SECTION 11

RADIOACTIVE WASTE MANAGEMENT

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SECTION 11

RADIOACTIVE WASTE MANAGEMENT

The purposes of this section are 1) to provide a complete description and performance evaluation of the Radioactive Waste Treatment Systems and 2) to demonstrate that, under normal plant operating conditions and during anticipated operational occurrences, the radioactive releases from the plant will be in conformance with applicable Nuclear Regulatory Commission (NRC) regulations.

11.1 SOURCE TERMS

Source terms describe the quantity, chemical species, and timing of radioactive materials released from the core and/or primary coolant system for a specific accident. The amounts of radioactive materials which are produced and stored in the reactor system are discussed in this section. These sources have been calculated for the design basis accidents and for normal operation using the ORIGEN computer code for nuclide concentrations and activities from fuel depletion.

11.1.1 Determination of Activity in Reactor Core

The total core activity was originally calculated for Salem with the ORIGEN computer code. The alternative source term (AST, Ref. 1) analysis required additional radionuclide groups than originally calculated by ORIGEN. The numerical values for core activity (60 isotopes) used in the AST LOCA analysis in UFSAR Section 15.4.1 are shown in Table 11.1-1 and were obtained from the default nuclide inventory file from the RADTRAD (3.02 and 3.03) computer program (Refs. 5, 6 & 7). The LOCA dose analysis is based on these 60 isotope activities and a core thermal power of 3632 MWt (=3459 x 1.05). The aerosol inventory is multiplied by 1.10 in some analysis applications to make the RADTRAD default file conservative with respect to the SNGS plant-specific ORIGEN inventory file.

11.1.2 Activities in The Fuel Rod Gap

The activity contained in the space (gap) between the fuel pellets and the cladding is released when the cladding is breached. Rod failure is typically caused by high fuel temperature and primary system depressurization.

The fraction of core activity assumed to be in the gap can vary depending on the specific application. Gap activity is the primary source term for the locked rotor, rod ejection and fuel handling accidents. The gap activity basis is discussed as part of the assumptions described in the specific accident section of Chapter 15.

11.1.3 Fuel Handling Sources

The inventory of fission products in a fuel assembly is dependent on the power rating of the assembly. The parameters used for calculation of the highest rated assembly in the core to be discharged are provided in Section 15.4.6.

11.1.4 Reactor Coolant Fission Product Activities

The parameters used in the calculation of the reactor coolant fission product concentrations, including pertinent information concerning the expected coolant cleanup flow rate, demineralizer effectiveness, and volume control tank noble gas stripping behavior, are presented in Table 11.1-7. The results of calculations are presented in Table 11.1-8. The table lists nuclides of fission and corrosioin products which are significant from a shielding standpoint as well as those nuclides which are listed in ANS Standard ANSI/ANS-18.1-1984. The values tabulated are the maximums that occur during the fuel cycle from startup through the equilibrium cycle.

In these calculations, small cladding defects in the equivalent of one percent of the fuel rods are assumed to be present at the initial core loading and uniformly distributed throughout the core. Similar defects are assumed to be present in all reload regions. The fission product escape rate coefficients are, therefore, based upon an average fuel temperature.

The fission product activity in the reactor coolant during operation with defects in the cladding of the fuel rods is computed using the following differential equations:

For parent nuclides in the coolant:

$$\frac{dN_{C_i}}{dt} = \frac{R_i \cdot N_{F_i}}{M_c} - \left[\lambda_i + D_i + \frac{Q_L}{M_C}\left(\frac{\Psi_i + DF_i - 1}{DF_i}\right)\right] \cdot N_{C_i}$$

For daughter nuclides in the coolant:

$$\frac{dN_{C_j}}{dt} = \frac{R_j \cdot N_{F_j}}{M_c} + f_i \cdot \lambda_i \cdot N_{C_i} - \left[\lambda_j + D_j + \frac{Q_L}{M_C}\left(\frac{\psi_j + DF_j - 1}{DF_j}\right)\right] \cdot N_{C_j}$$

Where:

N_c = Concentration of nuclide in the reactor coolant (atoms/gram)
N_F = Inventory of nuclide in the fuel (atoms)
t = Operating time (seconds)
R = Nuclide release coefficient (1/sec) - F . v
F = Fraction of fuel rods with defective cladding
v = Fission product escape rate coefficient (1/sec)
M_c = Mass of reactor coolant (grams)
λ = Nuclide decay constant (1/sec)
DF = Nuclide demineralizer decontamination factor
Q_L = Purification or letdown mass flow rate (grams/sec)
ψ = Nuclide volume control tank stripping fraction
f = Fraction of parent nuclide decay events that result in the formation of the daughter nuclide
D = Dilution coefficient for feed and bleed (1/sec) = :

$$\frac{\beta}{B_o - \beta \cdot t} \cdot \frac{1}{DF}$$

B_O = Initial boron concentration (ppm)
β = Boron concentration reduction rate (ppm/sec)

and where:

subscript i refers to the parent nuclide
subscript j refers to the daughter nuclide

11.1.5 Tritium Production

11.1.5.1 General - Overall Sources

Tritium is formed from several sources, the most abundant of which is the fissioning of uranium, which yields tritium as a ternary fission product. Tritium atoms are generated in the fuel at a rate of approximately 8×10^{-5} atoms per fission, or 1.05×10^{-2} curies/mwt/day. Boron-bearing control rods can also be a potential source of tritium. These potential sources of tritium are only present in the reactor coolant to the extent that they diffuse through the fuel or control rod cladding.

A direct source of tritium is the reaction of neutrons with dissolved boron in the reactor coolant. Neutron reactions with lithium are also a direct source of tritium. Lithium is present in a pressurized water reactor (PWR) for pH control and as a product of boron reactions with neutrons. An extremely small amount of tritium is also produced by neutron reactions with naturally occurring deuterium in light water.

11.1.5.2 Specific Individual Sources of Tritium

11.1.5.2.1 Ternary Fissions - Clad Diffusion

Because of the mode of operation of the PWR to minimize any liquid or gaseous discharges from the plant, it has been possible to very accurately determine the buildup of tritium from various sources in the plant and to identify their origin.

A program was undertaken by Westinghouse to determine the source of tritium in the reactor coolant in operating plants with both stainless steel and zircaloy cladding. This program clearly indicated that with the current generation of Westinghouse reactors with zircaloy-clad fuel, 1 percent or less of the tritium produced in the fuel will diffuse through the cladding into the coolant. For those plants containing stainless steel cladding, operational data have shown that as high as 80 percent of the ternary tritium produced will diffuse through the cladding.

The tritium concentration in the reactor coolant in those plants having stainless steel and zircaloy fuel cladding has been substantially different. Tritium concentrations at Yankee-Rowe (600 MWt), which has stainless steel cladding, has ranged from

about 4.5 to 5 µc/cc essentially throughout the core cycle. A total discharge from the plant during the core cycle of ~1500 curies of tritium was reported in the monthly operating reports. In addition, with the stainless cores, there has been a continuing source of tritium to the reactor coolant during the power coastdown period when all the boric acid has been removed from the system. This information, in particular, substantiates the high tritium diffusion through the stainless steel clad.

The experience at the R. E. Ginna plant has been substantially different. This plant operates at 1455 MWt and has zircaloy cladding. After approximately 8 months of operation at Ginna, the tritium concentrations were less than 0.3 µc/cc in the reactor coolant and the monthly discharges averaged ~5 curies/month. The experiences at Benzau and Zorita were comparable. An extensive program to follow the buildup of tritium in the Ginna plant was initiated, and the results indicated a potential source from the core which is 1 percent or less of the ternary fissions generated in the fuel. Based on this experience, the tritium sources during the operation of a PWR can be very accurately predicted.

In the past, Westinghouse has assumed that 30 percent of the tritium from ternary fissions would diffuse through the zircaloy fuel. This value was used as a basis for systems and operational design and is clearly conservative.

11.1.5.2.2 Tritium Produced from Boron Reactions

The neutron reactions with boron resulting in the production of tritium are:

B^{10} (n, 2α) T

B^{10} (n, α) Li^{7} (n, nα) T

B^{11} (n, T) Be^{9}

B^{10} (n, d) Be^{9*} (n, α) Li^6 (n, α) T

Of the above reactions, only the first two contribute significantly to the tritium production in a PWR. The B^{11} (n, T) Be^9 reactions have a threshold of 14 Mev and a cross section of ~5 mb. Since the number of neutrons produced at this energy is less than 10^9 n/cm^2-sec the tritium produced from this reaction is negligible. The B^{10}(n, d) reaction may be neglected since Be^{9*} has been found to be unstable.

11.1.5.2.3 Tritium Produced from Lithium Reactions

The neutron reactions with lithium resulting in the production of tritium are:

Li^7 (n, nα) T

Li^6(n, α) T

In Westinghouse designed reactors, lithium is used for pH adjustment of the reactor coolant. The reactor coolant is maintained at a maximum steady state level of 3.5 ± 0.15 ppm lithium by the addition of Li^7OH and by a cation demineralizer included in the Chemical and Volume Control System. This demineralizer will remove any excess of lithium such as could be produced in the B^{10}(n, α) Li^7 reaction.

The Li^6 (n, α) T reaction is controlled by limiting the Li^6 impurity in the Li^7 OH used in the reactor coolant and by lithiating the demineralizers with 99.9 atom percent Li^7.

11.1.5.2.4 Control Rod Sources

In a fixed burnable poison rod, there are two primary sources of tritium generation: the B^{10}(n, 2α) T and the P^{10} (n, α) Li^7(n, nα) T reactions. Unlike the coolant where the Li^7 level is controlled at a maximum steady state level of 3.5 ± 0.15 ppm, there is a buildup of Li^7 in the burnable

poison rod. The burnable poison rods are required during the first year of operation only. During this time the tritium production is 72 curies/pound B^{10}.

There are no tritium sources in Ag-In-Cd control rods.

11.1.5.2.5 Tritium Production from Deuterium Reactions

Since the fraction of naturally occurring deuterium in water is less than 0.0015, the tritium produced from this reaction is negligible (less than 1 curie per year)

11.1.5.2.6 Total Tritium Sources

Tritium sources released to the reactor coolant are listed in Table 11.1-9, based on 12 months of operation at full power (3558 MWt) and a 0.8 load factor.

Included in Table 11.1-9 is the amount of tritium produced in the reactor for all of the nuclear reactions described above.

Two columns of values of tritium released to the reactor coolant are given in Table 11.1-9, namely, a design value and an expected value. The design values are based on a release of 30 percent of the tritium produced being diffused through the fuel cladding. The present values are based on operating experience at existing PWR facilities where the data have indicated that the previous design values were unduly conservative. Based on this experience, the tritium released to the reactor coolant for a typical 3558 MWt reactor is reduced from ~3815 to ~690 curies/year.

Basic parameters employed to calculate the tritium inventory are given in Table 11.1-10.

11.1.6 Volume Control Tank Activity

The radiation sources in the volume control tank (VCT) are based on a nominal operating level in the tank of 200 cubic feet in the liquid phase and 200 cubic feet in the vapor phase, and on the stripping fractions given in Table 11.1-7, assuming no VCT purge. Table 11.1-11 lists the activities for the vapor phase of the VCT with clad defects in 1 percent of the fuel rods.

11.1.7 Gas Decay Tank Activity

The isotopic maximum inventories are determined in the RCS and VCT. Since there is no continuous purge from the volume control tank, the activity values are obtained based on the following considerations:

- At shutdown the radiogas inventory of the VCT is instantaneously transferred to the GDT.
- The RCS (operating with one percent fuel defects) is stripped to the VCT at the maximum letdown with a stripping fraction of 1.0 over a 3 hour period at the end of which the VCT is again instantaneously transferred to the GDT.
- The radioactive sources are calculated at the point where Kr-88 is a maximum in the GDT. This provides a limiting gamma source.

The GDT activities for noble gas nuclides are presented in Table 11.1-12.

11.1.8 Activity in Recirculated Sump Water

The concentration of iodine isotopes in the recirculation loop at initiation of recirculation phase after the design basis loss-of-coolant accident (LOCA) was replaced with an alternative source term (AST) pursuant to Section 50.67 of Title 10 of the Code Of Federal Regulations (10 CFR 50.67), "Accident Source Term", and the potential radiological consequences were re-evaluated. The guidance provided in Regulatory Guide 1.183, Alternative Radiological Source Terms For Evaluating Design Basis Accidents at Nuclear Power Reactors, July 2002, was used in the re-evaluation. The sump water volume is 328,600 gallons and 40% of the total core value of Iodine (in accordance with Reg. Guide 1.183, Table 2) is released to the containment sump. Of this total, 4.85%, which is elemental, is subject to becoming airborne in proportion to a flashing rate. The remainder is assumed to remain waterborne since the sump water pH is maintained > 7.

The radioactivity in the containment would be an additional source of radiation to the auxiliary building following a LOCA. The residual heat removal loop source and the containment source are used to calculate post-accident radiation doses in the Auxiliary Building. The radioactivity leaking out of the recirculation flow path in the Auxiliary Building is Engineered Safety Features (ESF) Leakage and is assumed to be a total of 0.45 gpm in the Section 15.4.1 accident analysis.

11.1.9 References for Section 11.1

1. Regulatory Guide 1.183, Alternative Radiological Source Terms For Evaluating Design Basis Accidents at Nuclear Power Reactors, July 2002

2. CCC-217, ORIGEN Isotope Generator and Depletion Code, Matrix Exponential Method, April, 1975.

3. Radioactive Source Term for Nominal Operation of Light Water Reactors, ANSI/ANS-18.1-1984, American Nuclear Society, December, 1984.

4. Section 50.67 of Title 10 of the Code Of Federal Regulations (10 CFR 50.67), "Accident Source Term"

5. "RADTRAD: A Simplified Model for Radionuclide Transport and Removal and Dose Estimation", NUREG/CR-6604, USNRC, April 1998

6. "RADTRAD: A Simplified Model for Radionuclide Transport and Removal and Dose Estimation", NUREG/CR-6604 Supplement 1, USNRC, June 8, 1999

7. "RADTRAD: A Simplified Model for RADionuclide Transport and Removal and Dose Estimation," W.C. Arcieri (ITSC), NUREG/CR-6604 Supplement 2 (October 2002)

TABLE 11.1-1

CORE ACTIVITIES

Isotope	Core Inventory curies/MWt
CO-58	2.553E+02
CO-60	1.953E+02
KR-85	3.056E+02
KR-85M	7.222E+03
KR-87	1.306E+04
KR-88	1.861E+04
RB-86	1.496E+01
SR-89	2.844E+04
SR-90	1.535E+03
SR-91	3.656E+04
SR-92	3.805E+04
Y-90	1.647E+03
Y-91	3.465E+04
Y-92	3.819E+04
Y-93	4.320E+04
ZR-95	4.377E+04
ZR-97	4.562E+04
NB-95	4.138E+04
MO-99	4.830E+04
TC-99M	4.169E+04
RU-103	3.598E+04
RU-105	2.340E+04
RU-106	8.175E+03
RH-105	1.621E+04
SB-127	2.208E+03
SB-129	7.820E+03
TE-127	2.132E+03
TE-127M	2.823E+02
TE-129	7.341E+03
TE-129M	1.935E+03

Isotope	Core Inventory curies/MWt
TE-131M	3.707E+03
TE-132	3.690E+04
I-131	2.750E+04
I-132	3.889E+04
I-133	5.556E+04
I-134	6.111E+04
I-135	5.278E+04
XE-133	5.556E+04
XE-135	1.389E+04
CS-134	3.425E+03
CS-136	1.042E+03
CS-137	1.915E+03
BA-139	4.976E+04
BA-140	4.924E+04
LA-140	5.032E+04
LA-141	4.615E+04
LA-142	4.449E+04
CE-141	4.476E+04
CE-143	4.352E+04
CE-144	2.697E+04
PR-143	4.273E+04
ND-147	1.911E+04
NP-239	5.120E+05
PU-238	2.902E+01
PU-239	6.545E+00
PU-240	8.254E+00
PU-241	1.390E+03
AM-241	9.181E-01
CM-242	3.514E+02
CM-244	2.056E+01

The LOCA dose analysis (Section 15.4.1) is based on these 60 isotope activities, which are from the RADTRAD default file. A core thermal power of 3632 MWt (=3459 x 1.05) was used. Aerosol inventory is multiplied by 1.10 to make the RADTRAD default file conservative with respect to the SNGS plant-specific ORIGEN inventory file.

TABLE 11.1-2

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TABLE 11.1-3

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TABLE 11.1-4

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TABLE 11.1-5

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TABLE 11.1-7

PARAMETERS USED IN THE CALCULATION OF REACTOR COOLANT FISSION PRODUCT ACTIVITIES(1)

1.	Core thermal power, max. calculated, MWt	3600
2.	Fraction of fuel containing clad defects	0.01
3.	Reactor coolant liquid volume, ft^3	10,892 (2)
4.	Reactor coolant average temperature, °F	568
5.	Purification flow rate (normal), gpm	77
6.	Effective cation demineralizer flow, gpm	7.5
7.	Volume control tank volumes	
	a. Vapor, ft^3	200
	b. Liquid, ft^3	200
8.	Fission product escape rate coefficients:	
	a. Noble gas isotopes, sec^{-1}	6.5×10^{-8}
	b. Br, I, and Cs isotopes, sec^{-1}	1.3×10^{-8}
	c. Te isotopes, sec^{-1}	1.0×10^{-9}
	d. Mo isotopes, sec^{-1}	2.0×10^{-9}
	e. Sr and Ba isotopes, sec^{-1}	1.0×10^{-11}
	f. Y, La, Ce, Pr isotopes, sec^{-1}	1.6×10^{-12}
9.	Mixed bed demineralizer decontamination factors:	
	a. Noble gases and Cs-134, 136, 137, Y-90, 91 and Mo-99	1.0
	b. All other isotopes	10.0
10.	Cation bed demineralizer decontamination factor for CS-134, 236, 237, Y-90, 91, and Mo-99	10.0

(1) The RCS volume was increased by the introduction of new Unit 2 AREVA NP Model 61/19T steam generators. However, the prior analysis remains conservative, therefore, the prior volumes and activities have not been adjusted.

(2) Conservatively bounds 20% tube plugging in Series 51 steam generator and 10% tube plugging in Model-F steam generator.

TABLE 11.1-7 (Cont.)

11. Volume control tank noble gas stripping fraction (closed system):

Isotope	Stripping Fraction
Kr-85	5.5×10^{-5}
Kr-85m	5.4×10^{-1}
Kr-87	8.0×10^{-1}
Kr-88	6.5×10^{-1}
Xe-133	2.9×10^{-2}
Xe-133m	6.7×10^{-2}
Xe-135	2.9×10^{-1}
Xe-135m	9.4×10^{-1}
Xe-138	9.4×10^{-1}

(No purge of VCT assumed in the calculation. Threfore, stripping fractions account only for radioactive decay.)

TABLE 11.1-8

REACTOR COOLANT EQUILIBRIUM FISSION AND CORROSION PRODUCT ACTIVITIES[(1)]

(Based on Parameters Given in Table 11.1-7)

Isotope	Activity µCi/gram	Isotope	Activity µCi/gram
Br-84	4.7×10^{-2}	Cs-136	2.9
Rb-88	4.8	Cs-137	1.5
Rb-89	2.1×10^{-1}	Cs-138	9.6×10^{-1}
Sr-89	4.3×10^{-3}	Ba-140	4.2×10^{-3}
Sr-90	1.2×10^{-4}	La-140	1.4×10^{-3}
Sr-91	6.2×10^{-3}	Ce-144	3.9×10^{-4}
Sr-92	1.3×10^{-3}	Pr-144	3.9×10^{-4}
Y-90	3.4×10^{-5}	Kr-85	8.2
Y-91	5.7×10^{-4}	Kr-85m	1.7
Y-92	1.2×10^{-3}	Kr-87	1.0
Zr-95	6.5×10^{-4}	Kr-88	3.0
Nb-95	6.5×10^{-4}	Xe-133	260
Mo-99	7.5×10^{-1}	Xe-133m	17.0
I-131	2.8	Xe-135	8.5
I-132	2.8	Xe-135m	4.9×10^{-1}
I-133	4.2	Xe-138	6.1×10^{-1}
I-134	5.7×10^{-1}	Mn-54	4.4×10^{-4}
I-135	2.3	Mn-56	2.0×10^{-2}
Te-132	2.9×10^{-1}	Co-58	1.5×10^{-2}
Te-134	3.0×10^{-2}	Co-60	1.9×10^{-3}
Cs-134	2.3	Fe-59	5.2×10^{-4}

(1) The RCS volume was increased by the introduction of new Unit 2 AREVA NP Model 61/19T steam generators. However, the prior analysis remains conservative, therefore, the prior volumes and activities have not been adjusted.

TABLE 11.1-9

TRITIUM PRODUCTION IN THE REACTOR COOLANT

Tritium Source	Total Produced	Released to the Coolant: Design Value	Released to the Coolant: Expected Value
Ternary Fissions	10,926	3240	107
Burnable Poison Rods (Initial Cycle)	973	287	10
Soluble Poison Boron			
(Initial Cycle)	397	397	397
(Equilibrium Cycle)	556	556	556
Li-7 Reaction	11	11	11
Li-6 Reaction	6	6	6
Deuterium Reaction	1	1	1
Totals Initial Cycle	12,314	3942	532
Totals Equilibrium Cycle	11,500	3814	685

(1) Weight of B_2O_3 = 221 (B^{10} = 13.58)

(2) Initial boron (hot, full power, equilibrium xenon) = 860 ppm

(3) Initial boron (hot, full power, equilibrium xenon) = 1200 ppm

TABLE 11.1-10

TRITIUM SOURCES FROM THE REACTOR EMPLOYING Ag-In-Cd ABSORBER RODS

Basic Assumptions and Plant Parameters:

1.	Core thermal power	3558 MWt
2.	Plant load factor	0.8
3.	Core volume	1153 ft^3
4.	Core volume fractions	
	a. UO_2	0.3052
	b. Zr + SS	0.1000
	c. H_2O	0.5948
5.	Initial reactor coolant boron level	
	a. Initial cycle	840 ppm
	b. Equilibrium cycle	1100 ppm
6.	Reactor coolant volume	12,560 ft^3
7.	Reactor coolant transport times	
	a. In-core	0.77 sec
	b. Out-of-core	10.87 sec
8.	Reactor coolant peak steady state lithium level (99 pure Li^7)	3.5 ± 0.15 ppm
9.	Core averaged neutron fluxes:	n/cm^2-sec
	a. $E > 6$ Mev	2.91×10^{12}
	b. $E > 5$ Mev	7.90×10^{12}
	c. 3 Mev $\leq E \leq$ 6 Mev	2.26×10^{13}
	d. 1 Mev $\leq E \leq$ 5 Mev	5.31×10^{13}
	e. $E < 0.625$ ev	2.26×10^{13}
10.	Neutron reaction cross-sections	
	a. B^{10} (n, 2α) T: σ(1 Mev ≤ E ≤ 5 Mev) = spectrum	31.6 mb (spectrum weighted)
	σ(E > 5 Mev) =	> 5 mb
	b. Li^7 (n, nαV) T: σ(3 Mev ≤ E ≤ 6 Mev) =	39.1 mb (spectrum weighted)
	σ(E > 6 Mev) =	400 mb

11.	Fraction of ternary tritium diffusing through zirconium cladding	
	a. Design value	0.30
	b. Expected value	0.01

Note: Although Unit 1 has Westinghouse Model-F and Unit 2 has AREVA NP Model 61/19T steam generators, the radioactivity values of Unit 1 and Unit 2 are bounded by the values in this Table. The values contained in this Table were based on the original Westinghouse Series 51 steam generators.

TABLE 11.1-11

VOLUME CONTROL TANK ACTIVITIES[(1)]

Assumptions are given previously under reactor coolant activity (Table 11.1-7)

Isotope	Total Activity (Curies)
Kr-83m	1.7×10^{1}
Kr-85	6.2×10^{1}
Kr-85m	1.0×10^{2}
Kr-87	2.7×10^{1}
Kr-88	1.4×10^{2}
Xe-131m	1.9×10^{2}
Xe-133	2.4×10^{4}
Xe-133m	1.5×10^{3}
Xe-135	6.6×10^{2}
Xe-135m	5.0×10^{1}
Xe-137	2.8×10^{-1}
Xe-138	3.4×10^{0}

(1) The RCS volume was increased by the introduction of new Unit 2 AREVA NP Model 61/19T steam generators. However, the prior analysis remains conservative; therefore, the prior volumes and activities have not been adjusted.

TABLE 11.1-12

GAS DECAY TANK ACTIVITY

Assumptions: Volume of the tank immaterial to this calculation.
Clad Defects in one percent of fuel rods.
Operation at 3600 MWt for 497 days.
Tank contains entire gaseous activity stripped off from the Reactor Coolant System.
Reactor Coolant System Volume is 12,446 ft^3.

Isotope	Total Activity Curies
Kr-85	1.5×10^3
Kr-85m	1.2×10^2
Kr-87	1.8×10^1
Kr-88	1.5×10^2
Xe-131m	2.9×10^2
Xe-133	3.5×10^4
Xe-133m	2.2×10^3
Xe-135	8.6×10^2

TABLE 11.1-13

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TABLE 11.1-14

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11.2 LIQUID WASTE SYSTEM

The Liquid Waste System (LWS) provides controlled handling and disposal of liquid wastes generated during plant operation. The system is designed to minimize exposure to plant personnel and the general public, in accord with Nuclear Regulatory Commission (NRC) regulations.

11.2.1 Design Objectives

The design objectives of the LWS are the following:

1. Maintain annual activity releases within the limits specified in 10CFR20

2. Protect the public health and safety by maintaining radioactive releases as low as practicable

3. Collect radioactive and potentially radioactive liquid wastes

4. Provide processing of liquid wastes such that operation and availability of the stations are not limited

5. Assure that exposures to the public are maintained below the design objectives set by Appendix I to 10CFR50

The design criteria for the LWS areas follows:

The facility design shall include those means necessary to maintain control over the plant radioactive liquid effluents. Appropriate holdup capacity shall be provided for retention of liquid, particularly where unfavorable environmental conditions can be expected to require operational limitations upon the release of radioactive effluents to the environment. In all cases, the design for radioactivity control shall be justified 1) on the basis of 10CFR20 requirements, for both normal operations

and for any transient situation that might reasonably be anticipated to occur and 2) on the basis of 10CFR50.67 dosage level limits for potential reactor accidents of exceedingly low probability of occurrence.

Liquid facilities are designed so that discharge of effluents and offsite shipments are in accordance with applicable governmental regulations.

Radioactive fluids entering the LWS are collected in tanks until determination of subsequent treatment can be made. They are sampled and analyzed to determine the quantity of radioactivity, with an isotopic breakdown if necessary. Liquid wastes are processed as required and then released under controlled conditions following isotopic analysis. The system design and operation are directed toward minimizing releases to unrestricted areas. Discharge streams are appropriately monitored and safety features are incorporated to preclude releases in excess of the limits of 10CFR20.

11.2.2 System Description

The bulk of the radioactive liquids discharged from the Reactor Coolant System (RCS) is processed and retained inside the plant by the Chemical and Volume Control System (CVCS) recycle train. This minimizes liquid input to the LWS which processes relatively small quantities of generally low activity level wastes. The processed water from waste disposal, from which most of the radioactive material has been removed, is discharged through a monitored line to the service water discharge header and then to the circulating water discharge.

During normal plant operation the LWS processes liquids from the following sources:

1. Equipment drains and leaks

2. Radioactive chemical laboratory drains

3. Hot shower drains

4. Decontamination area drains

5. CVCS demineralizer regenerant solutions and spent resins

6. Sampling System

In addition, piping has been installed to direct potential fluid leakage from valves in the following systems to the LWS: Residual Heat Removal (RHR), Safety Injection (SIS), Containment Spray, CVCS, Sampling Systems. This design minimizes the spread of highly radioactive liquid throughout the plant in the event of postulated accidents.

Additionally, continuous releases from steam generator blowdown primary to secondary leakage below the setpoint of the steam generator blowdown radiation monitor are processed to the condenser or to the non-radioactive waste treatment system.

The LWS also collects and transfers liquids from the following sources directly to the CVCS, to the waste holdup tanks, or back to the refueling water storage tank (depending on fluid content) for processing:

1. Reactor coolant loops

2. Pressurizer relief tank

3. Reactor coolant pump secondary seals

4. Excess letdown (during startup)

5. Accumulators

6. Valve and reactor vessel flange leakoffs

7. Refueling canal drains

These liquids flow to the reactor coolant drain tank and are discharged either directly to the CVCS holdup tanks or to the waste holdup tanks by the reactor coolant drain pumps which are operated automatically by a level controller in the tank. These pumps also return water from the refueling canal and cavity to the refueling water storage tank. There is one reactor coolant drain tank with two reactor coolant drain tank pumps located inside containment.

Where possible, waste liquids drain to the waste holdup tanks by gravity flow. Other waste liquids drain to the Auxiliary Building sump tank and are discharged to the waste holdup tanks by pumps operated automatically by a level controller for the Auxiliary Building sump tank.

With the exception of the shared pumps and tanks of the Laundry and Hot Shower Drains, the Chemical Drains, Portable Filter and the Portable Demineralizer, each unit has its own Liquid Waste Disposal System. The Laundry and Hot Shower Drain Tanks and the Chemical Drain Tank are pumped to one of the Waste Hold-up Tanks or the Waste Monitor Hold-up Tank of either unit.

When a Waste Hold-up or Waste Monitor Hold-up Tank is filled, it is isolated and sampled while another tank is in service. If analysis confirms that the activity level of the tank's contents is suitable for discharge, the tank's contents may be pumped through a flow meter and a radiation monitor to the Service Water System. Tanks requiring processing before release are routed on a batch basis through a portable filter and portable demineralizer. The effluent of the portable system is returned either to the Waste Monitor Hold-up Tanks or the CVCS Monitor Tanks to be sampled, analyzed, and either reprocessed or pumped through a flow meter and a radiation monitor to the Service Water System.

Although the Waste Monitor Hold-up or CVCS Monitor Tank analysis forms the basis for recording activity releases, the radiation monitor provides surveillance over the operation by closing the discharge valve if the liquid activity exceeds a preset value.

The system is capable of processing all liquid wastes generated during continuous operation of the primary system assuming that fission products escape to the reactor coolant by diffusion through defects in the cladding on one percent of the fuel rods.

At least two valves must be manually opened to permit discharge of liquid waste from the LWS. The control valve will trip closed on a high effluent radioactivity level signal.

The system is controlled from a central panel in the Auxiliary Building. Malfunction of the system is alarmed in the Auxiliary Building, and annunciated in the Control Room. All system equipment is located in the Auxiliary Building, except for the reactor coolant drain tank and drain tank pumps which are located in the reactor containment, and a 2-inch line from the drain tank pumps to the refueling water storage tank (RWST).

The LWS process flow diagram is shown on Plant Drawings 205239 and 205339. Performance data for the LWS is given in Table 11.2-1. The estimated annual liquid discharged to the LWS is given in Table 11.2-2.

11.2.3 System Design

The LWS code requirements are given in Table 11.2-3. A summary of component system data is given in Table 11.2-4. Note that Table 11.2-3 also contains code data for the Gaseous and Solid Radwaste Systems.

Hot Shower Tanks

Two stainless steel tanks collect liquid wastes originating from the hot shower and local sinks. These tanks and their associated pumps are common to both Units' LWS. The intention is that one tank will be available for filling, while the contents of the other tank are being pumped to a waste holdup tank to await processing. A basket type strainer is provided downstream of this pump to prevent discharge of lint to other tanks. The pump is started and stopped manually from a local control panel.

Chemical Drain Tank

The chemical drain tank is stainless steel and collects drainage from the chemistry laboratory. This potentially high activity waste is normally transferred to the waste holdup tanks to await processing. This tank and associated pump are common to both Units 1 and 2. The pump is started and stopped from a local control panel. The suction lines from the chemical drain tank and laundry tanks are interconnected to allow the laundry pump and chemical drain pump to substitute when necessary.

Reactor Coolant Drain Tank

The reactor coolant drain tank is a right circular cylinder with spherically dished heads. The tank is constructed of stainless steel with welded seams. The reactor coolant drain tank receives recyclable waste from the following sources:

1. Reactor coolant pump seal and head tank leakoffs

2. Drains from each of the four primary coolant loops

3. Reactor vessel flange leakage

4. Accumulator drains

5. Excess letdown

6. Refueling canal drains

During normal operation a nitrogen blanket, at a pressure of 0.5 psig, is maintained in the reactor coolant drain tank. The tank is normally vented to the Gaseous Waste Disposal System vent header so that changes in liquid level will cause the tank to breathe to and from this header. This eliminates the possibility of releasing hydrogen and radioactive gases to the containment. The contents of the reactor coolant drain tank can be transferred to one of the following systems:

1. CVCS holdup tanks

2. Emergency Core Cooling System (ECCS) RWST

3. LWS holdup tanks

Normally all waste collected in the reactor coolant drain tank is transferred to the CVCS holdup tanks by the Nos. 11, 21 and 12, 22 reactor coolant drain pumps. Operation of these pumps is automatically controlled by tank level instrumentation.

Valves WL-12 and WL-13 in the reactor coolant drain tank pump discharge line are maintained in the shut position following containment isolation until manually reset by the operator.

Waste Holdup Tank

Two waste holdup tanks are provided to accept liquid wastes from the CVCS, sump tank, chemical drain tank, reactor coolant drain tank, Steam Generator Blowdown System, floor and hot shower tanks. The tanks are of welded stainless steel construction. Individual air-operated valves in the common inlet manifold to these tanks are used to divert waste flow from one tank to the other. Two tanks are provided with the intention that one tank will be available to accept waste, while the contents of the other tank are being held to await processing. This holdup time provides an additional delay over and above that available from one tank, to allow shorter-lived radionuclides to decay. An additional 25,000-gallon waste monitor holdup tank is available to accept waste surges in the event of an emergency.

Containment Sump Tank and Pumps

The containment sump accumulates all floor drains, washdowns from refueling decontamination operations, drains and condensate from the fan coil units and miscellaneous equipment drains of a potentially radioactive but non-reactor coolant nature. The contents of this sump are pumped directly to the waste holdup tanks by two sump pumps that operate from sump level control instrumentation. Both pumps can also be started manually. All wetted parts of the pump are stainless steel. The tank is all welded stainless steel.

Waste Evaporator

The Unit 1 waste evaporator is abandoned in place. Installation of the Unit No. 2 waste evaporator has been cancelled.

Piping and Valves

Piping and valves which are in contact with liquid wastes are constructed of stainless steel. Piping connections are welded except where flanged connections are necessary to facilitate equipment maintenance. Isolation valves are provided to isolate each piece of equipment for maintenance, to direct the flow of waste through the system, and to isolate storage tanks for radioactive decay. Relief valves are provided for tanks containing radioactive wastes if the tanks might be over pressurized by improper operation or component malfunction.

Spent Resin Storage Tank

The spent resin storage tank retains the spent resin normally discharged from the mixed bed, evaporator feed ion exchange, spent fuel pit, deborating vessel and cation demineralizers. A layer of water is maintained over the resin storage to prevent resin degradation due to heat generation from decaying fission products. The contents can be removed any time by flushing with nitrogen. Resin sample connections are supplied upstream and downstream of the spent resin storage tank isolation valve. The tank is all welded austenitic stainless steel.

Waste Monitor and Waste Monitor Holdup Tanks

The Waste Monitor Tanks have been disabled and abandoned in place (West end of 84′ elevation). These tanks are of welded stainless steel construction.

The Waste Monitor Hold-up Tank, used as a 3rd WHUT, is also a welded stainless steel tank and it serves a dual purpose. Its normal function is as a third waste holdup tank to receive abnormally large quantities of waste discharged to the system, but it can also serve as a waste monitor tank.

Portable Processing System

Permanent provisions have been made to the 460 VAC supply, waste liquid piping, compressed air piping, and the demineralized water-restricted area piping to allow for the installation and operation of a portable system to process liquid radwaste from either unit. The Unit 2 liquid waste may be processed at a higher rate due to a difference in piping configuration. The system is installed and operated in the 100' elevation of the truck bay of the Auxiliary Building. The effluent of the portable system is returned to either the Waste Monitor Holdup Tanks or the CVCS Monitor Tanks to be sampled, analyzed, and either reprocessed or disposed of. Exhausted ion exchange media, or filtration, or both, is transferred to a burial site approved container after which it can be processed, classified, and shipped for disposal.

Steam Generator Blowdown

The steam generator blowdown system is described in section 10.4.8.

11.2.4 Operating Procedures

Verification is made to ensure that dilution flow sufficient to meet the requirements of 10CFR20 is available whenever radioactive liquid wastes are released to the Plant Discharge System.

Liquid waste releases are continuously monitored for gross activity during discharges to ensure that the activity limits specified in 10CFR20 for unrestricted areas are not exceeded. The maximum allowable release at the plant is specified in the Technical Specifications.

Radioactive liquid batch wastes are sampled prior to releases to the Plant Discharge System and records of all releases are kept.

Continuous releases from the steam generator blowdown system are monitored and controlled in accordance with the Salem Offsite Dose Calculation Manual.

11.2.5 Performance Tests

Samples are taken on each batch of liquid waste released. Station records contain the quantity and concentration of radioactive isotopes, the volume of each batch and estimates of the water flow for dilution. Each sample is analyzed for principal gamma emitters. Composites are prepared from each batch released during a month and analyzed for the principal gamma emitting nuclides, fission and activation products, gross alpha, and tritium. A quarterly composite analysis is also performed for Sr-89, Sr-90, and Fe-55. The sensitivities and frequencies of analyses comply with the requirements of Salem Technical Specifications.

Continuous releases from the steam generator blowdown system are sampled and analyzed to determine the quantity and concentration of isotopes present in the blowdown stream. Composites are prepared from the samples. The sensitivities and frequencies of analysis comply with the requirements of the Salem Technical Specifications.

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11.2.6 Estimated Releases

Liquid wastes are generated primarily by plant maintenance and service operations, and, consequently, the quantities and activity concentrations of influents to the system. Tables 11.2-5 and 11.2-6 are estimated values. Therefore, considerable operational margin has been assigned between the design capability and the estimated system load as indicated in Table 11.2-5. A conservative estimate of system ability to limit dissolved and suspended activity released in the liquid phase is summarized in Table 11.2-6: This tabulation is based on the following assumptions.

1. Fission product concentration in the reactor coolant is based on 1 percent defective fuel.

2. Non-recycleable leakage of reactor grade coolant which is processed and discharged from the plant is assumed to be a total of 2,102 gallons per day.

3. Non-reactor grade non-radioactive liquid leakage which is processed and discharged from the plant is assumed to be a total of 4,308 gallons per day.

4. It is assumed that the liquid wastes are accumulated in the waste holdup tank and then processed for discharge overboard with the raw cooling water.

5. A dilution factor is used in determining the discharge concentrations of liquids released from the waste monitor holdup and waste holdup tanks. This is discussed in Section 11.2.8.

6. The tritium that is formed in the fuel (the predominant source) diffuses through the zircaloy clad and eventually becomes available for dispersal to the environment. The expected release of ternary produced tritium is about 1 percent; the total annual tritium release expected is indicated in Table 11.2-6. All of the sources of tritium accumulating in the reactor coolant, shown in Section 11.1, are included in the annual release.

The release estimates given in Table 11.2-6 are based on continuous operation with 1 percent fuel defects in both units. Based on experience with operating pressurized water reactors to date, 0.2 percent is a more realistic estimate of fuel defects averaged over a year of operation. Hence, in order to evaluate expected releases, all release values, except for H-3 and corrosion activation products given in Table 11.3-6, could be reduced by a factor of 5.

The release estimates for continuous releases from the steam generator blowdown system are based on assumptions consistent with NUREG-0017. Releases are monitored by the steam generator radiation monitors which provide a signal to close the isolation valves to maintain releases within the requirements of the ODCM. At higher primary system activity levels, steam generator blowdown would be isolated at lower primary to secondary leakage rates.

11.2.7 Release Points

Release points are shown on the system flow diagram, Plant Drawings 205239 and 205339.

11.2.8 Dilution Factors

Monitored waste released from the Liquid Waste Disposal System is normally pumped into the Circulating Water System discharge lines via the Service Water System. The maximum release rates are calculated based on how many circulators are available in the release path. The release rate is controlled by throttling the discharge valve to maintain the maximum release rate or less.

Steam generator blowdown is directed to the non radioactive waste basin either from the blowdown flashtank or from the condensate polishing system and then to the circulating water system.

11.2.9 Estimated Doses (Potential)

There are generally two main pathways by which the general population could receive radiation exposure from liquid releases. One would be by drinking the water from the river in the vicinity of the station, the other would be by the consumption of marine life (such as fish and shellfish) that inhabit the general river and bay area (marine life has a tendency to reconcentrate certain radioactive isotopes above background water levels). Less important pathways would be from swimming in the river, boating or fishing, standing on the shoreline, etc.

Section 2 contains detailed information on public water supplies and known water wells in the Salem site vicinity. Because of the brackish condition of the river, no potable water supplies are drawn from it in the area. Ground water is the primary source of water supply in the vicinity with the exception of the city of Salem, New Jersey, which obtains about two-thirds of its water supply from Quinton, on Alloways Creek. This water supply is a dammed fresh water stream some 9 miles upstream from the Delaware River - Alloways Creek confluence. Hence, no radioactive releases would reach this public water supply from the Salem station as the flow is from Alloways Creek into the Delaware River.

Consideration was given to the possibility of radioactivity reaching public and private water supplies that are present in the Salem area. The Delaware River is not recharging aquifers that are in use. The upper sand layers in the region are saline aquifers. The artesian aquifers, located at much greater depths than the saline aquifers, are separated from the upper sand layers by an impervious clay strata. Hence, no Hydraulic communication exists. It is concluded that no liquid radioactive releases could reach ground water drinking supplies in the Salem area (additional information on ground water hydrology is given in Section 2).

Consideration was given to the radiation exposure that might be received through the fish food chain. The fundamental approach in evaluating this pathway is dependent on the radionuclide concentration provided by the fish in question. The concentration of the stable element (and the radionuclide) by the organism is related to the natural biological demand which the organism has for the element in question and the ratio of the concentration of the element to the elemental concentration in its water environment. Tables 11.2-7 and 11.2-8 relate the concentration factor (by the marine life), the radioactivity in the marine life and the resulting concentration to which the individual would be exposed upon consumption of fish and blue crabs, respectively.

The following assumptions and data were used to develop the concentration factors, ingestion factors and fraction of MPC in Tables 11.2-7 and 11.2-8:

1. Concentration factors are based on a literature review of the stable element chemistry and the radionuclide concentration contained in "Concentration Factors of Chemical Elements in Edible Aquatic Organisms," W. H. Chapman et al, UCRL-50564, December 30, 1968.

2. 100 percent of the effluent radionuclides are assumed to be ingested or absorbed by the marine life in question (the organisms are considered to remain in the vicinity of the discharge pipe 24 hours per day, 52 weeks per year).

3. Individual consumption will include approximately 4 - 5 meals per week (100 grams, or 140 grams per day, 52 weeks per year). The marine life in question are comprised of bullhead catfish, carp, weakfish, striped bass, American eel and blue crab. These organisms represent the major cross section of edible marine life caught in the area.

4. The ingestion factor is calculated based on the following assumptions:

   a. Equivalent density for "fish flesh" and water

   b. Individual daily consumption of "fish flesh" is approximately 140 grams

   c. MPC values given in 10CFR20 are based on a total daily intake of 2.2 liters of water

$$\text{Ingestion Factor} = \frac{140}{2200} = 6.36 \times 10^{-2}$$

This approach to an analysis of radiation exposure through the food chain is extremely conservative. The assumption that the marine life in question will remain adjacent to the discharge pipe 24 hours per day, 52 weeks per year, is highly conservative. The reduction in radioactivity due to radioactive decay and biological turnover was not assumed.

Based on Table 11.2-7 and 1 percent failed fuel, the potential exposure from eating 50,000 grams (110 lbs) per year of fish can be calculated as $6.1 \times 10^{-5} \times 500$ mrem = 0.030 mrem per year. Similarly, from Table 11.2-8, the potential exposure from eating 50,000 grams per year of crabmeat (net weight) can be calculated as $2.4 \times 10^{-4} \times 500$ mrem = 0.12 mrem per year.

Based on 0.2 percent fuel defects, the exposures from ingestion of fish and crabmeat would be 0.01 mrem/yr and 0.03 mrem/yr, respectively.

Consideration was also given to individuals swimming, fishing, and boating on the river. Calculation of these doses is given below.

The dose to an individual swimming near the discharge canal was calculated using the following basic equation for the dose in an "infinite" homogeneous source (1):

$$R = 51\ CE$$

where:

R = dose, rads/day

C = concentration of radioactive material, $\frac{uCi}{gm}$

E = decay energy, Mev/dis.

The release concentrations given in Table 11.2-6 (based on 1 percent fuel defects) would be adjusted to 2.0 x 10^{-11} uCi/cc, excluding H-3, and 4.5 x 10^{-7} uCi/cc for H-3 (based on 2 percent fuel defects).

It is assumed that an individual swims near the discharge canal for 200 hours per year. It is also assumed that the average decay energy is 1 Mev/dis for all isotopes except H-3. An average decay energy of 0.0057 Mev/dis is used for H-3.

The calculations to a maximum individual follow:

<u>Dose due to all isotopes except H-3</u>

$$D = 51(2.0\text{x}10^{-11}\ \frac{uCi}{cc})\ (1\frac{cc}{gm})\ (1\ \frac{Mev}{dis})\ (200\ \frac{hrs}{yr})\ (\frac{1\ day}{24\ hr})$$

$$D = 8.3\text{x}10^{-9}\ \frac{rads}{yr}$$

Dose due to H-3

$$D = 51(4.5\times 10^{-7}\,\frac{uCi}{cc})\,(1\ \frac{cc}{gm})\,(0.0057\,\frac{Mev}{dis})\,(200\,\frac{hrs}{yr})\,(\frac{1\ day}{24\ hrs})$$

$$D = 1.1\times 10^{-6}\,\frac{rads}{yr}$$

The total dose is thus 1.1×10^{-6} rads per year.

The dose to an individual fishing along the shore or on a boat on the river can be estimated in a manner similar to the one above. In this case, it is more appropriate to think in terms of a "semi-infinite" medium, since essentially no radioactivity from releases to the river is present in the air above the river (in the swimming case, it was assumed that the individual was "submerged" in the water; hence, the individual would be exposed to radiation from all directions).

The doses to an individual fishing 200 hours per year on shore or on a boat would thus be one-half of the doses previously calculated, or 4.3×10^{-9} rads per year due to all isotopes except H-3 and 5.5×10^{-7} rads per year due to H-3, for a total dose of 5.5×10^{-7} rads per year.

The dose pathways considered, and the resulting doses to the maximally exposed individual based on realistic liquid releases at 0.2 percent failed fuel defects, are summarized in Table 11.2-9.

Doses and releases, including steam generator blowdown continuous releases are controlled in accordance with the Salem Offsite Dose Calculation Manual.

11.2.10 Reference for Section 11.2

1. Evans, R. D. "The Atomic Nucleus," McGraw-Hill, 1955, p. 742.

TABLE 11.2-1

LIQUID WASTE SYSTEM PERFORMANCE DATA

Evaporator Design Life	39 years
Normal process capacity, liquids	16.5 gpm (Unit 1 only)
Evaporator load factor	
Average	Inactive
During peak week	Inactive
Annual liquid discharge(1)	
Reactor grade water	766,950 gal
Nonreactor grade water	1,572,500 gal
Total	2,339,450 gal
Activity other than tritium	0.145 curies
Tritium	690 curies
Portable Demineralizer normal	28 gpm (Unit 1)
Process capacity	38 gpm Unit 2)
Portable system normal process capacity	25-40 gpm
Portable system load factor	(at 28 gpm)
Average	6 percent
During Peak Week	42 percent

NOTES:

(1) Estimate based on Table 11.2-2.

TABLE 11.2-2

ESTIMATED ANNUAL LIQUID DISCHARGE TO WASTE DISPOSAL
(per unit)

Source	Total Annual Discharge, gal
BAE distallate	1,411,000
Hot showers	119,800
Laboratory	91,250
Equipment drains, leaks	702,400
Decontamination	15,000
Total waste disposal system	2,339,450

TABLE 11.2-3

WASTE DISPOSAL COMPONENTS CODE REQUIREMENTS

Component	Code
Chemical Drain Tank	ASME VIII (4) (not code stamped)
Reactor Coolant Drain Tank	ASME III, (1) Class C
Sump Tank	ASME III, (1) Class C
Waste Holdup Tank	ASME III, (1) Class C
Waste Monitor - Holdup Tank	ASME III, (1) Class C (9)
Waste Monitor Tank	ASME III, (1) Class C (10)
Laundry and Hot Shower Tank	ASME VIII (4) (not code stamped)
Waste Evaporator Forced Circulation Concentrator	ASME VIII (5)
Waste Filter	ASME III, (1) Class C
Piping and Valves (11)	ANSI B31.7 (2) Section 1 ANSI B31.1 (3)
Spent Resin Storage Tank	ASME III, (1) Class C
Pumps and Compressors (7) (11)	ASME Draft Code for Pumps and Valves for Nuclear Power, November, 1968
Evaporator Bottoms Holdup Tank	ASME VIII (5)
Reactor Coolant Drain Tank Pumps	NNS, Class D+ (6)
Portable Liquid Radwaste Processing System	(8)

WASTE DISPOSAL COMPONENTS CODE REQUIREMENTS

Component	Code

NOTES:

(1) ASME III - American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section III, Nuclear Vessels, 1968 Edition.

(2) For piping not supplied by the NSSS supplier, material inspection, fabrication, and quality control conform to ANSI B31.7. Where not possible to comply with ANSI B31.7, the requirements of ASME III-1971, which incorporated ANSI B31.7, were adhered to.

(3) ANSI B31.1 - Used for design and material selection.

(4) ASME VIII - American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section VIII, Pressure Vessels, 1968 Edition.

(5) ASME VIII - American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section VIII, Pressure Vessels, 1971 Edition.

(6) Quality Group D is augmented by the addition of Quality Assurance Requirements necessary to ensure an acceptable level of confidence that the pumps perform their intended functions.

(7) Does not include the Reactor Coolant Drain Tank Pumps (listed separately).

(8) Design criteria for the portable liquid radwaste processing system meets the intent of section 3 of the UFSAR and was specified to be compatible with Reg Guide 1.143, except for the seismic criteria which are addressed by the basic reactor facility, even though Salem does not commit to the Reg Guide.

(9) #12 Waste Holdup Tank (1WLE11) abandoned in place.

(10) Waste Monitor Tank (1&2) WLE (3&4) abandoned in place.

(11) Some Waste Gas (WG) System components can be reclassified to nonsafety-related, non-nuclear, and Seismic Class II in accordance with safety evaluation S-C-N510-MSE-0319, "Safety Classification of the Gas Waste Disposal System". However, all piping and valves providing isolation for the Waste Gas Decay Tanks shall be classified safety-related, Nuclear Class III, and Seismic Class I.

TABLE 11.2-4
COMPONENT DATA SUMMARY

Tanks	Quantity	Type	Volume	Design Pressure	Design Temp. °F	Material(1)
Reactor Coolant Drain (per unit)	1	Horiz	565 gal	25 psig	267	SS
Laundry and Hot Shower	2	Vert	600 gal	Atm	180	SS
Chemical Drain	1	Vert	600 gal	Atm	180	SS
Auxiliary Building Sump Tank	1	Horiz	1000 gal	15 psig	110	SS
Waste Holdup	2	Horiz	25,000 gal	15 psig	180	SS
Waste Monitor Holdup	1	Horiz	25,000 gal	15 psig	180	SS
Waste Monitor (4)	2	Vert	950 gal	15 psig	180	SS
Spent Resin	1	Vert	300 ft^3	100 psig	180	SS
Evaporator Bottoms Holdup Tank	1	Vert	1500 gal	0.5 psig	250	Inconel 625

TABLE 11.2-4

Tanks	Quantity	Type	Flow gpm	Head ft	Design Pressure psig	Design Temp °F	Material(1)
Reactor Coolant Drain	2	Horiz cent canned	50	175	150	200	SS
Chemical Drain	1(3)	Horiz cent(2)	20	100	150	200	SS
Laundry	1(3)	Horiz cent(2)	20	100	150	200	SS
Auxiliary Building Sump Tank	2	Horiz cent	40	60	150	180	SS
Waste Evaporator Feed	1	Horiz cent	16.5	150	150	200	SS
Waste Monitor (4)	2	Horiz cent	40	100	150	180	SS
Waste Monitor Holdup	1	Horiz cent	60	110	150	180	SS

NOTES:
(1) Material contacting fluid.
(2) Mechanical seal provided.
(3) Shared by Units 1 and 2.
(4) Waste Monitor Tanks and Pumps abandoned in place.

TABLE 11.2-5

ESTIMATED ANNUAL LIQUID DISCHARGE TO LIQUID WASTE SYSTEM

Source of Liquid Waste	Type	Amount (gal/yr)
Tank Drains	Reactor Grade	239,000
Filter Strainer Drains	Reactor Grade	51,000
Heat Exchanger Drains	Reactor Grade	63,700
Demineralizers	Reactor Grade	49,000
Valve Leakoffs	Reactor Grade	78,000
Pump Leakages	Reactor Grade	195,000
Sample Sink Drains	Reactor Grade	91,250
Total Reactor Grade		766,950
Component Cooling Heat Exchanger Drains	Nonreactor Grade	20,500
Component Cooling Valve Leakoffs	Nonreactor Grade	6,200
Hot Showers	Nonreactor Grade	119,800
Decontamination and Floor Scrubbing	Nonreactor Grade	15,000
BAE Distillate	Nonreactor Grade	1,411,000
Total Nonreactor Grade		1,572,500
Grand Total		2,339,450

TABLE 11.2-6

ESTIMATED ANNUAL LIQUID RELEASE BY ISOTOPE

(TWO UNIT BASIS)

Isotope	Annual Release	Discharge Canal Concentration	(MPC)w	Fraction of (MPC)w
	(μCi)	(μCi/cc)	(μCi/cc)	
H-3	1.37×10^{9}	4.54×10^{-7}	3×10^{-3}	1.51×10^{-4}
Cr-51	5.56×10^{1}	1.88×10^{-14}	2×10^{-3}	9.40×10^{-12}
Mn-54	7.24×10^{1}	2.45×10^{-14}	1×10^{-4}	2.45×10^{-10}
Mn-56	9.84×10^{0}	3.32×10^{-15}	1×10^{-4}	3.32×10^{-11}
Co-58	1.98×10^{3}	6.69×10^{-13}	1×10^{-4}	6.69×10^{-9}
Co-60	7.28×10^{1}	2.46×10^{-14}	5×10^{-5}	4.92×10^{-10}
Fe-59	7.34×10^{1}	2.48×10^{-14}	6×10^{-5}	4.13×10^{-10}
Sr-89	2.74×10^{2}	9.25×10^{-14}	3×10^{-6}	3.08×10^{-8}
Sr-90	1.34×10^{1}	4.53×10^{-15}	3×10^{-7}	1.51×10^{-8}
Sr-91	2.64×10^{0}	8.92×10^{-16}	7×10^{-5}	1.27×10^{-11}
Sr-92	2.84×10^{-1}	9.60×10^{-17}	7×10^{-5}	1.37×10^{-12}
Y-90	1.39×10^{0}	4.70×10^{-16}	2×10^{-5}	2.35×10^{-11}
Y-91	4.64×10^{2}	1.57×10^{-13}	3×10^{-5}	5.23×10^{-9}
Y-92	3.54×10^{-1}	1.20×10^{-16}	6×10^{-5}	2.00×10^{-12}
Zr-95	5.40×10^{-1}	1.82×10^{-14}	6×10^{-5}	3.03×10^{-10}
Mo-99	4.78×10^{4}	1.61×10^{-11}	2×10^{-4}	8.05×10^{-8}
Nb-95	4.40×10^{1}	1.49×10^{-14}	1×10^{-4}	1.49×10^{-10}
I-131	6.18×10^{4}	2.09×10^{-11}	3×10^{-7}	6.97×10^{-5}
I-132	2.86×10^{2}	9.66×10^{-14}	8×10^{-6}	1.21×10^{-8}
I-133	1.13×10^{4}	3.82×10^{-12}	1×10^{-6}	3.82×10^{-6}
I-134	7.12×10^{1}	2.41×10^{-14}	2×10^{-5}	1.20×10^{-9}
I-135	1.94×10^{3}	6.55×10^{-13}	4×10^{-6}	1.64×10^{-7}
Te-132	2.68×10^{3}	9.05×10^{-13}	3×10^{-5}	3.02×10^{-8}
Te-134	3.00×10^{0}	1.01×10^{-15}	------	-------

Isotope	Annual Release	Discharge Canal Concentration	(MPC)w	Fraction of (MPC)w
	(μCi)	(μCi/cc)	(μCi/cc)	
Cs-134	2.52×10^{4}	8.51×10^{-12}	9×10^{-6}	9.46×10^{-7}
Cs-136	6.18×10^{3}	2.09×10^{-12}	9×10^{-5}	2.32×10^{-8}
Cs-137	1.29×10^{5}	4.36×10^{-11}	2×10^{-5}	2.18×10^{-6}
Cs-138	7.30×10^{1}	2.46×10^{-14}	------	-------
Ba-140	1.58×10^{2}	5.34×10^{-14}	8×10^{-5}	1.78×10^{-9}
La-140	7.30×10^{0}	2.46×10^{-15}	2×10^{-5}	1.23×10^{-10}
Ce-144	2.96×10^{1}	1.00×10^{-14}	1×10^{-5}	1.00×10^{-9}
Pr-144	1.20×10^{-2}	4.05×10^{-18}	------	-------
Totals	2.90×10^{5}(1)	9.80×10^{-11}(1)	------	2.27×10^{-4}

NOTE:

(1) Excluding H-3

TABLE 11.2-7

CONCENTRATION OF RADIONUCLIDES IN FISH DUE TO PLANT OPERATION

Isotope	Discharge Canal Concentration (μCi/cc)	Concentration Factor	Activity in Fish (μCi/cc)	(MPC)w (μCi/cc)	Activity in Fish x Ingestion Factor (μCi/cc)	Fraction of (MPC)w
H-3	4.54×10^{-7}	9.3×10^{-1}	4.2×10^{-7}	3×10^{-3}	2.7×10^{-8}	9.0×10^{-6}
Cr-51	1.88×10^{-14}	4×10^{2}	7.5×10^{-12}	2×10^{-3}	4.8×10^{-13}	2.4×10^{-10}
Mn-54	2.45×10^{-14}	3×10^{2}	7.4×10^{-12}	1×10^{-4}	4.7×10^{-13}	4.7×10^{-9}
Mn-56	3.32×10^{-15}	3×10^{2}	1.0×10^{-12}	1×10^{-4}	6.4×10^{-14}	6.4×10^{-10}
Co-58	6.69×10^{-13}	5×10^{2}	3.3×10^{-10}	1×10^{-4}	2.1×10^{-11}	2.1×10^{-7}
Co-60	2.46×10^{-14}	5×10^{2}	1.23×10^{-11}	5×10^{-5}	7.9×10^{-13}	1.6×10^{-8}
Fe-59	2.48×10^{-14}	3×10^{3}	7.4×10^{-11}	6×10^{-5}	4.7×10^{-12}	7.8×10^{-8}
Sr-89	9.25×10^{-14}	5×10^{-1}	4.6×10^{-14}	3×10^{-6}	2.9×10^{-15}	9.7×10^{-10}
Sr-90	4.53×10^{-15}	5×10^{-1}	2.3×10^{-15}	3×10^{-7}	1.5×10^{-16}	5.0×10^{-10}
Sr-91	8.92×10^{-16}	5×10^{-1}	4.5×10^{-16}	7×10^{-5}	2.9×10^{-17}	4.1×10^{-13}
Sr-92	9.60×10^{-17}	5×10^{-1}	4.8×10^{-17}	7×10^{-5}	3.1×10^{-18}	4.4×10^{-14}
Y-90	4.70×10^{-16}	1×10^{2}	4.7×10^{-14}	2×10^{-5}	3.0×10^{-15}	1.5×10^{-10}
Y-91	1.57×10^{-13}	1×10^{2}	1.6×10^{-11}	3×10^{-5}	1.0×10^{-12}	3.3×10^{-8}
Y-92	1.20×10^{-16}	1×10^{2}	1.2×10^{-14}	6×10^{-5}	7.7×10^{-16}	1.3×10^{-11}
Zr-95	1.82×10^{-14}	1×10^{2}	1.8×10^{-12}	6×10^{5}	1.2×10^{-13}	2.0×10^{-9}
Mo-99	1.61×10^{-11}	1×10^{1}	1.6×10^{-10}	2×10^{-4}	1.0×10^{-11}	5.0×10^{-8}
Nb-95	1.49×10^{-14}	3×10^{4}	4.5×10^{-10}	1×10^{-4}	2.9×10^{-11}	2.9×10^{-7}
I-131	2.09×10^{-11}	1×10^{1}	2.1×10^{-10}	3×10^{-7}	1.3×10^{-11}	4.3×10^{-5}

TABLE 11.2-7 (Cont)

Isotope	Discharge Canal Concentration (μCi/cc)	Concentration Factor	Activity in Fish (μCi/cc)	(MPC)w (μCi/cc)	Activity in Fish x Ingestion Factor (μCi/cc)	Fraction of (MPC)w
I-132	9.66×10^{-14}	1×10^{1}	9.7×10^{-6}	8×10^{-6}	6.2×10^{-14}	7.8×10^{-9}
I-133	3.82×10^{-12}	1×10^{1}	3.8×10^{-11}	1×10^{-6}	2.4×10^{-12}	2.4×10^{-6}
I-134	2.41×10^{-14}	1×10^{1}	2.4×10^{-13}	2×10^{-5}	1.5×10^{-14}	7.5×10^{-10}
I-135	6.55×10^{-13}	1×10^{1}	6.6×10^{-12}	4×10^{-6}	4.2×10^{-13}	1.0×10^{-7}
Te-132	9.05×10^{-13}	1×10^{1} (1)	9.0×10^{-12}	3×10^{-5}	5.8×10^{-13}	1.9×10^{-8}
Te-134	1.01×10^{-15}	1×10^{1} (1)	1.0×10^{-14}	--	6.4×10^{-16}	--
Cs-134	8.51×10^{-12}	3×10^{1}	2.6×10^{-10}	9×10^{-6}	1.7×10^{-11}	1.9×10^{-6}
Cs-136	2.09×10^{-12}	3×10^{1}	6.3×10^{-11}	9×10^{-5}	4.0×10^{-12}	4.4×10^{-8}
Cs-137	4.36×10^{-11}	3×10^{1}	1.3×10^{-9}	2×10^{-5}	8.3×10^{-11}	4.2×10^{-6}
Cs-138	2.46×10^{-14}	3×10^{1}	7.4×10^{-13}	--	4.7×10^{-14}	--
Ba-140	5.34×10^{-14}	1×10^{1}	5.3×10^{-13}	3×10^{-5}	3.4×10^{-14}	1.1×10^{-9}
La-140	2.46×10^{-15}	1×10^{2}	2.5×10^{-13}	2×10^{-5}	1.6×10^{-14}	8.0×10^{-10}
Ce-144	1.00×10^{-14}	1×10^{2}	1.0×10^{-13}	1×10^{-5}	6.4×10^{-14}	6.4×10^{-9}
Pr-144	4.05×10^{-18}	1×10^{2}	4.0×10^{-16}	--	2.6×10^{-17}	--
TOTAL	9.80×10^{-11}(2)	--	2.6×10^{-9}(2)	--	--	6.1×10^{-5}

NOTES:

(1) Abstracted from reference "A Model for the Approximate Calculation of Safe Rates of Discharge of Radioactive Waste into Marine Environs," A. M. Freke, Health Physics, Vol. 13, pp. 743-758, 167.

(2) Excluding H-3.

TABLE 11.2-8

CONCENTRATION OF RADIONUCLIDES IN BLUE CRABS DUE TO PLANT OPERATION

Isotope	Discharge Canal Concentration (μCi/cc)	Concentration Factor	Activity in Crab (μCi/cc)	(MPC)w (μCi/cc)	Activity in Fish x Ingestion Factor (μCi/cc)	Fraction of (MPC)w
H - 3	4.54×10^{-7}	9.3×10^{1}	4.2×10^{-7}	3×10^{-3}	2.7×10^{-8}	9.0×10^{-6}
Cr - 51	1.88×10^{-14}	2×10^{3}	3.8×10^{-11}	2×10^{-3}	2.4×10^{-12}	1.2×10^{-9}
Mn - 54	2.45×10^{-14}	5×10^{3}	1.2×10^{-10}	1×10^{-4}	7.7×10^{-12}	7.7×10^{-8}
Mn - 56	3.32×10^{-15}	5×10^{3}	1.7×10^{-11}	1×10^{-4}	1.1×10^{-12}	1.1×10^{-8}
Co - 58	6.69×10^{-13}	1×10^{3}	6.7×10^{-10}	1×10^{-4}	4.3×10^{-11}	4.3×10^{-7}
Co - 60	2.46×10^{-14}	1×10^{3}	2.5×10^{-11}	5×10^{-5}	1.6×10^{-12}	3.2×10^{-8}
Fe - 59	2.48×10^{-14}	2×10^{4}	5.0×10^{-10}	6×10^{-5}	3.2×10^{-11}	5.3×10^{-7}
Sr - 89	9.25×10^{-14}	6.3×10^{0}	5.8×10^{-13}	3×10^{-6}	3.7×10^{-14}	1.2×10^{-8}
Sr - 90	4.53×10^{-15}	6.3×10^{0}	2.9×10^{-14}	3×10^{-7}	1.9×10^{-15}	6.3×10^{-9}
Sr - 91	8.92×10^{-16}	6.3×10^{0}	5.6×10^{-15}	7×10^{-5}	3.6×10^{-16}	5.1×10^{-12}
Sr - 92	9.60×10^{-17}	6.3×10^{0}	6.0×10^{-16}	7×10^{-5}	3.8×10^{-17}	5.4×10^{-13}
Y - 90	4.70×10^{-16}	1×10^{3}	4.7×10^{-13}	2×10^{-5}	3.0×10^{-14}	1.5×10^{-9}
Y - 91	1.57×10^{-13}	1×10^{3}	1.6×10^{-10}	3×10^{-5}	1.0×10^{-11}	3.3×10^{-7}
Y - 92	1.20×10^{-16}	1×10^{3}	1.2×10^{-13}	6×10^{-5}	7.7×10^{-15}	1.3×10^{-10}
Zr - 95	1.82×10^{-14}	1×10^{3}	1.8×12^{-11}	6×10^{-5}	1.2×10^{-12}	2.0×10^{-8}
Mo - 99	1.61×10^{-11}	1×10^{1}	1.6×10^{-10}	2×10^{-4}	1.0×10^{-11}	5.0×10^{-8}

TABLE 11.2-8 (Cont)

Isotope	Discharge Canal Concentration (μCi/cc)	Concentration Factor	Activity in Crab (μCi/cc)	(MPC)w (μCi/cc)	Activity in Fish x Ingestion Factor (μCi/cc)	Fraction of (MPC)w
Nb - 95	1.49×10^{-14}	1×10^{2}	1.5×10^{-12}	1×10^{-4}	9.6×10^{-14}	9.6×10^{-10}
I - 131	2.09×10^{-11}	5×10^{1}	1.0×10^{-9}	3×10^{-7}	6.4×10^{-11}	2.1×10^{-4}
I - 132	9.66×10^{-14}	5×10^{1}	4.8×10^{-12}	8×10^{-6}	3.1×10^{-13}	3.9×10^{-8}
I - 133	3.82×10^{-12}	5×10^{1}	1.9×10^{-10}	1×10^{-6}	1.2×10^{-11}	1.2×10^{-5}
I - 134	2.41×10^{-14}	5×10^{1}	1.2×10^{-12}	2×10^{-5}	7.7×10^{-14}	3.8×10^{-9}
I - 135	6.55×10^{-13}	5×10^{1}	3.3×10^{-11}	4×10^{-6}	2.1×10^{-12}	5.2×10^{-7}
Te - 132	9.05×10^{-13}	1×10^{1} (1)	9.0×10^{-12}	3×10^{-5}	5.8×10^{-13}	1.9×10^{-8}
Te - 134	1.01×10^{-15}	1×10^{1} (1)	1.0×10^{-14}	-	6.4×10^{-16}	-
Cs - 134	8.51×10^{-12}	2×10^{1}	1.7×10^{-10}	9×10^{-6}	1.1×10^{-11}	1.2×10^{-6}
Cs - 136	2.09×10^{-12}	2×10^{1}	4.2×10^{-11}	9×10^{-5}	2.7×10^{-12}	3.0×10^{-8}
Cs - 137	4.36×10^{-11}	2×10^{1}	8.7×10^{-10}	2×10^{-5}	5.6×10^{-11}	2.8×10^{-6}
Cs - 138	2.46×10^{-14}	2×10^{1}	4.9×10^{-13}	-	3.1×10^{-14}	-
Ba - 140	5.34×10^{-14}	2×10^{2}	1.1×10^{-11}	3×10^{-5}	7.0×10^{-13}	2.3×10^{-8}

TABLE 11.2-8 (Cont)

Isotope	Discharge Canal Concentration (μCi/cc)	Concentration Factor	Activity in Crab (μCi/cc)	(MPC)w (μCi/cc)	Activity in Fish x Ingestion Factor (μCi/cc)	Fraction of (MPC)w
La - 140	2.46×10^{-15}	1×10^{3}	2.5×10^{-12}	2×10^{-5}	1.6×10^{-13}	8.0×10^{-9}
Ce - 144	1.00×10^{-14}	1×10^{3}	1.0×10^{-11}	1×10^{-5}	6.4×10^{-13}	6.4×10^{-8}
Pr - 144	4.05×10^{-18}	1×10^{3}	4.0×10^{-15}	-	2.6×10^{-16}	-
TOTALS	9.80×10^{-11} (2)	-	4.0×10^{-9} (2)	-	-	2.4×10^{-4}

NOTES:

(1) Abstracted from reference "A Model for the Approximate Calculation of Safe Rates of Discharge of Radioactive Waste into Marine Environs," A.M. Freke, Health Physics, Vol. 13, pp. 743-758, 167.

(2) Excluding H-3.

TABLE 11.2-9

POTENTIAL RADIATION EXPOSURE PATHWAYS TO MAN (LIQUID)
(Based on 0.2 percent failed fuel defects)

Pathway	Individual Assumed Exposed	Calculated Exposure
Drinking River Water	None- water not potable	0
Radioactivity in Ground Water	None - See Section 11.1.3.1 of the FSAR	0
Ingestion of Fish	Individual eats 50,000 gm/yr of fish	0.01 mrem/yr
Ingestion of Crabmeat	Individual eats 50,000 gm/yr of crabmeat	0.03 mrem/yr
Swimming in River	Individual swims 200 hr/yr in discharge canal	0.0011 mrem/yr
Boating on River or Fishing on Shoreline	Individual boats or fishes 200 hr/yr	0.00055 mrem/yr

Figure F11.2-1A Sheets 1 through 5 of 5 intentionally deleted.

Refer to plant drawing 205239 in DCRMS

Figure F11.2-1B Sheets 1, 2 & 3 of 3 intentionally deleted.

Refer to plant drawing 205339 in DCRMS

11.3 GASEOUS WASTE SYSTEM

The Gaseous Waste System (GWS) provides controlled handling and disposal of gaseous wastes generated during plant operation. The system also supplies hydrogen and nitrogen to primary systems' components as required during normal operation. The system is designed to minimize exposure to plant personnel and the general public, in accordance with Nuclear Regulatory Commission (NRC) regulations. In this section, the system is described and evaluated.

11.3.1 Design Objectives

Design objectives for the GWS are the following:

1. To provide sufficient capacity and storage to process and store the volume of gaseous effluent expected for a period of 45 days

2. To provide cover gas for the liquid holdup tanks

3. To assure that releases of radioactive gaseous wastes are kept as low as practicable.

4. To maintain releases below the limits set by 10CFR20

5. To assure that exposures to the public are maintained below the design objective of 10CFR50 Appendix I

The design criteria for the GWS are as follows:

The facility design shall include those means necessary to maintain control over the plant radioactive gaseous effluents. Appropriate holdup capacity shall be provided for retention of gaseous effluents, particularly where unfavorable environmental conditions can be expected to require operational limitations upon the release of radioactive effluents to the environment. In all

cases, the design for radioactivity control shall be justified 1) on the basis of 10CFR20 requirements, for both normal operations and for any transient situation that might reasonably be anticipated to occur and 2) on the basis of 10CFR50.67 dosage level limits for potential reactor accidents of exceedingly low probability of occurrence.

Gaseous waste facilities are designed so that discharge of effluents are in accordance with applicable governmental regulations.

Radioactive gases entering the GWS are collected in tanks to allow for decay and isotopic analysis. The system design and operation is directed toward minimizing releases to unrestricted areas. Discharge streams are appropriately monitored and safety features are incorporated to preclude releases in excess of the limits of 10CFR20.

Radioactive gases are pumped by compressors through a manifold to one of the gas decay tanks where they are held for a suitable period of time to allow for decay. Cover gases in the Nitrogen Blanketing System can be reused to minimize gaseous wastes. During normal operation, decayed gases are discharged intermittently at a controlled rate from these tanks through the monitored plant vent. The system is provided with discharge controls.

11.3.2 System Description

During plant operations, gaseous wastes will originate from the following:

1. Degassing reactor coolant discharge to the Chemical and Volume Control System (CVCS)

2. Displacement of cover gases as liquids accumulate in various tanks

3. Miscellaneous equipment vents and relief valves

4. Sampling operations and automatic gas analysis for hydrogen and oxygen in cover gases.

The GWS consists of two waste gas compressors and four waste decay tanks. During normal operation, the GWS supplies nitrogen to plant components. Two liquid nitrogen storage tanks, each with a self contained (ambient) vaporizer are supplied. One storage tank and its vaporizer is used at a time to supply the operating headers for both units. The pressure regulator in the operating header of each unit is set for 100 psig discharge. Each operating header is backed up by a nitrogen (gaseous) cylinder manifold with a pressure regulator set at 90 psig. When the operating header is below 100 psig, an alarm will alert the operator. The backup header will come into service automatically at 90 psig to assure a continuous supply of gas. After the operating header has been switched over to the standby liquid nitrogen storage tank, and the operating header pressure restored to 100 psig, the flow from the backup header will drop to zero. In addition to use as a backup nitrogen supply to the Waste Disposal System (WDS), the nitrogen (gaseous) cylinder manifold also supplies high pressure nitrogen gas for recharging accumulators. A hydrogen cylinder manifold is included in the Gaseous Waste Disposal System. It serves as a backup supply for hydrogen feed to the volume control tank. Normal feed is from the bulk hydrogen Control System.

Most of the gas received by the WDS during normal operation is cover gas displaced from the CVCS holdup tanks as they fill with liquid. Since this gas must be replaced when the tanks are emptied during processing, facilities are provided to return gas from the decay tanks to the holdup tanks. A backup supply from the nitrogen header is provided for makeup if return flow from the gas decay tanks is not available. To avoid the possibility of hydrogen combustion in the vent header system while gas is being displaced from holdup tanks to the vent header, components discharging to the vent header system are restricted to those

containing no air or aerated liquids and the vent header itself is designed to operate at a slight positive pressure (0.5 psig minimum to 4.0 psig maximum) to prevent inleakage. Outleakage from the system is minimized with Saunders patent diaphragm valves, bellows seals, self-contained pressure regulators and soft-seated packless valves throughout the radioactive portions of the system.

Gases vented to the vent header flow to the waste gas compressor suction header. One of the two compressors is in continuous operation with the second unit instrumented to act as backup for peak load conditions or failure of the first unit. From the compressors, gas flows to one of four gas decay tanks. The control arrangement on the gas decay tank inlet header allows the operator to place one tank in service and to select one tank for backup. When the tank in service becomes pressurized to 92 psig, a pressure transmitter automatically closes the inlet valve to that tank, opens the inlet valve to the backup tank and sounds an alarm to alert the operator so he may select a new backup tank. Pressure indicators are provided to aid the operator in selecting the backup tank.

Gas held in the decay tanks can either be returned to the CVCS holdup tank or discharged to the atmosphere if it has decayed sufficiently for release. Generally, the last tank to receive gas will be the first tank emptied back to the holdup tanks which permits the maximum decay time before releasing gas to the environment. However, the header arrangement at the tank inlet gives the operator the option to fill, reuse or discharge gas to the environment simultaneously without restriction by operation of the other tanks.

During degassing of the reactor coolant prior to a cold shutdown, for example, it may be desirable to pump the gas purged from the volume control tank into a particular gas decay tank and isolate that tank for decay rather than reuse the gas in it. This is done merely by aligning the control to open the inlet valve to the

desired tank and closing the outlet valve to the reuse header. Simultaneously, one of the other tanks can be opened to the reuse header if desired, while another is discharged to atmosphere.

Before a tank is discharged to the environment, it is sampled and analyzed to determine and record the activity to be released, and then discharged to the plant vent at a controlled rate, and monitored for gross activity.

During operation, gas samples are drawn automatically from the gas decay tanks and automatically analyzed to determine their hydrogen and oxygen content. There should be no significant oxygen content in any of the tanks, and an alarm will warn the operator if any sample shows 2 percent or higher by volume of oxygen. This allows time to take required action before the combustible limits of hydrogen-oxygen mixtures are reached. Another tank is placed in service while the operator locates and eliminates the source of oxygen.

The system is controlled from a central panel in the Auxiliary Buildings. Malfunction of the system is alarmed in the Auxiliary Building, and annunciated in the Control Room. All system equipment is located in the Auxiliary Building.

The Unit 1 & Unit 2 auxiliary feedwater storage tanks are provided with a nitrogen purge/blanket system in order to control the dissolved oxygen concentration in the water. Each nitrogen purge/blanket system is provided with a dedicated nitrogen source.

The GWS process flow diagrams are shown on Plant Drawings 205240 and 205340.

11.3.3 System Design

Gas Decay Tanks

Four welded carbon steel tanks per unit are provided to contain waste gases (hydrogen, nitrogen, and fission gases). Each tank conforms to ASME Boiler and Pressure Vessel Code Section III, Class C. Design data are as follows:

Volume, each (ft^3)	525
Design pressure (psig)	150
Design temperature (°F)	180
Operating pressure (psig)	0 - 92
Operating temperature (°F)	50 - 150
Type	Vertical cylinder

Waste Gas Compressors

There are two waste gas compressors per system to provide continuous removal of gases discharged to the vent header. Only one unit is normally in operation. The second unit is provided for backup during peak load conditions, such as when degassing the reactor coolant or for service when the first unit is down for maintenance. The compressors are water sealed, rotary, positive displacement units in which the water is used to displace and compress the gas being moved. Each compressor has a capacity of 40 cfm at 105 psig. The seal water is cooled, in a heat exchanger, by the component cooling water. Makeup water for the seal is supplied to the compressor suction from the Component Cooling System.

Each compressor contains a mechanical seal to minimize leakage of seal water.

The compressor discharges a mixture of waste gas and water into the separator. In the separator, the water is centrifuged out of the mixture and is accumulated in the bottom of the separator.

The discharge from the separator is saturated at the discharge pressure and temperature of the gas. At 40 cfm, 105°F cooling water and 105 psig discharge, water vapor carryover is based on the following inlet conditions:

Gas	Inlet Temperature	Vapor Carryover
Saturated N_2	130°F	0.87 lb/min
Saturated N_2	140°F	0.352 lb/min
65% by Volume N_2 and 35% H_2	130°F	0.019 lb/min
65% by Volume N_2 and 35% H_2	140°F	0.355 lb/min

In order to assure that there will be sufficient pressure to circulate seal water at startup, the compressor discharge control valve on the separator is set to open at 50 psig. Proper water level in the separator is maintained by means of a liquid high level transmitter and a low level alarm and makeup switch. The liquid level transmitter actuates the high level drain valve. If the water level falls below the low level cutoff point, the low level switch opens the water makeup valve and water is introduced to the compressor through the inlet. Design data for the compressor are as follows:

Compressor

Number per unit	2
Type	Liquid piston rotary type
Design flow rate, N_2 (at 140°F, 2 psig) cfm	40
Design pressure (psig)	150
Design temperature (°F)	180
Normal operating pressure (psig)	
Suction	0.5 - 4.0
Discharge	0 - 92
Normal operating temperature (°F)	70 - 130

Compressor Motor

H.P.	25
RPM	3500
Volts	460
Phase	3
Cycle	60
Rise	90°C
Ambient temperature	40°C
Dripproof Enclosure Class B	
Powerhouse insulation	

Nitrogen Manifold

A supply header from the Liquid Nitrogen System supplies nitrogen gas to purge the vapor spaces of various components, to reduce hydrogen concentrations or replace fluid in emptying tanks. Pressure controllers (I-PIA-1066) which switch from the normal bulk supply to a backup gaseous cylinder header, assure a continuous flow of gas. Pressure regulator 1-PCV-1043 in the backup header is set at 90 psig which is lower than the 100 psig

in the operating header. When the operating header supply from one bulk tank is exhausted, the discharge pressure of this header will fall below the setpoint pressure of the backup header, which will come into service automatically.

This system has the additional function of supplying N_2 at 800 psi to the accumulator in the Safety Injection System. If the need ever arises, this pressurized gas will inject borated liquid from the accumulators into the reactor coolant loops. Design data for the manifold are as follows:

Type	Automatic switching dual header
Number per unit	1
Number of separate headers per package	2
Number of cylinders per header	18
Design flow rate, scfm	40
Design delivery pressure, psig	100

Station Bulk Low Pressure Nitrogen Supply

A station bulk low pressure (LP) nitrogen supply package has been added to the above system to provide additional capability. Two liquid nitrogen storage tanks, each with a self-contained vaporizer are supplied. One storage tank and its vaporizer are used at a time to supply the operating headers for both units. Design data are as follows:

Type	Vertical cylindrical, Double walled
N_2	55,866 scf
O_2	69,030 scf
Argon	67,470 scf
Operating pressure (Max)	245 psig
Design pressure (Max)	249 psig
Design temperature	-320°F - 100°F
Empty weight	4,400 lbs

Hydrogen Manifold

A dual manifold serves as a backup to the Bulk Hydrogen System to supply hydrogen to the volume control tank and to maintain the hydrogen partial pressure as hydrogen dissolves in the reactor coolant. A pressure controller (1-PIA-1065) which automatically switches from the normal system to the backup system, assures a continuous supply of gas. The operation of the backup header is essentially the same as for the Nitrogen Manifold System. Design data are as follows:

Type	Automatic switching dual header
Number per unit	1
Number of separate headers per package	2
Number of cylinders per header	6

Design flow rate, scfm 30

Design delivery pressure, psig 100

Gas Analyzer

Redundant gas analyzers, one in each Salem Unit and both cross-connected, are provided in accordance with the recommendations of NUREG-0472 to automatically monitor the concentrations of oxygen and hydrogen in the system, in order to indicate when the accumulation of these gases approaches an explosive mixture. Upon indication by alarm that the oxygen level is approaching a hazardous level, provisions must be made to either isolate the component or purge with nitrogen to the GWS. The gas analyzer has suitable connections for sampling when necessary from the following components:

Waste gas to plant vent

Reactor coolant drain tank

Spent resin storage tank

Gas decay tanks (2 points)

CVCS holdup tanks

Boric acid evaporator and gas stripper
Volume control tank

Pressure relief tank

Gas decay tank samples are analyzed continuously to ensure that the oxygen concentration remains less than or equal to 2 percent. Separate feed lines with calibration gases are provided for analyzer calibration purposes. The

high-span calibration gas is nominally 4% oxygen, and low-span calibration gas is nominally 1% oxygen. The balance of the calibration mixtures consists of nitrogen, except for small amounts of hydrogen (between 1% and 2.5%). The gas mixture allows calibration of the analyzer to the profile expected in the sample stream at alarm conditions. Design data for the analyzers are as follows:

Oxygen	By partial pressure measurement 0-5% O_2 Range
Hydrogen	By partial pressure measurement 0-25% H_2 Range
Recorder printout (chart)	Waste Gas Decay Tank: every 3 minutes Sequential sampling (cover gas): each point

All major equipment in the Gaseous Radwaste Disposal System is located outside of the Reactor Containment Building in the Auxiliary Building, Elevation 64 feet and 122 feet.

Piping

Gas piping is mainly carbon steel with stainless steel piping in some sections installed as part of modifications. Piping connections are welded except where flanged connections are necessary to facilitate equipment maintenance.

Valves exposed to gases are either carbon steel or stainless steel. Isolation valves are provided to isolate each piece of equipment for maintenance, to direct the flow of waste through the system, and to isolate storage tanks for radioactive decay.

Relief valves are provided for tanks containing radioactive wastes if the tanks might be over-pressurized by improper operation or component malfunction.

Codes and Standards

Additional information is presented in Table 11.2-3 for system piping, valves and compressors.

11.3.4 Operating Procedures

The gaseous wastes processed by this system consist primarily of hydrogen stripped from reactor coolant during boron recycle and degassing operations and nitrogen from the various tank cover gases and from the degassing operation. These gases are discharged to the vent header which feeds the suction of the waste gas compressors.

One of the two waste gas compressors will be operating with the other compressor being on standby. The operating compressor maintains a vent header pressure of 0.5 to 4.0 psig. If the vent header pressure rises to 4 psig, the standby compressor automatically energizes. The compressors can be used to: 1) pump gas to the waste decay tanks; 2) transfer gas between tanks; and 3) pump gas directly to the CVCS holdup tanks.

To pump gas to the gas decay tanks, the operator selects two tanks at the auxiliary control panel No. 104: one to receive gas, and one for standby. When the tank in-service is pressurized to 92 psig, flow is automatically switched to the standby tank and an alarm alerts the operator to select a new standby tank. The decay tank being filled is sampled automatically by the gas analyzer and an alarm will alert the operator to a high oxygen content. The tank must then be isolated and the operator is required to direct flow to the standby tank and select a new standby tank.

As the liquid in the CVCS holdup tanks is processed by the boric acid evaporator, gas must be provided as cover gas to replace the processed liquid. The cover gas may be provided from any of the gas decay tanks or from the nitrogen supply. The gas decay tank supplying the returning cover gas is selected manually at the auxiliary control panel No. 104 by opening the appropriate valve in the return line header. To maximize total residence time for gas decay in the system, the last tank filled should be the first tank returned as cover gas. A backup supply of gas to the holdup tanks is provided from the bulk nitrogen header for makeup when return flow is not available from the decay tanks.

Before a gas decay tank is discharged to the plant vent for release to atmosphere, a sample must be taken to determine activity concentration of the gas and total activity inventory in the tank. Total tank activity inventory is determined from the activity concentration and pressure in the tank. To release the gas, the appropriate local manual stop valve is opened to the plant vent and the gas discharge modulating valve is opened at the auxiliary control panel. If the Plant Vent Radiation Monitor detects high activity during release, the modulating valve automatically trips closed. To reopen the valve, the switch must first be reset by returning it to the closed position. The valve can then be repositioned.

The equipment which connects with the vent header system is limited in number. Under normal operating conditions no air is permitted to enter the vent header. During maintenance operations air could enter the boric acid evaporator vent condenser or the waste evaporator vent condenser. During maintenance operations on either of these pieces of equipment, the valve on the equipment discharge line to the vent header is closed. When maintenance operations are completed, and prior to opening the valves, the equipment is filled with nitrogen to purge the air. During discharge, the nitrogen purge is continued. No fluids can get into the vent header.

The maximum allowable release rate of gaseous radioactivity is specified in the Offsite Dose Calculation Manual.

A record of all releases is kept.

11.3.5 Performance Tests

Periodic inspection of waste gas compressors shall be done in accordance with the manufacturer's technical manual.

11.3.6 Estimated Releases

HISTORICAL NOTE:

The radiological release values originally contained in this section were calculated in support of initial licensing and have been deleted. Off-site releases during normal plant operations are controlled by the Radioactive Effluent Control Program.

Several sources of potential release of gaseous radioactivity to the environment have been identified. Each is discussed separately below.

Gas Decay Tanks

Gaseous wastes consist primarily of hydrogen stripped from coolant discharged to the CVCS holdup tanks during boron dilution, nitrogen and hydrogen gases purged from the CVCS volume control when degassing the reactor coolant, and nitrogen from the closed gas blanketing system. The gas decay tank capacity will permit adequate time to allow for decay of waste gas activity release based on 1-percent defective fuel clad, and 3423 MWt power with daily load reductions to 50-percent power for several hours.

Containment Purging

Purging of the containment will take place infrequently, on the order of two to three times a year per unit, to keep concentrations of radioactive gases in the containment within specified limits to allow plant personnel to enter the containment periodically for maintenance and inspection.

Section 9.4 describes the methods employed to minimize release to the environment from containment purging.

Diaphragm valves are used in the GWS vent header to eliminate steam leakage. All pipe connections in the vent headers are welded. Therefore, there will be no effect on the annual release of gaseous radioisotopes.

11.3.7 Release Points

Release points are shown on the system flow diagrams (Plant Drawings 205240 and 205340).

11.3.8 Dilution Factors

See Section 11.3.9.

11.3.9 Estimated Doses

The radiological release doses originally contained in this section were calculated in support of initial licensing and have been deleted. Off-site doses during normal plant operations are controlled by the Radioactive Effluent Control Program and the ODCM.

TABLE 11.3-1

(Historical Information)

ESTIMATED ANNUAL GASEOUS RELEASE BY ISOTOPE(1)
FROM GAS DECAY TANKS

(Per Unit)

Isotope	Activity Release to Environment Curies/yr
Kr 85	5450
Kr 85m, 87, 88	Negligible
Xe 133	2000
Xe 133m, 135, 135m 138	Negligible
Total	7450

NOTE:

(1) Based on 1 percent defective fuel, 3423 MWt core, load follow and 45 days holdup.

TABLE 11.3-2

(Historical Information)

ESTIMATED TOTAL RADIOACTIVE GASEOUS RELEASES
(TWO UNIT BASIS)

Isotope	Estimated Annual Release (Curies)
H-3	248
Kr-85	11450
Kr-85m	195
Kr-87	113
Kr-88	309
Xe-133	26850
Xe-133m	240
Xe-135	633
Xe-135m	10
I-131	0.23

TABLE 11.3-3

(Historical Information)

ESTIMATED OFF-SITE RADIATION EXPOSURES
GASEOUS RADIOACTIVE RELEASES

(Two Unit Basis)

		Radiation Exposure (mrem) at 1270 Meters, North Sector		
Isotope	Annual Release (Curies)	Finite Cloud* Whole Body Dose (mrem)	Semi-Infinite Cloud(1) Whole Body Dose (mrem)	Infinite Cloud(2) Skin Dose (mrem)
H-3	248	0	0	0.0007
Kr-85	11450	0.0033	0.012	1.12
Kr-85m	195	0.0039	0.008	0.023
Kr-87	113	0.0059	0.046	0.073
Kr-88	309	0.058	0.15	0.053
Xe-133	26850	0.227	0.55	1.94
Xe-133m	240	0.0062	0.0016	0.023
Xe-135	633	0.017	0.042	0.087
Xe-135m	10	0.0003	0.0011	0.0004
I-131	0.23	negl	negl	negl
Totals	40,000	0.269	0.81	3.32

NOTES:

(1) gamma energy
(2) beta energy

TABLE 11.3-4

(Historical Information)

POTENTIAL RADIATION EXPOSURE PATHWAYS TO MAN (GASEOUS)

Pathway	Individual Assumed Exposed	Calculated Exposure
External Exposure	Individual stands at nearest site boundary 100 percent of time	0.054 mrem/yr (whole body-finite cloud)
		0.16 mrem/yr (whole body-semi - infinite cloud)
		0.66 mrem/yr (skin - infinite cloud, beta energy only)
Ingestion of Milk	Young child drinks entire intake of milk from nearby dairy farms. Cows assumed to graze 9 months per year with grass making up to 100 percent of diet during that time.	0.26 mrem/yr (farm 4.1 miles NW of site)

Figure F11.3-1A Sheets 1, 2 & 3 of 3 intentionally deleted.

Refer to plant drawing 205240 in DCRMS

Figure F11.3-1B Sheets 1, 2 & 3 of 3 intentionally deleted.

Refer to plant drawing 205340 in DCRMS

11.4 RADIOLOGICAL MONITORING

11.4.1 Design Objectives

Design objectives for the Radiation Monitoring System (RMS) are as follows:

1. Warn of any radiation hazard which might develop

2. Give early warning of a plant malfunction which might lead to a radiation hazard or plant damage

3. Provide assurance that personnel exposure does not exceed 10CFR20 limits

4. Provide assurance that atmospheric releases will not exceed the design objectives of 10CFR50

5. Record the activity present at various plant locations

6. Provide data for radiological analyses and reports

7. Monitor process system filters for radiation buildups

11.4.2 Radiation Monitoring System

The RMS provides instrument channels, located at selected points in and around the plant to detect, compute, and record the radiation levels. In the event the radiation level should rise above a desired setpoint, an alarm will be initiated in the Control Room. The RMS operates in conjunction with regular and special radiation surveys and with chemical analyses performed by the plant staff to meet the radiological monitoring design objectives presented in Section 11.4.1.

The RMS signal processing equipment is centralized in six cabinets for Unit 1 and in three cabinets for Unit 2. High reliability and

ease of maintenance are emphasized in the design of this system. Cabinet equipment is equipped with sliding channel drawers for rapid replacement of units, assemblies and entire channels. It is possible to completely remove the various chasses from the cabinet, after disconnecting the cables from the rear of these units.

The components of the RMS are designed to meet or exceed the requirements of normal and DBA conditions for temperature, humidity, pressure and radiation, as stated in the Salem Generating Station Environmental Design Criteria.

The RMS is divided into the following subsystems:

1. The Process Radiation Monitoring System monitors various gaseous and liquid streams for indication of increasing radiation levels, and all identified effluent paths to establish the quantity of radioactivity being discharged to the environment.

2. The Process Filter Monitoring System monitors the buildup of radioactivity on various process filters to warn of unexpected radiation and to indicate the need for changing or cleaning the filter.

3. The Area Radiation Monitoring System monitors radiation levels in various locations of the plant to warn personnel of a deteriorating radiological condition. It is also useful in assessing the spread of radioactivity in a given area.

11.4.2.1 Radiation Monitoring System Description

11.4.2.1.1 Radiation Monitoring System - Unit 1

Except for 1R1B, 1R2, 1R3, 1R4, 1R5, 1R6A, 1R7, 1R9, 1R10A, 1R11A, 1R12A, 1R12B, 1R13A, 1R13B, 1R15, 1R17A, 1R17B, 1R18, 1R19A, 1R19B, 1R19C, 1R19D, 1R41A, 1R41B, 1R41C, 1R41D, 1R46A, 1R46B, 1R46C, 1R46D, 1R53A, 1R53B, 1R53C, and 1R53D, the Unit 1 RMS consists of analog channels which monitor radiation levels in various plant locations and operating systems. Monitors 1R1B, 1R41A, 1R41B, 1R41C, 1R46A, 1R46B, 1R46C, 1R46D, 1R53A, 1R53B, 1R53C and 1R53D have microprocessor-based electronics. A digital control and display module is located in the Control Equipment Room for monitors 1R1B, 1R41A, 1R41B and 1R41C, 1R46A, 1R46B, 1R46C, 1R46D, and in the Relay Room for monitors 1R53A, 1R53B, 1R53C and 1R53D. The output from each detector is transmitted via cables to the RMS cabinets in the Control Room area where the radiation level is indicated on a meter and pre-selected channels are recorded on a multipoint recorder. For area monitors, the radiation level is also indicated locally at the detector. High radiation level alarms are annunciated on the Control Room overhead annunciator and further identified at the RMS cabinets. For area monitors, a high radiation level is also alarmed at the detector location, except for area monitors located in the Control Room.

Each channel contains a completely integrated modular assembly, which includes the following.

1. Level Amplifier/Discriminator - Discriminates and amplifies the detector output to provide a discriminated and shaped pulse output to the log level amplifier.*

2. Log Level Amplifier - Accepts the shaped pulse of the level amplifier output,* performs a log integration (converts total pulse rate to a logarithmic analog signal), and amplifies the resulting output for suitable indication and recording.

* Note, monitors 1R1B, 1R41A/B/C/D and 1R53A/B/C/D have microprocessor based electronics that provide a direct digital conversion of detector output to CPM. 1R1B contains two channel inputs and monitors both Control Room area inlet ducts as illustrated on Figure 11.4-9.

3. Power Supplies - Individual power supplies are contained in each drawer for furnishing the positive and negative voltages for the transistor circuits, relays and alarm lights and for providing the high voltage for the detector.

4. Test-Calibration Circuitry - These circuits provide a pre-calibrated pulsed and/or analog signal to perform a channel test, and a solenoid operated radiation check source to verify the channel's operation. A light on the Control Room overhead annunciator indicates when any channel is in the test-calibrate mode, except 1R1B.

5. Radiation Level Meter - This meter, mounted on the assembly drawer, has a scale calibrated logarithmically from 10^1 to 10^6 counts per minute for process monitor channels and in mR/hr for area and filter monitor channels. For monitors 1R1B, 1R41A/B/C/D, and 1R53A/B/C/D, the displays are digital. Pre-selected signals are also recorded and displayed in the Control Room area.

6. Indicating Lights - These lights indicate high radiation levels and circuit failures. A light on the Control Room overhead annunciator is actuated on a high radiation signal and a yellow light on the RMS recorder panel indicates which channel. The Control Room alarm CRT provides discriminate 1R1B channel alarms. The yellow light is not applicable for 1R1B.

7. Bistable Circuits - Two bistable circuits are provided, one to alarm on high radiation (actuation point may be set at any level over the range of the instruments) and one to alarm on loss of signal (circuit failure).

8. Check Source - A remotely-operated long half-life radiation check source is furnished in each channel. The energy emissions are similar to the radiation energies being monitored. The source strength is sufficient to cause a visible increase in the meter indication. During checksource operation on R1B indication is frozen for both channels. If insufficient count rate is achieved (check source count rate compared against a setpoint), a norm failure alarm is provided.

RMS channels 1R2, 1R3, 1R4, 1R5, 1R6A, 1R7, 1R9, 1R10A, 1R13A, 1R13B, 1R15, 1R17A, 1R17B, 1R18, 1R19A, 1R19B, 1R19C, 1R19D, and 1R40 are microprocessor based digital instrumentation systems. This equipment is designed to power, operate, and monitor various types of radiation detectors. The system consists of a detector (GM tube, scintillation, or ion chamber), a local monitor, and a remote monitor. Local monitors perform pulse discrimination and shaping. Also, all calibration constants are stored in the local monitor. Remote monitors communicate with the local monitors via serial communication ports and provide analog outputs to the plant computer and indicators and contact outputs to alarm and interlocks.

RMS channels 1R11A, 1R12A, and 1R12B are microprocessor based digital instrumentation systems. A sample skid with a local microprocessor monitors containment atmosphere for particulate iodine and noble gas. Detector signals are processed locally. Alarm contacts are located at the microprocessor. Calibration and database constants are stored locally. A remote display unit communicates with the local microprocessor via a serial communication port. The remote display provides analog outputs to plant computers, indicators and contact outputs for annunciation.

11.4.2.1.2 Radiation Monitoring System - Unit 2

The Unit 2 RMS is primarily a microprocessor-based digital monitoring system. The system is basically a two-tiered structure with local field units and remote units in the Control Equipment Room. A simplified diagram of the general system structure is shown on Figure 11.4-1. 2R1B has microprocessor-based electronics and a digital control and display assembly in the Control Equipment room as illustrated on Figure 11.4-9.

Local Field Units

The local digital field units are located at selected points in the plant to detect airborne radioactivity, filter buildup radioactivity and process system radioactivity. Each field unit consists of a detector and a microprocessor/electronics cabinet. The units detect, compute, and indicate radiation data at their respective location. A digital display is provided for radiation level indication. Indicating lights are provided for alarm, warning, failure, and check source operation information. Auxiliary relay contacts are available for control system functions to indicate alarm, warning, and failure conditions. Keylock controls are used for testing, calibration, and entering data such as setpoints, conversion factors, and confidence levels.

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The system was designed to provide for the safe operation of the plant, to assure that personnel exposure does not exceed 10CFR20 limits, and to assure that environmental releases do not exceed Technical Specification limits. The Unit 2 system was designed to meet the same requirements as the Unit 1 system. The two Radiation Monitoring Systems perform essentially the same functions. There are, however, some differences in sensitivities, detector types, and monitoring channels.

11.4.2.2 Process Radiation Monitoring System Channel Description

The process monitors are utilized for monitoring process systems for potential radiation leakage and effluent discharge paths for normal releases and those following potential accidents. The monitors typically incorporate an offline liquid or gas sampling system. Some of the units monitor the process stream directly. Typical functional block diagrams of the process monitors are shown on Figures 11.4-2 through 11.4-7.

The Process Radiation Monitoring System is summarized in Tables 11.4-1 and 11.4-2 and consists of the following radiation

monitoring channels. The prefix numbers indicate monitors associated with Unit 1 or Unit 2.

Control Room Area Intake Duct Monitors (1-R1B and 2-R1B)

The Control Room Intake Air Radiation Monitoring System is a shared system. The R1B monitors provide redundant functions to monitor air drawn into the Control Room through the Unit 1 and Unit 2 air intakes to the Control Room. The dual channel processors with beta scintillation detectors in each of the Unit 1 and Unit 2 Control Room intake ducts provide the redundant initiation signals to place the ventilation system into its accident-pressurized mode of operation. In addition, the monitoring system provides continuous indication and recording of the radiation levels and annunciates in the Control Room the failures of the radiation monitoring equipment and the development of warning and alarm level radiation conditions. This is a safety-related channel.

Containment - Air Particulate Monitors (1-R11A and 2-R11A)

These monitors are provided to measure air particulate beta radioactivity in the containment and to ensure that the release rate through the plant vent during purging is maintained below specified limits. For Unit 1, channel R-11A takes a continuous air sample from the containment atmosphere. For Unit 2, channel 2-R11A takes a sample from the containment. The sample is drawn from the containment through a closed, sealed system and monitored by a scintillation counter-moving filter paper detector assembly. The filter paper collects 99 percent of all particulate matter greater than 1 micron in size on its constantly moving surface and is viewed by a photomultiplier-scintillation crystal combination.

The sample is returned to the containment or vent, depending on which source is being monitored. The detector assembly is in a completely enclosed housing. The detector is a hermetically sealed photomultiplier tube - scintillator combination. The filter paper has a 25-day minimum supply at normal speed. Lead shielding is provided to reduce the background level to where it does not interfere with the detector's sensitivity. The filter paper mechanism, an electro-mechanical assembly which controls the filter paper movement, is provided as an integral part of the detector unit.

Channel 1R11A (particulate) monitors containment for leak detection and effluent releases. Digital indication for 1R11A is located at the local and remote monitors. The monitors are set to indicate radiation from 10^1 to 10^6 cpm. The local monitor provides Normal, Warn and Alarm indications. The remote monitor in the Control Equipment Room, in addition to display and status indications on the monitor panel, provides analog outputs to the Safety Parameter Display System (SPDS), Plant Computer P250, indicators on Panel 1RP1, indication of high radiation on Panel 1RP1, and High/Trouble alarm on the Overhead Annunciator.

Containment/Plant Vent Radioactive Gas Monitors (1-R12A, 2-R12A, 1-R41C and 2-R41D)

These monitors are provided to measure gaseous radioactivity in the containment, and to ensure that the release rate through the plant vent during purging is maintained below specified limits. High radiation level initiates closure of the containment purge supply and exhaust duct valves and pressure relief line valves. For Unit 2, high radiation level also closes the waste gas discharge valve.

For Unit 1, channel 1-R12A takes a continuous air sample from the containment atmosphere. Channel 1-41D samples only the plant vent. For Unit 2, channel 2-R12A takes a sample from the containment and channel 2-R41D from the plant vent. All samples reach the gaseous detector after passing through the air particulate monitor or an air particulate sampler (1-R41D only). The sample is constantly mixed in the fixed, shielded volume, where it is viewed by beta scintillator. The sample is then returned to the source being monitored.

The detector assembly is in a completely enclosed housing containing a beta-gamma sensitive detector mounted in a constant gas volume container. Lead shielding is provided to reduce the background level to a point where it does not interfere with the detector's sensitivity.

Channel 1R12A (noble gas) monitors containment for effluent releases. Digital indication for 1R12A is located at the local and remote monitors. The monitors are set to indicate radiation from 10^1 to 10^6 cpm. The local monitor provides Normal, Warn and Alarm indications and provides an alarm relay contact for initiating closure of containment ventilation closure/isolation valves 1VC1, 4, 5, and 6 for Modes 1, 2, 3, 4 & 5. The remote monitor in the Control Equipment Room, in addition to display and status indications on the monitor panel, provides analog outputs to the Safety Parameter Display System (SPDS), Plant Computer P250, indicators on Panel 1RP1, indication of high radiation on Panel 1RP1, and High/Trouble alarm on the Overhead Annunciator.

Containment - Fixed Filter Iodine Monitor (1-R12B, 2-R12B)

Iodine is one of the more prominent isotopes requiring special surveillance. The containment monitoring system has been designed so that the sample flows first through the filter paper assembly and then through a charcoal cartridge. It is a scintillation type detector. For Unit 1, the sample is drawn from the containment for channel 1-R12B. Channel 1-R41C samples only the plant vent. For Unit 2, channel 2-R12B takes a sample from the containment.

High radiation level initiates closure of the containment purge supply and exhaust duct valves and pressure line relief valves. The abnormal conditions are alarmed in the Control Room and Control Equipment Room. A solenoid-operated check source is provided to give an instant checkout of the system functional status.

Channel 1R12B (iodine) monitors containment for effluent releases. Digital indication for 1R12B is located at the local and remote monitors. The monitors are set to indicate radiation from 10^1 to 10^6 cpm. The local monitor provides Normal, Warn and Alarm indications and provides an alarm relay contact for initiating closure of containment ventilation closure/isolation valves 1VC1, 4, 5, and 6. The remote monitor in the Control Equipment Room, in addition to display and status indications on the monitor panel, provides analog outputs to the Safety Parameter Display System (SPDS), Plant Computer P250, indicators on Panel 1RP1, indication of high radiation on Panel 1RP1, and High/Trouble alarm on the Overhead Annunciator.

Note

The containment radiation monitors (channels R11, R12A, and R12B) have elements common to all three channels of the particulate/noble gas/iodine monitoring assembly. These are described as follows:

1. The flow control assembly includes a pump unit and selector valves that provide a representative sample (or a "clean" sample) to the detectors.

2. The pump unit consists of:

   a. A pump to obtain the air sample

   b. A flowmeter to indicate the flow rate

   c. A flow control valve to provide flow adjustment

   d. A flow alarm assembly to provide low and high flow alarm signals

3. Selector valves are used to direct the desired sample to the detector for monitoring and to block flow from the sampling area when the channel is in the maintenance or "purging" condition.

4. A temperature sensor is used to protect the system from high temperature. This unit automatically closes the inlet motor operated valve upon a high temperature condition.

5. Purging is accomplished with a valve control arrangement whereby the normal sample flow is blocked and the detector purged with a "clean" sample. This facilitates detector calibration by establishing the background level and aids in verifying sample activity level.

6. For Unit 1, the flow control panel in the Control Room radiation monitoring racks permits remote operation of the flow control assembly. By operating a sample selector switch on the control panel, either the containment or a local "clean" sample may be monitored. For Unit 2, these controls are located on 2RP1 in the control room.

7. Deleted.

8. Containment isolation valves are provided for the containment sample piping. In addition, the containment isolation valves (for regular and backup flow paths) limit switch auxiliary relays are wired to the Auxiliary Annunciator System to provide a 1R11/12 loss of flow path alarm whenever both backup and regular flow paths are lost.

9. For Unit 1 and Unit 2, the containment particulate and gaseous monitors (1-R11A, 1-R12A, 2-R11A and 2-R12A) are also used as part of the Reactor Coolant Leak Detection System.

Alarm lights are actuated by the following:

1. Flow alarm assembly (low or high flow)

2. The pressure sensor assembly (high pressure)

3. The filter paper sensor (paper drive malfunction)

4. The pump power control switch (pump motor on)

Containment Fan Cooler Radiation Monitors (1-R13A and B and 2-R13A and B)

Service water is used as the cooling medium for the containment fan coolers and could be contaminated if the cooling coil leaks. Since the Service Water System discharges to the river, the fan cooler units will be monitored for radioactivity. This is done through the use of two monitors for the five fan coolers. Each monitor employs an in-line detector. Remote alarms and readout in the Control Room are also provided.

Condenser Air Removal Gas Monitors (1-R15 and 2-R15)

This channel monitors the discharge from the condenser air removal exhaust header for gaseous radiation which is indicative of a primary to secondary system leak. The gas discharge is routed to the plant vent. In Unit 1 a gamma sensitive scintillation detector is used to monitor the radiation level. The detector is inserted in an inline fixed volume container which includes adequate shielding to reduce the background radiation to where it does not interfere with the detector's sensitivity. In Unit 2, a gamma scintillator coupled to a photomultiplier tube is used.

Component Cooling Liquid Monitors (1-R17A, B and 2-R17A,B)

These channels continuously monitor the component cooling water for radiation. Leakage from the Reactor Coolant System and other systems' components to the component cooling water is detected by a scintillation counter located in an inline well. A high radiation level alarm signal initiates closure of the gas valve located in the component cooling surge tank vent line to prevent gaseous radiation release.

Waste Disposal System Liquid Effluent Monitors (1-R18 and 2-R18)

This channel continuously monitors all Waste Disposal System liquid releases from the plant. Automatic valve closure action is initiated by monitor after a high radiation level is indicated and alarmed in the Control Room. A scintillation counter in a fixed volume assembly monitors liquid effluent as it is discharged. Remote indication and annunciation are provided on the Waste Disposal System control board.

Steam Generator Blowdown Liquid Monitors (1-R19A, B, C, D, and 2-R19A, B, C, D)

Each of these channels (four channels per unit) monitors the liquid phase of the steam generators for radioactivity, which would indicate a primary-to-secondary system leak. The four steam generator blowdown sample lines each have a radiation monitor. In

Unit 1, the monitors are located in the Sampling Room where the blowdown sample has been cooled. In Unit 2, an offline sampling system is used.

A high radiation alarm signal will close the No. 12 (22) steam generator blowdown tank inlet valves and the steam generator blowdown isolation valves on the affected steam generator.

Letdown Line Monitors (1-R31A and 2-R31)

The Letdown Monitoring System for each unit consists of a single channel for monitoring total gross activity of the letdown line concentration. These monitors are also called failed fuel monitors. This purpose is to detect the failure of the cladding of one or more fuel elements by the gamma emission of fission products released into the reactor coolant.

1. Unit 1 System

   The system continuously measures gamma radiation intensity in a continuously flowing sample stream of reactor coolant water using a gamma scintillation detector. Besides continuous indication of the reactor coolant activity, high radiation conditions are alarmed in the Control Room.

   The detector is capable of measuring up to 10^6 cpm at which position it will reach the saturation condition. Provision is made for desensitizing the system by two or more decades to compensate for permanent activity buildup resulting from long-term normal operation. This

is accomplished by insertion of a lead spacer between the sensitive end of the detector and the letdown line.

2. Unit 2 System

A gamma scintillator is used to monitor the total gross gamma letdown line activity concentration. Signal processing is performed by the digital RMS to provide data on a significant increase of gross gamma activity. A significant increase of gross gamma activity would be indicative of a fuel cladding failure.

The detector is capable of measuring up to 1×10^{9} cpm. Provision is made for desensitizing the system by two or more decades to compensate for permanent activity buildup resulting from long-term normal operation. This is accomplished by insertion of a lead spacer between the sensitive end of the detector and the letdown line.

Evaporator and Feed Preheaters Condensate (1-R36)

Heating steam is supplied to the boric acid and waste evaporators and feed heater. Condensate from the evaporators and feed heater is returned to the condensate receivers from whence it is pumped back to the heating boiler. Steam is used in the tubes of the evaporators and in the heater for process heating. Since the evaporators and heater can contain radioactive fluids, a tube rupture could result in a contamination of the Condensate System, Heating Boiler System, and Heating Steam System.

This channel continuously monitors the activity in the common condensate piping from each unit's evaporators. This channel employs an offline sampler. A high radiation level alarm will automatically close the condensate line valve for each unit's evaporator packages. Alarm and indication are provided in the Control Room. A manually valved drain is provided for disposal of any contaminated condensate to the Waste Disposal System.

Nonradwaste Basin Discharge (2-R37)

The nonradwaste basin provides a potential release path due to the fact that steam generator blowdown is directed to the basin during plant startup. An offline sampler is provided to continuously monitor the discharge from the nonradwaste basin.

Plant Vent High Range Monitors (1-R41B-D and 2-R41B-D, 1-R45 and 2-R45)

The Plant Vent High Range Monitors consist of High Range Noble Gas Monitors (1-R41B-D and 2-R41B-D) and High Range Particulate and Iodine Sampling Skids (1-R45 and 2-R45).

The High Range Noble Gas Monitors comply with NUREG-0737, Item II.F.1, and the intent of Regulatory Guide 1.97, type C & E Category 2 requirements. The system provides a sampling capability of 10^{5} μci/cc for noble gases. The monitors are safety grade and qualified for the post-accident environment. However, the monitors do not perform any safety related function. The system is comprised of four noble gas channels as follows: low range noble gas (R41A), intermediate range noble gas (R41B), high range noble gas (R41C) and composite noble gas (R41D). Microprocessor software logic is used for the composite channel to display the effluent release rate based on low-, intermediate- or high-range noble gas concentrations (R41A, B or C, respectively) and the plant vent flow rate.

The High Range Particulate and Iodine Sampling Skids (1-R45 and 2-R45) comply with NUREG-0737, Item II.F.1, and the intent of Regulatory Guide 1.97, type E Category 3 requirements. The system provides a sampling capability of 10^{2} μci/cc for iodines and particulates.

Nonsafety-related heat tracing is provided to preclude freeze-up of the samplings lines during plant outages coincident with adverse weather conditions.

Main Steam High Range Monitors (1-R46A-D and 2-R46A-D)

The Main Steam High Range Monitoring System complies with NUREG-0737, Item II.F.1, and the intent of Regulatory Guide 1.97. The system provides a detection capability of 10^3 µci/cc. The monitors are safety grade and qualified for the post-accident environment. Channels R46A-D each monitor one of the main steam lines.

Main Steam Line N16 Radiation Monitors (1R53A/B/C/D and 2R53A/B/C/D)

The Main Steam Line N16 Radiation Monitoring System is one of the means used to detect and trend primary-to-secondary leakage in the main steam generators. It consists of four channels; each channel continuously monitors the N16 gamma radiation from one of the main steam lines. The detectors are high temperature NaI (T1) gamma scintillators with an integral Am241 check source. They are located upstream of the mixing bottle and as close to the main steam lines as practical. Detector output is processed by a multi-channel-analyzer, and the count rates in two energy windows are monitored for each channel. The high energy window is sensitive only to N16. Because N16 is present only during power operation, this monitor is used only during Mode 1.

The monitor is not included in the Reg. Guide 1.97 monitoring systems, has no interlocks with other monitors or systems, and is non-safety grade.

Technical Support Center (1R51)

The Technical Support Center monitors the radiation inside of the outside air intake duct.

11.4.2.3 Process Filter Monitoring System Channel Description

Area-type radiation monitors are provided on these liquid (process) filters to determine when they should be replaced by indicating the level of activity given off by the filter. A high radiation level alarm is initiated in the Control Room. A radiation indicator and alarm light are located at the filter.

The filters which are monitored include the following:

1. Reactor coolant filters (1-R26 and 2-R26)
2. Condensate filters (1-R40 and 2-R40)

All Unit 1 process filter monitors are GM tubes and have a range of 10^{-1}-10^{4} mR/hr. All Unit 2 process filter monitors are ion chambers or Geiger Mueller detectors and have a range of 10^{-1} -10^{6} mR/hr. They perform no control function.

The following liquid (process) filters are monitored for differential pressure and radioactivity to determine when the filters should be replaced:

1. Seal water injection filters
2. Seal water filters
3. Liquid waste filters
4. Spent fuel pool filters
5. Spent fuel pool skimmer filters
6. Refueling water purification filters
7. Ion exchange filters

11.4.2.4 Area Monitoring System Channel Description

Each channel consists of either a gamma sensitive GM tube detector or an ion chamber detector. Functional block diagrams of the area monitors are shown on Figures 11.4-7 and 11.4-8. For the Unit 1 analog channels, a meter is mounted on the front of each indicating-control module and is calibrated logarithmically from 0.1 mR/hr to 10 R/hr. In addition, one of the containment area monitors in Unit 1 is calibrated from 10 mR/hr to 1000 R/hr. A remote meter, calibrated logarithmically from 0.1 mR/hr to 10 R/hr (or from 10 mR/hr to 1000 R/hr) is mounted at the detector assembly. Radiation Monitoring System cabinet alarms consist of indicator lights for high radiation and detector or circuit failure. The remote meter and alarm assembly at the detector contains a red indicator light and an audible alarm which are actuated on high radiation. Both units have containment monitors capable of indicating area radiation from 1R/hr to 10^{7}R/hr.

Unit 1 digital channels 1R2, 1R3, 1R4, 1R5, 1R6A, 1R7, 1R9, 1R10A, and 1R40 provide digital indication at the local and remote monitors. The monitors are set to indicate radiation from 0.1 mR/h to 10 R/h. The local monitors provide Normal, Fail, and Alert indications and initiates buzzer and beacon light on high alarm. The remote monitors in the Control Equipment Room, in addition to display and status indication on the monitor panel, provide analog outputs to the plant computer (P250) and indicators on Panel 1RP1, indication of high radiation on Panel 1RP1, High/Trouble alarm on the Overhead Annunciator, and interlock contacts when required.

Tables 11.4-3 and 11.4-4 give a summary of the Area Radiation Monitoring System. A listing of the area monitors is also included below with any additional or special information added.

1. Control Room Area (Channel 2-R1A, 1-R1A) - This channel continuously monitors the Control Room area. This area monitor does not have its own integral flashing beacon and horn since it is located in the Control Room and an alarmed condition is indicated by the annunciator and audible alarm (Unit 2 is provided with LED alarm indication and an adjustable volume horn). This is a non-safety-related unit with a vital power supply.

2. Containment Area (Low Range) (1-R2, 2-R2)

3. Radiochemistry Laboratory (R3)

4. Charging Pump Room (1-R4, 2-R4)

5. Fuel Handling Building (Channels 1-R5 and 2-R5) - These channels continuously monitor the fuel storage areas. A high radiation alarm from either unit will initiate charcoal filtration of the Fuel Handling Building atmosphere. The Fuel Handling Accident in the Fuel Handling Building was analyzed without credit for filtration by the Fuel Handling Building Ventilation System. For Unit 2 the high radiation alarm will automatically start the exhaust fans. In addition to the integral alarm horn and flashing beacon, these units actuate an emergency evacuation horn in the building and radiation alert lights outside of the building. Each unit is on a separate vital power supply.

6. Sampling Room (R6A)

7. In-core Seal Table (1-R7, 2-R7)

8. Fuel Storage Area (1-R9, 2-R9) - These channels continuously monitor the fuel storage areas. A high radiation alarm from either unit will automatically start the exhaust fans (Unit 2 only) and initiate charcoal

filtration of the Fuel Handling Building atmosphere. The Fuel Handling Accident in the Fuel Handling Building was analyzed without credit for charcoal filtration by the Fuel Handling Building Ventilation System. In addition to the integral alarm horn and flashing beacon, these units actuate an emergency evacuation horn in the building and radiation alert lights outside of the building. Each unit is on a separate vital power supply.

9. Containment Personnel and Equipment Hatches (1-R10A, B and 2-R10A, B)

10. Counting Room (R20B)

11. Containment Area (High Range) (1-R44 A and B and 2-R44 A and B) -These channels continuously monitor the containment area and are provided with a special ion chamber detector for extended range capability in a post-accident environment. This is a safety-related unit with a vital power supply.

12. Public Service Control Point (R23)

13. Fuel Handling and Cask Handling Cranes (1-R32 A and B, 2-R32 A and B) - These channels are not connected to the central Radiation Monitoring System and are not provided with integral horns and flashing beacons. A flashing beacon and alarm bell on the cranes are initiated.

14. Mechanical Penetration Area (1-R34 and 2-R34)

15. Condensate Filter Area (1-R40 and 2-R40)

16. (Deleted)

17. (Deleted)

11.4.3 Sampling

Samples are taken as required by the plant Technical Specifications. The plant vent will be continuously monitored for gross radioactivity. Additionally, a fixed paper particulate filter followed by a charcoal cartridge (Cesco or equivalent) is installed, both of which will be changed weekly. The charcoal cartridge and particulate filter will be analyzed by gamma spectroscopy within 48 hours from change out. A sample will be taken manually at a frequency not to exceed monthly, and an isotopic analysis performed to determine the identity and quantity of noble gases.

The sample will be taken at a time when there are no gas decay tank releases or containment purges in progress, since these releases are not representative of a continuous release. Therefore, gas decay tank releases and containment purges will be analyzed isotopically on a batch basis.

The method employed will be a grab sample, utilizing a gas collection device, taken from either of two sampling points in the plant vent. These samplers are the same as those used in conjunction with the continuous plant vent gas monitors outlined in Section 11.4.2.2.

In order to ensure that the inline filter and cartridge sample is representative of the plant vent exhaust gas, a weekly isotopic analysis of the particulate filter and cartridge is performed. This isotopic inventory is used to determine the isotopic composition of the plant effluent. In the event that this equilibrium is upset by a refueling, process change, or a deviation of greater than 20 percent in the isotopic ratio established from the previous isotopic analysis (this does not apply while releasing gas decay tanks or during containment purging), a new isotopic analysis will be performed.

In order to ensure sampling during a radiological emergency that might render the normal plant vent sampling station uninhabitable, a supplemental Plant Vent Sampling System is provided. The supplemental Plant Vent Sampling System is located in a region of the plant that is expected to be habitable during all accident conditions (west side of the Fuel Handling Building). The supplemental system is designed such that it can be used during normal or accident conditions. The sample lines are heat traced to help ensure that a representative sample is being delivered to the supplemental sampling station.

The controlled facilities ventilation duct is equipped with offline particulate and iodine samples that operate in parallel with its system operation. The cartridges are removed and analyzed whenever necessary.

Samples are taken on each batch of liquid waste released. Station records contain the quantity and concentration of radioactive isotopes, the volume of each batch and estimates of the water flow for dilution. Each sample is analyzed for principal gamma emitters. Composites are prepared from each batch released during a month and analyzed for the principal gamma emitting nuclides, fission and activation products, gross alpha, and tritium. A quarterly composite analysis is also performed for Sr-89, Sr-90 and Fe-55. The sensitivities and frequencies of analyses comply with the requirements of Salem Technical Specifications.

Information on sampling of the containment atmosphere and the Reactor Coolant System is included in Section 9.3.

11.4.4 Inservice Tests and Calibrations

Radiation monitors of the Radiation Monitoring System are initially calibrated to standards traceable to the National Bureau of Standards, most often referred to as “primary calibration”. The area monitors undergo a range calibration by exposing the detectors to at least three radiation intensities from a Co-60 or Cs-137 source. The exception is the Containment High Range Area Monitors, which are calibrated with at least one intensity. The liquid and gas process monitors undergo "wet" isotopic calibrations with isotopes of an average energy comparable to those of the isotopes expected to be monitored. At least three concentrations of isotopes are used in calibrating the instruments. Beyond the initial primary calibration all radiation detectors undergo point source calibrations in a fixture with repeatable geometry, most often referred to as a “secondary calibration” or “transfer calibration”. This fixture and the button sources are utilized for periodic field calibrations. It is typical practice to use three secondary sources of the same isotope to validate detector linearity and stability; however, single point calibrations using secondary sources where detectors are inherently linear are acceptable. In addition, the detectors are provided with check sources which can be used to indicate functional operability. Tests, functional checks and calibrations are performed periodically in accordance with Technical Specification requirements and operating procedures.

TABLE 11.4-1
UNIT 1 PROCESS RADIATION MONITORING SYSTEM

Channel No.	Type of Detector	Channel Description	Minimum Detectable Level	Control Function/Interlocks
1-R1B(1)	Beta Scintillation	Control Room Intake Duct	10^{-6} μCi/cc ^{133}Xe	Isolates the Control Room Envelope from outside air and places the ventilation system in accident pressurized mode
1-R11A(1)	Scintillation	Containment or Vent Air	10^{-9} μCi/cc ^{137}Cs Particulate (moving filter)	Containment Ventilation Isolation (Mode 6)
1-R12A(1)	GM Tube	Containment or Vent Gas	10^{-6} μCi/cc ^{133}Xe Effluent	Containment Ventilation Isolation
1-R12B(1)	Scintillation	Containment or Vent Iodine	10^{-11} μCi/cc ^{131}I(2) Effluent (fixed filter)	Containment Ventilation Isolation
1R13A	Scintillation	Nos. 11, 12, & 13 Cont Fan Coil Unit Cooling Water	10^{-7} μCi/cc ^{137}Cs	—
1R13B	Scintillation	Nos. 13, 14 & 15 Cont Fan Coil Unit Cooling Water	10^{-7} μCi/cc ^{137}Cs	—
1-R15	Scintillation (In-Line)	Condenser Air Ejector	10^{-6} μCi/cc ^{133}Xe	—
1-R17A	Scintillation	Component Cooling-Liquid	10^{-5} μCi/cc ^{137}Cs	Surge Tank Vent Valve Closure

Channel No.	Type of Detector	Channel Description	Minimum Detectable Level	Control Function/Interlocks
1-R17B	Scintillation	Component Cooling-Liquid	10^{-5} µCi/cc ^{137}Cs	Surge Tank Vent Valve Closure
1-R18(1)	Scintillation	Liquid Waste Disposal Closure	10^{-5} µCi/cc ^{137}Cs	Liquid Waste Discharge Valve
1-R19A	Scintillation	No. 11 Steam Generator Blowdown	10^{-5} µCi/cc ^{137}Cs	High No.12 Blowdown Tank Inlet Valves High: SG Blowdown Isolation Valves
1-R19B	Scintillation	No. 12 Steam Generator Blowdown	10^{-5} µCi/cc ^{137}Cs	High No.12 Blowdown Tank Inlet Valves High: SG Blowdown Isolation Valves
1-R19C	Scintillation	No. 13 Steam Generator Blowdown	10^{-5} µCi/cc ^{137}Cs	High No.12 Blowdown Tank Inlet Valves High: SG Blowdown Isolation Valves
-R19D	Scintillation	No. 14 Steam Generator Blowdown	10^{-5} µCi/cc ^{137}Cs	High No.12 Blowdown Tank Inlet Valves High: SG Blowdown Isolation Valves
1-R31A	Scintillation	Letdown Line (Gross)	10^{-4} µCi/cc ^{60}Co	—

TABLE 11.4-1 (Cont.)

Channel No.	Type of Detector	Channel Description	Minimum Detectable Level	Control Function/Interlocks
1-R36	Gamma Scintillator	Evaporator and Feed Heater Condensate	10^{-5} μCi/cc ^{137}Cs	Condensate Line Valve
1-R41A	Beta Scintillator	Plant Vent Noble Gas (Low)	$3x10^{-7}$ μCi/cc ^{133}Xe *	
1-R41B	Beta-Gamma Scintillator	Plant Vent Noble Gas (Inter.)	$7x10^{-4}$ μCi/cc ^{133}Xe	
1-R41C	Beta-Gamma Scintillator	Plant Vent Noble Gas (High)	10^{-1} μCi/cc ^{133}Xe	
1-R41D	N/A	Plant Vent Noble Gas (composite)	N/A	Containment Ventilation Isolation Closes Waste Gas Discharge Valve

* This MDL reflects the design range of the detector. The actual detection level may be higher than the MDL due to the masking effect of background radiation at the installed location.

TABLE 11.4-1 (Cont.)

Channel No.	Type of Detector	Channel Description	Minimum Detectable Level	Control Function/Interlocks
1-R46A	Ion Chamber	Main Steam Line No. 11	0.1 mr/hr to 10,000 mr/hr(4)	---
1-R46B	Ion Chamber	Main Steam Line No. 12	0.1 mr/hr to 10,000 mr/hr(4)	---
1-R46C	Ion Chamber	Main Steam Line No. 14	0.1 mr/hr to 10,000 mr/hr(4)	---
1-R46D	Ion Chamber	Main Steam Line No. 13	0.1 mr/hr to 10,000 mr/hr(4)	---
1R51	Beta Scintillator	Technical Support Center	10 cpm (nominal) (5)	--
1-R53A	NaI(T1) Gamma Scint.	Main Steam Line No. 11	10 cpm (nominal) (5)	---
1-R53B	NaI(T1) Gamma Scint.	Main Steam Line No. 12	10 cpm (nominal) (5)	---
1-R53C	NaI(T1) Gamma Scint.	Main Steam Line No. 14	10 cpm (nominal) (5)	---
1-R53D	NaI(T1) Gamma Scint.	Main Steam Line No. 13	10 cpm (nominal) (5)	---

NOTES:

(1) Also performs a safety function
(2) Assumes 1-week collection time
(3) The upper range corresponds to at least 10^5 µCi/cc (Xe-133)
(4) The upper range corresponds to at least 10^3 µCi/cc (Xe-133)
(5) > Background (electrical noise)

TABLE 11.4-2

UNIT 2 PROCESS RADIATION MONITORING SYSTEM

Channel No.	Type of Detector	Channel Description	Minimum Detectable Level	Control Function/Interlocks
2-R1B	Beta Scintillator (2 channels)	Control Room Vent Intake Duct	10^{-6} µCi/cc ^{133}Xe	Isolates the Control Room Envelope from outside air and places the ventilation system in accident pressurized mode
2-R11A	Beta Scintillator	Containment Particulate	10^{-11} µCi/cc ^{137}Cs	Containment Ventilation Isolation (Mode 6)
2-R12A	Beta Scintillator	Containment Noble Gas	10^{-6} µCi/cc ^{133}Xe	Containment Ventilation Isolation
2-R12B	Gamma Scintillator	Containment Iodine	10^{-11} µCi/cc ^{131}I(2)	Containment Ventilation Isolation
2-R13A	Gamma Scintillator	21,22,23 Fan Cooler Service Water Discharge	10^{-7} µCi/cc ^{137}Cs	—
2-R13B	Gamma Scintillator	23,24,25 Fan Cooler Service Water Discharge	10^{-7} µCi/cc ^{137}Cs	—
2-R15	Gamma Scintillator	Condenser Air Ejector	10^{-6} µCi/cc ^{133}Xe	—
2-R17A	Gamma Scintillator	21 Component Cooling Loop	10^{-7} µCi/cc ^{137}Cs	Closes Surge Tank Vent Valve
2-R17B	Gamma Scintillator	22 Component Cooling Loop	10^{-7} µCi/cc ^{137}Cs	Closes Surge Tank Vent Valve
2-R18(1)	Gamma Scintillator	Liquid Waste Discharge	10^{-7} µCi/cc ^{137}Cs	Closes Liquid Waste Discharge Valve

TABLE 11.4-2 (Cont.)

UNIT 2 PROCESS RADIATION MONITORING SYSTEM

Channel No.	Type of Detector	Channel Description	Minimum Detectable Level	Control Function/Interlocks
2-R19A	Gamma Scintillator	21 Steam Generator Blowdown	$3.91x10^{-8}$ μCi/cc ^{137}Cs	Warn: Closed Blowdown Tank Inlet Valves High: Isolate 21 Steam Generator Blowdown
2-R19B	Gamma Scintillator	22 Steam Generator Blowdown	$3.91x10^{-8}$ μCi/cc ^{137}Cs	Warn: Closed Blowdown Tank Inlet Valves High: Isolate 22 Steam Generator Blowdown
2-R19C	Gamma Scintillator	23 Steam Generator Blowdown	$3.91x10^{-8}$ μCi/cc ^{137}Cs	Warn: Closed Blowdown Tank Inlet Valves High: Isolate 23 Steam Generator Blowdown
2-R19D	Gamma Scintillator	24 Steam Generator Blowdown	$3.91x10^{-8}$ μCi/cc ^{137}Cs	Warn: Closed Blowdown Tank Inlet Valves High: Isolate 24 Steam Generator Blowdown
2-R31(1)	Gamma Scintillator	Letdown Line	10^{-6} μCi/cc ^{137}Cs	---
2-R37	Gamma Scintillator	Non Radwaste Basin	10^{-8} μCi/cc ^{137}Cs	None
2-R41A	Beta Scintillator	Plant Vent Noble Gas (low)	$3x10^{-7}$ μCi/cc ^{133}Xe(2) *	None
2-R41B	Beta-Gamma Scintillator	Plant Vent Noble Gas (inter.)	$7x10^{-4}$ μCi/cc ^{133}Xe(2)	None
2-R41C	Beta-Gamma Scintillator	Plant Vent Noble Gas (high)	10^{-1} μCi/cc ^{133}Xe	None
2-R41D	N/A	Plant Vent Noble Gas (composite)	N/A	Containment Ventilation Isolation; Closes Waste Gas Discharge Valve

* This MDL reflects the design range of the detector. The actual detection level may be higher than the MDL due to the masking effect of background radiation at the installed location.

TABLE 11.4-2 (Cont.)

UNIT 2 PROCESS RADIATION MONITORING SYSTEM

Channel No.	Type of Detector	Channel Description	Minimum Detectable Level	Control Function/Interlocks
2-R46A	Ion Chamber	Main Steam Line No. 21	0.1 mr/hr to 10,000 mr/hr(4)	---
2-R46B	Ion Chamber	Main Steam Line No. 22	0.1 mr/hr to 10,000 mr/hr(4)	---
2-R46C	Ion Chamber	Main Steam Line No. 24	0.1 mr/hr to 10,000 mr/hr(4)	---
2-R46D	Ion Chamber	Main Steam Line No. 23	0.1 mr/hr to 10,000 mr/hr(4)	---

TABLE 11.4-2 (Cont.)

UNIT 2 PROCESS RADIATION MONITORING SYSTEM

Channel No.	Type of Detector	Channel Description	Minimum Detectable Level	Control Function/Interlocks
2-R53A	NaI(T1) Gamma Scint.	Main Steam Line 21	10 cpm (nominal) (5)	—
2-R53B	NaI(T1) Gamma Scint.	Main Steam Line 22	10 cpm (nominal) (5)	—
2-R53C	NaI(T1) Gamma Scint.	Main Steam Line 23	10 cpm (nominal) (5)	—
2-R53D	NaI(T1) Gamma Scint.	Main Steam Line 24	10 cpm (nominal) (5)	—

NOTES:

(1) Also performs a safety function

(2) Assumes 1 week collection time

(3) The upper range corresponds to 10^5 μCi/cc (Xe-133)

(4) The upper range corresponds to at least 10^3 μCi/cc (Xe-133)

(5) > Background (electrical noise)

TABLE 11.4-3

UNIT 1 AREA RADIATION MONITORING SYSTEM

Channel No.	Channel Description	Type of Detector	Range	Control Function/Interlocks
1-R1A	Control Room	GM Tube	10^{-1} -10^{4} mR/hr	---
1-R2	Containment	GM Tube	10^{-1} -10^{4} mR/hr	---
R3	Radio-Chem Laboratory	GM Tube	10^{-1} -10^{4} mR/hr	---
1-R4	Charging Pump Room	GM Tube	10^{-1} -10^{4} mR/hr	---
1-R5(1)	Fuel Handling	GM Tube	10^{-1} -10^{4} mR/hr	1) Fuel Handling Area Hi-Rad Evacuation Alarm 2) Fuel Handling Area Ventilation Exhaust Filter Units
R6A	Sampling Room	GM Tube	10^{-1} -10^{4} mR/hr	---
1-R7	In-core Seal Table	GM Tube	10^{-1} -10^{4} mR/hr	---
1-R9(1)	Fuel Storage Area	GM Tube	10^{-1} -10^{4} mR/hr	1) Fuel Handling Area Hi-Rad Evacuation Alarm 2) Fuel Handling Area Ventilation Exhaust Filter Units
1-R10A	Personnel Hatch to Containment (el 100 ft)	GM Tube	10^{-1} -10^{4} mR/hr	Containment Area High Radiation Alarm
1-R10B	Personnel Hatch to Containment (el 100 ft)	GM Tube	10^{-1} -10^{4} mR/hr	Containment Area High Radiation Alarm Actuates High Radiation Signal at Hatch
R20B	Counting Room	GM Tube	10^{-1} -10^{4} mR/hr	---

TABLE 11.4-3 (Cont.)

Channel No.	Channel Description	Type of Detector	Range	Control Function/Interlocks
R23	Monitoring Room	GM Tube	10^{-1} -10^{4} mR/hr	---
1-R32A (2)	Fuel Handling Crane Monitor	GM Tube	10^{-1} -10^{4} mR/hr	---
1-R34	Mechanical Penetration Area	GM Tube	10^{-1} -10^{4} mR/hr	---
1-R44A	Containment (High Range)		10^{0} - 10^{7} R/hr	---
1-R44B	Containment (High Range)		10^{0} - 10^{7} R/hr	---

NOTES:
(1) Also performs a safety function
(2) Local monitor only. Not indicated, recorded, or alarmed in the control room.

TABLE 11.4-4

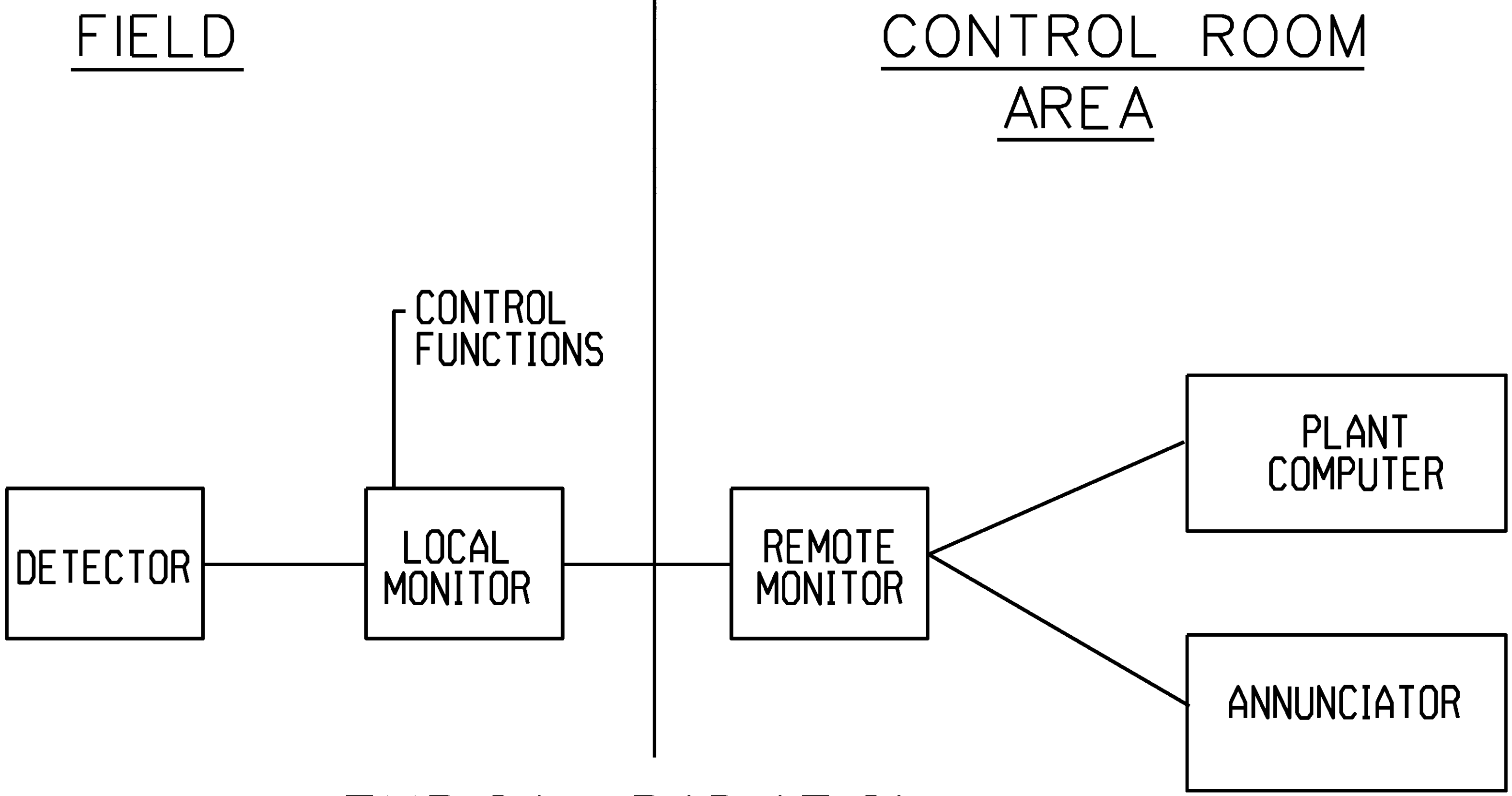
UNIT 2 AREA RADIATION MONITORING SYSTEM

Channel No.	Channel Description	Type of Detector	Range	Control Functions/Interlocks
2-R1A	Control Room	GM Tube	10^{-1} - 10^{4} mR/hr	
2-R2	Containment	GM Tube	10^{-1} - 10^{4} mR/hr	Actuates High Radiation Sign at Hatches
2-R4	Charging Pump Room	GM Tube	10^{-1} - 10^{4} mR/hr	(el 100 ft and el 130 ft)
2-R5(1)	Fuel Handling Building	GM Tube	10^{-1} - 10^{4} mR/hr	Initiates Charcoal Filtration and Evacuation Horns in Fuel Handling Building and automatically starts exhaust fans.
2-R7	Incore Seal Table	GM Tube	10^{-1} - 10^{4} mR/hr	---
2-R9(1)	Fuel Handling Building	GM Tube	10^{-1} - 10^{4} mR/hr	Initiates Charcoal Filtration and Evacuation Horns in Fuel Handling Building and automatically starts exhaust fans.
2-R10A	Containment Personnel Hatch (el 100 ft)	GM Tube	10^{-1} - 10^{4} mR/hr	Actuates High Radiation Sign at Hatch
2-R10B	Containment Personnel Hatch (el 130 ft)	GM Tube	10^{-1} - 10^{4} mR/hr	Actuates High Radiation Sign at Hatch
2-R32A(2)	Fuel Handling Crane	GM Tube	10^{-1} - 10^{4} mR/hr	---

TABLE 11.4-4 (Cont.)

Channel No.	Channel Description	Type of Detector	Range	Control Functions/Interlocks
2-R34	Mechanical Penetration Area	GM Tube	10^{-1} -10^{6} mR/hr	---
2-R44A	Containment (High Range)	Ion Chamber	10^{0} -10^{7} R/hr	---
2-R44B	Containment (High Range)	Ion Chamber	10^{0} -10^{7} R/hr	---

NOTES:
(1) Also performs a safety function.
(2) Local only - Not connected to RMS monitor in the Control Equipment Room.



TYPICAL RADIATION MONITORING STRUCTURE

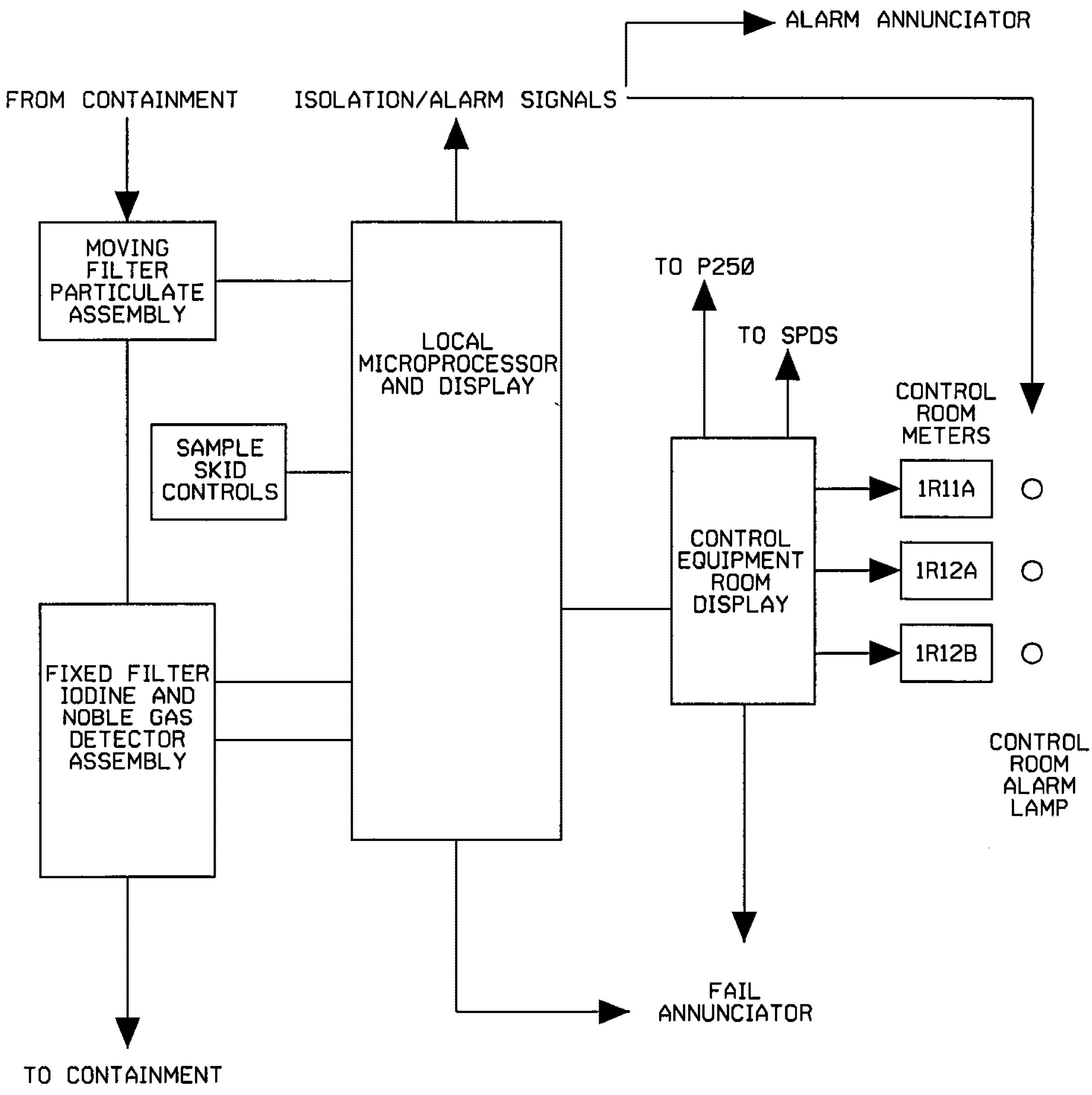
PSEG NUCLEAR LLC
SALEM GENERATING STATION

UNIT NO. 2 OVERALL RADIATION MONITORING SYSTEM MAKEUP

Updated FSAR Sheet 1 of 1

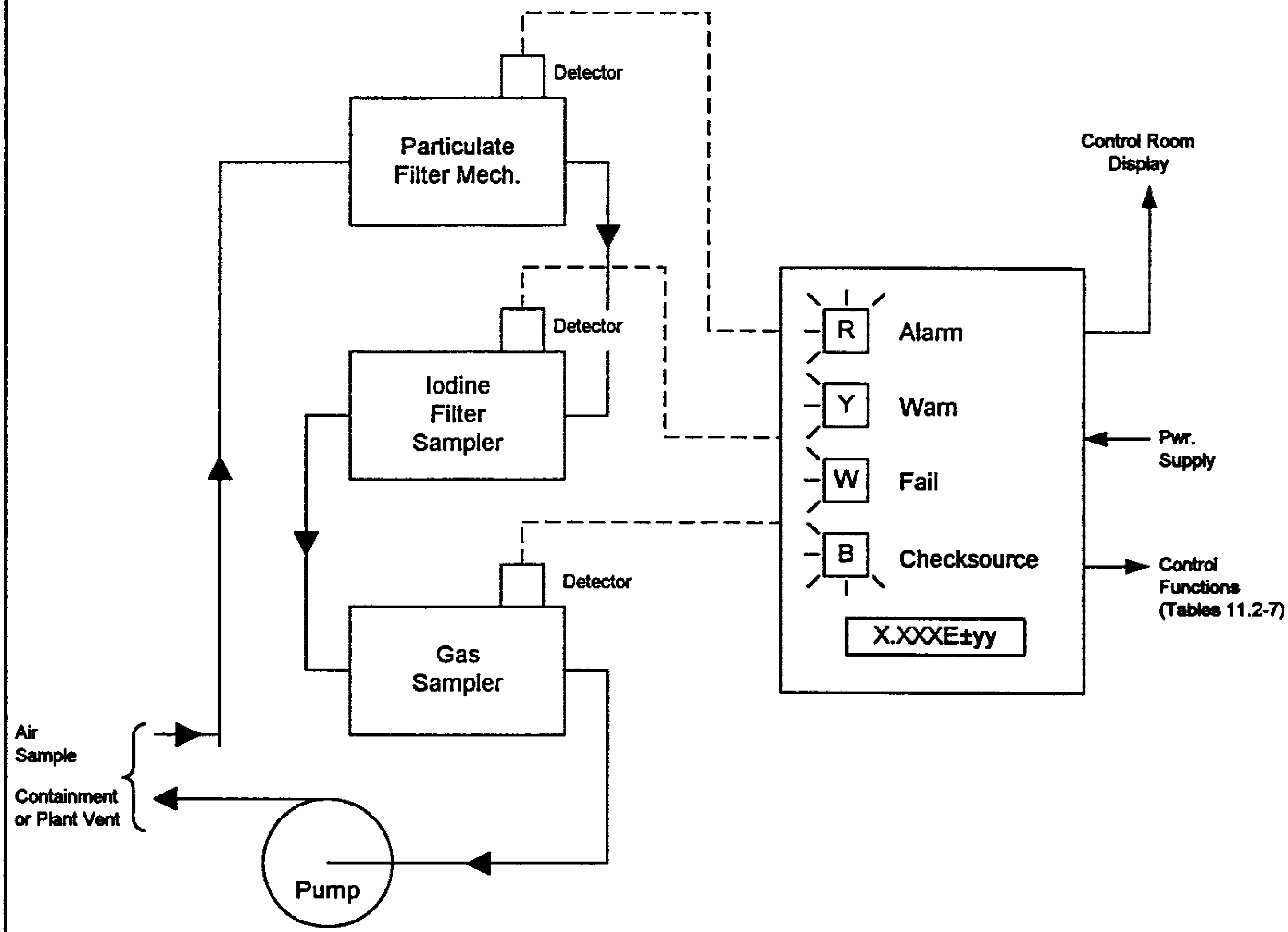
REVISION 27, NOVEMBER 25, 2013 Fig. 11.4-1

AIR PARTICULATE & GAS MONITOR



Revision 19, Nov. 19, 2001

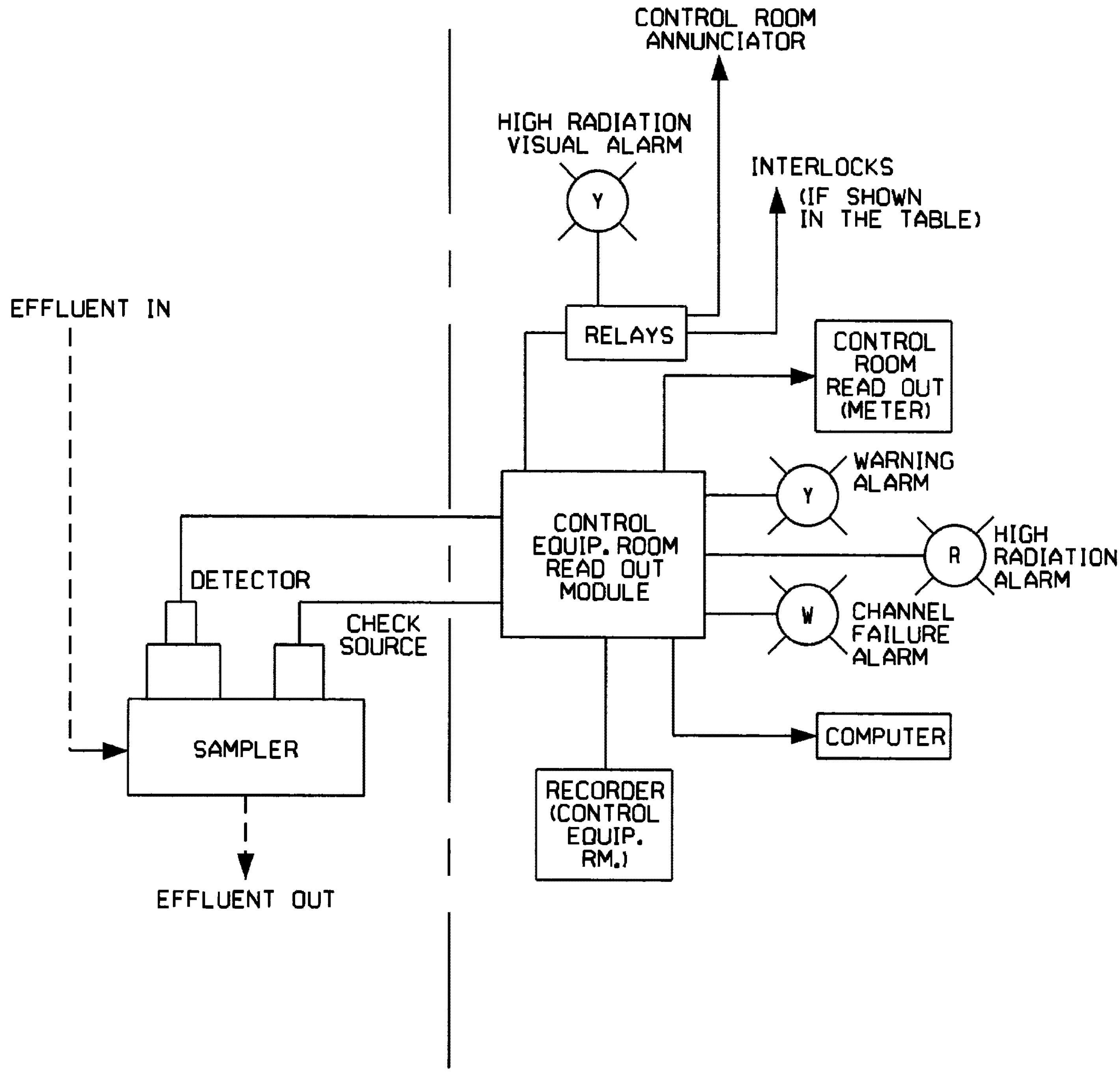
PSEG Nuclear, LLC SALEM NUCLEAR GENERATING STATION	Salem Nuclear Generating Station UNIT NO. 1 AIR PARTICULATE IODINE AND GAS MONITOR (TYPICAL)
	Updated FSAR Figure 11.4-2



2-R11A, 12A, B

PUBLIC SERVICE ELECTRIC AND GAS COMPANY SALEM NUCLEAR GENERATING STATION	Unit No. 2 Air Particulate, Iodine and Gas Monitor (Typical)
	Updated FSAR Revision 16 — Figure 11.4-3, Sheet 1 of 1, January 31, 1998

LIQUID PROCESS MONITOR
(TYPICAL)

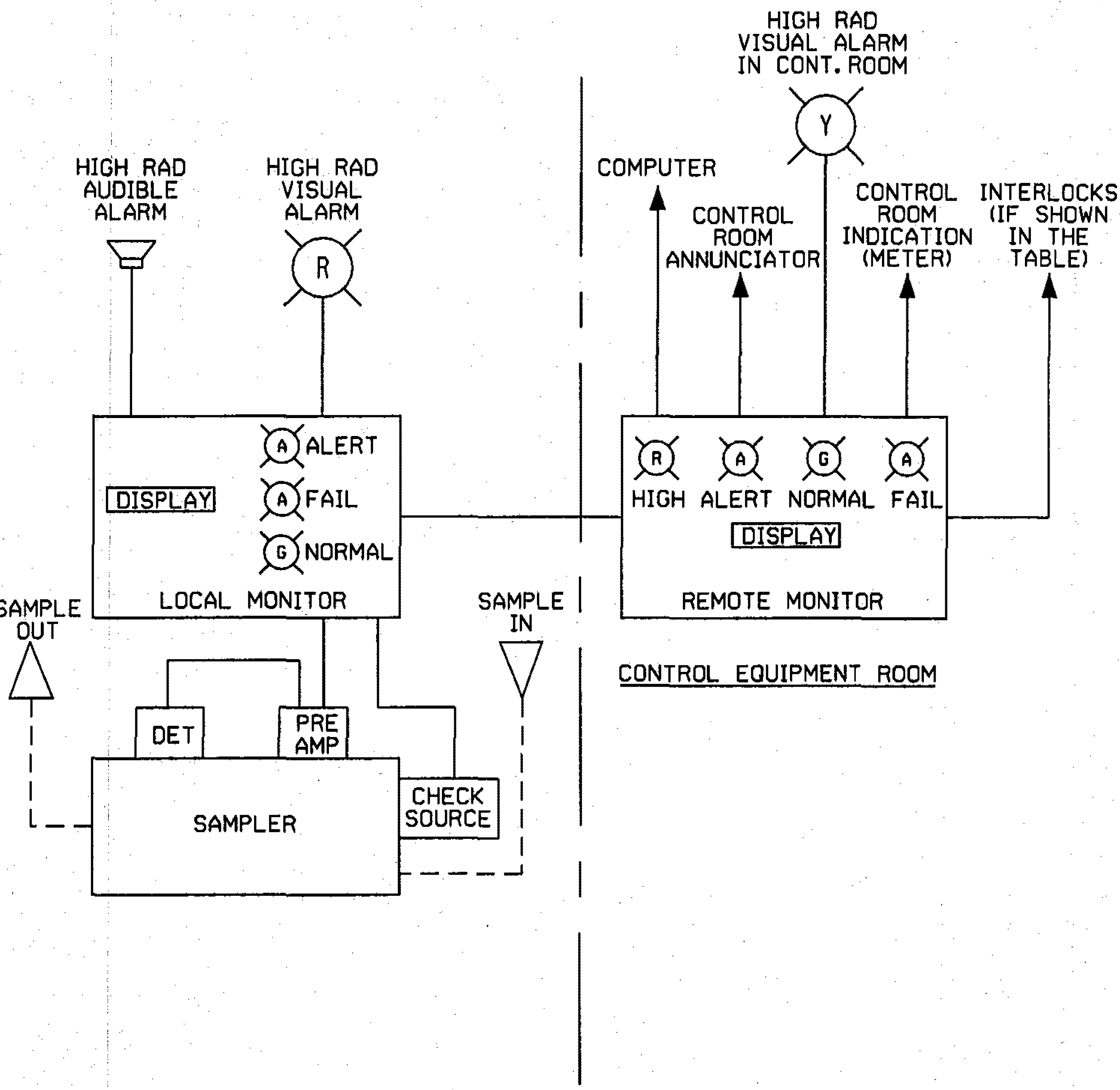


Revision 25, October 26, 2010

PSEG Nuclear, LLC SALEM NUCLEAR GENERATING STATION	Salem Nuclear Generating Station UNIT NO. 1 LIQUID MONITOR (TYPICAL)		
	Updated FSAR	SHEET 1 OF 4	Figure 11.4-4

DIGITAL LIQUID PROCESS MONITOR

(TYPICAL)

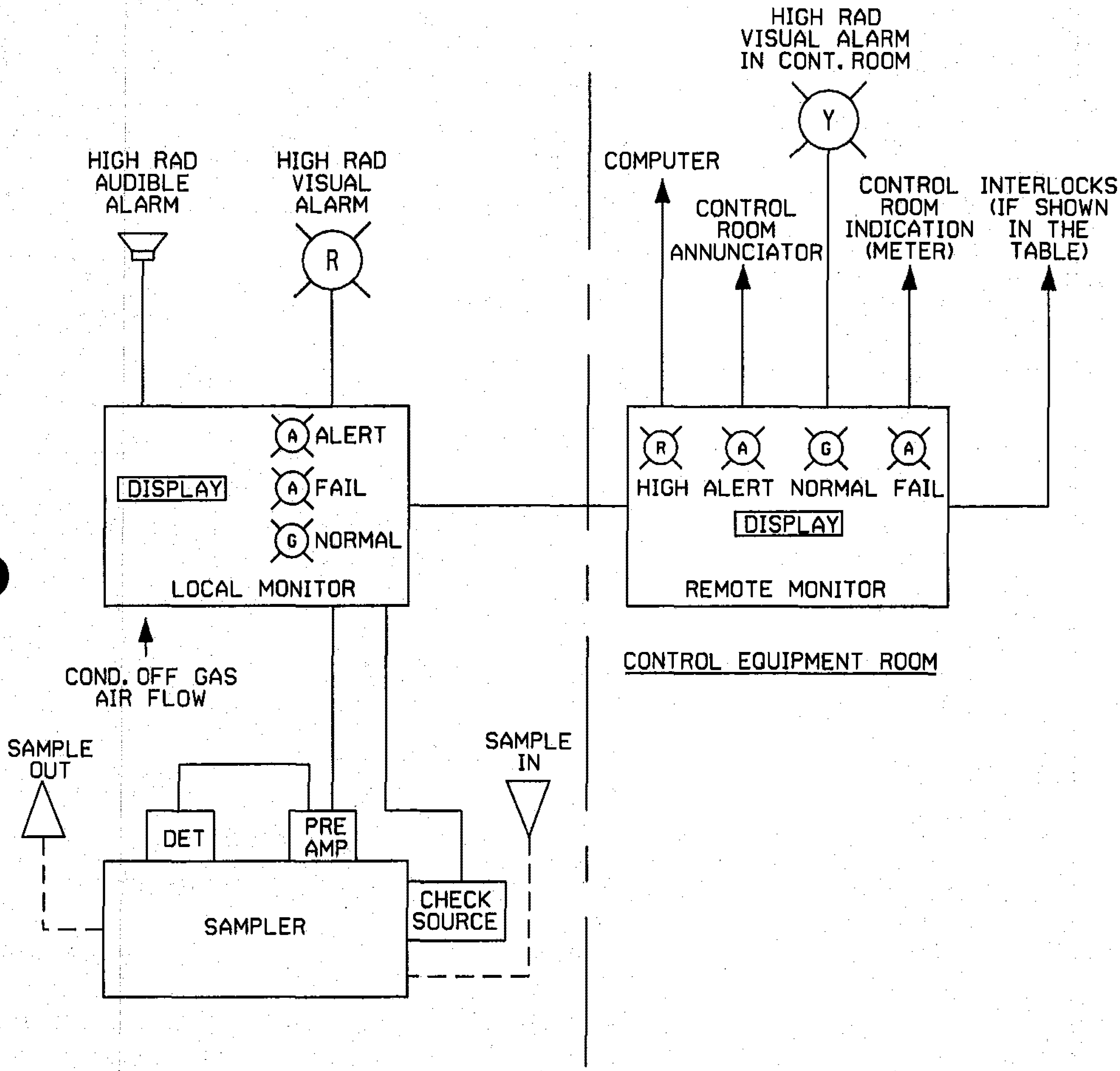


Revision 21, Dec. 6, 2004

PSEG Nuclear, LLC SALEM NUCLEAR GENERATING STATION	Salem Nuclear Generating Station UNIT NO. 1 LIQUID MONITOR (TYPICAL) Updated FSAR SHEET 2 Figure 11.4-4

DIGITAL LIQUID PROCESS MONITOR

(TYPICAL)

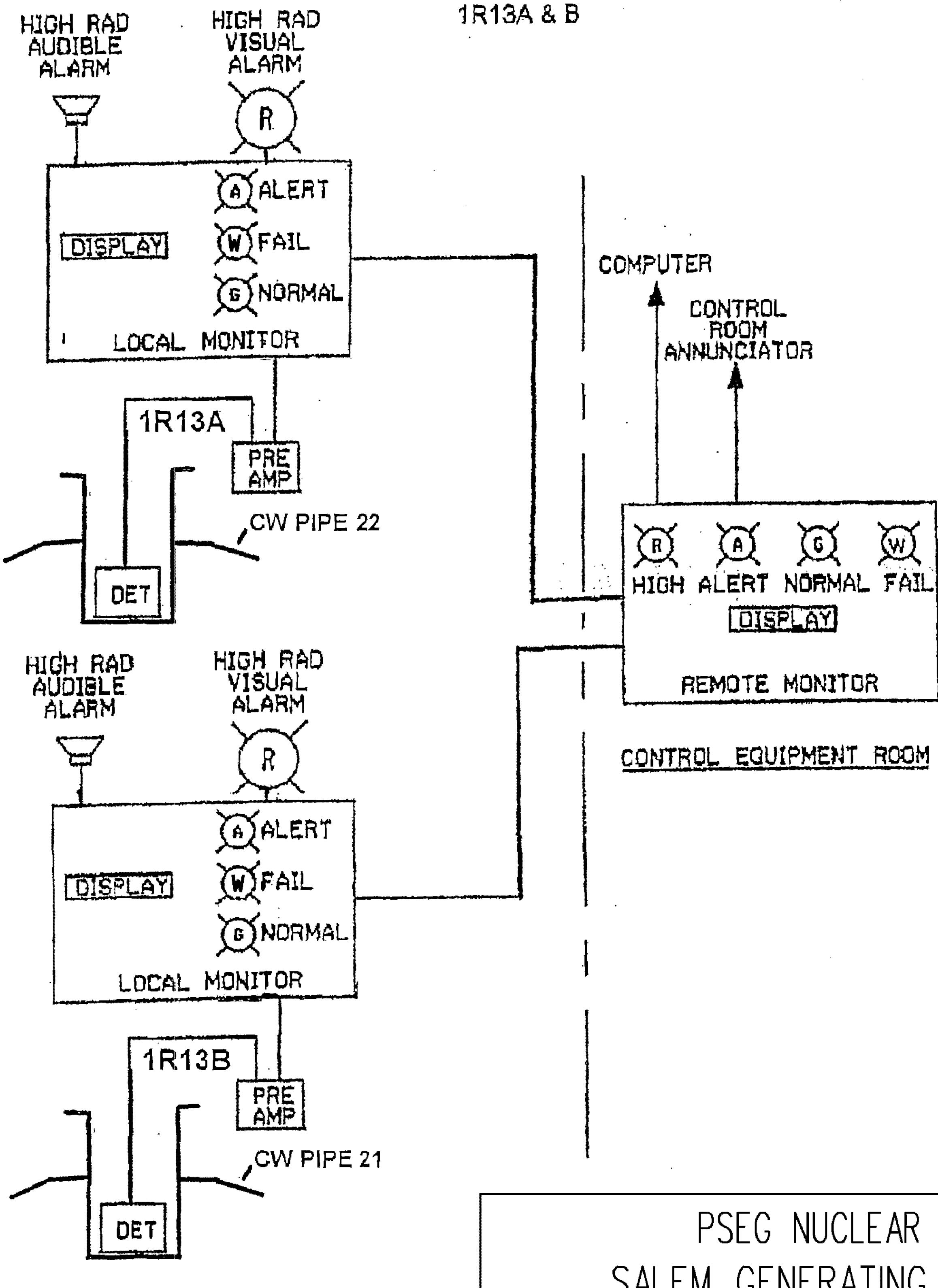


Revision 21, Dec. 6, 2004

PSEG Nuclear, LLC SALEM NUCLEAR GENERATING STATION	Salem Nuclear Generating Station UNIT NO. 1 COND. AIR EJECT RAD MONITOR 1R15 (TYPICAL)		
	Updated FSAR	SHEET 3	Figure 11.4-4

DIGITAL LIQUID PROCESS MONITOR

1R13A & B



PSEG NUCLEAR LLC
SALEM GENERATING STATION

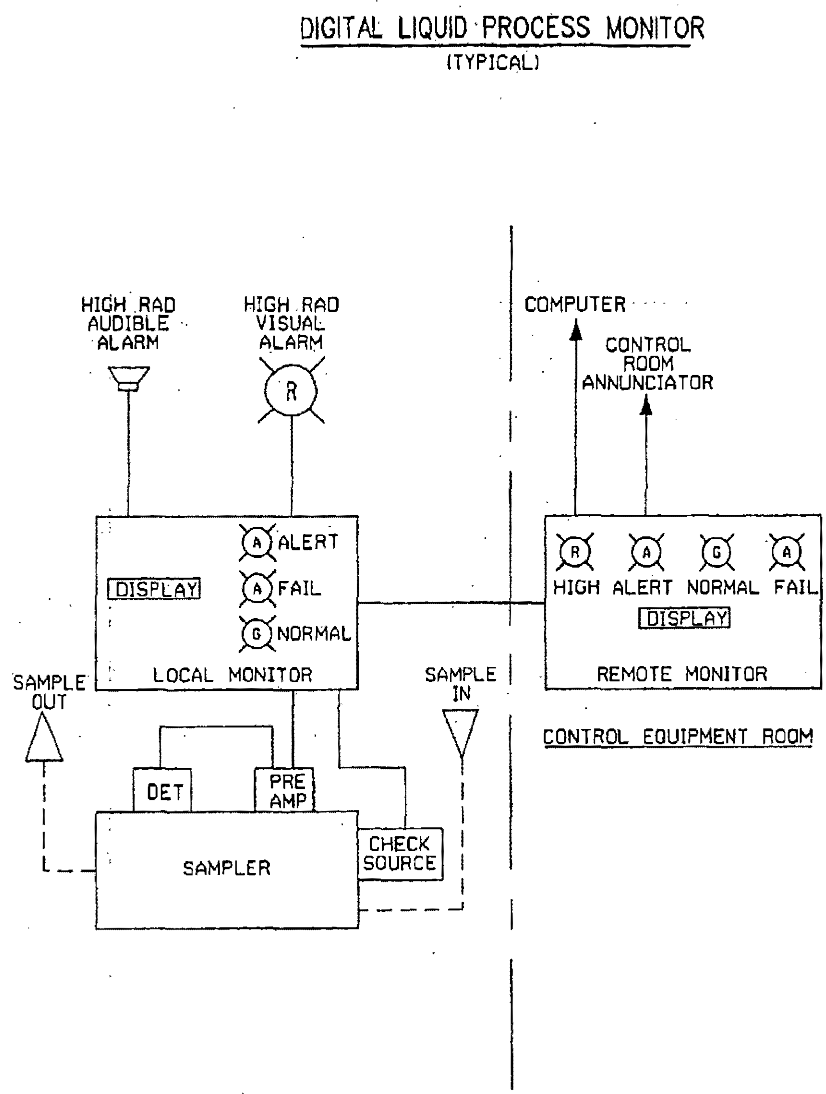
UNIT NO. 1 LIQUID MONITOR 1R13 A & B

Updated FSAR

Sheet 4

REVISION 27, NOVEMBER 25, 2013

Fig. 11.4-4



Note: 2R19A, B, C, D Share The Same Remote Field Unit and Have No Connections to the Central Processing Unit

Revision 26, May 21, 2012

PUBLIC SERVICE ELECTRIC AND GAS COMPANY
SALEM NUCLEAR GENERATING STATION

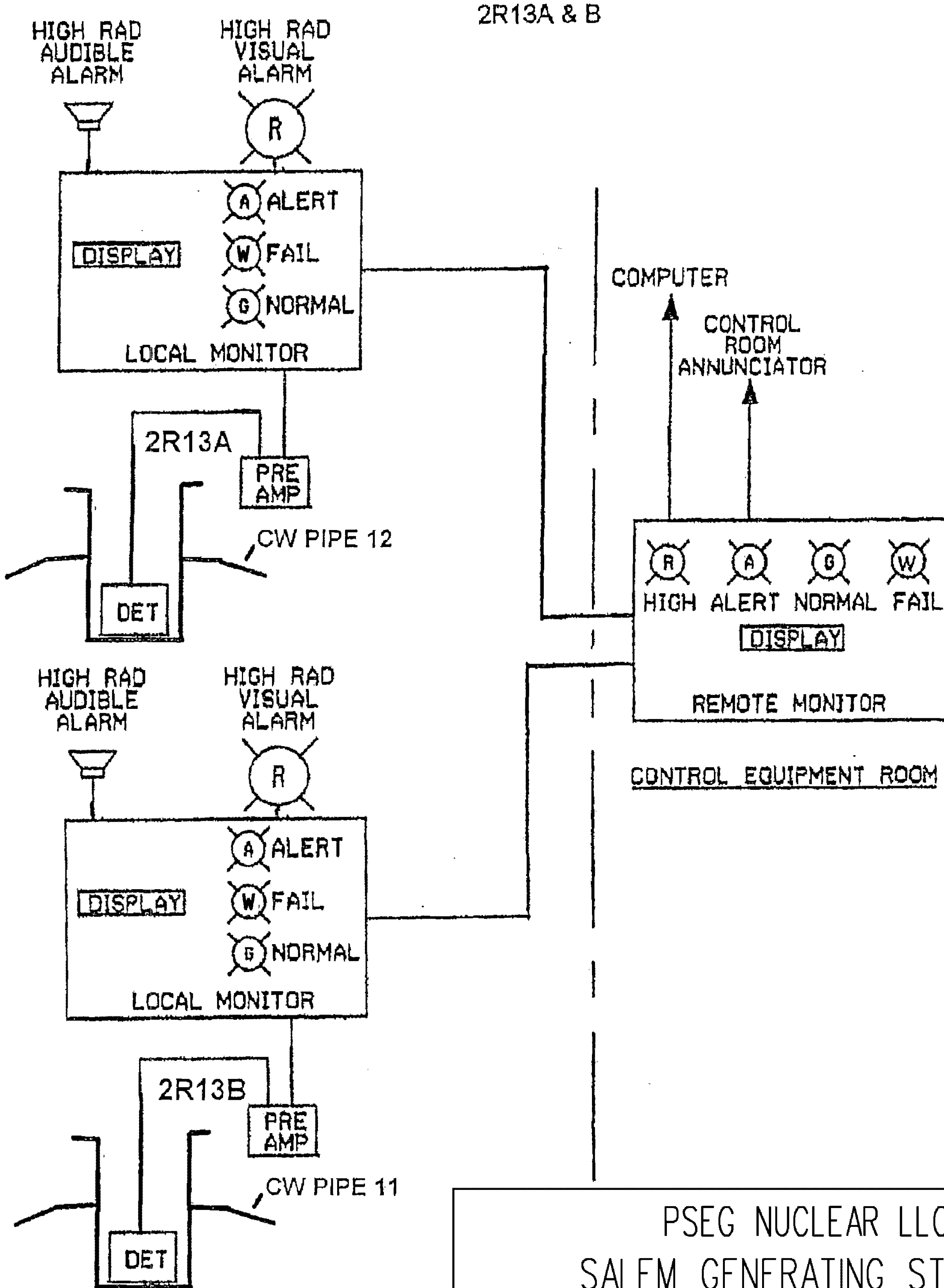
UNIT NO. 2 LIQUID MONITOR (TYPICAL)

Updated FSAR

Sheet 1 of 2
Figure 11.4-5

DIGITAL LIQUID PROCESS MONITOR

2R13A & B



PSEG NUCLEAR LLC
SALEM GENERATING STATION

UNIT NO. 2 LIQUID MONITOR 2R13A & B

Updated FSAR — Sheet 2

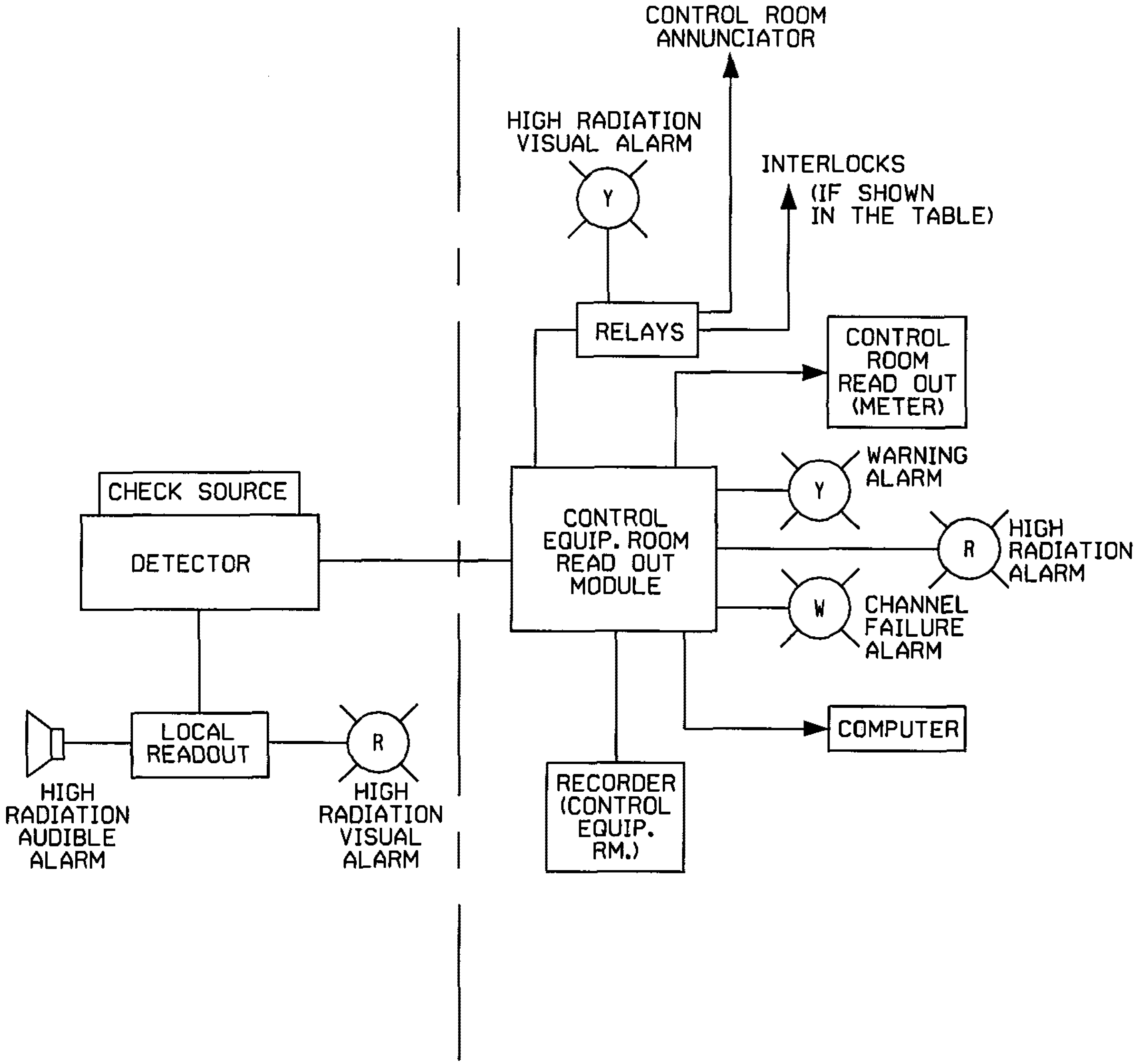
REVISION 27, NOVEMBER 25, 2013 — Fig. 11.4-5

THIS FIGURE HAS BEEN DELETED

Revision 21, Dec. 6, 2004

PSEG Nuclear, LLC SALEM NUCLEAR GENERATING STATION	Salem Nuclear Generating Station UNIT 2 IN-LINE GAS MODEL (TYPICAL)
	Updated FSAR — Figure 11.4-6

AREA MONITOR
(TYPICAL)

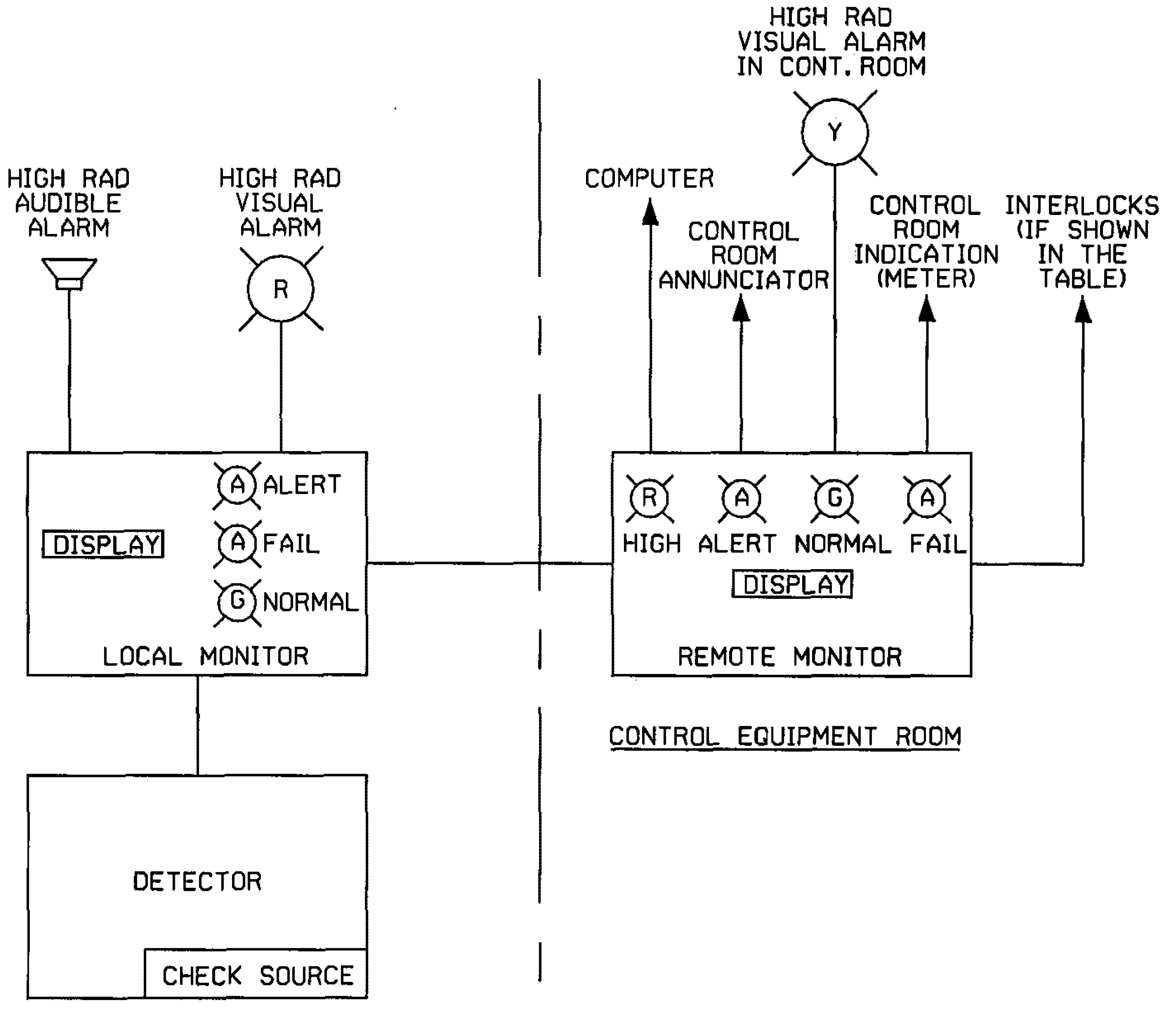


Revision 19, Nov. 19, 2001

PSEG Nuclear, LLC SALEM NUCLEAR GENERATING STATION	Salem Nuclear Generating Station UNIT NO. 1 AREA MONITOR (TYPICAL)		
	Updated FSAR	SHEET 1 OF 2	Figure 11.4-7

DIGITAL AREA MONITOR

(TYPICAL)

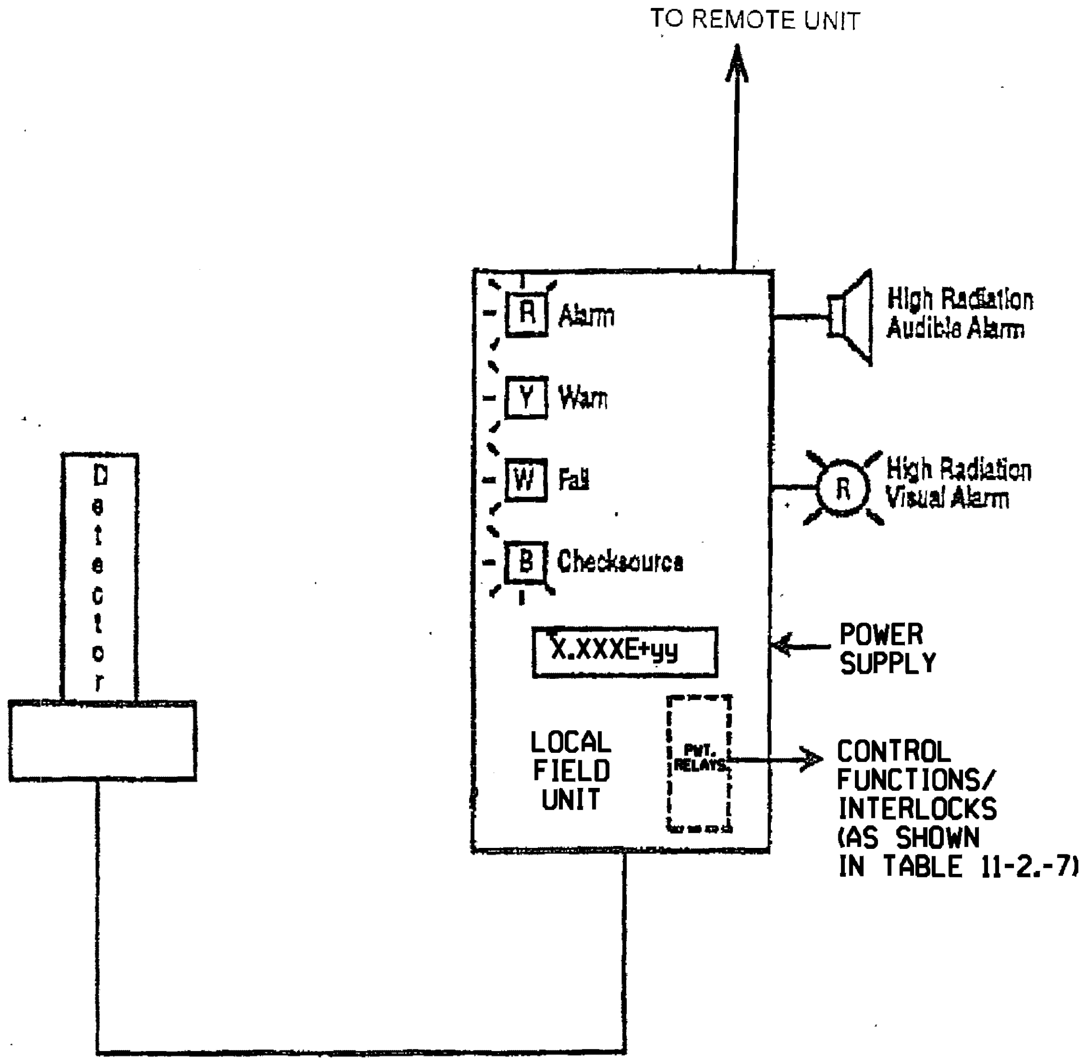


APPLICABLE RMS CHANNELS:

1R2, 1R3, 1R4, 1R5, 1R6A, 1R7, 1R9, 1R10A

Revision 19, Nov. 19, 2001

PSEG Nuclear, LLC SALEM NUCLEAR GENERATING STATION	Salem Nuclear Generating Station UNIT NO. 1 AREA MONITOR (TYPICAL) Updated FSAR SHEET 2 OF 2 Figure 11.4-7



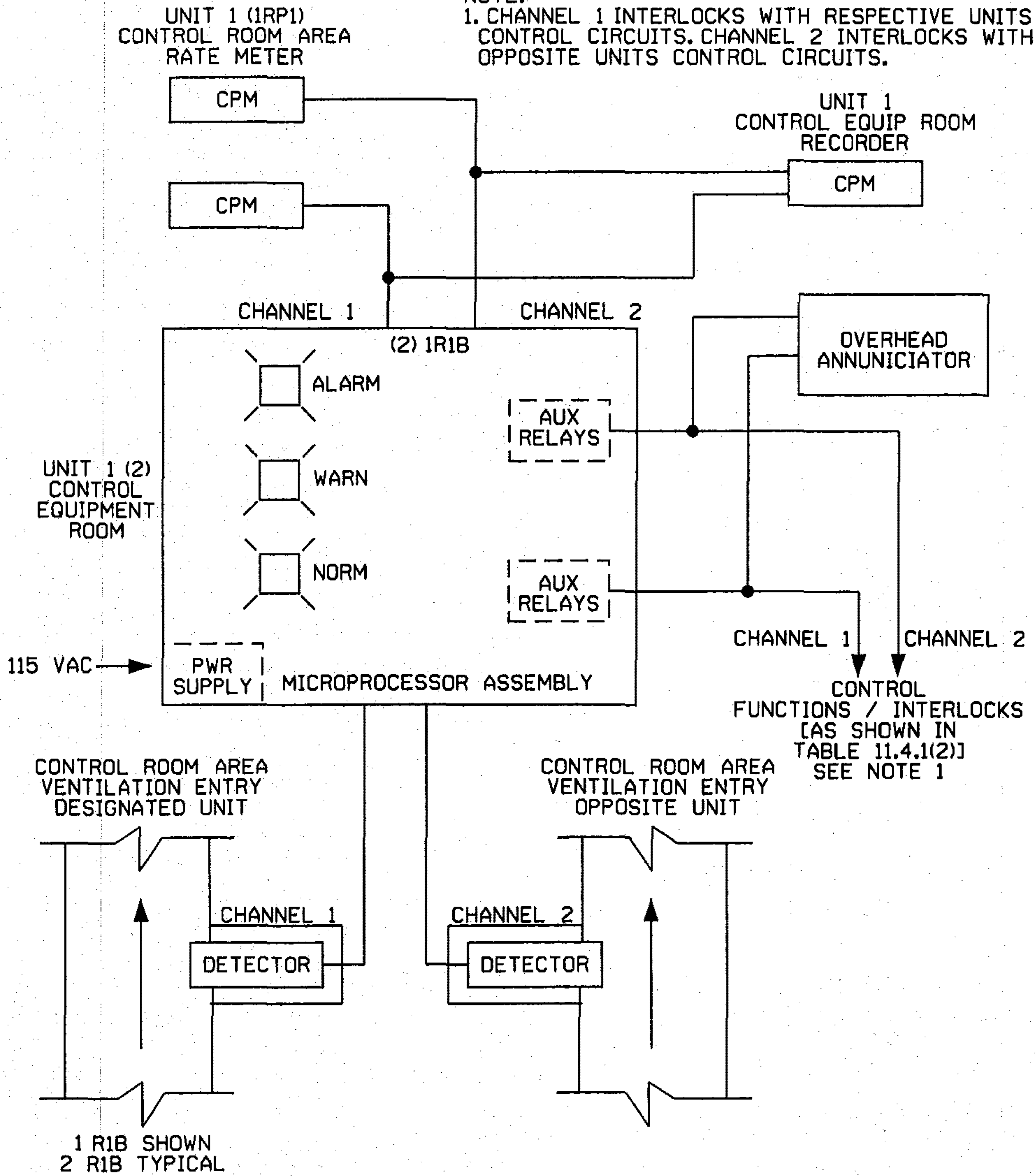
PSEG NUCLEAR LLC
SALEM GENERATING STATION

UNIT NO. 2 AREA MONITOR (TYPICAL)

Updated FSAR | Sheet 1 of 1

REVISION 27, NOVEMBER 25,2013 | Fig. 11.4-8

NOTE:
1. CHANNEL 1 INTERLOCKS WITH RESPECTIVE UNITS CONTROL CIRCUITS. CHANNEL 2 INTERLOCKS WITH OPPOSITE UNITS CONTROL CIRCUITS.



Revision 21, Dec. 6, 2004

PSEG Nuclear, LLC SALEM NUCLEAR GENERATING STATION	Salem Nuclear Generating Station CONTROL ROOM AREA INTAKE DUCT RADIATION MONITOR
	Updated FSAR — Figure 11.4-9

11.5 SOLID RADWASTE SYSTEM

The Solid Radwaste System collects, processes, packages, and provides temporary storage for radioactive solid wastes due for offsite shipment and permanent disposal.

11.5.1 Design Objectives

1. To provide a means of collecting spent demineralizer resins generated during plant operation.

2. To provide a means of packaging spent resins and expended filters in containers suitable for transfer from the plant site.

3. To provide a means of packaging low level contaminated solid wastes such as glassware and clothing.

4. To provide a means of processing spent resins and packaging for offsite shipment.

The design criteria for the Solid Radwaste Treatment System are as follows:

> The facility design shall include those means necessary to maintain control over the plant radioactive solid waste. Appropriate storage capacity shall be provided for retention of solid wastes. In all cases, the design for radioactivity control shall be justified (a) on the basis of 10CFR20 requirements, for both normal operations and for any transient situation that might reasonably be anticipated to occur and (b) on the basis of 10CFR50.67 dosage level limits for potential reactor accidents of exceedingly low probability of occurrence.

The solid waste facility is designed so that offsite shipments are in accordance with applicable governmental regulations.

The spent resins from the demineralizers and filter cartridges are packaged and stored onsite until shipment offsite for disposal.

11.5.2 System Design

The solid processing portion of the Waste Disposal System is designed to package all solid wastes for removal to volume reduction or burial facilities. All packaging shall meet DOT/NRC approval as applicable depending upon contents. Packages for burial will additionally conform to burial site facility criteria.

The resins are transferred to appropriate shipping containers for processing as necessary prior to transport.

Dry Active Wastes (DAW) are shipped to an off-site volume reduction facility.

11.5.3 Equipment Description

The Solid Radwaste System consists of components and subsystems described below.

Waste Processing Facility

The resin processing equipment is operated remotely from a control station that operates a fill head on the appropriate shipping container. The processing is done within the Auxiliary Building to control the release of air and liquid to the environment. Activity levels of the contents are monitored to limit the doses during shipment.

Off Site Volume Reduction

DAW is collected in Sea-vans and then shipped off-site for volume reduction processing by a licensed contractor. The volume-reduced DAW is placed into packages that will meet DOT/NRC approval and then shipped for disposal at a licensed burial site.

11.5.4 Expected Volumes

See Section 11.5.7

11.5.5 Packaging

Packaging is done in DOT/NRC approved packages, as appropriate, depending upon contents.

11.5.6 Storage Facilities

The storage areas are shielded to protect personnel in accessible portions of the solid radwaste area. The shielding is designed to meet the requirements of 10CFR20.

Low-level radwaste can be stored in the Low-Level Radwaste Storage Facility (LLRSF) for up to 5 years if shipping to a Low-Level Radwaste Disposal Facility is denied. The LLRSF has been specifically designed in accordance with guidelines provided in Generic Letter 81-38.

11.5.7 Shipment

The average annual volumes of solid wastes (on a two unit basis) shipped from the Salem Generating Station are as follows:

Spent resins, filter sludges	25 cubic meters
Dry compressible waste, contaminated equipment	100 cubic meters

The Process Control Program (PCP) has been approved by the Nuclear Regulatory Commission (NRC) which outlines the in-plant measures and controls to assure the suitability of radioactive waste products for transportation and/or disposal at a licensed burial site.

11.5.8 Reference for Section 11.5

1. Public Service Electric and Gas Letter, Liden to Varga, November 28, 1983

11.6 OFFSITE RADIOLOGICAL MONITORING PROGRAM

Since 1968 an Offsite Radiological Environmental Monitoring Program (REMP) has been conducted on the Artificial Island Site. Starting in December 1972, more extensive radiological monitoring programs were initiated with the purpose of identifying and quantifying the concentration of various radioactive elements in the different environmental media surrounding the Salem Generating Stations.

11.6.1 Program Objective

The objectives of the operational radiological environmental program are:

1. To fulfill the obligations of the radiological Environmental Monitoring sections of the Technical Specifications for the Salem Generating Station

2. To determine whether any significant increase occurs in the concentration of radionuclides in critical pathways

3. To determine if the station has caused an increase in the radioactive inventory of long lived radionuclides

4. To detect any change in ambient gamma radiation levels

5. To verify that the Salem Generating Station operations have no detrimental effects on the health and safety of the public or on the environment

The basis for the selection of sample locations was determined from site meteorological, hydrological, and geographical constraints enclosed by local demography and land use. Sample locations chosen are consistent with the U.S. Nuclear Regulatory Commission Branch Technical Position on the REMP. Sample locations were divided into two basic classes, indicator and control stations. Indicator stations were situated in areas most likely to be affected by a station release. All control stations were presumed to be located at a distance from the station to be unaffected by normal and accident plant operation. In this manner, fluctuations in the level of radionuclides and direct radiation at an indicator station could be compared to those, if any, at a control location to determine if the station was the cause of the fluctuations.

11.6.2 Preoperational and Operational Programs

The operational REMP for the Artificial Island Site was initiated in December 1976 after Unit 1 achieved criticality. The current operational program is described in the Offsite Dose Calculation Manual (ODCM). Table 11.6-2 provides the results obtained during the preoperational phase of the program.

11.6.3 Expected Pathways

The scope of the REMP is to determine whether any significant increases have occurred in the concentration of radioactive materials in the critical pathways as well as to determine if there is an increase in the radioactive inventory of long-lived radionuclei. During the months when cows are in pasture, the critical pathway has been determined to be the air, grass, milk, to man pathway. During those months when cows are not on pasture the air, broadleaf vegetation, to man pathways predominates. The selection of sample type was based upon these established critical pathways of the transfer of radionuclei, from the environment to man as well as experience of the preoperational phase of the program.

11.6.4 Physical Characteristics of Samples

The physical characteristics of the samples are summarized in the Offsite Dose Calculation Manual (ODCM).

The sensitivity of the analytical procedures used are consistent with the Branch Technical Position on the procedures for Environmental surveillance programs and are provided in the Radiological Monitoring sections of the Technical Specifications. The sensitivity of the analytical measurements was selected to verify that the measurable concentrations of radioactive materials are not higher than expected based on effluent monitoring and modeling.

The sensitivity requirements listed in the Radiological Monitoring sections of the Technical Specifications are sufficiently low to discern dose commitments which are a fraction of 10CFR50, Appendix I for there predicted critical pathways. It should be noted that deviations are permitted from the required sampling schedule if specimens are unobtainable due to hazardous conditions, seasonal unavailability, or the malfunction of automatic sampling equipment. All deviations from the sampling schedule are documented in the annual report. Descriptions of the analytical methods utilized are provided in the Annual Radiological Reports.

TABLE 11.6-1

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TABLE 11.6-2

SUMMARY OF RADIONUCLIDE CONCENTRATIONS IN ARTIFICIAL ISLAND PREOPERATIONAL RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM SAMPLES

Medium and Analysis Performed	Number of Samples Analyzed	Number Above MDL	Minimum	Maximum	Average ± 2 Sigma	Units
AQUATIC ENVIRONMENT						
Surface Water						
H-3	259	222	<80	600±100	213±165	pCi/l
Alpha (soluble)	136	6	<1.5	33±32	-	pCi/l
Alpha (insoluble)	136	5	<1.5	3.3±1.0		pCi/l
Alpha (total)	123	22	<1.5	27±20	-	pCi/l
Beta (soluble)	136	136	3.8±2.5	120±16	42±52	pCi/l
Beta (insoluble)	136	14	<3.0	4.8±3.2	-	pCi/l
Beta (total)	123	123	3.3±2.7	110±11	32±45	pCi/l
K-40	136	136	0.20±0.02	120±12	41±63	pCi/l
Sr-89	65	5	<0.37	1.5±0.8		pCi/l
Sr-90	77	20	<0.28	1.6±0.4	-	pCi/l
Gamma	259					
K-40		200	<6	200±30	48±64	pCi/l
ZrNb-95		2	<0.4	1.9±0.6	-	pCi/l
Cs-137		2	<0.5	0.86±0.56		pCi/l
Ra-226		9	<0.9	4.0±1.4	-	pCi/l
Solids	136	136	160±16	14000±1400	5748±7137	mg/l
Chlorides	136	136	32±3	17000±1700	3345±5567	mg/l
Edible Fish Flesh						
H-3 (aqueous)	28	15	<80	460±78	165±206	pCi/l
H-3 (organic)	19	17	<130	480±69	285±205	pCi/l
H-3 (organic)	8	4	<80	390±80	158±230	pCi/kg(dry)
Sr-89	9	0	<4.1	<100	-	pCi/kg(wet)
Sr-90	12	1	<4.1	67±11	-	pCi/kg(wet)

TABLE 11.6-2 (Cont)

Medium and Analysis Performed	Number of Samples Analyzed	Number Above MDL	Minimum	Maximum	Average ± 2 Sigma	Units
Edible Fish Flesh (Cont)						
Gamma	31					
K-40		31	1000±100	13000±1000	2914±4351	pCi/kg(wet)
Cs-137		5	8.7±6.6	11±6	-	pCi/kg(wet)
Ra-226		1	<4	130±80	-	pCi/kg(wet)
Edible Fish Bone						
Sr-89	18	3	100±60	100±70	-	pCi/kg(wet dry)
Sr-90	18	14	<24	940±100	335±614	pCi/kg(wet dry)
Blue Crab - Edible Hard Shell (Flesh)						
H-3 (Aqueous)	8	7	<80	330±71	180±168	pCi/l
H-3 (Organic)	2	2	95±78	420±120	259±455	pCi/l
H-3 (Organic)	4	1	<80	90±80	-	pCi/kg(dry)
H-3 (Total Tritium)	7	5	<80	420±60	219±259	pCi/l
Sr-89	10	0	<6.0	<51	-	pCi/kg(wet)
Sr-90	10	7	<5.1	150±26	40±103	pCi/kg(wet)
Gamma						
K-40	16	16	960±384	12000±1000	2835±5048	pCi/kg(wet dry)
ZrNb-95		1	<5	120±12	-	pCi/kg(wet)
Ra-226		3	<10	33±19	-	pCi/kg(wet)
Blue Crab - Edible Soft Shell (Total)						
H-3 (Aqueous)	4	3	<80	320±110	190±197	pCi/l
H-3 (Organic)	2	0	<140	<220	-	pCi/l
H-3 (Organic)	2	1	<80	130±80	105±71	pCi/kg(dry)
H-3 (Total Tritium)	4	3	<80	500±68	230±373	pCi/l
Sr-89	6	1	<7.5	<26	-	pCi/kg(wet)
Sr-90	8	5	<7.8	39±6	21±24	pCi/kg(wet)
Gamma						
K-40	7	7	770±462	3000±300	2040±1359	pCi/kg(wet)
Ra-226		1	<10	37±15	-	pCi/kg(wet)

TABLE 11.6-2 (Cont)

Medium and Analysis Performed	Number of Samples Analyzed	Number Above MDL	Minimum	Maximum	Average ± 2 Sigma	Units
Blue Crab - Nonedible Hard Shell (Shell)						
Sr-89	8	3	<57	210±170	-	pCi/kg(dry)
SR-90	8	8	330±30	990±87	614±511	pCi/kg(dry)
Prey Fish						
Sr-89	16	2	<5.8	320±22	-	pCi/kg(wet dry)
Sr-90	16	8	<4.8	66±11	28±40	pCi/kg(wet dry)
Gamma	29					
K-40		29	970±100	13000±1000	4604±6667	pCi/kg(wet dry)
ZrNb-95		2	<3	17±5	-	pCi/kg(wet)
Cs-137		1	<2	42±38	-	pCi/kg(wet)
Ra-226		1	<5	13±9	-	pCi/kg(wet)
Benthos						
Sr-89	12	1	<44	<36000	-	pCi/kg(dry)
Sr-90	12	5	<120	<30000	-	pCi/kg(dry)
Gamma						
K-40	4	2	<1	6.9±1.8	3.4±5.0	pCi/g(dry)
Mn-54		1	<0.03	0.13±0.09	-	pCi/g(dry)
Nb-95		2	<0.09	0.11±0.08	0.11±0.13	pCi/g(dry)
Ru-106		1	<0.3	0.91±0.65	-	pCi/g(dry)
Cs-137		1	<0.07	<0.1	-	pCi/g(dry)
Ra-226		4	0.26±0.08	0.48±0.16	0.38±0.18	pCi/g(dry)
Th-232		4	0.52±0.13	1.2±0.4	0.80±0.61	pCi/g(dry)
Zooplankton						
Sr-89	8	2	<0.21	4.6±4.4	-	pCi/g(dry)
Sr-90	8	5	<0.51	<4.9	1.3±2.9	pCi/g(dry)

TABLE 11.6-2 (Cont)

Medium and Analysis Performed	Number of Samples Analyzed	Number Above MDL	Minimum	Maximum	Average ± 2 Sigma	Units
Zooplankton (Cont)						
Gamma	20					
Be-7		1	-	3.0±2.8	-	pCi/g(dry)
K-40		3	<3	110±80	-	pCi/g(dry)
ZrNb-95		2	<0.09	<10	-	pCi/G(dry)
Ra-226		1	<0.2	<30	-	pCi/g(dry)
Sediment						
Sr-89	12	0	<0.03	<1.0	-	pCi/g(dry)
Sr-90	16	4	<0.03	0.32±0.05	-	pCi/g(dry)
Gamma	41					
Be-7		3	<0.1	2.3±0.3	-	pCi/g(dry)
K-40		41	3.4±0.4	21±2	15±23	pCi/g(dry)
Nb-95		8	<0.01	2.6±0.6	-	pCi/g(dry)
Zr-95		1	<0.02	0.70±0.30	-	pCi/g(dry)
Ru-103		1	<0.01	0.31±0.13	-	pCi/g(dry)
Ru-106		2	0.03±0.02	0.30±0.17	-	pCi/g(dry)
Sb-125		8	0.05±0.04	0.27±0.12	-	pCi/g(dry)
Cs-137		35	<0.01	0.40±0.04	0.15±0.22	pCi/g(dry)
Ce-144		2	<0.1	0.48±0.14	-	pCi/g(dry)
Ra-226		41	0.28±0.04	1.2±0.1	0.76±0.43	pCi/g(dry)
Th-232		41	0.21±0.11	1.3±0.1	0.84±0.54	pCi/g(dry)
ATMOSPHERIC ENVIRONMENT						
Air Particulates						
Alpha	1045	788	<0.16	7.9±3.1	1.1±2.8	10^{-3}pCi/m³
Beta	1088	1585	5.0	920±24	74±280	10^{-3}pCi/m³
Sr-89	60	18	<0.15	4.7±0.5	-	10^{-3}pci/m³
Sr-90	60	46	<0.20	3.0±0.3	0.9±1.6	10^{-3}pci/m³

Medium and Analysis Performed	Number of Samples Analyzed	Number Above MDL	Minimum	Maximum	Average ± 2 Sigma	Units
Air Particulates (Cont)						
Gamma	127					
Be-7		127	12±7	330±33	109±126	10^{-3} pCi/m³
Mn-54		31	<0.1	2.6±0.4	-	10^{-3} pCi/m³
Zr-95		80	<0.3	44±5	5±15	10^{-3} pCi/m³
Nb-95		74	<0.4	27±4	4±13	10^{-3} pCi/m³
Ru-103		90	<0.1	84±8	13±39	10^{-3} pCi/m³
Ru-106		60	<0.9	46±4	-	10^{-3} pCi/m³
Sb-125		55	<0.4	6.2±0.8	-	10^{-3} pCi/m³
Cs-237		122	<0.2	11±1	2.2±4.7	10^{-3} pCi/m³
BaLa-140		3	<0.4	27±8	-	10^{-3} pCi/m³
Ce-141		38	<0.2	46±5	-	10^{-3} pCi/m³
Ce-144		99	<1	120±15	19±54	10^{-3} pCi/m³
Ru-226		26	<0.6	16±7	-	10^{-3} pCi/m³
Th-232		24	<0.5	3.1±0.18	-	10^{-3} pCi/m³
Air Iodine						
I-131	519	20	<0.72	42±4	-	10^{-3} pCi/m³
Precipitation						
H-3	63	49	<80	610±90	216±290	pCi/1
Alpha	63	6	<0.45	4.7±1.8	-	pCi/1
Beta	63	57	<3.0	71±8	19±36	pCi/1
K-40	35	32	<0.01	0.52±0.05	0.13±0.21	pCi/1
Sr-89	16	7	<0.43	5.6±1.2	-	pCi/1
Sr-90	20	12	<0.48	3.8±1.1	1.5±2.2	pCi/1
Gamma	22					
Be-7		16	0.75±0.22	79±74	29±45	pCi/1
K-40		3	<5	18±5	-	pCi/1
Kr/Nb-95		9	<0.4	9.5±1.0	-	pCi/1
Ru-103		2	0.48±0.29	3.4±0.5	-	pCi/1
Cs-137		1	<0.5	1.2±0.4	-	pCi/1
BaLa-140		1	<0.4	2.2±0.9	-	pCi/1
Ce-144		1	<3	6.2±2.2	-	pCi/1

TABLE 11.6-2 (Cont)

Medium and Analysis Performed	Number of Samples Analyzed	Number Above MDL	Minimum	Maximum	Average ± 2 Sigma	Units
TERRESTRIAL ENVIRONMENT						
Well Water						
H-3	144	23	<40	380±77	-	pCi/l
Alpha	144	4	<1.0	9.6±2.5	-	pCi/l
Beta	144	126	<2.1	38±6	9±10	pCi/l
K-40	134	134	1.1±0.1	19±2	7.8±7.6	pCi/l
Sr-89	24	1	<0.49	<2.1	-	pCi/l
Sr-90	24	3	<0.36	0.87±0.52	-	pCi/l
Gamma	39					
Be-7		3	32±7	45±9	-	pCi/l
K-40		10	<0.5	30±14	-	pCi/l
ZrNb-95		2	<0.4	2.0±0.7	-	pCi/l
Ra-226		1	<0.8	2.0±1.2	-	pCi/l
Potable Water (Raw and treated)						
H-3	94	73	<80	350±80	179±173	pCi/l
Alpha	94	16	<0.53	2.7±1.3	-	pCi/l
Beta	94	63	<2.6	9.0±3.4	4.2±3.0	pCi/l
K-40	84	84	0.50±0.10	10±1	1.7±2.2	pCi/l
Sr-89	24	3	<0.33	1.1±0.9	-	pCi/l
Sr-90	28	10	<0.26	2.1±1.2	-	pCi/l
Gamma	32					
Ra-226		1	<0.8	1.4±1.0	-	pCi/l
Milk						
Sr-89	177	35	<0.55	14±2	-	pCi/l
Sr-90	193	169	<0.23	12±3	3.5±7.5	pCi/l
I-131	167	44	<0.03	65±6	-	pCi/l

Medium and Analysis Performed	Number of Samples Analyzed	Number Above MDL	Minimum	Maximum	Average ± 2 Sigma	Units
Milk (Cont)						
Gamma	270					
K-40		270	800±100	2000±200	1437±440	pCi/1
I-131		5	<0.4	63±14	-	pCi/1
Cs-137		147	<0.4	14±11	3.0±5.1	pCi/1
Ra-226		9	<0.9	<30		pCi/1
Food Products						
Sr-89	29	3	<2.5	10±6*	-	PCi/kg(wet)
Sr-90	45	24	<1.5	24±6*	7±11	PCi/kg(wet)
Gamma	52					
K-40		52	70±50	4800±480	2141±1628	PCi/kg(wet)
Cs-137		7	<1	59±33	-	PCi/kg(wet)
Fodder Crops (dry)						
Gamma	41					
Be-7		21	0.80±30	9.3±0.9	-	pCi/g(dry)
K-40		41	2.9±0.5	80±8	22±33	pCi/g(dry)
Co-58		3	<0.02	0.07±0.05	-	pCi/g(dry)
ZrNb-95		18	<0.02	9.5±4.8	-	pCi/g(dry)
Zr-95		3	4.3±0.4	6.3±0.6	-	pCi/g(dry)
Nb-95		3	3.0±0.3	4.2±0.4	-	pCi/g(dry)
Ru-103		4	0.09±0.02	1.3±0.1	-	pCi/g(dry)
RuRj-206		14	<0.01	2.5±0.2	-	pCi/g(dry)
I-131		3	<0.01	0.53±0.06	-	pCi/g(dry)
I-133		3	<0.02	0.46±0.11	-	pCi/g(dry)
Cs-137		6	<0.01	0.11±0.02	-	pCi/g(dry)
BaLa-140		3	<0.02	3.1±0.3	-	pCi/g(dry)

*Additional egg plant sampled 10-13-76 reported in units of pCi/kg(dry).
Results were: Sr-89, <55 pCi/kg(dry); Sr-90, 32±24 pCi/kg(dry).
(These data not included in ranges, or average).

TABLE 11.6-2 (Cont)

Medium and Analysis Performed	Number of Samples Analyzed		Number Above MDL	Minimum	Maximum	Average ± 2 Sigma	Units
Fodder Crops (dry) (Cont)							
Gamma							
Ce-141		4	0.28±0.03	5.1±0.5	-	pCi/g(dry)	
Ce-144		5	<0.1	1.4±0.3	-	pCi/g(dry)	
Ra-226		1	<0.04	0.10±0.04	-	pCi/g(dry)	
Th-232		3	<0.07	0.39±0.08	-	pCi/g(dry)	
Fodder Crops (wet)							
Gamma	5						
Be-7		4	0.47±0.17	4.7±0.5	2.0±3.7	pCi/g(wet)	
K-40		5	2.9±0.5	16±2	7±11	pCi/g(wet)	
ZrNb-95		1	-	0.03±0.02	-	pCi/g(wet)	
Zr-95		4	0.28±0.04	1.9±0.2	1.4±1.5	pCi/g(wet)	
Nb-95		4	0.04±0.02	0.36±0.04	0.27±0.30	pCi/g(wet)	
Mo-99		4	<0.01	2.2±0.2	1.1±1.9	pCi/g(wet)	
Ru-103		4	0.04±0.02	0.59±0.05	0.21±0.46	pCi/g(wet)	
I-131		4	<0.01	2.4±0.2	1.1±2.0	pCi/g(wet)	
I-132		4	0.23±0.06	1.4±0.1	0.79±0.96	pCi/g(wet)	
Te-132		4	<0.01	1.8±0.2	0.7±1.5	pCi/g(wet)	
I-133		4	<0.01	0.3±0.05	0.19±0.31	pCi/g(wet)	
Ba-140		4	0.50±0.22	2.8±2.0	1.8±2.0	pCi/g(wet)	
La-140		4	0.13±0.03	3.0±0.3	1.8±2.4	pCi/g(wet)	
Ce-141		4	0.06±0.03	1.1±0.1	0.77±0.96	pCi/g(wet)	
Ce-144		1	<0.08	0.45±0.25	-	pCi/g(wet)	
Np-239		3	0.72±0.20	3.2±0.3	2.0±2.5	pCi/g(wet)	
Game							
Sr-89	11	1	<90	<800	-	pCi/kg(dry)	
Sr-90	11	11	32±16	1800±600	249±540	pCi/kg(dry)	
Gamma	16						
K-40		16	1600±200	27000±800	4444±12124	pCi/kg(wet dry)	
RuRh-106		1	<10	<66±55	-	pCi/kg(wet)	
Cs-137		1	<3	620±50	-	pCi/kg(wet)	
Ra-226		2	<6	1000±100	-	pCi/kg(wet)	
Th-232		2	20±10	140±40	-	pCi/kg(wet)	

Medium and Analysis Performed	Number of Samples Analyzed	Number Above MDL	Minimum	Maximum	Average ±2 Sigma	Units
Fodder Crops (wet) (Cont.)						
Thyroid						
Gamma	7					
K-40		1	<400	1800±600	-	PCi/kg(wet)
I-131		1	<30	89±1	-	PCi/kg(wet)
Soil						
Sr-90	21	18	<0.03	1.1±0.1	0.26±0.50	pCi/g(dry)
Gamma	30					
Be-7		4	0.18±0.13	21±10	-	pCi/g(dry)
K-40		30	3.4±0.4	24±2	10±8	pCi/g(dry)
Mn-54		2	<0.01	0.01±0.01	-	pCi/g(dry)
ZrNb-95		1	<0.02	0.02±0.02	-	pCi/g(dry)
Zr-95		2	0.03±0.02	0.06±0.04	-	pCi/g(dry)
Nb-95		3	0.07±0.02	0.09±0.02	-	pCi/g(dry)
Sb-125		4	0.05±0.04	0.27±0.07	-	pCi/g(dry)
Cs-137		29	<0.1	2.8±0.3	0.8±1.5	pCi/g(dry)
Ce-144		2	<0.09	0.51±0.12	-	pCi/g(dry)
Ra-226		30	0.27±0.05	1.5±0.2	0.87±0.65	pCi/g(dry)
Th-232		30	0.14±0.06	1.4±0.1	0.74±0.58	pCi/g(dry)
DIRECT RADIATON						
Gamma Dose	3872M	3872	2.07±0.14	7.28±0.36	4.26±1.53	mrad/std.mon.
	1256Q	1256	1.07±0.24	5.60±0.54	4.62±1.18	mrad/std.mon.
	260SA	260	2.83±0.10	5.21±0.22	4.30±1.15	mrad/std.mon.