitors on the following airborne effluent streams and liquid effluent streams:

a. airborne effluent streams

    1. station vent stack, and

    2. standby gas treatment stack (stack within the station vent stack);

b. liquid effluent streams:

    1. service water,

2. radwaste effluent,

3. RHR service water effluent loop A, and

4. RHR service water effluent loop B.

The objective of the process radiological monitoring system is to monitor those sections of the plant process to control the release of radioactivity into the effluent streams. This is satisfied by the installation of process gaseous and airborne radiation monitors and liquid process monitors in the following locations:

a. process gaseous airborne monitors for:

   1. main steamline,

   2. off-gas pretreatment,

   3. off-gas posttreatment,

   4. fuel pool ventilation exhaust monitor,

   5. reactor building ventilation exhaust,

   6. carbon bed vault; and

b. process liquid monitors for:

   1. reactor building closed cooling water.

This group of monitors is used for surveillance and control of radioactive materials in gaseous and liquid effluents during normal reactor operations, including expected operational occurrences.

The process and effluent radiological monitoring and sampling system has been designed to provide for compliance with the requirements of 10 CFR 20 and 10 CFR 50 as well as the General Design Criteria of Appendix A, 10 CFR 70, and 10 CFR 100.

11.5.1.2 Design Criteria

The design of the process and effluent radiological monitoring and sampling system was based on the following criteria:

a. The airborne monitor detectors are beta-responsive to provide maximum sensitivity, except in the case of the pretreated off-gases, where gamma-sensitive chambers are used to measure the high concentration of radioactive noble gases. The off-gas post-treatment monitors are gamma responsive scintillation detectors.

b. Liquid monitors are gamma-responsive scintillation detectors to provide maximum sensitivity to a water medium.

c. Shielding is provided to reduce background and increase sensitivity.

d. The gaseous monitor range of detectability is based on actual experience at operating BWR's.

e. The detectors are nonsaturating.

f. All automatic controls and alarm trips annunciate in the main control room.

g. Monitors read out and record in the main control room.

h. Monitor components are readily accessible for maintenance.

i. The monitoring systems are designed for operability within the environmental conditions anticipated; the environmental design conditions are listed in Table 11.5-l.

j. Alarm setpoints are adjustable over the range of the instrument.

k. The following statements apply to the effluent monitors and samplers for airborne and gaseous radioactivity:

    1. They continuously withdraw a proportional, isokinetic, and representative sample as recommended by ANSI-N13.1-1969.

    2. The radioparticulates are concentrated on a high-efficiency filter and the radioiodines on an activated charcoal cartridge, each to be changed routinely for laboratory radio analysis.

    3. The radionoble gases are continuously monitored for gross beta activity.

4. Grab samples are taken periodically to correlate empirical data with monitor response.

l. Setpoints and ranges for effluent monitors are established to meet ODCM limits, which encompass 10 CFR 20 (including Table II of Appendix B) and 10 CFR 50 Appendix I objectives. Setpoints for process monitors are established to provide a warning of increased system activity and to initiate corrective action where appropriate.

Two independently adjustable radiation setpoints are provided for most monitors. The lower (High) set point normally activates only an alarm, while the upper (High-High) setpoint activates an alarm and initiates corrective action where appropriate. Setpoints are set to at least twice the background level to reduce the number of spurious trips. Alarm and trip functions associated with the various monitors are listed in Tables 11.5-2, 11.5-3, and 11.5-4. When used in conjunction with High-High setpoints, High setpoints are between background and High-High alarm setpoints. The setpoints are under the administrative control of the plant general manager or his authorized delegate and can be changed if needed within ODCM limits.

m. All process and effluent radiation monitors are annunciated in the main control room. A specific annunciator window alarms for a Low (failure), High, or High-High radiation alarm.

An operator can acknowledge the alarm and silence the audible alarm, but he cannot clear the annunciator window until the alarm condition is no longer present. Radiation alarms can be cleared only if the indication is within the setpoints.

11.5.2 <u>System Description</u>

11.5.2.1 <u>Gaseous Process Radiation Monitors</u>

11.5.2.1.1 <u>Reactor Building Ventilation Exhaust Radiation Monitor</u>

This monitor subsystem measures the radioactivity in the reactor building ventilation system exhaust duct prior to its discharge from the building. In doing so, it complies with General Design Criteria 13, 60, 63, and 64. The monitor system acts to detect a high activity level in the ductwork which could be due to fission gases from a leak or a refueling accident.

Two sets of independent redundant GM detectors are located inside the exhaust plenum ahead of the isolation dampers. Each of the four channels consists of the local detector (GM tube), and a preamplifier. A remote recorder, located in the main control room, trends all four channels. Two upscale (High-High) alarms (from each channel) start the SGTS, close the primary containment vent valves, and trip and isolate the reactor building vent system. Power is supplied from RPS Bus A for two channels, RPS Bus B for two channels, and from the 115-Vac instrument bus for the recorder. This system is Quality Group C and Seismic Category I. Arrangement details are shown on Drawing No. M-153 (sheets 1 and 6), while range, sensitivity, and other details are given in Table 11.5-2.

11.5.2.1.2 Off-Gas Posttreatment Radiation Monitor

This monitor subsystem measures the radioactivity in the off-gas vent pipe after the normal off-gas has been processed through the off-gas treatment (RECHAR) system and before it discharges into the elevated release duct. This complies with General Design Criteria 13, 60, 63, and 64.

A continuous representative sample is extracted from the vent pipe. It passes through a particulate filter, a sample pump, the two in-series gas monitors and then through a rotameter prior to being returned to the vent pipe. Each channel consists of a local detector and preamplifier and a count rate meter with a common recorder in the main control room. A switch located in the control room permits operation of solenoid valves to stop the sample flow and admit room air to purge the detectors. Two other switches can be used to move solenoid- operated check sources into position to check detector response. Three local control switches are also provided for the same purposes at the local instrument rack.

This subsystem provides control functions and is in addition a diagnostic tool which enables the main control room operator to take appropriate action. The monitor system has High and High- High alarms plus an associated valve closure function which closes the off-gas vent valve on a High-High high level alarm. Power is supplied from the ± 24-Vdc Buses A and B for the channels and from the 120-Vac instrument bus for the recorder. This subsystem is Quality Group C and non-Seismic Category I. Arrangement details are shown on Drawing No. M-153 (sheets 3 and 5), while range, sensitivity, and other details are given in Table 11.5-2.

11.5.2.1.3 Off-Gas Pretreatment Radiation Monitor

This monitor subsystem measures the radioactivity in the condenser off-gas at the discharge of the steam jet air ejector or the off-gas condenser. The monitor detects the radiation level which is attributable to the fission gases produced in the reactor and transported in the steam through the turbine to the condenser. It complies with General Design Criteria 13 and 63.

A continuous representative sample is extracted from the off-gas pipe via a stainless steel sample line. It is then passed through a sample chamber and returned to the process piping. 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 can be controlled locally or from a switch located in the main control room. A gamma-sensitive ion chamber is positioned adjacent to the vertical sample chamber and is connected to a digital readout channel. The chamber is listed in Table 11.5-2 for range and sensitivity. Arrangement details are shown in Drawing No. M-153 (sheets 3 and 5).

The channel has alarms but no trip function. Power is supplied from noninterruptible Bus A for the channel, and from the 120-Vac instrument bus for the recorder. This channel is classified Quality Group C and non-Seismic Category I.

The radiation level detected by the logarithmic channel can be directly correlated with the concentration of noble gases by using the semiautomatic sample system which is incorporated as part of this monitor. To use this sample system, a 14-cm serum bottle is inserted into a sample chamber, the vent is evacuated, and a solenoid-operated sample valve is opened to allow off-gas to enter the bottle. The bottle is then removed and counted in the counting room with a multichannel gamma pulse height analyzer to determine the concentration of the various noble gas radionuclides. A correlation between the observed activity and the monitor reading permits calibration of the monitor.

11.5.2.1.4 Main Steamline Radiation Monitor System

This monitor subsystem measures the radioactive gases coming from the reactor through the main steamlines. These gases are primary coolant activation gases plus minor quantities of fission product gases which come from fuel leaks and "tramp" uranium. If a gross release of fission products occurs, the monitoring subsystem provides an alarm in the main control room. The main steamline radiation monitoring system complies with General Design Criteria 13, 20, 21, 23, 24, 60, and 63.

Four detectors are physically located near the main steamlines just downstream of the outboard main steamline isolation valves in the space between the primary containment and secondary containment walls. The detectors are geometrically arranged so that this subsystem is capable of detecting significant increases in radiation level with any number of main steamlines in operation. Their location along the main steamlines allows the earliest practical detection of a gross fuel failure. Table 11.5-2 lists the sensitivity and range of the detectors.

The subsystem consists of four separate, redundant instrument channels. Each channel consists of a local detector (gamma-sensitive ion chamber) and a radiation analyzer. A recorder in the main control room is used to record the four channels.

Power is supplied from RPS Bus A for two channels, from RPS Bus B for two channels, and from the 115-Vac instrument bus for the recorder. This subsystem is Quality Group A and Seismic Category I. Arrangement details are shown on Drawing No. M-153 (sheets 4 and 6).

11.5.2.1.5 Fuel Pool Ventilation Exhaust Radiation Monitors

This monitor subsystem measures the radioactivity in the fuel pool ventilation exhaust duct prior to its discharge to the reactor building exhaust duct header system. This complies with General Design Criteria 13, 60, 63, and 64. The monitor system acts to detect a high activity level in the ductwork which could be due to fission gases from the fuel pool resulting from a refueling accident.

Two sets of independent redundant GM detectors are located inside the fuel pool area exhaust duct. Each of the four channels consist of the local detector (GM tube) and a preamplifier. A remote recorder in the main control room trends all four channels.

Two upscale (High-High) trips (from each set) start the SGTS, close the primary containment vent valves, and trip and isolate the reactor building vent system. A downscale low (failure) alarm results from a low trip. Power is supplied from RPS Bus A for two channels, RPS Bus B for two channels, and the 115-Vac instrument bus for the recorder. This system is Quality Group C and non-Seismic Category I. Arrangement details are shown on Drawing No. M-153 (sheets 1 and 6), while range, sensitivity, and other details are given in Table 11.5-2.

11.5.2.1.6 Carbon Bed Vault Monitoring System

This monitor subsystem measures the radioactivity in the charcoal adsorber vault area prior to passing through the off-gas filter train on its way to the reactor building exhaust duct header system. This complies with General Design Criterion 64. The monitor system acts to detect a high activity level in the charcoal adsorber bed vault area which could be due to fission gases released from off-gas system accidents.

Two redundant GM detectors are located in the adsorber vault. Each channel consists of the local detector, preamplifier, and indicating monitor in the control room. Separate alarms result from each channel in the event of a high trip or a downscale low (failure) trip. Power is supplied from RPS Bus A for one channel and RPS Bus B for the other channel. This system is Quality Group C and non-Seismic

Category I. Arrangement details are shown on Drawing No. M-153 (sheets 3 and 5), while range, sensitivity, and other details are given in Table 11.5-2.

11.5.2.2 Gaseous Effluent Radiation Monitors

11.5.2.2.1 Station Vent Stack Monitor

A wide range noble gas monitoring system has been installed to sample the effluent stream which leaves the station vent stack. This monitoring system has a measurement range for radioactive noble gas concentrations of 1×10^{-7} μCi/cc to 1×10^{5} μCi/cc and is designed to meet Class 1E requirements. It is also in the process of being qualified to IEEE 323-1974. Arrangement details for this system are shown in Drawing No. M-153 (sheets 1 and 7). This system has the following characteristics:

a. The system uses three beta sensitive scintillation detectors, a Low Range 1×10^{-7} μCi/cc to 1×10^{-1} μCi/cc, a Mid Range 1×10^{-4} μCi/cc to 1×10^{2} μCi/cc, and a High Range 1×10^{-1} μCi/cc to 1×10^{5} μCi/cc, which meets the requirements of NUREG-0737.

b. Off-line sampling is provided to the systems detectors using isokinetic probe OD18-N001 (2 scfm). An auxiliary sample pump is provided to maintain isokinetic sampling to the mid/hi detectors which uses a low volume 0.06 scfm pump for sampling.

c. Another requirement of NUREG-0737 is to have the ability to provide onsite analysis to determine the particulate and iodine release rates. The system is designed with two separate trains unshielded and shielded for the collection of particulates and iodines. Each train has three particulate and iodine collecting devices. The unshielded portion is used for collection during normal operations while the shielded portion is used for collection as a backup for post-accident conditions. The shielded samples can be removed from the skid and placed on a cart for transport to the laboratory for isotopic analysis. The shielding provides protection to the workers during collection and transport.

d. The system is designed to minimize background radiation to the detectors. The sample skid (OPLD4J) is separated from the detector skid (OPLD5J) to obtain as low a background as possible.

All monitor readouts of gas concentration and dose release rates are provided in the Technical Support Center and the Control Room continuously.

Table 11.5-3 lists the sensitivity and range of the detectors.

11.5.2.2.2 Standby Gas Treatment System Stack Effluent Monitor

A wide range noble gas monitoring system has been installed in the standby gas vent stack which is vertically supported by and physically located inside the station vent stack. This monitoring system has a measurement range for radioactive noble gas concentrations of 1×10^{-7} µCi/cc to 1×10^{5} µCi/cc and is designed to meet Class 1E requirements. It is also in the process of being qualified to IEEE 323-1974. Arrangement details for this system are shown in Drawing No. M-153 (sheets 1 and 7). This system has the following characteristics:

a. The system uses the same type of detectors as used in the Station Vent Stack Monitor to meet the requirements of NUREG-0737.

b. Off-line sampling is provided using one isokinetic probe OD18-N519 (2.0 scfm for low range sampling). An auxiliary sample pump is provided to maintain isokinetic sampling to the mid/hi detectors which uses a low volume 0.06 scfm pump for sampling.

c. Another requirement of NUREG-0737 is to have the ability to provide onsite analysis to determine the particulate and iodine release rates. The system is designed with two separate trains unshielded and shielded for the collection of particulates and iodines. Each train has three particulate and iodine collecting devices. The unshielded portion is used for collection during normal operations while the shielded portion is used for collection as a backup for post-accident conditions. The shielded samples can be removed from the skid and placed on a cart for transport to the laboratory for isotopic analysis. The shielding provides protection to the workers during collection and transport.

d. The system is designed to minimize background radiation to the detectors. The sample skid (OPLDIJ) is separated from the detector skid (OPLD2J) to obtain as low a background as possible.

All monitor readouts of gas concentration and dose release rates are provided in the Technical Support Center and the Control Room continuously. Table 11.5-3 lists the sensitivity and range of the detectors.

11.5.2.3 Liquid Process and Effluent Monitors

11.5.2.3.1 General Description

There are five process and effluent liquid monitors that monitor directly or indirectly the potentially radioactive contaminated discharges from the operating plant to the environment. They conform to 10 CFR 50 General Design Criteria 23, 60, 63, and 64 as well as to Appendix I Criteria. Each monitor system consists of a scintillation detector inserted into a well or adjacent to the process stream piping. Monitor locations are selected to obtain a reasonable geometry and are positioned away from crud trap and high associated background regions.

The detector preamplifier unit is designed to remain fully operational over the range of the temperatures shown in Table 11.5-1. If exposed to radiation transients which exceed the channel range, the channel maintains full-scale deflection and returns to normal functioning when the transient has subsided.

The component and equipment classifications are shown on Table 3.2-1. Liquid monitor system details are given in Table 11.5-4. Monitor arrangements are shown in Drawing No. M-153 (sheets 4 and 6). Specific descriptions of the individual monitors are given in the following subsections (11.5.2.3.2 through 11.5.2.3.5).

11.5.2.3.2 Service Water Monitor

The service water monitor is located at the inlet to the circulating water standpipe prior to discharge to the cooling lake. The location of this monitor permits detection of radioactive input to the service water from sources which could conceivably be contaminated by radioactive materials. It monitors the effluent being discharged from these systems to the cooling lake via the circulating water discharge flume. If an alarm condition exists, discharge line closure may be initiated manually. Monitor ranges and sensitivities are given in Table 11.5-4.

11.5.2.3.3 Radwaste Effluent Radiation Monitor System

This monitor subsystem measures the radioactivity in the radwaste effluent discharge line before mixing with the lake blowdown flow prior to entering the river.

Liquid waste can be discharged via the discharge tanks from several radwaste processed water tanks such as the floor drain sample tanks, waste sample tanks, or laundry drain sample tanks. These tanks contain the effluents of water that has been processed through one or more treatment systems such as evaporation, filtration, ion exchange and state-of-the-art vendor skids. Prior to discharge to the lake blowdown from the discharge tanks, the liquid is sampled and analyzed for radioactivity. Based upon this analysis, discharge is permitted at a specified release rate and dilution rate.

The detector is located on a sample line which takes a sample from the radwaste discharge line and returns it to the radwaste discharge line. The piping arrangement is designed so that the section of pipe in which the well is located can be flushed.

The system is equipped with a trip point which alarms and initiates automatic valve closure on the radwaste discharge line to prevent the violation of 10 CFR 20 limits for drinking water in the river downstream of the plant. Monitor ranges and sensitivities are given in Table 11.5-4.

11.5.2.3.4 RHR Heat Exchanger Cooling Water Effluent Monitor System

A process radiation monitor is located on the cooling water piping downstream of each of the two residual heat removal (RHR) heater exchangers (loops A and B). These monitors are designed to detect any radioactive leakage into the RHR cooling water during operation of the RHR heat exchangers.

The two RHR heat exchangers are 100% rated for reactor shutdown operation so that if a leak were to develop in one heat exchanger, the redundant unit could be brought into service by the control room operator.

During operation, the RHR heat exchanger cooling water system takes suction from the cooling lake via the CSCS water tunnel, and the discharge is returned to the lake.

The individual detectors are alarmed and annunciated in the main control room. Monitor details are given in Table 11.5-4.

11.5.2.3.5 Reactor Building Closed Cooling Water Monitor

The gamma-sensitive detector for this system is inserted in a well on the closed cooling water piping side of the RBCCW heat exchangers. Service water on the coolant side of the RBCCW is at a slightly higher pressure than the RBCCW, thus minimizing the potential for outleakage of the potentially contaminated closed cooling water.

The monitor system is a diagnostic tool to verify that no inleakage of primary plant water has occurred from the reactor water cleanup system nonregenerative heat exchanger system which uses RBCCW as its coolant. Since the RBCCW is a closed system, inleakage would be detected by the monitor system as well as surge tank level increase. Spillage or overflow from the RBCCW is directed to the radwaste system for processing and reuse.

Monitor details are given in Table 11.5-4.

11.5.2.4 Sampling

The following subsections present a detailed description of the radiological sampling procedures, frequencies, and objectives for all plant process and effluent sampling. This sample program provides the means to show compliance with the ODCM requirements for the process radiation monitoring system and radwaste system.

11.5.2.4.1 Process Sampling

Chapter 9.0 describes the design of sampling facilities provided for general sampling. Subsection 12.3.4 describes the in-plant airborne sampling system. The sample frequency, type of analyses, analytical sensitivity, and purpose of the sample are summarized in Table 11.5-5 for each liquid process sample location, and in Table 11.5-6 for each gas process sample location. The analytical procedures used in sample analysis are presented in Subsection 11.5.2.4.3. These samples serve to monitor radioactivity levels within various plant systems.

11.5.2.4.2 Effluent Sampling

Effluent sampling of all potentially radioactive liquid and gaseous effluent paths is conducted on a regular basis in order to verify the adequacy of effluent processing to meet the discharge limits to offsite areas. This effluent sampling program provides the information for the effluent measuring and reporting programs required by 10 CFR 50.36a. The frequency of the periodic sampling and analysis described herein is normal and may be increased if effluent levels approach ODCM limits. Tables 11.5-7 and 11.5-8 summarize the sample and analysis schedules.

11.5.2.4.3 Analytical Procedures

Samples of process and effluent gases and liquids are analyzed in the station laboratory or by an outside laboratory by means of the following techniques:

a. beta counting,

b. alpha counting,

c. gamma spectrometry, and

d. liquid scintillation counting.

Instrumentation which is available in the laboratory for the measurement of radioactivity at the time of initial fuel loading includes the following:

a. alpha/beta counter,

b. gamma spectrometer, and

c. liquid scintillation counter.

"Available" instrumentation and counting techniques change as other instruments and techniques become available. For this reason the frequency of sampling and the analysis of samples are generalized here but specifically identified in the station procedures. The following treatment is included as typical of those currently used at EGC generating stations.

Samples for beta and alpha counting are evaporated to dryness on planchets prior to counting on a proportional counter. Sample volume, counting geometry, and counting time are chosen to achieve the optimum measurement sensitivities. Correction factors are applied for sample-detector geometry, self-absorption, and detector efficiency.

Gaseous tritium samples are collected by condensations or adsorption, and the resultant liquid is analyzed by liquid scintillation counting techniques.

Radiochemical separations are used for the routine analysis of Sr-89 and Sr-90.

Liquid samples are collected in polyethylene bottles to minimize adsorption of nuclides onto container walls.

11.5.2.5 Instrument Inspection, Calibration, and Maintenance

During reactor operation, daily checks of effluent monitoring system operability are made by observing channel behavior. Routinely during reactor operation, the detector response is observed with a remotely positioned check source supplied with most of the monitors. Instrument background count rate is also observed to ensure proper functioning of the monitors. Any detector whose response cannot be verified by observation during normal operation or by using the remotely positioned check source will have its response checked with a portable check source. A record is maintained showing the background radiation level and the detector response.

The system has electronic testing and calibrating equipment which permits channel testing without relocating or dismounting channel components. An internal trip test circuit, adjustable over the full range of the readout meter, is used for testing.

Each channel is tested at least semiannually prior to performing a calibration check. Verification of valve operation ventilation diversion, or other trip action is done routinely if it can be done without jeopardizing the plant safety. The tests are documented.

11.5.2.5.1 Calibration

Calibration of the continuous radiation monitors is done with radionuclide standards reference to the National Bureau of Standards. Calibration intervals are detailed in the Technical Specifications, TRM, and ODCM.

11.5.3 Effluent Monitoring and Sampling

The implementation of the requirements of General Design Criterion 64 concerning monitoring of effluent discharge paths for radioactivity is covered in Subsections 11.5.2.2 and 11.5.2.3. Subsections 11.5.2.2.1 and 11.5.2.2.2 provide applicable discussions for gaseous effluent radiation monitors. Subsections 11.5.2.3.2 through 11.5.2.3.5 provide applicable discussions for liquid effluent radiation monitors.

11.5.4 Process Monitoring and Sampling

The implementation of the requirements of General Design Criteria 60 and 63 concerning automatic closure of isolation valves in gaseous and liquid effluent discharge paths (Criterion 60) and monitoring of radiation levels in radioactive waste process systems (Criterion 63) is covered in Subsections 11.5.2.1 and 11.5.2.3. Subsections 11.5.2.1.1 through 11.5.2.1.6 provide applicable discussions for gaseous process radiation monitors. Subsections 11.5.2.3.4 and 11.5.2.3.5 provide applicable discussions for liquid process radiation monitors.

11.5.5 High Radiation Sampling System (HRSS)

License Amendments No. 158 and No. 144 for LaSalle County Station, Units 1 and 2, respectively approve the elimination of the requirement to have and maintain the Post Accident Sampling System (or HRSS). The following items were committed to as part of license Amendments 144 and 158.

- LaSalle County Station has developed contingency plans for obtaining and analyzing highly radioactive samples of reactor coolant, suppression pool, and containment atmosphere. The contingency plans will be contained in the LaSalle County Station chemistry procedures and implemented with the implementation of the license Amendments. Establishment of contingency plans is considered a regulatory commitment.

- The capability for classifying fuel damage events at the Alert level threshold will be established at a level of core damage associated with radioactivity levels of 300 micro-curies per milliliter (ml) dose-equivalent iodine. This capability will be described in emergency plans and emergency plan implementing procedures and implemented with the implementation of the license Amendments. The capability for classifying fuel damage events is considered a regulatory commitment.

- LaSalle County Station has established the capability to monitor radioactive iodines that have been released offsite to the environs. This capability is described in emergency plans and emergency plan implementing procedures. The capability to monitor radioactive iodines is considered a regulatory commitment.

The information contained in UFSAR Section 11.5.5 regarding the regulatory requirements for post accident sampling is retained for historical purposes.

11.5.5.1 General System Description

The capability to obtain a sample for radioisotopic and chemical analyses of the reactor coolant and the containment atmosphere samples is provided by the High Radiation Sampling System (HRSS).

The system provides the capability to obtain samples from either Unit 1 or Unit 2 under degraded core accident conditions without excessive exposure.

The P&ID of the HRSS system is shown in Drawing Nos. M-156 (sheet 5) and M-115 (sheets 12 through 15). The system is installed in the auxiliary building (Drawing Nos. M-11 and M-12) and consists of a liquid sampling subsystem and an air sampling subsystem. The major components of the system are:

a. HRSS liquid sample panel;

b. containment air sample panel;

c. a cooling rack for thermally hot liquid samples;

d. an independent HVAC system;

e. a waste system for the HRSS to prevent wholesale contamination of secondary systems outside the primary containment;

f. valves and piping for the new sampling system;

g. an independent communication system to the control room; and,

h. controls for the entire system.

The actual sampling panels, the HVAC system and controls are installed at elevation 687 feet 6 inches (upper basement level) and the waste equipment (e.g., waste pumps, waste tank, etc.) are installed at elevation 663 feet 0 inch (basement level). The upper basement sampling room has shielded access independent from the reactor building proper and will allow removal of samples without excessive exposure to personnel. The access path to the HRSS is not common with the control room access path.

11.5.5.2 Liquid Sampling Subsystem

The HRSS liquid sampling panel is capable of sampling:

a. reactor coolant from the discharge side of the recirculation pump in the B recirculation loop;

b. reactor coolant from the discharge side of the residual heat removal heat exchangers (A and B);

c. reactor coolant from the discharge side of the cleanup nonregenerative heat exchangers before entering the reactor water cleanup demineralizers;

d. reactor coolant from the discharge side of the reactor water cleanup demineralizers (A, B, and C);

e. water from the drywell equipment drain sump;

f. water from the drywell floor drain sump; and

g. water from the HRSS tank.

Items c, d, e, f and g are systems that are not required for the implementation of the post accident sampling program.

Additional analysis capabilities were designed into the HRSS system. These capabilities are not required by the post accident sampling program and are no longer maintained.

11.5.5.4 Air Sampling Subsystem

The containment air sampling panel is capable of sampling:

a. air from the drywell;

b. air from the suppression chamber; and

c. air from the reactor building.

The containment air sampling panel utilizes a gas partitioner to capture an air sample and separate it into its gaseous, iodine, and particulate constituents. The iodine and particulate filters and gaseous samples will be managed based on radiation field conditions.

11.5.5.6 Offsite Chemical Analysis Capability

A contract is in place with an offsite vendor to provide backup chemical analysis capability if needed. Analysis results will be reported within 24 hours upon the vendor's receipt of the samples.

TABLE 11.5-1

PROCESS RADIATION MONITORING SYSTEM
ENVIRONMENTAL DESIGN CONDITIONS

RADIATION MONITOR SYSTEM	PRESSURE (psig)	TEMPERATURE (°F)	RELATIVE HUMIDITY (%)
Main steamline detectors	0 to 250	392 max.	--
Off-gas monitor sample systems	0 to 375	480 max.	--
All remaining subsystems, equipment, and instruments	0	32 to 140	10 to 98

TABLE 11.5-2

PROCESS RADIATION MONITORING SYSTEM (GASEOUS AND AIRBORNE MONITORS)

MONITOR	CONFIGURATION	TYPE	EXPECTED SENSITIVITY*	RANGE	PRINCIPAL RADIONUCLIDES MEASURED	EXPECTED ACTIVITY	ALARMS AND TRIPS
Off-gas post-treatment radiation monitor (GE)	Sample line between charcoal bed and discharge valve	Particulate filter Iodine filter Scintillator	10^{-6}-10^{-2} µCi/cm^3	10^{-1} to 10^6 cpm	Noble gas	Off-gas activity defined in Table11.3-1	Flow Low/High Downscale High High-High High-High-High Off-gas system outlet & drain isolated
Off-gas radiation monitor Pretreatment gas-log scale (GE)	Adjacent to sample chamber	γ-Ion chamber	3 x 10^{-10} Amp/R/hr	10^0 - 10^6 mR/h	Noble gas Fission products	Off-gas activity defined in Table 11.1-1	Downscale High High-High High-Low Flow
Main steamline radiation monitor (GE)	Adjacent to steamlines	γ-Ion chamber	3 x 10^{-10} Amp/R/hr	10^0 - 10^6 mR/hr	Coolant activation gases Noble gas	Steamline activity defined in Table 11.1-4	Downscale High High-High
Carbon bed vault radiation monitor	Inside vault	G-M	1m/hr	10^0 - 10^6 mR/hr	Fission products in carbon vaults	Approximately 100R/hr	High Downscale
Reactor Building exhaust plenum radiation monitor (GE) Air (N407)	Inline	G-M	250 cpm/pCi/cm^3	0.01-100 mR/hr	Noble gases	Reactor building activity defined in Table 11.3-8	Downscale High-High Isolate RB Vent System
Fuel pool ventilation exhaust monitor	Offline	G-M	250 cpm/pCi/cm^3	0.01-100 mR/hr	Noble gases	5 x 10^{-8} µCi/cc during power operation 5 x 10^{-6} µCi/cc during refueling	Downscale High-High Isolate RB Vent System

* Expected sensitivity is typical only. Actual sensitivity is to be determined as operational data becomes available.

TABLE 11.5-3

EFFLUENT RADIATION MONITORING SYSTEM (GASEOUS AND AIRBORNE MONITORS)

MONITOR	CONFIGURATION	TYPE	EXPECTED SENSITIVITY	RANGE	PRINCIPAL RADIONUCLIDES MEASURED	EXPECTED ACTIVITY	ALARMS
Station Vent Stack monitor (Wide Range)	Sample line from plant stack	Particulate filter Iodine filter Gas (low range) Plastic scintillator Gas (mid range) Solid state scintillator Gas (High range) Solid state scintillator	N/A N/A 4.32×10^{7} cpm/µCi/cc 2.84×10^{4} cpm/µCi/cc 1.19×10^{2} cpm/µCi/cc	N/A N/A 10^{-7} - 10^{-1} µCi/cc 10^{-4} - 10^{+2} µCi/cc 10^{-1} - 10^{+5} µCi/cc	Noble gases, Iodine and particulate	Gaseous discharge activity defined in Table 11.3-1	High INOP ALERT
Standby Gas treatment system effluent (Wide Range)	Sample lines from SGTS stack	Particulate filter Iodine filter Gas (low range) Plastic scintillator Gas (mid range) Solid state Scintillator Gas (high range) Solid state Scintillator	N/A N/A 4.32×10^{7} cpm/µCi/cc 2.84×10^{4} cpm/µCi/cc 1.19×10^{2} cpm/µCi/cc	N/A N/A 10^{-7} - 10^{-1} µCi/cc 10^{-4} - 10^{+2} µCi/cc 10^{-1} - 10^{+5} µCi/cc	Noble gases, Iodine and particulate	Reactor building activity defined in Table 11.3-8	High INOP

TABLE 11.5-4

PROCESS AND EFFLUENT RADIATION MONITORING SYSTEM (LIQUID MONITORS)

MONITOR	CONFIGURATION	TYPE	RANGE	PRINCIPAL RADIONUCLIDES MEASURED	EXPECTED ACTIVITY	ALARMS
Radwaste effluent radiation monitor	Sample line from radwaste discharge to blowdown line	γ-Scintillator	10^1 - 10^6 cpm	Cs-137* Co-60	Discussed in Section 11.2	High High/Low Flow Downscale Isolate
Service water effluent radiation monitor	Sample line from effluent pipe prior to discharge to other systems	γ-Scintillator	10^1 - 10^6 cpm	Cs-137* Co-60	Less than minimum detector sensitivity	INOP/Downscale High/Low Flow High
Reactor building closed cooling water radiation monitor	Sample line from effluent	γ-Scintillator	10^1 - 10^6 cpm	Cs-137* Co-60	Less than minimum detector sensitivity	INOP/Downscale High High/Low Flow
Residual heat removal service water radiation monitor (RHR - 2 each)	Sample line from effluent	γ-Scintillator	10^1 - 10^6 cpm	Cs-137* Co-60	Less than minimum detector sensitivity	Low/High Flow INOP/Downscale High

* Sensitivity based upon this radionuclide.

TABLE 11.5-5

RADIOLOGICAL ANALYSIS SUMMARY OF LIQUID PROCESS SAMPLES

SAMPLE DESCRIPTION	TYPICAL SAMPLE FREQUENCY*	ANALYSIS	EXPECTED SENSITIVITY** (μCi/ml)	PURPOSE
1. Reactor Coolant Iodine Isotopic Analysis Isotopic Analysis	 1/Week 1/Week	 Dose Equivalent I-131 Principle Gamma Emitters	 10^{-6} 5×10^{-7}	 Evaluate reactor water activity
2. Condensate storage tank A	Weekly ***	CS-137/Ba-137m Equivalent Activity	10^{-6}	Activity Monitoring
3. Condensate storage tank B	Weekly ***	CS-137/Ba-137m Equivalent Activity	10^{-6}	Activity Monitoring
4. Fuel pool filter-demineralizer Inlet and Outlet	Periodically when fuel is present ***	CS-137/Ba-137m Equivalent Activity	10^{-6}	Evaluate system performance

* Actual frequency is determined by plant needs and operational circumstances per the Technical Specifications.

** Expected sensitivity is typical only. Actual sensitivity to be determined as operational data become available.

*** NOT required per Technical Specifications

TABLE 11.5-6

RADIOLOGICAL ANALYSIS SUMMARY OF GASEOUS PROCESS SAMPLES

SAMPLE DESCRIPTION	TYPICAL SAMPLE FREQUENCY	TYPICAL ANALYSIS	EXPECTED SENSITIVITY** (µCi/ml)	PURPOSE
1. Containment atmosphere (drywell)	****Prior to entry	****Beta*	10^{-11} 10^{-6}	Determine need for respiratory equipment
2. Off-gas pretreatment sample	****Monthly	****Gamma isotopic	10^{-7}	Determine off-gas activity
3. Off-gas post treatment sample	Monthly ***	Gamma isotopic***	10^{-7}	Determine noble gas activity

* On particulate filter
** Expected sensitivity is typical only. Actual sensitivity to be determined as operational data becomes available.
*** As determined by plant needs and operational circumstances per the Technical Specifications.
**** Not required by Technical Specifications.

TABLE 11.5-7

RADIOLOGICAL ANALYSIS SUMMARY OF LIQUID EFFLUENT SAMPLES

SAMPLE DESCRIPTION	TYPICAL SAMPLE FREQUENCY*	TYPICAL ANALYSIS*	EXPECTED SENSITIVITY** (µCi/ml)	PURPOSE
1. River Discharge Tank	Each Batch	Cs-137/Ba-137m Equivalent Activity	10^{-6}	Radwaste Discharge
2. Liquid radwaste effluent Composite of all tanks discharged	Monthly Quarterly	Tritium Sr-89/90	10^{-5} 10^{-8}	
3. Circulating water	Weekly sample continuously collected proportional sample Monthly	Gamma Isotopic Tritium	10^{-7} 10^{-5}	Effluent discharge record (backup sample)

* As determined by plant needs and operational circumstances.

** Expected sensitivity is typical only. Actual sensitivity is to be determined as operational data becomes available.

TABLE 11.5-8

RADIOLOGICAL ANALYSIS SUMMARY OF GASEOUS EFFLUENT SAMPLES

SAMPLE DESCRIPTION	TYPICAL SAMPLE FREQUENCY***	TYPICAL ANALYSIS***	SENSITIVITY (μCi/ml)†	PURPOSE
1. Station vent stack	Weekly	I-131**, gamma isotopic	10^{-11} 10^{-10}	Effluent record
	Monthly	Gamma isotopic*	10^{-8}	
	Quarterly	Sr-89 and 90* Tritium	10^{-11} 10^{-10} 10^{-16}	

* On particulate filter
** On Charcoal cartridge
*** As determined by plant needs and operational circumstances.
† Expected sensitivity is typical only. Actual sensitivity to be determined as operational data becomes available.

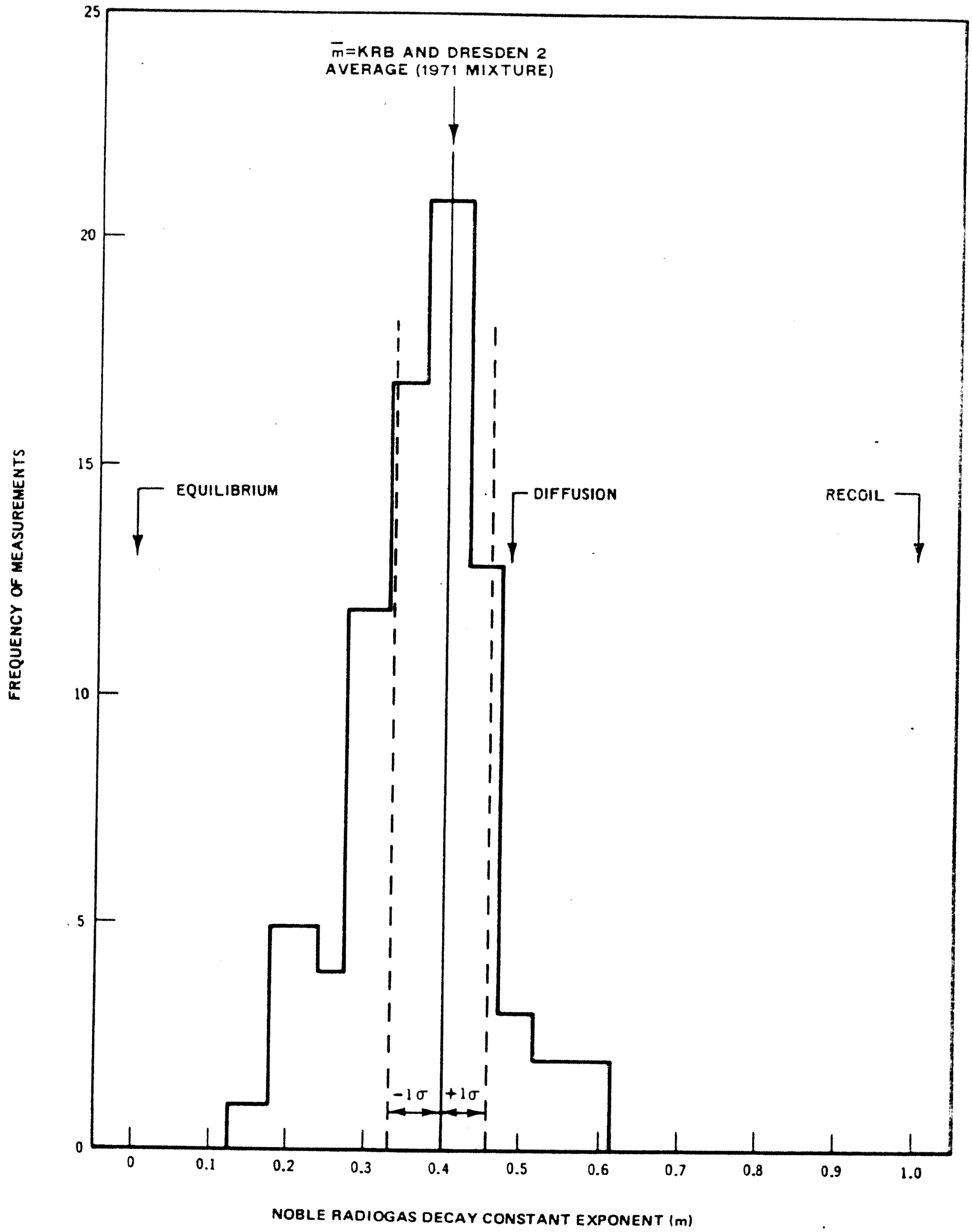

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FIGURE 11.1-1

NOBLE RADIOGAS DECAY CONSTANT EXPONENT FREQUENCY HISTOGRAM

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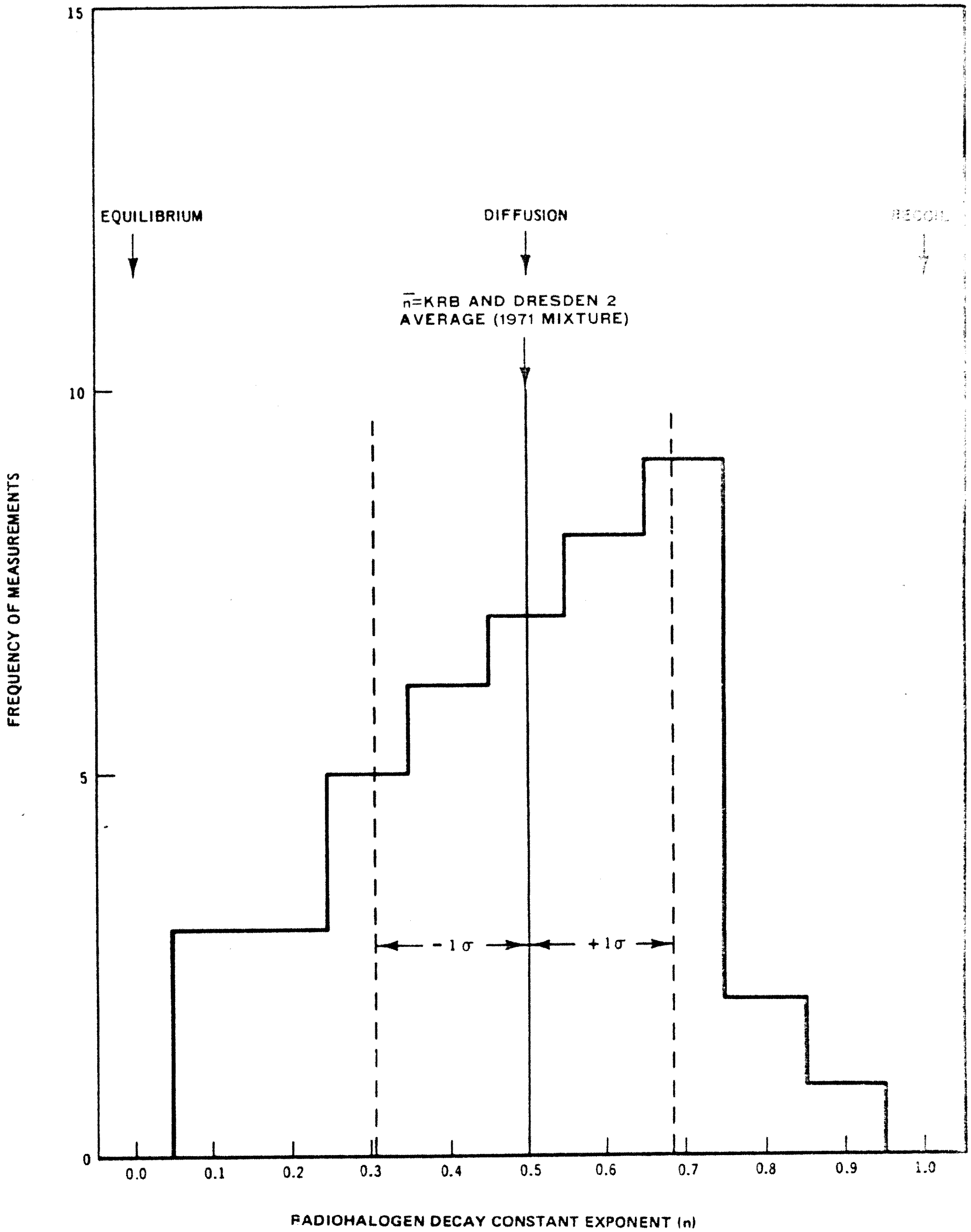

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FIGURE 11.1-2

RADIOHALOGEN DECAY CONSTANT EXPONENT
FREQUENCY HISTOGRAM

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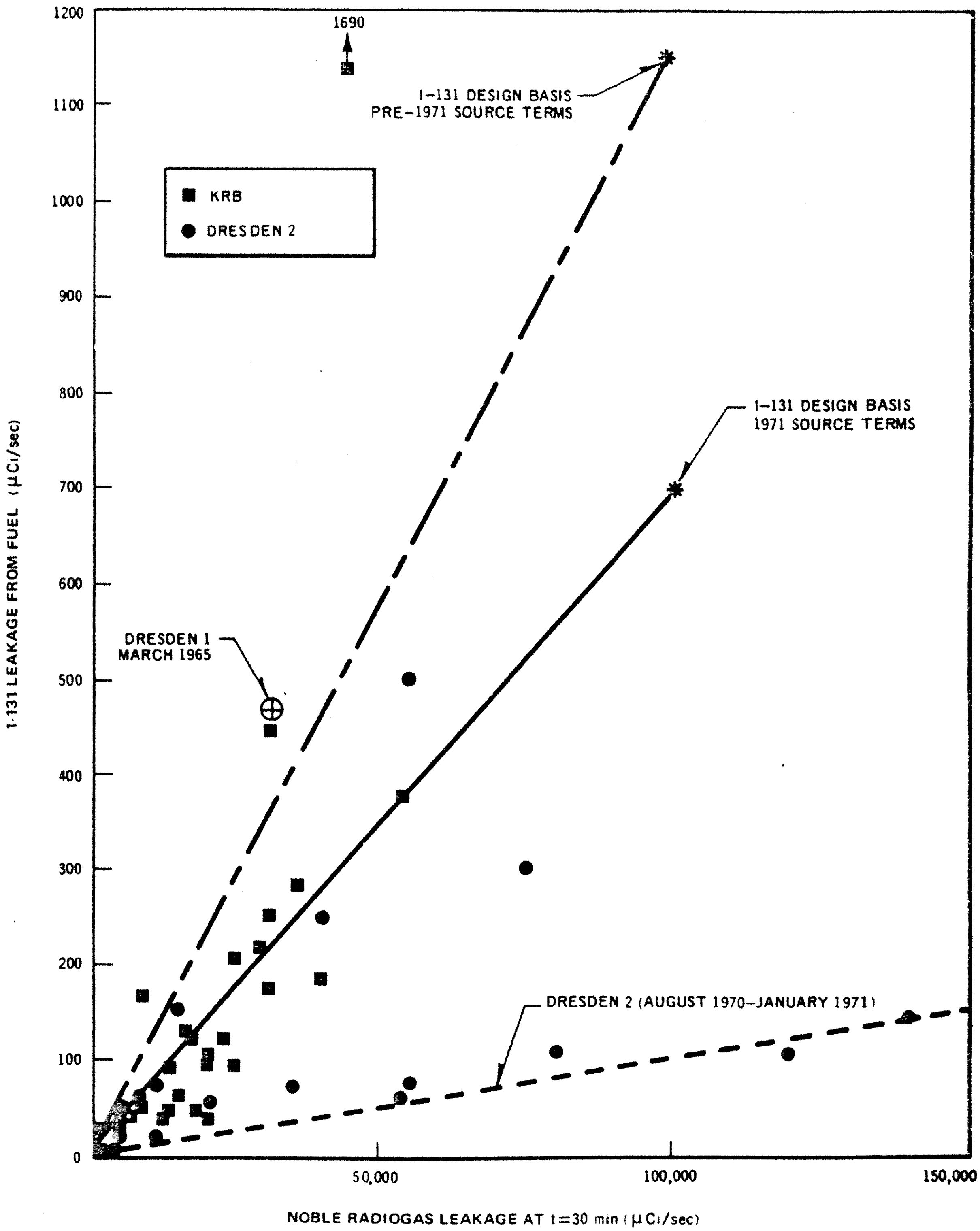

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FIGURE 11.1-3

NOBLE RADIOGAS LEAKAGE VS.
I-131 LEAKAGE

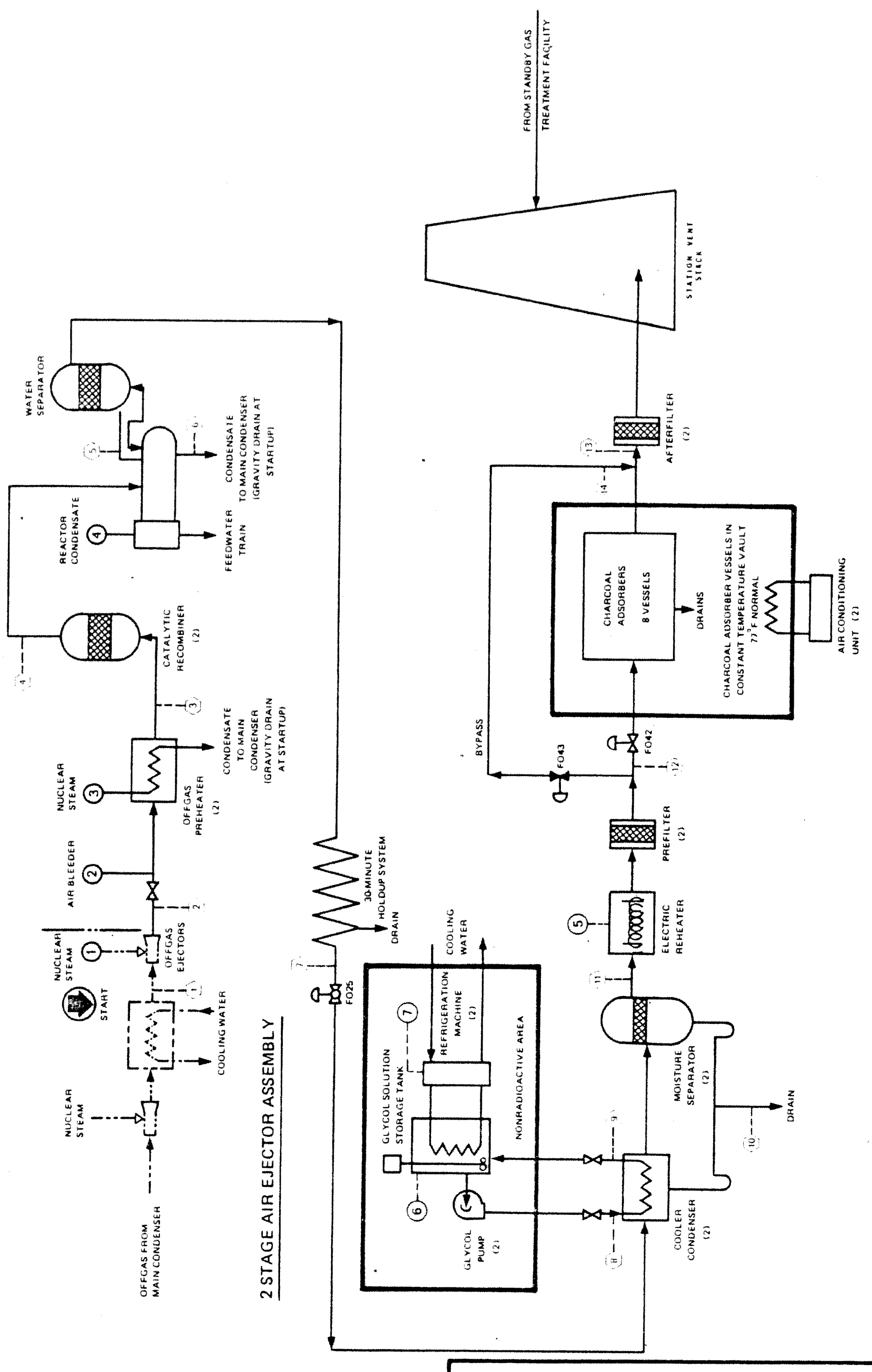

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FIGURE 11.3-1

OFF-GAS SYSTEM FLOW DIAGRAM

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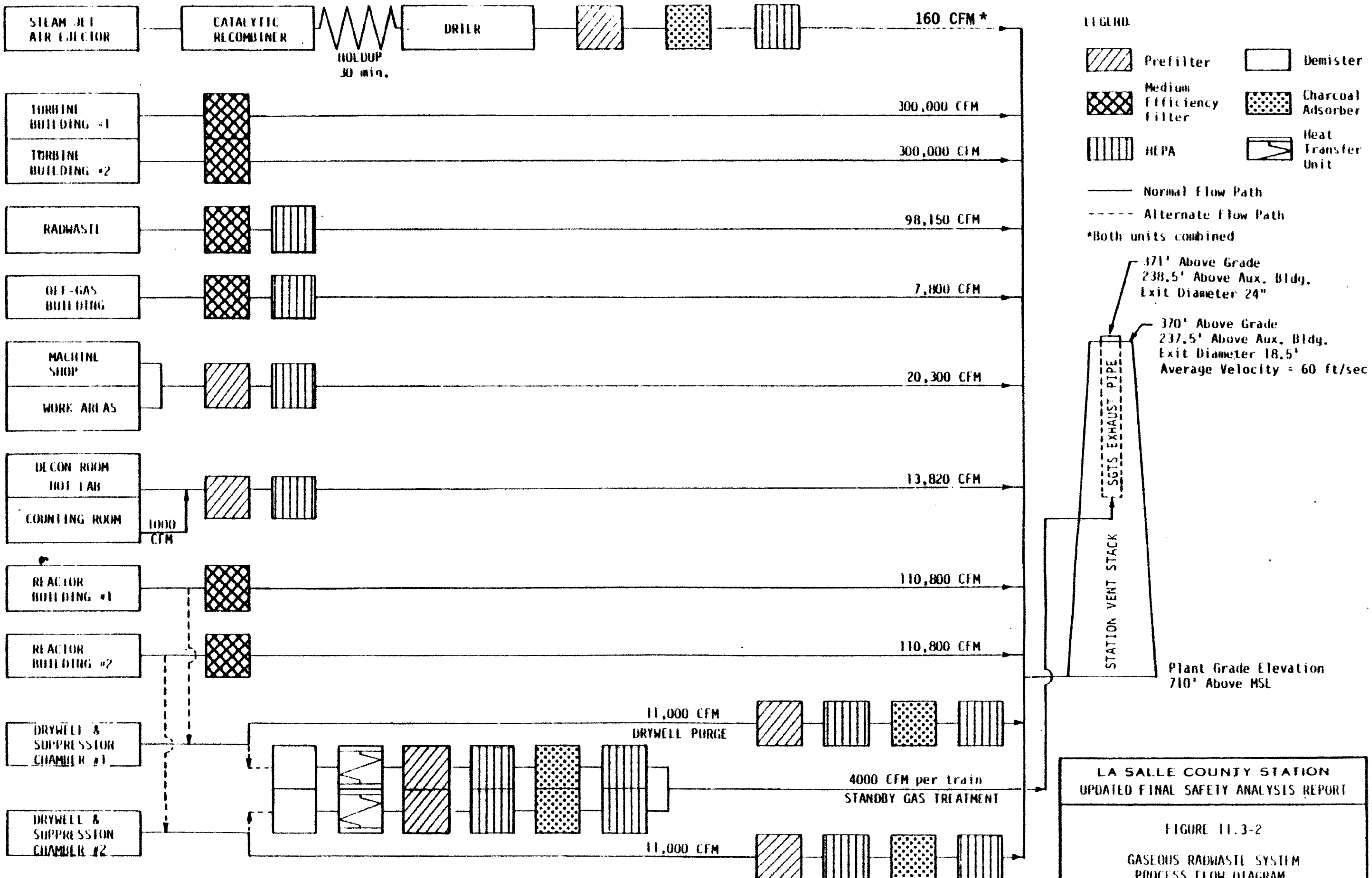

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FIGURE 11.3-2

GASEOUS RADWASTE SYSTEM
PROCESS FLOW DIAGRAM

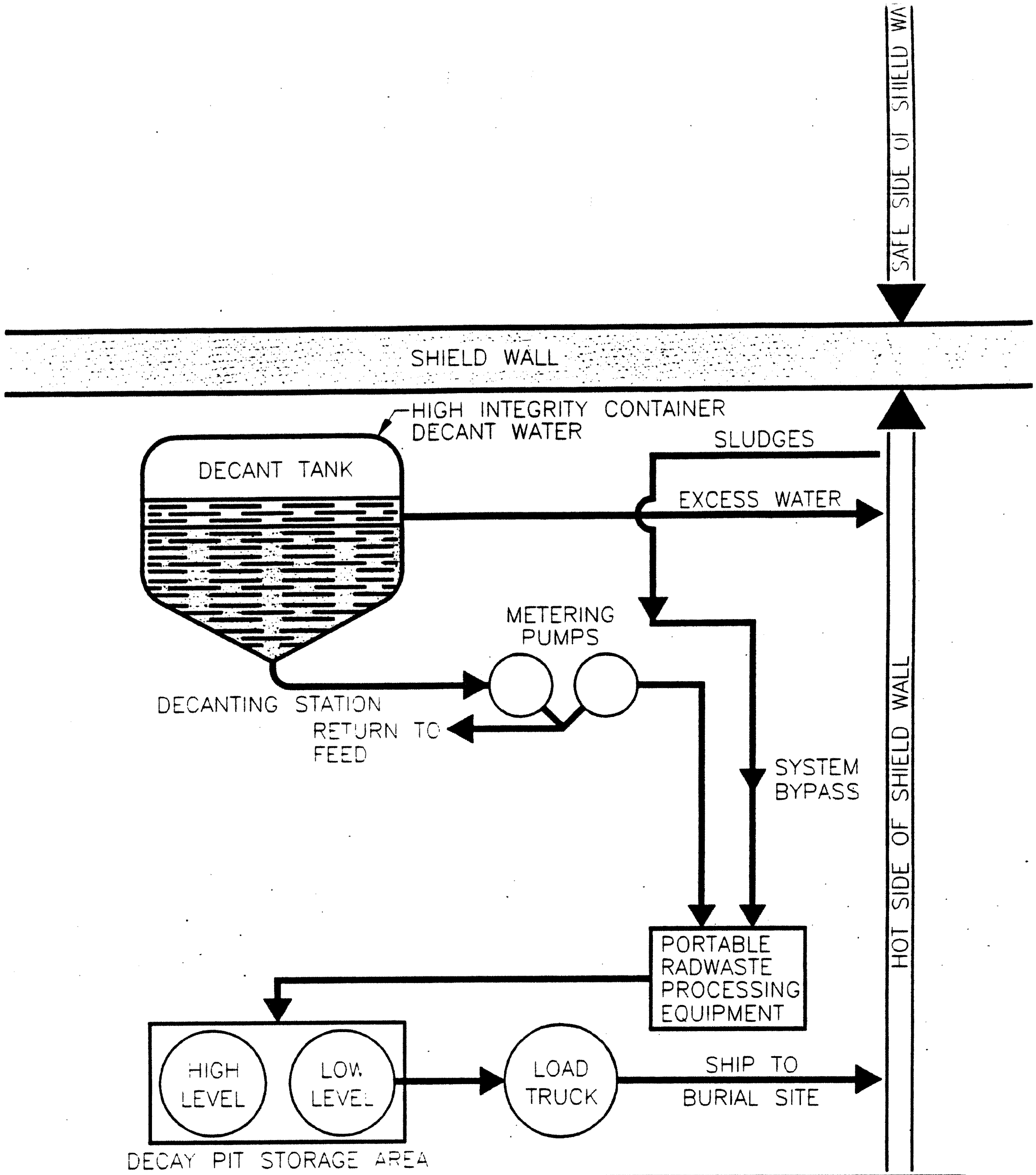

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FIGURE 11.4-1

RADWASTE DISPOSAL SYSTEM
FLOW DIAGRAM

REV. 12 - MARCH 1998