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11.1 SOURCE TERMS

Section 11.1.1 through 11.1.6 have been superseded and are retained for historical / informational purposes only. The development of the RCS activity at EPU conditions depicted in Table 11.1-1A is discussed in Section 11.1.7.

11.1.1 FISSION PRODUCTS (HISTORICAL)

EC282513

The mathematic model used for determining the specific concentration of nuclides in the reactor coolant involves a group of time dependent simultaneous equations. The fuel pellet region and reactor coolant region are analyzed by applying a mass balance of production and removal for each nuclide thereby establishing a set of first order, differential equations. In the fuel pellet region the mass balance includes the fission product production by direct fission yield, by parent fission product decay, and by neutron activation, and removal by decay, by neutron activation, and by escape to the reactor coolant. In the coolant region the analysis includes the fission product production by escape from the fuel through defective fuel rod cladding, by parent decay in the coolant, and by neutron activation of coolant fission products. Fission products are removed by decay, by coolant purification, by feed (makeup) and bleed (letdown to waste management system) for fuel burnup, and by leakage or other feed and bleed due to such operations as cold or hot startups and shutdowns or load follow operations.

The expression derived to determine the fission product inventory in the fuel pellet region is:

$$\frac{dN_{p,i}}{dt} = FY_i P + (f_{i-1}\lambda_{i-1} + \sigma_{i-1}\phi) N_{p,i-1} - (\lambda_i + \nu_i + \sigma_i \phi) N_{p,i}$$

and in the reactor coolant region is:

$$\frac{dN_{c,i}}{dt} = D\nu_i N_{p,i} + (f_{i-1}\lambda_{i-1} + \sigma_{i-1}\phi FCS) N_{c,i-1} - (\lambda_i + \frac{R}{W}\eta_i + \frac{(1-\eta_i)C}{C_o - Ct} + \frac{Q}{V}) N_{c,i}$$

where: N = population, atoms

F = average fission rate, fissions/"MWt"-sec

Y = U235 fission yield of nuclide, atoms/fission

P = core power, MWt

λ = decay constant, sec^{-1}

σ = microscopic capture cross section, cm^2

δ = thermal neutron flux, neutrons/cm^2 -sec

v = escape rate coefficient, sec^{-1}

f = branching fraction

t = time, sec

D = defective fuel rod cladding, fraction

FCS = core coolant volume to reactor coolant volume ratio

R = purification flow rate during power cycle, lb /sec

W = reactor coolant mass during power cycle, lb

η = resin efficiency of chemical and volume control system ion exchanger for a given nuclide, fraction

C_o = initial boron concentration, ppm

C = boron reduction rate by feed and bleed, compensating for fuel burnup, ppm/sec

Q = leakage or other feed and bleed from reactor coolant, lb/sec

Subscripts

p = pellet region

c = coolant region

i = designates the nuclide (i.e., $_{54}Xe^{133}$)

i-1 designates the parent nuclide (i.e., $_{53}I^{133}$)

This model does not involve the fuel plenum and gap region of the core because the escape rate coefficients represent the overall release from the fuel pellets to the reactor coolant. Plenum and gap region activities are calculated using diffusion theory. The production terms involving the microscopic capture cross section are used only to produce Cs^{134} from Cs^{133}, the stable end product of the 133 chain. The removal term involving the microscopic capture cross section is used only with Xe^{135} and only in the pellet region because of insignificant effects on other nuclides and in the coolant region.

The reactor coolant fission product activity concentrations used as basic source terms are given in Table 11.1-1 and the data bases for these calculations in Table 11.1-2. The effects of expected plant operation as a result of startups and shutdowns are simulated by using a constant liquid waste rate of 1.17 gpm or 617,000 gallons per year. The tritium concentration is based on no recycle of concentrator distillate.

The primary factor in determining the fission product inventories is the escape rate coefficient. This is an empirical coefficient which was derived from experiments initiated by Bettis and performed in the NRX and MTR reactors.[1] The escape rate coefficients derived from these tests are given in Table 11.1-2. The escape rate coefficients listed 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 post-irradiation examination showed that some test rods had experienced centerline melting. Later tests were conducted in the NRX reactor to determine the effect of rod length on the release of fission gases and iodines from defective fuel.[2] A by-product of these experiments was the effect of linear heat rate on the escape rate coefficient. The escape rate coefficient for several nuclides as a function of the linear heat rate is shown in Figure 11.1-1. Also shown in Figure 11.1-1 are the escape rate coefficients used for noble gases and halogens. Since the average heat rate for a fuel rod at full power is 6.14 kw/ft (Table 4.4-2), well below the crossover points in Figure 11.1-1, escape rate coefficients for xenon, krypton and iodine are conservative.

The activity of radioactive corrosion products (crud) and its thickness on reactor coolant system surfaces has been evaluated using measured data from six operating pressurized water reactors. These reactors are Connecticut Yankee, Indian Point 1, Yankee Rowe, Saxton, Shippingport, and SM-1.

Even though these reactors have different water chemistries and different materials in contact with the reactor coolant, their crud activity (dpm/mg) is markedly similar. The average and maximum activities of the long lived isotopes and the crud concentration for each of the reactors are shown in Table 11.1-3. The average activity for the six reactors and the maximum activity occurring in any one reactor are given at the bottom of the table. As can be seen from these numbers, there is less than a factor of ten difference between the average and maximum values.

The half lives, reactions and gamma decay energies for each of the long lived isotopes (significant isotopes remaining after 48 hours decay) are shown in Table 11.1-4.

The radioactive crud originates on in-core, and out-of-core surfaces. The crud plates out on the core surfaces and re-erodes after a short irradiation period. The irradiation period or core residence time is determined by the following expressions:

$$t_{res} = \frac{-1}{\lambda} \ln \left[1 - \frac{AS\ (16\ .67\ )}{\sum_{c} \Phi\ C} \right] \quad \text{secs}$$

$$\sum_{c} \Phi \frac{fN}{AMU} (\delta_{th} \Phi_{th} + \delta_{f} \Phi_{f}) = activation\ rate, \frac{dis}{g - \sec}$$

where: λ = decay constant, sec^{-1}

A = circulating crud activity, dpm/mg

s = total reactor coolant system surface area, cm^2

C = core surface area, cm^2

AMU = atomic mass unit of nuclide, gm/mole

N = Avogadro's Number, atoms/mole

f = fractional abundance of the parent nuclide in the in-core film

δ_{th} = thermal neutron capture cross section, cm^2

δ_f = fast neutron capture cross section, cm^2

Φ_{th} = thermal neutron flux, neutrons/cm^2-sec

Φ_f = fast neutron flux, neutrons/cm^2-sec

16.67 = units conversion $\frac{\text{min-mg}}{\text{sec-gm}}$

Residence time for each nuclide for each of the six reactors was evaluated using core and system parameters for each plant along with the maximum activity given in Table 11.1-3 for any plant. The longest residence time calculated for each nuclide is listed in Table 11.1-5. The maximum crud specific activities (dpm/mg) for the various long lived isotopes in the crud are shown in Table 11.1-5 which is based on maximum core residence times and parameters for the nuclear steam supply system (NSSS). The maximum 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 primary system areas are as shown in Table 11.1-6.

The activity concentrations of the crud in the reactor coolant are determined by utilizing the maximum specific activities given in Table 11.1-5 and the average crud concentration (75 ppb) identified in Table 11.1-3. The crud activity is shown in Table 11.1-1 along with the fission production activity concentrations.

A zinc injection system was installed in the Fall of 2009. The zinc injection system will displace cobalt from the resident RCS oxide films. This will result in a temporary increase in the RCS coolant activity 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.

11.1.3 TRITIUM PRODUCTION (HISTORICAL)

EC282513

Tritium is produced in the coolant or enters the coolant from a number of sources. One source is the fissioning of uranium within the fuel which yields tritium as a ternary fission product. Since Zircaloy fuel cladding reacts with tritium to form zirconium hydride, no tritium diffuses through the cladding (3) (4). Therefore, tritium is released to the coolant only from defective fuel.

Tritium is also produced by the reaction of neutrons with boron in the control element assemblies (CEAs). Data from operating plants using B_4C CEAs indicates that no tritium is released from the CEAs. The tritium may combine with carbon to form hydrocarbons and/or with lithium to form lithium hydride thereby preventing diffusion through the Inconel cladding. The low internal temperature of the B_4C CEAs may also prohibit tritium diffusion. To account for possible cladding defects, it is assumed that one percent of the tritium produced in the CEAs is released to the coolant.

Major sources of tritium are the activation of boron, deuterium, and nitrogen within the reactor coolant. Boron in the form of boric acid is used in the coolant for reactivity control and is the major source of tritium. The deuterium is a natural constituent of water. Nitrogen may be present due to aeration of the coolant during shutdown and due to aerated makeup water. Lithium is produced in the coolant as a result of neutron boron reaction and may be added as a pH control agent. Lithium activation, historically considered a source of tritium, is no longer considered to be a significant tritium source.

Table 11.1-7 identifies the contribution of each source to the total tritium concentration.

11.1.4 NITROGEN-16 PRODUCTION (HISTORICAL)

EC282513

Nitrogen-16 is produced from the reaction of fast neutrons with oxygen-16 forming unstable oxygen-17 which decays to nitrogen-16 by omitting a proton. Nitrogen-16 has a half life of 7.35 seconds and emits a gamma ray of high energy 82 percent of the time. The gamma energies are 6.13 and 7.10 Mev in a ratio of 12.5 to 1. The nitrogen activity at the reactor vessel coolant outlet nozzles is 3.02 x 10^8 dpm/cm^3 which is based on the nitrogen-16 production parameters given in Table 11.1-8.

11.1.5 FUEL EXPERIENCE (HISTORICAL) EC282513

Operation during the Mid 1970's and earlier with stainless steel clad fuel rods in the Connecticut Yankee reactor shows fuel failure rates on the order of 0.01 percent. Zircaloy-clad UO_2 fuel in the Obrigheim reactor in Germany sustained a fuel failure rate just over 0.1 percent in its first cycle; this has fallen in the second cycle to essentially zero (<0.001 percent). The fuel failure rate in the Dresden 1 reactor over a nine-year period has averaged <0.1 percent with the rate more recently being even lower. Fuel in the Mihama reactor in Japan and the Point Beach reactor has exceeded the burnup at which failures in fuel of similar design were observed in Ginna, without exhibiting increases in coolant activity (indicative of fuel defects).

The fact that widespread defects in some reactors, associated with fuel clad contamination have now been recognized and corrective measures taken, provides further assurance that failures at this frequency from this cause will not occur in the future. Existing licensing regulations limit coolant activity to that associated with 1 percent failed fuel. Over the lifetime of an operating reactor, it is expected that coolant activity levels corresponding to 0.1 percent failed fuel will predominate.

11.1.6 LEAKAGE SOURCES (HISTORICAL) EC282513

There are several potential sources of leakage from the plant systems that can contribute to the total release to the environs. If leakage occurs from systems containing reactor coolant, gaseous radioactivity could be released via several pathways.

Normal leakage from the reactor coolant system exposed to the containment atmosphere is expected to be 40 gallons per day or less. Reactor coolant nuclide activities given in Table 11.1-1 are used as source terms for all leakage calculations. Under equilibrium conditions, 10 percent or less of the iodine and particulates leaking into the containment remains in the atmosphere and is available for release. The other 90 percent of the iodine and particulate is either plated out in the containment or remains in the liquids and is collected in the containment sump. The annual average exposed leakage into the reactor auxiliary building is expected to be 10 gallons per day or less.

Regulatory Guide 1.42 (RG 1.42) recommends leakage values to be used in gaseous release pathway estimates. These suggested leakage values have been used in the gaseous release analysis.

Means of detecting leakages are discussed in Section 5.2.4.

Estimated liquid and gaseous releases due to leakage from various systems containing radioactivity are discussed in Sections 11.2.6 and 11.3.6, respectively.

11.1.7 REACTOR COOLANT ACTIVITY AT EPU CONDITIONS

The equilibrium nuclide source term in the RCA is based upon the core inventory, the percentage of cladding defects, and fission products escape rate coefficients. The St. Lucie Unit 1 Core/Fuel source term was developed from ORIGEN 2.1 computer code calculations for EPU conditions. Section 3.4 of Regulatory 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 Units 1 & 2 AST dose calculations approved by the NRC in References 15 and 16. 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 15.4.1-1e.

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=0). 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 stripping 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 was 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 was 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 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, 1-134 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 analysis will continue to use the Thyroid dose conversion factors from Table 2.1 of Federal Guidance Report 11 (Reference 16).

Although this 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 was not 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 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 was adjusted to achieve the Technical Specification limit of 100/E-bar for non-iodine activities. Implementation of 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:

$$\text{E-bar} = [\Sigma(E^{\beta}_i + E^{y}_i] / \Sigma\, a_i$$

where: a_i = activity of individual isotope (μCi/gm)
E^{β}_i = average beta emission energy (MeV/dis)
E^{y}_i = average gamma emission energy (MeV/dis)

The maximum allowed specific activity is 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 was greater than 100/E-bar, the specific activities were adjusted downward to be representative of plant operation at the Technical Specification limit to produce the final RCS specific activity provided in Table 11.1-1A. EC282513

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TABLE 11.1-1
REACTOR COOLANT SPECIFIC ACTIVITY (µCi/cc),577F
(Historical – see TABLE 11.1-1A for current values)

Nuclide	Half Life	Anticipated Operational Occurrences (1.0% Failed Fuel)	Normal Operations (0.1% Failed Fuel)	Nuclide	Half Life	Anticipated Operational Occurrence (1.0% Failed Fuel)	Normal Operations (0.1% Failed Fuel
H-3	12.3y	0.132	0.107	I-133	21h	5.66	0.566
Br-84	32m	4.66(-2)*	4.66(-3)	Xe-133	5.3d	181.0	18.1
Kr-85m	4.4h	1.49	0.149	Te-134	42m	2.62(-2)	2.62(-3)
Kr-85	10.8y	0.885	8.85(-2)	I-134	52m	0.62	6.2(-2)
Kr-87	76m	0.81	8.1(-2)	Cs-134	2.ly	0.10	1.0(-2)
Kr-88	2.8h	2.6	0.26	I-135	6.7h	2.7	0.27
Rb-88	18m	2.55	0.255	Xe-135	9.2h	7.53	0.753
Rb-89	15m	6.4(-2)	6.4(-3)	Cs-136	13d	2.55(-2)	2.55(-3)
Sr-89	51d	5.07(-3)	5.07(-4)	Cs-137	30y	0.32	3.2(-2)
Sr-90	28.8y	2.61(-4)	2.61(-5)	Xe-138	17M	0.36	3.6(-2)
Y-90	64h	1.02(-3)	1.02(-4)	Cs-138	32m	0.69	6.9(-2)
Sr-91	9.7h	3.56(-3)	3.56(-4)	Ba-140	12.8d	6.11(-3)	6.11(-4)
Y-91	59d	0.111	1.11(-2)	La-140	40.2h	5.85(-3)	5.85(-4)
Mo-99	67h	2.03	0.203	Pr-143	13.6d	5.84(-3)	5.84(-4)
Ru-103	39.6d	4.13(-3)	4.13(-4)	Ce-144	285d	4.13(-3)	4.13(-4)
Ru-106	367d	2.48(-4)	2.48(-5)	(Corrosion Products)			
Te-129	67m	2.51(-2)	2.51(-3)	Co-60	5.2y	5.19(-4)	5.19(-4)
I-129	1.7(7)y	7.21(-8)	7.21(-9)	Fe-59	45d	2.13(-5)	2.13(-5)
I-131	8.0d	3.97	0.397	Co-58	71d	4.66(-3)	4.66(-3)
Xe-131m	12d	1.48	0.148	Mn-54	312d	2.75(-5)	2.75(-5)
Te-132	78h	0.33	3.3(-2)	Cr-51	27d	3.8(-3)	3.8(-3)
I-132	2.3h	1.09	0.109	Zr-95	65d	9.35(-7)	9.35(-7)

* ( )Denotes Power of ten

TABLE 11.1-1A
REACTOR COOLANT SPECIFIC ACTIVITY (EPU)

Equilibrium RCS activity was adjusted to make the iodine inventory equivalent to the Technical Specification Dose Equivalent I-131 limit, and also adjusted to make the other relevant RCS isotopic inventory equivalent to the (proposed) Technical Specification limit on Dose Equivalent Xe-133, as defined in TSTF-490.

The RCS activity based on core inventory and the ANSI/ANS 18.1-1999 based corrosion products were combined to produce the total equilibrium RCS activity. In addition, the equilibrium RCS activity was adjusted to make the iodine inventory equivalent to the Technical Specification Dose Equivalent I-131 limit, and also adjusted to make the other relevant RCS isotopic inventory equivalent to the Technical Specification limit on Dose Equivalent Xe-133. This combined and adjusted fission and corrosion activity RCS source term is provided in the following table.

RCS FISSION & CORROSION PRODUCT TECHNICAL SPECIFICATION SOURCE TERM FOR ST. LUCIE

Nuclide	**µCuries/gm**	**Nuclide**	**µCuries/gm**
Co-58	4.394E-02	Cs-136	1.543E+00
Co-60	8.256E-02	Cs-137	2.899E+00
Cr-51	1.611E-01	Ba-139	4.800E-04
Fe-55	1.818E-01	Ba-140	7.124E-03
Fe-59	2.087E-02	La-140	3.299E-03
Mn-54	2.163E-01	La-141	2.646E-04
Zn-65	4.872E-03	La-142	7.483E-05
Kr-85	4.160E+01	Ce-141	1.147E-03
Kr-85m	1.268E+00	Ce-143	6.485E-04
Kr-87	7.574E-01	Ce-144	9.815E-04
Kr-88	2.247E+00	Pr-143	1.031E-03
Rb-86	3.403E-02	Nd-147	4.302E-04
Sr-89	5.691E-03	Kr-83m	3.826E-01
Sr-90	5.459E-04	Xe-138	5.168E-01
Sr-91	1.526E-03	Xe-131m	3.022E+00
Sr-92	6.105E-04	Xe-133m	3.479E+00
Y-90	7.799E-04	Xe-135m	8.553E-01
Y-91	3.327E-02	Cs-138	9.667E-01
Y-92	7.402E-04	Cs-134m	1.179E-01
Y-93	4.706E-04	Rb-88	2.326E+00
Zr-95	1.159E-03	Rb-89	1.022E-01
Zr-97	5.170E-04	Sb-124	1.910E-03
Nb-95	1.187E-03	Sb-125	1.277E-02
Mo-99	5.111E+00	Sb-126	9.173E-04
Tc-99m	3.885E+00	Te-131	1.741E-02
Ru-103	1.359E-03	Te-134	2.280E-02
Ru-105	1.934E-04	Te-125m	2.828E-03
Ru-106	7.973E-04	Te-133m	1.326E-02
Rh-105	7.047E-04	Ba-141	1.009E-04
Sb-127	5.187E-02	Rh-103m	1.350E-03
Sb-129	2.918E-02	Nb-97	8.889E-05

RCS FISSION & CORROSION PRODUCT TECHNICAL SPECIFICATION SOURCE TERM FOR ST. LUCIE
(CONTINUED)

Nuclide	**µCuries/gm**	**Nuclide**	**µCuries/gm**
Te-127	5.496E-02	Nb-95m	8.343E-06
Te-127m	8.677E-03	Pm-147	1.069E-04
Te-129	4.558E-02	Pm-148	1.866E-04
Te-129m	2.486E-02	Pm-149	3.485E-04
Te-131m	4.273E-02	Pm-151	1.021E-04
Te-132	5.233E-01	Pm-148m	2.556E-05
I-131	8.425E-01	Pr-144	9.816E-04
I-132	1.689E-01	Y-94	1.528E-05
I-133	8.713E-01	Y-91m	8.872E-04
I-134	7.726E-02	Br-82	6.096E-02
I-135	3.933E-01	Br-83	1.214E-01
Xe-133	2.381E+02	Br-84	5.002E-02
Xe-135	9.235E+00		
Cs-134	6.972E+00		

TABLE 11.1-2

BASES FOR REACTOR COOLANT RADIOACTIVITY
(Historical)

EC 282513

Core power level, Mwt	2700
Fuel cycle full power days	357**
Percent failed fuel	1.0
CVCS purification ion exchanger decontamination factor	10
Purification flow rate (CVCS purification ion exchanger) , gpm	40
Effective purification flow rate for lithium and cesium removal gpm	8
Fission product escape rate coefficients, Sec^{-1} (Based on centerline melting of fuel)	
Noble gases	6.5×10^{-8}
Halogens, Cs	2.3×10^{-8}
Te, Mo.	1.4×10^{-9}
All others	1.4×10^{-11}
Feed and bleed liquid waste for fuel burnup, gal/yr	216,000
Other feed and bleed liquid waste, gal/yr	617,000
Thermal neutron flux, n/cm^3 sec	4.3×10^{13}
Reactor coolant volume, ft^3	9662*

* Not including Pressurizer

** Due to the overall conservatism, values in Table 11.1-1 remain applicable for operation at 2700 MWt with extended burnup.

TABLE 11.1-3

CRUD SPECIFIC ACTIVITY (dpm/mg) - OPERATING REACTORS
(Historical)

Reactor		Nuclide									
		Co-60	Co-58	Mn-54	Cr-51	Fe-59	Hf-181	Zr-95	Cu-64	Crud ppb	References
Conn.	avg.	1.4(+7)*1.2(+8)		2.4(+6)	2.7(+7)	4.2(+6)	-	-	-	85	
Yankee	max.	3.5(+7)	4.0(+8)	1.2(+7)	1.4(+8)	1.5(+7)	-	-	-	4000	5
Indian	avg.	1.4(+6)	5.1(+6)	6.6(+5)	6.1(+6)	1.3(+6)	1.5(+5)	3.0(+5)	3.1(+9)	72	14
Point 1	max.	2.0(+6)	9.1(+6)	2.0(+6)	8.2(+6)	3.3(+6)		4.2(+5)	1.2(+10)	-	
Yankee	avg.	6.7(+6)	3.3(+7)	4.5(+6)	1.7(+7)	5.5(+6)	-	6.6(+5)	-	70	7,11
Rowe	max.	2.1(+7)	1.2(+8)	1.9(+7)	1.4(+8)	1.8(+7)	-	1.8(+6)	-	-	
Saxon	avg.	5.5(+6)	4.6(+7)	7.7(+6)	9.0(+7)	2.7(+6)	-	-	-	55	8, 9, 10, 11
	max.	2.2(+7)	1.5(+8)	1.4(+7)	1.1(+8)	6.0(+6)	-	-	-	250	
Shipping-	avg.	2.3(+7)	2.8(+6)	1.3(+6)	2.2(+6)	1.8(+6)	5.2(+5)	7.0(+5)	-	75	
port	niax.	4.8(+7)	3.2(+6)	1.7(+6)	2.2(+6)	1.8(+6)	7.6(+5)	9.7(+5)	-	-	12,13
San Onofre	avg.	2.0(+6)	2.2(+7)	1.4(+6)	3.1(+6)	6.7(+5)	-	-	-	90	6
	max.	2.0(+7)	1.2(+8)	4.2(+6)	6.7(+6)	3.8(+6)	-	-	-	400	
	avg.	8.8(+6)	3.8(+7)	3.0(+6)	2.4(+7)	2.7(+6)	3.4(+5)	5.5(+5)	3.1(+9)	75	
	max.	4.8(+7)	4.0(+8)	1.9(+7)	1.4(+8)	1.8(+7)	7.6(+5)	1.8(+6)	1.2(+10)	4000	

*Numbers in () are powers of ten.

TABLE 11.1-4

LONG LIVED ISOTOPES IN CRUD
(Historical)

Nuclide	$T_{1/2}$	Parent	Reaction	γ/dis	E(Mev)
Co-60	5.2y	Co-59	N, γ	2.00	1.25
Co-58	71d	Ni-58	N, P	1.00	0.81
Mn-54	312d	Fe-54	N, P	1.00	0.84
Cr-51	27d	Cr-50	N, γ	0.10	0.32
Fe-59	45d	Fe-58	N, γ	1.00	1.18
Hf-181	45d	Hf-180	N, γ	0.25	0.50
Zr-95	65d	Zr-94	N, γ	2.00	0.75
Cu-64	12.9h	Cu-63	N, γ	1.24	0.51

TABLE 11.1-5
LONG-LIVED CRUD ACTIVITY
(Historical)

Isotope	$T^{(b)}_{res}$ (sec)	Half Life	Crud Activity. (dpm/mG)(c)	Coolant Activity (μci/cc)(567F)(c)
Co-60	243	5.2y	2.3(+7)(a)	5.6(-4)
Co-58	27	71d	2.1(+8)	5.0(-3)
Mn-54	89	312d	2.3(+6)	5.6(-5)
Cr-51	17	27d	1.1(+8)	2.6(-3)
Fe-59	Sat.	45d	1.3(+6)	3.0(-5)

(a) Denotes power of ten.

(b) Based on maximum crud activities from operating plants. (See Table 11.1-3).

(c) Based on maximum crud activity levels and parameters for this plant.

TABLE 11.1-6

EQUILIBRIUM CRUD FILM THICKNESS
(Historical)

Location	Thickness (mg/cm^2)
Vessel Internals, Piping, Steam Generator Inlet Plenum	1.0
Pressurizer	0.4
CRDM, Vessel Head Incore Instrument Tops	0.3
Steam Generator Tubing	0.1

TABLE 11.1-7

SOURCES OF TRITIUM PRODUCTION
(Historical)

Source	Basis	Concentration* (µCi/cc at 577°F)***
Fission	1.0% failed fuel	0.028
CEA	1.0% cladding defects	0.0021
Boric Acid	Base loaded boron concentration	0.1008
Deuterium	150 ppm	0.0009
Lithium**	0.5 ppm	0.00036
Nitrogen	10 cc(N_2)/kg(H_2O)	0.0000112
Total		0.132

* Calculated on same basis as Table 11.1-1. These values remain applicable for use in Table 11.1-1 for operation at 2700 Mwt with extended burnup.

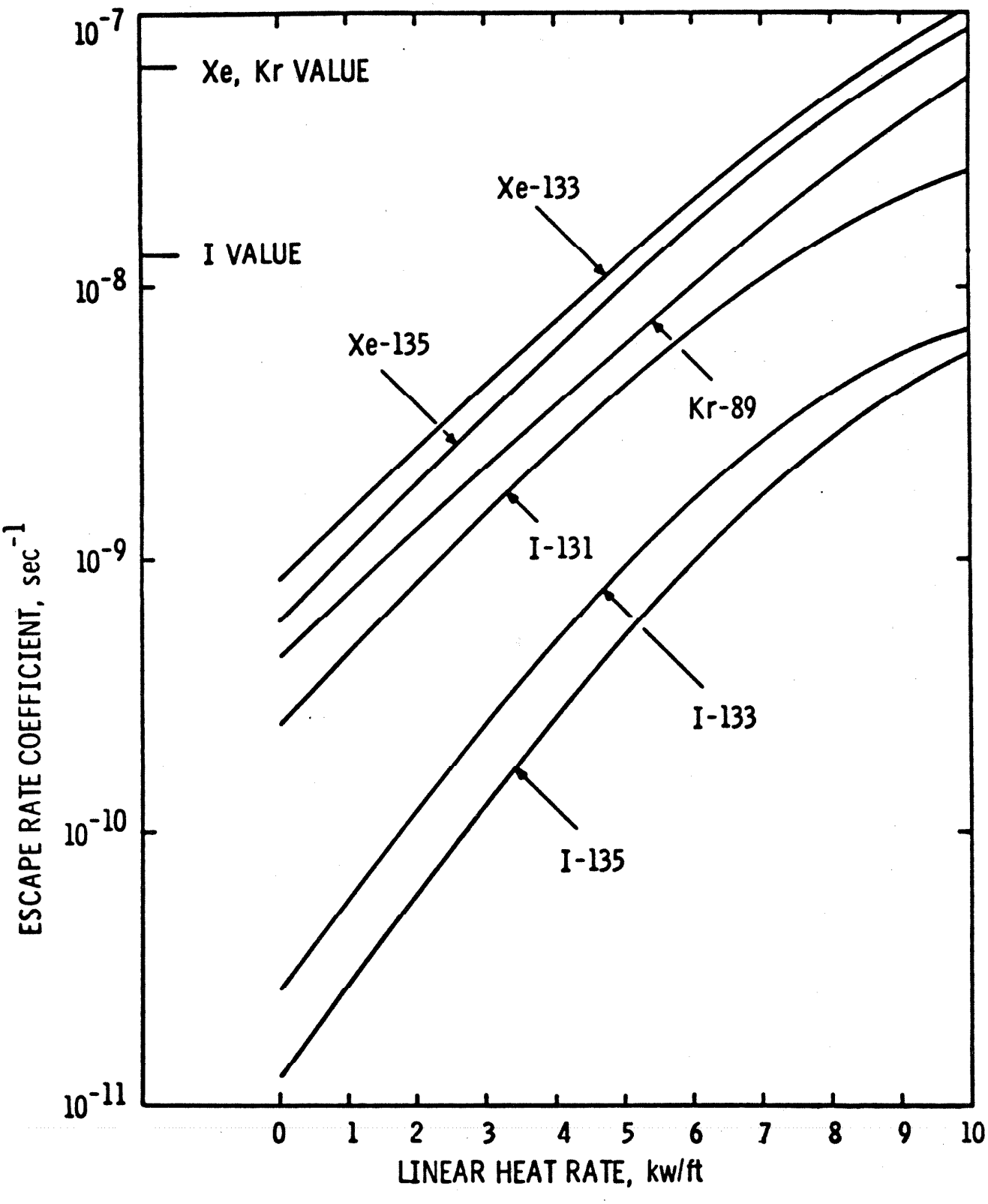
** Lithium activation is no longer considered to be a significant tritium source.

*** 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 11.1-8

NITROGEN-16 PRODUCTION PARAMETERS
(Historical)

Parameter	Value
Fast Neutron Flux, n/cm^2-sec	6.61×10^{13}
Coolant Density, g/cm^3	0.73
Core Transit Time, sec	0.853
Coolant Recirculation Time, sec	11.29
Reaction Cross Section, barns	2×10^{-5}
Reactor Power, Mwt	2700



FLORIDA POWER & LIGHT CO. St. Lucie Plant	Escape Rate Coefficients	Figure 11.1-1

11.2 LIQUID WASTE SYSTEMS

See updated information concerning the Liquid Waste Systems presented in the SL-1 10 CFR 50 Appendix I submittal, (6/1/76 Supplement to the St. Lucie Unit 1 Environmental Report).

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 1 and is retained here for historical purposes.

Continued compliance with the annual regulatory dose limits following 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 EPU 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-EPU conditions. This approach was utilized to estimate the maximum potential increase in effluent doses due to the EPU and to demonstrate that the estimated off-site doses following the EPU,
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 Radioactive Effluent Release Report.

11.2.1 DESIGN BASES

The boron recovery system and the liquid waste system (integral parts of the waste management system) are designed to:

a) process the various potentially radioactive liquid wastes such that the radioactivity release to the environs during normal operation will be as low as practicable. The numerical design objectives for releases during normal operation are to limit average annual liquid activity release quantity to 5 Ci and average annual activity release concentration to 2×10^{-8} µCi/cc excluding tritium and dissolved fission product gases.

b) limit the annual average tritium discharge concentration to 5×10^{-6} µCi/cc in accordance with the proposed Appendix I to 10 CFR 50.

c) limit releases due to anticipated operational occurrences within 10 CFR 20.

11.2.2 SYSTEM DESCRIPTION

Liquid waste influent to the waste management system, shown in Table 11.2-1 is segregated by chemistry and/or probable source activity for more efficient processing. Section 1.2.6 lists the liquid waste management system as a shared system which may, under certain conditions, be used by Unit 2. Tritiated, hydrogenated, borated reactor coolant quality wastes of potentially high activity are mainly processed in the boron recovery system. Aerated, chemically contaminated, and low activity liquid wastes are received and processed separately in the liquid waste system. Table 9.3-6 lists the drains routed to the drain tanks.

11.2.2.1 Boron Recovery System (BRS)

The major influent to the boron recovery system is reactor coolant from the chemical and volume control system letdown due to feed and bleed operations for shutdown, startups, and boron dilution over core life. During normal plant operation, reactor coolant quality water from valve and equipment leakoffs, drains and relief valve within the containment are collected in the reactor drain tank and subsequently processed by this system. Reactor coolant from leakoffs and drains in the reactor auxiliary building are collected in the equipment drain tank of the liquid waste system. An alternate drain path to return potentially radioactive leakage from the Emergency Core Cooling System (ECCS) pump area to the Containment Building is shown on Figures 6.2-28, 11.2-1 and 11.2-4.

The Leakage Collection and Return System (LCRS), collects leakage from equipment, valves and pumps, and provides an alternate path for returning this leakage to the containment reactor drain tank. Following a LOCA, operator action from the Control Room actuates the return system, thereby preventing the spread of radioactive sources to the waste management system. A manual CIS override is provided on containment isolation valves V6301 and V6302 to provide the flow path to the reactor drain tank. To ensure flow to the Equipment Drain Tank (EDT) during normal operation and to the Reactor Drain Tank (RDT) during accident conditions, an interlock insures that the normally open air operated valves HCV-06-7 and I-HCV-06-9 are closed, prior to opening the normally closed valve I-HCV-06-8, and initiating return flow to the containment.

The boron recovery P&I diagrams are shown on Figures 11.2-1, 11.2-2, and 11.2-3. The borated and hydrogenated water discharged by the reactor drain pumps or diverted by the chemical and volume control system volume control tank diversion valve (V2500) is sent to the waste management system holdup tanks. If RCS activity is above a pre-established threshold or if the nitrogen blanket in the holdup tanks is lost, RCS flow is diverted to the flash tank for processing. At the flash tank, dissolved hydrogen and fission gases are stripped by a counter current flow of nitrogen gas from the liquid and discharged to the gas surge tanks. Hydrogen is stripped from the water so that an explosive gas mixture does not occur in subsequent process equipment. Use of nitrogen cover gas in the holdup tanks provides additional protection. The nitrogen stripping medium maintains a slight overpressure in the flash tank to prevent air in-leakage, thus precluding potential formation of an explosive gas mixture in the flash tank. In the event of a high liquid level in the flash tank, the influent is automatically diverted to the holdup tanks. A low level in the flash tank stops the flash tank pumps automatically. The flash tank high and low pressure and level alarms are annunciated in the control room. A flow switch in the inlet line to the flash tank automatically opens the nitrogen supply valve and starts the flash tank pumps when water enters the tank. The degassed liquid is automatically pumped from the flash tank to the holdup tanks.

The holdup tanks provide storage capacity to accumulate discharges until a sufficient volume is available for further processing which is usually done on a batch basis. The radioactivity of the liquid is significantly reduced during storage by natural decay of the short lived radionuclides. During this period any degasification of radioactive materials can be monitored by the plant vent radiation monitors. Air in-leakage to the holdup tanks is precluded by a nitrogen overpressure maintained in the tanks. As the holdup tanks fill, nitrogen is displaced to other interconnected holdup tanks or to the gas collection header. The holdup tanks have high and low water level and pressure alarms which are annunciated in the control room. In the event process fluid has been routed directly into the holdup tanks, bypassing the flash tank , the stored liquid can be recirculated by the holdup drain or recirculation pumps to the flash tank to remove any hydrogen once the flash tank is placed in service. The holdup recirculation pump supplies flushing water to the preconcentrator ion exchangers and spent resin tank during resin sluice operations.

Normally, the contents of the holdup tanks are recirculated through the preconcentrator filter and preconcentrator ion exchanger prior to transfer to the waste management stream and/or RWT for reuse. 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. If necessary, the contents of one holdup tank may be recirculated through a preconcentrator filter and ion exchanger while a second holdup tank contents are processed through the other preconcentrator filter and ion exchanger prior to discharge to the waste management stream. The holdup pumps and holdup recirculation pump are stopped on low holdup tank level. The boric acid concentrators have both automatic sampling and local grab sample provisions to ensure control of the effluent chemistry.

NOTE: The information that follows in this subsection about the boric acid concentrators is maintained here for historical purposes only. These concentrators and supporting equipment are no longer used.

The boric acid concentrator is designed to concentrate a dilute solution of boric acid (30 to 1720 ppm) to 10,900 to 21,000 ppm. The resultant condensate (distillate) has a maximum boron concentration of 10 ppm.

The major components of the boric acid and waste concentrators are: the concentrator, vapor separator, feed preheater, vapor condenser, distillate cooler, concentrate cooler, and pumps as shown of Figures 11.2-2 and 11.2-4A.

When the boric acid concentrator is in use, the preheated boric acid solution is piped into the lower shell of the concentrator. Low pressure heating steam in the concentrator upper shell heats the boric acid solution to boiling. The water vapor along with remaining liquid rises through the tubes, into the steam chest and then out of the concentrator.

The vapor in the tubes causes a natural circulation upward at a flow rate of many times the boric acid feed rate. The entire volume of liquid slowly rises in concentration as water is boiled off and more solids enter with the feed boric acid solution until the desired concentration rate is reached. The concentrate fluid is pumped out of the concentrator into the boric acid holding tank at a rate that maintains the desired concentration.

In the vapor separator, the distillate is sprayed over the demisters to wash the vapors free of boric acid. Two stage vapor washing in the vapor separator is necessary to lower the boric acid content of the distillate below 10 ppm.

The distillate vapors are condensed in the vapor condenser and a distillate is formed which is pumped out of the system after being cooled by the distillate cooler.

The hot boric acid concentrate from the concentrator is pumped through the concentrate cooler and is cooled to the design temperature.

The bottoms from each boric acid concentrator are pumped via the boric acid discharge strainer to the boric acid holding tank for temporary storage and sampling. The recovered boric acid may then be returned to the makeup tanks.

The concentrator distillate passes through the boric acid condensate ion exchangers to remove boron carryover and into one of the two boric acid condensate tanks. While one boric acid condensate tank is filling, the other is sampled and recycled to the primary water tank. In the event that the contents of the tank do not meet the chemical or radioactivity limitations, the contents can be recycled to the holdup tanks for further processing, recycled through the boric acid condensate ion exchanger or drained to the liquid waste system.

A local high and low water level alarm is provided on the boric acid condensate tanks. The boric acid condensate pumps automatically stop on low water level in the tanks.

The liquid waste is pumped through a waste ion exchanger and then to the Waste Monitor Tanks. The waste can then be pumped to the circulating water discharge for release or returned to the aerated waste storage tank for reprocessing if the activity is too high.

Prior to controlled discharge of the treated liquid waste from the Waste Monitor Tanks the fluid must be analyzed and its activity verified as acceptably low. Discharge is accomplished through an effluent radiation monitor which records the release activity level and automatically terminates discharge on high radiation. If reuse in the plant is desired, the fluid is analyzed for acceptability of both chemistry and activity. The design criteria and controls for limiting radiation exposure of personnel from lines which normally carry radioactive fluids are discussed in Sections 12.1.3 and 12.1.5.

Design data for the major components are listed in Table 11.2-2. Flow, temperature and pressure data are given in Table 11.2-3 with the locations corresponding to process points on Figures 11.2-1, 11.2-2 and 11.2-3. Notice that the boric acid concentrators and associated components are no longer used.

The nuclide concentration for normal operation and for anticipated operational occurrences adjusted to 70°F are indicated for selected locations in the chemical and volume control system and boron recovery system in Tables 11.2-6 and 11.2-7, respectively. The selected locations are indicated by the process points on Figures 11.2-1, 11.2-2 and 11.2-3. Normal operation is defined as operating with 0.1 percent failed fuel, and an anticipated operational occurrence is defined as operation with 1 percent failed fuel.

Analysis is made assuming that the activities in Table 11.1-1 exists in the coolant upstream of the purification equipment in conjunction with the expected equipment performance given in Table 11.2-4 and in Table 11.2-5.

The DF's of 2 and 5 for fission gases in the BRS flash tank and concentrator are based on the removal of xenon and krypton dissolved in borated water. These values reflect test data and vendor information pertaining to the performance characteristics of the equipment. Field testing performed by CE indicates that a flash tank is approximately 50 percent effective for removing dissolved noble gases. Likewise, discussion with concentrator vendors have indicated that DF's of up to 10 are possible with boiling liquid type evaporators, noting that the vapor condensor is continuously vented to remove non-condensible gases. Thus, the DF values presented in Table 11.2-5 represent 1970's vintage technology for fission gas removal by this equipment.

Systems similar in function and design to the boron recovery system described herein have been used successfully at plants such as Connecticut Yankee and Ginna. Even with significant coolant radioactivity at Ginna, releases have been controlled well within the 10 CFR 20 limits.

All process components have been used extensively in the nuclear industry to remove radioactive contaminants from liquids. The performance of process units used in the analysis is in agreement with general industry experience.

11.2.2.2 Liquid Waste System

The liquid waste system is shown on Figures 11.2-4, through 11.2-4c. Liquid wastes include those from the laboratory sink drains, decontamination area drains, floor drains, building sumps, laundry effluent, and contaminated showers. The wastes are segregated for batch processing by collection in the equipment drain tank and chemical drain tank, and laundry drain tanks.

Low activity, aerated, and potentially dirty liquid drains and building sumps discharge to the chemical and equipment drain tanks. Low activity chemical drains from the sampling system, decontamination drains, and chemical laboratory drains flow to the chemical drain tank or to local drums. When a sufficient volume is collected in the drain tanks, the contents are pumped to the Aerated Waste Storage Tank (AWST). From the AWST, liquid wastes are normally processed through two to four waste ion exchangers, depending on waste chemistry and activity. These ion exchangers can be operated in series, parallel or a combination of the two. The ion exchanger vessels can contain mixed bed, cation, anionic resins, or and/or activated carbon or other filter media. Typically, the liquid waste is processed through two mixed bed ion exchangers in series, holding the other two in reserve.

The waste ion exchangers discharge to the Waste Monitor Tanks (WMT's). After a sample analysis has determined that the radioactivity of the WMT contents is within the discharge limit, the WMT is emptied at a controlled rate into the condenser water discharge. The radioactivity of the dischage is continuously monitored and recorded. Should the high set point of the monitor be exceeded, WMT release is automatically terminated.

In the event that discharge criteria are not met, the contents of the WMT are pumped back to the AWST for reprocessing through the waste ion exchangers.

NOTE: The information that follows in this subsection about the waste concentrator is maintained here for historical purposes only. The concentrator and supporting equipment are no longer used.

The major components of the waste concentrator are: the concentrator, vapor separator, vapor condenser and sub cooler, concentrate cooler and pumps.

Liquid waste containing dissolved solids and/or radioactive nuclides are collected in the aerated waste storage tank and are passed through a waste filter prior to entering the lower shell of the concentrator. Low pressure steam heats the waste solution in the concentrator causing water vapor and liquid waste solution to rise through the tubes into the steam chest and then out of the concentrator into the vapor separator. In the vapor separator, recycled distillate is sprayed over the demisters to scrub boric acid from the

water vapor. Two water vapor scrub stages are necessary to lower the boric acid concentration in the distillate to less than 10 ppm. The water vapor is then condensed in the vapor condenser and cooled by the distillate cooler. Approximately half of the condensed water vapor (distillate) is used for the boric acid scrub stages in the vapor separator. The balance is either collected in the waste condensate. tank and then pumped to WMT's through the waste ion exchangers or directly to the WMT's via the ion exchangers.

After the sample analysis has determined that the radioactivity of a WMT is within the discharge limit, the contents are pumped at a controlled rate into the condenser water discharge. The radioactivity of the discharge is continuously monitored and recorded. Should the high set point of the monitor be exceeded, the release is automatically terminated.

In the event that discharge criteria are not met, the waste is circulated through the waste ion exchanger and returned to the waste monitor tank for re-analysis. If the discharge criteria cannot be obtained by the waste ion exchanger, the waste may be processed via the waste concentrator. Should the volume of liquid wastes exceed the capacity of the waste concentrator, the boric acid concentrator(s) could be used as a waste evaporator.

The laundry wastes are collected in the laundry drain tank and are analyzed for radioactivity. If the radioactivity is within the established discharge limit, the laundry wastes are pumped from the tanks through a cartridge filter into the WMT's. Should the radioactivity exceed the established discharge limits, the laundry wastes are processed.

All tanks are equipped with water level instrumentation alarms and their respective pumps are tripped automatically on low level signals.

All piping 2½ inches and smaller is field run. Line sizes are shown on Figure 11.2-4.

Design data for the major components are given in Table 11.2-8. Flow, temperature, and pressure are given in Table 11.2-9 with the locations corresponding to data points on Figure 11.2-4. Expected performance of components is given in Table 11.2-10.

The design criteria and controls for limiting radiation exposure of personnel from lines which normally carry radioactive fluids are discussed in Section 12.1.3 and 12.1.5.

As indicated in Table 11.2-1, an estimated 156,000 gallons per year of liquid wastes, exclusive of laundry wastes are processed in the liquid waste system. The activity of the aerated liquid wastes collected in the equipment drain tank and chemical drain tank is approximately 1 percent of the reactor coolant activity, owing to dilution from washdown and decontamination procedures.

Radioactivity removal is accomplished by the waste ion exchangers. NUREG-0017 states that the decontamination factors given in Table 11.2-10 can be expected across mixed bed ion exchangers for dirty liquid wastes. Experience with waste ion exchangers prove these DF's to be reasonable. Experience indicates that a concentrator DF (bottom to distillate) of 10^4 can be obtained. The concentrators are specified at this rating and testing by the vendor has confirmed the performance.

Radioactivity removal by the waste ion exchangers is shown in Table 11.2-10.

Table 11.2-13 lists the leakage sources and the assumptions used in calculating estimated normal and anticipated operational occurrence releases due to leakage sources. Additional plant liquid releases can result from turbine system leakage.

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

$$(1)\ \frac{dA_{sg}^{i}}{dt} = Q_i + F_c A_c^i - (\lambda^i + B)A_{sg}^i) - (F_{sg})(PF_I^i) A_{sg}^i$$

$$(2)\ \frac{dA_T^i}{dt} = (F_{sg})(PF_I^i) - (\lambda^i + L_T)\ A_T^i - F_T A_T^i$$

$$(3)\ \frac{dA^{i}_{c}}{dt} = F_{T}(1 - PF^{i}_{2})A^{i}_{T} - \lambda^{i}A^{i}_{c} - F_{c}A^{i}_{c}$$

where:

A^{i}_{sg}	=	the total activity of isotope i in the steam generator liquid at time t , μCi.
Q_i	=	the steam generator leak rate for isotope i, μCi/sec
F_c	=	condensate flow rate, condenser volume/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^{i}_{1}	=	steam generator partition factor for the i^{th} isotope.
L_T	=	the turbine system leakage rate, vol/sec
A^{i}_{T}	=	activity in turbine steam for isotope i at time t, μCi
F_T	=	turbine flow rate, vol/sec
PF^{i}_{2}	=	condenser partition factor for isotope i

The above equations were solved for the equilibrium values of A^{i} A^{i}_{T} and A^{i}_{sg} 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.

11.2.3 SYSTEM DESIGN

The liquid waste system is designed on a batch mode basis for flexibility of operation. These batching operations proceed intermittently at faster flow rates than the annual average process rates and therefore the system components are sized per this criteria. Tables 11.2-2 and 11.2-8 list the design parameters for the major components of the liquid waste system.

Process and radiation instrumentation are depicted in Figures 11.2-1, 11.2-3, and 11.2-4 and are described in Table 11.2-16.

11.2.4 OPERATING PROCEDURES

11.2.4.1 Boron Recovery System

Operating procedures for the boron recovery system are written to logically reflect the different monitoring and operating functions required of the plant operators. Procedures include the following.

a) Procedures for the initial valve lineup to the holdup tanks and to the flash tank when RCS activity is high.

b) Monitoring requirements of reactor drain tank operations and procedures for processing the tank contents directly to a holdup tank, bypassing the flash tank.

c) Monitoring and actions to be taken in the event of flash tank bypass operation to avoid explosive mixtures of hydrogen and air in the holdup tanks.

d) Monitoring requirements for the conditions of the four holdup tanks and processing requirements for chemical and radioactivity sample analysis.

e) Procedures and valve lineups for processing a holdup tank contents through a preconcentrator filter and ion exchanger back to the same or different tank and processing sampling requirements.

f) Monitoring requirements for the condensate tanks conditions and procedures for valve lineups. Sampling and reprocessing procedures.

g) Operating procedures for the environmental discharge instrumentation and valves. The method for determining the discharge flow rate in terms of radioactivity discharge limits.

11.2.4.2 Liquid Waste System

Operating procedures for the liquid waste system include the following:

a) procedures for equipment drain tank and aerated waste storage tank influent valve lineups, monitoring tank conditions sampling, and waste discharge.

b) procedures for chemical drain tank influent valve lineups, monitoring tank conditions, and sampling.

c) procedures for laundry tank influent valve lineups, monitoring tank conditions, sampling, and valve lineup for discharge.

d) procedure for waste ion exchanger system valve lineups, operation, monitoring and sampling.

e) procedures for monitoring requirements for waste condensate tank conditions, valve lineups, sampling, and reprocessing. Discharge is done under the same procedure used for the boron recovery system.

11.2.5 PERFORMANCE TESTS

The boron recovery and liquid waste systems were tested prior to initial power plant start up to verify satisfactory flow characteristics through the equipment, to demonstrate satisfactory performance of pumps and instruments, to check for leak tightness of piping and equipment, and to verify proper operation controls. All piping and components were checked to ensure that they are properly installed. All manual and automatic valves were operated and checked for functionability. All alarms were checked for operability and verification of locations. The concentrators were tested for operation before installation at the site and after installation to assure proper integration with the system. The boric acid and waste concentrators were shop tested prior to shipment to demonstrate compliance with performance objectives. During hot functional testing, the boron recovery system operation was integrated with the chemical and volume control system. The purpose of this test was to check the procedures and system components used for receiving and processing waste water. Boric acid transfer operations and waste liquid disposal procedures were tested.

During normal plant operation, periodic testing as described below, verifies that the system components are operating as designed.

Filters are monitored for differential pressure and radiation level on a regular basis. Ion exchangers are monitored for

radiation level on a regular basis. To ensure that the flash tank is performing adequately, a liquid sample from the influent and effluent can be obtained to verify the hydrogen gas stripping function of the tank. All liquid discharges to the environs are sampled for radioactivity before discharge. The discharge radiation monitor is calibrated on a regular basis to assure accuracy.

11.2.6 ESTIMATED RELEASES

The curies released from the boron recovery and liquid waste systems are determined by multiplying the nuclide concentrations in the boric acid condensate tanks as shown in Tables 11.2-6 and 11.2-7 by the waste processed through each system. Table 11.2-14A gives estimated discharge activity in Ci/yr/nuclide for one percent (1%) failed fuel from the boron recovery system, and the chemical drain tank, equipment drain tank, aerated waste storage tank and laundry drain tanks. The waste schedule is shown in Table 11.2-1. Technical Specifications and the Offsite Dose Calculation Manual (ODCM) discuss the limits and conditions for controlled plant releases.

11.2.7 RELEASE POINTS

The only liquid release point from the waste management system to the environs is via the waste monitor tank to the circulating water discharge. Prior to discharge the contents of a tank are sampled for chemical and radioactivity concentrations. If the contents of the tank are found acceptable in terms of environmental discharge limits, the tank contents are discharged. A radiation monitor is provided in the discharge line to verify that the fluid discharge is below the applicable radioactivity limits. In the event that the discharged activity is unacceptable, the discharge monitor automatically terminates discharge operations. The discharge is located on the P & I diagram Figures 11.2-3 and 11.2-4. The release point from the plant is shown on Figure 11.2-5.

PC/M 05225 (Reference 5) 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 detention 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 conconcentration 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, and sealing of the DSC, is accomplished to ensure that no surface contamination remains on the DSC while transported to the storage module (Reference 6). 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 the dilute radioactive material were to be release 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.

An average annual dilution factor of 1.73×10^5 is obtained in the circulating water discharge based on 513,000 gpm flow in the discharge canal and an average annual liquid effluent flow rate of 2.95 gpm from the boron recovery system and the liquid waste system. The discharge and intake pipes are separated by 2400 feet of ocean shoreline which result in a negligible recirculation of discharge water. Recirculation of effluent is discussed in Section 2.4.12.

11.2.9 ESTIMATED DOSES

The maximum dose to an individual and total dose to the general public resulting from plant liquid releases after waste concentrator processing are shown in Table 11.2-15. Doses to the whole body and internal organs were calculated with the largest organ doses occurring to the gastrointestinal tract. This information is maintained in here for historical purposes only, since the waste concentrator is no longer used.

None of the liquid releases from the plant result in doses via drinking water. The surrounding municipalities do not depend on the sea as a source of potable water. All local groundwater runoff is towards the ocean which further diminishes the possible contamination of drinking water.

The only applicable exposure pathway comes from the ingestion of foods grown in the seawater containing plant releases and in recreational activities in these areas. Due to the reconcentration of radionuclides along the shoreline, the largest dose results from fishing along the shore. The bioaccumulation factors used in the dose calculations are taken from Reference 1 and summarized in Table 11.2-13. Maximum permissible dose factors are taken from Reference 2 and the liquid release dose calculation models from Reference 3. Individual sea food consumption, population fish harvest and population density data are listed in Table 11.2-13. The basis for the seafood consumption rates listed in Table 11.2-13 is Department of Agriculture data on food consumption in the South (Reference 4). The average amount of seafood ingested in the South during 1965-1966 was 9.17 kg fish and 1.15 kg shellfish. The maximum amount of seafood ingested by any one income group occurred among rural farmers, where seafood ingestion was 15.5 kg fish and 5.9 kg shellfish. No isotope natural decay times following release were assumed in the calculations.

The annual dose that would result from regular ingestion of foods grown in contaminated water is calculated according to the following equation:

$$D_{f_j} = \frac{(MPD)}{(MPCw)_i} \frac{B_{ij} G_i}{2200} Cw_j$$

where:

D_{fj} = food dose due to radionuclide j, mrem

B_{ij} = biological accumulation factor for radionuclide j in food i

G_i = grams of food i ingested per day

2200 = grams of water ingested daily

Cw_j = concentration of radionuclide j in ingested water, µCi/ml

MPD = maximum permissible dose, mrem/yr

$(MPCW)_j$ = maximum permissible concentration of nuclide j, µCi/Ml

REFERENCES FOR SECTION 11.2

1. Frike, A. M., "A Model for the Approximate Calculation of Safe Rates of Discharge of Radioactive Wastes into Marine Environments", Health Physics 13:743 (1967).

2. U. S. Department of Commerce, "Maximum Permissible Body Burdens and Maximum Permissible Concentrations of Radionuclides in Air and in Water for Occupational Exposure," National Bureau of Standards Handbook 69.

3. Gamertsfelder, Carl C., "Statement on the Selection of As Low As Practicable Design Objectives and Technical Specifications for the Operation of Light Water Cooled Nuclear Power Plants," U. S. AEC.

4. U. S. Department of Agriculture, "Food Consumption of Households in the South," Spring 1965, Agricultural Research Series Issued by Government Printing Office, July 1968.

5. PC/M 05225, “Grading and Drainage Modification for the ISFSI Project."

6. Certificate of Compliance For Spent Fuel Storage Casks, Certificate No. 1030, Docket No. 72-1030, Effective Date 1/10/2007, including Appendix A to Certificate of Compliance No. 1030, “NUHOMS® HD System Generic Technical Specifications” Technical Specification 5.2.4 d.

TABLE 11.2-1
SOURCES AND VOLUMES OF LIQUID WASTE
(HISTORICAL)

1. Boron Recovery System

Source	Waste Generating Operation		Volume (gallons/year)
Chemical and volume control system	Boron reduction for fuel burnup		216,000
	Cold shutdown and startups	335,800	
	Hot shutdown and startups		161,000
	Refueling shutdown and startup		68,000
Reactor coolant leakage	200 gpd for four reactor coolant pumps		62,600
			843,400

2. Liquid Waste System

Source	Waste Generating Operation	Volume (gallons/year)
Equipment drains and leakage	75 gal/day	28,000
Sample and laboratory sink drains	20 gal/day	7,000
Equipment decontamination	10 gpm for 20 minutes per day	73,000
Floor drains	5 gpm for 10 minutes per day	18,000
Fuel cask washdown	400 gal/cask per refueling	30,000
Subtotal		156,000
Laundry	1000 gal/day	365,000
Showers	4 showers per day at 30 gal per shower	44,000
Resin dewatering	Sluice and dewater 256 ft^3 resin per year at 2 ft^3 water/ft^3 resin	3,800
Total	Estimated normal operation	568,000

TABLE 11.2-2
(HISTORICAL)
DESIGN DATA FOR BORON RECOVERY SYSTEM COMPONENTS [2]

1. Ion Exchangers	Preconcentrator	Boric Acid Condensate[1]
Quantity	2	1
Type	Deep Resin Bed	Deep Resin Bed
Design Pressure, psig	150	150
Design Temperature, F	250	250
Normal Operating Pressure, psig	60	60
Normal Operating Temperature, F	120	120
Resin Volume, ft^3	36	36
Materials	ss	ss
ASME Code, Section	VIII	VIII
Resin	Mixed Bed, Specialty Resin Overlay	Anion Bed (No Longer Used)

2. Tanks	Reactor Drain	Flash	Holdup	Boric Acid Condensate[1]	Boric Acid Holding[1]
Quantity	1	1	4	2	1
Internal Volume, gal	1560	424	40,000	6,700	2,400
Design Pressure, psig	25	70	10	Atmos	Atmos.
Design Temperature, F	250	250	240	250	200
Normal Operating Pressure, psig	0.5 – 6.0	0.5	0.5-5 psig	Atmos.	Atmos.
Normal Operating Temperature, F	120	120	120	120	150
Blanket Gas	Nitrogen	Nitrogen	Nitrogen		
Material	55	55	ss	ss	ss
ASME Code, Class or Division	III,C	III,C	VIII,I	None	None

3. Pumps	Flash Tank	Reactor Drain	Holdup Drain, Holdup Recir. Boric Acid Cond.[1]	Boric Acid Holding[1]
Quantity-Full Capacity	2	2	5	2
Type	Centrifugal	Centrifugal	Centrifugal	Centrifugal
Design Pressure, psig	150	150	150	150
Design Temperature, F	200	200	200	200

TABLE 11.2-2 (Cont'd)
(HISTORICAL)

3. Pumps	Flash Tank	Reactor Drain	Holdup Drain, Holdup Recir. Boric Acid Cond.[1]	Boric Acid Holding[1]
Design Conditions				
Flow, gpm	150	50	40	50
Head, ft	63	140	154	96
Wetted Materials	SS	SS	SS	SS
Seal Type	Mechanical	Mechanical	Mechanical	Mechanical
Motor Horsepower	5	7.5	7.5	5
Motor Voltage,. volt	460	460	460	460
ASME Pump and Valve Code, Section	None	None	None	None

4. Filters	Preconcentrator
Quantity	2
Type of Elements	Replaceable Cartridge
Retention of 2 micron particles, %	98
Design Pressure, psia	150
Design Temperature, F	240
Design Flow, gpm	100
Material	Stainless Steel
ASME Code, Section	III, Class 'C'

5. Boric Acid Concentrator[1]	
Quantity	2
Design Pressure, psig	80
Design Temperature, F	250
Design Flow, gpm	20
Cooling Water Flow Rate, gpm	650
Steam Required at 15 psig, lb/hr	13,000
ASME Code, Section	VIII

(1) These items are no longer used.
(2) Original procurement information

TABLE 11.2-3
(HISTORICAL)
BORON RECOVERY SYSTEM PROCESS FLOW DATA

Mode #1 [1]								
Location:	1	2	3	4	5	6	7	8
Flow, gpm	200 gpd	50	50	50	50	60	50	50
Pressure, psig	1	4	12	7.5	6.5	4	45	2
Temperature, F	120	120	120	120	120	120	120	120
Mode #2 [1]								
Location:	1	2	3	4	5	6	7	8
Flow, gpm	200 gpd	-	-	40	40	50	40	40
Pressure, psig	1	-	-	5	4.5	4	44	2
Temperature, F	120	-	-	120	120	120	120	120
Mode #3 [1]								
Location:	1	2	3	4	5	6	7	8
Flow, gpm	200 gpd	-	-	84	84	94	84	84
Pressure,psig	1	-	-	21	18.5	4	34	2
Temperature, F	120	-	-	120	120	120	120	120
Mode #4 [1]								
Location:	1	2	3	4	5	6	7	8
Flow, gpm	200 gpd	-	-	128	128	138	128	128
Pressure,psig	1	-	-	51	45.5	4	18	2
Temperature, F	120	-	-	120	120	120	120	120

(Continued)

TABLE 11.2-3 (Cont'd)
(HISTORICAL)

Mode #5 (1)										
Location:	12	13	14	15	16(3)	17(3)	18(3)	19(3)	20(3)	21(3)
Flow, gpm	20	20	20	20	0-10	16-19	16-19	50	50	50
Pressure,psig	3	65	62	60	5	5	2	4	60	10
Temperature, F	120	120	120	120	120	120	120	120	120	120
Mode #6 (1)										
Location:	9	10	11	15a	16a					
Flow, gpm	50	50	50	50	50					
Pressure,psig	4	32	27	24	20					
Temperature, F	120	120	120	120	120					
Mode #7										
Location:	22(3)	23(3)								
Flow, gpm	50	50								
Pressure,psig	4	40								
Temperature, F	150	150								

(1) The modes of operation are defined as follows:

Mode No.	Description
1	Processing PDT Contents
2	CVCS Normal purification VCT Diversion Processing
3	CVCS Intermediate Purification VCT Diversion Processing
4	CVCS Maximum Purification VCT Diversion Processing
5	Processing holdup tank contents via the boric acid Concentrator
6	Holdup Tank Contents Recirculation
7	Pumping BAHT Contents to the BAMT in the CVCS.

(2) The pressure drop across the filters and ion exchangers will vary with loading. The pressure drops are typical.

(3) These flow paths are no longer used, as a result of discontinuing the use of the boric acid concentrators.

TABLE 11.2-4
(HISTORICAL)
EXPECTED FILTER AND ION EXCHANGER PERFORMANCE

Nuclide	Chemical and Volume Control System		Boron Recovery Waste Processing System	
	Filter DF	Ion Exchanger DF	Filter DF	Ion Exchanger DF
Br-84	1	10^3	1	10^2
Rb-88	1	1	1	10^2
Rb-89	1	1	1	10^2
Sr-89	1	10^2	1	10
Sr-90	1	10^2	1	10
Y-90	1	1	1	10
Sr-91	1	10^2	1	10
Y-91	1	1	1	10
Mo-99	1	1	1	10
Ru-103	1	10	1	10
Ru-106	1	10	1	10
Te-129	1	10	1	10
I-129	1	10^3	1	10^3
I-131	1	10^3	1	10^3
Te-132	1	10	1	10
I-132	1	10^3	1	10^3
I-133	1	10^3	1	10^3
Te-134	1	10	1	10
I-134	1	10^3	1	10^3
Cs-134	1	1	1	10^2
I-135	1	10^3	1	10^3
Cs-136	1	1	1	10^2
Cs-137	1	1	1	10^2
Cs-138	1	1	1	10^2
Ba-140	1	10^2	1	10
La-140	1	10	1	10
Pr-140	1	10	1	10
Ce-144	1	10	1	10
Co-60	10	1	1	1
Fe-59	10	1	1	1
Co-58	10	1	1	1
Mn-54	10	1	1	1
Cr-51	10	1	1	1
Zr-95	10	1	1	1

TABLE 11.2-5
(HISTORICAL)
BORON RECOVERY SYSTEM PERFORMANCE DATA

Flash Tank DF for Fission Gases	2
Boric Acid Concentrator	
DF for Liquid (Influent to Distillate)	200
DF for Fission Gases	5
Holdup Tank Delay Factor, days	9
Annual System Condensate Discharged to CWD[(1)]	
Volume, gal.	843,000
Activity, Curies H-3	339
Dissolved Fission Product Gases	24,500
All Others	0.89

Note: (1) The volume and activity values obtained are based on 843,000 gals of waste to BMS with 1% failed fuel.

TABLE 11.2-6
(HISTORICAL)
BORON RECOVERY SYSTEM (BRS) MAXIMUM NUCLIDE CONCENTRATIONS (70F) DURING NORMAL OPERATIONS (µCi/cc)

Nuclide	CVCS 6 BRS 1	CVCS 7	CVCS 10 BRS 2,3,4,5	BRS 6, 7, 8	BRS 12,13,14	BRS 15	BRS 16	BRS 18,19,20,21
H-3	1.47(-1)*	1.47(-1)	1.47(-1)	1.47(-1)	1.47(-1)	1.47(-1)	1.47(-1)	1.47(-1)
Br-84	6.40(-3)	6.40(-2)	6.43(-6)	6.43(-6)	0.0	0.0	0.0	0.0
Kr-85m	2.05(-1)	2.05(-1)	2.05(-1)	1.03(-1)	0.0	0.0	0.0	0.0
Kr-85	1.22(-1)	1.22(-1)	1.22(-1)	6.11(-2)	6.10(-2)	6.10(-2)	0.0	1.22(-2)
Kr-87	1.12(-1)	1.12(-1)	1.12(-1)	5.59(-2)	0.0	0.0	0.0	0.0
Kr-88	3.59(-1)	3.59(-1)	3.59(-1)	1.79(-2)	0.0	0.0	0.0	0.0
Rb-88	3.52(-1)	3.52(-1)	3.52(-1)	3.52(-1)	0.0	0.0	0.0	0.0
Rb-89	8.80(-3)	8.80(-3)	8.80(-3)	8.80(-3)	0.0	0.0	0.0	0.0
Sr-89	6.99(-4)	6.99(-4)	6.99(-6)	6.99(-6)	6.18(-6)	6.18(-7)	3.09(-5)	3.09(-9)
SR-90	3.60(-5)	3.60(-5)	3.60(-7)	3.60(-7)	3.60(-7)	3.60(-8)	1.80(-6)	1.80(-10)
Y-90	1.41(-4)	1.41(-4)	1.41(-4)	1.41(-4)	1.36(-5)	1.36(-6)	6.83(-5)	6.83(-9)
Sr-91	4.91(-4)	4.91(-4)	4.91(-6)	4.91(-6)	0.0	0.0	0.0	0.0
Y-91	1.53(-2)	1.53(-2)	1.53(-2)	1.53(-2)	1.38(-2)	1.40(-3)	6.89(-2)	6.89(-6)
Mo-99	2.80(-1)	2.30(-1)	2.80(-1)	2.80(-1)	3.00(-2)	3.00(-3)	1.50(-1)	1.50(-5)
Ru-103	5.70(-4)	5.70(-4)	5.70(-5)	5.70(-5)	4.87(-5)	4.87(-6)	2.43(-4)	2.43(-8)
Ru-106	3.42(-5)	3.42(-5)	3.42(-6)	3.42(-6)	3.36(-6)	3.36(-7)	1.68(-1)	1.68(-9)
Te-129	3.50(-3)	3.50(-3)	3.46(-4)	3.46(-4)	0.0	0.0	0.0	0.0
I-129	9.95(-9)	9.95(-9)	0.0	0.0		0.0	0.0	0.0
I-131	5.48(-1)	5.48(-1)	5.48(-4)	5.48(-4)	2.52(-4)	2.52(-7)	1.26(-5)	1.26(-9)
Xe-131m	2.04(-1)	2.04(-1)	2.04(-1)	1.02(-1)	6.07(-2)	6.07(-2)	0.0	1.21(-2)
Te-132	4.55(-2)	4.55(-2)	4.60(-3)	4.60(-3)	6.92(-4)	6.92(-5)	3.5(-3)	3.46(-7)
I-132	1.50(-1)	1.50(-1)	1.50(-4)	1.50(-4)	0.0	0.0	0.0	0.0

* Numbers in () are powers of ten.

TABLE 11.2-6 (Cont'd)
(HISTORICAL)

Nuclide	CVCS 6 BRS 1	CVCS 7	CVCS 10 BRS 2,3,4,5	BRS 6, 7, 8	BRS 12,13,14	BRS 15	BRS 16	BRS 18,19,20,21
I-133	7.81(-1)*	7.81(-1)	7.81(-4)	7.81(-4)	6.25(-7)	6.26(-10)	3.13(-8)	0.0
Xe-133	2.49(+1)	2.49(+1)	2.49(+1)	1.25(+1)	3.82	3.82	0.0	7.64(-1)
Te-134	3.60(-3)	3.60(-3)	3.61(-4)	3.61(-4)	0.0	0.0	0.0	0.0
I-134	8.56(-2)	8.56(-2)	8.55(-5)	8.55(-5)	0.0	0.0	0.0	0.0
Cs-134	1.38(-2)	1.38(-2)	1.38(-2)	1.38(-2)	1.37(-2)	1.37(-4)	6.80(-3)	6.84(-7)
I-135	3.72(-1)	3.72(-1)	3.72(-4)	3.72(-4)	7.35(-14)	0.0	0.0	0.0
Xe-135	1.04	1.04	1.04	5.19(-1)	0.0	0.0	0.0	0.0
Cs-136	3.50(-3)	3.50(-3)	3.50(-3)	3.50(-3)	2.20(-3)	2.18(-5)	1.10(-3)	1.09(-7)
Cs-137	4.42(-2)	4.42(-2)	4.42(-2)	4.42(-2)	4.41(-2)	4.41(-4)	2.21(-2)	2.21(-6)
Xe-138	4.97(-2)	4.97(-2)	4.97(-2)	2.48(-2)	0.0	0.0	0.0	0.0
Cs-138	9.52(-2)	9.52(-2)	9.52(-2)	9.52(-2)	0.0	0.0	0.0	0.0
Ba-140	8.43(-4)	8.43(-4)	8.43(-6)	8.43(-6)	5.18(-6)	5.18(-7)	2.59(-5)	2.59(-9)
La-140	8.07(-4)	8.07(-4)	8.07(-5)	8.07(-5)	1.95(-6)	1.95(-7)	9.74(-6)	9.74(-10)
Pr-143	8.06(-4)	8.06(-4)	8.06(-5)	8.06(-5)	5.11(-5)	5.11(-6)	2.55(-4)	2.55(-8)
Ce-144	5.70(-4)	5.70(-4)	5.70(-5)	5.70(-5)	5.57(-5)	5.57(-6)	2.79(-4)	2.79(-8)
Co-60	7.16(-4)	7.16(-5)	7.16(-5)	7.16(-5)	7.14(-5)	7.14(-5)	3.60(-3)	3.57(-7)
Fe-59	2.94(-5)	2.94(-6)	2.94(-6)	2.94(-6)	2.56(-6)	2.56(-6)	1.28(-4)	1.28(-8)
Co-58	6.40(-3)	6.43(-4)	6.43(-4)	6.43(-4)	5.89(-4)	5.89(-4)	2.94(-2)	2.94(-6)
Mn-54	3.79(-5)	3.79(-6)	3.79(-6)	3.79(-6)	3.72(-6)	3.72(-6)	1.86(-4)	1.86(-8)
Cr-51	5.20(-3)	5.24(-4)	5.24(-4)	5.24(-4)	4.19(-4)	4.19(-4)	2.10(-2)	2.09(-6)
Zr-95	1.29(-6)	1.29(-7)	1.29(-7)	1.29(-7)	1.17(-7)	1.17(-7)	5.86(-6)	5.86(-10)

* Numbers in () are powers of Ten

TABLE 11.2-7
(HISTORICAL)

BORON RECOVERY SYSTEM (BRS) MAXIMUM NUCLIDE CONCENTRATIONS (70°F) DURING ANTICIPATED OPERATIONAL OCCURRENCES (µCi/cc)

Nuclide	CVCS 6 1	CVCS 7	CVCS 10 2, 3, 4, 5	6, 7, 8	12, 13, 14	15	16	18, 19, 20, 21
H-3	1.82(-1)*	1.82(-1)	1.82(-1)	1.82(-1)	1.82(-1)	1.82(-1)	1.82(-1)	1.82(-1)
Br-84	6.43(-1)	6.43(-1)	6.43(-5)	6.43(-5)	0.0	0.0	0.0	0.0
Kr-85m	2.06	2.06	2.06	1.03	0.0	0.0	0.0	0.0
Kr-85	1.22	1.22	1.22	6.10(-1)	6.09(-1)	6.09(-1)	0.0	1.22(-1)
Kr-87	1.12	1.12	1.12	5.59(-1)	0.0	0.0	0.0	0.0
Kr-88	3.59	3.59	3.59	1.79	0.0	0.0	0.0	0.0
Rb-88	3.52	3.52	3.52	3.52	0.0	0.0	9.0	0.0
Rb-89	8.83(-2)	8.83(-2)	8.83(-2)	8.83(-2)	0.0	0.0	0.0	0.0
Sr-89	7.00(-3)	7.00(-3)	6.99(-5)	6.99(-5)	6.18(-5)	6.18(-6)	3.09(-4)	3.09(-8)
Sr-90	3.60(-4)	3.60(-4)	3.60(-6)	3.60(-6)	3.60(-6)	3.60(-7)	1.80(-5)	1.80(-9)
Y-90	1.40(-3)	1.40(-3)	1.40(-3)	1.40(-3)	1.36(-4)	1.36(-5)	6.83(-4)	6.83(-8)
Sr-91	4.90(-3)	4.90(-3)	4.91(-5)	4.91(-5)	0.0	0.0	0.0	0.0
Y-91	1.53(-1)	1.53(-1)	1.53(-1)	1.53(-1)	1.38(-1)	1.38(-1)	6.89(-1)	6.89(-5)
Mo-99	2.80	2.80	2.80	2.80	2.99(-1)	3.00(-2)	1.50	1.50(-4)
Ru-103	5.70(-3)	5.70(-3)	5.69(-4)	5.69(-4)	4.87(-4)	4.87(-5)	2.40(-3)	2.44(-7)
Ru-106	3.42(-4)	3.42(-4)	3.42(-5)	3.42(-5)	3.36(-5)	3.36(-6)	1.63(-4)	1.68(-8)
Te-129	3.46(-2)	3.46(-2)	3.50(-3)	3.50(-3)	0.0	0.0	0.0	0.0
I-129	9.95(-8)	9.95(-8)	0.0	0.0	0.0	0.0	0.0	0.0
I-131	5.48	5.48	5.50(-3)	5.50(-3)	2.50(-3)	2.52(-6)	1.26(-4)	1.26(-8)
Xe-131m	2.04	2.04	2.04	1.02	6.07(-1)	6.07(-1)	0	1.21(-1)
Te-132	4.55(-1)	4.55(-1)	4.55(-2)	4.55(-2)	6.90(-3)	6.92(-4)	3.46(-2)	3.46(-6)
I-132	1.50	1.50	1.50(-3)	1.50(-2)	0.0	0.0	0.0	0.0
I-133	7.81	7.81	7.80(-3)	7.80(-3)	6.26(-6)	6.26(-9)	3.13(-7)	0.0
Xe-133	2.49(+2)	2.49(+2)	2.49(+2)	1.25(+2)	3.82(+1)	3.82(+1)	0.0	7.64

* Numbers in () are powers of Ten

TABLE 11.2-7 (Cont'd)
(HISTORICAL)

Nuclide	CVCS 6 1	CVCS 7	CVCS 10 2, 3 4, 5	6, 7, 8	12, 13, 14	15	16	18, 19, 20, 21
Te-134	3.62(-2)*	3.62(-2)	3.60(-3)	3.60(-3)	0.0	0.0	0.0	0.0
I-134	8.55(-1)	8.55(-1)	8.55(-4)	8.55(-4)	0.0	0.0	0.0	0.0
Cs-134	1.38(-1)	1.38(-1)	1.38(-1)	1.38(-1)	1.37(-1)	1.40(-3)	6.84(-2)	6.84(-6)
I-135	3.73	3.73	3.70(-3)	3.70(-3)	0.0	0.0	0.0	0.0
Xe-135	10.39	10.39	10.39	5.19	8.18(-10)	8.18(-10)	0.0	1.64(-10)
Cs-136	3.52(-2)	3.52(-2)	3.52(-2)	3.52(-2)	2.18(-2)	2.18(-4)	1.09(-2)	1.09(-6)
Cs-137	4.42(-1)	4.42(-1)	4.42(-1)	4.42(-1)	4.41(-1)	4.40(-3)	2.20(-1)	2.21(-5)
Xe-138	4.97(-1)	4.97(-1)	4.97(-1)	2.48(-1)	0.0	0.0	0.0	0.0
Cs-138	9.52(-1)	9.52(-1)	9.52(-1)	9.52(-1)	0.0	0.0	0.0	0.0
Ba-140	8.40(-3)	8.40(-3)	8.43(-5)	8.43(-5)	5.18(-5)	5.18(-6)	2.59(-4)	2.59(-8)
La-140	8.10(-3)	8.10(-3)	8.07(-4)	8.07(-4)	1.95(-5)	1.95(-6)	9.74(-5)	9.74(-9)
Pr-143	8.10(-3)	8.10(-3)	8.06(-4)	8.06(-4)	5.11(-4)	5.11(-5)	2.60(-3)	2.55(-7)
Ce-144	5.70(-3)	5.70(-3)	5.70(-4)	5.70(-4)	5.70(-4)	5.57(-5)	2.80(-3)	2.79(-7)
Co-60	7.16(-4)	7.16(-5)	7.16(-5)	7.16(-5)	7.14(-5)	7.14(-5)	3.60(-3)	3.57(-7)
Fe-59	2.94(-5)	2.94(-6)	2.94(-6)	2.94(-6)	2.56(-6)	2.56(-6)	1.28(-4)	1.28(-8)
Co-58	6.40(-3)	6.43(-4)	6.43(-4)	6.43(-4)	5.89(-4)	5.89(-4)	2.94(-2)	2.94(-6)
Mn-54	3.79(-5)	3.79(-5)	3.79(-6)	3.79(-0)	3.72(-6)	3.72(-6)	1.86(-4)	1.86(-8)
Cr-51	5.20(-3)	5.24(-4)	5.24(-4)	5.24(-4)	4.19(-4)	4.19(-4)	2.10(-2)	2.09(-6)
Zr-95	1.29(-6)	1.29(-7)	1.29(-7)	1.29(-7)	1.17(-7)	1.17(-7)	5.86(-6)	5.86(-10)

*Numbers in () are powers of ten.

TABLE 11.2-8
(HISTORICAL)

DESIGN DATA FOR LIQUID WASTE SYSTEM COMPONENTS

1.	Ion Exchanger	Waste (A&D)	Waste (B&C)
	Quantity	2	2
	Type Resin	Mixed Bed	Mixed Bed
	Design Pressure, psig	150	150
	Design Temperature, F	250	200
	Normal Operating Pressure, psig	60	120
	Normal Operating Temperature, F	120	120
	Resin Volume, ft^3	36	50
	Materials	SS	SS
	ASME Code	VIII	VIII

2.	Tanks	Equipment Drain1A	Aerated Waste Storage	Chemical Drain	Waste Monitor	Laundry Drain	Waste Condensate[1]
	Quantity	1	1	1	2	2	2
	Internal Volume, gal.	1040	40,000	1040	40,000	1,040	1600
	Design Pressure, psig	Atmos	Atmos	Atmos	0.2	Atmos	Atmos
	Design Temperature, F	200	200	200	140	200	250
	Normal Operating Pressure, psig	Atmos	Atmos	Atmos	Atmos	Atmos	Atmos
	Normal Operating Temperature, F	120	120	120	120	120	120
	Material	SS	SS	SS	CS	SS	SS
	ASME Code, Division	VIII, I	AWWA	VIII,I	AWWA D 100	VIII,I	N.A.

3.	Pumps	Equipment Drain, 1A Chem.Drain,Wst.Condensate	Laundry Drain, Equip. Drn. 1B,1C	Waste Monitor
	Quantity	4	4	2
	Type	Centrifugal	Centrifugal	Centrifugal
	Design Pressure, psig	150	150	150
	Design Temperature, F	200	165	165
	Design Conditions			
	Flow, gpm	50	65	170
	Head, ft.	140	302	34
	Wetted Materials	SS	SS	SS
	Seal Type	Mechanical	Mechanical	Mechanical
	Motor horsepower	7.5	20	3
	Motor Voltage/Phase/Hz	460/3/60	460/3/60	460/3/60
	Code	N.A.	N.A.	N.A.

TABLE 11.2-8 (Cont'd)
(HISTORICAL)

	Waste	Laundry
4. Filters		
Quantity	2	2
Type of Elements	Replaceable Cartridge	Replaceable Cartridge
Particle Retention	25 micron, absolute	150 micron, nominal
Design Pressure, psig	150	150
Design Temperature, F	200	200
Design Flow, gpm	50	50
Material	Stainless Steel	Stainless Steel
ASME Code	Section III, Class 'C'	Section III, Class 'C'
5. Waste Concentrator[(1)]		
Quantity	1	
Design Pressure, psig	80	
Design Temperature, F	250	
Design Flow, gpm	2	
Cooling Water Flow Rate, gpm	130	
Steam Required at 15 psig, lb/hr	1300	
ASME Code	VIII	

(1) This component is no longer used.

TABLE 11.2-9
(HISTORICAL)
LIQUID WASTE SYSTEM

PRESSURE, TEMPERATURE AND FLOW DATA
(Reference Figure 11.2-4)

Mode 1 Processing equipment drain tank contents via two waste ion exchangers.

Location	1	2	3	4	5	6	7	8	9
Pressure, psig	0.5	0.5	2.0	61	0.5	2.0	130	0.5	2.0
Temperature, F	120	120	120	120	120	120	120	120	120
Flow, gpm	0.825	3.5	27.5	50	.25	27.5	50	0.75	0
Location	10	11	12	13	14	15	16		22
Pressure,psig	0	0	118	66	20	0	0		20
Temperature, F	120	120	120	120	120	120	120		120
Flow, gpm	0	0	50	50	50	0	0		105

TABLE 11.2-9
(HISTORICAL)

Mode 2 — Processing laundry waste through laundry filter, bypassing carbon filter/ion exchanger ID.

Location	1	2	3	4	5	6	7	8	9
Pressure, psig	0.5	0.5	2.0	0	0.5	2.0	0	0.5	2.0
Temperature, F	120	120	120	120	120	120	120	120	120
Flow, gpm	0.825	3.5	0	0	0.25	0	0	0.75	65

Location	10	11	12	13	14	15	16		22
Pressure, psig	130	115	0	0	0	0	0		20
Temperature, F	120	120	120	120	120	120	120		120
Flow, gpm	50	50	0	0	0	0	0		0

TABLE 11.2-10
(HISTORICAL)
LIQUID WASTE SYSTEM EXPECTED PERFORMANCE

	1st exchanger	2nd exchanger	Total
Ion Exchanger Decontamination Factor			
Tritium	1	1	1
Anion (Iodine)	100	10	1000
Cesium, Rubidium	2	10	20
Others	100	10	1000
Filter Decontamination Factor			
All Nuclides	1		
Waste Concentrator Decontamination Factor		500	
Holdup Time for Liquid Waste System, days			
Anticipated	1		
Normal	4.5		

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TABLE 11.2-13

ASSUMPTION'S USED IN CALCULATING ESTIMATED NORMAL AND ANTICIPATED OPERATIONAL OCCURRENCE RELEASES[1]

(Information presented in this table is historical)

	Estimated Normal Releases	Anticipated Operational Occurrence
1. Steam Generator Blowdown Releases		
Main Steam flow rate, lb/hr	11,206,000	11,206,000[2]
Liquid inventory per steam generator, lb	130,500	130,500
Blowdown rate, gal/day	200(26,567)	1,200(26,567)
Fuel failure, %	0.1	1.0(0.25)
Iodine partition coefficient	5×10^{-2} (10^{-2})	5×10^{-2} (10^{-2})
2. Releases to Reactor Auxiliary Building[3]		
Fuel failure, %	0.1	1.0(0.25)
Leakage to reactor auxiliary building, gal/day	10(19.2)	10(19.2)
Iodine partition coefficient	5×10^{-3}	5×10^{-3}
3. Secondary System Releases		
Steam generator tube leakage rate, gal/day	20 (18.1)	120(18.1)
Failed fuel, %	0.1	1.0(0.25)
Condenser iodine partition coefficient	5×10^{-4}	5×10^{-4}
Flow rate into condenser, lb/hr	7,840,000	7,840,000
Steam weight in main steam piping from steam generator to turbine stop valves (both loops), lb	7940	7940
Condenser Steam Space[4] lb	327	327
Turbine Steam[4], lb	87,000	87,000

(1) Regulatory Guide 1.42 assumptions utilized in the gaseous release analysis that differ from normally applied values are shown in parenthesis.

(2) Releases for the anticipated operational occurrence are assumed processed through the waste management system.

(3) Assumed reactor coolant activities corrected to 120F.

(4) Including hotwell

Table 11.2-13 (Cont'd)

Steam Generator Steam Space, [4]lb	9,500	9,500
Total Steam Space, lb	104,767	104,767
Condenser hotwell water weight, lb	680,000	680,000
Liquid inventory between hotwell and steam generator, lb	480,000	480,000
Steam leakage to turbine building, lb/hr	20 (1700)	20 (1700)

4. Containment Purge System

Reactor coolant system leak rate, lb/hr	50 (10)
Airborne activity cleanup flow rate	20,000 cfm = .48 volume/hr.
HEPA Filter Efficiency, %	99.9
Charcoal Filter Efficiency, %	
Organic Iodine	70
Inorganic Iodine	90
Operating time of removal system after shutdown, hr	10
Purge initiation after shutdown, hr	10
Purge flow rate	20,000 cfm = .48 volume/hr.
Purge frequency, times/yr.	4
Purge filter efficiencies, %	0
Iodine partition coefficient	0.1

5. Liquid Waste System Releases

Miscellaneous waste, gal	156,000
Laundry and showers	117,000

(4) Including piping

TABLE 11.2-13 (Cont'd.)

6) Decay Tanks

	Holdup Time days	Tank Inventory per cent
Reactor drain tank	3	50
Holdup tank	9	50
Boric acid condensate tank	3	50

7) Calculation Assumptions for Seafood Ingestion Dose

A. Individual Dose

1) Consumption

a) fish, kg/yr	18
b) crustacea, kg/yr	9
c) mollusk, kg/yr	9

2) Discharge canal flow, gpm 513,000

3) Reconcentration factors

Radionuclide	Fish	Crustacea	Mollusks
Cr-51	100	1,000	1,000
Mn-54	3,000	10,000	50,000
Fe-55	1,000	4,000	20,000
Fe-59	1,000	4,000	20,000
Co-58	100	10,000	300
Co-60	100	10,000	300
Rb-86	30	50	10
Sr-89	1	1	1
Sr-90	1	1	1
Y-90	30	100	100
Y-91	30	100	100
Zr-95	30	100	100
Zr-97	30	100	100
Nb-95	100	200	200
Mo-99	10	100	100
Ru-103	3	100	100
Ru-106	3	100	100
Rh-105	10	100	100
Sb-127	1,000	1,000	1,000
Te-127m	10	10	100
Te-129m	10	10	100
Te-131m	10	10	100
Te-132	10	10	100
I-131	20	100	100

TABLE 11.2-13 (Cont'd)

Radionuclide	Fish	Crustacea	Mollusks
I-132	20	100	100
I-133	20	100	100
I-135	20	100	100
Cs-136	30	50	10
Cs-137	30	50	10
Ba-140	3	3	3
Ce-141	30	100	100
Ce-143	30	100	100
Ce-144	30	100	100
Pr-143	100	1,000	1,000
Nd-147	100	1,000	1,000
Pm-147	100	1,000	1,000
Pm-149	100	1,000	1,000
Sm-153	100	1,000	1,000
H-3	1	1	1

B. Population Dose

1) Consumption Seafood Harvest in St. Lucie County

	Total	Contaminated Fish Consumed
a) fin fish, kg/yr	1.2×10^6	1.2×10^5
b) mollusk, kg/yr	1.0×10^5	1.0×10^4
c) crustacea, kg/yr	4.7×10^3	4.7×10^2

2) Seafood Harvested from Contaminated Waters, % 10

3) Ocean Dilution Factor 20

8. Population Density Data (1970)

Population within 5.0 mi. 1.165×10^3

Annular Population,

5 - 10 mi.	4.53×10^4
10 - 20 mi.	2.45×10^4
20 - 30 mi.	4.79×10^4
30 - 40 mi.	5.59×10^4
40 - 50 mi.	1.26×10^5
0 - 50 mi.	3.01×10^5

9. Recreational Doses

A. Swimming

1) Length of exposure, hr/yr 500

2) Shielding from beta's None, complete submersion

TABLE 11.2-13 (Cont'd)

B.	Boating		
	1)	Length of exposure, hr/yr	500
	2)	Source geometry	Semi-infinite geometry
	3)	Shielding from beta's	Boat assumed to shield all beta's
C.	Fishing		
	1)	Length of exposure	100
	2)	Shielding from beta's	No beta dose due to shielding of air between water and fisherman

TABLE 11.2-14

EXPECTED LIQUID RELEASES FROM THE LIQUID WASTE SYSTEM

(Information presented in this table is historical)

	Curies Released per Year		Activity in Circulating Water Discharge (μCi/cc)	
Nuclide	Normal Operation (0.1% Failed Fuel)	Anticipated Operation (1.0% Failed Fuel)	Normal Operation (0.1% Failed Fuel)	Anticipated Operation (1.0% Failed Fuel)
H-3	2.26 (2)*	3.39 (2)	2.21 (-7)	3.32 (-7)
Br-84	0.0	0.0	0.0	0.0
Kr-85m	1.09 (-13)	1.09 (-12)	0.0	0.0
Kr-85	3.89 (+1)	3.89 (+2)	3.81 (-8)	3.81 (-7)
Kr-87	0.0	0.0	0.0	0.0
Kr-88	0.0	0.0	0.0	0.0
Rb-88	0.0	0.0	0.0	0.0
Rb-89	0.0	0.0	0.0	0.0
Sr-89	1.41(-5)	2.71(-4)	1.38(-14)	2.66(-13)
Sr-90	1.29(-6)	1.80(-5)	1.26(-15)	1.76(-14)
Y-90	2.18(-5)	2.22(-4)	2.13(-14)	2.18(-13)
Sr-91	1.62(-12)	3 02(-7)	0.0	
Y-91	2.21(-2)	2.24(-1)	2.16(-11)	2.19(-10)
Mo-99	4.79(-2)	4.88(-1)	4.69(-11)	4.78(-10)
Ru-103	8.06(-5)	9.09(-4)	7.90(-14)	8.90(-13)
Ru-106	5.89(-6)	6.47(-5)	5.77(-15)	6.34(-14)
Te-129	0.0	5.04(-18)	0.0	
I-129	1.99(-10)	3.39(-9)	1.95(-19)	3.32(-18)
I-131	5.14(-4)	5.22(-2)	5.04(-13)	5.11(-111)
Xe-131m	3.87(+1)	3.87(+2)	3.79(-8)	3.79(-7)
Te-132	1.10(-3)	1.29(-2)	1.09(-12)	1.26(-11)
I-132	0.0	1.06(-9)	0.0	1.04(-18)
I-133	5.39(-7)	4.30(-3)	5.28(-16)	4.23(-12)
Xe-133	2.44(+3)	2.44(+4)	2.39(-6)	2.39(-5)
Te-134	0.0	0.0	0.0	0.0
I-134	0.0	4.68(-20)	0.0	
Cs-134	2.40(-1)	2.64(-2)	2.37(-12)	2.59(-11)
I-135	5.57(-14)	4.36(-5)	0.0	
Xe-135	5.22(-8)	5.22(-7)	5.11(-17)	5.11(-16)
Cs-136	3.53(-4)	3.90(-3)	1-46(-11)	3.86(-12)
Cs-137	7.90(-3)	8.54(-2)	7.75(-12)	8.37(-11)

*Numbers in () are powers of ten

TABLE 11.2-14 (Cont'd)

Nuclide	Curies Released per Year: Normal Operation (0.1% Failed Fuel)	Curies Released per Year: Anticipated Operation (1.0% Failed Fuel)	Activity in Circulating Water Discharge (μCi/cc): Normal Operation (0.1% Failed Fuel)	Activity in Circulating Water Discharge (μCi/cc): Anticipated Operation (1.0% Failed Fuel)
Xe-138	0.0	0.0	0.0	0.0
Cs-138	0.0	0.0	0.0	0.0
Ba-140	9.67 (-6)	1.95 (-4)	9.47 (-15)	1.91 (-13)
La-140	3.12 (-6)	4.51 (-5)	3.06 (-15)	4.41 (-14)
Pr-143	8.30 (-5)	9.27 (-4)	8.13 (-14)	9.08 (-13)
Ce-144	9.71 (-5)	1.10 (-3)	9.51 (-14)	1.05 (-12)
Co-60	1.30 (-3)	1.40 (-3)	1.25 (-12)	1.35 (-12)
Fe-59	4.24 (-5)	4.78 (-5)	4.15 (-14)	4.68 (-14)
Co-58	9.90 (-3)	1.11 (-2)	9.68 (-12)	1.09 (-11)
Mn-54	6.49(-5)	7.15(-5)	6.36(-14)	7.01(-14)
Cr-51	6.90(-3)	7.70(-3)	6.73(-12)	7.58(-12)
Zr-95	1.96(-6)	2.21(-6)	1.92(-15)	2.16(-15)
Total: H-3	2.26(+2)	3.39(+2)	2.21(-7)	3.32(-7)
Total: Noble Gas	2.51(+3)	2.51(+4)	2.47(-6)	2.46(-5)
Total: All Others	1.01 (-1)	9.22(-1)	9.87 (-11)	9.02 (-10)

*Numbers in () are powers of ten.

TABLE 11.2-14 (Cont'd)

EXPECTED LIQUID RELEASES FROM THE LIQUID WASTE SYSTEM

Turbine Building Floor Drainage

Nuclide	Curies Released Per Year Normal Operating (0.1% Failed Fuel)	 Anticipated Operation (1.0% Failed Fuel)
H-3	1.29 (-1)	5.47 (-1)
Br-84	3.18 (-7)	2.01 (-5)
Rb-88	9.93 (-6)	5.97 (-4)
Rb-89	2.18 (-7)	1.31 (-5)
Sr-89	4.32 (-5)	1.32 (-3)
Sr-90	1.84 (-6)	6.08 (-5)
Y-90	1.99 (-11)	1.20 (-9)
Sr-91	2.51 (-6)	1.40 (-4)
Y-91	6.50 (-4)	2.38 (-2)
Mo-99	1.49 (-3)	8.30 (-2)
Ru-103	2.03 (-5)	7.93 (-4)
Ru-106	2.07 (-6)	6.36 (-5)
Te-129	1.17 (-4)	4.60 (-3)
I-129	6.49 (-10)	1.93 (-8)
I-131	7.27 (-3)	3.62 (-1)
Te-132	2.80 (-4)	1.52 (-2)
I-132	8.60 (-4)	4.72 (-2)
I-133	1.37 (-3)	8.04 (-2)
Te-134	2.51 (-7)	1.51 (-5)
I-134	6.67 (-6)	4.03 (-4)
Cs-134	8.65 (-4)	2.62 (-2)
I-135	2.15 (-4)	1.28 (-2)
Cs-136	6.81 (-5)	3.43 (-3)
Cs-137	2.87 (-3)	8.55 (-2)
Cs-138	4.76 (-6)	2.86 (-4)
Ba-140	1.64 (-5)	7.53 (-4)
La-140	2.67 (-6)	1.52 (-4)
Pr-143	1.58 (-5)	7.28 (-4)
Ce-144	3.38 (-5)	1.05 (-3)
Co-60	4.59 (-5)	1.38 (-4)
Fe-59	1.15 (-6)	4.29 (-6)
Co-58	2.95 (-4)	1.04 (-3)
Mn-54	2.24 (-6)	7.02 (-6)
Cr-51	1.60 (-4)	6.51 (-4)
Zr-95	2.02 (-11)	1.22 (-10)

TABLE 11.2-14A
ESTIMATED RELEASES (Ci/Yr) FROM THE
LIQUID PROCESS STREAMS

Nuclide	Laundry Drain Tank	Boron Recovery System	Chemical Drain Tank	Equipment Drain Tank	Total
H3	5.5(-2)	2.3(2)*	8.6(1)	2.14(1)	339
BR84	0.0	0.0	0.0	0.0	0.0
KR85M	0.0	1.1(-12)	0.0	0.0	1.1(-12)
KR85	0.0	3.89(2)	0.0	0.0	3.89(2)
KR87	0.0	0.0	0.0	0.0	0.0
KR88	0.0	0.0	0.0	0.0	0.0
RB89	0.0	0.0	0.0	0.0	0.0
SR89	2.39(-5)	9.6(-5)	1.7(-4)	2.1(-5)	2.7(-4)
SR90	2.59(-8)	5.7(-6)	1.2(-5)	2.7(-7)	1.8(-5)
Y90	1.99(-7)	2.2(-4)	4.5(-6)	1.6(-6)	2.2(-4)
SR91	2.44(-8)	0.0	3.1(-7)	3.6(-8)	3.0(-7)
Y91	6.11(-4)	2.1(-1)	4.0(-3)	2.0(-1)	2.3(-1)
MO99	4.17(-4)	4.4(-1)	9.6(-3)	2.9(-2)	4.9(-1)
RU103	1.53(-5)	7.6(-4)	1.3(-4)	2.2(-6)	9.1(-4)
RU106	8.53(-6)	5.4(-5)	1.1(-5)	7.9(-6)	6.5(-5)
TE129	3.74(-14)	0	0	0	0
I129	3.40(-11)	0	0	0	3.4(-9)
I131	2.77(-3)	3.8(-5)	5.1(-2)	2.0(-3)	5.2(-2)
XE131M	0.0	3.8(2)	1.0(1)	0.0	3.9(2)
TE132	8.37(-5)	1.1(-2)	1.9(-3)	1.2(-4)	1.3(-2)
I132	7.12(-9)	0.0	6.0(-9)	0.0	1.1(-9)
I133	2.12(-4)	0.0	4.2(-3)	1.5(-4)	4.3(-3)
XE133	0.0	2.4(4)	0.0	0.0	2.4(4)
TE134	0.0	0.0	0.0	0.0	0.0
I134	0.0	0.0	0.0	0.0	0.0
CS134	7.09(-3)	2.2(-2)	4.7(-3)	7.4(-3)	2.6(-2)
I135	5.94(-6)		4.7(-5)	5.8(-5)	4.4(-5)
XE135	0.0	5.1(-7)	1.0(-8)	0.0	5.2(-7)
CS136	2.97(-5)	3.5(-3)	4.5(-4)	8.4(-5)	3.9(-3)
CS137	3.31(-1)	7.0(-2)	1.5(-2)	3.3(-1)	8.5(-2)
XE138	0.0	0.0	0.0	0.0	0.0

TABLE 11.2-14A (Cont'd)

Nuclide	Laundry Drain Tank	Boron Recovery System	Chemical Drain Tank	Equipment Drain Tank	Total
CS138	0.0	0.0	0.0	0.0	0.0
BA140	6.99(-6)	8.3(-5)	1.1(-4)	1.4(-6)	2.0(-4)
LA140	6.11(-7)	3.1(-5)	1.4(-5)	6.0(-7)	4.5(-5)
PR143	7.18(-6)	7.9(-4)	1.1(-4)	1.3(-6)	9.3(-4)
CE144	1.11(-4)	8.9(-4)	1.8(-4)	8.6(-5)	1.1(-3)
CO60	9.37(-5)	1.1(-3)	2.5(-4)	8.8(-5)	1.4(-3)
FE59	8.91(-8)	4.1(-5)	6.9(-6)	4.7(-7)	4.8(-5)
CO58	3.09(-5)	9.3(-3)	1.7(-3)	1.0(-4)	1.1(-2)
MN54	8.08(-7)	6.1(-5)	1.2(-5)	1.8(-6)	7.2(-5)
CR51	9.73(-6)	6.7(-3)	1.0(-3)	5.7(-5)	7.7(-3)
ZR95	5.68(-9)	1.9(-6)	3.3(-7)	2.2(-8)	2.2(-6)

*Numbers in ( ) are powers of ten

NOTE: The total annual releases are in Table 11.2-14

TABLE 11.2-15

OFFSITE DOSES DUE TO PLANT

LIQUID RELEASES

(Information presented in this table is historical)

	Normal Operation	Anticipated Operational Occurrence
Whole body dose due to ingestion of seafood:		
a) Maximum to individual	0.000165 mrem/yr	0.00354 mrem/yr
b) Total to population	0.298 man-rem/yr	2.19 man rem/yr
Gastrointestinal tract dose due to ingestion of seafood:		
a) Maximum to individual	0.00123 mrem/yr	0.00269 mrem/yr
b) Total to population	0.616 man-rem/yr	4.079 man-rem/yr
Exposure to individuals from recreational uses of the ocean in the vicinity of Hutchinson Island		
a) Swimming		
1) Gamma	0.49×10^{-4} mrem/yr	0.49×10^{-3}mrem/yr
2) Beta	0.46×10^{-4} mrem/yr	0.46×10^{-3}mrem/yr
b) Boating		
1) Gamma	0.25×10^{-4} mrem/yr	0.25×10^{-3}mrem/yr
2) Beta	0	0
c) Fishing		
1) Gamma	0.16×10^{-2}mrem/yr	0.16×10^{-1}mrem/yr
2) Beta	0	0

TABLE 11.2-16

WASTE MANAGEMENT SYSTEM INSTRUMENTATION APPLICATION
(Reference Figures 11.2-1, 3, 4, 4A, 4B, 4C & 11.3-1)

System Parameter and Location	Indication		Alarm			Control Function	Inst. Range[4]	Normal Operating Range	Instr. Accuracy[4]
	Local	Contr Room	High	Low	Rec[1]				
Boric Acid Holding Tank[5] Temperature	*		*[2]	*[2]		Boric Acid Holding Tank Heaters			
Gas Decay Tanks Temp.	*								
Gas Surge Tank Temp.	*								
Reactor Drain Tank Pressure	*								
Flash Tank Pressure		*	*	*					
Holdup Tank 1A Pressure		*	*	*					
Holdup Tanks 1B, 1C, 1D Pressure	*		*	*				2 psig	
Spent Resin Tank Press.	*		*[2]					0-15 psig	
Gas Decay Tanks Press.	*		*[2]					0-165 psig	
Gas Surge Tank Press.	*		*[2]			Waste Gas Compressor		0-5 psig	
Boric Acid Holding Pumps[5] Discharge Press	*							0-60 psig	
Reactor Drain Pump Discharge Pressure	*							20-60 psig	
Flash Tank Pumps Discharge Pressure	*							20-60 psig	

TABLE 11.2-16 (Cont'd)

System Parameter and Location	Indication Local	Indication Contr Room	Alarm High	Alarm Low	Rec[1]	Control Function	Inst. Range[4]	Normal Operating Range	Instr. Accuracy[4]
Holdup Drain Pumps Discharge Pressure	*							20-60 psig	
Boric Acid Condensate Pumps[5] Discharge Pressure	*							60 psig	
Equipment Drain Pump Discharge Pressure	*							60 psig	
Chemical Drain Pump Discharge Pressure	*							0-60 psig	
Laundry Drain Pumps Discharge Pressure	**							0-60 psig	
Waste Condensate Pumps Discharge Pressure	*							0-60 psig	
Resin Dewatering Pump[5] Discharge Pressure	*							0-60 psig	
Waste Gas Compressors Discharge Pressure	*							0-165 psig	
Waste Gas Discharge Pressure	*							0-15 psig	
Nitrogen Supply Press.	*		*	*				150-240 psig	
Hydrogen Supply Press.	*		*	*				90-110 psig	

TABLE 11.2-16 (Cont'd)

System Parameter and Location	Indication Local	Indication Contr Room	Alarm High	Alarm Low	Rec[1]	Control Function	Inst. Range[4]	Operating Range	Normal Instr. Accuracy[4]
Pre-Concentrator Filters ΔPs	*							0-10 psid	
Pre-Concentrator Ion Exchangers ΔPs	*							1-15 psid	
Boric Acid Condensate Ion Exchangers[5] ΔPs	*							1-15 psid	
Boric Acid Discharge Strainer ΔP	*							0-10 psid	
Laundry Filter ΔP	*							1-10 psid	
Waste Filter ΔP								1-15 psid	
Waste Ion Exchanger ΔP	*							1-15 psid	
Reactor Drain Tank Level		*	*	*		Reactor' Drain Pumps			
Flash Tank Level	*	*	*	*		Flash Tank Pumps, Isolate Tank & Bypass to Holdup Tanks			
Holdup Tanks Levels	*	*	*	*		Holdup Drain Pumps			
Boric Acid Holding Tank[5] Level	*		*[2]	*[2]		Boric Acid Holding Pump			
Boric Acid Condensate Tanks[5] Levels	*		*[2]	*[2]		Boric Acid Condensate Pumps			

TABLE 11.2-16 (Cont'd.)

System Parameter and Location	Indication Local	Indication Contr Room	Alarm High	Alarm Low	Rec[1]	Control Function	Inst. Range[4]	Normal Operating Range	Instr. Accuracy[4]
Waste Condensate Tanks Levels	*		*2	*2		Waste Condensate Pumps			
Equipment Drain Tank Level	*		*2	*2		Equipment Drain Pump 1A			
Aerated Waste Storage Tank Level	*		*2	*2		Equipment Drain Pumps 1B & 1C			
Chemical Drain Tank Level	*		*2	*2		Chemical Drain Pump			
Laundry Drain Tanks Levels	*		*2	*2		Laundry Drain Pump			
Spent Resin Tank Level	*		*2						
Gas Surge Tank Level			*						
Flash Tank N_2 Supply Flow	*							1.5 scfm	
Flash Tank Flow	*					Flash Tank Pumps, N_2 Supply Flow		-----	
Sluicing Water Flow	*							0-100 gpm	
Liquid Waste Discharge Flow		*				Liquid Waste Dischg. Isolation and Flow Control Valves		10-50 gpm	
Waste Gas Discharge Flow	*	*		*				.5-2.0 scfm	

TABLE 11.2-16 (Cont'd.)

WASTE MANAGEMENT SYSTEM INSTRUMENTATION APPLICATION

System Parameter and Location	Indication Local	Indication Contr Room	Alarm High	Alarm Low	Alarm Rec[1]	Control Function	Inst. Range[4]	Normal Operating Range	Instr. Accuracy[4]
Liquid Waste Dischg. (WMT) Radiation Monitor	*	*	*		*	Liquid Waste Dischg. Isolation Valves		Variable	
Waste Gas Dischg. Radiation Monitor		*	*		*	Waste Gas Dischg. Isolation Valves		Variable	
Boric Acid Concentrator 1A Pkg.[5] Alarms	*		*[3]					-	
Boric Acid Concentrator 1B Pkg.[5] Alarms	*		*[3]					-	
Radioactive Waste Concentrator Pkg.[5] Alarms	*		*[3]					-	
Waste Management System Annunciator Panel	*		*[3]					-	
Gas Analyzer	*		*[3]					-	
Equipment Drain Tank Differential Pres.	*		*[2]						
Waste Monitor Tank Level	*		*[2]	*		Waste Monitor Pump 1A & 1B; Drain & Condensate Pumps			
Waste Monitor Tank Pump Dischg. Pres.	*							0-60 psig	
WMT Strainer Delta P	*					----		----	

1 All alarm and recorders are in the control room unless otherwise indicated.
2 Alarmed on local Waste Management System annunciator panel.
3 Master alarm annunciated in control room for any individual system alarm condition.
4 Instrument ranges are selected in accordance with standard engineering practices. Instrument accuracies are selected such that existing instrument loop performance and safety analysis assumptions remain valid. Where applicable, instrument accuracies are also evaluated for their impact on setpoints in accordance with the FPL Setpoint Methodology.
5 These components are no longer used.

REFER TO DRAWING
8770-G-078, Sheets 160A, B, & C

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

FLOW DIAGRAM
WASTE MANAGEMENT SYSTEM

FIGURE 11.2-1

REFER TO DRAWING

8770-351

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

WASTE CONCENTRATOR-PIPING AND INSTRUMENTATION DIAGRAM

FIGURE 11.2-2

REFER TO DRAWING

8770-G-078, Sheets 161A & B

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

FLOW DIAGRAM
WASTE MANAGEMENT SYSTEM

FIGURE 11.2-3

REFER TO DRAWING

8770-G-078, Sheets 162A & B

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

FLOW DIAGRAM
WASTE MANAGEMENT SYSTEM
FIGURE 11.2-4

REFER TO DRAWING

8770-344

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

BORIC ACID CONCENTRATOR-PIPING AND INSTRUMENTATION

FIGURE 11.2-4a

REFER TO DRAWING

8770-G-078 Sheet 164

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

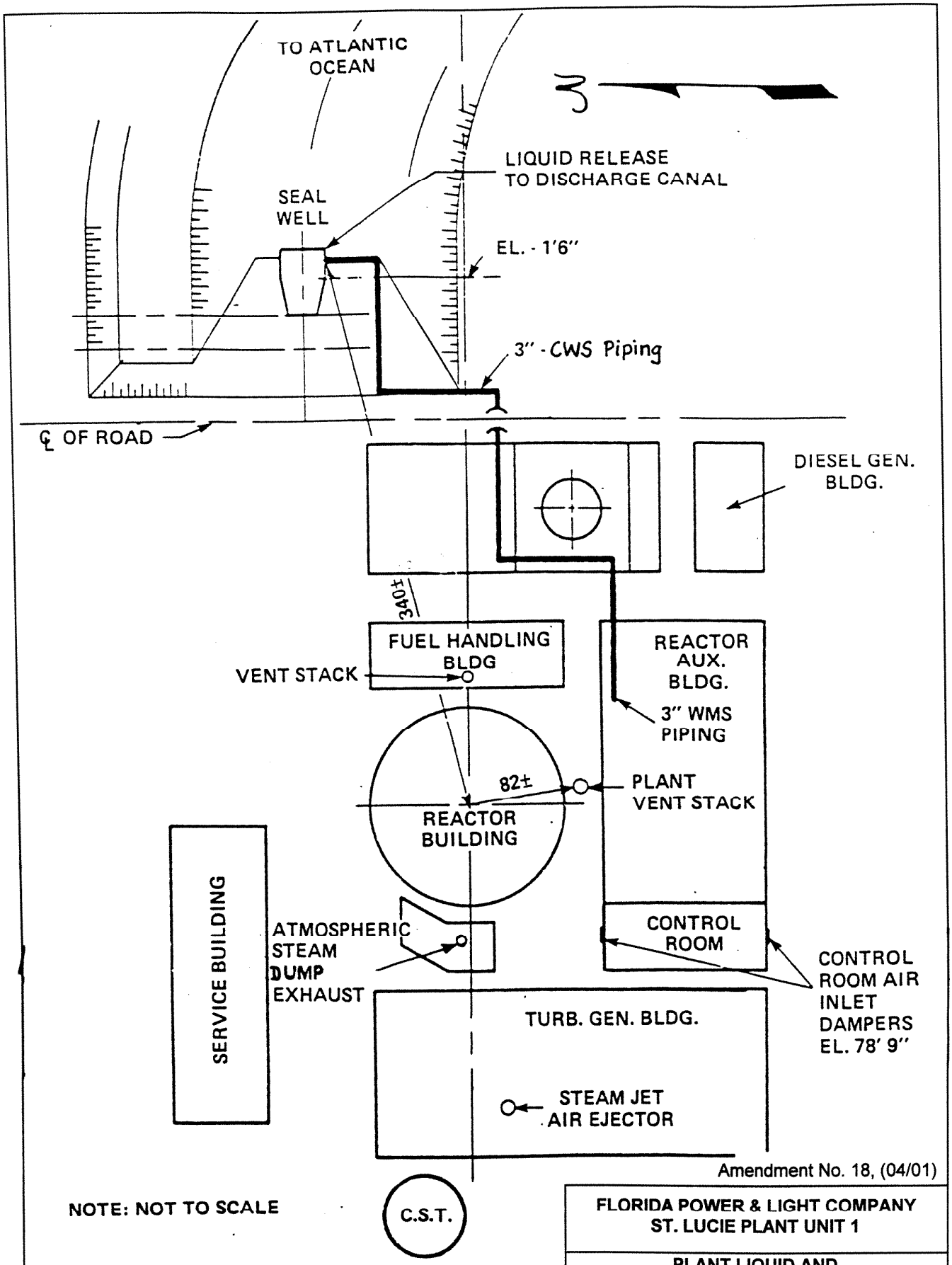
FLOW DIAGRAM WASTE MANAGEMENT SYSTEM
FIGURE 11.2-4b

REFER TO DRAWING

8770-G-078 Sheet 165

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

FLOW DIAGRAM WASTE MANAGEMENT SYSTEM
FIGURE 11.2-4c



Amendment No. 18, (04/01)

NOTE: NOT TO SCALE

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

PLANT LIQUID AND GASEOUS RELEASE POINTS
FIGURE 11.2-5

11.3 GASEOUS WASTE SYSTEM

See updated information concerning the Gaseous Waste System presented in the SL-1 10 CFR 50 Appendix I submittal, (6/1/76 Supplement to the St. Lucie Unit 1 Environmental Report).

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 1 and is retained here for historical purposes.

Continued compliance with the annual regulatory dose limits following extended 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 EPU 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 dose due to the EPU and to demonstrate that the estimated off-site doses following the EPU 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 Radioactive Effluent Release Report.

Estimations of the Gaseous Waste System processed volumes, releases and offsite doses are based on calculational data performed prior to operation of St. Lucie Unit 1. Actual data on Gaseous Waste effluents during operation is presented in the Annual Radiological Environmental Monitoring Report.

11.3.1 DESIGN BASES

The gaseous waste system processes the vent gases from equipment located in the chemical and volume control system, waste management system and fuel pool system, such that the radioactive gaseous release to the environs will be as low as practicable. The numerical design objective for releases during normal operation is to limit the site boundary noble gas dose to less than 10 mrem/year and iodine-131 and particulate site boundary concentrations to 10^{-5} times 10 CFR20 limits. Releases due to anticipated operational occurrences will be within 10 CFR 20 limits.

11.3.2 SYSTEM DESCRIPTION

11.3.2.1 Waste Gas Processing Systems

The principal flow paths of the waste gas system are shown on Figure 11.3-1. Process flow and activity data are given in Table 11.3-1 and component data in Table 11.3-2.

Plant gaseous releases come from the reactor auxiliary building ventilation, turbine system leakage, steam jet air ejector operation, gland steam condenser operation and containment purging in addition to releases from the gas collection header and gas surge header. Releases resulting from turbine leakage, the steam jet air ejector and gland system condenser are based on equations (1), (2) and (3) in Section 11.2.2.2. Gland steam has the same activity as turbine steam. Releases from the reactor auxiliary building ventilation system are based on leakage of unprocessed reactor coolant at 120 F. Containment purging results in a constant release regardless of percent failed fuel since the airborne radioactivity removal system is operated for as long as required to achieve the required activity level referred to in Section 12.2.

Waste gas is collected from the various source components by three headers; containment vent header, gas surge header, and gas collection header. The containment vent header receives hydrogenated potentially radioactive gas mixtures vented from the reactor drain tank, quench tank, refueling failed fuel detector vent, and reactor vessel head vent within the containment and directs the gases to the gas surge header. Hydrogenated and potentially radioactive gases vented from the volume control tank, flash tank, and boric acid concentrators in the reactor auxiliary building are also directed to the gas surge header along with the discharge gas from the gas analyzer. The vented gases flow to the surge tank where they are collected prior to being monitored for activity and released or compressed and sent to the Gaseous Radwaste Treatment System. If waste gases are entering the Gaseous Radwaste Treatment System they will remain in the surge tank until removed by the waste gas compressors. These compressors are automatically controlled by pressure instrumentation located on the surge tank. The surge tank is equipped with a drain line to remove any water that accumulates in the tank due to condensation. A level switch with a local alarm is on the surge tank and indicates to the operator when the drain trap is not functioning properly if continuously aligned.

Since the contents of the surge tank are expected to contain significant amounts of hydrogen, the gas stored in the surge tank is sampled frequently by the gas analyzer to determine the oxygen content of the tank. A nitrogen line connected to the surge tank and a pressure regulating valve in the line opens when pressure in the tank falls below 1.5 psig thereby maintaining a positive pressure above atmospheric and preventing air ingress.

Waste gas requiring holdup prior to release flows from the gas surge tank to a compressor where it is compressed to 165 psig and cooled by an aftercooler prior to entering the gas decay tanks. Holdup in these tanks permits additional radioactive decay. Chemistry procedures contain guidelines which are used to determine whether radioactive gaseous effluent requires holdup prior to release. Two compressors are available for transferring the gas to the gas decay tanks and are controlled by pressure instrumentation on the surge tank. One compressor starts when pressure in surge tank increases to 3 psig and stops when pressure falls to 1.5 psig. The second compressor starts at 7 psig and stops at 5 psig.

Aftercoolers supplied with each gas compressor cool the compressed gas prior to entering the gas decay tanks. There are three gas decay tanks (each provided with a pressure indicator including local alarm and temperature indicator) which receive the compressed gas from the waste gas compressors. The decay tanks have sufficient storage capacity for an average 30 day holdup, and after radioactivity has decayed to an acceptable level consistent with the design objective and has been verified by laboratory sample analysis, the gas is released to the environment via the plant vent at a controlled rate.

The fill procedure for the decay tanks is to have only one tank lined up to the compressor discharge. When the pressure in the tank increases to 165 psig the tank is isolated and manually switched over to an empty tank. The gaseous radioactivity in the filled isolated tank is allowed to decay until release can be made within the established limits. During this decay period the gas is periodically sampled and activity level determined. The sampling technique used prior to release of gas and the continuous monitoring systems are further discussed in Section 11.4.3.

The gas decay tank holdup time of 30 days is based on the following:

a. Three 144 ft^3 gas decay tanks operated at 165 psig provide a total storage capacity of approximately 5300 scf.

b. Three tanks permit simultaneous filling, holdup, and venting operations (i.e., while one tank is being filled, one filled tank is isolated for holdup, and the remaining tank may be discharged at a controlled rate to the vent pipe.

c. An annual waste gas volume of approximately 50,700 scf (0.097 scfm average) is conservatively estimated from processing reactor coolant degassing operations. The estimated annual volume of 50,700 scf conservatively assumes 4 cold shutdowns per year and 220 gpd reactor coolant leakage to the reactor drain tank with corresponding displacement of tank cover gas to the gas decay tanks.

d. The short term controlled discharges with 1 percent failed fuel conditions from the gas decay tanks are such that the resulting site boundary concentrations are less than 10 CFR 20 limits. Using the 0-2 hour short term χ/Q of 8.55×10^{-5} sec/m^3 (Section 2.3.4.3) and the gas decay tank activity given in Table 11.3-4, a minimum release time of approximately 1.5 days is required to meet the 10 CFR 20 limits. A discharge rate of approximately 1.2 scfm would empty a gas decay tank in 1.5 days. System design will allow up to approximately 5 scfm release rate.

Using the above assumptions, the holdup time is obtained as follows:

Gas Decay Tank	Elapsed Time (days)	Waste Gas Volume GDT (scf)	Average Decay Time (days)
1	12.6 fill time	1765	6.3 during filling
2	12.6 isolated	1765	12.6 during isolation
3	11.1 isolated	1765	11.1 during isolation
	1.5 discharge	1765	0.7 during discharge
Total	37.8	5295	30.7

Extra margin is also available in the overall holdup time in that gas production generally increases exponentially with core life because of boron dilution waste. For end of core cycle operation, sufficient gas decay tank capacity for 30 day holdup is provided assuming the following conservative operational conditions:

a. Boron dilution wastes are eliminated by use of the Chemical and Volume Control System ion exchanger(s) at approximately 97 percent core life.

b. 220 gpd reactor coolant leakage to the reactor drain tank.

c. Cold shutdown and startup at 90 percent life.

The following components are located in the discharge line from the gas decay tanks to the vent pipe; a pressure reducing valve, pressure indicator, needle valve, pneumatic operated fail closed on-off valve, an in-line radiation monitor, and a gas flow meter.

Prior to release, the required flow rate is determined, and the set point on the radiation monitor established. Initially the discharge valve from the gas decay tank, needle valve, and pneumatic operated valve are closed. The on-off pneumatic operated valve is opened and placed in automatic, and the discharge valve on the desired gas decay tanks is opened. The pressure reducing valve automatically closes when pressure from the decay tank in excess of 10 psig is sensed at the pressure reducing valve outlet. The needle valve is then opened as required to establish the desired flow rate to the vent pipe, and the pressure reducing valve opens to maintain constant downstream pressure of 10 psig. The on-off pneumatic operated valve automatically closes on high radioactivity level thus terminating discharge flow. An alarm will annunciate this event in the control room. When discharge flow decreases and the decay tank pressure decreases to approximately 10 psig, as noted by observing the gas flow meter and pressure indicator on the gas tank, the pressure reducing valve set point must be reduced to vent the tank down to atmospheric pressure.

The system flow paths and release points of the gases from the gas decay tanks and gas collection header are indicated on Figure 11.3-1. The diagram also shows the only flow bypass line in the waste gas system. This line from the gas surge tank to the vent pipe bypasses the waste gas compressor and gas decay tanks. This path is used to bypass compressors and gas decay tanks when Chemistry guidelines determine that holdup of gaseous effluent is not required prior to release, or when air or nitrogen is purged from process equipment after initial plant startup or for maintenance operations. A locked closed valve facilitates administrative control of this bypass line.

The waste gas system has connections for sampling the gas in the containment vent header, gas surge tank, and each gas decay tank. The gas to the gas surge header is primarily hydrogen and a gas analyzer located in the sampling system is used to monitor the oxygen concentration. Connections are available in the sampling system as outlined in Section 11.4.3 for taking a grab sample via gas analyzer to determine activity level prior to release.

The gas collection header collects the gases from primarily aerated vents of process equipment in the waste management system, chemical and volume control system, and fuel pool system. A listing of sources is given in Table 11.3-3. Because of the large volume of gas and the low activity level from the sources, the gases are routed directly to the vent pipe. The gases and expected activities to the vent pipe from the gas collection header are given in Table 11.3-1 at process data point No. 12. As a further check on activity from this source the vent pipe contains radioactivity monitors with alarms to indicate unexpected activity release.

The hydrogen and nitrogen required for plant operations are also a part of this system and redundant supply headers for each gas 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, and gas surge tank. 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 and valving to allow flexible operation. A low pressure alarm indicates when the backup source should be placed in service.

The design criteria and controls for limiting radiation exposure of personnel from lines which normally carry radioactive fluids are described in Sections 12.1.3 and 12.1.5.

11.3.3 SYSTEM DESIGN

The Gaseous Radwaste Treatment System is designed on a batch mode basis for flexibility of operation. These batching operations proceed intermittently at faster flow rates than the annual average process rates and therefore the system components are sized accordingly. Table 11.3-2 lists the design parameters for the major components of this system.

Process and radiation instrumentation are depicted in Figure 11.3-1 and are described in Table 11.2-16.

An accident analysis for the waste gas decay tanks is presented in Section 15.4.2.

11.3.4 OPERATING PROCEDURE

The Gaseous Radwaste Treatment System operating procedures include the following:

a) tank purging and venting procedures for all tanks venting to the waste gas system and for the tanks in the waste gas system;

b) gas surge tank to gas decay tank valve and compressor lineup procedures, requirements for monitoring proper automatic operation of the gas compressors, procedures for servicing the compressors, filters, and after coolers;

c) procedures for isolating a gas decay tank after filling and for monitoring the contents during the decay period;

d) procedures for setting the high trip set point on the discharge radiation monitor, for lining up the valves for discharge, and for determining the discharge flow rate and establishing it, monitoring requirements during the discharge period, procedure to reduce gas decay tank pressure to atmospheric at the end of the discharge period;

e) procedures for draining water from the gas surge tank and the gas decay tanks;

f) procedures for valve lineup, and establishing regulator set points and criteria for changing hydrogen gas cylinders;

g) procedures for valve lineup and establishing regulator set points and criteria for changing nitrogen gas cylinder; and

h) procedures for gas analyzer manual and automatic control of sample point selection, procedures for obtaining a gas sample in the sampling cylinder for laboratory analysis and for setting the sequential timer and for calibrating the gas analyzers to assure representative automatic sampling.

11.3.5 PERFORMANCE TESTS

The gas compressors including suction filter and discharge aftercooler are shop tested to assure proper operation. The gas analyzer system is shop tested.

The system preoperational tests are as follows:

a) instrumentation is checked for accuracy of readout and control and alarm set points;

b) control valves are checked for proper operation and relief and pressure regulation valve set points are confirmed;

c) operation of the entire system is checked by supplying nitrogen from various points such as the reactor drain tank and flash tank. Proper automatic operation of the compressors and controllability of discharge flow rate is verified; and

d) leak testing is done to assure negligible leakage.

During plant operation, periodic testing is done as follows:

a) the discharge radiation monitor is periodically calibrated with standard radiation source;

b) the gas analyzers are periodically calibrated with zero and upscale gases; and

c) process instrumentation is periodically calibrated.

11.3.6 ESTIMATED RELEASES

The gaseous releases have been calculated in consideration of the guidance provided by Regulatory Guide 1.42 (RG 1.42), "Interim Licensing Policy on as Low as Practicable for Gaseous Radioiodine Releases from Light-Water-Cooled Nuclear Power Reactors" (June 1973). The interim policy calculational methodology recommends an assumed failed fuel rate of 0.25 percent. Over the operating lifetime of this reactor, it is expected that failed fuel rates of 0.1% or less will predominate (see Section 11.1.5). Thus, average annual releases anticipated over the lifetime of the plant will be lower than those calculated by RG 1.42 methods by the ratio of the failed fuel rates (1/2.5). Likewise, releases during anticipated normal occurrences (1.0 percent) will be higher by the ratio of the failed fuel rates (10/2.5).

Gaseous releases based on RG 1.42 recommendations are provided in Table 11.3-4. The analysis assumed a steam generator blowdown rate per RG 1.42 of about 18 gpm and a cold water iodine partition factor of 10^{-4}.

The source terms used to determine these releases can be derived from Table 11.3-1 and Section 11.3.2. All noncondensable gases in the condenser are released to the environment. Releases from the reactor auxiliary building are calculated by multiplying the leak rate for reactor auxiliary building equipment by the appropriate partition factors and ventilation system filter efficiencies. Assumed leak rates, partition factors, and filter efficiencies are given in Table 11.2-13.

Any radioactivity released from failed fuel is reduced in concentration by the process equipment in the chemical and volume control system and waste management system. For iodine, the purification and preconcentrator ion exchanger decontamination factor (DF) is 10^3, and a liquid to gas partition factor of 10^4 is used for the reactor drain tank, flash tank, holdup tank, equipment drain tank, and boric acid concentrators(1). Noble gases are stripped by a factor of approximately 2 in the flash tank and 5 in the boric acid concentrators(1). Activity will also be reduced by natural decay due to residence time in the reactor drain tank, holdup tank, and gas decay tanks.

11.3.7 RELEASE POINTS

The release point to the vent pipe from the gas decay tanks or gas surge tank is indicated on Figure 11.3-1. The gaseous effluents from the reactor auxiliary building ventilation system and containment purge system are also released to the vent pipe. The vent pipe is shown on reactor auxiliary building general arrangement Figure 1.2-14, and on the ventilation systems flow diagram, Figure 9.4-1. The air ejectors are shown on Figures 1.2-5 and flow diagram Figure 10.1-3. All the release points are shown on the site plot plan, Figure 11.2-5.

11.3.8 DILUTION FACTORS

The value of χ/Q used for the plant gaseous releases at the site boundary during operation is determined from the guidance provided in the Offsite Dose Calculation Manual (ODCM). Generally, the χ/Q is determined based on the actual meteorological data that occured during the period of interest. Doses to the population at greater distances up to 50 miles are calculated using the worst average meteorological data given in Table 2.3-69. Factors used for specific dose estimates are provided in Section 11.3.9.

11.3.9 ESTIMATED DOSES(2)

Regulatory Guide 1.42 states that the limiting dose in achieving as low as practical releases is the dose calculated for a hypothetical child's thyroid and that the grass-cow-milk-child pathway is limiting for the calculation. This pathway has been analyzed for St. Lucie 1 in accordance with paragraph C.3 of Regulatory Guide 1.42 for cows actually in the area. In addition, the thyroid dose to an adult from ingestion of milk, the maximum whole body dose to an individual, the maximum thyroid dose to an individual as a result of inhalation, the whole body dose to the population within a 50 mile radius of the plant over the course of one year (8760 hours), and the thyroid-dose to an individual ingesting green leafy vegetation have been analyzed.

(1) The boric acid conentrators are no longer used.
(2) Estimated doses are based on information obtained from a study of the area surrounding the site prior to issuance of the operating license. Updated information concerning gaseous waste release doses is presented in the St. Lucie Unit 1 10 CFR 50 Appendix I submittal supplement to the site Environmental Report.

The offsite doses were calculated using the equation:

$$\text{Dose} = \sum\left(\frac{D}{A}\right)_i \quad R_i \quad \left(\frac{\chi}{Q}\right)$$

where

$\left(\frac{D}{A}\right)_i$ = Dose conversion factor for isotope i

R_i = The release in curies of isotope i

$\frac{\chi}{Q}$ = Meteorological dispersion factor (sec/m^3)

The dose conversion factors are provided in Tables 11.3-5A and 11.3-5B.

To estimate the dose potential via the milk and vegetable pathways, prior to issuance of the operating license, a study was made of the area surrounding the site. A Hutchinson Island zoning map is provided as Figure 11.3-2 and a five mile radius aerial photograph as Figure 11.3-3. The study indicated that at the time Unit 1 was initially licensed:

a) The closest existing dairy is located about 14 miles west-southwest of the site, and the present herd size is about 1000 head.

b) An existing, privately-owned cow is located about 7.5 miles from the site in the southwest direction.

c) The land close to the site is not available for and/or is not well suited for agricultural purposes.

With regard to the latter point, consider first Hutchinson Island:

a) All land within one mile is on the plant site.

b) To the south of the plant site the land is zoned residential to about 5 miles where a commercial camping ground is located.

c) To the north of the site the land is zoned residential to about 6 miles where City of Fort Pierce zoning commences.

d) There are currently no wells on Hutchinson Island and attempts to drill fresh water wells have been unsuccessful. Out to over 4 miles of the site, both north and south, there are no residents.

Turning now to the area west of the site across the Indian River. At about 1.75 miles mainland areas zoned residential begin. Further west are land areas zoned agricultural. However, the eastern portion of this area is covered with dense foliage and is primarily a swamp area. It is conceivable that at about 2.5 miles some form of agricultural activity could be initiated. Grazing would be quite limited and forage would be required to supplement existing vegetation.

Based on the above considerations and the guidance provided in Regulatory Guide 1.42, the milk-pathway calculation was performed at 7.3 miles for the closest existing cow in the plant site vicinity. In addition, vegetable pathway calculations at 2.5 miles were performed.

The atmospheric dilution factors (χ/Q) used in the vegetable and milk pathway calculations were:

a) at 7.5 miles from the plant in the southwest direction $\chi/Q = 7.57 \times 10^{-8}$ sec/m^3; and

b) at 2.5 miles from the plant in the southwest direction $\chi/Q = 4.04 \times 10^{-7}$ sec/m^3.

Atmospheric dilution factors do not account for the additional dilution afforded by building wake effects and depletion due to deposition from release point to receptor. At 2.5 miles, the former would decrease the dose in the order of 10 percent and the latter around 20 percent.

At the time of this calculation the nearest continually occupied dwelling is about 1.8 miles west southwest of the site. The χ/Q at this location is 5.07×10^{-7} sec/m^3.

The doses calculated utilizing the releases presented in Table 11.3-4 and the above assumptions are presented in Table 11.3-6.

11.3.10 CALCULATED OFFSITE DOSE

The Offsite Dose Calculation Manual (ODCM) describes the methodology used to calculate doses from offsite release during St. Lucie Unit 1 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 or as long as the current revisions of the Regulatory Guide 1.109 dose conversation factors and environmental transfer factors are substituted for those currently included in the ODCM.

TABLE 11.3-1
(HISTORICAL)
WASTE GAS SYSTEM FLOW DATA POINTS
(Reference Figure 11.3-1)
ANTICIPATED OPERATIONAL OCCURRENCE FOR 1% FAILED FUEL*

Position No.	1	2	3	4	5	6
Flow, SCFM	2.46(-2)	(1.90-9.5)x 10^{-4}	3.46(-3)	1.96(-2)	4.82(-2)	9.65(-2)
Pressure, psig	0.5-7	0.5-7	0.5-7	0.5-7	0.5-7	0.5-7
Temperature/F	70-120	70-120	70-120	70-120	70-120	70-120
Kr-85m $\mu\frac{Ci}{cc}$ **	0.12	2.19	0	0-22	4.16	2.19
Kr-85 "	0.464	2.24	21.8	0.16	2.56	2.24
Kr-87 "	0.206	1.24	0	8.29(-2)	2.30	1.24
Kr-88 "	0.140	3.81	0	0.35	7.24	3.81
Xe-131m "	0.728	3.18	21.5	0.26	4.26	3.18
Xe-133 "	81.80	257.0	996.	32.14	381.0	257.0
Xe-135 "	1.12	11.21	0	1.18	21.1	11.21
Xe-138 "	2.12(-3)***	0.51	0	3.98(-3)	1.01	0.51
I-129 "	4.80(-14)	2.70(-14)	3.98(-16)	1.29(-15)	2.84(-14)	2.70(-14)
I-131 "	2.31(-4)	1.20(-4)	7.76(-7)	7.06(-8)	1.19(-4)	1.20(-4)
I-132 "	4.50(-6)	2.63(-6)	0	1.38(-8)	2.89(-6)	2.63(-6)
I-133 "	1.57(-4)	8.26(-5)	9.53(-10)	9.72(-8)	8.25(-5)	8.26(-5)
I-134 "	1.00(-6)	7.21(-7)	0	5.37(-9)	9.07(-7)	7.21(-7)
I-135 "	3.03(-5)	3.16(-5)	0	4.24(-8)	1.68(-5)	3.16(-5)

* Concentrations of nuclides are reduced by a factor of 10 for the expected operation of 0.1% failed fuel.
** Activity values calculated at standard temperature and pressure.
*** Numbers ( ) are powers of ten.

TABLE 11.3-1 (Cont'd)
(HISTORICAL)
ANTICIPATED OPERATIONAL OCCURRENCE FOR 1% FAILED FUEL*

Position No.	7	8	9	10	11	12
Flow, SCFM	0.097	0.097	0.097	0.097	0.097	0.75
Pressure, psig	0.5-7	10-165	10-165	10	<0.1	<0.1
Temperature/F	70-120	70-110	80	80	80	70-120
Kr-85mμCi/cc**	2.19	2.19	0	0	0	1.33(-3)
Kr-85 "	2.24	2.24	2.22	2.22	2.22	4.31(-2)
Kr-87 "	1.24	1.24	0	0	0	2.80(-4)
Kr-88 "	3.81	3.81	0	0	0	1.70(-3)
Xe-131m "	3.18	3.18	0.56	0.56	0.56	4.41(-2)
Xe-133 "	257.0	257.0	4.98	4.98	4.98	2.29
Xe-135 "	11.21	11.21	0	0	0	9.92(-3)
Xe-138 "	0.51	0.51	0	0	0	3.52(-10)
I-129 "	2.70(-14)***	2.70(-14)	2.30(-13)	2.30(-13)	2.30(-13)	1.84(-15)
I-131 "	1.20(-4)	1.20(-4)	9.10(-6)	9.10(-6)	9.10(-6)	3.54(-6)
I-132 "	2.63(-6)	2.63(-6)	0	0	0	2.82(-10)
I-133 "	8.26(-5)	8.26(-5)	0	0	0	4.97(-9)
I-134 "	7.21(-7)	7.21(-7)	0	0	0	6.60(-12)
I-135 "	3.16(-5)	3.16(-5)	0	0	0	1.63(-10)

* Concentrations of nuclides are reduced by a factor of 10 for the expected operation of 0.1% failed fuel.
** Activity values calculated at standard temperature and pressure.
***Numbers in () are powers of 10.

TABLE 11.3-2

COMPONENT DATA[1]

1. Waste Gas Compressor

Type	Diaphragm Positive Displacement
Quantity	2
Capacity, SCFM	2
Discharge Pressure, psig	0-165
Codes	ASME P&V III for Nuclear Power Nov. 1968 ASME Power Test Code PTC-9 Displacement Compressors, Vacuum Pumps and Blowers.
Materials	Carbon Steel
Design Temperature, F	150 - Inlet; 350 - Outlet
Design Pressure, psig	200

2. Compressor Aftercooler

Type	Tube in Tube
Quantity	
Codes: Gas Side	ASME Boiler and Pressure Vessel Code 1968 Section III Unfired Pressure Vessels, Class C
Water Side	Section VIII of above code
Materials	Carbon Steel
Discharge Temperature, F	110

3. Compressor Inlet Filter

Type	Stainless Steel Screen
Quantity	2
Rating	5 micron
Clean Pressure Drop, psi @ 2 CFM	0.3
Code	ASME Boiler and Pressure Vessel Code 1968 Section III Unfired Vessels, Class C

TABLE 11.3-2 (Cont'd)

4. Gas Surge Tank

Type	Vertical
Quantity	1
Volume, ft 3	9
Design Pressure, psig	40
Design Temperature, F	200
Code	ASME Boiler and Pressure Vessel Code 1968, Section III, Class C
Material	Carbon Steel

5. Gas Decay Tank

Type	Vertical
Quantity	3
Volume, each, ft 3	144
Design Pressure, psig	190
Design Temperature, F	250
Codes	ASME Boiler and Pressure Vessel Code 1968, Section III, Class C
Material	Carbon Steel

(1) Original procurement information

TABLE 11.3-3

GAS COLLECTION HEADER SOURCE POINTS

1. Preconcentrator ion exchanger vent
2. Holdup tank vent
3. Boric acid condensate ion exchanger vent*
4. Boric acid holding tank vent*
5. Boric acid condensate tank vents*
6. Waste ion exchanger
7. Equipment drain tank vent
8. Chemical drain tank vent
9. Laundry drain tank vents
10. Waste condensate tank vents
11. Spent resin tank vent
12. Waste concentrator vent*
13. CVCS ion exchanger vents
14. Fuel pool system ion exchanger vent
15. Boric acid makeup tank vents
16. Charging pump vents
17. Boric acid makeup pump vents (capped per PC/M 86150)
18. Aerated water storage tank vent
19. Resin dewatering pump vent
20. Seal lube water tank vent
21. Post-accident sampling vent

*Note: These components are no longer used and vents have been isolated.

TABLE 11.3-4
(HISTORICAL)
ESTIMATED GASEOUS RELEASES BASED ON REGULATORY GUIDE 1.42 (Ci/sec)*
(0.25% failed fuel)

Isotope	Gas Surge Header	Gas Collection Header	Steam Generator Blowdown	Reactor Auxiliary Building Vent	Turbine Building Leakage From Secondary System	Steam Jet Air Ejector	Containment Purge	Turbine Steam Heating	Total Ci/sec.	Total Ci/yr.
Kr-85m	0	1.18(-7)	1.54(-10)	3.86(-7)	3.85(-11)	2.53(-7)	1.18(-8)	1.36(-12)	7.69(-7)	2.42(+1)
Kr-85	2.52(-5)**	3.84(-6)	9.13(-11)	2.29(-7)	2.29(-11)	1.51(-7)	3.22(-7)	8.10(-13)	2.97(-5)	9.36(+2)
Kr-87.	0	2.46(-8)	8.26(-11)	2.10(-7)	2.07(-11)	1.36(-7)	4.22(-9)	7.31(-13)	3.75(-7)	1.18(+1)
Kr-88	0	1.52(-7)	2.66(-10)	6.75(-7)	6.69(-11)	4.41(-7)	2.72(-8)	2.37(-12)	1.30(-6)	4.10(+1)
Xe-131m	6.15(-6)	3.94(-6)	1.52(-10)	3.83(-7)	3.83(-1)	2.53(-7)	1.11(-7)	1.35(-12)	1.08(-5)	3.40(+2)
Xe-133	6.37(-6)	2.03(-4)	1.87(-8)	4.69(-5)	4.69(-9)	3.09(-5)	6.76(-6)	1.66(-10)	2.94(-4)	9.26(+3)
Xe-135	0	8.81(-7)	7.58(-10)	1.91(-6)	1.91(-10)	1.20(-6)	6.90(-8)	6.73(-12)	4.06(-6)	1.28(+2)
Xe-138	0	3.18(-9)	3.48(-11)	9.34(-8)	8.53(-12)	5.62(-8)	7.16(-10)	3.00(-13)	1.54(-7)	4.85(0)
I-129	2.60(-18)	1.64(-19)	1.22(-18)	9.35(-17)	3.06(-17)	1.01(-16)	4.10(-17)	1.08(-18)	2.71(-16)	8.56(-9)
I-131	1.03(-10)	3.15(-10)	5.88(-11)	5.15(-9)	1.48(-9)	4.87(-9)	2.26(-9)	5.22(-11)	1.43(-8)	4.51(-1)
I-132	0	2.51(-14)	1.36(-11)	1.41(-9)	3.42(-10)	1.13(-9)	4.72(-10)	1.21(-11)	3.38(-9)	1.06(-1)
I-133	0	4.41(-13)	4.12(-11)	7.34(-9)	1.04(-9)	3.40(-9)	3.14(-9)	3.66(-11)	1.50(-8)	4.73(-1)
I-134	0	5.84(-16)	3.21(-13)	8.04(-10)	8.02(-12)	2.65(-11)	2.20(-10)	2.83(-13)	1.06(-9)	3.33(-2)
I-135	0	1.45(-14)	8.93(-12)	3.51(-9)	2.24(-10)	7.39(-10)	1.42(-9)	7.89(-12)	5.91(-9)	1.87(-1)

* Activity values calculated at standard temperature and pressure.
** Numbers in ( )are powers of ten

TABLE 11.3-5A
(HISTORICAL)
FACTORS FOR CONVERTING AIR CONCENTRATIONS OF

RADIOIODINE TO THYROID DOSE*

VIA INHALATION

(mrem/yr per pCi/m^3)

Age	I-129	I-130	I-131	I-132	I-133	I-134	I-135
1 yr.	44.6	1.62	12.7	0.452	3.31	0.227	1.09
Adult	34.3	1.57	10.9	0.414	2.70	0.209	0.985

VIA CONSUMPTION OF GREEN LEAFY VEGETABLES
(mrem/yr per pCi/m^3)

Age	I-129	I-130	I-131	I-132	I-133	I-134	I-135
4 yr.	2280	3.25	244	0.168	10.1	0.033	1.16
Adult	1300	2.13	150	0.106	5.99	0.021	0.731

VIA THE MILK PATHWAY
(mrem/yr per pCi/m^3)

Age	I-129	I-130	I-131	I-132	I-133	I-134	I-135
1 yr.	26,700	70.2	3560	2.35	151	0.468	16.7
Adult	2680	5.67	396	0.280	16.0	0.0561	1.96

TABLE 11.3-5B
WHOLE BODY DOSE CONVERSION FACTORS FOR PLUME IMMERSION

(rem/sec per C/m^3)

Kr-85m	Kr-85	Kr-87	Kr-88	Xe-131m	Xe-133	Xe-135	Xe-138
4.63(-2)	5.28(-4)	3.44(-1)	5.18(-1)	4.10(-2)	2.05(-2)	6.52(-2)	1.05(-1)

I-131	I-132	I-133	I-134	I-135
9.56(-2)	5.19(-1)	1.62(-1)	5.86(-1)	5.34(-1)

*Factors taken from "Final Environmental Statement Concerning Proposed Rule Making Action: Numerical Guides for Design Objectives and Limiting Conditions to Meet the Criterion "As Low As Practicable" for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents; Directorate of Regulatory Standards, U.S. Atomic Energy Commission."

TABLE 11.3-6
(HISTORICAL)
OFFSITE DOSES DUE TO PLANT
GASEOUS RELEASES

	Normal Operation	Regulatory Guide 1.42
Whole body dose to:		
a) Individual at site boundary	0.167 mrem/yr	0.415 mrem/yr
b) Population within 50 mile radius	0.70 man-rem/yr	1.75 man-rem/yr
Thyroid dose to:		
a) One year old individual at site boundary (Inhalation)	0.17 mrem	0.427 mrem
b) Adult-individual at site boundary (Inhalation)	0.144 mrem	0.361 mrem
c) One year old individual at nearest continually occupied dwelling (Inhalation at 1.8 miles)	0.050 mrem	0.125 mrem
d) Adult individual at nearest continually occupied dwelling (Inhalation at 1.8 miles)	0.042 mrem	0.106 mrem
e) Adult ingesting vegetables grown 2.5 miles from the plant	0.36 mrem/yr	0.9 mrem/yr
f) Four year old individual ingesting vegetables grown 2.5 miles from the plant	0.584 mrem/yr	1.46 mrem/yr
g) Adult ingesting milk from closest existing cow (7.5 miles)	0.178 mrem/yr	0.444 mrem/yr
h) One year old individual ingesting milk from closest existing cow (7.5 miles)	1.61 mrem/yr	4.02 mrem/yr

REFER TO DRAWING

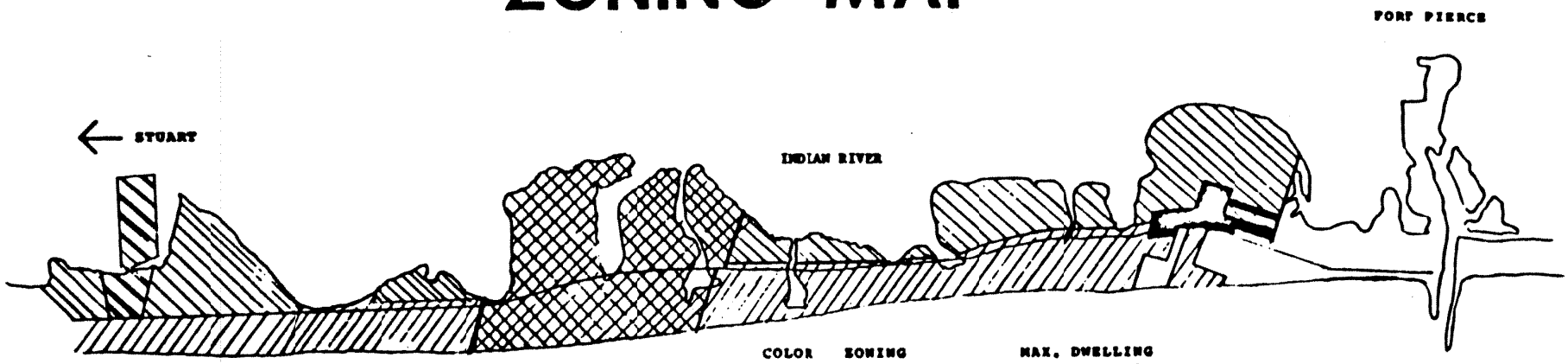
8770-G-078, Sheets 163A & B

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

FLOW DIAGRAM
WASTE MANAGEMENT SYSTEM

FIGURE 11.3-1

HUTCHINSON ISLAND
ZONING MAP



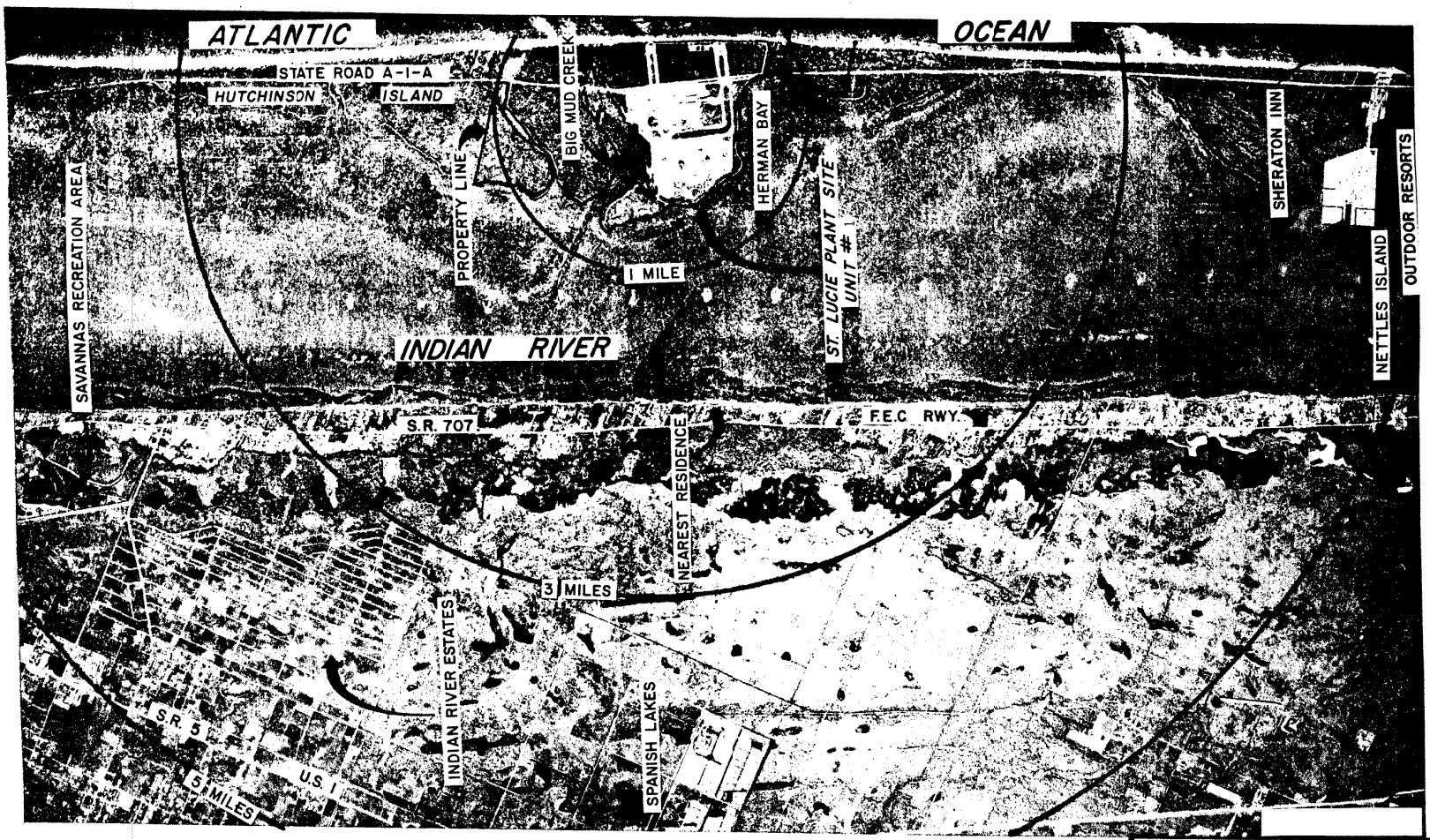
COLOR CODE	ZONING CLASSIFICATION	MAX. DWELLING UNITS PER ACRE
	FPL SITE	
	B-2	18
	A-1(campground)	15
	A-1	1
	R-1A	3
	R-4	18*

* 36 for Motels & Hotels

Note: Historical Information

FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

HUTCHINSON ISLAND
ZONING MAP
FIGURE 11.3-2



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PLANT LOCATION - 5 MILE RADIUS
FIGURE 11.3-3 SEPT. 1973

11.4 PROCESS AND EFFLUENT RADIOLOGICAL MONITORING SYSTEMS

11.4.1 DESIGN BASES

The process, effluent and area radiation monitoring systems are designed to warn of any radiation health violation which might develop, give early warnings of a malfunction which might lead to an unsafe health condition or unit damage and provide continual indication and recordings of radiation levels for normal operation, for anticipated operational occurrences and for a reasonable range of accident conditions. The system operates in conjunction with regular and special radiation surveys and with chemical and radiochemical analyses performed by the plant staff to provide timely, adequate information for continued safe operation and for assurance that personnel exposure does not exceed 10 CFR 20 guidelines.

Sampling systems, detectors, power supplies, ratemeters and recorders are designed to operate within the environmental conditions for the area in which they are located.

11.4.2 CONTINUOUS MONITORING

11.4.2.1 Process Radiation Monitor

Table 11.4-1 lists the process radiation monitors.

The primary purpose of the process radiation monitor is to alert plant operators to an increase in coolant radioactivity as quickly as possible. Such an increase in radioactivity would usually be caused by crud released in the reactor coolant system or chemical and volume control system 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 monitor is located in the chemical and volume control system letdown line in parallel with the purification filter but upstream of the ion exchangers. This location was selected because a continuous sample at relatively low temperature and pressure can be conveniently obtained and the sample effluent can be returned to the purification system without difficulty. This location also provides an optimum compromise between a minimum sample lag time and the required delay time for sample background radioactivity decay. The time lag from the reactor coolant system to the monitor is sufficient for all operating conditions to permit N^{16} to decay to a low level that will not interfere with monitor readings.

Gross gamma activity concentration and the activity concentration of a specific nuclide are monitored simultaneously. Iodine-135 is the specific nuclide chosen for activity monitoring for several reasons.

First, it is a fission product that is found in relative abundance in the reactor coolant in the event of fuel cladding failure. It is released from defective fuel with relative ease and does not plate out on the system surfaces. Finally, Iodine-135 is chosen because its 1.28 Mev gamma can be readily monitored using discrimination techniques, and its 6.7 hour half-life is long relative to sample lag time but short enough to provide indication of current fission product escape from the fuel. Some fission products that would contribute to gross gamma level plate out on the walls of the piping but since the gross gamma activity after minimum lag time is dominated by gaseous fission product activity, the gross gamma reading will be representative.

The reactor coolant specific activities as measured by the process radiation monitor are listed in Table 11.1-1 at a coolant temperature of 577F. The normal operating temperature of the process radiation monitor is 120F. Therefore, the data in Table 11.1-1 is increased by the density ratio of 1.39 to obtain the activity seen by the monitor.

The monitor consists of shielding, sampler, detector holder, gamma scintillation detector and preamplifier, logarithmic ratemeter, and single-channel linear ratemeter/analyzer. The gamma scintillation detector is a sodium iodide crystal, 1-1/2 inches diameter by 1 inch long, with photomultiplier tube and integral preamplifier which monitors gamma radiation in the energy range of 100 Kev to 3 Mev. The minimum and maximum detectable coolant activities are 10^{-4} and 10^{2} in µCi/cc of I-135 respectively. A remotely operated Cs-137 check source is used to test the overall operation of the system.

The sampler is of the in-line type with a sample volume 2-1/2 inches in diameter and 1-3/4 inches deep. The sampler and tubing are fabricated from stainless steel designed for 200 psig at 250°F. The detector assembly is completely shielded in 4 π geometry by approximately 4-1/2 inches of lead to reduce the background signal contribution. The shielding assembly also includes a removable collimating plug that must be placed between the detector and sample cell to increase the operating range of the monitor above 10 µCi/cc of I-135.

After amplification the detector output signal is fed to a linear ratemeter/analyzer and a logarithmic ratemeter located in the control room. The linear ratemeter/analyzer can monitor a specific fission product gamma activity (I^{135}) with a total range of 0 to 10^{6} cpm. The logarithmic ratemeter measures gross gamma activity with a range of 10 to 10^{6} cpm. Alarms are provided on both the linear ratemeter/analyzer and the logarithmic ratemeter with adjustable set points over the complete range.

The process radiation monitor is a trend monitor and its primary purpose is to indicate the possibility of fuel clad failure. Alarm set points are normally set at a value slightly above the current reading. It is expected that gross activity and perhaps Iodine-135 activity will periodically increase above the alarm set points due to normal plant transients. Consequently the alarm will periodically activate and the operator must determine the cause of the alarm. If an alarm is received and the Iodine-135 activity has increased and remains significantly above the prior steady state level, additional fuel failure can be assumed to have occurred. However, if an alarm is received due to high Iodine-135 activity concurrent with an increase in gross activity and a plant load increase, crud burst release can be suspected. In time, the coolant activity should return to the prior, lower, steady state value. If an alarm is received, the set points are immediately raised above the higher levels and the alarm circuits are reset. This assures that any successive increases will be brought to the operators attention. If coolant activity decreases, the set points are also lowered.

11.4.2.2 Gaseous Discharge Monitor

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 the gas decay tanks and to prevent radioactivity in excess of applicable limits from being released to the environment.

This monitor is located in the Gaseous Radwaste Treatment System in the gaseous discharge line downstream of the gas surge tank and the gas decay tanks but upstream of the plant stack. Therefore monitoring is accomplished before the gases are diluted in the plant stack. The concentrations of radioactivity expected to be released from the gas decay tanks are given in Table 11.3-1 under position number 10.

The monitor measures gross beta radioactivity. Even though some of the partially volatile fission products (iodines) will adhere to the walls of the piping and tanks this will not cause false low readings since the activity actually being released is the only concern. Shielding is provided with this monitor to minimize the effects of background radiation.

The monitor consists of an in-line sample detector assembly, shielding, check source, and logarithmic rate meter and recorder. Flow through the sampler is 1 to 10 scfm and all parts in contact with the process gas are stainless steel.

The detector is a beta scintillation detector which is shielded against background gamma radiation for a minimum sensitivity of 2×10^{-5} µCi/cc of Xe133. The total detector range is 10^{-5} to 10^{-3} µCi/cc based on discharged activity of unknown identity. A collimating plug is used in conjunction with the detector and is used for range of 10^{-1} to 10^{3} µCi/cc. Without the collimating plug detector range is 10^{-5} to 1.0 µCi/cc. The logarithmic rate meter receives the output signal from the detector. The range of rate meter is 10 to 10^{7} counts per minute.

There are two separate calibration check sources for the monitor. A remotely operated radiation check source can be exposed to the detector to calibrate the activity range of the detector. An internally generated count rate signal is used to calibrate the logarithmic rate meter.

The recorder receives an electrical signal from the logarithmic rate meter and continuously records the gaseous radioactivity that is released from the plant. The gas discharge flow rate is measured by a recorder in the control room.

The monitor has an alarm and trip system that is operated by the logarithmic rate meter. The set point of the alarm and trip is adjustable over the full range of the rate meter. Prior to the batch release of gaseous activity, a sample is taken for radioactive analysis as described in Section 11.4.3. Radioactivity analyses are performed to verify that the activity in the tank can be released to the environment without exceeding the applicable limits. The alarm set point of the monitor is set slightly greater than the level determined by the radioactivity analysis but below the applicable release limits.

If the activity should increase above the set point, an alarm is activated in the control room and valve V6565 on Figure 11.3-1 is tripped closed stopping discharge flow. The operator must then determine the cause of the alarm and trip. Another sample would be taken for radioactivity analysis to determine if the alarm and trip were due to high activity or instrument mis-calibration or malfunction. If high activity is the source of alarm and the contents of a gas decay tank are being released, discharge from that particular gas decay tank must either be justified at a higher activity level or the contents must be allowed to decay for an additional time period. If it has been established that the alarm was caused by instrument malfunction, discharging can continue only after the malfunction has been rectified.

11.4.2.3 Liquid Waste Discharge Monitor

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

This is an off-line monitor. Flow is caused by the differential pressure across flow element FE-6627 which is installed in the waste management system discharge line to the circulating water canal. Monitoring before dilution with circulating water allows greater accuracy of measurement.

The monitor for this service measures gross gamma radioactivity. The expected concentrations of each nuclide for 0.1 percent and 1.0 percent failed fuel conditions are given in Table 11.2-6 and 11.2-7, position 21, for boron recovery system releases.

If excessive radioactivity should be deposited in the sampler, the sampler may be removed for decontamination. Shielding is provided to reduce the effect of background radiation.

The monitor consists of an off-line sampler, detector assembly, shielding, check source, logarithmic ratemeter and recorder.

The monitor detector is a gamma scintillation crystal with photomultiplier tube and integral preamplifier shielded against background gamma radiation for a minimum detector sensitivity of 1 x 10^{-7} µCi/cc of Cs-137. The operating range is 10^{-7} to 10^{-2} µCi/cc based on discharged activity of unknown identity.

The digital ratemeter receives the output signal from the detector and has a range of 10^{-7} to 10^{-2} µCi/cc. Gammas with energies in the range of 70 Kev to 3 Mev can be detected.

The monitor contains a remotely operated radiation check source for calibrating the detector. A recorder receives an electrical signal from the digital rate meter and records the liquid radioactivity that is released from the plant. The liquid discharge flow rate is measured and is recorded in the control room.

If the activity reaches the set point, an alarm is activated in the control room and valve FCV-6627X shown on Figure 11.2-4C is tripped closed, stopping discharge flow. The operator must then determine the cause of the alarm and trip. Operator actions would be the same as those described for the gaseous discharge monitor.

11.4.2.4 Steam Generator Blowdown Monitor

The primary purpose of the steam generator blowdown radioactivity monitors is to continuously monitor and record the gross gamma activity discharged during blowdown of the steam generators.

One monitor is located in each blowdown sample line upstream of the blowdown line isolation valve.

One blowdown monitoring system is provided for each steam generator. Each consists of a sampler, gamma scintillation detector assembly with an integral photomultiplier tube and preamplifier, shielding, check source, logarithmic rate meter. One recorder serves both monitors.

The detector is a gamma scintillation crystal with photomultiplier tube and integral preamplifier shielded against background radiation for a minimum detector sensitivity of 1.0×10^{-6} µCi/cc of Cs-137. The operating range is 10^{-6} to 10^{-1} µCi/cc based on discharged activity of unknown identity.

The logarithmic ratemeter receives the output signal from the detector and has a range of 10 to 10^6 cpm. The ratemeter sensitivity is adjustable over the range of preamplifier output. Gammas with energies in the range of 100 Kev to 3 Mev can be detected.

The monitor contains two separate calibration check sources, a remotely operated radiation check source for calibrating the detector, and an internally generated count rate signal for calibrating the logarithmic ratemeter.

A recorder in the control room receives electrical signals from the two logarithmic ratemeters and records the radioactivity in the blowdown from each steam generator.

Each ratemeter activates a high radiation alarm in the control room and provides a signal to trip closed the steam generator blowdown and sample valves, FCV-23 - 3 and 23 - 7 if the 1A Steam Generator is affected and valves FCV-23-5 and 23 - 9 if the 1B Steam Generator is affected. The high radioactivity alarm set point is adjustable over the full range of the ratemeter.

Since blowdown water normally contains no radioactivity and very low chemical concentration, it is normally discharged directly to the circulating water canal. The blowdown radioactivity monitor alarm/trip set point is selected to assure that applicable discharge limits are not exceeded.

The monitor has an alarm and trip system controlled by the logarithmic ratemeter. The high radioactivity set point is adjustable over the full range of the ratemeter. The set point is determined as a result of sample analysis prior to the discharge of any fluid from the blowdown tank. The analysis must verify that the concentration of radioactivity to be discharged is less than applicable release limits. The alarm and trip set point of the logarithmic ratemeter is set slightly above the activity determined from the gross radioactivity analysis but below the applicable release limits.

11.4.2.5 Condenser Air Ejector Monitor

The condenser air ejector gas monitor measures noncondensable fission product gases in the condenser air ejector discharge to detect any primary to secondary leakage. Activity levels are recorded in the control room and alarms annunciated when the activity level exceeds predetermined limits. The channel consists of an off-line effluent sampler, a beta scintillation detector, a Cs-137 check source, a logarithmic ratemeter and power supply.

The sampler is shielded with 3" of lead. The operation limit for pressure is 50 psig and temperature limit 125 F. The ratemeter covers a range of 10-10^6 cpm with a meter accuracy of ±2 percent full scale. The channel activity level is recorded in the control room.

The ratemeter has lights to indicate detector failure, high radiation level warning and to indicate a high-high radiation alarm. A positive displacement, direct drive, dry vane 4 cfm pumping system is provided with this channel.

11.4.2.6 Component Cooling Water Monitor

The component cooling water monitors are designed to provide an indication to operation personnel whenever the activity in the component cooling system reaches or exceeds a preestablished level above the normal radiation level.

Each monitor consists of a shielded, gamma scintillation detector, a Cs-137 check source, preamplifier and ratemeter. The monitor dynamic range is 10^{-7} µCi/cc - 10^{-2} µCi/cc, with a sensitivity of 1.0 x 10^{-7} µCi/cc (Cs-137) in 1 mr/hr Co-60 backround. The monitor is of the offline type.

The detectors sample downstream of each of the component cooling system heat exchangers so that any possible release of radioactivity via a tube leak is monitored. This channel is annunciated in the control room whenever an abnormal level of activity is detected. The effected heat exchanger may then be isolated and removed from service by closing isolation valves.

The activity levels are recorded in control room. The alarm set points are set at a slightly higher radiation level than normally would be expected.

11.4.2.7 TMI Noble Gas Effluent Radiation Monitors

In compliance with the requirements of Section II.F.1 of NUREG-0737, a Noble Gas Effluent Monitoring System designed to function during post accident conditions as well as during normal operation has been provided. The Noble Gas Effluent Radiation Monitoring System is designed to measure the radioactive releases to the environment through four effluent paths and one accident release path.

The conformance of the high range effluent radiation monitoring system with the requirements of NUREG-0578, its clarifications and NUREG-0737 is detailed in Table 11.4-3. This table includes the criteria, safety

evaluation and remarks, and applicable guidance from NUREG-0737, Item II.F.1.

11.4.2.7.1 Design Basis

a) Potential pathways for the release of large quantities of gaseous activity following an accident are provided with a high range continuous noble gas monitor. These pathways are provided with the means for collecting iodine and particulate samples for laboratory analyses. Release points from the plant are shown on Figure 11.2-5.

b) Range of the accident effluent monitors exceeds postulated release concentrations resulting from an accident.

c) Range of the effluent monitors overlap with the range of the normal effluent monitor at the same location.

d) The monitors shall continue to function both during and after an accident.

e) The following effluent monitors are provided:

1 - Plant Vent Stack
2 - Fuel Handling Building Stack
3 - ECCS Area Ventilation System Exhaust (Train A & B)
4 - Atmospheric Steam Dump Exhaust

11.4.2.7.2 Design Description

Each noble gas effluent channel is equipped with a detector and its associated electronics. Each monitor has a dedicated operator interface. A schematic of the Noble Gas Effluent Monitoring System is shown in Figure 11.4-1. EC284322

The noble gas effluent monitors have dedicated operator interfaces that provide operational status and data and short term data storage (long term storage of radiation measurements is provided by an external system). Status conditions are annunciated by electronics within each channel. Ultimately, annunciation of channel status for all of the noble gas effluent monitors is via a common, multiple input annunciator window. A failure in any of these noble gas channels will not affect the remainder of the noble gas effluent monitors. EC284322

The effluent radiation monitors have normal to high range noble gas detectors. The Fuel Handling Building Stack Monitor has a normal range detector with provisions for continuous particulate and iodine sampling. The Plant Vent Stack Monitor and the ECCS Area Ventilation Exhaust Monitors have provisions for continuous particulate and iodine sampling. EC284322 EC284322

The following is a description of the monitors that are provided.

a) Plant Vent Stack Monitor

The plant vent stack effluent radiation monitor is a noble gas monitor with continuous particulate and iodine sampling capability. The accident monitor extends monitoring from normal monitor range, 10^{-7} to 10^{-2} µCi/cc, to the upper limit of 10^{5} µCi/cc of Xe-133 equivalent. The nozzle orifice size and sample flow rate were originally selected to provide an isokinetic sample velocity. However as discussed in Safety Evaluation PSL-ENG-SENS-00-108, an isokinetic sample velocity is not necessary to achieve representative sampling in this application. 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 Specifications limits.

b) Fuel Handling Building Stack Monitor

The FHB stack effluent radiation monitor is a three-stage particulate, iodine, and noble gas monitor with continuous particulate and noble gas sampling capacity high range of noble gas being 10^{5} µCi/cc of Xe-133 equivalent. The nozzle orifice size and sample flow rate were originally selected to provide an isokinetic sample velocity. However as discussed in Safety Evaluation PSL-ENG-SENS-00-108, an isokinetic sample velocity is not necessary to achieve representative sampling in this application. The primary purpose of this monitor is to continuously monitor and record the radioactivity level of effluent gases being released via the FHB stack.

c) ECCS Area Ventilation System Exhaust Monitors

Two ECCS ventilation system exhaust monitors measure the airborne effluent from the ECCS area. The monitor consists of a three stage monitor with a noble gas range of 10^{5} µCi/cc. The nozzle orifice size and sample flow rate were originally selected to provide an isokinetic sample velocity. However as discussed in Safety Evaluation PSL-ENG-SENS-00-108, an isokinetic sample velocity is not necessary to achieve representative sampling in this application. (See Figure 9.4-5).

d) Atmospheric Steam Dump Exhaust Monitor

The atmospheric steam dump exhaust monitor consists of energy compensated GM tubes mounted externally to each main steam line upstream of the discharge valves. (See Figure 11.4-2).

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.

Radiation levels are corrected for low energy gammas and have a range greater than 4×10^{2} µCi/cc of Xe-133 equivalent.

11.4.3 SAMPLING

Periodic sampling is required for a performance check on some of the process equipment, and to alert the operator to any abnormal condition that may be developing. The samples required are local liquid samples and gaseous samples taken directly by the gas analyzer. The location for these samples are as follows:

a) Local Liquid Samples

Flash tank influent and effluent

Preconcentrator filter influent and effluent

Preconcentrator ion exchanger influent and effluent

Boric acid concentrator distillate and bottoms*

Boric acid condensate ion exchanger influent and effluent*

Boric acid condensate tank recirculation*

Boric acid holding tank recirculation*

Equipment drain tank recirculation

Chemical drain tank recirculation

Laundry drain tank recirculation

Waste filter influent and effluent

Waste condensate tank recirculation

Waste ion exchanger discharge

Waste concentrator distillate and bottoms*

Liquid effluent discharge to the circulating water canal

b) Gas Analyzer Samples

Volume control tank

Flash tank

Holdup tank

Spent resin tank

Containment vent header

Gas surge tank

Gas decay tank

*Note: These components are not sampled since they are no longer used.

Additional samples are also taken at various strategic locations in the plant but these sample stations are connected directly to the sampling system and are discussed in Section 9.3.2.

The basis for selecting the above locations for sampling stations is to check the performance of process equipment thus maintaining the required chemistry throughout the operating plant. For liquid samples the expected composition of the nuclides taken at a specific point should be consistent with the specific activities and decontamination factors as listed in Section 11.2.

The frequency of taking samples from the various locations along with the analysis to be performed is listed in the ODCM. The frequency of sampling for the gaseous content is used as a preliminary guide and may be increased or decreased as operating personnel become more familiar with the plant. Gas samples to the gas analyzer are analyzed for gross oxygen concentration and alarms warn the operator if a potential explosive mixture exists at the sample point. A connection on the discharge side of the gas analyzer is available for taking a grab sample for isotopic analysis.

When obtaining local liquid samples, it is important to obtain representative samples. All local sample points are taken from vertically run pipe or from the top or side of horizontal run pipe. The local sample lines are as short as possible and the sampling isolation valves are located as close to the source as possible to limit the amount of purge water required before a representative sample is obtained.

11.4.3.1 Preconcentrator and Waste Filters

The influent and effluent samples are periodically taken and analyzed to ensure that the filters are removing suspended solids. Usually, a sample of suspended solids is obtained by filtering the liquid sample through a millipore filter. A count of filtered activity (gross count) is made and an analysis of crud weight is done by visual comparison with a known standard or the sample is dried and weighed. Only the portion of activity in particulate form is of direct interest, but dissolved activity would also normally be measured. Crud weight concentration at the filter inlets is normally 10 to 20 ppb but could be much higher at times. Effluent activity crud concentrations will normally be only a small percentage of influent values. The filters are normally changed on high radioactivity or high differential pressure. They are also replaced if testing reveals low decontamination factor (DF).

The procedures and calculations for determining suspended solids activity and crud weight are On-Site Review Group approved plant procedures, generated from: vendor procedures (CE, Westinghouse), ASTM procedures and EPA Standard Methods.

11.4.3.2 Flash Tank

The influent and effluent dissolved fission gas samples can be taken to verify the performance capability of the flash tank. The flash tank is operated when RCS activity is above a pre-established threshold or if the nitrogen blanket in the holdup tanks is lost due to a system malfunction. Plant procedures are available to recover from low nitrogen pressure in the holdup tanks. If the fission and hydrogen gas concentrations are not reduced in the holdup tanks when the flash tank is in operation, the flash tank controls would be checked and maintenance performed. If it is necessary to continue processing letdown waste, the flash tank can be routed to the holdup tanks. The contents of the holdup tanks can be recycled later through the flash tank as operational conditions permit.

11.4.3.3 Preconcentrator Ion Exchanger

Influent and effluent samples can be taken to determine the ion exchanger decontamination factor and to determine performance effectiveness. The decontamination factor is determined by measuring the gross beta-gamma activity of coolant entering the ion exchanger and comparing it to the gross beta-gamma activity of coolant leaving the ion exchanger. Influent and effluent concentration changes (by nuclide) are identified in Table 11.2-4. Measurements for lithium content can also be made.

Procedures for the determination of lithium in water are On-Site Review Group approved plant procedures, generated with input from: vendor procedures (CE, Westinghouse) ASTM procedures and EPA Standard Methods.

11.4.3.4 Boric Acid Concentrator

NOTE: Information about the Boric Acid Concentrators is maintained here for historical purposes only, since this item and supporting components (including those of Sections 11.4.3.5, 11.4.3.6 and 11.4.3.7) are no longer used.

The boric acid concentrator bottoms and distillate samples are obtained to determine if the unit is functioning as designed. The bottoms sample verifies that the boric acid is at the desired concentration required for transfer to the boric acid makeup tanks in the chemical and volume control system. This analysis will assure that the concentrator controls are operating adequately and that the automatic operation of the unit is satisfactory. The distillate is sampled to verify that the boron carryover is minimal and to estimate ion exchanger exhaustion rate.

By sampling both the bottoms and distillate for gross beta-gamma activity the concentrator DF can be verified. The same approach is used for the preconcentrator ion exchanger.

Procedures for the determination of boron measurement are FRG-approved plant procedures, generated with input from: vendor procedures (CE, Westinghouse), ASTM procedures and EPA Standard Methods.

11.4.3.5 Boric Acid Condensate Ion Exchanger

Influent and effluent samples can be taken to determine the effectiveness for boron removal and to verify the performance capability. Liquid samples are taken from the boric acid condensate tanks prior to discharge to the circulating water discharge, which would indicate functionability of the boric acid condensate ion exchangers. An increase in boron content or activity would indicate malfunction of the ion exchanger. The ion exchanger is designed for long operation periods and any carryover would be extremely infrequent. If high boron or high activity breakthrough should occur the ion exchanger resin will be replaced. The contents of the tank would then be recycled through the recharged ion exchanger prior to discharge, if necessary.

11.4.3.6 Boric Acid Condensate Tank

The contents of the boric acid condensate tank are recirculated, a representative sample of the tank contents is obtained, and a determination of the isotopic inventory is made. A sample is obtained by recycling the contents of the tank until a homogeneous mixture is attained and then a sample is drawn off the recirculation line. If the analysis indicates that release can be made within permissible limits, the quantity of activity to be released is recorded on the basis of liquid volume in the tank and its activity concentration. If release cannot be made within permissible limits, the waste is recycled for further processing. The concentrations of various nuclides are indicated in Tables 11.2-6 and -7.

11.4.3.7 Boric Acid Holding Tank

A sample is obtained by recirculating the contents of the tank to achieve a homogeneous mixture prior to drawing off a sample in the recirculation line. When concentration of boron matches that in the boric acid makeup tank the liquid can be transferred to the makeup tank. This station can also be used for taking a sample for activity analysis.

11.4.3.8 Equipment Drain Tank and Aerated Waste Storage Tank

Prior to drawing off a sample for analysis the contents of the tanks are recirculated to achieve mixing. The analytical results will determine if the liquid can be either processed by the waste concentrator or boric acid concentrator, or if activity is below the release limits of 10 CFR 20 it can be directly discharged via the circulating water discharge.

11.4.3.9 Chemical Drain Tank

The contents of the chemical drain tank are recirculated to achieve mixing prior to drawing off a sample.

The only analysis that is required is a gross beta-gamma count.

11.4.3.10 Laundry Drain Tank

The sampling and analytical techniques to be used for this tank is identical to that for the chemical drain tank.

11.4.3.11 Waste Concentrator

NOTE: Information about the Waste Concentrator is maintained here for historical purposes only, since this item and supporting components are no longer used.

The bottoms and distillate of the waste concentrator can be sampled for boron content and gross beta-gamma count to check the performance of the unit.

11.4.3.12 Waste Ion Exchanger

The discharge of the waste ion exchanger is sampled and analyzed for gross beta-gamma count. The sample is drawn off at the provided connection for analysis.

The sample analyzed at this point along with the sample for the distillate on the waste concentrator will determine the performance of the ion exchanger.

11.4.3.13 Waste Condensate Tank

The contents of the waste condensate tank are recirculated to achieve mixing prior to sampling. The samples are analyzed for a gross beta-gamma count prior to discharging the contents to the circulating water canal. However, if the count exceeds the acceptable limit the liquid is pumped to the waste concentrator or waste ion exchanger for additional processing.

11.4.3.14 Gas Analyzer

A gas analyzer is used to determine the concentration of oxygen in the gas to be sampled from the various sources as given in Section 11.4.3. The gas analyzer has an automatic timing device that periodically samples the gas from the tanks and vents by opening and closing appropriate solenoid valves. After the system is lined up to a particular sampling station and flow has started, entrained water is removed from the sample by flowing through a water separator trap. The excess water is drained from the trap and sent to the equipment drain tank, while the gas is pumped to a refrigeration unit where saturated water is removed. From the refrigeration unit the gas flows to the oxygen sensing device, and then another pump is used to discharge the gas to the gas surge header. The oxygen sensing element is installed in parallel with a flow meter installed upstream of each sensor so that the operator can manually balance the flow to each sensor. If a potentially explosive mixture is being approached, the gas analyzer alarm will annunciate in the control room to alert the operator that immediate action is required. An explosive mixture can be avoided by either purging or diluting with nitrogen. EC282237

Calibration gas for the gas analyzer is required. A zero gas (prepurified nitrogen) is required for the oxygen sensor. This gas is required to periodically calibrate the instrument. Oil-free air is required to purge the sensors prior to and after calibrating.

Sample connections are available in the system downstream of the pump that returns the gas to the gas surge header where a sample container (1 liter) is attached for taking a sample for isotopic analysis. An option is available to the operator for either bypassing the sensors to take the radioactive sample or the radioactive sample can be the same gas that has passed through the sensors for the oxygen determination.

11.4.4 INSERVICE INSPECTION, CALIBRATION AND MAINTENANCE

All radiation monitoring systems are calibrated periodically according to the schedule shown in the Technical Specifications and the ODCM. Any maintenance required will be conducted in accordance with the instructions provided by the instrument manufacturer.

The arrangements for obtaining independent audits and verification of the process and effluent monitoring system calibration will be made by the Quality Assurance Department. Details regarding this activity are outlined in Section 17.2, "Quality Assurance During the Operations Phase".

TABLE 11.4-1
PROCESS RADIATION MONITORS

Monitor	Type Detector	Minimum Sensitivity**	Isotope	Range	Maximum Area Radiation Level Detector	Remarks
Chemical and Volume Control System Process Radiation Monitor	Gamma Scintillation crystal with photomultiplier tube preamplifier	10^{-4} µCi/cc	I-135	10^{-4} to 10^{2} µCi/cc	5 mR/hr gamma Co-60	The setpoint can be adjusted as operations dictate
Waste Management System Liquid Effluent	Gamma Scintillation crystal with photomultiplier tube and integral preamplifier	1 x 10^{-7} µCi/cc	Cs-137	10^{-7} to 10^{-2} µCi/cc	0.1 mR/hr gamma Co-60	The monitor is adjusted to a setpoint following an isotopic analysis of the liquid waste to be discharged. The monitor will verify the analysis by alarming should the discharge activity exceed the predetermined value.
Waste Management System Gaseous Effluent	Beta Scintillation Detector	2 x 10^{-5} µCi/cc	Xe-133	10^{-5} to 10^{+3} µCi/cc	0.1 mR/hr gamma Co-60	The monitor is adjusted to a setpoint following an isotopic analysis of the gas to be discharged. The monitor will verify the analysis by alarming should the discharge activity exceed the predetermined value.
Steam Generator Blowdown	Gamma Scintillation Crystal with Photomultiplier Tube and Integral Preamplifier	1 x 10^{-6} µCi/cc	Cs-137	10^{-6} to 10^{-1} µCi/cc	0.1 mR/hr gamma Co-60	The monitors are adjusted to a setpoint following an isotopic analysis of liquid phase of the secondary side of the steam generator
Condenser Air Ejector	β Scintillation Crystal	9 x 10^{-5} µCi/cc	Xe-131	9(10^{-5}) to 4.5(10^{-2}) µCi/cc	0.1 mR/hr	*
Component Cooling Water	γ Scintillation Crystal	1 x 10^{-7} µCi/cc	Cs-137	10^{-7} to 10^{-2} µCi/cc	1.0 mR/hr	*

* Normally clean systems. If leakage across system heat exchangers occurs, setpoints are adjusted in order to provide early notification of leakage.

**Note: Minimum sensitivities based on purchase specification values. Radiation monitor performance is equal to or better than purchase specification value.

TABLE 11.4-2A

SAMPLE LOCATIONS AND ANALYSES - MONITORING OF
LIQUID & GASEOUS BATCH RELEASES EFFLUENTS

Refer to Offsite Dose Calculation Manual (ODCM) Chemistry Operating Procedure No. C-200.

TABLE 11.4-2B

SAMPLE LOCATIONS AND ANALYSES

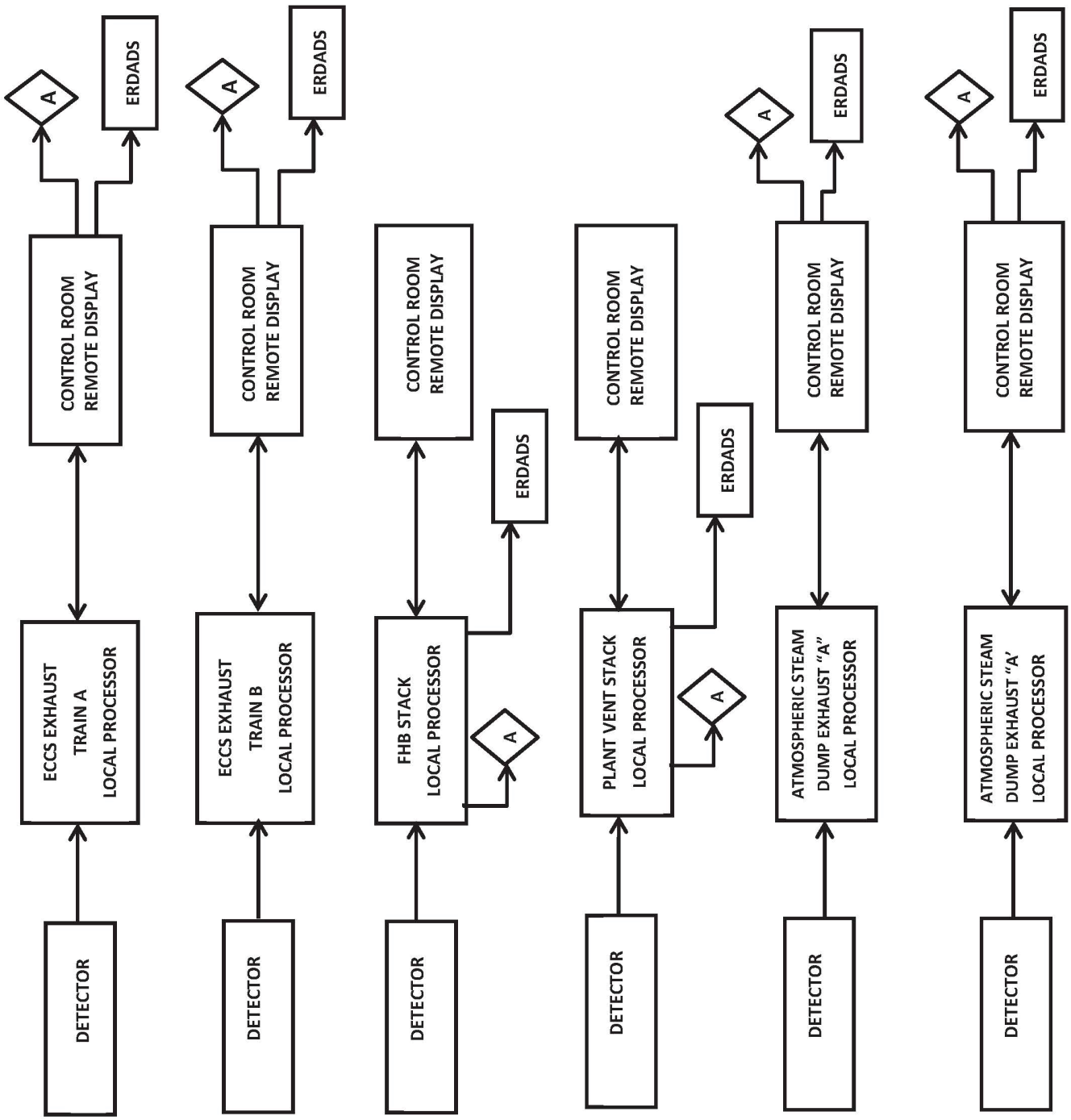
Location	Analysis
Preconcentrator Filters	Isotopic or Gross Activity
Preconcentrator Ion Exchangers	Isotopic or Gross Activity
Boric Acid Concentrator Bottoms and Distillate*	Isotopic or Gross Activity
Boric Acid Condensate Ion Exchangers*	Isotopic or Gross Activity
Waste Filter	Isotopic or Gross Activity
Waste Ion Exchanger	Isotopic or Gross Activity
Waste Concentrator Bottoms and Distillate*	Isotopic or Gross Activity
Flash Tank	Isotopic or Gross Activity
Equipment Drain Tank	Isotopic or Cross Activity
Chemical Drain Tank	Isotopic or Gross Activity
Holdup Tanks	Isotopic or Gross Activity

*Note: These components are not sampled since they are no longer used.

TABLE 11.4-3

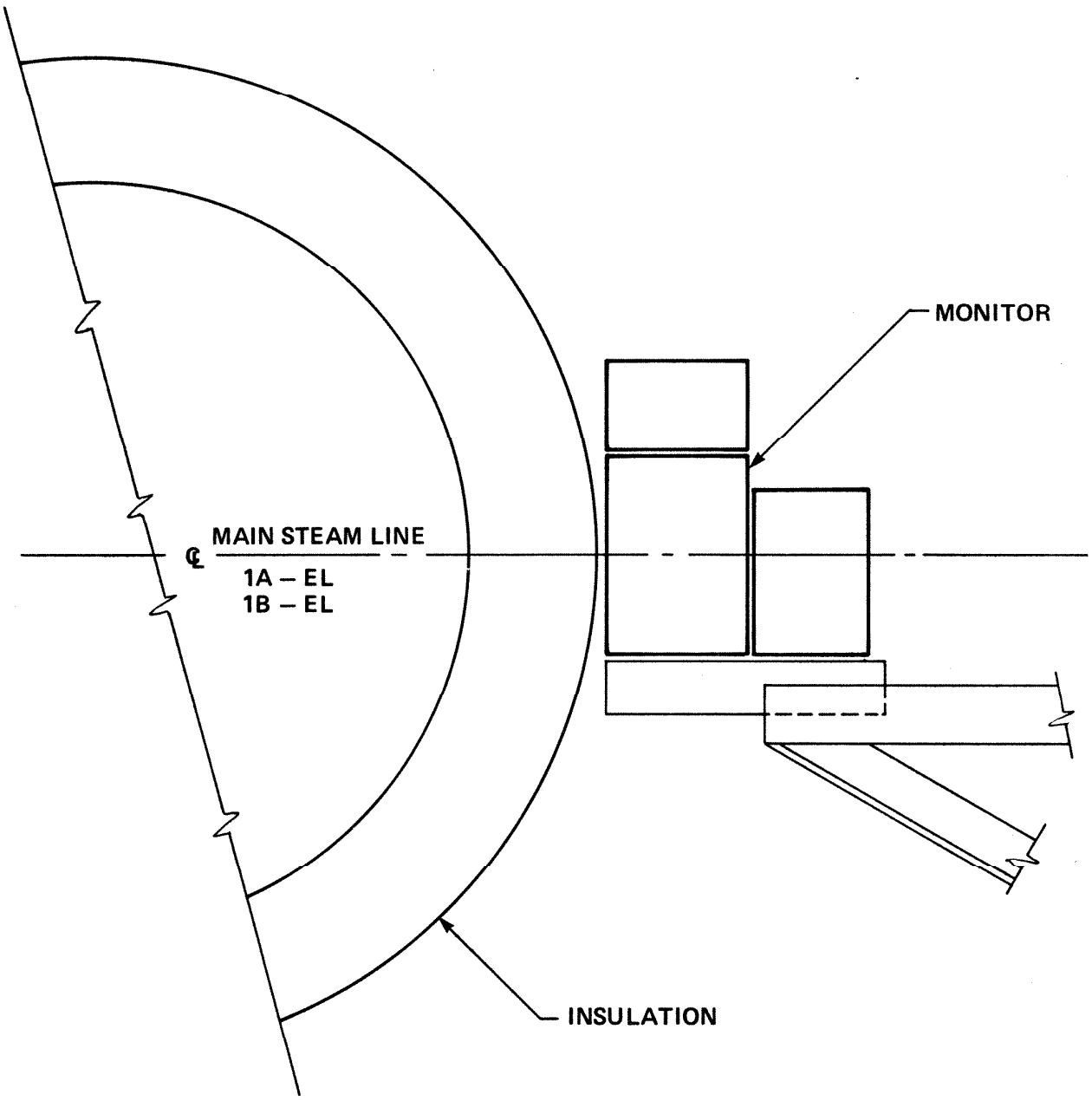
NOBLE GAS EFFLUENT RADIATION MONITORS
CONFORMANCE WITH TMI RELATED CRITERIA

Criteria	Safety Evaluation - Remarks	Reference
Monitoring of potential pathways for noble gas releases and iodine and particulate releases	The plant stack, ECCS, and Fuel Handling Building monitors have filtering for iodines and particulates and high range (10^5 µCi/cc) for the noble gases. The nozzle orifice size and sample flow rate were originally selected to provide an isokinetic sample velocity. However as discussed in Safety Evaluation PSL-ENG-SENS-00-108, an isokinetic sample velocity is not necessary to achieve representative sampling in this application.	Position, Clarification #1, Table II.F.1-1 Position, Changes #4, #3, Clarification #1, #3, Table II.F. 1-2
Range of noble gas monitors exceeds accident release	All noble gas ranges are 10^5 µCi/cc, in excess of require-ments. Expected maximum release is less than 10^3 µCi/cc.	Position, Clarification #2, #4, Table II.F.1-1
Effluent range overlaps with normal range	All monitors cover from ALARA to µCi/cc of noble gases.	Position, Clarification #4,, Table II.F.1-1



FLORIDA POWER & LIGHT COMPANY
ST. LUCIE PLANT UNIT 1

NOBLE GAS EFFLUENT RADIATION MONITORING SYSTEM BLOCK DIAGRAM
FIGURE 11.4-1



FLORIDA POWER & LIGHT COMPANY
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EXTERNALLY MOUNTED ATMOSPHERIC
STEAM DUMP EXHAUST MONITOR
FIGURE 11.4–2

11.5 SOLID WASTE SYSTEM

The Solid Waste Management System (SWMS) collects, controls, processes, packages, and temporarily stores solid radioactive waste and certain liquid radioactive waste generated as a result of normal plant operations, including anticipated operational occurrences. 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 1.

Operation at an assumed conservative core thermal power of 3030 MWt is expected to have minimal impact on waste quantities, on installed equipment performance, system operation, and maintenance. Activity levels for most of the solid waste are expected to vary proportional to the power level.

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

11.5.1 DESIGN OBJECTIVES

The solid waste system plan shown on Figures 1.2-13 and 11.2-4 is designed to collect and prepare solid radioactive wastes and certain radioactive liquid wastes for interim storage within the low level waste storage facility and/or for offsite shipment. The waste forms and quantities to be processed are given in Table 11.5-1. The waste activities to be processed are given in Tables 11.5-4 and 11.5-6.

11.5.2 SYSTEM DESCRIPTION

The solid waste system processes radioactive waste in the form of ion exchangers resins and filters, compressible and non-compressible solids. Ion exchanger resins are sluiced into the spent resin tank or shipping container, and dewatered in accordance with plant procedures. Filters are moved from their vessels into shipping containers. Compressible waste is compacted by a compactor located inside the drumming station or shipped sorted, uncompacted to an offsite radioactive waste volume reduction facility for processing. Non-compressible waste is packaged in boxes or bags. All of these wastes are packaged and shipped offsite in accordance with plant procedures.

A portable dewatering system is used when processing and packaging wet radioactive resin wastes.

The ion exchanger resin activity buildup was based on inlet activity concentration in Table 11.1-1, the maximum removal of each nuclide for each process ion exchanger at the expected flow rate through each ion exchanger.

For the purpose of this analysis, the following items were assumed prior to initial plant operation. CVCS ion exchanger(s) used for purification is (are) assumed to continuously process 40 gpm letdown over the full core cycle. The CVCS ion exchanger used for lithium removal is assumed to process 40 gpm letdown over 20 percent of the core cycle which is the expected usage of the unit to control lithium concentration. The CVCS ion exchanger(s) used for deboration is (are) assumed to process letdown over approximately 2.4 percent of the core cycle. This is the expected usage of the unit to remove the final 30 ppm of boron in the reactor coolant to compensate for fuel burnup.

Each of the two WMS preconcentrator ion exchangers were assumed to process one half of the 843,000 gallons of clean waste per year. The waste was assumed to be letdown reactor coolant which has already passed through a CVCS ion exchanger.

The boric acid condensate and the waste ion exchangers were not expected to retain a significant amount of activity due to the very low concentration of activity in the inlet process fluid.

The fuel pool ion exchanger was assumed to remove the portion of reactor coolant system end of life activity which enters the fuel pool water, assuming complete mixing, after being diluted by the refueling water.

The estimated activities on these ion exchangers by nuclide after one core cycle of operation with 1 percent failed fuel (anticipated operational occurrence)are given in Table 11.5-4. Normal operation with 0.1 percent failed fuel would result in 10% of the activity given for anticipated operational occurrence. Actually the CVCS purification bed activities represent the curies accumulated in two years of operation with one percent failed fuel. During the first year of operation, the resin is gradually converted from the $H\text{-}BO_3$ form to the $Li\text{-}BO_3$ form as lithium, produced by the B^{10} (η, α) Li^7 reaction in the core, is controlled below its upper concentration limit. During this time, monovalent cations such as cesium are removed by this bed from the coolant. By the end of one core cycle (~ 20 percent usage factor) this bed becomes saturated with lithium. Although the monovalent cations on the bed are retained, the bed loses capability to remove these cations from the coolant.

Estimated activities on the various ion exchanger resin beds after six months and one year decay are shown on Table 11.5-5. Cesium-137 dominates the activity that must be shipped, but without significant failed fuel, Co-60 would dominate. Assuming eight beds are disposed of per year (one purification, one deborating, two preconcentrator, two boric acid condensate, one waste condensate, and one fuel pool purification) the mixed activity would be about 23,000 Ci or 90 Ci/ft^3. Packaging requirements for this activity concentration and resin quantity are defined in Section 11.5.5.

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 operating cycles. Increased resin and filter usage during the first two cycles will occur, but resin depletion during cycle operation is not expected.

Filter vessels used in removing particulates from process streams are installed behind labyrinthed concrete cubicles roofed over by a concrete slab, except for the laundry filter which is not expected to have high levels of radioactivity. The roof slab is fitted with a removable concrete hatch. Holes in the hatch permit long handled tools to operate valves for draining vessel and to undo head bolt nuts to permit vessel head to swing open in a vertical arc. A monorail system above the cubicle roof allows for removing the hatch and setting it aside. Filters of high activity levels require a shielding device simulating an inverted bell to be positioned over the opened hatch. A winch device and appropriate opening allow the filter cartridge to be withdrawn from the filter vessel and into the cartridge (inverted bell) shield. The cartridge shield is placed on a transport device and removed to the drumming area where the cartridge will be prepared for disposal.

Low level activity cartridges may be removed without the use of a cartridge shield. Upon removal through the hatch it would be deposited in a shipping container positioned to receive the filter. The container would then be removed to the drumming area for final preparation for disposal.

The activity level on the filter elements is based on a conservatively high inlet concentration, the maximum removal for each nuclide and the expected flow rate through each unit. The purification filters 1A and 1B are each assumed to process 40 gpm in series (1A upstream and 1B downstream of ion exchangers) continuously over the entire core cycle. The fuel pool purification filter is assumed to remove the portion of reactor coolant system end of life activity which enters the fuel pool water, assuming complete mixing, after being diluted by the refueling water. A temporary portable filter/vacuum is also available to expedite outage activities and to help cleanup the fuel pool purification filter to improve water clarity. When depleted, filters can be remotely changed out while the unit remains underwater and stored in a remote floor mounted or hanging filter storage rack. The waste management system waste filter and the laundry filter are not expected to retain a significant amount of activity due to the very low concentration of activity in the inlet process fluid and the low quantity of liquid each unit must process. Each of the two preconcentrator filters are assumed to process one half of liquid waste processed by the boron recovery system. This waste is assumed to be reactor coolant which has been processed by the chemical and volume control system.

The estimated activity on these filters after one core cycle with source activities equivalent to one percent failed fuel is given in Table 11.5-6. These same filter activities after six months and one year decay are shown in Table 11.5-7. The total activity for a given filter may be divided between a number of cartridges that could be packaged separately. Packaging requirements for filter cartridges based on the activities of Table 11.5-7 are defined in Section 11.5.5.

Miscellaneous compressible solid waste such as contaminated clothing, plastic sheeting, and tape, accumulated as a result of health physics and maintenance activities and non-compressible solid waste such as tools and contaminated equipment are stored in designated areas and then shipped offsite in accordance with plant procedures.

Solidification of radioactive waste and processing of irradiated components are not activities routinely performed at the plant. However, if the need arises to perform these activities, plant procedures will be developed in accordance with Process Control Program requirements to ensure safety.

11.5.3 EQUIPMENT DESCRIPTION

A data summary of the solid waste system components is given in Table 11.5-8.

The spent resin tank can hold the equivalent of approximately eight to ten beds of spent resin from the various plant ion exchangers, and therefore, storage capacity in excess of one year is normally available. Higher activity resin will normally be sluiced to a shielded shipping container, but sluicing to shielded disposable container is also possible.

To sluice resin from the spent resin tank, water from the holdup pumps or primary makeup water enters the tank. The tank fills with water compressing any air in the tank forcing a water and resin slurry up the tank outlet line. The discharge pipe extends upward from near the tank bottom thus minimizing chances of resin binding while allowing efficient removal of nearly all the resin from the tank. A conductance type level sensor allows level to be monitored. The resin and water slurry enters the shipping container and the water is drawn off by a portable dewatering system. Ion exchanger resins can be sluiced into the spent resin tank or directly into a shipping container in accordance with plant procedures.

Before sluicing resin from an ion exchanger to the spent resin tank, the tank is vented and partially drained of water into the ECCS pump room sump. After resin is sluiced in,

the vent valve is closed.

Fresh ion exchange resin is added to the various ion exchangers by connecting the resin addition tank, essentially a funnel, to the resin fill flange and pouring the as-received resin directly into the ion exchanger vessel.

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

A compactor located inside the drumming room could be used for processing compressible wastes. The compactor is provided with HEPA filters to remove potentially radioactive airborne particulates.

No special equipment is employed for handling miscellaneous non-compressible solid waste.

The installed resin dewatering pump and waste baler compactor are no longer used.

11.5.3.1 Component Failure and System Malfunction Evaluation

The failure of any component in the solid waste system should not result in a direct "shine" dose to operating personnel because the floor drains, personnel evacuation, proper shielding, and the general remote location of switches and valves should isolate personnel from the accident and provide adequate shielding. However, the failure of some components in the solid waste system could release airborne activity. For example, failure of the spent resin tank could result in the release of noble gases produced from iodine decay. The following summarizes the results of the failures:

a) Failure of the spent resin tank could result in the release of 700 equivalent curies of Xe-131m with the following assumptions:

   1) The maximum Xe buildup occurs and is released instantaneously.

   2) The spent resin tank contains the combined resin volumes from the CVCS ion exchangers, and one fuel pool ion exchanger.

b) Failure of the spent resin tank filled with sluice water from the holdup tank could release the following activity:

   1) 470 equivalent curies of Xe-133.

   2) 3×10^{-5} equivalent curie of I-131.

The malfunction of any component in the solid waste system should not result in a "shine" or airborne dose to operating personnel. The system is designed for full backflushing capability and has remote handswitches and panels. Local shielding is provided where necessary and any auxiliary shielding can be temporarily installed when necessary. The following areas can be classified as high radiation zones which could result in a potential exposure to maintenance crews during system malfunction: (Isotope distribution is presented in Table 11.5-4)

a) The spent resin tank could contain one year's accumulation of spent resin equivalent to approximately 23,000 curies after 1 year of decay.

b) A single 55 gallon shipping container could contain approximately 600 curies after 1 year of decay in the spent resin tank.

11.5.4 EXPECTED VOLUMES

The following information respresents estimates which were developed based on industry experience and provided prior to initial plant operations. Actual solid waste volumes and activities are reported to the NRC in the Annual Radioactive Effluent Release Reports for their review.

The ion exchange resins are dewatered prior to placement in containers and stored within the Low Level Waste Storage Facility and/or shipped. Therefore, the stored/shipped volume is approximately 256 cubic feet per year with curie content per Table 11.5-5.

Miscellaneous non-compressible solid waste is estimated to range from 500 to 1000 cubic foot per year based on general plant operating experience. The need to dispose of a large component could significantly increase this estimate however. Shipped content ranges from 5 Ci to 15 Ci per year.

11.5.5 PACKAGING

The ion exchange resins, filters, compressible and noncompressible wastes are packaged, temporarily stored, and shipped in accordance with plant procedures to comply with 10 CFR and 49 CFR regulation. The waste is stored in high integrity containers (HICs) onsite within the Low Level Waste Storage Facility. The HICs are compatible with and inserted into shielded transport containers for transport offsite.

11.5.6 STORAGE FACILITIES

No plans have been made for storage of high level activity waste. Low activity waste is stored temporarily in the drumming station in the reactor auxiliary building with local shielding provided to reduce the dose rates to the operating personnel to below 10 CFR 20 limits. Solid waste is also temporarily stored in containers outdoors at locations designated by Health Physics prior to relocation onsite for interim storage within the Low Level Waste Storage Facility and ultimately for final shipment. Measures are established for these locations to minimize personnel radiation dose and to preclude the spread of contamination.

Additional storage is provided for dry active waste (DAW) in the Dry Storage Warehouse as described in safety evaluation JPN-PSL-SECP-92-055, consistent with Generic Letter 81-38. However, storage of DAW in this warehouse requires St. Lucie Management authorization.

11.5.7 SHIPMENT

Solid radwastes are shipped from the site by truck operated by carriers which are licensed contractors operating in accordance with Department of Transportation and other applicable regulations.

Shipping containers or vehicles are not normally stored on site. Short-term storage can occur on site, only at locations selected such that the security and radiological surveillance procedures for plant operation are met with respect to the activity involved. Longer term interim storage is also available on site within the Low Level Waste Storage Facility (LLWSF) until final shipment offsite can be made.

TABLE 11.5-1
(HISTORICAL)
ESTIMATED INPUTS TO SOLID WASTE SYSTEM PER CORE CYCLE
BASED ON CONTINUOUS OPERATION WITH 1 PERCENT FAILED FUEL

Source	Quantity (per year)	Operation
IX Resins, dewatered, ft^3	256	sluice ion exchangers once per year
Filter Elements	14 per year	change each filter cartridge twice per year
Miscellaneous Compressible Solid Waste, ft^3	1,000 to 2,500	compact
Miscellaneous Noncompressible Waste, ft^3	500 to 1,000	place in suitable containers

Table 11.5-2

Deleted.

Table 11.5-3

Deleted

TABLE 11.5-4
(HISTORICAL)
ESTIMATED ACTIVITY BUILDUP ON ION EXCHANGER RESINS PER CORE CYCLE
WITH 1% FAILED FUEL (CURIES[1])

Nuclide	CVCS (2) Purification	CVCS Deborating	WMS Pre Conc.(each)	FPS Purification
Br-84	0,455	1.1×10^{-2}	9.1×10^{-6}	5.11×10^{-2}
Rb-88	14	0	2.8×10^{-6}	1.66
Rb-89	0.29	0	5.8×10^{-8}	3.44×10^{-2}
Sr-89	110	0	0.22	0.473
Sr-90	24.5	0	4.9×10^{-2}	3.09×10^{-2}
Y-90	0.215	0	1.08×10^{-2}	4.27×10^{-2}
Sr-91	0.6	0	1.2×10^{-5}	5.94×10^{-2}
Y-91	500	0	25	4.24×10^{-2}
Mo-99	450	0	22.5	63.4
Ru-103	65	0	0.13	0.373
Ru-106	16.5	0	3.3×10^{-2}	2.54×10^{-2}
Te-129	0.47	1.11×10^{-2}	9.4×10^{-4}	6.34×10^{-2}
I-129	1.1×10^{-2}	5.24×10^{-6}	2.2×10^{-7}	7.92×10^{-6}
I-131	14,000	160.5	0.28	297
Te-132	430	8.12	0.86	18.7
I-132	45.5	1.03	9.1×10^{-4}	5.24
I-133	2150	52	4.3×10^{-2}	167
Te-134	0,305	7.2×10^{-3}	6.1×10^{-4}	3.92×10^{-2}
I-134	10	0.24	2.0×10^{-4}	1.18
Cs-134	4000	0	200	222
I-135	330	7.95	6.6×10^{-3}	33.5
Cs-136	37	0	1.85	2.46
Cs-137	18.500	0	925	90.3
Cs-138	1.3	0	6.5×10^{-2}	0.786
Ba-140	34	0	6.8×10^{-4}	0.497
La-140	3.95	0	7.9×10^{-3}	0.258
Pr-143	32	0	6.4×10^{-2}	0.321
Ce-144	265	0	0.53	0.382
Fission Product Total	41,000 Ci(2)	230 Ci	1150 Ci (3)	900 ci
Cr-51	6.0	0.144	1.1×10^{-2}	7.9×10^{-2}
Mn-54	0,300	0	0	0
Co-58	18.9	0.454	3.5×10^{-2}	0.107
Fe-59	5.6×10^{-2}	0	0	0
Co-60	8.30	0.199	1.3×10^{-2}	1.3×10^{-2}
Zr-95	3.49×10^{-3}	0	0	0
Corrosion Product Total:	32.6	0.797	5.9×10^{-2}	0.199
Overall Total:	41,000	231	1150	900

Notes: (1) Activity levels for normal operation (.1% failed fuel) would be 0.1 of the 1% failed fuel values.
(2) Based on operation for two core cycles. See Text.
(3) One-half of total.

TABLE 11.5-5
(HISTORICAL)
ESTIMATED ACTIVITY ON ION EXCHANGER RESINS AFTER DECAY (CURIES)

Nuclide	CVCS Purification		CVCS Deborating		WMS Pre Conc.(each)		FPS Purification	
	6 mo	1 yr	6 mo	1 yr	6 mo	1 yr	6 mo	1 yr
Br-84	0	0	0	0	0	0	0	0
Rb-88	0	0	0	0	0	0	0	0
Rb-89	0	0	0	0	0	0	0	0
Sr-89	9.0	0.736	0	0	1.81(-2)*	1.47(-3)	3.88(-2)	3.17(-3)
Sr-90	24.2	24.0	0	0	4.84(-2)	4.78(-2)	3.05(-2)	3.01(-2)
Y-90	0	0	0	0	0	0	0	0
Sr-90	0	0	0	0	0	0	0	0
Y-91	58.5	6.85	0	0	2.93	0.343	4.96(-3)	5.81(-4)
Mo-99	0	0	0	0	0	0	0	0
Ru-103	2.74	1.16(-2)	0	0	5.50(-3)	2.32(-4)	1.57(-2)	6.65(-4)
Ru-106	11.7	8.25	0	0	2.33(-2)	1.65(-2)	1.79(-2)	1.27(-2)
Te-129	0	0	0	0	0	0	0	0
I-129	1.10(-2)	1.10(-2)	5.24(-6)	5.24(-6)	2.27(-7)	2.2(-7)	7.92(-6)	7.92(-6)
I-131	2.08(-3)	0	2.38(-5)	0	0	0	4.41(-5)	0
Te-132	0	0	0	0	0	0	0	0
I-132	0	0	0	0	0	0	0	0
I-133	0	0	0	0	0	0	0	0
Te-134	0	0	0	0	0	0	0	0
I-134	0	0	0	0	0	0	0	0
Cs-134	3380	2850	0	0	169	142.5	188	159
I-135	0	0	0	0	0	0	0	0
Cs-136	2.18(-3)	0	0	0	1.51(-4)	0	1.16(-4)	0
Cs-137	18,280	18,040	0	0	914	904	89.4	88.4
Cs-138	0	0	0	0	0	0	0	0
Ba-140	1.72(-3)	0	0	0	0	0	2.52(-5)	0
La-140	0	0	0	0	0	0	0	0
Pr-143	3.10(-3)	0	0	0	0	0	3.11(-5)	0
Ce-144	170	109	0	0	0.338	0.216	0.245	0.157
Fission Product Total:	21,900	21,000	2.91(-5)	5.24(-6)	1065	1050	280	250
Cr-51	6.3(-2)	0	1.52(-3)	0	0	0	0	0
Mn-54	0.20	0.133	0	0	0	0	0	0
Co-58	3.18	0.534	7.65(-3)	1.28(-2)	0	0	1.8(-2)	3.05(-3)
Fe-59	0	0	0	0	0	0	0	0
Co-60	7.77	7.28	0.186	0.174	1.22(-2)	1.14(-2)	1.22(-2)	1.14(-2)
Zr-95	0	0	0	0	0	0	0	0
Corrosion Product Total:	11.21	7.95	0.264	0.187	1.22(-2)	1.14(-2)	3.02(-2)	1.45(-2)

Note: * denotes power of 10

TABLE 11.5-6
(HISTORICAL)
ESTIMATED ACTIVITY BUILDUP ON FILTER CARTRIDGE PER CORE CYCLE (CURIES)

Nuclide	CVCS Purification		WMS Pre-Concentrator (each)	WMS Waste	FPS Purification
	1A	1B			
Co-60	40.0	4.00	7.4(-2)*	**	7.33(-2)
Fe-59	0.317	3 .17 (-2)	**	**	**
Co-58	106.3	1.0.63	1.97(-1)	**	6.05(-1)
Mn-54	1.472	0.1472	**	**	**
Cr-51	34.5	3.45	6.3(-2)	**	4.46(-1)
Z r-95	1.98(-2)	**	**	**	**
Total	182	18.2	3.34(-1)	**	1.12

* ( ) power of 10

** less than 10^{-2}

TABLE 11.5-7
(HISTORICAL)
ESTIMATED ACTIVITY ON FILTER CARTRIDGES AFTER DECAY (CURIES)

Nuclide	CVCS Purification 1A		1B		WMS-Pre-Concentrator (Each)		WMS Waste		FPS Purification	
	6 mo	1 yr	6 mo	l yr	6 mo	1 yr	6 mo	1 yr	6 mo	1 yr
Co-60	37.4	35.0	3.74	3.5	6.92(-2)*	6.48(-2)	**	**	6.86(-2)	6.42(-2)
Fe-59	1.90(-2)	**	**	**	**	**	**	**	**	**
Co-58	17.9	3.00	1.79	0.330	3.32(-2)	**	**	**	1.02(-1)	1.71(-2)
Mn-54	0.98	0.655	9.8(-2)	6.55(-2)	**	**	**	**	**	**
Cr-51	0.364	**	3.64(-2)	**	**	**	**	**	**	**
Zr-95	2.82(-2)	**	**	**	**	**	**	**	**	**
Total	56.6	38.7	5.66	3.87	1.024(-1)	6.48(-2)	**	**	1.706(-1)	8.13(-2)

** less than 10^{-2}

* ( ) power of 10

TABLE 11.5-8

DESIGN DATA FOR SOLID WASTE SYSTEM COMPONENTS

1. Drum Roller (This equipment is no longer used.)

Quantity	2
Type	Horizontal, direct drive
Drum size, gal	55
Capacity, lb	1000
Code	None

2. Baler (This equipment is no longer used.)

Quantity	1
Type	Hydraulic, totally enclosed
Drum size, gal	55
Compacting force, psi	8.3
Exhaust fan capacity, CFM	40
Exhaust filter	Prefilter and HEPA filter
Code	None

3. Spent Resin Tank

Quantity	1
Type	Vertical
Volume, gal	3200
Design pressure, psig	50
Design temperature, F	200
Material	55
Code	ASME Boiler and Pressure Vessel Code 1968, Section III, Class C

4. Dewatering Pump (This equipment is no longer used.)

Quantity	1
Type	Self-priming centrifugal
Design flow, gpm	100
Design head, ft	95
Design temperature, F	200
Material	55
Code	None

5. Resin Addition Tank

Quantity	1
Volume, gal	100
Design pressure, psig	Atmospheric
Design temperature, F	N.A.
Material	55
Code	None

11.6 OFFSITE RADIOLOGICAL MONITORING PROGRAM

11.6.1 EXPECTED BACKGROUND

A preoperational radioactivity monitoring program was conducted to determine the magnitude and nature of the radioactivity in the environment surrounding the site prior to the startup of the plant. The information obtained serves as a base line in evaluating any changes in environmental radioactivity level that may possibly be attributed to the plant operation.

The preoperational program was closely coordinated with the State program for monitoring radioactivity levels. Discussions held with State agencies assisted in formulating an acceptable preoperational environmental radioactivity monitoring program. The resulting program is reviewed periodically to assure maximum effectiveness.

The expected annual dose from natural sources is 120 mrem.

11.6.2 CRITICAL PATHWAYS

Details of the potential exposure pathways to man are presented in Section 2.3.7.3 of the St. Lucie Plant Environmental Report.

Reconcentration of specific radionuclides by various trophic levels of marine ecology is considered in the environmental monitoring program by including samples of Indian River and Atlantic Ocean water, fish and shellfish, and bottom sediments. Interpretation of surveillance results are based on selected radionuclides, such as Cobalt 60, Manganese 54, Cesium 137 and Iodine 131. Factors to be considered in the selection of the radionuclides are:

a) The estimated composition and relative concentration of the various radionuclides in the plant liquid effluent

b) The reconcentration factors for these radionuclides in the species of fish and shellfish caught in the Indian River and in the ocean

c) The appropriate limits established by regulatory requirement.

FPL has a program to study marine ecology which includes radioactivity effects. Fish and shellfish samples are analyzed for gross radioactivity and for the controlling radionuclides. Recognition of the reconcentration factors, in conjunction with the 10 CFR 20 requirements, assures that discharges are within acceptable limits for environmental exposure.

11.6.3 SAMPLING MEDIA, LOCATIONS AND FREQUENCY

Sampling stations and sites are established on the basis of population density and distribution, meteorological, hydrological and topography conditions.

A comprehensive sampling program is established including definition of types of samples, number of samples, frequency of collections, and methods of analysis as detailed in the Offsite Dose Calculation Manual.

Marine organisms are collected where available near the site, with emphasis on obtaining samples near the plant discharge.

Bottom sediment samples,are collected at the plant outfall. Surface water samples from the outfall are collected and composited for analysis. Groundwater samples are collected from an appropriate nearby well. Soil samples are collected from the immediate plant environment. Air radiation dosimeters or air particulate samplers are located at various onsite or offsite positions. Selection of these positions considers the prevailing wind directions.

11.6.4 ANALYTICAL SENSITIVITY

The collected samples include samples of river, ocean and well water, soil, air particulates, farm and dairy products, fish and shellfish, and river and ocean bottom sediments from the vicinity of the plant. Sample radioactivity analyses, based on the type of sample and information desired, include one or more of the following: Gross alpha, gross beta, gross beta-gamma, Potassium 40, Iodine 131, Strontium 90 and others as appropriate.

The operational environmental radioactivity is similar to the preoperational program. A sampling and analysis schedule is based on the level of activity found in the plant effluent and in the environment samples. If the gross beta-gamma analysis shows an increase in gross gamma, a gamma spectrum analysis would be made to identify the nuclides present. Results of the sample analyses are evaluated to insure the effectiveness of plant radiation control and compliance with the requirements of all regulatory agencies, both Federal and State. This program is reviewed and revised as required.

Minimum levels of detectability in gaseous and liquid media for the following radionuclides assumes continuous monitoring with the level cited measurable within a four hour period are as shown in Table 11.6-1.

11.6.5 DATA ANALYSIS AND PRESENTATION

Environmental monitoring data shall be evaluated in terms of potential human radiation exposure. The value selected as the permissible dose to individuals living in the vicinity of the plant may be 5 or 10 mrem per year, corresponding to an "as low as reasonably achievable" (ALARA) consensus value.

11.6.6 PROGRAM STATISTICAL SENSITIVITY

Because of the normally fractional level of operations emission compared with background radioactivity in the plant vicinity, the program statistical sensitivity appears to be poor on an absolute activity level basis. Sufficient spectrographic analysis is done on enough samples to provide reasonable assurance that potential human exposure will not exceed 5 mrem per year ±25 percent from plant operations for most nuclides. The minimum detectable dose which could be determined from routine environmental measurements is not sufficiently low for the cesium isotopes, shoreline external dose from liquid pathway, and for I-131, airborne-milk pathway, to meet realistically the above stated sensitivity. Calculations based on measurable higher concentrations at known release points in-plant must be relied upon for these exceptions.

REFERENCES FOR SECTION 11.6

1. U.S. Atomic Energy Commission, WASH 1258, Final Environmental Statement concerning Proposed Rule Making Action: Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion "As Low As Practicable" for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents, Volume 3, EPA Review of Draft Appendix, Suitability of Environmental Methodology to Detect Environmental Radionuclide Concentrations from "As Low As Practicable" Discharges, pp. 301-308.

TABLE 11.6-1

PRACTICAL REPORTING LIMITS*

Gama Spectroscopy

Nuclide	Vegetation and Water 3.5 1 Geometry	Soil and Biota 1.0 1 Geometry	Fraction of MPC (H_2O) (Table II - 10 CFR 20)
Cerium 144	9.7×10^{-8} µCi/ml	2.0×10^{-7} µCi/ml	9.7×10^{-3}
Iodine 131	1.7×10^{-8} µCi/ml	4.0×10^{-8} µCi/ml	5.7×10^{-2}
Ruthenium 106	8.3×10^{-8} µCi/ml	1.8×10^{-7} µCi/ml	8.3×10^{-3}
Cesium 137	1.7×10^{-8} µCi/ml	4.0×10^{-8} µCi/ml	8.5×10^{-4}
Cesium 134	1.7×10^{-8} µCi/ml	6.0×10^{-8} µCi/ml	1.9×10^{-3}
Cobalt 58	1.7×10^{-8} µCi/ml	6.0×10^{-8} µCi/ml	1.7×10^{-4}
Cobalt 60	1.7×10^{-8} µCi/ml	6.0×10^{-8} µCi/ml	3.4×10^{-4}
Zirconium 95	1.4×10^{-8} µCi/ml	3.0×10^{-8} µCi/ml	2.3×10^{-4}
Manganese 54	1.4×10^{-8} µCi/ml	3.0×10^{-8} µCi/ml	1.4×10^{-4}
Zinc 65	3.1×10^{-8} µCi/ml	7.0×10^{-8} µCi/ml	3.1×10^{-4}
Potassium 40	1.8×10^{-7} µCi/ml	3.9×10^{-7} µCi/ml	6.0×10^{-2}
Barium 140	1.7×10^{-8} µCi/ml	4.0×10^{-8} µCi/ml	5.7×10^{-4}
Radium 226	5.7×10^{-8} µCi/ml	2.0×10^{-7} µCi/ml	
Thorium 232	2.8×10^{-8} µCi/ml	1.0×10^{-7} µCi/ml	1.4×10^{-2}

Other Analyses

Strontium 90	8.0×10^{-7} µCi/gm ash	
Tritium	2.0×10^{-7} µCi/ml (1 liter sample)	6.7×10^{-5}
Gross Alpha (water)	7×10^{-9} µCi/ml (1 liter sample)	
Gross Alpha (ash)	7×10^{-6} µCi/gm ash	
Gross Beta (water)	3×10^{-9} µCi/ml (1 liter sample)	
Gross Beta (ash)	1×10^{-5} µCi/gm ash	
Gross Beta - air particulate filters (Based on a total flow of 2200 m^3)	7.7×10^{-12} µCi/ml	
Air Iodine Cartridges	7.7×10^{-12} µCi/cm^3	7.7×10^{-2}

* Limits are representative of 10 CFR 20 values at the time of licensing.