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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.

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11.0 RADIOACTIVE WASTE MANAGEMENT

The radioactive waste control systems are designed to collect, process, control, and dispose of potentially radioactive waste in a safe manner without limiting unit or station operation or availability.

Equipment, instrumentation, and operating procedures are provided to assure that the discharge of radioactive materials will not exceed the limits as set forth in 10 CFR 20 and 10 CFR 50, Appendix I.

The performance objectives of the radioactive waste control systems are as follows:

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

B. To minimize the release of radioactive material to the environment;

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 as 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 gaseous radioactive waste. The major sources of gaseous radioactive wastes (condenser air ejector effluent and steam packing exhaust system effluent) are continuously decayed and filtered during operation and monitored to ensure that the release limits of 10 CFR 20 and 10 CFR 50, Appendix I are not exceeded.

B. Liquid radioactive wastes - 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 conductivity before being returned to the plant as condensate, sent to the condenser cooling water discharge canal, or reprocessed through the radioactive waste system for further purification (see Section 11.2).

C. Solid radioactive wastes - 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 suitable containers for storage onsite or disposal offsite.

The components of the system are designed and operated in such a manner as to minimize radiation exposure of personnel and significantly reduce the radioactivity levels below those limits set forth in 10 CFR 20; 10 CFR 50, Appendix I; and the regulations of the State of Illinois.

Discharge paths for the release of radioactive materials are monitored by the following systems.

A. Off-gas radiation monitor - The off-gas monitoring system actuates an alarm in the control room in the event that the gaseous discharge from the main turbine condenser significantly exceeds the normal emission rate. The monitoring system isolates the off-gas system after a time delay in the event that the release rate limit is exceeded. As a result of the action initiated by this system, the resultant doses will be below the guidelines set forth in 10 CFR 20 and 10 CFR 50, Appendix I.

B. Chimney effluent radiation monitor - The gaseous effluent discharged to the environment via the chimney is monitored for particulate, iodine, and noble gas activity. An alarm annunciates in the control room if the release rate limit is exceeded. Appropriate action, such as power reduction, etc., will be taken upon indication of the limits being exceeded.

C. Reactor building ventilation exhaust radiation monitor - In the event of high radiation levels in the reactor building ventilation exhaust duct or on the refueling floor, the monitors isolate the secondary containment and initiate the standby gas treatment system. The activity level necessary to isolate the secondary containment equates to a calculated dose rate less than the instantaneous effluent release limit of 500 mrem/year whole body and 3000 mrem/year skin.

D. Before any batch of liquid waste is discharged to the environment from the liquid waste treatment facility, the tank is isolated so that no additional water can be added to it. The batch of liquid waste is mixed by recirculation to assure that the sample obtained is representative. After mixing, the batch of liquid waste is sampled and analyzed for gamma isotopic activity. Factors for H-3, Fe-55, Sr-89 and Sr-90 which are based on previous discharges are calculated periodically. The factors may then be used to estimate H-3, Fe-55, Sr-89 and Sr-90 concentrations if the actual value is not known. Based upon these analyses, a discharge rate for the batch is determined so that when the batch is discharged and diluted by the plant circulating water discharge, the radioactivity level in the circulating water leaving the plant site will be less than the applicable effluent concentration limit (ECL), as stated in 10 CFR 20, Appendix B, Table 2. This ensures that the level of activity at the outlet of the discharge canal will be within the NRC limit for non-occupational use. Normally, the waste is a small percentage of the ECL, and at no time will waste discharge water leaving the discharge canal and entering the river exceed average ECL values over the course of a calender year.

Normally, an offline radiation detector monitors the radioactive system discharge line that feeds the circulating water discharge canal when a discharge is made. If an abnormal radioactivity level is detected, a grab sample is automatically collected, an alarm annunciates in the radwaste control room, and the operator terminates the discharge. Compliance with 10 CFR 20, 10 CFR 50 Appendix I, and 40 CFR 141 limits is further verified through programs delineated in the Offsite Dose Calculation Manual.

E. Service water effluent monitor - The normally uncontaminated service water effluent is monitored with a gross gamma radiation detection instrument for direct radiation. An alarm in the main control room is actuated if the service water becomes significantly contaminated. Appropriate action is taken to identify and repair the affected system.

Nonradioactive liquid wastes are controlled either administratively or by locating such sources outside of controlled access areas. These wastes are discarded by conventional means to the river via the waste water treatment system or through storm drains.

Radioactive and nonradioactive liquid waste discharges are monitored as required by the United States Environmental Protection Agency (EPA) and the Illinois EPA via the station's national pollution discharge elimination system (NPDES) permit.

The systematic evaluation program (SEP) did not address any issues related to radioactive waste management. Radwaste-related SEP Topics XI-1 and XI-2 originally under consideration were determined to be ongoing Office of Nuclear Reactor Regulation (NRR) generic issues, and were deleted from the SEP.

11.1 SOURCE TERMS

The basis selected for the design capacity of liquid, gaseous, and solid radioactive waste management systems is the design basis activity concentration of 5.5 μCi/cc of corrosion and fission products present in the reactor coolant. The corrosion and fission product input quantities are reduced by 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.

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

Radioactive nuclides in the reactor coolant system consist of fission products from a fuel cladding failure, radioactive corrosion products, and other radioactive products in the coolant.

Radioactive fission products in the 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.

Certain elements present as impurities in the reactor coolant are activated upon exposure in the reactor core. The principal activation products in the reactor coolant stream are listed in Table 11.1-2.

11.1.2 Radioactive Nuclide Concentration

Estimated concentrations of the fission product and activation product (which include the corrosion product) radioactive nuclides are tabulated in Table 11.1-1 and Table 11.1-2, respectively.

11.1.3 Mathematical Model and Parameters

This section discusses the mathematical equations and parameters that were used in obtaining the source terms which were, in turn, used as a basis for sizing and designing of the radioactive waste management systems.

The design basis numbers used for Dresden Units 2 and 3 are based on the diffusion model and an off-gas rate of 200,000 μCi/s. A design basis activity concentration of 5.5 μCi/cc of activation and fission products was applied in the design of Dresden Units 2 and 3.

The reactor design and recirculation system parameters listed in Table 11.1-3 were used to arrive at the fission product and activation product concentrations.

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:

Equilibrium: $R_g \sim k_1 Y$ (1)

Recoil: $R_g \sim k_2 Y \lambda$ (2)

The nomenclature in Section 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 (median between the equilibrium and recoil mixtures) is 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:

Diffusion: $R_g \sim k_3 Y \lambda^{0.5}$ (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, λ, is midway between the values for equilibrium 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. 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 of 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 using the same equation used for noble radiogases. In a fashion similar to that used with noble radiogases, a simplified equation can be shown to describe the leakage rate of each halogen radioisotope:

$R_h = K_h Y \lambda^n$ (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, λ. 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 coolant have been estimated conservatively. Carryover of these radioisotopes from the reactor water to the steam is estimated to be <0.1%. 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 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)

λ = 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 have been estimated conservatively.

11.1.3.6 Non-Coolant 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 non-coolant activation products have been estimated conservatively. Carryover of these isotopes from the reactor water to the steam is estimated to be <0.1%.

11.1.3.7 Tritium

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

A. Activation of naturally occurring deuterium in the primary coolant;

B. Nuclear fission of UO_2 fuel; and

C. 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 reactor 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 reactor coolant is available for release in liquid or gaseous effluents. The tritium formed from the activation of deuterium in a BWR can be calculated using the equation:

$$R_{act} = \frac{\Sigma \phi V \lambda}{3.7 \times 10^4 P} \tag{5}$$

where:

R_{act} = tritium formation rate by deuterium activation (μCi/s/MWt)

Σ = macroscopic thermal neutron cross section (cm^{-1})

ϕ = thermal neutron flux [neutrons/(cm^2)(s)]

V = coolant volume in core (cm^3)

λ = 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, the empirical relationship described as the "diffusion mixture" can be used 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 as follows:

$$R_{dif} = K Y \lambda^{0.5} \tag{6}$$

where:

R_{dif} = leakage rate of tritium from fuel (μCi/s)

Y = fission yield fraction (atoms/fission)

λ = 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 lesser 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, the reactor process coolant has a common tritium concentration.

Off-gases released from the plant will contain tritium, which is present as HT resulting from reactor coolant radiolysis, as well as HTO. In addition, water vapor from the turbine gland seal steam packaging exhauster and a smaller amount present in ventilation air (due to process steam leaks or evaporation from the sumps, tanks, 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 radiolytic 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 reactor process coolant. Reducing the amount of liquid effluent discharged will also result in a higher reactor process coolant equilibrium tritium concentration.

Essentially, all tritium in the reactor coolant will eventually be released to the environment, either as water vapor and gas to the atmosphere or as liquid effluent in the plant discharge. Reduction due to radioactive decay is negligible because of the 12-year half-life of tritium.

The USPHS study at Dresden Unit 1 estimated that approximately 90% of the total tritium released was observed in the liquid effluent, with the remaining 10% leaving in the 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. It is presently estimated that 45% of the total tritium released is in the liquid effluent discharged with approximately 55% leaving in the gaseous effluent.[1-10]

11.1.4 Fuel Fission Product Inventory

Information which is used in establishing fission product source terms for accident analysis is 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 are insignificant compared to off-gas releases from operating BWR plants.

Leakage of fluids from the reactor coolant process systems 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 plate out 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 radioactivity sources for the ventilation system are from the following systems:

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.

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 three site release points for gaseous effluent: the 310-foot chimney, the reactor building ventilation stack, and Unit 1 chemical cleaning building ventilation stack. There are no release points for gaseous effluent that are not a part of the radioactive waste management system.[11]

There is one site release point for liquid Potentially Radioactive and Radioactive effluent, which is the condenser cooling water discharge canal. There are no release points for liquid Potentially Radioactive and Radioactive effluent that are not a part of the radioactive waste management system. Radiation monitors are located for each line that discharges to the condenser cooling water discharge canal: Unit 2/3 liquid radwaste discharge line; Unit 2 service cooling water discharge line; and Unit 3 service cooling water discharge line.[11]

Radiation monitors are designed to continuously monitor the gaseous and liquid discharge streams and alert the control room operator in case the effluent stream exceeds the predetermined level of radioactivity. Requirements for continuing liquid discharge without the radiation monitors are specified in the ODCM.

The estimated quantity of tritium in the effluent stream discharged to the environment is discussed in subsection 11.1.3.7. and Section 11.2.

There are several areas of the Turbine Building that have floor drains that are routed to the non-radioactive wastewater processing systems. These areas include the Diesel Generator Rooms, Stator Cooling Rooms, Stator Cooling Area, EHC Skids, Feedwater Regulating Valve Area and the Trackways. Periodic sampling of the non-radioactive wastewater processing system effluent will detect intrusion of radioactive liquids from these floor drains.

11.1.9 References

1. Letter from N.H. Kalivianakis (CECo) to J.G. Keppler (NRC), dated June 24, 1986, transmitting the Dresden Station Annual Environmental Radiological and Meteorological Operating Report for 1985.

2. Dresden Nuclear Power Station Radioactive Waste and Environment Monitoring Annual Report for 1986, dated March 1987.

3. Letter from E.D. Eenigenburg (CECo) to A.B. Davis (NRC), dated March 18, 1989, transmitting Dresden Nuclear Power Station Radioactive Waste and Environmental Monitoring Annual Report for 1987, dated March 1988.

4. Letter from E.D. Eenigenburg (CECo) to U.S. Nuclear Regulatory Commission, dated February 24, 1989, transmitting Dresden Nuclear Power Station Radioactive Waste and Environmental Monitoring Annual Report for 1988, dated March 1989.

5. Letter from E.D. Eenigenburg (CECo) to A.B. Davis (NRC), dated March 19, 1990, transmitting Dresden Nuclear Power Station Radioactive Waste and Environmental Monitoring Annual Report for 1989, dated March 1990.

6. Letter from E.D. Eenigenburg (CECo) to A.B. Davis (NRC), dated March 18, 1991, transmitting Dresden Station Annual Radiological Environmental Operating Report for 1990, dated March 1991.

7. Letter from C.W. Schroeder (CECo) to A.B. Davis (NRC) dated February 20, 1992, transmitting Dresden Station Semiannual Radiological Effluent Report for July through December, 1991.

8. Dresden Station Semiannual Radiological Effluent Report for January through June, 1991.

9. Letter from E.D. Eenigenburg (CECo) to A.B. Davis (NRC), dated February 28, 1991, transmitting Semiannual Radioactivity Report for July through December 1990 for Dresden Nuclear Power Station.

10. Dresden Station Semiannual Radioactive Effluent Report for January through June, 1990.

11. Dresden Nuclear Power Station Offsite Dose Calculation Manual, Revision O.A, April 1991.

Table 11.1-1*

REACTOR COOLANT FISSION PRODUCTS
(Based on 7 x 10^5 μCi/s - Chimney Release Rate)

Radioisotopes	Half-Life	Concentration (μCi/cc)
I-138	5.9 s	8.1 x 10^{-2}
Br-88	16 s	1.1 x 10^{-1}
I-137	22 s	2.2 x 10^{-1}
Br-87	56 s	1.9 x 10^{-1}
I-136	86 s	2.7 x 10^{-1}
Br-85	3 min	1.8 x 10^{-1}
Br-84	32 min	2.6 x 10^{-1}
I-134	52 min	1.8 x 10^{0}
Br-83	2.3 hr	9.4 x 10^{-2}
I-132	2.3 hr	8.6 x 10^{-1}
I-135	6.7 hr	7.9 x 10^{-1}
I-133	21 hr	5.2 x 10^{-1}
I-131	8.05 days	7.9 x 10^{-2}
Tc-99	6.04 hr	8.4 x 10^{-2}
TOTAL		5.5

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

Table 11.1-2*

CONCENTRATIONS OF ACTIVATION PRODUCTS IN REACTOR COOLANT

Radioisotope	Half-Life	Concentration (µCi/cc)	
		Soluble	Insoluble
F-18	1.8 hr	4×10^{-3}	
Mn-56	2.56 hr	2×10^{-3}	5×10^{-2}
Ni-65	2.56 hr	5×10^{-5}	2×10^{-4}
Zn-69m	13.8 hr	1×10^{-5}	2×10^{-5}
Na-24	15 hr	2×10^{-3}	
W-187	24 hr	1×10^{-5}	3×10^{-3}
Co-58	70 days	4×10^{-4}	5×10^{-3}
Co-60	5 yr	4×10^{-5}	5×10^{-4}
Fe-59	45 days	2×10^{-7}	8×10^{-5}
P-32	14 days	2×10^{-5}	
Cr-51	27 days	2×10^{-4}	3×10^{-4}
Ag-110m	270 days	6×10^{-5}	3×10^{-6}
Mn-54	300 days	2×10^{-6}	4×10^{-5}
Zn-65	245 days	1×10^{-6}	1×10^{-6}
Total		0.01	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

REACTOR AND RECIRCULATION SYSTEM PARAMETERS USED FOR ORIGINAL FISSION PRODUCT ESTIMATES

Parameter	Reactor Nominal Design		Turbine Maximum Design	
Reactor power (MWt)	2255		2467	
Core - Active fuel length (in.)	144		144	
Equivalent diameter (in.)	182.2		186.2	
Circumscribed diameter (in.)	189.7		198.6	
Number of fuel assemblies	724		756	
Overall average core power density* (w/cc)	36.65		38.4	
Total coolant flowrate through the core (lb/hr)	9.8×10^7		9.8×10^7	
Primary steam flowrate (lb/hr)	8.62×10^6		9.43×10^6	
Core power peaking factors:	Max. at Core 6	At Core Boundary	Max. at Core 6	At Core Boundary
$\left.\frac{P_{max}}{P_{ave}}\right]_Z$ (axial)	1.57	0.7	1.57	0.7
$\left.\frac{P_{max}}{P_{ave}}\right]_R$ (radial)	1.5	0.7	1.5	0.7

Core Volume Fractions:

Material	Density (g/cc)	Volume Fraction	Volume Fraction
UO_2	10.4	0.254	0.254
Zr	6.4	0.130	0.130
H_2O	1.0	0.296	0.296
Void	0	0.320	0.320

Table 11.1-3 (Continued)

REACTOR AND RECIRCULATION SYSTEM PARAMETERS
USED FOR ORIGINAL FISSION PRODUCT ESTIMATES

Parameter	Reactor Nominal Design	Turbine Maximum Design
Reactor operating pressure (psia)	1015	1015
Average water density between core and vessel (g/cc)	0.73	0.73
Average water density below core (g/cc)	0.74	0.74
Average water-steam density above core		
In the plenum region (g/cc)	0.27	
Above the plenum (homogenized) (g/cc)	0.52	
Average steam density (g/cc)	0.036	0.036
Vessel inside radius (in.)	125.5	125.5
Vessel wall thickness (base material) (in.)	6⅛ min	6⅛ min
Vessel clad thickness (stainless steel) (in.)	⅛	⅛
Core shroud thickness (in.)	2	2
Nitrogen-16 activity of steam leaving the vessel (average gamma energy of 6.2 MEV/γ)	95 Ci/s	109 Ci/s

* This parameter is a function of a rated core thermal power (2255 or 2467 MWt), active fuel length (144 inches), number of fuel assemblies (724) and fuel assembly pitch (6 inches). This number does not represent the current licensed rated core thermal power nor active fuel length currently in use; it is only useful in the presented context.

11.2 LIQUID WASTE MANAGEMENT SYSTEMS

11.2.1 Design Objectives

The principal objectives for design and operation of the liquid waste management system include the following:

A. All wastes are sampled as batches to insure they meet established criteria and requirements prior to discharge from the system, either for reuse in the station as condensate or for discharge to the discharge canal.

B. Waste discharges to the discharge canal are at a rate such that the unidentified isotope mixture concentration in the canal does not exceed 10^{-7} μCi/cc including background. (This is a State of Illinois requirement that started with the Dresden Unit 1 permit for waste discharged to the river.)

C. The system is designed to handle one unit in operation and the other unit in startup. (Startup is defined as initial and any subsequent startups such as after refueling.)

D. Maximum quantity of liquid wastes include the additional wastes expected from startup, maintenance activities, unusual circumstances (e.g., condenser tube leaks), and radioactivity content due to design basis fuel leaks. Near full-time operation of the system is expected during such high-volume periods.

E. Table 11.2-1, extracted from design and process diagram data, shows the "normal" and maximum expected (design) throughputs for Unit 2 and Unit 3 operation. During a startup period volumetric throughputs are expected to be above normal and may approach maximum values. Performance of systems which produce radioactive wastes or which affect the discharge capability of radioactive wastes are also reflected in waste system performance.

F. System design anticipates that operation of the radwaste system is planned on a regular basis, especially the planning of transfers to insure against overflows and system overloading. Also anticipated are appropriate station surveillance and maintenance activities to determine and correct abnormal radwaste inputs.

The design provides for dewatering and solidification of liquid wastes, sludges, and resins to facilitate storage and disposal offsite as solid wastes. Radioactive solid waste is addressed in Section 11.4.

11.2.2 System Description

Radioactive materials are removed from the liquid waste streams by various mechanisms before the waste streams are discharged to condensate storage or are released to the discharge canal. Evaporators, demineralizers, filters or a portable waste treatment system may be utilized to remove contaminants.

Liquid radwaste is divided into three categories:

A. Low conductivity;

B. Moderate conductivity; or

C. High conductivity.

The four systems utilized to process the liquid radwaste are, respectively:

A. Equipment drain system;

B. Floor drain system ; or

C. Maximum recycle system (which is part of the floor drain system).

D. Portable waste treatment system;

Filter sludges and spent resins are treated as solid radwaste - decanted, placed in appropriate containers, and dewatered or solidified. This is described in Section 11.4.

Overall control of the radwaste system is exercised from a local control room situated in the radwaste building. A main panel in this room contains the instruments, controls, and alarms for the operation of the system. Various radwaste system alarm signals are received in the radwaste control room.

Table 11.2-2 shows the locations and capacities of the radwaste tanks. The total allowed activity in the six tanks in the tank farm outside the radwaste building is 3 curies; the allowed activity in any one tank is 0.7 curies.

In accordance with the Systematic Evaluation Program (SEP) Topic III-4.A, the liquid radioactive waste systems and tanks in the radwaste building are adequately protected from tornado generated missiles. See Section 3.3.2.3.2 for additional discussion.

Seismic damage was evaluated for the liquid radwaste system. The tanks of radioactive waste of concern were those at grade outside the radwaste building. Assuming failure of all outside tanks and structures, the evaluation showed that the Effluent Concentration Limits (ECL) were not exceeded. Although the activities resulting from a tank failure condition that are released to the river indicate the applicability of 10 CFR 100 limits, the actual expected concentrations are sufficiently low to allow 10 CFR 20 limits to be applied (see Section 15.7.2 for details of accident considerations).

The general arrangement drawings referenced in Section 1.2 show the general building layout of various pieces of equipment comprising the liquid radwaste system. Drawings M-39 and M-369 show the reactor building equipment drains and the drywell equipment drains which are the sources of liquid radwaste collected for normal processing in the equipment drain system. Drawings M-39 and M-369 also show the floor drain system in the reactor building that collects liquid radwaste for normal processing in the floor drain system. The drywell drain systems shown in these drawings are normally routed to the equipment drain system. Drawings M-40 and M-370 show the turbine building equipment drain system which collects liquid radwaste from the turbine building area for treatment in the equipment drain system. Drawings M-40 and M-370 also show the turbine building floor drain system which collects liquid radwaste for normal processing in the floor drain system. Drawing M-45, Sheets 1, 2, and 3 show the equipment drain liquid radwaste process piping, tanks, pumps, and instrumentation. Drawing M-44 shows the floor drain liquid radwaste processing system piping, pumps, tanks, and instrumentation. Drawing M-47, Sheet 1 shows the piping and instrumentation for the waste neutralizer tanks and pumps. Drawings M-720, Sheets 1 and 2, M-721, M-722, M-723, and M-724 show the process piping equipment and instrumentation for the maximum recycle system. These drawings also show the interfaces between this system and other liquid radwaste systems. The liquid radwaste process sampling is shown in Drawings M-178A and M-720A. Drawing M-3478 shows the liquid radwaste radiation monitor. Drawings M-3486 and M-3496 show the liquid radiation monitors for the service water systems. Figure 11.2-21 (Offsite Dose Calculation Manual [ODCM] Figure 2-1) shows a simplified liquid radwaste processing diagram and also shows the liquid radwaste discharge points leading to the river.

Cross-connection of the equipment and floor drain systems allows processing of wastes in various modes depending upon the water quality and/or equipment availability.

The liquid radwaste system is piped such that transfer of liquid wastes can be made directly from Dresden Unit 1 to Dresden Unit 2/3 radwaste system. Dresden Unit 1 is in a safe storage (SAFSTOR) condition, although some portions of the Unit 1 radioactive waste system remain operable.

11.2.2.1 Process Description

11.2.2.1.1 Equipment Drain System

Input for the equipment drain system, also known as the waste collector system, includes seal leakage from pump and valve glands which is collected in equipment drain sumps in the drywells, reactor building, and turbine building (Drawings M-39, M-40, M-369,and M-370). The wastes handled by this system typically have a low conductivity (on the order of 10 µmho or less) and a low-solids content, but may have a low or high activity.

Where appropriate, sources of waste water are provided with heat exchangers and/or multiple sumps and sump pumps. The drywell equipment drain sump, for example, has a heat exchanger which operates intermittently, and one sump, which has two sump pumps. The drywell floor drain sump also has two sump pumps. The drywell floor drain sump is normally pumped to the radwaste waste collector tank. During a refueling outage it may be aligned to the floor drain collection tank. Also during normal operation the drywell floor drain sump may be aligned to the floor drain collection tank, depending on the water conductivity. The reactor building equipment drain tanks, drywell equipment drain sump, and the drywell floor drain sump are prevented from automatic pumping of the sump and/or tank contents to the radwaste waste collector tank during an accident. This is discussed further in Section 11.2.3.4.

In the processing (Drawing M-45, Sheets 1, 2, and 3) of the liquid radwaste from the equipment drain system, the normal process path for the low-conductivity wastes are as follows. These low-conductivity wastes are collected in the waste collector tank. The waste is pumped through a filter and then to the demineralizer unit. The normal process flow is to the waste sample tanks where the processed water is sampled. If the processed liquid radwaste in the waste sample tank meets certain specifications (typical criteria are listed in Table 11.2-3), the processed water is pumped to the condensate storage tanks. Dependent upon the station water inventory, the water may be discharged to the river through the discharge canal. In the flow path for discharge to the river, the water from the waste sample tanks or floor drain sample tanks can be transferred to the waste surge tank for discharge to the river. Water processed into the floor drain sample tanks can be discharged directly to the river if required. Other processing flow paths exist in this processing system and are addressed in Section 11.2.2.4.

11.2.2.1.2 Floor Drain System

Input for the floor drain system (shown in Drawing M-44) includes water from the floor drain sumps in the reactor buildings, turbine building, and radwaste building, the maximum recycle drains and vents, and the heating boiler (Drawings M-39, M-40, M-369, and M-370). The wastes handled by this system are those having a higher conductivity than the water in the equipment drain system.

The liquid radwaste collected in the floor drain collector tank is transferred into the waste neutralizer tanks, which are sampled and batched to the maximum recycle neutralizer tanks, and then processed through the maximum recycle system to remove as much radioactive contamination as possible. The liquid from the floor drain collector tank can also be transferred to the floor drain surge tank or to the floor drain neutralizer tanks for processing through the maximum recycle system.

Drywell floor drains are routed to the equipment drain system because that waste is expected to have a low conductivity (leakage from reactor coolant system), and this routing removes a potential source of activity from the floor drain system.

Sample sink drains are segregated to minimize activity input to the floor drain system. The sample system is addressed in Subsection 11.2.2.6.

Curbs are placed around certain equipment drain sumps to prevent activity and high conductivity from entering these sumps in the event of floor drain sump overflow.

Input to the floor drain system collected in the floor drain collector tank (in the radwaste building basement) are those liquid wastes having a higher conductivity than the wastes in the equipment drain system. These liquid wastes are processed through the maximum recycle system.

11.2.2.1.3 Maximum Recycle System

The maximum recycle system (Drawings M-720, Sheets 1 and 2, M-721, M-722, M-723, and M-724) consists of two identical trains of components. Normally one train is used at a time, and the other is in standby. The maximum recycle trains can be used in parallel. The maximum recycle demineralizers can be operated in series as well as in parallel.

Waste water to be processed by the maximum recycle system is sampled and the pH adjusted if necessary. The solids or sludge, commonly called concentrated waste, are solidified or further processed as solid radwaste by contract services (Section 11.4). The distillate is demineralized if required and then sent to a sample tank. After being sent to a floor drain sample tank or waste sample tank for sampling, the water may be reused as condensate, or it may be discharged to the river by way of the floor drain sample tanks or waste surge tank, processed through the waste filter and demineralizer, or recycled for further distillation/demineralization.

Heat for the maximum recycle concentrators is supplied by steam from one of the three closed loop reboilers. The maximum recycle reboilers are shell and tube heat exchangers which are heated by steam from the nuclear steam supply system. The radwaste reboiler is also a shell and tube heat exchanger which is heated by steam from the auxiliary heating steam boiler.

11.2.2.1.4 Waste Concentrator System

The waste concentrator system, excluding the closed loop radioactive waste reboiler system, has been partly removed and partly abandoned in-place. Although this system was a part of the initial licensing basis of the plant, liquid radwaste treatment methods have been improved such that this system is no longer efficient in treating the waste for discharge and/or solidification for shipment for burial.

11.2.2.1.5. Portable Waste Treatment System

System taps are provided to allow connection to portable waste treatment systems that are capable of processing liquid radwaste. A portable waste treatment system is any system that enables efficient processing of liquid radwaste. Portable waste treatment systems can either be connected to plant installed radwaste equipment to augment processing capabilities or may be self contained, skid mounted equipment, capable of all liquid radwaste processing needs and requirements.

Processed liquid radwaste is discharged to radwaste system tanks or from portable waste treatment system tanks to the radwaste system discharge line to the cooling water discharge canal.

11.2.2.2 Description of Major Components

The components addressed in this subsection comprise the liquid radioactive waste systems described in Subsection 11.2.2.1.

11.2.2.2.1 Waste Collector Tank

The waste collector tank (2001-461) provides a storage volume of 33,000 gallons for liquid radioactive waste. The waste collector tank is a closed tank. Input to the tank comes from the sources (shown in Table 11.2-4) which are considered low-conductivity sources.

11.2.2.2.2 Waste Collector Pumps

The waste collector pumps (2005A and 2005B) perform the following functions:

A. Mix the contents of the waste collector tank;

B. Transfer tank contents to the floor drain collector tank;

C. Transfer tank contents to the waste sample tanks by way of the waste collector filters and demineralizer;

D. Provide redundant pump capability in the event that one of the pumps is not available;

E. Blowdown to the waste filter sludge tank;

F. Transfer to the maximum recycle system;

G. Supply the eductor driving force to decant the waste filter sludge tank ("A" cleanup filter sludge storage tank); and

H. Supply the eductor driving force to decant the resin cleaner sludge tanks ("B" cleanup filter sludge storage tank).

The pumps are designed for a 200-gal/min flow. The design temperature is 200°F and the design pressure is 150 psig. The pumps are provided with a seal water system to minimize maintenance requirements.

11.2.2.2.3 Waste Collector Filters

The waste collector filters (2043-1A and 2043-1B) remove fine particulates from the liquid waste. The filters use a filter aid (or precoat material) which, along with the filter sludge, is backwashed to the filter sludge tank where the sludge and filter media are processed further as solid waste. The filters have 400 square feet of filtering area each. The flow through these filters is normally 200 gal/min.

11.2.2.2.4 Waste Demineralizer

The waste demineralizer (2007) is used to purify the liquid radioactive waste to the water specifications for the condensate storage tanks. The demineralizer vessel typically contains a bed of mixed resin (H-OH) polishing type. When the resins are depleted they are normally transferred to the spent resin tank for disposal as radioactive waste. The demineralizer normal flowrate is 200 gal/min.

11.2.2.2.5 Waste Surge Tank

The waste surge tank (2001-463) is used for liquid radioactive waste discharge from the station to the river, or the water may be reprocessed. Excess low-conductivity water can also be stored in the water surge tank temporarily until it can be processed through the waste filter and waste demineralizer to the waste sample tanks. Normally the waste surge tank contents are processed through filters and demineralizers to the waste sample tanks and to storage in the condensate storage tanks however, if the total organic carbon content is high, or condensate storage tank is not available, the contents may be discharged to the river or reprocessed. Inputs to waste surge tank are from the following sources:

A. The floor drain sample tanks and

B. The waste sample tanks.

The 77,000-gallon capacity tank is normally the single source for discharge of liquid radioactive waste, however floor drain sample tanks or portable waste treatment system tanks can also be discharged, if required.

11.2.2.2.6 Waste Surge Tank Pump

The waste surge tank pump (2011) transfers the liquid waste either to the waste collection system for further processing or to the discharge canal to the river. The pump can also transfer the waste surge tank contents to the Unit 1 radwaste storage, to the "B" waste neutralizer tank, and to the maximum recycle system. The waste surge tank pump is also used to recirculate the waste surge tank contents for mixing. The pump capacity is 400 gal/min.

11.2.2.2.7 Waste Sample Tanks

The waste sample tanks (2001-468A, 2001-468B, and 2001-468C) collect the processed liquid radioactive waste from either the waste collector system or the floor drain system. The three tanks have a capacity of 33,000 gallons each. The redundancy allows one tank to be isolated for sampling and pumping of the liquid waste while the other tanks are available to receive processed liquid effluent. The waste sample tank contents are normally transferred to the condensate storage tanks. The tank contents can be transferred to the floor drain surge tank if required.

11.2.2.2.8 Waste Sample Tank Pumps

The waste sample tank pumps (2009-A, 2009-B, 2009-C) recirculate the tank contents for mixing prior to sampling and then transfer the processed liquid contents of the waste sample tanks to any of the following places:

A. The contaminated condensate storage tanks;

B. The waste collector tank;

C. The waste surge tank; and

D. The floor drain surge tank.

The design capacity for each pump is 400 gal/min.

11.2.2.2.9 Floor Drain Collector Tank

The floor drain collector tank (2001-459) provides a storage volume of 22,000 gallons for liquid radwaste. The floor drain collector tank is a closed tank. The input sources come from the areas shown in Table 11.2-5.

11.2.2.2.10 Floor Drain Collector Tank Pumps

The floor drain collector tank pumps (2/3-2013A and 2/3-2013B) provide the following functions:

A. Mix the contents of the floor drain collector tank by recirculation;

B. Transfer tank contents to the floor drain surge tank;

C. Transfer tank contents to the maximum recycle floor drain neutralizer tanks;

D. Transfer tank contents to waste neutralizer tanks A or B;

E. Transfer tank contents to "A" waste collector filter;

F. Transfer blowdown tank contents to floor drain filter sludge tank;

G. Supply the eductor driving force to decant the waste filter sludge tank ("A" cleanup filter sludge storage tank);

H. Supply the eductor driving force to decant the resin cleaner sludge tank ("B" cleanup filter sludge storage tank); and

I. Supply the eductor driving force to decant the floor drain filter sludge tank (filter sludge storage tank).

The pumps are designed for a 400-gal/min flowrate with a design temperature of 200°F and a design pressure of 150 psig. The pumps are provided with a seal water system to minimize maintenance requirements.

11.2.2.2.11 Floor Drain Filter

The floor drain filter was provided to remove fine particulate from the liquid radwaste stream, however this filter is no longer used. The equipment is bypassed.

11.2.2.2.12 Floor Drain Sample Tanks

The floor drain sample tanks (2001-484A and 2001-484B) collect the processed liquid radioactive waste from the maximum recycle system. The floor drain sample tanks have a capacity of 22,000 gallons each. The redundancy allows one tank to be isolated for sampling and pumping of the liquid waste while the other tank is available to receive processed liquid effluent. The contents of these tanks may be transferred to the contaminated condensate storage tank, reprocessed for additional cleanup, or discharged to the river as needed.

11.2.2.2.13 Floor Drain Sample Tank Pumps

The floor drain sample tank pumps (2016A and 2016B) recirculate the tank contents for mixing prior to sampling and then transfer the processed liquid contents from the floor drain sample tanks to any of the following places:

A. The maximum recycle floor drain neutralizer tanks;

B. The "B" waste neutralizer tank;

C. The waste collector tank;

D. The waste surge tank;

E. The floor drain surge tank;

F. The contaminated condensate storage tanks; and

11.2.2.2.14 Waste Neutralizer Tanks

The waste neutralizer tanks (2001-473A and 2001-473B) provide a 16,500-gallon capacity each for the storage, sampling, and processing of floor drain liquid wastes. Input to the waste neutralizer tanks are from the following sources:

A. Floor drain sample tanks;

B. Cask washdown;

C. Detergent drain from main turbine floor decontamination pit;

D. Floor drain collector tank/waste neutralizer tank A header;

E. Waste demineralizer area;

F. Unit 2 mechanical vacuum pump;

G. Unit 3 mechanical vacuum pump;

H. Drain from the 613' decontamination pit; and

I. Unit 1 radioactive waste system.

J. SBGT loop seal drains.

K. Condensate filtration system vent relief piping.

The redundancy allows one tank to be isolated for sampling and pumping of the neutralized liquid waste while the other tank is available to receive high- conductivity waste for neutralization.

11.2.2.2.15 Waste Neutralizer Pumps

The waste neutralizer pumps (2019A and 2019B) provide the following functions:

A. Recirculation of the tank liquid contents for mixing;

B. Transfer tank contents between the waste neutralizer tanks;

C. Transfer the tank contents to the maximum recycle floor drain neutralizer tanks;

D. Transfer the tank contents through the floor drain filter to the floor drain sample tanks;

E. Transfer the tank contents to the floor drain surge tank; and

F. Provide redundant pump capability in the event that one of the pumps is not available.

The design capacity of these waste neutralizer pumps is 400 gal/min with a design temperature of 200°F and a design pressure of 150 psig. These pumps are provided with seal water to minimize maintenance and replacement of these pumps seals due to wear from the chemical solutions.

11.2.2.2.16 Floor Drain Surge Tank

The floor drain surge tank (2/3-2012-359) provides the necessary surge volume (200,000 gallons) for the floor drain system. The tank is located outside the Unit 3 turbine building near the southwest corner. The tank bottom is sloped to reduce sludge buildup. The floor drain surge tank is considered a Class I Structure and is, therefore, not considered an above-ground tank for the purpose of the curies content requirements of the Technical Specifications. The following sources provide input to the floor drain surge tank:

A. The floor drain collector tank pumps;

B. The waste neutralizer pumps;

C. The emergency reject from the maximum recycle floor drain neutralizer tanks;

D. The floor drain demineralizers (recycled);

E. The waste sample tanks (recycled); and

F. The floor drain sample tanks (recycled).

11.2.2.2.17 Floor Drain Surge Tank Transfer Pumps

The floor drain surge tank transfer pumps (2/3-2012-357A and 2/3-2012-357B) provide the following functions:

A. Recirculate the floor drain surge tank contents for mixing, and

B. Transfer the tank contents to the maximum recycle neutralization tanks (also known as the floor drain neutralizer tanks).

The floor drain surge tank transfer pumps are each designed for 400-gal/min flow capacity. The pumps are designed for a temperature of 200°F and a pressure of 150 psig.

11.2.2.2.18 Floor Drain Neutralizer Tanks

The floor drain neutralizer tanks (2/3-2012-358A and 2/3-2012-358B) provide a storage volume for liquid radioactive waste and a mixing volume for chemical addition for neutralization. The capacity of each tank is 22,000 gallons.

The following sources are inputs to the floor drain neutralization tanks:

A. The floor drain surge tank;

B. The floor drain collector tank;

C. The waste neutralizer tanks A and B;

D. The floor drain demineralizers (recycled);

E. The waste surge tank; and

F. The waste collector tank.

11.2.2.2.19 DELETED

11.2.2.2.20 DELETED

11.2.2.2.21 Antifoam Addition Tank

The antifoam addition tank (2/3-2012-366) chemical solution is added to minimize the foaming and sudsing from detergents in the floor drain wastes. Antifoam chemical addition also minimizes liquid carryover from sudsing in the concentrator. The chemicals are gravity fed to the mixing pump suction line of the floor drain neutralizer tanks. Antifoam may also be added by using the vapor head spray and chemical feeder (2/3-3323) directly into the concentrator vapor heads.

11.2.2.2.22 Floor Drain Neutralizer Tank Mixing Pumps

The floor drain neutralizer tank mixing pumps (2/3-2012-401A and 2/3-2012-401B) are used to recirculate the liquid contents of the floor drain neutralizer tank as a means of mixing. These pumps can also transfer the waste solution from one floor drain neutralizer tank to the other floor drain neutralizer tank; they can also transfer the waste solution to the floor drain surge tank. The pumps have a design flow capacity of 200 gal/min each. The pumps also provide sample flow for grab sampling and mixing for chemical additions and for antifoam additions made to the tank contents.

11.2.2.2.23 Maximum Recycle Concentrator Feed Pumps

The maximum recycle concentrator feed pumps (2/3-2012-403A and 2/3-2012-403B) normally transfer the liquid waste to the maximum recycle concentrator vapor head. The pumps may also be used to recirculate the floor drain neutralizer tank contents. The pumps may transfer the floor drain neutralizer tank contents to the distillate tank without transfer to the maximum recycle concentrator. The maximum recycle concentrator feed pumps can transfer the floor drain neutralizer tank contents to either maximum recycle concentrator train.

11.2.2.2.24 Maximum Recycle Concentrator Vapor Heads

The maximum recycle concentrator vapor heads (2/3-2012-412A and 2/3-2012-412B) receive the radioactive liquid waste pumped from the floor drain neutralizer tanks. The maximum recycle concentrator is a stainless steel vessel which operates at atmospheric pressure. The vapor head which has a demisting pad contains the boiling liquid waste. The vapor exists the vessel at the top after passing through the demisting pad. The liquid is recirculated by a pump through a heat exchanger and returned to the vapor head vessel.

11.2.2.2.25 Maximum Recycle Concentrator Recirculation Pumps

The maximum recycle concentrator recirculation pumps (2/3-2012-413A and 2/3-2012-413B) circulate the liquid from the bottom of the concentrator vapor head through a heat exchanger and back to the bottom of the concentrator vapor head. The pump flowrate is approximately 11,500 gal/min so that a low differential fluid temperature is maintained to minimize fouling of the heat exchanger tubes.

11.2.2.2.26 Maximum Recycle Concentrator Heaters

The maximum recycle concentrator heaters (2/3-2012-419A and 2/3-2012-419B) heat the liquid radwaste circulated from the bottom of the concentrator vapor head. The normal operating temperature for the concentrator heaters is the boiling point of the liquid waste which is approximately 212°F. Steam heating for these shell and tube heat exchangers is provided by the maximum recycle reboilers and the radwaste reboiler.

11.2.2.2.27 Maximum Recycle Concentrated Waste Transfer Tanks

The maximum recycle concentrated waste transfer tanks (2/3-2012-416A and 2/3-2012-416B) collect the concentrated liquid waste from the maximum recycle concentrator system The liquid waste from the transfer tank may be recycled to the maximum recycle floor drain neutralizer tanks or the concentrated waste may be transferred to the concentrated waste tank.

11.2.2.2.28 Maximum Recycle Concentrator Condensers

The maximum recycle concentrator condensers (2/3-2012-414A and 2/3-2012-414B) condense the water vapor from the concentrator vapor head. The concentrator condensers are shell and tube heat exchangers cooled by the service water system. The condensed liquid (condensate) drains by gravity flow to the distillate tanks.

11.2.2.2.29 Maximum Recycle System Liquid Radwaste Reboilers

The maximum recycle system liquid radwaste reboilers (2/3-2012-354A and 2/3-2012-354B) are shell and tube heat exchangers which use nuclear steam from either unit to a heat closed loop. The secondary loop is heated to produce steam which supplies the heat to the concentrator heater for boiling the liquid radwaste in the bottom of the concentrator vapor head vessel.

11.2.2.2.30 Radwaste System Reboiler

The radwaste system reboiler (2001-484) is a shell and tube heat exchanger heated by steam from the heating steam boiler. The closed loop heating system within the radwaste reboiler provides alternate heating to the maximum recycle concentrator heaters.

11.2.2.2.31 Maximum Recycle Distillate Tanks

The maximum recycle distillate tanks (2/3-2012-410A and 2/3-2012-410B) collect the gravity draining condensate from the maximum recycle concentrator condensers. The distillate tanks also receive liquid waste from the floor drain neutralizer tanks when the maximum recycle concentrator system is bypassed. Recycled liquid waste from the floor drain filter is also received by the distillate tanks. The outlet flow from the floor drain demineralizers can also be recycled back to the distillate tanks for additional treatment through the demineralizers.

11.2.2.2.32 Maximum Recycle Distillate Tank Pumps

The maximum recycle distillate tank pumps (2/3-2012-409A and 2/3-2012-409B) transfer the distillate tank contents normally through the floor drain demineralizers and to either the floor drain sample tanks or the waste sample tanks. The pumps may also transfer the distillate tank contents around the demineralizers and to the waste sample tanks or the floor drain sample tanks. The liquid contents may also be recycled to the floor drain collector tank.

11.2.2.2.33 Floor Drain Surge Tank Transfer Pump House Sump and Sump Pumps

The floor drain surge tank transfer pump house sump collects liquid from the floor drain surge tank, and the sump pumps (2/3-2012-356A and 2/3-2012-356B) transfer the liquid from the sump to the floor drain collector tank.

11.2.2.2.34 Floor Drain Demineralizers

The floor drain demineralizers (2/3-2012-418A and 2/3-2012-418B) are typically mixed deep bed (H-OH) polishing type resin. When the resins are depleted they are normally transferred to the spent resin tank for disposal of radioactive waste. Flow through the resin bed is 200 gal/min to ensure proper ion exchange. The liquid flow from the system is about 25 gal/min with a recirculation flow of about 175 gal/min back to the liquid feed tank from the demineralizer outlet.

11.2.2.2.35 Discharge Flow to the Discharge Canal

The tank contents to be discharged are sampled and must have a minimal radioactivity content so that the calculated discharge flowrate is greater than the pump capacity with dilution flow from the main condenser circulating water flow to the discharge canal. An effluent radiation monitor is located off-stream to warn of high-activity water being discharged. Requirements for continuing liquid discharge without the effluent radiation monitor are specified in the ODCM.

11.2.2.2.36 Discharge Line to the Discharge Canal

The above ground segment of the discharge line (2/3-2019) is carbon steel and the underground segment is High Density Polyethylene (HDPE) (PE4710).

11.2.2.2.37 Liquid Radwaste System Piping

A major upgrade of the radwaste system piping (about 7000 feet) replaced approximately 2500 feet of piping and permanently removed approximately 1500 feet of piping. About 3000 feet of the piping remains in place. The replacement piping, fittings, and valves are stainless steel or HDPE (PE4710) for underground portions. The original piping is carbon steel.

11.2.2.3 Redundancy of Major Equipment

Redundancy of the major pieces of equipment discussed in Section 11.2.2.2 facilitates operation of the systems while a pump is down for maintenance; a filter is being backwashed; or an ion-exchange resin is being cleaned or sluiced to the spent resin tank. The redundancy in tanks provides for an available process tank while the contents of one tank are being recirculated, sampled, or transferred.

11.2.2.4 Alternate Process Pathways

There are several process pathways within the liquid radwaste system. The normal flow path for liquids collected in the waste collector tank is through the waste sample tanks to condensate storage. An alternate pathway exists for the waste sample tanks to be transferred to the waste surge tank. The floor drain collector tank flow pathway is through the maximum recycle process system to the floor drain sample tanks that transfer liquid to the waste collector tank. An alternate processing pathway exists such that the floor drain sample tank contents can be transferred to the waste surge tank through the waste collector tank and waste collector processing system.

11.2.2.5 Instrumentation and Control

The system is instrumented with temperature, pressure, and flow indicators. There are differential pressure transmitters across such equipment as the demineralizers and the filters. The high differential pressure alarms in the radwaste control room. There are temperature, pressure, and flow recorders located in the radwaste control room. Tank level indicators transmit the tank level indication to the radwaste control room where a high-level alarm sounds at a preset value to minimize tank overflow. The tank high-level alarms also annunciate in the main control room.

11.2.2.6 Process Monitoring and Sampling

The liquid radwaste system has a radiation monitoring and sampling station for the discharged effluent. This system is discussed in Section 11.2.3.

The liquid radwaste process sampling system is provided in three parts (see Drawing M-178A):

A. The radwaste sample sink;

B. The maximum recycle sample sink; and

C. The maximum recycle demineralizer sample sink.

The sample sources for the radwaste sample sink are listed in Table 11.2-6.

These sample points provide means at a centralized location to obtain liquid grab samples for radioisotopic analysis in the chemical laboratory. In addition to these liquid sample points, seven conductivity cells are located in the radwaste sample sink cabinet. The conductivity cells monitor the liquid conductivity for the following streams:

A. Waste sample tank pump "A" recirculation (range 0 to 10 μmho);

B. Waste sample tank pump "B" recirculation (range 0 to 10 μmho);

C. Waste sample tank pump "C" recirculation (range 0 to 10 μmho);

D. Floor drain collector tank pump discharge (range 0 to 500 μmho);

E. Waste collector pump discharge (range 0 to 100 μmho);

F. Floor drain filter outlet (range 0 to 100 μmho); and

G. Waste demineralizer outlet (range 0 to 10 μmho).

The sample sources for the maximum recycle sample sink are listed below:

A. Maximum recycle floor drain neutralizer tank "A";

B. Maximum recycle floor drain neutralizer tank "B";

C. Concentrator condenser "A" outlet; and

D. Concentrator condenser "B" outlet.

These sample points provide means to obtain liquid grab samples at a centralized location for radioisotopic analysis in the chemical laboratory. The analytical results determine the water quality and system operation.

The sample sources for the maximum recycle demineralizer sample sink are listed below:

A. Floor drain demineralizer inlet "A";

B. Floor drain demineralizer outlet "A";

C. Floor drain demineralizer inlet "B"; and

D. Floor drain demineralizer outlet "B."

These sample points provide means to obtain liquid grab samples at a centralized location for radioisotopic analysis in the chemical laboratory. The analytical results determine water quality and demineralizer efficiency.

11.2.2.7 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. Effluent monitor calibration is performed periodically as required by procedure and the Offsite Dose Calculation Manual (ODCM)

11.2.2.8 Protection Against Accidental Discharge

Protection against accidental discharge is provided by design redundancy, instrumentation for detection and alarm of abnormal conditions, and procedural controls. The arrangement of the radwaste building and the methods of waste processing provide a substantial degree of immobility of the wastes within the station. This arrangement assures 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 small. For example, the waste collector tanks, filter and demineralizers, and other equipment within the radwaste building are contained within rooms so that leakage is contained within the building.

All Potentially Radioactive and Radioactive liquid waste discharges to the environment are routed through a single line to the discharge canal. This line has flowmeters, an offline radiation monitor, and double valves which are kept locked closed except when in use. The normal flow of liquid waste to the river is from the waste surge tank. The floor drain sample tank(s) or portable waste treatment system tank(s) can be discharged if necessary. Locked closed valves (2/3-2001-91, 2/3-2018A-504, and 2/3-2018B-501) prevent transfer to the discharge canal when transferring waste sample tank contents and floor drain sample tank contents to the waste surge tank. Procedurally, the waste surge tank must be sampled, analyzed, and a discharge rate determined prior to allowing discharge to the canal. The discharge procedure also requires the valve lineup for discharge to be independently verified and the discharge rate calculation to be independently verified. Once a transfer is initiated, the operator checks the flowmeter, the effluent radiation monitor, and the level recorder for the waste surge tank. Thus, the operator has a number of means of confirming the correct routing.

The only error that could cause inadvertent waste discharge from the radioactive waste management system to the canal would be leaving the canal discharge valves open so subsequent tank filling could result in flow to the canal. Most tank contents transferred to the waste surge tank are to be discharged. In the highly unlikely event that the canal discharge valves were left open so that a subsequent tank filling resulted in flow to the canal, no effluent concentration limit (ECL) should be exceeded.

It should be noted that procedural controls require the discharge valves to be locked closed at the end of a discharge. Also the discharge valves must be verified locked closed when transferring a tank of liquid to the waste surge tank.

11.2.2.9 Leakage Detection Systems

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 Section 5.2.

11.2.2.10 Ultrasonic Resin Cleaners

The ultrasonic resin cleaner is a device for cleaning the ion-exchange resins in the deep bed condensate treatment system. An ultrasonic resin cleaner (URC) is installed in the Unit 2 condensate demineralizer system. Both Unit 2 and Unit 3 condensate demineralizer systems are cross-connected to permit the Unit 2 URC to service either unit. The Unit 3 URC was replaced with an advanced resin cleaner. The accumulated crud (principally iron oxides) is removed as a thin water slurry to the waste collector subsystem of radwaste.

11.2.2.11 Advanced Resin Cleaner

The advanced resin cleaner (ARC) is a device for cleaning the ion-exchange resins in the deep bed condensate treatment system. An advanced resin cleaner is installed in the Unit 3 condensate demineralizer system and is cross-connected to permit use with either unit. The ARC features a vibrating screen system containing a coarse screen and fine screen in series and a high pressure water spray both of which aid in removing corrosion products. The spray water and cleaned resin transfer water are recycled continuously to minimize the volume of wastewater generated.

11.2.3 Radioactive Releases

11.2.3.1 Release Concentrations

Activity released with the liquid wastes is difficult to characterize since liquid wastes come from a number of sources and the quantity of activity is a strong 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.

Table 11.2-7 shows the typical isotopic content which may be present in the radioactive liquid waste discharged to the river based on historical data and pre-uprate conditions. Table 11.2-8 shows the corresponding typical tritium content in the liquid waste discharged from the station. Table 11.2-9 shows the typical isotopic content discharged in the containment cooling service water.[1-10]
Core uprate to 2957 MWt is expected to increase the activity in the liquid waste discharged including tritium by the percentage of the uprate, i.e., 17%.

When discharged to the river, the liquid radioactive wastes from Units 1, 2 and 3 are diluted in the condenser cooling water discharge canal. This dilution of the liquid radioactive wastes lowers the concentration at the time of discharge to a level which is in accord with the limits prescribed by 10 CFR 20 and State of Illinois regulations. The expected average annual activity discharge is significantly less than that permissible under 10 CFR 20. Since this estimate assumes that the activity discharged consists only of radioisotopes Sr-90 and Pb-210, the estimate overstates the actual contribution to the environment radioactivity.

Since additional dilution of wastes by the normal river flow further reduces radioactivity, concentrations of waste activity actually in the river are of the order of one-hundredth of the ECL per 10 CFR 20 for the mixtures generally discharged.

On the basis that the tritium activity in the liquid waste originates in the reactor and is essentially associated with the water molecule, tritium concentrations in liquid wastes will be about the same as in reactor water. Because of the infrequent need to discharge wastes to the river, and the large dilution factor which results in a tritium concentration several orders of magnitude below the ECL, the radiological consequences of tritium activity in the liquid wastes are nil.

11.2.3.2 Effluent Monitoring and Sampling

The radwaste discharge monitor is an offline sampling type monitor (see Drawing M-347B). When a discharge occurs, the radwaste operator valves-in the monitor and energizes the instrumentation. The water sample is taken from the discharge to river line and is fed through the process detector, a grab sample valve, and into a receiver tank. Failure by any of several means provides audible annunciation in the radwaste control room. The high-activity alarm annunciates in both the radwaste control room and the main control room. The monitor is addressed in more detail in Section 11.5.

The liquid contents of the waste surge tank, floor drain sample tank or portable waste treatment system tank are sampled and analyzed before being discharged to the river. The discharge rate is determined based on the chemical analysis and the dilution flow in the discharge canal. If the radiation monitor high-radiation alarm sounds, an automatic sample is taken in the monitor cabinet for chemical analysis by the laboratory.

The service water system discharge stream for both Unit 2 and Unit 3 is monitored for radioactivity and is sampled for chemical analysis periodically as required by the ODCM (see Drawings M-3486 and M-3496).

11.2.3.3 Liquid Waste Release Points

There are three liquid Potentially Radioactive and Radioactive waste release points from the station to the discharge canal. The following release points (see Figure 11.2-21) are shown in the ODCM:[11]

A. The Unit 2/3 liquid radwaste discharge point;

B. The Unit 2 service water system discharge point; and

C. The Unit 3 service water system discharge point.

The typical isotopic quantities of activity found in these streams are given in Tables 11.2-7 and 11.2-9.

There are radiation monitors on each of these three discharge streams.

11.2.3.4 Liquid Waste Discharge from Containment During Accident Conditions

Following the Three Mile Island accident, the NRC requested that each licensee evaluate the possibility of an inadvertent transfer of potentially highly radioactive liquids from inside containment to the liquid radwaste area. The evaluation was conducted and the appropriate modifications were completed.

Plant design ensures that highly radioactive fluids are confined to the reactor building during a loss-of-coolant accident. Transfer of radioactive water from the following systems and components is prevented during an accident:

A. Reactor building equipment drain tank (RBEDT) pump;

B. East reactor building floor drain sump (RBFDS) pump;

C. West reactor building floor drain sump (RBFDS) pump;

D. Southeast core spray/LPCI corner room sump pump;

E. Southwest core spray/LPCI corner room sump pump;

F. High pressure coolant injection (HPCI) room sump pump;

G. Drywell equipment drain sump; and

H. Drywell floor drain sump.

The RBEDT and RBFDS pumps trip upon a receipt of a Group II primary containment isolation (PCI) signal and do not automatically restart after a Group II PCI is reset. Since the emergency core cooling system (ECCS) room sumps (Items D, E, and F above) are required to be operational during a Group II PCI condition, they are not tripped. The ECCS corner room sump pump discharges are routed to the RBFDSs to confine radioactive fluids to the reactor building while these ECCS sump pumps are operable. A PCI Group II signal bypass is provided so that the RBEDT pump and valves can be operated in both the discharge and recirculation modes with the PCI Group II signal still present, i.e., a separate PCI group reset is provided for the RBEDT system. A control room control switch and an annunciator is provided for each of the RBEDT, east RBFDS, and west RBFDS pumps and for the RBEDT recirculation and discharge valves.

11.2.4 References

1. Letter from N.H. Kalivianakis (CECo) to J.G. Keppler (NRC), dated June 24, 1986, transmitting the Dresden Station Annual Environmental Radiological and Meteorological Operating Report for 1985.

2. Dresden Nuclear Power Station Radioactive Waste and Environment Monitoring Annual Report for 1986, dated March 1987.

3. Letter from E.D. Eenigenburg (CECo) to A.B. Davis (NRC), dated March 18, 1989, transmitting Dresden Nuclear Power Station Radioactive Waste and Environmental Monitoring Annual Report for 1987, dated March 1988.

4. Letter from E.D. Eenigenburg (CECo) to U.S. Nuclear Regulatory Commission, dated February 24, 1989, transmitting Dresden Nuclear Power Station Radioactive Waste and Environmental Monitoring Annual Report for 1988, dated March 1989.

5. Letter from E.D. Eenigenburg (CECo) to A.B. Davis (NRC), dated March 19, 1990, transmitting Dresden Nuclear Power Station Radioactive Waste and Environmental Monitoring Annual Report for 1989, dated March 1990.

6. Letter from E.D. Eenigenburg (CECo) to A.B. Davis (NRC), dated March 18, 1991, transmitting Dresden Station Annual Radiological Environmental Operating Report for 1990, dated March 1991.

7. Letter from C.W. Schroeder (CECo) to A.B. Davis (NRC) dated February 20, 1992, transmitting Dresden Station Semiannual Radiological Effluent Report for July through December, 1991.

8. Dresden Station Semiannual Radiological Effluent Report for January through June, 1991.

9. Letter from E.D. Eenigenburg (CECo) to A.B. Davis (NRC), dated February 28, 1991, transmitting Semiannual Radioactivity Report for July through December 1990 for Dresden Nuclear Power Station.

10. Dresden Station Semiannual Radioactive Effluent Report for January through June, 1990.

11. Dresden Nuclear Power Station Offsite Dose Calculation Manual, Revision O.A, April 1991.

Table 11.2-1

VOLUMETRIC THROUGHPUT - RADIOACTIVE WASTE CONTROL SYSTEM

	DESIGN BASIS (gal/day)
Waste Collector	Units 2 and 3
Normal	28,800
Maximum	180,000
Floor Drain Collector System	
Normal	12,000
Maximum	44,000
Waste Neutralizer System	
Normal	4,200
Maximum	21,000

Table 11.2-2

TANK CAPACITIES IN LIQUID RADWASTE SYSTEM

Radwaste Tanks Outside:	Location	Capacity (gal)
Waste Sample Tanks	Outside radwaste	
A	building at grade	33,000
B		33,000
C		33,000
Floor Drain Sample Tanks	Outside radwaste	
A	building at grade	22,000
B		22,000
Waste Surge Tank	Outside radwaste building at grade	77,000
Tanks Inside Radwaste Building:		
Waste Collector Tank	Basement Floor	33,000
Floor Drain Collector Tank	Basement Floor	22,000
Waste Neutralizer Tanks		
A	Basement Floor	16,500
B	Basement Floor	16,500
Concentrated Waste Tank	Basement Floor	6,000
Cleanup Filter Sludge Storage Tanks		
A	Basement Floor	27,000
B	Basement Floor	27,000
Filter Sludge Storage Tank	Basement Floor	27,000
Spent Resin Storage Tank	Basement Floor	15,000
Tanks Inside Maximum Recycle Building:		
Floor Drain Neutralizer Tanks		
A	Ground Floor	22,000
B	Ground Floor	22,000
Concentrated Waste Transfer Tanks	Ground Floor	1,000
A	Ground Floor	1,000
B		
Distillate Tanks		
A	Ground Floor	2,000
B	Ground Floor	2,000

Table 11.2-3

TYPICAL REQUIREMENTS FOR PLANT RECYCLE OF LIQUID RADWASTE RECLAIMED WATER FOR THE CONTAMINATED CONDENSATE STORAGE TANKS

Parameter	Limits
pH	5.6 to 8.6
Conductivity	<1 μmho/cm
Silicon dioxide (SiO_2)	<=100 ppb
Chloride (Cl_2)	<=20 ppb
Radioactivity	8 x 10^{-4} μCi/cc
Turbidity	<=10 nephelometric turbidity units (NTU)[1,2]
Sulfate	<=20 ppb
Total organic carbon	<=0.40 ppm

Notes:

1. If the NTU is 10 or less and all other chemistry criteria are met, the water is acceptable for storage.

2. If the NTU is greater than 10, the tank should be reprocessed. The water will not be returned to storage without prior specific approval from the Shift Engineer, after consulting with a chemist.

Table 11.2-4

INPUTS TO THE WASTE COLLECTOR TANK

A. The cleanup demineralizers;

B. The pressurized drain pumps;

C. Unit 2 ultrasonic resin cleaner; Unit 3 advanced resin cleaner;

D. Unit 2 turbine building equipment drain sumps and pumps;

E. Unit 2 reactor building equipment drain tanks and pumps;

F. Unit 2 drywell floor drain sumps and pumps;

G. Unit 2 reactor building floor drain sumps and pumps;

H. Unit 2 drywell equipment drain sumps and pumps;

I. Unit 2 drywell equipment drain sump heat exchangers;

J. Unit 2 condensate demineralizer system;

K. Unit 3 condensate demineralizer system;

L. Unit 2 condensate demineralizer equipment relief valve discharge;

M. Unit 3 condensate demineralizer equipment relief valve discharge;

N. Unit 3 turbine building equipment drain sumps and pumps;

O. Unit 3 reactor building equipment drain tanks and pumps;

P. Unit 3 drywell equipment drain sumps and pumps;

Q. Unit 3 drywell floor drain sumps and pumps;

R. Unit 3 reactor building floor drain sumps and pumps;

S. Unit 3 drywell equipment drain sump heat exchangers;

T. Unit 3 recycle drain;

U. Unit 3 fuel pool heat exchanger;

V. Unit 2 fuel pool heat exchanger;

W. The waste demineralizer;

Table 11.2-4 (Continued)

INPUTS TO THE WASTE COLLECTOR TANK

X. The off-gas drain pump;

Y. The fuel pool filter;

Z. The floor drain filter;

AA. The floor drain sample tank;

AB. Unit 2 recycle drain;

AC. Samples system sinks;

AD. The spent resin tank;

AE. The waste filter sludge tank and the resin cleaner sludge tank;

AF. Unit 2 reactor bulding equipment drain tank drains;

AG. Unit 3 reactor building equipment drain tank drains;

AH. Unit 2 turbine building equipment drain sump drains; and

AI. Unit 3 turbine building equipment drain sump drains.

Table 11.2-5

INPUT TO THE FLOOR DRAIN COLLECTOR TANK

A. Waste collection tank/floor drain collection tank south header;

B. Waste collection tank/floor drain collection tank north header;

C. Unit 3 turbine building floor drain sump;

D. Unit 3 drywell floor drain sumps and pumps;

E. Unit 3 reactor building floor drain sumps and pumps;

F. Unit 2 turbine building floor drain sump;

G. Unit 2 drywell floor drain sumps and pumps;

H. Unit 2 reactor building floor drain sumps and pumps;

I. Surge tank transfer pumphouse sump;

J. Floor drain filter sludge tank;

K. Waste filter sludge tank and resin cleaner sludge tank;

L. Off-gas filter house floor drain sump;

M. Sample system sinks;

N. Floor drain demineralizers;

O. Radwaste floor drain sump and pump A;

P. Radwaste floor drain sump and pump B; and

Q. Turbine building floor drain.

R. Unit 2 Turbine Building Equipment Drain Sump and Pumps (see section 9.3.3.5)

S. Unit 3 Turbine Building Equipment Drain Sump and Pumps (see section 9.3.3.5)

T. 100% Condensate Filtration System Backwash Drain (see Section 10.4.7.2)

Table 11.2-6

RADWASTE SAMPLE SINK SAMPLE SOURCES

A. Floor drain collector pump outlet;

B. Floor drain filter outlet;

C. Waste collector pump outlet;

D. Waste collector filter "A" outlet;

E. Waste collector filter "B" outlet;

F. Waste neutralizer tank "A";

G. Waste neutralizer tank "B";

H. Waste surge tank;

I. Waste demineralizer outlet;

J. Waste sample tank "A";

K. Waste sample tank "B";

L. Waste sample tank "C";

M. Floor drain sample tank "A"; and

N. Floor drain sample tank "B."

Table 11.2-7

TYPICAL RADIOACTIVE ISOTOPIC CONTENT DISCHARGED IN THE LIQUID RADWASTE EFFLUENT

Nuclides Released	Quantity (Ci/yr)
Sr-89	8.06×10^{-4}
Sr-90	2.35×10^{-3}
Ar-41	5.87×10^{-4}
Mn-54	6.65×10^{-2}
Co-58	1.57×10^{-3}
Fe-59	4.40×10^{-3}
Co-60	2.72×10^{-1}
Zn-65	7.87×10^{-5}
Ru-103	3.97×10^{-5}
Sb-122	1.74×10^{-3}
Sb-124	1.01×10^{-2}
I-131	4.11×10^{-4}
I-133	1.47×10^{-4}
I-135	2.72×10^{-5}
Cs-134	1.38×10^{-2}
Cs-137	2.92×10^{-1}
Ba-140	2.42×10^{-4}
La-140	6.51×10^{-4}
Ce-141	4.21×10^{-4}
Cr-51	8.30×10^{-4}
Ag-110m	1.88×10^{-6}
Xe-133	1.97×10^{-3}
Xe-135	1.21×10^{-3}
Tc-99m	1.72×10^{-4}
Mo-99	3.16×10^{-4}
Fe-55	4.02×10^{-2}

Table 11.2-7 (Continued)

TYPICAL RADIOACTIVE ISOTOPIC CONTENT DISCHARGED IN
THE LIQUID RADWASTE EFFLUENT

Nuclides Released	Quantity (Ci/yr)
Kr-88	5.96×10^{-6}
Nb-95	1.59×10^{-6}
Zr-95	3.05×10^{-6}
Cs-138	5.43×10^{-5}
Rb-88	5.24×10^{-5}
Cu-64	4.64×10^{-7}
Br-82	2.16×10^{-4}
I-132	8.59×10^{-6}
I-134	4.30×10^{-6}
As-76	9.81×10^{-6}
Hf-181	1.82×10^{-6}

Note: These typical annual values were averaged over 7 years of station operation (1985 through 1991).

Table 11.2-8

TYPICAL TRITIUM CONTENT DISCHARGED
IN THE LIQUID RADWASTE EFFLUENT

Year	Quantity (curies)
1985	7.45
1986	12.71
1987	22.84
1988	21.93
1989	17.18
1990	20.37
1991	12.77
Total	115.25
Average	16.46 Ci/yr

Table 11.2-9

TYPICAL RADIOACTIVE ISOTOPIC CONTENT DISCHARGED IN THE CONTAINMENT COOLING SERVICE WATER

Nuclides Released	Quantity (Ci/yr)
Sr-89	3.04×10^{-6}
Sr-90	9.11×10^{-7}
Mn-54	1.17×10^{-4}
Co-58	5.79×10^{-6}
Fe-59	8.67×10^{-6}
Co-60	4.77×10^{-4}
Zn-65	1.28×10^{-5}
Sb-124	2.86×10^{-9}
I-133	1.73×10^{-7}
I-135	3.20×10^{-7}
Cs-134	1.82×10^{-5}
Cs-137	9.12×10^{-5}
Ba-140	8.00×10^{-8}
La-140	8.00×10^{-8}
Cr-51	3.00×10^{-8}
Fe-55	4.17×10^{-5}

Note: These typical annual values were averaged over 7 years of station operation (1985 through 1991).

11.3 GASEOUS WASTE MANAGEMENT SYSTEMS

This section describes the capabilities to control, collect, process, handle, and dispose of the gaseous radioactive waste generated as a result of normal operation and anticipated operational occurrences.

The systems addressed in this section are the off-gas system, the turbine gland seal exhaust system, and the mechanical vacuum pump system. The effects of hydrogen addition for hydrogen water chemistry are also addressed in this section.

The off-gas system collects, contains, and processes the radioactive gases extracted from the steam condenser. The gases are exhausted by the steam jet air ejectors and flow through a preheater to a catalytic recombiner where all of the hydrogen is recombined with oxygen to form steam. All steam from the off gas stream is condensed for return as condensate and the non-condensable gases flow to a holdup pipe. The gas flow continues through a cooler condenser, a moisture separator, electric reheaters, a prefilter, activated charcoal adsorber vessels, high efficiency particulate air (HEPA) filters, and then to the 310-foot chimney for discharge to the environment. An alternate off-gas system flow path allows flow to bypass the catalytic recombiners and the activated charcoal adsorber vessels.

The gland seal exhaust system removes steam, air and radioactive gases from the turbine gland sealing system(see section 10.4.3) exhaust header. The steam is condensed and the condensate returned to the main condenser. The gases are discharged to the chimney via a holdup volume in the base of the stack shared by Units 2 and 3

The mechanical vacuum pump system rapidly establishes main condenser vacuum during startup. The vacuum pump effluent is discharged to the gland seal exhaust system line to the holdup volume in the stack base. If the mechanical vacuum pump is not available, the SJAES can establish vacuum.

The hydrogen water chemistry (HWC) system, including the hydrogen addition and oxygen addition systems, is described in Section 5.4.

The gaseous waste treatment facilities, including the 310-foot chimney, were evaluated under the Systematic Evaluation Program (SEP) Topic III-2, Wind and Tornado Loadings and Topic III-4.A Tornado Missiles. Two cases were evaluated independently by the NRC "staff, with the assistance from the Franklin Research Center….using the ACI ultimate strength method," and it was determined that the Dresden Station Unit 2 gaseous waste treatment facilities were adequately protected. The NRC determined that the loss of the reactor building ventilation stack would not result in an inability to achieve safe shutdown or in an adverse offsite radiological impact. Upgrading of the reactor building ventilation stack to withstand the design basis tornado was not recommended.[1][13]

11.3.1 Design Objectives

11.3.1.1 Off-Gas System

The design objectives of the off-gas system are as follows:

A. To provide effective control of process off-gases with capability for preventing releases over limits prescribed 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 abnormal conditions;

D. To minimize the release of the normally occurring activated radioactive gases by suitable short-term decay; and

E. To minimize the hazard of explosion of hydrogen and oxygen gas in the off-gas system. The off-gas system is designed to eliminate the potential for a hydrogen explosion resulting from radiolytic decomposition of reactor water. By using catalytic recombiner and restricting to hydrogen concentration to less than 0.5 percent in the remaining off-gas system piping.

To achieve these objectives, the off-gas system is designed using the following bases:

A.	Design pressure	300 psi
B.	Original equipment capability	300 psi
C.	Recombiner/activated charcoal adsorber vessels	350 psi
D.	Operating pressure	6 psig
E.	Off-gas system design flowrate	250 ft^3/min
F.	Minimum size of particulates filtered	0.3 μm
G.	Release point above ground	310-foot (chimney)
H.	Piping design code	ASA B31.1

The original off-gas system design was modified in 1972 to reduce the radioactive gaseous effluent discharged from the chimney. For the off-gas system with the recombiner and activated charcoal adsorber vessel addition, the design objectives for offsite doses were 10 mrem/yr for noble gases and 1.0×10^{-5} times the old (prior to January 1994)10 CFR 20 limits for iodines during normal operations. The Off Gas System is not Safety-related. The system was designed Seismic Class 2. The equipment and piping (including valves) added to the system during the recombiner/charcoal adsorber vessel addition modification were designed per ASME Section III, Nuclear Power Plant Components, Subsection ND, Code Class 3, even though the original piping was designed to USAS B31.1. Based on Regulatory Guides 1.26 and 1.143, future repairs or replacement shall be carried out using applicable codes as follows: Piping (ANSI B31.1), Valves (ANSI B16.5 or 16.34), Vessels or tanks (ASME VIII, Division 1). These changes in the applicable Codes or their effective dates will require documentation of comparison of quality requirements (Code reconciliation) as delineated in Section 3.2.10.

11.3.1.2 Turbine Gland Seal Exhaust System

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

11.3.1.3 Mechanical Vacuum Pump System

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

11.3.1.4 Plant Features Which 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 are summarized below:

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

B. Use of water-to-steam partition to retain halogens in the coolant.

C. Holdup of off-gas to allow decay of short half-lived activities before discharge. The nominal holdup with the recombiner bypassed reduces the potential radiation effects on the order of a factor of 10 as compared to no holdup.

D. Monitoring of the air ejector off-gas stream and initiating automatic isolation of the holdup piping when the radioactivity release rate exceeds limits. The holdup provides ample time to prevent release of fission product gases in excess of the limits.

E. Use of HEPA filters at the end of the holdup piping to remove particulate radioisotopes formed by the decay of the noble gas radioisotopes in the holdup pipe.

F. Elevated release from the 310-foot chimney, which is approximately 1½ 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 in addition to the air ejector monitors. A separate high-range noble gas monitor (SPING) and grab sampling point for monitoring accident effluents has also been installed (see Section 11.5).

H. Use of activated charcoal adsorber beds to delay the discharge of noble gases in the effluent and for the adsorption of any radioactive iodine. The activated charcoal adsorber vessels and beds are designed such that the temperature increase from radioactive isotopic decay is limited well below the activated charcoal ignition temperature, thus precluding overheating of the activated charcoal bed and the vessel or a fire in the activated charcoal bed and consequent escape of radioactive materials. Although the iodine input into the off-gas system is small by virtue of its retention in the reactor coolant and steam condensate, the activated charcoal bed effectively removes it by adsorption and prevents its release.

To protect the recombiner:
- The recombiner temperatures are monitored and alarmed providing advanced indication of degrading component performance.
- The off-gas trains that have three SJAEs (2A & 3B) are equipped with an alarm and trip on low SJAE steam pressure, and an alarm and trip on low dilution steam pressure.
- The off-gas trains that have two SJAEs (2B & 3A) are equipped with an alarm and trip on low SJAE steam pressure, and, alarm only on low dilution steam pressure.

11.3.2 System Description

There are 16 sources of radioactive gaseous effluent, all of which exhaust through the 310-foot chimney. These sources are listed in Table 11.3-1.

Major sources of gaseous waste radioactivity are the off-gas system and the turbine gland seal system.

The off-gas system is discussed in Section 11.3.2.1. The ventilation systems for the off-gas recombiner rooms for Units 2 and 3, the turbine building for Units 2 and 3, 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 these systems is discharged to the environment through the 310-foot chimney. The SBGTS discharges treated radioactive gases to the environment through the 310-foot chimney. The SBGTS is discussed in more detail in Section 6.5. The turbine gland seal system for Units 2 and 3 and the mechanical vacuum pump system for Units 2 and 3 are discussed in Sections 11.3.2.2 and 11.3.2.3 respectively.

11.3.2.1 Off-Gas System

11.3.2.1.1 Process Description

The off-gas system is shown in Drawings M-43, Sheets 1, 2, and 3, and M-371, Sheets 1, 2, and 3. The general arrangement drawings referenced in Section 1.2 show the elevation and plan views for the off-gas systems. Drawings M-12, Sheet 2 and M-345, Sheet 2 show more detail of the piping for the steam jet air ejectors. In brief, the condenser off-gas, or air ejector effluent, passes through a recombiner for radiolytically produced H_2, and O_2, followed by moisture separators, a shielded holdup line, a treatment system including additional moisture separation, a bed of activated charcoal filters, and through final particulate filters. It is then discharged at a height of about 40 feet inside of the chimney from a line that enters at the base. The effluents are diluted by a large volume of ventilation exhaust from several buildings (Table 11.3-2); the ventilation exhaust enters the side of the chimney at a height of about 40 feet and contains little activity by comparison.

The off-gas system operates at a pressure of approximately 6 psig or less, so the differential pressure that could cause leakage is small. To preclude leakage of radioactive gases, the system is welded wherever possible, and bellows seal valve stems or equivalent are used. The entire system is designed to maintain its integrity in the event of a hydrogen-oxygen detonation.

The fission gases, activation gases, radiolytic hydrogen and oxygen, and air inleakage are removed from the main condenser by the air ejectors and/or the main vacuum pump when in use. The mixture is diluted with steam during the process and passes through condensers for moisture removal. The mixture is then routed through a preheater to the catalytic recombiner. Preheating the mixture is necessary to ensure optimum recombiner performance.

In the recombiner, the radiolytic hydrogen and oxygen are catalytically converted to water in the form of superheated steam. This steam (along with steam used for the air ejector driving flow, 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 main condenser.

The noncondensible effluent is then routed to the holdup piping. In the holdup piping, the shorter lived radioactive isotopes (principally N-13, N-16, O-19 and certain isotopes of xenon and krypton) decay either to nonradioactive isotopes or radioactive particulate daughter products.

A sustained high radiation level in the off-gas holdup line (which would be expected to occur in the event of a significant fuel failure, for example) will first cause annunciation of an alarm in the main control room, then after a time delay, will cause isolation of the off-gas system. Isolation of the off-gas system results in a loss of condenser vacuum and a scram. The time delay gives the operator time to take action to reduce the radiation level in the off-gas system below the trip point (by reducing power, for example) to avoid a scram of the unit.

After leaving the holdup pipe, the effluent is cooled again for removal of water, reprocessed through a moisture separator for further drying, and then heated in a reheater for humidity control.

Before entering the activated charcoal adsorber vessels, the effluent is filtered by prefilters to remove the previously noted particulate daughter products.

The 12 activated charcoal adsorption beds permit selective adsorption and delay of the xenons and kryptons from the carrier gas (principally air). Following the activated charcoal adsorber vessels, the effluent is filtered again by afterfilters to remove particulate daughter products and then is discharged through the 310-foot concrete chimney.

The afterfilter system, which is located just before the chimney, consists 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 and tested to remove 99% of the particulates in the off-gas system based on PSL sphere testing.

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

The off-gas system startup and process operational limitations and requirements are covered in the ODCM and operating procedures.

11.3.2.1.2 Description of Major Components

11.3.2.1.2.1 Steam Jet Air Ejectors

The 2A and 3B trains have a two stage air ejector unit with an inter and after condenser, which discharges to a booster jet. Dilution steam is added to the discharge flow to the preheater. Either or both first stage jets may be used depending on the capacity needed and condensate temperature, but both second stage jets must be used at all times. (This arrangement resulted from a modification made to the original system when the plant was modified for closed cycle circulating water system operation.) Drawings M-43, Sheets 2 and 3, and M-371, Sheets 1 and 2.

The 2B and 3A trains have two first stage jets whose use is the same as in 2A and 3B trains. There are two second stage jets whose discharge bypasses the after condenser. There is no booster jet, but dilution steam is also added to the flow path. (This arrangement resulted from a modification made to prevent Off Gas fires, by maintaining the gas mixture diluted.) Steam is never condensed out of the flow stream by the after condenser, so there is never a combustible mixture present in the 2B and 3A trains. Figures 11.3-2, 11.3-4, 11.3-5, 11.3-8, 11.3-9, and 11.3-10.

Steam for the jets is from the turbine throttle header via 125 psig pressure control valves.

11.3.2.1.2.2 Preheaters

The preheaters are U-tube heat exchangers using steam on the tube side to superheat the off-gas mixture of steam and gases on the shell side. The off-gas mixture is heated to ensure recombination. 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. The steam source is the turbine throttle steam, and the steam passes through a pressure-reducing valve set at 250 psig. This limits the steam temperature at or below 410°F in case of loss of off-gas flow.

11.3.2.1.2.3 Catalytic Recombiners

The gaseous hydrogen and oxygen are combined catalytically into superheated steam at a variable temperature (nominally 800°F) based on the hydrogen and oxygen volume. The inlet temperature to the recombiner is nominally 350°F. Water will quench the catalytic reaction, but drying restores capability (temporary poison). Freon gases, oil and halogens act as permanent poisons to the catalyst.

A source for oxygen addition is provided to ensure that proportionate amounts of hydrogen and oxygen are available for recombination when the hydrogen water chemistry system is used. See Section 5.4 for a detailed discussion of the hydrogen and oxygen addition systems.

11.3.2.1.2.4 Off-Gas Condensers

The off-gas condensers are U-tube heat exchangers which use condensate to cool the gases and condense the steam in the off-gas piping. The effluent temperature is less than approximately 155°F. The condensed water drains to the main condenser.

The gas flow exiting the off-gas condenser consists mainly of air inleakage and fission product gases.

11.3.2.1.2.5 Water Separators

The water separator removes any entrained moisture from the gaseous mixture exiting the off-gas condenser. The water separator drains to the off-gas condenser.

11.3.2.1.2.6 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, which are removed by the prefilters downstream. The holdup pipe is 36 inches in diameter and 965 feet long. With the recombiner bypassed and a design flowrate of 250 ft^3/min (the normal flowrate is about 150 ft^3/min - see Table 11.3-3), the pipe provides a holdup time of about 30 minutes. The traverse time is increased from 30 minutes to approximately 6 hours by removal of the hydrogen and oxygen as water from the gaseous stream.

11.3.2.1.2.7 Cooler Condensers

The cooler condenser, is a shell and tube heat exchanger where the gas makes multiple shell side passes, it further cools the gas to remove as much moisture as possible. The gas stream is cooled by a chilled ethylene glycol-water mixture to a temperature of 45°F. This results in a moisture content of the off-gas of less than 1%.

11.3.2.1.2.8 Moisture Separators

The moisture separator removes any entrained moisture from the gas stream exiting the cooler condenser. Removal of condensible water vapor and entrained moisture is essential because the activated charcoal efficiency is a function of moisture content.

11.3.2.1.2.9 Reheaters

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

11.3.2.1.2.10 Prefilters

The prefilters consist of full-flow HEPA filters designed to remove 99.97% of the radioactive and non-radioactive particulates that are greater than 0.3 μm in size from the gas stream. There is a high differential pressure alarm which annunciates in the control room on a high differential pressure across the HEPA filter unit.

11.3.2.1.2.11 Charcoal Adsorbers

The activated charcoal adsorber beds provide for radioactive decay of the major activation gases and fission gases in the main condenser off-gas. The activated charcoal adsorber beds provide a retention time of 14.6 days for xenon holdup and 19.4 hours for krypton holdup. There are 12 activated charcoal adsorption beds. Each bed of activated charcoal is contained in a steel vessel which is 4 feet in diameter with an overall height of 21 feet. The 12 activated charcoal adsorption vessels contain approximately 74,000 pounds of activated charcoal, and are designed for 350 psig. The absorbers may be operated with 12 beds in series or 3 parallel trains of 4 beds in series.

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

The activated charcoal adsorber beds and vessels are designed to limit the temperature of the activated charcoal bed to well below the ignition temperature, thus precluding overheating or fire and consequent escape of radioactive materials. The activated charcoal adsorber vessels and beds are located in a shielded room, maintained at a constant temperature by an air conditioning system designed to remove the decay heat generated in the adsorber vessels. The maximum centerline temperature of each of the activated charcoal adsorber beds 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 activated charcoal adsorber vessels in the vault. Even if the vault cooling is lost, the temperature rise is not sufficient to cause activated charcoal ignition. The activated charcoal adsorber beds are maintained at 77°F by the vault air conditioning system. Due to the thermal capacitance of the activated charcoal adsorber beds and the massive concrete vault walls, temperature changes caused by failure of the vault air conditioning system will be sufficiently slow so that the resulting changes in the activated 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 flexibility. During a plant outage when the condenser is not maintained at vacuum, there is no gas flow through the activated charcoal adsorber beds and the holdup is very high, even if the activated charcoal reaches ambient temperatures. High radiation level in the activated charcoal adsorber bed vault will cause an alarm in the control room.

11.3.2.1.2.12 Afterfilters

Afterfilters provide the final filtration of the off-gas before its release to the 310-foot chimney.

The filters, located just before the chimney, consist of two 100% capacity HEPA filtering units. The second filter (spare) provides backup and assures availability of filtration. These filters are designed to remove from the off-gas 99.97% of the particulates 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 is 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 in.H_2O. 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 systems, moisture separator, particulate filters, and activated charcoal adsorber vessel vault air conditioning units is provided for operating convenience and maintenance. There are two, 100% redundant SJAE and recombiner trains. Provision is made for the two hydrogen analyzers to sample the effluent from either or both recombiner trains. Either or both cooler condenser trains (cooler condenser, moisture separator, reheater, and prefilter) may be selected for operation. The activated charcoal adsorber beds can be operated in one of three modes: all 12 activated charcoal adsorber beds in series; three parallel strings of four activated charcoal adsorber beds or bypassing of all 12 activated charcoal adsorber beds.

11.3.2.1.4 Alternate Off-Gas Discharge Pathway

Alternate pathways for the radioactive gases, as shown in Drawings M-43, Sheets 1 and 3 and M-371, Sheets 1 and 3, exist from the SJAE to the main chimney for discharge to the environment. These alternate pathways can bypass the recombiner train and the activated charcoal adsorber vessels and establish the original design pathway with only the holdup pipe and discharge filters to account for nuclide decay and for capturing particulates in the gas stream. Valving is provided to bypass and isolate the off-gas treatment system (recombiner and/or activated charcoal adsorber beds) and to operate with just the holdup line. 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 activated charcoal and HEPA filter system.

A more desirable alternate pathway is to bypass only the activated charcoal adsorber system. Use of the recombiners in the off-gas system would allow up to 6 hours holdup due to the removal of the hydrogen and oxygen content as water. Due to the high moisture content of the off-gas stream the activated charcoal adsorber beds cannot be employed in the system if the recombiners are bypassed.

The 2B recombiner can not be bypassed due to the fire prevention modification, and the 3A recombiner cannot be bypassed due to cutting and capping of Unit 3 SJAE crosstie.

11.3.2.1.5 Instrumentation and Control

The off-gas system is monitored by flow, humidity, and temperature instrumentation and by hydrogen analyzers for operation and control. Table 11.3-4 lists process instruments that cause alarms and notes whether the parameters are indicated or recorded in the main control room.

Drawings M-43, Sheet 5 and M-371, Sheet 5 show the hydrogen analyzer and oxygen analyzers for the off-gas system.

11.3.2.1.6 Process Monitoring and Sampling

The activity of the effluent entering and leaving the off-gas treatment system is continuously monitored.

The off-gas sampling system sample racks are shown in Drawings M-178, M-179, M-420, and M-421.

The process radiation monitoring includes the air ejector off-gas monitoring system and the area radiation monitors for the activated charcoal adsorber vessel vault. The air ejector off-gas monitoring system is discussed in Section 11.5, and the activated charcoal adsorber vessel vault area radiation monitor is discussed in Section 12.3. The activated charcoal adsorber vessel vault radiation monitor provides a local high-radiation alarm. A low alarm indicates malfunction of this monitoring system.

A manual sample of the process treated off-gas flow stream is taken downstream of the activated charcoal adsorber beds, see Drawings M-179 and M-421. At other sample points shown in Drawings M-43, Sheets 2 and 3, and M-371, Sheets 2 and 3, sample vials of gas are collected manually from the off-gas sampling system at a common point located in the off-gas filter building.

11.3.2.1.7 Inspection and Testing

The off-gas and exhaust ventilation filters are replaced when the pressure drop across the filter exceeds the normal operating range. Test connections are available for checking the efficiency of the installed filters. Adequate tests to determine filter efficiency are conducted as necessary. They are only of importance when fission gas release rates are significant. The off-gas system prefilters are also included in the testing requirements.

The gaseous waste disposal systems are used on a routine basis and do not require specific testing to assure operability.

Monitoring equipment and process instrumentation are calibrated and maintained on a specific schedule or when indication of malfunction occurs. The systems were functionally tested to verify their initial operability prior to placing them in service. The radioactive gaseous radiation monitoring instruments listed in the ODCM are demonstrated operable by performance of an instrument check on a daily basis. The offgas system air operated valves are tested every refuel outage in conjunction with the offgas radiation monitor calibration and main steam line high radiation offgas valves logic testing.

11.3.2.2 Turbine Gland Seal Exhaust System

11.3.2.2.1 System Description

The turbine gland seal exhaust system (see Figure 10.4-2 and Drawings M-43, Sheet 1 and M-371, Sheet 1) consists of the gland steam condenser and the gland steam condenser exhauster. There are two turbine gland seal exhaust systems for each unit.

The turbine gland sealing steam, along with substantial quantities of air (which is drawn through the outer seals), is drawn to the gland steam condenser by the exhauster. Approximately 95% of the steam used in the turbine gland seals is condensed in the gland steam condenser and returned to the main condenser.

The remaining steam, air and noncondensibles (including any radioactive gases) present in the gland seal off-gas is discharged to the Unit 2/3 common hold-up volume for gland seal exhaust/main vacuum pumps in the base of the chimney. The small quantity of radioactive gases released by way of the gland seal off-gas system does not require a long decay time. A minimum holdup time of 1.75 minutes in the hold up volume is used for decay of the major activation gases (N-16 and O-19), which have half-lives on the order of seconds.

The gland seal steam condenser exhauster maintains a vacuum on the gland seal steam condenser and the sealing steam exhaust header. The effluents from the gland seal system cannot be routed to the air ejector recombiner or 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 to 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.

11.3.2.2.2 Description of Major Components

11.3.2.2.2.1 Turbine Gland Seal Exhaust System Condenser

The turbine gland seal exhaust system condenser, using main condensate water through double-pass tubes, condenses about 95% of the steam in the gas stream. A bypass flapper in the water box causes most of the main condensate to bypass the tubes except at low flow. Level is maintained by a control valve. There are high and low level alarms.

11.3.2.2.2.2 Turbine Gland Seal Steam Condenser Exhauster

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

11.3.2.3 Mechanical Vacuum Pump System

The mechanical vacuum pump system (see Drawings M-43, Sheet 1 and M-371, Sheet 1) rapidly establishes the main condenser vacuum at 20 to 25 in.Hg. This system is used at 5% reactor thermal power and below. It exhausts through a discharge silencing tank at about 2320 scfm 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 exhaust system piping. There is one condenser vacuum pump and silencer for each unit. If it is not available, the SJAEs can draw vacuum, but this takes considerably longer.

11.3.2.4 Hydrogen Ignition Control

Because the off-gas system contains mixtures of hydrogen and oxygen, it contains a potentially explosive and burnable gas stream. Some of the precautions taken to minimize the potential for these explosions, pre-ignitions, and fires are as follows:

A. The off-gas afterfilters are grounded to prevent static build-up and sparks;

B. Operating procedures exist for controlling and extinguishing an off-gas system fire. Normally an explosive mixture exists only between the second stage air ejector discharge and the booster jet in 2A, 3A and 3B strains. 2B train was modified to always have a diluted, non combustible mixture, to prevent off gas fires.

11.3.3 Radioactive Releases

11.3.3.1 Plant Release Points

There are three release points to the atmosphere for gaseous effluent and ventilation exhaust - the reactor building vent stack, the 310-foot chimney, and Unit 1 chemical cleaning building stack.

11.3.3.1.1 Reactor Building Ventilation Stack

The physical and process characteristics of the two principal gaseous release points are shown in Table 11.3-5. The limitations for release of gaseous effluents from the station are set in the ODCM. Table 11.3-6[2-11] presents the typical radioactive isotopes and quantities discharged from Units 2 and 3. Based on historical data and pre-uprate conditions, uprate to 2957 MWt is expected to increase the activity in the gaseous effluents by the percentage of the uprate, i.e., 17%.

Air from the reactor building ventilation exhaust (approximately 110,000 ft^3/min per unit) is normally released through the reactor building vent stack, which is common to Units 2 and 3 (see Drawings M-269, Sheet 1 and M-529). If activity is present in any significant quantity, secondary containment is isolated and normal ventilation flow to the vent stack is automatically terminated as discussed in Section 6.2.3. Flow from the standby gas treatment system (SBGTS) (4000 ft^3/min) is directed to the base of the chimney. The SBGTS has its own particulate and charcoal filters.

Air or nitrogen from inerting or deinerting the drywell is normally discharged with the reactor building ventilation exhaust from the vent stack. If radioactivity is present in any significant quantity (because of activation products such as Argon-41 for example) the purge air can be discharged separately through the SBGTS to avoid a high-radiation trip of the reactor building ventilation.

11.3.3.1.2 Plant 310-Foot Chimney

The ventilation system air flow through the chimney is approximately 430,910 ft^3/min during normal operation of both units. The radioactive gaseous flow from the off-gas systems, 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 main chimney are shown on Figure 11.3-17 (Figure 10-2 from the Offsite Dose Calculation Manual [ODCM]).[12] The chimney dilution flow of ventilation air is shown on Drawing M-272.

Natural dispersion of gases into the atmosphere is achieved in an efficient manner by discharge through the chimney. The combination of the height of the chimney, 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 dose by a factor of approximately 100 as compared with release of the gaseous wastes at ground level.

Air ejector off-gases are normally expected to have the composition shown in Table 11.3-3.

The activation gases listed in Table 11.3-7 (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. For Units 2 and 3, the combined release rate is approximately 500 μCi/s.

The fission product gases may arise 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. The principal gaseous fission product isotopes discharged from the chimney based on historical data and pre-uprate conditions are shown in Table 11.3-8. Typical quantities, including the isotopic analysis, of the radioisotopes discharged from the 310-foot chimney are presented in Table 11.3-8.[2-11] Core uprate to 2957 MWt is expected to increase the activity in the gaseous effluents by the percentage of the uprate, i.e., 17%.

In the absence of fuel rod leaks, N-13 from the air ejector off-gases and the N-16 and O-19 from the gland seal system are the principal contributors to the environs radiation dose. The aggregate of these three corresponds to a radiation dose of less than 0.1 mrem/yr. If fuel rod leaks do occur, the noble radioactive gases xenon and krypton become the principal contributors. The solid daughter products of the noble gases are removed in the filter of the off-gas system before release of the gases to the 310-foot chimney.

The holdup of the condenser air ejector off-gas provides sufficient time between detection of high-radiation levels and isolation of the holdup line to prevent release of fission product gases in excess of the release limits. When such a release rate is detected, the holdup line is automatically isolated after a 15-minute delay. This time interval is provided to permit corrective action to be taken to obviate plant shutdown. The holdup time is established to provide for decay of short half-lived noble gases to reduce chimney release.

Similarly the 1.75-minute holdup time for the gland seal off-gas system is chosen to provide sufficient decay of the activation gases. The holdup time is shorter because the activity present in this system is three orders of magnitude less than that from the condenser air ejector. The short holdup time allows decay of N-16 and O-19, which have half-lives of 7 and 27 seconds, respectively. The 1.75 minute holdup time is provided by a five chamber holdup volume in the base of the chimney. This holdup volume is common to Units 2 and 3.

11.3.3.1.3 Unit 1 Chemical Cleaning Building Ventilation Stack

The Unit 1 chemical cleaning building and interim radwaste storage facility (IRSF) ventilation systems exhaust into the Unit 1 chemical cleaning building stack. The exhaust discharge is monitored to support the IRSF.

11.3.3.2 Effluent Monitoring and Sampling

The off-gas system provides ample monitoring and control to ensure that limits set forth in 10 CFR 20 are not exceeded. The off-gas holdup, effluent sampling, calibrating of the off-gas monitors, particulate filtering, and excessive release alarm are all protection measures taken to meet standards set by 10 CFR 20.

Normal monitoring of the chimney effluent is by a sampling radiation monitor suited to measuring a low concentration of activity in a large flow. In the event that operation of the SBGTS is required and, coincidentally, the turbine building ventilation is shut off, the activity in the small undiluted flow could exceed the sensitivity of the chimney monitor. It is therefore unsuited for measurement of an accident effluent. Additionally, the fission product mixture for an accident effluent is energetically quite varied relative to the normal noble gas mixture in the off-gas.

The original chimney monitoring system is intended for normal audit. A high-range noble gas monitor has been added for monitoring of any accident effluent (not only to the chimney, but to all potential release points). It is also an offline-sampling-type monitor. The sampling system for the 310-foot chimney is shown in Drawing M-422, Sheet 2.

Control of air ejector off-gas release rates is accomplished by duplicate continuous radiation monitor recorders on the off-gas line, which alarm in the control room. This monitoring instrumentation is described in Section 11.5. Samples of the air ejector off-gas can be taken for laboratory analysis and can be used to calibrate and check the air ejector off-gas monitors. The chimney monitors provide backup alarms for and supply data to the processor about the chimney release activity.

Similarly, the reactor building ventilation stack is monitored for the total release of radionuclides from this system. This stack monitor (see Figure 11.3-19 [Drawing M-422, Sheet 1]) has only an alarm function. The two monitors upstream of the reactor building ventilation duct isolation dampers also monitor the individual unit ventilation gas activity and, upon reaching a predetermined setpoint, causes secondary containment isolation. These monitors are discussed in more detail in Section 11.5. Secondary containment isolation is addressed in Section 6.2.3.

11.3.3.3 Effects of Hydrogen Addition

Commonwealth Edison has reviewed the effects that HWC has on offsite dose. The results of these calculations are based on conservative assumptions and should be considered approximate. The anticipated, calculated, offsite dose to the nearest individual is summarized below:

A.	Units 2 and 3 without HWC	1.7 mrem/year
B.	Unit 2 with HWC and Unit 3 without	4.8 mrem/year
C.	Units 2 and 3 with HWC	8.0 mrem/year

These values will increase by approximately the percentage of uprate. During the first HWC test (performed in May and June of 1982) it was determined that injecting hydrogen into the feedwater increases the carry-over of N-16 with the steam. This phenomenon results in higher than normal radiation levels in all areas of the plant that contain steam piping. This effect raised concerns about an increase in offsite dose due to the "sky-shine" of the turbine.

During operation of Unit 2 with Cycle 9 reload, an assessment of the effects of hydrogen injection on dose rate was made. In order to assess the effects of hydrogen injection, dose rate measurements were taken under the following conditions:

A. With hydrogen injection;

B. Without hydrogen injection; and

C. With Unit 2 shutdown.

Units 1 and 3 were shut down during this operating period.

The data indicate that the three plant areas most significantly influenced by hydrogen injection are the main turbine floor, the area above the main turbine floor, and the condensate pump room area. The largest average increase is seen on the turbine deck where dose rates rise by 450%. Additional decay time in the condenser and hotwell lessen the N-16 contribution in the condenser pump room so the dose rates increase by only 340%.

The area that shows the most significant increase is the turbine crane cab. The radiation shine off the top of the turbine increases the dose rate to the crane operator to as much as 100 mrem/hr. This dose rate is a function of positioning over the turbine as well as the amount of hydrogen being injected into the feedwater.

Other areas surveyed in the turbine building realize an insignificant increase in dose rates. All of these areas are well-shielded from reactor steam and condensate lines.

To assess the HWC impact on the environmental dose, thirty locations were selected to be surveyed based on their positions relative to one reference point. The reference point, the intersection of the turbine axis and center line between the D-2 low pressure turbines B and C, was assumed to be the center of the N-16 source for the environs. Measurements were taken for 5 to 30 minutes using a multiplying ion chamber.

Based on the data obtained, the contributions from HWC to the environment dose rate is a function of measurement location. Significant variation exists in the dose rate contributions at similar distances. This is a result of the shielding effect of various onsite structures and the dose contributed from radioactive onsite storage (such as holding tanks).

The dose rates within 750 feet of the measurement point generally follow an expected pattern. Dose rates close to the reactor building are low due to the shielding of the plant walls. As the distance from the reactor building increases the dose rates increase to a maximum and then decrease gradually. The largest contribution found due to hydrogen injection was 25 μR/hr at 440 feet south of the reference point. At distances greater than 1500 feet, the dose rate contribution to the environment is typically below 1 μR/hr.

11.3.4 References

1. Integrated Plant Safety Assessment Systematic Evaluation Program; Dresden Nuclear Power Station Unit 2, NUREG-0823, October 1982, Nuclear Regulatory Commission.

2. Letter from N.J. Kalivianakis (CECo) to J.G. Keppler (NRC), dated June 24, 1986, transmitting the Dresden Station Annual Environmental Radiological and Meteorological Operating Report for 1985.

3. Dresden Nuclear Power Station Radioactive Waste and Environmental Monitoring Annual Report for 1986, dated March 1987.

4. Letter from E.D. Eenigenburg (CECo) to A.B. Davis (NRC), dated March 18, 1989, transmitting Dresden Nuclear Power Station Radioactive Waste and Environmental Monitoring Annual Report for 1987, dated March 1988.

5. Letter from E.D. Eenigenburg (CECo) to U.S. Nuclear Regulatory Commission, dated February 24, 1989, transmitting Dresden Nuclear Power Station Radioactive Waste and Environmental Monitoring Annual Report for 1988, dated March 1989.

6. Letter from E.D. Eenigenburg (CECo) to A.B. Davis (NRC), dated March 19, 1990, transmitting Dresden Nuclear Power Station Radioactive Waste and Environmental Monitoring Annual Report for 1989, dated March 1990.

7. Letter from E.D. Eenigenburg (CECo) to A.B. Davis (NRC), dated March 18, 1991, transmitting Dresden Station Annual Radiological Environmental Operating Report for 1990, dated March 1991.

8. Letter from C.W. Schroeder (CECo) to A.B. Davis (NRC) dated February 20, 1992, transmitting Dresden Station Semiannual Radiological Effluent Report for July through December 1991.

9. Dresden Station Semiannual Radiological Effluent Report for January through June 1991.

10. Letter from E.D. Eenigenburg (CECo) to A.B. Davis (NRC), dated February 28, 1991, transmitting Semiannual Radioactive Report for July through December 1990 for Dresden Nuclear Power Station.

11. Dresden Station Semiannual Radioactive Effluent Report for January through June 1990.

12. Dresden Nuclear Power Station Offsite Dose Calculation Manual, Revision O.A, April 1991.

13. Integrated Plant Safety Assessment Systematic Evaluation Program; Dresden Nuclear Power Station Unit 2, NUREG-0823, Supplement No. 1, U.S. Nuclear Regulatory Commission, October 1989.

Table 11.3-1

RADIOACTIVE GASEOUS EFFLUENT SOURCES EXHAUSTED
THROUGH THE 310-FT CHIMNEY

A. Off-gas system for Unit 2;

B. Off-gas system for Unit 3;

C. Turbine gland seal system for Unit 2;

D. Turbine gland seal system for Unit 3;

E. Mechanical vacuum pump system for Unit 2;

F. Mechanical vacuum pump system for Unit 3;

G. Standby gas treatment system (SBGTS), Train A;

H. Standby gas treatment system (SBGTS), Train B;

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

J. Off-gas system recombiner rooms' ventilation system for Unit 3;

K. Turbine building ventilation system for Unit 2;

L. Turbine building ventilation system for Unit 3;

M. Off-gas building ventilation system;

N. Radwaste building ventilation system;

O. Maximum recycle building ventilation system; and

P. Solidification building ventilation system.

Table 11.3-2

CHIMNEY DILUTION AIR

Source	Flowrate (ft^3/min)
U2 turbine building and recombiner rooms ventilation exhaust	177,200
U3 turbine building and recombiner rooms ventilation exhaust	194,200
Radwaste building ventilation exhaust	28,760
Radwaste solidification building ventilation exhaust	16,000
Maximum recycle building ventilation exhaust	6,750
Off-gas filter building ventilation exhaust	5,500
* U2 turbine gland seal exhaust	4,000
* U3 turbine gland seal exhaust	4,000
Standby gas treatment system exhaust	4,000
U2 high-radiation sampling system building ventilation exhaust	1,000
U3 high-radiation sampling system building ventilation exhaust	1,000
TOTAL(1)	442,910

Notes:

1. In addition, 250 ft^3/min of off-gas per unit (total 500 ft^3/min) is exhausted through the 310-ft chimney.

* Figure reference value per FSAR - not located on drawing.

Table 11.3-3

AIR EJECTOR OFF-GAS COMPOSITION

	Flowrate (ft^3/min)
Hydrogen	0.0273/MWt = 81 at 2957 MWt
Oxygen	0.0137/MWt = 40.5 at 2957MWt
Air (assumed condenser leakage)	10 to 30
Water vapor (to saturate)	20 to 24
Activated noble gases	Negligible
TOTAL	152 to 176

Note: To be revised after operating data is collected.

Table 11.3-4

PROCESS INSTRUMENT ALARMS FOR OFF-GAS TREATMENT SYSTEM

Parameter	Main Control Room Indicated	Main Control Room Recorded
Preheater discharge temperature — low	X	
Recombiner catalyst temperature — high/low		X
Off-gas condenser drain well (dual) level — high/low (alarm only)		
Off-gas condenser gas discharge temperature — high (alarm only)		
H_2 analyzer (off-gas condenser discharge) (dual) — high		X
Cooler-condenser discharge temperature — high		X
Glycol solution temperature — high/low		X
Glycol storage tank level — low (alarm only)		
Prefilter differential pressure — high	X	
Charcoal bed inlet humidity — high		X
Charcoal bed temperature — high		X
Charcoal vault temperature — high/low		X
After filter inlet gas flow — high/low		X
After filter differential pressure — high (alarm only)		

Table 11.3-5

PHYSICAL AND PROCESS CHARACTERISTICS
OF PRINCIPAL GASEOUS RELASE POINTS

Characteristics	Gaseous Release Point	
	Chimney	Reactor Building Ventilation Stack[1]
Height (above grade)	310 ft (95 m)	160 ft (48 m)
Inside diameter, exit	11 ft (3.36 m)	9 ft (3 m)
Exit velocity	77.7 ft/s (23.7 m/s)	52 ft/s (16 m/s)
Discharge volume	Variable: can range from 200,000 to 442,910 ft^3/min	Variable: can range from 220,000 to 223,000 ft^3/min
Heat rate control	4 x 10^6 calories/s at 375,000 ft^3/min and is proportional to flow	-

Note:

1. The reactor building ventilation stack itself is 55 feet tall and is mounted on the turbine building.

Table 11.3-6

TYPICAL RADIOACTIVE ISOTOPES AND QUANTITIES DISCHARGED FROM THE REACTOR BUILDING VENTILATION SYSTEM STACK

	Nuclides Released	Quantity (Ci/yr)
A. Fission Gases	Kr-87	1.47×10^{-1}
	Kr-88	1.09×10^{0}
	Kr-85m	6.37×10^{-1}
	Kr-85	0.15×10^{-6}
	Xe-135	4.23×10^{1}
	Xe-133	6.78×10^{0}
B. Iodines	I-131	12.10×10^{-3}
	I-133	11.37×10^{-3}
	I-135	2.04×10^{-2}
C. Particulates	Sr-89	1.57×10^{-2}
	Sr-90	3.31×10^{-4}
	Cr-51	3.41×10^{-3}
	Mn-54	3.0×10^{-3}
	Co-58	1.23×10^{-3}
	Fe-59	1.22×10^{-3}
	Co-60	1.33×10^{-2}
	Zr-95	1.63×10^{-5}
	Nb-95	1.07×10^{-6}
	Ru-103	1.46×10^{-4}
	Ag-110m	9.79×10^{-5}
	Sb-124	5.29×10^{-5}
	I-131	7.04×10^{-4}
	Cs-134	4.99×10^{-7}
	Cs-136	1.19×10^{-5}
	Cs-137	1.18×10^{-4}

Table 11.3-6 (Continued)

TYPICAL RADIOACTIVE ISOTOPES AND QUANTITIES DISCHARGED FROM THE REACTOR BUILDING VENTILATION SYSTEM STACK

Nuclides Released	Quantity (Ci/yr)
Ba-140	1.40×10^{-3}
Ce-141	1.90×10^{-5}
Ce-144	4.83×10^{-6}
Zn-65	3.29×10^{-4}
Ba-133	1.98×10^{-5}
Sb-125	3.31×10^{-6}
Mo-99	3.81×10^{-3}
La-140	1.93×10^{-4}
Fe-55	1.87×10^{-2}

Note: These typical annual values were averaged over 7 years of station operation (1985 through 1991).

Table 11.3-7

TYPICAL OFF-GAS COMPOSITION FOR A SINGLE UNIT[(1)]

A. Activation Gases

The emission rate values for the activation gases are estimates. The high gamma energy and short half-lives for the activation gases make them unsuitable for routine measurements at the plant.

Isotope	Half-Life	Emission Rate (μCi/s)
N-17	4.1 s	1×10^{0}
N-16	7.35 s	1×10^{0}
O-19	29 s	1×10^{0}
N-13	10 min	2×10^{2}
Ar-41	1.83 hr	6×10^{0}
Ar-37	34.3 day	2×10^{-4}
H-3 (Tritium)	12.36 yr	1×10^{-3}

B. Noble Gases

		Emission Rate (μCi/cc)	
Isotope	Half-Life	Before Recombiner	After Treatment System
Kr-85m	4.40 hr	700	34
Kr-87	76 min	5,800	9
Kr-88	2.79 hr	3,200	63
Xe-133	5.27 day	240	41
Xe-135m	5.70 min	22,000	360
Xe-135	9.16 hr	4,800	200
Xe-138	4.20 min	140,000	1,400

Note:

1. Data must be doubled for Unit 2 and 3 combined maximum release rates.

Table 11.3-8

TYPICAL RADIOACTIVE ISOTOPES AND QUANTITIES DISCHARGED FROM THE 310-FOOT CHIMNEY

	Nuclides Released	Quantity (Ci/yr)
A. Fission Gases	Xe-138	1.86×10^{2}
	Xe-135m	3.53×10^{1}
	Kr-87	1.37×10^{1}
	Kr-88	2.02×10^{1}
	Kr-85m	8.91×10^{0}
	Kr-85	2.51×10^{-2}
	Xe-135	2.25×10^{2}
	Xe-133	1.55×10^{1}
	Ar-41	8.89×10^{-3}
B. Iodines	I-131	1.29×10^{-2}
	I-133	6.66×10^{-2}
	I-135	1.02×10^{-1}
C. Particulates	Sr-89	1.37×10^{-1}
	Sr-90	2.92×10^{-4}
	Cr-51	1.66×10^{-5}
	Mn-54	2.25×10^{-4}
	Co-58	4.45×10^{-6}
	Fe-59	1.11×10^{-5}
	Co-60	1.20×10^{-3}
	Ru-103	7.03×10^{-4}
	Ag-110m	1.02×10^{-5}
	Sb-124	2.79×10^{-7}
	I-131	5.88×10^{-3}
	Cs-134	2.74×10^{-7}
	Cs-137	2.03×10^{-4}

Table 11.3-8 (Continued)

TYPICAL RADIOACTIVE ISOTOPES AND QUANTITIES DISCHARGED
FROM THE 310-FOOT CHIMNEY

Nuclides Released	Quantity (Ci/yr)
Ba-140	1.33×10^{-2}
Ce-141	2.61×10^{-5}
Mo-99	2.19×10^{-6}
La-140	1.68×10^{-3}
Fe-55	8.33×10^{-4}

Note: These typical annual values were averaged over 7 years of station operation (1985 through 1991).

11.4 SOLID WASTE MANAGEMENT SYSTEM

This section describes the capabilities of the station for collecting, processing, and packaging wet and dry solid radioactive waste generated as a result of normal operation, including anticipated operational occurrences, for shipment offsite or storage onsite.

Contract services are used for processing Class A unstable waste and waste which requires stability for burial offsite due to the requirements of 10 CFR 61. The process control program (PCP)[1] is used, as applicable, to process all low-level radioactive wet wastes that are solidified or dewatered to meet the applicable federal, state, and burial site requirements.

The solid radwaste area is shown in the general arrangement drawings referenced in Section 1.2. The treatment and flow of wet solid waste is shown in Drawing M-46, Sheet 1.

PROCESS CONTROL PROGRAM (PCP) – Shall contain the current formulas, sampling, analysis, test, and determinations to be made to ensure that processing and packaging of solid radioactive wastes based on demonstrated processing of actual or simulated wet solid wastes will be accomplished in such a way as to assure compliance with 10 CFR parts 20, 61, and 71, State regulation, burial ground requirements, and other requirements governing the disposal of solid radioactive waste.

- Written procedures are established, implemented, and maintained covering the Process Control Program.

Changes to the PROCESS CONTROL PROGRAM (PCP) shall be documented and records of reviews performed shall be retained. This documentation shall contain:

- Sufficient information to support the change together with the appropriate analyses or evaluation 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, or other applicable regulations.

Changes shall become effective after review and acceptance, including approval by the Station Manager.

11.4.1 Design Objectives

The design objectives of the solid radioactive waste control system are to process, package, and provide shielded facilities for temporary storage of the radioactive waste. If needed, the waste may be allowed to radioactive decay prior to shipment and/or placed in extended storage prior to shipment offsite. These radioactive wastes are prepared for shipment in approved shipping containers and shipped offsite using common or contract carriers in compliance with the United States Department of Transportation (DOT) regulations (49 CFR) and 10 CFR 20, 10 CFR 61, and 10 CFR 71 as applicable.

11.4.2 System Description

The solid radioactive waste control system is a series of mechanical operations that are designed to process the solid wastes remotely with a minimum of personnel handling and exposure. The equipment supplied to accomplish this handling is designed to be remotely operated in order to accomplish the functions described below. The handling and processing are capable of being performed without exceeding established exposure limits.

The following are typical solid radioactive wastes:

A. Filter sludges and spent resins;

B. Concentrated wastes;

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, etc., from contaminated areas; and

F. Used reactor equipment such as spent control rod blades, temporary control curtains, fuel channels, and incore ion chambers.

The processing, packaging, and handling, both prior and subsequent to storage, are performed in facilities in accordance with procedures, the objectives of which are to minimize personnel radiation exposure, prevent spillage of radioactive wastes, and provide for necessary cleanup and maintenance of equipment.

The reactor wastes, such as spent control rod blades and fuel channels, and other irradiated hardware components are stored in the fuel storage pool to allow decay, then packaged, and transferred in approved shipping containers for offsite burial or for long-term onsite storage. Used reactor equipment is stored in the spent fuel storage pool before shipment. Storage in the spent fuel pool for any period of time permits the decay of the short-lived isotopes.

The maintenance wastes such as contaminated clothing and tools are packed in suitable containers (approved by the Department of Transportation) and may be stored prior to shipment.

The process wastes, such as filter sludges and spent resins, are collected in tanks, processed, and stored prior to shipment. Waste loading is accomplished by using remotely operated equipment. When required, shipping casks are used to shield the radioactive waste.

The general procedure for handling all solid radioactive waste is to temporarily store the waste onsite in appropriate containers. Then ship the containers per DOT Regulations to an offsite processor and/or to a licensed burial facility.

Radioactive Waste may be stored in the Interim Radwaste Storage Facility (IRSF) (shielded as necessary) for an extended time period. Storage in this site may prolong if the burial site is closed.

Equipment too large to be handled by contact requires special procedures. Since the need for handling large equipment is quite infrequent, providing storage facilities in advance of need is not justified. The handling of such equipment depends upon the radiation level, transportation requirements, and available storage sites. Suitable procedures for decontamination, shielding, shipment, and storage of such items are developed as necessary.

To assure personnel safety during operation, ample shielding of the processing and storage areas is provided. Television cameras are provided in locations where visible control is required in a radioactive area. In addition, ventilation is provided for contamination control during maintenance and cleanup.

11.4.2.1 Contractor-Supplied Solidification System

Contractor solidification services may be utilized for wastes. Contractor solidification services are often used for wastes that are required to be classified as ‘stable’ per 10 CFR 61 and/or the burial site licenses. The Process Control Program (PCP) assures vendor procedures are reviewed and approved and comply with 10 CFR 61 criteria.

When performing waste solidification on site, a sample is obtained in accordance with the contractor's Process Control Program (PCP).

During the solidification process a ‘fill head’ assembly is placed onto the liner or HIC opening. A seal limits the spread of contamination and dust.

The system control panel is easy to operate and is equipped with interlocks and alarms to preclude inappropriate operator actions. Specifics are provided in the contractor's PCP.

After solidification, the liner is closed, surveyed, and either stored onsite in the IRSF or shipped offsite for burial.

The waste streams considered for this processing method include ion exchange resin, sludge’s, filter media and concentrated waste. Concrete or other solidifying agents are used to solidify the waste. Improvements in the current de-watering technology have limited the need for this waste processing method.

11.4.2.2 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 resins, sludges that can be dewatered and filter media. Generally the ion exchange resins and filter media (sludge) are dewatered or solidified in a liner or a high integrity container (HIC).

Liners or High Integrity Containers (HIC’s) are normally shipped to the plant by truck.

The contractor's fill head is positioned on the liner or HIC and waste is transferred into the liner or HIC. Once the liner or HIC is full the waste can be dewatered (when applicable).

Both Station and Contractor personnel are normally involved in the transfer of waste from the liquid radwaste processing systems to a liner or HIC. During waste transfer a representative waste sample can be obtained from an 'in-line' sampler or other methods to obtain a waste sample.

After the liner or HIC is full, the contractor starts the de-watering cycle(s) to either "gross de-water" or "de-water to meet burial limits". During the de-watering process, the liquid is filtered through the liner filters before returning to the liquid radwaste system.

When the de-watering process is complete, the fill head is removed and the liner or HIC is capped. The liner or HIC is placed in temporary storage prior to being moved to the IRSF or shipped offsite to a processor, or burial site. Radiation Protection performs a liner survey to determine container dose rates and total curies.

Ion exchange resins can either be de-watered to meet burial site criteria, or, de-watered, shipped offsite and thermally treated for volume reduction. Once the resin volume is reduced the end product is packaged and shipped to a burial site or sent back to the station for extended storage in the IRSF.

Ion exchange resins can also be thermally destroyed and volume reduced prior to burial. This process is performed off-site by a contractor. The resins that are processed by this method will be transferred to a HIC or other suitable container and shipped to the contractors off-site processing facility and ultimately to burial.

11.4.2.3 Contractor Encapsulation of Waste

Contractor encapsulation services may be utilized for wastes that require classification as stable wastes 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.

The waste to be encapsulated is placed in an approved liner, and then the liner is filled with cement. After cooling, the waste is inspected to ensure adequate encapsulation and the container is capped, secured, surveyed moved to the IRSF or shipped for burial.

11.4.2.4 Dry Active Waste

Dry active waste (DAW) occurs from many sources:

A. Air filters from the off-gas systems, the various plant ventilation systems, and the standby gas treatment systems;

B. Contaminated clothing, tools, and small equipment;

C. Miscellaneous contaminated paper, rags, plastic, metal, wood, etc.;

D. Contaminated concrete chippings and dirt;

E. Cartridge/sock filters from liquid radwaste processing; and

F. Activated plant components; i.e., control blades, local power range monitors (LPRMs), fuel channels, etc.

The dry active 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 are then collected and moved to a controlled-access location where the wastes are packaged in suitable containers. Compressible wastes can be compacted into drums by a hydraulic press to reduce their volume or packaged un-compacted into suitable shipping containers. They can be stored until shipped offsite for burial or to a waste-processing facility. Ventilation is provided to maintain control of contaminated particles when operating packaging equipment or during equipment maintenance and cleanup.

Large radioactive components and/or equipment such as a steam dryer require special handling, storage containers and/or shipping considerations. Handling of such equipment depends upon the radiation level and available storage sites. Suitable procedures for decontamination, shielding, shipment, and storage of such items are developed as necessary.

<u>Waste Water Treatment and/or Sewage Treatment Plant Sludges</u>

Waste Water Treatment and Sewage Treatment Sludge can be transferred to a drying area, processing into containers, or land applied (when applicable). The waste containers are closed and secured and moved to a transport vehicle or a 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.

11.4.2.5 Classification of Radioactive Wet Waste

Radioactive wet wastes which are solidified, dried or dewatered are classified as either Class A, Class B, or Class C to determine the acceptability for disposal and for 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 and 10 CFR 20.

11.4.3 Inspection and Testing

Proper operation of the handling, drumming, solidification, and dewatering 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 as necessary, and equipment that is operated only periodically is tested to assure proper operation of valves and equipment to minimize the chance of a failure or malfunction during operation.

When the contractor's solidification system is used onsite, prior to installing the lid, both the contractor and station personnel perform a visual inspection. The visual inspection, in conjunction with the contractor's PCP requirements, verifies that the processed waste meets the solidification criteria for shipping and burial. If the processed waste does not meet burial criteria, the contractor is required to provide an acceptable resolution.

When using the contractor's dewatering system, verification of an acceptable dewatered product is performed according to the contractor's 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 filled 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 filled liner is not an acceptable product, the contractor is required 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 container dose rates.

11.4.4.1 Dry Active Waste

The DAW is normally stored in a designated storage area near the processing area if the dose rate from the containers is less than or equal to 100 mrem/hr container contact radiation level. The DAW may be stored in various work areas while waiting to be moved to the process area. If the DAW container contact radiation level is greater than 100 mrem/hr, the DAW is normally stored in the radwaste building container storage areas. DAW may also be stored at an interim storage location which is away from the processing area while awaiting shipment to the processor or burial site.

11.4.4.2 Contractor Solidified, Dewatered, or Encapsulated Waste

The contractor solidified, dewatered, encapsulated waste containers are normally shipped when processing has been completed. If storage is required for any of these types of wastes, the containers may be temporarily stored onsite at an interim storage location. If processed waste is required to be stored after the waste is processed off-site, the containers will be shipped back from the processor and stored at an interim storage location in acceptable burial containers per station requirements.

11.4.4.3 Interim Radwaste Storage Facility

The interim radwaste storage facility (IRSF) was constructed to facilitate continued nuclear power station operation should the existing burial facilities shut down.

The IRSF is located inside the protected area. Figure 11.4-2 shows the location of the facility. Figure 11.4-3 shows the general arrangement of the IRSF.

A portion of the existing chemical cleaning facility was used in the construction of the IRSF. The major IRSF areas are the truck bay, control room, equipment room, and storage bay. The truck and storage bays are serviced by a 10-ton crane.

Closed-circuit television (CCTV) cameras are located on the crane; two of them are permanently fixed to observe the grid system coordinates for proper placement of the low level waste (LLW) containers. The other CCTV cameras can be moved to several orientations to facilitate container placement and remote container surveillances.

Storage bay access is limited to access through the normally locked container decontamination area or via the crane through the storage bay/truck bay interface notch.

The IRSF truck bay is used for receiving LLW material for storage. It is also used as a truck loading area for LLW material being shipped to the burial site.

The control room contains the IRSF crane control panel and CCTV monitors. The control room and the equipment room are located adjacent to the IRSF but in the chemical cleaning building. The ventilation system for the IRSF is an extension of the chemical cleaning building ventilation system. The ventilation system exhausts through a prefilter/HEPA filter arrangement and then through the chemical cleaning building exhaust stack. The exhaust discharge is monitored for radioactivity.

11.4.4.4 New Storage Building

The new storage building is designed to provide long term storage and radiation shielding for the old Unit 2 and Unit 3 steam dryers/transportation container assemblies. It is 55 feet long and 30 feet wide and stands 21 feet and 3 inches tall. The new storage building is a reinforced concrete structure with 30 inch thick walls and a 27 inch reinforced concrete roof.

The new storage building is located in the owner controlled area outside of the existing station security protected area (PA), northeast of the 345kV switchyard at coordinates N 9,565 and E 14,425 and approximate elevation 525'.

The new storage building is designed with two compartments, each having a water collection sump. The building is equipped with permanent pipe penetrations to allow for connection to a portable sump pump. The old steam dryers/transportation container assemblies are placed in the new storage building through the roof opening. Removable reinforced concrete roof slabs are used to close each roof opening.

The new storage building is non-safety related and is not required to be designed as Seismic Category I and is located outside the PA. The new storage building is designed for UBC seismic per the design basis document TDBD-DQ-1. It is also designed for the PMF, tornado winds and tornado missiles.

The building is not designed to be occupied and as such there are no doors, lighting, HVAC or electricity.

11.4.5 Shipping of Radioactive Waste

The solid radioactive waste is shipped from the plant to a licensed offsite processor for further processing or directly to a burial site licensed to receive such material. The Unit 2 and 3 core uprates to 2957 MWth implemented in 2001 and 2002 did not significantly increase radioactive waste volumes. Typical solid radioactive waste volumes and isotopic quantities shipped offsite can be obtained from the Station's Annual Radioactive Effluent Release Report as described in Reg Guide 1.21[2-11]

Each burial site requires the shipper to obtain a burial permit before the site will accept solid waste.

Procedures cover the various aspects of loading different types of shipping casks and trucks (or trailers) with specific types of wastes. The marking of containers, the shipping papers, the determination of the radionuclides, and the method for classifying the waste are all described in and performed according to station procedures.

11.4.6 Process Sampling, Control, and Surveying

Process waste is sampled to determine its classification and to determine the quantity of radionuclides (curies of activity) to be shipped in the container. Sampling is performed to determine the appropriate formula for solidification, when necessary. Process sampling, analysis, and container surveying are performed in accordance with station procedures.

11.4.7 Planning

Planning and scheduling are coordinated with shipping contractor's for the movement of solid waste packaged by the station or by a contractor. Planning is essential when the contractor is performing the containerization of the waste (either dewatering or solidification) 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 allowing station operations, refueling, and maintenance activities to be coordinated.

11.4.8 References

1. Dresden Nuclear Power Station - Process Control Program, Latest Revision

2. Letter from N.J. Kalivianakis (CECo) to J.G. Keppler (NRC), dated June 24, 1986, transmitting the Dresden Station Annual Environmental Radiological and Meteorological Operating Report for 1985.

3. Dresden Nuclear Power Station Radioactive Waste and Environmental Monitoring, Annual Report for 1986, dated March 1987.

4. Letter from E.D. Eenigenburg (CECo) to A.B. Davis (NRC), dated March 18, 1989, transmitting Dresden Nuclear Power Station Radioactive Waste Environmental Monitoring, Annual Report for 1987, dated March 1988.

5. Letter from E.D. Eenigenburg (CECo) to U.S. Nuclear Regulatory Commission, dated February 24, 1989, transmitting Dresden Nuclear Power Station Radioactive Waste and Environmental Monitoring Annual Report for 1988, dated March 1989.

6. Letter from E.D. Eenigenburg (CECo) to A.B. Davis (NRC), dated March 19, 1990, transmitting Dresden Nuclear Power Station Radioactive Waste and Environmental Monitoring Annual Report for 1989, dated March 1990.

7. Letter from E.D. Eenigenburg (CECo) to A.B. Davis (NRC), dated March 18, 1991, transmitting Dresden Nuclear Power Station Annual Radiological Environmental Operating Report 1990, dated March 1991.

8. Letter from C.W. Schroeder (CECo) to A.B. Davis (NRC), dated February 20, 1992, transmitting Dresden Nuclear Power Station Semiannual Radiological Effluent Report for July through December, 1991.

9. Dresden Station Semiannual Radiological Effluent Report for January through June, 1991.

10. Letter from E.D. Eenigenburg (CECo) to A.B. Davis (NRC), dated February 28, 1991, transmitting Semiannual Radioactive Report for July through December, 1990 for Dresden Nuclear Power Station.

11. Dresden Nuclear Power Station Semiannual Radioactive Effluent Report for January through June 1990.

11.5 PROCESS AND EFFLUENT RADIOLOGICAL MONITORING AND SAMPLING SYSTEMS

Process and effluent monitoring is provided throughout the radioactive waste systems to assure proper operation and control. Alarms are provided to warn of abnormal conditions so that corrective action can be taken to prevent overflows, equipment malfunction, damage, or process errors. Where necessary, automatic shutdown occurs upon the alarm signal.

The process and effluent radiation monitoring system consists of several individual monitoring systems:

A. Main steam line (MSL) monitoring system;

B. Air ejector off-gas monitoring system;

C. Chimney effluent monitoring system;

D. Reactor building ventilation Geiger-Mueller monitoring system;

E. Reactor building ventilation stack monitoring system;

F. Reactor building closed cooling water system monitoring system;

G. Service water system monitoring system;

H. Liquid radioactive waste discharge monitoring system; and

I. Isolation condenser vent monitoring system.

The process and effluent sampling and monitoring systems addressed are for the liquid and gaseous radioactive waste systems.

Area monitoring (see Section 12.3) is provided as necessary to inform personnel of local radiation conditions. Readouts and high-radiation alarms are placed in the control room and locally as necessary.

See Section 7.3.2 for more information concerning drywell radiation monitors.

11.5.1 Design Objectives

The general design objectives of the radioactive process and effluent radiological monitoring and sampling systems are as follows:

A. To provide continuous indication of radiation levels or releases of radioactive material;

B. To give warning when radiation monitors malfunction; and

C. To provide an alarm when radiation levels or releases exceed preselected levels.

Additional specific performance objectives are stated for each system as they apply.

11.5.1.1 Main Steam Line Monitoring System

The MSL monitoring system is designed to continuously monitor the radiation from the MSLs to permit the prompt indication of gross release of fission products from the fuel to the reactor primary system coolant and subsequently to the turbine-generator system.

The monitoring system automatically initiates a trip and isolation of the mechanical vacuum pump, if activity levels in the MSLs indicate that such action is required. Isolation of the mechanical vacuum pump is achieved by closure of the steam jet air ejector suction valves.

In addition to the MSL monitoring, gross fuel failure is detected by the off-gas monitoring and chimney effluent monitoring systems. These systems are described in Sections 11.5.2.2 and 11.5.2.3.

11.5.1.2 Air Ejector Off-Gas Monitoring System

The air ejector off-gas monitors are designed to provide the following functions:

A. Continuously monitor, indicate, and record the radioactivity level of the effluent gases removed from the main condenser by the air ejector off-gas system;

B. Alarm in the control room on high-radiation level in the off-gas system; and

C. Initiate closure (after a time delay) of the off-gas system isolation valve when the radiation level in the off-gas system exceeds the prescribed limit.

11.5.1.3 Chimney Effluent Monitoring System

In order that the operator can be continuously aware of activity being released from the plant, the chimney effluent monitoring system is designed to continuously monitor, indicate, and record the radioactivity level of the effluent gases being discharged from the chimney to the atmosphere. The chimney discharge includes particulate, iodine, and noble gases released during both normal operating conditions and for the worst postulated accident release rate. The system is also designed to alarm in the main control room if the chimney effluent radioactivity release exceeds the prescribed limit. The particulates and iodine are also analyzed in the station laboratory using removable filter and activated charcoal canister samples.

11.5.1.4 Reactor Building Ventilation Geiger-Mueller Monitoring System

The reactor building ventilation monitoring system is designed to provide automatic isolation of the secondary containment when the concentration of radioactive materials in the ventilation exhaust exceeds prescribed levels. Secondary containment isolation is addressed further in Section 6.2.3. The reactor building ventilation stack radiation monitor has only an alarm function. The radiation monitor in the reactor building ventilation ducts and the intertied radiation monitors at the spent fuel pool have isolation and alarm functions.

11.5.1.5 Reactor Building Ventilation Stack Monitoring System

In order that the operator can be continuously aware of activity being released from the plant, the reactor building ventilation stack monitoring system is designed to continuously monitor, indicate, and record the radioactivity level of the exhaust air discharged from the reactor building ventilation stack to the atmosphere. The reactor building ventilation stack discharge includes particulate, iodine, and noble gases released during both normal operating conditions and for the worst postulated accident release rate. The system is also designed to alarm in the control room if the ventilation stack effluent radioactivity level exceeds the prescribed limit. The particulates and iodine are also analyzed in the station laboratory using removable filter and charcoal canister samples.

11.5.1.6 Reactor Building Closed Cooling Water System Monitoring System

In order that the operator can be continuously aware of activity within the reactor building closed cooling water (RBCCW) system, this system has been designed to continuously monitor, indicate, and record the radioactivity concentration levels and to alarm in the main control room if the RBCCW system radioactivity level exceeds a prescribed setpoint.

11.5.1.7 Service Water System Monitoring System

In order that the operator can be continuously aware of activity being released from the plant through the service water system, this monitoring system is designed to continuously monitor, indicate, and record the radioactivity levels within the service water system and to alarm in the main control room if radiation levels approach limitation for station discharge.

11.5.1.8 Liquid Radioactive Waste Discharge Monitoring System

The liquid radioactive waste discharge monitor continuously measures, indicates, and records the radioactivity concentration levels during a discharge to the river. The monitor alarms in both control rooms (see Drawing M-347B) when the radiation level approaches limitation for station discharge. Requirements for continuing liquid discharge without the monitor are specified in the ODCM.

11.5.1.9 Isolation Condenser Vent Monitoring System

The isolation condenser vent monitor is designed to detect and warn the operator of a tube leak. To meet the design requirement, the shell-side vent monitor records the radioactivity of the vent effluent and alarms in the main control room if a preset level is exceeded.

11.5.1.10 Onsite/Offsite Environmental Monitoring

Onsite and offsite monitoring stations which measure the gamma radiation level and collect airborne particulates for periodic analysis are provided to confirm that releases of airborne radioactive materials have been controlled within the limits established by license or 10 CFR 20 and the design criteria specified in 10 CFR 50, Appendix I.

11.5.1.11 Linear Monitoring (Flux Tilt Monitor) System

The linear monitoring (flux tilt monitor) system is designed to assist in determining the location of leaking fuel elements in the reactor core.

11.5.1.12 High Radiation Sampling System

The high radiation sampling system (HRSS) is designed to provide contingency sample points for the reactor coolant and associated reactor waste streams. Sampling these streams enables the operator to assess the extent of reactor coolant leakage throughout the station during post-accident operations.

11.5.2 System Description

The MSL monitoring system provides indication, alarm, and isolation functions. The air ejector monitors and the reactor building ventilation monitors also perform an automatic isolation or closure function. The following systems, which do not perform an automatic isolation function, are intended to provide an information and alarm function: the chimney effluent monitor, process liquid monitors, isolation condenser monitor, and reactor building ventilation stack monitor. Table 11.5-1 presents the parameters for the radiation monitoring system equipment.

The systems which provide an automatic isolation function can be classified into two radiological source categories. The first category is the reactor building monitors, which are intended to detect abnormal amounts of radioactive material in the reactor building air which could be released to the environment untreated if normal ventilation were not terminated. Thus this system isolates secondary containment. The reactor building monitors are addressed further in Section 6.2.3.

The second category of automatic isolation systems includes the MSL monitors and air ejector monitors. Both of these systems sample essentially the same potential source of abnormal amounts of radioactive material, i.e., gaseous fission products released from the reactor core. The steam line monitors are intended to provide rapid detection of gross fuel failure.

The Mechanical Vacuum Pump Trip Instrumentation initiates a trip of the main condenser mechanical vacuum pump breaker following events in which main steam line radiation exceeds predetermined values. Tripping the mechanical vacuum pump limits the offsite and control room doses in the event of a control rod drop accident (CRDA).

The Mechanical Vacuum Pump Trip Instrumentation includes detectors, monitors, and relays that are necessary to cause initiation of a mechanical vacuum pump trip. The channels include electronic equipment that compares measured input signals with pre-established setpoints. When the setpoint is exceeded, the channel output relay actuates, which then outputs an isolation signal to the mechanical vacuum pump trip logic.

The trip logic consists of two independent trip systems, with two channels of Main Steam Line Radiation–High in each trip system. Each trip system is a one-out-of-two logic for this Function. Thus, either channel of Main Steam Line Radiation–High in each trip system is needed to trip a trip system. The outputs of the channels in a trip system are combined in a one-out-of-two taken twice logic so that both trip systems must trip to result in a pump trip signal.

There is one mechanical vacuum pump breaker associated with this Function.

The Mechanical Vacuum Pump Trip Instrumentation is assumed in the safety analysis for the CRDA. The Mechanical Vacuum Pump Trip Instrumentation initiates a trip of the mechanical vacuum pump to limit offsite and control room doses resulting from fuel cladding failure in a CRDA.

The air ejector monitors provide a dual function. One is an alarm function in the control room when the high-radiation setpoint is exceeded; the other is an automatic isolation function (after a 15-minute delay) when the high-high radiation setpoint is reached. This latter function, with the associated holdup prior to actual release of off-gas to the atmosphere, assures that the normal operating limits of 10 CFR 20 are not exceeded and, in addition, provides a backup isolation function to the steam line monitors to further assure that the fission products from a gross fuel failure are retained in the plant.

All monitors are capable of problem self-indication, i.e., they give an alarm when downscaled or deenergized. 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 or radioactive check sources.

All monitoring systems provide continuous indication in the control room. 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 preestablished limits are approached.

11.5.2.1 Main Steam Line Monitoring System

The main steam line monitoring system (see Figures 11.5-1 and 11.5-2) incorporates four channels of instrumentation for the group of four MSLs with each channel consisting of the following components:

A. A gamma-sensitive ionization chamber;

B. A dc log radiation monitor complete with fail-safe operational alarms, appropriate high- and low-voltage power supplies, and control and alarm trip contacts; and

C. A continuous strip chart recorder.

Each channel is continuously indicated and recorded in the main control room. Each channel also alarms in the main control room.

The detection points are immediately downstream of the outboard isolation valves in the primary containment structure. A channel reading of 1.5 times normal background level with hydrogen addition provides an alarm on any of the four channels. A high channel reading will trip and isolate the mechanical vacuum pump analytical limit: 8000 mr/hr.

The main steam line monitors are located such that they are in the radiation field of the four MSLs. The range and sensitivity of the monitors have been chosen such that the monitors are capable of detecting increases of radiation in the environment near the MSLs due to the activity release following a gross fuel failure.

A gross fuel failure would result in a significant increase in the MSL radiation levels. The redundancy of detector channels and the general location of the detectors in the MSL radiation field assure the reliability of the system.

The Channel A MSL radiation monitors, A and C, are powered from the reactor protection system (RPS) bus. The Channel B MSL radiation monitors, B and D, are powered from the essential service system (ESS) bus.

11.5.2.2 Air Ejector Off-Gas Monitoring System

The air ejector off-gas monitoring system (see Figure 11.5-3) incorporates two identical channels of instrumentation, each consisting of the following components:

A. A gamma-sensitive ionization chamber;

B. A dc log radiation monitor complete with fail-safe operational alarms, appropriate high- and low-voltage power supplies, and control and alarm trip contacts; and

C. A shared two-pen continuous strip chart recorder, complete with alarm-trip contacts.

The noncondensible gases are drawn from the main condenser by the steam jet air ejectors (which are arranged in two stages) then routed through a preheater to a recombiner which catalytically combines the hydrogen and oxygen gases radiolytically produced in the reactor core. Hydrogen added to the feedwater and oxygen added to the off-gas system are also combined in the recombiner. The remaining noncondensible gases exiting from the recombiner pass through moisture separators, then into a holdup line. The gases are monitored by two detectors which are located in separate shielded housings on the holdup line. The radioactivity level is indicated and recorded continuously in the control room. The off-gas system is described in more detail in Section 11.3.

The radioactivity levels of N-16 and O-19 in the MSLs are normally relatively high, but they quickly decay due to their short half-lives. Therefore, to obtain a more accurate indication of the activity levels of radioisotopes which affect the gaseous discharge limits for the chimney release point, the air ejector off-gas sample is monitored after a transportation time delay. Due to the position of the detectors at the inlet to the holdup line, this delay is at least 2 minutes when the recombiner is bypassed, or about 30 minutes when the recombiner is used. If the high-radiation setpoint is reached, a signal is initiated to close the isolation valve in the off-gas line after a time delay of 15 minutes. The time delay allows the operator to evaluate the data and prevent an unwarranted isolation valve closure or reactor shutdown if the signal is false. The holdup pipe volume in the off-gas line after the sample point and before the isolation valve allows a delay between the time of the high-radiation signal and the release to the chimney discharge point. The delay time is approximately 30 minutes with the recombiner and the charcoal adsorber beds bypassed and approximately 6 hours with the recombiner operating alone or with the recombiner and charcoal adsorbers both in operation (which is the normal case). A downscale trip gives warning of instrumentation malfunction. The two channels are arranged so that they operate independently of each other. The logic is arranged so that a closure of the off-gas line is initiated by two upscale or one upscale trip signals and one downscale trip signal.

A third channel using a linear count rate meter is provided to give a more sensitive indication when flux tilting is used to assist in locating leaking fuel assemblies. Currently, flux tilting is not being used.

Provisions are made for collecting grab samples of steam jet air ejector off-gas for more sensitive and quantitative laboratory analysis.

The redundancy incorporated into the monitoring system provides assurance that abnormal releases of radioactive material are detected, annunciated, and isolated.

To calibrate the monitors, the results of analysis of a grab sample 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.

11.5.2.3 Chimney Effluent Gas Monitoring System

The chimney effluent monitor (see Figure 11.5-4 and Drawing M-422, Sheet 2) consists of a single multiple-range system particulate, iodine, and noble gas (SPING) monitor and a backup system which incorporates two channels of instrumentation.

The release rates (μCi/s) from the 310-foot chimney and the reactor building ventilation stack are calculated from the instrument readouts (counts per second) and totalled by the operator to assure compliance with gaseous release rate limits for the plant. The isotopic quantities are reported as required by the ODCM.

The chimney flow consists of air ejector off-gas (approximately 20 ft^3/min with the recombiner operating; 150 ft^3/min without the recombiner operating) mixed with ventilation air (approximately 430,910 ft^3/min)(see Section 11.3.3.1.2 for additional information). A representative sample is drawn continuously from the chimney through an isokinetic sample probe located at two-thirds of the chimney height. The placement of the probe is in accordance with good engineering design practice, i.e., probe height is at least 10 times the chimney diameter. The SPING monitor and its backup system use the same isokinetic probe.

11.5.2.3.1 SPING Monitoring Instrumentation

The SPING monitor is computerized instrumentation (see Figure 11.5-4 and Drawing M-422, Sheet 2) with sufficient range to accurately monitor the chimney effluent for the worst postulated accident releases as well as for normal operating conditions. The installation of the SPING monitor is a result of the events occurring at Three Mile Island (TMI) on March 28, 1979.

The SPING monitor is a microprocessor-based radiation detection system. The programs (software) which control the system are stored in read-only memory (ROM) and therefore are fixed. Only the parameters of the system can be varied. The microcomputer performs the tasks of data acquisition, history file management, operational status check, and alarm determination. The microcomputer communicates with the operator through a terminal in the control room.

The isokinetic probe, a four-tube stainless steel assembly, is designed to obtain a representative sample over the chimney cross-sectional flow. The SPING monitor sampling system draws the sample from the isokinetic probe in the chimney using a constant-flow pumping system.

After being drawn from the chimney through the isokinetic sample probe and then into the monitor, the sample goes through a filter paper on which any particulates are deposited, then through a charcoal cartridge which traps the iodines, then into a gas chamber for low- and medium-range noble gas measurement before returning to the 310-foot chimney. The sample only passes through the high-range noble gas detector once the count rate on the medium range detector exceeds a preset limit. At this time the sample flow bypasses the low- and mid-range detectors. The sample flow proceeds through the Victoreen particulate sample filter and through the high-range noble gas detector before returning to the 310-foot chimney.

The particulate filter is monitored by a beta scintillation detector from one side and a solid-state alpha detector from the other side.

The charcoal cartridge is monitored by a 2-inch x 2-inch NaI(Tl) gamma scintillation detector. A single-channel analyzer (SCA) which is calibrated to the 364-keV energy level of I-131 is used. An additional SCA which is calibrated to measure energy levels above the I-131 energy level provides a measure of the background in the iodine window. This measurement is subtracted to compensate for the effects of a fluctuating background.

Noble gas measurement is performed by low-range, medium-range, and high-range detectors viewing a sample volume. The low- and medium-range detectors view the same volume. The high-range detector views a separate sample volume contained in 1-inch diameter stainless steel tubing. Low-range noble gas monitoring is performed by a beta scintillation detector. Medium-range noble gas monitoring is performed by an energy-compensated GM detector whose output is proportional to the gamma emission from the sample. High-range noble gas monitoring is performed by an energy-compensated GM detector monitoring a section of 1-inch diameter stainless steel tubing. The detector output is proportional to the gamma emission of the sample.

There are two background subtracting detectors on the SPING. Both of these detectors are energy compensated GM detectors. One of these is shielded, one is not. These detectors may be used to compensate for the effects of background.

Check sources are provided for some channels as listed in Table 11.5-2.

The basic unit of all calculations on data within the SPING monitor is counts per minute. These individual 1-minute values are instantaneous values. Any background subtraction sources specified are calculated and subtracted from the count rate. The result is the net counts per minute and the data from these individual 1-minute intervals are used in the dose assessment model calculations. History files are maintained on each channel for three time intervals: 23 ten-minute intervals, 24 one-hour intervals, and 24 one-day intervals.

The data for any maintained interval are the average of the accumulated data in that interval. Abnormal status (but not alarm conditions) of the instrument for any interval is stored and indicated.

The SPING monitor and the control terminal in the control room continuously exchange messages and/or data via a communications line. The operator can view the radiation level on any or all channels, retrieve history files, set or reset the pump, flush the instrument, synchronize the clock, and activate check sources on specified channel(s) via the control terminal.

The SPING monitor is provided with a self-contained battery backup power system.

11.5.2.3.2 <u>Backup System</u>

The backup monitoring system (see Figure 11.5-4 and Drawing M-422, Sheet 2) incorporates two channels of instrumentation, each of which includes:

A. An isokinetic sampling probe shared by both channels and the SPING monitor;

B. A particulate and iodine filter assembly shared by both channels;

C. A shielded radiation sampler;

D. A sample pumping assembly shared by both channels;

E. A scintillation crystal-photomultiplier counter;

F. A preamplifier;

G. A log count rate meter in the main control room, range 10^{-1} to 10^{6} cps, with one downscale and two upscale alarms; and

H. A two-pen recorder in the main control room shared by both instrument channels.

After being drawn from the chimney through the isokinetic sample probe by the sample pumping assembly, the sample flows through a two-part assembly: one, an absolute particulate filter, and the other, an activated charcoal pack for iodine collection. The sample continues into the two shielded radioactive gas detection samplers with fixed holdup volumes and then returns to the chimney via the flow control unit. The effluent is monitored in each sampler by a detector. The in-line particulate filters are periodically removed for detailed radiological quantitative analyses.

A check source is included with each of the two individual monitoring channels. The source is located externally to a cylinder, which is internal to the chamber shield and adjacent to the detector. The source is normally hidden from the view of the detector by the shielding; however, by rotating the cylinder the source comes into view, allowing verification of detector operation. The cylinder is rotated by a small electric motor, which is actuated from the control room.

11.5.2.4 Reactor Building Ventilation Geiger Mueller Monitoring System

The reactor building ventilation GM monitoring system continuously monitors the reactor building air in two locations: in the ventilation exhaust plenum and on the refueling floor near the spent fuel pool. The GM monitor in each unit's ventilation exhaust plenum monitors the reactor building ventilation exhaust air and when high radioactivity is detected, the secondary containment isolation is initiated. initiated. The GM monitors located on the refueling floor (one each side of the spent fuel storage pool) monitor the environment around the spent fuel storage pool and when high radioactivity is detected, secondary containment is isolated. See Section 6.2.3 for further discussion of secondary containment isolation. There are two channels of instrumentation in each location. Each channel includes the following components:

A. A GM tube (sensor and converter);

B. An indicator and trip unit;

C. A low- and a high-voltage power supply;

D. An auxiliary trip unit; and

E. A shared recorder.

A high-radiation level trip on any monitor or downscale trips on both monitors in a given location initiates secondary containment isolation. The reactor building ventilation GM monitoring system is completely redundant, i.e., it meets the single failure criterion for active components.

The range of the ventilation duct GM monitors is 10^{-2} to 10^{2} mrem/hr. The range of the refueling floor GM monitors is 1 to 10^{6} mrem/hr. All channels are indicated, alarmed, and recorded in the main control room. The ventilation duct monitors share a two-pen recorder. The refueling floor monitors share a multipoint recorder.

This equipment is designed for a mean time between failure of 1 year per point or channel. This includes the power supply and other components listed above.

Power is supplied from the 120-V RPS buses. For each pair of monitors, one channel is powered from one RPS bus and the other channel from the other RPS bus.

This power is very stable, but in the event of a power failure, a downscale alarm occurs in the control room to inform the operator. Should one of the power supplies fail such that no downscale alarm were annunciated, the one remaining power supply and its associated monitors would still give positive indication and a one-out-two trip for secondary containment isolation.

The reactor building ventilation monitoring system is set to isolate secondary containment (see Section 6.2.3) upon detection of a refueling accident. A refueling accident offers the greatest potential for radioactive release via the reactor building ventilation exhaust. The high-level setpoint is chosen sufficiently above refueling operations background radiation level to avoid spurious trips but low enough to trip from the radiation level resulting from the design basis refueling accident.

The refueling accident is evaluated in Section 15.7.3. The reactor building ventilation monitoring system is effective in preventing radioactive release in excess of 10 CFR 50.67 limitations.

The sensitivity, accuracy, and range capability of the reactor building ventilation GM monitors permit the monitors 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.

Failure of a monitor which results in a downscale trip will not prevent isolation of the secondary containment (see Section 6.2.3) when the other monitor detects a high-radiation level.

The capability to calibrate and test the monitors is provided by built-in, electronic calibration equipment.

11.5.2.5 Reactor Building Ventilation Stack Monitoring System

The ventilation stack monitor (see Drawing M-422, Sheet 1) is a single, multirange SPING monitor identical to that described for the chimney in Section 11.5.2.3. As a backup to the iodine and particulate sampling capability of the SPING monitor, each unit is equipped with a sampler which extracts a portion of the reactor building ventilation exhaust for sampling. Each sample skid consists of two redundant pumps, a flowmeter, a vacuum gauge, an iodine sample cartridge, and a particulate sample cartridge. Low sample flow is alarmed in the main control room.

The ventilation stack monitor has sufficient range to monitor the reactor building ventilation air exhausting from the stack under normal operating conditions and for the worst postulated accident release rate. The monitor supplements the indication provided by the reactor building ventilation duct monitoring system for each unit. Secondary containment isolation (see Section 6.2.3) is manually initiated by the operator if the automatic functions of the reactor building ventilation duct monitoring system fail. Adequate backup is also provided for the SPING monitor by the reactor building ventilation exhaust samplers. These samples are routinely removed and counted in the chemistry lab area. The results of SPING monitor and sample testing are reported as required by the ODCM.[1]

11.5.2.6 Reactor Building Closed Cooling Water System Monitoring System

The reactor building closed cooling water system is primarily utilized to provide cooling for potentially contaminated systems such as the reactor water cleanup (RWCU) system, reactor concrete shielding, non-regenerative heat exchangers, and recirculation pumps. The system contains activity due to design inleakage from heat exchangers or other components which contain radioactive water. Changes in the normal radiation level signify an increase in activity concentration in the system.

The process liquid monitor (see Figure 11.5-5) incorporates one channel of instrumentation consisting of the following components:

A. A scintillation crystal-photomultiplier counter;

B. A log count rate meter;

C. A continuous strip chart recorder;

D. Trip auxiliaries; and

E. A control room alarm.

At the mounting installation, a scintillation detector is located in a shielded sampler which is positioned on a vertical section of the process liquid piping. A vertical section of piping is used to minimize background radiation due to plate out.

Trip circuits are also included to indicate abnormal concentrations of fission and radioactive corrosion products so that action can be taken to prevent the accidental transfer of highly radioactive materials. The readout consists of a seven decade meter display. This system shares a common two-pen strip chart recorder with the service water radiation monitoring system.

The reactor building closed cooling water system radiation monitor provides seven- decade monitoring with the lowest decade established below the normal background of the system. The high-radiation alarm setpoint is based upon the normal, full power, operating background but is considerably less than the operating limit. Table 11.5-1 lists the specific data pertaining to the sensitivities and accuracies of the monitoring equipment.

11.5.2.7 Service Water System Monitoring System

The service water (SW) and the containment cooling service water (CCSW) provide cooling to numerous plant systems via heat exchangers (see Sections 9.2.1 for CCSW and 9.2.2 for SW). The Unit 2 and Unit 3 service water system discharge points for SW and CCSW are monitored for radioactivity. High radioactivity detected in the normally non-radioactive SW or CCSW discharges would indicate leakage into the SW or CCSW from one or more of the systems they service.

The service water radiation monitors are a GA-ESI skids (See Drawing M-3496 for Unit 2 and Drawing M-3486 for Unit 3). Each service water system monitor incorporates one channel of instrumentation consisting of the following components:

A. An NaI(T1) gamma scintillation process detector;

B. High-radiation annunciation in the control room;

C. Monitor failure annunciation in the control room;

D. Safety Parameter Display System (SPDS) indication; and

E. A continuous strip chart recorder.

The water sample is taken from the service water standpipe and provided to the process radiation detector. The Unit 2 liquid monitor assembly consists of a skid mounted RD-53A-50PD Liquid Sampler, a RM-2000 Microprocessor Assembly, a Customer Interface Junction Box Assembly, the Power Control Center Assembly, and a pump and plumbing for the sample. The Unit 3 liquid monitor assembly consists of a skid mounted RD-53A-50PD Liquid Sampler, a Customer Interface Junction Box Assembly, the Power Control Center Assembly, a pump and sample. The RM-2000 receives and converts both Unit 2 and Unit 3 detector signals to analog and digital data for display and processing by the radiation monitoring system equipment. The RM-2000 controls monitor function and alarm relays and maintains monitor and channel databases information containing monitor-operating parameters. If the High-radiation setpoint is exceeded, the high-radiation annunciator in the control room activates the SPDS “RAD RELEASE” box turns red.

The radiation monitor provides six decades of monitoring. The lowest decade is established below the normal background of the monitored system. The high-radiation alarm setpoint is based upon the normal, full-power, operating background but is considerably less than the upper range of the detector. Table 11.5-1 lists the specific data pertaining to the range of the monitoring equipment.

Other monitor instrumentation in the control room includes an RM-2300 operator interface module, annunciator for monitor failure, and a continuous strip chart recorder for providing an historical record.

The process sample flow begins at the scoop tube inserted into the service water standpipe and ends with a return connection to the standpipe (see Drawings M-3496 and M-3486). The liquid radiation monitor skid sample pump is used to move the process sample water from the scoop tube through the process radiation detector bowl, through the flow switch and then return it to the standpipe. The GA-ESI liquid rad monitor provides a check source and purge features that can be programmed for automatic activation. The check source is programmed to activate once every twenty-four hours to test the detector if the monitor detects a detector failure it will provide an alarm. The check source function can be manually activated at the RM-2000 or RM-2300 panel in the control room. The purge feature is programmed to be a manual function that can be manually activated at the RM-2000 or RM-2300 in the control room. Purge feature is used to flush the skid or the sample probe using the Domestic Water supply.

Automatic backwash feature has been provided to backwash the sample probe when a low flow condition is sensed. If a low flow condition exists for greater than two-minutes a backwash is automatically initiated and flushes the sample probe and skid for two minutes. After the completion of the backwash if flow is not re-established above the predefined setpoint the service water radiation monitor will trip after a three and half minute delay. During the two-minute backwash the monitor is not sampling the service water and is not functional. The purge cycle is less than the 2-hour time delay authorized by the Offsite Dose Calculation Manual (ODCM) section 12.2.1 Radioactive Liquid Effluent Monitoring Instrumentation before entry into the associated Conditions and Required Actions must be initiated.

RM-2000 Monitor provides the following Operating Alarms:

Monitor Alarm Conditions:

a. Database Incomplete
b. Battery Low or Fail
c. RAM Test Fail
d. Checksum Fail
e. Board Dead

Channel Alarm Conditions:

a. Out of Service
b. Checksource Failed
c. No Pulses
d. Loss of Flow
e. Over Range
f. Data Unavailable
g. Channel Not Calibrated
h. Degraded High Voltage

Manually operated grab sample valves are provided for obtaining samples. Samples may be required when the radiation monitor alarms on high-radiation or at times when the monitor is inoperable.

11.5.2.8 Liquid Radioactive Waste Discharge Monitoring System

Liquid radioactive waste is occasionally discharged to the environment. Liquid releases are made on a batch basis from the waste surge tank, floor drain sample tank or portable waste treatment system tank which has been isolated so that no additional water may be inadvertently discharged. The batch is processed below the maximum concentration given in 10 CFR 20 and discharged into the circulating water leaving the plant. The tank is recirculated to assure a representative sample and then analyzed for gamma activity and H-3, Fe-55, Sr-89 and Sr-90 activity concentrations are estimated based on the gamma activity to determine a discharge rate to ensure that 10 CFR 20 limits are not exceeded. Further dilution occurs when the water leaves the discharge canal and enters the river.

The radwaste discharge monitor (see Figure 11.5-7 and Drawing M-3478) is an offline sampling type monitor. When a discharge is made, the radwaste operator valves in the monitor and energizes the instrumentation. Requirements for continuing liquid discharge without the radwaste discharge monitor are specified in the ODCM.

The process liquid monitor incorporates one channel of instrumentation consisting of the following components:

A. An NaI(Tl) gamma scintillation process detector;

B. An NaI (Tl) gamma scintillation background detector;

C. A float-type flow indicator/switch;

D. High-radiation annunciation in the control room;

E. Low receiver tank level annunciation in the radwaste control room;

F. High receiver tank level annunciation in the radwaste control room;

G. Low-flow annunciation in the radwaste control room;

H. Monitor failure annunciation in the radwaste control room;

I. Chart recorder; and

J. A grab sample station with an automatically operated solenoid valve.

The water sample is taken from the discharge to the river line, fed through the process detector, a grab sample valve, and into a receiver tank. Tank level is maintained between high and low setpoints by a discharge pump which feeds the sample back into the discharge line.

The process and background monitor radiation signals are fed through local interface boxes (IB-2) to a DAM in the radwaste control room. The DAM accumulates the count data for each detector and calculates the count rate through background subtraction. The DAM is operated from a control terminal in the main control room which feeds computer information to a chart recorder. The DAM provides local indication and alarms and radwaste control room panel annunciation. Both the discharge pump and the grab sample valve have local, manual control.

The system, which has been designed to maintain a constant flow and a preset band of tank level, provides annunciation in the radwaste control room to alert the operator of any deviations from the normal operating status. A low-level annunciation is received if the discharge pump continues to run below the low setpoint. Low-flow annunciation will be received if the sample flow drops below the prescribed setpoint. If the high-radiation setpoint is exceeded, annunciation is received and a grab sample is automatically obtained by a solenoid valve which opens for a set period of time to allow part of the sample flow leaving the detector to flow into a container. In addition, a loss of monitor power, loss of a monitor radiation signal, or loss of a high-radiation signal results in a monitor failure annunciation.

The procedures for liquid radioactive waste discharge to the river, along with the monitor failure, low flow, high and low receiver-tank level, and high-radiation annunciation in the radwaste control room, will assure that the liquid radioactive waste discharges are monitored properly. This assures that the activity in the water leaving the discharge canal and entering the river is within federal limits for nonoccupational use.

The use of applicable procedures assures that the valve lineup for the discharge of liquid radwaste is correct. After initiating the discharge the lineup can be further verified by noting the level drop in the waste surge tank, floor drain sample tank or portable waste treatment system tank.

11.5.2.9 Isolation Condenser Vent Monitoring System

Monitoring of gross radiation is provided at the isolation condenser vent line by two channels of instrumentation. Each channel is powered from one of the RPS buses. The amplifiers associated with the detectors are logarithmic and have ranges of 10^{-2} to 10^{3} mrem/hr and 1 to 10^{5} mrem/hr, respectively. The detectors are identical to those used for the area radiation monitoring system addressed in Section 12.3. The output of each monitor is indicated and recorded in the control room. When the gross activity in the condenser vent line reaches a preset level (indicating tube leaks in the isolation condenser) an alarm is sounded. Failure of the monitoring equipment, either upscale or downscale, is annunciated.

The isolation condenser vent monitor is of sufficient range and sensitivity to detect radiation increases in the condenser which indicate a tube leak. The alarm level is set sufficiently above background to be representative of a leak. Since the background is continuously recorded, any abnormal increase is noted by the operator. Following an alarm, the operator may isolate the condenser.

11.5.2.10 Process Liquid Sampling System

The process liquid sampling system is provided in three parts at three locations. The process liquid sampling system is addressed in more detail in Section 11.2.

11.5.2.11 Process Gaseous Sampling System

The process gaseous sampling system is addressed in Section 11.3. Hydrogen analyzers are a part of the control instrumentation monitoring the hydrogen concentration of the off-gas downstream of the off-gas condenser. The hydrogen addition requires a corresponding oxygen addition such that there is minimal, if any, residual hydrogen in the off-gas downstream of the hydrogen recombiners. Oxygen analyzers are also used in monitoring and control of the gaseous systems. The hydrogen water chemistry system is addressed further in Section 5.4. Drawings M-178, M-179, M-421 depict the process sampling in the off-gas system. Drawings M-43, Sheet 5 and M-371, Sheet 5 show the off-gas hydrogen and oxygen analyzer flow and sample conditioning.

11.5.2.12 High Radiation Sampling System

The HRSS is used to sample a few streams during normal operation and numerous additional points following an accident. The HRSS system is addressed in more detail in Section 9.3.2.

11.5.2.13 Linear Monitoring Subsystem (Flux Tilt Monitor)

The air ejector off-gas radiation monitor contains a third channel which uses a linear count rate meter. This channel is provided to give a more sensitive indication when flux tilting is used to assist in locating leaking fuel assemblies. Currently, flux tilting is not being used (see Section 11.5.2.2).

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.

11.5.3.1 Liquid Effluent Monitoring and Sampling

The liquid radwaste discharge stream, the Unit 2 service water system, and the Unit 3 service water system are all monitored by an Eberline radiation monitor and sampler. The Unit 2 reactor building closed cooling water system and the Unit 3 reactor building closed cooling water system are monitored by the original GE-installed monitoring system. Details of the monitoring instruments are addressed in Section 11.5.2. Additional details for the sampling are addressed in Section 11.2. For the discharge of liquid radwaste from the station, the tank of liquid for discharge to the river must be sampled and analyzed such that a discharge flowrate can be determined before discharge of the liquid begins. The

ODCM defines sampling and sampling frequency dependent on operation of the discharge monitor.

11.5.3.2 Gaseous Effluent Monitoring and Sampling

The two continuous gaseous release points (the 310-foot chimney and the reactor building ventilation stack) are monitored continuously by SPING radiation monitors. The monitors are addressed in more detail in Section 11.5.2. Additional details of sampling are addressed in Section 11.3. The requirements for sampling and laboratory analysis are addressed in the plant Technical Specifications. The SPING monitors provide activity information and provide an alarm when a preset activity level is reached.

11.5.3.3 Environmental Radiation Monitoring and Radiological Sampling Program

The environmental radiation monitoring and radiological sampling program is conducted in accordance with the requirements of the ODCM. Analyses of the results are determined as required by the ODCM. The results are reported annually to the NRC in the Annual Radiological Environmental Operating Report.

11.5.3.4 Response to Draft of the AEC-Proposed General Design Criteria

For the details of the response to AEC-Proposed General Design Criteria, see Section 3.1. In addition to the discussion in Section 3.1, the gaseous and liquid release paths are monitored during normal operation including any anticipated operational occurrences.

11.5.4 Process Monitoring and Sampling

11.5.4.1 Process Liquid Monitoring and Sampling

The liquid process monitoring and sampling system is addressed in Section 11.2 for the liquid radwaste system, the service water system, and the reactor building closed cooling water system.

11.5.4.2 Process Gaseous Monitoring and Sampling

The crossties between the reactor building ventilation system, the SBGTS system, and the refueling floor monitors is addressed in Sections 11.3, 11.5.2.4, and 6.2.3. The MSL monitors and/or the steam jet air ejector radiation monitors provide the isolation function for the 310-foot chimney releases from the off-gas system, the vacuum pump system, and the turbine steam gland seal system.

11.5.4.3 Response to Draft of the AEC-Proposed General Design Criteria

For details of the response to the AEC-Proposed General Design Criteria, see Section 3.1. In addition to the discussion in Section 3.1, the effluent samplers and monitors for the two gaseous release points provide only an alarm function when the appropriate radiation alarm signal is given.

11.5.5 References

1. Dresden Nuclear Power Station Offsite Dose Calculation Manual, Revision O.A, April 1991.

Table 11.5-1

RADIATION MONITORING SYSTEMS EQUIPMENT PARAMETERS

Radiation Monitoring System	General Monitor Type	Number of Channels	Detector Type	Range	Indicator and Alarm Location	Radiation Alarm Types	Equipment Alarm Types	Recorder		Sampling	Logic
								Type	Location		
Main Steam	Area	4	Ionization chamber	0 to 10^6 mrem/hr	Main control room	High and high-high	Downscale	Dual pen	Main control room	In-line	mechanical vacuum pump trip
Air-Ejector (Off-Gas)	Radioactive gas	2	Ionization chamber	0 to 10^6 mrem/hr	Main control room	High and high-high	Downscale	Dual pen	Main control room	In-line	Off-gas isolation (after 15-min delay)

Table 11.5-1 (Continued)

RADIATION MONITORING SYSTEMS EQUIPMENT PARAMETERS

Radiation Monitoring System	General Monitor Type	Number of Channels	Detector Type	Range	Indicator and Alarm Location	Radiation Alarm Types	Equipment Alarm Types	Recorder		Sampling	Logic
								Type	Location		
Effluent Gas (SPING)	Air particulate	1	Solid-state alpha and	Up to 10^5 μCi/cc for high noble gas	Main control room and locally	High, high-high, and trend	Fail	Computer memory to strip chart type for noble gases	Local and main control room	Offline	
		1	Scintillation								
(one for main chimney, one for reactor building ventilation stack)	Iodine	1	Scintillation								
	Noble gas	1	Scintillation (low)								
		1	G-M tube (medium)								
		1	G-M tube (high)								
	Area	1	G-M tube								

Table 11.5-1 (Continued)

RADIATION MONITORING SYSTEMS EQUIPMENT PARAMETERS

Radiation Monitoring System	General Monitor Type	Number of Channels	Detector Type	Range	Indicator and Alarm Location	Radiation Alarm Types	Equipment Alarm Types	Recorder		Sampling	Logic
								Type	Location		
Effluent Gas (Chimney GE System)	Noble gas	2	Scintillation	10^{-1} to 10^{6} cps	Main control room	High	Downscale	Dual pen	Main control room	Offline	
	Air particulate	1	None								
	Iodine	1	None								
Reactor Building Ventilation Exhaust Plenum	Area	2	G-M tube	10^{-2} to 10^{2} mrem/hr	Main control room	High and high-high	Downscale	Dual pen	Main control room	In-line	SBGTS initiation and reactor building ventilation isolation
Refueling Floor	Area	2	G-M tube	1 to 10^{6} mrem/hr	Main control room	High	Downscale	Multi-point (with ARMs)	Main control room	Offline	SBGTS initiation and reactor building ventilation isolation
Reactor Building Closed Cooling Water System	Liquid effluent	1	Scintillation	10^{-1} to 10^{6} cps	Main control room	High and high-high	Downscale	Strip chart	Main control room	In-line	

Table 11.5-1 (Continued)

RADIATION MONITORING SYSTEMS EQUIPMENT PARAMETERS

Radiation Monitoring System	General Monitor Type	Number of Channels	Detector Type	Range	Indicator and Alarm Location	Radiation Alarm Types	Equipment Alarm Types	Recorder		Sampling	Logic
								Type	Location		
Service Water Effluent	Liquid effluent	1	Scintillation	10^1 to 10^6 cps (10^{-6} to 10^{-2} μCi/cc)	Main control room	High and high-high	Monitor failure	Strip chart	Main control room	Offline	
Radwaste Liquid	Liquid effluent	1	Scintillation	10^1 to 10^6 cps	Main control room	Alert (2) High	Downscale, loss of power	Alpha-numeric printer	Main control room	Offline	
					Radwaste control room	High		Chart recorder	Radwaste control room		
Isolation condenser	Area	2	G-M tube	10^{-2} to 10^3 and 1 to 10^5 mrem/hr	Main control room	High	Downscale	Multi-point (with ARMs)	Main control room	In-line	

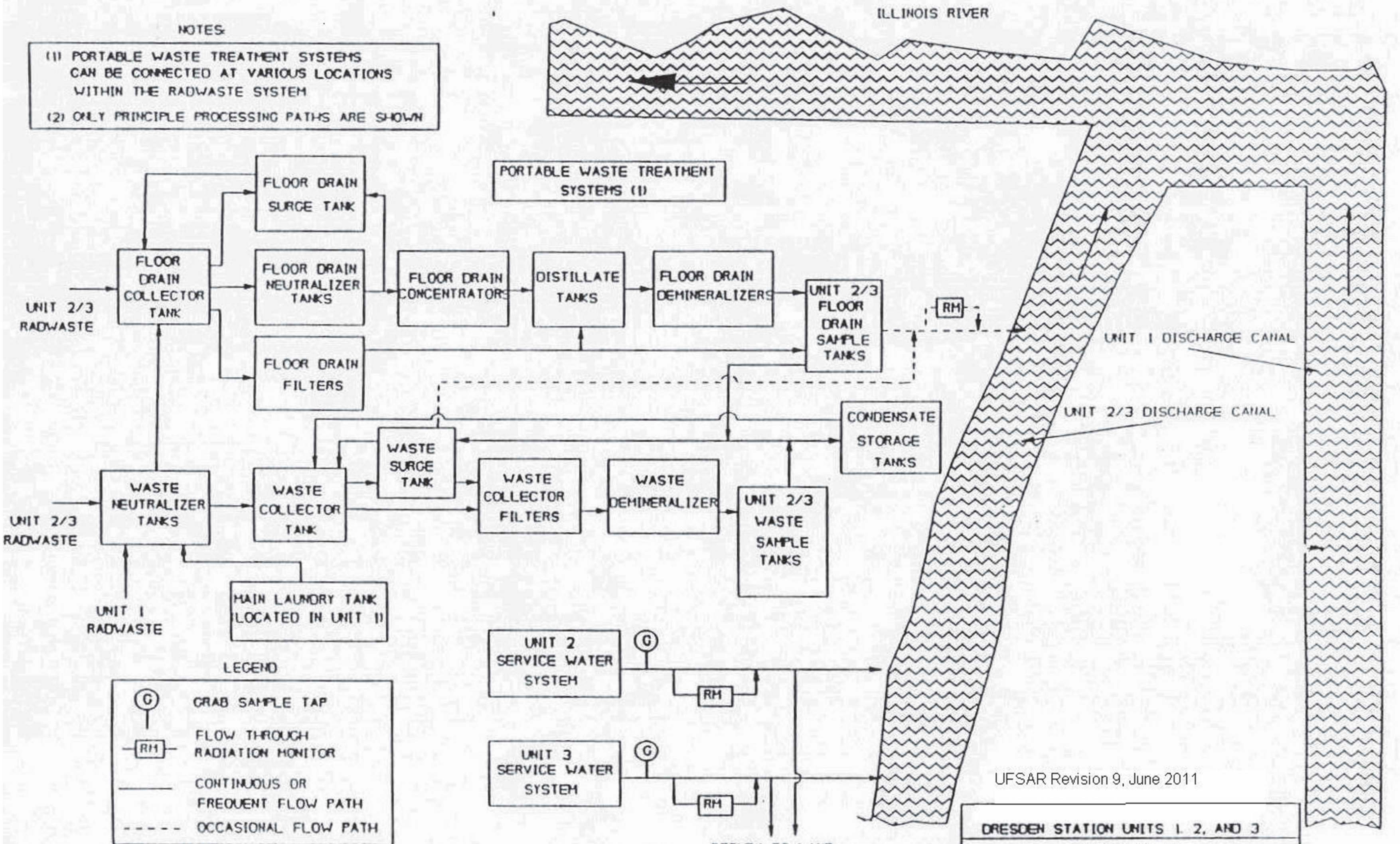
Notes:

1. Return pump failure alarms for Unit 3 only.
2. Software driven alarms from the Eberline Control Terminals. Refer to printer for monitor identification

Table 11.5-2

CHECK SOURCES FOR SPING MONITORS

Channel		Check Source	
Number	Type	Content (μCi)	Isotope
1	Beta particulate	30	Cs-137
2	Alpha particulate	-	-
3	Iodine (gamma)	0.5	Ba-133
4	Iodine subtraction (gamma)	-	-
5	Beta gas (low-range noble gas)	30	Cs-137
6	Gamma area	0.5	Sr-90 – Y-90
7	Gamma gas (medium-range noble gas)	-	-
8	Gamma background	-	-
9	Gamma gas (high-range noble gas)	0.5	Sr-90 – Y-90

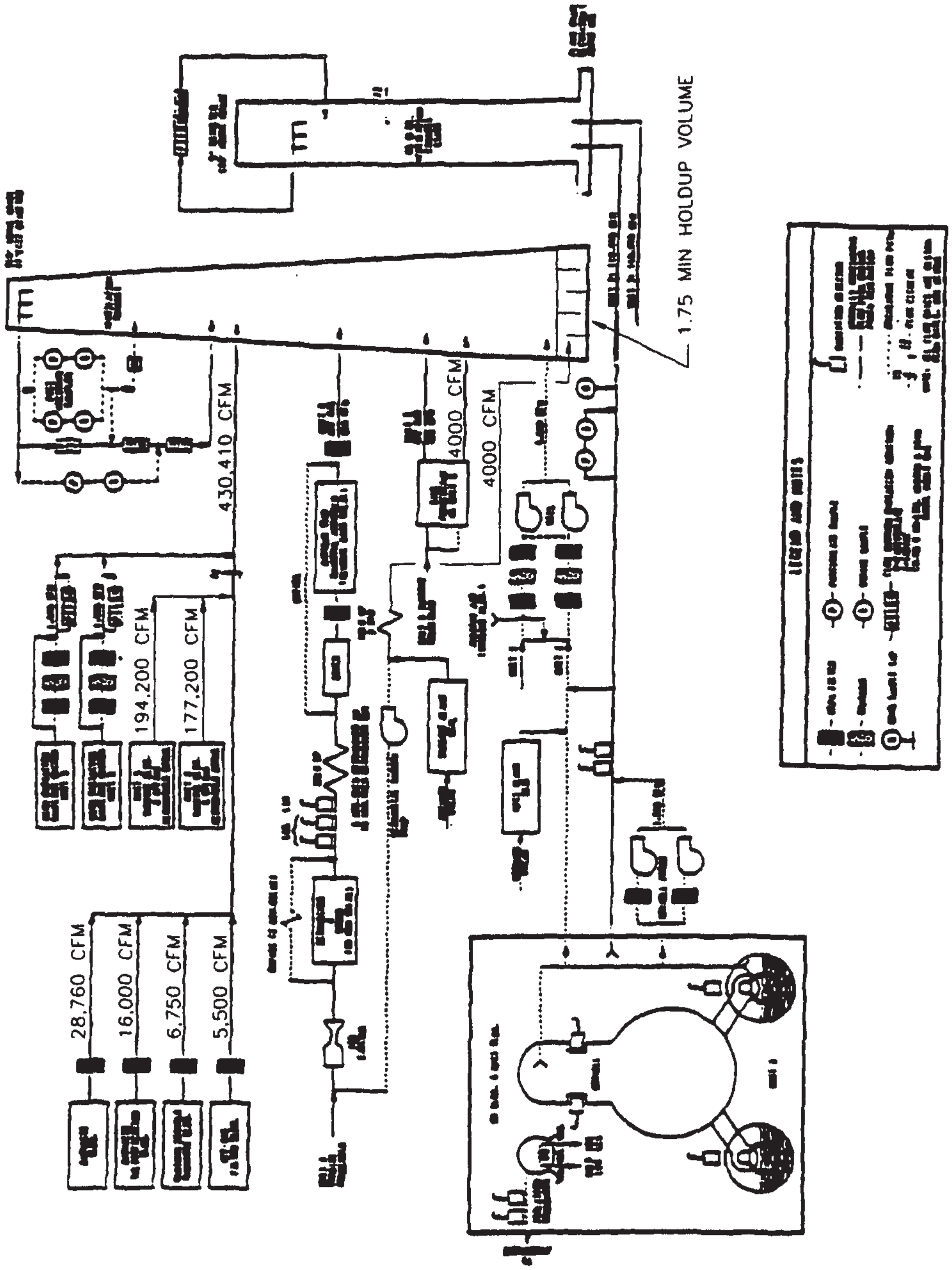


UFSAR Revision 9, June 2011

DRESDEN STATION UNITS 1, 2, AND 3

FIGURE 11.2-21

SIMPLIFIED LIQUID RADWASTE PROCESSING AND LIQUID EFFLUENT FLOW DIAGRAM

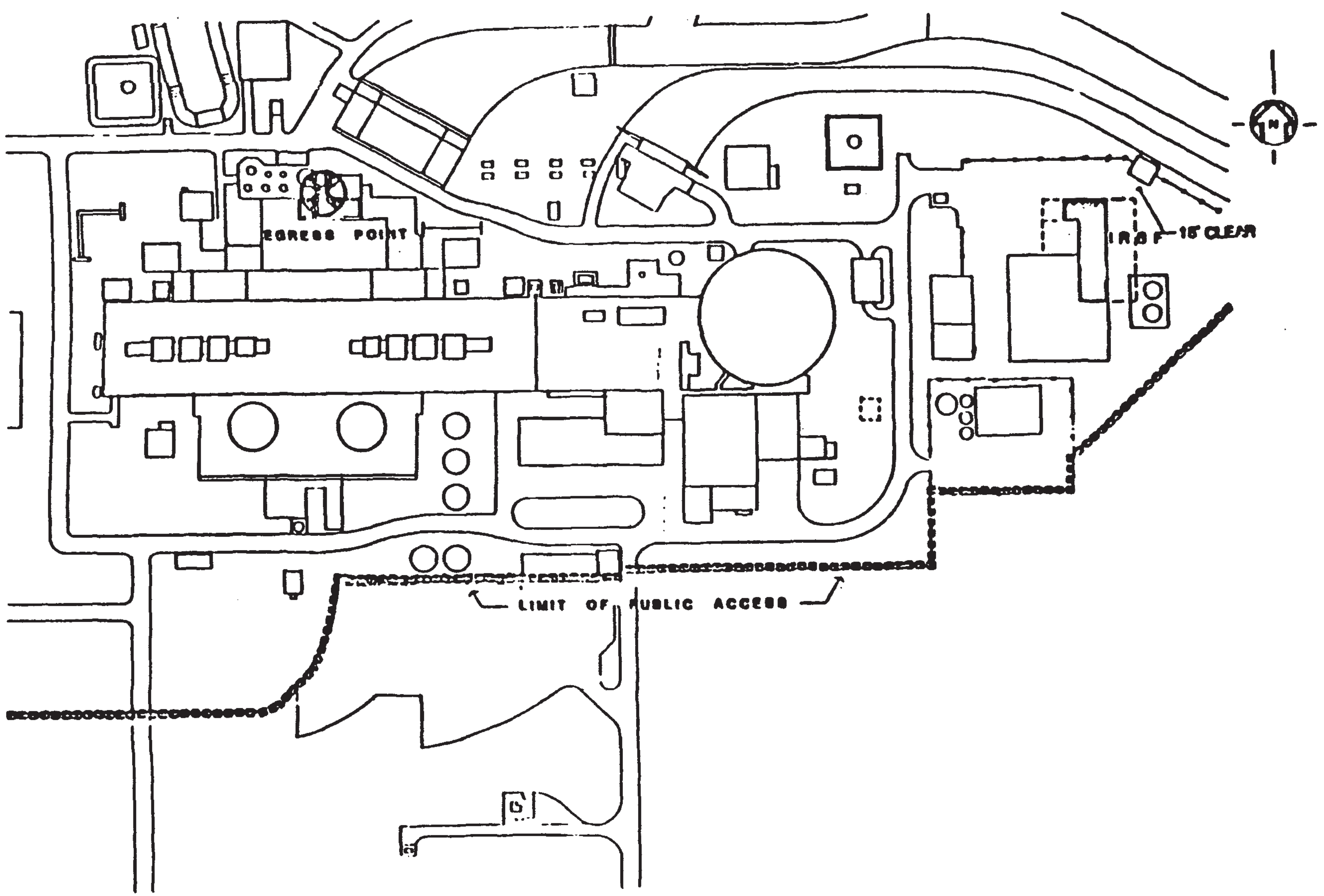


UFSAR REVISION 2, JUNE 1997

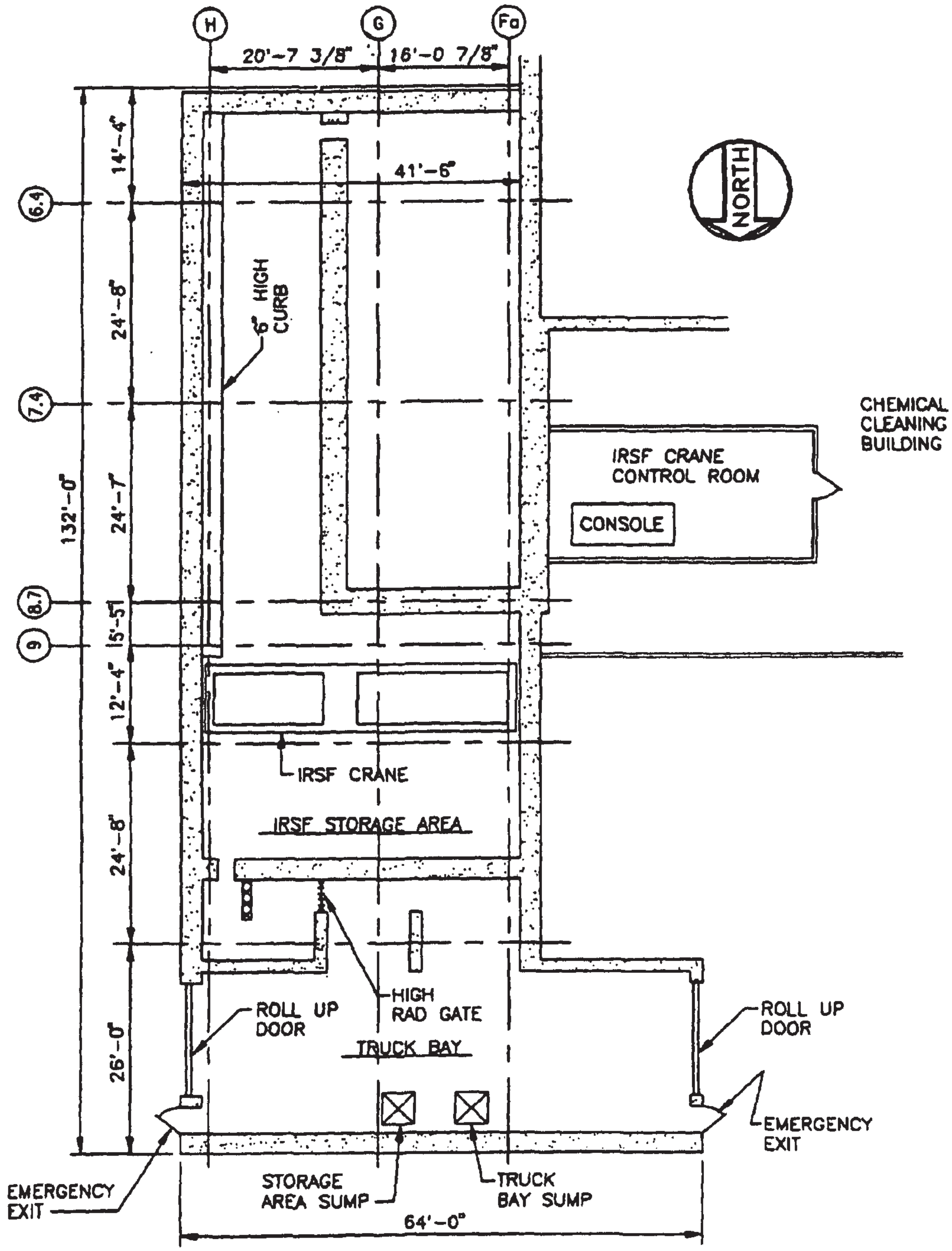
DRESDEN STATION
UNITS 2 & 3

SIMPLIFIED GASEOUS RADWASTE AND
GASEOUS EFFLUENT FLOW DIAGRAM

FIGURE 11.3-17



DRESDEN STATION UNITS 2 & 3
I R S F LOCATION DRESDEN SITE
FIGURE 11.4-2

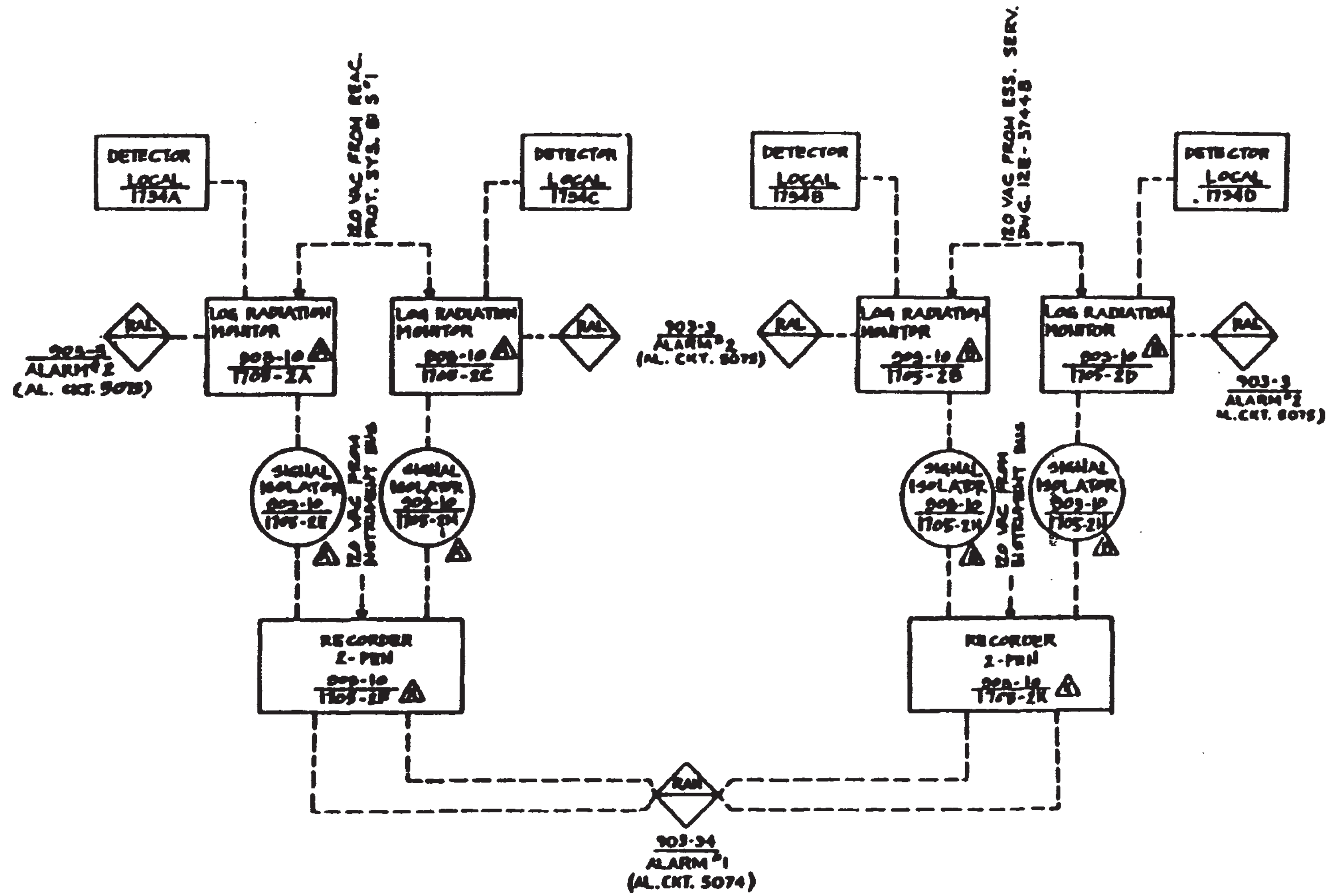


UFSAR REVISION 2, JUNE 1997

DRESDEN STATION
UNITS 2 & 3

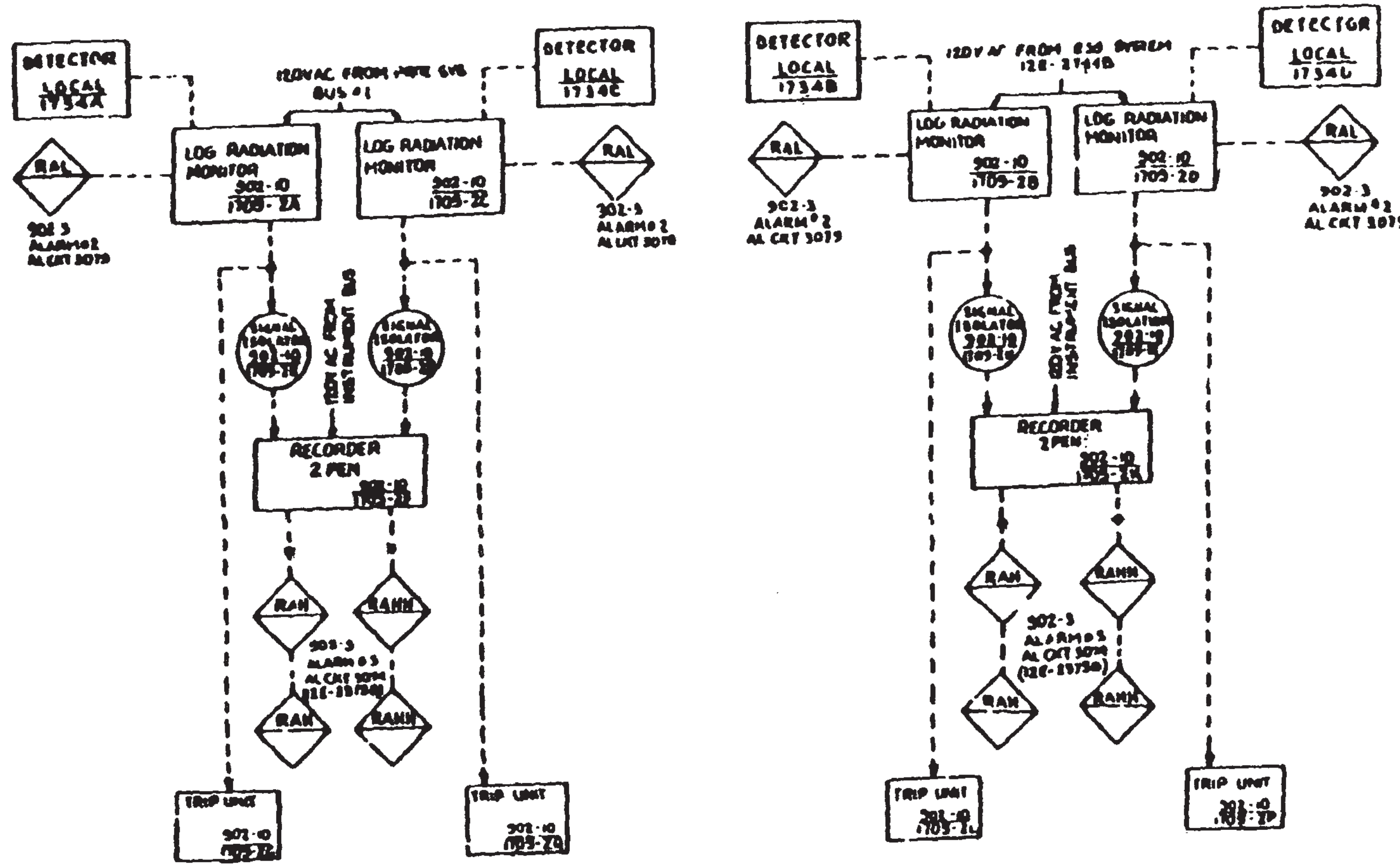
IRSF PLAN OF DRIVE THRU TRUCK BAY

FIGURE 11.4-3



UFSAR REVISION 4, JUNE 2001

DRESDEN STATION UNITS 2 & 3
UNIT 3 MAIN STEAM LINE RADIATION MONITORING SYSTEM
FIGURE 11.5-1

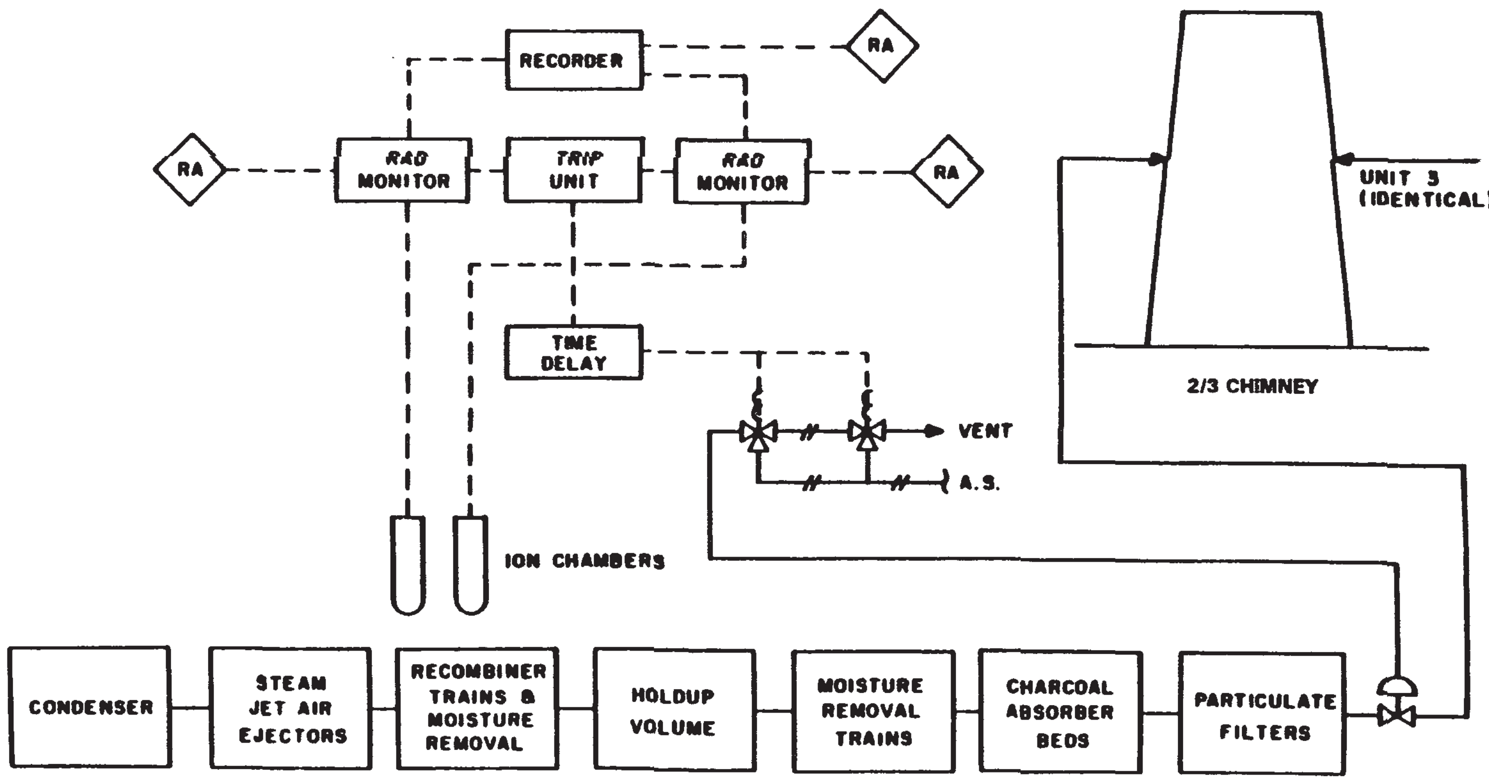


UFSAR REVISION 4, JUNE 2001

DRESDEN STATION
UNITS 2 & 3

UNIT 2 MAIN STEAM LINE RADIATION MONITORING SYSTEM

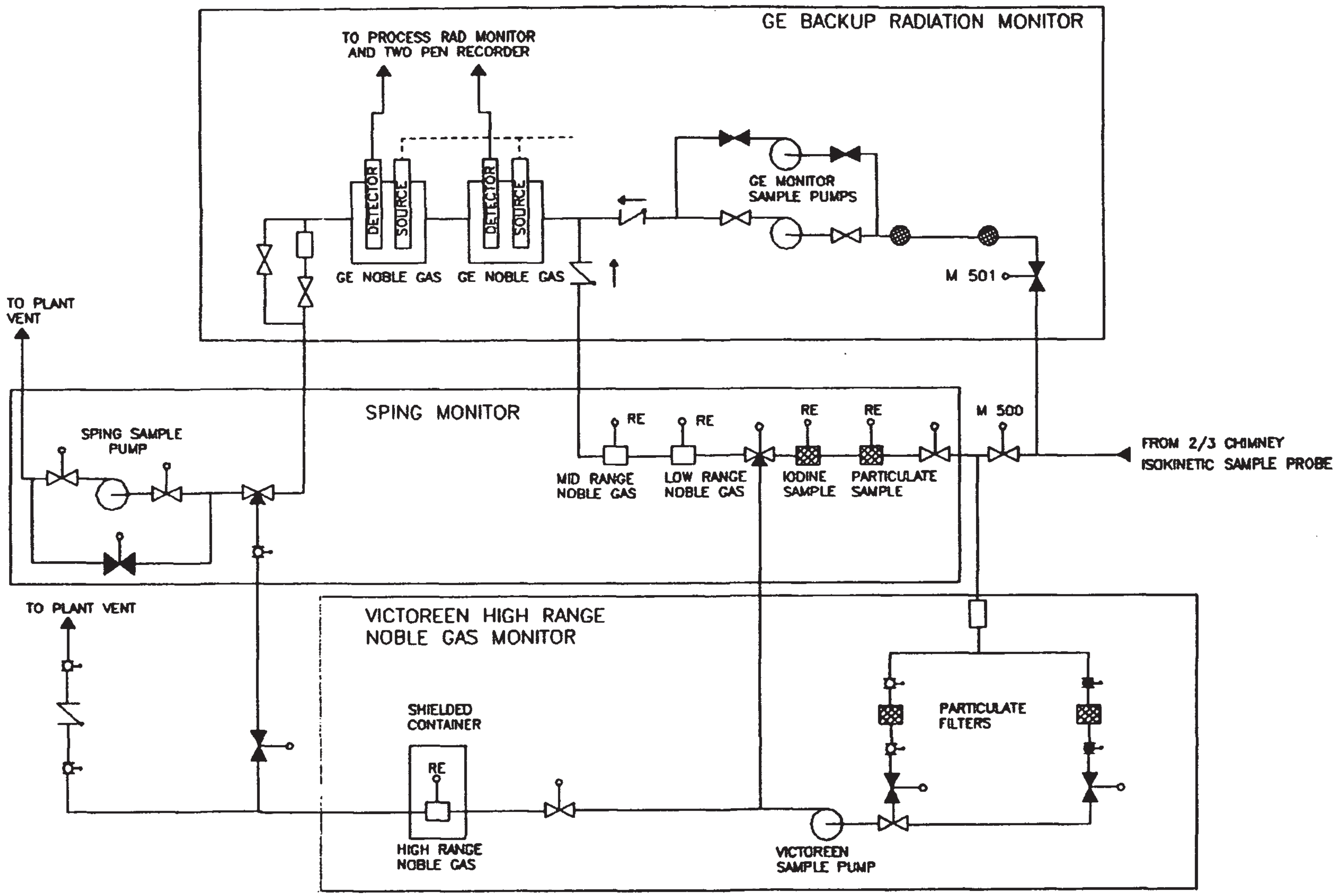
FIGURE 11.5-2



DRESDEN STATION
UNITS 2 & 3

AIR EJECTOR OFF-GAS RADIATION MONITORING SYSTEM

FIGURE 11.5-3

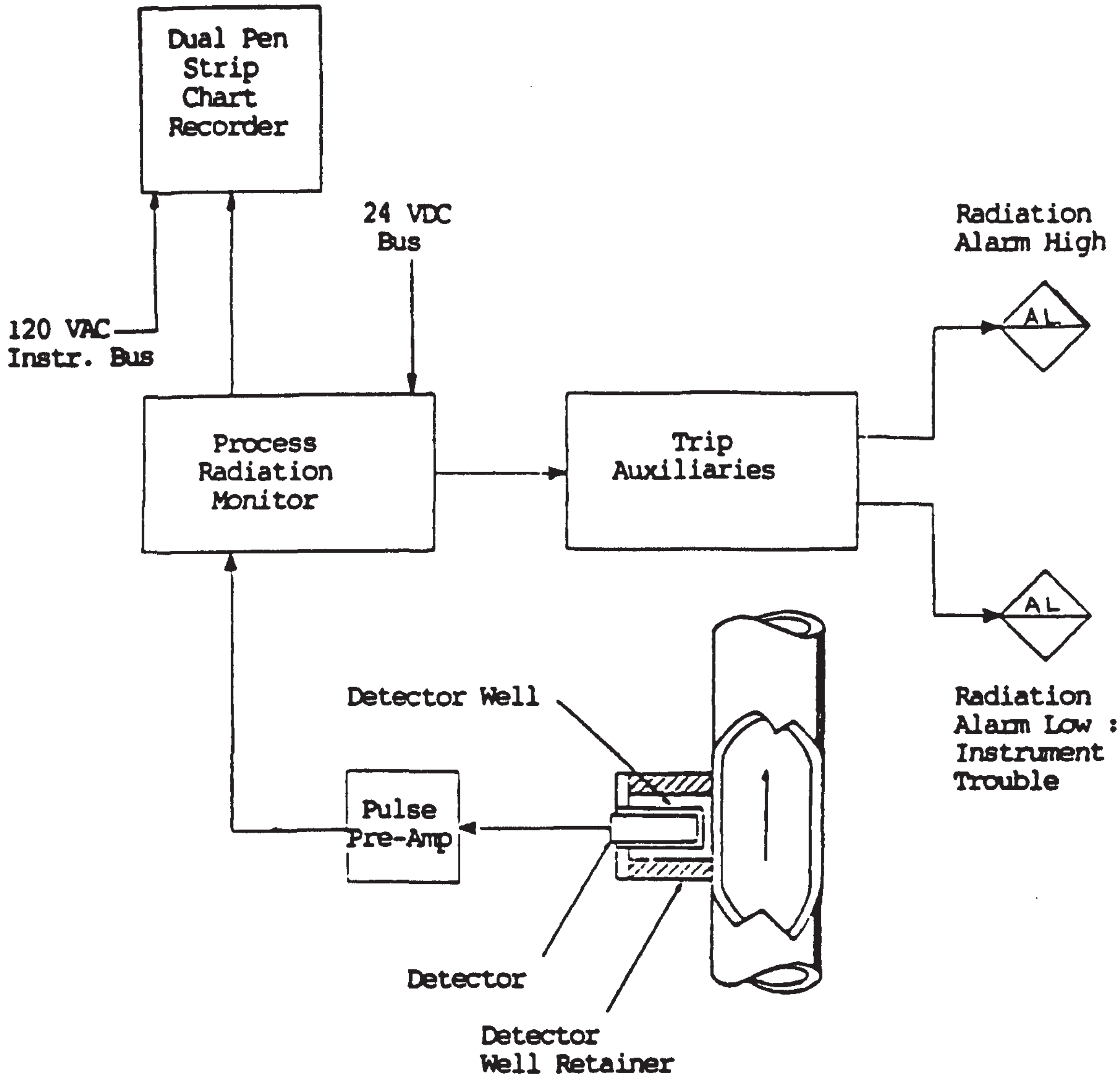


RE – RADIATION DETECTOR ELEMENT

DRESDEN STATION
UNITS 2 & 3

CHIMNEY EFFLUENT GAS MONITORING SYSTEM

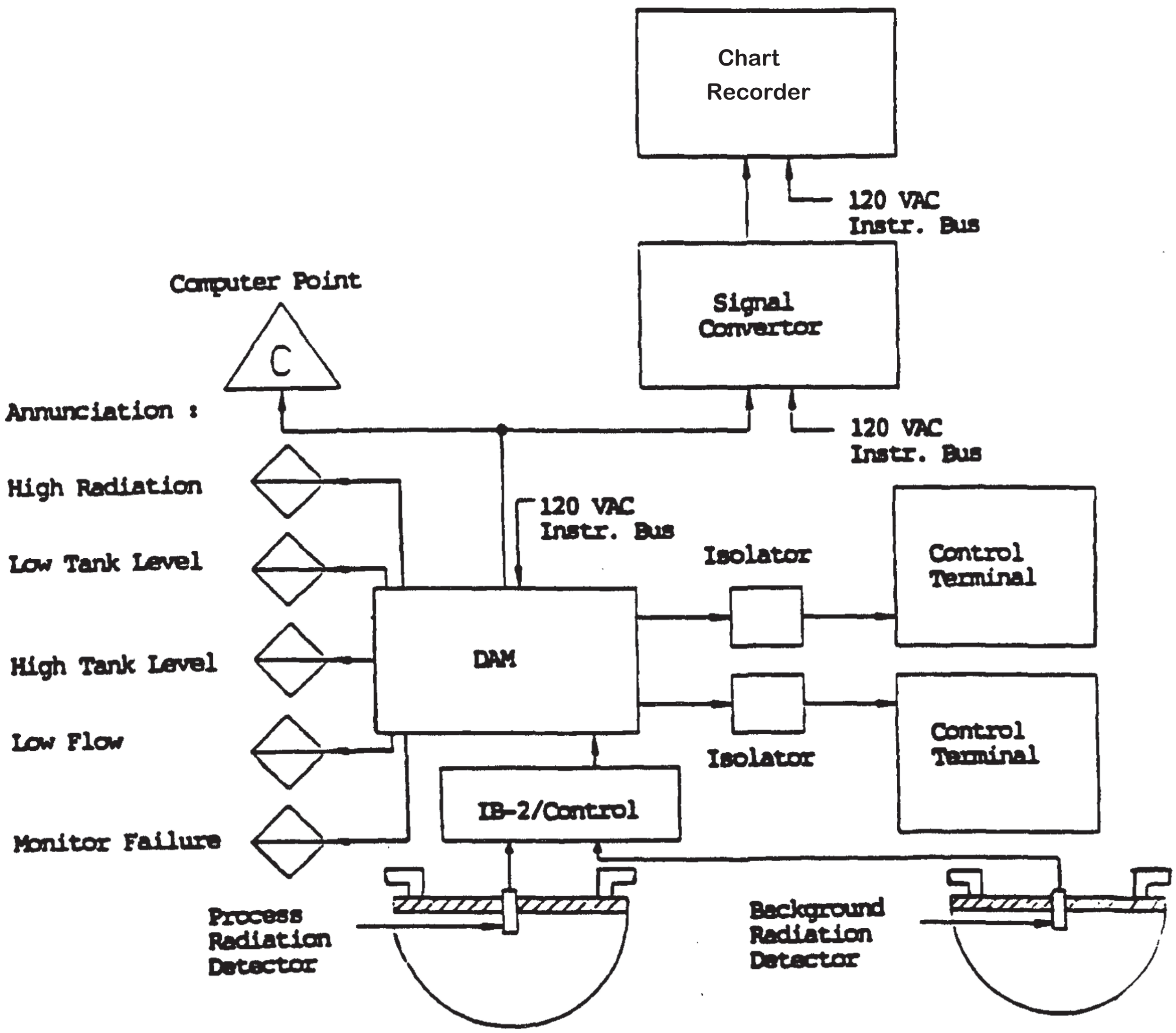
FIGURE 11.5-4



DRESDEN STATION
UNITS 2 & 3

FUNCTIONAL BLOCK DIAGRAM
REACTOR BUILDING CLOSED COOLING
WATER LIQUID RADIATION MONITOR

FIGURE 11.5-5



UFSAR Revision 11, June 2015

DRESDEN STATION
UNITS 2 & 3

FUNCTIONAL BLOCK DIAGRAM
LIQUID RADIOACTIVE WASTE
DISCHARGE MONITOR

FIGURE 11.5-7