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DRAWINGS CITED IN THIS CHAPTER*

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

CHAPTER 11.0 RADIOACTIVE WASTE MANAGEMENT

Radioactive waste systems are designed to collect, process, control and dispose of radioactive waste in a safe manner without limiting unit or station operations or availability. Equipment, instrumentation, and operating procedures are provided to assure that the discharge of radioactive wastes will not exceed the limits as set forth in 10 CFR 20 (now also 10 CFR 50, Appendix I). [11.1-1]

The performance objectives of the radioactive waste systems are:

A. To provide effective control of processes to prevent the release of radioactive materials in excess of limits prescribed in 10 CFR 20 (Now also 10 CFR 50, Appendix I);

B. To minimize the radioactive waste release to the environs;

C. To provide sufficient time for operator decision and action in the event of off-standard conditions; and

D. To minimize the radiation hazards to the station personnel and the public.

Radioactive wastes resulting from station operation are classified liquid, gaseous and solid wastes. The following descriptions pertain to radioactive wastes as used herein:

A. Gaseous Radioactive Wastes

Airborne particulates, gases vented from process equipment, and under certain conditions, the building ventilation exhaust air, are considered as gaseous radioactive waste. The major source of gaseous radioactive waste (condenser air ejector effluent) is continuously decayed and filtered during operation and monitored to ensure that release limits of 10 CFR 20 and 10 CFR 50, Appendix I are not exceeded.

B. Liquid Radioactive Waste [11.1-2]

Liquids from the reactor process systems, or liquids which have become contaminated with these process system liquids are considered liquid radioactive waste. The liquid radioactive wastes are processed according to their purity (essentially conductivity, organic content, and activity level) before being returned to the plant as condensate, sent to the diffuser pipe, sent to the discharge bay, or reprocessed through the radioactive waste system for further purification.

C. Solid Radioactive Waste [11.1-3]

Solids recovered from the reactor process system, solids in contact with reactor process system liquids or gases, and solids used in the reactor process system operation are considered solid radioactive waste. The solid radioactive wastes are processed and put into drums, liners, high-integrity containers, bins, or boxes for storage onsite or disposal offsite.

The system's components are designed and operated in such a manner as to minimize radiation exposure of personnel, as well as to significantly reduce the radioactivity levels

below those limits set forth in 10 CFR 20 and the regulations of the states of Illinois and Iowa. The process control program (PCP) contains the approved methods for handling solid radioactive wastes.

By virtue of location, nonradioactive liquid wastes are kept separate from controlled access areas. These wastes are discharged by conventional means.

11.1 Source Terms

The basis selected for the design capacity of the liquid, gaseous, and solid radioactive waste management systems is the design basis activity concentration of 4.5 μCi/cc of corrosion and fission products present in the reactor coolant. The corrosion and fission product input quantities are reduced in processing through the radioactive liquid and gaseous systems. The reduced values then become the estimated quantities of radionuclides disposed offsite as solids or released to the environment. Further details of the liquid, gaseous, and solid waste management systems are presented in Sections 11.2, 11.3, and 11.4, respectively. [11.1-4]

This section discusses the sources of radionuclides and the amount of those radioactive materials produced in the reactor system.

11.1.1 Source of Radioactive Nuclides

Sources of radioactive nuclides in the reactor coolant system consist of fission products from a fuel cladding failure, radioactivated corrosion products, and radioactivated products in the coolant. [11.1-5]

Radioactive fission product nuclides arise from minor amounts of "tramp" uranium on the surface of the fuel cladding and from either imperfections or perforations which might develop in the fuel cladding. The principal radioactive fission product nuclides in the reactor coolant are listed in Table 11.1-1. [11.1-6]

Certain elements present in the reactor coolant are activated upon exposure in the reactor core. The principal activated products in the reactor coolant stream are listed in Table 11.1.2. [11.1-7]

11.1.2 Radioactive Nuclide Concentrations

Estimated concentrations of the fission product, activation product, and corrosion product radioactive nuclides are tabulated in Table 11.1-1 and Table 11.1-2. [11.1-8]

11.1.3 Mathematical Model and Parameters

This section discusses the mathematical equations and parameters used both in obtaining the source terms used as a basis for sizing and design of the radioactive waste management systems.

The design basis numbers used for the Quad Cities Station are based on the diffusion model and an off-gas rate of 200,000 μCi/s. A design basis activity concentration of 4.5 μCi/cc of corrosion and fission products was applied in the Quad Cities design.

The reactor design and operating parameters used to arrive at the fission product and activated product concentrations are listed in Table 11.1-3. [11.1-9]

11.1.3.1 Noble Radiogas Fission Products

The noble radiogas fission product source terms observed in operating BWRs are generally complex mixtures with sources varying from minuscule defects in cladding to "tramp" uranium on external cladding surfaces. The relative concentrations or amounts of noble radiogas isotopes can be described as follows: [11.1-10]

$$\text{Equilibrium: } R_g = k_1 Y \qquad (11.1\text{-}1)$$

$$\text{Recoil: } R_g = k_2 Y \lambda \qquad (11.1\text{-}2)$$

The nomenclature in Subsection 11.1.3.4 defines the terms in these and succeeding equations. The constants $k_{\{1\}}$ and $k_{\{2\}}$ describe the fractions of the total fissions that are involved in each of the releases. The equilibrium and recoil mixtures are the two extremes of the mixture spectrum that are physically possible. When a sufficient time delay occurs between the fission event and the release time of the radiogases from the fuel to the coolant, the radiogases approach equilibrium levels in the fuel and the equilibrium mixture results. When there is no delay or impedance between the fission event and the release of radiogases, the recoil mixture is observed.

It was assumed that noble radiogas leakage from the fuel would be the equilibrium mixture of the noble radiogases present in the fuel.

The intermediate decay mixture was termed the "diffusion" mixture. It must be emphasized that this "diffusion" mixture is merely one possible point on the mixture spectrum,ranging from the equilibrium to the recoil mixture, and does not have the absolute mathematical and mechanistic basis for the possible calculational methods for equilibrium and recoil mixtures. However, the "diffusion" distribution pattern which has been described, is as follows:

$$\text{Diffusion: } R_g = k_3 Y \lambda^{0.5} \qquad (11.1\text{-}3)$$

The constant $k_{\{3\}}$ describes the fraction of total fissions that are involved in the release. The value of the exponent of the decay constant, lambda, midway between the values for equilibrium, 0, and recoil.

Though the previously described "diffusion" mixture was used by GE as a basis for design since 1963, the design basis release magnitude used has varied from 0.5 Ci/s to 0.1 Ci/s as measured after a 30-minute decay (t = 30 min.). The noble radiogas source-term rate after a 30-minute decay was used as the conventional measure for the design basis fuel leakage rate since it is conveniently measurable and was consistent with the nominal design basis 30 minutes of gas holdup system used on a number of plants.

11.1.3.2 Radiohalogen Fission Products

Historically, the radiohalogen design basis source term was established by the same equation used for noble radiogases. In a fashion similar to that used with gases, a simplified equation can be shown to describe the leakage rate of each halogen radioisotope:

$$R_h = K_h Y \lambda^n \qquad (11.1\text{-}4)$$

The constant, $K_{\{h\}}$, describes the magnitude of leakage from fuel. The relative rates of halogen radioisotope leakage is expressed in terms of n, the exponent of the decay constant, lambda. As was done with the noble radiogases, the average value was determined for n. The value for n is 0.5 with a standard deviation of +-0.19.

11.1.3.3 Other Fission Products

The observations of the fission products (and transuranic nuclides, including Np-239) in operating BWRs are not adequately correlated by simple equations. For these radioisotopes, design basis concentrations in reactor water have been estimated conservatively. Carryover of these radioisotopes from the reactor water in the steam is estimated to be <0.1% (<0.001 fraction). In addition to carryover, however, decay of noble radiogases in the steam leaving the reactor will result in production of noble gas daughter radioisotopes in the steam and condensate systems.

11.1.3.4 Nomenclature

The following list of nomenclature defines the terms used in equations for source-term calculations:

R_g = leakage rate of noble gas radioisotope (μCi/s)
R_h = leakage rate of halogen radioisotope (μCi/s)
Y = fission yield of a radioisotope (atoms/fission)
lambda = decay constant of a radioisotope (s^{-1})
n = radiohalogen decay constant exponent (dimensionless)
K_g = a constant establishing the level of noble radiogas leakage from fuel
K_h = a constant establishing the level of radiohalogen leakage from fuel

11.1.3.5 Coolant Activation Products

The coolant activation products are not adequately correlated by simple equations. Design basis concentrations in reactor water and steam and have been estimated conservatively. [11.1-11]

11.1.3.6 Noncoolant Activation Products

The activation products formed by activation of impurities in the coolant, or by corrosion of irradiated system materials, are not adequately correlated by simple equations. The design basis source terms of noncoolant activation products have been estimated conservatively. Carryover of these isotopes from the reactor water to the steam is estimated to be <0.1% (<0.001 fraction).

11.1.3.7 Tritium

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

1. Activation of naturally occurring deuterium in the primary coolant;
2. Nuclear fission of $UO_{\{2\}}$ fuel; and
3. Neutron reactions with boron used in reactivity control rods.

The tritium, formed in control rods (which may be released from a BWR in liquid or gaseous effluents), is believed to be negligible. Activation of deuterium in the primary coolant is a prime source of tritium available for release from a BWR. Some fission product tritium may also transfer from fuel to primary coolant. This discussion is limited to the uncertainties associated with estimating the amounts of tritium generated in a BWR which are available for release.

All of the tritium produced by activation of deuterium in the primary coolant is available for release in liquid or gaseous effluents. The tritium formed in BWR can be calculated using the equation:

$$R_{act} = \frac{\Sigma \phi V \lambda}{3.7 \times 10^4 P} \qquad (11.1\text{-}5)$$

where

R_{act} = tritium formation rate by deuterium activation (μCi/s/MWt)
SIGMA = macroscopic thermal neutron cross section (cm^{-1})
phi = thermal neutron flux (neutrons/(cm^2)(s))
V = coolant volume in core (cm^3)
lambda = tritium radioactive decay constant (1.78×10^{-9} s^{-1})
P = reactor power level (MWt)

The fraction of tritium produced by fission which may transfer from fuel to the coolant (which will then be available for release in liquid and gaseous effluents) is much more difficult to estimate. However, since zircaloy-clad fuel rods are used in BWRs, essentially all fission product tritium will remain in the fuel rods unless defects are present in the cladding material.

The study made at Dresden Unit 1 in 1968 by the U.S. Public Health Service (USPHS) suggests that essentially all of the tritium released from the plant could be accounted for

by the deuterium activation source. For purposes of estimating the leakage of tritium from defective fuel, it can be assumed that it leaks in a manner similar to the leakage of noble radiogases.

Thus, one can use the empirical relationship described as the "diffusion mixture" when predicting the source term of individual noble gas radioisotopes as a function of the total noble gas source term. The equation which describes this relationship is:

$$R_{dif} = KY\lambda^{0.5} \tag{11.1-6}$$

where

$R_{\{dif\}}$ = leakage rate of tritium from fuel (μCi/s)
Y = fission yield fraction (atoms/fission)
lambda = radioactive decay constant (s^{-1})
K = a constant related to total tritium leakage rate

Tritium formed in the reactor is generally present as tritiated oxide (HTO) and to a less degree as tritiated gas (HT). Tritium concentration in the steam formed in the reactor will be the same as in the reactor water at any given time. This tritium concentration will also be present in condensate and feedwater. Since radioactive effluents generally originate from the reactor and power cycle equipment, radioactive effluents will also have this tritium concentration. Condensate storage receives treated water from the radioactive waste system and reject water from the condensate system. Thus, all plant process water will have a common tritium concentration.

Off-gases released from the plant will contain tritium, which is present as tritiated gas (HT) resulting from reactor water radiolysis as well as tritiated water vapor (HTO). In addition, water vapor from the turbine gland seal steam packaging exhauster, and a less amount present in ventilation air (due to process steam leaks or evaporation from the sumps, tanks, and spills, and floors), will also contain tritium. The remainder of the tritium will leave the plant in liquid effluents or with solid wastes.

Recombination of radiolysis gases in the air ejector off-gas system will form water, which is condensed and returned to the main condenser. This tends to reduce the amount of tritium leaving in gaseous effluents. Reducing the gaseous tritium release will result in a slightly higher tritium concentration in the plant process water. Reducing the amount of liquid effluent discharged will also result in a higher process coolant equilibrium tritium concentration.

Essentially, all tritium entering the primary coolant will eventually be released to the environs, either as water vapor and gas to the atmosphere, a liquid effluent to the plant discharge, or as solid waste. Reduction due to radioactive decay is negligible due to the 12-year half-life of tritium.

The USPHS study at Dresden Unit 1 estimated that approximately 90% of the tritium released was observed in liquid effluent, with the remaining 10% leaving as gaseous effluent. Efforts to reduce the volume of liquid effluent discharges may change this distribution so that a greater amount of tritium will leave as gaseous effluent. From a practical standpoint, the fraction of tritium leaving as liquid effluent may vary between 5 and 15% with the remainder leaving in the gaseous effluent.

11.1.4 Fuel Fission Product Inventory

Fuel fission product inventory information is used in establishing fission product source terms for accident analysis and is, therefore, addressed in Chapter 15.

11.1.5 Process Leakage Sources

Process leakage results in potential release paths for noble gases and other volatile fission products via ventilation systems. Liquid from process leaks are collected and routed to the liquid-solid radwaste system. Radionuclide releases via ventilation paths are at extremely low levels and have been insignificant compared to process off-gas releases from operating BWR plants.

Leakage of fluids from the process system will result in the release of radionuclides into plant buildings. In general, the noble radiogases will remain airborne and will be released to the atmosphere with little delay via the building ventilation exhaust ducts. The radionuclides will partition between air and water, and airborne radioiodines may "plateout" on metal surfaces, concrete, and paint. A significant amount of radioiodine remains in the air or is desorbed from surfaces. Radioiodines are found in ventilation air as methyl iodide and as inorganic iodine which is defined here as particulate, elemental, and hypoidodus acid forms of iodine. Particulates will also be present in the ventilation exhaust air.

An evaluation of the radioactive releases from ventilation systems, for compliance with 10 CFR 50, Appendix I and 10 CFR 20, is given in Section 11.3.

11.1.6 Other Releases

All other releases are covered in Section 11.2.

11.1.7 Radioactivity Sources for Ventilation Systems

The potential for radioactivity sources for the ventilation system are from the following systems: [11.1-12]

A. Drywell equipment drain sump system,

B. Reactor building equipment drain tank system,

C. Radwaste building equipment drain sump system,

D. Turbine building equipment drain sump system,

E. Drywell floor drain sumps system,

F. Reactor building floor drain sumps system,

G. Radwaste building floor drain sumps system, and

H. Turbine building floor drain sumps system.

The quantity of potential liquid leakage or drainage and the point source are identified in Tables 11.1-4, 11.1-5, 11.1-6, 11.1-7, 11.1-8, 11.1-9, 11.1-10, 11.1-11. Any dissolved radioactive gases will come to equilibrium between the liquid and compartment atmosphere and provide the source of any radioactivity in the ventilation systems during normal plant operation.

11.1.8 Sources Not Normally Part of the Radioactive Waste Management Systems

There are two site release points for gaseous effluent: the ventilation chimney and the reactor building vent stack. There are no release points for gaseous effluent that are not a part of the radioactive waste management system.

There are three site release points for liquid effluent, spray canal blowdown, south diffuser pipe, and the discharge bay. There are no release points for liquid effluent that are not a part of the radioactive waste management system.

Radiation monitors continuously monitor the gaseous and liquid discharge stream and alert the control room operator in case the effluent stream exceeds the predetermined level of radioactivity.

The estimated quantity of tritium in the effluent stream discharged to the environment is discussed in subsection 11.1.3.7.

Table 11.1-1**

REACTOR WATER FISSION PRODUCTS
(Based on 200,000 μCi/s - 30 Minute Holdup Time)

Isotope	Half-Life	Concentration (μCi/cc)*
I-131	8.05 days	1.3×10^{-1}
I-132	2.3 hr.	6.8×10^{-1}
I-133	21 hr.	7.6×10^{-1}
I-134	52 min.	9.4×10^{-1}
I-135	6.7 hr.	9.1×10^{-1}
I-136	86 sec.	7.7×10^{-2}
I-137	22 sec.	6.2×10^{-2}
I-138	5.9 sec.	2.2×10^{-2}
Br-83	2.3 hr.	7.4×10^{-2}
Br-84	32 min.	1.1×10^{-1}
Br-85	3 min.	5.4×10^{-2}
Br-87	56 sec.	5.5×10^{-2}
Br-88	16 sec.	3.2×10^{-2}
Tc-m99	6.04 hr.	9.5×10^{-2}
Mo-Tc-99m	2.78 days	4.5×10^{-1}
Total		4.5 μCi/cc

* Concentrations per plant

** The values noted in this table represent design basis concentrations and remain valid for core uprate to 2957 MWt.

Table 11.1-2*

CONCENTRATIONS OF ACTIVATION
PRODUCTS IN REACTOR WATER

Radioisotope	Half-Life	Concentration in Reactor Water (μCi/cc)
Soluble		
F-18	1.8 hr.	0.004
Mn-56	2.56 hr.	0.002
Ni-65	2.56 hr.	0.00005
Zn-69m	13.8 hr.	0.00001
Na-24	15 hr.	0.002
W-187	24 hr.	0.00001
Co-58	70 days	0.0004
Co-60	5 yr.	0.00004
Fe-59	45 days	0.0000002
P-32	14 days	0.00002
Cr-51	27 days	0.0002
Ag-110m	270 days	0.00006
Mn-54	300 days	0.000002
Zn-65	245 days	0.000001
Total Soluble		0.01
Insoluble		
Mn-56	2.56 hr.	0.05
Co-58	70 days	0.005
Co-60	5 yr.	0.0005
Fe-59	45 days	0.00008
Mn-54	300 days	0.00004
Cr-51	27 days	0.0003
Ag-110m	270 days	0.000003
Zn-69m	13.8 hr.	0.00002

Table 11.1-2*

CONCENTRATIONS OF ACTIVATION
PRODUCTS IN REACTOR WATER

Radioisotope	Half-Life	Concentration in Reactor Water (µCi/cc)
W-187	24 hr.	0.003
Ni-65	2.56 hr.	0.0002
Zn-65	245 days	0.000001
Total Insoluble		0.06
Total Activity (Soluble and Insoluble)		0.07

* The values noted in this table represent design basis concentrations and remain valid for core uprate to 2957 MWt.

Table 11.1-3

ORIGINAL REACTOR AND RECIRCULATION SYSTEM

	Parameter	Reactor Design		Turbine Maximum
1.	Reactor Power (MWt)	2237		2511
2.	Core - active fuel length - in.	144 (145.25 new length)		---
	equivalent diameter - in.	182.2		---
	circumscribed diameter - in.	189.7		---
3.	Number of Fuel Assemblies	724		---
4.	Overall Average Core Power Density watts/cc	36.65 (34.2 for new length)		40.48
5.	Total Coolant Flow Rate through the Core - lb/hr	98×10^6		98×10^6
6.	Primary Steam Flow Rate - lb/hr	8.605×10^6		9.764×10^6
7.	Core Power Peaking Factors:	Max. at Core □	At Core Boundary	
	$\left.\frac{Pmax}{Pave}\right]_z$ (Axial)	1.57	0.7	
	$\left.\frac{Pmax}{Pave}\right]_R$ (Radial)	1.50	0.7	

Table 11.1-3 (Continued)

ORIGINAL REACTOR AND RECIRCULATION SYSTEM

Parameter	Reactor Design		Turbine Maximum
8. Core Volume Fractions:			
Material	Density (gm/cc)	Volume Fraction	Volume Fraction
$UO_{\{2\}}$	10.4	0.254	.0254
Zr	6.4	0.130	0.130
H_2O	1.0	0.334	0.296
Void	0	0.282	0.320
9. Reactor Operating Pressure - psia	1015		1015
10. Average Water Density Between Core and Vessel - gm/cc	0.73		0.73
11. Average Water Density Below Core - gm/cc	0.74		0.74
12. Average Water-Steam Density Above Core:			
In the Plenum Region - gm/cc	0.27		---
Above the Plenum (homogenized) gm/cc	0.52		---
13. Nitrogen-16 activity of steam leaving the vessel (average gamma energy of 6.2 MEV/()	4.15×10^5 MEV/cc-sec 54.6 Ci/sec		4.73×10^5 MEV/cc-sec 70.3 Ci/sec

Table 11.1-4*

DRYWELL EQUIPMENT DRAIN SUMP SOURCES
FOR RADIOACTIVE MATERIAL**

Leak or Drain (Rates are for one Reactor)	Gal. per day	Dchrg. Vol. Gal.	Activity Concentration (μCi/cc)		Daily Activity (μCi/day)	
			Norm.	Max.	Norm.	Max.
Recirc. Pump Seal Leakage	1400	---	1×10^{-1}	5	6×10^{5}	3×10^{7}
Recirc. Valve Seal Leakage	380	---	1×10^{-1}	5	2×10^{5}	7×10^{6}
Steam Valve Seal Leakage	190	---	1×10^{-4}	5×10^{-3}	7×10^{1}	3×10^{3}
RCIC and HPCI System Valve Leakage	190	---	1×10^{-4}	5×10^{-3}	7×10^{1}	3×10^{3}
Clean Up Valve Seal Leakage	100	---	1×10^{-1}	5	4×10^{4}	2×10^{6}
Shut Down Valve Seal Leakage	190	---	1×10^{-4}	5×10^{-3}	7×10^{1}	3×10^{5}
Control Rod Drive Valve Seal Leakage	100	---	5×10^{-6}	1×10^{-3}	2	4×10^{2}
Totals of Continuous Inputs	2600	---	7×10^{-2}	3	7×10^{5}	3×10^{7}
Relief Valve Blowoff (Infrequent)	---	<300	10^{-4}	---	---	---
Recirc. Drains (Infrequent)	---	<800	10^{-1}	2	---	---
Vent Reactor for HydroTest (1/yr.)	---	<100	$<10^{-3}$	---	---	---
Steam Line Drains (Startup)	---	---	$<10^{-3}$	---	(Heat Exchanger)	
Control Rod Drive Drains (Infrequent)	---	100	$<10^{-3}$	---	---	---
Bellows Drain (Startup)	---	<1000	$<10^{-3}$	---	---	---

* See notes following this Table.

** Original estimated numbers.

Table 11.1-4*

DRYWELL EQUIPMENT DRAIN SUMP SOURCES
FOR RADIOACTIVE MATERIAL**

NOTES

THE FOLLOWING ARE APPLICABLE TO TABLES 11.1-4 THROUGH 11.1-11:

1. The following definitions apply to Tables 11.1-4 through 11.1-11:

   Normal Volume - Expected volume during steady state normal operation.

   Maximum Volume - Maximum expected volume during unsteady state operation such as startup, shutdown, high equipment leakage, etc.

   Normal Activity - Activity level expected during operation with no fuel leaks — corrosion product reactor water activity concentration of 0.1 μCi/cc.

   Maximum Activity - Activity level expected during operation with fuel leak rate equivalent to reactor water activity concentration of 4.6 μCi/cc (corrosion and fission products present).

   Caution: Maximum volume and maximum activity are not necessarily concurrent.

2. Waste system input activities are based on a reactor water-to-steam decontamination factor of 10^{-3}.

3. Activity shown as of plant origin and does not include background.

Table 11.1-5*

REACTOR BUILDING EQUIPMENT DRAIN TANK
SOURCES FOR RADIOACTIVE MATERIAL**

Leak or Drain (Rates are for one Reactor)	Gal. per day	Dschrg. Vol. Gal.	Activity Concentration (µCi/cc)		Daily Activity (µCi/day)	
			Norm.	Max.	Norm.	Max.
Clean Up Pump Seal Leakage	480	---	1×10^{-3}	5×10^{2}	2×10^{3}	8×10^{4}
Control Rod Drive Sump Seal Leakage	240	---	5×10^{-6}	1×10^{-3}	5	9×10^{2}
Scram Valve Seal Leakage	510	---	5×10^{-6}	1×10^{-3}	10	2×10^{3}
Feed Valve Seal Leakage	290	---	1×10^{-6}	5×10^{-5}	1	5×10^{2}
Miscellaneous Valve Seal Leakage	1400	---	1×10^{-2}	5×10^{-1}	6×10^{4}	3×10^{6}
Clean Up Sample Drains	430	---	3×10^{-2}	2	6×10^{4}	3×10^{6}
Clean Up Sludge Pump Leakage	---	---	---	---	---	---
Clean Up Decant Pump Leakage	---	---	---	---	---	---
Totals of Continuous Inputs	3400	---	9×10^{-3}	4×10^{-1}	1×10^{5}	5×10^{6}
Shutdown — Tube Side	---	350	$<10^{-1}$	---	---	---
Regenerative — Shell and Tube Side	---	1000	$<10^{-1}$	---	---	---
Non-Regenerative — Tube Side	---	250	$<10^{-4}$	---	---	---
Control Rod Hydraulic System Drains (Infrequent)	---	200	5×10^{-6}	---	---	---
Clean Up Relief Valve Drains (Infrequent)	---	---	---	---	---	---
Clean Up Phase Separator Drains and Overflow	---	---	2×10^{-2}	---	---	---

* See notes following Table 11.1-4.

** Original estimated numbers.

Table 11.1-6*

RADWASTE BUILDING EQUIPMENT DRAIN SUMP
SOURCES FOR RADIOACTIVE MATERIAL**

Leak or Drain (Rates are for two Reactors)	Gal. per day	Dschrg. Vol. Gal.	Activity Concentration (µCi/cc)		Daily Activity (µCi/day)	
			Norm.	Max.	Norm.	Max.
Condensate Sludge Pump Leakage	---	---	---	---	---	^^---
Waste Collector Pump Leakage	---	---	---	---	---	---
Waste Sample Pump Leakage	---	---	---	---	---	---
Condensate Decant. Pump Leakage	---	---	---	---	---	---
Waste Surge Pump Leakage	---	---	---	---	---	---
Precoat and Filter Aid Pump Leakage	---	---	---	---	---	---
Condensate Phase Separator Tanks, Drains, and Overflow	---	---	---	---	---	---
Waste Sample and Waste Sludge Tanks, Drain, and Overflow	---	---	---	---	---	---
Radwaste Filter Head Drain	---	---	---	---	---	---
Waste Demineralizer and Spent Resin Tank, Drain and Overflow	---	---	---	---	---	---
Totals of Continuous Inputs	1000	---	1×10^{-4}	5×10^{-3}	4×10^{2}	2×10^{4}

* See notes following Table 11.1-4.

** Original estimated numbers.

Table 11.1-7*

TURBINE BUILDING EQUIPMENT DRAIN SUMP
SOURCES FOR RADIOACTIVE MATERIAL**

Leak or Drain (Rates are for one Reactor)	Gal. per day	Dschrg. Vol. Gal.	Activity Concentration (μCi/cc)		Daily Activity (μCi/day)	
			Norm.	Max.	Norm.	Max.
Condensate Pump Seal Leakage	1900	---	1×10^{-4}	5×10^{-3}	7×10^{2}	3×10^{4}
Feed Pump Seal Leakage	720	---	1×10^{-6}	5×10^{-5}	3	1×10^{2}
Off Gas Drains	2100	---	0	8×10^{-2}	0	6×10^{5}
Condensate and Feed System Sample Drain	960	---	1×10^{-5}	5×10^{-4}	4×10^{1}	2×10^{3}
Totals of Continuous Inputs	5700	---	4×10^{-5}	3×10^{-2}	8×10^{2}	7×10^{5}
Condensate Backwash Receiving Tank Drain and Overflow	---	---	5×10^{-6}	---	---	---
Heater Vents and Drains (6/yr.)	---	500	$<10^{-4}$	---	---	---
Heater Maintenance Drains (Infrequent)	---	1500	$<10^{-4}$	---	---	---
Condensate Maintenance Drains	---	100	$<10^{-4}$	---	---	---
Feed Heater Relief Valves (Infrequent)	---	---	5×10^{-6}	---	---	---

* See notes following Table 11.1-4.

** Original estimated numbers.

Table 11.1-8[*]

DRYWELL FLOOR DRAIN SUMPS SOURCES
FOR RADIOACTIVE MATERIAL[**]

Leak or Drain (Rates are for one Reactor)	Gal. per day	Dschrg. Vol. Gal.	Activity Concentration (µCi/cc)		Daily Activity (µCi/day)	
			Norm.	Max.	Norm.	Max.
Control Rod Drive Flange Leakage	2500	---	5×10^{-6}	1×10^{-3}	5×10^{1}	1×10^{4}
Floor Drains	500	---	5×10^{-6}	1×10^{-3}	10	2×10^{3}
Totals of Continous Inputs	3000	---	5×10^{-6}	1×10^{-3}	6×10^{1}	1×10^{4}
Control Blade Backseat Leakage (Infrequent)	---	---	$<10^{-3}$	---	---	---
Closed Cooling Water System Drains (Infrequent)	---	100	$<10^{-3}$	---	---	---
Vent Cooler Drains (Infrequent)	---	---	10^{-4}	---	---	---

* See notes following Table 11.1-4.

** Original estimated numbers.

Table 11.1-9*

REACTOR BUILDING FLOOR DRAIN SUMPS
SOURCES FOR RADIOACTIVE MATERIAL**

Leak or Drain (Rates are for one Reactor)	Gal. per day	Dschrg. Vol. Gal.	Activity Concentration (μCi/cc)		Daily Activity (μCi/day)	
			Norm.	Max.	Norm.	Max.
Floor Drains	2000	---	1×10^{-4}	2×10^{-2}	8×10^{2}	1×10^{5}
Total of Continuous Inputs	2000	---	1×10^{-4}	2×10^{-2}	8×10^{2}	1×10^{5}
Non-regenerative Shell Side	---	350	$<10^{-3}$	---	---	---
Shutdown □ Shell Side	---	650	$<10^{-3}$	---	---	---

* See notes following Table 11.1-4.

** Original estimated numbers.

Table 11.1-10*

RADWASTE BUILDING FLOOR DRAN SUMPS
SOURCES FOR RADIOACTIVE MATERIAL**

Leak or Drain (Rate is for two Reactors)	Gal. per day	Dschrg. Vol. Gal.	Activity Concentration (μCi/cc)		Daily Activity (μCi/day)	
			Norm.	Max.	Norm.	Max.
Floor Drain Collector Tank Pump Leakage	---	---	---	---	---	---
Waste Collector Tank Drain and Overflow	---	---	---	---	---	---
Floor Drain Collector Tank Drain and Overflow	---	---	---	---	---	---
Chemical Waste Pump Leakage	---	---	---	---	---	---
Chemical Waste Tank Drain and Overflow	---	---	---	---	---	---
Floor Drain Filter Head Drain	---	---	---	---	---	---
Floor Drain Sample Pump Leakage	---	---	---	---	---	---
Floor Drain Sample Tank Drain and Overflow	---	---	---	---	---	---
Totals of Continuous Inputs	1000	---	2×10^{-5}	9×10^{-4}	8×10^{1}	3×10^{3}

---

* See notes following Table 11.1-4.

** Original estimated numbers.

Table 11.1-11*

TURBINE BUILDING FLOOR DRAIN SUMPS
SOURCES FOR RADIOACTIVE MATERIAL**

Leak or Drain (Rates are for one Reactor)	Gal. per day	Dschrg. Vol. Gal.	Activity Concentration (μCi/cc)		Daily Activity (μCi/day)	
			Norm.	Max.	Norm.	Max.
Floor Drains	2000	---	1×10^{-6}	5×10^{-5}	8	4×10^{2}
Total of Continuous Inputs	2000	---	1×10^{-6}	5×10^{-5}	8	4×10^{2}

* See notes following Table 11.1-4.

** Original estimated numbers.

11.2 LIQUID WASTE MANAGEMENT SYSTEMS

The various liquid radwaste treatment methods utilized to reduce the discharge of radioactivity to the lowest practicable limit are discussed in detail in this section. P&IDs M-51, M-52 and M-57 show the major flow paths of the liquid radwaste systems. P&IDs M-53, M-54, M-57 and M-59 show the interfaces with the solid radwaste systems. P&IDs M-51 and M-543, show the liquid radioactive waste discharge pathway. Radioactive Effluent Release Reports are submitted to the NRC in accordance with Technical Specifications requirements and specifies the quantities of each radionuclide released to the unrestricted areas in both the liquid and gaseous (see Section 11.3) effluents during the period. These reports shall be in accordance with the format and content required by Regulatory Guide 1.21, Revision 1, dated June 1974. [11.2-1]

11.2.1 Design Bases

The liquid radioactive waste system collects, treats, stores, and disposes as necessary all radioactive liquid wastes. Liquid wastes are collected in sumps and drain tanks in the various buildings, then transferred to the appropriate tanks in the radwaste building for further treatment or temporary storage, and discharge. If the waste meets the requirements for re-use, it is recycled back into the contaminated condensate storage tanks. If it does not meet recycling requirements, the contents are either returned for reprocessing or discharged from the plant. [11.2-2]

The Quad Cities radwaste system was designed to achieve a radionuclide concentration, for discharge of any batch, when diluted with the circulating water flow, of less than 1×10^{-7} μCi/cc on an unidentified basis. The radionuclide concentration on an annual average basis are expected to be even lower than five mrem which meets "low as practicable" criterion. [11.2-3]

Batches with radioactivity concentrations low enough to allow discharge to the river are released to the discharge flume weir. Wastes to be discharged from the system are handled on a batch basis with each one being analyzed and handled appropriately. These batches are diluted with condenser circulating water effluent in order to achieve a discharge concentration, at the point of entry into the river, below the limits set by 10 CFR 20, and Illinois and Iowa state regulations. [11.2-4]

The design provides for dewatering and solidification of sludges and ion exchange resins to facilitate their storage and disposal offsite as solid wastes. The dewatering and solidification of radioactive sludges and ion exchange resins is covered in Section 11.4. [11.2-5]

A waste demineralizer decontamination factor of 100 was used for designing the demineralizer resin bed depth and volume. This decontamination factor was also used in the original determination of the individual nuclide concentrations in the effluent based on the original estimated influent nuclide concentrations. [11.2-6]

11.2.2 System Description

The process and instrumentation diagrams P&IDs M-51, M-52 and M-57 show the collection and processing flow paths of the liquid radioactive waste system. The liquid radioactive waste system interfaces with the solid radioactive waste system are shown in P&IDs M-57, M-53 and M-59. The liquid radioactive waste discharge pathway is shown in P&IDs M-51 and M-543. The radwaste tank capacities are given in Table 11.2-1. [11.2-7]

Table 11.2-1 shows the approximate inventory by isotope for each of the radwaste tanks shown in P&IDs M-51, M-52, M-57, M-53 and M-54. This table assumes that all tanks are filled to the maximum listed tank volume, which is not the case under normal operating conditions. These values are based on operating data gathered at Quad-Cities. Table 11.2-2 shows the estimated inventory by isotope for solid radioactive material collected in radwaste tankage. Because tanks may have freshly collected batches, the waste and floor drain sample tanks' isotopic inventory had a 12-hour decay time applied to more realistically show what will be present after process time through the respective systems. [11.2-8]

The liquid radioactive waste disposal system, a batch-type system, collects and processes waste in an efficient manner. Processed liquid wastes are returned to the plant for re-use, discharged from the plant with dilution from the discharge flume weir flow or returned to the station's radwaste systems for additional treatment. Processed liquid wastes can be discharged in batches that are verified as meeting the necessary standards of radioactivity concentration and chemical purity. Additionally, the required composite samples are accumulated from samples taken from the discharge sample tanks prior to liquid discharge to the river. Furthermore, liquid waste discharge records are maintained onsite. [11.2-9]

In an effort to realize the most effective treatment methods the various wastes are collected separately by type. This segregates lower activity wastes from higher activity wastes and limits the volume and activity of radioactive waste discharged to the environment. The Quad Cities liquid radwaste system is designed to treat all liquid wastes prior to discharge. Four subsystems are involved and include: [11.2-10]

1. Floor drain,

2. Waste collector,

3. Chemical waste, and

4. Laundry drain.

The sources of liquid radioactive waste at Quad Cities are from the equipment drains, the floor drains, the laboratory drains, chemical decontamination solutions, and laundry drains. [11.2-11]

Radioactive liquid wastes are received and processed in the four subsystems listed previously and shown in P&IDs M-51, M-52, M-57, M-53 and M-543. To ensure that wastes are processed through the equipment provided in each of these systems, the following system features are incorporated: [11.2-12]

A. Processing equipment is designed and selected so maintenance requirements are minimized, i.e., minimal rotating parts and shielding for access while other equipment is operational.

B. The floor drain filter, the waste filter, and the spare floor drain filter are crossconnected so one may be used in place of the other if a unit requires maintenance.

C. Pumps are crossconnected with other pumps so a malfunctioning pump does not adversely affect process flow capability or hinder repair/replacement time.

D. Since the subsystems are batch operations, rather than continuous operations, they are preceded by collection tanks that can accumulate wastes. A floor drain surge tank is also provided to accumulate system surges. Normally, the floor drain surge tank, is less than 20% full except during outages. After collection, the tank's contents can be further processed in the radwaste system by filtration and ion-exchange processes.

E. Certain operations are also subject to scheduling and can be delayed in the event of mechanical problems. Examples are:

   1. Transfer to and from the cleanup and condensate phase separators,

   2. Transfer to and from the waste sludge tank and spent resin tanks,

F. Steam cleaning connections are provided for the waste, floor drain, and spare filters so that in-place cleaning can be performed in about three hours.

G. Cycle times allowed for in the design permit filter backwashing and precoating as a part of normal procedure.

Waste and floor drain demineralizer resins can be replaced in less than a shift, the major task being handling of the resins from container to demineralizer. Resin replacement is an infrequent operation. An essential factor is to maintain an appropriate resin inventory at the station for replacement. [11.2-13]

Wastes that accumulate in the floor drain collector tank are considered low purity waste. These wastes, which are moderately conductive and generally have low radioactive concentrations, are processed through a filter (to remove insoluble material) and one or more demineralizers (to remove soluble material) and routed to the floor drain sample tanks. Following batch sampling, the wastes are normally outside the station criteria for re-use in the plant and are returned to the radwaste system for reprocessing or to river discharge. If wastes are within the station criteria for re-use in the plant, then wastes can be returned to the condensate storage system for re-use. [11.2-14]

Wastes accumulated in the waste collector tank are considered high purity waste. These low conductivity wastes have variable radioactive concentrations dependent upon their area of collection. Normally, these wastes are processed through a filter, (to remove insoluble material) and one or more demineralizers (to remove soluble material) and routed to the waste sample tanks. Following batch sampling, the wastes are normally returned to the condensate storage system for reuse. Water outside the station criteria for re-use in the plant is returned to the radwaste system for reprocessing or to river discharge.

The waste collected in the chemical waste tank may be transferred to the floor or equipment drain system or the chemical waste sample tank. From the chemical waste sample tank, the waste may be transferred to the new spent resin tank for further processing or to the floor or equipment drain systems. Inputs to the chemical waste system include laboratory drains, leakage from Reactor Water Cleanup and Fuel Pool Demin manual drain valves and decon operations.[11.2-15]

Detergent wastes collected in the laundry drain tanks are generally low in activity and produced in small volumes. These factors result in a low total activity discharge to the environment. Laundry wastes do not require treatment other than filtration. Laundry wastes are filtered and sent to the laundry sample tank for sampling and further filtering, if required, prior to transfer to the floor drain collector tank or river discharge tank. [11.2-16]

Resins which are basically "spent" from processing waste or floor drain collector waste can be further used for initial processing of chemical wastes. Batch discharges from the river discharge tank consist mainly of laundry water filtered prior to discharge, or floor drain, collector, waste collector water which has been processed, but does not meet the criteria (such as high organic content) for re-use. Predicted daily average radioactivity quantities anticipated concurrent with design basis fuel leaks are given in Table 11.2-3. [11.2-17]

Liquid effluents are released from the river discharge tank in batches, which is the only liquid release path from Radwaste utilized after sampling and analysis, through a monitored radioactive liquid waste line. The monitor station has an alarm. The discharge line directs the effluent to the discharge flume weir and provides adequate dilution with station condenser circulating water to assure that the effluent reaching the river is below the regulatory requirements for the mixture of all the nuclides released. This line and the now abandoned discharge line directing effluent to the south diffuser line are the only release paths used since May 1984, when the Quad Cities station condenser circulating water system was changed to discharge directly into the river rather than to the spray canal. This change increases the dilution flow when radioactive liquids are batch released from the river discharge tank to the circulating water system. Table 11.2-4 shows the expected activity in the discharge bay and the Mississippi River based on historical data and pre-uprate conditions. Core uprate to 2957 MWt is expected to increase the activity in the liquid waste discharged by the percentage of the uprate, i.e., 18%. The river discharge tank and either the river discharge pump or the waste surge pump provide a single discharge point from the station. The discharge is monitored by a radiation monitor which has no automatic functions and is provided for record and alarm only. The radiation monitor is described in Section 11.5.2.7. Since the river discharge tank is the only tank used for discharge, the chances for inadvertent discharges are minimized. Spool pieces are provided to allow water in the river discharge tank to be returned to radwaste for additional treatment. Under discharge conditions, the discharge line is isolated from the radwaste process line. [11.2-18]

The expected concentration of each principal radionuclide (half-life greater than 30 minutes) which are contained in the various components of the liquid radwaste treatment system and the associated piping and valves for each of the above components, are shown in Tables 11.2-1 and 11.2-2. [11.2-19]

Four deep-bed demineralizers can be used in the radwaste system. Pressure precoat-type filter/demineralizers, using powdered ion exchange resins and filter media, are used in the fuel pool cleanup system, reactor water cleanup (RWCU) system condensate feedwater system, and radwaste system filters. The deep-bed ion exchange resins and the powdered ion exchange resins are not regenerated. [11.2-20]

All radwaste filter sludges are either collected in the waste sludge tank or the condensate phase-separators. Spent resins from the waste demineralizer are collected in the waste spent resin tank. Spent resins from the A, B, and C demineralizers are collected in the max-recycle spent resin tank. [11.2-21]

The waste sludge tank solids isotopic inventory is an approximation dependant upon which streams have contributed to the total batch activity mixture. Normal input to the waste sludge tank is from the floor drain, spare floor drain, and waste filter backwashes, and floor drain collector tank blowdowns. [11.2-22]

Cleanup system filter/demineralizer sludge is collected in the cleanup phase-separators. Condensate filter/demineralizer sludge and fuel pool system filter demineralizer sludge are generally collected in the condensate phase-separators. Excess backwash water is removed by decantation and the sludge is accumulated for radioactive decay and further processing. The sludge consists of filter precoat material (powered resins and filter media), activated corrosion products, fission products, and other insoluble materials. Decant water from the phase separators is transferred to the waste collector tank. Decant water from the condensate phase separators may be transferred to the floor drain collector tank or waste collector tank. [11.2-23]

Overall control of the radwaste processing system is exercised from the radwaste building control room. A main panel in this room contains the instrumentation and control components for system operation. In addition, local control stations are provided for the filter backwash and precoating operations, and waste processing. Alarm annunciation is in the radwaste control room or at local panels. [11.2-24]

11.2.2.1 Protection Against Accidental Discharge

Design redundancy, instrumentation for detection and alarm of abnormal conditions, and procedural controls provide protection against accidental discharge. The arrangement of the radwaste building and waste processing methods substantially immobilize the wastes within the station. This is to assure that in the event of a failure of any of the liquid waste equipment or errors in operation of the system, the potential for inadvertent release of liquids is extremely small. For example, the system's tanks, which may contain high-level activity inventories, are located in the radwaste building basement at a floor elevation 21 feet 7 inches below grade level. The tanks include: [11.2-25]

A. Four condensate phase separator tanks,

B. Chemical waste tank,

C. Floor drain collector tank,

D. Waste collector tank,

E. Waste spent resin tank, and

F. Waste sludge tank.

If the largest tank in the radwaste basement (collector tank) failed, 22,000 gallons of water would be released and would result in a basement liquid waste level of less than 4 feet. This level of water would not be capable of raising any other basement tank since all tanks are supported by legs about 3 feet above floor level. [11.2-26]

Other equipment such as filters, demineralizers, centrifuges, pumps, max-recycle spent resin tank, etc., are contained within cells or rooms so that leakage is contained within the building. [11.2-27]

The following tanks are located in the radwaste tank farm: [11.2-28]

A. River discharge tank,

B. Two floor drain sample tanks,

C. Two waste sample tanks,

D. Laundry sample tank, and

E. Chemical waste sample tank.

These tanks contain filtered or otherwise treated water with the exception of the chemical waste sample tank.

The tank farm is contained in a concrete basin. This basin is sized to retain all of the water contained in the above tanks should any or all of the tanks leak or fail. There is a drain line which leads from the bottom of the basin and drains into the radwaste building floor drain sump, located in the radwaste basement. A second drain line from the bottom of the basin that leads to the storm sewer has been capped downstream of valve 1/2-2001-804. The estimated radioactivity content in these tanks is given in Table 11.2-5. [11.2-29]

Consequently, in the event of leaks or spills from radwaste tank farm tanks or equipment, control of the liquid radioactive waste is assured to the extent that there will be no spread of radioactivity to the grounds or other areas outside the confines of the station. Sumps and pumps collect any such waste and return them to the appropriate system for processing. The waste surge tank was renamed the river discharge tank and the piping was modified such that it is the single discharge tank for radioactive liquid waste from the station to the river. [11.2-30]

11.2.2.2 Administration

The principal administrative areas involved in maintaining an operational system are daily planning of radwaste processing, control of the station water inventory and conducting a preventive maintenance program. [11.2-31]

Radwaste system planning assures that wastes are processed in a timely manner and that station operations, such as refueling, and maintenance activities (draining, flushing, decontamination, etc.,) are coordinated to prevent the flooding of the radwaste system with unexpected quantities of water.

A daily planning of radwaste operations is conducted, to instruct operators on the various waste water movements to be made. This planning can anticipate inputs such as from cleanup and condensate filter/demineralizer backwashing, and draining and flushing of equipment for maintenance. The net result of such planning is to keep the wastes moving through the system, recognize and correct difficulties as they may occur, and keep tank inventories low. Proper planning and its subsequent execution leaves capability and capacity available for unusual conditions (e.g., equipment malfunctions, and condenser tube leaks.) [11.2-32]

In addition to operations planning, daily log sheets provide data on waste volumes processed through the various subsystems. Charting of such data shows trends in station performance. Thus, a systematic increase in floor drain volume would lead to a search for its cause and a plan for corrective maintenance. Waste volumes are thus kept in the "normal" design range so system capacity is not impaired by continued abnormal inputs.

Station inventory control is done to minimize the necessity for discharging waste water because of excessive inputs via the makeup system. Both the planning and water inventory control activities are also useful in detecting abnormal inputs to radwaste (thus revealing the causes of such inputs and the need for correction). [11.2-33]

Provisions are made in the design of the station to detect leakage from vital fluid carrying systems at and beyond the reactor coolant pressure boundary. These leakage detection methods are discussed in detail in Sections 5.2.5 and 5.2.6. [11.2-34]

The preventive maintenance program has the obvious objective of minimizing unplanned equipment conditions which would affect radwaste performance. The crossties and equipment spares noted previously accommodate such conditions in critical flow paths. [11.2-35]

11.2.2.3 Inspection and Testing

Testing of this system is precluded by its normal day-to-day operation. Inspection is performed per equipment requirements and normal maintenance procedures. [11.2-36]

11.2.3 Radioactive Releases

In passing through the various tanks of the radwaste system, the wastes are subjected to a holdup time which varies from approximately 1 hour to 1 week, and permits decay of the numerous short half-life constituents. [11.2-37]

The procedure for computing the discharge rate involves independent calculation by the Chemistry Technician and Field Supervisor. The results are compared by the Shift Engineer who also determines the discharge rate. This provides additional protection against improper discharge to the river. [11.2-38]

A comparison of liquid radioactive waste discharges and their potential effects on the environment is shown in Table 11.2-6. Table 11.2-6, Line IIB, shows the environs dose rate which would occur for the conditions shown in Table 11.2-3. The dose rate in the discharge flume assumes this to be a person's sole source of drinking water. The dose rates in the river are given assuming each of the two river flows were to occur all year. The two flow cases are obviously conservative, using minimum and average flows for

dilution in the river. The last columns show station contributed dose as a fraction of natural background. [11.2-39]

The station liquid radwaste discharge contribution to population radiation dose is a small fraction of natural background and the natural variations therein (~5 mrem/yr). Thus, the criteria for "as low as reasonably achievable" (ALARA) is satisfied. For conditions where fuel leakage is less than the design basis amount, smaller activity discharges and resultant potential dose rates will occur.

Lines I and IIa of Table 11.2-6 show, respectively, the Curies per year which could be discharged within the legal maximum permitted under 10 CFR 20 and under the ODCM limit of a discharge flume concentration of 10^{-7} μCi/cc.

Based upon operating experience in other BWRs, the maximum permissible concentration (MPC) of the isotopes present is approximately 1.6 x 10^{-6} μCi/cc; thus, discharge flume concentrations of 10^{-7} μCi/cc or less, represent a safety factor of approximately 20 compared to the identified mixture concentration of 1.6 x 10^{-6} μCi/cc.

To show that the liquid wastes discharged continue to be representative of an MPC of about 1.6 x 10^{-6} μCi/cc, quarterly analyses of representative wastes are made. Since station operation is characterized by periodic gross activity, gross iodine analyses of reactor water, and by continuous monitoring of off-gas fission gases, the activity in the station is known. Liquid radwaste activities are characterized by this information and the quarterly liquid waste analyses. Such information and analysis is in keeping with recent revisions to 10 CFR 20 and 10 CFR 50, which require that waste discharge information be known so that estimates of radiation dose exposure of offsite persons can be made. Records of station operation will thus be able also to show the insignificance of liquid waste discharges and that they are ALARA."

The radioactivity released with the liquid wastes is difficult to define since the liquid wastes come from a number of sources and the quantity of activity is a major function of plant operation, including holdup time. The total amount of activity and the relative quantities of each isotope will vary significantly from day to day with varying power levels and leakage from fuel elements. [11.2-40]

The expected average annual activity discharged should be less than one-fourth of that permissible under 10 CFR 20. This estimation assumes that the activity discharged consists only of radioisotopes Sr-90 and Pb-210 which overstates the actual radioactivity contribution to the environs. [11.2-41]

The river discharge tank entry in Table 11.2-1 shows the most significant isotopes which may be present in the combined liquid waste discharged offsite.

Because neither Ra-226 nor Ra-228 of station origin will be present, the discharge concentration for an otherwise unidentified mixture is set at a fixed maximum. However, if certain other radioisotopes which, when determined by the methods set forth in 10 CFR 20, Appendix B, are considered absent, then higher permissible concentrations may be used for discharge. Waste discharges are averaged for the calendar year. Normal river flow further dilutes the specific radioactivity concentrations present. The waste activity actually in the river is of the order of one-thousandth of the maximum permissible concentration per 10 CFR 20 for the mixtures generally discharged. [11.2-42]

Table 11.2-1

EXPECTED INDIVIDUAL NUCLIDE CONCENTRATION μCi/cc
OF LIQUID RADIOACTIVE WASTE IN RADWASTE TANKAGE BASED ON OPERATIONAL DATA

	Waste Collector Tank	Floor Drain Collector Tank	Waste Sample Tank	Floor Drain Sample Tank	Cleanup Phase Separators	Condensate Phase Separators	Chemical Waste Tank	River Discharge Tank	Laundry Drain Sample Tank
Number of tanks	1	1	2	2	4	4	1	1	2
Expected μCi contained in all full tanks*	7×10^3	7×10^3	1×10^4	1×10^4	5×10^7	2×10^8	1×10^3	2×10^4	6×10^2
Expected average μCi contained in all full tanks *	1×10^3	1×10^3	2×10^3	2×10^3	8×10^6	2×10^7	2×10^2	3×10^3	9×10^1
Maximum tank volume per tank (gallon)	22,000	21,000	22,000	21,000	4,500	12,500	5,000	65,000	1,000

* The table assumes that all tanks are filled to the maximum listed tank volume, which is not the case under normal operating conditions.

Table 11.2-1 (Continued)

EXPECTED INDIVIDUAL NUCLIDE CONCENTRATION μCi/cc
OF LIQUID RADIOACTIVE WASTE IN RADWASTE TANKAGE BASED ON OPERATIONAL DATA

Half Life (Days)	Nuclide	Waste Collector Tank	Floor Drain Collector Tank	Waste Sample Tank	Floor Drain Sample Tank	Cleanup Phase Separators	Condensate Phase Separators	Chemical Waste Tank	River Discharge Tank	Laundry Drain Sample Tank
27.7	Cr-51	5×10^{-7}	5×10^{-7}	5×10^{-7}	5×10^{-7}	5×10^{-3}	5×10^{-3}	5×10^{-7}	5×10^{-7}	5×10^{-7}
2.75	Tc-99m	2×10^{-6}	2×10^{-6}	2×10^{-6}	2×10^{-6}	2×10^{-2}	2×10^{-2}	2×10^{-6}	2×10^{-6}	2×10^{-6}
2.36	Np-239	9×10^{-7}	9×10^{-7}	9×10^{-7}	9×10^{-7}	9×10^{-3}	9×10^{-3}	9×10^{-7}	9×10^{-7}	9×10^{-7}
1.38	Ce-143	1×10^{-8}	1×10^{-8}	1×10^{-8}	1×10^{-8}	1×10^{-4}	1×10^{-4}	1×10^{-8}	1×10^{-8}	1×10^{-8}
8.04	I-131	4×10^{-7}	4×10^{-7}	4×10^{-7}	4×10^{-7}	4×10^{-3}	4×10^{-3}	4×10^{-7}	4×10^{-7}	4×10^{-7}
0.185	Ru-105	3×10^{-7}	3×10^{-7}	3×10^{-7}	3×10^{-7}	3×10^{-3}	3×10^{-3}	3×10^{-7}	3×10^{-7}	3×10^{-7}
39.4	Ru-103	1×10^{-8}	1×10^{-8}	1×10^{-8}	1×10^{-8}	1×10^{-4}	1×10^{-4}	1×10^{-8}	1×10^{-8}	1×10^{-8}
0.867	I-133	6×10^{-6}	6×10^{-6}	6×10^{-6}	6×10^{-6}	6×10^{-2}	6×10^{-2}	6×10^{-6}	6×10^{-6}	6×10^{-6}
12.8	Ba-140	5×10^{-7}	5×10^{-7}	5×10^{-7}	5×10^{-7}	5×10^{-3}	5×10^{-3}	5×10^{-7}	5×10^{-7}	5×10^{-7}
0.0954	I-132	3×10^{-9}	3×10^{-9}	3×10^{-9}	3×10^{-9}	3×10^{-5}	3×10^{-5}	3×10^{-9}	3×10^{-9}	3×10^{-9}
0.996	W-187	7×10^{-9}	7×10^{-9}	7×10^{-9}	7×10^{-9}	7×10^{-5}	7×10^{-5}	7×10^{-9}	7×10^{-9}	7×10^{-9}
64	Zr-95	2×10^{-8}	2×10^{-8}	2×10^{-8}	2×10^{-8}	2×10^{-4}	2×10^{-4}	2×10^{-8}	2×10^{-8}	2×10^{-8}
2.75	Mo-99	1×10^{-6}	1×10^{-6}	1×10^{-6}	1×10^{-6}	1×10^{-2}	1×10^{-2}	1×10^{-6}	1×10^{-6}	1×10^{-6}
0.704	Zr-97	2×10^{-7}	2×10^{-7}	2×10^{-7}	2×10^{-7}	2×10^{-3}	2×10^{-3}	2×10^{-7}	2×10^{-7}	2×10^{-7}

Table 11.2-1 (Continued)

EXPECTED INDIVIDUAL NUCLIDE CONCENTRATION μCi/cc
OF LIQUID RADIOACTIVE WASTE IN RADWASTE TANKAGE BASED ON OPERATIONAL DATA

Half Life (Days)	Nuclide	Waste Collector Tank	Floor Drain Collector Tank	Waste Sample Tank	Floor Drain Sample Tank	Cleanup Phase Separators	Condensate Phase Separators	Chemical Waste Tank	River Discharge Tank	Laundry Drain Sample Tank
0.396	Sr-91	6 x 10^{-6}	6 x 10^{-6}	6 x 10^{-6}	6 x 10^{-6}	6 x 10^{-2}	6 x 10^{-2}	6 x 10^{-6}	6 x 10^{-6}	6 x 10^{-6}
3.50	Nb-95	1 x 10^{-8}	1 x 10^{-8}	1 x 10^{-8}	1 x 10^{-8}	1 x 10^{-4}	1 x 10^{-4}	1 x 10^{-8}	1 x 10^{-8}	1 x 10^{-8}
70.8	Co-58	2 x 10^{-8}	2 x 10^{-8}	2 x 10^{-8}	2 x 10^{-8}	2 x 10^{-4}	2 x 10^{-4}	2 x 10^{-8}	2 x 10^{-8}	2 x 10^{-8}
312	Mn-54	3 x 10^{-}	3 x 10^{-8}	3 x 10^{-8}	3 x 10^{-8}	3 x 10^{-4}	3 x 10^{-4}	3 x 10^{-8}	3 x 10^{-8}	3 x 10^{-8}
0.148	Y-92	2 x 10^{-5}	2 x 10^{-5}	2 x 10^{-5}	2 x 10^{-5}	2 x 10^{-1}	2 x 10^{-1}	2 x 10^{-5}	2 x 10^{-5}	2 x 10^{-5}
44.6	Fe-59	2 x 10^{-9}	2 x 10^{-9}	2 x 10^{-9}	2 x 10^{-9}	2 x 10^{-5}	2 x 10^{-5}	2 x 10^{-9}	2 x 10^{-9}	2 x 10^{-9}
0.275	I-135	1 x 10^{-5}	1 x 10^{-5}	1 x 10^{-5}	1 x 10^{-5}	1 x 10^{-1}	1 x 10^{-1}	1 x 10^{-5}	1 x 10^{-5}	1 x 10^{-5}
1930	Co-60	2 x 10^{-7}	2 x 10^{-7}	2 x 10^{-7}	2 x 10^{-7}	2 x 10^{-3}	2 x 10^{-3}	2 x 10^{-7}	2 x 10^{-7}	2 x 10^{-7}
0.113	Sr-92	7 x 10^{-6}	7 x 10^{-6}	7 x 10^{-6}	7 x 10^{-6}	7 x 10^{-2}	7 x 10^{-2}	7 x 10^{-6}	7 x 10^{-6}	7 x 10^{-6}
1.68	La-140	2 x 10^{-7}	2 x 10^{-7}	2 x 10^{-7}	2 x 10^{-7}	2 x 10^{-3}	2 x 10^{-3}	2 x 10^{-7}	2 x 10^{-7}	2 x 10^{-7}
244	Zn-65	1 x 10^{-8}	1 x 10^{-8}	1 x 10^{-8}	1 x 10^{-8}	1 x 10^{-4}	1 x 10^{-4}	1 x 10^{-8}	1 x 10^{-8}	1 x 10^{-8}
0.0511	Nb-97	1 x 10^{-8}	1 x 10^{-8}	1 x 10^{-8}	1 x 10^{-8}	1 x 10^{-4}	1 x 10^{-4}	1 x 10^{-8}	1 x 10^{-8}	1 x 10^{-8}
0.379	Xe-135	2 x 10^{-5}	2 x 10^{-5}	2 x 10^{-5}	2 x 10^{-5}	2 x 10^{-1}	2 x 10^{-1}	2 x 10^{-15}	2 x 10^{-15}	2 x 10^{-15}
13.1	Cs-136	1 x 10^{-7}	1 x 10^{-7}	1 x 10^{-7}	1 x 10^{-7}	1 x 10^{-3}	1 x 10^{-3}	1 x 10^{-7}	1 x 10^{-7}	1 x 10^{-7}
0.625	Na-24	4 x 10^{-7}	4 x 10^{-7}	4 x 10^{-7}	4 x 10^{-7}	4 x 10^{-3}	4 x 10^{-3}	4 x 10^{-7}	4 x 10^{-7}	4 x 10^{-7}

Table 11.2-1 (Continued)

EXPECTED INDIVIDUAL NUCLIDE CONCENTRATION μCi/cc
OF LIQUID RADIOACTIVE WASTE IN RADWASTE TANKAGE BASED ON OPERATIONAL DATA

Half Life (Days)	Nuclide	Waste Collector Tank	Floor Drain Collector Tank	Waste Sample Tank	Floor Drain Sample Tank	Cleanup Phase Separators	Condensate Phase Separators	Chemical Waste Tank	River Discharge Tank	Laundry Drain Sample Tank
32.5	Ce-141	8×10^{-9}	8×10^{-9}	8×10^{-9}	8×10^{-9}	8×10^{-5}	8×10^{-5}	8×10^{-9}	8×10^{-9}	8×10^{-9}
251	Ag-110m	8×10^{-8}	8×10^{-8}	8×10^{-8}	8×10^{-8}	8×10^{-4}	8×10^{-4}	8×10^{-8}	8×10^{-8}	8×10^{-8}
11,000	Cs-137	4×10^{-8}	4×10^{-8}	4×10^{-8}	4×10^{-8}	4×10^{-4}	4×10^{-4}	4×10^{-8}	4×10^{-8}	4×10^{-8}
367	Ru-106	2×10^{-8}	2×10^{-8}	2×10^{-8}	2×10^{-8}	2×10^{-4}	2×10^{-4}	2×10^{-8}	2×10^{-8}	2×10^{-8}
753	Cs-134	2×10^{-8}	2×10^{-8}	2×10^{-8}	2×10^{-8}	2×10^{-4}	2×10^{-4}	2×10^{-8}	2×10^{-8}	2×10^{-8}
5.24	Xe-133	2×10^{-7}	2×10^{-7}	2×10^{-7}	2×10^{-7}	2×10^{-3}	2×10^{-3}	2×10^{-7}	2×10^{-7}	2×10^{-7}
284	Ce-144	1×10^{-8}	1×10^{-8}	1×10^{-8}	1×10^{-8}	1×10^{-4}	1×10^{-4}	1×10^{-8}	1×10^{-8}	1×10^{-8}
60.2	Sb-124	4×10^{-8}	4×10^{-8}	4×10^{-8}	4×10^{-8}	4×10^{-4}	4×10^{-4}	4×10^{-8}	4×10^{-8}	4×10^{-8}
0.224	Cs-138	4×10^{-7}	4×10^{-7}	4×10^{-7}	4×10^{-7}	4×10^{-3}	4×10^{-3}	4×10^{-7}	4×10^{-7}	4×10^{-7}
TOTAL		8×10^{-5}	8×10^{-5}	8×10^{-5}	8×10^{-5}	8×10^{-1}	8×10^{-1}	8×10^{-5}	8×10^{-5}	8×10^{-5}

Table 11.2-1 (Continued)

EXPECTED INDIVIDUAL NUCLIDE CONCENTRATION μCi/cc
OF LIQUID RADIOACTIVE WASTE IN RADWASTE TANKAGE BASED ON OPERATIONAL DATA

	Condensate Backwash Receiving Tank	Floor Drain Surge Tank	Laundry Drain Sample Tank	Max Recycle Spent Resin Tank	Radwaste Mixing Tank	Chemical Waste Sample Tank	Waste Sludge Tank	Waste Demin Spent Resin Tank	Total of All Tanks
Number of tanks	2	1	1	1	1	1	1	1	---
Expected μCi contained in all full tanks*	7×10^7	6×10^4	3×10^2	3×10^3	8×10^2	2×10^4	5×10^3	4×10^2	3×10^8
Expected average μCi contained in all full tanks*	1×10^7	9×10^3	5×10^1	5×10^2	1×10^2	2×10^3	7×10^2	6×10^1	4×10^7
Maximum tank volume per tank (gallon)	12, 000	200,000	1,000	10,000	2,500	5,000	15,000	1,200	528,700

Table 11.2-1 (Continued)

EXPECTED INDIVIDUAL NUCLIDE CONCENTRATION μCi/cc
OF LIQUID RADIOACTIVE WASTE IN RADWASTE TANKAGE BASED ON OPERATIONAL DATA

Half Life (Days)	Nuclide	Condensate Backwash Receiving Tank	Floor Drain Surge Tank	Laundry Drain Sample Tank	Max Recycle Spent Resin Tank	Radwaste Mixing Tank	Chemical Waste Sample Tank	Waste Sludge Tank	Waste Demin Spent Resin Tank	Total of All Tanks
27.7	Cr-51	5×10^{-3}	5×10^{-7}	5×10^{-7}	5×10^{-7}	5×10^{-7}	5×10^{-7}	5×10^{-7}	5×10^{-7}	5×10^{-7}
2.75	Tc-99m	2×10^{-2}	2×10^{-6}	2×10^{-6}	2×10^{-6}	2×10^{-6}	2×10^{-6}	2×10^{-6}	2×10^{-6}	2×10^{-6}
2.36	Np-239	9×10^{-3}	9×10^{-7}	9×10^{-7}	9×10^{-7}	9×10^{-7}	9×10^{-7}	9×10^{-7}	9×10^{-7}	9×10^{-7}
1.38	Ce-143	1×10^{-4}	1×10^{-8}	1×10^{-8}	1×10^{-8}	1×10^{-8}	1×10^{-8}	1×10^{-8}	1×10^{-8}	1×10^{-8}
8.04	I-131	4×10^{-3}	4×10^{-7}	4×10^{-7}	4×10^{-7}	4×10^{-7}	4×10^{-7}	4×10^{-7}	4×10^{-7}	4×10^{-7}
0.185	Ru-105	3×10^{-3}	3×10^{-7}	3×10^{-7}	3×10^{-7}	3×10^{-7}	3×10^{-7}	3×10^{-7}	3×10^{-7}	3×10^{-7}
39.4	Ru-103	1×10^{-4}	1×10^{-8}	1×10^{-8}	1×10^{-8}	1×10^{-8}	1×10^{-8}	1×10^{-8}	1×10^{-8}	1×10^{-8}
0.867	I-133	6×10^{-2}	6×10^{-6}	6×10^{-6}	6×10^{-6}	6×10^{-6}	6×10^{-6}	6×10^{-6}	6×10^{-6}	6×10^{-6}
12.8	Ba-140	5×10^{-3}	5×10^{-7}	5×10^{-7}	5×10^{-7}	5×10^{-7}	5×10^{-7}	5×10^{-7}	5×10^{-7}	5×10^{-7}
0.0954	I-132	3×10^{-5}	3×10^{-9}	3×10^{-9}	3×10^{-9}	3×10^{-9}	3×10^{-9}	3×10^{-9}	3×10^{-9}	3×10^{-9}
0.996	W-187	7×10^{-5}	7×10^{-9}	7×10^{-9}	7×10^{-9}	7×10^{-9}	7×10^{-9}	7×10^{-9}	7×10^{-9}	7×10^{-9}
64	Zr-95	2×10^{-4}	2×10^{-8}	2×10^{-8}	2×10^{-8}	2×10^{-8}	2×10^{-8}	2×10^{-8}	2×10^{-8}	2×10^{-8}
2.75	Mo-99	1×10^{-2}	1×10^{-6}	1×10^{-6}	1×10^{-6}	1×10^{-6}	1×10^{-6}	1×10^{-6}	1×10^{-6}	1×10^{-6}

Table 11.2-1 (Continued)

EXPECTED INDIVIDUAL NUCLIDE CONCENTRATION μCi/cc
OF LIQUID RADIOACTIVE WASTE IN RADWASTE TANKAGE BASED ON OPERATIONAL DATA

Half Life (Days)	Nuclide	Condensate Backwash Receiving Tank	Floor Drain Surge Tank	Laundry Drain Sample Tank	Max Recycle Spent Resin Tank	Radwaste Mixing Tank	Chemical Waste Sample Tank	Waste Sludge Tank	Waste Demin Spent Resin Tank	Total of All Tanks
0.704	Zr-97	2×10^{-3}	2×10^{-7}	2×10^{-7}	2×10^{-7}	2×10^{-7}	2×10^{-7}	2×10^{-7}	2×10^{-7}	2×10^{-7}
0.396	Sr-91	6×10^{-2}	6×10^{-6}	6×10^{-6}	6×10^{-6}	6×10^{-6}	6×10^{-6}	6×10^{-6}	6×10^{-6}	6×10^{-6}
3.50	Nb-95	1×10^{-4}	1×10^{-8}	1×10^{-8}	1×10^{-8}	1×10^{-8}	1×10^{-8}	1×10^{-8}	1×10^{-8}	1×10^{-8}
70.8	Co-58	2×10^{-4}	2×10^{-8}	2×10^{-8}	2×10^{-8}	2×10^{-8}	2×10^{-8}	2×10^{-8}	2×10^{-8}	2×10^{-8}
312	Mn-54	3×10^{-4}	3×10^{-8}	3×10^{-8}	3×10^{-8}	3×10^{-8}	3×10^{-8}	3×10^{-8}	3×10^{-8}	3×10^{-8}
0.148	Y-92	2×10^{-1}	2×10^{-5}	2×10^{-5}	2×10^{-5}	2×10^{-5}	2×10^{-5}	2×10^{-5}	2×10^{-5}	2×10^{-5}
44.6	Fe-59	2×10^{-5}	2×10^{-9}	2×10^{-9}	2×10^{-9}	2×10^{-9}	2×10^{-9}	2×10^{-9}	2×10^{-9}	2×10^{-9}
0.275	I-135	1×10^{-1}	1×10^{-5}	1×10^{-5}	1×10^{-5}	1×10^{-5}	1×10^{-5}	1×10^{-5}	1×10^{-5}	1×10^{-5}
1930	Co-60	2×10^{-3}	2×10^{-7}	2×10^{-7}	2×10^{-7}	2×10^{-7}	2×10^{-7}	2×10^{-7}	2×10^{-7}	2×10^{-7}
0.113	Sr-92	7×10^{-2}	7×10^{-6}	7×10^{-6}	7×10^{-6}	7×10^{-6}	7×10^{-6}	7×10^{-6}	7×10^{-6}	7×10^{-6}
1.68	La-140	2×10^{-3}	2×10^{-7}	2×10^{-7}	2×10^{-7}	2×10^{-7}	2×10^{-7}	2×10^{-7}	2×10^{-7}	2×10^{-7}
244	Zn-65	1×10^{-4}	1×10^{-8}	1×10^{-8}	1×10^{-8}	1×10^{-8}	1×10^{-8}	1×10^{-8}	1×10^{-8}	1×10^{-8}
.0511	Nb-97	1×10^{-4}	1×10^{-8}	1×10^{-8}	1×10^{-8}	1×10^{-8}	1×10^{-8}	1×10^{-8}	1×10^{-8}	1×10^{-8}

Table 11.2-1 (Continued)

EXPECTED INDIVIDUAL NUCLIDE CONCENTRATION μCi/cc
OF LIQUID RADIOACTIVE WASTE IN RADWASTE TANKAGE BASED ON OPERATIONAL DATA

Half Life (Days)	Nuclide	Condensate Backwash Receiving Tank	Floor Drain Surge Tank	Laundry Drain Sample Tank	Max Recycle Spent Resin Tank	Radwaste Mixing Tank	Chemical Waste Sample Tank	Waste Sludge Tank	Waste Demin Spent Resin Tank	Total of All Tanks
0.379	Xe-135	2×10^{-1}	2×10^{-5}	2×10^{-5}	2×10^{-5}	2×10^{-5}	2×10^{-5}	2×10^{-5}	2×10^{-5}	2×10^{-5}
13.1	Cs-136	1×10^{-3}	1×10^{-7}	1×10^{-7}	1×10^{-7}	1×10^{-7}	1×10^{-7}	1×10^{-7}	1×10^{-7}	1×10^{-7}
0.625	Na-24	4×10^{-3}	4×10^{-7}	4×10^{-7}	4×10^{-7}	4×10^{-7}	4×10^{-7}	4×10^{-7}	4×10^{-7}	4×10^{-7}
32.5	Ce-141	8×10^{-5}	8×10^{-9}	8×10^{-9}	8×10^{-9}	8×10^{-9}	8×10^{-9}	8×10^{-9}	8×10^{-9}	8×10^{-9}
251	Ag-110m	8×10^{-4}	8×10^{-8}	8×10^{-8}	8×10^{-8}	8×10^{-8}	8×10^{-8}	8×10^{-8}	8×10^{-8}	8×10^{-8}
11,000	Cs-137	4×10^{-4}	4×10^{-8}	4×10^{-8}	4×10^{-8}	4×10^{-8}	4×10^{-8}	4×10^{-8}	4×10^{-8}	4×10^{-8}
367	Ru-106	2×10^{-4}	2×10^{-8}	2×10^{-8}	2×10^{-8}	2×10^{-8}	2×10^{-8}	2×10^{-8}	2×10^{-8}	2×10^{-8}
753	Cs-134	2×10^{-4}	2×10^{-8}	2×10^{-8}	2×10^{-8}	2×10^{-8}	2×10^{-8}	2×10^{-8}	2×10^{-8}	2×10^{-8}
5.24	Xe-133	2×10^{-3}	2×10^{-7}	2×10^{-7}	2×10^{-7}	2×10^{-7}	2×10^{-7}	2×10^{-7}	2×10^{-7}	2×10^{-7}
284	Ce-144	1×10^{-4}	1×10^{-8}	1×10^{-8}	1×10^{-8}	1×10^{-8}	1×10^{-8}	1×10^{-8}	1×10^{-8}	1×10^{-8}
60.2	Sb-124	4×10^{-4}	4×10^{-8}	4×10^{-8}	4×10^{-8}	4×10^{-8}	4×10^{-8}	4×10^{-8}	4×10^{-8}	4×10^{-8}
0.224	Cs-138	4×10^{-3}	4×10^{-7}	4×10^{-7}	4×10^{-7}	4×10^{-7}	4×10^{-7}	4×10^{-7}	4×10^{-7}	4×10^{-7}
	TOTAL	8×10^{-1}	8×10^{-5}	8×10^{-5}	8×10^{-5}	8×10^{-5}	8×10^{-5}	8×10^{-5}	8×10^{-5}	8×10^{-5}

Table 11.2-2

RADIONUCLIDE ACTIVITY OF SOLID RADIOACTIVE
MATERIAL IN LIQUID RADWASTE TANKS BASED ON OPERATIONAL DATA
(μCi in Solid Form/Tank)

	Cleanup Phase Separators	Condensate Phase Separators	Condensate Backwash Rec. Tank	Waste Sludge Tank	Max-Recycle Spent Resin Tank	Waste Demin Spent Resin Tank
Number of tanks	4	4	2	1	1	1
Expected μCi in solid form per tank	5×10^{7}	6×10^{8}	6×10^{6}	6×10^{6}	1×10^{3}	2×10^{8}
Expected average μCi in solid form per tank	5×10^{6}	6×10^{7}	2×10^{5}	3×10^{5}	6×10^{7}	1×10^{6}
Nuclides						
Cr-51	2×10^{6}	2×10^{7}	2×10^{5}	2×10^{5}	5×10^{7}	7×10^{6}
Mn-54	2×10^{6}	2×10^{7}	2×10^{5}	2×10^{5}	3×10^{7}	5×10^{6}
Co-58	6×10^{5}	7×10^{6}	7×10^{4}	7×10^{4}	1×10^{7}	2×10^{6}

Table 11.2-2

RADIONUCLIDE ACTIVITY OF SOLID RADIOACTIVE
MATERIAL IN LIQUID RADWASTE TANKS BASED ON OPERATIONAL DATA
(μCi in Solid Form/Tank)

Error! Bookmark not defined.	Cleanup Phase Separators	Condensate Phase Separators	Condensate Backwash Rec. Tank	Waste Sludge Tank	Max-Recycle Spent Resin Tank	Waste Demin Spent Resin Tank
Co-60	3×10^7	4×10^8	4×10^6	4×10^6	7×10^8	1×10^8
Zn-65	2×10^6	2×10^7	2×10^5	2×10^5	4×10^7	5×10^6
Cs-134	6×10^6	7×10^7	7×10^5	7×10^5	1×10^8	2×10^7
Cs-137	1×10^7	1×10^8	1×10^6	1×10^6	3×10^8	4×10^7
Ba/La-140	3×10^5	3×10^6	3×10^4	3×10^4	6×10^6	9×10^5
Sr-90	2×10^6	3×10^7	3×10^5	3×10^5	5×10^7	8×10^6
Pu-239, 240	9×10^3	1×10^5	1×10^3	1×10^3	2×10^5	3×10^4
Pu-238	2×10^4	2×10^5	2×10^3	2×10^3	4×10^5	6×10^4
Am-241	3×10^2	4×10^3	4×10^1	4×10^1	7×10^3	1×10^3
Cm-242, 243	3×10^3	3×10^1	3×10^2	4×10^2	5×10^4	8×10^3
Cm-244	2×10^3	2×10^4	2×10^2	2×10^2	3×10^4	5×10^3

Table 11.2-3

PREDICTED ACTIVITY PRESENT IN WASTES TO BE
DISCHARGED TO THE DIFFUSER PIPE OR DISCHARGE FLUME WEIR
CONCURRENT WITH DESIGN BASIS FUEL LEAKS

Waste Type	Daily Average Curies	Daily Average Concentration
Floor drains	0.08	
Laboratory drains		~2 x 10^{-8} μCi/cc
Decontamination solutions	[0.04]*	
Laundry wastes	7 x 10^{-5}	

* Only when chemical decontaminations are performed.

Table 11.2-4

EXPECTED ACTIVITY CONCENTRATIONS
IN DISCHARGE BAY AND MISSISSIPPI RIVER

Isotope	Half Life (Days)	Release Rate (Ci/yr)*	Environmental Inventory Ci After 1 Yr	MPC Per 10 CFR 20 (μCi/cc)	Concentration After Mixing With Two Unit Circulating Water (μCi/cc)**	Maximum Expected Routine Concentration In Mississippi River (μCi/cc)‡	Expected Annual Average Concentration In Mississippi River (μCi/cc)♯
Sr-91	.396	3.41×10^{-3}	0	5×10^{-5}	2×10^{-12}	3×10^{-13}	8×10^{-14}
Cs-134	753	$2.49 \times 10^{+0}$	8.49×10^{-3}	9×10^{-6}	1×10^{-9}	2×10^{-10}	6×10^{-11}
Cs-137	11000	$6.81 \times 10^{+0}$	$6.66 \times 10^{+0}$	2×10^{-5}	3×10^{-9}	6×10^{-10}	2×10^{-10}
I-131	8.04	5.56×10^{-2}	1.17×10^{-15}	3×10^{-7}	3×10^{-11}	5×10^{-12}	1×10^{-12}
Co-58	70.8	3.36×10^{-2}	1.02×10^{-3}	9×10^{-5}	2×10^{-11}	3×10^{-12}	9×10^{-13}
Co-60	1930	$2.80 \times 10^{+0}$	$2.46 \times 10^{+0}$	3×10^{-5}	1×10^{-9}	3×10^{-10}	7×10^{-11}
Fe-59	44.6	LLD	---	5×10^{-5}	---	---	---
Zn-65	244	2.12×10^{-2}	1.11×10^{-2}	1×10^{-4}	2×10^{-11}	3×10^{-12}	7×10^{-13}
Mn-54	312	1.04×10^{-1}	4.62×10^{-2}	1×10^{-4}	5×10^{-11}	1×10^{-11}	2×10^{-12}

* Release rate for 1980

** 1,000,000 gal/min

‡ Based on a very low flow of 12,000 ft^3^/s

♯ Based on an average flow of 47,000 ft^3^/s

Table 11.2-4 (Continued)

EXPECTED ACTIVITY CONCENTRATIONS
IN DISCHARGE BAY AND MISSISSIPPI RIVER

Isotope	Half Life (Days)	Release Rate (Ci/yr)*	Environmental Inventory Ci After 1 Yr	MPC Per 10 CFR 20 (μCi/cc)	Concentration After Mixing With Two Unit Circulating Water (μCi/cc)**	Maximum Expected Routine Concentration In Mississippi River (μCi/cc)‡	Expected Annual Average Concentration In Mississippi River (μCi/cc)‡‡
Cr-51	27.7	LLD	---	2×10^{-3}	---	---	---
Zr-95	64.0	9.50×10^{-4}	1.82×10^{-5}	6×10^{-5}	5×10^{-13}	5×10^{-14}	2×10^{-14}
Nb-95	35.2	LLD	---	1×10^{-4}	---	---	---
Mo-99	275	LLD	---	4×10^{-5}	---	---	---
Y-92	.148	9.30×10^{-3}	0	6×10^{-5}	5×10^{-12}	9×10^{-13}	2×10^{-13}
Ag-110m	251	7.47×10^{-4}	2.72×10^{-4}	3×10^{-5}	4×10^{-13}	7×10^{-14}	2×10^{-14}
Tc-99m	2.75	3.17×10^{-4}	3.29×10^{-44}	3×10^{-4}	2×10^{-13}	3×10^{-14}	8×10^{-15}
Ba-140	12.8	2.76×10^{-1}	7.10×10^{-10}	2×10^{-5}	1×10^{-10}	3×10^{-11}	7×10^{-12}
I-135	.275	3.66×10^{-3}	0	4×10^{-6}	2×10^{-12}	3×10^{-13}	9×10^{-14}
Cs-136	13.1	LLD	---	6×10^{-5}	---	---	---
I-133	.867	1.99×10^{-2}	0	1×10^{-6}	1×10^{-11}	2×10^{-12}	5×10^{-13}
La-140	12.8	1.93×10^{-1}	4.96×10^{-10}	2×10^{-5}	1×10^{-10}	2×10^{-11}	5×10^{-12}

Table 11.2-4 (Continued)

EXPECTED ACTIVITY CONCENTRATIONS
IN DISCHARGE BAY AND MISSISSIPPI RIVER

Isotope	Half Life (Days)	Release Rate (Ci/yr)*	Environmental Inventory Ci After 1 Yr	MPC Per 10 CFR 20 (μCi/cc)	Concentration After Mixing With Two Unit Circulating Water (μCi/cc)**	Maximum Expected Routine Concentration In Mississippi River (μCi/cc)‡	Expected Annual Average Concentration In Mississippi River (μCi/cc)‡‡
Ce-141	32.5	2.09×10^{-3}	8.653×10^{-7}	9×10^{-5}	1×10^{-12}	2×10^{-13}	5×10^{-14}
Np-239	2.36	2.99×10^{-2}	7.67×10^{-49}	13×10^{-4}	23×10^{-12}	33×10^{-13}	73×10^{-14}
Xe-133	5.24	8.38×10^{-2}	8.70×10^{-23}	1.2×10^{-4}	4×10^{-11}	8×10^{-12}	2×10^{-12}
Xe-135	.379	7.15×10^{-2}	0	4×10^{-5}	5×10^{-11}	7×10^{-12}	2×10^{-12}
H-3	4504	$1.02 \times 10^{+1}$	$9.74 \times 10^{+0}$	3×10^{-3}	5×10^{-9}	1×10^{-9}	2×10^{-10}
Sr-89	50.5	1.21×10^{-2}	8.04×10^{-5}	3×10^{-5}	6×10^{-12}	1×10^{-12}	3×10^{-13}
Sr-90	10593	5.79×10^{-2}	5.65×10^{-2}	4×10^{-5}	3×10^{-11}	5×10^{-12}	1×10^{-12}
	TOTAL	23.4	19.0	---	1×10^{-8}	2×10^{-9}	5×10^{-10}

Table 11.2-5

ESTIMATED CURIE CONTENT FOR RADIOACTIVE LIQUID WASTE TANKS
IN THE TANK FARM SYSTEM

	Waste Sample*	Floor Drain Sample	River Discharge	Laundry Sample	Chem. Waste Sample	River µCi/cc	10 CFR 20 MPC
Number of tanks	2	2	1	1	1	---	---
Estimated µCi (all tanks)	1×10^{4}	1×10^{4}	2×10^{4}	3×10^{2}	2×10^{4}	3×10^{5}	---
Estimated average µCi (all tanks)	2×10^{3}	2×10^{3}	3×10^{3}	5×10^{1}	2×10^{3}	1×10^{3}	---
Maximum volume (each)	22,000	21,000	65,000	1,000	5,000	5×10^{12} cc/hr**	---
Cr-51	5×10^{-7}	5×10^{-7}	5×10^{-7}	5×10^{-7}	5×10^{-7}	5.5×10^{-11}	2×10^{-3}
Tc-99m	2×10^{-6}	2×10^{-6}	2×10^{-6}	2×10^{-6}	2×10^{-6}	2.2×10^{-10}	3×10^{-4}
I-131	4×10^{-7}	4×10^{-7}	4×10^{-7}	4×10^{-7}	4×10^{-7}	4.4×10^{-11}	3×10^{-7}
Np-239	9×10^{-7}	9×10^{-7}	9×10^{-7}	9×10^{-7}	9×10^{-7}	9.9×10^{-11}	1×10^{-4}
Ce-143	1×10^{-8}	1×10^{-8}	1×10^{-8}	1×10^{-8}	1×10^{-8}	1.1×10^{-12}	4×10^{-5}
Ru-105	3×10^{-7}	3×10^{-7}	3×10^{-7}	3×10^{-7}	3×10^{-7}	3.3×10^{-11}	1×10^{-4}
Ru-103	1×10^{-8}	1×10^{-8}	1×10^{-8}	1×10^{-8}	1×10^{-8}	1.1×10^{-12}	8×10^{-5}

Table 11.2-5 (Continued)

ESTIMATED CURIE CONTENT FOR RADIOACTIVE LIQUID WASTE TANKS
IN THE TANK FARM SYSTEM

	Waste Sample*	Floor Drain Sample	River Discharge	Laundry Sample	Chem. Waste Sample	River µCi/cc	10 CFR 20 MPC
I-133	6×10^{-6}	6×10^{-6}	6×10^{-6}	6×10^{-6}	6×10^{-6}	6.6×10^{-10}	1×10^{-6}
Ba-140	5×10^{-7}	5×10^{-7}	5×10^{-7}	5×10^{-7}	5×10^{-7}	5.5×10^{-11}	2×10^{-5}
I-132	3×10^{-9}	3×10^{-9}	3×10^{-9}	3×10^{-9}	3×10^{-9}	3.3×10^{-13}	8×10^{-6}
W-187	7×10^{-9}	7×10^{-9}	7×10^{-9}	7×10^{-9}	7×10^{-9}	7.7×10^{-12}	5×10^{-5}
Zr-95	2×10^{-8}	2×10^{-8}	2×10^{-8}	2×10^{-8}	2×10^{-8}	2.2×10^{-12}	6×10^{-5}
Mo-99	1×10^{-6}	1×10^{-6}	1×10^{-6}	1×10^{-6}	1×10^{-6}	1.1×10^{-10}	4×10^{-5}
Zr-97	2×10^{-7}	2×10^{-7}	2×10^{-7}	2×10^{-7}	2×10^{-7}	2.2×10^{-11}	2×10^{-5}
Sr-91	6×10^{-6}	6×10^{-6}	6×10^{-6}	6×10^{-6}	6×10^{-6}	6.6×10^{-10}	5×10^{-5}
Nb-95	1×10^{-8}	1×10^{-8}	1×10^{-8}	1×10^{-8}	1×10^{-8}	1.1×10^{-12}	1×10^{-4}
Co-58	2×10^{-8}	2×10^{-8}	2×10^{-8}	2×10^{-8}	2×10^{-8}	2.2×10^{-12}	9×10^{-5}
Mn-54	3×10^{-8}	3×10^{-8}	3×10^{-8}	3×10^{-8}	3×10^{-8}	3.3×10^{-12}	1×10^{-4}
Y-92	2×10^{-5}	2×10^{-5}	2×10^{-5}	2×10^{-5}	2×10^{-5}	2.2×10^{-09}	6×10^{-5}
Fe-59	2×10^{-9}	2×10^{-9}	2×10^{-9}	2×10^{-9}	2×10^{-9}	2.2×10^{-12}	5×10^{-5}
I-135	1×10^{-5}	1×10^{-5}	1×10^{-5}	1×10^{-5}	1×10^{-5}	1.1×10^{-09}	4×10^{-6}

Table 11.2-5 (Continued)

ESTIMATED CURIE CONTENT FOR RADIOACTIVE LIQUID WASTE TANKS
IN THE TANK FARM SYSTEM

Co-60	2×10^{-7}	2×10^{-7}	2×10^{-7}	2×10^{-7}	2×10^{-7}	2.2×10^{-11}	3×10^{-5}
Sr-92	7×10^{-6}	7×10^{-6}	7×10^{-6}	7×10^{-6}	7×10^{-6}	7.7×10^{-10}	2×10^{-4}
La-140	2×10^{-7}	2×10^{-7}	2×10^{-7}	2×10^{-7}	2×10^{-7}	2.2×10^{-11}	2×10^{-5}
Zn-65	1×10^{-8}	1×10^{-8}	1×10^{-8}	1×10^{-8}	1×10^{-8}	1.1×10^{-12}	1×10^{-4}
Nb-97	1×10^{-8}	1×10^{-8}	1×10^{-8}	1×10^{-8}	1×10^{-8}	1.1×10^{-12}	4×10^{-4}
Xe-135	2×10^{-5}	2×10^{-5}	2×10^{-5}	2×10^{-5}	2×10^{-5}	1.1×10^{-9}	4×10^{-5}
Cs-136	1×10^{-7}	1×10^{-7}	1×10^{-7}	1×10^{-7}	1×10^{-7}	1.1×10^{-11}	6×10^{-5}
Na-24	4×10^{-7}	4×10^{-7}	4×10^{-7}	4×10^{-7}	4×10^{-7}	4.4×10^{-11}	3×10^{-5}
Ce-141	8×10^{-9}	8×10^{-9}	8×10^{-9}	8×10^{-9}	8×10^{-9}	8.8×10^{-13}	9×10^{-5}
Ag-110m	8×10^{-8}	8×10^{-8}	8×10^{-8}	8×10^{-8}	8×10^{-8}	8.8×10^{-12}	3×10^{-5}
Cs-137	4×10^{-8}	4×10^{-8}	4×10^{-8}	4×10^{-8}	4×10^{-8}	4.4×10^{-12}	2×10^{-5}
Ru-106	2×10^{-8}	2×10^{-8}	2×10^{-8}	2×10^{-8}	2×10^{-8}	2.2×10^{-12}	1×10^{-5}
Cs-134	2×10^{-8}	2×10^{-8}	2×10^{-8}	2×10^{-8}	2×10^{-8}	2.2×10^{-12}	9×10^{-6}
Xe-133	2×10^{-7}	2×10^{-7}	2×10^{-7}	2×10^{-7}	2×10^{-7}	2.2 xx 10^{-11}	1.2×10^{-4}

Table 11.2-5 (Continued)

ESTIMATED CURIE CONTENT FOR RADIOACTIVE LIQUID WASTE TANKS
IN THE TANK FARM SYSTEM

	Waste Sample*	Floor Drain Sample	River Discharge	Laundry Sample	Chem. Waste Sample	River μCi/cc	10 CFR 20 MPC
Ce-144	1×10^{-8}	1×10^{-8}	1×10^{-8}	1×10^{-8}	1×10^{-8}	1.1×10^{-12}	1×10^{-5}
Sb-124	4×10^{-8}	4×10^{-8}	4×10^{-8}	4×10^{-8}	4×10^{-8}	4.4×10^{-12}	2×10^{-5}
Cs-138	4×10^{-7}	4×10^{-7}	4×10^{-7}	4×10^{-7}	4×10^{-7}	4.4×10^{-11}	5×10^{-6}
Totals	8×10^{-5}	8×10^{-5}	8×10^{-5}	8×10^{-5}	8×10^{-5}		

* Data is from Unit 1 coolant isotopics, averaged for 1980 divided by 10,000 (average dilution factor).

**

$$\frac{\mu\text{Ci/cc x gallons x } 3.785 \text{ x } 10^{3} \text{ cc/gal}}{6.0 \text{ x } 10^{7} \text{ gal/hr cond. flow x } 3.785 \text{ x } 10^{3} \text{ cc/gal} + 5 \text{ x } 10^{12} \text{ cc/hr river flow}}$$

Table 11.2-6

COMPARISON OF LIQUID RADIOACTIVE
WASTE DISCHARGES AND THEIR POTENTIAL EFFECTS ON ENVIRONMENT

Case	Dis. Flume Concen. (μCi/cc)	Dis. Period (hrs/day)	Dis. Period (days/yr)	Activity Dis. (Ci/yr)	Environs Dose Rate (mrem/yr): In Dis. Flume	Environs Dose Rate (mrem/yr): In River Min. Flow*	Environs Dose Rate (mrem/yr): In River Ave. Flow*	Fraction of Natural Background Exposure**: In Dis. Flume	Fraction of Natural Background Exposure**: In River at Ave. Flow*
I. 10 CFR 20 - Identified Mixture-Maximum Continuous Discharge Legal Maximum	1.6×10^{-6}	24	365	3200	500	89	23	3.6	0.16
II. 10 CFR 20 - Unidentified Mixture:									
A. Maximum per Tech Spec	10^{-7}***	24	365	200	30	5.4	1.4	0.018	0.0097
B. Expected Release - with design basis fuel leaks	1×10^{-8}***	As Required	365	24	2.9	---	---	---	---

* Minimum Flow = 7 day low flow = 11,900 ft^3/s
Average Flow = Average annual flow = 47,000 ft^3/s
Doses calculated assume these flows persisted all year

** Natural background averages = 140 mrem/year

*** Discharge flume concentration at 10^{-7} μCi/cc during discharge, but actual MPC of mixture being discharged is ~1.6×10^{-6} μCi/cc, and minimum MPC with no fission products is ~ 10^{-5} μCi/cc.

11.3 GASEOUS WASTE MANAGEMENT SYSTEMS

This section describes the capabilities of the Quad Cities Station to control, collect, process, handle, and dispose of the gaseous radioactive waste generated as a result of normal operation and anticipated operational occurrences. [11.3-1]

The Unit 1 and 2 systems addressed in this section are the off-gas system, the turbine gland seal system, and the mechanical vacuum pump system.

The off-gas system collects, contains, and processes the radioactive gases extracted from the steam condenser. The gases are exhausted by the steam jet ejectors (see Section 10.4) and flow through a preheater to a catalytic recombiner where most of the hydrogen and oxygen present recombine to form steam. The steam is condensed for return as condensate and the noncondensible gases flow to a holdup pipe. The gas flow continues through a cooler condenser, moisture separator, electric reheaters, prefilter, activated carbon adsorbers, high efficiency particulate air (HEPA) filters, and then to the 310-foot chimney for discharge to the environment. Two alternate off-gas system flow paths allow flow to bypass the catalytic recombiners, or the activated carbon adsorbers or both.

The turbine gland seal system provides steam to the shaft seals using steam to exclude air from the turbine cavities. The steam in the gas mixture is condensed and the air and radioactive gases are discharged to the 310-foot chimney.

The mechanical vacuum pump system establishes and maintains the main condenser vacuum when steam is not available. The vacuum pump effluent is discharged to the 310-foot chimney.

11.3.1 Design Objectives

11.3.1.1 Off-Gas System

The design objectives of the off-gas handling system are: [11.3-1a]

A. To provide effective control of process off-gases with capability for preventing releases over limits defined in 10 CFR 20;

B. To minimize radioactive particle release to the atmosphere;

C. To provide sufficient time for operator decision and action when continuous monitoring indicates development of off-standard conditions;

D. To minimize the release of normally occurring activation radiogases by suitable short-term decay; and

E. To minimize the hazard of explosion of hydrogen and oxygen gas in the off-gas system.

To achieve these objectives the air ejector off-gas system was designed using the following bases:

Design pressure	350 psig
Design flow rate (air ejector flow)	250 ft^3/min
Nominal size of particulate removed	0.3 µm and greater
Release point above ground	310 feet (chimney)
Piping design code	USAS B31.1.0 (original system) ASME B&PV Section III Subsection ND, Class 3 (Rechar System) (Maintained to ANSI B31.1)

The original off-gas system design was modified in 1972 to reduce the radioactive gaseous effluent discharged from the chimney. The new design objectives were given as 10 mrem/yr for noble gases and 1.0 x 10^{-5} times 10 CFR 20 limits for iodines. [11.3-2]

11.3.1.2 Turbine Gland Seal System

The design objective and description for the turbine gland seal system are given in Section 11.3.2.2.

11.3.1.3 Mechanical Vacuum Pump System

The design objectives and description for the mechanical vacuum pump system are given in Section 11.3.2.3.

11.3.1.4 Plant Features to Minimize the Amounts of Radioactive Effluents

The plant design includes several specific features or effects which minimize the amounts of radioactive materials released to the environment. These include: [11.3-3]

A. The use of high-integrity Zircaloy-clad fuel rods to contain fission products within the fuel.

B. The water to steam partition, retaining halogens in the coolant.

C. The provision of 4-hour holdup of the off-gas to allow decay of short-lived products before discharge. This reduces the potential radiation effects by a factor of approximately 10 as compared to a no holdup scheme and provides ample time to prevent release of fission product gases in excess of the instantaneous permissible release rate limits.

D. The provisions for monitoring the air ejector off-gas stream and initiating automatic isolation of the holdup piping when the radioactivity level is high.

E. High efficiency filters at the end of the off-gas piping to remove particulate radioisotopes formed by the decay of the noble gas radioisotopes in the holdup pipe.

F. Elevated release of gases from the 310-foot chimney, which is approximately 1-1/2 times the height of nearby structures, to reduce direct radiation dose rates on the ground and to maximize the atmospheric dispersion of the gas plume before it reaches ground level.

G. Continuous monitoring of the chimney effluent with appropriate alarms as a backup to the air ejector monitors.

11.3.2 System Description

There are eighteen sources of radioactive gaseous effluent all of which exhaust through the 310-foot chimney. The sources are listed below: [11.3-4]

1. Off-gas system for Unit 1;

2. Off-gas system for Unit 2;

3. Turbine gland seal system for Unit 1;

4. Turbine gland seal system for Unit 2;

5. Mechanical vacuum pump system for Unit 1;

6. Mechanical vacuum pump system for Unit 2;

7. Standby gas treatment system (SBGTS) for Unit 1;

8. SBGTS for Unit 2;

9. Off-gas system recombiner rooms' ventilation system for Unit 1;

10. Off-gas system recombiner rooms' ventilation system for Unit 2;

11. Turbine building ventilation system for Unit 1;

12. Turbine building ventilation system for Unit 2;

13. Off-gas building ventilation system;

14. Radwaste building ventilation system;

15. Maximum recycle building ventilation system; and

16. Solidification building ventilation system;

17. High radiation sampling system (HRSS) building ventilation system for Unit 1; and

18. HRSS building ventilation system for Unit 2.

The off-gas system is discussed in Section 11.3.2.1. The ventilation systems for the off-gas recombiner rooms for Units 1 and 2, the turbine building for Units 1 and 2, the off-gas building, the radwaste building, the maximum recycle building, and the solidification building are discussed in detail in Section 9.4. The potentially radioactive ventilation air from the systems is discharged to the environment through the 310-foot chimney. The SBGTS is discussed in detail in Section 6.5. The SBGTS discharges treated radioactive gases to the environment through the 310-foot chimney. The gland seal system for Units 1 and 2 and the mechanical vacuum pump system for Units 1 and 2 are discussed in Sections 11.3.2.2 and 11.3.2.3 respectively. The HRSS building ventilation system for Units 1 and 2 is discussed in Section 9.3.2.1.4.

11.3.2.1 Off-Gas System

11.3.2.1.1 Process Description

The off-gas system is shown in P&IDs M-42 and M-84. The main air ejectors, consisting of primary and secondary jets, remove the fission gases, activation gases, and radiolytic hydrogen and oxygen from the main condenser. The gaseous mixture is then routed through the preheater and into the catalytic recombiner. Preheating the gaseous mixture is necessary to ensure optimum recombiner performance. [11.3-5]

In the recombiner, most of the radiolytic hydrogen and oxygen are catalytically recombined to form water (in the form of superheated steam). This steam, along with the steam used for dilution, trace quantities of unreacted hydrogen and oxygen, air, and radioactive gases, exits the recombiner to a condenser where the steam is condensed to liquid and returned to the reactor condensate system. The noncondensible effluent is then routed to the holdup piping where the shorter-lived radioactive isotopes (principally N-13, N-16, 0-19, and certain isotopes of Xenon and Krypton) decay to either nonradioactive isotopes or radioactive particulate daughter products. [11.3-6]

Upon leaving the holdup pipe, the effluent is cooled further in the cooler condenser using a chilled glycol system for removal of water vapor, passed through a moisture separator for further drying, and then heated in a reheater for humidity control. Before entering the charcoal adsorbers, the effluent is filtered to remove the previously noted particulate daughter products. Following the charcoal adsorbers, the effluent is filtered again to remove particulate daughter products and then discharged through the 310-foot chimney.

The off-gas system provides ample monitoring and control to ensure that the limits set forth in 10 CFR 20 are not exceeded. The off-gas holdup pipe, effluent calibration of the off-gas monitors, particulate filtering, and alarms are all protective measures taken to meet the standards set by 10 CFR 20. [11.3-7]

Shielding is provided for the off-gas system equipment to maintain safe radiation exposure levels for plant personnel.

11.3.2.1.2 Description of Major Components

11.3.2.1.2.1 Steam Jet Air Ejectors

The steam jet air ejector (SJAE) is comprised of a 2-stage unit. The first stage (primary jet) utilizes an intercondenser while the second stage (secondary jet) serves as the booster air ejector to provide the fluid driving force. In this train, the after condenser is bypassed. These stages remove the noncondensible gases from the condenser and provide flow to the off-gas systems. The steam jet driving flow is from the turbine throttle through a pressure regulating valve set at a minimum of 125 psig. [11.3-8] [11.3-9]

A source for oxygen injection is provided to ensure that proportionate amounts of hydrogen and oxygen are available for recombination for both off-gas trainsfor both offgas trains.

11.3.2.1.2.2 Preheaters

The preheater is a U-tube heat exchanger using steam to superheat the off-gas mixture of steam and gases to ensure the absence of water which is a recombiner catalyst poison. The preheaters are heated with steam rather than electricity to eliminate the presence of potential ignition sources and to limit the temperature of the gases in the event of cessation of gas flow.

11.3.2.1.2.3 Catalytic Recombiners

The gaseous hydrogen and oxygen is combined catalytically into superheated steam at a variable temperature. The inlet temperature to the recombiner is 350°F.

11.3.2.1.2.4 Off-Gas Condensers

The off-gas condenser is a U-tube heat exchanger which uses condensate water to cool and condense the steam in the off-gas piping.

11.3.2.1.2.5 Holdup Pipe

The radioactivity of the gaseous stream is reduced in the off-gas system holdup pipe. The holdup allows the shorter-lived Xenons and Kryptons to decay to particulate daughter products. The removal of hydrogen and oxygen as water increases the holdup time to approximately four hours as a result of the lowered flow rate.

11.3.2.1.2.6 Cooler Condensers

The cooler condenser, a multipass tube side heat exchanger, further cools the gas to remove as much moisture as possible. Cooling is supplied by a chilled ethylene glycol-water mixture.

11.3.2.1.2.7 Moisture Separators

A moisture separator removes entrained moisture from the gas stream exiting the heat exchanger. Moisture separators are located at the exit of the off-gas condenser and the exit of the cooler condenser. Removal of condensible water vapor and entrained moisture is essential because the activated carbon efficiency is a function of moisture content.

11.3.2.1.2.8 Reheaters

The electric reheater heats the off-gas stream to the optimal temperature for activated carbon adsorption. Heating of the off-gas stream to approximately 70°F assures that any residual water vapor does not interfere with the activated carbon adsorption process.

11.3.2.1.2.9 Prefilters

The prefilters consist of full-flow HEPA filters designed to remove 99.97% of the particulates in the off-gas greater than 0.3 μm in size. [11.3-10]

11.3.2.1.2.10 Charcoal Adsorbers

The charcoal adsorbers provide for radioactive decay of the major activation gases and fission gases in the main condenser off-gas. The adsorbers provide a retention time of 14.6 days for Xenon and 19.4 hours for Krypton holdup. There are 12 charcoal adsorption beds. Each bed is 4 feet in diameter with an overall height of 21 feet. The 12 charcoal beds contain approximately 74,000 pounds of charcoal. The vessel is designed for 350 psig. [11.3-11]

Although iodine input into the off-gas system is small by virtue of its retention in reactor water and condensate the charcoal will effectively remove it by adsorption and prevent its release.

The charcoal adsorbers are designed to limit the temperature of the charcoal to well below the ignition temperature, thus precluding overheating or fire and consequent escape of radioactive materials. The charcoal adsorbers are located in a shielded room, maintained at a constant temperature by an air conditioning system (see Section 9.4) designed to remove the decay heat generated in the adsorbers. The maximum centerline temperature of the charcoal is less than 10°F above room temperature when the flow is stopped. The decay heat of 50 Btu/hr is sufficiently small compared to the thermal mass of the charcoal vault. Even if vault cooling is lost, the temperature rise will not be sufficient to cause charcoal ignition. The charcoal is maintained at 77°F by the vault air conditioning system. Due to the thermal capacitance of the charcoal beds and the massive concrete vault walls, temperature changes caused by failure of the vault air conditioning system will be sufficiently slow that the resulting changes in the charcoal adsorption coefficient will not produce a rapid release of adsorbed radioactive nuclides. In order to maintain consistent system operation, a redundant vault air conditioning system is supplied to allow for maintenance and operational convenience. During a plant outage when the condenser is not maintained at vacuum, there is no gas flow through the charcoal and the holdup is very high, even if the charcoal reaches ambient temperatures. High radiation level in the charcoal bed vault will cause an alarm in the control room. [11.3-12]

11.3.2.1.2.11 Afterfilters

These filters provide the final filtration of the off-gas before its release to the 310-foot chimney. [11.3-13]

The filters, located just before the chimney, consist of two parallel sets of full-flow, HEPA filters. The spare set of filters provides backup and assures availability of filtration. These filters are designed to remove 99.97% of the particulates in the off-gas greater than 0.3 μm in size. Static grounding wires are installed on the filter to minimize the potential for an off-gas explosion at this point. A loop seal was installed on the drain line from the filters to eliminate a leakage point for the radioactive gaseous effluent. The maximum operating differential pressure across these filter units is 4 inches of water. The pressure switches alarm in the control room on high differential pressure across the filter unit.

11.3.2.1.3 Redundancy of Equipment

Redundancy of the air ejector, preheater, recombiner, off-gas condenser, water separator, cooler-condenser, moisture separator, particulate filters, and charcoal vault air conditioning units is provided for operating convenience and maintenance. Valving is provided for selecting either one or both recombiner trains. Each recombiner train consists of a third-stage air ejector, preheater, recombiner, off-gas condenser, and a water separator, except for the 1B train which has no third-stage air ejector. [11.3-14]

Either one or both cooler condenser trains (cooler condenser, moisture separator, reheater, and prefilter) may be selected for operation. The charcoal beds can be operated in one of three modes:

1. All 12 charcoal beds in series;
2. Three parallel strings of 4 charcoal beds; or
3. Bypassing of all 12 charcoal beds.

11.3.2.1.4 Alternate Off-Gas Discharge Pathway

An alternate pathway, P&IDs M-42 and M-84, exists for the radioactive gases from the steam jet air ejector to the main chimney for discharge to the environment which involves bypassing the hydrogen recombiners and the charcoal adsorbers. This train configuration establishes the original design pathway with only the holdup and discharge filters for capturing particulates in the gas stream. Using this alternate pathway, the radioactive gases entering the off-gas system are held up to allow decay of the short-lived isotopes before being discharged to the environment through the 310-foot chimney. The radioactive gases from the main condenser air ejectors are delayed a minimum of 30 minutes in shielded piping before entering the filter system. [11.3-15]

11.3.2.1.5 Instrumentation and Control

The activity of the effluent entering and leaving the modified off-gas system is continuously monitored. This system is also monitored by flow, humidity, and temperature instrumentation, plus hydrogen analyzers and oxygen analyzer (see Section 5.4), for operation and control. Table 11.3-1 lists process instruments that cause alarms and notes whether the parameters are indicated or recorded in the control room. [11.3-16]

The off-gas system operates at a pressure of approximately 5 psig or less so the differential pressure that could cause leakage is small. To preclude radioactive gas leaks, the system is welded wherever possible, and bellows seal valve stems or equivalents are used. The entire system is designed to maintain its integrity in the event of a hydrogen-oxygen explosion.

Operational control is maintained by the use of radiation monitors to keep the release rate within the limits established by the Technical Specifications. A radiation monitor at the beginning of the holdup pipe continuously monitors gaseous radioactivity release from the reactor and, therefore, continuously monitors the degree of fuel leakage and input to the charcoal adsorbers. This radiation monitor is used to isolate the off-gas system upon detection of high radioactivity to prevent unacceptably high levels of radioactive gases from entering the plant vent chimney and being discharged to the environment. A continuous gas sampling radiation monitor is also provided at the outlet of the charcoal adsorbers to continuously monitor the effluent of the off-gas system. This gas sampling radiation monitor is used to provide an alarm upon detection of high radiation in the off-gas system effluent. Provision is also made for sampling of the influent and effluent gases. [11.3-17]

To protect the recombiner, the dilution steam supply pressure is monitored and alarmed on low pressure. The recombiner temperatures are monitored and alarmed to indicate any deterioration of performance.

The holdup of the condenser air ejector radioactive off-gas provides sufficient time between detection and release to permit isolation of the holdup line. The holdup line is isolated to prevent release of fission product gases when the monitor alarm/trip setpoints are exceeded. These trip setpoints are established in accordance with the Offsite Dose Calculation Manual (ODCM) as required by the Technical Specifications. When a monitor trip occurs, the holdup line is automatically isolated after a 15-minute delay. This time interval permits corrective action to be taken to avoid plant shutdown. [11.3-18]

11.3.2.1.6 Inspection and Testing

The gaseous waste disposal systems are used on a routine basis and do not require specific testing to ensure operability. Monitoring equipment is calibrated and maintained on a specific schedule and on indication of malfunction. [11.3-19]

The off-gas filters are replaced or changed over when the pressure drop across the filter exceeds the normal operation range. Instrumentation is provided for checking the installed efficiency of the filters. [11.3-20]

The off-gas systems are sampled weekly for performance and any adverse change is investigated further. Furthermore, the air dose due to release of radioactive noble gases in gaseous effluents is determined on a monthly basis by allocating the effluents between units and using the methods prescribed in the offsite dose calculation manual (ODCM). Doses due to the treated gases released to unrestricted areas, at or beyond the site boundary, are projected on a monthly basis in accordance with the ODCM methods of calculation. To minimize the potential for an explosive gas mixture to develop, the off-gas holdup system is monitored by the explosive gas monitoring system which is channel checked, functionally tested, and calibrated per the Technical Requirements Manual. The radioactive gaseous radiation monitoring instruments listed in the ODCM are demonstrated operable by performance or instrument check. [11.3-21]

11.3.2.2 Turbine Gland Seal System

11.3.2.2.1 System Description

The turbine gland seal system consists of the gland steam condenser and the gland steam condenser exhauster. There are two turbine gland seal systems for each unit. The turbine gland seal system provides a seal between the moving shaft and the stationary casing parts of the turbine to prevent air inleakage to the turbine during operation. The seals basically consist of a series of close tolerance rings in the casing which surround the turbine shaft at either end. By passing steam (at a pressure higher than atmospheric) from inside the casing through the seals, air is prevented from leaking into the turbine. Rather than exhaust this gland seal steam to the turbine building atmosphere, normal steam turbine design directs it to the gland steam condenser along with substantial quantities of air, which are drawn through the outer seals. Approximately 95% of the steam used in the turbine gland seals is condensed in the gland steam condenser and returned to the unit condensate system. The remaining steam, air and noncondensibles (including any radioactive gases) present in the gland steam are exhausted with nominal hold-up (approximately 1.75 minutes) to the plant chimney. The gland seal steam condenser exhauster maintains a vacuum on the gland seal steam condenser and exhausts the noncondensed portion of the gaseous volume to the holdup pipe prior to entering the chimney for discharge to the environment. [11.3-22]

The effluents from the gland seal system cannot be routed to the air ejector recombiner and charcoal beds. The relative absence of hydrogen renders a recombiner useless for reducing the effluents from this system. In using charcoal to delay radioactive noble gases the volume required for a given delay time is directly proportional to the gas flow. The noncondensible air and gas flow from the gland seals is about 30-50 times larger than the flow of noncondensibles exiting a recombiner in the off-gas system. Therefore, dynamic charcoal adsorption is not practical for treatment of the gland seal effluent discharged to the chimney. The shorter holdup time is adequate because the activity present in this system is three orders of magnitude less than that from the condenser air ejector. The holdup time allows decay of N-16 and O-19, which have half-lifes on the order of seconds.

11.3.2.2.2 Description of Major Components

11.3.2.2.2.1 Turbine Gland Seal Steam Condenser

The condenser using cooling water through double-pass tubes condenses the steam in the gas stream. [11.3-23]

11.3.2.2.2.2 Turbine Gland Seal Steam Condenser Exhauster

The exhauster maintains a vacuum on the turbine gland seal steam condenser. The exhauster vents to the 1.75-minute holdup volume.

11.3.2.3 Mechanical Vacuum Pump System

The mechanical vacuum pump system establishes and maintains the main condenser vacuum at 20—25 inches of mercury. This system, which is used when steam to operate the air ejectors is not available, exhausts through a discharge silencing tank at about 2320 standard ft^3/min of gas (air) at 15 in.Hg. The pump discharges this flow of contaminated gaseous effluent to the base of the 310-foot chimney via the gland seal holdup piping. There is one condenser vacuum pump and silencer for each unit. [11.3-24]

11.3.3 Radioactive Releases

11.3.3.1 Plant Release Points

Radioactive gaseous releases to the environment occur from only two release points, the plant 310-foot chimney and reactor building ventilation stack. [11.3-25]

11.3.3.1.1 Reactor Building Ventilation Stack

The reactor building ventilation system, including the drywell ventilation system and the drywell purge system, discharges the ventilation air and radioactive gases at an approximate flow rate of 209,000 ft^3/min for both units through the reactor building ventilation stack. The typical radioactive gases discharged on an annual basis, based on historical data and pre-uprate conditions, are shown in Table 11.3-2. Core uprate to 2957 MWt is expected to increase the activity in the gaseous effluents by the percentage of the uprate, i.e., 18%. The reactor building and drywell ventilation systems are discussed in greater detail in Section 9.4. The physical and process design characteristics of the two gaseous release points are shown in Table 11.3-3. The limitations for release of gaseous effluents from the plant are set in the ODCM. [11.3-26]

11.3.3.1.2 Plant 310-Foot Chimney

The ventilation system air flow through the chimney is approximately 300,000 ft^3/min during normal operation of both Units 1 and 2. The radioactive gaseous flow from the off-gas system, the turbine gland seal systems and the SBGTS is estimated to be 12,000 ft^3/min during operation of both units. The radioactive gaseous system flows for the reactor building vent stack and the main chimney are shown on Figure 11.3-1 and in the ODCM. [11.3-27]

Natural dispersion of gases into the atmosphere is achieved in an efficient manner by discharge through the chimney. The combination of its height, the exit velocity of the effluent, and the buoyancy of the exit gases, promotes favorable plume behavior for efficient dispersal. The height of the chimney assures that diffusion of the plume will not be influenced by the eddy currents occurring around the station structures. Based upon diffusion characteristics of the gases, and considering the meteorological characteristics of

the site and surroundings, it is calculated that release from the top of the 310-foot chimney contributes to a reduction in offsite ground-level concentration by a factor of approximately 100 as compared with release of the gaseous wastes at ground level.

The major source of gaseous waste activity is the steam jet air ejector effluent. Turbine building ventilation exhausts contribute large volumes of air to the chimney effluent but very little activity in comparison to the major sources. Likewise, reactor building ventilation exhausts contribute little activity via the reactor building vent stack. Drywell gases are purged and exhausted from the reactor building vent stack to reduce residual airborne contamination prior to personnel access. Provisions are made to automatically divert the purge exhaust through the SBGTS at a reduced flow rate corresponding to system capacity if activity is present in any significant quantity. These gases are then discharged from the ventilation chimney. Air ejector off-gases are normally expected to have the composition shown in Table 11.3-4.

The activation gases listed in Table 11.3-5 (principally N-13) are released from the chimney at the rate of approximately 250 μCi/s per unit during rated power operation. The rate of release of these gases is proportional to the thermal output of the reactor and the holdup time in the system before release at the chimney.

The fission product gases may come from minor amounts of tramp uranium on the surface of the fuel cladding, from imperfections, or from perforations which might develop in the fuel cladding. In the absence of fuel rod leaks, N-13 from the air ejector off-gases, and the N-16, O-19 from the gland seal system are the principal contributions to environs radiation dose. The aggregate of these three isotopes corresponds to a radiation dose of less than 0.1 mrem/yr at the site boundary. If fuel rod leaks do occur, the noble radioactive gases, Xenon and Krypton, become the principal contributors. The particulate daughter products of the noble gases are removed by the off-gas system charcoal adsorbers and filters prior to release of gases to the chimney. [11.3-28]

The principal gaseous isotopes from fission product sources which are discharged from the chimney are shown in Table 11.3-6. The emission rate of fission gases presented in Table 11.3-6 totals approximately 600 μCi/s and is representative of the total leakage during a year.

The most thorough tritium balance made on a BWR was made at Dresden Unit 1 by the U.S. Public Health Service.[1] This study suggests that the activation of deuterium in the coolant water within the reactor is probably the main source of tritium in the primary coolant. The formation rate was calculated to be 0.06 μCi/s for this 700 MWt reactor. The measured release rate of tritium in liquid radwaste was 0.05 μCi/s which accounted for approximately 90% of the release with about 10% leaving as gaseous waste. Thus, essentially all of the Dresden Unit 1 tritium formation can be accounted for by the activation of deuterium. This is further confirmed by tritium concentrations measured in the reactor water at other boiling water reactors using zircaloy fuel with varying numbers of fuel failures. Based on these studies, the tritium formation rate in each Quad Cities unit is expected to be about 0.3 μCi/s. [11.3-29]

Most tritium formed is recombined with oxygen to form tritiated water. This tritium is drained to the main condenser where it becomes condensate for the feedwater system. It is removed from the system by its radioactive decay. The tritium which is not recombined is released from the main chimney. [11.3-30]

The limitations for release of gaseous effluents from the plant are set in the ODCM. These quantities of radioisotopes released provide a dose rate in the unrestricted areas surrounding the site boundary (including at the site boundary) that is below the allowable limits set by 10 CFR 20 and 10 CFR 50. The methodology for calculating the release doses is presented in the ODCM and applies methods and equations consistent with Regulatory Guides 1.109 and 1.111 and NUREG 0133. The typical annual gaseous release, including tritium, from the plant chimney based on historical data and pre-uprate conditions is shown in Table 11.3-7. Core uprate to 2957 MWt is expected to increase the activity in the gaseous effluent by the percentage of the uprate, i.e., 18%. [11.3-31]

11.3.3.2 Process Monitoring and Sampling

The gaseous effluent is sampled on a continuous basis at both points of release. Provisions are also available for sampling the gaseous effluent manually using laboratory techniques. The radioactive gaseous effluent can be sampled at various process points such as at the steam jet air ejector, the exit of the recombiner, the exit of the various carbon adsorption beds and the chimney. The reactor building vent effluent is sampled continuously. Provisions for laboratory sample analysis are also included. Sampling also includes isotopic analyses and isokinetic sampling. Additional details of the process monitoring instrumentation is given in Section 11.5. [11.3-32]

A remote radioactive gaseous effluent monitoring system has been installed and housed in a specially constructed facility for the State of Illinois Department of Nuclear Safety as required by the Illinois Safety Preparedness Act No. 83-1342 of September 4, 1984. This independent facility monitors the radioactive gases released from the chimney and relays the information directly to the IDNS office in Springfield, Illinois. [11.3-33]

11.3.4 References

11.3-2 Kahn, B., et al, "Radiological Surveillance Studies at a Boiling Water Nuclear Power Reactor," BRH/DER 70-l, March, 1970.

Table 11.3-1

PROCESS INSTRUMENT ALARMS

Parameter	Main Control Room Indicated	Main Control Room Recorded
Recombiner inlet temperature - low	X	
Recombiner catalyst temperature - high/low		X
Off-gas condenser drain well (dual) level - high/low	X	
Off-gas condenser gas discharge temperature - high	X	
Cooler - condenser discharge temperature - high		X
Glycol solution temperature - high/low		X
Glycol storage tank level - low		
Prefilter DELTA-P - high	X	
Moisture (charcoal bed inlet) - high		X
Charcoal bed temperature - high		X
Charcoal vault temperature - high/low		X
Adsorber vault radiation - high	X	
Adsorber outlet radiation - high/high-high		
Gas flow (post filter inlet) - high/low		X
High efficiency particulate air filters DELTA-P - high	X	

Table 11.3-2

REACTOR BUILDING VENTILATION STACK TYPICAL ANNUAL GASEOUS EFFLUENTS

		Nuclides Released	Quantity, Ci/yr
(A)	Fission Gases	Kr-85	<LLD**
		Kr-85m	<LLD
		Kr-87	<LLD
		Kr-88	<LLD
		Xe-133	<LLD
		Xe-135	1.78E-01
		Xe-135m	4.93E-01
		Xe-138	<LLD
(B)	Iodines	I-131	2.62E-03
		I-133	1.52E-03
		I-135	1.69E-03
(C)	Particulates	Sr-89	5.54E-05
		Sr-90	3.57E-06
		Cs-134	<LLD
		Cs-137	2.63E-04
		Ba-140	<LLD
		La-140	3.54E-05
		Cr-51	1.06E-02
		Mn-54	4.46E-03
		Co-58	6.41E-04
		Co-60	9.70E-03
		I-131	<LLD
		Ag-110m	<LLD
(C)	Particulates (continued)	Zm-65	4.17E-05
		Mo-99	4.17E-03
		I-133	3.92E-05
		I-135	1.23E-03

Table 11.3-2

REACTOR BUILDING VENTILATION STACK TYPICAL ANNUAL GASEOUS EFFLUENTS

	Nuclides Released	Quantity, Ci/yr
	FE-59	3.03E-05
	Nb-95	1.54E-05
	HF-181	2.03E-05

* Based on 1990 Semi Annual Effluent Reports.

** LLD - Lower Limit of Detection.

Table 11.3-3

PHYSICAL AND PROCESS CHARACTERISTICS
OF GASEOUS RELEASE POINTS

Characteristics	Gaseous Release Point	
	Chimney	Reactor Building Vertical Stack*
Height (above grade)	310 ft (95m)	159 ft (48m)
Inside diameter	11 ft (3.35m)	9 ft (2.74m)
Exit velocity	70 ft/s (21.4 m/s)	54.8 ft/s (16.7 m/s)
Discharge volume	421,300 cfm	209,000 cfm
Heat rate control	9 x 10^4 cal/sec at 399,900 cfm and is proportional to flow	---

* The reactor building vent stack itself is 55 feet tall and is mounted on the turbine building.

Table 11.3-4

AIR EJECTOR OFF-GAS COMPOSITION

	Flow Rate (ft^3/min at 130°F, 1 atm.)
Hydrogen	81
Oxygen	40.5
Air (assumed condenser leakage)	18-42
Water vapor (to saturate)	33-37
Activated noble gases	Negligible
TOTAL	172.5 – 200.5

Table 11.3-5

TYPICAL OFF-GAS ACTIVATION GAS COMPOSITION FOR A SINGLE UNIT
(Data must be doubled for Units 1 and 2 Combined Release Rates)

Isotope	Half-Life	Emission Rate μCi/s
H-3	12.4 yr.	3×10^{-2}
N-17	14 sec.	1×10^{0}
N-16	7.35 sec.	1×10^{0}
O-16	29 sec.	1×10^{0}
N-13	10 min.	2×10^{2}
Ar-41	1.83 hr.	2×10^{1}
Ar-37	34.3 day	1×10^{-4}

Table 11.3-6

TYPICAL OFF-GAS FISSION GAS COMPOSITION

Isotopes	Half-Life Days	Emission Rate µCi/s
Xe-135m	0.0106	5×10^1
Xe-138	0.0098	1×10^2
Kr-87	0.053	2×10^1
Kr-88	0.118	9×10^1
Kr-85m	0.187	6×10^1
Xe-135	0.379	1×10^2
Xe-133	5.24	1×10^2

Table 11.3-7

TYPICAL MAIN CHIMNEY ANNUAL CHIMNEY GASEOUS EFFLUENTS*

	Nuclides Released	Quantity, Ci/yr
(A) Fission Gases	Kr-85	<LLD
	Kr-85m	9.88×10^{-1}
	Kr-87	1.93×10^{0}
	Kr-88	1.10×10^{0}
	Xe-133	4.48×10^{0}
	Xe-135	2.54×10^{0}
	Xe-135m	9.87×10^{0}
	Xe-138	4.25×10^{1}
	Ar-41	1.10×10^{0}
(B) Iodines	I-131	1.87×10^{-3}
	I-133	7.77×10^{-3}
	I-135	1.34×10^{-2}
(C) Particulates	Sr-89	1.54×10^{-3}
	Sr-90	1.04×10^{-5}
	Cs-134	<LLD**
	Cs-137	9.70×10^{-6}
	Ba-140	7.51×10^{-4}
	La-140	3.24×10^{-3}
	Cr-51	1.02×10^{-4}
	Mm-54	1.56×10^{-4}
	Co-58	<LLD
	Co-60	3.97×10^{-4}
	I-131	1.65×10^{-4}
	Ag-110m	<LLD
	Mo-99	2.19×10^{-4}
	I-133	1.21×10^{-3}
	I-135	2.09×10^{-3}
	Fe-59	1.29×10^{-5}
(D) Tritium	H-3	1.16×10^{2}

* Based on 1990 Semi-annual Effluent Reports.
** LLD - Lower limit of detection.

11.4 SOLID WASTE MANAGEMENT SYSTEM

This section describes the capabilities of the station to collect, process, and package wet and dry solid radioactive waste generated as a result of normal operation, including anticipated operational occurrences for shipment to burial or onsite storage.

The in-plant cement solid waste system presently installed is no longer used. Contract services are normally used in lieu of the in-plant cement solid waste system for processing Class A unstable waste and waste which requires stability for burial offsite. The process control program (PCP) is used as applicable to process all low level radioactive wet wastes to meet the applicable federal, state, and burial site requirements. [11.4-1]

The PCP contains the current formulas, analysis, test, and determinations to be made to ensure that processing of solid radioactive waste based on demonstrated processing of actual or simulated wet solid radioactive wastes will be accomplished in such a way as to assure compliance with 10 CFR Parts 20, 61, and 71, state regulations, burial ground requirements, and other requirements governing the disposal of solid radioactive waste. [11.4-1a]

Changes to the PCP are documented and records of reviews performed are retained. This documentation contains: sufficient information to support the change together with the appropriate analyses or evaluations justifying the change(s); and, a determination that the change will maintain the overall conformance of the solidified waste product to existing requirements of Federal, State, other applicable regulations. The changes to the PCP shall become effective after review and acceptance, including approval by the Station Manager.

Written procedures have been established, implemented, and maintained covering the activities of the PCP.

11.4.1 Design Objectives

The design objective of the solid radioactive waste disposal system is to process, package and provide facilities for temporary storage of solid wastes prior to shipment from the station for offsite disposal. These solid radioactive wastes are shipped offsite, for subsequent disposal, on vehicles suitable to comply with limits set forth by NRC, DOT, and state regulations. [11.4-2]

11.4.2 System Description

The processing, packaging and handling, prior and subsequent to shipping or on-site storage, are performed in shielded and ventilated facilities and using procedures, the objectives of which are to minimize personnel radiation exposure and prevent spillage of radioactive wastes, while simultaneously providing for necessary cleanup and for equipment maintenance. [11.4-3]

The reactor wastes such as spent control rod blades and fuel channels, are stored for decay in the fuel storage pool, or packaged, and transferred to permanent disposal offsite in suitable approved shipping containers. Fuel channels may also be packaged and transferred to the Interim Radwaste Storage Facility for temporary storage prior to offsite disposal.

The maintenance wastes such as contaminated clothing, tools, paper, and plastic can be compressed in approved drums for disposal offsite.

The process wastes such as filter sludges and spent resins, are collected in tanks. These wastes can be solidified in 55 gallon drums, steel liners, or high integrity containers (HIC). HICs can be dewatered onsite, partially dewatered onsite, and shipped offsite for final dewatering, or shipped offsite for complete dewatering. Chemical treatment of the HICs may be performed if the presence of biological activity is detected. The drum filling equipment is operated remotely with viewers, and conveyors that transport the drums through the drum filling line and within the storage areas. The steel liners or high integrity containers (HIC) are processed in a truck mounted shipping cask or process shield. The following are typical of wet and dry solid radioactive wastes: [11.4-4]

A. Filter sludges;

B. Spent resins;

C. Air filters from off-gas and radioactive ventilation systems;

D. Contaminated clothing, tools, and small pieces of equipment which cannot be economically decontaminated;

E. Miscellaneous paper, rags, plastic, etc., from contaminated areas;

F. Dry wastes from equipment that has been activated during reactor operation;

G. Oily sludges; and

H. Cartridge filter elements.

11.4.2.1 In-Plant Cement Solid Waste System

During early plant history the cement solid waste system was used to solidify class A unstable waste forms. Due to improvements in current dewatering technology, this system is no longer used at Quad Cities Station. [11.4-5]

11.4.2.2 Contractor Supplied Solidification System

Contractor solidification services may be utilized at the station for wastes which are required to be classified as stable waste per 10 CFR 61 and/or the burial site licenses. These services may also be used to process wastes that are not required to be stable. Vendor procedures and the process control program are reviewed to assure compatibility with the station systems and procedures. Specific station procedures are developed and approved prior to use of the contract services. [11.4-6]

The waste to be processed by this system is transferred to the mixing tank where it is concentrated by a series of fills and decantations. The waste is then transferred to a disposable liner or HIC where the appropriate contractor solidification materials are added to solidify the radioactive waste. In certain cases (for example, waste resulting from tank cleaning), the waste is sent directly to the disposable waste container. After solidification is completed, the liner is closed, surveyed, and either shipped for offsite burial or stored onsite in the Interim Radwaste Storage Facility (IRSF).

11.4.2.3 Contractor Supplied Dewatering System

Contractor dewatering services may be used at the station in lieu of solidification for stable and unstable waste forms. The wastes considered for dewatering are ion exchange bead resins, charcoal, sludge, and filter precoat media. The liner or high integrity container (HIC) is normally shipped to the plant in a suitable transportation cask. The filling head is positioned on the liner or HIC and the transfer and dewatering process is started. The contractor receives waste from the station radwaste mixing tank via the radwaste pump or other appropriate method (e.g., chemical decon resin columns). After the radwaste has been transferred to the liner or HIC, the balance of the dewatering process is completed by the contractor. The liquid waste resulting from this process is returned to the liquid radwaste system. Upon completion of the dewatering process, the filling head is removed and the container is inspected to verify that it is acceptable. [11.4-7]

Samples are obtained and the container is capped, secured, and surveyed. The container may be stored in the interim radwaste storage facility shipped for offsite dewatering and possible chemical treatment, or shipped to the burial site.

11.4.2.4 Contractor Encapsulation of Waste

Contractor encapsulation services may be utilized by the station for waste which are required to be classified as stable waste per 10 CFR 61 and/or burial site licenses. The contractor supplied procedures or other documents which support the encapsulation process are used by the station to prepare specific station procedures for review, prior to use, to assure compatibility with the station systems and procedures. [11.4-8]

The item to be encapsulated is placed inside an approved liner and the liner is filled with a stable formula of cement. After sufficient cooling, the encapsulation is inspected and the container is capped, secured, surveyed and shipped for burial or stored onsite in the Interim Radwaste Storage Facility (IRSF).

11.4.2.5 Dry Active Waste

Dry active waste (DAW) occurs from many sources: [11.4-9]

A. Air filters from the off-gas systems;

B. Air filters from the various plant ventilation systems;

C. Air filters from the standby gas treatment systems;

D. Contaminated clothing, tools, and small equipment;

E. Miscellaneous contaminated paper, rags, plastic, wood, etc.; and

F. Contaminated concrete chippings and dirt.

The dry active waste is collected from throughout the plant and normally taken to the Laundry, Tool, and Decontamination (LTD) Building compactor area. When the activity is greater than the level acceptable for packaging in the LTD Building, the waste is then taken to the radwaste building to be packaged into 55-gallon drums.

The activity of dry active waste is normally low enough to permit handling by manual contact. These wastes are collected in containers located in appropriate zones around the plant, as dictated by the volume of wastes generated during plant operation and maintenance. The containers or their contents are then collected and moved to the LTD Building. At this location, the dry active waste can be packaged by various methods. Noncompressible wastes can be handpacked into metal boxes or loaded into C-vans. Compressible wastes can be compacted into drums by a hydraulic press to reduce their volume or can be handpacked into metal boxes or C-vans. Metal boxes or drums can be stored in the LTD Building or DAW Building, shipped offsite for burial, or shipped to a waste processing facility. C-vans are normally shipped offsite to a waste processing facility. Ventilation is provided to maintain control of contaminated

particles when operating packaging equipment or during equipment maintenance and also cleanup. [11.4-10]

Equipment too large to be handled in this way will require special procedures. Since the need for handling of large equipment is quite infrequent, providing storage facilities in advance is not justified. Handling of such equipment depends upon the radiation level, transportation facilities, and available storage sites. Suitable procedures for decontamination, shielding, shipment, and storage of such items are developed as necessary. [11.4-11]

11.4.2.6 Waste Water Treatment and/or Sewage Treatment Plant Sludge

Waste water treatment sludge is transferred to a drying area while sewage treatment sludge may either be land applied or transferred to a drying area. If drying of the sludge is utilized, the dried sludge is packaged by plant personnel in 55-gallon drums or other suitable waste container. The drums or other waste containers are closed and secured and moved to a transport vehicle or storage area to await inspection and shipment. Inspection verification of an acceptable product for shipment and burial is normally made at the time the transporting vehicle is loaded. [11.4-12]

11.4.2.7 Classification of Radioactive Wet Waste

Radioactive wet wastes which are solidified or dewatered are classified as either Class A, Class B, or Class C to determine the acceptability for disposal and for the purpose of segregation at the disposal site. The waste class is based on the concentration of certain radionuclides in the waste as outlined in 10 CFR 61.55. The most commonly used methods for waste classification are the direct measurement of individual radionuclides and the use of scaling factors to determine the radionuclide concentration. Station procedures are used to determine the radionuclide concentration for the classification of radioactive waste for burial. [11.4-13]

11.4.3 Inspection and Testing

Proper operation of this equipment is demonstrated prior to the actual handling of radioactive wastes. Normal operations preclude the necessity for testing equipment continually in use. Periodic inspection is performed, and equipment that is operated only periodically is tested as necessary to assure proper operation of valves and equipment to minimize the chance of failure or malfunction during operation. [11.4-14]

If the in-plant solidification process is in use, each drum of solidified waste is verified to be void of freestanding water prior to shipping or storage. If a drum is found to contain freestanding water, dry cement will be added to solidify the free water or the drum will be recycled through the mixing line as required. Drums of solidified waste will not be shipped with more than 0.5% freestanding water. [11.4-15]

When using an onsite contractor's solidification system, a visual inspection of the waste container contents is performed by both the contractor and station personnel prior to installing the lid. The visual inspection in conjunction with the contractor's PCP verifies that the product is acceptable. If the contractor's PCP or visual inspection does not verify solidification, the contractor will be required to provide the station an acceptable resolution.

When using an onsite contractor's dewatering system, verification of an acceptable dewatered product is performed by both the contractor and station personnel according to the contractor's and station procedures. The acceptance criteria is dependent upon the type of dewatering system used and the material dewatered.

When contractor encapsulation is performed, a visual inspection of each liner is performed to verify encapsulation prior to installing the lid. The visual inspection verifies that the product meets the acceptance criteria of the contractor's procedures. If the liner is not an acceptable product, the contractor will be required by the station to provide an acceptable resolution.

11.4.4 Storage of Solid Radwaste

Solid radwaste may be stored in several places onsite while awaiting shipment. The storage place depends upon the particular type of solid waste. [11.4-16]

11.4.4.1 Drummed Solidified Bead Resin

QCNPS no longer solidifies bead resin or other radioactive wastes. Although a General Electric in-plant cement solidification system was installed during initial construction and utilized for many years, QCNPS currently uses only commercial, vendor-supplied processing systems for the processing of the liquid wastes generated by the station. The GE in-plant cement system had initially functioned as intended; however, due to changing regulations on waste forms, its use was eventually restricted to processing only wastes which did not require stability, which accounts for only a small fraction of the total wet wastes. If solidification or encapsulation is to be performed onsite at QCNPS, then specific station procedures shall be developed as well as appropriate revisions made to the Process Control Program (PCP). [11.4-17]

11.4.4.2 Drummed Dry Active Waste

The DAW drums are normally stored in a designated storage area near the Laundry, Tool, and Decontamination (LTD) Building processing area if the waste is <=100 mrem/hr or in the DAW storage facility for on-site storage. The DAW may be stored in various work areas while waiting to be moved to the process area. If the DAW radiation level is >100 mrem/hr, the DAW is normally stored in the radwaste building drum storage areas, or in the DAW storage facility with Radiation Protection approval. [11.4-18]

11.4.4.3 Dry Active Waste Boxes

The filled DAW boxes are normally stored either in the designated storage area near the LTD Building process area or may be stored in various waste locations while awaiting shipment or may be stored in the DAW storage facility for on-site storage. Waste placed into the boxes is normally <=100 mrem/hr. Waste >100 mrem/hr can be placed into DAW boxes and placed in the DAW storage facility with Radiation Protection approval. [11.4-19]

11.4.4.4 Contractor Solidified, Dewatered, or Encapsulated Waste

Fuel channels and contractor solidified, dewatered, or encapsulated waste containers are normally shipped when processing is completed. If storage is required for any of these types of wastes, the containers of waste may be stored onsite, in the IRSF. [11.4-20]

11.4.4.5 Interim Radwaste Storage Facility

The interim radwaste storage facility (IRSF) is located inside the protected area and is used to temporarily store solid waste. The IRSF building is a stand-alone, reinforced concrete building.

The general layout of the IRSF is shown on drawing B-1804. The major IRSF areas are the truck bay, control room, equipment room, and storage bay. The truck and storage bays are serviced by a 20-ton crane. Remote cameras are used to aid in the operation of the crane.

The IRSF is designed for storage of radioactive waste in a ready-to-ship form. The IRSF is designed to store 416 containers. The containers are generally expected to average below 15 R/hr on contact.

11.4.4.6 Dry Active Waste Storage Facility

The DAW Building is located inside the owner controlled area. The DAW Building is a pre-engineered metal sided, cleared-span, steel frame building.

The metal drums and boxes intended for use in the DAW facility shall meet “General Design” (formerly Strong Tight) requirements at a minimum.

DAW containers are generally expected to average less than or equal to 30 millirem per hour at contact. This is based on filling the DAW Storage facility with approximately four years of generated DAW having an isotopic activity of 1.10 millicurie per cubic foot.

11.4.5 Shipping of Radioactive Waste

The solid radioactive waste is shipped from the plant to burial sites licensed and available to receive such material. Wastes that need further processing may be shipped via an offsite vendor. The typical volumes of solid radioactive waste shipped from the station based on historical data and pre-uprate conditions are given in Table 11.4-1. Core uprate to 2957 MWt will not significantly impact these estimates. [11.4-21]

Each burial site requires the shipper to obtain a burial permit before the site will accept solid waste. [11.4-22]

The lower activity radioactive wastes are normally loaded onto unshielded shipping vehicles for transporting to waste processors or a burial site. Higher activity wastes are normally shipped in an appropriate shielded cask on a transport vehicle. Procedures cover the various aspects of loading different types of shipping casks and different types of trucks, trailers, or C-vans with specific types of wastes. The marking of containers, the shipping papers, the determination of the radionuclides, and classification of the waste are all described and performed according to station procedures. [11.4-23]

11.4.6 Process Sampling

Process waste is sampled to determine the classification discussed previously and to determine the quantity of radionuclides (Ci of activity) to be shipped in the container. If onsite contractor solidification services are utilized, sampling is performed as necessary to determine the appropriate formula for solidification. Onsite sampling is performed in accordance with station procedures. [11.4-24]

11.4.7 Planning

Planning and scheduling are exercised in coordination with shipping contractors for the movement of waste packaged by the station or by a contractor. Planning is essential when the contractor is performing the containerization of the waste and the transporting of the packaged waste in one all-encompassing operation. The radwaste system planning is done to assure that wastes are processed in a timely manner to assure that station operations, refueling, and maintenance activities are coordinated. [11.4-25]

11.4.8 References

None

Table 11.4-1

TYPICAL VOLUMES OF RADIOACTIVE SOLID WASTE SHIPPED FROM THE STATION

Year	Volume (Cubic Feet)	Radioactivity (Milli Curies)
1985	42,508.30	55,401,907.86
1986	46,518.40	2,136,314.06
1987	32,459.80	28,815,338.19
1988	31,281.50	637,386.49
1989	34,569.10	133,112,042.77
1990	42,697.30	1,240,311.71

11.5 PROCESS AND EFFLUENT RADIOLOGICAL MONITORING AND SAMPLING SYSTEMS

This section describes the systems and equipment that monitor and sample the process and effluent streams in order to control releases of radioactive materials. These radioactive materials are generated as the result of normal operations, including anticipated operational occurrences, and during postulated accidents.

Process monitoring is provided throughout the system to assure proper operation and control. Alarms are provided to warn of off-standard conditions such that corrective action can be taken to prevent overflows or equipment damage, and to correct equipment malfunction or process errors. Where necessary, automatic shutdown occurs on an alarm signal.

Area monitoring is provided as necessary in the accessible radwaste areas to inform personnel of local radiation conditions. The area radiation monitoring system is discussed in Section 12.3.

The plant onsite and offsite environs (locations identified in the Offsite Dose Calculation Manual) are monitored for gamma radiation levels and airborne particulates for periodic analysis. The environmental radiological monitoring and sampling program provides radiological information on the food chain materials consumed by area inhabitants.

11.5.1 Design Objectives

The design objectives of the radiation monitoring systems are: [11.5-1]

A. To continuously monitor and record in the control room the radiation level in accessible work areas of the plant to provide personnel protection;

B. To continuously monitor and record locally the radiation level of the surrounding area of the station in compliance with the ODCM to determine the effects of station operations on the background radiation of the area; and

C. To continuously monitor the operating systems of the plant, where the process fluid or gas is normally radioactive, and those that have the potential for radioactive release to the environs.

These process and environmental monitoring functions are performed by seven process-monitoring subsystems and a series of detectors strategically located throughout the plant and offsite.

The radiation monitoring systems provide indication in the control room, with the exception of the environs monitoring stations. As a general requirement, the various process monitors are capable of initiating appropriate alarms and actuating control equipment to assure containment of radioactive materials if pre-established limits are approached.

The monitoring subsystems which provide automatic protective functions, such as isolation valve closure or reactor shutdown, are designed so that a single component failure does not prevent the required action, or means are provided for back-up manual action to be taken prior to exceeding 10 CFR 20 release limitations, as defined in Section 11.3. All monitors are capable of self-supervision, i.e., give an alarm when downscale or de-energized.

Alarms are also provided to give warning if the monitor's sampling system malfunctions. All monitors are capable of operational verification by means of test signals, radioactive check sources, or natural background.

11.5.2 System Description

The purposes of the process radiation monitoring systems are to ensure that the release of fission products to the atmosphere and to unrestricted areas of the plant is within permissible limits, to provide continuous measurement of this release, and to alarm or to initiate automatic action to contain the radioactive release if the predetermined release rates are exceeded. Plant process lines monitored by this system include the main steam lines, main condenser air ejector off-gas line, 310-foot chimney discharge, reactor building ventilation exhaust, reactor building closed cooling water discharge, service water discharge, and radioactive waste (radwaste) system effluent. [11.5-2]

Functional subsystems of the process radiation monitoring system are as follows:

A. Air ejector off-gas monitoring subsystem,

B. Linear (flux tilt) monitoring subsystem,

C. Chimney gas monitoring subsystem,

D. Reactor building ventilation radiation monitoring subsystem,

E. Reactor building exhaust sampling,

F. Main steam line monitoring subsystem, and

G. Process liquid monitoring subsystem.

All subsystems except linear monitoring provide control room alarm annunciations upon detecting high radiation. Linear monitoring, an adjunct subsystem used in locating leaking fuel assemblies, provides a visual display and continuous record in the control room but no alarm.

11.5.2.1 Air Ejector Off-Gas Monitoring

To guard against the atmospheric release of fission products that might result from a fuel element failure, off-gas from the main condenser is monitored for gross gamma activity downstream of the steam jet air ejectors and catalytic recombiner and prior to release of the off-gas to the chimney in the holdup volume. Continuous radiation monitoring is maintained on the off-gas vent line. Provision is made for the removal of process samples for laboratory testing. A radiation monitor also monitors the charcoal adsorber vaults and alarms a high radiation condition in the control room. [11.5-3]

The off-gas radiation monitoring subsystem is shown in Figure 11.5-1. Off-gas radiation monitoring is performed in two channels, each of which includes a gamma sensitive ion chamber detector, and a logarithmic (log) radiation monitor with integral power supply. The two channels share a common recorder and a trip auxiliaries unit whose output feeds

an interval timer. The radiation level is displayed on a control room panel meter. Subsystem alarm annunciators and recorder are also located on panels in the main control room. [11.5-4]

The ion-chamber detectors in both channels are mounted adjacent to and just ahead of the off-gas holdup, a 36-inch diameter header whose function is to delay the off-gas volume. Flow through this header requires at least 30 minutes to reach the vent stack without the recombiner and 4 hours with the recombiner. Gamma radiation in the inlet flow ionizes gas in the detector chambers, producing a small direct current signal proportional to the radiation absorbed. Ion chambers will operate with an increasing output to at least 10^4 R/hr. This signal in each channel goes to a high-gain, dc amplifier in the log radiation monitor, whose output voltage is scaled to the usual radiation level experienced in the off-gas flow during normal plant operations. An upscale output from the log radiation monitor denotes high radiation in the off-gas flow. A downscale output from the monitor denotes a possible equipment malfunction. Output signals from the monitors go to the recorder and to the trip auxiliaries unit. [11.5-5]

A downscale output (radiation low) from either monitor actuates a control room equipment malfunction alarm and a trip in the trip auxiliaries unit. As described in sub-section 11.5.2.1.1 an upscale output from either monitor trips a switch in the recorder signaling that radiation has exceeded the prescribed limit. A gross upscale output (equivalent to the maximum instantaneous release rate) from either monitor actuates a high-high radiation alarm in the control room and a trip in the trip auxiliaries unit. Control logic in the auxiliaries unit is such that two upscale trips or one upscale and one downscale trip are required in order to initiate a time-delayed closure of the off-gas holdup valve, to prevent release of off-gas from the holdup volume to the chimney. This delay (15 minutes) is half the time required for the gas to travel through the holdup volume without the recombiner in operation. The operator thus has the opportunity to evaluate the high radiation condition before valve closure, without danger that hazardous radiation will be released to atmosphere. [11.5-6]

The time delay is provided by an adjustable delay switch in the interval timer. The timer output is applied to the off-gas isolation valve through a manual switch in the control room, which is set to AUTO during normal plant operations. Setting this switch to CLOSED immediately isolates the off-gas system. [11.5-7]

Closing of the holdup valve is accompanied by simultaneous closing of an air-operated valve that shuts the off-gas drain to the radwaste building equipment drain sump. For Unit 2, both valves are of the fail-close type, designed to close immediately on the loss of control power. For Unit 1, the drain valve is of the fail-close type. The Unit 1 holdup valve is of the fail-open type, designed to remain open on the loss of control power/air. In the event that the Unit 1 holdup valve fails open concurrent with a valid valve closure (high radiation) signal, operations would manually isolate off-gas effluent to the chimney prior to exceeding 10 CFR 20 release limitations. Closure resulting from a main steam line high-high radiation trip is accompanied also by the closing of pilot valves which control off-gas flow into the steam jet air ejectors and off-gas holdup from the main condenser. [11.5-8]

Power to one log radiation monitor is 120 Vac regulated from the RPS. Power for the second log radiation monitor is 120 Vac from the essential service bus. Integral power supplies provide high voltage to the detectors. The recorder draws 120 Vac from the instrument bus. Valve-closure power to the holdup and pilot valves is 120 Vac from the essential service bus.

Off-gas sampling used in conjunction with the air ejector off-gas radiation monitors is an off-gas vial sampler system. Figure 11.5-1 shows this system which includes two pumps, a sample vial with hypodermic type connector, a vial positioner, four normally- closed

solenoid-operated valves, manual valves, and a control panel. On the control panel are a pressure gauge, manual control switches, and indicator lights. The system is used to remove off-gas samples for laboratory analysis. Samples can be taken during normal plant operations and also during reactor shutdown. [11.5-9]

The radioactivity levels of N-16 and O-19 in the main steam lines are normally relatively high, but quickly decay due to their short half lives. Therefore, to obtain a more accurate indication of the activity levels of radioisotopes which affect the gas discharge limits through the chimney release point, the air ejector off-gas sample is monitored after a transportation time delay of at least 2 minutes in the holdup volume.

As indicated by Table 11.5-1, the air ejector off-gas monitors are of sufficient range and accuracy to detect an increase in air ejector off-gas radiation level. The redundancy incorporated into the monitoring system provides assurance that abnormal release of radioactive material will be detected, annunciated, and isolated. [11.5-10]

To calibrate the monitors, grab sample analyses are compared to the monitor indications at the time of sampling. Since the radioactivity levels of N-16 and O-19 in the main steam are normally relatively high, the transportation time delay to the air ejector off-gas monitor location allows for the rapid decay of the short-lived gases. The delay permits a more accurate indication of the activity levels of the longer-lived gases of interest.

11.5.2.1.1 Off-gas and main Steam Line Monitor Recorders Alarm Function

The Off-gas and Main Steam Line (MSLRM) Radiation Monitoring sub-systems have sensitivity to fission product release events. As documented by GE BWR safety evaluation NEDO-31400A, independent calculations were performed which indicated the MSLRM monitors capability. The calculations show that fuel cladding failures can be detected on the order of 10 fuel rods. [11.5-10a]

In compliance with NEDO 31400A requirements of Off-gas and MSLRM recorders alarm setpoints will be set to 1.5 times normal full power background radiation with hydrogen addition. This will provide prompt notification of impending flow blockage or other fuel cladding failure events to the operator. Operators can use the valid alarm signal to take appropriate actions required to minimize plant contamination and radiological releases. Administrative controls will be employed to ensure prompt sampling of reactor coolant is performed when the recorder alarm setpoint is exceeded.

11.5.2.2 Linear Monitoring Subsystem (Flux Tilt Monitor)

As shown in Figure 11.5-1, an off-gas linear monitoring subsystem is provided as an adjunct to the air ejector off-gas monitoring subsystem. Components of this subsystem are a gamma-sensitive ion chamber detector, a linear radiation monitor with a plug-in high-voltage power module, and a recorder. The detector is mounted adjacent to the off-gas header, near the gas holdup inlet. The linear (wide-range) monitor and the recorder are located in the control room. [11.5-11]

The detector senses changes in the off-gas radiation level and applies its signal to the monitor, whose output is on an expanded scale comparable to that of a logarithmic meter. Integral with the monitor is a range switch. The subsystem is used in conjunction with a planned series of control rod manipulations which, in the event of a failed or leaking fuel element, cause coinciding changes in the off-gas radiation level.

11.5.2.3 Chimney Gas Monitoring Subsystem

Gamma activity in the chimney discharge is monitored by two instrumentation channels, each of which uses a scintillation counter detector as its sensing element (see Figure 11.5-2). Samples of the gas are taken high in the chimney, where good mixing is ensured, and drawn through a constant-flow network past the detectors. Signals from the detectors pass through the pulse preamplifiers and radiation monitors to a recorder, where permanent record is made of gamma radiation in the samples. Purposes of the monitoring subsystem are to maintain continuous surveillance for gross radiation in the 310-foot chimney and to collect samples of iodine and particulates, if any, in the off-gas discharge. [11.5-12]

Each monitoring channel comprises a scintillation counter, shielded sample chamber, pulse preamplifier, and process radiation (log count) monitor with integral power supply for providing high voltage to the counter. The channels share the recorder, and a trip auxiliaries unit whose outputs initiate control room high and low radiation alarm annunciations. The recorder, alarms, and remote control switches for the chimney gas monitoring subsystem are located in the control room.

The scintillation counter is a gamma-sensitive crystal with photomultiplier, which views the sample flow through a thin retaining wall in the sample chamber. Radiation in the sample flow produces output pulses from the detector, which are applied through the pulse preamplifier and a double-shielded cable to the process radiation monitor. At around 10^7 cps the scintillation detector output will very likely have saturated and reversed due to internal charge build up and subsequent loss of distinguishable pulses. This is an inherent characteristic of any counting detector. Within the monitor, a logarithmic feedback element functions such that the output voltage of the monitor is proportional to the logarithm of the input current. This output signal is displayed on a front panel meter and applied to the recorder. Trip circuits in the monitor function through the trip auxiliaries unit to annunciate panel alarms. Upscale trips denote high radiation in the sampling line. Downscale trips denote a possible equipment malfunction. [11.5-13]

Each sample chamber is provided with a shielded check source (Cs-137) used for an operability check of the scintillation counter detector. The positioning of these check sources is controlled locally and also by two remote switches in the control room. [11.5-14]

Gas samples are continuously withdrawn from the 310-foot chimney by the chimney gas sample panel and pump assembly through a four-tube stainless steel isokinetic probe high in the chimney. This probe is designed to take a sample across the chimney proportional to the cross-sectional flow through the chimney such that a representative sample is obtained by the constant-flow pumping system. An in-line filter assembly is provided for periodic insertion in the sample line close to the isokinetic probe to determine the plateout between the probe and the filter assembly in the sample panel. [11.5-15]

Within the panel, flow is routed through a removable particulate and halogen filter assembly before admission to the sample chambers. The millipore particulate filter is designed to trap radioactive particles and the charcoal halogen filter to extract iodine from the input flow. These filters are removed periodically for analysis. [11.5-16]

The monitoring system designed for the Quad Cities units does not detect iodine in the presence of other gases. A sample is taken from the stack and is filtered to remove particulates before it is passed through a charcoal filter which removes the iodine. Originally, this charcoal filter was removed from the sample line and allowed to decay prior to its being counted to determine the activity. The decay period permitted short half-life isotopes to be diminished to a point where the iodine content can be successfully determined. This step is no longer necessary as the spectrum of isotopes from the fuel presently in use does not consist of a significant amount of the short lived radionuclides. [11.5-17]

The chimney effluent is also monitored by a separate particulate iodine and noble gas (SPING) 4 and a Victoreen sampling system. The Victoreen sampling system has high range capabilities to deal with accident type conditions. The Victoreen sampling system automatically begins taking samples after a high signal has been received from the SPING 4 low range noble gas monitor. The Victoreen sampling system has no trip functions associated with it. Output from the Victoreen monitor is obtainable in the control room. [11.5-18]

The Eberline SPING monitor is a self-contained microprocessor-based radiation detection system used to monitor for the radioactive noble gases in the air. A microcomputer performs the tasks of data acquisition, history file management, operational status check and alarm determination. A digital display is provided, which indicates the numerical data and status of the channel to which it is connected. Radiation is detected and the signals processed by the interface boxes. These output signals are converted to count rate by the microcomputer which then performs all mathematical calculations and control functions. [11.5-19]

The noble gas measurement is performed by several detectors viewing the gas sample. The low range and medium range detectors view the same sample volume. The low range monitor is a beta scintillation detector. The medium range monitor is an energy compensated Geiger-Mueller (G-M) tube detector. An energy-compensated G-M detector monitors the high range noble gas sample; its output is proportional to the gamma emissions of the sample. On a high alarm of each range, the SPING monitor automatically changes to the next higher range. Output from the SPING monitor can be obtained in the control room.

The SPING monitors, for the main chimney and the reactor building vent gaseous effluent sampling and monitoring, are controlled through the radiation monitoring system control terminal on control room panel 912-4. Both the liquid radwaste, service water, and gaseous effluent monitors are controlled by this terminal. The microcomputer associated with the SPING monitor is vulnerable to radiation damage from a total integrated dose greater than 1000 rads. [11.5-20]

11.5.2.4 Reactor Building Ventilation Radiation Monitoring Subsystem

This subsystem provides continuous gamma-radiation monitoring of the reactor building ventilation system exhaust and fuel pool area. Two sensor and converter (detector) units are located in each reactor building exhaust duct, and two other sensor and converter units are located where they can monitor the environment of each of the spent fuel pools. Shutdown and isolation of the reactor building ventilation system (described in Section 6.2.3) and startup of the standby gas treatment system (SBGTS), (described in Section 6.5.3) is initiated on any one upscale or two downscale trip signals from either set of monitors. The reactor building ventilation system will trip to the standby gas treatment system (SBGTS) at a release rate of 1.44×10^4 μCi/s. [11.5-21]

The combined reactor building ventilation effluent is also monitored by a SPING 4. This monitor has high range capabilities to deal with accident-type conditions. It has no trip functions associated with it. The SPING low range noble gas detector has a range 2.3×10^{-1} μCi/cc — 2.3×10^6 μCi/cc for Xe-133 which envelopes the range for the reactor building vent effluent trip point. [11.5-22]

As shown in Figure 11.5-3, the monitoring subsystem comprises four channels, each consisting of one sensor-and-converter detector and indicator and trip unit. The indicator and trip units in channels A and B of the exhaust duct monitors are four-decade units. The indicator and trip units in channels A and B of the fuel pool radiation monitors are six-decade units. To guard against monitoring signal loss, input power to one of the channels of each type is derived from a different source. Both A channel monitors are served by a common power supply which draws 120 Vac from RPS bus A. Both B channel monitors are served by a second common power supply which draws 120 Vac from ESS bus. The exhaust duct monitors provide output signals to a recorder in the control room. Output signals from the fuel pool monitors are not recorded. Various control room alarm annunciations are generated by the monitoring subsystems in the control room, but no RPS trips are provided to shutdown the reactor.

The sensor in each channel is a G-M tube, polarized by high voltage from the power supply. A G-M tube will operate with an increasing output (to at least 10^4 R/hr). The G-M tube circuitry is designed utilizing both the pulse and dc G-M modes with the outputs summed into the indicator and trip units. Additionally, a special circuit has been included to prevent meter fall-off at flux levels considerably in excess of full scale. The circuit output continues to increase until the amplifier saturates, and then remains at that level for further increases in flux. Under extended periods of operation at high levels, the detector could be subject to failure by deterioration of the ionization gas or by heat-induced mechanical failure. Pulses produced in the tube by radiation are processed by the converter to produce a dc current proportional to the logarithm of the input count rate. [11.5-23]

This signal is maintained approximately 5% of full scale above electrical zero by a low intensity radioactive (bug) source near the tube. The output signal from the sensor and converter is applied to the indicator and trip unit, where it is amplified and used to drive a meter. This unit also provides trip functions for upscale and downscale alarms through four auxiliary device units. A downscale trip in any channel annunciates a low radiation (malfunction) alarm. An upscale trip in channel A or B initiates a fuel pool high radiation alarm. A gross high-high trip in Channel A or B initiates a reactor building ventilation system high-high radiation alarm. Control logic is such that one high level trip or two low level trips in either set of monitors isolates the reactor building ventilation system and starts the SBGTS. [11.5-24]

High radiation in the exhaust duct also is annunciated in the control room through a switch in the recorder. This is an alarm condition only, and does not initiate protective action. [11.5-25]

The reactor building ventilation monitoring subsystem is set to isolate the reactor building ventilation and start the SBGTS upon detection of a refueling accident. The refueling accident offers the greatest potential for radioactive release via the reactor building ventilation exhaust. The high level set point is chosen sufficiently above refueling operations background radiation levels to avoid spurious trips, but low enough to detect and initiate a trip from the radiation level resulting from a design basis refueling accident. The trip setpoint will be at approximately 2 mR/hr above background for the RB vent rad monitor. The fuel pool rad monitors are set at ≤100 mR/hr trip setpoint. [11.5-26]

The refueling accident is evaluated in Section 15.7.2. The reactor building ventilation monitoring subsystem is effective in preventing radioactive release in excess of 10 CFR 100 limitations. [11.5-27]

The sensitivity, accuracy, and range capability of the reactor building ventilation exhaust monitors permit the monitor to detect radioactivity increases in the reactor building ventilation. The monitors are selected with physical and electrical characteristics permitting them to function in the reactor building ventilation environment. The range of the instrument is from 0.01 mR/hr to 100 mR/hr; therefore, the minimum sensitivity is 0.01 mR/hr. If the assumption is made that a release rate corresponding to the minimum detector sensitivity occurred under very stable 2m/sec meteorological conditions, the offsite dose at the nearest site boundary would be 9.0×10^{-3} mR/hr. [11.5-28]

Failure of a monitor which results in a downscale trip will not prevent isolation of the reactor building ventilation and initiation of the SBGTS if the other monitors detect a high radiation level. [11.5-29]

11.5.2.5 Reactor Building Exhaust Sampling

In conjunction with the reactor building monitors is a reactor building vent exhaust sampling station (Figure 11.5-4). The reactor building ventilation exhaust is sampled and monitored prior to release out the reactor building vent stack. The sample is taken through an isokinetic probe located in the exhaust duct to obtain a well-mixed sample. The sample passes through a particulate filter and a charcoal filter by means of a vacuum pump and flow control unit. The filter assembly is removable and is located at the sampling probe to minimize plateout. The filters are removed periodically for laboratory analysis. [11.5-30]

11.5.2.6 Main Steam Line Monitoring Subsystem

To guard against the significant release of fission products from the fuel to the reactor coolant and subsequently to the turbine, the main steam lines are continuously monitored for gross gamma activity immediately downstream of the primary containment outer isolation valves in these lines. Functions of the main steam line radiation monitoring subsystem are to provide immediate indication of gross radiation in the lines and to initiate automatic action to contain the radiation, or limit the release of radioactivity. [11.5-31]

As shown in Figure 11.5-5 the main steam line detectors, which are gamma-sensitive ion chambers, supply signals to four logarithmic (log) radiation monitors with integral power supply. Each steam line is viewed by all detectors. The power supplied provides ionizing voltage to the detectors. The radiation monitors function like those of the air ejector off-gas monitoring subsystem, producing an output voltage compatible with the usual radiation level of the steam flow during plant operations. Output signals from the monitors are applied to the process computer and to 2 recorders in the control room. All 4 MSL radiation control monitor signals are recorded in the control room.

Each log radiation monitor has a downscale alarm trip and an upscale trip. Another alarm is preset at a value 1.5 times the normal background radiation with hydrogen addition and annunciates a main steam line high radiation alarm in the control room. This alarm is activated by a switch on the recorder. The log radiation monitor upscale trip (preset at 15 times normal background without hydrogen addition) is connected to the PCIS relay scheme, which has two safety channels both of which must be activated to initiate a protective action. Activating either A or B safety channels annunciates a channel A or B main steam line high-high radiation alarm in the Main Control Room. Activating both channels results in a trip of the condenser mechanical vacuum pump.

As a safeguard against signal loss in the main steam line monitoring subsystem, 120-Vac input power is supplied to the radiation monitors in both RPS safety channels from both an RPS power source and the essential service bus. Monitors A and C (one in each safety channel) derive input power from RPS bus A; monitors B and D (one in each channel) draw power from the essential service bus. The recorder obtains 120-Vac input power from the instrument bus. [11.5-33]

The main steam line monitors are located such that they are in the radiation field of the four main steam lines. The range and sensitivity of the monitors (Table 11.5-1) have been chosen such that the monitors are capable of detecting increases of radiation due to the activity release following gross fuel failure. [11.5-34]

Continuous recording of the main steam line radiation levels is available to the operator. Abnormal radiation levels are annunciated in the control room.

Gross fuel failure would result in significant increase of the main steam line radiation levels. The monitors would detect the increase and initiate the protective trip functions. The redundancy of detector channels and the general location of the detectors in the main steam line radiation field assure the reliability of the system in carrying out its protection functions.

11.5.2.7 Process Liquid Monitoring Subsystem

This subsystem provides continuous gamma-radiation monitoring of three process liquid streams or a side stream: the radioactive waste disposal system, the reactor building closed cooling water system, and the service water discharge system. As shown in Figure 11.5-6, one channel of instrumentation is provided for each process stream. The sensing element in each channel is a scintillation counter, positioned on a vertical (section) of the process liquid piping (RBCCW), or positioned in the sample bowl assembly (SW & RW). For the RBCCW rad monitor only, pulses from the detector are amplified by a pulse preamplifier in the vicinity of the detector and applied to a control room process radiation (log count) monitor. The monitor counts the pulses and produces an output voltage that is proportional to the logarithm of the input current, for application to a trip auxiliaries unit

and a control room recorder. The pulse count in each channel is displayed on a back panel meter. The SW and RW rad monitors were modified to Eberline monitors, which is discussed below. [11.5-35]

Monitors in the reactor building closed cooling water system and service water discharge piping share a common recorder and common alarm annunciators in the control room. An upscale trip from either channel initiates a high-radiation alarm. A downscale trip from either channel initiates an alarm, denoting an equipment malfunction. [11.5-36]

The radwaste effluent monitor supplies signals to a recorder located in the radwaste building. An upscale trip or a downscale trip from this channel annunciates alarms both in the control room and the radwaste building. The upscale trip provides a high radiation alarm and the downscale trip provides a separate monitor failure alarm, for both the control room and the RW control room.

The RBCCW process monitor obtains 24-Vdc power from the + 24-Vdc bus. Both recorders obtain 120-Vac power from the instrument bus. The SW rad monitor obtains 120-Vac power from the MCC 17-1-1 (U-1) and 27-1-1 (U-2), and the RW rad monitor from MCC 17-27-3.

Process sampling is accomplished manually by collecting samples for laboratory analyses from a number of process locations such as the waste sample tanks, the floor drain sample tanks, the river discharge tank, various sump tanks and equipment drain tanks, as well as the water systems such as the RHR loops A and B, RHR heat exchangers A and B, the reactor water cleanup system, the off-gas system, the chimney, reactor building vent stack, and the radioactive liquid waste discharge line. [11.5-37]

An Eberline monitor and sampler was installed on the liquid radwaste discharge and the service water discharge lines. These monitors consist of a liquid sampler and a sodium iodide crystalline detector and are controlled by the SPING control terminal. The liquid from these lines is continuously monitored during discharges from the respective systems.

The reactor building closed cooling water system is monitored for leakage of radioactive material into the system.

The control of the radioactivity concentration in liquid wastes which are processed in the plant or released to the diffuser pipe is achieved by analyses of samples from individual batches of waste liquids. The monitors provide an additional check to ensure that deviations do not occur as discharge is performed. Table 11.5-1 lists specific data pertaining to the sensitivities and accuracies of the monitoring equipment. [11.5-38]

11.5.2.8 High Radiation Sampling System

The high radiation sampling system (HRSS) provides sampling functions during normal operation. The HRSS also provides post-accident sampling capability. The HRSS is presented in detail in Section 9.3.2. [11.5-39]

11.5.3 Effluent Monitoring and Sampling

The effluent monitoring and sampling pertains to the liquid radwaste monitoring and sampling and to the gaseous radwaste monitoring and sampling as it relates to the discharge of radioactive effluent from the station. The liquid effluent monitoring is

addressed in Subsection 11.5.2.7. The gaseous effluent monitoring is addressed in Subsections 11.5.2.3 and 11.5.2.4. Prior to any liquid discharge from the river discharge tank, the liquid in the tank is sampled and analyzed. [11.5-40]

11.5.3.1 Environmental Radiation Monitoring

Onsite and offsite monitoring stations which measure the gamma radiation level and collect airborne particulates from the air for periodic analysis are provided to confirm that releases of airborne radioactive materials have been controlled within the limits established by license. [11.5-41]

Environs radiation monitors of two types are used at the Quad Cities plant: radiation monitors and sampling monitors. Direct radiation monitoring consists of two dosimeters which are provided at each location to monitor the integrated radiation exposure. Sampling monitors are particulate and iodine air samplers.

Monitoring is performed at onsite and offsite locations as described in the Offsite Dose Calculation Manual (ODCM). The site meteorological monitoring program is discussed in Section 2.3.

11.5.3.2 Response to Draft of the AEC Proposed General Design Criteria

For details of the response to these criteria, see Section 3.1. In addition, the gaseous and liquid release paths are monitored during normal operation including any anticipated operational occurrences. [11.5-42]

11.5.3.3 Environmental Radiological Sampling Program

The Environmental Radiological Sampling Program as presented in the original FSAR (Section 2.7) and Amendments to the FSAR is historical. The facility Technical Specifications define the current Environmental Radiological Sampling Program. The Offsite Dose Calculation Manual (ODCM) expands upon the Environmental Radiological Sampling Program as defined in the facility Technical Specification. The results of the Environmental Radiological Sampling and Analytical Program are reported in the Annual Radiological Environmental Operating Report. This report also includes the results of the annual land use census. The program is tailored specifically to the Quad Cities site as described in the ODCM which lists the details of the sampling and analyses to be performed. In the Final Environmental Statement (FES) for the station it is stated that the post-operational program is basically a continuation of the preoperational program and the details of the monitoring program will be included in the Technical Specifications. [11.5-43]

11.5.4 Process Monitoring and Sampling

This section addresses the isolation function of the process gaseous radwaste monitors in the chimney and the reactor building vent stack. The intertie between the reactor building ventilation system, the SBGTS system and the fuel pool monitors is discussed. The action of a high radiation alarm from the main steam line monitors is also addressed. [11.5-44]

11.5.4.1 Response to Draft of the AEC Proposed General Design Criteria

For details of the response to these criteria, see Section 3.1. In addition, the effluent samplers and monitors for the two gaseous release points cause isolation at the discharge points when the appropriate radiation alarm signal is given. The reactor building ventilation radiation monitoring subsystem (Section 11.5.2.4) isolates the reactor building ventilation system and starts up the SBGTS system when a high radiation level trip occurs. The spent fuel pool radiation monitors (Section 11.5.2.4) also isolate the reactor building ventilation system and start up the SBGTS system when a high radiation level trip occurs. [11.5-45]

The radiation monitor and samples on the liquid radioactive waste discharge line monitors the activity as the waste is discharged. The two liquid discharge points are the south diffuser pipe and the discharge bay. The canal discharge point is present but not used; the canal is currently used as a fish hatchery for various stocking programs.

Table 11.5-1

RADIATION MONITORING SYSTEM PRINCIPAL DESIGN PARAMETERS

		DETECTING						
	General Monitoring Type	Type	Radiation Detected	Detector Sensitivity	Maximum Temperature °F	Relative Humidity	Check Source	Energy Response
Air ejector off-gas	Radioactive gas	Ionization chamber	Gamma	$3.7x10^{-10}$ Amps/R/hr +/-20%	110	98	Built-in electronic	
Radioactive effluent gas	Radioactive gas	Scintillation	Gamma	Per ODCM requirements	110	98	Built-in radiation	
Main steam line	Area	Ionization chamber	Gamma	$3.7x10^{-10}$ Amps/R/hr +/-20%	110	98	Built-in electronic	
Process liquid/radwaste	Liquid effluent	Scintillation	Gamma	Per ODCM requirements	110	98	Built-in electronic	70 keV — 7 MeV
Process gas	Radioactive gas	G-M tube	Gamma	Refuel floor Rad Monitors: $\leq$ 1 mR/hr RB Vent Rad Monitors: $\leq$ 0.01 mR/hr	140	98	Manual radiation built-in electronic	80 keV — 7 MeV
Area radiation monitoring system all channels	Area	G-M tube	Gamma	$\leq$ Bottom of Range per Tables 12.3-3, 12.3.-4, and 12.3-5	140	98	Manual radiation Built-in electronic	80 keV — 7 MeV
Environs radiation monitoring system all channels	Air particulate		Beta gamma					

Table 11.5-1 (Continued)

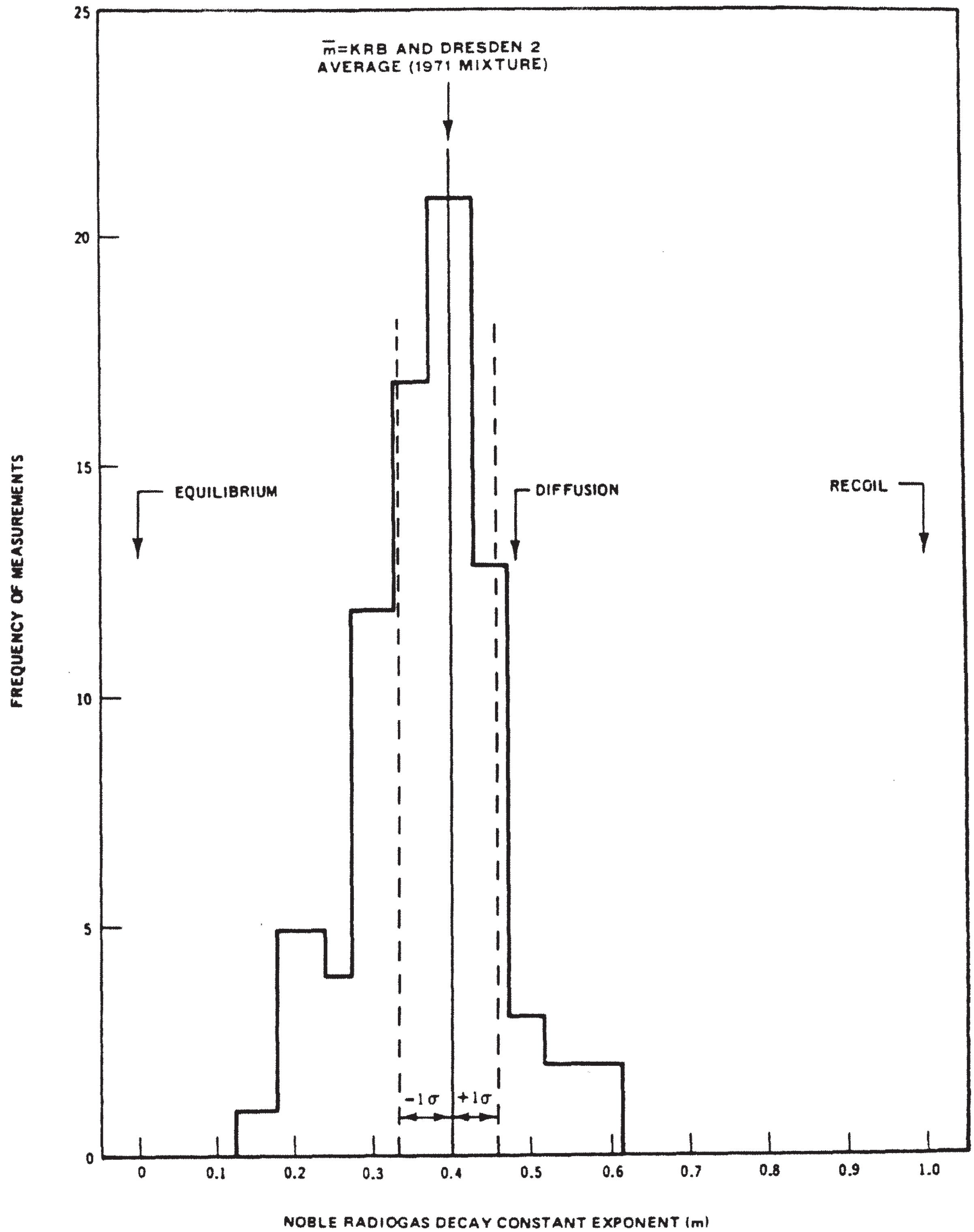
RADIATION MONITORING SYSTEM PRINCIPAL DESIGN PARAMETERS

		INDICATING			ANNUNCIATING			
	Type	Scale	Power	Location	Type	Location	Alarm Hi	Alarm Low
Air ejector off-gas	Picoammeter	10^{-3} — 10^{3} R/hr Log	Station supplied	Remote in control room	Visual and audio	Remote in control room	X	
Radioactive effluent gas	Count rate meter	10^{-1} — 10^{6} cps Log	Station supplied	Remote in control room	Visual and audio	Remote in control room	X	
Main steam line	Picoammeter	10^{-3} — 10^{3} R/hr Log	Station supplied	Remote in control room	Visual and audio	Remote in control room	X	
Process liquid/radwaste	Count rate meter	10 — 10^{6} cpm Log	Station supplied	Remote in control room	Visual and audio	Remote in control room	X	
Process gas	Count rate meter	10^{-2} — 10^{2} (RB Vent) 10^{0} — 10^{6} (Refuel floor)	Station supplied	Remote in control room	Visual and audio	Remote in control room	X	
Area radiation monitoring system all channels	Count rate meter	Various Log	Station supplied	Remote in control room	Visual and audio	Remote in control room same local	X	

Table 11.5-1 (Continued)

RADIATION MONITORING SYSTEM PRINCIPAL DESIGN PARAMETERS

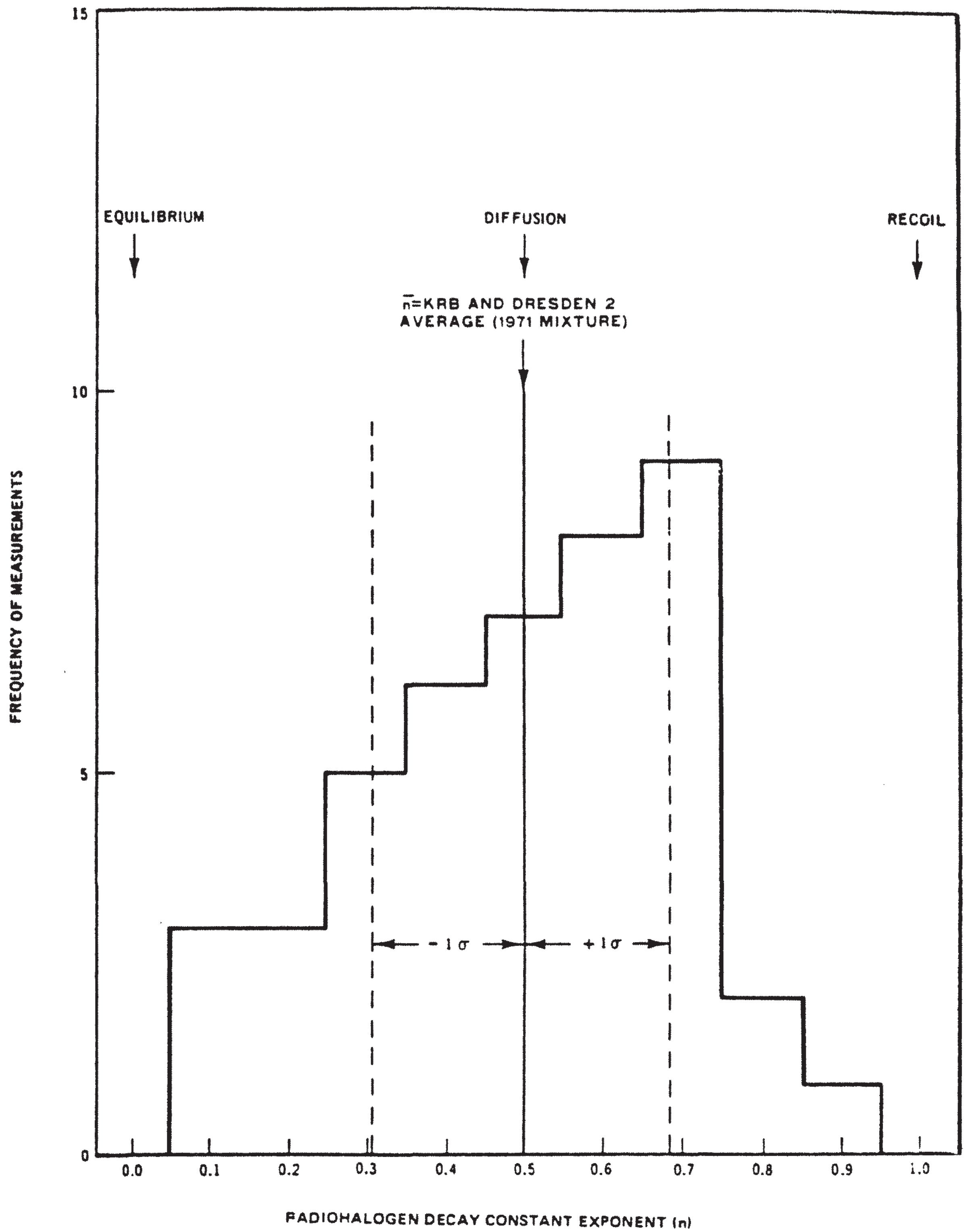
	RECORDING			SAMPLING		CONTROLLING
	Type	Response	Scale	Location	Measured Media	
Air ejector off-gas	Recorder	Chart speed 1 in./hr per speed 1 or 5 s/scale	6 Decade log	Off line	Air	Isolates off-gas isolation valve
Radioactive effluent gas	Recorder	Chart speed 2 in./hr	7 Decade log	Off line	Air shielded	
Main steam line	Recorder	Chart speed 1 in./hr per speed 1 or 5 s/scale	6 Decade log	In line	Steam	Trips cond. Vac. pump
Process liquid and radwaste	Recorder (2)	Chart speed 2 in./hr per speed 1 or 5 s/scale	5 Decade log	In line	Water	
Process gas	Recorder (RB Vent Rad Monitors only)	Chart speed 2 in./hr per speed 1 or 5 s/scale	4 Decade log	in	Air	
Area radiation monitoring system all channels	40-point multipoint recorder	Chart speed 2 in./hr	Various Log		Air	



QUAD CITIES STATION
UNITS 1 & 2

NOBLE RADIOGAS DECAY CONSTANT
EXPONENT FREQUENCY HISTOGRAM

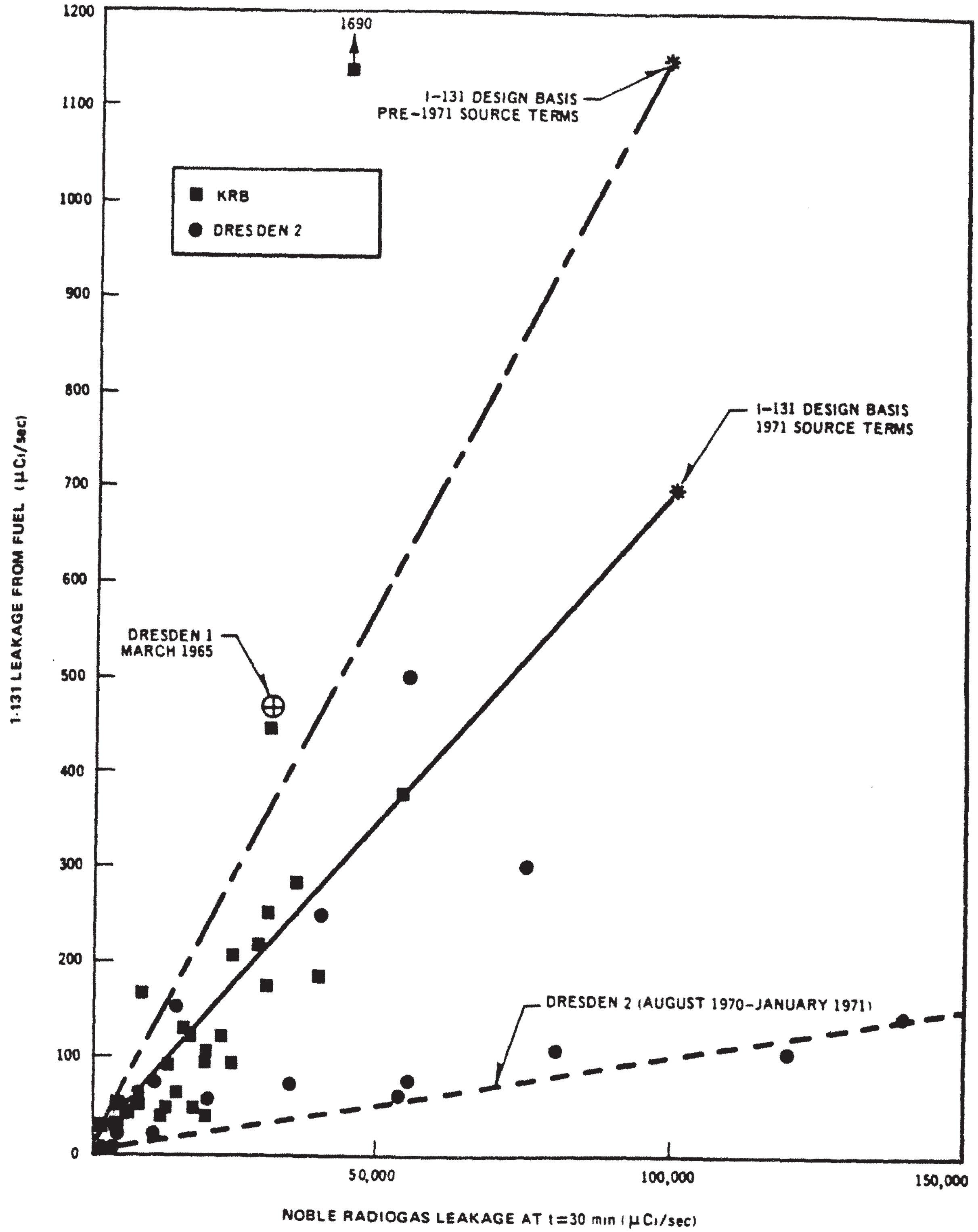
FIGURE 11.1-1



QUAD CITIES STATION
UNITS 1 & 2

RADIOHALOGEN DECAY CONSTANT
EXPONENT FREQUENCY HISTIGRAM

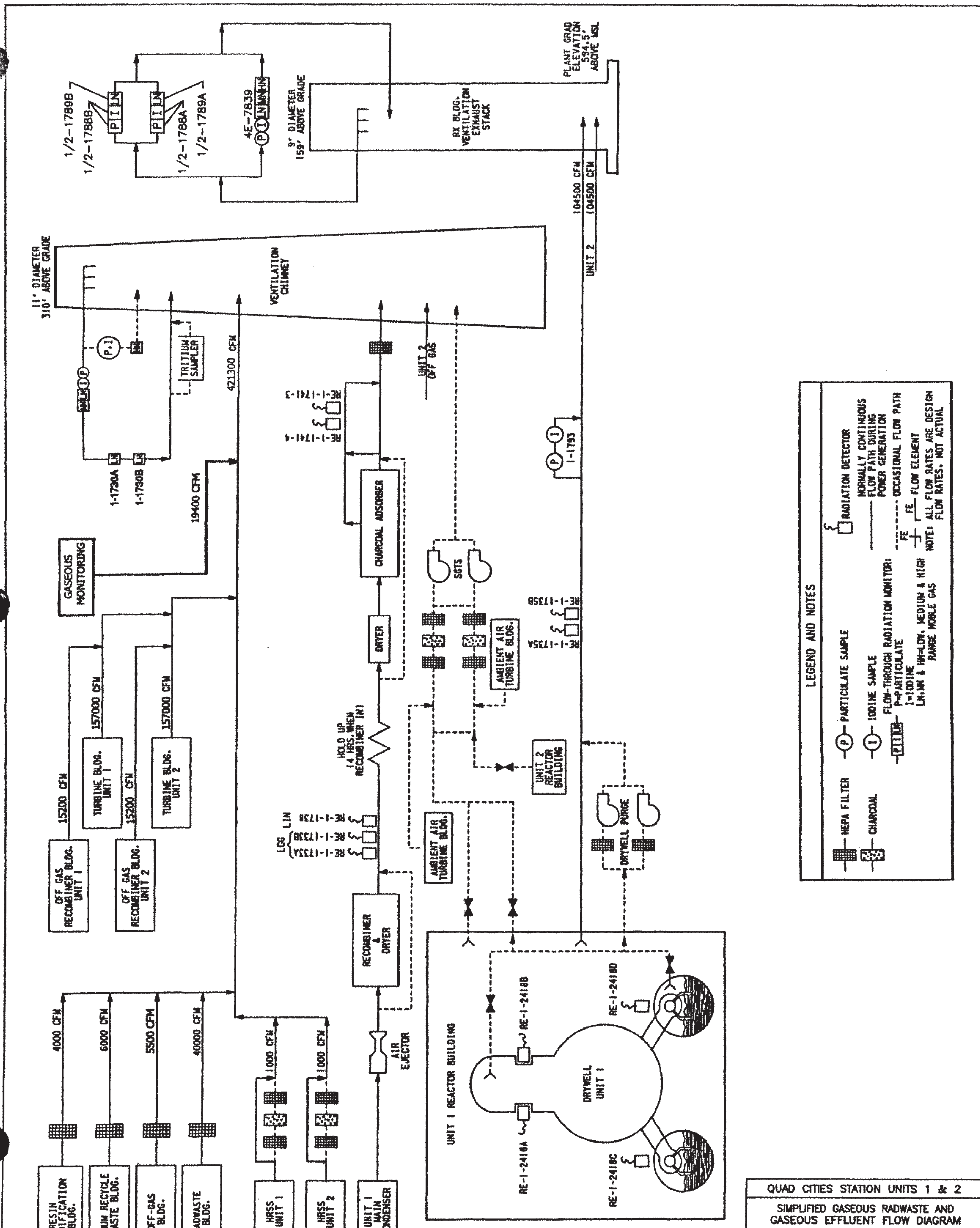
FIGURE 11.1-2



QUAD CITIES STATION
UNITS 1 & 2

NOBLE RADIOGAS LEAKAGE VERSUS
I-131 LEAKAGE

FIGURE 11.1-3

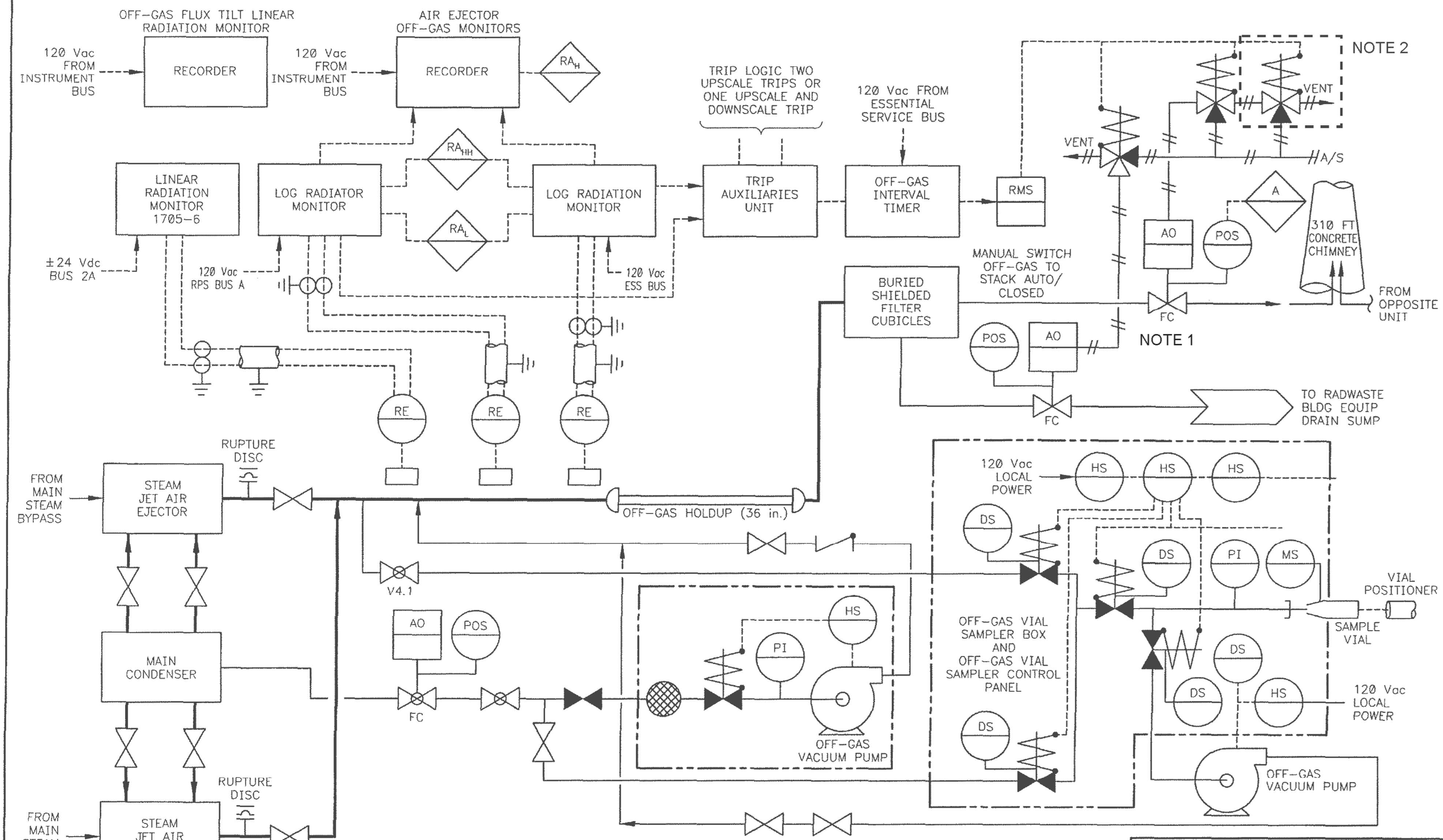


QUAD CITIES STATION UNITS 1 & 2

SIMPLIFIED GASEOUS RADWASTE AND GASEOUS EFFLUENT FLOW DIAGRAM

FIGURE 11.3–1

REVISION 6, OCTOBER 2001



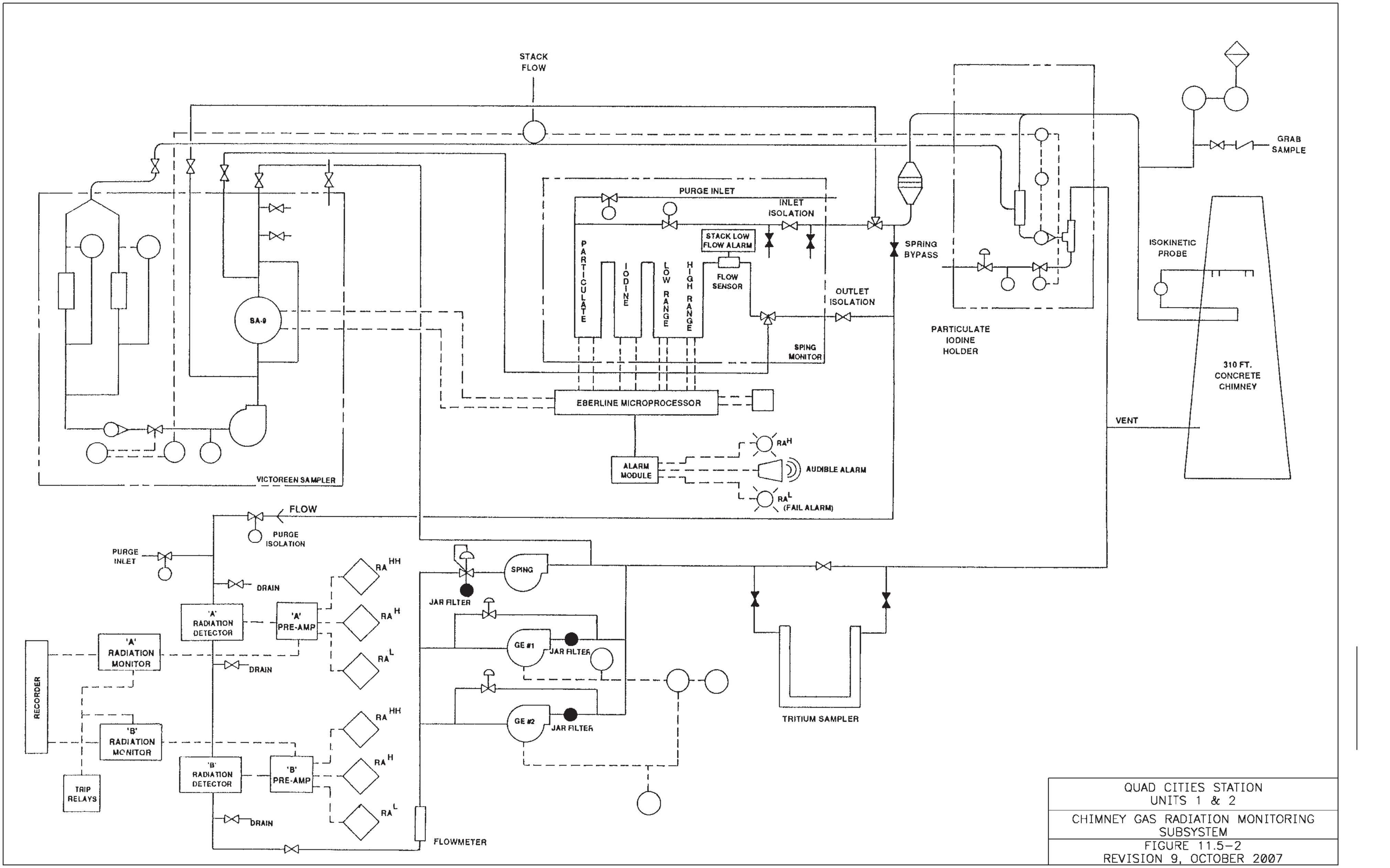
NOTES:
1. UNIT 1 FAILS IN THE OPEN POSITION. UNIT 2 FAILS IN THE CLOSED POSITION.
2. REDUNDANT SOLENOID VALVE INSTALLED ON UNIT 2 ONLY. UNIT 1 HAS ONLY ONE SOLENOID VALVE.

QUAD CITIES STATION UNITS 1 & 2

AIR EJECTOR OFFGAS AND LINEAR RADIATION MONITORING SUBSYSTEMS

FIGURE 11.5-1

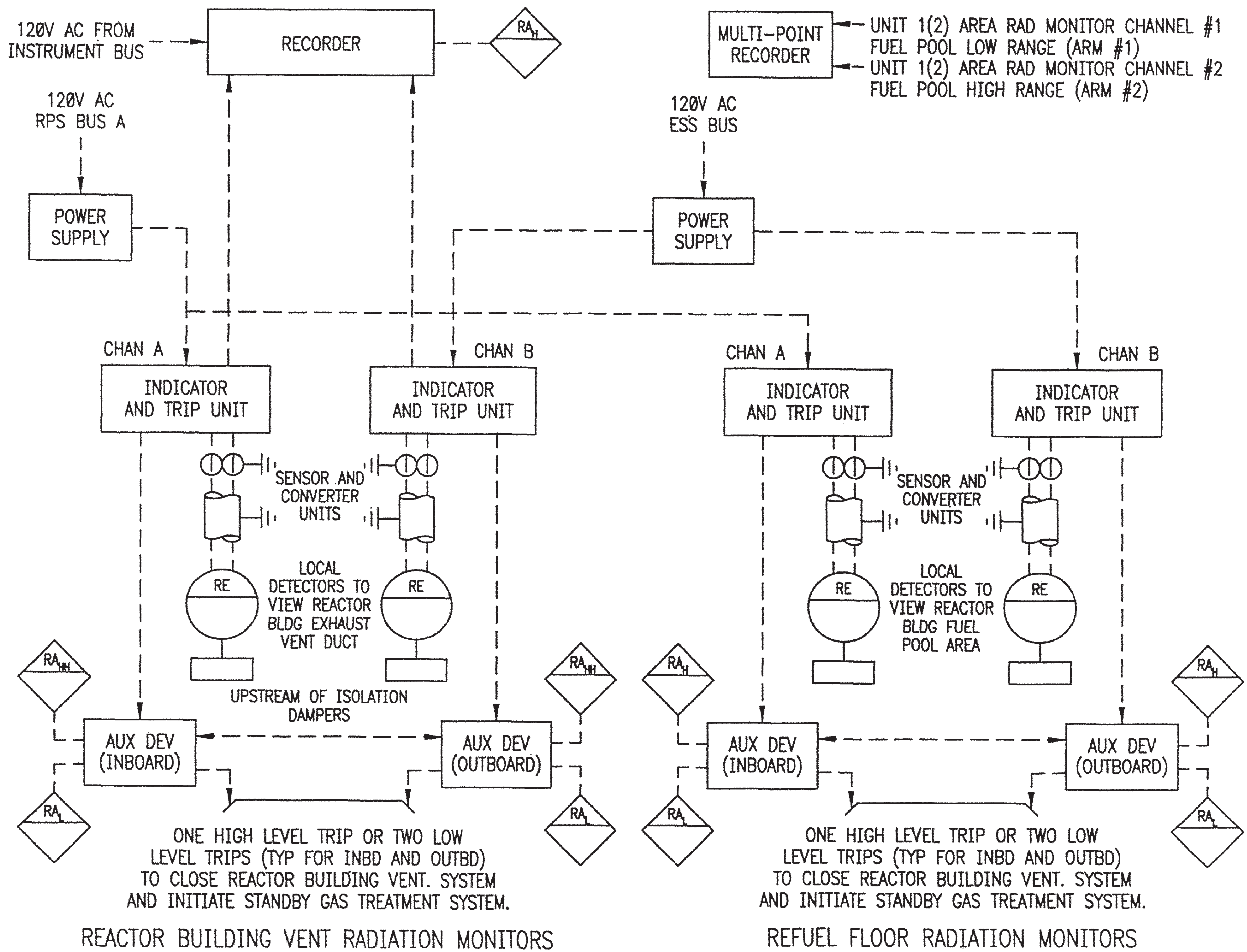
REVISION 15, OCTOBER 2019



QUAD CITIES STATION
UNITS 1 & 2

CHIMNEY GAS RADIATION MONITORING SUBSYSTEM

FIGURE 11.5-2
REVISION 9, OCTOBER 2007

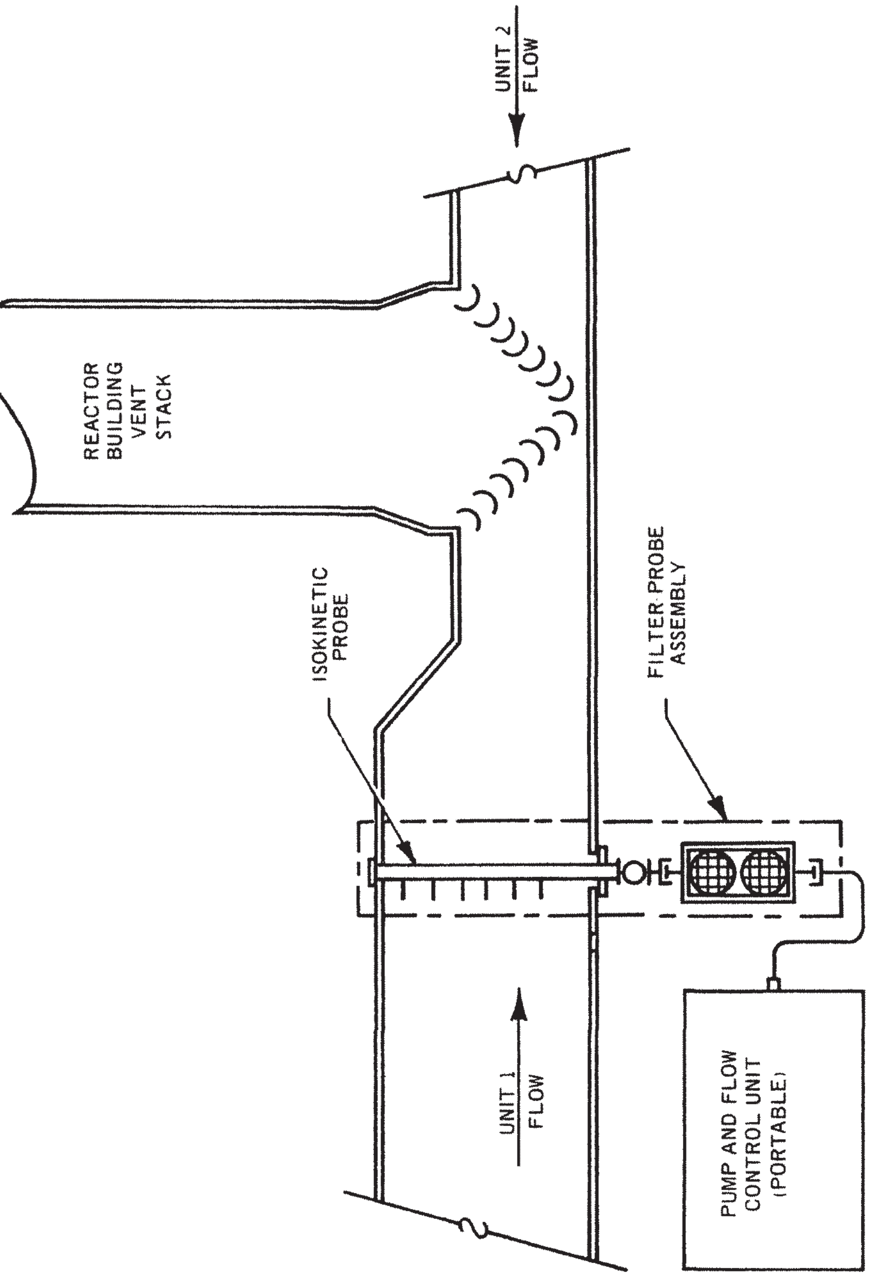


QUAD CITIES STATION
UNITS 1 & 2

REACTOR BUILDING VENTILATION RADIATION MONITORING SUBSYSTEM

FIGURE 11.5-3

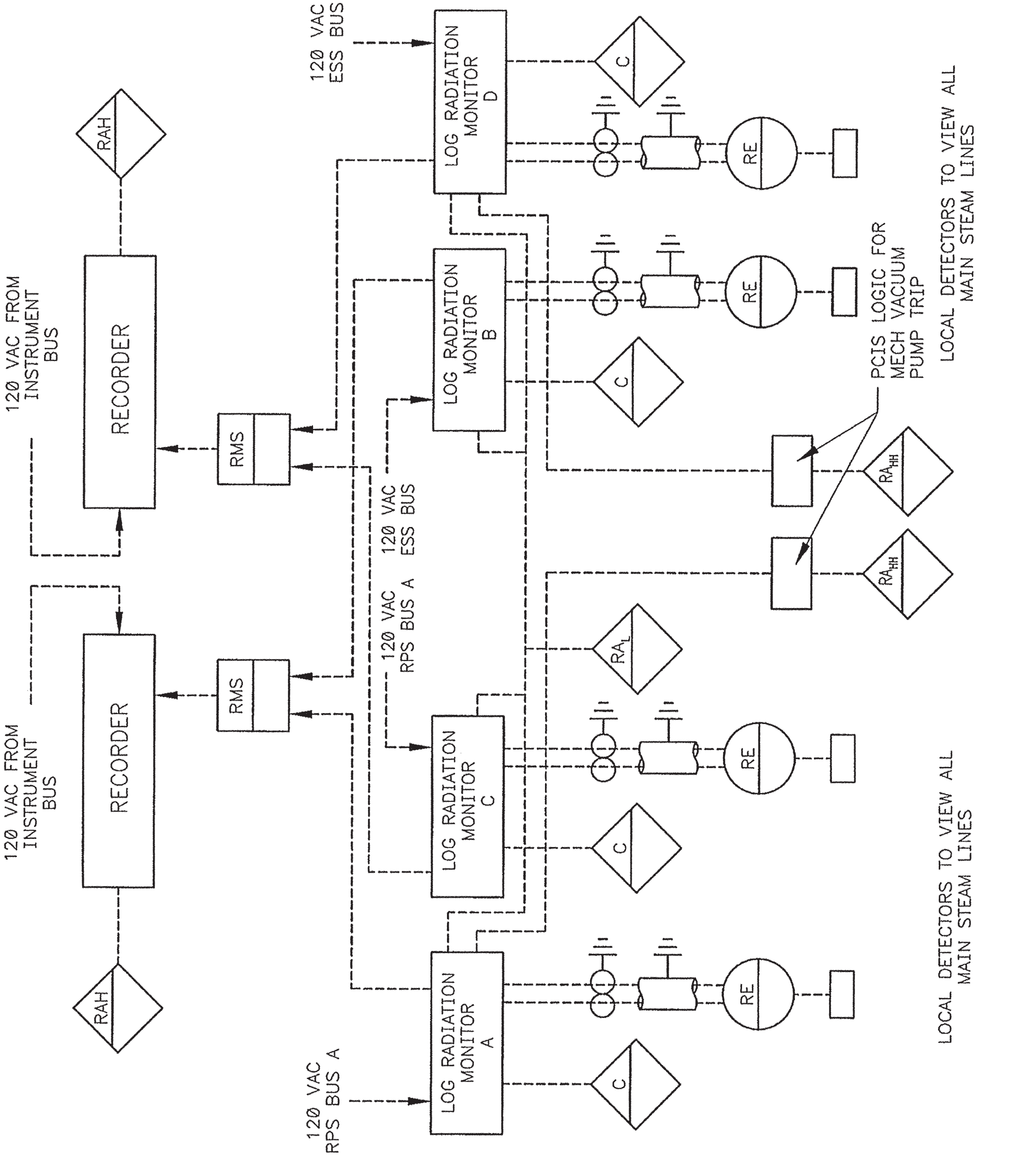
REVISION 9, OCTOBER 2007



QUAD CITIES STATION
UNITS 1 & 2

REACTOR BUILDING EXHAUST VENT
SAMPLE SYSTEM

FIGURE 11.5-4

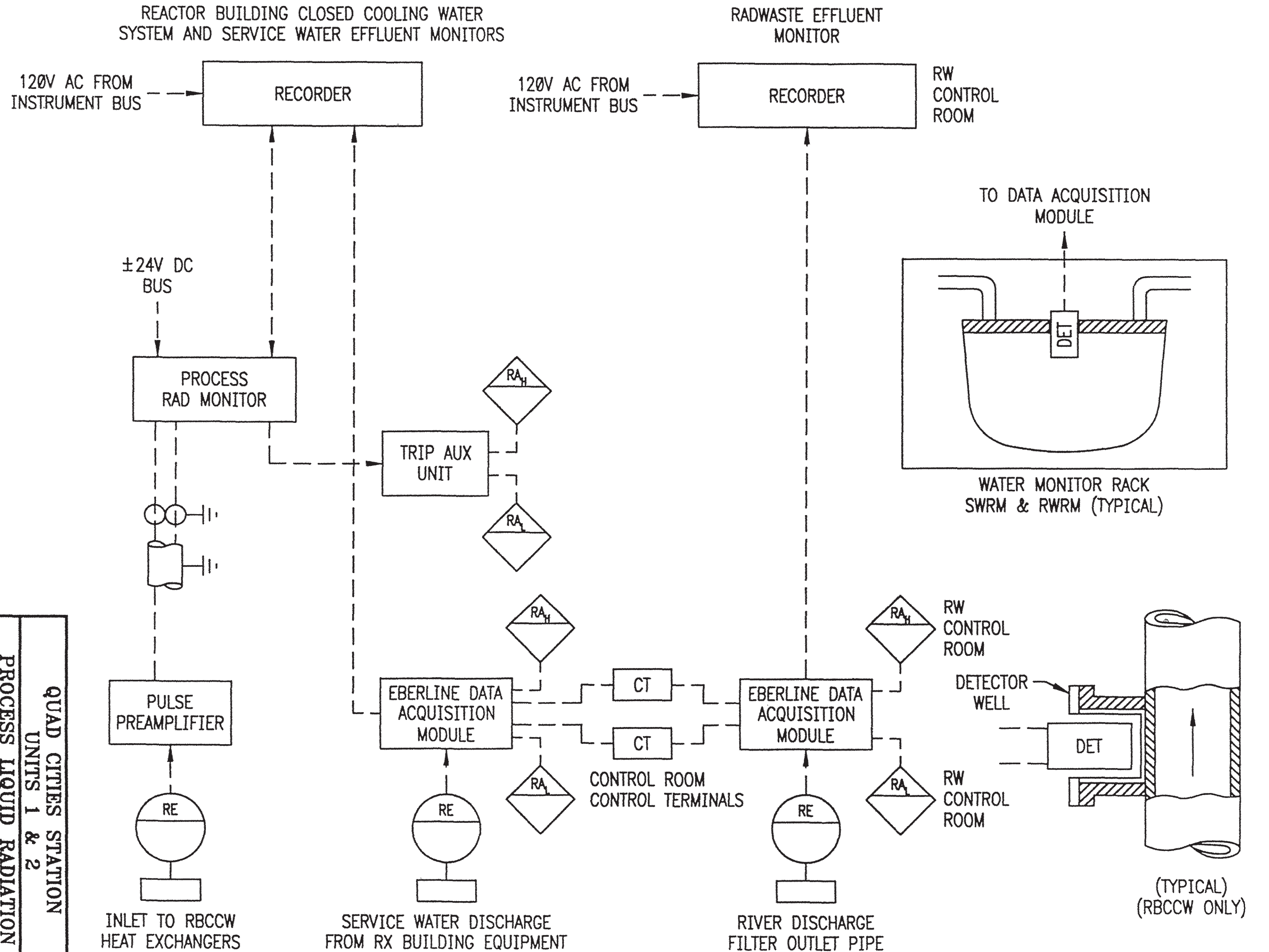


QUAD CITIES STATION UNITS 1 & 2

MAIN STEAM LINE RADIATION MONITORING SUBSYSTEM

FIGURE 11.5-5

REVISION 9. OCTOBER 2007



QUAD CITIES STATION
UNITS 1 & 2

PROCESS LIQUID RADIATION MONITORING SUBSYSTEM

FIGURE 11.5-6
REVISION 9, OCTOBER 2007