11.0 RADIOACTIVE WASTE MANAGEMENT

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11.0 RADIOACTIVE WASTE MANAGEMENT

11.1 SOURCE TERMS

The fission product inventory in the reactor core and the diffusion to the fuel pellet/cladding gap are presented in chapter 15. In this section two source terms are presented. The first is a conservative design base that utilizes a conventional fuel clad defect model. This design model serves as a basis for system and shielding requirements and calculations of the maximum offsite doses resulting from credible accidents.

The second source term is a realistic model used to predict expected long-term average concentrations of radionuclides in the primary and secondary fluid stream and an average plant's releases over its lifetime. This realistic model, based on available measured nuclide concentrations during normal operation, was formulated as a standard for the American National Standard Source Term Specifications, ANSI N237,[1]and is the source term model used in NUREG-0017.[2][3]

11.1.1 REACTOR COOLANT AND SECONDARY SIDE ACTIVITY

11.1.1.1 Design Basis Model

The parameters used in the calculation of the reactor coolant fission product inventories, together with the pertinent information concerning the expected coolant cleanup flowrate and demineralizer effectiveness, are summarized in table 11.1-1. The results of the calculations are presented in table 11.1-2. In these calculations the defective fuel rods are assumed to be present at the initial core loading. The fission product escape rate coefficients are based upon average fuel temperature and are further based on fuel defect tests performed at the Saxton reactor. Recent experience at two plants operating with fuel rod defects has verified the escape rate coefficients listed in table 11.1-1.

For further information on core fission product calculations see subsection 15.1.7.

For fuel failure and burnup experience see chapter 4.

The fission product activities in the reactor coolant during operation with small cladding defects (fuel rods containing pinholes or fine cracks) are computed using the following differential equations.

For parent nuclides in the coolant:

$$\frac{dN_{wi}}{dt} = D\nu_i N_{Ci} - (\lambda_i + R\eta_i + \frac{B'}{B_o - tB'}) N_{wi}$$

For daughter nuclides in the coolant:

$$\frac{dN_{wj}}{dt} = D\nu_j N_{Cj} - (\lambda_j + R\eta_j + \frac{B'}{B_o - tB'}) N_{wj} + \lambda_i N_{wi}$$

where:

N_o = Nuclide concentration.

D = Clad defects, as a fraction of rated core thermal power being generated by rods with clad defects.

R = Purification flow (coolant system volumes/s).

B_o = Initial boron concentration (ppm).

B' = Boron concentration reduction rate by feed and bleed (ppm/s).

η = Removal efficiency of purification cycle for nuclide.

λ = Radioactive decay constant (s^{-1}).

ν = Escape rate coefficient for diffusion into coolant.

t = Time (s).

C = Refers to core.

w = Refers to coolant.

i = Refers to parent nuclide.

j = Refers to daughter nuclide.

Table 11.1-3 lists the activities in the volume control tank using the assumptions summarized in table 11.1-1.

The activities in the pressurizer are separated between the liquid and the steam phase, and the results obtained are given in table 11.1-4 using the assumptions summarized in table 11.1-1.

The activities to be found in the gaseous waste processing system are given in table 11.1-5.

As a necessary part of the effort to reduce effluent of radioactive liquid wastes, Westinghouse has been surveying various pressurized water reactor (PWR) facilities which are in operation to identify design and operating problems influencing reactor coolant and nonreactor grade leakage and hence the load on the waste processing system.

Leakage sources have been identified in connection with pump shaft seals and valve stem leakage.

When packed glands are provided a leakage problem may be anticipated, while mechanical shaft seals provide essentially zero leakage. Valve stem leakage was experienced when the originally specified packing was used. A combination of graphite filament yarn packing sandwiched with asbestos sheet packing is used with improved results in several plants. A bellows seal being utilized in later plants eliminates all stem leakage.

In addition, seat leakage was experienced on some pressurizer power-operated relief valves. However, this was found to be due to a manufacturing error and has been corrected.

Current PWR design is based on a reactor coolant leakage of 20 gal/day/unit into the floor drain tank. Nonreactor grade leakage entering the floor drain tank from the containment and the auxiliary building is assumed to be 40 gal/day. In addition, an excessive reactor coolant leakage of 1 gal/min can be handled under abnormal operating conditions. Although leakage from the primary system to the secondary side in the steam generator is unlikely, secondary side fission product activity is evaluated by applying an expected 0.2 percent defective fuel and 20 gal/day primary to secondary side leakage.

Table 11.3-1 gives conservatively estimated leakages from the gaseous waste processing system with corresponding design activity discharges from the plant vent stack. The activity releases are based on a leakage of 100 sf^3/year with cladding defects in fuel rods generating 1 percent of the rated core thermal power. The leak rate is based on the sensitivity of commercially available portable leak detectors.

11.1.1.2 Realistic Model

The parameters used to describe the realistic model are given in table 11.1-6 together with the range of values utilized by ANSI N237-1976. Corrections have been made according to the ANSI N237-1976 standard formulas. Operation of a Westinghouse gaseous waste management system is assumed.

Regulatory Guide 1.112[4], appendix B, recommends input parameters needed to execute the gaseous and liquid effluents (GALE) computer code[2] for pressurized water reactors. These values are listed in table 11.1-7.

11.1.2 RADIOACTIVE RELEASE SOURCES

Total plant liquid and gaseous releases are discussed in subsections 11.2.6 and 11.3.6, respectively. Release pathways for gaseous effluents are described in subsection 12.2.2. Liquid release paths include the steam generator blowdown processing system, as described in subsection 10.4.8, and the turbine building drains. The turbine building sumps are periodically pumped out at a rate of approximately 800 gal/min per each pump. This flowrate is utilized to determine the total volume released from the turbine building sumps. The liquid release paths are shown in figure 9.2-1.

REFERENCES

1. American National Standard Source Term Specification, ANSI N237-1976/ANS-18.1, approved May 11, 1976.

2. U.S. Nuclear Regulatory Commission, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors," NUREG-0017, Office of Standard Development, April 1976.

3. Alabama Power Company letter to the Nuclear Regulatory Commission, "Dose Calculations to Conform with Appendix I Requirements," U.S. NRC Docket Nos. 50-348 and 50-364, June 3, 1976.

4. U.S. NRC Regulatory Guide 1.112, Revision O-R, “Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Power Reactors,” April 1976.

TABLE 11.1-1 (SHEET 1 OF 3)

PARAMETERS USED IN THE CALCULATION OF REACTOR COOLANT FISSION AND CORROSION PRODUCT ACTIVITIES[c]

Parameter	Value
Reactor power (MWt)	2774
Clad defects, as a percent of rated core thermal power being generated by rods with clad defects	1.0[d]
Reactor coolant liquid volume (ft^3)	9723.3
Reactor coolant full power average temperature (°F)	577
Normal purification flowrate (gal/min)	60[d]
Effective cation demineralizer flow (gal/min)	6.0[d]
Volume tank volumes	
Vapor (ft^3)	175
Liquid (ft^3)	125
Fission product escape rate coefficients	
Noble gas isotopes (s^{-1})	6.5×10^{-8}
Br, I, and Cs isotopes (s^{-1})	1.3×10^{-8}
Te isotopes (s^{-1})	1.0×10^{-9}
Mo isotopes (s^{-1})	2.0×10^{-9}
Sr and Ba isotopes (s^{-1})	1.0×10^{-11}
Y, La, Ce, and Pr isotopes (s^{-1})	1.6×10^{-12}
Mixed bed demineralizer decontamination factors	
Noble gases and Cs-134, Cs-136, Cs-137, Y-90, Y-91, and Mo-99	1.0
All other isotopes including corrosion products	10.0

TABLE 11.1-1 (SHEET 2 OF 3)

Parameter	Value
Cation bed demineralizer decontamination factor for Cs-134, Cs-136, Cs-137, Y-90, Y-91, and Mo-99	10.0
Boron concentration and reduction rates	
B_o, initial cycle (ppm)	700
B', initial cycle (ppm/day)	2.5
B_o, equilibrium cycle (ppm)	1000
B', equilibrium cycle (ppm/day)	3.5
Pressurizer volumes	
Vapor (ft^3)	560
Liquid (ft^3)	840
Spray line flow (gal/min)	1.0
Pressurizer stripping fractions	
Noble gases	1.0
All other elements	0
Volume control tank noble gas and iodine stripping fractions	

Isotope	Stripping Fraction[(a)]
Kr-85	2.5×10^{-1}
Kr-85m	2.9×10^{-1}
Kr-87	6.0×10^{-1}
Kr-88	4.3×10^{-1}
Xe-131m	2.5×10^{-1}
Xe-133	2.5×10^{-1}
Xe-133m	2.6×10^{-1}
Xe-135	2.8×10^{-1}
Xe-138	8.0×10^{-1}

TABLE 11.1-1 (SHEET 3 OF 3)

Isotope	Partition Coefficient[b]
I-131	100
I-132	100
I-133	100
I-134	100
I-135	100

a. Fraction of inlet isotope concentration stripped off in the volume control tank.

b. Ratio between isotopic concentration in the liquid phase and isotope concentration in the vapor phase.

c. Reviewed for power uprate, RCS activities in table 11.1-2 remain bounding for power uprate

d. Evaluations of the impact of increasing letdown flow to 145 gpm (and cation demineralizer flow to 14.5 gpm) indicate this original combination of parameters continues to give conservatively high design basis source terms.

TABLE 11.1-2 (SHEET 1 OF 2)

REACTOR COOLANT EQUILIBRIUM FISSION AND CORROSION PRODUCT ACTIVITIES[a]

Isotope	Activity (μCi/g)
Br-84	4.4×10^{-2}
Rb-88	3.8
Rb-89	0.10
Sr-89	4.1×10^{-3}
Sr-90	1.4×10^{-4}
Sr-91	2.0×10^{-3}
Y-90	1.7×10^{-4}
Y-91	6.1×10^{-3}
Y-92	7.3×10^{-4}
Zr-95	7.0×10^{-4}
Nb-95	6.9×10^{-4}
Mo-99	5.5
I-131	2.5
I-132	0.9
I-133	4.0
I-134	0.6
I-135	2.2
Te-132	0.3
Te-134	3.0×10^{-2}
Cs-134	0.26
Cs-136	0.15
Cs-137	1.3
Cs-138	0.96
Ba-140	4.2×10^{-3}
La-140	1.4×10^{-3}
Ce-144	3.3×10^{-4}
Pr-144	3.2×10^{-4}
Kr-85	0.14
Kr-85m	2.1
Kr-87	1.3
Kr-88	3.6
Xe-131m	0.21
Xe-133	7.98×10^{1}
Xe-133m	1.5
Xe-135	5.7
Xe-135m	0.19
Xe-138	0.68
Mn-54[b]	7.8×10^{-4}
Mn-56[b]	2.9×10^{-2}
Co-58[b]	1.3×10^{-1}
Co-60[b]	7.5×10^{-4}

TABLE 11.1-2 (SHEET 2 OF 2)

Isotope	Activity ($\mu Ci/g$)
Fe-59[(b)]	1.0×10^{-3}
Cr-51[(b)]	9.5×10^{-4}
Zn-65	8.0×10^{-3}

a. Based on parameters given in table 11.1-1.

b. Corrosion product activities based on activity levels measured at operating reactors.

TABLE 11.1-3

EQUILIBRIUM VOLUME CONTROL TANK ACTIVITIES[a]

Isotope	Vapor Activity (Ci)
Kr-85	4.3
Kr-85m	3.0×10^{1}
Kr-87	1.5×10^{1}
Kr-88	6.0×10^{1}
Xe-131m	4.5
Xe-133	2.24×10^{3}
Xe-133m	4.2×10^{1}
Xe-135	1.2×10^{2}
Xe-135m	0.35
Xe-138	1.3
I-131	0.0122
I-132	0.00448
I-133	0.0196
I-134	0.00266
I-135	0.0105

Isotope	Liquid Activity (Ci)
I-131	0.87
I-132	0.32
I-133	1.4
I-134	0.19
I-135	0.75

a. Based on parameters given in table 11.1-1.

TABLE 11.1-4

PRESSURIZER ACTIVITIES[a]

Isotope	Vapor Activity ($\mu Ci/cm^3$)
Kr-85	1.5
Kr-85m	1.4×10^{-1}
Kr-87	2.6×10^{-2}
Kr-88	1.6×10^{-1}
Xe-131m	3.6×10^{-1}
Xe-133	1.4×10^{2}
Xe-133m	1.2
Xe-135	7.9×10^{-1}
Xe-135m	8.0×10^{-4}
Xe-138	3.2×10^{-3}

Isotope	Liquid Activity ($(\mu)Ci/cm^3$)
N-16 (max)	1.5
Rb-88	1.7×10^{-2}
Mo-99	2.7
I-131	1.8
I-132	2.9×10^{-2}
I-133	0.9
I-134	7.5×10^{-3}
I-135	0.17
Cs-134	2.6×10^{-1}
Cs-136	1.4×10^{-1}
Cs-137	1.3
Cs-138	7.7×10^{-3}

a. Based on parameters given in table 11.1-1.

TABLE 11.1-5

TOTAL WASTE GAS PROCESSING SYSTEM INVENTORY[a][b]

Isotope	Activity (Ci)
Kr-85	4.75×10^4
Kr-85m	2.2×10^1
Kr-87	3.0
Kr-88	2.8×10^1
Xe-131m	4.3×10^2
Xe-133	4.8×10^4
Xe-133m	3.9×10^2
Xe-135	1.8×10^2
Xe-135m	0.03
Xe-138	0.13
I-131	0.6624
I-132	0.00273
I-133	0.1036
I-134	0.000576
I-135	0.02016

a. Assuming 40 years of full power operation and no releases or leakages from the system.

b. The renewed operating licenses authorize an additional 20-year period of extended operation for both FNP units, resulting in a plant operating life of 60 years. Since the GWPS has not been operated in the continuous purge (and holdup) mode in the past, the inventory accumulated in the GWPS up to the date of the renewed licenses (over 20 years of plant operation) is essentially nil. Should the system begin to be operated in the continuous purge mode at any time for the remaining life of the plant, the stated 40-year inventory of the GWPS is bounding for the period of extended operation. Refer to section 11.3 for a description of GWPS operation.

TABLE 11.1-6

PARAMETERS USED TO DESCRIBE THE REACTOR SYSTEM-REALISTIC BASIS

				ANSI N237 Range	
Parameter	Symbol	Units	Value	Maximum	Minimum
Thermal power	P	MWt	3565	3800	3000
Steam flowrate	FS	lb/h	1.2×10^{7}	1.7×10^{7}	1.3×10^{7}
Weight of water in reactor coolant system	WP	lb	4.0×10^{5}	6.0×10^{5}	5.0×10^{5}
Weight of water in all steam generators	WS	lb	2.7×10^{5}	5.0×10^{5}	4.0×10^{5}
Reactor coolant letdown flow (purification)	FD	lb/h	3.0×10^{4}	4.2×10^{4}	3.2×10^{4}
Reactor coolant letdown flow (yearly average for boron control)	FB	lb/h	538	1.0×10^{3}	2.5 x 10
Steam generator blowdown flow (total)	FBD	lb/h	3.8×10^{4}	1.0×10^{5}	5.0×10^{4}
Fraction of radioactivity in blowdown steam that is not returned to the secondary coolant system	NBD	-	1.0	1.0	0.9
Flow through the purification system cation demineralizer	FA	lb/h	3.0×10^{3}	7.5×10^{4}	0.0
Ratio of condensate demineralizer flowrate to the total stream flowrate	NC	-	0.0	0.01	0.0
Ratio of the total amount of noble gases routed to gaseous radwaste from the purification system to the total amount routed from the primary coolant system (not including the boron recycle system)	Y	-	See table 11.1-1	0.01	0.0
Primary-to-secondary leak rate	-	lb/day	100	-	100

TABLE 11.1-7 (SHEET 1 OF 3)

PARAMETERS SPECIFIED BY REGULATORY GUIDE 1.112 APPENDIX B (INPUT PARAMETERS FOR THE GALE COMPUTER CODE)[3]

Description	Value
Thermal power level (MWt)	2785
Mass of primary coolant (lb)	4.2×10^5
Primary system letdown rate (gal/min)	60[a]
Letdown cation demineralizer flowrate (gal/min)	6.0[a]
Number of steam generators	3
Total steam flow (lb/h)	12.3×10^6
Mass of steam in each steam generator (lb)	6.5×10^3
Mass of liquid in each steam generator (lb)	1.0×10^5
Total mass of secondary coolant	1.52×10^6
Total blowdown rate (lb/h)	3.75×10^4
Condensate demineralizer regeneration time	0.0
Condensate demineralizer flow fraction	0.0
Maximum radwaste dilution flow (gal/min)	16.0×10^3
Shim Bleed	
Shim bleed flowrate (gal/day)	1.56×10^3
Decontamination factor for I	10^5
Decontamination factor for Cs and Rb	4×10^4
Decontamination factor for others	10^6
Collection time (day)	1.03
Process and discharge time (day)	0.148
Fraction discharged	1.0
Equipment Drains	
Equipment drains flowrate (gal/day)	250
Fraction of reactor coolant activity	0.005
Decontamination factor for I	10^3
Decontamination factor for Cs and Rb	10^4
Decontamination factor for others	10^4
Collection time (day)	1.03
Process and discharge time (day)	0.148
Fraction discharged	1.0
Clean Waste	
Clean waste input flowrate (gal/day)	1.64×10^2
Fraction of reactor coolant activity	0.005
Decontamination factor for I	10^3
Decontamination factor for Cs and Rb	10^4
Decontamination factor for others	10^4

TABLE 11.1-7 (SHEET 2 OF 3)

Description	Value
Collection time (day)	24.4
Process and discharge time (day)	0.37
Fraction discharged	1.0
Dirty Waste	
Dirty waste input flowrate (gal/day)	1.38×10^2
Fraction of reactor coolant activity	0.002
Decontamination factor for I	10
Decontamination factor for Cs and Rb	2.0
Decontamination factor for others	10
Collection time (day)	0
Process and discharge time (day)	15.0
Fraction discharged	1.0
Blowdown Waste	
Blowdown fraction processed	1.0
Decontamination factor for I	10^2
Decontamination factor for Cs and Rb	4.0×10^2
Decontamination factor for others	10^4
Collection time	0.0
Process and discharge time	0.0
Fraction discharged	1.0
Regenerant flowrate	
Decontamination factor for I	N/A
Decontamination factor for Cs and Rb	N/A
Decontamination factor for others	N/A
Collection time	N/A
Process and discharge time	N/A
Fraction discharged	N/A

TABLE 11.1-7 (SHEET 3 OF 3)

Description	Value
Gaseous Waste System	
Holdup time for xenon (day)	90
Holdup time for krypton (day)	90
Fill time of decay tanks for gas stripper	0.0
Gas waste system: HEPA?	Yes
Auxiliary building: charcoal?	Yes
Auxiliary building: HEPA?	Yes
Containment volume (ft^3)	2.15×10^6
Containment atmosphere cleanup rate (ft^3/min)	20×10^3
Containment shutdown purge: charcoal?, HEPA?	Yes, Yes
Number purge per year	8
Containment normal purge rate (ft^3/min); charcoal?, HEPA?	0; yes, yes
Fraction of iodine released from blowdown tank vent	0.05
Fraction of iodine released from main condenser air ejector	0.1
Detergent waste decontamination factor	0.1

a. Evaluation of the impact of increasing letdown flow to 145 gpm and cation demineralizer flow to 14.5 gpm indicate the releases shown in Tables 11.2-7, 11.2-8, 11.3-9, and 11.3-10 and the resultant doses shown in Tables 11.2-9 and 11.3-11 remain bounding.

11.2 LIQUID WASTE SYSTEMS

11.2.1 DESIGN OBJECTIVES

The liquid waste processing system (LWPS) is designed to receive, segregate, process, recycle, and discharge liquid wastes. The system design considers potential personnel exposure and ensures that quantities of radioactive releases to the environment are as low as reasonably achievable. Under normal plant operation, the total activity from radionuclides leaving the LWPS does not exceed a small fraction of the discharge limits defined in column 2, Table II, Appendix B to 10 CFR 20.

Further, overall radioactive release limits are established as a basis for controlling plant discharges during operation with the occurrence of a combination of equipment faults of moderate frequency. A combination of equipment faults that could occur with moderate frequency include operation with fuel cladding defects in combination with such occurrences as:

A. Steam generator tube leaks.

B. Malfunction in LWPS.

C. Excessive leakage in reactor coolant system equipment.

D. Excessive leakage in auxiliary system equipment.

The radioactive releases from the plant resulting from equipment faults of moderate frequency are within the column 2, Table II, Appendix B to 10 CFR 20 limits on a short-term basis and do not exceed four to eight times the limits stated previously for normal operation on an annual average basis.

11.2.2 SYSTEM DESCRIPTIONS

The LWPS collects and processes potentially radioactive wastes for recycle or for discharge. Provisions are made to sample and analyze fluids before they are recycled or discharged. Based on the laboratory analysis, these wastes are either released under controlled conditions via the cooling water system or retained for further processing. A permanent record of liquid releases is provided by analyses of known volumes of waste.

The radioactive liquids discharged from the reactor coolant system can be processed by the boron recycle system, the portable demineralizer system as described in paragraph 11.2.3.1.8, or the Liquid Radwaste Processing System (LRWPS) as described in paragraph 11.2.3.1.9. The limited amount of fuel leakage experienced in the plant operating history has enabled the use of the portable demineralizer system and/or the LRWPS to process the bulk of the reactor coolant system radioactive liquid discharges. The operation of either the demineralizer system or the LRWPS results in a smaller volume of waste to be shipped offsite for disposal. The permanently installed boron recycle system remains available for use to ensure that the technical specification limits are met. The use of the portable demineralizer system, the

LRWPS, or the boron recycle system limits input to the LWPS and results in processing of relatively small quantities of generally low activity wastes.

The LWPS is arranged to recycle reactor grade water if possible. This is implemented by the segregation of equipment drains and waste streams, which prevents the intermixing of liquid wastes. The LWPS consists of two main subsystems designated as drain channel A and drain channel B. Drain channel A processes water which can be recycled, and drain channel B processes water which is to be discharged. A drain system is also provided inside the containment to collect drains and leaks and transfer them to an appropriate tank. Capability for handling and storage of spent demineralizer resins is also provided.

The plant has a Liquid Radwaste Processing System (LRWPS) that uses a series of filters, demineralizers, reverse osmosis, and other methods to process water from any waste stream and produce an exceptionally low activity effluent that may be returned to the plant or discharged to the environment. This system is described in paragraph 11.2.3.1.9.

Additionally, the plant has been equipped with a portable demineralizer system described in paragraph 11.2.3.1.8. This system is capable of processing water from any of the waste streams and producing a very low activity effluent. Water processed through the disposable demineralizer system is routed to the waste monitor tank for analysis prior to release.

Instrumentation and controls necessary for the operation of the LWPS are located on a control board in the auxiliary building. Any alarm on this control board is relayed to the main control board in the control room. Additionally, alarms from the Liquid Radwaste Processing System (LRWPS) are local to the Radwaste Control Room (RCR).

Process flow diagrams and piping and instrumentation diagrams are shown in figure 11.2-1 and drawings D-175042, sheets 1, 2, 3, and 4, and D-205042, sheets 1, 2, 3, 4, and 7. All lines in the LWPS, including field run, are considered potential carriers of significant radioactivity.

Table 11.2-1 also gives process parameters for key locations in the system. Expected volumes to be processed by the LWPS are given in table 11.2-2. Assuming the volumes presented in table 11.2-2 are processed at a uniform rate, the input to the waste evaporator will be approximately 0.2 gal/min, while the evaporator is designed to handle 15 gal/min. Hence, excess capacity is available to handle abnormal operating conditions. This will only change the load on the system; otherwise the operating features will not change. Component failures in the LWPS are taken care of during system shutdown. The system is designed so that interchange of components is possible.

11.2.2.1 Recycle Portion (Drain Channel A - Tritiated and Aerated Water Sources)

Drain channel A is provided to process borated water which enters the LWPS via equipment leaks and drains, valve leakoffs, pump seal leakoffs, tank overflows, and other tritiated and aerated water sources.

Deaerated, tritiated water inside the reactor containment (from sources such as valve leakoffs), which is collected in the reactor coolant drain tank, need not enter drain channel A. These may be routed directly to the boron recycle holdup tanks for processing and/or reuse.

Administratively controlled equipment drains are the major contributor of water that may be recycled. Valve and pump leakoffs outside the reactor containment are also collected in the recycle holdup tank for processing and recycle. Abnormal liquid sources include leaks that may develop in the reactor coolant and auxiliary systems. Considerable surge and processing capacity is incorporated in the recycle portion of the LWPS to accommodate abnormal operations.

The basic composition of the liquid collected in the recycle holdup tank is boric acid and water with some radioactivity. Liquid collected in this tank is sampled and recycled or drained to the waste holdup tank for processing via the LWPS. Evaporator bottoms are normally processed at a low boron concentration to the waste holdup tank unless found acceptable for boric acid recycle. The condensate leaving the waste evaporator may pass through the waste condensate demineralizer and then enter the condensate tank. When a sufficient quantity of water has collected in the waste condensate tank, it is normally transferred to the reactor makeup water storage tank for reuse. Samples are taken at sufficiently frequent intervals to ensure proper operation of the system to minimize the need for reprocessing. If a sample indicates that further processing is required, the condensate may be passed through the waste condensate demineralizer or, if necessary, returned to the recycle holdup tank for additional evaporation.

The water collected in the recycle holdup tank may be routed to the portable demineralizer for processing rather than processing the water through the evaporator. Water processed through the demineralizer is not normally recycled.

The water collected in the recycle holdup tank may be routed to the disposable demineralizer or the Liquid Radwaste Processing System (LRWPS) for processing rather than processing the water through the evaporator. Water processed through the demineralizer is not normally recycled.

11.2.2.2 Waste Portion (Drain Channel B - Nonreactor Grade Water Sources)

Drain channel B is provided to collect and process nonreactor grade liquid wastes. These include floor drains, equipment drains containing nonreactor grade water, laundry and hot shower drains, and other nonreactor grade sources. Drain channel B equipment includes a floor drain tank and filter, laundry and hot shower tank and filter, chemical drain tank, waste monitor tank demineralizer and filter, Liquid Radwaste Processing System (LRWPS), disposable demineralizer system, and two waste monitor tanks.

Nonrecyclable reactor coolant leakage enters the waste holdup tank from system leaks inside the containment via the containment sump and enters the floor drain tank from system leaks in the auxiliary building via the floor drains. Unless an extremely large leak develops, this liquid would not be recycled because it is diluted and contaminated by water entering the floor drain tank from other sources, e.g., laboratory equipment rinses, hose water, component cooling leaks, etc. Nonreactor grade leakage enters the floor drain tank from the auxiliary building floor

drains. Sources of water to the drains are fan cooler leaks, secondary side steam and feedwater leaks, component cooling water, and hose water. This leakage is assumed not to contribute significantly to activity release. The activity level is normally much less than 10^{-7} μCi/cm^3.

Normally, the activity of the floor drain tank contents is well below permissible levels. Hence the contents may be transferred directly to the waste monitor tanks after sampling. Following analysis to confirm the acceptable low level, the tank contents are discharged without further treatment. However, should spills, leaks, or equipment failures cause radioactive water to enter the floor drain tank, this water is processed through the waste evaporator, the Liquid Radwaste Processing System (LRWPS), or disposable demineralizer.

In general, if the activity in the floor drain tank is greater than 10^{-5} μCi/cm^3, the liquids should be processed. If such a case should occur, the waste evaporator concentrate is drummed or directed to the waste holdup tank and the condensate returned to drain channel B for ultimate discharge, or the waste is processed via the disposable demineralizer system or the Liquid Radwaste Processing System (LRWPS), and the effluent returned to drain channel B for ultimate discharge.

Laundry and hot shower drains are the largest source of liquid wastes and normally need no treatment for removal of radioactivity. This water is transferred to one of the waste monitor tanks via the laundry and hot shower filter. The laundry and hot shower water may be processed through the disposable demineralizer or the Liquid Radwaste Processing System (LRWPS) if required. A sample is taken, the results logged after analysis, and the water discharged if the activity level is below acceptable limits.

The basic criterion for the laboratory drain subsystem is that strict segregation of radioactive and nonradioactive liquid wastes be maintained. Two separate drains are provided for this control. One is used to dispose of spent and excess radioactive samples directly to the chemical drain tank for later processing by the disposable demineralizer system or the Liquid Radwaste Processing System (LRWPS). The second drain is provided for normal laboratory equipment decontamination and rinsing. This liquid waste is directed to the floor drain tank. The sampling room contains two sinks. Excess sample purges of reactor grade coolant are drained from one sink to the waste holdup tank for recycle. The other sink is used for draining nonreactor grade excess samples to the floor drain tank.

Liquid wastes are released from the waste monitor tanks through a discharge valve interlocked with a process radiation monitor. The valve closes automatically and annunciates in the control room when the radioactivity concentration in the liquid discharge exceeds a preset limit. Liquid waste discharge flow volume is recorded.

11.2.2.3 Waste from Spent Resin

The spent resin sluice portion of the LWPS consists of a spent resin storage tank, a spent resin sluice pump, and a spent resin sluice filter. The equipment is arranged in such a way that the resin sluice water, after entering a demineralizer vessel, returns to the spent resin storage tank

for reuse. The basic criterion for the system is to transport spent resin to the spent resin storage tank without generating large volumes of waste liquid. This is accomplished by reusing the sluice water for subsequent resin sluicing operations. Spent resins from the disposable demineralizer system are disposed of with the demineralizer vessel. Spent resins from the Liquid Radwaste Processing System (LRWPS) can be sent to either the Solids Handling System (SHS) Spent Resin Tank or to the Solidification and Dewatering Facility (SDF).

11.2.3 SYSTEM DESIGN

11.2.3.1 Component Design

In accordance with its safety classification, the LWPS components are designed to meet the design requirements of the codes and standards listed in table 3.2-1. However, it should be noted that the components of the LWPS listed in table 3.2-1, as a minimum, are designed and manufactured to the requirements given in Regulatory Guide 1.143, Revision 1, with the exception of the seismic design criteria given in Regulatory Position C.5. The components, systems, and structures of the LWPS are designed to the seismic design criteria given in section 3.7. For further information on the safety and seismic classification of this system, see chapter 3.

The materials of construction along with the essential design parameters are given in table 11.2-3. All parts of components in contact with borated water are fabricated or clad withaustenitic stainless steel. In addition, all pumps are provided with vent and drain connections.

Paragraph 3.9.2.7 gives the general design criteria for field run piping.

11.2.3.1.1 Pumps

A. Reactor Coolant Drain Tank Pumps

Due to the relative inaccessibility of the containment and the loop drain requirement, two pumps are provided. One pump provides sufficient flow for normal tank operation, with one pump for standby.

B. Waste Evaporator Feed Pump

One standard pump is used. The waste evaporator feed pump supplies feed to the evaporator based on level control in the waste holdup tank.

C. Waste Evaporator Condensate Tank Pump

The waste evaporator condensate tank pump is a transfer pump. One standard pump is used to transfer the contents of the waste condensate tank to reactor makeup water storage tanks or the boron recycle system holdup tank.

D. Deleted.

E. Spent Resin Sluice Pump

This pump is similar, with regards to performance characteristics, to the reactor coolant drain tank pumps. Its delivery flow is based on the required velocity to sluice resin.

F. Laundry and Hot Shower Tank Pump (Unit 1 only)

One standard pump is used to transfer the water to the waste monitor tank.

G. Floor Drain Tank Pump

One standard pump is used to transfer water normally to the waste monitor tank. The pump can also be used to supply the waste evaporator or for pumping the waste back to the waste holdup tank.

H. Waste Monitor Tank Pumps

One standard pump is used for each tank to discharge water or to recycle water if further processing is required. The pump may also be used for circulating the water in the waste monitor tank in order to obtain uniform tank contents and hence a representative sample before discharge. The pump can be throttled to achieve the desired flowrate.

11.2.3.1.2 Reactor Coolant Drain Tank Heat Exchangers

The reactor coolant drain tank heat exchanger is a U-tube type with one shell pass and two tube passes. Although the heat exchanger is normally used in conjunction with the reactor coolant drain tank, it can also cool the pressurizer relief tank from 200°F to 120°F in < 8 h.

11.2.3.1.3 Tanks

A. Reactor Coolant Drain Tank

One tank is provided for each unit. The purpose of the reactor coolant drain tank is to collect leakoff type drains inside the containment at a central collection point for further disposition through a single containment penetration via the reactor coolant drain tank pumps. The tank provides surge and net positive suction head requirements to the pumps.

The water entering the reactor coolant drain tank may be of adequate purity to allow direct recycling to the boron recycle system holdup tank. If this water is not

compatible or if it contains dissolved air or nitrogen, it must be processed in the LWPS channel A.

Sources of water entering the reactor coolant drain tank include the reactor vessel flange leakoff, valve leakoffs, reactor coolant pump 2 and 3 seal leakoffs, and the excess letdown heat exchanger flow. No continuous leakage is expected from the reactor vessel flange during operation.

The system is designed to maintain a constant level in the tank to minimize the amount of gas sent to the waste gas processing system and also to minimize the amount of hydrogen required. One pump runs continuously. The level in the tank is maintained by a control valve in the discharge line. The valve operates on signals from a level controller connected to the tank and regulates flow fractions back to the tank and out of the system, respectively.
As an alternate mode of operation, the reactor coolant drain tank (RCDT) pumps may both be secured and manually actuated at the necessary intervals. Reactor coolant drain tank parameters are provided at the system control station. If the manual mode of operation is chosen, procedural requirements will ensure that the reactor coolant drain tank level is monitored at regular intervals and the pumps are actuated as needed.

With the waste gas system out of service, the RCDT can be vented via polytubing through the sample connection on the RCDT vent line to the waste gas decay tank sample station and into a waste gas decay tank.

B. Waste Holdup Tank

One atmospheric pressure tank is provided outside the containment to collect equipment drains, valve and pump seal leakoffs, recycle holdup tank overflows, and other water from tritiated, aerated sources.

C. Waste Evaporator Condensate Tank

One tank is provided to collect condensate from the waste evaporator.

D. Chemical Drain Tank

One tank is provided to collect chemically and radiologically contaminated water from the laboratories and decontamination room wastes.

E. Spent Resin Storage Tanks

The purpose of the spent resin storage tanks (one for primary spent resins and one for secondary spent resin) is to provide a collection point for spent resin to allow for decay of short lived radionuclides before drumming. The tank serves also as a head tank for the spent resin sluice pump. Vertical, cylindrical tanks are used because the symmetrical bottom facilitates the removal of resin. The tank is designed so that sufficient pressure can be applied in the gas space of the tank to

push resin out and to the drumming station or solidification and dewatering building, which may be at a higher elevation than the spent resin storage tank.

The spent resin storage tank and associated equipment that may contain radioactive material are shielded to limit the dose to personnel.

F. Laundry and Hot Shower Tank (Unit 1 only)

One atmospheric pressure tank is used to collect laundry and hot shower drains within the controlled areas.

G. Floor Drain Tank

One atmospheric pressure tank is used to collect floor drains from the controlled areas of the reactor plant.

H. Waste Monitor Tanks

The two atmospheric pressure waste monitor tanks are provided for monitoring liquid discharges from the LWPS. Each tank is sized to hold a volume large enough so that sampling requirements are minimized, thus minimizing further laboratory effluent.

I. Waste Evaporator Reagent Tank

One tank is used for adding chemicals to the plant for such things as cleaning of the waste evaporator tubes.

11.2.3.1.4 Demineralizers

As part of a continuous pressurized water reactor (PWR) operating plant following, Westinghouse has obtained operational data on demineralizer decontamination factors for selected isotopes. The measured range of decontamination factors for these isotopes is given in table 11.2-4.

These values were observed across mixed-bed demineralizers containing cation resin in the Li-7 form and anion resin in the borated form.

In considering the waste evaporator condensate demineralizer and the waste monitor tank demineralizer, it can be assumed that greater decontamination factors would be realized because the resin in both the demineralizers are in the hydrogen hydroxyl form. The minimum values in table 11.2-4 were generally observed just prior to resin flushing and recharging, while during the operating life of the demineralizer, decontamination factors were consistently closer to the maximum values.

Although specific operating decontamination factors have not as yet been measured for other isotopes, their behavior in a mixed-bed demineralizer may be inferred from this data. One

would anticipate, for example, tellurium and bromine to have decontamination factors similar to those given above for the iodine and fluorine.
The process decontamination factors used for design are given in table 11.2-4.

A. Waste Evaporator Condensate Demineralizer

One mixed-bed demineralizer in the hydrogen hydroxyl form is provided to remove ionic contaminants from the waste condensate which is intended to be recycled to the reactor coolant system.

B. Waste Monitor Tank Demineralizer

One mixed-bed demineralizer is provided to remove trace contaminants from the water if evaporation is not deemed necessary.

11.2.3.1.5 Filters

Filters are provided with easily disposable filter media.

Each filter has a capacity to pass 100 gal/min of water at pressure ratings compatible with filter design and not to exceed a pressure differential across the filter of 200 ft head.

The filters provided will be of material and construction that will meet all system parameters as listed in FSAR tables 9.1-2, 9.3-6, and 11.2-3.

The methods employed to change filters and screens are dependent on activity levels. Filters are valved out of service with a pressure indicator between the isolation valves to ensure that the valves are not leaking and that the filter is not at system pressure. The filter is drained to the appropriate tank and vented locally. If the radiation level of the filter is low enough, it is changed manually. If activity levels do not permit manual change, the spent cartridge is removed remotely with temporary shielding to protect personnel. The spent cartridge is placed in a shielded drum for removal to the solid waste disposal area. A new cartridge is installed, the housing is reassembled, vent and drain valves are closed, and the filter is valved into service. Filters are normally changed because of high ΔP rather than high radiation levels.

11.2.3.1.6 Strainers

Strainers are provided in two different types: basket and Johnson screen. The basket type is a mesh or screen construction, and the Johnson screen is a wound wire construction. The basket type strainers are not given an absolute rating because the tolerances in the size range of particles are not critical; it would not be feasible to put meaningful absolute rating on these strainers. The Johnson screen type strainer also does not have an absolute rating but a nominal rating with a plus/minus tolerance. Actually, the largest absolute particle that can pass through the screen is the rating plus the tolerance.

The basket type laundry and hot shower strainer is not replaced after use but is cleaned and put back in service. Because this screen traps only large particles, it contains only negligible activity and provides no hazard to personnel. It is cleaned as necessary.
The drumming header strainer is a Johnson screen type and is backflushable. The system in which it is installed provides an easy method for backflushing without removal of the strainer.

11.2.3.1.7 Waste Evaporator

One waste evaporator is used. The waste and the boron recycle evaporators are identical units and are interconnected so that they serve as a standby for each other under abnormal conditions.

11.2.3.1.8 Portable Demineralizer System(s)

Units 1 and 2 are served by a LWPS capable of processing the same liquid streams as the evaporator and the chemical drains. Therefore, when the portable demineralizer system(s) is in use, the waste evaporators are not needed.

Presently, the plant portable demineralizer system consists of the atmosphere demineralizer system (ADS).

The effluent from the disposable demineralizer system is routed to the waste monitor tank for sampling and analysis prior to release. The liquid released is subject to the same release restrictions as evaporator distillate, so the offsite dose is accounted for in the same manner for either system.

The expended media (resin, charcoal, filters) can be transferred to appropriate shipping containers and dewatered or solidified as necessary and shipped to a licensed burial ground for ultimate disposal.

The volume of waste to be shipped offsite for burial is significantly lower for the disposable demineralizer system than for the solidified evaporator concentrates, therefore, the system acts as an effective volume reduction device.

11.2.3.1.9 Liquid Radwaste Processing System (LRWPS)

The LRWPS can be used by either Unit 1 or 2 to process the same waste steams as the associated evaporator or the disposable demineralizer system. Therefore, when the LRWPS is in use neither the waste evaporator nor the disposable demineralizer system is needed.

The LRWPS processes incoming waste streams in several steps. Incoming fluid from the waste holdup tanks is passed through a carbon filter to removal of gross TOC (Total Organic Carbon), oil, and greases, which can lead to adverse fouling conditions in later processes. An ozone generator is then used to remove any remaining TOCs. The fluid then continues to the Tubular Ultra-Filtration (TUF) system where it passes through its first series of reverse osmosis equipment. Reject from the TUF system can be sent to the spent resin tank for later processing.

The process stream exiting the TUF system is then passed through a carbon bed and then a series of demineralizers via the Roughing Ion Exchange Module. This module removes the majority of the dissolved solids. The fluid exiting the roughing module then enters the second series of reverse osmosis equipment that removes any remaining dissolved solids.

Finally, the processes stream is passed through a final series of demineralizers called the Polishing Ion Exchange Module which removes any remaining impurities leaving the water with close to 99.99% removal of the initial waste stream constituents. This water is then sent to the Waste Monitor Tanks for reuse in the plant or discharge to the environment.

11.2.3.2 Instrumentation Design

The system instrumentation is described in table 11.2-5 and shown in the flow and piping diagrams in drawings D-175042, sheets 1, 2, 3, and 4, and D-205042, sheets 1, 2, 3, 4, and 7.

The instrumentation readout is located chiefly on the waste processing system panel in the auxiliary building. Some instruments are read where the equipment is located. Instrumentation related to the Liquid Radwaste Processing System (LRWPS) is displayed in the radwaste control room.

All alarms are shown separately on the waste processing system panel and further relayed to one common system annunciator on the main control board of the plant. Alarms related to the Liquid Radwaste Processing System (LRWPS) are sent to the radwaste control room.

All pumps are protected against loss of suction pressure by a control setpoint on the level instrumentation for the respective vessels feeding the pumps. The reactor coolant drain tank pumps are interlocked with flowrate instrumentation and stop operating when the delivery flows reach minimum setpoints.

Pressure indicators upstream and downstream of filters, strainers, and demineralizers provide local indications of pressure drops across each component.

All liquid releases from the LWPS are monitored for radioactivity by a scintillation counter. This instrumentation is further described in section 11.4.

11.2.4 OPERATING PROCEDURES

The LWPS is manually operated except for some functions of the reactor coolant drain tank circuit, Liquid Radwaste Processing System (LRWPS), and the disposable demineralizer system. The system includes adequate control equipment to protect the system components and adequate instrumentation and alarm functions to provide operator information to ensure proper system operation. All pumps in the system have low level shutoffs, and all filters and demineralizers have pressure indication upstream and downstream to indicate fouling.

11.2.4.1 Normal Operation

Operation of the LWPS is essentially the same during all phases of normal reactor plant operation; the only differences are in the load on the system. The following sections discuss the operation of the system in performing its various functions. In this discussion, the term "normal operation" should be taken to mean all phases of operation except operation under emergency or accident conditions. The LWPS is not regarded as an engineered safety feature system.

11.2.4.1.1 Recycle Portion

Water is accumulated in the waste holdup tank until sufficient quantity exists to warrant an evaporator startup, to switch the evaporator operation from the floor drain tank to the waste holdup tank, or to process through the disposable demineralizer system.

During evaporation the distillate is checked for boron and activity concentration, and if the analysis shows compatibility with reactor makeup grade water, it is transferred to the reactor makeup water storage tank. If the distillate is high in boron concentration or activity, it may be passed through the waste evaporator condensate demineralizer before being transferred to the reactor makeup water storage tank. If reevaporation is required and the waste evaporator is not available, then the distillate can be transferred to the boron recycle holdup tanks for processing by the boron recycle evaporator. The bottoms from the waste evaporator may be concentrated to approximately 12-percent boric acid but are normally concentrated to a low boric acid concentration and dumped to the waste holdup tank. Should the bottoms be acceptable for recycle, they are concentrated to approximately 4-percent boric acid and transferred to the boric acid tanks.

During normal operation, the reactor coolant drain tank level regulation and pressure control are automatic and require no operator action.

Operation of the recycle portion of the LWPS during refueling is the same as for power operation, although the load on the system may be increased when refueling is complete. The water remaining in the canal following normal drain down is pumped to the suction of the refueling water purification pump by the RCDT pumps.

11.2.4.1.2 Waste Portion

The waste portion of the LWPS consists of two subsystems: the laundry and hot shower system and the floor drain tank system.

Laundry and hot shower water enters the laundry and hot shower tank for holdup. The water is filtered and transferred to the waste monitor tank where it is sampled and discharged.

The water in the floor drain tank is sampled to determine the degree of processing required. It can be sent directly to the waste monitor tank provided for floor drain tank water or to the waste monitor tank via the waste monitor tank demineralizer, or it can be processed through the waste evaporator, the Liquid Radwaste Processing System (LRWPS), or the disposable demineralizer system. If the water is evaporated, the distillate is sent to the waste monitor tank and the

concentrate is recycled or solidified. The water in the waste monitor tank is again sampled and can be recirculated through the waste monitor tank demineralizer if further processing is required. When this water has been sufficiently processed, it is discharged into the plant discharge line at a rate so as not to exceed a small fraction of Technical Specification limits. A process decontamination factor in the range of 7.2×10^3 to 7.2×10^5 is expected for the evaporator demineralizer combination. Chemical trace element tests as well as operating experience on similar evaporators have justified this process decontamination factor.

Water leaving this system to the discharge canal is monitored for radiation. Should the radiation monitor close the discharge valve, it must be cleared before the valve can be reopened. The monitor element can be cleared by flushing it with demineralized water from the temporary connection back to the waste monitor tank. During refueling the load on the waste portion of the LWPS is increased, but there is no change in operation.

11.2.4.1.3 Laboratory Drain Portion

The laboratory drain portion consists of three sinks in the laboratory (one main sink and two sinks in the fume hoods), one sink in the gas analysis room fume hood, and one sink in the chemical drain tank and pump room. Spent and excess reactor coolant samples which cannot be recycled are disposed of via the chemical drain tank sink. When sufficient waste is collected in the tank, the waste is drummed or processed via the Liquid Radwaste Processing System (LRWPS) or the disposable demineralizer system. Equipment rinse water and other nonreactor grade water is disposed of via the floor drain tank sink.

11.2.4.1.4 Spent Resin Handling Portion

This portion of the system sluices resin from the demineralizers and transports resin from the spent resin storage tank to the drumming room or bulk shipping facility.

A. Resin Sluicing

Before resin sluicing begins, the demineralizer is valved out of service and the flow path is aligned from the resin sluice pump through the process line of the demineralizer, through the screen at the top of the demineralizer, and back to the spent resin storage tank. This process loosens the bed in preparation for sluicing. After about 10 min of backflushing, the pump is shut off and the valves in the backflush circuit are closed. The sluice line is opened, and the resin sluice pump is restarted. The resin then flows to the spent resin storage tank. After the resin sluice pump is shut off, the fresh resin is added via the resin fill line and the valve is then closed. The valves are then realigned for normal process operation. Resins are never sluiced through the spent resin sluice pump.

B. Resin Drumming

When sufficient resin is accumulated for processing, the valves in the line to the solidification and dewatering facility are opened and all other valves are closed.

The tank is then pressurized with N_2. The appropriate valves are then opened and the resin is forced into the disposal container. During drumming, N_2 is forced through the spargers in the tank bottom to level the resin and maintain the tank pressure. When the containers are full, the valves are closed and the tank pressure is relieved to the plant vent. The valves are then flushed using reactor makeup water.

C. Disposable Demineralizer

The disposable demineralizers and filters consist of shipping cask liners equipped with underdrains and filled with the appropriate resin or filter material. The disposable demineralizer system consists of an activated carbon filter and two mixed-bed demineralizers or three demineralizers connected in series. When the demineralizer or filter is depleted it is dewatered and/or solidified, placed in a shipping cask, and shipped off site for disposal. This allows expended units to be disposed of without rehandling the resins or filter media. This procedure results in reduced radiation exposure to operating personnel.

D. Liquid Radwaste Processing System (LRWPS)

The Liquid Radwaste Processing System consists of a separate series of demineralizers that are sluiced to the Solids Handling System (SHS) spent resin tank that is independent of the plant spent resin tank. The SHS spent resin tank is internal to the LRWPS. Sluiced resins from the LRWPS can also be sent to the Solidification and Dewatering Facility (SDF) for processing. Concentrates from the liquid radwaste system and resins from the SHS spent resin tank can then be transferred to the LRWPS concentrate dryer system where they will have their water content removed and then be transferred to a disposal liner for eventual offsite shipment

The level indicating system in the spent resin storage tank is a conventional system, the only difference being that there is a bellows to keep resin fines away from the instrument. The method of operation limits the resin to a maximum level and the water to a minimum level, the minimum water level being some distance above the resin level. The lower level tap is located in the area which contains water and no resin. This arrangement minimizes the possibility of plugging the level tap. Because the system indicates only total level and not the amount of resin and the amount of water, an inventory of spent resins in the tank is maintained. Since the resin volumes flushed from demineralizers are known and the resin volumes drummed are known, the resin level in the tank is also known.

11.2.4.2 <u>Faults of Moderate Frequency</u>

The system can handle the occurrence of equipment faults of moderate frequency such as:

A. Malfunction in the LWPS

Malfunction in this system could include such things as pump or valve failures or evaporator failure. Because of pump standardization throughout the system, a spare pump can be used to replace most pumps in the system. There is sufficient surge capacity in the system to accommodate waste until the failures can be fixed and normal plant operation resumed. Also, the Liquid Radwaste Processing System (LRWPS) and the disposable demineralizer system provides additional surge capacity for the LWPS.

B. Excessive Leakage in Reactor Coolant System Equipment

The system can handle a 1-gal/min reactor coolant leak in addition to the expected leakage during normal operation. Operation of the system is almost the same as for normal operation except that the load on the system is increased. A 1-gal/min leak into the RCDT is handled automatically but may increase the load factor of the recycle evaporator. If the 1-gal/min leak enters the waste holdup tank, operation is the same as normal except for the increased load on the evaporator or disposable demineralizer system. Abnormal liquid volumes of reactor coolant resulting from excessive reactor coolant or auxiliary building equipment leakage (1 gal/min) can also be accommodated by the floor drain tank and processed by the LWPS.

C. Excessive Leakage in Auxiliary System Equipment

Leakage of this type could include water from steam side leaks and fan cooler leaks inside the containment which are collected in the containment sump and sent to the waste holdup tank. Other sources could be component cooling water leaks, service water leaks, and secondary side leaks. This water would enter the waste holdup tank and would be processed and discharged as during normal operation.

D. Steam Generator Tube Leaks

During periods of operating with fuel defects coincident with steam generator tube leaks, radioactive liquid is processed via the steam generator blowdown system. This system is described in subsection 10.4.8.

11.2.4.3 <u>Operating Experience</u>

Different processing systems with evaporators have been tested for feed to distillate decontamination factors. A 2-gal/min evaporator was operated with the feed in the pH range of 5.2 to approximately 11.6. Gross beta and gamma activity as well as I-131 activity were measured in the feed and the distillate. The decontamination factors obtained were in the range of 5×10^3 to 5×10^4. The same evaporator was also tested with sodium; the decontamination

factors obtained for sodium were, in general, 10^5 or higher. A second evaporator was tested at different pH levels with sodium for gross beta and gamma activity. The test confirmed the results previously obtained. A Westinghouse-designed evaporator similar to the one to be used for Farley Nuclear Plant (FNP) has been shop tested measuring decontamination factor between bottom and distillate for sodium. The decontamination factors obtained were in the range of 10^5 to 10^6. Hence, a feed to distillate evaporator decontamination of 10^3 used for design is considered conservative.

For operational decontamination factors obtained on demineralizers, see paragraph 11.2.3.1.4.

11.2.5 PERFORMANCE TESTS

Initial performance tests are performed to verify the operability of the components, instrumentation and control equipment, and applicable alarms and control setpoints.

Operability testing of the LWPS is conducted periodically in accordance with the plant Technical Specifications to determine the reliability of components designed to reduce liquid activity levels. A decontamination factor for the waste evaporator is obtained by measuring the concentrations of I-131, Cs-137, and Co-60 before and after processing to monitor the efficiency of the LWPS. Demineralizer efficiency will be monitored in a similar manner. The waste evaporator output may be recycled in order to reduce contamination to the design levels. Either the Liquid Radwaste Processing System (LRWPS) or the disposable demineralizer system may be used to process the liquid waste to reduce the load on the LWPS.

The radiological analyses conducted to assess the performance of the LWPS and to determine discharge concentrations are discussed in detail in section 11.4.

11.2.6 ESTIMATED RELEASES

11.2.6.1 Nuclear Regulatory Commission Requirements

The following documents have been issued by the Nuclear Regulatory Commission to provide regulations and guidelines for release of radioactive liquids:

A. 10 CFR 20, Standards for Protection Against Radiation.

B. 10 CFR 50, Licensing of Production and Utilization Facilities.

C. Regulatory Guide 1.109, Revision 1, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR 50, Appendix I," October 1977.

D. Regulatory Guide 1.113, Revision 1, "Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I," April 1977.

During plant operations, radioactive liquid releases will be controlled in accordance with Technical Specifications. For nuclear power plants, the NRC acceptance criteria for compliance with the dose limits stated in 10 CFR 20.1301 for individual members of the public may be demonstrated by complying with the limits of 10 CFR 50, Appendix I, and 40 CFR 190. Therefore, it is acceptable that the limits associated with the release rate Technical Specifications are based on ten times the effluent concentration limits given in column 2, Table 2, Appendix B to 10 CFR 20.1001 - 20.2401, since operational history at Farley Nuclear Plant has demonstrated that the calculated maximum individual doses to members of the public are small percentages of the values given in 10 CFR 50, Appendix I, and 40 CFR 190.

11.2.6.2 Westinghouse PWR Experience Releases

The liquid releases are highly dependent upon administrative activities which control the use of water for decontamination, equipment and floor rinsing, and other uses in the controlled areas.

The plants operating at the time of Unit 1 licensing were reporting liquid discharges as shown in table 11.2-6 for years 1970 and 1971.

11.2.6.3 Expected Liquid Waste Processing System Releases

The equipment utilized during liquid waste processing is at the discretion of the operator; therefore, the calculated releases do not address all possible treatment processes but only the process which was the basis for the original plant design. Liquid releases from FNP were calculated using the PWR-GALE computer code[2] and parameters listed in table 11.1-7, which are discussed in more detail below. Releases calculated assuming operation with expected levels of fuel cladding defects of 0.12 percent are presented in table 11.2-7. Primary and secondary coolant activity levels are discussed in section 11.1 for the realistic case. In agreement with reference 2, the total releases include an adjustment factor of 0.15 Ci/year, using the same isotopic distribution as the calculated release, to account for anticipated operational occurrences.

The tables list the calculated annual release from each of the process paths discussed below as well as the total annual release. A comparison of annual average effluent concentrations with values stated in column 2, Table II, Appendix B to 10 CFR 20 is provided in table 11.2-8 for operation with expected fuel leakage.

The releases are calculated for one unit, assuming that both units are operating. This is done to reflect the impact of the second unit's operation on the operation of systems and components shared between the two units. To obtain the combined releases of the two units, simply double the values listed in table 11.2-7.

A survey has been performed of liquid discharges from different Westinghouse pressurized water reactor plants, with results presented in table 11.2-6 for years 1970 and 1971. The data include radionuclides released on an unidentified basis and are all within the permissible concentration for release of liquid containing an unidentified radionuclide mixture. The data in

table 11.2-6 clearly indicate that actual releases are highly dependent upon the actual operation of the plant and can vary significantly from year to year for a given plant as well as from plant to plant.

The LWPS is assumed to operate as described in subsection 11.2.4.

11.2.6.4 Steam Generator Blowdown System

The secondary side activity used in the offsite release analysis is given in table 11.2-7.

The blowdown from the secondary side is normally released to the environment; however, the liquid may be recycled to the main condenser if required.

The estimated activity released per unit to the environment from such discharges is given in table 11.2-7. The system is further described in subsection 10.4.8.

11.2.6.5 Turbine Building Drains

The concentration of isotopes in steam or liquid leaked to the turbine building is considered a factor of 100 lower than secondary side concentration in table 11.2-8 for all isotopes except tritium. Tritium concentration in leakage is assumed to be the same as in the secondary side. The factor of 100 accounts for limited carryover in steam. Steam leakage of 5 gal/min (condensed) and liquid leakage of 12 gal/min is assumed to be discharged through turbine building drains. Discharge rates for each isotope are given in table 11.2-7.

11.2.6.6 Estimated Total Releases

The potential releases from each source have been evaluated as indicated in above sections. As shown in table 11.2-8, the total expected liquid release from one unit of the plant is a small fraction of the regulations as outlined in paragraph 11.2.6.1. It is further shown that the expected liquid releases from FNP are well below releases in presently-operating plants, as shown in table 11.2-6 for years 1970 and 1971. Hence, the releases from the plant are in accordance with the design objectives as outlined in subsection 11.2.1 and the plant Technical Specifications.

11.2.7 RELEASE POINTS

The LWPS is designed to minimize the total radioactive fluid released to the environment by processing and recycling as much water as possible. This design allows only one release point, as shown in drawings D-175042, sheet 4 and D-205042, sheet 4. Drawing D-170180, sheet 1 shows the physical location of this release point.

11.2.8 DILUTION FACTORS[1]

[Historical]

[*The volume of the mixing zone will be small and will contain concentrations ranging from full dilution to concentrations approaching those in the discharge pipe. The discharge pipe concentrations, less than 161 times higher than full dilution concentrations, would be confined to an extremely small volume.*

For doses received near the plant, the annual average flowrate of the Chattahoochee River (11,500 ft^3/s) is used. For performance of the 10 CFR 50 Appendix I analysis,[1] liquid effluent isotopes from the FNP were assumed to be diluted by a flow of approximately 16,000 gal/min (71.4 ft^3/s for 2 units) from each unit prior to discharge. Dilution flow of less than 16,000 gal/min may occur during certain operational conditions such as plant outages. Compliance with the FNP ODCM and Technical Specifications ensures that release concentrations are within acceptable limits.

The mixing ratio (inverse of the dilution factor) was taken as 0.2 for fish ingestion and recreational pathways in accordance with recommendations in Regulatory Guide 1.109, Table A-1.[3] The resultant concentrations roughly approximate those at the edge of the initial mixing zone.

The mixing ratio for other pathways was taken as 3.1×10^{-3}, which is equivalent to dilution of the 16,000 gal/min effluent stream in the full flow of the river (11,500 ft^3/s).[4]

Full dilution is warranted because the nearest downstream water usage for the other pathways is at least 20 miles downstream.

*Mixing near the discharge structure and discharge characteristics are discussed further in subsection 2.4.12 and OLSER subsection 3.4.3.****]***

The historical information above was utilized in the dose calculations to confirm FNP compliance with the requirements of 10 CFR 50, Appendix I prior to plant operation (see reference 1).

11.2.9 ESTIMATED DOSES[1]

[HISTORICAL]

[*Dose models and values for usage rates, holdup times, and other parameters used to estimate maximum doses to individuals from discharges to the hydrosphere are those described in Regulatory Guide 1.109.[3]*

Pathways evaluated include fish ingestion, shoreline recreation, boating, and swimming. Freshwater invertebrates are not normally consumed by humans. At present, there are no public

water supply intakes closer than 50 miles downstream of the plant, and there are no known plans to construct any. Separate evaluation of a hypothetical drinking water pathway, based on full river flow dilution of plant effluents, indicates that no organ dose or total body dose would exceed 0.03 mrem/year, if the pathway existed.

River water is occasionally used for irrigation at several points 20 miles or more downstream from the plant. It is extremely unlikely that any of the product reaches the individual with the maximum doses from plant liquid effluents, so the pathway was not included in the estimation of maximum doses to individuals. A separate analysis of the ingestion pathway, assuming full river flow dilution of plant liquid effluents and 5-inches-per-month irrigation (comparable to normal summer rainfall), indicates that no organ or total body dose in any age group would exceed 0.006 mrem/year from consumption of irrigated vegetables (fresh or stored), milk, or meat.

The historical dose estimates above were calculated to confirm that the Farley Nuclear Plant conforms with the requirements of 10 CFR 50, Appendix I prior to plant operation (see reference 1).]

The estimated doses for the appropriate pathway are outlined in table 11.2-9 along with the Appendix I design objective doses for comparison. It is clear that the estimated doses follow the design objective doses in each case. Actual plant releases during normal operation are governed by the Farley Nuclear Plant Technical Specifications and Offsite Dose Calculations Manual.

REFERENCES

1. Alabama Power Company Letter to the Nuclear Regulatory Commission, "Dose Calculations to Conform with Appendix I Requirements," USNRC Docket Nos. 50-348, 50-364, June 3, 1976.

2. U.S. Nuclear Regulatory Commission, "Calculation of Releases from Pressurized Water Reactors," PWR-GALE Computer Code, NUREG-0017, April 1976.

3. Regulatory Guide 1.109, Revision 1, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR 50, Appendix I," October 1977.

4. Regulatory Guide 1.113, Revision 1, "Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I," April 1977.

TABLE 11.2-1

PROCESS FLOW DIAGRAM - LWPS – TABULATED ACTIVITIES

LINE NUMBER	LOCATION OF LINE		TEMP (°F)	PRESS. (PSIG)	FLOW RATE (GPM)	ISOTOPIC CONCENTRATION (μCi/GRAM)
1	CVCS RCP SEAL LEAKOFF INPUT TO RC DRAIN TANK		130	5	0.156	[illegible]
2	RC DRAIN TANK LOOP AFTER HX - INPUT TO BRS RH TANK		170	120	0.156	[illegible]
3	DRAIN HEADER INPUT TO WASTE HOLDUP TANK		AMB	ATM	0.138	[illegible]
4	SAMPLE ROOM SINK - INPUT TO WASTE HOLDUP TANK		AMB	ATM	0.006	[illegible]
5	SUM OF INPUTS 3 AND 4		AMB	ATM	[illegible]	[illegible]
6	AFTER RH TANK - BEFORE WASTE EVAP. PACKAGE		AMB	ATM	15	[illegible]
7	EVAPORATOR CONCENTRATES	A	150	50	0.006	[illegible]
		B	150	50	[illegible]	[illegible]
8	AFTER WASTE EVAPORATOR PACKAGE	A	120	45	15	[illegible]
		B	20	45	15	[illegible]
9	AFTER WASTE EVAPORATOR CONDENSATE DEMINERALIZER	A	20	VAR	15	[illegible]
		B	120	VAR	15	[illegible]
10	TREATED WATER AFTER WASTE EVAP. CONDENSATE TANK		120	VAR	0.138	[illegible]
11	INPUT FROM FLOOR HEADER TO FLOOR DRAIN TANK		AMB	ATM	0.032	[illegible]
12	INPUT FROM AUXILIARY BLDG. SUMPS TO FLOOR DRAIN TANK		AMB	ATM	0.032	[illegible]
13	SUM OF INPUTS 11 AND 12		AMB	ATM	0.064	[illegible]
14	INPUT FROM LAB RINSE WATER TO FLOOR DRAIN TANK		AMB	ATM	0.032	[illegible]
15	OUTPUT FROM FLOOR DRAIN TANK		AMB	VAR	15	[illegible]
16	OUTPUT FROM WASTE MONITOR TANK 1		AMB	VAR	0.094	[illegible]
17	INPUT FROM SPENT SAMPLES TO CHEMICAL DRAIN TANK		AMB	ATM	0.002	[illegible]
18	OUTPUT FROM CHEMICAL DRAIN TANK		AMB	VAR	0.002	[illegible]
19	LAUNDRY AND HOT SHOWER DRAINS INPUT TO LAUNDRY AND HOT SHOWER TANK		160	ATM	0.229	[illegible]
20	OUTPUT FROM WASTE MONITOR TANK 2		AMB	VAR	0.229	[illegible]
21	SUM OF INPUTS 18 AND 20		AMB	ATM	0.323	[illegible]

VENT NUMBER	VENT DESCRIPTION	TEMP (°F)	PRESS. (PSIG)	FLOW RATE (CC/MIN)	Kr-85m	Kr-85	Kr-87	Kr-88	Xe-133m	Xe-133	Xe-135	I-131	I-133
1	REACTOR COOLANT DRAIN TANK TO GWPS	130	5	6.6	3.3(-2)	2.5(+2)	5.7(-1)	3.5(-1)	5.8(0)	[illegible]	[illegible]	4.0(-4)	[illegible]
2	WASTE HOLDUP TANK VENT TO PLANT VENT	AMB	ATM	431	NEG	NEG	NEG	NEG	NEG	NEG	NEG	[illegible]	9.0(-4)
3	WASTE EVAPORATOR PACKAGE VENT TO PLANT VENT	150	2	6.0	NEG	NEG	NEG	NEG	NEG	NEG	NEG	NEG	NEG
4	FLOOR DRAIN TANK VENT TO PLANT VENT	AMB	ATM	383	NEG	NEG	NEG	NEG	NEG	NEG	NEG	[illegible]	[illegible]
5	WASTE MONITOR TANK 1 LOCAL VENT	AMB	ATM	754	NEG	NEG	NEG	NEG	NEG	NEG	NEG	[illegible]	[illegible]
6	CHEMICAL DRAIN TANK VENT TO PLANT VENT	AMB	ATM	[illegible]	NEG	NEG	NEG	NEG	NEG	NEG	NEG	[illegible]	[illegible]
7	LAUNDRY AND HOT SHOWER TANK LOCAL VENT	VAR	ATM	466	NEG	NEG	NEG	NEG	NEG	NEG	NEG	NEG	NEG
8	WASTE MONITOR TANK 2 LOCAL VENT	AMB	ATM	466	NEG	NEG	NEG	NEG	NEG	NEG	NEG	NEG	NEG

NEG - NEGLIGIBLE
AMB - AMBIENT
VAR - VARIABLE
N.D. - NO DATA AVAILABLE

TANK NUMBER	TANK IDENTIFICATION	TEMP. (°F)	PRESS. (PSIG)	TANK VOLUME (GAL.)	TANK INVENTORIES (CURIES)
1	REACTOR COOLANT DRAIN TANK	130	5	350	[illegible]
2	WASTE HOLDUP TANK	AMB	ATM	10000	[illegible]
3	WASTE EVAPORATOR CONDENSATE TANK	120	ATM	5000	[illegible]
4	FLOOR DRAIN TANK	AMB	ATM	10000	[illegible]
5	WASTE MONITOR TANK 1	AMB	ATM	5000	[illegible]
6	CHEMICAL DRAIN TANK	AMB	ATM	600	[illegible]
7	LAUNDRY AND HOT SHOWER TANK	VAR	ATM	15000	[illegible]
8	WASTE MONITOR TANK 2	AMB	ATM	5000	[illegible]
9	SPENT RESIN STORAGE TANK	AMB	VAR	2918	[illegible]

TABLE 11.2-2

PARAMETERS USED IN THE CALCULATION OF ESTIMATED ACTIVITY IN LIQUID WASTES

Collector Tank with Sources [(a)]	Volume of Liquid Wastes	Basis	Collection (Period Assumed Before Processing)	Comments
Reactor coolant drain tank	225 gal/day	0.05 gal/min/ reactor coolant pump 2 seal leak; 0.002 gal/min/reactor coolant pump 3 seal leak	Feed and bleed	Recycled to BRS
Waste holdup tank				
Equipment drains	57,000 gal/year	Filter drains, heat exchanger drains, tank drains, demineralizer drains		
Excess samples	3000 gal/year	3000 samples/year at 1 gal/sample		
Total	60,000 gal/year		20 days	Recycled to RMW
Floor drain tank				
Decontamination water	15,000 gal/year	40,000 ft^2 section once per week with 20 gal of water per 5000 ft^2 and remainder for fuel cask, vessel head, etc.		
Laboratory equipment	16,000 gal/year	60 gal/day for 5 day/week		
Nonrecyclable	7000 gal/year	20 gal/day		
Nonreactor grade leaks	13,000 gal/year	40 gal/day		
Total	51,000 gal/year		30 days	Discharged
Chemical drain tank	1000 gal/year	3000 samples/year at 1/8 gal/sample plus rinse water	90 days	Sent to the waste holdup tank
Laundry and hot shower tank	120,000 gal/year	300 gal/day with remainder for abnormal and refueling operations	7 days	Discharged

(a) These sources may be processed via disposable demineralizer system.

TABLE 11.2-3 (SHEET 1 OF 10)

EQUIPMENT PRINCIPAL DESIGN PARAMETERS

Component	Value
Pumps	
Reactor coolant drain tank pumps	
Number (per unit)	2
Type	Horizontal, centrifugal
Design pressure (psig)	150
Design temperature (°F)	200
Design flow (gal/min)	
Design point 1	100
Design point 2	140 (150 for N2G21P001B-N)
Design head (ft)	
Design point 1	300
Design point 2	250
Material	Stainless steel
Waste evaporator feed pump	
Number (per unit)	1
Type	Horizontal, centrifugal
Design pressure (psig)	150
Design temperature (°F)	200
Design flow (gal/min)	
Design point 1	35
Design point 2	100
Design head (ft)	
Design point 1	250
Design point 2	200
Material	Stainless steel
Waste evaporator condensate tank pump	
Number (per unit)	1
Type	Horizontal, centrifugal
Design pressure (psig)	150
Design temperature (°F)	200
Design flow (gal/min)	
Design point 1	35
Design point 2	100
Design head (ft)	
Design point 1	250
Design point 2	200
Material	Stainless steel

TABLE 11.2-3 (SHEET 2 OF 10)

Component	Value
Spent resin sluice pump	
Number (per unit)	2
Type	Horizontal, centrifugal
Design pressure (psig)	150
Design temperature (°F)	200
Design flow (gal/min)	
Design point 1	100
Design point 2	140
Design head (ft)	
Design point 1	300
Design point 2	250
Material	Stainless steel
Laundry and hot shower tank pump	
Number (per unit)	1 (Unit 1)
Type	Horizontal, centrifugal
Design pressure (psig)	150
Design temperature (°F)	200
Design flow (gal/min)	
Design point 1	35
Design point 2	100
Design head (ft)	
Design point 1	250
Design point 2	200
Material	Stainless steel
Floor drain tank pump	
Number (per unit)	1
Type	Horizontal, centrifugal
Design pressure, (psig)	150
Design temperature (°F)	200
Design flow (gal/min)	
Design point 1	35
Design point 2	100
Design head (ft)	
Design point 1	250
Design point 2	200
Material	Stainless steel

TABLE 11.2-3 (SHEET 3 OF 10)

Component	Value
Waste monitor tank pumps	
Number (per unit)	2
Type	Horizontal, centrifugal
Design pressure (psig)	150
Design temperature (°F)	200
Design flow (gal/min)	
Design point 1	35
Design point 2	100
Design head (ft)	
Design point 1	250
Design point 2	200
Material	Stainless steel
<u>Heat Exchangers</u>	
Reactor coolant drain tank heat exchanger	
Number (per unit)	1
Type	U-tube
Est. UA (Btu/h)(°F)	70,000
Design pressure (psi)	
Shell	150
Tube	150
Design temperature (°F)	
Shell	250
Tube	250
Design flow (lb/h)	
Shell	112,000
Tube	44,600
Temperature in (°F)	
Shell	105
Tube	180
Temperature out (°F)	
Shell	125
Tube	130
Material	
Shell	Carbon steel
Tube	Stainless steel

TABLE 11.2-3 (SHEET 4 OF 10)

Component	Value
Tanks	
Reactor coolant drain tank	
Number (per unit)	1
Usable volume (gal)	350
Type	Horizontal
Design pressure (psig)[(a)]	100
Design temperature (°F)	250
Material	Stainless steel
Diaphragm	No
Waste holdup tank	
Number (per unit)	1
Usable volume (gal)	10,000
Type	Vertical
Design pressure (psig)	Atmospheric
Design temperature (°F)	200
Material	Stainless steel
Diaphragm	No
Waste evaporator condensate tank	
Number (per unit)	1
Usable volume (gal)	5000
Type	Vertical
Design pressure (psig)	Atmospheric
Design temperature (°F)	200
Material	Stainless steel
Diaphragm	Optional
Chemical drain tank	
Number (per unit)	1
Usable volume (gal)	600
Type	Vertical
Design pressure (psig)	Atmospheric
Design temperature (°F)	200
Material	Stainless steel
Diaphragm	No

TABLE 11.2-3 (SHEET 5 OF 10)

Component	Value
Spent resin storage tank	
Number (per unit)	2
Usable volume (ft^3)[(b)]	350
Type	Vertical
Design pressure (psig)	100
Design temperature (°F)	200
Radiation level inside compartment (R/h)	1000
Material	Stainless steel
Diaphragm	No
Laundry and hot shower tank	
Number	1 (Unit 1)
Usable volume (gal)	10,000
Type	Vertical
Design pressure (psig)	Atmospheric
Design temperature (°F)	200
Material	Stainless steel
Diaphragm	No
Floor drain tank	
Number (per unit)	1
Usable volume (gal)	10,000
Type	Vertical
Design pressure (psig)	Atmospheric
Design temperature (°F)	200
Material	Stainless steel
Diaphragm	No
Waste monitor tank	
Number (per unit)	2
Usable volume (gal)	5000
Type	Vertical
Design pressure (psig)	Atmospheric
Design temperature (°F)	200
Material	Stainless steel
Diaphragm	No

TABLE 11.2-3 (SHEET 6 OF 10)

Component	Value
Waste evaporator reagent tank	
Number (per unit)	1
Usable volume (gal)	5
Type	Vertical
Design pressure (psig)	150
Design temperature (°F)	200
Material	Stainless steel
Diaphragm	No
Demineralizers	
Waste evaporator condensate demineralizer	
Number (per unit)	1
Type	Flushable
Design pressure (psig)	150
Design temperature (°F)	200
Design flow (gal/min)	35
Resin volume (ft^3)	30
Material	Stainless steel
Resin type	IRN-150[(C)]
Design process decontamination factor	100
Waste monitor tank demineralizer	
Number (per unit)	1
Type	Flushable
Design pressure (psig)	150
Design temperature (°F)	200
Design flow (gal/min)	35
Resin volume (ft^3)	30
Material	Stainless steel
Resin type	IRN-150[(C)]
Design process decontamination factor	100
Disposable demineralizer system	
Number (shared)	1
Design flow (gal/min)	32
Design pressure (psig)	
Piping	250
Feed concentrations (ppm boron)	10-2500
Effluent concentrations (ppm boron)	10-2500
Average process decontamination factor for all isotopes	100

TABLE 11.2-3 (SHEET 7 OF 10)

Component	Value
Filters	
Waste evaporator feed filter	
Number (per unit)	1
Design pressure (psig)	150
Design temperature (°F)	200
Design flow (gal/min)	35
ΔP at design flow (psi)	
Fouled	20
Unfouled	5
Size of particles 98 percent retained (μm, nominal)	25
Radiation level (R/h)	100
Materials	
Housing	Stainless steel
Filter element	Polypropylene
Waste evaporator condensate filter	
Number (per unit)	1
Design pressure (psig)	150
Design temperature (°F)	200
Design flow (gal/min)	35
ΔP at design flow (psi)	20
Size of particles, 98 percent retained (μm, nominal)	25
Radiation level (R/h)	<1
Materials	
Housing	Stainless steel
Filter Element	Polypropylene
Spent resin sluice filter	
Number (per unit)	2
Design pressure (psig)	150
Design temperature (°F)	200
Design flow (gal/min)	150
ΔP at design flow (psi)	
Fouled	20
Unfouled	5
Size of particles, 98 percent retained (μm, nominal)	25[d]/6[e]
Radiation level (R/h)	100
Materials	
Housing	Stainless steel
Filter element	Polypropylene

TABLE 11.2-3 (SHEET 8 OF 10)

Component	Value
Laundry and hot shower tank filter	
Number	1 (Unit 1)
Design pressure (psig)	150
Design temperature (°F)	200
Design flow (gal/min)	35
ΔP at design flow (psi)	
Fouled	20
Unfouled	5
Size of particles, 98 percent retained (μm, nominal)	25
Radiation level (mR/h)	<100
Materials	
Housing	Stainless steel
Filter element	Polypropylene
Floor drain tank filter	
Number (per unit)	1
Design pressure (psig)	150
Design temperature (°F)	200
Design flow (gal/min)	35
ΔP at design flow (psi)	
Fouled	20
Unfouled	5
Size of particles, 98 percent retained (μm, nominal)	25
Radiation level (R/hr)	100
Materials	
Housing	Stainless steel
Filter element	Polypropylene
Waste monitor tank filter	
Number (per unit)	1
Design pressure (psig)	150
Design temperature (°F)	200
Design flow (psi)	
Fouled	20
Unfouled	5
Size of particles, 98 percent retained (μm, nominal)	25
Radiation level (R/h)	90
Materials	
Housing	Stainless steel
Filter element	Polypropylene

TABLE 11.2-3 (SHEET 9 OF 10)

Component	Value
<u>Strainers</u>	
Laundry and hot shower tank strainer	
Number	1 (Unit 1)
Type	Basket
Design pressure (psig)	150
Design temperature (°F)	200
Design flow (gal/min)	35
ΔP at design flow (psi)	Negative
Mesh number	40
Nominal rating (in.)	1/16
Radiation level	Negative
Materials	Stainless steel
Floor drain tank strainer	
Number (per unit)	1
Type	Basket
Design pressure (psig)	150
Design temperature (°F)	200
Design flow (gal/min)	35
ΔP at design flow (psi)	Negative
Mesh number	40
Nominal rating (in.)	1/16
Radiation level	Negative
Materials	Stainless steel
Drumming header strainer	
Number	1 (Unit 1)
Type	Johnson
Design pressure (psig)	150
Design temperature (°F)	200
Design flow (gal/min)	40
ΔP at design flow (psi)	Negative
Mesh number	100
Nominal rating (in.)	0.003 ± 0.001
Radiation level	Negative
Materials	Stainless steel

TABLE 11.2-3 (SHEET 10 OF 10)

Component	Value
<u>Evaporators</u>	
Waste evaporator	
Number (per unit)	1
Design flow (gal/min)	15
Steam design pressure (psig)	50
Feed concentrations (ppm boron)	10-2500
Bottoms concentrations (ppm boron)	7000-21,000
Design process decontamination factor	1000

a. External design pressure - 60 psig.
b. Total for resin and liquid.
c. Rohm and Haas Amberlite or equivalent.
d. Applies to Cuno filters.
e. Applies to ultipor GF Plus filter.

TABLE 11.2-4

RANGE OF MEASURED DECONTAMINATION FACTORS FOR SELECTED ISOTOPES[a]

Isotope	Minimum	Maximum
F-18	1.73×10^{1}	1.5×10^{3}
Mn-54	$>2.5 \times 10^{1}$	$>1.3 \times 10^{2}$
Co-58	3.2×10^{1}	8.2×10^{3}
I-131	1.1×10^{1}	1.6×10^{4}
I-133	1.1×10^{1}	1.8×10^{4}
I-135	1.4×10^{1}	2.0×10^{4}
Cs-137	2.4	1.3×10^{3}

a. These values were observed across mixed bed demineralizers containing cation resin in the Li-7 form and anion resin in the borated form.

TABLE 11.2-5 (SHEET 1 OF 5)

LIQUID WASTE PROCESSING SYSTEM INSTRUMENTATION DESIGN PARAMETERS

Channel Number[a]	Location of Primary Sensor	Design Pressure (psig)	Design Temperature (°F)	Range	Alarm Setpoint	Control Setpoint	Location of Readout
Flow Instrumentation							
FI-1007	Waste evaporator feed pump discharge flow	150	200	0-30 gal/min			Local
FIC-1008	Reactor coolant drain tank pump discharge flow	150	250	0-250 gal/min		Low, 85 gal/min	WPS panel
FIA-1009	Reactor coolant drain tank recirculation flow	150	250	0-250 gal/min	Low, 85 gal/min		WPS panel
FICA-1011	Spent resin sluice pump discharge flow	150	200	0-150 gal/min	Low, 50 gal/min	Low, 50 gal/min	WPS panel
FI-1085A	Waste monitor tank pump 1 discharge flow	150	200	0-100 gal/min			WPS panel
FI-1085B	Waste monitor tank pump 2 discharge flow	150	200	0-100 gal/min			WPS panel
Pressure Instrumentation							
PIA-1004	Reactor coolant drain tank	150	250	0-100 psig	High, 8 psig		WPS panel
PIA-1006	Spent resin storage tank	150	200	0-100 psig	High, 90 psig		WPS and drumming panel
PI-1016	Waste evaporator feed pump discharge pressure	150	200	0-150 psig			Local
PI-1017	Waste evaporator feed header pressure	150	200	0-150 psig			Local

TABLE 11.2-5 (SHEET 2 OF 5)

Channel Number[a]	Location of Primary Sensor	Design Pressure (psig)	Design Temperature (°F)	Range	Alarm Setpoint	Control Setpoint	Location of Readout
PI-1018A	Reactor coolant drain tank pump 1 discharge pressure	150	250	0-150 psig			Local
PI-1018B	Reactor coolant drain tank pump discharge pressure	150	250	0-150 psig			Local
PI-1018C	Laundry and hot shower tank pump discharge pressure	150	200	0-150 psig			Local
PI-1018G	Waste evaporator condensate pump discharge pressure	150	200	0-150 psig			Local
PI-1074	Waste evaporator outlet pressure	150	200	0-150 psig			Local
PI-1075	Waste evaporator condensate demineralizer outlet pressure	150	200	0-150 psig			Local
PI-1076	Waste evaporator condensate filter outlet pressure	150	200	0-150 psig			Local
PI-1078	Floor drain tank filter inlet pressure	150	200	0-150 psig			Local
PI-1079	Floor drain tank filter outlet pressure	150	200	0-150 psig			Local
PI-1080	Laundry and hot shower tank filter inlet pressure	150	200	0-150 psig			Local
PI-1081	Laundry and hot shower tank filter outlet pressure	150	200	0-150 psig			Local
PI-1084A	Waste monitor tank pump 1 discharge pressure	150	200	0-150 psig			Local

TABLE 11.2-5 (SHEET 3 OF 5)

Channel Number[(a)]	Location of Primary Sensor	Design Pressure (psig)	Design Temperature (°F)	Range	Alarm Setpoint	Control Setpoint	Location of Readout
PI-1084B	Waste monitor tank pump 2 discharge pressure	150	200	0-150 psig			Local
PI-1086	Resin sluice filter inlet pressure	150	200	0-150 psig			Local
PI-1087	Resin sluice filter outlet pressure	150	200	0-150 psig			Local
PI-1088	Waste monitor tank filter inlet pressure	150	200	0-150 psig			Local
PI-1089	Waste monitor tank filter outlet pressure	150	200	0-150 psig			Local
PI-1090	Floor drain tank pump discharge pressure	150	200	0-150 psig			Local
<u>Level Instrumentation</u>							
LICA-1001	Waste holdup tank	150	200	0-100%	High-high, 90% High, 30% Low, 10%	Low, 15%	Local and WPS panel
LICA-1002	Chemical drain tank	150	200	0-100%	High, 90%	Low, 10% Low, 5%	Local, WPS, and drumming panels
LICA-1003	Reactor coolant drain tank	150	250	0-100%	High, 75% Low, 5%	Low, 30%	WPS panel
LICA-1005	Spent resin storage	150	200	0-100%	High, 60%	Low, 35% Low, 30%	WPS and drumming panels
LICA-1010	Laundry and hot shower tank	150	200	0-100%	High, 90% Low, 10%	Low, 15%	Local and WPS panels
LICA-1012	Waste evaporator condensate	150	250	0-100%	High, 68%	Low, 15%	Local and WPS

TABLE 11.2-5 (SHEET 4 OF 5)

Channel Number[a]	Location of Primary Sensor	Design Pressure (psig)	Design Temperature (°F)	Range	Alarm Setpoint	Control Setpoint	Location of Readout
	tank				Low, 10%		panels
LICA-1077	Floor drain tank	150	200	0-100%	High, 55% Low, 10%	Low, 15%	Local and WPS panels
LICA-1082	Waste monitor tank 1	150	200	0-100%	High, 90% Low, 10%	Low, 10%	Local and WPS panels
LICA-1083	Waste monitor tank 2	150	200	0-100%	High, 90% Low, 10%	Low, 10%	Local and WPS panels
<u>Temperature Instrumentation</u>							
TIA-1058	Reactor coolant drain tank	150	250	50-250°F	High, 170°F		WPS panel
<u>Radiation Instrumentation</u>							
RICA-18	Waste discharge line	150	200	10-1 million counts/min	High	Variable	WPS and radiation monitor panels
<u>Disposable Demineralizer Instrumentation</u>							
	Influent conductivity	250	200	0-1000 μmho/cm	-	-	Disposable demineralizer control panel (N2G21L001-N)
	Effluent conductivity	250	200	0-100 μmho/cm	-	-	Disposable demineralizer control panel (N2G21L001-N)
	Flowrate	250	200	0-72 gal/min	-	-	Disposable demineralizer control panel (N2G21L001-N)

TABLE 11.2-5 (SHEET 5 OF 5)

Channel Number[a]	Location of Primary Sensor	Design Pressure (psig)	Design Temperature (°F)	Range	Alarm Setpoint	Control Setpoint	Location of Readout
	Flow integrator	250	200	0-10 million gal	-	-	Disposable demineralizer control panel (N2G21L001-N)

a. The following abbreviations are used:
F - flow
P - pressure
L - level
t - temperature
R - radiation
I - indication
C - control
A - alarm

TABLE 11.2-6

RADIOACTIVE LIQUID RELEASES FROM WESTINGHOUSE-DESIGNED PWR PLANTS

Plant	Year	Cladding	Average Percentage Fuel Defects	Total Released (B+Y) Ci	Average Discharge Concentration (µCi/ml)	Fraction of Column 2, Table II, Appendix B to 10 CFR 20.1-20.601 Concentration
Yankee Rowe	1970	Stainless steel	Negligible	0.036	1.5×10^{-10}	1.5×10^{-3}
	1971	Stainless steel	0.001	0.00034	1.25×10^{-12}	1.25×10^{-5}
Connecticut Yankee	1970	Stainless steel	0.01	29.5	4.02×10^{-8}	4.02×10^{-1}
	1971	Stainless steel	0.03	5.85	7.75×10^{-9}	7.75×10^{-2}
San Onofre	1970	Stainless steel	0.007	3.41	6.1×10^{-9}	6.1×10^{-2}
	1971	Stainless steel	0.015	9.21	1.34×10^{-8}	1.34×10^{-1}
R. E. Ginna	1970	Zircaloy	0.4	9.35	1.43×10^{-8}	1.43×10^{-1}
	1971	Zircaloy	0.26	0.96	1.45×10^{-9}	1.45×10^{-2}
H. B. Robinson Unit 2	1970	Zircaloy				
	1971	Zircaloy	<0.001	0.74	1.01×10^{-9}	1.01×10^{-2}
Point Beach Unit 1	1970	Zircaloy				
	1971	Zircaloy	<0.01	0.14	2.48×10^{-10}	2.48×10^{-3}

TABLE 11.2-7 (SHEET 1 OF 2)

CALCULATED RELEASES OF RADIOACTIVE MATERIAL IN LIQUID EFFLUENTS (PER UNIT) ASSUMING EXPECTED FUEL LEAKAGE

FNP – 1 & 2 UPRATE CASE   LIQUID EFFLUENTS   ANNUAL RELEASE TO DICHARGE CANAL

		Coolant Concentrations									
Nuclide	Half-Life (Days)	Primary (Micro Ci/ML)	Secondary (Micro Ci/ML)	Boron RS (Curies)	Misc. Wastes (Curies)	Secondary (Curies)	Turb Bldg (Curies)	Total LWS (Curies)	Adjusted Total (Ci/Yr)	Detergent Wastes (Ci/Yr)	Total (Ci/Yr)
CORROSION AND ACTIVATION PRODUCTS											
CP-51	2.78E+01	1.92E-03	2.02E-07	0.00000	0.00005	0.00000	0.00000	0.00006	0.00011	0.00000	0.00011
MN 54	3.03E+02	3.13E-04	4.86E-08	0.00000	0.00001	0.00000	0.00000	0.00001	0.00002	0.00010	0.00012
FE 55	9.50E+02	1.61E-03	1.70E-07	0.00000	0.00006	0.00000	0.00000	0.00007	0.00012	0.00000	0.00012
FE 59	4.50E+01	1.01E-03	1.24E-07	0.00000	0.00003	0.00000	0.00000	0.00003	0.00006	0.00000	0.00006
CO 58	7.13E+01	1.61E-02	1.72E-06	0.00001	0.00053	0.00003	0.00002	0.00058	0.00109	0.00040	0.00150
CO 60	1.92E+03	2.02E-03	2.18E-07	0.00000	0.00008	0.00000	0.00000	0.00008	0.00015	0.00087	0.00100
ZR 95	6.50E+01	0.00E+00	0.00E+00	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00014	0.00014
NB 95	3.50E+01	0.00E+00	0.00E+00	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00020	0.00020
FISSION PRODUCTS											
BR 83	1.00E-01	5.30E-03	1.98E-07	0.00000	0.00000	0.00029	0.00000	0.00030	0.00056	0.00000	0.00056
BR 84	2.21E-02	2.91E-03	3.39E-08	0.00000	0.00000	0.00005	0.00000	0.00005	0.00009	0.00000	0.00010
RB 86	1.87E+01	8.58E-05	1.03E-08	0.00000	0.00001	0.00000	0.00000	0.00002	0.00003	0.00000	0.00003
RB 88	1.24E-02	2.25E-01	1.41E-06	0.00000	0.00000	0.00053	0.00000	0.00053	0.00098	0.00000	0.00098
SR 89	5.20E+01	3.53E-04	4.96E-08	0.00000	0.00001	0.00000	0.00000	0.00001	0.00002	0.00000	0.00002
MO 99	2.79E+00	8.60E-02	1.04E-05	0.00003	0.00008	0.00016	0.00010	0.00036	0.00068	0.00000	0.00068
TC 99M	2.50E-01	5.19E-02	2.87E-05	0.00002	0.00008	0.00043	0.00019	0.00072	0.00134	0.00000	0.00130
RU103	3.96E+01	4.54E-05	4.99E-09	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00001	0.00002
RU106	3.67E+02	1.01E-05	1.21E-09	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00024	0.00024
AG110M	2.53E+02	0.00E+00	0.00E+00	0.00000	0.00000	0.00000	0.00000	0.00000	0.00000	0.00004	0.00004
TE127M	1.09E+02	2.82E-04	2.20E-08	0.00000	0.00001	0.00000	0.00000	0.00001	0.00002	0.00000	0.00002

TABLE 11.2-7 (SHEET 2 OF 2)

FNP – 1 & 2 UPRATE CASE LIQUID EFFLUENTS ANNUAL RELEASE TO DICHARGE CANAL

		Coolant Concentrations									
Nuclide	Half-Life (Days)	Primary (Micro Ci/ML)	Secondary (Micro Ci/ML)	Boron RS (Curies)	Misc. Wastes (Curies)	Secondary (Curies)	Turb Bldg (Curies)	Total LWS (Curies)	Adjusted Total (Ci/Yr)	Detergent Wastes (Ci/Yr)	Total (Ci/Yr)
TE127	3.92E-01	9.09E-04	1.57E-07	0.00000	0.00001	0.00000	0.00000	0.00001	0.00002	0.00000	0.00002
TE129M	3.40E+01	1.41E-03	1.51E-07	0.00000	0.00004	0.00000	0.00000	0.00004	0.00008	0.00000	0.00008
TE129	4.79E-02	1.78E-03	9.27E-07	0.00000	0.00003	0.00001	0.00000	0.00004	0.00008	0.00000	0.00008
I130	5.17E-01	2.23E-03	1.72E-07	0.00000	0.00000	0.00026	0.00001	0.00027	0.00051	0.00000	0.00051
TE131M	1.25E+00	2.59E-03	2.30E-07	0.00000	0.00000	0.00000	0.00000	0.00001	0.00001	0.00000	0.00001
TE131	1.74E-02	1.23E-03	8.64E-07	0.00000	0.00000	0.00001	0.00000	0.00001	0.00003	0.00000	0.00002
I131	8.05E+00	2.74E-01	3.08E-05	0.00100	0.00300	0.04590	0.00300	0.05290	0.09883	0.00001	0.09900
TE132	3.25E+00	2.76E-02	2.66E-06	0.00001	0.00004	0.00004	0.00003	0.00012	0.00022	0.00000	0.00022
I132	9.58E-02	1.10E-01	1.47E-05	0.00006	0.00005	0.02188	0.00026	0.02225	0.04156	0.00000	0.04200
I133	8.75E-01	3.98E-01	3.49E-05	0.00094	0.00002	0.05205	0.00285	0.05586	0.10435	0.00000	0.10000
I134	3.67E-02	5.25E-02	9.16E-07	0.00000	0.00000	0.00137	0.00000	0.00137	0.00256	0.00000	0.00260

a. From GALE-CODE calculations.

TABLE 11.2-8 (SHEET 1 OF 2)

COMPARISON OF CALCULATED CONCENTRATIONS IN EFFLUENT WATER DISCHARGE WITH CONCENTRATION VALUES STATED IN COLUMN 2, TABLE II, APPENDIX B TO 10 CFR 20

ASSUMING EXPECTED FUEL LEAKAGE

Isotope	Annual Release to Discharge Pipe (μCi)[a]	Concentration in Circulating Water Discharge[b] (μCi/ml)	Maximum Permissible Concentration[c] (μCi/ml)	Fraction of Maximum Permissible Concentration
Cr-51	1.10E+02	4.32E-12	5.00E-04	8.64E-09
Mn-54	1. 20E+02	4. 71E-12	3.00E-05	1.57E-07
Fe-55	1.20E+02	4.71E-12	1.00E-04	4.71E-08
Fe-59	6.00E+01	2.36E-12	1.00E-05	2.36E-07
Co-58	1.50E+03	5.89E-11	2.00E-05	2.95E-06
Co-60	1.00E+03	3.93E-11	3.00E-06	1.31E-05
Zr-95	1.40E+02	5.50E-12	2.00E-05	2.75E-07
Nb-95	2.00E+02	7.86E-12	3.00E-05	2.62E-07
Br-83	5.60E+02	2.20E-11	9.00E-04	2.44E-08
Br-84	1.00E+02	3.93E-12	4.00E-04	9.82E-09
Rb-86	3.00E+01	1.18E-12	7.00E-06	1.68E-07
Rb-88	9.80E+02	3.85E-11	4.00E-04	9.62E-08
Sr-89	2.00E+01	7.86E-13	8.00E-06	9.82E-08
Mo-99	6.80E+02	2.67E-11	8.00E-06	3.34E-06
Tc-99m	1.30E+03	5.11E-11	1.00E-03	5.11E-08
Ru-103	2.00E+01	7.86E-13	3.00E-05	2.62E-08
Ru-106	2.40E+02	9.43E-12	3.00E-06	3.14E-06
Ag-110m	4.00E+01	1.57E-12	6.00E-06	2.62E-07
Te-127m	2.00E+01	7.86E-13	9.00E-06	8.73E-08
Te-127	2.00E+01	7.86E-13	1.00E-04	7.86E-09
Te-129m	8.00E+01	3.14E-12	7.00E-06	4. 49E-07
Te-129	8.00E+01	3.14E-12	4.00E-04	7.86E-09
I-130	5.10E+02	2.00E-11	2.00E-05	1.00E-06
Te-131m	1.00E+01	3.93E-13	8.00E-05	4.91E-09
Te-131	2.00E+01	7.86E-13	8.00E-05	9.82E-10
I-131	9.90E+04	3.89E-09	1.00E-06	3.89E-03
Te-132	2.20E+02	8.64E-12	9.00E-06	9.60E-07
I-132	4.20E+04	1.65E-09	1.00E-04	1.65E-05
I-133	1.00E+05	3.93E-09	7.00E-06	5.61E-04
I-134	2.60E+03	1.02E-10	4.00E-04	2.55E-07
Cs-134	1.30E+04	5.11E-10	9.00E-07	5.67E-04
I-135	3.70E+04	1.45E-09	3.00E-05	4.84E-05

TABLE 11.2-8 (SHEET 2 OF 2)

Isotope	Annual Release to Discharge Pipe (μCi)[a]	Concentration in Circulating Water Discharge[b] (μCi/ml)	Maximum Permissible Concentration[c] (μCi/ml)	Fraction of Maximum Permissible Concentration
Cs-136	3.70E+03	1.45E-10	6.00E-06	2.42E-05
Cs-137	1.10E+04	4.32E-10	1.00E-06	4.32E-04
Ba-137m	7.30E+03	2.87E-10	--	--
Ce-144	5.20E+02	2.04E-11	3.00E-06	6.81E-06
All Others	6.00E+01	2.36E-12	2.00E-09	1.18E-03
TOTAL	3.24E+05	1.27E-08	--	6.75E-03
H-3	5.50E+08	2.16E-05	1.00E-03	2.16E-02
Total+H-3	5.50E+08	2.16E-05	--	2.84E-02

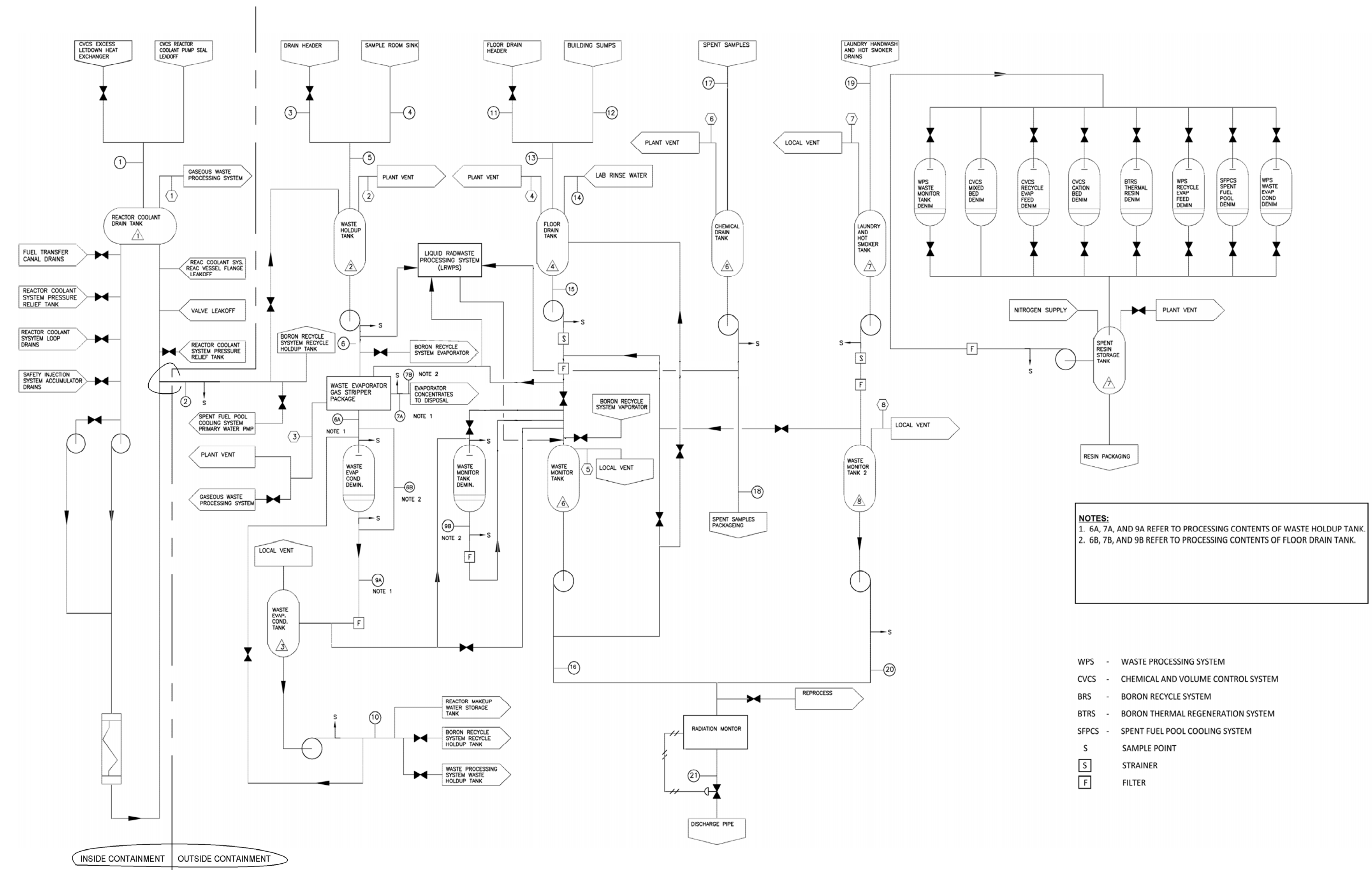
a. Based on the estimated isotopic liquid effluents in Table 11.2-7.
b. Based on the 16,000 gal/min/unit discharge in FSAR subsection 11.2.8.
c. Column 2, Table II, Appendix B to 10 CFR 20.

TABLE 11.2-9

ESTIMATED MAXIMUM INDIVIDUAL DOSE FROM LIQUID EFFLUENTS (mrem/year)[a]

Pathway	Age Group	Total Body	GI Tract	Bone	Liver	Kidney	Thyroid	Lung	Skin
Drinking water	Adult								
	Teen				Not Applicable (See Text)				
	Child								
	Infant								
Fish Ingestion	Adult	6.45E-01	4.30E-02	4.51E-01	8.54E-01	2.94E-01	3.89E-01	9.87E-02	0.00E+00
	Teen	3.75E-01	3.14E-02	4.72E-01	8.73E-01	2.96E-01	3.65E-01	1.13E-01	0.00E+00
	Child	1.53E-01	1.39E-02	5.83E-01	7.55E-01	2.48E-01	3.82E-01	8.88E-02	0.00E+00
	Infant	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Shoreline Recreation	Adult	4.40E-04	4.40E-04	4.40E-04	4.40E-04	4.40E-04	4.40E-04	4.40E-04	5.14E-04
	Teen	2.46E-03	2.46E-03	2.46E-03	2.46E-03	2.46E-03	2.46E-03	2.46E-03	2.87E-03
	Child	5.13E-04	5.13E-04	5.13E-04	5.13E-04	5.13E-04	5.13E-04	5.13E-04	5.99E-04
	Infant	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Boating	Adult	-	-	-	-	-	-	-	-
	Teen	-	-	-	-	-	-	-	-
	Child	-	-	-	-	-	-	-	-
	Infant	-	-	-	-	-	-	-	-
Swimming	Adult	-	-	-	-	-	-	-	-
	Teen	-	-	-	-	-	-	-	-
	Child	-	-	-	-	-	-	-	-
	Infant	-	-	-	-	-	-	-	-
Totals	Adult	6.45E-01	4.34E-02	4.51E-01	8.54E-01	2.94E-01	3.89E-01	9.91E-02	5.14E-04
	Teen	3.77E-01	3.39E-02	4.74E-01	8.75E-01	2.98E-01	3.67E-01	1.15E-01	2.87E-03
	Child	1.54E-01	1.44E-02	5.84E-01	7.56E-01	2.49E-01	3.83E-01	8.93E-02	5.99E-04
	Infant	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00

a. Appendix I Design Objectives for Liquid Effluents: total body dose = 3 mrem/year per unit from all pathways; dose to any organ = 10 mrem/year per unit from all pathways. Docket RM-50-2 Annex to Appendix I Design Objectives: total body dose = 5 mrem/year per site from all pathways; dose to any organ = 5 mrem/year per site from all pathways; nontritium releases = 5 Ci/year per unit.b. Dash (-) indicates dose less than 0.001 mrem/year.



REV 30 10/21

SOUTHERN COMPANY
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JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2

PROCESS FLOW DIAGRAM
LWPS TABULATED ACTIVITIES

FIGURE 11.2-1

11.3 GASEOUS WASTE SYSTEMS

11.3.1 DESIGN OBJECTIVES

The gaseous waste processing system (GWPS) was designed to remove fission product gases from the reactor coolant and have the capacity to contain these throughout the 40-year plant life. This was based on continuous operation with reactor coolant system activities associated with operation, with cladding defects in the fuel rods generating 1% of the rated core thermal power. The system was also designed to collect and store expected fission gases from the boron recycle evaporator and reactor coolant drain tank throughout the plant life.[a]

Although the system has the design capacity to contain fission product gases for the life of the plant, operating experience has demonstrated that the waste gas decay tanks must be released periodically due to nitrogen buildup. These releases are necessary following degassification of the reactor coolant system (RCS) prior to each RCS maintenance or refueling outage and again following deoxygenation at the end of each RCS maintenance or refueling outage. These releases are monitored and quantified in accordance with the radiological effluent technical specifications. At a preset level in the plant vent stack, waste gas decay tank releases are automatically terminated, as described in paragraph 11.4.2.2.5.

Experience has also shown that fuel defect levels have been extremely low. This greatly reduces the benefit of continuous purging of the volume control tank (VCT). Thus, during normal operation with low fuel defect levels, the VCT purge may not be performed. Without this continuous input of hydrogen with trace fission gases, there is no need to continuously operate the GWPS. When the GWPS is required during periods of high RCS fission gas concentrations and the VCT purge is initiated, the compressor and gas decay tanks can be utilized in a compressed storage mode of operation.

Gaseous activity released due to equipment leakage during normal operation of the plant is mixed with ventilation exhaust and is further diluted due to atmospheric dilutions. Table 11.3-9 gives estimated activity discharges from the plant vent stack.

The plant design considers potential personnel exposure and ensures that quantities of gaseous radioactive releases to the environment are as low as reasonably achievable. Under normal plant operation, the total activity from gaseous radionuclides leaving the GWPS does

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a. The renewed operating licenses authorize an additional 20-year period of extended operation for both FNP units, resulting in a plant operating life of 60 years. Since the GWPS has not been operated in the continuous purge mode in the past, the inventory accumulated in the GWPS up to the date of the renewed licenses (over 20 years of plant operation) is essentially nil. Should the system begin to be operated in the continuous purge mode at any time for the remaining life of the plant, the stated 40-year capacity of the GWPS remains sufficient.

not exceed a small fraction of the discharge limits as defined in column 1, Table II, Appendix B to 10 CFR 20. Although plant operating procedures, equipment inspection, and preventive maintenance are performed during plant operations to minimize equipment malfunction, overall radioactive release limits have been established as a basis for controlling plant discharges during operation with the occurrence of a combination of equipment faults of moderate frequency. A combination of equipment faults which could occur with moderate frequency include operation with fuel defects in combination with such occurrences as:

A. Steam generator tube leaks.

B. Malfunction in liquid waste processing system.

C. Malfunction of GWPS.

D. Excessive leakage in reactor coolant system equipment.

E. Excessive leakage in auxiliary system equipment.

The radioactive releases from the plant resulting from equipment faults of moderate frequency are within column 1, Table II, Appendix B to 10 CFR 20 limits on the short-term basis and do not exceed four to eight times the limits stated previously for normal operation.

11.3.2 SYSTEM DESCRIPTION

The GWPS consists mainly of a closed loop comprised of two waste gas compressors, two catalytic hydrogen recombiners, and gas decay tanks to accumulate the fission product gases.

The major input to the GWPS during normal operation is taken from the gas space in the volume control tank. The volume control tank gas space may be purged at a rate of 0.7 sf^3/min. Table 11.3-2 lists the rate of activity input to the GWPS during normal operation. There are no liquid seals in the system. The system is designed to preclude explosions by keeping the concentration of hydrogen and oxygen below the explosive limits. The Technical Requirements Manual contains limits for the concentration of hydrogen and oxygen.

Process flow diagrams and piping and instrumentation diagrams are shown in figure 11.3-1 and drawings D-175042, sheets 5, 6, 11, and 12, and D-205042, sheets 5, 6, 9, and 10. All lines in the gaseous waste system, including field run, are considered potential carriers of significant radioactivity. Only non-Category I pipe of Class B31.1 of the American National Standards Institute (size 2 in. and under) will be field run. This piping is shown on the piping and instrumentation drawings and is designated as Safety Class NNS. Table 11.3-3 gives process parameters for key locations in the system.

The GWPS includes two waste gas compressors and eight gas decay tanks to accumulate fission gases. Seven of these gas decay tanks are used during operation in the compressed gas storage mode, while the eighth one is normally used to accept relief valve discharges and administratively approved inputs.

11.3.3 SYSTEM DESIGN

11.3.3.1 Component Design

Gaseous waste processing equipment parameters are given in Table 11.3-4. For further information on safety and seismic classification see chapter 3. Paragraph 3.9.2.7 gives the general design criteria for field run piping.

11.3.3.1.1 Waste Gas Compressors

Two waste gas compressor packages are provided. One unit is normally used, with the other on a standby basis.

The units are centrifugal displacement machines which are skid-mounted in a self-contained package. Construction is primarily of carbon steel. Mechanical seals are provided to minimize the outleakage of seal water. The compressor has been used in Westinghouse pressurized water reactor (PWR) plants with excellent experience.

11.3.3.1.2 Recombiners

Two catalytic hydrogen recombiners are provided. As shown on drawing D-175042, sheets 11 and 12, one of the two recombiners is normally used to remove hydrogen from the hydrogen-nitrogen fission gas mixtures by oxidation to water vapor, which is removed by condensation. The other recombiner is available on a standby basis. Both units are self-contained and designed for continuous operation. In the compressed storage operating mode, the recombination function is not used. However, the recombiners and H_2 and O_2 analyzers remain installed.

11.3.3.1.3 Gas Decay Tanks

Gas decay tanks are provided as described in table 11.3-4. The tanks used during power operation are of vertical cylindrical type, and the shutdown tanks are horizontal cylindrical. All the gas decay tanks are constructed of carbon steel.

11.3.3.1.4 Valves

Each valve in the system is designed to meet the temperature, pressure, and code requirements for the specific application in which it is used. Special consideration is given to leaktightness. The recombiner circuits use manual valves provided with a diaphragm to prevent stem leakage and control valves with leakoffs returned to the gas system. Other parts of the gas system use Saunders valves and control valves with bellows seal.

11.3.3.2 Instrumentation Design

The main system instrumentation is described in table 11.3-5 and shown in the flow and piping diagrams of drawings D-175042, sheets 5, 6, 11, and 12, and D-205042, sheets 5, 6, 9, and 10.

The instrumentation readout is located mainly on the waste processing system panel in the auxiliary building. Some instruments are read where the equipment is located.

All alarms are shown separately on the waste processing system panel and further relayed to one common system annunciator on the main control board of the plant.

When used, the catalytic recombiner system is designed for automatic operation with a minimum of operator attention. Each package includes four online gas analyzers which are the primary means of recombiner control. A multipoint temperature recorder monitors temperatures at several locations in the recombiner packages.

Process gas flowrate is measured by an orifice located upstream of the recombiner preheater. Local pressure gauges indicate pressures at the recombiner inlet and the oxygen supply pressure.

11.3.4 OPERATING PROCEDURES

The gaseous waste processing system at Farley Nuclear Plant and systems of similar design have been operating for several years with excellent experience, as far as components and overall system performance are concerned.

Systems constructed from carbon steel have been in service for many years, and no failure due to corrosion damage has been reported.

Components of identical design to those used for the FNP are in use on several Westinghouse-designed GWPS. The performance and operating history of the compressors have been excellent.

11.3.4.1 Operation with Continuous VCT Purge

Prior to the system being put into operation, the GWPS is flushed free of air and filled with nitrogen. During normal power operation, nitrogen gas is continuously circulated around the loop by one of the two compressors. Fresh hydrogen gas is charged to the volume control tank, where it is mixed with fission gases which are stripped from the reactor coolant into the tank gas space. The contaminated hydrogen gas is then vented from the tank into the circulating nitrogen stream to transport the fission gases into the GWPS. The resulting mixture of nitrogen-hydrogen fission gas is pumped by the compressor to the recombiner where enough oxygen is added to reduce the hydrogen to a low residual level by oxidation to water vapor on a catalytic surface. After the water vapor is removed, the resulting gas stream is circulated to the gas decay tanks and back to the compressor suction to complete the loop circuit.

Each gas decay tank is capable of being isolated, and the number of tanks valved into operation at any time is restricted to diminish the amount of radioactive gases which could be released as a consequence of any single failure, such as the rupture of any single tank or connected piping. By alternating use of these tanks, the accumulated activity is contained in approximately equal parts.

When the hydrogen contained in the reactor coolant must be removed in preparation for a cold shutdown, two methods are available for removal. In the first method, the gas decay tanks are valved out of service and one of the two shutdown tanks is placed in service between the compressor discharge and the recombiner suction. The first degassing operation will require that the shutdown tank be pressurized with nitrogen before degassing begins. In addition, the flow of hydrogen to the volume control tank is stopped, the bypass on the volume control tank vent line is opened, and purge flow from the shutdown tank to the volume control tank is initiated, thus establishing a recirculation path between the GWPS and the volume control tank. The flow of gas through the volume control tank is controlled at the flowrate required to support RCS, CVCS, and GWPS operational requirements.

Initially, the flow will be predominantly hydrogen, but as degassing progresses the gas will become primarily nitrogen. Because of the difference in density between the gases, the throttle valve in the bypass line may require adjustment during the degassing operation to maintain a constant flowrate.

The alternative method of dissolved hydrogen removal consists of the controlled addition of hydrogen peroxide, which reacts with the dissolved hydrogen to form water. In this method, the reactor coolant system is sampled and analyzed for the dissolved hydrogen concentration. The gas spaces of various tanks which may contain hydrogen are sampled to establish initial hydrogen or oxygen concentration. Then, the stoichiometric amount of hydrogen peroxide is calculated and added to the charging pump suction via the chemical addition tank. Sampling is again performed to confirm that the reactor coolant dissolved hydrogen concentration is less than 5 cc/kg and that no hazardous mixtures of hydrogen and oxygen have been created.

11.3.4.2 Operation Without Continuous VCT Purge

Although the GWPS was designed for continuous purge of the VCT and 40-year holdup of fission gases, operating experience at FNP has shown that the GWPS can be operated without a continuous purge while maintaining personnel exposure within limits and maintaining releases within concentration and offsite dose limits. Many other operating PWRs are not designed with continuous purge capability and have operated for many years with gaseous releases from the GWPS well within MPC and offsite dose limits.

Fission gas production is directly related to fuel integrity. Fuel defects have been minimal at plants with Westinghouse fuel and, therefore, fission gas RCS concentrations are normally well below design limits (1-percent fuel defects).

The purpose of the VCT purge is to strip fission gases from the reactor coolant to reduce the exposure to personnel from fission gases which escape with reactor coolant leakage. However,

the primary contributor to exposure from leakage is Co-60, which is dissolved in the liquid. The only fission gases with significant half-lives are Xe-133 and Kr-85. Without a continuous VCT purge, Xe-133 will reach an equilibrium value in the RCS in about 30 days. Without fuel defects, the equilibrium concentration will be orders of magnitude less than design values and will not contribute to doses as a result of reactor coolant leakage. Kr-85 will accumulate in the RCS because of its long half-life. However, being a beta-emitter, it is not expected to contribute significantly to personnel exposure. Therefore, the benefit of the continuous purge is limited when fission gas concentrations are already low.

When RCS fission gas concentrations are low, there is no need for continuous inputs to the GWPS in the compressed gas storage mode. Therefore, the system is shut down. When it is necessary to vent gases from other systems to the GWPS, a waste gas compressor and gas decay tank are aligned in a recirculation loop or remain shutdown. These small inputs are normally accumulated in a gas decay tank, isolated for a period of decay, then released. Refer to plant procedures for information regarding the transfer of gases to the GWPS.

When RCS fission gas concentrations are high, the VCT purge may be initiated in the compressed gas mode to allow the hydrogen steam to carry fission gases into the GWPS recirculation loop. Each gas decay tank is aligned prior to the previous tank exceeding its pressure limit. Filled GDTs are isolated for decay of the fission gases. VCT purge operations may be planned such that by the time the seventh tank is filled, the contents of the first tank are released and the tank is available to collect gases again. This process may continue until RCS fission gas concentrations reach an acceptable level.

Doses resulting from a steam generator tube rupture accident have been evaluated without taking credit for VCT purge (see section 15.4). Regarding the gas decay tank rupture accident, the offsite dose will be limited by maintaining the activity in each gas decay tank within the Technical Requirements Manual limit of 70,500 Ci. Without the VCT purge, gas decay tank activity accumulation will be drastically reduced during normal operation but will increase during plant shutdown, due to RCS fission gas activity buildup. In spite of this buildup, the gas decay tank curie limit can still be maintained by the normal practice of spreading the activity among two or more tanks during shutdown degassing.

Therefore, the VCT purge is not required during normal power operation and can be aligned in or out of service at the operator's discretion. This philosophy allows for a simpler and more reliable GWPS operation as described below.

11.3.4.2.1 Plant Startup

This operation remains the same as with a VCT purge. The nitrogen gas space in the VCT is replaced with hydrogen by burping the gas space to the GWPS until the required RCS dissolved hydrogen concentration is achieved.

11.3.4.2.2 Normal Power Operation

The VCT purge is normally isolated. Without the major hydrogen input, the need for the recombiner operation is reduced. One compressor and one gas decay tank may be placed in service as necessary to accommodate the very small flow and volumetric inputs from the reactor coolant drain tank vent, the recycle evaporator vent, and the recycle holdup tank diaphragm vent. The VCT purge can be initiated as required to reduce RCS radiogas inventory at the discretion of the plant operators. At a nominal flowrate of 0.7 sf^3/min, a gas decay tank will be filled in 4 days. Therefore, a recombiner must eventually be placed into service, depending upon the available capacity in the remaining gas decay tanks.

11.3.4.2.3 Plant Shutdown

If shutdown is required such that the RCS is opened to the containment atmosphere (e.g., refueling or maintenance), the RCS fission gas and hydrogen concentration must be reduced to required levels. The VCT gas space is burped to the GWPS and fresh nitrogen is aligned. With the reactor shut down, no additional fission gases are produced. Therefore, the VCT burping needs to remove only residual fission gases, if their concentration is above the required shutdown limit. In this mode, the GWPS compressor and gas decay tank must be available to receive and collect the VCT burp volume. The gases can be stored for decay as required, then released as during normal plant operation.

11.3.5 PERFORMANCE TESTS

Initial performance tests are performed to verify the operability of the components, instrumentation, and control equipment.

Periodic testing of the oxygen monitors and hydrogen monitors is conducted in accordance with the Technical Requirements Manual.

11.3.6 ESTIMATED RELEASES

11.3.6.1 <u>Nuclear Regulatory Commission Requirements</u>

The following documents have been issued by the Nuclear Regulatory Commission to provide regulations and guidelines for radioactive releases:

A. 10 CFR 20, Standards for Protection Against Radiation.

B. 10 CFR 50, Licensing of Production and Utilization Facilities.

C. Regulatory Guide 1.42, Revision 1, "Interim Licensing Policy on As Low As Practicable for Gaseous Radioiodine Releases from Light Water Cooled Nuclear Power Reactors," March 1974.

Regulatory Guide 1.42, which was in effect during plant design, was withdrawn by the NRC in 1976 and replaced by guidance presented in the following regulatory guides:

A. Regulatory Guide 1.109, Revision 1, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR 50, Appendix I," October 1977.

B. Regulatory Guide 1.111, Revision 1, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors," July 1977.

C. Regulatory Guide 1.112, Revision O-R, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Reactors," April 1976.

D. NUREG-0133, "Preparation of Radiological Effluent Technical Specifications Nuclear Power Plants," October 1978.

During plant operations, radioactive gaseous releases will be controlled in accordance with technical specifications. For nuclear power plants, the NRC acceptance criteria for compliance with the dose limits stated in 10 CFR 20.1301 for individual members of the public may be demonstrated by complying with the limits of 10 CFR 50, Appendix I, and 40 CFR 190. Therefore, the use of dose rate values based on the guidance contained in NUREG-0133[(4)] is acceptable for use as a technical specification limit for gaseous effluent release rates since operational history at Farley Nuclear Plant has demonstrated that, with these dose rate limits in effect, the calculated maximum individual doses to members of the public are small percentages of the limits of 10 CFR 50, Appendix I and 40 CFR 190.

11.3.6.2 Radioactive Noble Gas Releases From FNP

A summary of gaseous discharges from FNP for years 1977 to 1983 is presented in table 11.3-6.

11.3.6.3 Expected GWPS Releases (Recombiner or Compressed Gas Mode)

The GWPS is designed to remove fission product gases from the volume control tank and recycle evaporator and was designed with the capacity to contain them through the lifetime of the plant. Since the VCT purge to the GWPS reduces fission product gas concentrations in the reactor coolant during unit operation, it reduces the escape of radioactive gases arising from any possible reactor coolant leakage. Design is based on continuous operation, with reactor coolant system activities associated with operation, with cladding defects in fuel rods generating

1% of the rated core thermal power. Table 11.3-7 shows the maximum fission product inventory in the GWPS over the 40-year plant life based on a 1-unit plant.[a]

Figure 11.3-2 shows that, for a given power rating, the quantity of fission gas activity accumulated in the gas system after 40 continuous years of operation is only twice the activity accumulated after 30 days of operation.[a] This is because most of the accumulated activity arises from short-lived isotopes reaching equilibrium in 1 month or less.

The difference between the 30-day and 40-year accumulations is essentially all Kr-85. This accumulation of Kr-85 is not a hazard to the plant operator because:

A. Radiation background levels in the plant are not noticeably affected by the accumulation of Kr-85, which is a beta emitter for which the tanks themselves provide adequate shielding.

B. The system activity inventory is distributed in several tanks so that the maximum permissible inventory in any single tank is actually less than that of earlier GWPS designs.

C. Since this system permits fission gas removal from the reactor coolant during normal operation, it is expected to reduce plant activity levels caused by a leakage of reactor coolant.

The capability to release a waste gas decay tank directly to the plant vent stack was provided as part of the original design of each unit. Automatic shutoff for such release occurs at a preset vent stack radiogas monitor setpoint as described in paragraph 11.4.2.2.5.

To further ensure design basis releases in accordance with the "as low as reasonably achievable" philosophy, the plant Offsite Dose Calculation Manual establishes limits for the releases. The quantity of radioactivity contained in each waste gas storage tank is limited by the Technical Requirements Manual to 70,500 Ci.

11.3.6.4 Releases from Ventilation Systems

A detailed analysis of one unit of the plant has been made to ascertain those items that could possibly contribute to airborne radioactive releases. Results of the analysis are presented in Section 11.3.9.

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a. The renewed operating licenses authorize an additional 20-year period of extended operation for both FNP units, resulting in a plant operating life of 60 years. Since the GWPS has not been operated in the continuous purge mode in the past, the inventory accumulated in the GWPS up to the date of the renewed licenses (over 20 years of plant operation) is essentially nil. Should the system begin to be operated in the continuous purge mode at any time for the remaining life of the plant, the stated 40-year capacity of the GWPS remains sufficient.

During normal plant operations, airborne noble gases and/or iodines can originate from reactor coolant leakage, equipment drains, venting and sampling, secondary side leakage, condenser air ejector and gland seal condenser exhausts, GWPS leakage, refueling operations, and evaporations from the spent fuel pool.

11.3.6.5 Estimated Total Releases

The potential release from the sources discussed in subsections 11.3.6.3 and 11.3.6.4 has been evaluated. Radioactive effluent releases from the plant for normal operation are given in table 11.3-9. These release rates were calculated using a composite of the PWR-GALE code[(1)] and plant operating parameters referenced in paragraph 11.1.1.2 and table 11.1-7 for operation in the continuous purge mode (section 11.3.4.1) or in the compressed storage mode (section 11.3.4.2). The releases are calculated for one unit; to obtain the combined releases for the two units, double the values listed in table 11.3-9.

The dose calculations, based on the estimated total plant releases, show that the releases are in accordance with the design objectives in subsection 11.3.1 and meet the regulations and guidelines as outlined in subsection 11.3.6.1. Further, the total plant releases, noted in table 11.3-10, are within the plant technical specifications, which are developed to be consistent with the "as low as reasonably achievable" criterion and the concentration limits specified in column 1, Table II, Appendix B to 10 CFR 20.

11.3.7 RELEASE POINTS

The GWPS is designed to contain all fission product gases generated during the plant lifetime. Any gases that do leak from the system are swept up by the radwaste area ventilation system, which discharges the gas to the plant vent stack.

The vent stack is the principal release point of gaseous waste to the environment. However, in the event of primary to secondary leakage, the power-operated atmospheric relief valve vents and the turbine building vent could become release points.

The vent stack is shown as part of the ventilation system in drawings D-175045 and D-205045. The physical location of the stack, shown in the plant general arrangement drawings of figures 1.2-1 through 1.2-9, exhausts at a height of 145 ft 9 in. above grade.

The main exhaust line from the radwaste area ventilation system to the vent stack is a 6-ft diameter duct which is flanged into the vent stack.

The vent stack parameters are as follows:

A. Base elevation - 155 ft (same as ground elevation).

B. Orifice elevation - not applicable.

C. Orifice inside diameter - not applicable.

D. Effluent velocity at flange of main exhaust line 2650 ft/min.

E. Heat input to stack - 39,078 Btu/h.

11.3.8 DILUTION FACTORS

Gaseous and particulate radioactive effluents may be normally released from the plant vent and turbine building vent as discussed in 11.3.7. Subsection 2.3.5 outlines the methodology and information used to determine the long-term atmospheric dilution (X/Q) and deposition (D/Q) for these release points. Effluent sources and associated vents are listed in table 2.3-15. Vent design information and input assumptions utilized for the long-term diffusion estimates are given in tables 2.3-16 and 2.3-17.

11.3.9 ESTIMATED DOSES[(1)]

[HISTORICAL]

[*Dose models, usage factors, and other parameters used to estimate the maximum doses to individuals from discharges of gaseous and particulate radioactive effluents are those described in Regulatory Guide 1.109.[(3)] These models were applied to the FNP using the source terms and atmospheric dilution factors discussed in subsections 11.3.6.5 and 11.3.8.*

Pathways included are plume exposure, ground shine, inhalation, and food ingestion (cow or goat milk, vegetation, and meat). Beta and gamma radiation doses to air were determined for the offsite location with the highest potential dose.

Receptor locations were selected by inspection of dispersion parameter values at locations tabulated in Tables 2.3-21 and 2.3-22. Doses were evaluated for a number of locations at which real receptors exist. Results were reviewed to identify the maximally exposed individual. This process is considered to yield doses which are unlikely to be substantially underestimated.

*The historical dose estimates above were calculated to confirm that the Farley Nuclear Plant conforms with the requirements of 10 CFR 50, Appendix I prior to plant operation (see reference 1).****]***

Detailed results are presented in Table 11.3-11 for the maximally exposed individual. This table provides a breakdown by organ and pathway, and doses include the summation from both assumed plant discharge points given in Table 2.3-16. Furthermore, the total dose to each organ is given in Table 11.3-11 along with the Appendix I design objective doses for comparison. It is clear that the estimated doses follow the design objective dose in each case. Actual plant releases during normal operation are governed by the Farley Nuclear Plant Technical Specifications and Offsite Dose Calculations Manual.

REFERENCES

1. Alabama Power Company letter to the Nuclear Regulatory Commission, "Dose Calculations to Conform with Appendix I Requirements," USNRC Docket Nos. 50-348, 50-364, June 3, 1976.

2. U.S. Nuclear Regulatory Commission, "Calculation of Releases from Pressurized Water Reactors," PWR-GALE Computer Code, NUREG-0017, April 1976.

3. Regulatory Guide 1.109, Revision 1, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR 50, Appendix I," October 1977.

4. U.S. Nuclear Regulatory Commission, "Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants," NUREG-0133, October 1978.

TABLE 11.3-1

(This table has been intentionally deleted.)

TABLE 11.3-2

ACTIVITY INPUT TO THE GASEOUS WASTE PROCESSING SYSTEM DURING NORMAL OPERATION[(a)]

Isotope	Input (Ci/year)
Kr-85	3.21×10^3
Kr-85m	3.16×10^4
Kr-87	1.58×10^4
Kr-88	6.31×10^4
I-131	20.5
I-132	7.53
I-133	33
I-134	4.47
I-135	17.66
Xe-131m	4.73×10^3
Xe-133	2.36×10^6
Xe-133m	4.42×10^4
Xe-135	1.26×10^5
Xe-135m	3.7×10^2
Xe-138	1.37×10^3

a. The table is based on 1 percent fuel defects and no decay.

TABLE 11.3-3

PROCESS FLOW DIAGRAM - GWPS – TABULATED ACTIVITIES

WASTE PROCESSING SYSTEM (GAS) - OPERATING PARAMETERS

BASIS: FUEL DEFECTS - 0.25%; POWER LEVEL - 2774; NO. OF UNITS - 1; GAS DECAY TANKS (NOTE 4) - 8; OPERATING INTERVAL - 2 DAYS; STRIPPING EFFICIENCY - 0.4

ITEM	DESCRIPTION	TEMP	PRESS	FLOW	N_2	H_2	ISOTOPIC CONCENTRATION, µCi/CC (NOTE 1)						
	(GAS STREAMS)	°F	PSIG	SCFM	%	%	KR85 (NOTE 3)	KR85M	KR87	KR88	XE-133	XE-133M	XE-135
1	VOLUME CONTROL TANK PURGE	130	15	0.7	0	100	0.11	0.73	0.38	1.45	54.3	1.35	2.65
2	GAS DECAY TANK DISCH. TO COMP.	AMB	0.5	40	99.9	0.1	25	0.29	0.04	0.37	225	5.10	2.38
3	COMPRESSOR SUCTION	AMB	0.5	40.7	98.2	1.8	24.5	0.30	0.05	0.39	221	5.03	2.38
4	COMP. DISCH. TO RECOMBINER	140	45	40.7	98.2	1.8	24.5	0.30	0.05	0.39	22	5.03	2.38
5	RECOMBINER DISCH. TO GAS DECAY TANKS	140	30	40	99.9	0.1	25	0.29	0.04	0.37	225	5.10	2.38
6	MISC. VENTS - EVAPS. RCDT, RHUTEDUCTOR	140	0.5		0	100							
7	RECOMBINER OXYGEN SUPPLY	AMB	50	0.35	0	0	0	0	0	0	0	0	0
8	RECOMBINER CALIBRATING GAS	AMB	15	0.004	94	6	0	0	0	0	0	0	0
9	RECOMBINER CALIBRATING GAS	AMB	ATM	0.004	94	6	0	0	0	0	0	0	0
10	WASTE GAS SYSTEM NITROGEN SUPPLY	AMB	100	0	100	0	0	0	0	0	0	0	0
11	NSSS NITROGEN SUPPLY	AMB	100	0	100	0	0	0	0	0	0	0	0
12	NITROGEN RELIEF TO PLANT VENT	AMB	100	0	100	0	0	0	0	0	0	0	0
13	NSSS HYDROGEN SUPPLY	AMB	100	0.7	0	100	0	0	0	0	0	0	0
14	VOLUME CONTROL TANK HYDROGEN	AMB	100	0.7	0	100	0	0	0	0	0	0	0
15	HYDROGEN RELIEF TO PLANT VENT	AMB	100	0	0	100	0	0	0	0	0	0	0
16	WASTE GAS DISCH. TO PLANT VENT	AMB	ATM	0	100	0	25	0	0	0	0	0	0
17	RECYCLE GAS TO VOLUME CONTROL TANK	AMB	100	0	100	0	0	0	0	0	0	0	0
18	PRESSURIZER RELIEF TANK VENT AND RETURN	120	3	0	100	0							
19	SHUTDOWN TANK RELIEF	AMB	ATM	0	100	0	0	0	0	0	0	0	0

ITEM	DESCRIPTION	TEMP	PRESS	FLOW	ISOTOPIC CONCENTRATION µCi/CC (NOTE 2)						
	LIQUID STREAMS	°F	PSIG	GPD	KR85 (NOTE 3)	KR85M	KR87	KR88	XE-133	XE-133M	XE-135
1	WASTE GAS COMPRESSOR DRAIN	140	45	0	5.96	.073	.01	.096	44.5	1.0	0.48
2	RECOMBINER DRAIN	140	30	6	4.33	.053	.008	.068	32.5	.75	0.35
3	GAS DECAY TANK DRAINS	AMB	40	36	1.78	.023	.003	.028	15.3	0.3	0.15
4	SYSTEM DRAINS TO VOL. CONTROL TANK	140	30-45	42	2.28	.028	.005	.035	17.0	0.38	0.18
5	RECOMBINER REACTOR MAKEUP WATER	AMB		0	0	0	0	0	0	0	0
6	COMPRESSOR MAKEUP WATER	AMB		36	0	0	0	0	0	0	0

ITEM	COMPONENT	TEMP	PRESS	N_2	H_2	COMPONENT INVENTORY, CURIES						
		°F	PSIG	%	%	KR85 (3)	KR85M	KR87	KR88	XE-133	XE-133M	XE-135
A	COMPRESSOR	140	45	98.2	1.8	11.25	0.13	0.018	0.17	101.8	2.53	1.08
B	RECOMBINER	140	30	99.9	0.1	8.5	0.098	0.014	0.13	78	1.73	0.80
C	GAS DECAY TANK	AMB	1.0	99.9	0.1	1981	5.25	0.73	6.8	4875	92.3	45.3
	TOTAL SYSTEM					11885	5.50	0.75	7.0	12000	98.3	46.0

NOTES:

1. CONCENTRATIONS IN µCi PER CC OF GAS AT ATMOSPHERIC PRESSURE AND 140°F.
2. CONCENTRATIONS IN µCi PER CC LIQUID AT ROOM TEMPERATURE.
3. KR - 85 CONCENTRATIONS ARE MAX VALUES BUT DO NOT OCCUR SIMULTANEOUSLY WITH OTHER ISOTOPE MAX CONCENTRATIONS.
4. INCLUDES TWO SHUTDOWN TANKS

TABLE 11.3-4

GASEOUS WASTE PROCESSING SYSTEM COMPONENT DATA[a]

Waste gas compressors	
Type	Centrifugal
Quantity	2
Design pressure (psig)	150
Design temperature (°F)	180
Operating temperature (°F)	70-140
Operating suction pressure	0.5
Design discharge pressure (psig)	100
Design flow, N_2 at 140°F (sf^3/min)	40
Gas decay tanks	
Type	Vertical, horizontal
Quantity	8
Design pressure (psig)	150
Design temperature (°F)	180
Volume, each (ft^3)	600
Material of construction	Carbon steel
Recombiners	
Type	Catalytic
Quantity	2
Design pressure (psig)	150
Design temperature (°F)	(b)
Design flowrate (sf^3/min)	50
Operating hydrogen recombiner rate (sf^3/min)	0.7
Material of construction	Stainless steel

a. The above components are designed and manufactured to the requirements given in Regulatory Guide 1.143, Revision 1, with the exception of the seismic design criteria given in Regulatory Position C.5. The components, systems, and structures are designed to the seismic design criteria given in section 3.7.

b. Varies by component, but exceeds component operating temperature by 100°F.

TABLE 11.3-5 (SHEET 1 OF 2)

GASEOUS WASTE PROCESSING SYSTEM INSTRUMENTATION DESIGN PARAMETERS

Channel Number[a]	Location of Primary Sensor	Design Pressure (psig)	Design Temperature (°)	Range	Alarm Setpoint	Control Setpoint	Location of Readout
Flow Instrumentation							
QIA-*1091	Gas decay tank water flush	150	180	0-6000 gal	Adjustable 3000-6000 gal		Local
Pressure Instrumentation							
PIA-1036	Gas decay tank 1	150	180	0-150 psig 0-30 psig	100 psig 20 psig		WPS panel
PIA-1037	Gas decay tank 2	150	180	0-150 psig 0-30 psig	100 psig 20 psig		WPS panel
PIA-1038	Gas decay tank 3	150	180	0-150 psig 0-30 psig	100 psig 20 psig		WPS panel
PIA-1039	Gas decay tank 4	150	180	0-150 psig 0-30 psig	100 psig 20 psig		WPS panel
PIA-1052	Gas decay tank 5	150	180	0-150 psig 0-30 psig	100 psig 20 psig		WPS panel
PIA-1053	Gas decay tank 6	150	180	0-150 psig 0-30 psig	100 psig 20 psig		WPS panel
PIA-1054	Gas decay tank 7	150	180	0-150 psig 0-30 psig	100 psig 20 psig		WPS panel

*Unit 2 only.

TABLE 11.3-5 (SHEET 2 OF 2)

Channel Number[a]	Location of Primary Sensor	Design Pressure (psig)	Design Temperature (°)	Range	Alarm Setpoint	Control Setpoint	Location of Readout
PIA-1055	Gas decay tank 8	150	180	0-150 psig 0-30 psig	100 psig 20 psig		WPS panel
PIA-1065	Hydrogen supply header	150	180	0-150 psig	90 psig		WPS panel
PIA-1066	Nitrogen supply header	150	180	0-150 psig	90 psig		WPS panel
PICA-1092	Compressor suction header	150	180	2 psi vacuum 2 psig	0.5 psi	0.5 psi	WPS panel
PI-1094	Volume control tank discharge pressure	150	250	0-20 psig			Local
PI-1047	Gas decay tank inlet nitrogen pressure						
<u>Level Instrumentation</u>							
LICA-1030	Compressor moisture separator	150	180	0-30 in. water	15 in. water	15, -10, 8, -5, -1 in. water	WPS panel and local
LICA-1032	Compressor moisture separator	150	180	0-30 in. water	15 in. water	15, -10, 8, -5, -1 in. water	WPS panel and local

a. The following abbreviations are used:

F	flow	T	temperature	I	indication
Q	water integrator	L	level	C	control
P	pressure	R	radiation	A	alarm

TABLE 11.3-6

RADIOACTIVE NOBLE GAS RELEASES FROM FNP

Year	Total Release (Ci)	Annual Boundary Dose (mR/year)	Annual Tech Spec Fraction %
UNIT 1			
1977	1.00×10^{3}	7.85×10^{-3}	0.08
1978	3.53×10^{3}	9.89×10^{-2}	0.99
1979	3.18×10^{3}	5.08×10^{-2}	0.51
1980	1.96×10^{4}	2.54	25.40
1981	2.21×10^{2}	1.26×10^{-1}	1.26
1982	3.81×10^{4}	1.13×10^{-2}	0.11
1983	2.20×10^{4}	5.00×10^{-2}	5.00
UNIT 2			
1977	-	-	-
1978	-	-	-
1979	-	-	-
1980	-	-	-
1981	2.60	2.03×10^{-3}	0.02
1982	3.54×10^{3}	4.24×10^{-1}	4.24
1983	8.47×10^{2}	1.18×10^{-1}	1.18

TABLE 11.3-7

ACCUMULATED RADIOACTIVITY IN THE GASEOUS WASTE PROCESSING SYSTEM AFTER 40 YEARS OPERATION[(a) (b)]

Activity (Ci) Following Plant Shutdown

Isotope	Zero Decay	30 Days	50 Days
Kr-85	11,890	11,820	11,780
Kr-85m	5.5	~0	~0
Kr-87	0.75	~0	~0
Kr-88	7.0	~0	~0
I-131	0.1656	0.0126	0.00226
I-132	0.000684	~0	~0
I-133	0.0259	~0	~0
I-134	0.000144	~0	~0
I-135	0.00504	~0	~0
Xe-131m	108	19	5.8
Xe-133	12,000	232	17
Xe-133m	96	0.01	~0
Xe-135	45	~0	~0

a. The table is based on 40 years continuous operation with 0.25% fuel defect.

b. The renewed operating licenses authorize an additional 20-year period of extended operation for both FNP units, resulting in a plant operating life of 60 ears. Since the GWPS has not been operated in the continuous purge mode in the past, the inventory accumulated in the GWPS up to the date of the renewed licenses (over 20 years of plant operation) is essentially nil. Should the system begin to be operated in the continuous purge mode at any time for the remaining life of the plant, the stated 40-year capacity of the GWPS remains sufficient.

TABLE 11.3-8

REDUCTION IN REACTOR COOLANT SYSTEM GASEOUS FISSION PRODUCTS RESULTING FROM NORMAL OPERATION OF THE GASEOUS WASTE PROCESSING SYSTEM[(a)]

Reactor Coolant Gaseous Fission Product Activities (μCi/g)		
Isotope	GWPS Operating	GWPS Not Operating
Kr-85m	2.1	2.2
Kr-85	0.14	5.5
Kr-87	1.3	1.3
Kr-88	3.6	3.8
Xe-131m	0.21	1.8
Xe-133m	1.5	3.2
Xe-133	79.8	290
Xe-135	5.7	6.1

a. Based on operation with cladding defects in fuel generating 1 percent of the rated core thermal power; power - 2774 MWt; purification letdown rate - 60 gal/min; purge rate -0.7 sf^3/min.

TABLE 11.3-9 (SHEET 1 OF 2)

EXPECTED ANNUAL AVERAGE RELEASE OF AIRBORNE RADIONUCLIDES[(a)(b)]

GASEOUS RELEASE RATE (CURIES PER YEAR)[(c)]

Nuclide	Primary Coolant (μCi/gm)	Secondary Coolant (μCi/gm)	Gas Stripping Shutdown	Gas Stripping Continuous	Building Ventilation Reactor	Building Ventilation Auxiliary	Building Ventilation Turbine	Blowdown Vent Offgas	Air Ejector Exhaust	Total
KR-83m	1.973E-02	6.646E-09	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
KR-85m	8.438E-02	2.900E-08	0.0E+00	0.0E+00	2.0E+00	4.0E+00	0.0E+00	0.0E+00	2.0E+00	8.0E+00
KR-85	2.064E-03	7.048E-10	5.5E+01	5.6E+02	2.4E+02	3.0E+00	0.0E+00	0.0E+00	1.0E+00	8.6E+02
KR-87	5.952E-02	1.936E-08	0.0E+00	0.0E+00	0.0E+00	4.0E+00	0.0E+00	0.0E+00	2.0E+00	6.0E+00
KR-88	1.735E-01	5.818E-08	0.0E+00	0.0E+00	2.0E+00	7.0E+00	0.0E+00	0.0E+00	3.0E+00	1.2E+00
KR-89	5.600E-03	1.912E-09	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
XE-131M	5.366E-03	1.844E-09	1.0E+01	6.2E+02	3.8E+02	1.5E+01	0.0E+00	0.0E+00	7.0E+00	1.0E+03
XE-133M	3.859E-02	1.326E-08	0.0E+00	0.0E+00	8.0E+00	2.0E+00	0.0E+00	0.0E+00	0.0E+00	1.0E+01
XE-133	1.606E+00	5.441E-07	1.0E+01	5.1E+02	6.7E+02	5.8E+01	0.0E+00	0.0E+00	2.7E+01	1.3E+03
XE-135M	1.424E-02	4.810E-09	0.0E+00	0.0E+00	0.0E+00	3.0E+00	0.0E+00	0.0E+00	1.0E+00	4.0E+00
XE-135	1.997E-01	6.749E-08	0.0E+00	0.0E+00	1.7E+01	2.0E+01	0.0E+00	0.0E+00	1.0E+01	4.7E+01
XE-137	1.007E-02	3.410E-09	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00	0.0E+00
XE-138	4.810E-02	1.600E-08	0.0E+00	0.0E+00	0.0E+00	3.0E+00	0.0E+00	0.0E+00	1.0E+00	4.0E+00
TOTAL NOBLE GASES										3.3E+03

IODINE RELEASE RATE - CURIES PER YEAR

Nuclide	PrimaryCoolant (μCi/gm)	Secondary Coolant (μCi/gm)	Building Ventilation Fuel Handling	Building Ventilation Reactor	Building Ventilation Auxiliary	Building Ventilation Turbine	Blowdown Vent Offgas	Air Ejector Exhaust	Total
I-131	2.739E-01	3.079E-05	6.0E-04	1.4E-03	1.4E-02	1.7E-03	1.8E-01	2.7E-03	2.0E-01
I-133	3.979E-01	3. 491E-05	1.9E-03	4.6E-03	4.6E-02	1.9E-03	2.1E-01	4.0E-03	2.7E-01

TRITIUM GASEOUS RELEASE 560 CURIES/YR

TABLE 11.3-9 (SHEET 2 OF 2)

AIRBORNE PARTICULATE RELEASE RATE - CURIES PER YEAR

Nuclide	Waste Gas System	Building Ventilation: Fuel Handling	Building Ventilation: Reactor	Building Ventilation: Auxiliary	Total
CR-51	1.4E-07	1.8E-06	3.2E-07	3.2E-06	5.5E-06
MN-54	4.5E-05	(e)	9.2E-07	1.8E-04	2.3E-04
FE-59	1.5E-05	(e)	3.2E-07	6.0E-05	7.5E-05
CO-57	0.0E+00	0.0E+00	2.9E-08	0.0E+00	2.9E-08
CO-58	1.5E-04	2.1E-04	3.2E-06	6.0E-04	9.6E-04
CO-60	7.0E-05	(e)	1.4E-06	2.7E-04	3.4E-04
SR-89	3.3E-06	2.1E-05	4.6E-07	1.3E-05	3.8E-05
SR-90	6.0E-07	8.0E-06	1.8E-07	2.9E-06	1.2E-05
ZR-95	4.8E-08	3.6E-08	0.0E+00	1.0E-05	1.0E-05
NB-95	3.7E-08	2.4E-05	6.3E-08	3.0E-07	2.4E-05
RU-103	3.2E-08	3.8E-07	5.6E-08	2.3E-07	7.0E-07
RU-106	2.7E-08	6.9E-07	0.0E+00	6.0E-08	7.8E-07
SB-125	0.0E-05	5.7E-07	0.0E+00	3.9E-08	6.1E-07
CS-134	4.5E-05	1.7E-05	9.2E-07	1.8E-04	2.4E-04
CS-136	5.3E-08	0.0E+00	1.1E-07	4.8E-07	6.4E-07
CS-137	7.5E-05	2.7E-05	1.6E-06	3.0E-04	4.0E-04
BA-140	2.3E-07	0.0E+00	0.0E+00	4.0E-06	4.2E-06
CE-141	2.2E-08	4.4E-09	4.6E-08	2.6E-07	3.3E-07

(a) For one unit.

(b) Twenty-five Ci/year of argon-41 are released from the containment, and 8 Ci/year of carbon-14 are released from the waste gas processing system (from PWR-GALE code, section 11.1, reference 1). 710 Ci/year of tritium are released via vapor pathways.

(c) The appearance of 0.0 in the table indicates release is less than 1.0 Ci/year for noble gas and 0.0001 Ci/year for I.

(d) Composite highest value GALE data as described in section 11.3.6.

(e) Nuclide amount insignificant; does not contribute to nuclide total value.

TABLE 11.3-10 (SHEET 1 OF 2)

COMPARISON OF CALCULATED MAXIMUM OFFSITE AIRBORNE CONCENTRATION WITH CONCENTRATION VALUES STATED IN COLUMN 1, TABLE II, APPENDIX B TO 10 CFR 20 ASSUMING EXPECTED FUEL LEAKAGE

Isotope	Total Annual Release from One Unit[a] (Ci/year)	Maximum Site Boundary Concentration[b] (μCi/ml)	Maximum Permissible Concentration (MPC)[c] (μCi/ml)	Fraction of MPC
Kr-83m	0.00E+00	0.00E+00	5.00E-05	0.00E+00
Kr-85m	8.00E+00	3.17E-12	1.00E-07	3.17E-05
Kr-85	8.60E+02	3.41E-10	7.00E-07	4.87E-04
Kr-87	6.00E+00	2.38E-12	2.00E-08	1.19E-04
Kr-88	1.20E+01	4.76E-12	9.00E-09	5.29E-04
Kr-89	0.00E+00	0.00E+00	N/A	--
Xe-131m	1.00E+03	3.96E-10	2.00E-06	1.98E-04
Xe-133m	1.00E+01	3.96E-12	6.00E-07	6.61E-06
Xe-133	1.30E+03	5.15E-10	5.00E-07	1.03E-03
Xe-135m	4.00E+00	1.59E-12	4.00E-08	3.96E-05
Xe-135	4.70E+01	1.86E-11	7.00E-08	2.66E-04
Xe-137	0.00E+00	0.00E+00	N/A	--
Xe-138	4.00E+00	1.59E-12	2.00E-08	7.93E-05
I-131	2.00E-01	7.93E-14	2.00E-10	3.96E-04
I-133	2.70E-01	1.07E-13	1.00E-09	1.07E-04
C-14	8.00E+00	3.17E-12	3.00E-07	1.06E-05
Ar-41	3.40E+01	1.35E-11	1.00E-08	1.35E-03
H-3	5.60E+02	2.22E-10	1.00E-07	2.22E-03
Cr-51	5.50E-06	2.18E-18	3.00E-08	7.27E-11
Mn-54	2.30E-04	9.12E-17	1.00E-09	9.12E-08
Fe-59	7.50E-05	2.97E-17	5.00E-10	5.95E-08
Co-57	2.80E-08	1.11E-20	4.00E-09	2.77E-12
Co-58	9.60E-04	3.81E-16	2.00E-09	1.90E-07
Co-60	3.40E-04	1.35E-16	2.00E-10	6.74E-07
Sr-89	3.80E-05	1.51E-17	1.00E-09	1.51E-08
Sr-90	1.20E-05	4.76E-18	3.00E-11	1.59E-07
Zr-95	1.00E-05	3.96E-18	5.00E-10	7.93E-09
Nb-95	2.40E-05	9.51E-18	2.00E-09	4.76E-09
Ru-103	7.00E-07	2.77E-19	2.00E-09	1.39E-10

TABLE 11.3-10 (SHEET 2 OF 2)

Isotope	Total Annual Release from One Unit[a] (Ci/year)	Maximum Site Boundary Concentration[b] (μCi/ml)	Maximum Permissible Concentration (MPC)[c] (μCi/ml)	Fraction of MPC
Ru-106	7.80E-07	3.09E-19	8.00E-11	3.86E-09
Sb-125	6.10E-07	2.42E-19	3.00E-09	8.06E-11
Cs-134	2.40E-04	9.51E-17	2.00E-10	4.76E-07
Cs-136	6.40E-07	2.54E-19	9.00E-10	2.82E-10
Cs-137	4.00E-04	1.59E-16	2.00E-10	7.93E-07
Ba-140	4.20E-06	1.66E-18	2.00E-09	8.32E-10
Ce-141	3.30E-07	1.31E-19	1.00E-09	1.31E-10
TOTAL				6.82E-03

a. Total Ci/year from table 11.3-9.

b. Based on the sum of contributions from the plant vent and turbine building vent using a ground release dilution factor (X/Q) of 1.0 x 10^{-5} s/m^3.

c. From column 1, Table II, Appendix B to 10 CFR 20.

d. 0.0 indicates release <1.0 Ci/yr for noble gas.

TABLE 11.3-11 (SHEET 1 OF 2)

ESTIMATED ANNUAL DOSES TO A MAXIMUM EXPOSED INDIVIDUAL FROM GASEOUS AND PARTICULATE EFFLUENTS (MREM/YR)[(a)(b)]

Pathway	Group	Body	Tract	Bone	Liver	Kidney	Thyroid	Lung	Skin
Plume (.9 mi WSW)	All	3.10E-02	3.10E-02	3.10E-02	3.10E-02	3.10E-02	3.10E-02	3.22E-02	1.04E-01
Ground Shine + Inhal +Veg. (0.9 Mile WSW)	Adult	9.49E-02	9.21E-02	1.90E-01	9.67E-02	9.85E-02	1.55E+00	9.02E-02	9.10E-02
	Teen	1.21E-01	1.18E-01	3.01E-01	1.25E-01	1.25E-01	1.36E+00	1.17E-01	1.18E-01
	Child	2.21E-01	2.17E-01	7.13E-01	2.27E-01	2.26E-01	2.07E+00	2.16E-01	2.16E-01
	Infant	1.51E-02	1.50E-02	6.37E-03	1.53E-02	1.53E-02	1.01E-01	1.50E-02	1.61E-02
Cow or Goat	Adult								
	Teen				None within 5 miles				
	Child								
	Infant								
Meat (1.1 Miles WSW)	Adult	8.36E-03	8.30E-03	3.02E-02	8.55E-03	8.69E-03	1.29E-01	8.02E-03	8.00E-03
	Teen	6.49E-03	6.42E-03	2.55E-02	6.70E-03	6.81E-03	9.38E-02	6.27E-03	6.25E-03
	Child	1.12E-02	1.10E-02	4.79E-02	1.15E-02	1.17E-02	1.43E-01	1.10E-02	1.09E-02
	Infant	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Totals[(c)] (excluding plume)	Adult	1.03E-01	1.00E-01	2.20E-01	1.05E-01	1.07E-01	1.68E+00	9.82E-02	9.90E-02
	Teen	1.28E-01	1.25E-01	3.27E-01	1.32E-01	1.32E-01	1.45E+00	1.23E-01	1.24E-01
	Child	2.32E-01	2.28E-01	7.61E-01	2.38E-01	2.38E-01	2.21E+00	2.27E-01	2.27E-01
	Infant	1.51E-02	1.50E-02	6.37E-03	1.53E-02	1.53E-02	1.01E-01	1.50E-02	1.61E-02

TABLE 11.3-11 (SHEET 2 OF 2)

a. Highest offsite annual Beta air dose = 0.13 mrad[(e)]
   Highest offsite annual Gamma air dose = 0.05 mrad[(e)]

b. All data are on a per unit basis.

c. Evaluated at a location where an exposure pathway and dose receptor actually exist at the time of licensing.

   Appendix I design objectives - radioiodines and particulates:
      Dose to any organ from all pathways - 15 mrem/year per unit

   Annex to Appendix I Docket RM-50-2 design objectives:

      Dose to any organ from all pathways - 15 mrem/year per site
      I-131 releases - 1 Ci/year per unit (reference table 11.3-9)

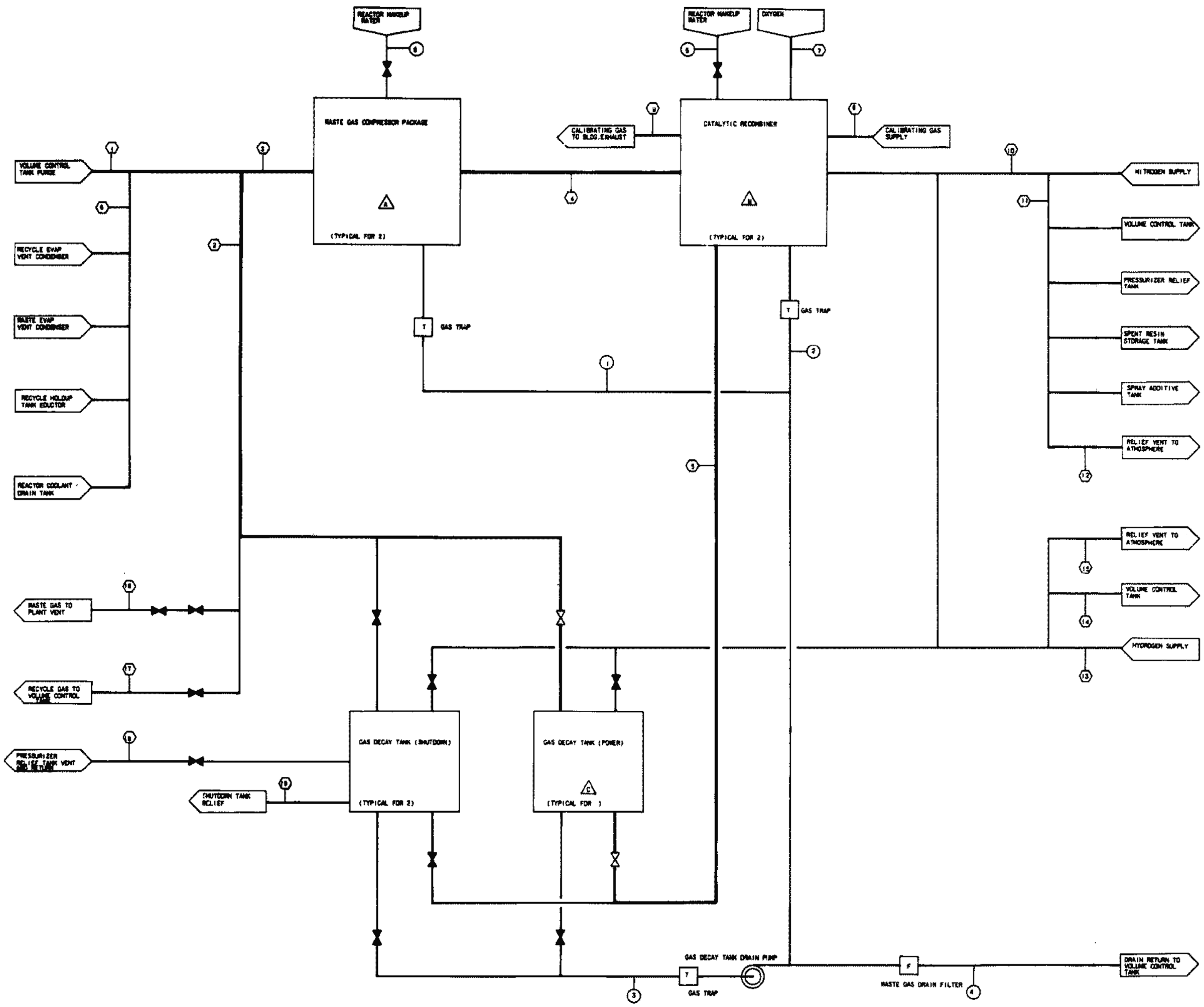
d. Evaluated at a location that could be occupied during the term of plant operation.

   Appendix I design objectives - gaseous effluents (noble gases only):

      Gamma dose in air - 10 mrad/year per unit
      Beta dose in air - 20 mrad/year per unit
      Dose to total body of individual - 5 mrem/year per unit
      Dose to skin of individual - 15 mrem/year per unit

   Annex to Appendix I, Docket RM-50-2. Design objectives are the same as Appendix I except on a per-site basis; therefore, calculated doses should be multiplied by 2.

e. Provided for information only; a receptor is assumed present at the location of a potential pathway. This evaluation is based on the worst case X/Q at the site boundary.



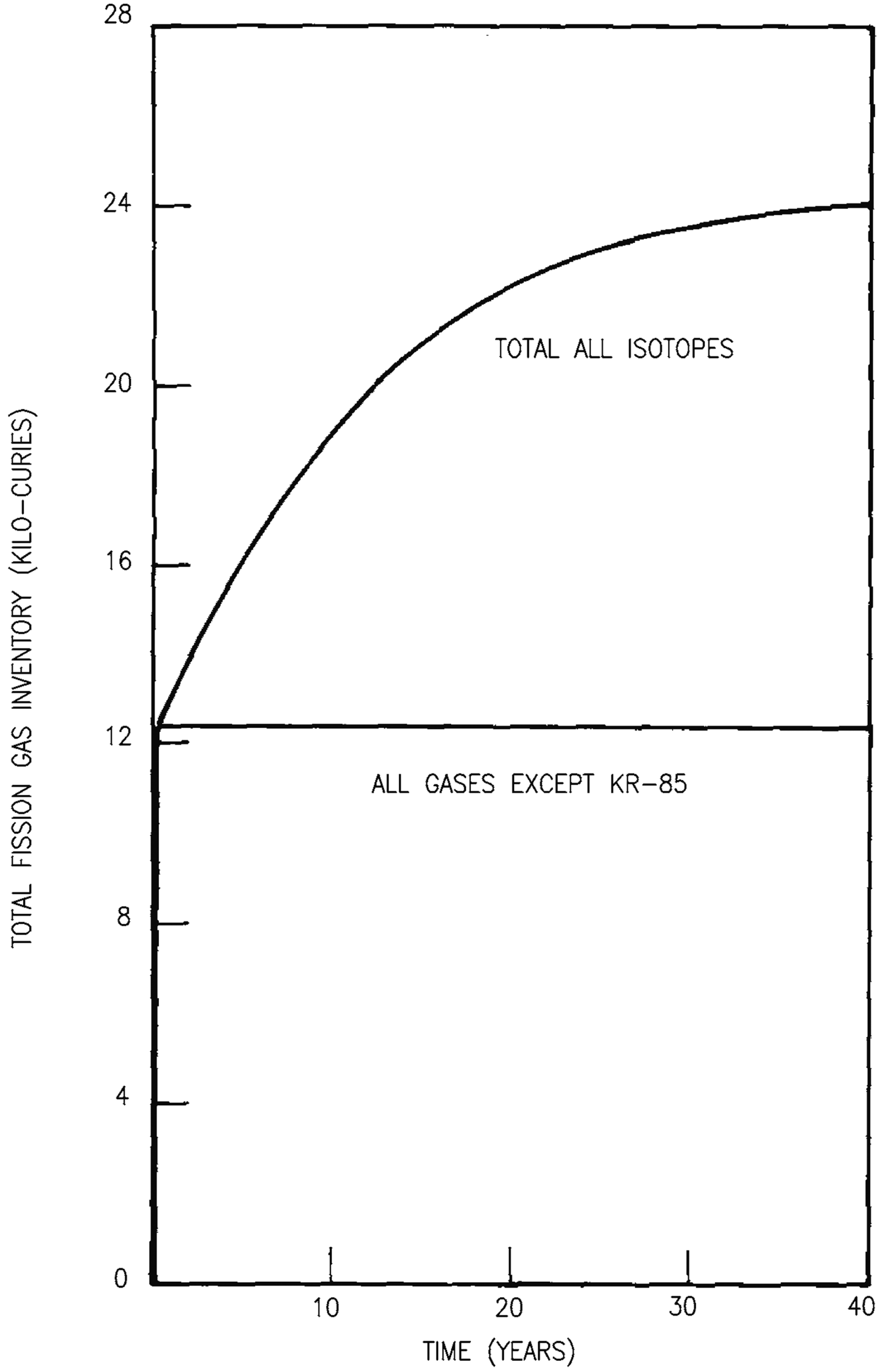
REV 21 5/08

SOUTHERN COMPANY
Energy to Serve Your World®

JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2

PROCESS FLOW DIAGRAMS FOR GASEOUS WASTE PROCESSING SYSTEM

FIGURE 11.3-1



(BASED ON CONTAINMENT CORE OPERATION WITH 0.25% FUEL DEFECTS)

REV 21 5/08

SOUTHERN COMPANY *Energy to Serve Your World®*	**JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2**	ESTIMATED WASTE GAS PROCESSING SYSTEM FISSION GAS ACCUMULATION
		FIGURE 11.3-2

11.4 PROCESS AND EFFLUENT RADIOLOGICAL MONITORING SYSTEMS

11.4.1 DESIGN OBJECTIVES

The radiation monitoring system consists of the following subsystems:

A. The process and effluent radiological monitoring system (PERMS), which includes both continuous process and periodic sampling systems.

B. The area radiation monitoring system, which monitors radiation fields in various areas within the plant. This system is further described in subsection 12.1.4.

C. The airborne radioactivity monitoring system, which is described in subsection 12.2.4.

The PERMS is designed to enable plant operation to be in compliance with Table II, Appendix B to 10 CFR 20.1 - 20.601, the low as reasonably achievable criterion of 10 CFR 50, and the Technical Specification limits of the Farley Nuclear Plant, in addition to being in accordance with the NRC acceptance criteria contained in 1971 General Design Criteria 60, 63, and 64 as documented in section 11.5 of NUREG 75/034, dated May 2, 1975.

The radiation monitoring system does not meet the guidelines of NRC Regulatory Guide 1.21 in its entirety. Specifically, continuous isotopic analysis and measurement of radionuclides to exceedingly low sensitivities and monitoring of all potential paths for radioactive release are not within the current state of the art and are therefore not addressed in the design of this system.

The general design objectives for the PERMS are to:

A. Warn of any radiation health hazard to operating personnel.

B. Warn of leakage from process systems containing radioactivity.

C. Monitor activity released in effluents and provide alarm and termination of the release when radiation levels exceed setpoint limits. Where the termination of release is not feasible, the monitors provide a continuous indication of the magnitude of activity released.

The accomplishment of these general objectives by the PERMS will provide assurance that exposures to individuals in restricted and unrestricted areas are as low as reasonably achievable during all modes of plant operation and during accidents.

Except for the containment purge exhaust line monitors, the spent fuel pool exhaust flow gas monitors, and the containment air particulate and noble gas monitor, the PERMS is not designed to Seismic Category I or to Institute of Electrical and Electronics Engineers (IEEE) accident grade standards and, therefore, no credit is taken for these monitors in the accident evaluations in chapter 15. The containment purge exhaust line monitors and the spent fuel pool

exhaust flow gas monitors are designed to Seismic Category I and to IEEE accident grade standards. The containment purge exhaust line monitors mitigate the consequences of a fuel handling accident inside the containment by isolating the containment purge and mini-purge lines. The spent fuel pool exhaust flow gas monitors mitigate the consequences of a fuel handling accident in the spent fuel pool area by realigning the spent fuel pool ventilation exhaust to the penetration room filtration (PRF) unit. The containment air particulate and noble gas monitor is part of the nonsafety-related reactor coolant pressure boundary leakage detection system and is procured under a quality assurance program consistent with Seismic Category I and IEEE Class 1E requirements. The containment air particulate and noble gas monitor does not perform a safeguard function and is not required to operate during or after a seismic event.

The design objectives for PERMS periodic sampling include the following:

A. Enable manual collection of representative samples of planned gaseous and liquid effluents prior to discharge to unrestricted areas during normal reactor operation and during anticipated operational occurrences in order to allow laboratory measuring and recording of the quantity of each of the principal radionuclides present in these discharges as required by 10 CFR 50.36a.

B. Enable manual collection of representative samples of gaseous and liquid process streams during normal reactor operation and during anticipated operational occurrences in order to allow laboratory measuring and recording of the quantity of each of the principal radionuclides present to verify and supplement the continuous process system monitors.

Written procedures shall be established, implemented, and maintained covering the quality assurance for effluent and environmental monitoring using the guidance in Regulatory Guide 4.15, dated February 1979.

11.4.2 PROCESS AND EFFLUENT RADIATION MONITORING

11.4.2.1 General Description

The components of the PERMS are designed for the following environmental conditions:

A. Temperature - An ambient temperature range of 40°F to 120°F.

B. Humidity - Relative humidity of 0 to 95 percent.

C. Pressure - Components in the auxiliary building and control room are designed for normal atmospheric pressure. Area monitoring system components inside the containment are designed to withstand containment test pressure.

D. Radiation - Process and effluent radiation monitors are of a nonsaturating design so that they will register full scale if exposed to radiation levels up to 100 times full-scale indication.

E. Radiation monitoring equipment is designed and located in such a way that radiation damage to electrical insulation and other materials will not affect its usefulness over the life of the plant. The only components of this system that are located in the containment are the detectors and associated local alarm and indication equipment for certain area monitoring channels. They are not expected to operate following a major loss-of-coolant accident (LOCA) and are not designed for this purpose.

Figure 11.4-1 contains the overall function block diagram for the radiation monitoring system.

All continuous process radiation monitors are indicated in the control room. They provide instrument malfunction and high activity annunciation in the main control room.

High reliability and ease of maintenance are emphasized in the design of this system. Sliding channel drawers are used for rapid replacement of units, assemblies, and entire channels. It is possible to completely remove the various chassis from the cabinet after disconnecting the cable connectors from the rear of these units.

Radiation monitoring system cabinet alarms consist of a red indicator light for high radiation and detector or circuit failure. Except for the R-10, R-11/12, Unit 1 R-29B & C, and Unit 2 R-29B & C monitors, the local meter and alarm assembly at the area monitor detector locations contains a red indicator light and a buzzer type alarm annunciator which are actuated on high radiation. The monitors R-10, R-11/12, Unit 1 R-29B & C, and Unit 2 R-29B & C have a local display which contains a red indicator light which is actuated on high radiation. [See figures 11.4-2, 11.4-3, 11.4-4, 11.4-5, and 11.4-6 and drawings U-167650, U-167651, U-167652, B-507156 (Unit 1 only), B-507157 (Unit 1 only), B-356750 (Unit 2 only), and B-356751 (Unit 2 only)]. An indicating light is provided on each drawer and a common annunciator is provided on the control board to indicate a channel placed in the test mode.

Radiation levels are recorded by a data acquisition system computer which can display data, on demand, to the operator.

Table 11.4-1 indicates the detector medium and temperature conditions. The different operating temperature ranges are within the design limits of the system.

Sensitivity of a radiation monitor is defined as the minimum signal level which is discernible above environmental noise (background). The sensitivity of all process and effluent monitors is designed to be two times the environmental signal level.

Each channel monitors gross concentrations, and detector output is measured in counts per minute (cpm), microCuries per cubic centimeter (μCi/cc), milliRad per hour (mRad/h), or roentgens per hour (R/h). Each channel has a minimum range of three decades.

The anticipated concentrations of radionuclides in the various process and effluent streams will be normal background radiation. The sensitivity of the detectors monitoring these streams will

ensure that abnormal conditions will be detected before they cause an undue hazard to the operators or the general public.

The relation of the radiation monitoring channels to the systems with which they are associated is given in the section describing those systems. Periodic tests will ensure that the channels operate properly.

11.4.2.2 System Description

This system consists of multiple channels which monitor radiation levels in various plant operating systems. The output from each channel detector is transmitted to the radiation monitoring system cabinets located in the control room where the radiation level is indicated by a meter or dot-matrix display and recorded. High radiation level alarms are indicated on the radiation monitoring system cabinets with annunciation at the control board in the control room.

A main control board annunciator provides a single window which alarms for any channel (process or area) detecting high radiation. A second common main control board annunciator is actuated for any channel failure. A third common main control board annunciator is actuated when any channel is placed in test mode. Verification of which channel has alarmed is made at the radiation monitoring system cabinets. (See figure 11.4-1.) Individual annunciators are located near the base of the radiation monitoring system cabinets.

A tabulation of the process radiation monitoring channels is found in table 11.4-2. The minimum sensitivity listed in the table is based on a Co-60 background level of at least 2 mR/h.

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

A. Log Level Amplifier or Microprocessor Controller/Display

   With the exception of R-10, R-11, R-12, Unit 1 R-29B & C, and Unit 2 R-29B & C, the log level amplifier accepts detector pulses, performs a log integration (converts total pulse rate to a logarithmic analog signal), and amplifies the resulting output for suitable indication and recording. For R-10, R-11, R-12, Unit 1 R-29B & C, Unit 2 R-29B & C, and R-60 (A through C) detector output is processed by the microprocessor and transmits the resulting output for indication and recording.

B. Power Supplies

   Furnishes the positive and negative voltages for the circuits, relays, and alarm lights and provides the high voltage for the detector.

C. Test Calibration Circuitry

   A precalibrated pulse signal to test channel electronics and a solenoid-operated radiation check source to verify channel operation are provided. A common annunciator on the main control board indicates when a channel is in the test

mode as shown on table 11.4-2. Channels R-15 (B and C), R-66 (A through F), R-10, R-11, Unit 1 monitors R-29B & C, Unit 2 monitors R-29B & C, and Channel R-60 (A through C) have a solenoid-operated radiation checksource to verify channel operation and an alarm light that indicates when an abnormal detector signal or power conditions exist.

D. Radiation Level Meter, LCD, or Dot-Matrix Display

Provides level indication calibrated in either cpm, µCi/cc, mRad/h, or R/hr for the minimum range shown on table 11.4-2. The level signal is also recorded.

E. Indicating Lights

Indicate high radiation alarms, circuit failures, and for the R-10, R-11/12, Unit 1 R-29B & C, Unit 2 R-29B & C, and R-60 (A through C) radiation monitoring system, “OPER” fault.

F. Bistable Circuits

Two bistable circuits are provided, one to alarm on high radiation (actuation point may be set at any level within the range of the instruments) and one to alarm on loss of signal (circuit failure or system “OPER” fault).

G. Checksource

A remotely operated, long half-life radiation checksource is furnished in each channel. The energy emission ranges are similar to the radiation energy spectra being monitored. The source strength is sufficient to cause approximately 30 percent of full scale indication. For R-10, R-11/12, Unit 1 R-29B & C, Unit 2 R-29B & C, and R-60 (A through C), the checksource activity is compared with a stored value in the local skid-mounted microprocessor and must exceed 85 percent of that value to pass.

11.4.2.2.1 Penetration Room Air Particulate Monitor - Channel R-10

The penetration room air particulate monitor (figure 11.4-2) detects air particulate beta radioactivity in the penetration room ventilation system discharge ducting. This monitor is functionally similar to the containment air particulate monitor R-11.

11.4.2.2.2 Containment Air Particulate Monitor - Channel R-11

This monitor (figure 11.4-6) is provided to continuously monitor air particulate beta radioactivity in the containment. This channel takes a continuous air sample from the containment atmosphere. The sample is drawn outside the containment in a closed system monitored by a scintillation counter fixed filter paper detector assembly. The filter paper collects 99 percent of

all particulate matter > 1 μm in size and is viewed by a photomultiplier beta scintillation combination. The air sample is returned to the containment after it passes through the series connected (channel R-12) gas monitor.

The detector assembly is in a completely enclosed housing. The detector is a photomultplier tube beta scintillation combination and the output is transmitted through the microprocessor to the radiation monitoring system control room cabinets.

Lead shielding is provided to reduce the effect of background radiation to where it does not interfere with the detector's sensitivity. A fixed filter paper system is part of the detector unit.

11.4.2.2.3 Containment Radioactive Gas Monitor - Channel R-12

This monitor (figure 11.4-6) is provided to continuously measure gaseous beta-gamma radioactivity in the containment.

This channel takes the continuous air sample from the containment atmosphere after it passes through the air particulate monitor and draws the sample through a closed system to the gas monitor assembly. The sample is monitored by a beta detector located in a fixed, shielded volume. The sample is then returned to the containment.

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 effect of background radiation to a point where it does not interfere with the detector's sensitivity.

The signal is processed by a microprocessor. The output signal is transmitted to the radiation monitoring system cabinets in the control room.

The containment air particulate and radioactive gas monitors (channels R-11 and R-12) have assemblies that are common to both channels. They are described as follows:

A. The flow control assembly includes a pump unit and selector valves that provide containment sample or a clean sample to the detectors.

B. The pump unit consists of:

   1. A pump to obtain the air sample.

   2. A digital mass flow indicator/controller to adjust and indicate the flowrate.

   3. A flow control valve to provide steady flow.

   4. High and low flow alarms provided on the radiation monitoring system rack.

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

D. A pressure sensor is used to protect the system from high pressure. This unit automatically closes the inlet and outlet valves upon a high pressure condition.

E. Purging is accomplished with a valve control arrangement whereby the normal sample flow is blocked and the detector "purged" with a clean sample.

F. The flow control panel in the control room radiation monitoring system racks permits operation of the flow control assembly. By operating a sample selector switch on the control panel, either the containment or "purge" sample may be monitored. Indicator lights are actuated by the following:

- Flow alarm assembly (low or high flow).
- The pressure sensor assembly (high pressure).
- The pump power control switch (pump motor on).

The containment air particulate and noble gas monitor (channels R-11 and R-12) is part of the nonsafety-related reactor coolant pressure boundary leakage detection system. These monitors provide continuous monitoring of the containment atmosphere to comply with the requirements of GDC 30 and GDC 64 for normal plant operation.

In the appendix 3A conformance statement for Regulatory Guide (RG) 1.45, it is noted that the reactor coolant pressure boundary leakage detection system does not perform a safeguard function and, contrary to the guidance of RG 1.45 for airborne particulate equipment, the system is not required to operate during or after a seismic event. The NRC accepted this position as noted in the original SER (NUREG-75/034, May 2, 1975, pages 5-6 and 5-7).

The R-11 and R-12 monitors do not serve a safety-related function and are not required to function during or after a seismic event. The monitors are Q-listed only for the purpose of documenting repairs and modifications. As permitted by FSAR subsection 17.3.5, the replacements can be purchased to meet the quality requirements of the original specification. These quality requirements are the appropriate quality assurance provisions for the purpose of meeting FSAR subsection 17.3.1 requirements for replacements.

Replacement of the existing Unit 1 R-11 and R-12 monitors with equipment procured under a quality assurance program consistent with a safety classification of safety-related and to Seismic Category I safe shutdown earthquake (SSE) requirements is advantageous for possible future changes to accident analysis requirements. Power to the skid-mounted equipment will be from Class 1E MCCs. The control panel in the control room will retain its current classification of nonsafety-related, Seismic II/I. An additional advantage would be to provide a higher

assurance for the closure function of the monitor's isolation valves after an SSE, allowing the associated containment isolation valves to be opened for operation of post-accident grab sampler R-67.

Post-accident grab sampler R-67 taps off upstream of the R-11/12 monitor inlet and returns to the system downstream of the R-11/12 monitor outlet. The purpose of this grab sample point is to provide a means for sampling the atmosphere in containment post-accident for particulates, iodines, and gases with a minimum of exposure. This sample point consists of:

A. Remote-operated valves on the inlet line and the outlet line to the grab sample point.

B. Filter holder with quick disconnects (holds filter disk and silver zeolite cartridge).

C. Gaseous collection vessel.

D. Vacuum pump with throttle valve.

E. Flow indication.

F. Control panel to allow remote operation of the valves and vacuum pump.

As documented in NRC SERs(1)(2)(3), post-accident sampling of containment atmosphere either conforms to NRC acceptance criteria contained in NUREG-0578, NUREG-0737, and Regulatory Guide 1.97 or deviations have been justified.
As documented in NRC SER(4), dated May 22, 2002, issuance of amendments 156 and 148 to the Units 1 and 2, respectively, Facility Operating Licenses supersedes the post-accident sampling system (PASS) specific requirements imposed by post-TMI confirmatory orders. However, the capability to obtain grab samples of the containment atmosphere will be maintained.

11.4.2.2.4 Waste Gas Processing Monitor - Channel R-13

The input line to the waste gas system compressor (drawing U-167652) is monitored for gaseous activity by a Geiger-Mueller tube to ensure the Technical Specification limit for waste gas decay tank storage is not exceeded and to provide the capability to establish the radioactive gas inventory in the waste gas processing system. Remote indication and annunciation are provided on the waste processing system control panel. The alarm setpoint will be based upon the waste gas decay tank rupture accident such that the resulting dose at the site boundary will be limited to 0.5 rem.

11.4.2.2.5 Deleted

11.4.2.2.6 Condenser Air Ejector Gas Monitor - Channel R-I5

This channel (drawing U-167650) monitors the discharge from the air ejector exhaust header of the condenser for gaseous radioactivity, which is indicative of a primary to secondary system leak. The gas discharge is routed to the turbine building vent. A beta-gamma sensitive Geiger-Mueller tube is used to monitor the gaseous radioactivity level. The detector is inserted into an inline fixed-volume container which includes adequate lead shielding to reduce the effect of background radiation so that it does not interfere with the detectors' maximum sensitivity.

11.4.2.2.7 Intermediate Range Condenser Air Ejector Monitor - Channel R-15B

This channel utilizes a beta-gamma sensitive Geiger-Mueller tube to monitor the gaseous radioactivity level in the air ejector exhaust. The monitor is set to alarm at one-half decade below the upper scale limit of R-15, thus providing an indication of the severity of a primary to secondary system leak that may be out of the measurement range of R-15.

11.4.2.2.8 High Range Condenser Air Ejector Monitor - Channel R-15C

This channel utilizes an ion chamber to monitor the gaseous radioactivity level in the air ejector exhaust. The monitor is set to alarm at one-half decade below the upper scale limit of R-15B, thus providing an indication of the severity of a primary to secondary system leak that may be out of the measurement range of R-15 or R-15B.

11.4.2.2.9 Deleted

11.4.2.2.10 Component Cooling Liquid Monitor - Channel R-17A and B

This channel (drawing U-167651) continuously monitors the component cooling system for radiation indicative of a leak of reactor coolant from the reactor coolant system and/or the residual heat removal system.

Scintillation counters are located in inline wells. A high radiation level alarm signal initiates closure of the valve located in the component cooling surge tank vent line to prevent gaseous radiation release.

Adequate lead shielding is provided to reduce the effect of background radiation so that it does not interfere with the detectors' maximum sensitivity.

11.4.2.2.11 Waste Processing System Liquid Effluent Monitor - Channel R-18

This channel (drawing U-167651) continuously monitors all waste processing system liquid releases from the plant. Automatic valve closure action is initiated by this monitor to prevent further release after a high radiation level is indicated and alarmed. A scintillation counter in an

inline sampler assembly monitors these effluent discharges. Remote indication and annunciation are provided on the waste processing system control board.

Adequate lead shielding is provided to reduce the effect of background radiation so that it does not interfere with the detectors' maximum sensitivity.

The monitor is located in the discharge line, prior to dilution, to provide superior measurement of radioactivity. Should the instrument fail during a release or should the activity exceed the instrument setpoint, the discharge valves will close and stop the release.

11.4.2.2.12 Steam Generator Liquid Sample Monitor - Channel R-19

This channel (figure 11.4-3) monitors the liquid phase of the secondary side of the steam generator for radioactivity that would indicate a primary to secondary system leak, providing backup information to that of the condenser air ejector gas monitor. Samples from the bottom or surface of each of the steam generators are combined in a common header and the resulting common sample is continuously monitored by a scintillation counter in an inline sampler assembly.

Adequate lead shielding is provided to reduce the effect of background radiation so that it does not interfere with the detectors' maximum sensitivity.

High activity alarm indications are displayed at the radiation monitoring system cabinets and at the monitor location, with annunciation at the main control board.

In the event of a high activity alarm, isolation valves in the sample lines would close. Subsequent identification of the steam generator that is leaking would be made by manual override of sample line isolation and by drawing samples from each steam generator for analysis.

11.4.2.2.13 Service Water Liquid Monitor - Channel R-20A and B

A scintillation counter is located in an offline sampler assembly.

Adequate lead shielding is provided to reduce the effect of background radiation so that it does not interfere with the detectors' maximum sensitivity.

Radiation monitors are provided in the discharge line from the containment air coolers, which are the main source of radioactivity discharged via the service water system to the environment (figure 11.4-6). Sensitivity of these instruments is given in table 11.4-2 (channel R-20). The only time there could be radioactive leakage into the service water system through the containment coolers is for a very short period of time when the containment pressure is higher than the service water pressure following a LOCA.

11.4.2.2.14 Deleted

11.4.2.2.15 Deleted

11.4.2.2.16 Steam Generator Blowdown Processing System Monitors – Channel R-23A and B

Steam generator blowdown process radiation monitor channel R-23A (figure 11.4-3) is provided to continuously monitor the liquid activity level of the blowdown fluid entering the surge tank. This full flow, inline monitor detects large fluctuations in activity concentration due to variations in steam generator inleakage conditions or to radioactive breakthrough of the system demineralizer train. A high signal from this instrument sounds an alarm and stops blowdown by closing the system's process controlled isolation valve.

Steam generator blowdown discharge radiation monitor channel R-23B (figure 11.4-3) is provided to continuously monitor the liquid activity level of the discharged or recycled fluid. This full flow, inline monitor provides the final radioactive control of the system's effluent. A high signal from this instrument sounds an alarm and closes the discharge valve if it should be open. Being downstream of channel R-23A, this instrument provides redundant protection against the discharge of excessive amounts of activity.

11.4.2.2.17 Containment Exhaust Flow Gas Monitors – Channel R-24A and B

A. Introduction

The containment exhaust flow gas monitors (figure 11.4-4) act to limit the radioactive releases associated with a fuel handling accident in the containment during purge operations. Design requirements were derived by analyses of the radioactive releases associated with the fuel handling accident discussed in chapter 15.

B. Identification of Safety Criteria

The documents listed below were considered in the design of the containment exhaust flow gas monitors:

1. <u>General Design Criteria for Nuclear Power Plants</u>, Appendix A, Title 10 CFR 50, July 7, 1971.

2. Regulatory Guide 1.22, "Periodic Testing of Protection System Actuation Functions."

3. The Institute of Electrical and Electronics Engineers, "Standard Criteria for Protection Systems for Nuclear Power Generating Stations," IEEE 279-1971.

4. The Institute of Electrical and Electronics Engineers, "Standard Criteria for Class 1E Electrical Systems for Nuclear Power Generating Stations," IEEE 308-1971.

5. The Institute of Electrical and Electronics Engineers, "Trial-Use Standard: General Guide for Qualifying Class 1 Electrical Equipment for Nuclear Power Generating Stations," IEEE 323-1971.

6. The Institute of Electrical and Electronics Engineers, "Standard Installation, Inspection, and Testing Requirements for Instrumentation of Nuclear Power Generating Stations," IEEE 336-1971.

7. The Institute of Electrical and Electronics Engineers, "Trial-Use Criteria for the Periodic Testing of Nuclear Power Generating Station Protection Systems," IEEE 338-1971.

8. The Institute of Electrical and Electronics Engineers, "Trial-Use Guide for Seismic 11.4-14 Qualification of Class 1 Electrical Equipment for Nuclear Power Generating Stations," IEEE 344-1971.

C. Independence of Redundant Safety-Related Systems

The criteria, including separation criteria, are given in paragraph 7.1.2.2.

D. Physical Identification of Safety-Related Equipment

The criteria are given in paragraph 7.1.2.3.

E. Conformance to IEEE 317-1971

The criteria are not applicable.

F. Conformance to IEEE 323-1971

The degree of conformance is given in paragraph 7.1.2.5.

G. Conformance to IEEE 336-1971

The degree of conformance is given in paragraph 7.1.2.6.

H. Conformance to IEEE 338-1971

The degree of conformance is given in paragraph 7.1.2.7.

I. Conformance to Regulatory Guide 1.22

The containment purge isolation system is designed to provide the greatest possible flexibility for periodic tests of the system.

J. Initiating Circuits

As shown in figure 11.4-4, when the gaseous activity in the containment purge exhaust line reaches the high level setpoint, automatic isolation of the containment purge and exhaust lines is initiated. Following the isolation of the purge system, radiation monitoring of subsequent releases into the containment is provided by the nonprotection grade air particulate and gas monitors (channels R-11 and R-12).

K. Bypasses

Each channel can be bypassed by means of front panel-mounted electromechanical switches for the purpose of testing and calibration. Visual indication is provided to aid the operator while performing this function. The bypass is indicated.

L. Interlocks

The criteria are not applicable.

M. Sequencing

The criteria are not applicable.

N. Redundancy

Redundant monitors and actuation circuits are provided. This arrangement allows testing of one actuation channel through its final output device. During such testing, administrative controls will ensure that only one actuation channel is bypassed. The other channel is available to effect isolation if required.

O. Diversity

The criteria are not applicable.

P. Actuated Devices

The actuated devices and their characteristics are shown in subsection 6.2.3.

Q. Supporting Systems

Supporting systems for the redundant containment purge exhaust gas monitors are the interruptible ac instrument power supply, the dc control power supplies, and the instrument air system. The isolating function is fail-safe with respect to all of these supporting systems.

R. Nonsafety Systems

The nonsafety-related systems associated with the radiation monitors are the instrument air system, the station annunciator, and the station computer.

S. Description

Redundant offline gas monitors are provided to continuously measure gaseous radioactivity levels of the containment purge exhaust flow during containment purge operations.

A motor-driven positive displacement pump is used to draw a continuous sample from the containment purge exhaust flow line and direct the sample through a particulate removal prefilter. The sample is then routed to a 4π shielded sample chamber of sufficient volume to accomplish the design purpose of the system. The sample effluent is monitored for radioactivity by a thin beta crystal scintillation detector assembly placed within the sample chamber in contact with the effluent, prior to being returned to the exhaust line to the stack.

The beta crystal is optically coupled to a photomultiplier tube, which responds to the light scintillations emitted from the crystal as a result of incident radiation giving up its energy within the crystal. The photocathode of the photomultiplier tube absorbs energy in the form of a pulse (current) which is fed directly into a preamplifier at the base of the detector assembly and in turn provides a signal to the control room readout instrumentation.

The control room readout instrument consists of a five-decade log level amplifier and associated circuitry as required to convert total pulse rate to a logarithmic analog signal output for suitable indication, recording, and alarm trip circuits.

The setpoint is based upon a release in which Kr-85 and Xe-133 are the predominant radionuclides, site boundary concentration values as presented in column 1, Table 2, Appendix B to 10 CFR 20.1001 - 20.2401, and the highest annual average mixed-mode X/Q value at the site boundary. Isolation of releases from the containment at or below concentration levels which correspond to these site boundary concentrations ensures that dose rates at the site boundary will not exceed limits established by Technical Specifications. The programmatic controls contained in the Technical Specifications represent the NRC acceptance criteria for radioactive gaseous effluent release rates.

Power for channels R-24A and B is provided from vital motor control centers A and B, respectively.

A "loss of power" and "channel failure" are monitored for each detector providing annunciation in the control room.

A channel performance test is available to the operator. An electronic pulse signal is used to verify the performance of the readout instrumentation.

A radioactive checksource, controlled from the readout instrument in the control room, can be actuated to check system integrity. This checksource is used as a convenient operational and gross calibration check of the detection system.

The checksource is of the same or similar energy and range of the isotopes to be monitored.

A three-way, solenoid-operated valve at the sample chamber inlet is operable from the control room. It is provided to permit air purging of the sample chamber to facilitate background activity checks.

Visual/audible indication of channel failure and/or high radiation is provided in the control room.

11.4.2.2.18 Spent Fuel Pool Exhaust Flow Gas Monitors - Channel R-25A and B

A. Introduction

The spent fuel pool exhaust flow gas monitors (figure 11.4-5) act to limit the radioactive releases associated with a fuel handling accident in the spent fuel pool. Design requirements were derived by analyses of the radioactive releases associated with the fuel handling accident discussed in chapter 15.

B. Identification of Safety Criteria

The documents listed below were considered in the design of the spent fuel pool gas monitors.

1. General Design Criteria for Nuclear Power Plants, Appendix A, Title 10 CFR 50, July 7, 1971.

2. Regulatory Guide 1.22, "Periodic Testing of Protection System Actuation Functions."

3. The Institute of Electrical and Electronics Engineers, "Standard Criteria for Protection Systems for Nuclear Power Generating Stations," IEEE 279-1971.

4. The Institute of Electrical and Electronics Engineers, "Standard Criteria for Class 1E Electrical Systems for Nuclear Power Generating Stations," IEEE 308-1971.

5. The Institute of Electrical and Electronics Engineers, "Trial-Use Standard: General Guide for Qualifying Class 1 Electrical Equipment for Nuclear Power Generating Stations," IEEE 323-1971.

6. The Institute of Electrical and Electronics Engineers, "Standard Installation, Inspection, and Testing Requirements for Instrumentation and Electronic Equipment During Construction of Nuclear Power Generating Stations," IEEE 336-1971.

7. The Institute of Electrical and Electronics Engineers, "Trial-Use Criteria for the Periodic Testing of Nuclear Power Generating Station Protection Systems," IEEE 338-1971.

8. The Institute of Electrical and Electronics Engineers, "Trial-Use Guide for Seismic Qualification of Class 1 Electrical Equipment for Nuclear Power Generating Stations," IEEE 344-1971.

C. Independence of Redundant Safety-Related Systems

The criteria, including separation criteria, are given in paragraph 7.1.2.2.

D. Physical Identification of Safety-Related Equipment

The criteria are given in paragraph 7.1.2.3.

E. Conformance to IEEE 317-1971

The criteria are not applicable.

F. Conformance to IEEE 323-1971

The degree of conformance is given in paragraph 7.1.2.5.

G. Conformance to IEEE 336-1971

The degree of conformance is given in paragraph 7.1.2.6.

H. Conformance to IEEE 338-1971

The degree of conformance is given in paragraph 7.1.2.7.

I. Conformance to Regulatory Guide 1.22

The isolation system is designed to provide the greatest possible flexibility for periodic tests of the system.

J. Initiating Circuits

As shown in figure 11.4-5, when the gaseous activity in the spent fuel handling building exhaust line reaches the high level setpoint, automatic isolation of the ventilation lines is initiated and the penetration room filtration system automatically starts, taking suction from the spent fuel area.

K. Bypasses

Each channel can be bypassed by means of front panel-mounted electromechanical switches for the purpose of testing and calibration. Visual indication is provided to aid the operator while performing this function. The bypass is indicated.

L. Interlocks

The criteria are not applicable.

M. Sequencing

The criteria are not applicable.

N. Redundancy

Redundant monitors and actuation circuits are provided. This arrangement allows testing of one actuation channel through its final output device. During such testing, administrative controls will ensure that only one actuation channel is bypassed. The other channel is available to effect isolation if required.

O. Diversity

The criteria are not applicable.

P. Actuated Devices

The actuated devices and their characteristics are shown in paragraph 9.4.2.2.2.

Q. Supporting Systems

Supporting systems for the gas monitors are the interruptible ac instrument power supply, the dc control power supplies, and the instrument air system. The isolating function is fail safe with respect to all of these supporting systems.

R. Nonsafety systems

The nonsafety-related systems associated with the radiation monitors are the instrument air system, the station annunciator, and the station computer.

S. Description

Redundant offline gas monitors are provided to continuously measure gaseous radioactivity releases to the environs by the ventilation fans exhausting the spent fuel pool area of the auxiliary building. The offline monitors incorporate a positive displacement pump that draws a continuous air sample from the spent fuel pool ventilation exhaust duct. This sample is then directed through a particulate removal prefilter. The sample is then routed to a 4π lead-shielded sample chamber of sufficient volume to accomplish the design purpose of the system. The sample effluent is then monitored for radioactivity by a thin beta scintillation detector assembly placed within the sample chamber in contact with the effluent, prior to being returned to the common vent duct exhausting all spent fuel pool spaces.

The beta crystal is optically coupled to a photomultiplier tube, which responds to the light scintillations emitted from the crystal as a result of incident radiation giving up its energy within the crystal. The photocathode of the photomultiplier tube absorbs the energy photons and emits the absorbed energy in the form of a pulse (current) fed directly into a preamplifier at the base of the detector assembly and, in turn, provides a signal to the control room readout instrumentation.

The control room readout instrument consists of a log level amplifier and associated circuitry as required to convert total pulse rate to a logarithmic analog signal output for suitable indication, recording, and alarm trip circuits. The ratemeters have a five-decade range (10^1 to 10^6 cpm).

The setpoint is based upon a release in which Kr-85 and Xe-133 are the predominant radionuclides, site boundary concentration values as presented in column 1, Table 2, Appendix B to 10 CFR 20.1001 - 20.2401, and the highest annual average mixed-mode X/Q value at the site boundary. Isolation of releases from the fuel handling area at or below concentration levels which correspond to these site boundary concentrations ensures that dose rates at the site boundary will not exceed limits established by Technical Specifications. The programmatic controls contained in the Technical Specifications represent the NRC acceptance criteria for radioactive gaseous effluent release rates.

Power for channel R-25A and B is provided from vital motor control centers A and B, respectively.

11.4.2.2.19 Deleted

11.4.2.2.20 Noble Gas Effluent Monitors

A1. **Deleted**

A2. Plant Vent Stack Monitor - GGG Monitor Skid Assembly R-29B & Sample Conditioning Skid Assembly R-29D

The GGG monitor skid assembly R-29B and sample conditioning skid assembly R-29D are located in the mechanical MCC room at el 155 ft of the auxiliary building. The GGG monitor system measures the radiation activity in the plant vent stack by drawing a representative sample via the monitor pumps, feeding it to the assembly's detectors, and exhausting it back to the plant vent. The system consists of a low- and mid/high-range gas monitoring with sample/filtering of particulate and iodine.

1. GGG Monitor - Low (R-29E)/Mid (R-29F)/High (R-29G) Noble Gas Channels with Composite Channel for R-29B:

   The GGG vent stack monitoring system has three noble gas detectors with a combined range of 10^{-7} to 10^{5} μCi/cc. The GGG monitor has low- and mid/high-range gas detectors with two different sample paths, low (normal) and mid/high (accident). When the plant vent gas activity concentration is near the top of the range of the low-range gas detector, the mid/high-range sample path starts automatically. A fourth channel, the composite channel, represents the entire system combined range to cover the low-, mid-, and high-gas detector ranges and displays the current activity of the selected gas detector.

   The low-range channel uses a beta scintillator/PhotoMultiplier tube (PMT) to detect beta radiation. The mid- and high-range channels use solid-state cadmium telluride (chlorine-doped) elements for detecting beta-gamma radiation. The radiation in the gas sample causes an electrical output that is in direct proportion to the level of the beta (low gas) and beta-gamma (mid/high gas) radioactivity present in the sample. The signal from these detectors is processed by a microprocessor. Radiation activity data, indication, and alarms are displayed locally at the skid and remotely in the main control room in the radiation monitoring system cabinets. Annunciation is provided on the main control board annunciation panel.

   Remote indication and annunciation are provided on the waste gas processing system control board. A high alarm initiates the automatic closure of the gas decay tank discharge valve.

   The assembly has lead shielding to minimize the effects of background radiation. The detectors have the fixed type of background subtract capability.

2. Sample Conditioning Skid Assembly 29D:

The sample conditioning skid (SCS) - The plant vent stack particulate/iodine grab samplers on the R-29D function to filter particulates and iodine from the vent stack sample air prior to entering the GGG gas detector chambers. There are two sets of filters, one set for the low-range (normal) sample path and another set for the mid/high-range (accident) sample path for a total of 4 filters. The two filters in each sample path allow the filters to be changed without stopping the monitors. The two mid/high-range filters are each enclosed in a 2-in. lead shield assembly to protect personnel from radiation exposure during filter removal. Grab sample ports are also provided in the GGG monitor assembly's outlet sample return line to facilitate collection of required plant vent gaseous samples via a portable gas sample apparatus.

Calibration of all detector channels is by use of external calibration sources and is performed upon installation and at established intervals. The GGG monitor is capable of functioning both during and following an accident.

In addition, checksources are provided to verify channel operation. The following summarizes by channel number and type which checksources are provided:

- Channels R-29E - 0.1 µCi, Cl-36 Beta Checksource.

- Channels R-29F and R-29G - 50 µCi, Cs137 Gamma Checksource.

The plant vent noble gas concentration in $\mu Ci/cm^3$ is determined by sampling and/or by obtaining a value from the monitor. The plant vent flowrate is determined by the number of operating auxiliary building exhaust fans. The release rate in Ci/s is determined by the following equation:

Release rate (Ci/s) = Concentration ($\mu Ci/cm^3$) x flowrate (ft^3/min) x conversion factor

The above method to determine noble gas release rate is described in emergency implementing procedures. During emergencies the release rate is calculated periodically as directed by the emergency director to determine whether the accident classification should be upgraded.

A3. Unit 1 and Unit 2 Plant Vent Stack Monitor - Particulate, Iodine, and Gas (PIG) Monitor Skid Assembly Unit 1 and Unit 2 R-29C

The PIG monitor skid assembly Unit 1 and Unit 2 R-29C is located in the mechanical equipment room at el 175 ft of the auxiliary building. The system consists of particulate, iodine, and noble gas detectors designed to continuously

sample the air from the plant vent stack. Monitor Unit 1 and Unit 2 R-29C contains Channels R-29H, R-29I, and R-29J. The PIG measures particulate, iodine, and noble gas radioactive nuclide concentrations in fixed volume samples. It includes two beta detector subassemblies, one for particulate (fixed filter) and one for gas radiation countrate. These channels use a scintillator and PhotoMultiplier Tube (PMT) subassembly to produce an electrical signal proportional to the level of beta radioactivity sensed by the particulate and gas detectors. The third detector subassembly is a gamma detector for iodine radiation countrate. This channel uses a NaI crystal and PMT subassembly to produce an electrical signal proportional to the level of gamma radioactivity sensed by the iodine detector. A lead shield assembly surrounds the three detectors, minimizing the effects of background radiation on activity data. The detectors have the fixed type of background subtract capability.

The signal from these detectors is processed by a microprocessor. Radiation activity data, indication, and alarms are displayed locally at the skid and remotely in the main control room radiation monitoring system cabinets. Annunciation is provided on the main control board annunciation panel.

Grab sample ports are provided in the PIG monitor assembly's outlet sample return line to facilitate collection of required plant vent gaseous samples via a portable gas sample apparatus.

Calibration of all detector channels is by use of external calibration sources and is performed upon installation and at established intervals.

In addition, checksources are provided to verify channel operation. The following summarizes by channel number and type which checksources are provided:

- Channel R-29H - 0.1 µCi, Cl-36 Beta Checksource.
- Channel R-29I - 8 µCi Ba-133 Gamma Checksource.
- Channel R-29J - 0.1 µCi Cl-36 Beta Checksource.

A4. The R-29B/R-29D monitor performs a RG 1.97 function. The Unit 1 R-29B/R-29D and R-29C and Unit 2 R-29B/R-29D and R-29C monitors are Class 1E. The monitors have been environmentally qualified by the vendor for the environment in which they are located. The monitors are Q-Listed for the purpose of documenting repairs and modifications. As permitted by FSAR subsection 17.3.5, the replacement can be purchased to meet the quality requirements of the original specification. These quality requirements are the appropriate quality assurance provisions for the purpose of meeting the FSAR subsection 17.3.1 requirements for replacements.

B. Main Condenser Air Removal Monitor - Channel R-15B and C

The main condenser air removal exhaust systems for Units 1 and 2 are monitored using the existing monitor (described in paragraph 11.4.2.2) on the steam jet air ejector exhaust for the normal range of radioactivity. The accident range of radioactivity will be monitored for Units 1 and 2 by intermediate and high range detectors with overlapping ranges and located at the common vent duct for the turbine building. The accident monitor consists of two detectors and readouts. The intermediate range detector readout module has a range of indication of 0.1 to 100 mR/h. The high range detector readout module has a range of 10 mR/h to 1000 R/h. The relationship between mR/h and $\mu Ci/cm^3$ will be established for the noble gas isotopes present during an accident. The range of the accident monitors in $\mu Ci/cm^3$ is from 10^{-5} to 10^3, with the normal range monitor measuring concentrations down to 10^{-6} $\mu Ci/cm^3$. This is the required range for the case where the steam jet air ejector exhaust is combined with turbine building ventilation exhaust. The readout modules are located in the control room and provide continuous indication. The accident detectors are shielded from background radiation with 6 in. of lead. Calibration is by use of an external calibration source and is performed upon installation and at intervals specified in the Technical Requirements Manual.

C. Main Steam Line Monitors - Channel R-60 (A through C)

The steam generators main steam lines will be monitored by measuring the radiation levels in these steam lines. There are three detectors located adjacent to the main steam lines in the main steam valve room. One detector will be used to monitor the main steam line from each steam generator The monitors have a range of 0.01 mRad/h to 1000 mRad/h. Each detector will be connected to a readout module in the control room, providing continuous indication. Each detector is collimated and background shielded with 2.5 in. of lead.

Calibration for both units is by use of an external calibration source and is performed upon installation and at intervals specified in the Technical Requirements Manual.

D. Design for Noble Gas Effluent Monitors

The noble gas effluent monitors are powered from a vital instrument bus. Procedures have been developed for use, calibration of the system, and dissemination of release rate information. The original APC position was to monitor the main condenser air removal exhaust and the discharge from the steam generator safety relief valves and atmospheric relief valves with a portable gamma survey instrument. However, APC finalized the above position based on NRC questions during the latter part of 1980 and purchased the best available monitors upon finalization of this position. To ensure accurate reading of each of these

monitors, a complex shielding design is required to discriminate actual readings from background, including containment shine.

11.4.2.2.21 Control Room Makeup Air Inlet Gas Monitors – Channel R-35A and B

A. Introduction

During normal plant operation, control room air is recirculated through air conditioning units to maintain control room design conditions of temperature and relative humidity. Fresh air makeup is provided by a supply duct from the computer room air conditioning unit. Redundant radiation monitors are provided on the makeup air supply duct. When a high radiation level is sensed by the monitors, a high radiation alarm is actuated in the control room, the air path is isolated to prevent entry of radioactive contaminants, and the control room ventilation system is manually aligned to the emergency recirculation mode. In addition, the radiation monitor signal also isolates the outside air intake to the technical support center (TSC) and realigns the TSC ventilation to the emergency filtration mode. After isolation of the control room and when conditions permit, fresh air can be brought in manually through redundant control room pressurization charcoal filter systems.

B. Identification of Safety Criteria

The documents listed below were considered in the design of the control room makeup air gas monitors:

1. General Design Criteria for Nuclear Power Plants, Appendix A, Title 10 CFR 50, July 7, 1971.

2. Regulatory Guide 1.22, "Periodic Testing of Protection System Actuation Functions."

3. The Institute of Electrical and Electronics Engineers, "Standard Criteria for Protection Systems for Nuclear Power Generating Stations," IEEE 279-1971.

4. The Institute of Electrical and Electronics Engineers, "Standard Criteria for Class 1E Electrical Systems for Nuclear Power Generating Stations," IEEE 308-1971.

5. The Institute of Electrical and Electronics Engineers, "Trial-Use Standard: General Guide for Qualifying Class 1 Electrical Equipment for Nuclear Power Generating Stations," IEEE 323-1971.

6. The Institute of Electrical and Electronics Engineers, "Standard Installation, Inspection, and Testing Requirements for Instrumentation of Nuclear Power Generating Stations," IEEE 336-1971.

7. The Institute of Electrical and Electronics Engineers, "Trial-Use Criteria for the Periodic Testing of Nuclear Power Generating Station Protection Systems," IEEE 338-1971.

8. The Institute of Electrical and Electronics Engineers, "Trial-Use Guide for Seismic Qualification of Class 1 Electrical Equipment for Nuclear Power Generating Stations," IEEE 344-1971.

C. Independence of Redundant Safety-Related Systems

The criteria, including separation criteria, are given in paragraph 7.1.2.2.

D. Physical Identification of Safety-Related Equipment

The criteria are given in paragraph 7.1.2.3.

E. Conformance to IEEE 317-1971

The criteria are not applicable.

F. Conformance to IEEE 323-1971

The degree of conformance is given in paragraph 7.1.2.5.

G. Conformance to IEEE 336-1971

The degree of conformance is given in paragraph 7.1.2.6.

H. Conformance to IEEE 338-1971

The criteria are not applicable.

I. Conformance to Regulatory Guide 1.22

The control room makeup air isolation system is designed to provide the greatest possible flexibility for periodic tests of the system.

J. Initiating Circuits

When the gaseous activity in the control room makeup air duct reaches the high level setpoint, automatic isolation of the outside air intake to the control room and the TSC ventilation system is shifted to the emergency filtration mode. Following the isolation of the normal control room intake, monitoring of activity in the control room and TSC is provided by the control room area radiation monitor (channel R-1), TSC area radiation monitor (R-01B), and health physics surveys.

K. Bypasses

Each channel can be bypassed by means of front panel-mounted electromechanical switches for the purpose of testing and calibration. Visual indication is provided to aid the operator while performing this function. The bypass is indicated.

L. Interlocks

The criteria are not applicable.

M. Sequencing

The criteria are not applicable.

N. Redundancy

Redundant monitors and actuation circuits are provided. This arrangement allows testing of one actuation channel through its final output device. During such testing, administrative controls will ensure that only one actuation channel is bypassed. The other channel is available to effect isolation if required.

O. Diversity

The criteria are not applicable.

P. Actuated Devices

The actuated devices and their characteristics are shown in subsection 9.4.1.

Q. Supporting Systems

Supporting systems for the redundant control room makeup air gas monitors are the interruptible ac instrument power supply, the dc control power supplies, and the instrument air system. The isolating function is fail-safe with respect to all of these supporting systems.

R. Nonsafety Systems

The nonsafety-related systems associated with the radiation monitors are the instrument air system, the station annunciator, and the station computer.

S. Description

Redundant offline gas monitors are provided to continuously measure gaseous radioactivity levels of the control room makeup air flow during normal plant operations.

A motor-driven positive displacement pump is used to draw a continuous sample from the control room makeup air line and direct the sample through a particulate removal prefilter. The sample is then routed to a 4π shielded sample chamber of sufficient volume to accomplish the design purpose of the system. The sample is monitored for radioactivity by a thin beta crystal scintillation detector assembly placed within the sample chamber in contact with the sample, prior to being returned to the makeup air duct.

The beta crystal is optically coupled to a photomultiplier tube, which responds to the light scintillations emitted from the crystal as a result of incident radiation giving up its energy within the crystal. The photocathode of the photomultiplier tube absorbs energy in the form of a pulse (current) which is fed directly into a preamplifier at the base of the detector assembly and in turn provides a signal to the control room readout instrumentation.

The control room readout instrument consists of a log level amplifier and associated circuitry as required to convert total pulse rate to a logarithmic analog signal output for suitable indication, recording, and alarm trip circuits. The ratemeters have a five-decade range (10^1 to 10^6 cpm).

Power for channels R-35A and B is provided from vital motor control centers A and B, respectively.

A "loss of power" and "channel failure" are monitored for each detector providing annunciation in the control room.

A channel performance test is available to the operator. An electronic pulse signal is used to verify the performance of the readout instrumentation.

A radioactive checksource, controlled from the readout instrument in the control room, can be actuated to check system integrity. This checksource is used as a convenient operational and gross calibration check of the detection system.

The checksource is of the same or similar energy and range of the isotopes to be monitored.

A three-way, solenoid-operated valve at the sample chamber inlet is operable from the control room. It is provided to permit air purging of the sample chamber to facilitate background activity checks.

Visual/audible indication of channel failure and/or high radiation is provided in the control room.

11.4.2.3 Alarm Setpoint Basis

The alarm setpoints for the process radiation monitors are based on the following:

A. The methodology used to calculate setpoints for RE-13, RE-15, RE-18, RE-23B, RE-24, Unit 1 Channels R-29E and R-29J and Unit 2 Channels R-29E and R-29J is specified in the Offsite Dose Calculation Manual (ODCM). The RE-23A setpoint methodology, while not specified in the ODCM, will be the same as that for RE-23B.

B. The RE-15B setpoint will be based such that the monitor will alarm at one half decade before RE-15 goes offscale. The RE-15C setpoint will be based such that the monitor will alarm at one half decade before RE-15B goes offscale.

C. Detector response which will provide warning to the operator of leakage of activity into a normally low activity system. This includes channels RE-11, RE-12, RE-17A and B, RE-19, RE-20A and B, and RE-60A, B, and C.

D. The detector response which will provide to the operator warning of plant vent stack effluent in accordance with Regulatory Guide 1.97. This includes R-29B/R-29D. In addition, the R-29B low-range noble gas channel and the Unit 1 and Unit 2 Monitor RE-29C iodine channel setpoints are based on the NOUE emergency classification criteria, annual average meteorology, on ODCM-based dose conversion factors, and maximum plant vent stack flowrate.

Typical alarm setpoints for the process radiation monitors are listed in table 11.4-3.

11.4.2.4 Design Evaluation

The liquid and gaseous waste discharge monitors are provided to maintain surveillance over the release of radioactivity with the following features:

A. A checksource is provided to permit the operator to check the monitor before discharge by operating a switch on the radiation monitor system panel.

B. If the reading falls off scale at any time, an indicator visible to the operator in the control room will alarm.

C. If the power supply to the channel fails, a high radiation alarm will annunciate. Control valves associated with the channel will also close.

An evaluation of instrumentation function relative to monitoring and for controlling release of radioactivity from various plant systems is discussed below.

A. Fuel Handling

For activity releases inside the containment and in the fuel handling area, the offline gas monitors (channel 24A and B or R-25A and B) would function. Each of these monitors initiates alarms in the control room and initiates ventilation isolation when the radiation level exceeds a preset level. Activity releases within the auxiliary building ventilation exhaust flow would cause the plant vent and area monitors to alarm on an increase in radiation level.

B. Liquid and Gas Wastes

For ruptures or leaks in the waste processing system, plant area monitors and the vent stack monitor will alarm on an increase in radiation level over a preset level. For cases where leaks are involved, the operator may control activity release by system isolation. For more severe postulated accident cases, such as rupture of waste tanks, activity release is not controlled. The environmental consequences of the postulated accidents are based on no-instrument action. For inadvertent releases relative to violation of administrative procedures, monitors provide means for limiting radioactivity release as well as alarming functions. The plant vent monitor will close the flow control valve in the waste decay tanks discharge line when the radiation level in the plant vent exceeds a preset level. Where liquid waste releases are involved, the waste processing system liquid discharge monitor trips shut a valve in the liquid waste discharge line when the radiation level in the discharge line exceeds a preset level.

C. Waste Gas Release Procedures

There is normally no need to vent the waste gas processing system, although occasional discharges will be required to perform maintenance. The waste gas release is an operator decision based on weather conditions and activity contained in the waste gas. When the operator has decided to release waste gas, he first samples the gas to determine its activity concentration. With this information and total pressure in the tank, the operator knows the quantity of activity to be released as well as the rate at which the gas can be released. To make the actual release, he must unlock and then open the manual isolation valve at the tank discharge and set the discharge flow control valve at the desired rate based on the plant vent activity monitor. Discharge flow is maintained at a constant rate by a pressure regulator upstream of the flow control valve. If the discharge flowrate results in an excessive radioactivity release rate, the flow control valve is tripped shut by the plant vent monitor.

D. Liquid Waste Release Procedure

The release of liquid waste is under administrative control. The normal procedure for discharging liquid waste is as follows:

1. A batch of waste is collected in one waste monitor tank.

2. The tank is isolated.

3. The tank contents are recirculated to mix the liquid.

4. A sample is taken for analysis.

5. If analysis indicates that release can be made within permissible limits, the quantity of activity to be released is recorded on the basis of the liquid volume in the tank and its activity concentration. Release is made when it is determined that the release is "as low as practicable" of necessity below permissible limits.

6. To release the liquid, an operator must unlock and open a stop valve in the discharge line, which is normally locked shut; open a second valve, which trips shut automatically on high radiation signal from the monitor (channel R-18); start a waste monitor tank pump and establish the normal flowrate using the flow indicator provided; and, finally, close the recirculation valve. Liquid is now being discharged.

E. Steam Generator Blowdown Release Procedure

See subsection 10.4.8 for a description of the steam generator blowdown release procedure.

F. Turbine Building Sump Release Procedure

There are two collection sumps located in each turbine building, each having a 30,000-gal capacity and isolated from each other. The radiation activity levels in these sumps are expected to be minimal. Each sump is provided with recirculation capability and will be grab sampled for composite prior to or during its discharge in conformance with Regulatory Guide 1.21. The discharge from this source is constant flow during a batch process. Radiation monitors at the discharge line would provide no meaningful information since the laboratory analysis of grab samples will be much more accurate; also, the instrument performance would be questionable as to the quality of water discharged from this system. Therefore, no in-process radiation monitors are justified, and none are provided.

G. Main Condenser Blowdown

The main condenser hotwell is blown down on occasion to assist in the maintenance of secondary water chemistry. The release is via the condensate pump discharge to the turbine building sump pump discharge line. This pathway is monitored as described in table 11.4-6.

11.4.3 SAMPLING

The following paragraphs present a detailed description of the radiological sampling procedures, frequencies, and objectives for all reactor plant process and effluent sampling. The process sampling system is described in subsection 9.3.2.

11.4.3.1 Process Sampling

The sample frequency, type of analyses, analytical sensitivity, and purpose of the sample are summarized in table 11.4-4 for each liquid process sample location and in table 11.4-5 for each gas process sample location. The analytical procedures used in sample analysis are presented in paragraph 11.4.3.3. This sampling monitors activity levels within various plant systems.

11.4.3.2 Effluent Sampling

11.4.3.2.1 Normal Operation Sampling

Effluent sampling of all potentially radioactive liquid and gaseous effluent paths will be conducted on a regular basis in order to verify the adequacy of effluent processing to meet the discharge limits to unrestricted areas. This effluent sampling program will be of such a comprehensive nature as to provide the information for the effluent measuring and reporting programs required by 10 CFR 50.36a, in annual reports to the NRC. The frequency of the periodic sampling and analysis described herein is a minimum. Table 11.4-6 summarizes the sample and analysis frequency schedules presented in the following paragraphs.

The following sample regime will apply to all potentially radioactive liquid effluents released to the plant discharge header from the liquid waste processing system.

A. Measurements will be made on a representative sample of each batch of effluent released and kept as a record together with the volume of the batch, the average dilution water flow used during discharge, and the time and date of release.

B. At least monthly a batch that is typical of average releases of radioactivity will be analyzed for dissolved fission and activation gases. This analysis has a minimum detectable concentration (MDC) as specified in table 11.4-6.

C. Proportional composite samples will be made up periodically to calculate total activity released. These will be samples in which the quantity of liquid added to the composite from each batch released will be proportional to the quantity of liquid in that batch. The composite will represent the average concentration prior to release and, by multiplying by the total volume released, will represent the quantity of radioactivity released during the compositing period. Such composite samples will be made up and analyzed in accordance with table 11.4-6

The steam generator blowdown system sample regime will be as specified in table 11.4-6.

Turbine building sump releases and condenser blowdown (during releases via this pathway) will be sampled and analyzed in accordance with table 11.4-6.

The following sample regime will apply to any intentional release from each containment purge exhaust:

A. The meteorological conditions of wind speed, wind direction, and atmospheric stability will be determined and averaged on an hourly basis for the purpose of determining the atmospheric dispersion during the period of release.

B. A representative gaseous sample of each release will be analyzed for individual noble gas nuclides in accordance with table 11.4-6. The gross noble gas activity released from the containment will be determined using grab samples.

C. A representative sample of each release will be analyzed for tritium in accordance with table 11.4-6. The samples will be collected by condensation or adsorption.

The following sample regime will apply to the potentially radioactive gaseous releases continuously discharged from the plant vent stack system and the condenser steam jet air ejector system:

A. Meteorological measurements of wind speed, wind direction, and atmospheric stability will be made and averaged over each 1-h period.

B. Gaseous activity releases will be quantitatively determined based on gaseous sample analyses and release flowrates for each of these effluent streams. The accumulated releases will be reported on a quarterly basis.

C. Within 1 month of initial criticality, at least monthly thereafter, and then following each refueling, process change, or other occurrence that could alter the mixture of radionuclide gas, samples will be analyzed for principal gamma emitting nuclides and tritium in accordance with table 11.4-6.

The following sample regime will also apply to the gaseous releases from the plant vent stack system:

A. A sample will be drawn through an iodine sampling device to determine the quantity of radioactive iodine isotopes released. The device will be analyzed at least weekly for I-131 and I-133 in accordance with table 11.4-6.

B. A continuous sample will be drawn through a particulate filter device and analyzed weekly for principal gamma emitting nuclides (Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141, Ce-144, and I-131) in accordance with table 11.4-6.

C. A monthly analysis of gross alpha will be made on a composite of particulate filters for a duration of 1 month in accordance with table 11.4-6.

D. A quarterly analysis for Sr-89 and Sr-90 will be made on a composite of particulate filters for a duration of one calendar quarter in accordance with table 11.4-6.

11.4.3.2.2 Post-Accident Sampling and Analysis of Plant Effluents

Southern Nuclear Operating Company has the capability to provide continuous sampling of plant gaseous effluent for post-accident releases of radioactive iodine and particulates at the plant vent and the condenser air removal system. The sampling method involves passing a portion of the effluent gases through a filter assembly and transporting the filter to a counting room for analysis. The sampling system has the following capabilities:

A. Effective iodine absorption of > 90 percent for all forms of gaseous iodine.

B. Greater than 90-percent retention of particulates for 0.3-mm diameter particulates.

C. Design intent meets sampling requirements of American National Standards Institute N13.1-1969.

D. Continuous collection whenever exhaust flow occurs.

E. Analytical facilities and procedures considered in the design basis sample.

F. Shielding factors considered in the design.

Onsite laboratory capability exists to analyze or measure these samples. The sampling system design is such that plant personnel can remove samples, replace sampling media, and transport the samples to the onsite analysis facility with radiation exposures that are not in excess of the guidelines of General Design Criterion 19 of 5 rem whole body and 75 rem to the extremities during the duration of the accident, assuming the design basis shielding envelope of NUREG 0737.

The post-accident vent stack monitor draws its sample into its filter system downstream of HEPA filters that limit sample particle size to 0.3 μm and smaller. The smaller particulates behave like gas and eliminate the need for isokinetic sampling. In addition, there is a post-accident plant vent stack grab sampling system with particulate and iodine filters and a gas sampler that allow a technician to draw samples to be analyzed in the laboratory.

The steam jet air ejector sample point is located on the vertical section of the turbine building exhaust ventilation duct. Locating the sample point on the vertical section of the exhaust duct ensures that the absorber material is not degraded with entrapped water.

11.4.3.3 Analytical Procedures

Samples of process and effluent gases and liquids will be analyzed in the laboratory by the following techniques:

A. Gross alpha counting.

B. Gamma spectrometry.

C. Liquid scintillation counting.

D. Radiochemical separations.

Instrumentation that will be available in the laboratory for the measurement of radioactivity includes:

A. End-window Geiger-Mueller counter.

B. Windowless or low-background windowed 2μ internal proportional counter.

C. Liquid scintillation counter.

D. Gamma spectrometer:

1. Intrinsic germanium detector.

2. Multichannel analyzer system.

Gross alpha analysis of all liquid effluent samples may be performed with the internal proportional counter. Samples will be evaporated onto planchets for counting. Sample volume and counting time will be chosen to give the desired sensitivities. Corrections will be made for sample detector geometry, sample self-absorption, and other parameters as necessary to ensure accuracy.

Gross alpha analysis of air particulate will be performed by counting a portion of the filter paper in the internal proportional counter.

Gamma spectrometry will be used for isotopic analysis of gaseous, air particulate, and liquid samples. A high efficiency, high resolution intrinsic germanium detector will be available for resolving complex gamma spectra.

Gaseous tritium samples may be collected by condensation, adsorption, or freeze out. Liquid samples for tritium analysis with interfering impurities will be purified prior to analysis by ion exchange, distillation, and/or filtration. Samples will be counted on the liquid scintillation counter.

Liquid samples will be collected in polymer bottles to minimize adsorption of nuclides onto container walls.

Gaseous radioactive iodine sampling devices will be tested to determine sampling efficiency and/or sampling line losses. In lieu of in-plant testing, data from tests performed by other organizations will be used if available.

When abnormal activity levels are detected by the above procedures, plant conditions will be studied to determine the cause of the abnormal activity and corrective action taken to bring the abnormal conditions back to normal.

11.4.4 INSERVICE INSPECTION, CALIBRATION, AND MAINTENANCE

11.4.4.1 Definitions

Radiation monitor channel check - A channel check shall be the qualitative assessment, by observation, of channel behavior during operation. This determination shall include, where possible, comparison of the channel indication and status to other indications or status derived from independent instrument channels measuring the same parameter.

Radiation monitor channel operational test - A channel operational test (COT) shall be the injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify the operability of required alarm, interlock, and trip functions. The COT shall include adjustments, as necessary, of the required alarm, interlock, and trip setpoints so that the setpoints are within the required range and accuracy.

Radiation monitor channel calibration - A channel calibration shall be the adjustment, as necessary, of the channel so that it responds within the required range and accuracy to known input. The channel calibration shall encompass the entire channel, including the required sensor, alarm, interlock and trip functions. The channel calibration may be performed by means of any series of sequential, overlapping calibrations or total channel steps so that the entire channel is calibrated.

11.4.4.2 Calibration Procedure

A primary calibration is performed on a one-time basis, utilizing typical isotopes of interest to determine proper detector response. Further primary calibrations are not required since the geometry cannot be significantly altered within the sampler. Calibration of samplers is then performed based on a known correlation between the detector responses and a secondary standard.

A remotely operated, long half-life radiation checksource is furnished as a secondary source in each channel. The energy emission ranges are similar to the radiation energy spectra being

monitored. The source strength is sufficient to cause approximately 30 percent of full scale indication.

11.4.4.3 Test Frequencies

The radiation monitoring system channels will have the following tests performed: channel check, source check, channel operational testing, and channel calibration. Intervals for source checks, channel checks, channel operational tests, and channel calibrations are contained in plant procedures, Technical Specifications, ODCM, or the Technical Requirements Manual, as applicable.

REFERENCES:

1. NRC Safety Evaluation Report, J. M. Farley Nuclear Plant Unit 1 and Unit 2, NUREG-0117 Supplement No. 5 to NUREG-75/034, dated March 1981.

2. Letter from NRC, dated March 26, 1985, and enclosed SER related to the Post-Accident Sampling System.

3. Letter from NRC, dated January 7, 1987, and enclosed SER related to Regulatory Guide 1.97.

4. Letter from NRC, dated May 22, 2002, and enclosed SER related to FOL amendments 156 and 148 for Units 1 and 2, respectively.

TABLE 11.4-1

DETECTING MEDIUM CONDITIONS

Channel	Medium	Temperature Range (°F)
R-1	Air	40-95
R-1B	Air	40-104
R-2	Air	40-120
R-3	Air	40-104
R-4	Air	40-104
R-5	Air	40-104
R-6	Air	40-104
R-7	Air	40-120
R-8	Air	40-104
R-9	Air	40-104
R-10	Air	50-110
R-11	Air	50-120
R-12	Air	50-120
R-13	Air	50-500
R-15	Air	40-150
R-15B and C	Air	50-105
R-17A and B	Water	60-200
R-18	Water	60-500
R-19	Water	<100
R-20A and B	Water	30-110
R-23A and B	Air	50-120
R-24A and B	Air	40-120
R-25A and B	Air	40-110
R-27A and B	Air	60-367
R-29B	Air	40-120
1R-29C	Air	40-120
2R-29C	Air	40-120
R-35A and B	Air	8-107 Note 1
R-60A through C	Air	1-107
R-66A through F	Air	Note 2

Notes:

1) 8°F is the lowest allowable outside air temperature that can be heated to the detector low limit of 32°F before reaching the detector.

2) The monitors are located inside the low level radwaste building which has no air conditioning. Ventilation fans change the building air once each hour; therefore, the medium conditions are assumed to be the same as the outdoor temperature conditions of -1°F to 107°F.

TABLE 11.4-2 (SHEET 1 OF 5)

Channel	Monitor	Monitor Type	Indicating Devices and Alarms	Detector Type	Sensitivity Range μCi/cm3	Principal Isotopes Monitored	Detector Energy Response	Process or Effluent Stream Flowrate	Sample Flowrate and Velocity	Service	Back Ground mr/h	Normal Activity	Maximum Activity	Alarms and Their Basis	Monitor Control Function
R-10	Penetration room filtration monitoring system	Air particulate	Display and control module and alarms on control room radiation monitoring panel	Scintillation detector	$1.0x10^{-12}$ to $1.0x10^{-6}$ *	Co-58 Co-60 Rb-88 I-131 I-133 Cs-134 Cs-137	85 kev to 2.28 mev-β	5000 cfm (Exh) 500 cfm (Recir)	2.0 cfm	Continuous	2.5	Normal operational penetration room activity	Post-LOCA penetration room activity	1. "OPER" fault 2. High radiation level 3. Test mode	None
R-11	Containment monitoring system	Air particulate	Display and control module and alarms on control room radiation monitoring panel	Scintillation detector	$1.0x10^{-12}$ to $1.0x10^{-6}$ *	Co-58 Co-60 Rb-88 I-131 I-133 Cs-134 Cs-137	85 kev to 2.28 mev-β	N/A	2.0 cfm	Continuous	2.5	Normal operational containment activity	Post-LOCA containment activity levels	1. "OPER" fault 2. High radiation level 3. Test mode	None
R-12	Containment monitoring system	Radioactive gases	Display and control module and alarms on control room radiation monitoring panel	Scintillation detector	$1.0x10^{-12}$ to $1.0x10^{-6}$ *	Ar-41 Kr-85 Xe-133 Xe-135	85 kev to 2.28 mev-β	N/A	2.0 cfm	Continuous	2.5	Normal operational containment activity	Post-LOCA containment activity levels	1. "OPER" fault 2. High radiation level 3. Test mode	None
R-13	Waste gas processing system	Radioactive gases	Log rate meter and alarms on controls room monitoring panel	Geiger-Mueller	$1.0x10^{-1}$ to $1.0x10^{+4}$ *	Kr-85 Kr-87 Xe-133 Xe-135	100 kev to 3.0 mev-γ 200 kev to 3.0 mev-β	40 scfm @ 110 psig 140^{o} F 2-in. SCH 40	10 cfm	Continuous	2.0		Maximum expected waste gas decay tank activity	1. Circuit failure 2. High radiation level 3. Test mode	None
R-15	Condenser air ejector monitoring system	Radioactive gases	Log rate meter and alarms on control room monitoring panel	Geiger-Mueller	$1.0x10^{-6}$ to $1.0x10^{-3}$ ($5x10^{-7}$ for Kr^{85})	Ar-41 Kr-85 Xe-133 Xe-135	60 kev to 3.0 mev-γ (Kr^{85}) 200 kev to 3.0 mev-β (A^{41})		10 cfm	Continuous	2.0	Normal condenser exhaust activity levels	Post steam generator tube rupture accident activity levels	1. Circuit failure 2. High radiation level 3. Test mode	None

TABLE 11.4-2 (SHEET 2 OF 5)

Channel	Monitor	Monitor Type	Indicating Devices and Alarms	Detector Type	Sensitivity Range $\mu Ci/cm^3$	Principal Isotopes Monitored	Detector Energy Response	Process or Effluent Stream Flowrate	Sample Flowrate and Velocity	Service	Back Ground mr/h	Normal Activity	Maximum Activity	Alarms and Their Basis	Monitor Control Function
R-15B	Condenser air ejector monitoring system	Radioactive gases	4-decade logarithmic scale meter and alarms on control room monitoring panel	Geiger-Mueller	$1.1x10^{-4}$ to $1.4x10^{-1}$	Ar-41 Kr-85 Xe-133 Xe-135	60 kev to 3.0 mev-γ (Kr^{85}) 200 kev to 3.0 mev-β (Ar^{41})		10 cfm	Continuous	2.0	Normal condenser exhaust activity levels	Post steam generator tube rupture accident activity levels	1. Circuit failure 2. High radiation level 3. Test mode	None
R-15C	Condenser air ejector monitoring system	Radioactive gases	5-decade logarithmic scale meter and alarms on control room monitoring panel	Pressurized ion chamber	$1.1x10^{-2}$ to $1.4x10^{3}$	Ar-41 Kr-85 Xe-133 Xe-135	60 kev to 3.0 mev-γ (Kr^{85}) 200 kev to 3.0 mev-β (Ar^{41})		10 cfm	Continuous	2.0	Normal condenser exhaust activity levels	Post steam generator tube rupture accident activity activity	1. Circuit failure 2. High radiation level 3. Test mode	None
R-17 (A-B)	Component cooling water monitoring system	Liquid	Log rate meter and alarms on control room monitoring panel	NaI scintillation detector	$1.0x10^{-5}$ to $1.0x10^{-2}$	Co-58 Co-60 I-131 I-133 Cs-134 Cs-137	100 kev to 3.0 mev-γ	3000	1-5 gpm	Continuous	2.0	Less than minimum detector sensitivity	Maximum expected CCW activity	1. Circuit failure 2. High radiation level 3. Test mode	Close CCW surge tank vent
R-18	Liquid waste processing monitoring system	Liquid	Log rate meter and alarms on control room monitoring panel	NaI scintillation detector	$1.0x10^{-5}$ to $1.0x10^{-2}$	Co-58 Co-60 I-131 I-133 Cs-134 Cs-137	100 kev to 3.0 mev-γ	Max 100 gpm at 200 ft head 2-in SCH 40	1-5 gpm	Continuous	2.0	Normal radioactive waste activity level	Anticipated operational occurrences radioactive waste activity level	1. Circuit failure 2. High radiation level 3. Test mode	Close RCV-018 on Hi-Radiation
R-19	Steam generator radiation monitoring system	Liquid	Log rate meter and alarms on control room monitoring panel	NaI scintillation detector	$1.0x10^{-5}$ to $1.0x10^{-2}$ *	Co-58 Co-60 I-131 I-133 Cs-134 Cs-137	100 kev to 3.0 mev-γ	1-5 gpm	1-5 gpm	Continuous	2.0	Normal steam generator activity	Post steam generator tube rupture accident activity level	1. Circuit failure 2. High radiation level 3. Test mode	Isolate steam generator sample lines

TABLE 11.4-2 (SHEET 3 OF 5)

Channel	Monitor	Monitor Type	Indicating Devices and Alarms	Detector Type	Sensitivity Range μCi/cm3	Principal Isotopes Monitored	Detector Energy Response	Process or Effluent Stream Flowrate	Sample Flowrate and Velocity	Service	Back Ground mr/h	Normal Activity	Maximum Activity	Alarms and Their Basis	Monitor Control Function
R-20 (A-B)	Service water	Liquid	Log rate meter and alarms on control room radiation monitoring panel	NaI scintillation detector	$1.0x10^{-5}$ to $1.0x10^{-2}$ *	Co-58 Co-60 I-131 I-133 Cs-134 Cs-137	100 kev to 3.0 mev-ϒ	1600 to 4000 gpm	1-5 gpm	Continuous	2.0			1. Circuit failure 2. High radiation level 3. Test mode	None 55 55
-23 (A-B)	Steam generator blowdown processing monitoring system	Liquid	Log rate meter and alarms on control room radiation monitoring panel	NaI scintillation detector	$1.0x10^{-6}$ to $1.0x10^{-3}$ *	Co-58 Co-60 I-131 I-133 Cs-134 Cs-137	100 kev to 3.0 mev-ϒ	15 to 37.5 gpm 2” and 3” SCH 40	1-5 gpm	Continuous	1.0	Normal steam generator activity <1.0x10 μc/cc	Post steam generator tube rupture accident activity level	1. Circuit failure 2. High radiation level meeting low as practicable 10 CFR 50 for continuous discharge 3. Test mode	Isolate steam generator blowdown process system and discharge line on high alarm
R-24 (A-B)	Containment purge monitoring system	Gaseous	Log rate meter and alarms on control room radiation monitoring panel	Beta scintillation detector	$1.0x10^{-6}$ to $1.0x10^{-3}$ *	Ar-41 Kr-85 Xe-133 Xe-135	200 kev to 3.0 mev-β	50,000 cfm 25,000 cfm	10 cfm	Continuous	2.0	Normal operational containment activity	Post-LOCA containment activity levels	1. Circuit failure 2. High radiation level 3. Test mode	Containment purge vent isolation
R-25 (A-B)	Fuel handling monitoring system	Gaseous	Log rate meter and alarms on control room radiation monitoring panel	Beta scintillation detector	$1.0x10^{-6}$ to $1.0x10^{-3}$ *	Ar-41 Kr-85 Xe-133 Xe-135	200 kev to 3.0 mev-β	20,000 cfm	10 cfm	Continuous	2.0	Normal operational auxiliary building vent exhaust activity levels	Post fuel handling accident activity levels	1. Circuit failure 2. High radiation level 3. Test mode	Fuel handling area vent isolation
R-29B[a]	Plant vent stack effluent monitors, wide range gas	Gaseous	Display and control module and alarms on control room radiation monitoring panel	Beta scintillation/photo multiplier detector (1); gamma cadmium telluride solid-state detector (2)	$1.0x10^{-4}$ to $1.0x10^{5}$ *	Ar-41 Kr-85 Xe-133 Xe-135	85 kev to 2.28 mev-β 75 kev to 2.4 mev-ϒ	150,000 cfm (maximum) 5000 cfm (minimum)	Normal Skid Flow 2.5 cfm Accident Skid Flow 0.06 cfm	Continuous	2.0	Normal operational auxiliary building vent exhaust activity levels	Post-LOCA containment activity levels	1. Oper failure 2. High radiation level 3. Test mode	Closure of gas release valve in waste gas processing system

TABLE 11.4-2 (SHEET 4 OF 5)

Channel	Monitor	Monitor Type	Indicating Devices and Alarms	Detector Type	Sensitivity Range μCi/cm3	Principal Isotopes Monitored	Detector Energy Response	Process or Effluent Stream Flowrate	Sample Flowrate and Velocity	Service	Back Ground mr/h	Normal Activity	Maximum Activity	Alarms and Their Basis	Monitor Control Function
1R-29C[b1]	Plant vent stack monitoring system	Particulate/ iodine/low range gas	Display and control module and alarms on control room radiation monitoring panel	Beta scintillation/photo multiplier detector (2); NaI crystal/photomulti plier detector (1)	1.0×10^{-9} to 1.0×10^{-6} (Particulate) 1.0×10^{-10} to 1.0×10^{-6} (iodine) 5.0×10^{-7} to 1.0×10^{-4} (Noble Gas)*	Co-58 Co-60 Rb-88 I-131 I-133 Cs-134 Cs-137 Xe-133 Xe-135 Kr-85 Ar-41	85 kev to 2.28 mev-β 329 kev to 407 mev-γ	150,000 cfm (maximum) 5000 cfm (minimum)	2.5 cfm	Continuous	2.0	Normal operational auxiliary building vent exhaust activity levels		1. Oper failure 2. High radiation level 3. Test mode	None
2R-29C[b2]	Plant vent stack monitoring system	Particulate/ iodine/low range gas	Display and control module and alarms on control room radiation monitoring panel	Beta scintillation/photo multiplier detector (2); NaI crystal/photomulti plier detector (1)	1.0×10^{-9} to 1.0×10^{-6} (Particulate) 1.0×10^{-10} to 1.0×10^{-6} (iodine) 5.0×10^{-7} to 1.0×10^{-4} (Noble Gas)*	Co-58 Co-60 Rb-88 I-131 I-133 Cs-134 Cs-137 Xe-133 Xe-135 Kr-85 Ar-41	85 kev to 2.28 mev-β 329 kev to 407 mev-γ	150,000 cfm (maximum) 5000 cfm (minimum)	2.5 cfm	Continuous	2.0	Normal operational auxiliary building vent exhaust activity levels		1. Oper failure 2. High radiation level 3. Test mode	None
R-35 (A-B)	Control room makeup air inlet	Gaseous	Log rate meter and alarms on control room radiation monitoring panel	Beta scintillation detector	1.0×10^{-6} to 1.0×10^{-3} *	Ar-41 Kr-85 Xe-133 Xe-135	200 kev to 3.0 mev-β	20,000 cfm	10 cfm	Continuous	2.0	Normal operational outside air activity	< 800 cpm	1. Circuit failure 2. High radiation level 3. Test mode	Control room/TSC ventilation isolation
R-60 (A-C)	Main steam line monitors	Gaseous	LCD display with scale meter and alarm indications and alarms on control room monitoring panel	NaI(TI) scintillation detector	1.0×10^{-2} to 1.0×10^{4} *	Ar-41 Kr-85 I-131 I-133 Xe-133 Xe-135	80 kev to 3.0 mev-γ			Continuous	2.0	Normal operational main steam line activity levels	Post steam generated tube rupture accident activity levels	1. Oper failure 2. High radiation level 3. Test mode	None

TABLE 11.4-2 (SHEET 5 OF 5)

Channel	Monitor	Monitor Type	Indicating Devices and Alarms	Detector Type	Sensitivity Range µCi/cm3	Principal Isotopes Monitored	Detector Energy Response	Process or Effluent Stream Flowrate	Sample Flowrate and Velocity	Service	Back Ground mr/h	Normal Activity	Maximum Activity	Alarms and Their Basis	Monitor Control Function

(a) This refers to monitor R-29B. Channels R-29E (Low), R-29F (Mid), R-29G (high), and R-29B - Composite Gas.

(b1) This refers to Monitor Unit 1 R-29C. Channels R-29H (Particulate), R-29I (Iodine), and R-29J (Low Gas).

(b2) This refers to Monitor Unit 2 R-29C. Channels R-29H (Particulate), R-29I (Iodine), and R-29J (Low Gas).

* range not for all isotopes

TABLE 11.4-3 (SHEET 1 OF 2)

PROCESS MONITOR ALARM SETPOINTS

Channel	Monitor	Alarm Level[a]
R-10	Penetration room filtration air particulate	1420 cpm
R-11	Containment air particulate	1420 cpm
R-12	Containment radioactive gas	320 cpm
R-13	Waste gas processing	18,000 cpm
R-15	Condenser air ejector	3000 cpm
R-15B	Condenser air ejector, intermediate range	4.5 mR/h
R-15C	Condenser air ejector, high range	50 mR/h
R-17A and B	Component cooling liquid	320 cpm
R-18	Waste processing liquid effluent	320 cpm
R-19	Steam generator liquid sample	320 cpm
R-20A and B	Service water liquid	320 cpm
R-23A and B	Steam generator blowdown processing system liquid	320 cpm
R-24A and B	Containment purge monitor	13,000 cpm
R-25A and B	Spent fuel pool exhaust flow gas monitor	55,000 cpm
R-29B[b]	Plant vent stack effluent monitors, wide-range gas	R-29E - 4.44×10^{-4} μCi/cc R-29B - Composite Gas 1.31×10^{-4} μCi/cc
1R-29C[c1]	Plant vent stack monitor particulate/iodine/low-range gas	R-29H - 6.7×10^{-8} μCi/cc R-29J - 1.31×10^{-4} μCi/cc
2R-29C[c2]	Plant vent stack monitor particulate/iodine/low-range gas	R-29H - 6.7×10^{-8} μCi/cc R-29J - 1.31×10^{-4} μCi/cc
R-60A through C	Main steam line monitors	7×10^{-2} μCi/cc

TABLE 11.4-3 (SHEET 2 OF 2)

PROCESS MONITOR ALARM SETPOINTS

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a. These are typical values. Actual values are contained in the plant Technical Specifications, the Technical Requirements Manual, and/or the plant radiological control and protection procedures.

b. This refers to Monitor R-29B, Channels R-29E and R-29B - Composite Gas.

c1. This refers to Unit 1 Monitor R-29C, Channels R-29H and R-29J.

c2. This refers to Unit 2 Monitor R-29C, Channels R-29H and R-29J.

TABLE 11.4-4 (SHEET 1 OF 5)

RADIOLOGICAL ANALYSIS SUMMARY OF LIQUID PROCESS SAMPLES

Sampling Location	Sampling[a] Frequency	Analysis Performed	MDC[b] ($\mu Ci/cm^3$)	Reason for Analysis
Reactor Coolant System				
Hot leg reactor coolant	Weekly	Gamma spectrometric	10^{-6}	Evaluate integrity of fuel cladding and monitor levels of fission and activated corrosion products.
	Weekly	Tritium	10^{-5}	
	Monthly	Gross ∝	10^{-6}	
	Weekly	Selected noble gases	10^{-5}	
	Monthly	Activated corrosion products	-	Determine purification requirements.
	Biweekly	Dose equivalent I-131	10^{-6}	
	Semiannually	Average energy $\overline{E}$	-	Determine limits for activity in the reactor coolant system.
Chemical and Volume Control System				
Upstream of mixed bed demineralizer	Monthly	Gamma spectrometric	-	Evaluate performance of letdown purification demineralizers.
Downstream of mixed bed and cation bed demineralizers[c]	Weekly	Gamma spectrometric	-	Evaluate performance of letdown purification demineralizers.
	Monthly	Gamma spectrometric	-	
Holdup tanks	Each batch	Gamma spectrometric	10^{-6}	Determine purification requirements of waste water. Evaluate performance of recycle evaporator feed demineralizer.
	Monthly	Gamma spectrometric	10^{-5}	

TABLE 11.4-4 (SHEET 2 OF 5)

Sampling Location	Sampling[a] Frequency	Analysis Performed	MDC[b] ($\mu Ci/cm^3$)	Reason for Analysis
Downstream recycle evaporator feed demineralizers	Each batch	Gamma spectrometric	10^{-6}	Determine purification requirements of waste water. Evaluate performance of recycle evaporator feed demineralizer.
	Monthly	Gamma spectrometric	10^{-5}	
Downstream recycle/holdup tanks	Each batch	Gamma spectrometric	10^{-6}	Evaluate performance of recycle evaporator.
	Monthly	Gamma spectrometric	10^{-5}	
Recycle evaporator distillate	Each batch	Gamma spectrometric	10^{-6}	Evaluate performance of recycle evaporator.
	Monthly	Gamma spectrometric	10^{-5}	Evaluate performance of recycle evaporator.
Downstream recycle evaporator condensate demineralizer	Each batch	Gamma spectrometric	10^{-6}	Evaluate performance of recycle evaporator condensate demineralizer
	Monthly	Gamma spectrometric	10^{-5}	
Recycle evaporator concentrates	Each batch	Gamma spectrometric	10^{-6}	Determine whether to recycle or solidify.
Safety Injection System				
Refueling water storage tank	Monthly	Gamma spectrometric	10^{-6}	General monitoring.
	Monthly	Tritium	10^{-5}	Evaluate in-plant buildup of tritium.
Accumulator tanks	Monthly	Gamma spectrometric	10^{-6}	General monitoring.
Component Cooling Water System				
Downstream of component cooling water pumps	Weekly	Gamma spectrometric	10^{-6}	Check on component cooling water system radiation monitors.

TABLE 11.4-4 (SHEET 3 OF 5)

Sampling Location	Sampling[(a)] Frequency	Analysis Performed	MDC[(b)] ($\mu Ci/cm^3$)	Reason for Analysis
Service Water System				
Downstream of component cooling water heat exchangers	Weekly	Gamma spectrometric	10^{-6}	Monitor inleakage from component cooling water system.
Residual Heat Removal System	Once/72 h	Gross $\beta\gamma$	10^{-6}	Evaluate integrity of fuel cladding and monitor levels of fission and activated corrosion products.
	Once/72 h	Gamma spectrometric	10^{-6}	
Recyclable (Channel A) Waste Treatment System				
Reactor coolant drain tank	Each batch	Gamma spectrometric	10^{-6}	Determine whether to recycle or process.
Downstream waste holdup tank	Each batch	Gamma spectrometric	10^{-6}	Determine processing requirements
	Monthly	Gamma spectrometric	10^{-5}	Evaluate performance of waste evaporator.
Waste evaporator distillate	Each batch	Gamma spectrometric	10^{-6}	Evaluate performance of waste evaporator.
	Monthly	Gamma spectrometric	10^{-5}	
Downstream waste evaporator distillate demineralizer	Each batch	Gross $\beta\gamma$	10^{-6}	Evaluate performance of waste evaporator condensate demineralizer
	Monthly	Gamma spectrometric	10^{-5}	
Waste condensate tank	Each batch	Gamma spectrometric	10^{-6}	Determine disposition of processed condensate
Waste evaporator concentrates	Each batch	Gamma spectrometric	10^{-6}	Determine whether to return to boron recycle system or to solid waste disposal.
	Monthly	Gamma spectrometric	10^{-5}	

TABLE 11.4-4 (SHEET 4 OF 5)

Sampling Location	Sampling[a] Frequency	Analysis Performed	MDC[b] (μCi/cm^3)	Reason for Analysis
Nonreactor Grade (Channel B) Waste Treatment System				
Laundry and hot shower tank	Each batch	Gamma spectrometric	10^{-6}	Determine whether to process or discharge.
Floor drain tank	Each batch	Gamma spectrometric	10^{-6}	Determine whether to process or discharge.
Downstream waste monitor tank demineralizer	Each batch	Gamma spectrometric	10^{-6}	Evaluate performance of demineralizer.
Waste monitor tanks	Each batch	Gamma spectrometric	5×10^{-7}	Determine whether to reprocess or discharge.
Primary Makeup Water System				
Primary water storage tank	Monthly	Gamma spectrometric	10^{-6}	Evaluation of systems for recycling waste water.
	Monthly	Tritium	10^{-5}	Evaluate in-plant buildup of tritium.
Spent Fuel Pool Cooling and Demineralizer System				
Spent fuel pool water	Biweekly	Gamma spectrometric	10^{-6}	Determine purification requirements and evaluate leakage from spent fuel.
	Monthly	Tritium	10^{-5}	Evaluate in-plant buildup of tritium.
Upstream fuel pool demineralizer	Monthly	Gamma spectrometric	10^{-6}	Evaluate performance of demineralizer.
	Monthly	Gamma spectrometric	10^{-5}	Evaluate performance of demineralizer.
Downstream fuel pool demineralizer	Monthly	Gamma spectrometric	10^{-6}	Evaluate performance of demineralizer.
	Monthly	Gamma spectrometric	10^{-5}	Evaluate performance of demineralizer.
Condensate and Feedwater System				
Steam generator feedwater	Periodically	Gamma spectrometric	10^{-7}	Determine radionuclide carryover in main

TABLE 11.4-4 (SHEET 5 OF 5)

Sampling Location	Sampling[a] Frequency	Analysis Performed	MDC[b] ($\mu Ci/cm^3$)	Reason for Analysis
				steam.
	Periodically	Gamma spectrometric	5×10^{-7}	Determine radionuclide carryover in main steam.
Steam Generator Blowdown System				
Downstream of steam generator ion exchangers when in use	5 days/week	Gamma spectrometric	10^{-7}	Evaluate performance of demineralizer.
	Weekly	Gamma spectrometric	5×10^{-7}	Evaluate performance of demineralizer.

a. These are typical frequencies. Actual frequencies are contained in the plant Technical Specifications and the plant operating procedures.

b. For principle γ emitters as per STS.

c. When in use.

TABLE 11.4-5

RADIOLOGICAL ANALYSIS SUMMARY OF GAS PROCESS SAMPLES

Sampling Location	Sampling Frequency[(a)]	Analysis Performed	MDC ($\mu Ci/cm^3$)	Purpose of Sample
Chemical volume and control system holdup tanks vapor space	Weekly	Noble gas isotopic	10^{-4}	Evaluate tank leakage; determine whether to purge to GCH.
Containment atmosphere	Prior to personnel entry	Noble gas isotopic	10^{-7}	Personnel radiation protection; determine necessity for operation of cleanup recirculation units and/or containment purge systems.
		Halogens	10^{-9}	
		Air particulate	10^{-9}	
		Tritium	10^{-6}	

a. These are typical frequencies. Actual frequencies are contained in plant operating procedures.

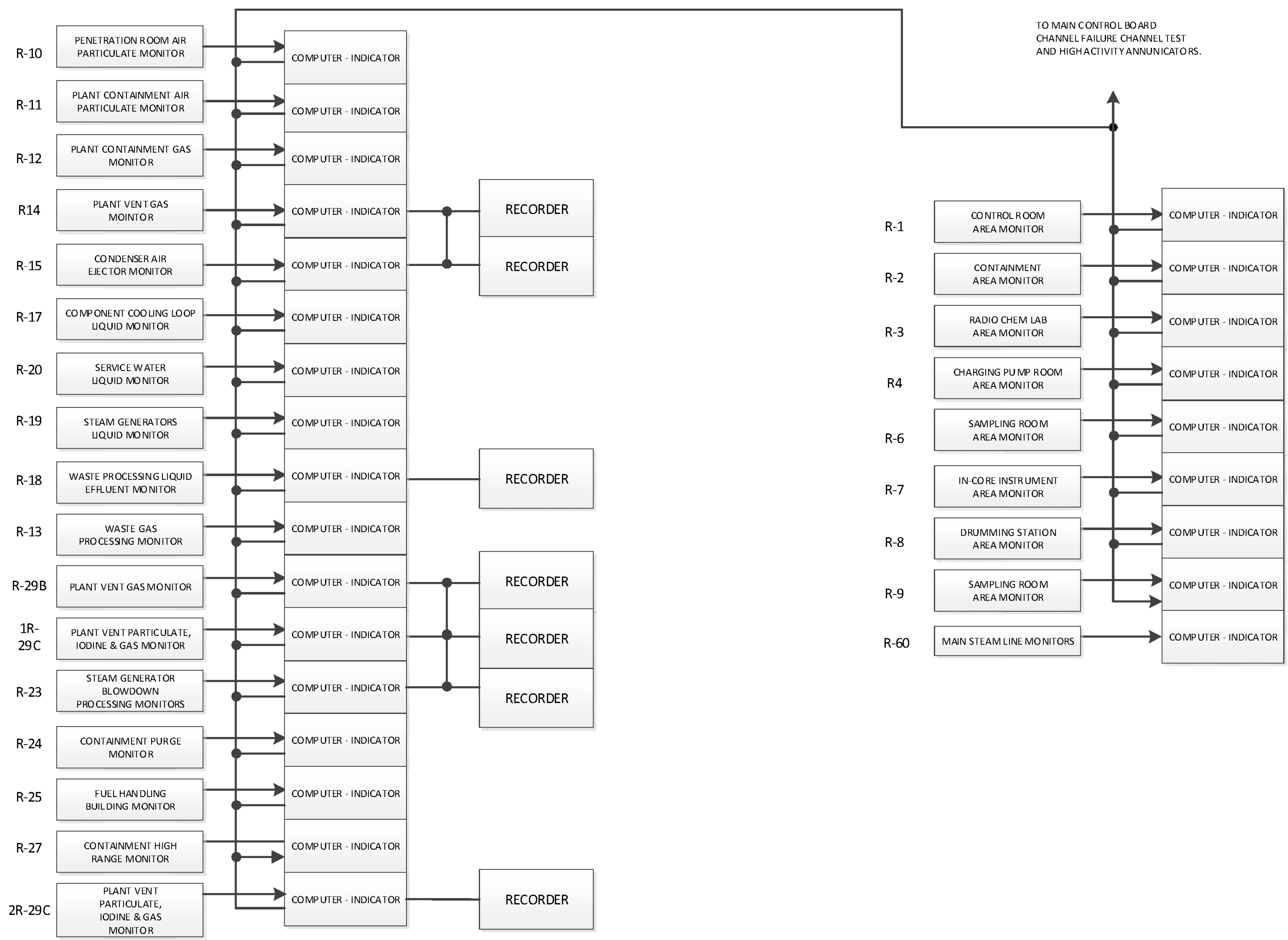
TABLE 11.4-6 (SHEET 1 OF 2)

EFFLUENT SAMPLE AND ANALYSIS SCHEDULE

Effluent Stream	Sample Frequency	Minimum Analysis Frequency	Type of Activity Analysis Performed	Minimum Detectable Concentration (MDC) (μCi/ml)
Radioactive waste processing system discharges (Batch Waste Release Tanks)	Each batch prior to discharge	Each batch prior to discharge	Principal gamma emitters I-131	5×10^{-7} 1×10^{-6}
	One batch/month prior to discharge	Monthly	Dissolved gases (gamma emitters)	1×10^{-5}
	Each batch prior to discharge	Monthly composite	H-3 Gross alpha	1×10^{-5} 1×10^{-7}
	Each batch prior to discharge	Quarterly composite	Sr-89, Sr-90, Fe-55	5×10^{-8} 1×10^{-6}
Steam generator blowdown processing system discharge	Daily grab sample	Weekly composite	Principal gamma emitters I-131	5×10^{-7} 1×10^{-6}
	Monthly grab sample	Monthly	Dissolved gases (gamma emitters)	1×10^{-5}
	Daily grab sample	Monthly composite	H-3 Gross alpha	1×10^{-5} 1×10^{-7}
	Daily grab sample	Quarterly composite	Sr-89, Sr-90, Fe-55	5×10^{-8} 1×10^{-6}
Condenser Blowdown	Daily grab sample During discharge	Weekly composite	Principal gamma emitters H-3	5×10^{-7} 1×10^{-5}
Turbine building sump	Grab sample prior to or during release	Weekly composite	Principal gamma emitters H-3	5×10^{-7} 1×10^{-5}

TABLE 11.4-6 (SHEET 2 OF 2)

Effluent Stream	Sample Frequency	Minimum Analysis Frequency	Type of Activity Analysis Performed	Minimum Detectable Concentration (MDC) (μCi/ml)
Waste gas storage tank releases	Grab sample each tank prior to release	Each tank prior to release	Principal gamma emitters	1×10^{-4}
Containment purge	Grab sample each purge prior to release	Prior to each purge	Principal gamma emitters H-3	1×10^{-4} 1×10^{-6}
Condenser steam jet air ejector discharge	Monthly grab samples	Monthly	Principal gamma emitters H-3	1×10^{-4} 1×10^{-6}
Plant vent stack	Monthly grab samples	Monthly	Principal gamma emitters H-3	1×10^{-4} 1×10^{-6}
Plant vent stack, Containment purge	Continuous (charcoal)	Weekly charcoal sample	I-131 I-133	1×10^{-12} 1×10^{-10}
	Continuous	Weekly particulate sample	Principal gamma emitters (I-131 and others)	1×10^{-11}
	Continuous	Monthly composite particulate sample	Gross alpha	1×10^{-11}
	Continuous	Quarterly composite particulate sample	Sr-89, Sr-90	1×10^{-11}
	Continuous	Noble gas monitor	Noble gases Gross Beta & Gamma	1×10^{-6}



REV 28 10/18

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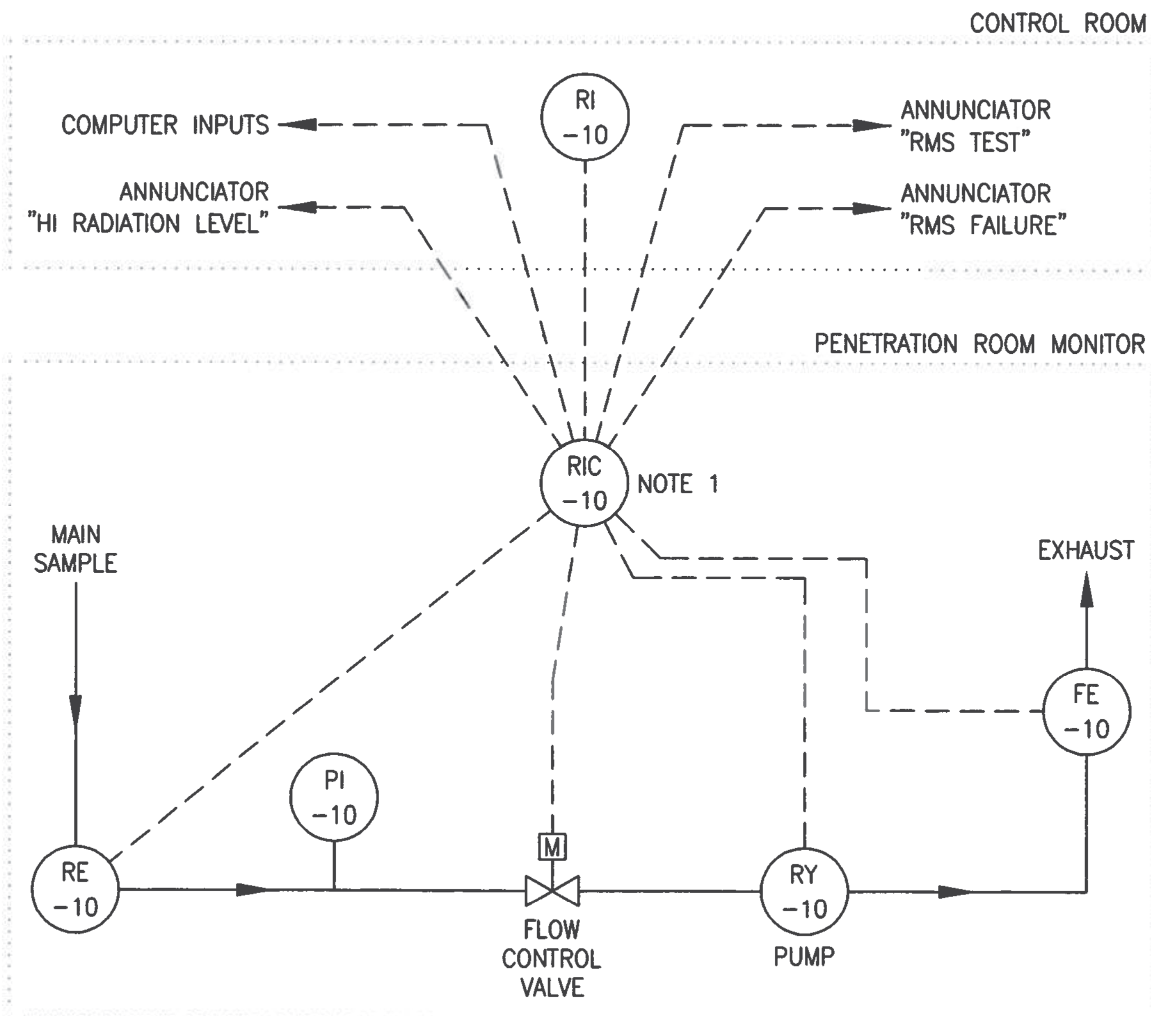
JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

RADIATION MONITORING SYSTEM

FIGURE 11.4-1



NOTES:
1. RIC-10 PROVIDES DETECTOR DATA PROCESSING, CONTROL OF THE PUMP, & FLOW CONTROL.

REV 23 5/11

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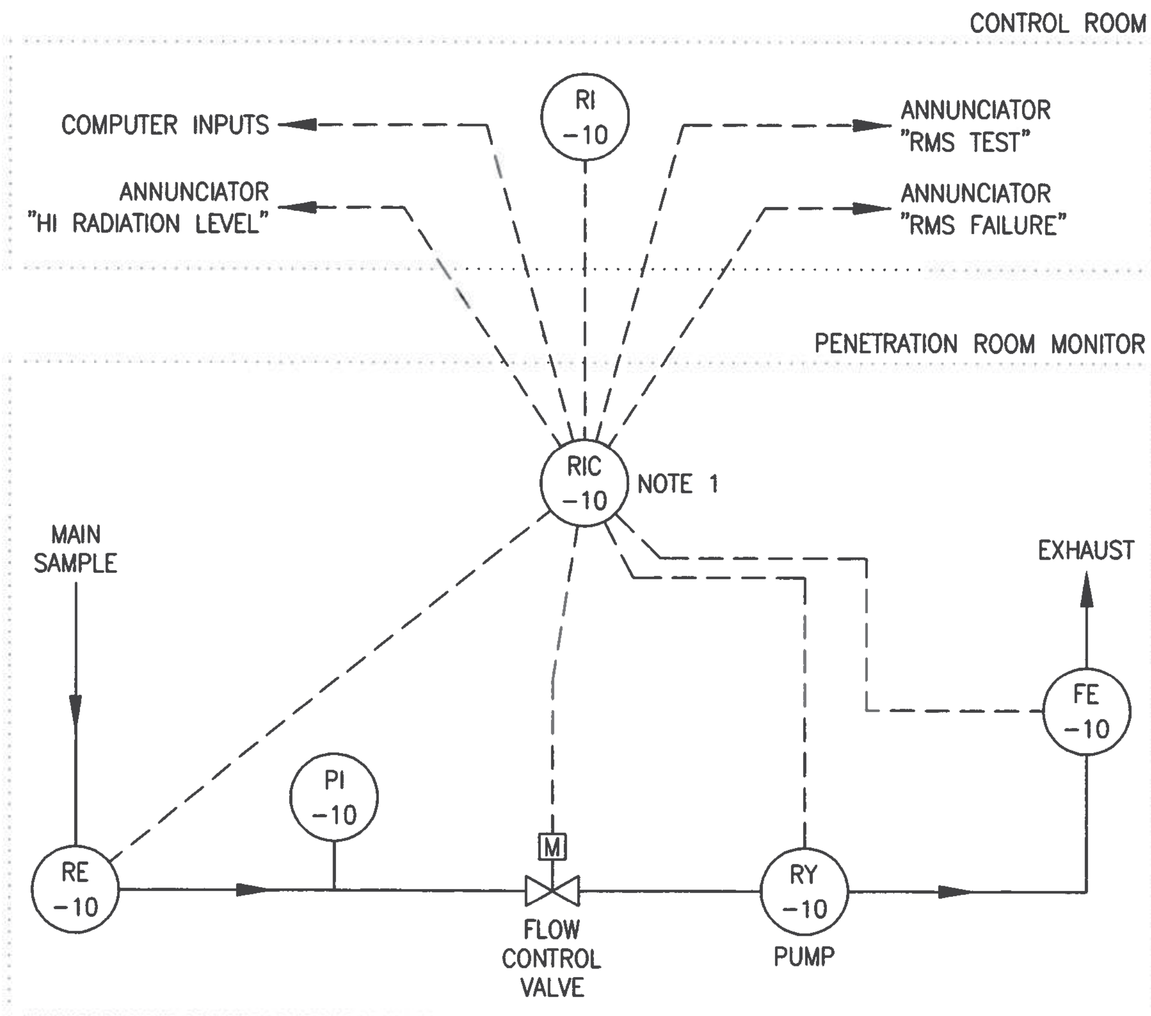
JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

RADIATION MONITORING SYSTEM
AREA RANGE R-10

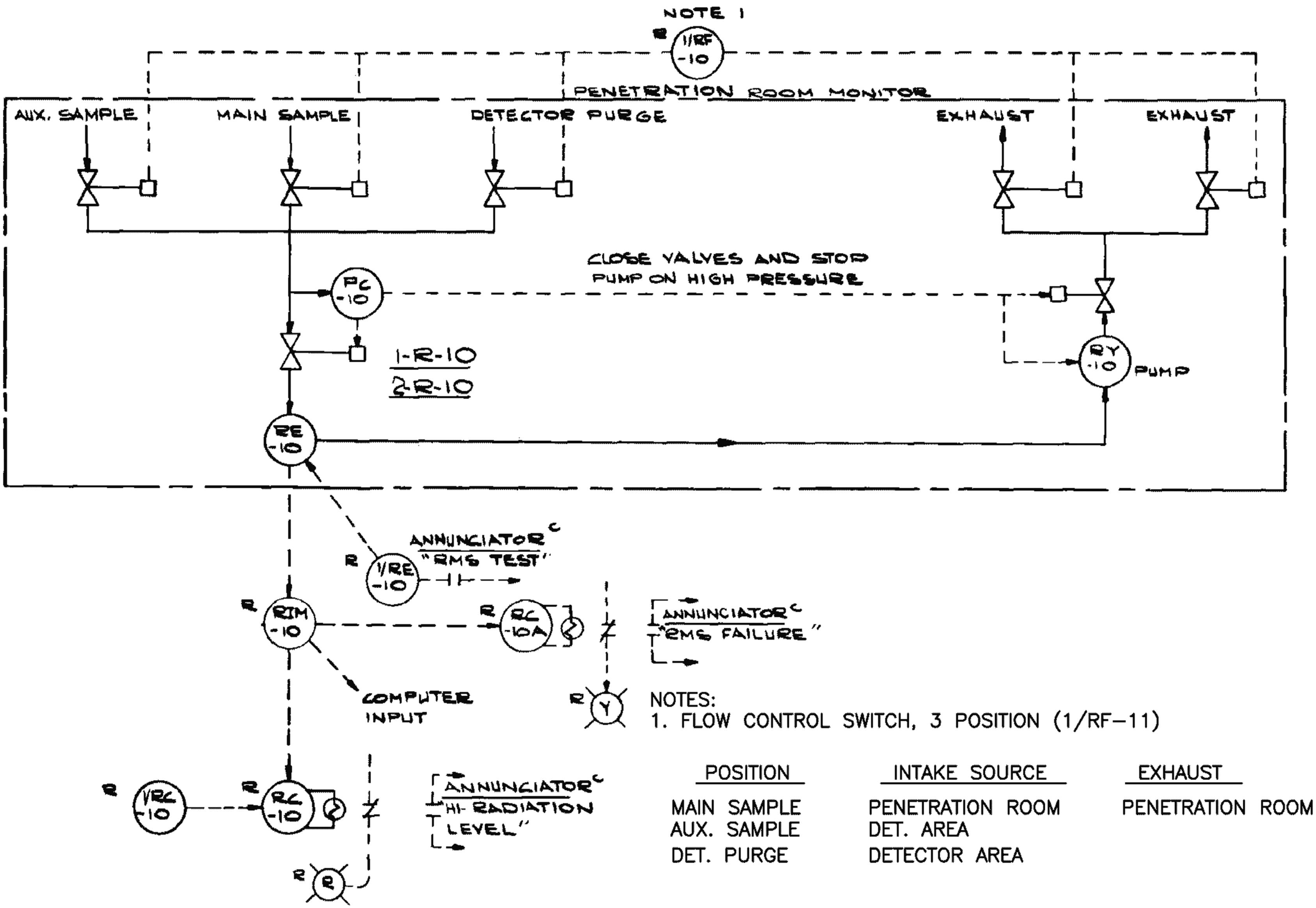
FIGURE 11.4-2



NOTES:
1. RIC-10 PROVIDES DETECTOR DATA PROCESSING, CONTROL OF THE PUMP, & FLOW CONTROL.

REV 23 5/11

SOUTHERN COMPANY *Energy to Serve Your World®*	JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1	RADIATION MONITORING SYSTEM AREA RANGE R-10
		FIGURE 11.4-2a



NOTES:
1. FLOW CONTROL SWITCH, 3 POSITION (1/RF–11)

POSITION	INTAKE SOURCE	EXHAUST
MAIN SAMPLE	PENETRATION ROOM	PENETRATION ROOM
AUX. SAMPLE	DET. AREA	
DET. PURGE	DETECTOR AREA	

REV 23 5/11

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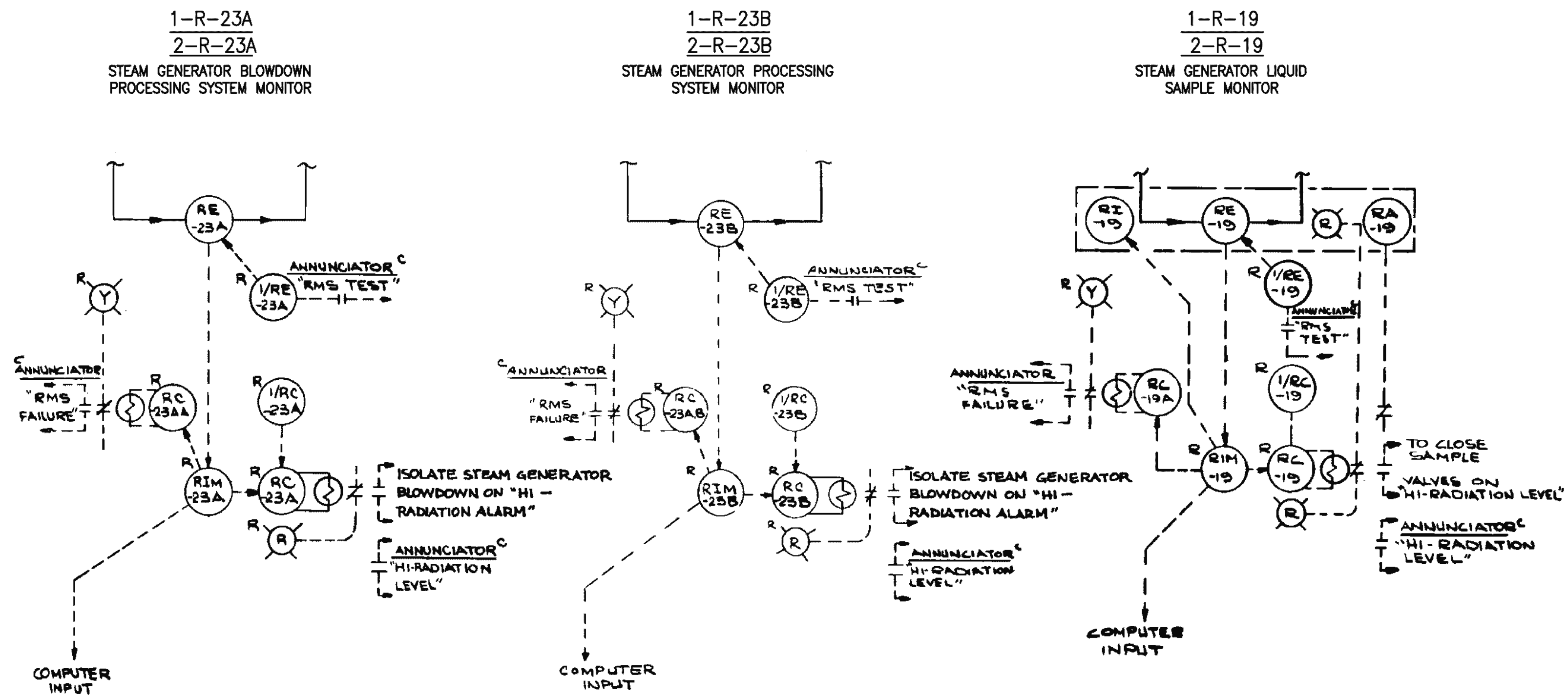
JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 2

RADIATION MONITORING SYSTEM
AREA RANGE R-10

FIGURE 11.4-2b



REV 21 5/08

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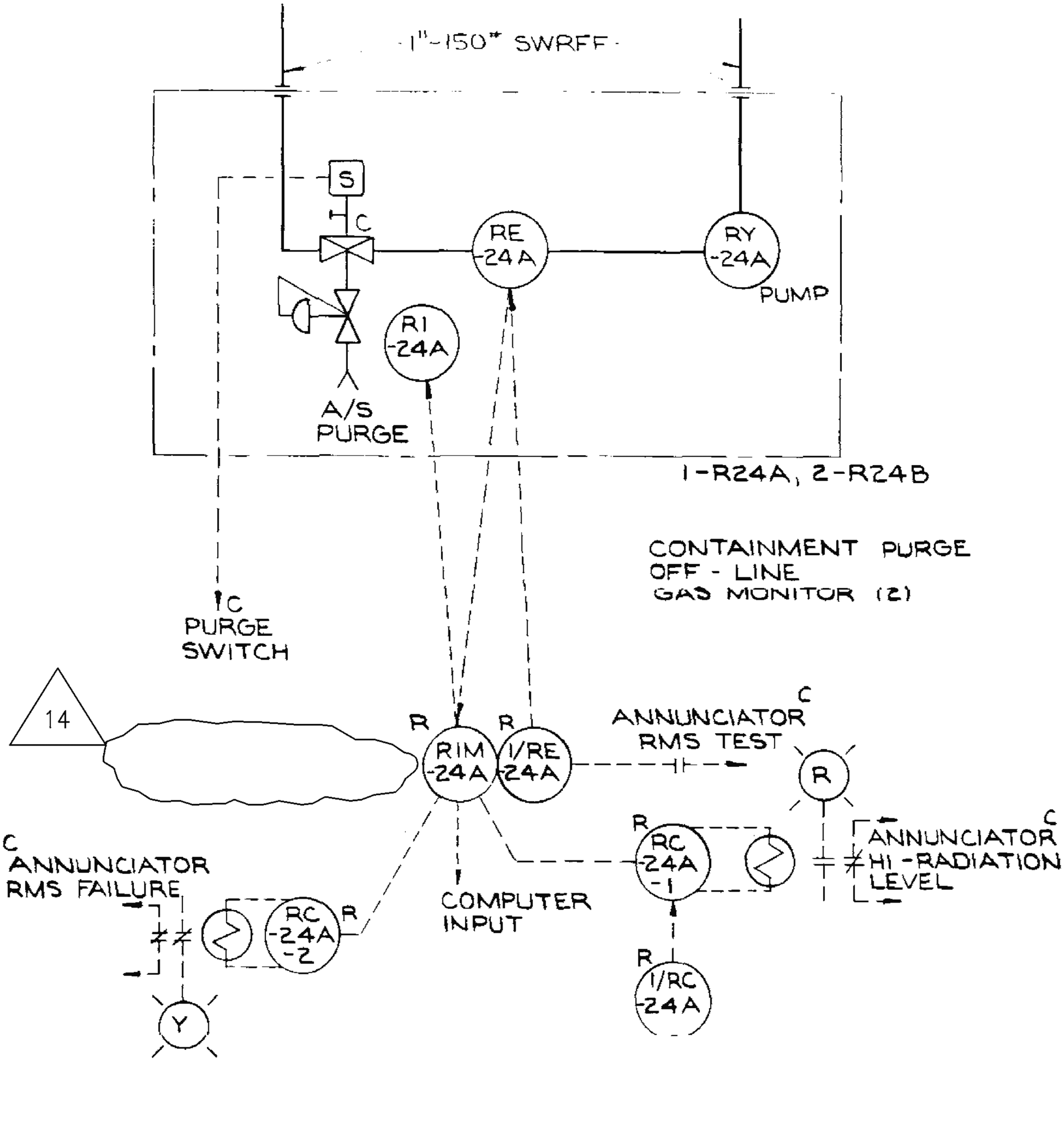
JOSEPH M. FARLEY NUCLEAR PLANT UNIT 1 AND UNIT 2

RADIATION MONITORING SYSTEM
AREA RANGE R-19, R-23A, AND R-23B

FIGURE 11.4-3



REV 21 5/08

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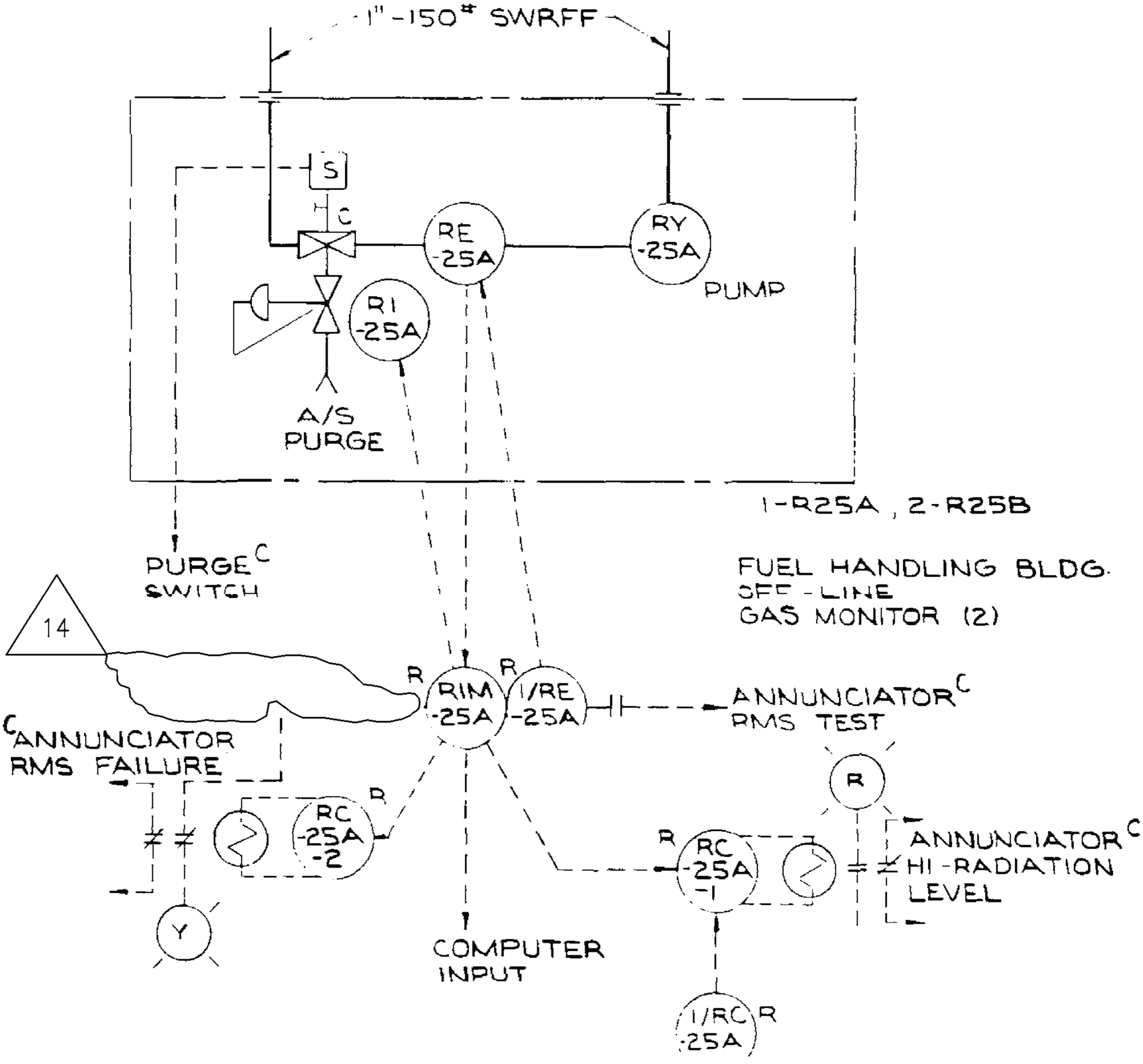
JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

RADIATION MONITORING SYSTEM
FUNCTIONAL BLOCK DIAGRAM (R-24)
CONTAINMENT PURGE MONITOR

FIGURE 11.4-4



1"-150# SWRFF
S
C
RE
-25A
RY
-25A
PUMP
RI
-25A
A/S
PURGE
1-R25A , 2-R25B
PURGE C
SWITCH
FUEL HANDLING BLDG.
OFF-LINE
GAS MONITOR (2)
14
R
RIM
-25A
R
1/RE
-25A
ANNUNCIATOR C
RMS TEST
C ANNUNCIATOR
RMS FAILURE
R
RC
-25A
-2
R
Y
R
RC
-25A
-1
ANNUNCIATOR C
HI-RADIATION
LEVEL
COMPUTER
INPUT
1/RC R
-25A

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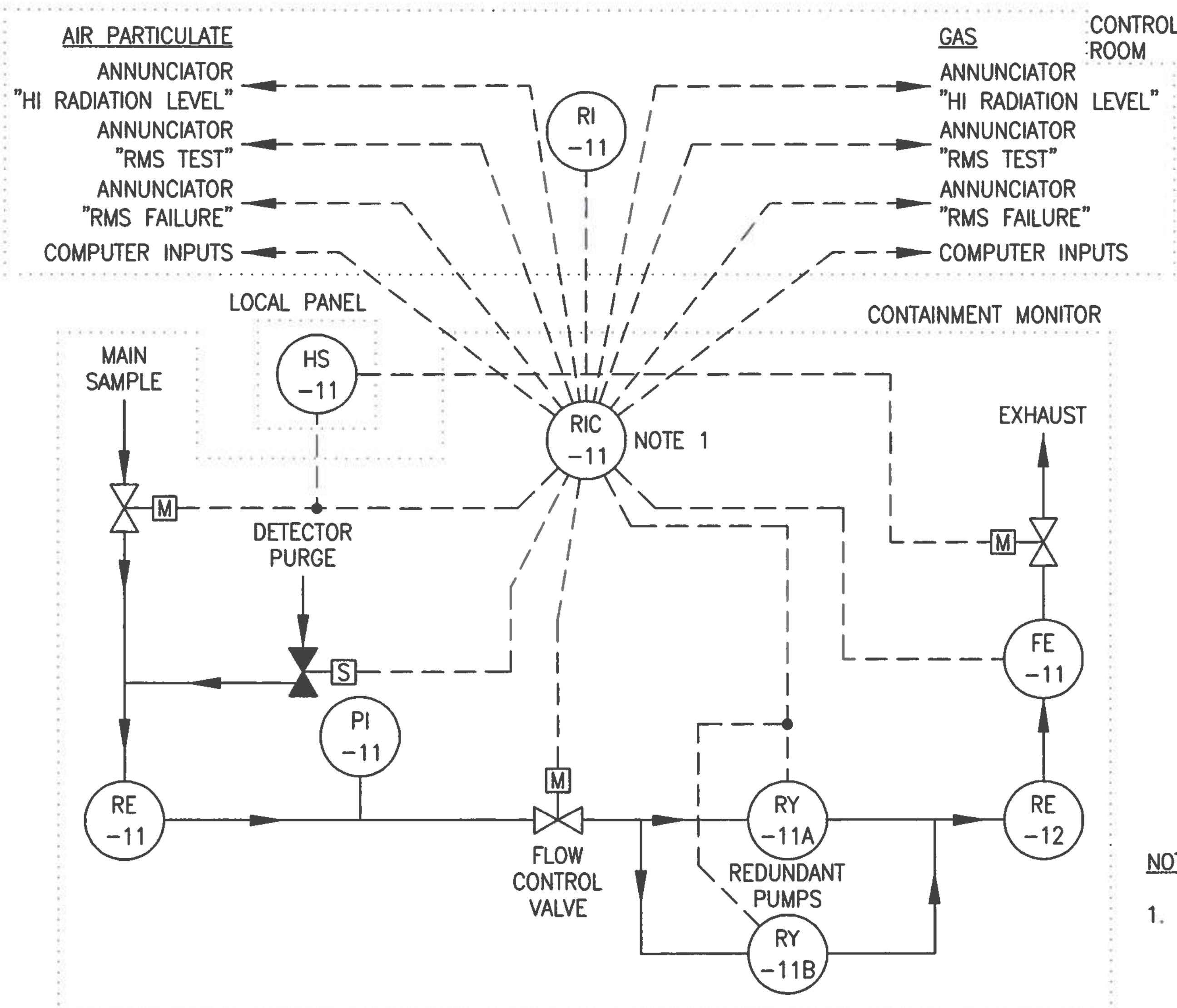

NOTES:

1. RIC-11 PROVIDES DETECTOR DATA PROCESSING, CONTROL OF THE PUMPS, FLOW CONTROL, & PURGE VALVE OPERATION.

REV 23 5/11

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JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 1 AND UNIT 2

RADIATION MONITORING SYSTEM
AREA RANGE R-11 AND R-12

FIGURE 11.4-6

11.5 SOLID WASTE SYSTEM

11.5.1 DESIGN OBJECTIVES

The solid waste system is designed to transfer spent resins, evaporator concentrates, and chemical tank effluents. This system is installed in Unit 1 and has adequate capacity to serve both units.

To provide more efficient solidification and to ensure compliance with current burial ground license requirements (including volume restrictions), provision has been made for the use of a portable cement solidification system. The portable system is operated in the solidification and dewatering facility outside the Unit 1 and Unit 2 auxiliary building and is capable of solidifying resins, evaporator concentrates, and chemical drains from both units. The system also serves as a solidification system for the disposable demineralizer system, should solidification be required prior to shipment.

A separate system is available to compact dry active wastes such as paper, disposable clothing, rags, towels, floor coverings, shoe covers, plastics, cloth smears, and respirator filters.

It is estimated that about 15,000 ft^3 of solid waste are produced for burial each year which includes dry active waste (DAW).

The systems are used to package radioactive wastes within the limitations specified by 10 CFR 20, 10 CFR 61, 10 CFR 71, and 49 CFR 170 through 178. Bulk waste may be shipped to a licensed waste processor or to a disposal facility without encapsulation or solidification in accordance with these regulations and per applicable license and regulations for the receiver of the waste.

Shielding is designed to limit general area radiation levels in the drumming rooms, Liquid Radwaste Processing System (LRWPS), and the low level radwaste building.

During normal work activities, tools, scrap, and other miscellaneous equipment and materials may become radioactively contaminated. The Solidification/Dewatering Facility (SDF) can be used as a decontamination area when needed. Appropriate administrative controls as determined by FNP personnel will be reviewed and implemented to ensure compliance with regulatory guidelines if used as a decontamination area.

Solidification via the portable system is accomplished with the liner inside a shipping cask or a shielded enclosure in the solidification and dewatering facility, which provides the necessary personnel shielding.

11.5.2 SYSTEM INPUTS

Input to the solid waste system comes from the spent resin storage tanks, waste evaporator, the Liquid Radwaste Processing System (LRWPS), and chemical drain tank. Solid, compressible wastes are products of the plant operation and maintenance. Input points are identified in figure 11.5-1.

Volumes and isotopic inventories are discussed in subsection 11.5.4.

11.5.3 EQUIPMENT DESCRIPTION

11.5.3.1 Processing System Design

The process flow diagram for the in-plant solid waste system is shown in figure 11.5-1. The solid waste system is designed to package all solid wastes in standard drums in Unit 1 and Unit 2 for removal to disposal facilities. In addition, the system has been designed to permit bulk shipment of wastes by transfer to a disposal liner at the SDF. The solidification and dewatering facility consists of a building with shielded pits and process lines located east of the Unit 1 and Unit 2 auxiliary building. This facility is capable of receiving waste from either unit. Bulk shipment will be performed by an approved carrier and sent to a licensed waste processor or waste disposal site. Procedures used for solidification will be reviewed and approved by Southern Nuclear Operating Company (SNC).

Spent resin and evaporator concentrates may be encapsulated in containers, while solid waste such as paper, clothing, rags, towels, etc., is compressed directly into the drums. The chemical drain tank effluent is sent to the waste holdup tank for further processing. In the case of metals, wood, etc., the material will be loaded into an appropriate sized container to facilitate shipment and burial.

A. Encapsulation Process - In-Plant System

The evaporator bottoms and spent resins are transported in pipes to the drumming area.

The evaporator bottoms and spent resins are dispensed from a common manifold using six separate valves. The chemical drain tank effluent is dispensed from a single valve on the tank drain line. These valves are fail-safe, air-operated diaphragm valves.

Waste evaporator bottoms are encapsulated in 55-gal drums that are prepared in a nonradiation area separate from the drumming room.

The drums are positioned upright and an injector assembly is suspended within the drum. A vibrator which is strapped to the vertical surface of the drum is energized, and four bags of vermiculite cement are gradually poured into the drum. This mixture completely surrounds the liquid injector assembly. The drum

lid is installed, and the clamping ring is secured in position. The drum is now ready for use.

Spent resin slurries are encapsulated in 55-gal drums that are prepared in a nonradiation area separate from the drumming room.

The drums are positioned upright, and a mixture of water and cement is poured into the drum until the bottom surface is covered with a 1-in. thick layer. This operation is followed by placing a 16-gauge thick, carbon steel casting sleeve in the drum and filling the annulus between the casting sleeve and the inside diameter of the drum with the water cement mixture for a height of 29 in. After the cement liner has become compact, the drum vibrator is strapped to the outside surface of the drum and then energized. A 1-in. layer of dry vermiculite cement is then poured into the bottom of the casting sleeve. A resin cage assembly, fabricated of 12-gauge thick carbon steel and resembling a DOT-2R container, is suspended inside the casting sleeve. The void between the cage and sleeve and the area above the cage, extending to the top of the drum, is filled with the dry vermiculite cement. The drum lid is then installed, and the clamping ring is secured in position. The drum is now ready for use.

B. Encapsulation Process - Portable System

The in-container cement solidification system utilizes specially designed liners. These liners are equipped with internal mixer assemblies and fit into cylindrical shielded shipping casks. Figure 11.5-2 shows a cross-sectional view of a liner in place within a cask.

In-container cement solidification can be utilized in conjunction with the disposable demineralizers and filters. Liquid radwaste streams are processed, and then the depleted processing medium is solidified or dewatered, transported, and disposed of, all in the same vessel. Resin handling operations, such as sluicing from the demineralizer to the solidification container, are minimized, thereby resulting in savings in both cost and personnel radiation exposure. In addition, using this approach of disposable demineralizers and filters offers significant waste volume reduction benefits as compared to other techniques for concentrating liquid wastes, such as evaporation.

C. Baling Process

The baling process involves the use of drums. The baler is equipped with a dust shroud to prevent the escape of radioactive particulate matter during the compaction process. This shroud is connected to the building exhaust system. After the drum has been filled with compacted wastes, it is sealed and transferred to the storage area.

The in-plant radwaste encapsulation process described in paragraph 11.5.3.1.A has not been qualified in accordance with the stability criteria of 10 CFR 61.56(b). Subsequent to the startup of the disposable demineralizer system (described in

paragraph 11.2.3.1.8) in 1981 and due to the advent of high integrity containers, the need for solidification has diminished substantially. Thus, the in-plant radwaste encapsulation system is not needed and will not be used at the Farley Nuclear Plant (unless subsequently qualified).

11.5.3.2 Component Design - In-Plant System

A. Drum Vibrator Package

The drum vibrator consists of an electric-operated ball race vibrator, mounting bracket, and mounting strap. The vibrator is mounted on the drum and is used to compact the vermiculite cement inside the drum.

B. Drum

The drum consists of a DOT-17H, 55-gal drum, drum lid with 2-in. bung, lid gasket, and closing ring. After proper assembly, the drum is placed at a filling position in the drumming room, enclosed in shielding if required, and connected to process piping.

C. Vermiculite Cement Blend

The vermiculite cement is a mixture of vermiculite and Portland cement. This blend, which is both a desiccant and a solidifying medium, can either be made on the plant site or obtained from commercial suppliers.

D. Liquid Injector Assembly

The liquid injector assembly is composed of standard, commercially available components and can be assembled on the plant site. The disposal item distributes the liquid waste evaporator bottoms with the vermiculite cement in the drum.

E. Drum Package Tool and Fixture

The drum package tool and fixtures consist of separate metal and wooden support and positioning devices. The wooden fixture holds either the resin cage assembly or the liquid injector in position inside the drum, while the vermiculite cement is being added to the drum. The metal positioning device maintains the position of the injector or resin cage in the prepackaged drum during transit from the prepackaging area to the drumming room filling position.

F. Resin Cage Assembly

The resin cage assembly, which resembles a DOT-2R container, can be fabricated at the plant site. This item aids in removing water from the spent resins and is encapsulated within the solidified cement vermiculite mixture in the drum.

G. Impact Wrench Package

The impact wrench is an air-operated impact wrench with a deep socket. This wrench is used primarily to obtain adequate and uniform application of torque to the closing ring bolt on drum closing.

H. Drum Shields

The drum shield consists of a two-piece, cylindrical lead shield, jacketed with steel. The shield provides radiological protection for operating personnel during the injection of waste liquid into the drums and during subsequent in-plant handling of the drums. Under the maximum radioactivity loading, the calculated surface contact reading is 10 mR/h.

I. Piping Header Assembly

The piping header assembly consists of flexible hose with drum filling header, a liquid level device (vacuum switch), and a gauge. This assembly directs liquid flow to the packages. It automatically controls the volume of liquid added to the packages and can be used whether or not the shield is in place.

J. Drum Header Installation and Removal Tool

The drum header installation and removal tool is of steel fabrication and is used to install and remove the drum filling header into or out of the drum packages.

K. Vacuum Pump Package

The vacuum pump package consists of a commercially available vacuum pump with motor, filter, manifold, vent valve, suction hose, gauge, and service cart. This pump evacuates the drum package prior to the filling operation.

L. Resin Cage Plug Installation Tool

The resin cage plug installation tool is a magnetic socket with a long extension; it also includes a shielding adapter. This tool permits an operator to install a plug in the top of the resin cage assembly which is enclosed by a shield. The shielding adapter provides radiological protection for the operator's hands.

M. Drum Plug Installation Tool

The drum plug installation tool consists of a magnetic socket with a long extension and also includes a shielding adapter. This tool permits an operator to install a plug in the 2-in. bung of the drum lid which is enclosed by a shield. The shielding adapter provides radiological protection for the operator's hands.

N. Drum Lifting Grab

The drum lifting grab is a lifting device with a 5-ton capacity. It is capable of both automatic and remote operation. It can also be used for other nonremote operations involving the handling of the drums.

O. Shield Spreader Bar

The load beam or spreader bar has a 6-ton capacity. The spreader bar is used, under normal conditions, to maneuver the shield assembly which may contain the drum. The design of the shield spreader bar is shown in figure 11.5-3.

P. Drumming Station Control Panel

The drumming station control panel is approximately 42 in. long by 42 in. wide by 16 in. deep and is used for all electrical controls. It has a mounted board for indicating instruments and a wall-mounted cabinet with a piano-hinged front door for access to the cabinet interior. The panel contains the switches that control the operation of the spent resin storage tank and the filling valves. It also contains indicators for open or closed positions of the valves. The arrangement of the drumming station control panel is shown in figure 11.5-4.

11.5.3.3 Operation of Equipment

A. Encapsulation - In-Plant System

The prepared drums are placed in the drumming room at the filling positions and connected to the dispensing manifolds and the vacuum pump. If required, the drums are enclosed in shields. Manifold arrangement and drum area layout are shown in figure 11.5-5 and figures 1.2-1 and 1.2-2, respectively.

Six 55-gal drums can be filled simultaneously with evaporator bottoms or spent resin slurries.

The vacuum pump is energized and the individual drums are evacuated. During the processing of waste evaporator concentrates or chemical tank effluent, the drums are evacuated to a 23-in. Hg vacuum. If spent resins are being processed, the drums are evacuated to a 21-in. Hg vacuum.

After each drum is evacuated, the vacuum pump is disconnected and the required capping performed. Following an interval of 5 min, the vacuum in each drum is recorded. The vacuum is then inspected at 2-min intervals for a minimum of three readings. If the loss in vacuum does not exceed 0.1-in. Hg V-ac/min and the package vacuum level is within limits, the drums are ready to be filled. The remaining encapsulation process is remotely controlled from the panel located outside the drumming area.

When the filling operation is complete, the vacuum switch opens and thus closes the dispensing valve to terminate the filling.

The pressure used to transfer the wastes to the manifold is released in the following manner: The waste evaporator bottom transferring operation is stopped by turning the transfer pump switch to the off position. The spent resin pressure is released by venting the spent resin storage tank.

The major advantages of the vacuum filling technique is that radioactive particulate or gaseous matter is not released to the work area environment.

After transfer of wastes and before the operator enters the drumming area, the filling hoses are flushed to the drums. The flush water is provided by the makeup water system and is manually operated by valves located in a shielded area of the drumming room. The system is so designed that all interconnecting lines between the storage drum, dispensing valve, and manifold are flushed back to the spent resin storage tank or to the remaining spaces in the individual drums. The waste in the chemical drain tank effluent valve and drain piping can also be flushed back into the chemical drain tank. A radiation detector, located in the drumming area with a readout instrument on the control panel, enables the operator to determine remotely the radiation levels in the drumming area. Throughout these operations, additional radiation surveys can be made by using portable survey equipment.

The header assemblies are removed from each drum. Using the shielded tools provided, a plug is inserted into the resin cage guide tube and tightened securely. This is followed by the insertion of a 2-in. bung closure and shielded plug. After the plugs are installed, the drum with shield is transferred from the drumming areas to the shield stripping area, using the overhead crane and spreader bar. To remove the shield from the drum, the operator releases the latching mechanism attaching the shield cylinder to the shield base. Following this operation, the operator moves to the observation area where succeeding operations may be performed remotely. Using the crane, the shields are stripped from the drums.

B. Encapsulation - Portable System

A typical configuration of the in-container cement solidification system is shown in figure 11.5-2. The design and operational features of the system, when utilized to solidify resins, are as follows:

1. The solidification liner is in place within the shielded shipping cask or shielded pit. The cask cover is in place, and the shield plug in the center of the cask cover is removed or the liner is in a shielded pit in the solidification facility.

2. The drive and fill-flange assembly is bolted to the liner opening, forming a sealed closure. The mixer drive unit is an integral part of the drive and fill-flange assembly, and it remains in place during the sluicing and dewatering

operations. The mixer drive mates with the mixer shaft in the solidification liner.

3. Vent and overflow lines are used to prevent overpressurization of the liner and overflow into the cask, respectively. The vent line is directed to a filter/dust collector.

4. If necessary, resins are slurried into the liner through the waste inlet connection in the drive and fill-flange assembly.

5. Suction is maintained on the dewatering line, so that the slurry is dewatered during the sluicing operation unless material is already in a disposable liner.

6. High levels within the liner are monitored during the sluicing operation using a contact level probe.

7. Dewatering is continued after sluicing is completed in order to remove as much water as possible from the resin (as necessary).

8. Upon completion of dewatering, the resin inlet line is disconnected and the cement fill line inlet valve is opened.

9. Water is added in specified amounts, and the liner internal mixer is rotated using the drive unit. Cement is added, in amounts dictated by the process control plan, in order to produce the desired mixture.

10. Mixing is continued until the torque required to rotate the mixer increases, indicating setup of the mixture.

11. The drive and fill-flange assembly is removed, the solidification liner is capped, and if cask is used, the shipping cask shield plug is reinstalled. The cask is then prepared for shipment. If pit is used after solidification, the container is transferred into the shipping cask.

C. Baler Operation

The baler is used to compress low radiation level solid wastes into drums. The drum is placed in the baler, solid wastes are inserted in the open drum, and the shroud door is closed. The drum is automatically positioned to be coaxial with the baler ram. An operator initiates the compaction process by positioning an up/down switch in the down position, thus energizing the hydraulic pump motor. The hydraulic pressure forces the ram down into the drum, thereby compressing the wastes. The shroud door is opened, and additional wastes are added to the drum. The cycle is repeated until the drum is full and the lid is installed, the clamping ring tightened, and the drum stored pending shipment. An anti-spring hood device may be used to help keep compressed materials in the barrels during compaction.

The shroud is ducted to the plant ventilation system to remove dust or particles that may be emitted from the drum during compression of the wastes.

11.5.4 EXPECTED VOLUMES

[HISTORICAL]

[*Table 11.5-1 gives the total solid waste generated per year through 1986. The associated Ci content of solid waste processed from Farley Nuclear Plant (FNP) to date is also given.*

The principal nuclides shipped from the plant site include the following:

Chromium-51	*Molybdenum-99*
Manganese-54	*Iodine-131*
Manganese-56	*Cesium-134*
Iron-59	*Cesium-136*
Cobalt-58	*Cesium-137*
Cobalt-60	*Iron-55***]**

The total solid waste generation at FNP is documented/reported in the Annual Radiological Effluent Report as required by the Technical Specifications.

11.5.5 PACKAGING

The solid waste system product is shipped to the appropriate burial ground facility using the necessary package, as required by Department of Transportation (DOT) regulations. These packages include but are not limited to: strong, tight containers; type A containers; type B containers; and large quantity containers.

11.5.6 STORAGE FACILITIES

Solid radwastes, dewatered resins, and waste oil are contained in their appropriate packages and stored at the following designated locations:

A. Low-level radwaste storage building (outside of auxiliary building). The low-level radwaste storage building (LLRB) has been constructed to supplement the plant's storage capabilities for equipment and some types of radwaste. This building will be utilized as the primary storage point for dry active waste as well as the loading facility for shipments of dry active waste. Storage containers normally will be

transported to and placed in the storage building by means of a fork lift. The LLRB will utilize a first-in, first-out method of inventory control, when practical, since the building is designed to accommodate temporary material storage of up to a 4-year duration. Other waste forms may be stored in the LLRB as needed should primary storage locations not be available. Contaminated equipment and materials may be stored in the LLRB provided that suitable contamination control practices are implemented.

B. Drumming rooms (rooms 2420 and 2421 at el 155 ft in the Unit 2 auxiliary building).

C. New fuel storage rooms (Unit 1 auxiliary building room 459 and Unit 2 auxiliary building room 2459 at el 155 ft).

D. Liquid Radwaste Processing Facility (LRWPF) (Unit 1 auxiliary building room 603 and Unit 2 auxiliary building room 2603 at el 130 ft).

E. Nuclear laundry rooms (rooms 420 and 421 at el 155 in the Unit 1 auxiliary building).

F. Solidification dewatering facility oil and paint storage room.

G. Temporary storage during and after filling prior to transport to the LLRB is normally in the east end of Unit 1 or Unit 2 155-ft hallway.

H. Temporary storage of waste in process inside the solidification and dewatering facility.

I. Temporary storage of disposable demineralizer liners, oil drums, or other wastes may be in suitable containers outside the solidification and dewatering facility until shipment. This storage time will normally be kept to a minimum.

J. Low-level contaminated construction materials in suitable containers may be temporarily staged in the Complex III warehouse until shipment or movement to other storage locations.

11.5.7 SHIPMENT

Under normal operating conditions, solid radioactive wastes will be shipped from the site by carriers licensed to perform offsite disposal of such wastes. The normal mode of shipment will be by truck in containers which meet DOT standards. Facilities are available for shipment by rail, if necessary. Typically, the shipper will supply any special DOT-approved casks that he may have available for the shipment of radioactive material with an unusual configuration, with a high Ci content, and/or with a high radiation level. The shipper will be contracted to pick up waste as required.

The containers are transferred from the storage area to the shipper's vehicle by mechanical means. Large liners, including disposable demineralizers, will be handled by the spent fuel cask crane or by appropriate mobile cranes. Loading will be performed by SNC and/or contractor personnel and the vehicle will be assigned for sole use of SNC. Once the waste is loaded and secured on the truck, surveys are made to ensure compliance with 49 CFR 170 through 199, specifically, if shipped as "Radioactive - LSA" material:

A. The dose rate will be less than 2 mrem/h in the normally occupied areas of the cab, less than 10 mrem/h at 2 meters from the vehicle vertical surfaces, and less than 200 mrem/h at any point on the surface of the vehicle.

B. The vehicle shall smear less than 2200 dpm/100 cm2 beta gamma and less than 220 dpm/100 cm2 alpha.

C. Containers will be labeled or marked as required, and shipping papers that describe the hazardous materials that are being transported will be provided to the carrier.

Although it will not be a normal procedure to store full or partially loaded shipment vehicles on the site for extended periods of time, if the occasion should arise, the vehicles will normally be parked inside the owner controlled fenced area. Normally, drummed waste will not be stored in any areas other than those mentioned in subsection 11.5.6. Some exceptions to this are when drums are stored for super compactor support or if plant management gives permission to store in other areas due to unforeseen circumstances.

[HISTORICAL]
[TABLE 11.5-1 (SHEET 1 OF 3)

ACTUAL TOTAL SOLID WASTE GENERATED PER UNIT AT FNP

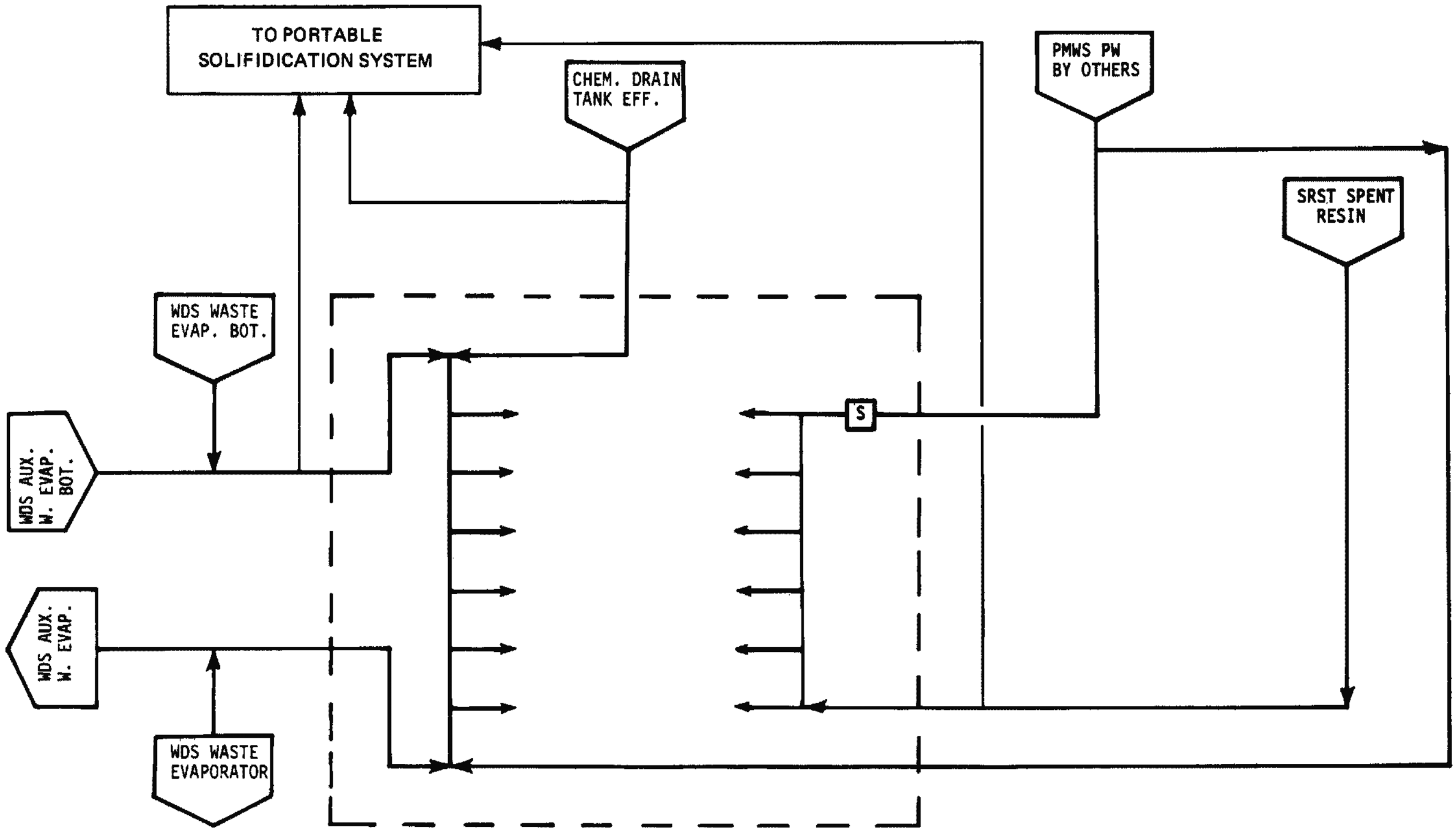
Period	*Total Solid Volume (m^3)*	*Total Ci*
1977, 3rd and 4th quarters		
Spent resin, filter sludges, evaporator bottoms	*221*	*3.90*
Dry compressible waste	*0*	*0*
1978, 1st and 2nd quarters		
Spent resin, filter sludges, evaporator bottoms	*68*	*0.117*
Dry compressible waste	*0*	*0*
1978, 3rd and 4th quarters		
Spent resin, filter sludges, evaporator bottoms	*110*	*3.40*
Dry compressible waste	*91*	*2.20*
1979, 1st and 2nd quarters		
Spent resin, filter sludges, evaporator bottoms	*340*	*150*
Dry compressible waste	*200*	*30*
1979, 3rd and 4th quarters		
Spent resin, filter sludges, evaporator bottoms	*260*	*390*
Dry compressible waste	*310*	*20*
1980, 1st and 2nd quarters		
Spent resin, filter sludges, evaporator bottoms	*88*	*180*
Dry compressible waste	*140*	*340*
1980, 3rd and 4th quarters		
Spent resin, filter sludges, evaporator bottoms	*73*	*9.8*
Dry compressible waste	*140*	*1.8*

TABLE 11.5-1 (SHEET 2 OF 3)

Period	*Total Solid Volume (m^3)*	*Total Ci*
1981, 1st and 2nd quarters		
Spent resin, filter sludges, evaporator bottoms	*133*	*589*
Dry active waste	*226*	*13.1*
1981, 3rd and 4th quarters		
Spent resins and filter sludges	*16.4*	*115*
Dry active waste	*189*	*2.47*
1982, 1st and 2nd quarters		
Spent resins and filter sludges	*19.3*	*25.1*
Dry active waste	*183*	*14.4*
1982, 3rd and 4th quarters		
Spent resins and filter sludges	*9.63*	*3.23*
Dry active waste	*163.2*	*60.2*
1983, 1st and 2nd quarters		
Spent resins and filter sludges	*33.13*	*788.04*
Dry active waste	*190.28*	*22.33*
1983, 3rd and 4th quarters		
Spent resins and filter sludges	*22.2*	*213.2*
Dry active waste	*214.8*	*20.1*
Irradiated components, control rods, etc.	*1.0*	*5.1*
1984, 1st and 2nd quarters		
Spent resins and filter sludges	*20.02*	*113.8*
Dry active waste	*288.4*	*12.09*

TABLE 11.5-1 (SHEET 3 OF 3)

Period	*Total Solid Volume (m^3)*	*Total Ci*
1984, 3rd and 4th quarters		
Spent resins and filter sludges	*13.93*	*153.65*
Dry active waste	*187.0*	*15.2*
1985, 1st and 2nd quarters		
Spent resins and filter sludges	*35.68*	*27.47*
Dry active waste	*277.0*	*14.85*
1985, 3rd and 4th quarters		
Spent resins and filter sludges	*35.39*	*773.5*
Dry active waste	*129.81*	*5.50*
Other (absorbed oil)	*17.20*	*0.0022*
1986, 1st and 2nd quarters		
Spent resins and filter sludges	*16.1*	*306.0*
Dry active waste	*60.1*	*0.201*
1986, 3rd and 4th quarters		
Spent resins and filter sludges	*26.03*	*920.3*
Dry active waste	*141.3*	*1.77*
Irradiated components (control rods, etc.)	*1.37*	*570.0]*



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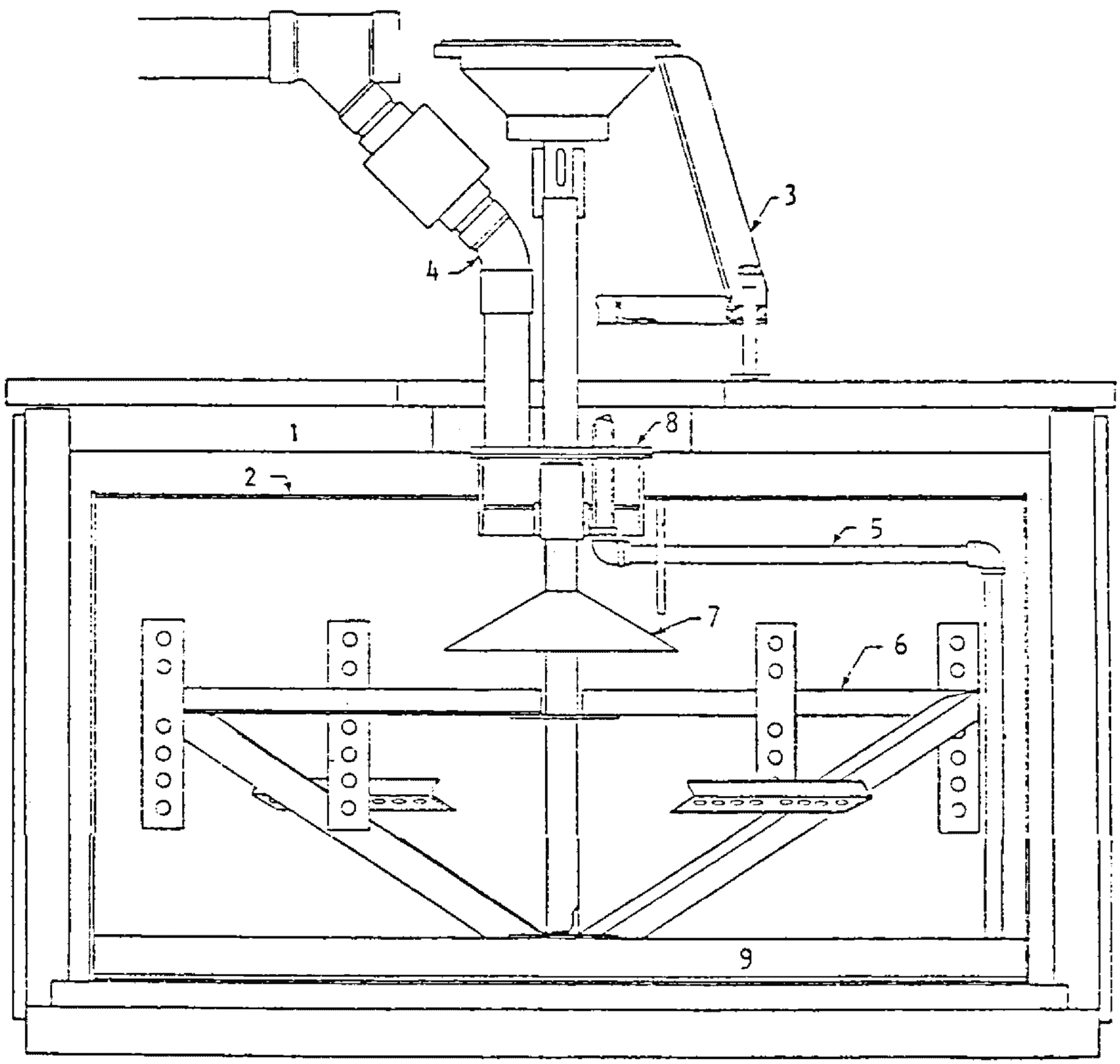
NUCLEAR PLANT
UNIT 1 AND UNIT 2

PROCESS FLOW DIAGRAM
SOLID WASTE PROCESSING SYSTEM

FIGURE 11.5-1

1 CASK
2 LINER
3 DRIVE FRAME AND ASSEMBLY
4 CEMENT FEEDLINE
5 DEWATERING LINE
6 MIXER ASSEMBLY
7 DEFLECTOR PLATE
8 ACCESS CLOSURE PLATE
9 FILTER ELEMENTS



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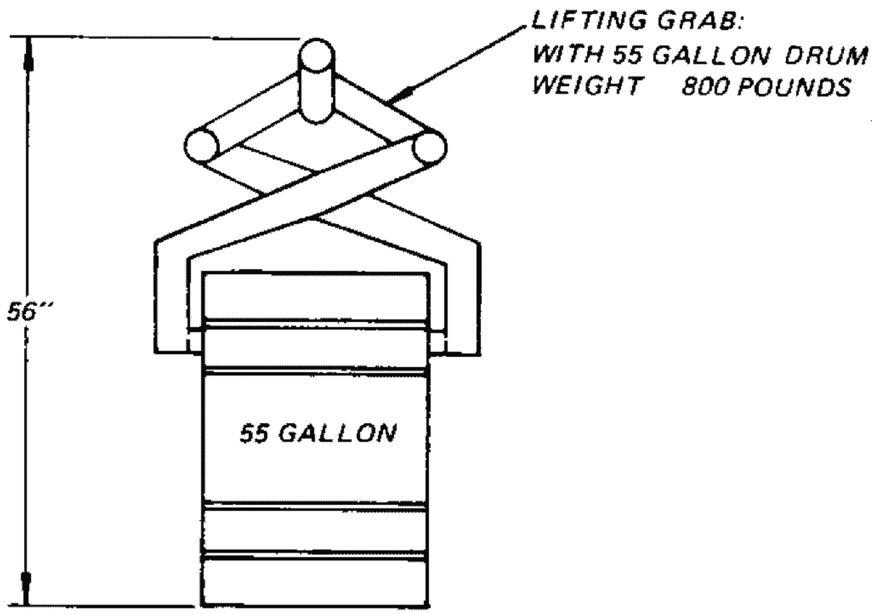
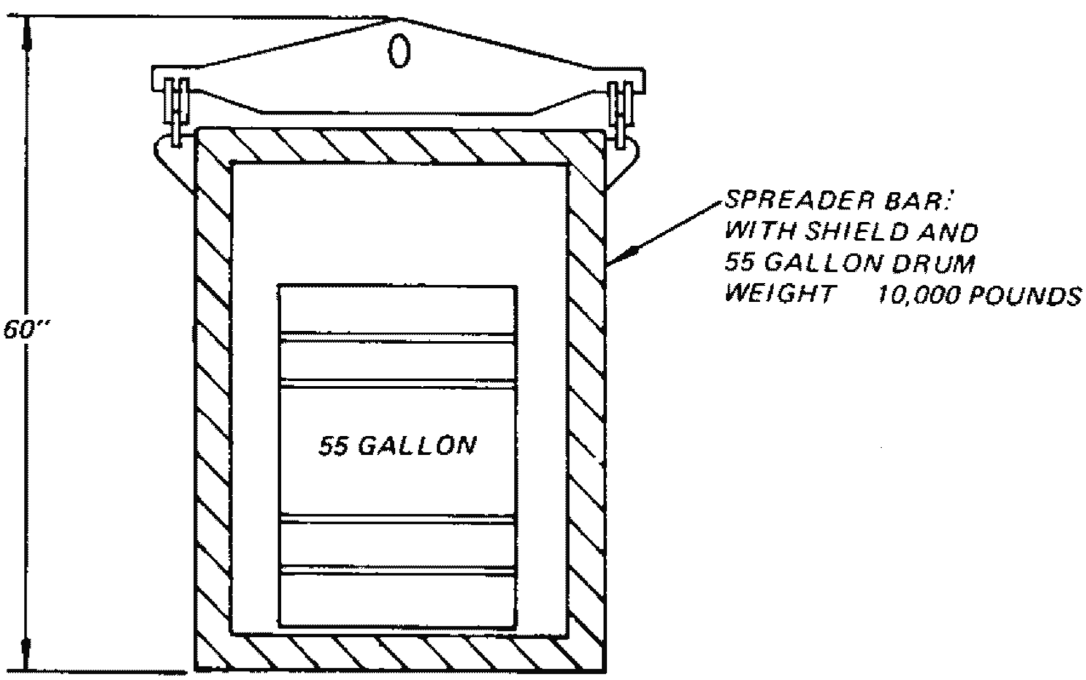
UNIT 1 AND UNIT 2

PORTABLE IN-CONTAINER CEMENT SOLIDIFICATION SYSTEM

FIGURE 11.5-2





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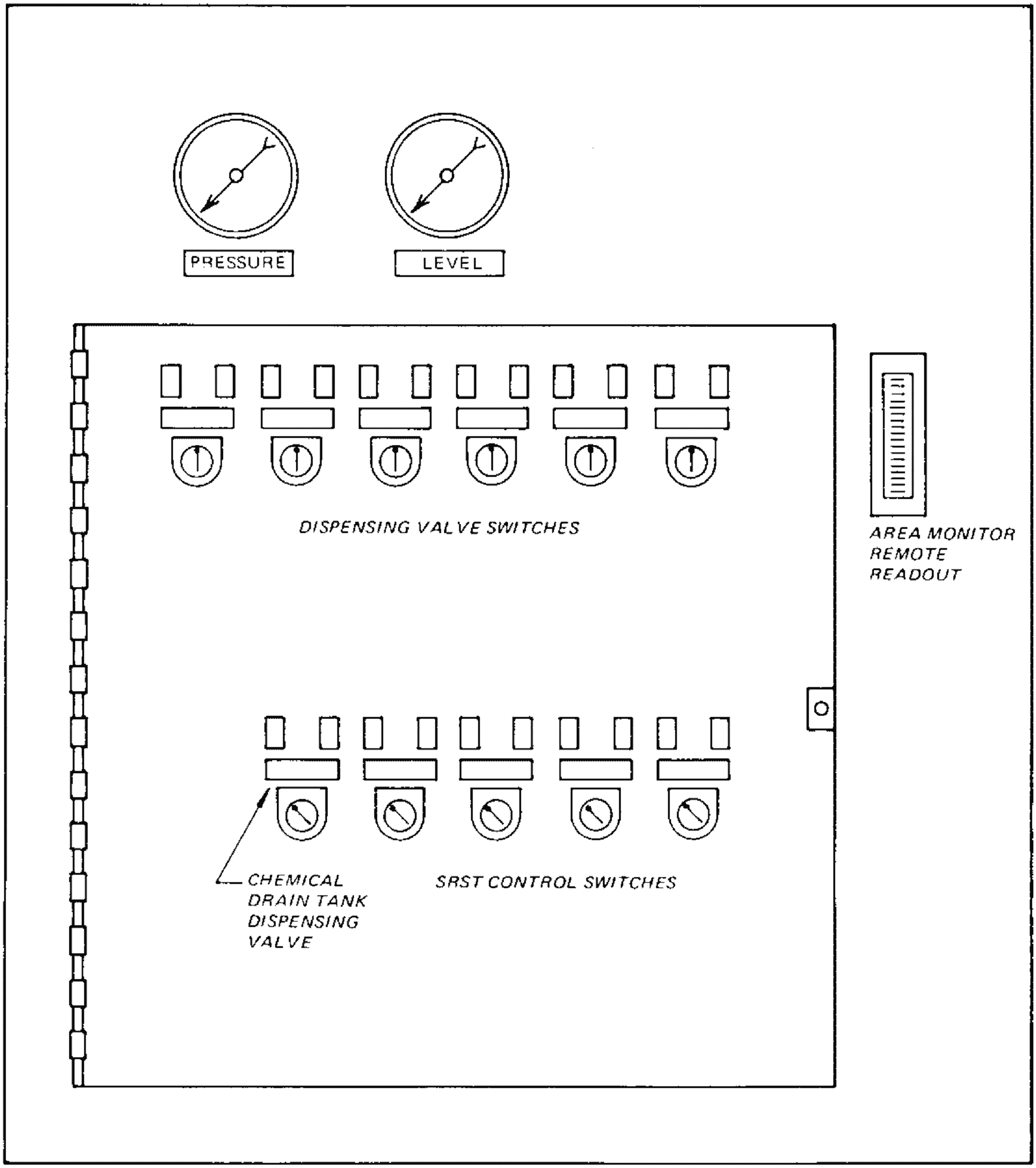
UNIT 1 AND UNIT 2

DRUM HANDLING ASSEMBLIES

FIGURE 11.5-3



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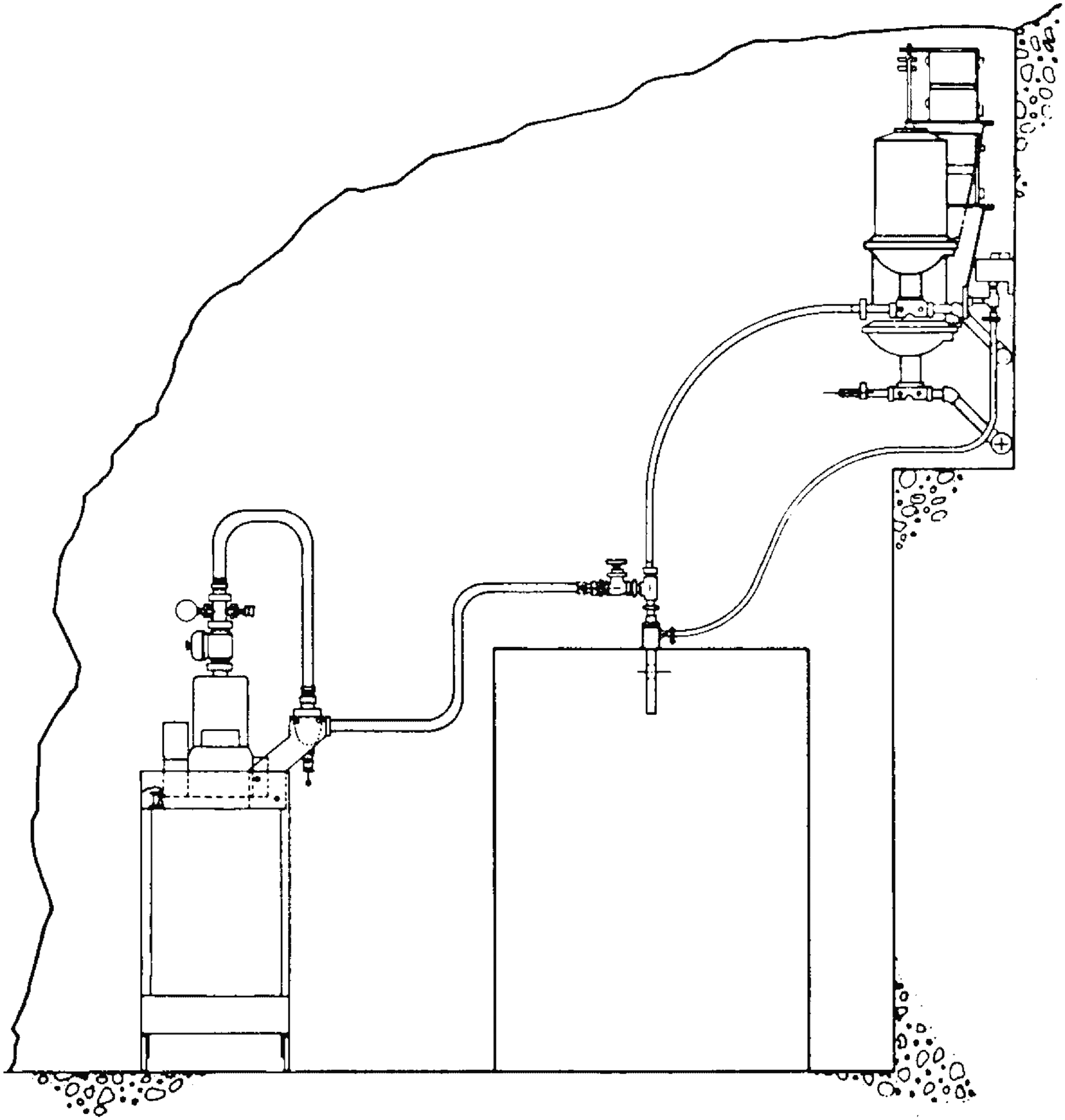
DRUMMING ROOM CONTROL PANEL

FIGURE 11.5-4



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UNIT 1 AND UNIT 2

SHIELDED DRUM PACKAGE INSTALLED
AND READY FOR EVACUATION

FIGURE 11.5-5

11.6 RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

The Radiological Environmental Monitoring Program (REMP) is described in Chapter 4 of the Offsite Dose Calculation Manual (ODCM).

APPENDIX 11A TRITIUM CONTROL

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APPENDIX 11A

TRITIUM CONTROL

The release of tritium to the environment from operating Westinghouse pressurized water reactors (PWRs) has always been well below 10 CFR 20.1 - 20.601 limits. This section discusses the reduced tritium production in the plant as a result of employing zircaloy-clad fuel and silver-indium-cadmium control rods.

11A.1 SYSTEM SOURCES

There are two principal contributors to tritium production within the PWR system: the ternary fission source and the dissolved boron in the reactor coolant. Additional small contributions are made by Li-6, Li-7, and deuterium in the reactor water. Tritium production from different sources is shown in table 11A-1.

11A.1.1 THE FISSION SOURCE

This tritium is formed within the fuel material and may do one of the following:

A. Remain in the fuel rod uranium matrix.

B. Diffuse into the cladding and become hydrated and fixed there.

C. Diffuse through the clad for release into the primary coolant.

D. Release to the coolant through macroscopic cracks or failures in the fuel cladding.

Previous Westinghouse design has conservatively assumed that the ratio of fission tritium released into the coolant to the total fission tritium formed was approximately 0.30 for zircaloy-clad fuel. The operating experience at the R. E. Ginna Plant of the Rochester Gas and Electric Company, and at other operating reactors using zircaloy-clad fuel, has shown that the tritium release through the zircaloy fuel cladding is substantially less than earlier estimates predicted. Consequently, the release fraction may be revised downward from 30 percent to 10 percent based on this data.[1]

11A.1.2 CONTROL ROD SOURCE

The full and part length rods for this plant are of silver-indium-cadmium. There are no reactions in these absorber materials which would produce tritium, thus eliminating any contribution from this source.

11A.1.3 BORIC ACID SOURCE

A direct contribution to the reactor coolant tritium concentration is made by neutron reaction with the boron in solution. The concentration of boric acid varies with core life and load follow, so that this is a steady decreasing source during core life. The principal boron reactions are the B-10(n, 2α)H-3 and B-10(n, α)Li-7(n, nα)H-3 reactions. The Li-7 reaction is controlled by limiting the overall lithium concentration to approximately 2 ppm during operation. Li-6 is essentially excluded from the system by utilizing 99.9 percent Li-7.

11A.1.4 BURNABLE SHIM ROD SOURCE

These rods are in the core only during the first operating cycle and their tritium contribution is potential only during this period.

11A.1.5 LITHIUM AND DEUTERIUM

Lithium and deuterium reactions contribute only minor quantities to the tritium inventory, as shown in table 11A-1.

11A.2 DESIGN BASES

The design intent is to reduce the tritium sources in the reactor coolant system to a practical minimum in order to permit longer retention of the reactor coolant within the plant. Reduction of source terms is provided by utilizing silver-indium-cadmium control rods and the determination that the quantity of tritium released from the fuel rods with zircaloy cladding is less than originally expected.

11A.3 DESIGN EVALUATION

Table 11A-1 is a comparison of a typical design basis tritium production, which has been utilized in the past to establish system and operational requirements of the plant and present expected values.(1) It is noted that there are two principal contributors to the tritium production: ternary fission source and the dissolved boron in the reactor coolant. Of these sources it is noted that the 30 percent release of ternary fission through the cladding was the predominant contributor in past design considerations.

Because of the importance of this source on the operation of the plant, Westinghouse has been closely following operating plant data. Table 11A-2 represents tritium releases during one calendar year for different Westinghouse PWR plants. Further, a program is being conducted at the R. E. Ginna Plant to follow this in detail. The R. E. Ginna Plant has a zircaloy-clad core with silver-indium-cadmium control rods. The operating levels of boron concentration during the startup of the plant are approximately 1100 to 1200 ppm of boron. In addition, burnable poison rods in the core contain boron which will contribute some tritium to the coolant, but only during the first cycle. Data during the operation of the plant have indicated very clearly that the present

design sources were indeed conservative. The tritium released is essentially from the boron dissolved in the coolant and a ternary fission source which is less than 10 percent. In addition to this data, other operating plants with zircaloy-clad cores have also reported very low tritium concentrations in the reactor coolant system after considerably longer operation.

For a leakage from the primary coolant system into the containment of 40 lb/day, with an assumed tritium concentration in the coolant of 2.5 $\mu Ci/cm^3$ (no containment ventilation purge), the tritium concentration in the atmosphere of the containment would be low enough to permit access without protective equipment by plant maintenance personnel for an average of 2 h/week.

Leakage into the containment atmosphere is based on leakages from equipment such as pumps and valves. Abnormal leakages in excess of the design estimate have occurred in operating plants. The leaking components have been identified and corrective measures have been taken. For example, bellows sealed valves, diaphragm sealed valves, and pump seal purge systems have been employed.

The total activity that would be released from the containment purge during refueling operations would range in the order of 20 to 40 Ci of tritium, depending on the core cycle, relative humidity, etc. It is not proposed that this amount of activity from evaporative losses be collected but that it be discharged from the plant. Similarly, any radioactive gases in the containment would be discharged. Evaporation of tritium from the refueling pool has been considered in evaluating the consequences to tritium on both operators and environmental releases. This indicates maximum tritium concentration in the containment consistent with 40 h/week occupancy and total tritium release of about 30 Ci/refueling.

The tritium source terms in the reactor coolant are at a low level (approximately 1110 Ci/cycle) such that it is possible to discharge tritium in amounts to preclude in-plant exposure problems without exceeding the "as low as practicable" design objective. Alternatively, without any intentional removal of tritiated water:

A. Tritium levels should not cause a problem during refueling through the 40-year original operating license term or the 60-year operating life resulting from the renewed licenses. Assuming no change in tritium production or in system leakage, the plant activities (and consequently, releases) would increase only 9% for the additional 20-year period of extended operation due to the short half-life. Considering system leakage, the actual increase is even less.

B. Special procedures (purging, etc.) prior to containment access may be required.

Credit is taken for dilution of the reactor coolant system water by refueling water and spent-fuel pool. Discharge of the tritiated water from the plant is therefore possible at extended intervals or, if discharged on a regular basis, would below the 10 CFR 20.1 - 20.601 limits because of the reduced production rates.

Based on the above, the following conclusions have been reached:

A. The tritium levels in plants operating with zircaloy-clad cores will be substantially lower than previous design predictions.

B. The tritium source in the plants will be reduced by utilizing silver-indium-cadmium control rods.

C. Containment access during power operation and refueling with continued storage of the tritium in the plant is possible with the application of special procedures (purging, etc.) prior to containment access.

D. The containment tritium purge is relatively small compared to the total available and will be discharged.

REFERENCE

1. Westinghouse Electric Corporation, "Source Term Data for Westinghouse Pressurized Water Reactors," WCAP-8253, Revision 1, July 1975.

TABLE 11A-1

TYPICAL DESIGN BASIS TRITIUM PRODUCTION

Tritium Source	Total Produced (Ci/year)	Release Expected to Reactor Coolant (Ci/year)
Ternary fission	8160	816
Burnable poison rods		
Initial cycle	599	60
Soluble boron		
Initial cycle	152	152
Equilibrium cycle	217	217
Lithium and deuterium reactions	82	82
Total initial cycle	8993	1110
Total equilibrium cycle	8459	1120

Basis

Power level, core thermal power (Mwt)	2766
Load factor	0.8
Release fraction from fuel (percent)	10
Release fraction from burnable poison rods (percent)	10
Burnable poison rod B-10 mass (g)	2374
Reactor coolant boron concentration, initial cycle (ppm)	700
Reactor coolant boron concentration, equilibrium cycle (ppm)	1000

TABLE 11A-2

TRITIUM RELEASES FOR 1971 FROM WESTINGHOUSE-DESIGNED OPERATING REACTORS

Plant	Type of Cladding	Total Released (Ci)	Average Discharge Concentration ($\mu Ci/cm^3$)	Fraction Column 2, Table II, Appendix B, 10 CFR 20.1-20.601 (3×10^{-3} $\mu Ci/cm^3$)
Yankee Rowe	Stainless steel	1633	5.9×10^{-6}	2.0×10^{-3}
Connecticut Yankee	Stainless steel	5830	7.7×10^{-6}	2.6×10^{-3}
San Onofre	Stainless steel	4570	6.7×10^{-6}	2.4×10^{-3}
Robert E. Ginna	Zircaloy	154	2.3×10^{-7}	7.7×10^{-5}
H. B. Robinson Unit 2	Zircaloy	118	1.7×10^{-7}	6.0×10^{-5}
Point Beach Unit 1	Zircaloy	266	4.7×10^{-7}	1.6×10^{-4}

APPENDIX 11B

MATHEMATICAL MODEL OF RADIOACTIVITY DOSES

[THIS SECTION INTENTIONALLY DELETED]