RADIOACTIVE WASTE MANAGEMENT

CHAPTER 11

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11.0 RADIOACTIVE WASTE MANAGEMENT

The Waste Management Systems (WMS) are designed to collect, monitor, and process all liquid, gaseous and solid radioactive wastes originating from the plant operation. The principal design objective is to protect plant personnel, the general public and the environment by assuring that all releases of radioactive materials both in plant and to the environs, are in accordance with the regulations of 10 CFR 20 and 50. To best accomplish this objective, functions are divided and solids, liquids and gases are handled separately. Thus, the WMS consists of three subsystems. The liquid wastes are handled via the Liquid Waste Management System (LWMS), the gaseous wastes are handled via the Gaseous Waste Management System (GWMS), and the solid wastes are handled via the Solid Waste Management System (SWMS).

A general design objective is to provide an ability to monitor, control and treat all potentially radioactive plant liquid and gaseous effluents to assure that their particulate and dissolved content complies not only with Federal standards, but also with state and local air and water quality standards where applicable.

The Waste Management Systems are designed to reduce operator radiation exposure to as low as is reasonably achievable (ALARA).

11.1 SOURCE TERMS

11.1.1 DESIGN BASIS SOURCE TERMS

The development of the RCS activity at EPU conditions used for the AST radiological assessments of Chapter 15 (Table 15.0-32a) is discussed in Section 11.1.6.

11.1.1.1 Fission Product Activities in Reactor Coolant (Maximum Values)

Maximum fission product activities are based on one percent failed fuel and are used as design basis source terms for shielding and facilities design and for calculating the consequences of postulated accidents. The isotopes chosen for consideration in the maximum case are those which are significant for design purposes by reason of a combination of energy, half-life, and/or abundance.

The mathematical model used to determine the maximum concentration of nuclides in the Reactor Coolant System (RCS) involves a group of linear, first order differential equations. These equations are obtained by applying a mass balance for production and removal of fission products from the fuel pellet region as well as the reactor coolant region.

In the fuel pellet region the production sources for a given fission product are;

a. Direct Fission Yield

b. Parent Fission Product Decay

c. Neutron Activation.

The removal paths for a given fission product are:

a. Decay

b. Neutron activation

c. Escape to coolant.

In the reactor coolant region the production sources for a given fission product are:

a. Escape from the fuel through defective fuel rod cladding

b. Parent decay of reactor coolant fission products

c. Neutron activation of reactor coolant fission products.

The removal paths for a given fission product are:

a. Decay

b. Reactor coolant purification

c. Feed and bleed for fuel burnup

d. Leakage and other feed and bleed operations (i.e., startups and load follow operations).

The expression utilized in determining the fission product inventory in the fuel pellet region is:

$$\frac{dN_{p,i}}{dt} = (F)(Y_i)(P) + (f_{i-1}\lambda_{i-1} + \sigma_{i-1}\phi)N_{p,i-1} - (\lambda_i + v_i + \sigma i\phi)N_{\rho,i} \quad (1)$$

The expression for the fission product inventory in the reactor coolant region is:

$$\frac{dN_{c,i}}{dt} = (D)(v_i)(N_{\boldsymbol{\rho},i}) + (f_{i-1}\boldsymbol{\lambda}_{i-1} + \boldsymbol{\sigma}_{i-1}\boldsymbol{\phi}.CVR)(N_{c,i-1}) - \left(\lambda_i + \frac{Q}{W}n_i + \frac{(1-n_i)C}{C_o - C_t} + \frac{L}{W}\right)N_{c,i} \quad (2)$$

where the variables are identified as:

N = Population, atoms

F = Average fission rate, fissions/MWt-sec.

Y = U-235 fission yield of nuclide, fraction(1)

P = Core power, MWt

λ = Decay constant, sec. $^{-1}$ (2) (3)

σ = Microscopic capture cross-section, cm^2 (3) (4)

$\emptyset$ = Thermal neutron flux, neutrons/cm$^{(2)}$- sec.

v = Escape rate coefficient, sec.$^{-1}$

f = Branching fraction

t = Time, sec.

D = Fraction of failed fuel

CVR = Core coolant volume to Reactor Coolant System volume ratio, fraction

Q = CVCS purification flowrate during power operation, lbm/sec.

w = RCS mass during power operation, lbm

n = Resin efficiency of CVCS ion exchanger and flash tank efficiency (subscripted for a particular nuclide)

C_o = Beginning of core life boron concentration, ppm

C = Boron concentration reduction rate because of feed and bleed, ppm/sec.

L = Leakage or other feed and bleed from the reactor coolant, lbm/sec and where the subscripts are identified as:

i = i^{th} nuclide

i-1 = precursor to i^{th} nuclide

p = pellet region

c = reactor coolant region

Escape rate coefficients (v_i) are used to represent the overall release from the fuel pellets to the reactor coolant. The escape rate coefficient for a given isotope is an empirical value derived from experiments initiated by Bettis Atomic Power Laboratory and run in the NRX and MTR reactors. The escape rate coefficients were obtained from test rods which were operated at high linear heat rates. The linear heat rates were uniform over the 10.25 inch long test sections. The exact linear heat rates were not precisely known but postirradiation inspection showed that some test specimens had experienced centerline melting. Later tests were done in Canada(5) to determine the effect of rod length on the release of fission gases and iodines from defective fuel rods. A byproduct of these experiments was the relationship between linear heat rate and escape rate coefficient. The conservative escape rate coefficients are given in Table 11.1-1 for noble gases, halogens, and other elements.

The values of the parameters used to calculate the reactor coolant maximum fission product activities are shown in Table 11.1-1. The maximum fission product activities are presented in Table 11.1-2 are for 2700 MW operation. RCS activities shown in Table 11.1-3 are for 2560 MW operation.

11.1.1.2 Maximum Corrosion Product (Crud) Activities in Reactor Coolant

The activity of radioactive crud and the crud thickness on Reactor Coolant System surfaces are evaluated using measured data from various operating pressurized water reactors.

Even though the reactors have different water chemistries and different materials in contact with the reactor coolant, the crud activity (dpm/mg-crud), crud film thicknesses and dose rates due to crud are remarkedly similar. The half-lives, parent nuclide reactions, and gamma decay energies for each of the long-lived isotopes, (the long-lived isotopes are defined as those isotopes remaining after 48 hours of decay) in the radioactive crud are shown in Table 11.1-4.

The radioactive crud originates on both in-core and out-of-core surfaces. The crud plates out on the in-core surfaces and re-erodes after a short irradiation period. This irradiation period or core residence time for each isotope is determined by the following equations (see Appendix 11.1A for the derivation of these equations):

Circulating Crud:

$$t_{res} = -\frac{1}{\lambda}\ln\left[1 - \frac{A_i A_T (16.67)}{\sum_c \phi C}\right] \text{ (sec)} \qquad (3)$$

Deposited Crud:

$$t_{res} = -\frac{1}{\lambda}\ln\left[1 - \frac{A_j (16.67)}{\sum_c \phi}\right] \text{ (sec)} \qquad (4)$$

where:

A_i, A_j + the crud activities for each isotope (dpm/mg-crud)

$\sum_c \phi$ = the activation rate (d/gm-sec)

C = the core surface area (cm^2)

λ = decay constant (sec^{-1})

A_T = total reactor coolant system area (cm^2)

16.67 = units conversion (min-gm/sec.-mg)

The activation cross-section Σ_i is as follows:

$$\sum_i = \frac{\left(\frac{a}{i}\right)_i \left(\frac{w}{o}\right)_i N_o \sigma_i}{[A]_i}, \text{ cm}^2/\text{gm} \quad (5)$$

where:

$(a/o)_i$ = the isotopic abundance

$(w/o)_i$ = the elemental abundance in the crud or the elemental abundance in the bass metal

N_o = Avogadro number (0.6023×10^{24} atoms/gm-mole)

σ_i = the microscopic cross-section (cm^2)

$[A]_i$ = the atomic weight of isotope (i)

Circulating crud is taken to be all crud in the reactor coolant. Deposited crud is taken to be all crud which plates out on in-core surfaces.

Table 11.1-5 lists the average and maximum crud activities (based on Reference 6 through 19) for various operating reactors.

Table 11.1-6 indicates the activation rates for crud on a plant-by-plant basis for those reactors.

Table 11.1-7 presents the average and maximum core residence times for crud (based on the data of Tables 11.1-5 and 11.1-6) as calculated from the preceeding two equations. (Here Fe-59 residence times are long enough to permit its activity to be assumed saturated).

The crud activities based on a power level of 2700 Mwt can be calculated using the following equation:

$$A_i = \sum_i \phi \left(1 - e^{-\lambda_i t_{res}}\right) \frac{A_c}{A_T} (0.06), dpm / mg .crud \quad (6)$$

where:

(A_c / A_T) = is the ratio of core surface area to total RCS surface area.

$\sum_i \phi$ = is the activation rate, values for which are given in Table 11.1-8.

The (t_{res}) values used in the above equation are calculated by averaging the data given in Table 11.1-7. The maximum residence time values of Table 11.1-7 are generally a factor of 2 to 4 times higher than the average values.

Therefore, for conservatism, the maximum values are the basis for the (t_{res}) figures used. For each nuclide, (t_{res}) is the average of the maximum values. The resulting calculated crud

activities are conservative. The data in Table 11.1-9 is applicable to both the deposited crud and the circulating crud.

For circulating crud, the activity in the reactor coolant is calculated as follows:

$$A - \frac{A_i}{60}\left(75x10^{-9}\right) \quad \rho(2.7x10^{-5})\ 1 \times 10^3,\ \mu Ci/cm^3 \tag{7}$$

Where:

ρ= is density of water (g/cm^3) and 1 x 10^3 is mg/gm

2.7 x 10^{-5} converts disintegrations/sec to micro-curies.

The crud concentration in the reactor coolant is taken to be the average of the values listed on Table 11.1-5 (75 ppb).

The values obtained from equation (7) for the average reactor coolant activities are presented in Table 11.1-10.

The maximum reactor coolant activities can be higher due to "crud bursts" during shutdowns or changes in power, however, these "bursts" occur over short periods of time, and therefore, the average values are more reasonable to use for long term operation.

The equilibrium thickness of radioactive crud film (mg-crud/cm^2) has been determined by two methods:

a. The direct measurement of the film during maintenance and/or tests in operating reactors.

b. Calculating crud film thickness from measured dose rates and specific activities (dpm/mg-crud) of deposited crud.

The equilibrium crud film thicknesses for various Reactor Coolant System are expected to be as shown in Table 11.1-11.

The calculated crud activities (Table 11.1-10) are reasonable values and together with measured plateout thicknesses (Table 11.1-11) match measured shutdown dose rates around various equipment associated with operating reactors. However, both the crud concentrations and plateout thicknesses do have rather wide variations for operating reactors and many combinations of activities and plateout thicknesses could reproduce the measured shutdown dose rates. It is for this reason that the crud activities may be periodically reviewed as more measured crud activities, plateout thicknesses and dose rates become available.

The conservative evaluation of the above operating data yields circulating crud concentrations (Table 11.1-10) which are lower for all isotopes, except Cr-51, than the concentrations given in ANSI N237 (Reference 20). ANSI standard N237 is intended for use in evaluating only normal operations including anticipated operational occurrences. However, the N237 values for circulating crud are used as design source terms as well as for normal operations, since they yield a design that is more conservative than that resulting from the calculations outlined in the

preceeding discussion. The circulating corrosion product activity concentrations in the reactor coolant, which are used as design source terms, are listed in Tables 11.1-2 and 11.1-3.

For deposited corrosion products in the RCS, the deposition thicknesses given in Table 11.1-11 are used to derive the shutdown dose rates for the RCS reported in Subsection 12.2.1.

A zinc injection system has been installed to allow injection of a zinc acetate solution into the RCS. The zinc injection system will displace cobalt from the resident RCS oxide films. This will result in a temporary increase in the RCS coolant activities from ^{58}Co and ^{60}Co. The elevation in the radiocobalt activity will be temporary, lasting only until the plant oxide layers are fully conditioned. The period of increase in activity will be approximately two operating cycles depending upon the rate of zinc injection.

11.1.1.3 Neutron Activation Products

11.1.1.3.1 Nitrogen-16 Activity

Nitrogen-16 is produced by the $0^{16}(n,p)N^{16}$ reaction which beta decays, emitting high energy gammas 78 percent of the time. The gamma energies are 6.13 and 7.10 MeV in a ratio of 12.5 to 1.0. The N^{16} half-life is 7.13 seconds. The threshold for the reaction is 10.2 MeV.

The N^{16} activity at the reactor vessel outlet nozzle is 4.82×10^6 disintegrations/cm^3 -sec. This activity is based on the following expression and reactor parameters:

$$\text{Activity (d/cm}^3\text{-sec)} = \frac{\sum\phi\left(1-e^{-\lambda t_c}\right)e^{-\lambda t_r}}{\left(1-e^{-\lambda t_T}\right)} \qquad (8)$$

where:

$\sum \phi$ = the reaction rate, 4.54×10^7 d/cm^3-sec

t_C = the core transit time, 0.78 sec

t_T = the total reactor coolant loop transit time, 10.24 sec

t_r = the time from the core outlet to the outlet nozzle, 0.90 sec

λ = the N-16 decay constant, 0.097 sec^{-1}

11.1.1.3.2 Carbon-14 Production

Carbon-14 is produced in the RCS primarily via the following pathways;

$O^{17}(n,\alpha)C^{14}$	and	$N^{14}(n,p)C^{14}$
σ = .24 x $10^{-24} cm^2$		σ = 1.8 x $10^{-24} cm^2$
% Abundance = 0.037	and	% Abundance = 99.634
N = 1.3×10^{22} Atoms/KgH_2O	and	N = 2.75×10^{20} Atoms/KgH_2O

where:

σ = reaction cross-section

N = Atom concentration of C^{14}

The equation which yields the contribution per parent nuclide is:

$$Q_o = N_o \ \sigma_o \ \phi \ m \ t \ p \ s \qquad (9)$$

where:

Q_o = Production rate from parent (i), (Ci/yr)

N_o = Atom concentration of parent (i) in the RCS water, (atoms/ KgH_2O)

σ_o = Thermal cross-section of parent (i), (cm^2)

ϕ = Thermal neutron flux, 5.12 x 10^{13}, (N/cm^2 sec)

m = Mass of core water, 1.45 x 10^4, (KgH_2O)

t = Conversion factor, 3.15 x 10^7 (sec/yr.)

P = Plant capacity factor, 0.8*

S = Activity conversion factor, 1.03 x 10^{-22}, $\frac{(\text{Ci from C14})/}{\text{Atom } C^{14}}$

Based on the above data, the C-14 production contributions are:

O^{17} (n, $\propto$) C^{14} 6.02Ci/Yr

N^{14}(N,P) C^{14} 0.95 Ci/Yr

Total C^{14} production 6.97 Ci/Yr

11.1.1.4 Maximum Fuel Pool Fission and Corrosion Product Activities (Operation at 2700 MW)

Fuel pool maximum fission and corrosion product specific activities are given in Table 11.1-12 for the start of the refueling period. It is assumed that upon shutdown for refueling the Reactor Coolant System is cooled down for a period of approximately two days. During this period the reactor coolant is letdown through the purification filter, purification ion exchanger and volume control tank. This flow path serves two purposes.

First, removal of noble gases from the reactor coolant via the volume control tank avoids large activity releases to the containment following reactor vessel head removal. Second, the reduction of dissolved fission and corrosion products in the reactor coolant (by ion exchange

* C^{14} production could be adjusted for the appropriate capacity factor

and filtration) reduces the amount which would otherwise enter the fuel pool, refueling cavity and refueling water tank.

At the end of this period, the coolant in the RCS is drained until the water level is just below the reactor vessel head flange. The reactor vessel head is then removed, and the refueling water cavity is filled with 445,000 gallons of water from the refueling water tank. For conservatism the remaining reactor coolant is assumed to mix completely with the fuel pool water, and the water in the refueling water cavity. After refueling, the spent fuel pool and refueling canal are isolated, and the water is pumped from the refueling cavity back to the refueling water tank.

The preceding series of events, along with a spent fuel pool and refueling canal water volume of 351,000 gallons, form the basis for the calculated fuel pool activities. These activities are given in Table 11.1-12. The Table 11.1-12 values are taken to be the initial activities, which are subsequently reduced by decay or operation of the fuel pool purification system.

The maximum tritium activity in the spent fuel is also shown in Table 11.1-12 and is calculated using the maximum RCS activity discussed in Subsection 11.1.4.

11.1.1.5 Maximum Steam Generator Activities (2700 MW)

The radionuclide concentrations in the steam generators under maximum design basis conditions are presented in Table 11.1-13. The assumptions upon which these values are based are presented in Table 11.1-14.

The time dependent equations used for calculating the activities in the steam generator, condenser, and turbine system are:

$$\frac{dA^{i}_{sg}}{dt} = Q_i + F_c A^{i}_{c} - (\lambda^i + B)A^{i}_{sg} - (F_{sg})\,(PF^{i}_{1})\,A^{i}_{sg} \qquad (10)$$

$$\frac{dA^{i}_{T}}{dt} = A^{i}_{sg}\,(F_{sg})\,(PF^{i}_{1}) - (\lambda^i + L_T)\,A^{i}_{T} - F_T A^{i}_{T} \qquad (11)$$

$$\frac{dA^{i}_{c}}{dt} = F_T\,(1-PF^{i}_{2})\,A^{i}_{T} - \lambda^i A^{i}_{c} - F_c A^{i}_{c} \qquad (12)$$

where:

A^{i}_{sg} = the total activity of isotope i in the steam generator liquid at time t, μCi

Q_i = the steam generator tubing leak rate for isotope i, μCi/sec

F_C = condensate flow rate, condenser vol/sec

A^{i}_{c} = condenser condensate activity of isotope i at time t, μCi

λ^i = decay constant for the i^{th} isotope, sec^{-1}

B = steam generator blowdown rate, vol/sec

F_{sg} = steam generator flow rate, vol/sec

PF_1^i = steam generator partition factor for the i^{th} isotope

L_T = the turbine system leakage rate, vol/sec

A_T^i = activity in turbine steam for isotope i at time t, μCi

F_T = turbine flow rate, vol/sec

PF_2^i = condenser partition factor for isotope i

The above equations are solved for the equilibrium values of A_c^i A_T^i and A_{sg}^i. For the specific case of tritium, the solution of these equations yields a value of 7.3 x 10^{-4} μCi/cc. For conservatism this value has been increased to 1.0 x 10^{-3} μCi/cc, which is consistent with the recommendations of NUREG-0017, April 1976 (R1) (21). The turbine system volume for this analysis includes the steam generator steam space, piping between the steam generator and turbine steam space, and condenser steam space. Condenser activity is based on the condensate volume in the hotwell and the piping between the hotwell and steam generator. Steam generator activity is based on the liquid inventory of the steam generator.

11.1.1.6 RCS Source Term Used for AST Dose Assessments

The equilibrium nuclide source term in the RCS is based upon the core inventory, the percentage of cladding defects, and fission product escape rate coefficients. The St. Lucie Units 2 Core/Fuel source term was developed from ORIGEN 2.1 computer code calculations for EPU conditions. Section 3.4 of Reg. Guide 1.183 specifies the radionuclide groups that should be considered for AST analyses. The nuclide list evaluated in this calculation is consistent with, but not identical to, the list of elements given in Section 3.4 of Regulatory Guide 1.183. This list of 107 nuclides was used in the St. Lucie Unit 2 AST dose analysis approved by the NRC in Reference 30. To ensure that the range of specified enrichments was bounded, 1.5 w/o, 3.0 w/o, 4.5 w/o or 5.0 w/o cases were analyzed. For each isotope, the maximum activity from any of the enrichment cases was determined. The end of cycle (49,000 MWD/MTU) activities for each isotope were extracted from all four ORIGEN cases. The resulting core inventory of radionuclide groups is documented for the LOCA dose assessment in Table 15.0-32.

To compute the RCS primary coolant activity, a GOTHIC model of the Chemical and Volume Control System (CVCS) purification loop was developed to account for extraction of nuclides by the mixed bed demineralizer, the second mixed bed demineralizer, and degassing in the volume control tank (VCT), as well as dilution of the nuclide concentration by normal Primary Makeup Water System operation for RCS boron control.

Since every nuclide is removed by at least one filter, all nuclides will come to equilibrium when the removal rate through the filters in the CVCS purification loop and by decay matches the nuclide source term production rate. Filter efficiency converted to "decontamination factors" is input for each nuclide for every filter. The mixed bed demineralizer is modeled with a mixed bed DF of 2 for Cs, and a DF of 10 for all isotopes except Y, Mo and noble gases. None of the other isotopes (Y, Mo and noble gases) will be removed and have a DF of 1 (η=O). The lithium

removal mixed bed demineralizer is modeled with a mixed bed DF of 2 for Cs, and a DF of 10 for all isotopes except Y, Mo and noble gases.

The degassing in the VCT is modeled with the lower bound WG Processing System not in service. Gas shipping fraction values are shown below used directly as the filter efficiencies.

Isotope	Stripping Fraction (RCS Inventory)
Kr-83m	0.88
Kr-85m	0.74
Kr-85	0.00014
Kr-87	0.91
Kr-88	0.82
Xe-131m	0.03
Xe-133	0.07
Xe-133m	0.15
Xe-135m	0.97
Xe-135	0.50
Xe-138	0.975

GOTHIC models the removal of ‘contaminated’ RCS fluid and the introduction of ‘clean’ makeup water during the boron dilution process. Since all isotopes in this flow stream are totally removed from the system, the decontamination factor is infinite (η=1) for all isotopes. The GOTHIC calculation is executed for 100 million seconds to ensure equilibrium activities are achieved.

Note that the results for the seven corrosion product isotopes (Co-58, Co-60, Cr-51, Fe-55, Fe-59, Mn-54, and Zn-65) are determined separately in accordance with ANSI/ANS-18.1-1999.

The above calculated iodine activities must be adjusted to achieve the Units 1 and 2 Technical Specification 3.4.8 limit of 1.0 µCi/gm dose equivalent I-131. The definition for DOSE EQUIVALENT I-131 in the approved for AST application Technical Specifications states that:

> *"DOSE EQUIVALENT I-131 shall be that concentration of I-131 (µcuries/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134 and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Federal Guidance Report 11, 'Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion. "*

It is noted that St. Lucie’s Technical Specification definitions specifically call out thyroid dose equivalent. Accordingly, this new analysis will continue to use the thyroid dose conversion factors from Table 2.1 of Federal Guidance Report 11 (Reference 31).

Although this new AST based EPU dose analysis evaluates the dose impact of I-130 when released as core inventory, the DE I-131 definition applicable to RCS inventory will not be changed to include I-130. As described in the 100/E-bar adjustment factor section below, the 100/E-bar (as well as the DE Xe-133 equivalent) scale factor definition also excludes this I-130 isotope. Since I-130 is excluded from both the DE I-131 and the 100/E-bar definitions, this nuclide is, by definition, non-dose-significant and will therefore be excluded from the "adjusted" RCS nuclide inventories.

The Technical Specification definition of Dose Equivalent 1-13 1 (D.E. 1-131) can be represented by the following equation: D.E. 1-131 (pCi1gm) = C(ai x DCFi) I DcF1.131

The Technical Specification definition of Dose Equivalent I-131 (D.E. I-131) can be represented by the following equation:

$$\text{D.E. I-131 } (\mu Ci/gm) = \Sigma(a_i \times DCF_i) / DCF_{I\text{-}131}$$

where:

a_i = activity of individual isotope (μCi/gm)

DCF_i = thyroid dose conversion factor for iodine isotope

$DCF_{I\text{-}131}$ = dose conversion factor for I-131

The non-iodine species must also be adjusted to achieve the Technical Specification limit of 100/E-bar for non-iodine activities. Implementation of Technical Specification Task Force TSTF-490 (as approved by the NRC in TSTF-490, Revision 0, Deletion of E Bar Definition and Revision to RCS Specific Activity Technical Specification) will convert that 100/E-bar inventory to an equivalent Dose Equivalent Xe-133 value as a proposed replacement for the definition of 100/E-Bar. By choosing an equivalent inventory, no relaxation of the allowable RCS inventory is being requested.

The Technical Specification definition of E-bar Average Disintegration Energy is:

> *"E-bar shall be the average (weighted in proportion to the concentration of each radionuclide in the reactor coolant at the time of sampling) of the sum of the average beta and gamma energies per disintegration (in MeV) for isotopes, other than iodines, with half lives greater than 15 minutes, making up at least 95% of the total non-iodine activity in the coolant."*

The Technical Specification definition of E-bar can be represented by the following equation:

$$\text{E-bar} = [\Sigma(E^{\beta}_i + E^{\gamma}_i) \times a_i) / \Sigma a_i$$

where:

a_i = activity of individual isotope (μCi/gm)

E^{β}_i = average beta emission energy (MeV/dis)

E^{γ}_i = average gamma emission energy (MeV/dis)

The maximum allowed specific activity related to the 1% fuel failure specific activity limit by the following derived adjustment factor:

342.64* Adj Factor < 327.30

Adj factor = 0.9552

Since the total specific activity of 342.64 µCi/gm is greater than 100/E-bar, the specific activities must be adjusted downward to be representative of plant operation at the Technical Specification limit to produce the final RCS specific activity provided in Table 15.0-32a.

11.1.2 NORMAL OPERATION INCLUDING ANTICIPATED OPERATIONAL OCCURRENCES (AVERAGE VALUES)(Cycle 1)

11.1.2.1 Reactor Coolant System Activities

The concentration of nuclides in the Reactor Coolant System under normal operating conditions including anticipated operational occurences are derived from the concentrations given in ANSI Standard N237. The Reactor Coolant System activities calculated in accordance with N237 are used as the basis for the radionuclide concentrations in all other plant systems during normal operations and are used as the basis for calculating the radioactive releases from the plant. Table 11.1-1 gives operating parameters but the parameters are not within the range of those given in N237. Therefore, adjusted N237 values based on the correction factors given in N237, are used. RCS activities are presented in Table 11.1-3.

11.1.2.2 Fuel Pool Activities

The bases for spent fuel pool and refueling canal activities are the same as for the maximum case described in Subsection 11.1.1.4. The concentrations are based on the RCS activities of N237 as discussed in Subsection 11.1.2.1. Spent fuel pool and refueling canal activities are presented in Table 11.1-15.

11.1.2.3 Steam Generator Activities

The radionuclide concentrations in the steam generators under 2560 MW operating conditions, are presented in Table 11.1-16. The values were obtained using the GALE code and the assumptions presented in Table 11.1-17.

11.1.3 TRITIUM PRODUCTION

The principal sources of tritium production in a pressurized water reactor or (PWR) are from ternary fission and neutron induced reactions in boron, lithium and deuterium that are present in the reactor coolant, borated shim rods and control element assemblies (CEA). The tritium produced in the reactor coolant contributes immediately to the overall tritium activity while the tritium produced by fission and neutron capture in the CEAs and borated shim rods contributes to the overall tritium activity via release through the cladding.

11.1.3.1 Activation Sources of Tritium

The activation reactions producing tritium are as shown in Table 11.1-18. Reactions 1 through 4 from boron -10, lithium, and deuterium are the major sources of tritium in the reactor coolant,

CEAs, and borated shim rods. The tritium production from reactions 5 and 6 (B-11 and N-14 sources) is insignificant due to low cross-section and/or abundance and can be neglected.

The tritium production from the above sources is determined by the following expressions:

Tritium formation rate = Production rate-decay

$$\frac{dN}{dt} = \sum_a \phi - N \qquad (13)$$

$$N = \frac{\sum_a \phi (1 - e^{-\lambda t})}{\lambda}$$, atoms/cm^3 at time (t)

activity (curies) $= V \lambda N \times 2.7 \times 10^{-11}$ (14)

$$= \sum_a \phi \ (1\text{-}e^{\lambda t}) \ V \times 2.7 \times 10^{-11} \qquad (15)$$

where:

$\Sigma_a \phi$ = the production rate (atoms/cm^3-sec)

λ = tritium decay constant

t = the reactor operating period of interest

v = the effective core volume, borated shim rod volume or CEA volume (cm^3)

2.7×10^{-11} converts disintegrations/sec. to curies.

The appropriate parameters used in the production calculation are shown in Table 11.1-19. Based on these parameters, the total amount of tritium produced is given in Table 11.1-20. For conservatism, the method used in NUREG-0017(21) is used in developing the values presented in Table 11.1-3 for tritium concentration in the RCS.

11.1.3.2 Tritium From Fission

The ternary fission production of tritium in the core is expressed simply by:

$$\frac{dN}{dt} = YF - \lambda N \qquad (16)$$

$$N = \frac{YF}{\lambda}\left(1 - e^{-\lambda t}\right)$$, atoms at time (t) (17)

activity (curies) = $\lambda N \times 2.7 \times 10^{-11} = YF \ (1\text{-}e^{-\lambda t}) \ 2.7 \times 10^{-11}$ (18)

where:

Y = the tritium fission yield (tritium atoms per fission)

F = the fission rate (f/sec)

A = tritium decay constant

T = the reactor operating period of interest, 2.7x 10^{-11} convert disintegrations/sec to curies

Tritium as a product of fission[1], [22] has a yield of 8.0 x 10^{-5} atoms/fission for U-235 and a yield of 2.6x10^{-4} atoms/fission for U-238, Pu-239 and Pu-241. The amount of tritium that is released through fuel cladding can be indirectly determined using measured tritium levels from operating PWRs subtracting the calculated tritium activity produced by neutron capture in the reactor coolant, and attributing the remaining tritium activity to release from the cladding of the fuel rods, borated shim rods, and CEAs. Due to the large number of fuel rods as compared to the number of borated shim rods and CEAs within the core during operation, any amount of tritium released to the system is principally from the fuel rods. The total amount of tritium produced per fuel cycle can be determined by summing the total tritium discharged in the gaseous, liquid, and solid waste discharges of the plant and the tritium inventories in the RCS and other waste or refueling tanks that can contain tritium at the end of the fuel cycle of interest. This method has been used to analyze operating data from various PWRs [6] [23 to 28]. The results of the analysis are shown in Table 11.1-21. Buildup of plutonium in the fuel with burnup was accounted for in the analysis. Based on this data, an average expected one percent tritium release from the fuel and a maximum two percent design value are used to estimate the annual tritium production in Table 11.1-20.

11.1.4 TRITIUM CONCENTRATIONS

The concentration of tritium in the reactor coolant is a function of:

a. the inventory of tritiated liquids in the plant,

b. the rate of production of tritium due to activation in the reactor coolant as well as release from the fuel, and

c. the extent to which tritiated water is recycled or discharged from the plant. The design basis concentration in the reactor coolant is determined from the expression:

$$\frac{dA}{dt} = P - (\lambda_1 + \lambda_2)A$$

where:

A = Tritium activity, (Ci)

t = time, (years)

p = tritium production rate, (Ci/year)

λ_1 = natural radioactive decay factor, (year-1)

λ_2 = loss factor for all nonrecoverable leakage and evaporation from the reactor coolant

$$\left(\frac{\text{total water loss, gal/yr}}{\text{reactor coolant volume, gal}} \right) \quad (\text{year}^{-1})$$

The value of λ_1 the decay factor for tritium, is 0.0563 year^{-1}. The value of λ_2 is a function of all the leakage pathways for loss of RCS fluid.

11.1.5 LEAKAGE SOURCES

Systems containing radioactive liquids are potential sources of leakage to the environment. Table 11.1-22 provides a listing of leakage values from valves and pumps. Leakage of reactor coolant into the containment atmosphere, which is ultimately exhausted to the environment at times of containment purge, is assumed to be one percent per day of the reactor coolant noble gas activity and .001 percent per day of the iodine activity in the reactor coolant. An additional potential source of gaseous discharge results from reactor coolant leakage into the Reactor Auxiliary Building. A leakage rate of 180 lbs/day of a mixture of hot and cold reactor coolant leakage is assumed, with an iodine and noble gas partition factor of .0075 and 1.0 respectively. The liquid from these leakage sources is collected and processed in the Liquid Waste Management System which is described in Section 11.2. Other features which reduce leakage are; the use of packless diaphragm valves, double packing, and lantern rings on valves which have packing.

Steam generator tube leakage can result in the buildup of radionuclides in the Main Steam and Feedwater Systems and the Steam Generator Blowdown System (SGBS). Under normal operation a leakage rate of 100 lbs/day is assumed. This activity can ultimately result in discharge of small amounts of liquid and gaseous wastes to the environment. The discharge of liquid waste can result from liquid leakage to the Turbine Building sump and the release of portions of processed blowdown. It is assumed that leakage to the Turbine Building sump is five gpm and that all of the steam generator blowdown is processed and released to the circulating water discharge. The SGBS is discussed in Subsection 10.4.8.

Gaseous releases can result from main steam leakage and the gland seal system exhaust. Overall main steam leakage is assumed to be approximately 1700 lbs/hour and originates from many sources, each too small to identify. Turbine gland seal steam flow is sent to a gland steam condenser resulting in negligible discharges. Since steam generator blowdown temperature and pressure is reduced via heat exchanger, rather than a flash tank, an insignificant quantity of noncondensible gases are produced. The above leakage rates and partition coefficients are based on the recommendations and experience presented in NUREG-0017(R1).

Releases inside the plant are handled by the appropriate ventilation system. Plant ventilation systems are discussed in Section 9.4 and the Radiation Monitoring System is discussed in Subsection 12.3.4. The source terms used as design bases for evaluating these systems are provided in Subsection 12.2.2.

Means of detecting reactor coolant pressure boundary leakage are discussed in Subsection 5.2.5.

Estimated liquid and gaseous releases due to leakage from various systems containing radioactivity are derived using GALE Code and are discussed in Subsections 11.2.3 and 11.3.3 respectively. The various anticipated operational occurrences that may result in radioactive release to the environment are as described in NUREG 0017 (R1). The Environmental Report identifies the transport mechanism and describes the release path.

SECTION 11.1: REFERENCES

1. Meek, M.E. and Rider, B.F., "Summary of Fission Product Yields for U235, U238, Pu241 at Thermal, Fission Spectrum and 14 MeV Neutron Energies," APED-5398, Class 1, March 1, 1968.

2. Lippincott, E.P., Pitner, A.L., and Kellogg, L.S., "Measurement of 10^B (n,t) Cross Section in a Fast Neutron Spectrum," HEDL-TME-73^B-49, May 1973.

3. "Chart of Nuclides," USAEC, modified by Batelle-Northwest, May 1969 and May 1970.

4. "Neutron Cross Sections," BNL 325 Supplement No. 2, May 1964.

5. G.M. Allison and H.K. Rae, "The Release of Fission Gases and Iodines from Defected UO ^{2} Fuel Elements of Different Lengths," AECL-2206, June 1965.

6. Grant, P.J., et al., "Oconee Radiochemistry Survey Program," RDTPL-75-4, May 1975.

7. Weisman, J., and Bartnoff, S., "The Saxton Chemical Shim Experiment," WCAP-3269-24, July 1965.

8. "Large Closed-Cycle Water Reactor Research and Development Program," Progress Report, WCAP-3269-13, April 1, 1965 - June 30, 1965.

9. "Corrosion Product Behavior in Stainless-Steel-Clad Water Reactor Systems," Nuclear Applications, Vol. 1, October 1965.

10. Abrams, C.S., and Salterelli, E.A., "Decontamination of the Shippingport Atomic Power Station," WAPD-299, January 1966.

11. Weingart E., "Radiation Buildup on Mechanisms and Thermal Barriers," WAPD-PWR-TE- 145, June 1963.

12. Indian Point 1 Semi-Annual Operations Reports, September 1966, September 1967, March 1968, September 1968.

13. "Test Data Sheets, Maine-Yankee Core Crud Removal," CENPD-113, August 13, 1973.

14. Uhl, D.L., "Oconee Radiochemistry Survey Program," Semi-annual Report July-December 1973, May 1974.

15. Uhl, D.L., "Oconee Radiochemistry Survey Program," Semi-annual Report January-June 1974, May 1975.

16. Connecticut Yankee Monthly Operating Reports, February 1968, March 1968, June 1968, July 1968, December 1968, January 1969, March-May 1969, August 1969, October 1969, December 1969, March 1970, October 1970, November 1970.

17. San Onofre Monthly Operating Reports, December 1969, January 1970, January-March 1971, June-September 1971, November 1971.

18. Yankee Rowe Monthly Operating Reports, February-June 1969, August-December 1969, January-December 1970, January 1972, April-July 1972.

19. "Large Closed-Cycle Water Reactor Research and Development Program," Progress Report WCAP-3620-12, January 1, 1965 - March 31, 1965.

20. ANSI N237 Rev. 00, "Radioactive Materials in Principal Fluid Streams of Light-Water Cooled Nuclear Power Plants."

21. Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors (PWR-GALE Code), NUREG-0017, REV.01.

22. ANL-7450 Chemical Engineering Division Research Highlights, May 1967 - April 1968.

23. Omaha Semi-Annual Reports, 1973-1975.

24. Maine Yankee Daily Log Sheets, June 1972 - April 1975.

25. Point Beach Semi-Annual Reports, June 1971 - January 1974.

26. H.B. Robinson Semi-Annual Reports, June 1971 - January 1975.

27. Ginna Semi-Annual Reports, June 1971 - January 1975.

28. "Source Term Data for Westinghouse Pressurized Water Reactors," WCAP-8253, May 1974.

29. K.H. Lin, "Use of Ion Exchange for the Treatment of Liquids in Nuclear Power Plants," ORNL 4792.

30. USNRC "St. Lucie Plant, Unit 2 Issuance of Amendment Regarding Alternative Source Term (TAC No. MD6202)," September 29, 2008.

31. Federal Guidance Report No. 11, "Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion," September, 1988.

TABLE 11.1-1

BASIS FOR REACTOR COOLANT FISSION PRODUCT ACTIVITIES(1)

Duration of Reactor Operation	5 core cycles(4)	
Equilibrium Fuel Cycle, Equivalent Full Power Days	303(4)	
Average Thermal Fission Rate (Fissions/Mw-second)	2.51×10^{16} (4)	
Thermal Neutron Flux - Average (n/cm^2-second)	5.1×10^{13}	
Reactor Coolant Mass, lbm (2560 Mwt)	4.52×10^{5}	
Core Coolant Volume to Reactor Coolant Volume Ratio	0.076	
Purification Flow, gpm (normal letdown)	40	
Beginning of Life Coolant Boron Concentration, ppm	1400	
Boron Concentration Reduction Rate, ppm/second	5.35×10^{-5}	
Ion Exchangers; Removal Effectiveness	DF[2]	Efficiency
CVCS Purification Ion Exchanger #1		
Xe, Kr, H-3	1	0
Cs, Rb	2	50
All other nuclides	10	90
CVCS Purification Ion Exchanger #2(3)		
Xe, Kr, H-3	1	0
All other nuclides	10	90
CVCS Flash Tank		
Xe, Kr	2	50
All other nuclides	1	0
Fission Product Escape Rate Coefficients, sec^{-1}		
Noble gases	6.5×10^{-8}	
Halogens	1.3×10^{-8}	
Cs	2.3×10^{-8}	
Te,Mo	1.4×10^{-9}	
All Others	1.4×10^{-11}	
Percent Failed Fuel	1.0	

	Normal	Maximum
Power Level, Mwt	2560	2700

(1) Some parameters are used in analysis of maximum activities only. Values for normal operations including anticipated operational occurrences, are assumed to be consistent with the ANSI Standard N237 Model.

(2) Ion exchanger decontamination factors (DF) are derived from findings of a generic review of the nuclear industry by ORNL(29)

(3) This ion exchanger, used primarily for lithium removal is used in series with the #1 Purification Ion Exchanger during approximately 20 percent of the core cycle.

(4) Although these values may not represent current operation, the results in Table 11.1-2 remain applicable for 2700 MW operation.

(5) The replacement steam generators have a larger primary mass than the original steam generators. Using this liquid mass is conservative in determining the radionuclide concentration based on failed fuel fractions.

TABLE 11.1-2

MAXIMUM ACTIVITIES IN THE REACTOR COOLANT
DUE TO CONTINUOUS OPERATION AT 2700 MWT WITH 1 PERCENT FAILED FUEL

Nuclide	Specific Activity @ 70F, µCi/cc
H-3	3.5 (0)*
N-16	1.3 (+02)
KR-85M	1.1 (0)
KR-85	2.3 (0)
KR-87	8.9 (-1)
KR-88	2.7 (0)
XE-131M	4.4 (0)
XE-133	2.9 (2)
XE-135	6.4 (0)
XE-138	5.1 (-1)
BR-84	2.9 (-2)
RB-88	2.7 (0)
RB-89	6.5 (-2)
SR-89	5.7 (-3)
SR-90	1.9 (-4)
Y-90	1.9 (-4)
SR-91	3.7 (-3)
Y-91	6.1 (-3)
ZR-95	7.7 (-3)
MO-99	4.6 (-1)
RU-103	6.4 (-3)
RU-106	1.7 (-3)
TE-129	1.3 (-2)
I-129	3.4 (-8)
I-131	3.7 (0)
TE-132	3.5 (-1)
I-132	7.7 (-1)
I-133	4.6 (0)
TE-134	3.3 (-2)
I-134	4.6 (-1)
CS-134	1.6 (-1)
I-135	2.3 (0)
CS-136	1.1 (-1)
CS-137	4.4 (-1)
CS-138	9.2 (-1)
BA-140	8.8 (-3)
LA-140	8.5 (-3)
PR-143	7.3 (-3)
CE-144	4.6 (-3)
CR-51	2.6 (-3)
MN-54	4.3 (-4)
FE-55	2.2 (-3)

* numbers in ( ) are powers of 10

TABLE 11.1-2 (Cont'd)

Nuclide	Specific Activity @ 70F, µCi/cc
FE-59	1.4 (-3)
CO-58	2.2 (-2)
CO-60	2.8 (-3)

TABLE 11.1-3

RCS ACTIVITIES DURING NORMAL OPERATIONS
INCLUDING ANTICIPATED OPERATION OCCURRENCES
(Cycle 1 - 2560 MW)

Nuclide	Specific Activity @ 70°F µCi/cc	Nuclide	Specific Activity @ 70°F µCi/cc
H-3	1.0 (0)*	Y-91M	3.4 (-4)
N-16	1.24 (+2)	Y-93	3.7 (-5)
KR-83M	1.9 (-2)	ZR-95	8.2 (-5)
KR-85M	1.0 (-1)	NB-95	6.8 (-5)
KR-85	1.1 (-1)	MO-99	1.1 (-1)
KR-87	5.5 (-2)	TC-99M	5.0 (-2)
KR-88	1.8 (-1)	RU-103	6.2 (-5)
KR-89	4.6 (-3)	RU-106	1.4 (-5)
XE-131M	9.5 (-2)	RH-103M	4.2 (-5)
XE-133M	2.0 (-1)	RH-106	9.2 (-6)
XE-133	1.6 (+1)	TE-125M	4.0 (-5)
XE-135M	1.2 (-2)	TE-127M	3.9 (-4)
XE-135	3.2 (-1)	TE-127	9.2 (-4)
XE-137	8.3 (-3)	TE-129M	1.9 (-1)
XE-138	4.0 (-2)	TE-129	1.5 (-3)
BR-83	4.7 (-3)	TE-131M	3.0 (-3)
BR-84	2.4 (-3)	TE-131	1.0 (-3)
BR-85	2.8 (-4)	TE-132	3.5 (-2)
I-130	2.3 (-3)	BA-137M	1.5 (-2)
I-131	3.6 (-1)	BA-140	3.0 (-4)
I-132	9.7 (-2)	LA-140	1.9 (-4)
I-133	4.4 (-1)	CE-141	9.6 (-5)
I-134	4.4 (-2)	CE-143	4.9 (-5)
I-135	2.0 (-1)	CE-144	4.5 (-5)
RB-86	1.2 (-4)	PR-143	6.8 (-5)
RB-88	1.8 (-1)	PR-144	3.1 (-5)
CS-134	3.7 (-2)	NP-239	1.5 (-3)
CS-136	1.9 (-2)	CR-51	2.6 (-3)
CS-137	2.7 (-2)	MN-54	4.3 (-4)
SR-89	4.8 (-4)	FE-55	2.2 (-3)
SR-90	1.4 (-5)	FE-59	1.4 (-3)
SR-91	7.0 (-4)	CO-58	2.2 (-2)
Y-90	1.5 (-6)	CO-60	2.8 (-3)
Y-91	8.8 (-5)		

*numbers in ( ) are powers of 10.

TABLE 11.1-4

PHYSICAL PARAMETERS OF LONG-LIVED ISOTOPES IN CRUD

Isotope	$T_{1/2}$	$\lambda(days)^{-1}$	Parent	Reaction	λ/dis	E(mev)
Co-60	5.26yr	3.6 (-4)	Co-59	n,γ	2.00	1.25
Co-58	71.4d	9.73 (-3)	Ni-58	n,p	1.00	0.81
Mn-54	313d	2.21 (-3)	Fe-54	n,p	1.00	0.84
Cr-51	27.8d	2.49 (-2)	Cr-50	n,γ	0.10	0.32
Fe-59	45d	1.54 (-2)	Fe-58	n,γ	1.00	1.18
Zr-95	65.5d	1.06 (-2)	Zr-94	n,γ	2.00	0.75

TABLE 11.1-5

MEASURED RADIOACTIVE CRUD ACTIVITY (dpm/mg-crud)

Reactor	Co-60	Co-58	Mn-54	Cr-51	Fe-59	Hf-181	Zr-95	Cu-64	Crud ppb	Ref
Conn. Yankee [a]										
Ave	9.1(+6)[c]	9.9(+7)	2.3(+6)	1.3(+7)	2.8(+6)	--	--	--	85	16
Max	2.5(+7)	4.0(+8)	1.2(+7)	3.6(+7)	1.5(+7)	--	--	--	4000	
San Onofre, [a]										
Ave	2.0(+6)	2.2(+7)	1.4(+6)	3.1(+6)	6.7(+5)	--	--	--	90	17
Max	2.0(+7)	1.2(+8)	4.2(+6)	2.0(+7)	3.8(+6)	--	--	--	4000	
Yankee Rowe, [a]										
Ave	6.7(+6)	3.3(+7)	4.5(+6)	1.7(+7)	5.5(+6)	--	6.6(+5)	--	70	18,9
Max	2.1(+7)	1.2(+8)	1.9(+7)	1.4(+8)	1.8(+7)	--	1.8(+6)	--	--	
Saxton, [b]										
Ave	4.3(+6)	2.7(+7)	3.9(+6)	9.0(+7)	1.2(+6)	--	--	--	55	7,8
Max	2.2(+7)	1.5(+8)	1.4(+7)	1.1(+8)	6.0(+6)	--	--	--	250	9,19
Shippingport, [a]										
Ave	2.3(+7)	2.8(+6)	1.3(+6)	2.2(+6)	1.8(+6)	5.2(+5)	7.0(+5)	--	75	10,11
Max	4.8(+7)	3.2(+6)	1.7(+6)	2.2(+6)	1.8(+6)	7.6(+5)	9.7(+5)	--	--	
Indian Point I, [a]										
Ave	1.8(+6)	4.6(+6)	7.7(+5)	5.7(+6)	2.2(+6)	1.5(+5)	2.3(+5)	3.1(+9)	77	12
Max	2.9(+6)	9.1(+6)	2.0(+6)	8.2(+6)	3.3(+6)	--	4.2(+5)	1.2(+10)	--	
Maine-Yankee, [a]										
Ave	--	--	--	--	--	--	1.29(+6)	--		
Max	2.22(+6)	4.53(+7)	9.70(+5)	4.24(+7)	2.03(+6)	--	7.26(+6)	--	41	13
Oconee, [a]										
Ave	2.8(+6)	5.1(+7)	5.5(+5)	2.9(+7)	2.4(+5)	--	5.6(+6)	--	25	
Max	2.3(+7)	1.9(+8)	1.1(+7)	1.5(+8)	1.7(+6)		8.7(+6)		100	14,15
Average Crud (ppb)									68/75[d]	

a. Circulating crud.
b. Deposited crud on fuel rods with exception of Cr-51 (ave, max) and Fe-59 (ave) which are circulating.
c. ( ) denotes power of 10.
d. Does not include Oconee data.

TABLE 11.1-6

ACTIVATION RATES

Reactor	Activation Rates, (d/gm-sec) Co-60	Co-58	Mn-54	Cr-51	Fe-59	Zr-95	A_T/A_C
Conn. Yankee[a]	1.90(+10)[b]	7.00(+10)	1.40(+9)	2.90(+10)	1.90(+8)	-	4.10
San Onofre[a]	1.90(+10)	7.00(+10)	1.40(+9)	2.90(+10)	1.90(+8)	-	4.10
Yankee Rowe	1.70(+10)	1.50(+10)	4.34(+9)	1.90(+10)	3.50(+8)	7.50(+8)	3.13
Saxton	8.00(+9)	1.00(+10)	2.95(+9)	1.30(+10)	2.40(+8)	-	5.26
Indian Point 1	6.6(+9)	1.3(+10)	3.7(+9)	1.1(+10)	2.0(+9)	1.4(+8)	4.53
Maine-Yankee	6.5(+9)	6.1(+10)	5.2(+8)	1.9(+10)	6.3(+7)	3.8(+8)	5.44
Oconee	1.3(+10)	1.00(+11)	3.1(+9)	9.8(+10)	9.5(+8)	3.1(+9)	4.00

a. Conn. Yankee and San Onofre fluxes and area ratios assumed the same.
b. ( ) denotes power of 10.

TABLE 11.1-7

AVERAGE AND MAXIMUM RESIDENCE TIMES, DAYS

Reactor	Co-60	Co-58	Mn-54	Cr-51	Fe-59[a]	Zr-95
Conn. Yankee						
Ave	92	10	54	1	-	-
Max	262	51	390	4	-	-
San Onofre						
Ave	20	2	32	1	18	-
Max	207	13	104	2	-	-
Yankee Rowe						
Ave	58	13	25	2	111	6
Max	185	56	116	19	-	17
Saxton						
Ave	25	5	10	38	38	-
Max	136	30	38	54	-	-
Shippingport						
Ave	115	1	8	1	10	2
Max	246	1	11	1	10	3
Indian Point I						
Ave	58	3	7	2	115	13
Max	94	6	19	2	-	24
Maine-Yankee						
Ave	-	-	-	-	-	34
Max	87	7	84	9	-	-
Oconee						
Ave	41	3	5	1	1	12
Max	356	13	118	4	8	66
Ave of Max T_{res}	66	22	110	12	-	29
	140[c]	23	110	13	-	20

a. Fe-59 isotope reaches equilibrium before erosion from core surfaces.
b. Included in Zr-95 of Maximum (T_{res}).
c. Lower values do not include Oconee data.

TABLE 11.1-8
ASSUMED ACTIVATION RATES

Isotope	Activation Rate Σ_id(d/g-sec)
60_{Co}	8.78 (+9)[(a)]
58_{Co}	8.06 (+10)
54_{Mn}	7.33 (+8)
51_{Cr}	2.65 (+10)
59_{Fe}	9.36 (+7)
95_{Zr}	7.09 (+8)

(a) Denotes power of ten (10).

TABLE 11.1-9

LONG LIVED CRUD ACTIVITY AS
DERIVED FROM OPERATING DATA

Isotope	T_{res} (days)	Half life	Act, dpm/mg
60_{Co}	166	5.26 y	.92 (+7)[(a)]
58_{Co}	22	71.4 d	.28 (+9)
54_{Mn}	110	31.3 d	.29 (+7)
51_{Cr}	12	27.8 d	.12 (+9)
59_{Fe}	Saturation	45 d	.17 (+7)
95_{Zr}	29	65.5 d	.34 (+7)

(a) Denotes power of ten (10).

TABLE 11.1-10

REACTOR COOLANT ACTIVITY AS DERIVED FROM OPERATING DATA[a]

Isotope	Activity (µci/cc)
60_{Co}	3.1 (-4) [b]
58_{Co}	9.4 (-3)
54_{Mn}	9.6 (-5)
51_{Cr}	4.2 (-3)
59_{Fe}	5.7 (-5)
95_{Zr}	1.1 (-4)

(a) Reactor coolant temperature is 70 F. Crud level 75 ppb.

(b) Denotes power of ten (10).

TABLE 11.1-11

EQUILIBRIUM CRUD FILM THICKNESS

Location	Thickness (mg/cm^2)
Vessel Internals, Piping, SG Inlet Plenum	1.00(+0)*
Pressurizer	
Lower Head	6.5(-1)
Surge Line	1.20(+0)
CEDM, Vessel Head ICI Tops	3.00(-1)
SG Tubing	1.00(-1)
Regenerative HX	3.50(-1)
Letdown HX	3.00(-2)
Shutdown Cooling HX	3.00(-2)
Spent Fuel Pool HX	3.00(-3)

*( ) Denotes power of 10

TABLE 11.1-12

MAXIMUM FISSION AND CORROSION PRODUCT ACTIVITIES
IN THE SPENT FUEL FOOL
(2700 MW)

Nuclide	Specific Acitivity @ 70°F (μCi/cc)
H-3	3.5 (0)*
N-16	0.
KR-85M	1.3(-05)
KR-85	5.8(-02)
KR-87	6.4(-14)
KR-88	3.9(-07)
XE-131M	9.9(-02)
XE-133	5.6(+0)
XE-135	4.1(-03)
XE-138	0.
BR-84	0.
RB-88	0.
RB-89	0.
SR-89	1.4(-04)
SR-90	4.8(-06)
Y-90	2.8(-06)
SR-91	2.9(-06)
Y-91	1.5(-04)
ZR-95	1.9(-04)
MO-99	7.0(-03)
RU-103	1.6(-04)
RU-106	4.3(-05)
TE-129	0.
I-129	8.6(-10)
I-131	7.9(-02)
TE-132	5.7(-03)
I-132	7.7(-09)
I-133	2.3(-02)
TE-134	0.
I-134	0.
CS-134	4.0(-03)
I-135	3.8(-04)
CS-136	2.5(-03)
CS-137	1.1(-02)
CS-138	0.
BA-140	2.0(-04)
LA-140	9.3(-05)
PR-143	1.7(-04)
CE-144	1.2(-04)
CR-51	6.2(-05)
MN-54	1.1(-05)
FE-55	5.6(-05)
FE-59	3.4(-05)
CO-58	5.5(-04)
CO-60	7.1(-05)

* ( ) Denotes power of 10

TABLE 11.1-13

DESIGN BASIS RADIONUCLIDE CONCENTRATIONS
IN THE STEAM GENERATORS (μCi/gm)
(2700 MW)

Isotope	Water	Steam
Kr-85m	-	4.07(-7)
Kr-85	-	8.50(-7)
Kr-87	-	3.28(-7)
Kr-88	-	1.00(-6)
Xe-131m	-	1.63(-6)
Xe-133	-	1.08(-4)
Xe-135	-	2.38(-6)
Xe-138	-	1.79(-7)
Br-84	3.17(-7)	3.17(-9)
I-129	5.62(-12)	5.62(-14)
I-131	5.89(-4)	5.89-(6)
I-132	2.98(-5)	2.98(-7)
I-133	5.60(-4)	5.60(-6)
I-134	7.88(-6)	7.88(-8)
I-135	1.79(-4)	1.79(-6)
Rb-88	1.78(-5)	1.78(-8)
Rb-89	3.70(-7)	3.70(-10)
Sr-89	9.42(-7)	9.42(-10)
Sr-90	3.17(-8)	3.17(-11)
Sr-91	3.52(-7)	3.52(-10)
Y-90	2.84(-8)	2.84(11)
Y-91	1.01(-6)	1.01(-9)
Z-95	1.28(-6)	1.28(-9)
Mo-99	6.91(-5)	6.91(-8)
Ru-103	1.06(-6)	1.06(-9)
Ru-106	2.82(-7)	2.82(-10)
Te-129	2.95(-7)	2.95(-10)
Te-132	5.33(-5)	5.33(-8)
Te-134	4.83(-7)	4.83(-10)
Cs-134	2.66(-5)	2.66(-8)
Cs-136	1.78(-5)	1.78(-8)
Cs-137	7.32(-5)	7.32(-8)
Cs-138	1.06(-5)	1.06(-8)
Ba-140	1.43(-6)	1.43(-9)
La-140	1.20(-6)	1.20(-9)
Pr-143	1.19(-6)	1.19(-9)
Ce-144	7.65(-7)	7.65(-10)
Cr-51	4.28(-7)	4.28(-10)
Mn-54	7.15(-8)	7.15(-11)
Fe-55	3.66(-7)	3.66(-10)
Fe-59	2.32(-7)	2.32(-10)
Co-58	3.65(-6)	3.65(-9)
Co-60	4.66(-7)	4.66(-10)

TABLE 11.1-13 (Cont'd)

Isotope	Water	Steam
H-3	7.26(-4)	7.26(-4)*
N-16	-	5.27(-6)

* The tritium concentration of 1.0(-3) µCi/gm for normal operating conditions is generated following the NUREG-0017 guideline (see Table 11.1-21) and is higher than the best estimated value for maximum concentrations. Hence in subsequent calculations the value used for maximum tritium concentrations is 1.0(-3).

TABLE 11.1-14

ASSUMPTIONS AND PARAMETERS FOR DESIGN BASIS ACTIVITIES
FOR THE STEAM GENERATOR

Reactor Coolant Activities	1% Failed Fuel See Table 11.1-2
Primary to Secondary Leakage	100 lb/day
Steam Generator Blowdown	50 gal/min*
Steam Generator Liquid Mass***	2.61×10^5 lbs
Steam Generator Partition Factors	Iodines .01 Particulates .001 Noble Gases 1.0 Tritium 1.0
Main Steam Rate	1.12×10^7 lbs/hr**

Notes:

* The design blowdown rate of 190 gpm has not been used in this calculation because that would result in lower steam generator activities.

** This value is conservative for power operation at 2700 MW for activity calculations.

*** The replacement steam generators have a larger primary mass than the original steam generators. Using this liquid mass is conservative in determining the radionuclide concentration based on failed fuel fractions.

TABLE 11.1-15

FISSION AND CORROSION PRODUCT ACTIVITIES IN THE SPENT FUEL POOL
UNDER NORMAL CONDITIONS INCLUDING ANTICIPATED OPERATIONAL OCCURRENCES
(Cycle 1 - 2560 MW)

Nuclide	Specific Activity @ 70°F (μCi/cc)	Nuclide	Specific Activity @ 79°F (μCi/cc)
H-3	1.0(0)*	Y-91M	0.
N-16	0.	Y-93	3.4(-08)
KR-83M	6.6(-12)	ZR-95	2.0(-06)
KR-85M	1.2(-06)	NB-95	1.7(-06)
KR-85	2.8(-03)	MO-99	1.7(-03)
KF-87	4.0(-15)	TC-99M	4.6(-06)
KR-88	2.6(-08)	RU-103	1.5(-06)
KR-89	0.	RU-106	3.5(-07)
XE-131M	2.1(-03)	RH-103M	0.
XE-133m	2.7(-03)	RH-106	0.
XE-133	3.1(-01)	TE-125M	9.9(-07)
XE-135M	0.	TE-127M	9.7(-06)
XE-135	2.0(-04)	TE-127	6.2(-07)
XE-137	0.	TE-129M	4.6(-05)
XE-138	0.	TE-129	0.
BR-83	9.6(-11)	TE-131M	2.5(-05)
BR-84	0.	TE-131	0.
BR-85	0.	TE-132	5.7(-04)
I-130	3.8(-06)	BA-137M	0.
I-131	7.6(-03)	BA-140	6.8(-06)
I-132	9.7(-10)	LA-140	2.1(-06)
I-133	2.2(-03)	CE-141	2.3(-06)
I-134	0.	CE-143	4.5(-07)
I-135	3.3(-05)	CE-144	1.1(-06)
PB-86	2.8(-06)	PR-143	1.6(-06)
RB-88	0.	PR-144	0.
CS-134	9.3(-04)	NP-239	2.1(-05)
CS-136	4.3(-04)	CR-51	6.2(-05)
CS-137	6.8(-04)	MN-54	1.1(-05)
SR-89	1.2(-05)	FE-55	5.6(-05)
SR-90	3.5(-07)	FE-59	3.4(-05)
SR-91	5.4(-07)	CO-58	5.5(-04)
Y-90	2.2(-08)	CO-60	7.1(-05)
Y-91	2.2(-06)		

* Numbers in are powers of 10

TABLE 11.1-16

RADIONUCLIDE CONCENTRATIONS IN THE STEAM GENERATORS UNDER NORMAL OPERATING CONDITIONS
(Cycle 1 - 2560 MW)

Isotope	Water (μCi/gm)	Steam (μCi/gm)
Kr-83m	-	6.8(-9)
Kr-85m	-	3.5(-8)
Kr-85	-	4.0(-9)
Kr-87	-	1.9(-8)
Kr-88	-	6.3(-8)
Xe-131m	-	8.9(-9)
Xe-133m	-	4.2(-8)
Xe-133	-	2.2(-6)
Xe-135m	-	4.3(-9)
Xe-135	-	1.0(-7)
Xe-137	-	3.0(-9)
Xe-137	-	1.4(-8)
Cr-51	5.2(-7)	5.2(-10)
Mn-54	1.3(-7)	1.3(-10)
Fe-55	4.4(-7)	4.4(-10)
Fe-59	3.2(-7)	3.2(-10)
Co-58	4.5(-6)	4.5(-9)
Co-60	5.7(-7)	5.7(-10)
Np-239	2.4(-7)	2.4(-10)
H-3	1.0(-3)	1.0(-3)
Br-83	2.1(-7)	2.1(-9)
I-130	2.8(-7)	2.8(-9)
I-131	7.5(-5)	7.5(-7)
I-132	1.6(-5)	1.6(-7)
I-133	6.4(-5)	6.4(-7)
I-134	8.6(-7)	8.6(-9)
I-135	1.7(-5)	1.7(-7)
Rb-86	2.7(-8)	2.7(-11)
Rb-88	1.2(-6)	1.2(-9)
Cs-134	7.9(-6)	7.9(-9)
Cs-136	3.4(-6)	3.4(-9)
Cs-137	5.3(-6)	5.3(-9)
Sr-89	1.3(-7)	1.3(-10)
Sr-91	7.0(-8)	7.0(-11)
Y-91M	4.6(-8)	4.6(-11)
Y-91	1.9(-8)	1.9(-11)
Zr-95	1.9(-8)	1.9(-11)
Nb-95	1.9(-8)	1.9(-11)

TABLE 11.1-16 (Cont'd)

Isotope	Water (μCi/gm)	Steam (μCi/gm)
Mo-99	2.4(-5)	2.4(-8)
Tc-99m	3.8(-5)	3.8(-8)
Ru-103	1.3(-8)	1.3(-ll)
Rh-103m	3.0(-8)	3.0(-11)
Ru-106	3.2(-9)	3.2(-12)
Te-127m	5.7(-8)	5.7(-11)
Te-127	2.3(-7)	2.3(-10)
Te-129m	3.9(-7)	3.9(-10)
Te-129	9.0(-7)	9.0(-10)
Te-131m	4.6(-7)	4.6(-10)
Te-131	7.7(-7)	7.7(-10)
Te-132	6.1(-6)	6.1(-9)
Ba-137m	1.3(-5)	1.3(-8)
Ba-140	6.0(-8)	6.0(-11)
La-140	6.9(-8)	6.9(-11)
Ce-141	1.9(-8)	1.9(-11)
Pr-143	1.3(-8)	1.3(-11)
Ce-144	1.3(-8)	1.3(-11)
Pr-144	3.1(-8)	3.1(-11)

TABLE 11.1-17

ASSUMPTIONS FOR NORMAL RADIONUCLIDE CONCENTRATIONS
IN THE STEAM GENERATORS

Cycle 1

ST. LUCIE 2	PWR
Thermal Power Level (MWT)	2560.
Plant Capacity Factor	0.8
Mass of Coolant in Primary System (10^3 lbs)	452.
Percent Fuel with Cladding Defects	0.12
Primary System Letdown Rate (GPM)	40.
Letdown Cation Demineralizer Flow Rate (GPM)	0.
Number of Steam Generators	2.
Total Steam Flow Rate (10^6 lbs/hr)	11.2
Mass of Steam in each Steam Generator (10^3 lbs)	9.5
Mass of Liquid in each Steam Generator (10^3 lbs)	130.5
Mass of Water in Steam Generator (Thousand lbs)	261.
Total Mass of Secondary Coolant (10^3 lbs)	1106.
Blowdown Rate (gal/min)	40.
Primary to Secondary Leak Rate (lbs/day)	100.
Condensate Demineralizer Regeneration Time (days)	0.
Fission Product Carry-Over fraction	0.001
Halogen Carry-Over Fraction	0.01
Fraction of Feed Water through Condensate Demineralizer	0.

TABLE 11.1-18

TRITIUM ACTIVATION REACTIONS

Reaction	Threshold Energy (MeV)	Cross-section (mb)[a]
1. ^{10}B (n, 2α) ^{3}H	1.9	1.13 (+1) [b]
2. ^{7}Li (n, nα)^{3}H	3.9	9.80
3. ^{6}Li (n,α) ^{3}H	Thermal	9.45 (+2) barns
4. ^{2}H (n,γ) ^{3}H	Thermal	5.70 (-1)
5. ^{11}B (n, ^{3}H)^{9}Be	10.4	8.00 (-3)
6. ^{14}N (n, ^{3}H) ^{12}C	4.3	3.00 (-1)

a. Threshold cross sections

b. ( ) denotes power of 10

TABLE 11.1-19

PARAMETERS USED IN TRITIUM PRODUCTION DETERMINATION[b]

Parameters	Value
Effective Core Volume, cm^3	1.87 (+7)[a]
Fast Neutron flux > 0.625 ev, $n.cm^2$-sec	2.2 (+14)
Thermal Neutron flux n/cm^2-sec	5.2 (+13)
Lithium Concentration, ppm	
Average	0.6
Maximum	1.0
Lithium - 6 abundance, %	1.6
Boron Concentration, ppm,	400
Power level (Mwt)	
Average	2560
Maximum	2688

(a) Denotes power of ten (10).

(b) Although some parameter values may not be representative of current operation at 2700 MW, the tritium production used in subsequent calculations remains acceptable due to the small effect of tritium on dose consequences.

TABLE 11.1-20

TRITIUM PRODUCTION IN REACTOR COOLANT

Reaction	Expected (Cycle 1) (Ci/yr)	Maximum[d] (Ci/yr)
^{2}H (n,Y) ^{3}H	2.5	2.5
$^{6}L_{i}$ (n,$\propto$) ^{3}H	348	580
^{7}Li (n,n$\propto$) ^{3}H	2	3.3
^{10}B (n,2$\propto$) ^{3}H	197	197
Fission	159[a]	334[b]
	709[c]	1117

(a) 1 percent fuel release
(b) 2 percent fuel release
(c) The conservative value of 1024 Ci/yr calculated using the guidelines of NUREG-0017 is used in all subsequent analysis.
(d) Although tritium production for operation at 2700 MW and current operating practices may be higher, the use of these tabular values is acceptable due to the small effect of tritium on doses.

TABLE 11.1-21

TRITIUM PRODUCTION AND RELEASE AT OPERATING PWRs

Operating PWR	Cycle No.	Measured[a] Production (Ci/Cycle)	Total[b] Calculated Production due to Fissions (Ci/Cycle)	Total Calculated to Capture in RCS (Ci/Cycle)	Release From Fuel (Percent)
Maine Yankee	1	305.3	11,720	370.0	-
Li conc approx 0	2	59.8	6,510	155.3	-
Omaha					
Li conc approx 0	1	192.6	6,100	153.9	0.6
Palisades					
Li conc approx 0	1	440	10,890	343.8	0.9
Calvert Cliffs 2					
Li conc 0.5 ppm	1	576	20,900	469	0.5
Millstone 2[c]	1	625	19,360	400	1.1
Li conc approx 0	2	338	11,040	175	1.2

a) Production is total measured tritium discharges plus measured system inventories.
b) Fission curies are based on approximate cycle average fraction fission of U-235, U-238 and Pu.
c) Includes 24 curies tritium released from poison shims.

TABLE 11.1-22

EQUIPMENT LEAKAGE ASSUMPTIONS

Valves	
Seat Leakage	10 cc/hr/in. Seat Diameter
Steam Leakage	10 cc/hr/in. Stem Diameter
Pumps	
Centrifugal	50 cc/hr
Positive Displacement	1 gallon/hr
Pump Flanges	30 cc/hr

APPENDIX 11.1A

DERIVATION OF RESIDENCE TIMES

The derivation of the core residence times for circulating crud and deposited crud as shown in Subsection 11.1.1 is as follows:

11.1A.1 CIRCULATING CRUD:

The number of radioactive atoms (N_f) in the crud film on in-core surfaces at any time is:

$$\frac{dN_f}{dt} = \sum_i \phi - \lambda_i N_f \qquad \text{(A-1)}$$

Solving for N_f yields the following:

$$N_f = \frac{\sum_i \phi}{\lambda_i}\left(1 - e^{-\lambda_i t_{res}}\right) \quad \text{atom/g} \qquad \text{(A-2)}$$

where:

$\sum_i \phi$ = the activation rate for each isotope i (d/g-sec)

λ_i = the decay constant for each isotope (sec^{-1}) and

t_{res} = the desired core residence time (sec).

The number of radioactive atoms (N_C) released to the reactor coolant at any time is:

$$\frac{dN_c}{dt} = N_f\,[ER]\,A_c - (\alpha + \beta + \lambda_i)N_c \text{ atoms/sec}$$

Solving for N_C yields the following:

$$N_c = \frac{N_f\,[ER]\,A_c}{(\alpha + \beta + \lambda_i)}\left(1 - e^{-(\alpha + \beta + \lambda_i)t}\right)$$

where:

ER = the erosion rate (g/cm^2)

A_C = the core surface area (cm^2)

a = the plateout rate (sec^{-1})

β = the purification cleanup rate (sec^{-1})

λ_i = the decay constant (sec^{-1})

The total amount of crud (M_c) released to the reactor coolant any time is:

$$\frac{dM_c}{dt} = [ER]\ A_T - (\alpha + \beta)\, M_c \qquad \text{(A-4)}$$

where:

M_C includes both radioactive and nonradioactive material

Solving for M_C yields:

$$M_c = \frac{[ER]\ A_T}{(\alpha + \beta)} \left(1 - e^{-(\alpha + \beta)t}\right) \text{ grams} \qquad \text{(A-5)}$$

where:

ER = the erosion rate (g/cm^2)

A_T = the total system area (cm^2)

α = the plateout rate (sec^{-1})

β = the purification cleanup rate (sec^{-1})

The activity (A_i) of the crud released to the reactor coolant is:

$$A_i = \frac{\lambda_i N_c}{M_c} \text{ , dps per gram of crud in reactor coolant} \qquad \text{(A-6)}$$

Substituting the values of N_C and M_C into the above expression and assuming λ_i is small when compared to α and β, the activity of the crud is as follows:

$$A_i = \sum_i \phi (1 - e^{-\lambda_i\ t_{res}})\ \frac{A_c}{A_T}(0.06) \text{ dpm/mg-crud} \qquad \text{(A-7)}$$

where:

0.06 = a constant changing dps/g-crud to dpm/mg-crud

This activity (A_i) is also assumed to be the activity of the crud which plates out on out-of-core surfaces.

Solving equation (A-7) for t_{res} yields equation (3).

11.1A.2 DEPOSITED CRUD

The activity (A_j) of the deposited crud is:

$$A_j = \lambda_i N_f = \sum_i \phi (1 - e^{-\lambda_i t_{res}}) \quad 0.06 \qquad \text{(A-8)}$$

Solving equation (A-8) for t_{res} yields equation (4).

11.2 LIQUID WASTE SYSTEM

Estimations of Liquid Waste System volumes, radioactivity concentration, and offsite doses presented in the tables and text of this chapter are based on calculational data or on similar systems in use at other plants, prior to operation of St. Lucie Unit 2 and is retained here for historical purposes.

Continued compliance with the annual regulatory dose limits following core extended power uprate has been demonstrated using scaling factors that address NUREG-0017 equations and assumptions and the reported liquid effluent and dose data during the years 2003 to 2007, taking into consideration the associated annual average core power level during that period extrapolated to 100 percent availability. For the uprate condition, the system parameters used reflected the flow rates and coolant masses at an NSSS power level of 3034 MWt and a core power level of 3030 MWt. To estimate an upper bound impact on off-site doses, the highest factor found for any chemical group pertinent to the release pathway was applied to the average doses previously determined as representative of operation at pre-uprate conditions. This approach was utilized to estimate the maximum potential increase in effluent doses due to the uprate and to demonstrate that the estimated off-site doses following the uprate, although increased, will continue to remain significantly below the annual design objectives for liquid radwaste effluents set by 10 CFR 50 Appendix I and 40 CFR 190.

It is noted that for an operating plant, the actual performance and operation of installed equipment, the reporting of actual offsite releases and doses, and compliance with the regulatory limits of 10 CFR 50 Appendix I and 40 CFR 190 is controlled by the Offsite Dose Calculation Manual.

Actual data on liquid waste effluents and dose to the public resulting from plant operation is presented annually in the Annual Radiological Effluent Release Report.

11.2.1 DESIGN BASES

Radioactive liquid wastes which are discharged from the plant are first processed by the Liquid Waste Management System (LWMS). The LWMS design bases are as follows:

a. The discharge activity level is in accordance with 10 CFR 20 criteria. An evaluation is provided in Subsection 11.2.3. The analysis takes into consideration normal and design basis operations.*

b. The principal design criteria for the LWMS is that it provides for handling of liquid wastes in such a manner as to meet the requirements of 10 CFR 20 and 10 CFR 50, Appendix I.

c. The St. Lucie Unit 2 Environmental Report (CP) in Amendment 7, dated October 1975 and Amendment 8, dated June 1976 provides a detailed evaluation to show that the LWMS is capable of controlling releases of radioactive materials within the numerical design objectives of Appendix I to 10 CFR 50. A review of the plant design and site usage characteristics reveal that no change has occurred which

*Note: "Design" means 2700 MWt and 1 percent failed fuel for source terms. "Normal" means 2560 MWt and NUREG-0017, April 1976, (R1) source terms.

would require a re-evaluation (see Subsection 11.2.3). Amendment 8 of the ER (CP) also provides a cost benefit analysis in Subsection 10.7.8 which is still applicable. The NRC, however, elected to evaluate the final design of the LWMS based on the requirements of the September, 1975 Annex to Appendix I.

d. The estimated total LWMS releases are summarized in Table 11.2-1. Assumed equipment decontamination factors are shown in Table 11.2-2. The concentration of radiological releases at the site boundary and a comparison with 10 CFR 20 limits for normal and design basis conditions are shown in Tables 11.2-3 and 4. The RCS activities are given in Section 11.1.

e. The individual component design parameters are given in Table 11.2-5.

f. The expected liquid borated waste inputs are described in Table 11.2-6 which gives the yearly inputs. Non-borated liquid waste inputs are shown in Table 11.2-7. The LWMS is designed to handle both sets of influents on a batch mode basis for flexibility of operation. "Batching" allows the LWMS system to handle "surges" in the influent rate, which may exceed the annual average flow rate.

g. The design, quality assurance, construction and testing criteria for the LWMS meet or exceed the guidelines of Regulatory Guide 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants," October 1979 (Rev. 1) particularly Position 1 (Systems Handling Radioactive Materials in Liquids) Position 4 (Additional Design, Construction and Testing Criteria) and Position 6 (Quality Assurance for Radwaste Management Systems). Furthermore, the seismic design classification of the structures housing the LWMS meet the guidelines of Regulatory Guide 1.143 as described in Section 3.2 and in Table 11.2-5.

h. General Design criteria 60 and 64 requirements are met for the LWMS.

i. The LWMS is designed with sufficient capacity, redundancy, and operational flexibility that it can accept waste and process influent during periods of equipment downtime.

j. The LWMS influent stream specific activities are given in Table 11.2-8. The component inventories (as presented in Section 12.2) are determined using these values andapplying the decontamination factors of Table 11.2-2, the component parameters of Table11.2-5, the waste volumes of Tables 11.2-6 and 11.2-7, the flow paths shown on Figures 11.2-1 to 11.2-9, and the process flow data of Tables 11.2-9 and 11.2-10 (see System Description, Subsection 11.2.2).

k. All LWMS tanks that may potentially contain radioactivity are provided with liquid level indicators and automatic control for diversion of the process stream from tanks exceeding predetermined level, and/or alarm. These alarms inform the operator of tank high and low liquid level. Low level indication of a waste tank terminates its associated pumps operation, to prevent damage to the pump.

   Each LWMS tank is located in separate cubicle and curbs are provided to contain tank overflow. Tank overflows are routed back to the LWMS for processing. See

Table 11.2-11 for the design provisions and controls provided to preclude inadvertent or uncontrolled releases of radioactivity. The release points of the LWMS are provided with radiation monitors which automatically close valves to prevent release of radioactivity to the environment when the radiation exceeds an acceptable level.

l. The LWMS cannot have inadvertent releases of radioactive liquids, due to the incorporation of the provisions stated in Item (h).

Should an operator inadvertently initiate the discharge of fluid with high activity, the effluent line radiation monitor would isolate the discharge stream and alert the operator. The automatic isolation valves mentioned are "fail closed" valves that must be "energized" to be opened. This means that should the radiation monitor fail or loss of electricity/air supply occur, these valves would close. LWMS design also incorporates manually operated valves located in all lines feeding the discharge line, which can be employed as backups.

m. All tanks in the LWMS are closed atmospheric tanks that are vented to the gas collection header (GCH) and overflow to a building sump. The overflow line is sized to handle the maximum pumped influent to the tank. High level alarms warn the operator of the tank reaching an overflow condition, upon receipt of a high level alarm the operator manually shifts influents to the second tank.

n. The LWMS is designed for operator ALARA as discussed in Section 12.3.

11.2.2 LWMS SYSTEM DESCRIPTION

Liquid wastes produced in the plant are collected and processed by the Liquid Waste Management System (LWMS). Two major processing schemes are present. They are the Boron Management Subsystem (BMS) and the Liquid Waste Subsystem (LWS). The liquid waste influents to the LWMS is segregated by chemistry and/or probable source activity for more efficient processing. Section 1.2.4 lists the liquid waste management system as a shared system which may, under certain conditions, be used by Unit 1. The equipment design data are provided in Table 11.2-5.

11.2.2.1 Boron Management Subsystem

Reactor coolant wastes of potentially high activity are handled as follows:

11.2.2.1.1 Normal Mode

Those wastes resulting from Chemical and Volume Control System operations (as listed on Table 11.2-6) enter the LWMS via the CVCS letdown line (Figure 11.2-1) to the holdup tanks. If RCS activity is above a pre-established threshold or if the nitrogen blanket in the holdup tanks is lost, RCS inventory is directed to the flash tank for processing. The flash tank uses a countercurrent flow of nitrogen to remove hydrogen and fission gases from the liquid. The gases are vented to the gaseous waste management system (see Section 11.3). This precludes the buildup of combustible gases within the flash tank and downstream components. A nitrogen overpressure is maintained within the flash tank, when in use, and the down stream hold up tanks to prevent air in-leakage. Thus, the possibility of forming a potentially explosive mixture within these tanks is precluded. The degassified liquid waste is pumped from the flash tank,

using the flash tank pumps to the holdup tanks (Figure 11.2-1) where it is stored prior to further processing. In this way the radioactivity of the stored liquid is significantly reduced by natural decay of the short lived radionuclides.

Those wastes resulting from valve and equipment leakage and those which enter the various containment drains, are collected in the reactor drain tank (Figure 11.2-1). When the liquid in the reactor drain tank (RDT) reaches a high level, an alarm is annunciated and the reactor drain tank pumps (Figure 11.2-1) are manually started to transfer the RDT contents to the holdup tanks, when RCS activity is low. With high RCS activity, RDT inventory is directed to the flash tank. This waste is handled in a similar fashion to letdown waste. A nitrogen overpressure is maintained in the reactor drain tank.

Liquid wastes stored in the holdup tank are then thoroughly mixed using the holdup tank recirculation loop. The loop consists of the holdup tanks (Figure 11.2-1), the holdup recirculation pump, a preconcentrator filter and preconcentrator ion exchangers (Figure 11.2-2). Preconcentrator ion exchangers normally contain mixed bed resin. As permitted by Engineering Evaluation, preconcentrator ion exchangers may contain an overlay of specialty resin to target removal of fine particulates and/or specific ionic species. The recirculation loop components remove suspended solids and insure a uniform fluid chemistry within the holdup tank. The holdup tank contents are sampled, to determine what further processing, if any, is required. Depending on the results of the sample analysis, the contents may be transferred to the Unit 1 holdup tanks for discharge or to the RWT for reuse. The liquid in the holdup tanks could be used, if available, as flushing water during resin sluicing operations for the preconcentrator ion exchanger and the spent resin tank. However, primary makeup water is normally used for this operation in accordance with plant procedures.

11.2.2.1.2 Waste Surges

The holdup tanks are sized with sufficient capacity to handle the waste surges which would occur during the normal operation of the plant, and be received in the BMS. There is also sufficient redundancy and interconnections to allow the contents of one holdup tank to be processed while another is being recirculated prior to processing, and the remaining two are receiving incoming wastes. This flexibility allows the BMS to create additional holdup capacity by simultaneously eliminating stored waste while accepting new waste.

The holdup tank capacity is adequately sized to handle the relief valve discharge (see interfaces on Figure 11.2-1) in the unlikely event that they should lift.

11.2.2.1.3 Boric Acid Concentrator Operation

Note: The Boric Acid Concentrators are no longer used. Information provided below for this component and associated equipment is maintained here for historical purposes.

The holdup tank contents are transferred to either of the boric acid concentrators (Figures 11.2-3 or 11.2-4) by the holdup drain pumps (Figure 11.2-2), which first pass the fluid through a pre-concentrator filter and the pre-concentrator ion exchanger (Figure 11.2-2).

Once the waste stream enters the boric acid concentrator, it is separated into two effluent streams by a simple evaporation process. The boric acid concentrator is designed to concentrate a dilute boric acid solution with a boron concentration of 30 to 1720 ppm, into a

bottoms stream with a concentration of 10,900 to 21,000 ppm boron and a distillate stream with a concentration of 10 ppm boron maximum.

11.2.2.1.3.1 Distillate

The distillate is condensed before it leaves the boric acid concentrator and is then pumped to the boric acid condensate ion exchanger (Figure 11.2-5) where any trace impurities remaining in the distillate are removed. The water continues on to the boric acid condensate tanks (Figure 11.2-5), where it is stored prior to sampling and reuse or discharge. Should high levels of impurities still be present in the condensate there are provisions to recycle it back through the boric acid concentrator.

11.2.2.1.3.2 Bottoms

The concentrated boric acid solution leaving the boric acid concentrator is pumped through the boric acid strainer and then to the boric acid holding tank (Figure 11.2-5). All piping, components, and tanks which handle the concentrated boric acid solution are heat-traced to keep the boric acid in solution.

Once in the boric acid holding tank the concentrate (bottoms) is recirculated using the boric acid holding pumps (Figure 11.2-5), to insure a uniform chemistry. A sample is taken and analyzed to determine the purity and concentration of the boric acid solution. Depending on the analysis and plant requirements, the boric acid can be either reused in the plant (transferred to the boric acid makeup tanks in the CVCS), sent to the liquid Waste Subsystem prior to solidification in the Solid Waste Management System or recycled to the boric acid concentrators for further concentration if the solution is too dilute.

11.2.2.2 Liquid Waste Subsystem

Liquid waste from sources outside of containment, usually of low activity and low purity are collected in either the equipment drain tank, chemical drain tank, or laundry drain tanks.

Prior to processing, the contents of these tanks are thoroughly mixed via recirculation, and representative samples are taken. The samples are analyzed to determine what, if any, processing is required for that liquid.

11.2.2.2.1 Normal Mode

The equipment drain tank (EDT), as shown on Figure 11.2-6, receives wastes from the various equipment drains outside containment. When the tank reaches a preset level it is emptied via the equipment drain pumps (Figure 11.2-6). The waste liquid is normally aligned to the Unit 1 aerated waste storage tank (AWST). However, the waste can be manually aligned to the waste condensate tanks.

The chemical drain tank (CDT) seen on Figure 11.2-6, receives liquid waste inputs from the lab drains and decontamination area drains, which are normally high in impurities. The chemical drain tank is emptied upon reaching a preset level, by the chemical drain pump. The waste liquid is normally aligned to the Unit 1 AWST. However, the waste can be manually aligned to the waste condensate tanks.

The laundry drain tanks (LDT), shown on Figure 11.2-6, store the influents from the plant showers, contaminated sinks, laundry operations, and potential inputs from the Steam Generator Blowdown System. When a laundry drain tank reaches a preset level, it is emptied by a laundry drain tank pump (Figure 11.2-6) and filtered via the laundry filter (Figure 11.2-6). From there it goes on to the Unit 2 AWST for processing.

Piping connections exist for the EDT, CDT and LDT to release their contents from the plant via the circulating water discharge. If needed, plant procedures will be developed to perform this action in accordance with the Offsite Dose Calculation Manual (ODCM).

11.2.2.2.2 Waste Surges

When waste surges occur in the influent volumes to the LWMS tanks, there is sufficient capacity, redundancy and system interconnections to provide adequate handling of these wastes.

Smaller quantity surges would only cause the waste tanks to fill quicker, hence more frequent processing of each tank's contents. Larger quantity surges which would exceed the capacity of those tanks could be routed to the aerated waste storage tank, for holdup until operations permit its processing or the results of sampling allow release of effluents to the circulating water discharge. The Unit 1 aerated waste storage tank specifically acts as a surge tank for the equipment drain tank, chemical drain tank and waste condensate tank. The Unit 2 AWST acts as a surge tank for the laundry drain tanks.

Because of a common discharge with the chemical drain tank, the equipment drain tank may become filled at any time and require emptying to handle further inputs. The equipment drain pump is automatically started on high equipment drain tank level and the contents are pumped to the Unit 1 aerated waste storage tank. Discharge of the laundry drain tank and waste condensate tanks, if necessary, is a manual operation.

The Unit 1 aerated waste storage tank's contents would be mixed using the equipment drain pumps. A representative sample would be obtained and analyzed to determine what processing, if any, is required for the waste liquid. The liquid would be pumped from the Unit 1 aerated waste storage tank via the equipment drain pumps to either the circulating water discharge, or through the waste filter for reprocessing, depending on the water quality.

11.2.2.2.3 Waste Concentrator Operation

Note: The Waste Concentrator is no longer used. Information related to this component and supporting equipment contained below and in the next two sections is maintained for historical purposes.

Liquid wastes enter the waste concentrator (Figure 11.2-7), and the waste stream is separated into two effluent streams by a simple evaporation process. One stream, the distillate, is a very pure water stream. The other, the bottoms, is a highly concentrated solution of impurities.

11.2.2.2.3.1 Distillate

The distillate may still contain some trace impurities. It is then condensed and pumped from the waste concentrator to the waste condensate ion exchanger (Figures 11.2-5 & 11.2-9), where

those trace impurities are removed. The condensate is stored in the waste condensate tanks, mixed via the waste condensate pumps (Figure 11.2-9), and sampled to analyze for purity.

The condensate may then be either sent to the circulating water discharge for release from the plant, or routed back through the waste system for further treatment if needed. Reuse within the plant is also possible (see Figures 11.2-1, 2 and 9 for process routes).

11.2.2.2.3.2 Bottoms

Most of the impurities which are present in the influent stream to the concentrator leave in the bottoms stream. The bottoms are pumped from the concentrator (Figure 11.2-7) via the concentrate pumps into the liquid waste concentrator bottoms storage tank.

Once in the liquid waste concentrator bottoms storage tank the concentrate (bottoms) is recirculated using the liquid waste concentrator bottoms pump to insure a uniform chemistry. This tank will provide adequate holdup under expected processing conditions. During liquid waste processing, the concentrate is pumped via a liquid waste concentrator bottoms pump directly to the drumming station (Figure 11.2-9) where it is prepared for offsite disposal (see Section 11.4). After processing, provisions are made to flush the tank, the pump, and the associated piping with clean water. This washdown water is then discharged to either the equipment drain tank or the aerated waste storage tank.

The boric acid concentration in the liquid waste concentrator bottoms storage tank/pump and the associated piping and valves containing boric acid solution is administratively controlled to a lower boron concentration of up to 3.5 weight percent. No heat tracing is required for these areas because at or below a concentration of 3.5 weight percent boric acid, the ambient temperature of the Auxiliary Building (where the above components are located) will be sufficient.

11.2.2.3 Miscellaneous Waste Control

The LWMS components are either vented to the gas collection header (Figure 11.2-8) or the gas surge header (see Section 11.3). A nitrogen overpressure is maintained in the reactor drain tank, flash tank, and holdup tanks. This prevents air in-leakage and provides a dilutent for the hydrogen which may diffuse out of the water they contain, thus precluding the formation of potentially explosive gas mixtures.

Resin is added to the ion exchangers via the resin addition tank (Figure 11.2-9). This is a portable tank which is moved to the resin fill line of the ion exchanger to be filled. It is connected to this line, and the new resin drains from the resin addition tank to fill the ion exchanger. No activity is associated with this tank, as it only stores and transports unused resin.

Outdoor storage tanks which could contain potentially radioactive liquid are the Refueling Water Tank (RWT), Primary Water Storage Tank (PWST), Condensate Storage Tank (CST) and Steam Generator Blowdown Monitor Storage Tanks (SGBMST). Provisions have been incorporated for these tanks to prevent, collect and/or process spills in the unlikely event of contamination, and are described in Subsections 6.3.2.2.4, 9.2.3.1.5, 9.2.6.3 and 10.4.8.3, respectively.

During normal operation, the Turbine Building floor drains are discharged to the storm water basin without processing or monitoring. However, storm basin grab samples are collected weekly per plant procedures if a primary to secondary leak is detected.

The condenser air ejector radiation monitor detects very low levels of radioactivity in the plant secondary and is a much more sensitive method of radioactivity detection for effluent monitoring purposes, than sampling the turbine building floor drains. The radioactivity in the turbine building floor drains will be at least an order of magnitude lower than that in the plant secondary because of other system dilution and rain water in the storm water basin.

The Steam Generator Blowdown is described in Subsection 10.4.8.

PC/M 05225 (Reference 1) designed the installation of the Independent Spent Fuel Storage Installation (ISFSI) pad drainage equipment, a detention pond, and an outfall through the Intake Canal berm to spill into the plant Intake Canal. The drainage system is designed to detain and treat storm water to remove any surface water contaminants prior to discharging overboard. The discharge from the detection pond occurs automatically by gravity draining through a specially-designed control structure.

Releases from the ISFSI Pad are expected to be of low probability and quite small because of the following attributes, each of which is discussed below:

- The probability of transporting radioactive material to the ISFSI site is very small;

- The concentration of radiological effluents from the detention pond would be very small based on dilution available in the transporting surface waters and the detention pond itself; and

- The concentration of any radiological effluents from the pond outfall would be extremely small.

    1. The dry shielded canister (DSC) which holds the spent fuel is designed to withstand accidents and natural phenomena without rupture. Transfer of fuel into the DSC, andsealing of the DSC, is accomplished to ensure that no surface contamination remains on the DSC while transported to the storage module (Reference 2). Thus, the storage module will have no external radioactive contamination on its surface.

    2. Even if there were slight contamination of the storage module, it would be diluted by runoff into the detention pond.

    3. Even if dilute radioactive material were to be released from the outfall, it would be further diluted in the plant discharge canal seawater flowrate.

Based on the above, 10 CFR 20.1302, “Compliance With Dose Limits For Individual Members Of The Public,” is met via routine radiological surveys of the ISFSI pad and detention pond. Routine surveys of the ISFSI pad and pond are sufficient for the measurement and quantification of any potential radiological release.

11.2.3 RADIOACTIVE RELEASES

During liquid processing by the Boron Management Subsystem (BMS) and Liquid Waste Subsystem, radioactivity is removed so that the bulk of the liquid is restored to usable quality which is either recycled in the plant or discharged. The radioactivity removed from the liquids is concentrated in filters and ion exchange resin. These concentrated wastes may be shipped to an approved offsite disposal location or released from the site as part of liquid effluent following appropriate dilution. If the water is to be recycled back to the reactor, it must meet the purity requirements for reactor coolant. If the liquid does not meet the reactor coolant purity requirements, it is discharged. The activity of any released effluent must be consistent with the discharge criteria of 10 CFR 20 and Appendix I to 10 CFR 50. The BMS and WMS are capable of monitoring radioactive liquid discharge from the systems to ensure that activity concentrations do not exceed predetermined limits. The control valves on the discharge lines automatically close to terminate the discharge, once the radiation monitor exceeds an acceptable level.

The analysis and demonstration of compliance with Appendix I to 10 CFR 50 was submitted to the NRC in Amendments 7 and 8 to the Environmental Report (CP). A review of plant design and site usage characteristics reveals that no major changes have occurred since this submittal which would require a reanalysis of compliance with the cost-benefit requirements of Appendix I to 10 CFR 50. The NRC, however, elected to evaluate the final design of the LWMS based on the requirements of the September, 1975 Annex to Appendix I.

Table 11.2-12 presents the assumptions used to calculate releases of radioactive material in liquid effluents. The estimate of the normal liquid effluent from the facility, including anticipated operational occurrences, is presented in Table 11.2-1. These releases are obtained using the guidance presented in NUREG-0017, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from PWRS," April 1976 (R1) and are comparable to those reported in ER (CP) Amendment 8.

The estimated amount of tritium released via the liquid pathway is calculated from the volume of reactor coolant that is released in the nonrecyclable waste streams. The concentration of tritium in wastes originating from reactor coolant during normal operation is taken as 1 μCi/gm, consistent with Table 11.1-3. Tritium in liquid that leaks into, or is used as makeup to, the secondary system is considered to be released in liquid effluents through the Turbine Building floor drain discharge. The parameters for reactor coolant activity prior to processing are used to calculate the tritium concentration in the waste streams.

The estimated liquid effluent concentrations after uniform dilution in the circulating discharge flow are presented in Table 11.2-3 for normal operation, including anticipated operational occurrences. These concentrations are compared to limits of 10 CFR 20 and are shown to be much less. The estimated doses caused by the release of radioactivity in the liquid effluents as given in Table 11.2-13 are historical. The exposures are well within the limits of Appendix I to 10 CFR 50 and are comparable to those reported in ER (CP) Amendment 8. The doses were calculated in accordance with Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I," March 1976 (R0).

With a design basis fuel leakage of 1 percent failed fuel for an entire year, the estimated annual release of radionuclides in the liquid effluents is given in Table 11.2-4. The concentrations in the circulating water discharge of this release and a comparison with 10 CFR 20 maximum

permissible concentrations are also provided in Table 11.2-4, which shows that the released concentrations are only a very small fraction of 10 CFR 20 allowables.

The discharge points are shown on the general site plan of Figure 1.2-2.

TABLE 11.2-1

ESTIMATED LIQUID EFFLUENTS
ANNUAL RELEASES TO DISCHARGE CANAL

NUCLIDE	BORON RS (CURIES)	MISC. WASTES (CURIES)	SECONDARY (CURIES)	TURB BLDG (CURIES)	TOTAL LWS (CURIES)	ADJUSTED TOTAL (CI/YR)	LAUNDRY WASTES (CI/YR)	TOTAL (CI/YR)
CORROSION AND ACTIVITATION PRODUCTS								
CR 51	0.00001	0.00000	0.00041	0.00001	0.00042	0.00066	0.00000	0.00066
MN 54	0.00000	0.00000	0.00010	0.00000	0.00010	0.00016	0.00100	0.00120
FE 55	0.00001	0.00000	0.00035	0.00000	0.00037	0.00057	0.00000	0.00057
FE 59	0.00000	0.00000	0.00025	0.00000	0.00026	0.00041	0.00000	0.00041
CO 58	0.00007	0.00002	0.00354	0.00004	0.00368	0.00568	0.00400	0.00970
CO 60	0.00001	0.00000	0.00045	0.00001	0.00047	0.00073	0.00870	0.00940
ZR 95	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00140	0.00140
NB 95	0.00000	0.00000	0.00000	0.00000	0.00000	0.0000	0.00200	0.00200
NP239	0.00000	0.00000	0.00018	0.00000	0.00019	0.00029	0.00000	0,00029
FISSION PRODUCTS								
BR 83	0.00000	0.00002	0.00004	0.00000	0.00007	0.00011	0.00000	0.00011
RB 86	0.00001	0.00001	0.00002	0.00000	0.00003	0.00005	0.00000	0.00005
RB 88	0.00001	0.00006	0.00000	0.00000	0.00007	0.00011	0.00000	0.00011
SR 89	0.00000	0.00000	0.00010	0.00000	0.00010	0.00016	0.00000	0.00016
SR 91	0.00000	0.00000	0.00004	0.00000	0.00004	0.00006	0.00000	0.00006
Y 91M	0.00000	0.00000	0.00003	0.00000	0.00003	0.00004	0.00000	0.00004
Y 91	0.00001	0.00000	0.00002	0.00000	0.00003	0.00004	0.00000	0.00004
ZR 95	0.00000	0.00000	0.00002	0.00000	0.00002	0.00002	0.00000	0.00002
NB 95	0.00000	0.00000	0.00002	0.00000	0.00002	0.00002	0.00000	0.00002
MO 99	0.00004	0.00008	0.01776	0.00022	0.01810	0.02797	0.00000	0.02800
TC 99M	0.00004	0.00008	0.02407	0.00029	0.02447	0.03781	0.00000	0.03800
RU 103	0.00000	0.00000	0.00001	0.00000	0.00001	0.00002	0.00014	0.00016
RH 103M	0.00000	0.00000	0.00001	0.00000	0.00001	0.00002	0.00000	0.00002
RU106	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00240	0.00240
AG110M	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00044	0.00044
TE127M	0.00000	0.00000	0.00005	0.00000	0.00005	0.00007	0.00000	0.00007
TE127	0.00000	0.00000	0.00014	0.00000	0.00015	0.00023	0.00000	0.00023
TE 129M	0.00000	0.00000	0.00031	0.00000	0.00032	0.00049	0.00000	0.00049
TE 129	0.00000	0.00000	0.00022	0.00000	0.00023	0.00036	0.00000	0.00036
I130	0.00000	0.00006	0.00017	0.00002	0.00025	0.00039	0.00000	0.00039
TE131M	0.00000	0.00000	0.00032	0.00000	0.00033	0.00051	0.00000	0.00051
TE131	0.00000	0.00000	0.00006	0.00000	0.00006	0.00009	0.00000	0.00009
I131	0.00173	0.03375	0.05874	0.00732	0.10153	0.15688	0.00006	0.16000
TE132	0.00002	0.00003	0.00466	0.00006	0.00476	0.00735	0.00000	0.00740
I132	0.00016	0.00323	0.00655	0.00030	0.01025	0.01584	0.00000	0.01600
I133	0.00059	0.01805	0.04317	0.00519	0.06730	0.10398	0.00000	0.10000
I134	0.00000	0.00008	0.00002	0.00000	0.00010	0.00016	0.00000	0.00016
CS134	0.00354	0.00192	0.00629	0.00008	0.01183	0.01828	0.01300	0.03100
I135	0.00014	0.00284	0.00839	0.00093	0.01230	0.01900	0.00000	0.01900
CS136	0.00070	0.00089	0.00271	0.00003	0.00434	0.00671	0.00000	0.00670
CS137	0.00260	0.00138	0.00419	0.00005	0.00823	0.01271	0.02400	0.03700
BA137M	0.00047	0.00003	0.00392	0.00005	0.00447	0.00691	0.00000	0.00690
BA140	0.00000	0.00000	0.00005	0.00000	0.00005	0.00009	0.00000	0.00007
LA140	0.00000	0.00000	0.00005	0.00000	0.00006	0.00009	0.00000	0.00009

TABLE 11.2-1 (Cont'd)

NUCLIDE	BORON RS (CURIES)	MISC. WASTES (CURIES)	SECONDARY (CURIES)	TURB BLDG (CURIES)	TOTAL LWS (CURIES)	ADJUSTED TOTAL (CI/YR)	LAUNDRY WASTES (CI/YR)	TOTAL (CI/YR)
FISSION PRODUCTS (Cont'd)								
CE141	0.00000	0.00000	0.00002	0.00000	0.00002	0.00002	0.00000	0.00002
PR143	0.00000	0.00000	0.00001	0.00000	0.00001	0.00002	0.00000	0.00002
CE144	0.00000	0.00000	0.00001	0.00000	0.00001	0.00002	0.00520	0.00520
PR144	0.00000	0.00000	0.00001	0.00000	0.00001	0.00002	0.00000	0.00002
ALL OTHERS	0.00000	0.00000	0.00002	0.00000	0.00002	0.00004	0.0	0.00004
TOTAL (EXCEPT H-3)	0.01047	0.06256	0.18749	0.01463	0.27515	0.42515	0.06234	0.49000

TRITION RELEASE 430 CURIES PER YEAR.

TABLE 11.2-2

ASSUMED LWMS EQUIPMENT DECONTAMINATION FACTORS

NUCLIDE	BORIC ACID (a)(e) CONCENTRATOR(d) [BOT/DIST]	BORIC ACID (a)(e) CONCENTRATOR(d) (INF/DIST]	WASTE (b)(e) CONCENTRATOR (d) [BOT/DIST]	WASTE (b)(e) CONCENTRATOR (d) [INF/DIST]	FLASH FILTERS TANK	FLASH FILTERS (c)	BORIC ACID(e) CONDENSATE ION EXCHANGER (ANION BED)	BORIC ACID PRE-CONCENTRATOR ION EXCHANGER (MIXED BED)	WASTE ION EXCHANGER (MIXED BED)
I, Br	10^3	10^2	10^3	50	1	1	100	10	100
Rb,Cs	10^4	10^3	10^4	500	1	1	1	2	2
Zr,Nb	10^4	10^3	10^4	500	1	1	1	10	100
Mo,Tc	10^4	10^3	10^4	500	1	1	1	10	100
Y	10^4	10^3	10^4	500	1	1	1	10	100
OTHER CA^+	10^4	10^3	10^4	500	1	1	1	10	100
OTHER AN^-	10^4	10^3	10^4	500	1	1	100	10	100
CRUD	10^4	10^3	10^4	500	1	1	10	10	10
TRITIUM	1	1	1	1	1	1	1	1	1
NOBLE GASES	1	1	-	-	2	1	1	1	

(a) Boric Acid Concentrator Factor = 10
(b) Waste Concentrator Factor = 20
(c) Includes Laundry, Waste, and Pre-concentrator Filters
(d) BOT = Bottoms Conc.
DIST = Distillate Conc.
INF = Influent Conc.
(e) Equipment is no longer used.

TABLE 11.2-3

ESTIMATED LIQUID RADIOLOGICAL EFFLUENTS
SITE BOUNDARY CONCENTRATIONS - NORMAL OPERATION

ISOTOPE	CONC.* (µCI/ML)	MPC** (µCI/ML)	C/MPC
CR 51	8.12E-13	2.00E-03	4.06E-10
MN 54	1.48E-12	1.00E-04	1.48E-08
FE 55	7.0IE-13	8.00E-04	8.76E-10
FE 59	5.04E-13	5.00E-05	1.01E-08
CO 58	1.19E-11	9.00E-05	1.33E-07
CO 60	1.16E-11	3.00E-05	3.85E-07
ZR 95	1.72E-12	6.00E-05	2.87E-08
NB 95	2.46E-12	1.00E-04	2.46E-08
NP239	3.57E-13	1.00E-04	3.57E-09
BR 83	1.35E-13	3.00E-06	4.51E-08
RB 86	6.15E-14	2.00E-05	3.08E-09
RB 88	1.35E-13	MPC NOT GIVEN IN 10 CFR 20	
SR 89	1.97E-13	3-00E-06	6.56E-08
SR 91	7.38E-14	5.00E-05	1.48E-09
Y 91m	4.92E-14	3-00E-03	1.64E-11
Y 91	4.92E-14	3.00E-05	1.64E-09
MO 99	3.44E-11	4.00E-05	8.61E-07
TC 99m	4.67E-11	3.00E-03	1.56E-08
RU103	1.97E-13	8.00E-05	2.46E-09
RH103m	2.46E-14	1.00E-02	2.46E-12
RU106	2.95E-12	1.00E-05	2.95E-07
AG110m	5.41E-13	3.00E-05	1.80E-08
TE127m	3.61E-14	5.00E-05	1.72E-09
TE127	2.83E-13	2.00E-04	1.41E-09
TE129m	6.03E-13	2.00E-05	3.01E-08
TE129	4.43E-13	8.00E-04	5.54E-10
I130	4.80E-13	3.00E-06	1.60E-07
TE131m	6.27E-13	4.00E-05	1.57E-08
TE131	1.11E-13	MPC NOT GIVEN IN 10 CFR 20	
I131	1.97E-10	3.00E-07	6.56E-04
TE132	9.10E-12	2.00E-05	4.55E-07
I132	1.97E-11	8.00E-06	2.46E-06
I133	1.23E-10	1.00E-06	1.23E-04
I134	1.97E-13	2.00E-05	9.84E-09
CS134	3.81E-11	9.00E-06	4.24E-06
I135	2.34E-11	4.00E-06	5.84E-06
CS136	8.24E-12	6.00E-05	1.37E-07
CS137	4.55E-11	2.00E-05	2.28E-06
BA137m	8.49E-12	MPC NOT GIVEN IN 10 CFR 20	
BA140	8.61E-14	2.00E-05	4.31E-09
LA140	1.11E-13	2.00E-05	5.54E-09
CE141	2.46E-14	9.00E-05	2.73E-10
PR143	2.46E-14	5.00E-05	4.92E-10
CE144	6.40E-12	1.00E-05	6.40E-07
PR144	2.46E-14	MPC NOT GIVEN IN 10 CFR 20	
H 3	5.29E-07	3.00E-03	1.76E-04
TOTAL C/MPC			9.73E-04

* With a dilution flow of 510,000 gpm.
** Limits are representative of 10 CFR 20 values at the time of licensing. Refer to note in Section 12.2.2 concerning present use of MPC.

TABLE 11.2-4

ESTIMATED LIQUID RADIOLOGICAL RELEASES AND
SITE BOUNDARY CONCENTRATIONS - DESIGN BASIS

ISOTOPE	RELEASE (CI/YR)	CONC. * (µCI/ML)	MPC** (µCI/ML)	C/MPC
CR 51	6.60E-04	8.12E-13	2.00E-03	4.06E-10
MN 54	1.20E-03	1.48E-12	1.00E-04	1.48E-08
FE 55	5.70E-04	7.01E-13	8.00E-04	8.76E-10
FE 59	4.10E-04	5.04E-13	5.00E-05	1.01E-08
CO 58	9.70E-03	1.19E-11	9.00E-05	1.33E-07
CO 60	9.40E-03	1.16E-11	3.00E-05	3.85E-07
ZR 95	1.30E-01	1.60E-10	6.00E-05	2.66E-06
Rb 88	1.65E-03	2.03E-12	MPC NOT GIVEN IN 10 CFR 20	
SR 89	1.90E-03	2.34E-12	3.00E-06	7.79E-07
SR 91	3.20E-04	3.94E-13	5.00E-05	7.87E-09
Y 91	2.77E-03	3.41E-12	3.00E-05	1.14E-07
MO 99	1.17E-01	1.44E-10	4.00E-05	3.60E-06
RU103	1.65E-02	2.03E-11	8.00E-05	2.54E-07
RU106	2.90E-02	3.57E-11	1.00E-05	3.57E-06
TE129	3.12E-03	3.84E-12	8.00E-04	4.80E-09
I131	1.64E+00	2.02E-09	3.00E-07	6.72E-03
TE132	7.40E-02	9.10E-11	2.00E-05	4.55E-06
I132	1.27E-01	1.56E-10	8.00E-06	1.95E-05
I133	1.05E+00	1.29E-09	1.00E-06	1.29E-03
I134	1.67E-03	2.05E-12	2.00E-05	1.03E-07
CS134	1.34E-01	1.65E-10	9.00E-06	1.83E-05
I135	2.19E-01	2.69E-10	4.00E-06	6.73E-05
CS136	3.88E-02	4.77E-11	6.00E-05	7.95E-07
CS137	6.03E-01	7.42E-10	2.00E-05	3.71E-05
BA140	2.05E-03	2.52E-12	2.00E-05	1.26E-07
LA140	4.03E-03	4.96E-12	2.00E-05	2.48E-07
PR143	2.15E-03	2.64E-12	5.00E-05	5.29E-08
CE144	5.32E-01	6.54E-10	1.00E-05	6.54E-05
H 3	4.69E+02	5.77E-07	3.00E-03	1.92E-04
TOTAL C/MPC				8.43E-03

* With a dilution flow of 510,000 gpm.

** Limits are representative of 10 CFR 20 values at the time of licensing. Refer to note in Section 12.2.2 concerning present use of MPC.

TABLE 11.2-5

EQUIPMENT DESCRIPTION **

Flash Tank	
Quantity	1
Type	Vertical, cylindrical
Internal Volume, gallons	424
Design Pressure	70 (internal)/15 (external)
Design Temperature,°F	250
Normal Operating Pressure, psig	0-5
Normal Operating Pressure, °	120
Code	Class 3, ASME III, 1974 Edition with Summer 1974 Addenda
Material	Stainless Steel (SS)
Blanket Gas	Nitrogen
Holdup Tanks	
Quantity	4
Type	Vertical, cylindrical
Internal Volume, gallons	40,000
Design Pressure, psig	10
Design Temperature, °F	240
Normal Operating Pressure	0.5 to 5.0
Normal Operating Temperature,°F	120
Code	ASME VIII
Material	SS
Blanket Gas	Nitrogen
Reactor Drain Tank	
Quantity	1
Type	Horizontal cylindrical
Internal Volume, gallons	1618
Design Pressure, psig	25 (internal) / 15 (external)
Design Temperature,°F	250
Normal Operating Pressure, psig	0.5 – 6.0
Normal Operating Temperature, °F	120
Code	Class 3, ASME III, 1974 Edition with Winter 1975 Addenda
Material	SS
Blanket Gas	Nitrogen
Boric Acid Condensate Tanks (Equipment is no longer used)	
Quantity	2
Type	Vertical, cylindrical.
Internal Volume, gallons	7,300
Design pressure, psig	Atmospheric
Design Temperature, °F	250
Normal Operating Pressure, psig	Atmospheric
Normal Operating Temperature, °F	120
Code	None
Material	SS

* Designed, fabricated and inspected per ASME Section VIII, not code stamped.

** Some information provided in this table reflects data used for procurement of components prior to initial plant operations.

TABLE 11.2-5 (Cont'd)

Boric Acid Holding Tank (Equipment is no longer used)

Quantity	1
Type	Vertical, cylindrical
Internal Volume, gallons	2,400
Design Pressure, psig	Atmospheric
Design Temperature, °F	250
Normal Operating Pressure, psig	Atmospheric
Normal Operating Temperature, °F	150
Code	None*
Material	SS

Equipment Drain Tank

Quantity	1
Internal Volume, gallons	1,040
Design Pressure, psig	Atmospheric
Design Temperature. °F	200
Normal Operating Pressure, psig	Atmospheric
Normal Operating Temperature, °F	120
Material	SS
Code	None

Chemical Drain Tank

Quantity	1
Internal Volume, gallons	1,040
Design Pressure, psig	Atmospheric
Design Temperature, °F	200
Normal Operating Pressure, psig	Atmospheric
Normal Operating Temperature, °F	120
Material	SS
Code	None*

Laundry Drain Tank

Quantity	2
Internal Volume, gallons	2,000
Deign Pressure, psig	Atmospheric
Design Temperature, °F	200
Normal Operating Pressure, psig	Atmospheric
Normal Operating Temperature, °F	120
Material	SS
Code	None

Aerated Waste Storage Tank

Quantity	1
Internal Volume, gallons	40,000
Design Pressure, psig	Atmospheric
Design Temperature, °F	200
Normal Operating Pressure, psig	Atmospheric
Normal Operating Temperature,	120
Material	SS
Code	AWWA D100

* Designed, fabricated and inspected per ASME Section VIII, not code stamped.

TABLE 11.2-5 (Cont'd)

Waste Condensate Tank	
Quantity	2
Internal Volume, gallons	1,725
Design Pressure, psig	Atmospheric
Design Temperature, °F	250
Normal Operating Pressure, psig	Atmospheric
Normal Operating Temperature °F	120
Material	SS
Code	None*
Spent Resin Tank	
Quantity	1
Type	Vertical, Cylindrical
Internal Volume, Gallons	3,213
Design Pressure, psig	50
Design Temperature, °F	200
Normal Operating Pressure, psig	12
Normal Operating Temperature, °F	120
Code	Class 3, ASME III, 1974 Edition with Winter 1975 Addenda.
Material	SS
Resin Addition Tank	
Type	Vertical
Quantity	1
Internal Volume, Gallons	114
Design Pressure, psig	Atmospheric
Design Temperature, °F	N/A
Operating Temperature, °F	100
operating Pressure, psig	Atmospheric
Wetted Material	SS
Code	None
Flash Tank Pumps	
Quantity	2
Type	Centrifugal
Design Pressure	150
Design Temperature,°F	200
Normal Operating Temperature,°F	120
Capacity, Rate, gpm	150
Rated Head, feet	63
Motor Horsepower	5.0
Wetted Materials	SS
Code	Manufacturer's Standard
Reactor Drain Pumps	
Quantity	2
Type	Centrifugal
Design Pressure, psig	150

* Designed, fabricated and inspected per ASME Section VIII, not code stamped,

TABLE 11.2-5 (Cont'd)

Reactor Drain Pumps (Cont'd)	
Design Temperature °F	200
Normal Operating Temperatures °F	120
Capacity, rates gpm	50
Fated Head, feet	140
Motor Horsepower	7.5
Wetted Materials	SS
Code	Manufacturer's Standard
Holdup Drain and Recirculation Pump	
Quantity	3
Type	Centrifugal
Design Pressure, psi	150
Design Temperature,	200
Normal Opeating Temperature,°F	120
Capacity, rated, gpm	40
Rated Head, feet	154
Motor Horsepower	7.5
Wetted Materials	SS
Code	Manufacturer's Standard
Boric Acid Holding Pumps (Equipment is no longer used)	
Quantity	2
Type	Centrifugal
Design Pressure, psig	150
Design Temperature, °F	200
Normal Operating Temperature, °F	160
Capacity, reated, gpm	50
Rated Head, feet	96
Motor Horsepower	5
Wetted Material	SS
Code	Manufacturer's Standard
Boric Acid Condensate Pumps (Equipment is no longer used)	
Quantity	2
Type	Centrifugal
Design Pressure, psi	150
Design Temperature, °F	200
Normal Operating Temperature, °F	120
Capacity, rated, gpm	50
Rated Head, feet	140
Motor Horsepower	7.5
Wetted Materials	SS
Code	Manufacturer's Standard
Equipment Drain Pumps	
Quantity	3
Type	Centrifugal
Design Pressure, psig	150

TABLE 11.2-5 (Cont'd)

Equipment Drain Pumps (Cont'd)	
Design Temperature °F	200
Design Conditions	
Flow, gpm	50
Head, feet	140
Wetted Materials	SS
Motor Horsepower	7.5
Normal Operating Temperature, °F	120
Code	Manufacturer’s standard
Chemical Drain Pump	
Quantity	1
Type	Centrifugal
Design Pressure, psi	150
Design Temperature, °F	200
Design Conditions	
Flow, gpm	50
Head, feet	140
Wetted Materials	SS
Motor Horsepower	7.5
Normal Operating Temperature °F	120
Code	Manufacturer’s standard
Laundry Drain Pumps	
Quantity	2
Type	Centrifugal
Design Pressure, psig	150
Design Temperature, °F	200
Design Conditions	
Flow, gpm	50
Head, feet	140
Wetted materials	SS
Motor Horsepower	7.5
Normal Operating Temperature,°F	120
Code	Manufacturer’s Standard
Waste Condensate Pumps	
Quantity	2
Type	Centrifugal
Design Pressure, psi	150
Design Temperature, °F	200
Design Conditions	
Flow, gpm	50
Head, feet	140
Wetted Materials	SS
Motor Horsepower	7.5
Operating Temperature, °F	120
Code	Manufacturer’s Standard

TABLE 11.2-5 (Cont'd)

Resin Dewatering Pump (Equipment is no longer used)

Type	Centrifugal
Quantity	1
Design Pressure, psig	150
Design Temperature, °F	200
Normal Operating Temperature, °F	120
Capacity, GPM	100
Fated Head, feet	95
Motor Horsepower	10
Wetted Materials	SS
Code	Manufacturer's standard

Boric Acid Concentrators (Equipment is no longer used)

Quantity	24
Design DF (Bottoms/Distillate)	10^4(Minimum)
Design Pressure psig	80
Design Temperature, °F	250
Normal Operating Pressure (process), psig	27
Normal Operating Temperature (process), °F	120
Design Flow (process), gpm	20
Normal Operating Flow (process), gpm	20
Code	ASME VIII
Material	SS

Waste Concentrator (Equipment is no longer used)

Quantity	1
Design DF (Bottoms/Distillate)	10^4
Design Pressure, psig (Process)	80
Normal Operating Pressure (Process), psig	27
Design Temperature, °F (Process)	250
Normal Operating Temperature,°F	120
Design Flow, gpm	20
Normal Operating Flow, gpm	20
Material	SS
Code	ASME VIII

Liquid Waste Concentrator Bottoms Storage Tank (Equipment is no longer used)

Quantity	1
Type	Vertical, Cylindrical
Internal Volume, gallons	900.0
Design Pressure, psig	Atmospheric
Design Temperature, °F	250
Normal Operating Pressure, psig	Atmospheric

TABLE 11.2-5 (Cont'd)

Liquid Waste Concentrator Bottoms Tank (Cont'd)

Normal Operating Temperature, °F	160-170
Code	AWWA D-100
Material	Inconel 625

Liquid Waste Concentrator Bottoms Pump (Equipment is no longer used)

Quantity	
Type	Centrifugal
Design Pressure, psig	50
Design Temperature, °F	250
Normal Operating Temperature, °F	160-170
Capacity Rate, gpm	50
Rated Head, feet	140
Motor Horsepower	7.5
Wetted Materials	ASTM A744 CN7M
Code	Manufacturer's Standard

Preconcentrator Ion Exchangers

Quantity	2
Type	Flushable
Design Pressure, psi	150
Design Temperature, °F	250
Normal Operating Pressure, psig	60
Normal Operating Temperature °F	120
Resin Volume (Useful) Required, ft^3	34
Design Flow, gpm	100
Normal Flow, gpm	20
Code for Vessel	ASME VIII
Material	SS
Resin	Mixed Bed, Specialty Resin overlay

Boric Acid Condensate Ion Exchanger (Equipment is no longer used)

Quantity	1
Type	Flushable
Design Pressure, Psig	150
Design Temperature, °F	250
Normal Operating Pressure, psig	60
Normal Operating Temperature, °F	120
Resin Volume (Useful) Required, ft^3	32
Design Flow, gpm	50
Normal Flow, gpm	20
Code for Vessel	ASME VIII
Material	SS
Resin	Anion Bed (OH Form)

Waste Ion Exchangers

Quantity	2
Type	Flushable deep resin bed

TABLE 11.2-5 (Cont'd)

Waste Ion Exchangers (Cont'd)

Design Pressure, psig	150
Normal operating Pressure, psig	60
Design Temperature, °F	250
Design Operating Temperature °F	120
Design Flow, gpm	50
Normal Operating Flow, gpm	20
Resin Volume (Useful), ft^3	32
Code for Vessel	ASME VIII
Material	SS
Resin	Mixed Bed

Preconcentrator Filters

Quantity	2
Type Element	Replaceable Cartridge
Particle Retention	2 micron, 98%
Normal Operating Pressure, psig	67
Design Pressure, psig	200
Design Temperature, °F	250
Normal Operating Temperature, °F	120
Design Flow, gpm	40
Normal Flow, gpm	20
Code	Class 3, ASME III, 1977 Edition with Summer 1977 Addenda
Material	SS

Waste Filter

Quantity	1
Type of Elements	Replaceable Cartridge
Particle Retention	20 to 150 Micron, Absolute as discerned by Chemistry Department
Design Pressure, psig	200
Design Temperature, °F	250
Design Flow, gpm	50
Material	ss
Code	Class 3, ASME III, 1977 Edition with Summer 1977 Addenda

Laundry Filter

Quantity	1
Type of Elements	Replaceable Cartridge
Particle Retention	150 Micron, Absolute
Design Pressure, psig	200
Design Temperature, OF	250
Design Flow, gpm	50
Material	ss
Code	Class 3, ASME III, 1977 Edition with Summer 1977 Addenda

TABLE 11.2-6

CVCS ESTIMATED WASTE INPUTS TO THE LWMS - BORATED WASTES

Part A: Component Sizing Inputs (1)

Source	Volume (Gal/Yr)
1. Boron Dilution to adjust for fuel burnup	237,750
2. Cold shutdown and startups (3 per year @30%, 60% 90% of core life)	274,300
3. Hot shutdowns and startups (2 per year @55%, 65% of core life)	128,300
4. Refueling (1 per yr. to refuel reactor)	104,150
5. Leakage to reactor drain tank (200 gal/day 313 days/yr)	62,600
6. Resin Dewatering waste water (not CVCS input directly, results from sluicing of spent resin to spent resin tank. Sluice water is directed to holdup tanks for processing).	3,800
Total	810,900

Part B: Component Shielding and Process Points Activities Inputs(2)

Source	
1. Boron Dilution to adjust for fuel burnup	237,750
2. Cold shutdown and startup at 60% of core life	75,600
3. Hot shutdowns and startups (2 Per Yr. @ 55%, 65% of core life)	128,300
4. Refueling (1 Per Yr. to refuel reactor)	104,150
5. Leakage to reactor drain tank (200 gal/day 292 days/yr)	58,400
6. Back-to-back cold shutdown at 80% of core life	216,450
Total	820,650

Part C: Anticipated Operational Occurrence Waste Input (3)

Back-to-back cold shutdown at 80% of core life 65 hours, 55.5 gpm average flow	216,466

(1) These inputs form the basis for BMS equipment capacity sizing and system operation.

(2) These inputs form the basis for the shielding source terms and process point activities, as noted in Subsection 11.2.1, Item (j).

(3) This input was used in evaluating the ability of the LWMS to handle waste surges.

TABLE 11.2-7

SOURCES AND ESTIMATED VOLUMES OF LWS WASTE INFLUENTS

Source	Waste Generating Operation	Volume gallons/year) Expected	Maximum*
Equipment drains and leakage	75 gal/day	28,000	1,000
Sample and laboratory sink drains	20 gal/day	7,000	500
Equipment decontami-nation	10 gpm for 20 minutes per day	73,000	3,000
Floor drains	5 gpm for 10 minutes per day	18,000	1,000
Fuel cask washdown	400 gal/cask per refueling	30,000	N/A
Sub-total		156,000	
Laundry	200 gal/day	73,000	2,000
Showers	4 Showers per day at 30 gal per shower	44,000	1,000
Turbine building floor drains	Secondary system leakage 5 gpm	2,628,000	50,000
Total	Estimated Normal Operation	2,901,000	

*Maximum volume is in gal/event.

TABLE 11.2-8

SPECIFIC ACTIVITIES OF LWS INFLUENTS

Source	Fraction of RCS
Equipment Drains	0.2(1)
Sample and Laboratory Sink Drains	0.02(1)
Equipment Decontamination	0.1(1)
Floor Drains	0.2(1)
Fuel Cask Washdowns	.001(2)
Laundry	10^{-4}(3)
Showers	10^{-4}(3)
Steam Generator Blowdown System	(4)
Turbine Building Floor Drains	10^{-6}(1)

(1) Based on ANSI N199, November 1975

(2) C-E Assumption

(3) Based on WASH 1258, July 1973

(4) See Steam Generator Liquid Concentration per Table 11.1-16

TABLE 11.2-9

BORATED WASTE TRAIN PROCESS FLOW DATA

Mode #1	Processing RDT Contents [1]							
Location:	1	2	3	4	5	6	7	8
Flow (gpm)	50	50	50	50	50	50	50	50
Pressure (psig)	0.5	0.5	67	67	0.5	0.5	67	65
Temperature (°F)	120	120	120	120	120	120	120	120

Mode #2	CVCS Normal Purification VCT Diversion Processing[1]				
Location:	4	5	6	7	8
Flow (gpm)	40	40	40	40	40
Pressure (psig)	60	0.5	0.5	67	65
Temperature (°F)	120	120	120	120	120

Mode #3 Processing Holdup Tank Contents Via the Boric Acid Concentrator [2][4]							
Location:	12	13	14	15	16	17	18
Flow (gpm)	20	20	20	20	0-2	20	20
Pressure (psig)	60	60	60	60	20	60	67
Temperature (°F)	120	120	120	120	170	140	140

Mode #4	Discharging Boric Acid Condensate Tank Contents[3][4]		
Location	19	20	21
Flow (gpm)	50	50	50
Pressure (psig)	0	60	60
Temperature (°F)	120	120	120

Mode #5	Pumping BAHT Contents to the BAMT in the CVCS [3][4]	
Location:	22	23
Flow (gpm)	50	50
Pressure (psig)	0	60
Temperature (°F)	170	170

(1) See Figure 11.2-1 for process point location.

(2) Pts. 12-15 are found on Figure 11.2-7, Pts. 16-18 are found on Figure 11.2-5.

(3) Pts. 19-23 are found on Figure 11.2-5.

(4) This path is no longer used since equipment is no longer used.

TABLE 11.2-10

GENERAL WASTE TRAIN PROCESS FLOW DATA

Mode #1 Processing EDT Contents Via the Waste Concentrator[4]							
Location:	1	2	3	4	12	13	14
Flow (gpm)	*2,784,000	*2,784,000	20	20	20	20	20
Pressure (psig)	0	0	0	167	67	60	59
Temperature (°F)	120	120	120	120	120	120	120

Mode #2 Processing CDT Contents Via the Waste Concentrator[4]						
Location:	5	6	7	12	13	14
Flow (gpm)	*110,000	20	20	20	20	20
Pressure (psig)[3]	0	0	67	66	60	59
Temperature (°F)	120	120	120	120	120	120

Mode #3 Discharging the LDT's				
Location:	8	9	10	11
Flow (gpm)	*197,080	50	50	57
Pressure (psig)[3]	0	0	67	67
Temperature (°F)	120	120	120	120

Mode #4 Discharging W.C.T.'s[4]			
Location:	15	16	21[21]
Flow (gpm)	50	50	50
Pressure (psig)[3]	0	67	20
Temperature (°F)	120	120	170

Mode #5 Transfer of Concentrator Bottoms to the Concentrator Bottoms Tank [4]	
Location:	21
Flow (gpm)	1
Pressure (psig)[3]	20
Temperature (°F)	170

* Number indicates annual flow rate, in gallons/year.

(1) Process Flow Points 1 thru 12 are located on Figure 11.2-6. Process Flow Points 13 thru 21 are located on Figure 11.2-9.
(2) This Point 21 is in the Boron Management Subsystem located on Figure 11.2-5.
(3) The pressure drop across filters and ion exchangers will vary with loading. The drops indicated are typical.
(4) This path is no longer used since equipment is no longer used.

Mode #6 Transfer of SRT Contents to Outside Shipping Container		
Location:	20	
Flow (gpm)	100	
Pressure (psig)	60	
Temperature (°F)	120	

Mode #7	Transfer of Spent Resin to the SRT	
Location:	17	
Flow (gpm)	100	
Pressure (psig)	60	
Temperature (°F)	120	

Mode #8	Dewatering the SRT Contents(1)	
Location:	18	19
Flow (gpm)	100	100
Pressure (psig)	0	60
Temperature (°F)	120	120

Notes:
(1) This information is considered to be historical.

TABLE 11.2-11

DESIGN PROVISIONS TO CONTROL RELEASE OF RADIOACTIVE MATERIALS
DUE TO OVERFLOW FROM ALL LIQUID TANKS OUTSIDE CONTAINMENT

	TANK	LEVEL DETECTION	ALARM	OVERFLOW	OVERFLOW ROUTED TO	CONTROL FUNCTION
(1)	Flash Tank	LIC 6603, 6604	Hi and Lo	None	-	Stops Flash Tank Pump on low level. Bypass to Holdup Tank on high level. Controls Flash Tank Pump discharge valve.
(2)	Hold Up Tanks 2A, 2B, 2C, 2D	LIC 6607, 6608, 6609 and 6610	Hi and Lo	Yes	No ED. Each compartment provided with 3" FD routed to Sump Tank 2B	Stops all holdup pumps on low level.
(3)	Boric Acid Condensate(a) Tank 2A, 2B	LIC 6624, 6626	Hi and Lo*'	Yes	3" ED for each tAnk routed to Sump Tank 2B	Stops Boric Acid Condensate Pumps on low level
(4)	Boric Acid holding Tank (a)	LIC 6622	Hi and Lo'*	Yes	No ED 3", normally plugged FD routed to Sump Tank 2B	Stops Boric Acid Holding Pump on low level
(5)	Equipment Drain Tank	LIC 6634	Hi and Lo'*	Yes	1-1/2" line to Chem Drain Sump	Stops associated pump on low level
(6)	Chemical Drain Tank	LIC 6635	Hi and Lo'*	Yes	1-1/2" line routed to Chemical Drain Sump Tank	Stops associated pump on low level
(7)	Laundry Drain Tanks 2A, 2B	LIC 6636, 6637	Hi and Lo*	Yes	1-1/2" line for each tank routed to laundry drain Sump Tank	Stops associated pump on low level
(8)	Waste Condensate Tanks 2A, 2B (a)	LIC 6640, 6641	Hi and Lo'*	Yes	Two 3" ED routed to Sump Tank 2A	Stops associated pump on low level
(9)	Spent Resin Tank	LIA 6644	Hi'*	None	4" ED routed to Sump Tank 2B	None
(10)	Resin Addition Tank	-	---	None	---	---
(11)	Aerated Waste Storage Tank	LIC 6632	Hi'*	Yes	3" ED routed to Sump Tank 2A	Stops associated pump on low level
(12)	Concentrator Bottoms Tank (a)	LI 06-50	Hi'*	None	---	Alarm Function only
(13)	Concentrator Bottoms Tank (a)	LS 06-51	None	None	---	Stops associated pump on low level

Note: * Local Tank WMS annunciator panel, all others in control room.
ED = Equipment Drain
FD = Floor Drain
EDT = Equipment Drain Tank

(a) Equipment is no longer used.

TABLE 11.2-12

ESTIMATED LIQUID WASTE INPUTS

STREAM	FLOW RATE GAL/DAY	FRACTION OF RCA	FRACTION DISCHARGED	COLLECTION TIME (DAYS)	DECONTAMINATION FACTORS[3] I	CS	OTHERS
Shimbleed Rate	2.78E+03	1.000	0.1	46.0	1E+05	2E+03	1E+04
Equipment Drains	9.60E+01	0.200	0.1	3.1	5E+02	1E+03	5E+04
Clean Wastes	2.74E+02	0.093	1.0	2.9	5E+02	1E+03	5E+04
Dirty Wastes	1.65E+02	0.076	1.0	3.1	5E+02	1E+03	5E+04
Blowdown	5.75E+04	(4)	1.0	0.0	1E+02	1E+02	1E+02

Notes

(1) See Tables 11.2-6 & 11.2-7
(2) See Table 11.2-8
(3) See Table 11.2-2
(4) See Steam Generator Liquid Concentration per Table 11.1-16.

TABLE 11.2-13

MAXIMUM INDIVIDUAL DOSES FROM EXPOSURE
TO NORMAL OPERATIONAL LIQUID RADIOLOGICAL RELEASES (Estimated)
(Information provided in this table is historical)

ADULT DOSE (MREM PER YEAR INTAKE)

Pathway	SKIN	BONE	LIVER	TOTAL BODY	THYROID	KIDNEY	LUNG	GI-LLI
Fish		4.43E-04	8.78E-04	6.60E-04	6.35E-03	3.73E-04	1.74E-04	3.11E-03
Invertebrate		1.18E-03	8.22E-04	6.69E-04	8.12E-03	6.08E-03	4.98E-05	2.63E-02
Shoreline	7.90E-04	6.76E-04	6.76E-04	6.76E-04	6.76E-04	6.76E-04	6.76E-04	6.76E-04
Swimming	0.	3.12E-06	3.12E-06	3.12E-06	3,12E-06	3.12E-06	3.12E-06	3.12E-06
Boating	0.	1.56E-06	1.56E-06	1.56E-06	1.56E-06	1.56E-06	1.56E-06	1.56E-06
Total	7.90E-04	2.30E-03	2.38E-03	2.01E-03	1.52E-02	7.14E-03	9.05E-04	3.01E-02

TEENAGER DOSE (MREM PER YEAR INTAKE)

Pathway	SKIN	BONE	LIVER	TOTAL BODY	THYROID	KIDNEY	LUNG	GI-LLI
Fish		4.41E-04	8.48E-04	4.02E-04	5.76E-03	2.84E-04	1.64E-04	2.06E-03
Invertebrate		1.24E-03	8.25E-04	6.46E-04	7.42E-03	4.62E-03	4.25E-05	2.06E-02
Shoreline	4.41E-03	3.77E-03	3.77E-03	3.77E-03	3.77E-03	3.77E-03	3.77E-03	3.77E-03
Swimming	0.	3.12E-06	3.12E-06	3.12E-06	3.12E-06	3.12E-06	3.12E-06	3.12E-06
Boating	0.	1.56E-06	1.56E-06	1.56E-06	1.56E-06	1.56E-06	1.56E-06	1.56E-06
TOTAL	4.41E-03	5.46E-03	5.45E-03	4.83E-03	1.70E-02	8.68E-03	3.98E-03	2.65E-0

CHILD DOSE (MREM PER YEAR INTAKE)

	SKIN	BONE	LIVER	TOTAL BODY	THYROID	KIDNEY	LUNG	GI-LLI
Pathway		5.30E-04	7.17E-04	2.01E-04	6.02E-03	1.22E-04	1.28E-04	7.73E-04
Invertebrate		1.55E-03	7.31E-04	6.97E-04	8.16E-07	2.07E-03	3.04E-05	9.14E-03
Shoreline	9.22E-04	7.88E-04	7.88E-04	7.88E-04	7.88E-04	7.88E-04	7.88E-04	7.88E-04
Swimming	0.	1.74E-06	1.74E-06	1.74E-06	1.74E-06	1.74E-06	1.74E-06	1.74E-06
Boating	0.	8.71E-07	8.71E-07	8.71E-07	8.71E-07	8.71E-07	8.71E-07	8.71E-07
TOTAL	9.22E-04	2.87E-03	2.24E-01	1.69E-03	1.50E-02	2.98E-03	9.50E-04	1.07E-02

Note: The radioactive effluent monitoring program as described in the St. Lucie Offsite Dose Calculation Manual is used to determine the dose to members of the public during reactor operation.

Refer to Drawings
2998-G-078 SH. 160A & B

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOW DIAGRAM
WASTE MANAGEMENT SYSTEM

FIGURE 11.2-1

Refer to Drawing
2998-G-078 SH. 168

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOW DIAGRAM
WASTE MANAGEMENT SYSTEM

FIGURE 11.2-2

Refer to Drawings
2998-G-078 SH. 165A & B

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOW DIAGRAM
WASTE MANAGEMENT SYSTEM

FIGURE 11.2-3

Refer to Drawings
2998-G-078 SH. 166A & B

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOW DIAGRAM
WASTE MANAGEMENT SYSTEM

FIGURE 11.2-4

Refer to Drawings
2998-G-078 SH. 161A & B

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOW DIAGRAM
WASTE MANAGEMENT SYSTEM

FIGURE 11.2-5

Refer to Drawing
2998-G-078 SH. 162

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOW DIAGRAM
WASTE MANAGEMENT SYSTEM

FIGURE 11.2-6

Refer to Drawing
2998-G-078 SH. 167A & B

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOW DIAGRAM
RADIOACTIVE WASTE CONCENTRATOR
WASTE MANAGEMENT SYSTEM

FIGURE 11.2-7

Refer to Drawing
2998-G-078 SH. 171

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOW DIAGRAM
WASTE MANAGEMENT SYSTEM

FIGURE 11.2-8

Refer to Drawing
2998-G-078 SH. 169

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOW DIAGRAM
WASTE MANAGEMENT SYSTEM

FIGURE 11.2-9

11.3 GASEOUS WASTE SYSTEM

Estimations of Gaseous Waste System volumes, radioactivity concentration, and offsite doses presented in the tables and text of this section are based on calculational data or on similar systems in use at other plants, prior to operation of St. Lucie Unit 2 and is retained here for historical purposes.

Continued compliance with the annual regulatory dose limits following core power uprate has been demonstrated using scaling factors that address NUREG-0017 equations and assumptions and the reported gaseous effluent and dose data during the years 2003 to 2007, taking into consideration the associated annual average core power level during that period extrapolated to 100 percent availability. For the uprate condition, the system parameters used reflected the flow rates and coolant masses at an NSSS power level of 3034 MWt and a core power level of 3030 MWt. To estimate an upper bound impact on off-site doses, the highest factor found for any chemical group pertinent to the release pathway was applied to the average doses previously determined as representative of operation at pre-uprate conditions. This approach was utilized to estimate the maximum potential increase in effluent doses due to the uprate and to demonstrate that the estimated off-site doses following the uprate, although increased, will continue to remain significantly below the annual design objectives for gaseous radwaste effluents set by 10 CFR 50 Appendix I and 40 CFR 190.

It is noted that for an operating plant, the actual performance and operation of installed equipment, the reporting of actual offsite releases and doses, and compliance with the regulatory limits of 10 CFR 50 Appendix I and 40 CFR 190 is controlled by the Offsite Dose Calculation Manual.

Actual data on gaseous waste effluents and dose to the public resulting from plant operation is presented annually in the Annual Radiological Effluent Release Report.

11.3.1 DESIGN BASES

Radioactive gaseous wastes which are to be discharged from the plant are first collected by the Gaseous Waste Management System (GWMS). This system is referred to in the Unit 2 Technical Specification definitions as the Gaseous Radwaste Treatment System. The GWMS design bases are as follows:

a. The principal design objective of the GWMS is to protect plant personnel, the general public, and the environment by ensuring that all releases of radioactive gases both in the plant and to the environment are as low as is reasonably achievable (ALARA).

b. The principal design criteria for the GWMS is that it provides for the handling of the plant's gaseous wastes in such a manner as to meet the requirements of 10 CFR 20 and 10 CFR 50, Appendix I.

c. The St. Lucie Unit 2 Environmental Report (CP) in Amendment 7, dated October 1975 and Amendment 8 dated June 1976 provides a detailed evaluation to show that the GWMS is capable of controlling releases of radioactive materials within the numerical design objective of Appendix I to 10 CFR 50. A review of the plant design and site usage characteristics reveals that no significant change has

occurred which would require a re-evaluation (see Subsection 11.3.3). Amendment 8 of the ER (CP) also provides a cost benefit analysis in Subsection 10.7.8 which is still applicable. The NRC, however, elected to evaluate the final design of the GWMS based on the requirements of the September, 1975 Annex to Appendix I.

d. The estimated annual GWMS releases are summarized in Tables 11.3-1 and 2. These are based on the process points activities for the GWMS, (see item (j)) specifically the inventories for the gas decay tanks adjusted to account for decay due to holdup. (The basis for these activities are the Reactor Coolant System activities given in Section 11.1 and releases from liquid tanks to the gas collection header.)

e. The individual component design parameters are given in Table 11.3-3.

f. The expected inputs to the GWMS are listed on Table 11.3-4. Surges to the GWMS may be handled by utilizing the holdup capability of the gas decay tanks or by increasing the rate of filling the gas decay tanks.

g. The design, quality assurance, construction and testing criteria for the GWMS meet or exceed the guidelines of Regulatory Guide 1.143, "Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants," October, 1979 (Rev. 1) particularly Position 2 (Gaseous Radwaste Systems) Position 4 (Additional Design, Construction and Testing Criteria) and Position 6 (Quality Assurance for Radwaste Management Systems). Furthermore, the seismic design classification of the structures housing the GWMS meets the guidelines of Regulatory Guide 1.143 (Rev 1) and is described in Section 3.2.

h. General design criteria 60 and 64 are met.

i. Redundant waste gas compressors insure that at least one compressor is available to keep the holdup capability of the GWMS functioning properly when needed. Multiple gas decay tanks allow for sufficient flexibility so that one tank can be filled while another is discharged and a third holds up gas for decay.

j. Specific activities for the influent streams to the GWMS are given in Table 11.3-5. The locations of the points are noted on Figure 11.3-1, as circled numbers corresponding to groups of data given in Table 11.3-6.

   The basis for these values are the Reactor Coolant System activities given in Section 11.1 and are calculated applying appropriate component decontamination factors (Table 11.2-2), the component parameters of Table 11.3-3, the waste volumes of Table 11.3-4, the flow paths shown on Figure 11.3-1, the process flow data of Table 11.3-6, and the process activities of Section 11.2 (Table 11.2-8).

   The radionuclide inventories themselves serve as the basis for the GWMS releases (see item (d) and Tables 11.3-1 and 2).

k. Gaseous releases due to equipment malfunction or operator error are mitigated by several design features. To discharge a gas decay tank, the operator must manually open the gas decay tank isolation valve and "activate to open" the discharge line isolation valve. Also, should activity in the gaseous effluent being discharged exceed a predetermined level, the discharge line radiation monitor signals the isolation valve to close. Radiation monitor failure also closes the isolation valve. The isolation valve itself fails closed. There are also manually operated valves available to cut off flow. These features combine to create a reliable discharge control scheme which prevents in advertant radioactive gaseous releases.

   The Process and Effluent Radiological Monitoring and Sampling System is described in Section 11.5.

l. HEPA filtration is provided in the Reactor Auxiliary Building and Fuel Handling Building exhausts to collect radioactive material in the particulate form. The system is capable of reducing the radiation dose below the limits of 10 CFR 20 and 50.

m. The GWMS is not designed to withstand the effects of an explosion. Rather, it is designed to detect and preclude the formation of potentially explosive mixtures of hydrogen and oxygen by an extensive combination of design features. The gas analysis: can be performed by two automatic, continuously operating monitors in the GWMS; is made of nonsparking components; uses fail-closed valves for potential oxygen sources; and alarms on local panels and in the main control room. The GWMS is designed to operate below four percent oxygen; thereby, the system is analyzed for oxygen. Both of the gas analyzers are provided with automatic control functions each of which can independently supply measurements verifying that oxygen is not present in potentially explosive concentrations. Alarms annunciate both locally and in the control room at "High alarm" and "High-high alarm", setpoints. "High alarm" is set at 1.0% oxygen by volume for Continuous Oxygen Analyzer and 1.75% oxygen by volume for Auto Gas Analyzer. "High-high alarm" is set at two percent oxygen by volume for both the Continuous Oxygen and the Auto Gas Analyzers. Control features to reduce the potential for explosion are automatically initiated at the "High-high alarm" setting. The control features are as follows: (1) automatic isolation of the source of oxygen from the system, through the use of a fail-closed valve; (2) introduction of Nitrogen as a diluent to reduce concentrations below the four percent by volume limit specified.

   The system uses pressurized Gas Decay Tanks (GDTs) when additional radioactive decay of effluent prior to release is required; therefore, a gas analyzer sample point is provided between the Waste Gas Compressors and the GDTs. The Auto Gas Analyzer is used to sample the in-service GDT. The Continuous Oxygen Analyzer can be used to sample the combined stream of the sources at the gas surge header. The Auto Gas Analyzer can also be manually aligned to measure several other points in the system.

Both gas analyzers are non-sparking and have daily sensor checks, monthly functional checks and quarterly calibrations.

n. The GWMS is designed for operator ALARA as described in Section 12.3.

11.3.2 SYSTEM DESCRIPTION

The design of the Gaseous Waste Management System (GWMS) is shown on Figure 11.3-1 and 11.3-1a. Principal flow paths (heavy lines) through the system are clearly indicated in the figure. Process data is presented in Table 11.3-6. System component design parameters are contained in Table 11.3-3.

Waste gases which are routed to the GWMS are mainly radioactive or potentially radioactive gases from various sources throughout the plant. Gaseous wastes are generated from reactor coolant degassing operations, processing of radioactive liquid wastes and tank purgings. Waste gases enter the GWMS by way of three headers; the gas collection header (GCH), the containment vent header (CVH) and the gas surge header (GSH).

The gas collection header receives low activity gases containing oxygen from aerated tanks and components. Sources and volumes to the gas collection header are given in Table 11.3-7. These gases are then directed to the plant vent for monitoring and discharge. The containment vent header collects potentially radioactive waste gases from the reactor drain tank and the quench tank. The gas surge header collects the radioactive gases with negligible oxygen content from the flash tank, boric acid concentrator*, the containment vent header, the volume control tank, and both of the gas analyzer discharges. Sources, volumes, and flow rates to the gas surge header are given in Table 11.3-4. The activities of these sources are given in Table 11.3-5.

Gases from the gas header flow into the gas surge tank where they are collected. The gas can remain in the gas surge tank until the pressure builds to a point which actuates a single waste gas compressor or aligned to the plant vent for monitoring and discharge. For waste gas holdup operation, the waste gas compressor feeds a preselected gas decay tank until the pressure in the gas surge tank drops to a point where the waste gas compressor stops. A second waste gas compressor starts if the pressure in the gas surge tank increases above a certain level. This automatic operation of the waste gas compressors continue until a gas decay tank is observed to approach its upper operating pressure. At this point another gas decay tank is manually lined up to receive the waste gas compressor's discharge, and the first tank is isolated. The just filled gas decay tank is analyzed by the gas analyzer for oxygen content. Grab samples can also be taken for radioactivity analysis.

To reduce the possibility of water intrusion into the Gas Surge Tank (GST), the gas surge header has been provided with a drain. This drain is located at a point downstream of all the influent lines, except for the flash tank vent line and is directed to the flash tank. In addition, the GST contains three liquid level switches which enables monitoring of liquid in the GST. In the event of Hi-Hi liquid level In the GST, a level switch trips the waste gas compressor. A second switch annunciates locally on Hi level when GST draining is required. A third switch operates an indicating light which advises an operator to cease draining. This light is off when the liquid level is below the low level point.

*Note: The boric acid concentrators are no longer used.

The gas analyzer package is provided to monitor oxygen concentrations in various plant components where potentially explosive mixtures could develop. The Auto Gas Analyzer can be aligned to sample the following sources: gas surge header, flash tank, holdup tanks, waste gas compressor effluent, gas decay tanks and volume control tank. The Auto Gas Analyzer is normally selected to continuously monitor a single sample point, i.e., the in-service GDT. The sampling of any source includes line purging, sample analysis, and results display. Any oxygen concentration above a predetermined setpoint ("high-high alarm") will initiate alarms on the gas analyzer panel, and in the main control room. If automatic analyzer operation is interrupted, samples can be obtained from the grab sample port on the analyzer, and from local sample lines at selected locations in the GWMS. The sample port on the analyzer will allow manual sample collection if the analyzer pump or instrumentation is inoperable.

A second Continuous Oxygen Analyzer is incorporated into the GWMS. The Continuous Oxygen Analyzer is located on the Gas Surge Tank influent line. The Continuous Oxygen Analyzer provides alarms at the "high-high alarm" set point and provides automatic control functions at the high-high oxygen concentration to preclude the admission of excess oxygen to the GWMS, as does the Auto Gas Analyzer Package. For either unit, if the oxygen concentration reaches a preset level ("high-high alarm"), the unit will automatically isolate the influent sources (through the use of a fail-close valve) and an alarm is also annunciated. All alarms are annunciated locally, and in the control room.

The combination of the Auto Gas Analyzer and the Continuous Oxygen Analyzer provides a degree of redundant monitoring of the GWMS. First, the Auto Gas Analyzer normally provides continuous monitoring of the in-service gas decay tank, but can be aligned to several other sample points, including the gas surge header. Second, the Continuous Oxygen Analyzer provides redundancy by continuously monitoring the influent, combined stream to the Gas Surge Tank (i.e., gas surge header). Both gas analyzers are also provided with the means to introduce a diluent (nitrogen) to the system in order to lower the O_2 concentration, thus preventing the reaching of flammability limits.

For both gas analyzers in the GWMS, all components in contact with the process stream are non-sparking.

The gas which has been held in the isolated gas decay tank, is allowed to decay for a period of time to reduce the activity of the gas. GWMS design provides for an average 25 day holdup for all gaseous wastes, even during periods of waste surges. (For conservatism, a 9.3 day collection time with a 9.3 day holdup time is used in calculating the releases in Subsection 11.3.3).

The only process flow bypass line that exists in the GWMS leads from the gas surge tank directly to the gas discharge header and bypasses the waste gas compressor and gas decay tanks. This flow path is used when Chemistry guidelines have determined that no holdup of gaseous effluent is required prior to release, or when it is necessary to purge air from components after maintenance operations. The valve on this bypass line is locked closed to facilitate administrative control. The bypass flow passes through the radiation monitor in the gas discharge line.

Independent supply headers for the hydrogen and nitrogen required for plant operations are provided. Hydrogen gas is supplied to the volume control tank gas space to maintain the desired concentration of reactor coolant dissolved hydrogen to suppress the net decomposition of water in the reactor. Nitrogen cover and/or purge gas is provided to the holdup tanks, quench

tank, reactor drain tank, safety injection tanks, spent resin tank, gas surge tank and gas decay tanks. A nitrogen stream is supplied to the flash tank for degassing liquid waste when the flash tank is operated, and periodic purges with nitrogen are provided as required for various Waste Management System and Chemical and Volume Control System components. The two gas supply systems include relief valves, regulators, and instrumentation with alarms. Check valves are provided in both gas supply headers to provide a means of preventing back leakage of contaminated gases to the uncontaminated systems.

Description of ventilation systems serving Reactor Building, Reactor Auxiliary Building, Fuel Handling Building and Turbine Building with potential for discharging radioactive gaseous waste are discussed in Section 9.4. Radioactive sources for these areas are discussed in Section 12.2.

The operation and description of the Main Condenser Evacuation System and Turbine Gland Sealing System are discussed in Subsections 10.4.2 and 10.4.3 respectively.

11.3.3 RADIOACTIVE RELEASES

The expected gaseous releases during normal operations, including anticipated operational occurrences from plant release points, per nuclide, are shown in Tables 11.3-1 and 2. The assumptions are presented in Table 11.3-8 and the values are calculated in accordance with NUREG-0017, April 1976 (R1). The calculated average annual airborne radioactivity concentrations, at the site boundary, are compared with the limits of 10 CFR 20 in Table 11.3.9.

The tritium released through the ventilation exhaust system during normal operation is also calculated. The annual quantity of tritium available for release during normal operation, is calculated using a functional relationship derived from measured liquid and vapor tritium releases at operating PWRs and the integrated thermal power output during the calendar year in which the release occurs. This relationship expresses total tritium as a function of power output, as discussed in NUREG-0017 (R1). It is assumed that the tritium released through the ventilation exhaust systems is the total tritium available for release minus the tritium calculated to be released through the liquid pathway.

With a design basis fuel leakage for an entire year, the annual release of radionuclides in the gaseous effluents is given in Table 11.3-10. The concentrations at the site boundary of the release and the fraction of 10 CFR 20 maximum permissible concentrations are also provided in Table 11.3-10.

The analysis and demonstration of compliance with Appendix I to 10 CFR 50 was submitted to the NRC in Amendments 7, October 1975 and 8, June 1976, to the Environmental Report (CP). A review of the plant design and site usage characteristics reveals that the only significant change which has occurred since these submittals is to the containment ventilation system. The NRC, however, elected to evaluate the final design of the GWMS based on the requirements of the September, 1975 Annex to Appendix I.

The containment atmospheric clean-up system has been removed, as recommended in ER (CP) Amendment 8, and a continuous low volume containment purge system added. The radioactive releases in Tables 11.3-1 and 2 include the effect of the continuous low volume containment purge system and are virtually the same as those provided in Table 10.8-3 of ER (CP) Amendment 8, therefore a reanalysis of compliance with the cost-benefit requirements of

Appendix I to 10 CFR 50 is not required. Accordingly, ER (CP) Amendments 7 and 8 should be consulted for a detailed description of the analysis.

The maximum doses to an individual are presented in Table 11.3-11. These doses differ from those presented in ER (CP) Amendment 8 in that these are based on preoperational meteorological data and critical dose locations. The site boundary air doses, not presented in ER (CP) Amendment 8, are provided in Table 11.3-12. The value of χ/Q to be used for gaseous releases at the site boundary during operation is determined from the guidance provided in the Offsite Dose Calculation Manual (ODCM).

Figure 11.3-2 presents location of all gaseous release points. Table 11.3-13 provides the height and inside dimensions of each release point along with the effluent temperature and exit velocity. For additional discussion on ventilation systems see Section 9.4 and Subsection 12.3.3.

11.3.4 CALCULATED OFFSITE DOSE

The ODCM describes the methodology used to calculate doses from offsite gaseous releases during St. Lucie Unit 2 operation. The Methodology section of the ODCM uses the models suggested by NUREG-0133 and Regulatory Guide 1.109 to provide calculation methods and parameters for determining results in compliance with the Controls section of the ODCM. Simplifying assumptions have been applied where applicable to provide a more workable document for implementing the Control requirements. Alternate calculation methods may be used from those presented in the ODCM as long as the overall methodology does not change and as long as the most up-to-date revisions of the Regulatory Guide 1.109 dose conversion factors and environmental transfer factors are substituted for those currently included in the ODCM.

TABLE 11.3-1

EXPECTED GASEOUS RELEASE RATE - CURIES PER YEAR

	Gas Stripping		Building Ventilation					
	Shutdown	Continuous	Reactor	Auxiliary	Turbine	Blowdown Vent Offgas	Air Ejector Exhaust	Total
KR-83M	0.	0.	1.0E+00	0.	0.	0.	0.	1.0E+00
KR-85M	0.	0.	1.4E+01	2.0E+00	0.	0.	1.0E+00	1.7E+01
KR-85	5.0E+00	1.9E+02	7.0E+00	0.	0.	0.	0.	2.0E+02
KR-87	0.	0.	3.0E+00	1.0E+00	0.	0.	0.	4.0E+00
KR-88	0.	0.	1.8E+01	4.0E+00	0.	0.	2.0E+00	2.4E+01
KR-89	0.	0.	0.	0.	0.	0.	0.	0.
XE-131m	6.0E+00	1.9E+02	1.4E+01	0.	0.	0.	0.	2.1E+02
XE-133M	3.0E+00	3.9E+01	5.7E+01	2.0E+00	0.	0.	1.0E+00	1.0E+02
XE-133	7.9E+02	1.9E+04	3.5E+03	1.3E+02	0.	0.	8.0E+01	2.4E+04
XE-135M	0.	0.	0.	0.	0.	0.	0.	0.
XE-135	0.	0.	7.0E+01	6.0E+00	0.	0.	4.0E+00	8.0E+01
XE-137	0.	0.	0.	0.	0.	0.	0.	0.
XE-138	0.	0.	0.	0.	0.	0.	0.	0.
Total Noble Gases								2.5E+04
1-131	0.	0.	2.5E-02	6.1E-02	7.9E-04	0.	3.8E-02	1.2E-01
1-133	0.	0.	1.9E-02	7.2E-02	8.3E-04	0.	4.5E-02	1.4E-01

Tritium Gaseous Release 594 Curies/Yr

0.0 Appearing in the table indicates release is less than 1.0 Ci/Yr for noble gas, 0.0001 Ci/Yr for I

TABLE 11.3-2

AIRBORNE PARTICULATE RELEASE RATE-CURIES PER YEAR

Nuclide	Waste Gas System	Building Reactor	Ventilation Auxiliary	Total
MN-54	4.5E-03	2.2E-04	1.8E-04	4.9E-03
FE-59	1.5E-03	7.5E-05	6.0E-05	1.6E-03
CO-58	1.5E-02	7.5E-04	6.0E-04	1.6E-02
CO-60	7.0E-03	3.4E-04	2.7E-04	7.6E-03
SR-89	3.3E-04	1.7E-05	1.3E-05	3.6E-04
SR-90	6.0E-05	3.0E-06	2.4E-06	6.5E-05
CS-134	4.5E-03	2.2E-04	1.8E-04	4.9E-03
CS-137	7.5E-03	3.8E-04	3.0E-04	8.2E-03

TABLE 11.3-3

COMPONENT DATA **

Gas Surge Tank	
Type	Vertical
Quantity	1
Volume, ft^3	9
Design pressure, psig	40
Design temperature, °F	200
Code	Class 3, ASME III, 1974 Edition with Summer 1974 Addenda
Material	Carbon Steel
Gas Decay Tank	
Type	Vertical
Quantity	3
Volume, each, ft^3	138
Design pressure, psig	190
Design temperature, °F	250
Codes	Class 3, ASME III, 1974 Edition with Summer 1974 Addenda
Material	Carbon Steel
Auto Gas Analyzer*	
Type	Automatic hydrogen/oxygen analyzer
Quantity	1
Measurement Range:	
Hydrogen	0-5%, 0-50%, 0-100% by volume
Oxygen	0-5% by volume
Sample Inlet Pressure, psig	1-20
Sample Discharge Pressure, psig	1-15
Code	Manufacturers Standard

* Auto Gas Analyzer is used to measure oxygen content only (hydrogen analyzer is not used).
** Note: Some information provided in this table reflects data used for procurement of components prior to initial plant operations.

TABLE 11.3-3 (Cont'd)

Waste Gas Compressor	
Type	Diaphragm Positive Displacement
Quantity	2
Capacity, scfm	2
Discharge pressure, psig	0-165
Code	Class 3, ASME III, 1971 Edition with Summer 1973 Addenda
Materials	Stainless Steel
Design temperature, °F	200 - Inlet; 250 - Outlet
Design pressure, psig	190
Continuous Oxygen Analyzer	
Type	Continuous Analyzer for Oxygen
Quantity	1
Measurement Range	0-5% by volume
Sample Inlet Pressure, psig	1.5-20
Sample Discharge Pressure, psig	1.5-10
Code	Manufacturer's Standard

TABLE 11.3-4

EXPECTED ANNUAL INPUTS TO THE GWMS SURGE HEADER[2]

A. Average Annual Input

Source	Input (SCF/yr)
Reactor Drain Tank	7,800
Flash Tank	22,160
Volume Control Tank	4,450
Boric Acid Concentrators	1,650
Refueling Failed Fuel Detector	2,000
	38,060

B. Surge Volumes [1]

Source	Input
Back-to-back Cold Shutdown at 67% Core Life.	
Inputs to GWMS	1.76 SCFM for 25 hours 0.06739 SCFM for 114 hours
Net Input	3100 SCF

(1) This is the maximum waste surge for which the GWMS is designed to handle.
(2) This table contains original plant Design Basis information. Some sources listed may no longer be valid inputs.

TABLE 11.3-5

GWMS PROCESS POINTS ACTIVITIES[1][2] (µCi/cc)

Nuclide	Flash Tank	Volume Control Tank	Reactor Drain Tank	Refueling Failed Fuel Detector
KR-83M	2.9(-01)*	3.2(-01)	1.7(-02)	1.7(-02)
KR-85M	1.5(+00)	1.9(+00)	9.1(-02)	9.1(-02)
KR-85	1.8(+00)	2.3(+00)	1.0(-01)	1.0(-01)
KR-87	8.5(-01)	8.3(-01)	5.0(-02)	5.0(-02)
KR-88	2.8(+00)	3.2(+00)	1.6(-01)	1.6(-01)
KR-89	7.1(-02)	9.2(-03)	4.2(-03)	4.2(-03)
XE-131M	1.6(+00)	1.1(+00)	8.6(-02)	8.6(-02)
XE-133M	3.2(+00)	2.3(+00)	1.8(-01)	1.8(-01)
XE-133	2.6(+02)	1.8(+02)	1.5(+01)	1.5(+01)
XE-135M	1.8(-01)	4.6(-02)	1.1(-02)	1.1(-02)
XE-135	5.0(+00)	3.5(+00)	2.9(-01)	2.9(-01)
XE-137	1.3(-01)	1.1(-02)	7.5(-03)	7.5(-03)
XE-138	6.1(-01)	1.4(-01)	3.6(-02)	3.6(-02)
BR-83	1.2(-06)	3.3(-07)	4.7(-06)	4.7(-06)
BR-84	6.1(-07)	1.0(-07)	2.4(-06)	2.3(-06)
BR-85	7.1(-08)	2.1(-09)	2.8(-07)	2.8(-07)
I-130	6.7(-07)	1.9(-07)	2.3(-06)	2.3(-06)
I-131	1.6(-04)	3.1(-05)	3.6(-04)	3.6(-04)
I-132	2.5(-05)	6.8(-06)	9.7(-05)	9.7(-05)
I-133	1.4(-04)	3.7(-05)	4.4(-04)	4.4(-04)
I-134	1.1(-05)	2.4(-06)	4.4(-05)	4.4(-05)
I-135	5.5(-05)	1.6(-05)	2.0(-04)	2.0(-04)

(1) Boric Acid Concentrator activities not used for released due to extremely low volume and activity.

(2) This table contains original plant Design Basis information. Some sources listed may no longer be valid inputs.

*() denotes powers of 10.

TABLE 11.3-6

GWMS PROCESS POINTS

Location:	1	2	3	4	5	6	7	8	9	10	11
Flow (SCFM)	9800*	0.3	1650*	4450*	22,160*	38,060*	2	2	10	10	10
Pressure (psig)	0.5	0.5	0.5	0.5	20	0.5	3.0	165	165	15	0
Temperature (°F)	120	120	120	120	120	120	120	120	120	120	120

*Flow for these points is SCF per year.

TABLE 11.3-7

EXPECTED ANNUAL INPUTS TO THE GWMS GAS COLLECTION HEADER(1)

Source	Volume (SCFY)
Waste Concentrator (2)	2,600
Waste Condensate Tanks	353,600
Laundry Drain Tanks	26,350
Chemical Drain Tank	14,700
Equipment Drain Tank	357,500
Boric Acid Holding Tank (2)	1,100
Boric Acid Condensate Tank (2)	9,900

(1) Items listed are those used for activity released from the plant.

(2) Item is no longer used.

TABLE 11.3-8

ASSUMPTIONS USED TO CALCULATE
RADIONUCLIDE RELEASE THROUGH THE GWMS

GASEOUS WASTE INPUTS

There is continuous low vol purge of Vol. Control Tk		
There is continuous stripping of full Letdown Flow		
Holdup Time (days) for XE From Reactor Coolant Sys		9.3
Holdup Time (days) for KR From Reactor Coolant Sys		9.3
Fill Time (days) for Holdup System for Gas Stripping		9.3
Gas Waste System	Particulate Release Fraction	1.0
Auxiliary Bldg	Iodine Release Fraction	1.0
	Particulate Release Fraction	0.01
Containment Free Volume (10**6 Ft**3)		2.5
Frequency of CNTMT Bldg High VoL Purge (Times/Yr)		4.
CNTMT-High Vol. Purge Iodine Release Fraction		1.0
	Particulate Release Fraction	0.01
CNTMT-Low Vol. Purge Rate (Cfm)		2000.0
CNTMT-Low Vol.	Iodine Release Fraction	0.1
	Particulate Release Fraction	0.01
Steam Leak to Turbine Bldg (Lbs/Hr)		1700.0
Fraction of Iodine Released From Condenser Air Ejector		
Offgas Treatment System		1.0
There is no Cryogenic Offgas System		

TABLE 11.3-9

GASEOUS RADIOLOGICAL EFFLUENT SITE BOUNDARY CONCENTRATIONS (NORMAL OPERATION)[a][d]

Isotope	Concentration (μCi/cc)	Ci/MPC[c]
Kr-83m	5.1(-14)	-[b]
Kr-85m	8.6(-13)	8.6(-6)
Kr-85	1.0(-11)	2.9(-6)
Kr-87	2.0(-13)	1.0(-5)
Kr-88	1.2(-12)	6.0(-5)
Xe-131m	1.1(-11)	2.8(-5)
Xe-133m	5.1(-12)	1.7(-5)
Xe-133	1.2(-9)	4.0(-3)
Xe-135	4.1(-12)	4.1(-5)
I-131	6.1(-15)	6.1(-5)
I-133	7.1(-15)	1.8(-5)
Mn-54	2.5(-16)	2.5(-7)
Fe-59	8.1(-17)	1.6(-8)
Co-58	8.1(-16)	2.7(-8)
Co-60	3.9(-16)	3.9(-8)
Sr-89	1.8(-17)	6.0(-8)
Sr-90	3.3(-18)	1.1(-7)
Gs-134	2.5(-16)	2.5(-7)
Cs-137	4.2(-16)	2.1(-7)
H-3	2.6(-11)	1.3(-4)
ΣC/MPC		4.4(-3)

(a) See Note in Table 11.3-12

(b) MPC not given

(c) Refer to note in Section 12.2.2 concerning present use of MPC

(d) This table contains original plant Design Basis information which represents expected maximum concentrations. Actual effluent data is reported to the NRC in the Annual Radioactive Effluent Release Report.

TABLE 11.3-10

DESIGN BASIS GASEOUS RADIOLOGICAL RELEASE[a][c]

Nuclide	Release (Ci/Yr)	Concentration (μCi/cc)	Ci/MPC[b]
Kr-85m	1.8(+2)	9.1(-12)	9.1(-5)
Kr-85	3.6(+4)	1.8(-9)	6.1(-3)
Kr-87	6.4(+1)	3.2(-12)	1.6(-4)
Kr-88	3.4(+2)	1.7(-11)	8.6(-4)
Xe-131m	3.4(+4)	1.7(-9)	4.3(-3)
Xe-133	1.0(+6)	5.1(-8)	1.7(-1)
Xe-135	1.7(+3)	8.6(-11)	8.6(-4)
I-131	1.2(+0)	6.1(-14)	6.1(-4)
I-133	1.3(+0)	6.6(-14)	1.6(-4)
Mn-54	4.7(-3)	2.4(-16)	2.4(-7)
Fe-59	1.6(-3)	8.1(-17)	1.6(-8)
Co-58	1.5(-2)	7.6(-16)	2.5(-8)
Co-60	7.3(-3)	3.7(-16)	3.7(-8)
Sr-89	3.8(-3)	1.9(-16)	6.4(-7)
Sr-90	6.5(-4)	3.3(-17)	1.1(-6)
Cs-134	1.8(-2)	9.1(-16)	9.1(-7)
Cs-137	1.1(-1)	5.6(-15)	2.8(-6)
H-3	6.48(+2)	2.8(-11)	1.4(-4)
Total C/MPC			1.8(-1)

(a) See note in Table 11.3-12

(b) Refer to note in Section 12.2.2 concerning present use of MPC

(c) This table contains original plant Design Basis information which represents expected maximum concentrations. Actual effluent data is reported to the NRC in the Annual Radioactive Effluent Release Report.

TABLE 11.3-11

INDIVIDUAL DOSES FROM GASEOUS RELEASES[5]
ALL PATHWAYS - NORMAL OPERATION
(mrem/yr)

	Adults	Teenagers	Children	Infants
Total Body				
Immersion[1]	3.9(-2)	3.9(-2)	3.9(-2)	3.9(-2)
Ground Deposition[1]	1.3(-2)	1.3(-2)	1.3(-2)	1.3(-2)
Vegetables[1]	4.7(-3)	4.3(-3)	4.0(-3)	-
Milk - Cow[2]	1.2(-3)	1.2(-3)	1.3(-3)	1.7(-3)
Milk - Goat[3]	7.7(-3)	7.6(-3)	6.7(-3)	7.7(-3)
Meat[4]	2.2(-4)	1.1(-4)	9.0(-5)	-
Skin				
Immersion[1]				
Ground Deposition[1]	1.5(-2)	1.5(-2)	1.5(-2)	1.5(-2)
Thyroid*				
Immersion	3.9(-2)	3.9(-2)	3.9(-2)	3.9(-2)
Ground Deposition[1]	1.3(-2)	1.3(-2)	1.3(-2)	1.3(-2)
Vegetables[1]	1.0(-1)	7.9(-2)	1.2(-1)	-
Milk - Cow[2]	8.2(-2)	1.2(-1)	2.5(-1)	5.9(-1)
Milk - Goat[3]	2.4(-1)	3.6(-1)	7.1(-1)	1.7(+0)
Meat[4]	4.4(-3)	3.1(-3)	4.6(-3)	-

(1) At critical residence 1.9 miles WSW.
(2) At critical cow 5.0 miles NW.
(3) At critical goat 2.2 miles SW.
(4) At critical meat animal - 3.2 miles W.
(5) This table contains original plant Design Basis information which represents expected maximum values. Actual effluent data is reported to the NRC in the Annual Radioactive Effluent Release Report.
(*) All other organ doses are lower than thyroid doses.

TABLE 11.3-12

SITE BOUNDARY ANNUAL AVERAGE AIR DOSES*(1)

Gamma Dose	4.6(-1) mrad/yr
Beta Dose	1.3(+0) mrad/yr

*All values based on the highest average atmospheric dispersion factor at the EZ over a period of record of 9/1/76 to 8/31/78. The χ/Q is 1.6(-6) sec/m^3. This value is terrain and recirculation corrected, see Subsection 2.3.5 for additional discussion of atmospheric dispersion.

(1) This table contains original plant Design Basis information which represents expected maximum values. Actual effluent data is reported to the NRC in the Annual Radioactive Effluent Release Report.

TABLE 11.3-13

GASEOUS EFFLUENT RELEASE POINT PARAMETERS

Containment Building Purge

Location - Plant Vent (see Figure 11.3-2)

Height - Elevation 203 feet (1)

Inside Diameter - 70.75 inches

Temperature - 120 °F at 93 °F ambient(2)

Exit Velocity - 3184 ± 10% feet/minute(3)
4631 ± 10% feet/minute(4)

Reactor Auxiliary Building (Normal Operation)

Location - Plant Vent (see Figure 11.3-2)

Height - Elevation 203 feet

Inside Diameter - 70.75 inches

Temperature - 104 °F at 93 °F ambient(2)

Exit Velocity - 3184 feet/min(3)

Fuel Handling Building

Location - Plant stack above FHB (see Figure 11.3-2)

Height - Elevation 109.5 feet

Inside Dimensions - 41.0 inch diameter

Temperature - 104 °F at 93 °F ambient

Exit Velocity - 2671.5 feet/minute

Turbine Generator Building

Location - Steam Jet Air Ejector (see Figure 11.3-2)

Height - Elevation 71.5 feet

Inside Diameter - 8 inches

TABLE 11.3-13 (Cont'd)

Turbine Generator Building (Cont'd)

Temperature - 160 °F

Exit Velocity - 2253 feet/minute

NOTES

(1) All release points are above mean low water. Plant Grade = +18' MLW

(2) When ambient temperatures are 70 °F or above, the effluent temperature is generally 11 °F degrees greater than ambient.

(3) This exit velocity is based on the combined CFM flow of both the Continuous Containment Purge and the RAB Ventilation main exhaust.

(4) This exit velocity is based on the combined cfm flow of both the Containment Purge (refueling) and the RAB Ventilation main exhaust.

Refer to Drawings
2998-G-078 SH. 163A & B

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOW DIAGRAM
WASTE MANAGEMENT SYSTEM

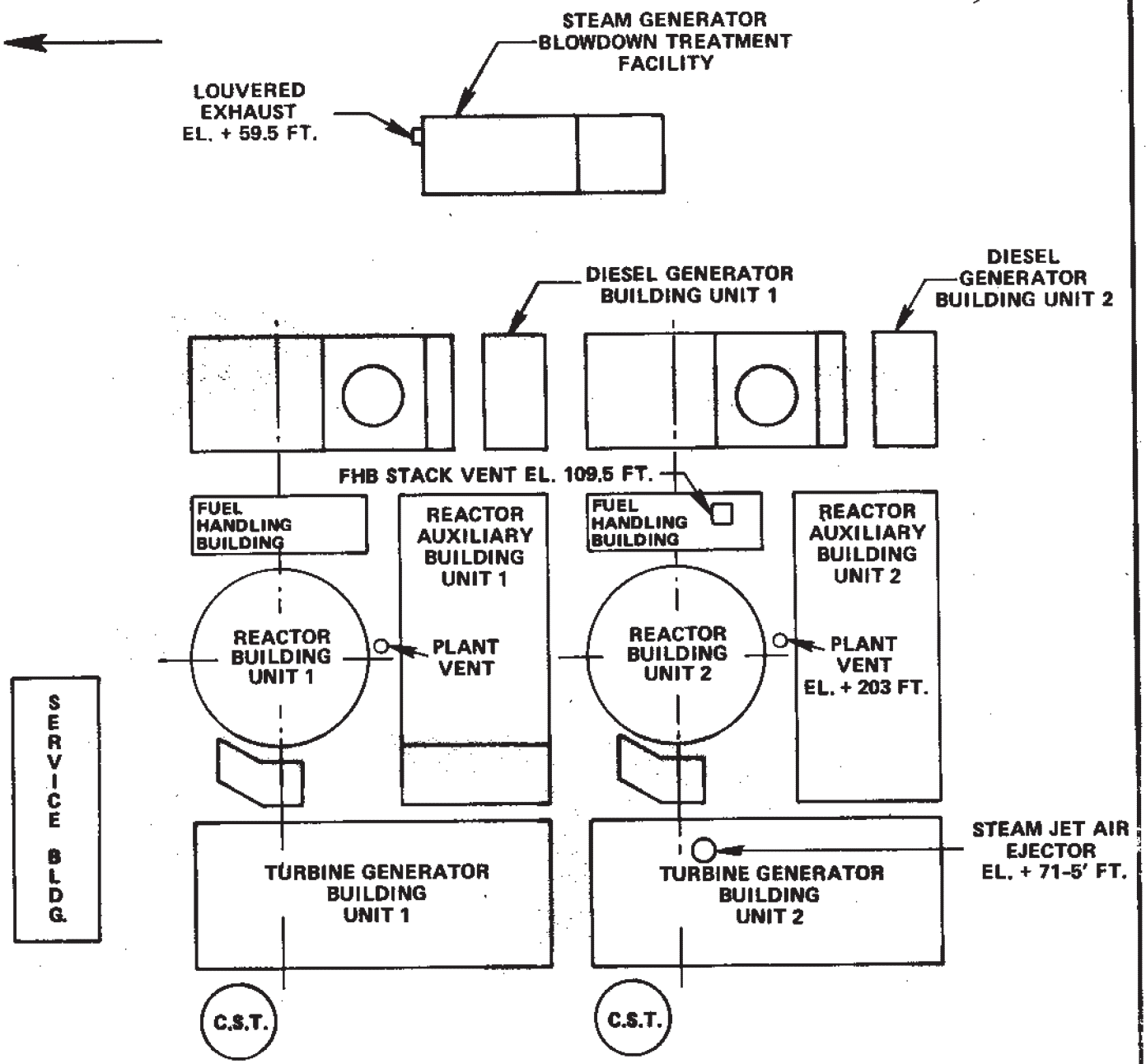
FIGURE 11.3-1

Refer to Drawing
2998-G-078 SH. 164

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

FLOW DIAGRAM
WASTE MANAGEMENT SYSTEM

FIGURE 11.3-1a



NOTE: ALL RELEASE POINTS ARE ABOVE
MEAN LOW WATER.
PLANT GRADE = 18' MLW.

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

VENTILATION RELEASE
POINTS
FIGURE 11.3-2

11.4 SOLID WASTE MANAGEMENT SYSTEM

The Solids Waste Management System (SWMS) collects, controls, processes, packages, handles, and temporarily stores solid radioactive waste and certain liquid radioactive waste generated as a result of normal operation of the plant, including anticipated operational occurrences. Spent resins are expected to be sluiced to a shipping container and dewatered with a portable dewatering system. When solidification is required, procedures will be developed in accordance with the Process Control Program.

Estimations of the Solid Waste System forms, offsite doses and quantities reported in the text and tables of this section are based on calculational data prior to operation of St. Lucie Unit 2. Operation at power uprate conditions is expected to have minimal impact on installed equipment performance, system operation and maintenance. Thus, only minor changes are expected in waste volume generation. However, the activity levels for most of the solid waste will increase proportionately to the increase in long-life coolant activity, bounded by the percentage increase in power level.

Actual data on solid waste effluents obtained during operation is presented annually in the Annual Radiological Effluent Release Report.

11.4.1 DESIGN BASES

The objective of the SWMS is to process solid wastes from the Liquid Waste Management System (liquid waste), the Chemical and Volume Control System and the Fuel Pool Purification System. The SWMS also collects, packages, stores, and prepares for transport to an offsite disposal facility any disposable solid radwaste (e.g., contaminated clothing, rags, paper, lab equipment, and supply items) generated in the operation of the plant.

In order to accomplish these design objectives, the following specific criteria are satisfied:

a. The SWMS provides for the collection, processing, packaging, and storage of solid wastes resulting from plant operations without limiting the operation or availability of the plant. Types of wastes, quantities (maximum and expected volumes) and radionuclide distributions given in Table 11.4-1 as inputs to the SWMS are accommodated in the system design. The bases for the values are included on the table.

b. The SWMS storage area is capable of providing sufficient space to allow liners and drums to be temporarily stored to allow decay prior to shipment offsite in accordance with Section B (III) of Branch Technical Position ETSB 11-3 (Rev. 1).

c. The SWMS provides a reliable means of remotely handling spent resins, and filter cartridges as required. The handling of this solid radwaste will be done while maintaining the exposure levels to plant personnel within the permissible limits of 10 CFR 20, and will reduce operator radiation exposure to as low as is reasonably achievable in accordance with Regulatory Guide 8.8 (Rev. 3).

d. The SWMS prevents the release of significant quantities of radioactive materials to the environs in order to keep the exposure to the public and operating personnel within the requirements of 10 CFR 50 Appendix I and 10 CFR 20.

e. All radioactive waste is packaged (including the shipping container) in a manner which will allow interim storage onsite within the low level waste storage facility and/or shipment and disposal in accordance with 49 CFR 170-179, 10 CFR 20, 10 CFR 61, 10 CFR 71, and applicable state regulations.

f. The SWMS provides remote handling of large size shipping containers (70-200 cubic feet each). These containers are used in packaging of spent resins and filter cartridges.

g. Portions of the SWMS are located in the Reactor Auxiliary Building, which is a seismic Category I building.

   The portable dewatering system is located outside and adjacent to the drumming storage area when processing and packaging wet radioactive wastes.

h. The permanently installed equipment and piping are in accordance with Regulatory Position 4.0 of Regulatory Guide 1.143 (Rev 1). Quality assurance requirements for the permanently installed equipment and piping are in accordance with Regulatory Position 6.0 of Regulatory Guide 1.143 (Rev 1). Procurement of the portable resin dewatering system is in accordance with the requirements of NUREG-0800, Section 11.4, Branch Technical Position ETSB 11-3.

i. The SWMS also provides for onsite interim storage of low level waste within the low level waste storage facility (LLWSF).

11.4.2 SYSTEM DESCRIPTION

The SWMS consists of spent resin tank (described in Section 11.2) to receive ion exchanger resins, and piping and valve connections to a shipping container and to the ECCS sump for resin drain/dewatering operations. A filter transfer cask is also provided.

Portable dewatering systems are used to dewater and dry spent resins.

Exhausted resins from ion exchangers in the Chemical and Volume Control System, the Liquid Waste Management System and the Fuel Pool Purification System are sluiced to the spent resin tank. After storage for decay the resins are transferred to a shipping container for dewatering operations by the portable dewatering system.

The temporary increased activity in the RCS coolant as a result of zinc injection will cause an increase in resin and filter activity for approximately two cycles. Increased resin and filter usage during the first two cycles will occur, but resin depletion during cycle operation is not expected.

The St. Lucie plant site Process Control Program (PCP) implements the requirements of 10 CFR Part 50.36a and General Design Criterion 60 of Appendix A to 10 CFR 50. Specifically, the PCP applies to waste form classification of radioactive waste destined for land burial in accordance with 10 CFR 20, dewatering of bead resins for disposal and vendor supplied processes for solidification, encapsulation or absorption of liquid radioactive waste. Refer to the Process Control Program administrative procedure for details.

In order to transport the filled liner to interim storage within the low level waste storage facility and/or an offsite disposal facility, the containers and the transport vehicle are monitored for loose surface radioactivity and decontaminated as required for offsite shipment. The radioactive content of the containers is determined and additional packaging used, if necessary, to allow shipment and burial in accordance with 49 CFR 170-179, 10 CFR 20, and 10 CFR 71 and other state regulations. The expected volumes of solid waste to be shipped offsite are given in Table 11.4-3. The expected volumes of wastes to be shipped were calculated using the inputs to the Solid Waste Management System (Table 11.4-1) and a ratio of two volumes of waste to one volume of solidification material. The associated curie content, including a listing by principal nuclides is given in Table 11.4-4 for spent resins, Table 11.4-5 for filter cartridges, Table 11.4-6 for waste concentrates and Table 11.4-7 for boric acid concentrates. The activities are based on the radionuclide removed from the liquid processing streams. The basis for each waste is detailed in each respective table.

A comparison of the process capacity of the solid waste management system and its capacity utilization is provided in Table 11.4-9.

11.4.2.1 System Component Descriptions Tanks, Valves, and Piping

The SWMS consists of an installed portion and a portable resin dewatering system. The installed system consists of a spent resin tank provided with pressure and level instrumentation, connections to a shipping container, the vent gas header collection header, primary makeup water system, holdup tanks, and ECCS sump. Resin from CVCS, fuel pool and liquid waste ion exchanger is transferred into this tank for temporary storage prior to dewatering operations by the portable dewatering system from the shipping container.

A resin dewatering pump was originally installed discharging into the holdup tanks; however, the equipment is no longer used.

Portable Shielding

Portable shielding may be used to reduce the radiation exposure to operating and maintenance personnel to as low as is reasonably achievable in accordance with 10 CFR 20 and Regulatory Guide 8.8. The shields are placed in position with the lifting equipment.

Filter Transfer Cask

A top loading filter transfer cask is provided to safely transport spent filter elements from the bottom loaded filter housing to the filter drumming area. The filter transfer cask is mounted on an electric powered transfer vehicle and is equipped with a movable shield to reduce the radiation exposure to maintenance personnel during filter removal.

Containers for Waste

The containers used for shipment includes dry waste containers and disposable liners. If necessary these containers can be placed in shielded transportation casks for interim storage within the low level waste storage facility and/or for offsite shipment. The quantity of radioactivity shipped will determine if shielding is required and the strength of the shielding cask or overpack (i.e., low specific activity, Type A, or Type B). The containers used for shipment are in compliance with 49 CFR 170-179, 10 CFR 20, and 10 CFR 71 and other state regulations.

11.4.2.2 Dry Waste Processing

Dry wastes, which become contaminated with radioactivity, are collected throughout the plant. The solid disposable wastes are collected, stored in containers and prepared for interim storage within the low level waste storage facility and/or for shipment offsite. The containers are then stored in the onsite low level waste storage facility. After sufficient containers have accumulated for a shipment, the containers are shipped offsite to a disposal facility. Spent filter cartridges are transferred to the filter transfer cask. A monorail is used to lift the filter transfer cask to elevation 19.5 ft. Each cartridge is transferred into a liner or container.

Handling and packaging of large waste material (e.g., core components, HEPA filters and activated charcoal or equipment which become activated during reactor operation) will be handled using qualified personnel with appropriate radiation protection measures. Since each such item handled would have unique problems, the personnel, the procedures and the packaging are determined for each case separately.

11.4.2.3 Design Features to Control Releases, Reduce Maintenance Improve Operations

The SWMS provides a reliable means for processing the wastes by including equipment, interlocks, alarms and fail-safe mechanisms so that exposure of individuals to radiation in restricted areas is below the limit specified in 10 CFR 20.101.

Pressure instrumentation and level instrumentation are used where necessary in pump suction lines, pump discharge lines, and tanks to detect abnormal operation. If abnormal operation is detected, the system controls provide alarms and stop operation in a fail-safe manner.

Temporary curbings are provided around the portable dewatering system area to prevent spread of liquid from the cubicle in the case of overflow. The floors in the cubicles are pitched to floor drains located at low points to facilitate floor drainage. The floor drains inside the Reactor Auxiliary Building drain to a floor drain sump for collection and processing by the Liquid Waste Management System. Leakage from the spent resin dewatering equipment is collected by the curbing provided during dewatering operations. Portable pumps are provided to transfer the waste to the Liquid Waste Management System for processing.

The drumming storage area inside the Reactor Auxiliary Building has floor drains to drain waste water to the equipment drain tank for processing by the Liquid Waste Management System.

Equipment is arranged and shielded to permit operation, inspection, and maintenance with as low as is reasonably achievable personnel exposure. Tanks and processing equipment which contain large quantities of radwaste are shielded and air flows are from low activity areas to higher activity areas. The drumming storage area is vented to the plant stack.

Pressure retaining components of process systems utilize welded construction to the maximum practicable extent. Flanged joints or suitable rapid disconnect fittings are used only where maintenance or operational requirements clearly indicate that such construction is preferable.

Screwed connections in which threads provide the only seal will not be used except for instrumentation connections where welded connections are not suitable. Process lines will not be less than 3/4 inch (nominal I.D). Screwed connections backed up by seal welding, socket welding or mechanical joints may be used on lines 3/4 inch or greater, but less than 2 1/2 inch, nominal size. For lines 2 1/2 inch and above, pipe welds are of the buttjoint type.

Completed process systems are initially pressure tested to the maximum practicable extent. Piping systems are hydrostatically tested in their entirety except at atmospheric tank connections where no isolation valves exist. Testing of piping systems are performed in accordance with applicable ASME or ANSI codes. Testing provisions are incorporated to enable periodic evaluation of the operability and required functional performance of active components of the system.

Materials for the pressure components conform to materials specifications of Section II of the ASME Code or ASTM materials.

11.4.2.4 Quality Assurance Program

Since the impact of these systems on safety is limited, a quality assurance program corresponding to the full extent of Appendix B to 10 CFR Part 50 is not required. However, to ensure that system will perform their intended function, a quality assurance program is established that is sufficient to ensure that all, design, construction, and testing provisions are met. Quality assurance requirements for the permanently installed equipment and piping are in accordance with Regulatory Position 6 of Regulatory Guide 1.143, Rev. 1. The procurement of a solidification system will require that the system comply with the quality assurance requirements of Regulatory Guide 1.143, Rev. 1.

TABLE 11.4-1

ESTIMATED INPUTS TO SOLID WASTE MANAGEMENT SYSTEM

Source	Form	Table Reference for Radionuclide Distribution	Quantity (ft^3/yr) Note 1
Spent Resins			
CVCS	Dewatered	Table 12.2-20	208
Fuel Pool	Dewatered	Table 12.2-34	96
Liquid Waste Management System	Dewatered	Table 12.2-12	144
Concentrator Bottoms (Note 2)			
Liquid Waste		Tables 12.2-12, 26	2780
Filters			
Cartridges	Cartridges	Table 12.2-10 12.2-24 12.2-34 (Note 3)	160
Compressible Waste	Plastic, Bags Paper, Etc.	Negligible	36000
Non-Compressible Wastes	Tools, Etc.	Negligible	3000

Notes:

1) Quantities are based on actual radioactive waste outputs of St. Lucie 1 adjusted for liquid waste concentrator operations.

2) Equipment is no longer used.

3) Additional filter cartridges may result from the use of a portable filter/vacuum in the Spent Fuel Pool; however, this does not impact the nuclide inventories.

TABLE 11.4-2

This table has been deleted.

TABLE 11.4-3

ESTIMATED QUANTITIES OF OUTPUT FROM
SOLID WASTE MANAGEMENT SYSTEM

Source	Form	Quantity(ft^3/yr)
Spent Resins		
CVCS	Dewatered	312(1)
Fuel Fool	Dewatered	144(1)
Liquid Waste Management System	Dewatered	216(1)
Concentrator Bottoms		
Liquid Waste	12% $Na_2 B_4 O_7$	4170(1)
Filters		
Cartridges	Cartridges	160
Compressible Wastes	Plastic, Bags Paper, Etc.	7200(2)
Non-Compressible Wastes	Tools etc.	3000

Notes:

1) Based on two volumes of waste per volume solidification agent.

2) Based on a compaction factor of five.

3) Equipment is no longer used.

TABLE 11.4-4

ESTIMATED SPENT RESIN ACTIVITY CURIES/FT$^{3(2)}$

Nuclide	Dewatered	Solidified w/cement$^{(1)(3)}$
I 129	0	0
I 131	3.4 E-00	2.3 E-00
RB 86	1.7 E-03	1.1 E-03
CS 134	5.1 E-00	3.4 E-00
CS 136	1.8 E-01	1.2 E-01
CS 137	3.9	2.6
SR 89	2.7 E-02	1.8 E-02
SR 90	3.2 E-03	2.1 E-03
ZR 95	5.8 E-03	3.9 E-03
NB95	2.7 E-03	1.8 E-03
RU103	2.8 E-03	1.8 E-03
RU106	2.5 E-03	1.6 E-03
TE125M	2.6 E-03	1.7 E-03
TE127M	3.9 E-02	2.6 E-02
TE129M	7.4 E-02	4.9 E-02
BA140	4.3 E-03	2.9 E-03
CE141	3.6 E-03	2.4 E-03
CE144	7.4 E-03	4.9 E-03
PR143	1.1 E-03	7.0 E-04
CR51	8.2 E-03	5.5 E-03
MN54	7.4 E-03	5.0 E-03
FE55	4.7 E-02	3.1 E-02
FE59	7.0 E-03	4.7 E-03
CO58	1.7 E-01	1.1 E-01
CO60	6.2 E-02	4.2 E-02

Bases:
(1) 0.667 ft^3 spent esins when solidified with cement will have a volume of 1.0 ft^3
(2) 256 ft^3 resin/resin tank; input activities from Table 12.2-14
(3) Spent resins are no longer solidified.

TABLE 11.4-5

ESTIMATED SPENT FILTERS ACTIVITY - SHIPPED CURIES/BATCH[1]

Nuclide	CVCS[2] Letdown	CVCS Pre-con-[2] centrator	LWMS[3] Waste	Fuel Pool[4] Laundry	Purification
Co-60	4.0 E+01	5.8 E-01	1.2 E-01	7.1 E-01	9.4 E-02
Fe-59	1.2 E+01	5.6 E-02	1.4 E-02	8.2 E-03	4.0 E-02
Co-58	2.3 E+02	1.4 E+00	3.3 E-01	1.9 E-01	6.6 E-01
Mn-54	5.7 E+00	6.7 E-02	1.5 E-02	8.4 E-03	1.4 E-02
Cr-51	1.7 E+01	5.9 E-02	1.6 E-02	9.5 E-03	6.9 E-02
Fe-55	3.1 E+01	4.3 E-01	9.1 E-02	5.3 E-02	7.3 E-02

Bases:

(1) 1 filter per 55 gallon container, encapsulated with solidification agent
(2) input activities from Table 12.2-24
(3) input activities from Table 12.2-10
(4) input activities from Table 12.2-34

TABLE 11.4-6

SOLIDIFIED WASTES CONCENTRATES
µCi/CC solidified waste(1)

This table contains historical information

Nuclide	Waste Concentrates(2)	Waste Nuclide	Concentrates
Br-83	1.3 E-04	RH-103M	6.4 E-07
84	2.2 E-05	106	1.4 E-09
85	2.4 E-07	TE-125M	2.9 E-06
I-130	9.9 E-05	127M	2.6 E-05
131	2.4 E-02	127	3.7 E-05
132	2.6 E-03	129M	1.3 E-04
133	2.1 E-02	129	2.6 E-05
134	6.4 E-04	131M	1.6 E-04
135	7.5 E-03	132	2.1 E-03
RB-86	8.1 E-06	BA-137M	1.1 E-05
88	9.5 E-04	140	2.0 E-05
CS-134	2.6 E-03	LA-140	1.1 E-05
136	1.3 E-03	LE-141	6.6 E-06
137	1.9 E-03	143	2.6 E-06
SR-89	3.3 E-05	144	3.1 E-06
90	9.9 E-07	PR-143	4.6 E-05
91	2.9 E-05	144	1.6 E-07
Y-90	9.0 E-08	NP-239	8.8 E-05
91	6.2 E-06		
91M	4.8 E-06		
93	1.5 E-06		
ZR-95	5.7 E-06	CR-51	1.8 E-05
NB-95	4.6 E-06	MN-54	3.1 E-06
M0-99	6.6 E-03	FE-55	1.5 E-05
TC-99M	1.8 E-03	59	9.7 E-06
RU-103	4.4 E-06	CO-58	1.5 E-04
106	9.9 E-07	60	2.0 E-05

Bases:

(1) 0.667 cc waste when solidified has a volume of 1 cc.

(2) Input activity from Table 12.2-12; volume of waste concentrator = 800 gallons.

TABLE 11.4-7

SOLIDIFIED BORIC ACID CONCENTRATES
(μcuries/cc solidified waste[1])

This table contains historical information

Nuclide	Active[2]	Nuclide	Activity[2]	Nuclide	Activity[2]
KR-83M	8.8 E-08	CS-134	1.1 E-03	BA-137M	6.7 E-11
85M	2.6 E-06	136	3.0 E-04	140	1.5 E-05
85	3.2 E-03	137	P.1 E-04	LA-140	8.1 E-07
87	1.2 E-07	SR-89	4.0 E-05	CE-141	7.1 E-06
88	1.9 E-06	90	7.3 E-07	143	1.4 E-07
89	1.8 E-11	91	1.7 E-07	144	4.3 E-06
XE-131M	1.5 E-03	Y-90	1.4 E-08	PR-143	3.5 E-06
133M	5.5 E-04	91	7.3 E-06	144	6.3 E-12
133	1.3 E-01	91M	5.9 F-10	NP-239	1.1 E-05
135M	1.1 E-09	93	1.0 E-08	CR-51	9.5 E-06
135	3.4 E-05	ZR-95	7.0 E-06	MN-54	2.3 E-06
137	4.6 E-11	NB-95	5.1 E-06	FE-55	1.2 E-05
138	3.0 E-09	MO-99	1.1 E-03	59	6.1 E-06
BR-83	6.8 E-08	TC-90M	4.6 E-06	CO-58	1.0 E-04
84	1.7 E-09	RU-103	4.8 E-06	60	1.5 E-05
85	1.7 E-12	106	1.3 E-06		
I-130	8.8 E-07	103M	9.5 E-11		
131	1.2 E-02	106	1.5 E-15		
132	1.2 E-06	TE-125M	3.4 E-06		
133	5.0 E-04	127M	3.5 E-05		
134	8.1 E-08	127	2.1 E-07		
135	2.3 E-05	129M	1.4 E-04		
RB-86	2.3 E-06	129	4.9 E-09		
88	1.9 E-08	131M	7.0 E-06		
		131	4.3 E-10		
		132	4.3 E-04		

Bases:

(1) 0.6667 cc waste when solidified has a volume of 1cc.
(2) Input activity from Table 12.2-22; volume of Boric Acid Holdup Tank = 2400 gallons.

TABLE 11.4-8

This table has been deleted.

TABLE 11.4-9

SOLID WASTE MANAGEMENT PROCESS CAPACITY
AND CAPACITY UTILIZATION

(Information provided in this table is historical)

		Solidification System Capability		
Type of Waste	Output of Solidified Waste Ft^3/Yr	Process Flow Rate GPM (Note 2)	System Output Ft^3/Yr	Fraction of Process Capacity Used
St. Lucie No. 2 Design Basis (Note 1)				
Spent Resins	672			
Concentrator Bottoms	4,170			
Filters	160			
Total	5,002	13.3	128,031 (Note 3)	4%
Compressible Solids	-7,200	(Note 4)	23,531	31%
Values Given by NRC				
Solidified Concentrator Bottoms & Spent Resins	13,000	13.3	128,031	10%
Dry Waste	10,000	(Note 4)	23,531	42.5%

Notes:

1) Per Table 11.4-3 of FSAR

2) 2/3 of System Design Flow Rate (Table 11.4-2 of FSAR)

3) 100 Workdays/yr @ 2 shifts/day, 0.5 operation coefficient, and 2:1 ratio of waste to solidification agent

4) Sixteen 55 gal drums/day, 200 workdays/yr.

11.5 PROCESS AND EFFLUENT RADIOLOGICAL MONITORING AND SAMPLING SYSTEMS

The continuous Process and Effluent Radiological Monitoring System is an integral part of a Radiation Monitoring System which also includes the area and airborne radiation monitoring instrumentation (refer to Subsection 12.3.4).

The Process and Effluent Radiological Monitoring and Sampling Systems monitor and furnish information to operators concerning activity levels in selected plant process systems and plant effluents.

The systems consist of permanently installed continuous off-line type monitoring devices together with provisions for specific routine sample collections and laboratory analyses. The overall systems are designed to assist the operator in providing information for evaluating and controlling the radiological consequences of normal plant operation, anticipated operational occurrences, and postulated accidents such that resultant radiation exposures and releases of radioactive materials in effluents to unrestricted areas are maintained as low as reasonably achievable.

These systems are supplemented by the Area and Airborne Radioactivity Monitoring Systems described in Subsection 12.3.4.

11.5.1 DESIGN BASES

11.5.1.1 Process Radiological Monitoring System

The continuous Process Radiological Monitoring System, supplemented by the Sampling System, is designed to perform the following functions:

a. Provide assistance to operators to insure the proper functional performance of the selected systems being monitored.

b. Provide for early detection of radioactivity leakage into normally nonradioactive systems, including primary-to-secondary leakage; and process system leakage into normally nonradioactive systems, such as component cooling water loops.

c. Provide information to plant personnel of radiation levels in liquid or gaseous process lines to permit proper control.

d. Provide information to plant personnel of any abnormal increase in normally radioactive or potentially radioactive process lines.

11.5.1.2 Effluent Radiological Monitoring System

The Effluent Radiological Monitoring System is designed to perform the following functions in order to meet the requirements of 10 CFR 20, 10 CFR 50 General Design Criteria 60 and 64, and follow the recommendations of Regulatory Guide 1.21, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants," June 1974 (R1), and Regulatory Guide 4.15, "Quality Assurance for Radiological Monitoring Programs (Normal Operations) – Effluent Streams and the Environment," for major and potentially significant paths

for release of radioactive material during normal operations, including anticipated operational occurrences:

a. Provide continuous representative sampling, monitoring, storage of information, indication and if necessary, alarm of liquid and gaseous radioactivity levels.

b. Provide the capability, during the batch release of radioactive liquid and gaseous wastes, to alarm and initiate automatic closure of the appropriate waste discharge valves before Technical Specifications limits are approached or exceeded.

c. Provide radiation level indication and alarm annunciation to the control room operators whenever Technical Specifications limits for release of radioactivity are approached or exceeded.

Additional details of the Radioactive Effluent Monitoring Program may be found in the St. Lucie Plant Offsite Dose Calculation Manual (ODCM).

11.5.1.3 Sampling System

The Sampling System provides grab samples to supplement the continuous Process and Effluent Radiological Monitoring System, and in particular is designed to provide specific information regarding specific radionuclide composition of process and effluent streams and to monitor tritium as required in Regulatory Guide 1.21 (R1). Results of routine laboratory analysis of process samples are used to monitor the operational performance of unit equipment and to provide additional information for making operating decisions.

The basis for selecting sampling locations for liquid and gaseous streams is to permit laboratory analysis for confirmation of readings from the stream monitors, to provide more precise information than may be obtained from the continuous monitors, and to verify effectiveness of processes.

The post accident effluent release points are provided with filter assemblies to collect samples of suspended radioactive particulates and gaseous iodine. The sampling system design is such that plant personnel could remove samples, replace sampling media, and transport the sample to an onsite analysis facility with radiation exposures less than those of GDC 19.

11.5.2 SYSTEM DESCRIPTION

11.5.2.1 Process and Effluent Radiological Monitoring

The requirements of the system design bases for continuous monitoring are satisfied by a system of off-line-type monitoring channels for the inplant liquid and gaseous process lines. The system includes single-stage gaseous monitors, single-stage liquid monitors, three-stage particulate, iodine and noble gas monitors, high range multistage gaseous monitors and externally mounted monitors.

Continuous monitoring means that the monitor operates uninterrupted for extended periods during normal plant operation. The monitor may occasionally be out of service for maintenance, repair, calibration, etc, during which time the frequency of sampling of the particular stream may be increased, depending on the past history of the radioactivity level of the stream.

Off-line radiation detection instrumentation is associated with liquid and gaseous process and effluent streams, in order to monitor radionuclide concentrations in such streams. Radiation measurements can be obtained through measurements of gross beta/gamma activity and/or a select gamma energy basis dependent upon the isotopic composition of a particular stream. The monitoring system operates in conjunction with regular and special radiation surveys and with radiochemical information for continued operation. Continual indication and recordings of radiation levels for normal operation, for anticipated operational occurrences and for a range of accident conditions is maintained for each channel associated with the monitoring system. Readout recording, alarm annunciation and alarm set point adjustment are centralized within an area of the control room complex.

The radiation monitors required to operate during and after an accident are qualified to operate in the accident environment that they experience. Procedures are used to convert the instrument readings to release rates per unit time.

11.5.2.1.1 Radiation Monitoring System

The Process and Effluent Radiation Monitoring System is a digital computerbased system and consists of various monitor channels located throughout the plant. Each channel is equipped with a detector and its associated electronics, a local control and display unit, a power supply, and a microprocessor per monitor which may consist of more than one channel as in the case of airborne type monitors.

All channel information is processed through a dedicated local microprocessor per monitor and then transmitted to the computer system for the purpose of data logging, processing, editing and displaying of information obtained from the radiation sensors. Those channels identified as safety related are further indicated and recorded on digital ratemeters and recorders located in the control room. A schematic of the system is shown on Figure 11.5-1. A dual computer and loop configuration allows any component to fail without affecting the remainder of the system.

11.5.2.1.2 Continuous Sampler Assembly

All continuous process and effluent radiation monitors are located in an off-line sampler assembly.

Each sampling assembly consists of a sampler and the associated piping, fittings, and other components as required to transport the sample through the system. All samplers include radiation detection equipment and a check source. The monitor cabinet is a skid mounted system and includes such items as the microprocessor, a sampling pump, valves, interconnecting piping, fittings, flow and pressure indicators.

The sample chamber is sized and shielded in a 4 π geometry as required to achieve the specified minimum system sensitivities. Each sampler is constructed of stainless steel and is located as close as practical to the process stream, such that sample line interference or losses are insignificant.

Samplers are designed so that they have flush capability for decontamination purposes where practicable.

Each continuous gaseous and liquid monitor is provided with a solenoid-operated check source that simulates a radioactive sample in the detector sample chamber and may be used for

operational and gross calibration checks of the detector and readout equipment. All check sources have half-lives greater than seven years.

11.5.2.1.3 Radiation Detectors

The detector assembly is a completely weatherproof assembly, housing a detector, photomultipliers, and radiation check source. The assembly is capable of withstanding the design pressure and temperature of the piping system of which it is a part.

The detector assembly is incorporated in the sampler assembly. All detector assemblies are designed to detect over the range addressed in Table 11.5-1 in a 2.5 mr/hr (1 MeV gamma) external field.

A shielded photomultiplier is provided integral with the detector to ensure reliable transmission of a high signal-to-noise ratio.

Scintillation detectors are beta- or gamma-sensitive detectors suitable for analysis of photopeaks up to 2.5 MeV and beta energy up to 5.0 MeV.

The detector is one of the following types as noted in Table 11.5-1 and below:

a. Single-Stage Liquid Monitor (SSLM)

   A single-stage liquid monitor consists of a gamma sensitive scintillation detector, coupled to a photomultiplier tube which is protected by an electromagnetic shield. The minimum detection limit of the monitor for Cs-137 in a 1 mr/hr background of Co-60 gamma radiation at a 95 percent confidence level is 1 x 10^{-6} µCi/cm for one minute counting time. The resolution of the detector is less than 10 percent Full Width at Half Maximum (FWHM) at 0.662 MeV (Cs-137).

   Each SSLM is provided with an off-line liquid sampling system as shown in Figure 11.5-2.

b. Single-Stage Gaseous Monitor (SSGM)

   A single-stage gaseous monitor consists of a beta sensitive plastic scintillation radiation detector, coupled to a photomultiplier tube which is protected by an electromagnetic shield. Figure 11.5-3 is a block diagram showing the components of the monitor. Those monitors designated as safety related have redundant pumps.

   The minimum detectable limit of the monitor for Xe-133 in a 1 mr/hr background at a 95 percent confidence level is 1 x 10^{-6} µCi/cm^3, based on a sample flow rate of 1 scfm and a one-half minute counting time. The response of the detector is at least 3x $\sqrt{background}$ above background.

c. Three-Stage Particulate Iodine and Noble Gas (P-I-G) Monitor

   Three-stage particulate, iodine and noble gas monitor consists of three detectors, one each for noble gases, airborne particulates, and iodine (refer to Figure 11.5-4). The gaseous monitor is similar to the SSGM described in b) above.

The particulate monitor is a beta sensitive plastic scintillation radiation detector, coupled to a photomultiplier tube which is protected by an electromagnetic shield. The minimum detectable limit of the monitor for Sr-90 in a 1 mr/hr background at 95 percent confidence level is 10^{-9} µCi/cm^3, based on a sample flow rate of 2 scfm and a one-half minute counting time. The response of the detector is at least 3x $\sqrt{background}$ above background. The movable filter increases the time to achieve the same count rate.

Filters for the particulate monitor are at least 99 percent efficient for particles 0.3 microns and larger. All of the monitors utilize moving filters. The filter paper advances in either a continuous or a step mode, or can be used as a fixed filter that is advanced manually.

The iodine monitor is a gamma-sensitive NaI(T1) crystal, coupled to a photomultiplier tube which is protected by an electromagnetic shield. The minimum detectable limit of the monitor for I-131 in a 1 mr/hr background at a 95 percent confidence level is 10^{-9} µCi/cm^3, based on a sample flow rate of 2 scfm and a 3 minute counting time. The response of the detector is at least 3x $\sqrt{background}$ above background. The resolution of the detector does not exceed 10 percent FWHM at 0.662 MeV (Cs-137).

A fixed filter cartridge assembly is used for the iodine channels. It is easily accessible for replacement. It is at least 85 percent efficient for the collection of iodine.

These monitors incorporate redundant sample flow pumping systems in order to increase the monitors availability.

d. Multi-Stage Gaseous Monitor (MSGM)

The multi-stage gaseous monitor consists of three detectors with overlapping ranges. The low range detector is similar to the SSGM described in b) above. The mid range detector uses a solid state detector and has a range overlapping by more than one decade of the low range detector. The high range detector also uses a solid state detector and its range overlaps the mid range detector and extends to 10^5 µCi/cm^3. (See Figure 11.5-5)

e. Externally Mounted Monitor (EXT)

The Atmospheric steam dump monitor consists of gamma detecting GM tubes viewing the main steam lines and a background subtraction detector (see Figure 11.5-6). The minimum detectable limit for noble gas activity in the main steam in a 5mR/hr background at a 95 percent confidence level is 6 x 10^{-3} µCi/cm^3, based on a one minute counting time.

A conversion factor has been developed to correct for the low energy gammas the external monitors do not detect.

11.5.2.1.4 Controls and Alarms

All monitors are provided with either a local control and display unit located near the monitor or a portable indicator control box capable of accessing the monitor control features and data base. Either of the two units provide information relating to operational mode, alarm status and data output. Purging, check source actuation, valve and pump control, and various test mode actuations may be done locally and, with the exception of valve control, within the cabinets at the various operator's terminals.

The digital information from all channels is stored by the redundant computers and displayed at the operator consoles on computer terminal displays. If an alarm condition is detected, a status change occurs at each of the computer terminals and logging of the alarm occurs automatically. Monitor status, radiation level, and alarm status are displayed. Alarms include two up-scale trips to indicate high radiation levels and one downscale trip to indicate instrument trouble. For those channels designated as safety related, data displays and recorders are also present in a safety related panel in the control room.

For those channels which perform control action, any one of the following automatically sends an isolation signal to the valve located on the monitored line to prevent further flow: radionuclide concentrations above the preset "high" radiation trip point, failure of the detector or sample pump, or loss of flow to the sampling chamber.

Alarm set points are variable over the entire dynamic range and are set from the control room. Alarm setpoints may be introduced or changed from the following locations, a) for safety related monitors: from the individual channel control and display units located in the control room safety cabinets, and b) for non-safety related monitors: from any of the computer terminals, locally by means of the local control and display unit, or by the portable indicator control unit. All alarm set points are protected and changed only by means of proper access identification. Exact setpoint depends on background and plant conditions. For effluent monitors, high-high alarms indicate before 10 CFR 20 limits are reached.

11.5.2.1.5 Power Supplies

Each monitoring channel is provided with an independent power supply, designed such that a failure in that channel does not affect any other channel. Monitors that are identified as safety-related monitors are redundant and are supplied power from safety-related ac buses. The power supplies for these channels are identified in Table 11.5-1. Power to the non-safety related channels, that monitor only normal operations, is supplied from non-safety related ac buses as indicated on Table 11.5-1.

11.5.2.2 Monitors Locations and Specifications

Table 11.5-1 gives a tabulation of basic information describing each of the continuous process and effluent radiological monitors. The basis for the ranges of concentration are as follows:

a. Process Monitors

    1. Maximum expected concentrations during normal operations and anticipated operational occurrences, as well as range of expected concentrations.

2. The highest sensitivity commercially available when purchased in order to detect process system leakage contamination as early as possible.

b. Effluent Monitors

1. Range of expected concentrations during normal operations and anticipated operational occurrences.

2. Sensitivity commensurate with detecting gross amounts of activities which when diluted by atmospheric dispersion or discharge canal water, would be below the limits specified in 10 CFR 20.

Actual values of the alarm limits depend on various plant conditions; therefore, the values listed in Table 11.5-1 should be interpreted as theoretical preliminary values.

The following process monitors are provided:

1. Component Cooling Water
2. Chemical and Volume Control System (CVCS)
3. Boric Acid and Waste Evaporator Holding Tank Condensate Return

The following effluent monitors are provided:

1. Steam Generator Blowdown
2. Liquid Waste Discharge
3. Gaseous Waste Discharge
4. Condenser Air Ejector
5. Plant Vent
6. Fuel Handling Building Stack
7. ECS Ventilation System exhaust
8. Plant Stack Accident Monitor
9. Atmospheric Steam Dump Exhaust Monitor

11.5.2.2.1 Component Cooling Water Monitor

Two radiation monitors are provided in the Component Cooling Water System. Each is a single-stage liquid monitor as described in Subsection 11.5.2.1.3a. The purpose of these monitors is to detect leakage of radioactive water into the normally non-radioactive component cooling water loop.

The monitors sample the component water supply downstream of each of the Component Cooling Water System heat exchangers and return the sample to the component cooling water

pumps' suctions, thus enabling the monitors to utilize the pressure drop and obtain samples sufficiently cooled for the detectors to prevent detector damage.

The monitors provide a high alarm when concentration levels reach preset-limits, which constitutes positive indication of contaminated leakage to the system. The receipt of these alarms alert the operator to the presence of leakage so that additional radiation surveys, sampling, and equipment isolation can be effected in order to locate and repair the leakage source.

The atmospheric vent valve of the component cooling water surge tank is automatically closed by means of a high radiation signal from these same detectors. The system then operates unvented with relief to the Waste Management System.

These monitors are seismically qualified and relay their information directly to the control room.

11.5.2.2.2 CVCS Process Monitor

Note: This monitor has been isolated and no longer used per safety evaluation PSL-ENG-SENS-98-094.

The CVCS process monitor is a single-stage liquid monitor as described in Subsection 11.5.2.1.3a.

The primary purpose of the CVCS process radiation monitor is to alert plant operators to an increase in reactor radioactivity as quickly as possible. Such an increase in radioactivity would usually be caused by crud released in the Reactor Coolant System or CVCS letdown line. However, an increase in specific fission product nuclide activity along with an increase in gross gamma activity would be indicative of failed fuel cladding.

The CVCS monitor is a continuous monitor located in the CVCS letdown line upstream of the purification filter.

Gross gamma activity concentration and the activity concentration of a specific nuclide are monitored simultaneously.

As noted in Table 11.5-1, the expected activities monitored are the same as that for reactor coolant. Since the CVCS monitor is a trend monitor, alarm setpoints vary, and normally are set slightly higher than the current steady-state coolant activity.

11.5.2.2.3 Boric Acid and Waste Evaporator Condensate Monitor

A liquid process radiation monitor samples the condensate recovery tank drain line receiving condensate flow from the boric acid and waste evaporators. This assures that a failure of the heating coils whereby contaminated condensate could be introduced into the uncontrolled secondary makeup water system via the recovery tank drainline can be isolated automatically. The monitor is a single-stage liquid monitor as described in Subsection 11.5.2.1.3a.

The process monitor draws a sample upstream of the cation conductivity cell from the recovery tank drainline. Alarm signals from either the process monitor or the conductivity cell automatically close system outlet valves in order to ensure further processing of any contaminated concentrate collected.

11.5.2.2.4 Steam Generator Blowdown Monitor

Two steam generator blowdown monitors are provided to continuously monitor steam generator primary-to-secondary leakage. One monitor is located in each blowdown sample line downstream of the blowdown Line isolation valve. The monitors are single-stage liquid type described in Subsection 11.5.2.1.3a. Each monitor provides a signal to close the blowdown valves on high radiation. The alarm set point is determined by the applicable dilution and discharge limit. Discharge from steam generator blowdown is via the Steam Generator Blowdown Treatment Facility (common to both units). Releases from this system are performed by administrative controls.

11.5.2.2.5 Liquid Waste Discharge Monitor

The liquid waste discharge monitor is a single-stage liquid monitor as described in Subsection 11.5.2.1.3a. The monitor is located in the Liquid Waste Management System discharge line to the circulating water canal.

The primary purpose of the liquid waste discharge monitor is to continuously monitor and record the radioactivity that is being discharged in the liquid waste being released to the circulating water canal. The monitor also terminates the liquid discharge from the plant if the radioactivity being released exceeds the monitor setpoint which is set below the applicable activity release limits.

11.5.2.2.6 Gaseous Waste Discharge Monitor

This gaseous waste discharge monitor is a single-stage gaseous monitor as described in Subsection 11.5.2.1.3b. This monitor is located in the Gaseous Waste Management System discharge line downstream of the gas surge tank and gas decay tanks but upstream of the vent pipe. Therefore monitoring is accomplished before the gases are diluted in the vent pipe.

The primary purpose of the gaseous waste discharge monitor is to continuously monitor and record all gaseous radioactivity released from the gas surge tank or gas decay tanks and to prevent radioactivity in excess of applicable limits from being released to the environment. The alarm set points are adjusted based on sample analyses and gaseous activity discharge limits. If the activity exceeds the setpoint, the discharge valve is automatically closed.

11.5.2.2.7 Condenser Air Ejector Monitor

The condenser air ejector monitor is a single-stage gaseous monitor, as described in Subsection 11.5.2.1.3b. The monitor measures noncondensable fission product gases in the condenser air ejector discharge to detect any primary-to-secondary leakage. The presence of radioactivity in this line indicates a primary-to-secondary leak in the steam generators. The predominant isotopes would be Kr-85 and Xe-133, with the presence of iodine. The function of this monitor is to alarm in the event of a primary-to-secondary steam generator tube leak.

The monitor is located on the discharge of the Steam Jet Air Ejector. The alarm setpoint would be set slightly higher than expected plant background.

11.5.2.2.8 Plant Vent Monitor

The two seismically mounted Class 1E plant vent monitors are three stage particulate, iodine and noble gas monitors, as described in Subsection 11.5.2.1.3c. The primary purpose of the plant vent monitors is to continuously monitor and record the radioactivity level of plant effluent gases being discharged from the plant vent in order to assure that the plant releases do not exceed Technical Specification limits. The alarm setpoint is based on applicable discharge limits. These monitors are seismically qualified and radioactivity levels are indicated and recorded in the Control Room. These two monitors, RS-26-13 and RS-26-14, are shown on Figure 9.4-1.

Isokinetic sample nozzles were originally provided to insure that a representative sample is withdrawn from the vent. However, Engineering Evaluation PSL-ENG-SENS-05-033 documents that isokinetic sampling is not necessary to achieve representative sampling in this application. Periodic testing and inspection of the Plant Vent HEPA filters is performed to ensure integrity and efficiency of the filters is maintained.

11.5.2.2.9 Fuel Handling Building (FHB) Stack Monitor

The FHB stack monitor is a three-stage particulate, iodine, and noble gas monitor as described in Subsection 11.5.2.1.3c. The primary purpose of this monitor is to continuously monitor and record the radioactivity level of effluent gases being released via the FHB stack. The alarm setpoint is set slightly higher than plant background conditions since this release point is not considered a normal release mode. An isokinetic sample nozzle is provided. This monitor is shown on Figure 9.4-11. Isokinetic sample nozzles were originally provided to ensure that a representative sample is withdrawn from the vent. However, engineering evaluation PSL-ENG-SENS- 05-033, Rev. 2, documents that isokinetic sampling is not necessary to achieve representative sampling in this application. EC278372

11.5.2.2.10 ECCS Area Ventilation System Exhaust Monitors

Two safety-related monitors are provided to measure the airborne effluent from the ECCS area. A sample is withdrawn from the ECCS area ventilation exhaust ducts to an off-line monitor. These monitors consist of the multistage gaseous monitors as described in Subsection 11.5.2.1.3d. The alarm setpoint is based on applicable discharge limits. These monitors are seismically qualified and radioactivity levels are indicated and recorded in the control room. These two monitors, RS-26-69 and RS-26-70, are shown on Figure 9.4-1.

11.5.2.2.11 Plant Vent Accident Range Radiation Monitor

The plant vent accident range radiation monitor is a multistage gaseous monitor as described in Subsection 11.5.2.1.3d. Upstream of the detectors are iodine and particulate prefilters. Sampling nozzles withdraw a sample from the stack for radiation analysis. The alarm setpoint is based on applicable discharge limits. These monitors are seismically qualified and radioactivity levels are indicated and recorded in the non-safety portion of the auxiliary panel. This monitor, RS-26-90 is shown on Figure 9.4-1.

11.5.2.2.12 Atmospheric Steam Dump Exhaust Monitor

The atmospheric steam dump monitor is described in Subsection 11.5.2.1.3e. The primary purpose of this monitor is to continuously monitor and record the radioactivity level in the main

steam that is discharged to the environment via the atmospheric steam dump valves. The alarm setpoint is set at the lowest range since this release point is not considered a normal release mode.

11.5.2.3 Calibration and Inspection

A remotely- or manually-operated check source is provided with each detector assembly. The check source isotope has a half-life greater than seven years, with emissions in the energy range and of the same type as being monitored, and is usable as a convenient operational and gross calibration check of the associated detection and readout equipment. The check source strength provides a count rate of approximately 1.5 decades above background. The check source controls are mounted on the channel indicator module in the control cabinets. These check sources can be activated automatically through the computer terminal in the control room or the hot chemistry laboratory.

Isotopic calibration of the complete radiation monitoring system are performed at the factory. Field calibration sources, with their decay curves, are provided with the system hardware. For the high range in containment monitor, a current source will be used for calibration of the radiation ranges above 10R/hr.

Further isotopic calibrations are not requires, since the geometry cannot be altered significantly within the sampler. Calibration of samplers is then performed, based on a known correlation between the detector responses and field calibration standards.

This single-point calibration confirms the detector sensitivity. The field calibration is performed by removing the detector and placing the calibration source on the sensitive area of the detector.

The radiation monitoring channels are checked and inspected in accordance with the Technical Specifications. Grab samples are periodically collected for isotopic analysis as described in the plant procedures. Setpoint adjustment and functional testing are done on a monthly or quarterly basis depending on the detector channel and calibration is performed at each refueling shutdown or indication of equipment malfunction.

11.5.2.4 Noncontinuous Sampling for Radioactivity

To augment the information provided by the continuous process and effluent monitors, samples are taken at specified intervals at selected locations in the process and effluent streams.

These samples are then taken to the radiochemistry laboratory for analysis. Although a number of the analyses are for other than radioactivity content, each sample can be analyzed for its isotope content or gross activity by use of instrumentation available in the counting room. This instrumentation consists of proportional counters, liquid scintillation detector, and Ge(Li) semiconductor detector and associated data analysis computer.

The sensitivity of the liquid scintillation spectrometer and Ge(Li) semiconductor-detector spectrometer are sufficient to enable detection of the isotopes in the samples within the limits specified by Regulatory Guide 1.21, (R1), Position B3.

There are three kinds of samples taken at the plant: samples from the Process Sample System (Subsection 9.3.2), local liquid grab samples, and gas analyzer grab samples. In addition grab

samples taken directly from all process and effluent radiation monitors and the particulate and iodine filters in the gaseous monitors may be removed for laboratory analysis. The location and other data for the specific sampling points are listed in Table 11.5-2 for primary samples, Table 11.5-3 for secondary samples, and Table 11.5-4 for local and gas analyzer samples.

Sample point locations are based on one or more of the following requirements:

a. to check the performance of process equipment,

b. to alert the operator to any abnormal condition such as leakage, and/or

c. to insure effluent releases are below applicable limits.

To insure representative samples all liquid points are taken from vertically run pipe or from the top of horizontal run pipe. The local sample lines are as short as possible to limit the amount of purge water required before a representative sample is obtained. Vent samples are taken from straight duct runs. Liquid tanks are recirculated prior to sampling.

11.5.2.5 Review of Requirements of PERMSS

A review of the monitoring and sampling provisions in the gaseous process and effluent radiological monitoring and sampling system with the systems described in the Standard Review Plan, Section 11.5, Table 1A is tabulated in Table 11.5-5.

A review of the monitoring and sampling provisions in the liquid process and effluent radiological monitoring and sampling system with the systems described in the Standard Review Plan, Section 11.5, Table 1B is tabulated in Table 11.5-6.

11.5.2.6 Continuous Sampling and Analysis of Plant Effluent

The post accident effluent release points are provided with filter assemblies for collection of suspended particulates and.gaseous iodine. The sampler assemblies are easily removable, self supporting in nature and surrounded by a radiation shield to protect the operator. The radiation shield design assumes that 10^2 μCi/cm of gaseous radioiodine and particulates is deposited on the sampling medium for a 30 minute sampling time with an average gamma energy of 0.5 MeV. This design basis sample will be used to calculate the occupational dose to personnel during sample handling and transport, and analysis of sample. Sampling will be provided at the ECCS exhaust and plant vent stack exhaust. Continuous and grab samples will be provided at these points.

The iodine adsorbing cartridge uses activated charcoal with at least 90 percent effective adsorption for all forms of gaseous iodine, as its active ingredient. The particulate filter is added upstream of the iodine cartridge in order to prevent the radioactive particulates from entering the iodine cartridge.

The sampling medium for particulates is at least 90 percent effective for retention of 0.3 micron diameter particles.

Design of the analytical facilities and preparation of analytical procedures will consider the design basis sample.

11.5.3 EFFLUENT MONITORING AND SAMPLING

For a discussion of implementation of General Design Criteria 64, Subsections 11.5.1 and 11.5.2 contain a detailed description of the means which are provided for monitoring effluent discharge paths for radioactivity that may be released for normal operations, including anticipated operational occurrences, and from postulated accidents. Additional details may be found in the St. Lucie Plant Offsite Dose Calculation Manual (ODCM).

11.5.4 PROCESS MONITORING AND SAMPLING

For a discussion of implementation of General Design Criteria 60, Subsections 11.5.1 and 11.5.2 contain a detailed description of the means which are provided for automatic closure of isolation valves in gaseous and liquid effluent paths.

For a discussion of General Design Criterion 63, Subsections 11.5.1 and 11.5.2 contain a detailed description of the means which are provided for monitoring of radiation levels in radioactive waste process systems.

TABLE 11.5-1

PROCESS AND EFFLUENT RADIATION MONITORS

Monitors	Number	Type(1)	Location(2)	Control Function	Power Supply	Range(6) µCi/cc	Minimum Sensitivity(3) (7)	Typical Alarms & Control Setpoint (8)	At Detector (8)
a) Component Cooling Water	2	SSL	I-1"-CC-227 I-1"-CC-227	Close Surge Tank vent	Safety AC bus		$1.3x10^{8}$ cpm/µCi/ccCs137	$1.4x10^{-4}$µCi/cc	2.5 mR/hr
b) Chemical and Volume Control Letdown*	1	SSL	1"-CH-432	None	Nonsafety AC bus		$1.3x10^{8}$ cpm/µCi/ccCs137	$1.4x10^{2}$µCi/cc	10 mR/hr
c) Steam Generator Blowdown	2	SSL	-	Close blowdown valves FCV-23-3, -5,-7, & -9	NonSafety AC bus		$1.3x10^{8}$ cpm/µCi/ccCs137	$5x10^{-5}$µCi/cc	1 mR/hr
d) Liquid Waste Discharge	1	SSL	1"-WM-E19	Close discharge valves FCV-6627X & Y	NonSafety AC bus		$1.3x10^{8}$ cpm/µCi/ccXe137	$4x10^{-6}$µCi/cc	2.5 mR/hr
e) Gaseous Waste Discharge	1	SSG	2"-WM-D40	Close discharge Valve V6565	NonSafety AC bus		$4.3x10^{7}$ cpm/µCi/ccXe133	500µCi/cc	2.5 mR/hr
f) Condenser Air Ejector	1	SSG	1"-AE-45	None	NonSafety AC bus		$4.3x10^{7}$ cpm/µCi/ccXe133	$3x10^{-7}$µCi/cc	1 mR/hr
g) Plant Vent	2	P-I-G	-	None	Safety AC bus		$8.6x10^{4}$ cpm/µCi/ccCs137 $1x10^{5}$ cpm/µCi/ccI131 $2.1x10^{7}$ cpm/µCi/ccXe133	$1x10^{-7}$µCi/cc $5x10^{-7}$µCi/cc $5x10^{-6}$µCi/cc	1 mR/hr
h) Fuel Handling Building Stack	1	P-I-G	-	None	NonSafety AC bus		$8.6x10^{4}$ cpm/µCi/ccCs137 $1x10^{5}$ cpm/µCi/ccI131 $2.1x10^{7}$ cpm/µCi/ccXe133	$1x10^{-7}$µCi/cc $1x10^{-6}$µCi/cc $1x10^{-5}$µCi/cc	1 mR/hr
i) ECCS Area Ventilation System Exhaust	2	MSG	HVE-9A HVE-9B	None	Safety AC bus		$4.32x10^{7}$ cpm/µCi $2.84x10^{4}$ cpm/µCi $1.19x10^{2}$ cpm/µCi	$1x10^{-6}$µCi/cc $1x10^{-3}$µCi/cc $1x10^{0}$µCi/cc	2.5 mR/hr normal <1000 mR/hr accident

* Note: This monitor is no longer used. See Section 11.5.2.2.2.

TABLE 11.5-1(Cont'd)

Monitors	Number	Type(1)	Location(2)	Control Function	Power Supply	Range(6) µCi/cc	Minimum Sensitivity(3) (7)	Typical Alarms & Control Setpoint (8)	At Detector (8)
j) Boric Acid and Waste Holdup Condensate Tank	1	SSL	1"-CR-9	None	Nonsafety AC bus		$1.3x10^{8}$ cpm/µCi/ccCs137	$1.4x10^{-4}$µCi/cc	2.5 mR/hr
k) Plant Vent (high Range Noble Gas Monitor)	1	MSG	-	None	Nonsafety AC bus (on zero load block)		$4.32x10^{7}$ cpm/µCi/cc $2.84x10^{4}$ cpm/µCi/cc $1.19x10^{2}$ cpm/µCi/cc	$5x10^{-6}$µCi/cc $1x10^{-3}$µCi/cc 1.0 µCi/cc	1 mR/hr normal 20 mR/hr accident
l) Atmospheric Steam Dump Exhaust	3(4)	G-M tube	Main Steam(5) Trestle	None	Nonsafety AC bus		10 cpm/mR/hr	$1x10^{-1}$µCi/cc	0.25 mR/hr normal <1000 mR/hr accident
m) Deleted									
n) Post-Accident Failed Fuel	1	SSL	RAB(6)	None	Non-safety		$1.3x10^{8}$ cpm/µCi/ccCs137	1.0µCi/cc	20mR/hr

Notes:

(1) SSL = Single Stage Liquid, SSG = Single Stage Gaseous, MSG = Multi-Stage Gaseous, P-I-G = Particulate, iodine and noble gas (refer to Subsection 11.5.2.1.3)
(2) All monitors (except for e and l) are off-line type. Location indicates sample line take-off.
(3) Sensitivity listed is for counting time and background states. In addition, all monitors meet the sensitivities indicated in Subsection 11.5.2.1.3.
(4) One detector is required for background subtraction.
(5) Detectors view the main steam line.
(6) Instrument ranges are selected in accordance with standard engineering practices.
(7) Minimum sensitivities based on purchase specification values. Radiation monitor performance is equal to or better than purchase specification value.
(8) The “Typical Alarms & Control Setpoint” and “At Detector” column information is kept for historical purposes. This information is maintained by Plant procedures.

TABLE 11.5-2

PRIMARY SYSTEM SAMPLE POINTS

Location	Basis for Selection	Radioactivity Analysis
Hot Leg Loop 2A	Sample reactor coolant	Gross activity, Tritium, isotope identification crud activity, gaseous activity.
Pressurizer Surge Line	-	As required
Pressurizer Steam Space	Sample gas and non-Condensibles	As required
High Pressure Safety Injection Pump Mini Flow Line 2A	LOCA coolant	Gross activity
High Pressure Safety Injection Pump Mini Flow Line 2B	LOCA coolant	Gross activity
LPSI Pump Discharge 2A	LOCA coolant	Gross activity
LPSI Pump Discharge 2B	LOCA coolant	Gross activity
Shutdown Cooling Suction Line	Reactor coolant during shutdown	Gross activity Crud Activity
Purification Filter 2A - Outlet	Verify filter Performance	As required
Purification Filter 2A - Inlet	Verify filter Performance	As required
Purification Filter 2B - Outlet	Verify filter Performance verify ion exchanger Performance	As required
Purification IX -	Verify filter and ion exchangers Performance	As required
Safety Injection Tanks 2A1,2A2,2B1,2B2	Verify quality water in tanks	As required

TABLE 11.5-3

SECONDARY SYSTEMS SAMPLE POINTS

Location	Basis for Selection	Radioactivity Analysis
Main Steam 2A	Steam purity	As required
Main Steam 2B	Steam purity	As required
Heaters 5A & 5B	Feedwater quality	As required
Condensate Pump Discharge	Condensate quality	As required
Condenser Hotwell 2A1	Condensate quality	As required
Condenser Hotwell 2A2	Condensate quality	As required
Condenser Hotwell 2B1	Condensate quality	As required
Condenser Hotwell 2B2	Condensate quality	As required
Steam Generator 2A Blowdown	SG chemistry, P/S Leakage	Gross activity
Steam Generator 2B Blowdown	SG chemistry, P/S Leakage	Gross activity

Note:
P/S = Primary to Secondary

TABLE 11.5-4

LOCAL GRAB SAMPLE POINTS AND GAS ANALYZER SAMPLE POINTS

Location	Basis for Selection	Expected Concentration	Sampling Frequency	Sample Analysis
1) Flash Tk - Influent and Effluent 1/2-WM-522,540	Verify performance of flash tank: increase in H2 content indicates flash tank malfunction	Table 12.2-26	As required	As required
2) Pre-Concetrator Filter - Influent and Effluent 1/2-WM-599,901,905,913	Ensure filter is removing suspended solids and determine decontamination factor	Table 12.2-24	As required	As required
3) Pre-Concentrator Ion Exchanger - Influent and Effluent 1/2-WM-905,913,932,934	Determine ion exchanger - decontamination factor and performance Effectiveness	Table 12.2-20	As required	Gross activity, Lithium
4) Boric Acid Concentrator(b) Bottoms and Distillate 1/2-WM-X41,-Y41,-703,-704	Determine concentrator performance and verify boric acid is at desired concentration required for transfer to CVCS	Table 12.2-26	Bottoms: As required by feed concentration Distillate: Continuous Conductivity analyzer	Boron -
5) Boric Acid Condensate Ion Exchanger(b) - Influent and Effluent 1/2-WM-943,966	Determine the effectiveness for boron removal and to verify performance capability: increase in boron content or activity indicates exchanger malfunction	Table 12.2-20	As required	Boron, Gross activity
6) Boric Acid Condensate Tank(b) Recirculation 1/2-WM-997,973	Determine if release can be made within permissible limits or to Recirculate for further processing	Table 12.2-22	As required	Routine water quality
7) Discharge Line to Circ. Water Discharge 1/2-WM-987,B36	Provide verification that process radiation monitor is functioning. Also provides redundant verification of isotopic analysis made in BA condensate tanks.	Table 12.2-8	As required	Gross activity, Isotopic analysis
8) Boric Acid Holding Tank(b) 1/2-WM-A14	Verify boric acid is at desired Concentration prior to transfer to BA makeup tank. If contaminated, boric acid can be pumped to disposal.	Table 12.2-22	Prior to transfer to BAMTS	Boron
9) Equipment Drain Tank 1/2-WM-A76	Determine if liquid can be released below 10 CFR 20 limits or be Processed by either waste concentrator or boric acid concentrator.	Table 12.2-8	As required for influent DF	Gross activity, Isotopic analysis

TABLE 11.5-4 (Cont'd)

	Location	Basis for Selection	Expected Concentration	Sampling Frequency	Sample Analysis
10)	Chemical Drain Tank 1/2-WM-A75	Determine if liquid can be released below 10 CFR 20 limits or be processed by waste concentrator	Table 12.2-8	As required to influent DF	Gross activity, Isotopic analysis
11)	Laundry Drain Tanks 1/2-WM-A97,B05	Determine if liquid can be released below 10 CFR 20 limits or be processed by waste concentrator	Table 12.2-8	As required by operation of the laundry	Gross activity, Isotopic analysis
12)	Waste Filter - Influent and Effluent 1/2-WM-A75,A76,A81	Ensuring filter is removing suspended solids and determine decontamination factor	Table 12.2-10	As required (normally changed out only on radiation level)	Gross activity
13)	Waste Condensate Tanks 1/2-WM-B26,B35	Determine if liquid can be released below 10 CFR 20 limits or be processed by waste concentrator or ion exchanger	Table 12.2-8	As required	Gross activity or Isotopic analysis
14)	Waste Ion Exchanger 1/2-WM-705,D16	Determine ion exchanger decontamination factor	Table 12.2-12	Monthly	Gross activity or Isotopic analysis
15)	Waste Concentrator (b) Distallate and Bottoms 1/2-W,4-Z41,-705	Verify concentrator performance	Table 12.2-12	As required (per batch) based on influent activity	Boron, Gross activity or Isotopic analysis
16)	Volume Control Tank 1/2-CH-591	Verify performance of volume control tank	Table 12.2-22	Weekly or as required by system operation	Isotopic analysis of gas space, H_2, O_2, N_2 concentration of gas space
17)	Makeup to Volume Control Tank 1/2-CH-564	Used to verify performance of volume control tank	-	Weekly	Routine water quality, Gross activity, Tritium
18)	Boric Acid Batching Tank 1/2-CH-586	Verify boron concentration	-	1 per batch as required	Boron
19)	Boric Acid Makeup Tanks 1/2-CH-582,575	Verify boron concentration	-	Weekly	Boron, Gross activity, Chloride, Flouride
20)	Shutdown Cooling Downstream SDC Heat Exchangers I-SI-471,170	Verify performance of shutdown cooling heat exchangers	Table 12.2-28	Daily when shutdown 5 out of 7 days when shutdown Weekly when shutdown	Boron Routine water quality, Gross activity or Isotopic analysis, Lithium Suspended solids, Crud activity, Tritium
21)	Safety Injection Tanks I-SI-403	Verify quality of water in safety Injection tank	-	As required by Technical Specifications when sampled	General water quality

TABLE 11.5-4 (Cont'd)

	Location	Basis for Selection	Expected Concentration	Sampling Frequency	Sample Analysis
22)	Fuel Pool Ion Exchanger 1/2-FS-543,531	Determine DF performance of the ion exchanger	Table 12.2-34	As required	Gross activity or Iso analysis
23)	Fuel Pool Purification Filter 1/2-FS-531,525	Determine performance of Filter efficiency	Table 12.2-34	Monthly for radiation levels; as required by water quality measurements of spent fuel pool	Radiation surveys
24)	Volume Control Tank - gas sample[(a)] Gas Analyzer	Analyze for potential explosive mixture and radioactivity concentration	95% $H_2$when operating 95% $N_2$when shutdown	Weekly	H_2, O_2, N_2 isotopic or Gross activity
25)	WMS Flash Tank - gas sample[(a)] Gas Analyzer	Analyze for potential explosive mixture and radioactivity concentration		As required	H_2, O_2 and isotopic
26)	Holdup Tanks - gas sample[(a)] Gas Analyzer	Analyze for potential explosive mixture	95% H_2	Weekly	H_2, O_2
27)	Spent Resin Tank - gas sample[(a)] Gas Analyzer	Analyze radioactivity concentration	4% O_2	As required by operation	H_2, O_2 and isotopic
28)	Containment Vent Header - gas sample[(a)] Gas Analyzer	Analyze radioactivity concentration	4% O_2	Weekly	H_2, O_2 and isotopic
29)	Gas Surge Tank - gas sample[(a)] Gas Analyzer	Analyze for potential explosive mixture and radioactivity concentration	4% O_2	Weekly	H_2, O_2 and isotopic
30)	Gas Decay Tank - gas sample [(a)] Gas Analyzer	Analyze for potential explosive mixture and radioactivity concentration	4% O_2	Weekly	H_2, O_2 and isotopic

(a) The monitoring of radioactivity is accomplished by obtaining a grab sample of the gas to be analyzed at the gas analyzer. Provisions are provided on the gas analyzer to manually mount and dismount a grab sample bottle so that it can be filled and then removed for laboratory analysis of its contents.
(b) Equipment is no longer used.

TABLE 11.5-5

RADIATION MONITORING SYSTEM PROVISIONS

(This table is maintained as historical information. Radiation monitoring is performed in accordance with the Offsite Dose Calculation Manual and associated procedures.)

No.	Process System	Monitor Provisions			Sample Provisions		
		In Process		In Effluent	In Process		In Effluent
		Continuous	ACF	Continuous	Grab	Grab	Continuous
1.	Waste Gas Holdup System		N			NT	
2.	Condenser Evacuation System			N	I	NT	
3.	Vent & Stack Release Pt. System			N		NIT	I
4.	Containment Purge Systems[a]	G	G[x]			NT	(I)
5.	Aux. Bldg. Ventilation System			(N)	I	(NI)	(I)
6.	Fuel Storage Area Vent. System[a]	G	G[x]	N	I	NT	I
7.	Radwaste Area Vent. Systems			(N)	(I)	(NT)	(I)
8.	Turb. Gland Seal Cond. Vent System			(N)	(I)	(NT)	
9.	Mechanical Vacuum Pump Exhaust Hogging System			(N)	(I)	(NT)	
10.	Evaporator Vent Systems			(N)	(I)	(NIT)	(I)

TABLE 11.5-5 (Cont'd)

No.	Process System	Monitor Provisions			Sample Provisions		
		In Process	In Effluent		In Process	In Effluent	
		Continuous	ACF	Continuous	Grab	Grab	Continuous
11.	Pre-Treatment Liquid Radwaste Tank Vent Gas Systems			(N)	(I)	(NIT)	(I)
12.	Flash Tank and Steam Generator Blowdown Vent Systems	*	*	*	*	*	*
13.	Turbine Bldg. Vent. Systems	N/A	N/A	N/A	N/A	N/A	N/A
14.	Pressurized Boron Recovery Vent Systems		(N)	(N)		(NT)	(I)

ACF - Automatic Control Feature

N - Noble gas radioactivity

T - Tritium radioactivity (carbon-14 analysis in gaseous effluents may be considered)

I - Radioiodine radioactivites and radioactivity of materials in particulate form and alpha emitters

G - Gross radioactivity

x - The automatic control feature is provided by the process continuous radiation monitor.

* - Monitored and Sampled by Unit 1 equipment

() - These provisions are required only for systems not monitored, sampled or analyzed (as indicated) prior to release by downstream provisions

a - Process monitoring by area monitors

TABLE 11.5-6

ST LUCIE 2 RADIATION MONITORING SYSTEM PROVISIONS
(This table is maintained as historical information. Radiation monitoring is performed in accordance with the Offsite Dose Calculation Manual and associated procedures.)

No.	Process System	Monitor Provisions			Sample Provisions		
		In Process	In Effluent		In Process	In Effluent	
		Continuous	ACF	Continuous	Grab	Grab	Continuous
1.	Liquid Radwaste (Batch) Effluent System	-	G	-	GR	GRT	-
2.	Liquid Radwaste (cont'd) Effluent System	N/A	N/A	N/A	N/A	N/A	N/A
3.	Service Water System	(1)	(1)	(1)	-	GRT	(1)
4.	Component Cooling Water System	G	G[x]	-	GR	-	-
5.	Spent Fuel Pool Treat-Ment System	G[**]	-	-	GR	-	
6.	Equip & Floor Drain Collection & Treatment Systems	-	(G)	-	(GR)	(GRT)	-
7.	Phase Separator Decant & Holding Basin Systems	N/A	N/A	N/A	N/A	N/A	N/A
8.	Chemical & Regeneration Solution Waste System	N/A	N/A	N/A	N/A	N/A	N/A
9.	Laboratory & Sample System Waste Systems	-	(G)	-	GR	(GRT)	-

TABLE 11.5-6 (Cont'd)

No.	Process System	Monitor Provisions			Sample Provisions		
		In Process	In Effluent		In Process	In Effluent	
		Continuous	ACF	Continuous	Grab	Grab	Continuous
10.	Laundry & Decontainment Waste Systems	-	(G)	-	(GR)	(GRT)	-
11.	Resin Slurry, Solidification & Baling Drain Systems	N/A	N/A	N/A	N/A	N/A	N/A
12.	Radwaste Liquid Tanks (outside the buildings)	N/A	N/A	N/A	N/A	N/A	N/A
13.	Storm & Underdrain Water System	-	-	-	-	(GRT)	-
14.	Tanks and Sumps inside containment	-	(G)	-	(GR)	(GRT)	-
15.	Boron Recovery System Liquid Effluent	N/A	N/A	N/A	N/A	N/A	N/A
16.	Steam Generator Blowdown (Batch) Liquid Effluent System	N/A	N/A	N/A	N/A	N/A	N/A
17.	Steam Generator Blowdown Continuous Liquid Effluent System	(2)	(2)	(2)	(2)	(2)	(2)
18.	Sec. Coolant Treatment Waste & Turbine Bldg Drain Systems	-	(G)	-	(GR)	(GRT)	-

TABLE 11.5-6 (Cont'd)

No.	Process System	Monitor Provisions			Sample Provisions		
		In Process	In Effluent		In Process	In Effluent	
		Continuous	ACF	Continuous	Grab	Grab	Continuous
19.	Ultrasonic Resin Cleanup Waste Systems	N/A	NIA	N/A	N/A	N/A	N/A
20.	Non Containment Waste Water and Pwr Turb Bldg Cold Drain System	-	-	-	-	(GRT)	(GRT)

G - Gross radioactivity

R - Principal identification and concentration of radionuclides and alpha emitters

T - Tritium radioactivity and radioactivity of materials in particulate form and alpha emitters.

** Area Monitoring

Notes

(1) - System not in contact with potentially contaminated systems, therefore no monitors.

(2) - Process and effluent by Unit 1 equipment

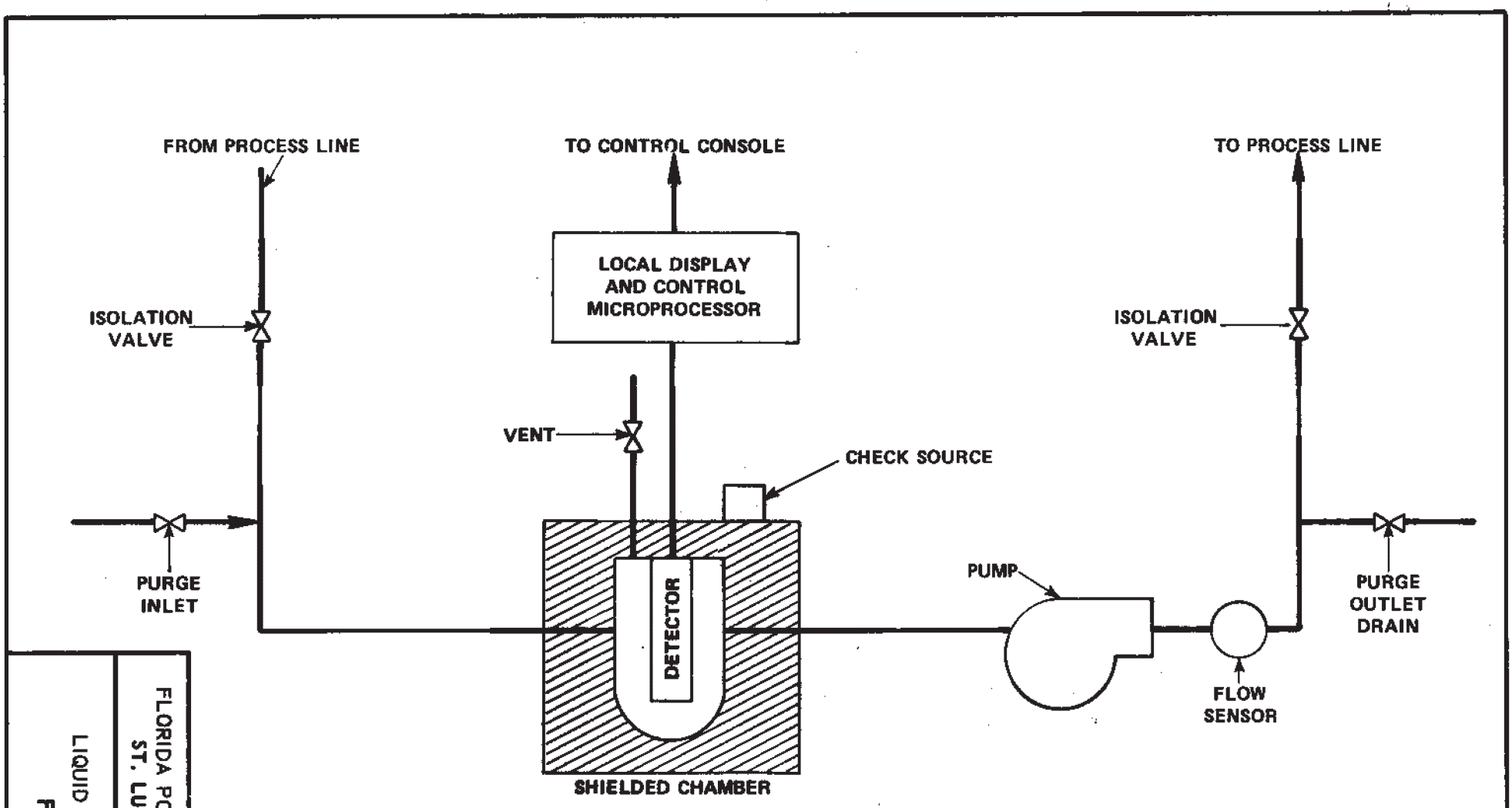
( ) - These provisions are required only for systems not monitored, sampled or analyzed (as indicated) prior to release by downstream provisions

x - The automatic control features is provided by the process continuous radiation monitor.

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

RADIATION MONITORING SYSTEM
BLOCK DIAGRAM
FIGURE 11.5-1

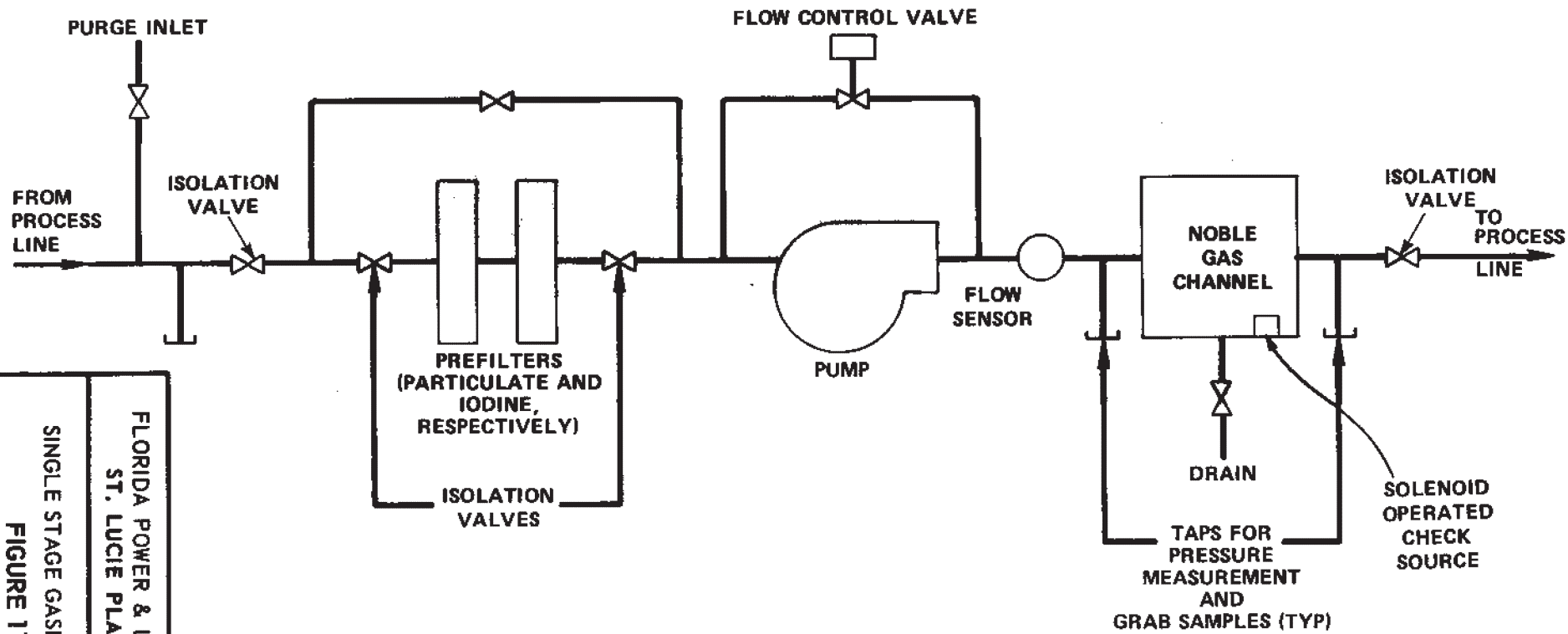
Amendment No. 18 (01/08)



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

LIQUID MONITOR SCHEMATIC

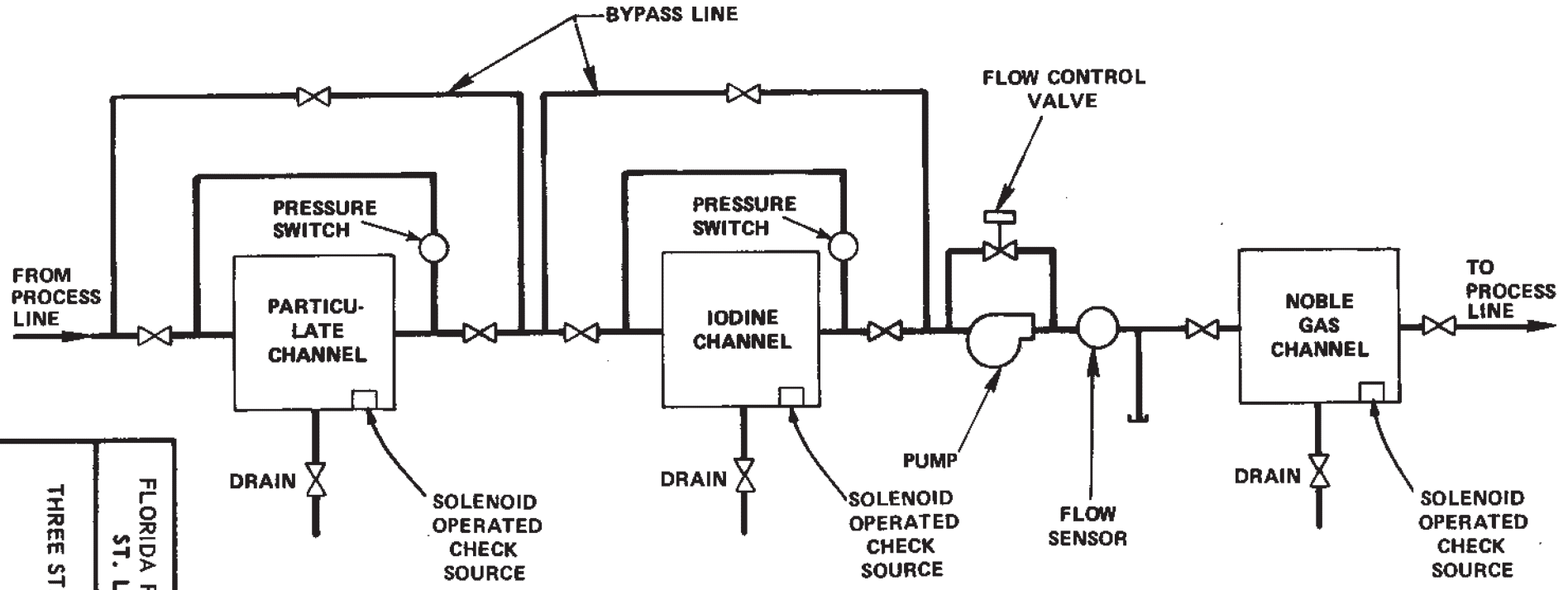
FIGURE 11.5–2



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

SINGLE STAGE GASEOUS MONITOR

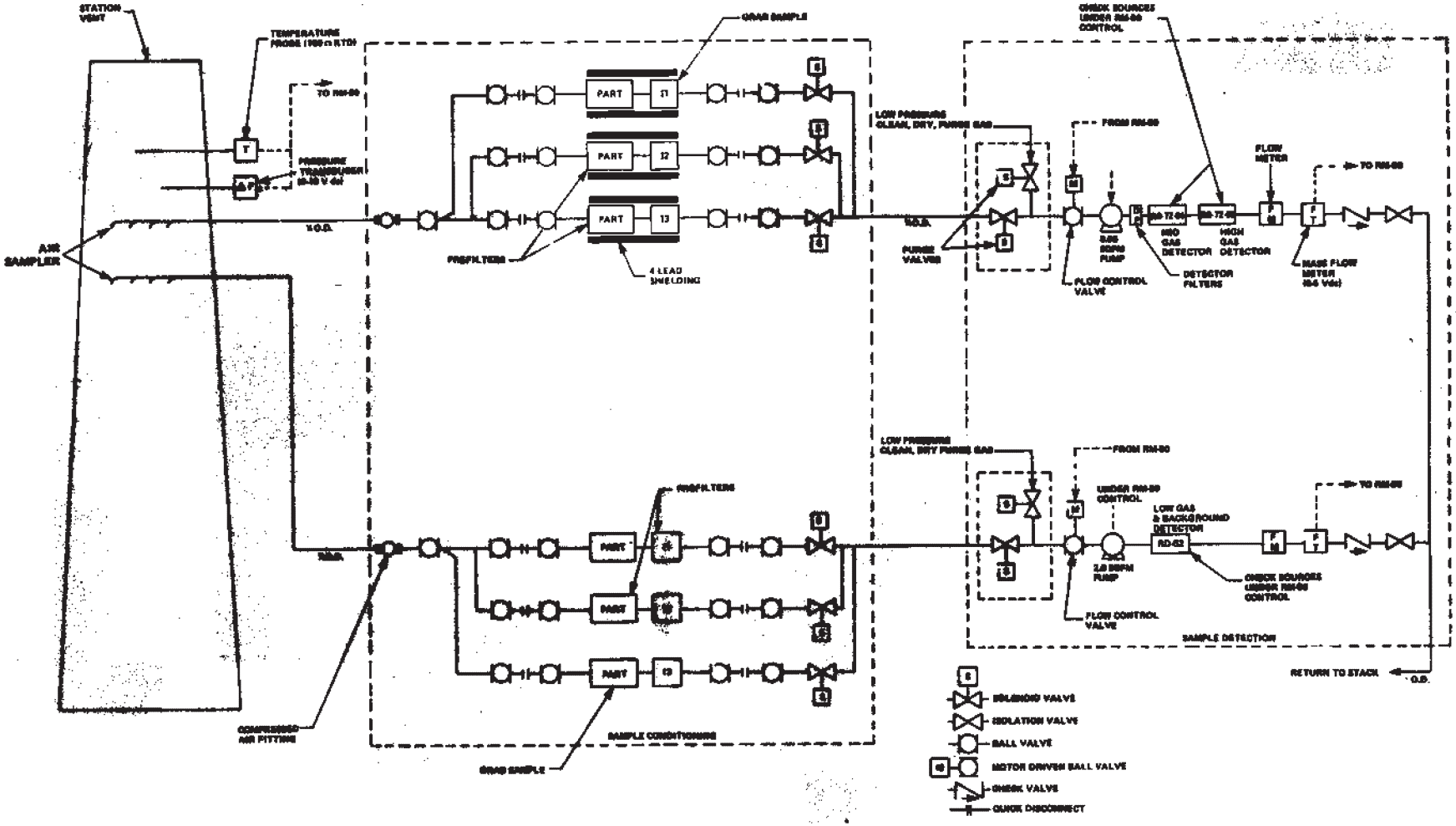
FIGURE 11.5-3



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

THREE STAGE AIRBORNE MONITOR.

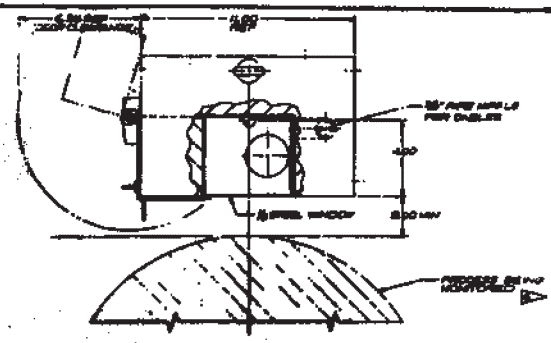
FIGURE 11.5–4

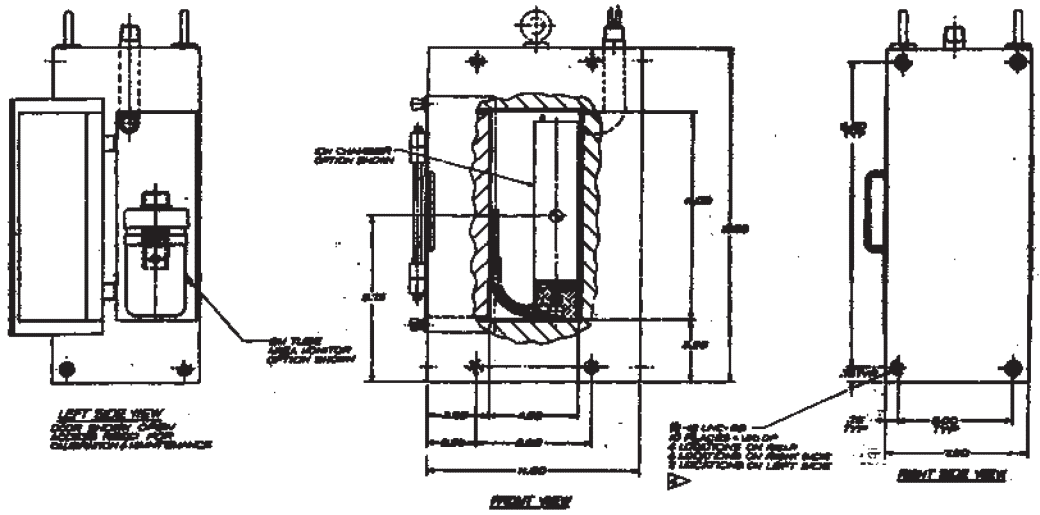


FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 2

MULTISTAGE GASEOUS MONITOR

FIGURE 11.5-5





PART NO.	AREA MONITOR DETECTOR OPTION	RANGE
0860-6186 -01	RD-BAMON CHAMBER	[illegible]
0860-6186 -02	RD-AB [illegible]	[illegible]
0860-6186 -03	RD-AB (GM TUBE)	[illegible]
0860-6186 -04	RD-11 (GM TUBE)	[illegible]
0860-6186 -05	RD-IB (GM TUBE)	[illegible]

AMENDMENT NO. 9 (10/94)

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT - UNIT 2

EXTERNALLY MOUNTED MONITOR

FIGURE 11.5-6