

DSAR-11.1

**Radioactive Waste and Radiation
Protection and Monitoring**

Radioactive Waste Disposal System

Rev 4

Safety Classification:

Safety

Usage Level:

Information

Change No.:	EC 69954
Reason for Change:	Updated to reflect LWTF reanalysis due to reduced EAB.
Preparer:	C. Waszak

Fort Calhoun Station

Table of Contents

11.	RADIOACTIVE WASTE AND RADIATION PROTECTION AND MONITORING.....	5
11.1	Radioactive Waste Disposal System.....	5
11.1.1	Design Bases.....	5
11.1.1.1	General.....	5
11.1.1.2	Radioactive Waste Inventory.....	6
11.1.1.3	Deleted.....	6
11.1.1.4	Deleted.....	6
11.1.2	Liquid Wastes.....	7
11.1.2.1	Sources and Characteristics of Liquid Wastes.....	7
11.1.2.2	Collection and Handling of Liquid Wastes.....	8
11.1.2.3	Liquid Waste Treatment.....	13
11.1.2.4	Liquid Waste Disposal.....	14
11.1.2.5	System Components.....	14
11.1.2.6	System Operation.....	19
11.1.2.7	Design Evaluation.....	19
11.1.2.8	Availability and Reliability.....	31
11.1.2.9	Operation.....	32
11.1.2.10	Tests and Inspections.....	32
11.1.3	Gaseous Wastes.....	33
11.1.3.1	General.....	33
11.1.3.2	Sources of Waste Gas.....	34
11.1.3.3	Processing of Waste Gases.....	36
11.1.3.4	Deleted.....	36
11.1.3.5	Deleted.....	36
11.1.3.6	Deleted.....	36
11.1.3.7	Design Evaluation.....	36
	Radiological Gases Released from Auxiliary Building.....	36
11.1.3.8	Availability and Reliability.....	46
11.1.3.9	Operation.....	47
11.1.3.10	Tests and Inspections.....	47
11.1.4	Solid Wastes.....	48
11.1.4.1	General.....	48
11.1.4.2	Sources of Solid Waste.....	48
11.1.4.3	System Components.....	50
11.1.4.4	System Operation.....	51
11.1.4.5	Design Evaluation.....	51
11.1.4.6	Availability and Reliability.....	52
11.1.4.7	Tests and Inspections.....	52

List of Tables

Table 11.1-2 - Radioactive Waste Volumes	6
Table 11.1-10 - Reactor Coolant Waste Volume Estimates	9
Table 11.1-11 - Liquid Waste Volumes.....	10
Table 11.1-12 - Component Design Data, Waste Disposal System Tanks.....	15
Table 11.1-13 - Component Design Data, Waste Disposal System Pumps	16
Table 11.1-14 - Component Design Data, Waste Disposal System Process Equipment	18
Table 11.1-15 - Fission Product Activity in the Waste Treatment System at STP	21
Table 11.1-16 - Anticipated Quantities and Concentrations of Principle Radionuclides in the Discharge Tunnel Including Hypothetical Bounding Case and Comparison to 10 CFR 20 Limits	26
Table 11.1-18 - Deleted	35
Table 11.1-21 - Deleted	36
Table 11.1-23 - Maximum Gaseous Release, Spent Regenerant Tanks.....	38
Table 11.1-25 - Annual Releases of Radioactive Gases and Particulates.....	40
Table 11.1-26 - Solid Waste Volumes	50

List of Figures

The following figures are controlled drawings and can be viewed and printed from the applicable listed aperture card.

<u>Figure No.</u>	<u>Title</u>	<u>Aperture Card</u>
11.1-1	Calculated Maximum Source Concentration Downstream of Fort Calhoun Station.3657

11. **RADIOACTIVE WASTE AND RADIATION PROTECTION AND MONITORING**

11.1 Radioactive Waste Disposal System

11.1.1 Design Bases

11.1.1.1 General

The radioactive waste disposal system (RWDS) is designed to protect plant personnel and the public from exposure to radioactive wastes in accordance with 10 CFR Part 20; 10 CFR 50, Appendix I; 40 CFR Part 190; 10 CFR 50 Appendix A General Design Criterion 60, and Draft GDC Criteria 17 and 18; 10 CFR 50 Appendix B for reviews and audits; and the intent of NUREG-1301 (see [Section 11.3](#)).

The RWDS has been reviewed against the requirements of NUREG-1301 "Offsite Dose Calculation Manual Guidance: Standard Radiological Effluent Controls for Pressurized Water Reactors". As a result of the review, Technical Specifications were approved to govern effluent instrumentation calibration and operation, allowable dose rates, approved methodology to calculate dose rates, limiting conditions for operating the RWDS, requirements for environmental monitoring programs and requirements for maintaining records, ensuring adequate review and audits and reporting information as required. The details of RETS commitments for the liquid, gaseous and solid radioactive treatment systems are discussed in Sections 11.1.2, 11.1.3, and 11.1.4. RETS commitments for Radiation Monitoring are discussed in [Section 11.2.3](#). [Section 11.3](#) addresses overall requirements such as the Offsite Dose Calculation Manual (ODCM), reporting requirements, and summarizes the requirements of RETS as they are addressed by the Technical Specifications.

NRC Generic Letter 89-01 allowed licensees to remove the procedural details of the Radiological Effluent Technical Specifications from the Technical Specifications and place them in the ODCM. The administrative section of Technical Specifications was updated to include the programmatic controls necessary to ensure compliance with Federal Regulations. This change has placed the procedural requirements for equipment, sampling, analyses, monitoring, and dose limitations in the ODCM. Reference to specific sections of the ODCM will not be made in this document.

The RWDS includes equipment to collect, store, process and treat as required, monitor, and dispose of liquid, solid, and gaseous radioactive wastes.

The RWDS was designed to process and remove radioactive wastes from the plant adequately and safely when 1 percent of the core fuel elements have failed and corrosion and fission product concentrations in the reactor coolant were at design values. The design of the RWDS was based on the plant operating cycle.

Note: Since Fort Calhoun Station is permanently shutdown and defueled, the radioactive waste generation and removal needs are reduced. As a result, radioactive waste estimates for a running plant are considered to bound the needs of the decommissioning station.

11.1.1.2 Radioactive Waste Inventory

The waste volumes that were estimated to accumulate during one refueling cycle are shown in Table 11.1-2. Since Fort Calhoun Station is permanently shutdown and defueled, the radioactive waste generation and removal needs are reduced. As a result, radioactive waste estimates for an operating plant are considered to bound the needs of the decommissioning station.

Table 11.1-2 - Radioactive Waste Volumes

	Volume (ft ³ /cycle)	Basis
Liquids	150,000	Processed liquid at 70°F
Gases	51,135	At 70°F and 1 atm
Solids	5,000	Dry Activated Waste, filters, spent resins, depleted filtration/ion exchange media

11.1.1.3 Deleted

11.1.1.4 Deleted

11.1.2 Liquid Wastes

11.1.2.1 Sources and Characteristics of Liquid Wastes

The liquid waste collection and storage system is divided into two sections; auxiliary systems process wastes, and hotel wastes. The original sources of liquid wastes and their routing to the collection points are shown in the flow diagrams, P&ID's 11405-M-6, 11405-M-7 and 11405-M-99.

Auxiliary Systems Process Wastes

The principal sources for the liquids are:

- a. Spent regenerant;
- b. Auxiliary building floor drain header;
- c. Auxiliary building sump flows;
- d. Laboratory and decontamination area drain header;
- e. Spent resin sluice water;
- f. Monitor tanks contaminated return flows;
- g. Waste holdup tank relief valves;
- h. Containment building sump flows.
- i. Radioactive Waste Processing Building sump flows.
- j. Chemical and Radiation Protection Building Laboratory drains.
- k. Fuel transfer canal drains and safety injection system drains.

Wastes from these sources are subject to contamination. They are collected in the spent regenerant tanks which are vented to the auxiliary building ventilation system.

Aerated Domestic Wastes

The principal sources for these liquids are:

- a. Laundry facility drains
- b. Shower drains
- c. Hand sink drains

These wastes all originate in the auxiliary building and are transported in the laundry drain header which discharges to the hotel waste tanks. Domestic wastes are normally low in activity.

11.1.2.2 Collection and Handling of Liquid Wastes

Reactor Coolant Liquids

The volume of waste entering the radioactive waste disposal system from this source was estimated for a core cycle and shown in Table 11.1-10.

NOTE: This estimate supports the radiological consequences analysis in DSAR section 14.20 which is for a running plant. As such the cycle waste volume estimates are being maintained even though Fort Calhoun Station is permanently shutdown and actual radioactive waste generation and removal needs are reduced.

Table 11.1-10 - Reactor Coolant Waste Volume Estimates

Reactor Coolant Discharged to Waste Between Refuelings
Based on Postulated Events During an Equilibrium Core Cycle

<u>Event</u>	<u>Elapsed Time (equivalent full power days)</u>	<u>Waste Volume (liquid @ 70°F, ft³/event)</u>
1. Reactor refueled at 70°F through heatup to 570°F, initial full power and xenon equilibrium	2	4,230
2. Cold shutdown No. 1 and restart following attainment of samarium equilibrium	23	3,278
3. Hot shutdown No. 1 and restart	118	1,167
4. Hot shutdown No. 2 and restart	168	1,717
5. Hot shutdown No. 3 and restart	248	1,938
6. Cold shutdown No. 2, partial drain for maintenance and restart	254	9,463
7. Hot shutdown No. 4 and restart	490	5,671
8. Initiate operation of deborating demineralizer No.3	500	--
9. Cold shutdown	530	<u>763</u>
Total from events		28,227
Total from control of coolant boron concentration during 530 full power days		<u>24,514</u>
Total per equilibrium cycle		52,741*

Assumptions: (a) Base loaded plant; boron adjustment for load following is not required.
(b) Reactivity effect of xenon during shutdown is not compensated by boron adjustment.

Other reactor coolant type wastes are variable in flow and occur chiefly as periodic drains, leak-offs, and occasional relief valve discharges. All liquid waste volume estimates are shown in Table 11.1-11.

* This value is used to conservatively calculate the environment radioactive release inventories for a liquid waste tank failure in calculation FC06802.

Table 11.1-11 - Liquid Waste Volumes

	Volume liquid @ 70°F, (ft ³ /482 full power days)	<u>Remarks</u>
1.	<u>Reactor Coolant Wastes</u>	
	Boron control	From Table 11.1-10
	Reactor coolant pump seal leak-offs	--
	CEDM leak-offs	Design value for RWDS purpose
	Charging pump seal leak-offs	--
	Stored energy safety injection	Flow to RWDS based tanks, check valve leak-offs on 0.1% leak-off
	Purification filters drain	20
	CVCS ion exchangers, drain and sluice water	400
	Reactor coolant and CVCS sample wastes	16,494
	Valve leak-offs and safety relief) Valve discharge) Quench tank drain)	normally zero
2.	<u>Spent Regenerant Chemicals</u>	
	Deborating exchangers	700
		Based on two regenerations per cycle
3.	<u>Hotel Wastes</u>	49,482
		--
4.	<u>Spent Fuel Pool Cooling System</u>	
	Filter drain	50
	Ion exchanger drain and sluice water	60
		Two replacements per cycle
		--
5.	<u>Radiochemical Lab Drains</u>	-
		Accounted for in sampling wastes
6.	<u>Secondary Plant Steam Generator Blowdown</u>	16,494
		Normally zero
		Assumes discharge of water inventory of two steam generators per year.
Total		141,711 ft ³
		Rounded to 150,000 ft ³

Reactor coolant liquids are collected as follows:

- a. Deleted
- b. Auxiliary building sump tank (WD-25): This tank is the collection point for equipment drains in the auxiliary building (equipment drain header). The ABST is vented and drained. All incoming liquid is directed to the Room 23 sump, then pumped to the spent regen tanks.
- c. Waste holdup tanks (WD-4A/B/C): These tanks receive the output from the spent regenerant tanks (WD-13A/B). The function of these tanks is to provide temporary storage capacity. Three waste holdup tanks are provided.

When one of the waste holdup tanks becomes filled, the waste flow is diverted to a second tank. The accumulated batch may be then thoroughly mixed by means of a recirculation pump. The recirculation pump is also capable of transferring the contents of one tank to another. Normally, the third waste holdup tank is on standby, ready to receive waste flow if the second tank becomes filled before the contents of the first tank have been discharged.

Two waste holdup pumps take suction from the tanks and deliver the waste to the treatment inlet header or to the monitor tanks. The two pumps are manually controlled from the waste treatment control panel (see [Section 7.6.3](#)).

Auxiliary Systems Process Wastes

These wastes are collected in the spent regenerant tanks and include: floor drain header flow from the auxiliary building, sump flows from the auxiliary building, radioactive waste processing and containment buildings, and spent resin sluice water. Gravity drains from the floor drain header and the drain header above floor elevation 971'-0" are collected directly in the spent regenerant tanks, whereas gravity drains from the sub-basement floor elevation and floor drains within the containment are collected in sumps and are delivered automatically by level-controlled pumps to the spent regenerant tanks.

Two spent regenerant tanks are provided and they are constructed of type 304 stainless steel due to the variety of liquids they might contain. Connection to the caustic dilution tank is provided for neutralization purposes, if required. The tanks are vented to the building ventilations exhaust system. Checked vent lines permit atmospheric inflow to the tanks on falling liquid level and exhaust to the ventilation system on rising liquid level.

A completed waste batch is normally delivered to the waste holdup tanks or the treatment inlet header. Delivery is made by two spent regenerant pumps, manually controlled, that also serve to mix the tank contents by recirculation. The tanks can also be transferred directly to the monitor tanks or the other spent regenerant tank or be recirculated and sampled if desired.

Hotel Wastes

These flows are chiefly from the laundry drain header, are usually low in activity, and are collected in the hotel waste tanks.

A filter has been placed in this line to prevent the passage of radioactive solids to the hotel waste tank from the laundry washers.

Two hotel waste tanks are provided, each designed to hold approximately one day's hotel waste flow. They are constructed of carbon steel, since corrosive liquids do not enter the laundry drain header. The tanks are simply vented to the atmosphere.

Mixing is accomplished by use of the hotel waste pumps as circulators, after which the waste batch is sampled and analyzed. The batch is then delivered to either the treatment inlet header or the monitor tanks or the overboard discharge header by the two manually controlled hotel waste pumps.

11.1.2.3 Liquid Waste Treatment

General

The RWDS is designed to provide filtration, and demineralization as needed to ready the waste for ultimate disposal. The process flow diagrams are shown in P&ID's 11405-M-8 and 11405-M-9.

Filtration

Suspended solids are removed by two waste filters. Solids are retained on the disposable filter element. Filter effluent is directed to the next treatment step or to the monitor tanks.

Filtration/Ion-Exchange

Filtration/ion-exchange (FIX) services are presently being used as the preferred method for liquid waste treatment and is located in the Radioactive Waste Processing Building.

The FIX system is designed to remove specific radioisotopes in the liquid waste stream.

The treated effluent from the FIX system is transferred to the monitor tanks.

Monitor Tanks

The two monitor tanks normally receive processed liquid wastes from the waste holdup tanks. The wastes are sampled and analyzed isotopically to confirm acceptability for controlled release to the overboard header. One tank can be undergoing recirculation for sampling while the other tank is being released to the overboard header.

11.1.2.4 Liquid Waste Disposal

During releases of radioactive liquid waste, the equipment and conditions shall be in accordance with the ODCM. The doses resulting from liquid releases shall not exceed, during any calendar year, 3 millirem to the total body (10 millirem to any organ) as required by 10 CFR Part 50 Appendix I.

The requirements for sample monitoring and testing prior to release and the requirements to ensure monitors are calibrated are included in the ODCM. Records of liquid releases must be maintained and are subject to the review, audits, and reporting requirements discussed in [Section 11.3](#).

The overboard header is the only path through which the liquid rad wastes can be released from the containment, auxiliary, Radioactive Waste Processing and CARP buildings. It receives liquid from the monitor tanks, & the hotel waste tanks. The overboard header originates at the monitor tanks or the hotel waste tanks and terminates in the condenser circulating water discharge tunnel, entering the tunnel in the section downstream of the warm water recirculation return (see P&ID 11405-M-257). Effluent from the monitor tanks or the hotel waste tanks is moved by two monitor tank pumps or hotel waste pumps and the flow rate is monitored on a recorder.

The overboard header is equipped with a radiation monitor that interrupts flow if waste activity reaches a predetermined setpoint (see [Section 11.2.3](#)).

11.1.2.5 System Components

The various components of the RWDS are divided into three groups for convenience of listing; tanks, pumps, and process equipment. These are shown in Tables 11.1-12, 13, and 14.

Table 11.1-12 - Component Design Data, Waste Disposal System Tanks

<u>Tank</u>	<u>No. Installed/ Item No.</u>	<u>Tank Capacity gallons/ft³</u>	<u>Pressure, psig Design/ Operating</u>	<u>Temperature °F Design/ Operating</u>	<u>Material*</u>	<u>Code</u>
Waste Holdup Tanks	3/WD-4A, B&C	45,800/6,100	15/Atmos	200/120	CS	ASME Section III, Class C, Feb. 1968
Spent Regenerant Tanks	2/WD-13A&B	5,530/739	5/Atmos	200/70	304 SS	ASME Section VIII, Feb. 1968
Hotel Waste Tanks	2/WD-15A&B	1,200/160	15/Atmos	200/140	CS	ASME Section VIII, Feb. 1968
Monitor Tanks	2/WD-22A&B	6,770/905	5/Atmos	200/140	304 SS	ASME Section VIII, Feb. 1968
Auxiliary Building Sump Tank	1/WD-25	700/95	25/Atmos	200/120	304 SS	ASME Section VIII, Feb. 1968
Spent Resin Storage Tank	1/WD-33	3,250/434	25/Atmos	250/120	304 SS	ASME Section VIII, Feb. 1968
Waste Metering Tank	1/WD-46	688/92	Atmos	-	316 SS	ASME Section VIII, Feb. 1968

* SS= Stainless Steel, CS= Carbon Steel

Pumps were designed in accordance with the Standards of the Hydraulic Institute and all motors conformed to NEMA standards. Materials were in accordance with the appropriate ASTM specifications. Other codes and standards are listed in the tables referenced above.

Table 11.1-13 - Component Design Data, Waste Disposal System Pumps

<u>Pump</u>	<u>No. Installed/ Item No.</u>	<u>Type</u>	<u>Capacity</u>	<u>Fluid Side Material*</u>
Containment Sump Pumps	2/WD-3A&B	Vertical Centrifugal	50 gpm @ 40 ft.	AI
Waste Holdup Tank Pumps	2/WD-5A&B	Horizontal Centrifugal, Canned Rotor	50 gpm @ 177 ft.	316 SS
Waste Holdup Recirculation Pump	1/WD-6	Horizontal Centrifugal	500 gpm @ 85 ft.	AI
Spent Reg. Pumps	2/WD-14A&B	Horizontal Centrifugal	50 gpm @ 157 ft.	304 SS
Hotel Waste Pumps	1/WD-16A	Horizontal Centrifugal	50 gpm @ 120 ft.	AI
	1/WD-16B	Horizontal Centrifugal	50gpm @ 130 ft.	AI

* AI = All Iron
SS = Stainless Steel
CS = Carbon Steel

Table 11.1-13 (Continued)

<u>Pump</u>	<u>No. Installed/ Item No.</u>	<u>Type</u>	<u>Capacity</u>	<u>Material*</u>	<u>Fluid Side</u>
Monitor Tank Pumps	2/WD-23A&B	Horizontal Centrifugal	50 gpm @ 160 ft.		304 SS
Auxiliary Bldg. Sump Tank Pumps	2/WD-26A&B	Horizontal Centrifugal	35 gpm @ 110 ft.		304 SS
Auxiliary Bldg. Room 21 and 22 Sump Pumps	4/WD-27A&B, 40A&B	Vertical Centrifugal	20 gpm @ 34 ft.		CI
Auxiliary Bldg. Room 23 Sump Pumps	2/WD-41A&B	Vertical Centrifugal	20 gpm @ 36 ft.		CI
Spent Resin Pump	1/WD-34	Horizontal Centrifugal	30 gpm @ 106 ft.		304 SS
Radioactive Waste Processing Bldg. Sump Pumps	4/WD-30A&B, WD/31A&B	Vertical Centrifugal	65 gpm @ 40 ft.		304SS

* SS = Stainless Steel
CS = Carbon Steel
CI = Cast Iron

Table 11.1-14 - Component Design Data, Waste Disposal System Process Equipment

<u>Waste Filters, Item No's WD-17A&B</u>	<u>Description</u>
Number	2
Type type	Expendable element pressure
Materials of Construction	304 stainless steel vessel
Vessel Design Pressure, psig	150
Vessel Design Temperature, °F	250
Vessel Code	ASME Section III, Class C, Feb. 1968
Flow Rate (filter), each, gpm	150
Average Efficiency, % (particles 50 microns)	43

11.1.2.6 System Operation

The operation of the liquid waste section of the RWDS involves a combination of automatic and manual controls. The flow of liquids from one of the collection tanks (the auxiliary building sump tank) and the four drain sumps can be controlled automatically by liquid level. The control panels are described in [Section 7.6.3](#).

At the waste holdup tanks, the hotel waste tanks, the spent regenerant tanks, and the monitor tanks, the operator must decide where to send the contents of a tank. The operator can send it through various tanks, filters, or the Filtration Ion Exchange System, depending on the processing required. Therefore, the flow leaving these tanks is manually controlled at the waste disposal control panel.

The waste filters are equipped with differential pressure indication and the filters are replaced when a predetermined pressure drop is reached.

The filtration/ion exchange system is designed to provide any flow logic through the system's pressure vessels. The flow logic is dependent upon the type of waste to be processed and is accomplished by manually valving the hose setup between vessels.

11.1.2.7 Design Evaluation

The anticipated performance of the liquid waste system has been calculated in accordance with the following assumptions.

The maximum cycle quantity of liquid waste containing significant activity was approximately 141,711 cu. ft. As shown in Table 11.1-10, 52,741* cu. ft. of the total liquid waste was from the chemical and volume control system and had already passed through the purification ion exchangers. The activity of this liquid waste was assumed to be reduced by a factor of 10 for each nuclide except rubidium, molybdenum, noble gases and tritium for which a factor of unity has been assumed. An additional volume of 22,184* cu. ft., shown in Table 11.1-11, had an activity equal to that of reactor coolant. Hotel wastes are low in activity and with the addition of a filter which collects radioactive solids, will remain low in activity at discharge to the hotel waste tanks. Waste volumes resulting from steam generator blowdown while normally zero, was estimated on the basis that primary-to-secondary leakage requires that the zero load liquid inventory of both steam generators (6,000 cu. ft.) was discharged to the RWDS once per cycle and that the activity was consistent with having operated for 45 days with a 1 gph primary-secondary leak and one percent fuel failure based on historical calculations [Ref. 11.4.16](#).

The two waste filters are designed to remove insoluble corrosion products, some of which may be radioactive. However, no credit has been assumed for these filters in the system evaluation. The Filtration Ion Exchange System average total decontamination factor is 364. The normal liquid waste holdup time is 30 days. The fission product activities in the liquid waste treatment system are shown in Table 11.1-15.

NOTE: This estimate supports the radiological consequences analysis in DSAR section 14.20 which is for a running plant. As such the cycle waste volume estimates are being maintained even though Fort Calhoun Station is permanently shutdown and actual radioactive waste generation and removal needs are reduced.

Dilution flow rates are also reduced due to the plants shutdown and permanently defueled condition. Administrative measures account for reduced dilution flow by limiting the undiluted effluent discharge flow rate. This ensures that the concentration of diluted radioactive effluents remain within the limits specified in 10 CFR Part 20. The capacity to remove liquid waste at these reduced flow rates remains well above the expected average annual liquid waste discharge.

* These values are used to conservatively estimate the environmental radioactivity release inventories in calculation FC06802.

Table 11.1-15 - Fission Product Activity in the Waste Treatment System at STP				
NUCLIDE	SPECIFIC ACTIVITY AT STP with ion exchange RCS Concentration $\mu\text{Ci/cc}$	LRWS As-Received Concentration $\mu\text{Ci/cc}$	Annual 1 Day Decay $\mu\text{Ci/cc}$	Annual 30 Day Decay $\mu\text{Ci/cc}$
Kr-83m	2.20E+00	2.20E+00	1.72E-04	5.87E-119
Kr-85	1.94E-01	1.94E-01	1.33E-01	1.33E-01
Kr-85m	4.67E+00	4.67E+00	7.85E-02	1.37E-48
Kr-87	9.37E+00	9.37E+00	1.35E-05	2.69E-170
Kr-88	1.31E+01	1.31E+01	2.58E-02	4.50E-76
Kr-89	1.64E+01	1.64E+01	2.17E-136	0.00E+00
Kr-90	1.77E+01	1.77E+01	0.00E+00	0.00E+00
Xe-131m	1.63E+00	1.63E+00	1.06E+00	1.96E-01
Xe-133	1.13E+02	1.13E+02	6.79E+01	1.47E+00
Xe-133m	1.09E+00	1.09E+00	5.49E-01	5.68E-05
Xe-135	1.30E+01	1.30E+01	1.45E+00	1.75E-23
Xe-135m	7.27E+00	7.27E+00	2.26E-28	0.00E+00
Xe-137	3.18E+01	3.18E+01	6.72E-113	0.00E+00
Xe-138	3.03E+01	3.03E+01	3.45E-30	0.00E+00
Br-82	7.49E-04	2.75E-03	1.18E-03	1.37E-09
Br-83	3.16E-02	1.16E-01	7.79E-05	4.09E-92
Br-85	6.66E-02	2.44E-01	1.65E-152	0.00E+00
I-129	9.16E-09	3.36E-08	2.31E-08	2.31E-08
I-130	5.52E-03	2.02E-02	3.63E-03	4.09E-20
I-131	1.19E+00	4.35E+00	2.75E+00	2.25E-01
I-132	3.55E-01	1.30E+00	6.16E-04	1.19E-95
I-133	5.02E-01	1.84E+00	5.70E-01	4.84E-11
I-134	5.61E-01	2.06E+00	8.16E-09	9.33E-248
I-135	4.77E-01	1.75E+00	9.57E-02	1.25E-33
I-136	2.24E-01	8.19E-01	0.00E+00	0.00E+00
Cs-132	9.27E-06	3.40E-05	2.10E-05	9.44E-07
Cs-134	4.17E-02	1.53E-01	1.05E-01	1.02E-01
Cs-134m	9.45E-03	3.46E-02	7.86E-05	8.16E-77
Cs-135m	9.75E-03	3.57E-02	1.64E-10	1.19E-247
Cs-136	1.30E-02	4.77E-02	3.12E-02	6.77E-03
Cs-137	3.13E-02	1.15E-01	7.91E-02	7.89E-02
Cs-138	4.70E-01	1.72E+00	4.12E-14	0.00E+00
Cs-139	4.39E-01	1.61E+00	1.96E-47	0.00E+00
Cs-140	3.95E-01	1.45E+00	0.00E+00	0.00E+00
Rb-86	4.12E-03	4.12E-03	2.73E-03	9.30E-04
Rb-88	1.93E+00	1.93E+00	5.61E-25	0.00E+00
Rb-89	2.53E+00	2.53E+00	5.36E-29	0.00E+00
Rb-90	2.35E+00	2.35E+00	1.78E-170	0.00E+00
Rb-90m	7.19E-01	7.19E-01	8.06E-102	0.00E+00
Ag-110	4.07E-02	1.49E-01	0.00E+00	0.00E+00

Table 11.1-15 - Fission Product Activity in the Waste Treatment System at STP				
NUCLIDE	SPECIFIC ACTIVITY AT STP with ion exchange RCS Concentration $\mu\text{Ci/cc}$	LRWS As-Received Concentration $\mu\text{Ci/cc}$	Annual 1 Day Decay $\mu\text{Ci/cc}$	Annual 30 Day Decay $\mu\text{Ci/cc}$
Ag-110m	1.02E-03	3.73E-03	2.56E-03	2.36E-03
Ag-111	1.49E-02	5.47E-02	3.43E-02	2.31E-03
Ag-112	6.84E-03	2.51E-02	8.64E-05	1.68E-71
As-76	4.96E-06	1.82E-05	6.65E-06	7.32E-14
Cd-115	2.03E-03	7.43E-03	3.75E-03	4.53E-07
Cd-115m	9.39E-05	3.44E-04	2.33E-04	1.49E-04
Ga-72	4.05E-06	1.48E-05	3.14E-06	4.37E-21
Ge-77	1.72E-04	6.30E-04	9.95E-05	2.89E-23
In-115m	2.03E-03	7.43E-03	1.26E-04	2.54E-51
Sb-122	2.40E-04	8.78E-04	4.68E-04	2.74E-07
Sb-124	1.86E-04	6.82E-04	4.64E-04	3.33E-04
Sb-125	2.15E-03	7.89E-03	5.43E-03	5.32E-03
Sb-127	2.13E-02	7.80E-02	4.49E-02	2.43E-04
Sb-129	7.85E-02	2.88E-01	4.52E-03	1.12E-50
Sb-130	2.61E-02	9.57E-02	7.03E-13	0.00E+00
Sb-130m	1.09E-01	4.01E-01	4.45E-70	0.00E+00
Sb-131	1.93E-01	7.06E-01	6.98E-20	0.00E+00
Sb-132	1.14E-01	4.18E-01	1.87E-104	0.00E+00
Sb-132m	1.11E-01	4.07E-01	4.63E-156	0.00E+00
Sb-133	1.63E-01	5.97E-01	1.81E-174	0.00E+00
Se-83	1.47E-02	5.38E-02	1.36E-21	0.00E+00
Sn-121	1.99E-03	7.28E-03	2.71E-03	4.92E-11
Sn-123	1.56E-04	5.73E-04	3.93E-04	3.36E-04
Sn-125	1.22E-03	4.47E-03	2.86E-03	3.56E-04
Sn-127	8.56E-03	3.14E-02	7.85E-06	1.40E-105
Te-127	2.10E-02	7.69E-02	8.94E-03	3.53E-25
Te-127m	3.49E-03	1.28E-02	8.75E-03	7.28E-03
Te-129	7.49E-02	2.75E-01	1.12E-07	2.95E-188
Te-129m	1.51E-02	5.53E-02	3.73E-02	2.05E-02
Te-131	2.05E-01	7.50E-01	2.38E-18	0.00E+00
Te-131m	4.86E-02	1.78E-01	7.05E-02	7.34E-09
Te-132	3.49E-01	1.28E+00	7.12E-01	1.49E-03
Te-133	2.75E-01	1.01E+00	1.48E-35	0.00E+00
Te-133m	2.27E-01	8.30E-01	8.59E-09	1.17E-235
Te-134	4.57E-01	1.68E+00	4.94E-11	0.00E+00
Ba-137m	2.98E-02	1.09E-01	1.13E-171	0.00E+00
Ba-139	4.49E-01	1.65E+00	8.57E-06	2.65E-154
Ba-140	4.49E-01	1.65E+00	1.07E+00	2.22E-01
Ba-142	3.91E-01	1.43E+00	1.28E-41	0.00E+00
Sr-89	2.63E-01	9.63E-01	6.54E-01	4.40E-01

Table 11.1-15 - Fission Product Activity in the Waste Treatment System at STP				
NUCLIDE	SPECIFIC ACTIVITY AT STP with ion exchange RCS Concentration $\mu\text{Ci/cc}$	LRWS As-Received Concentration $\mu\text{Ci/cc}$	Annual 1 Day Decay $\mu\text{Ci/cc}$	Annual 30 Day Decay $\mu\text{Ci/cc}$
Sr-90	2.45E-02	8.98E-02	6.18E-02	6.17E-02
Sr-91	3.26E-01	1.19E+00	1.43E-01	1.42E-23
Sr-92	3.41E-01	1.25E+00	1.86E-03	9.41E-81
Sr-93	3.79E-01	1.39E+00	3.95E-59	0.00E+00
Sr-94	3.75E-01	1.37E+00	0.00E+00	0.00E+00
Mo-99	4.57E+00	4.57E+00	2.45E+00	1.63E-03
Mo-101	4.13E+00	4.13E+00	5.89E-30	0.00E+00
Pd-109	9.33E-02	3.42E-01	7.00E-02	3.59E-17
Rh-103m	3.88E-01	1.42E+00	1.85E-08	2.06E-232
Rh-105	2.49E-01	9.13E-01	3.93E-01	4.68E-07
Rh-105m	7.61E-02	2.79E-01	0.00E+00	0.00E+00
Rh-106	1.54E-01	5.64E-01	0.00E+00	0.00E+00
Ru-103	3.88E-01	1.42E+00	9.63E-01	5.77E-01
Ru-106	1.40E-01	5.14E-01	3.53E-01	3.35E-01
Tc-99m	4.04E-01	1.48E+00	6.41E-02	8.97E-37
Tc-101	4.13E-01	1.51E+00	3.15E-31	0.00E+00
Tc-104	3.20E-01	1.17E+00	1.68E-24	0.00E+00
Tc-105	2.65E-01	9.72E-01	6.32E-58	0.00E+00
Ce-141	4.14E-01	1.52E+00	1.02E+00	5.52E-01
Ce-143	3.91E-01	1.43E+00	5.96E-01	2.67E-07
Ce-144	3.20E-01	1.17E+00	8.05E-01	7.50E-01
Np-239	5.14E+00	1.88E+01	9.66E+00	1.90E-03
Pu-238	8.32E-04	3.05E-03	2.10E-03	2.10E-03
Pu-239	1.07E-04	3.92E-04	2.70E-04	2.70E-04
Pu-240	1.40E-04	5.12E-04	3.53E-04	3.53E-04
Pu-241	3.60E-02	1.32E-01	9.09E-02	9.06E-02
Pu-242	5.93E-07	2.17E-06	1.50E-06	1.50E-06
Th-228	9.33E-10	3.42E-09	2.35E-09	2.29E-09
Am-241	4.33E-05	1.59E-04	1.09E-04	1.09E-04
Cm-242	1.22E-02	4.49E-02	3.08E-02	2.72E-02
Cm-244	1.18E-03	4.34E-03	2.99E-03	2.98E-03
Eu-154	1.82E-03	6.67E-03	4.59E-03	4.56E-03
Eu-155	7.91E-04	2.90E-03	1.99E-03	1.97E-03
Eu-156	5.71E-02	2.09E-01	1.38E-01	3.67E-02
Eu-157	6.06E-03	2.22E-02	5.12E-03	8.12E-17
Eu-158	2.13E-03	7.80E-03	1.94E-12	2.94E-286
Eu-159	1.08E-03	3.94E-03	3.09E-27	0.00E+00
Gd-159	1.46E-03	5.34E-03	1.50E-03	7.76E-15
Ho-166	2.02E-05	7.41E-05	2.74E-05	4.19E-13
La-140	4.61E-01	1.69E+00	7.69E-01	4.84E-06

Table 11.1-15 - Fission Product Activity in the Waste Treatment System at STP				
NUCLIDE	SPECIFIC ACTIVITY AT STP with ion exchange RCS Concentration $\mu\text{Ci/cc}$	LRWS As-Received Concentration $\mu\text{Ci/cc}$	Annual 1 Day Decay $\mu\text{Ci/cc}$	Annual 30 Day Decay $\mu\text{Ci/cc}$
La-141	4.11E-01	1.51E+00	1.49E-02	5.45E-56
La-142	4.03E-01	1.48E+00	1.78E-05	1.94E-143
La-143	3.88E-01	1.42E+00	2.19E-31	0.00E+00
Nb-95	4.33E-01	1.59E+00	1.07E+00	6.03E-01
Nb-95m	4.93E-03	1.81E-02	1.03E-02	3.92E-05
Nb-97	4.04E-01	1.48E+00	9.93E-07	4.77E-181
Nb-97m	3.81E-01	1.39E+00	0.00E+00	0.00E+00
Nd-147	1.65E-01	6.06E-01	3.92E-01	6.28E-02
Pm-147	5.18E-02	1.90E-01	1.31E-01	1.28E-01
Pm-148	4.13E-02	1.51E-01	9.17E-02	2.17E-03
Pm-148m	8.09E-03	2.96E-02	2.01E-02	1.23E-02
Pm-149	1.40E-01	5.12E-01	2.58E-01	2.92E-05
Pm-151	4.89E-02	1.79E-01	6.87E-02	2.89E-09
Pr-142	1.47E-02	5.40E-02	1.56E-02	1.73E-13
Pr-143	3.82E-01	1.40E+00	9.17E-01	2.08E-01
Pr-144	3.21E-01	1.18E+00	6.73E-26	0.00E+00
Sm-153	1.09E-01	4.01E-01	1.93E-01	5.72E-06
Tb-160	2.45E-04	8.98E-04	6.12E-04	4.64E-04
Y-90	2.52E-01	2.54E-01	1.35E-01	7.29E-05
Y-91	3.35E+00	3.38E+00	2.30E+00	1.63E+00
Y-91m	1.89E+00	1.91E+00	2.51E-09	3.69E-262
Y-92	3.44E+00	3.47E+00	2.17E-02	1.46E-61
Y-93	2.57E+00	2.60E+00	3.44E-01	6.27E-22
Y-94	4.04E+00	4.07E+00	1.87E-23	0.00E+00
Y-95	4.17E+00	4.20E+00	1.54E-41	0.00E+00
Zr-95	4.31E-01	1.58E+00	1.08E+00	7.86E-01
Zr-97	4.01E-01	1.47E+00	3.78E-01	1.52E-13
H-3	3.32E+00	3.32E+00	2.29E+00	2.28E+00
Rn-220	9.39E-10	3.44E-09	0.00E+00	0.00E+00
Total	3.25E+02	3.91E+02	1.05E+02	1.11E+01

****NOTE:** All noble gases are assumed to be released from solution immediately after entering the LRWS.

The operating plant anticipated annual quantities of liquid waste releases and the corresponding annual average concentrations in the discharge tunnel are given in Table 11.1-16 for those nuclides expected to have annual average concentrations greater than 1×10^{-20} $\mu\text{Ci/cc}$. As illustrated by the table, it is expected that no single nuclide will exceed 1 percent of 10 CFR Part 20 limits on an annual average basis. Cumulative dose contributions from radioactive materials in liquid effluents released to unrestricted areas shall be determined on a quarterly basis in accordance with the ODCM. The total annual average concentration of liquid wastes discharged, excluding tritium, is not expected to exceed 8.2×10^{-8} $\mu\text{Ci/cc}$. The operating plant expected annual average concentration of tritium in the discharge tunnel was approximately 2.14×10^{-8} $\mu\text{Ci/cc}$.

For the purposes of calculating the operating plant anticipated concentrations, an annual average discharge tunnel flow of 305,000 gpm was used. This average flow was obtained by assuming the use of two circulating water pumps and one raw water pump during six cold months of the year and use of three circulating water pumps and one raw water pump during the six warmer months. For Bounding worst case scenario of reduced circulating water (120,000 gpm) the expected annual average concentration of liquid wastes is still below the 10 CFR Part 20 limits.

Effluents shall be limited to ten times 10 CFR Part 20, Appendix B, Table 2, Column 2 concentrations at discharge.

Calculations have been made to determine the downstream concentration of radionuclides discharged in the circulating water discharge from the Fort Calhoun Station into the Missouri River. These calculations were based on a model developed and experimentally verified by Yotsukura, Fischer and Sayre in: "Measurements of Mixing Characteristics of the Missouri River between Sioux City, Iowa, and Plattsmouth, Nebraska, U. S. Geological Survey Water Supply paper 1899-G, U. S. Government Printing Office, Washington: 1970". The computer code described in this publication was obtained by OPPD and its applicability confirmed by comparison with experimental data contained in the paper for a center-of-stream source of dye and its dispersion in the river reach adjacent to the plant site.

The calculated maximum concentration of wastes is shown in Figure 11.1-1 as a function of distance. Conditions are shown for a maximum distance of 19.5 miles, which corresponds to the location of the municipal water intake for the city of Omaha.

The source is assumed to be a continuous release of material from the bank which is uniformly mixed with 5% of the total river discharge and the 5% stream tube has the same concentration from the point of injection to 200 feet downstream.

Table 11.1-16 - Anticipated Quantities and Concentrations of Principle Radionuclides in the Discharge Tunnel Including Hypothetical Bounding Case and Comparison to 10 CFR 20 Limits

NUCLIDE	Total Activity Released Annually	Average Annual Concentration	Bounding Case Annual Concentration	10 CFR 20 Limits Appendix B Table 2, Col 2	Fraction of Limits Annual Concentration
	Ci	µCi/cc	µCi/cc	µCi/cc	Limit
Kr-83m	3.42E-118	5.64E-127	2.01E-125		NA
Kr-85	7.73E-01	1.27E-09	4.55E-08		NA
Kr-85m	8.00E-48	1.32E-56	4.71E-55		NA
Kr-87	1.57E-169	2.58E-178	9.23E-177		NA
Kr-88	2.63E-75	4.33E-84	1.55E-82		NA
Kr-89	0.00E+00	0.00E+00	0.00E+00		NA
Kr-90	0.00E+00	0.00E+00	0.00E+00		NA
Xe-131m	1.14E+00	1.88E-09	6.74E-08		NA
Xe-133	8.57E+00	1.41E-08	5.05E-07		NA
Xe-133m	3.31E-04	5.46E-13	1.95E-11		NA
Xe-135	1.02E-22	1.68E-31	6.02E-30		NA
Xe-135m	0.00E+00	0.00E+00	0.00E+00		NA
Xe-137	0.00E+00	0.00E+00	0.00E+00		NA
Xe-138	0.00E+00	0.00E+00	0.00E+00		NA
Br-82	8.01E-09	1.32E-17	4.72E-16	4.00000000E-05	0.000000%
Br-83	2.38E-91	3.93E-100	1.40E-98	9.00000000E-04	0.000000%
Br-85	0.00E+00	0.00E+00	0.00E+00		NA
I-129	1.35E-07	2.22E-16	7.93E-15	2.00000000E-07	0.000000%
I-130	2.38E-19	3.93E-28	1.40E-26	2.00000000E-05	0.000000%
I-131	1.31E+00	2.17E-09	7.74E-08	1.00000000E-06	0.216565%
I-132	6.96E-95	1.15E-103	4.10E-102	1.00000000E-04	0.000000%
I-133	2.82E-10	4.65E-19	1.66E-17	7.00000000E-06	0.000000%
I-134	5.44E-247	8.96E-256	3.20E-254	4.00000000E-04	0.000000%
I-135	7.30E-33	1.20E-41	4.30E-40	3.00000000E-05	0.000000%
I-136	0.00E+00	0.00E+00	0.00E+00		NA
Cs-132	5.50E-06	9.07E-15	3.24E-13	4.00000000E-05	0.000000%
Cs-134	5.97E-01	9.84E-10	3.52E-08	9.00000000E-07	0.109352%
Cs-134m	4.76E-76	7.84E-85	2.80E-83		NA
Cs-135m	6.93E-247	1.14E-255	4.08E-254	1.00000000E-03	0.000000%
Cs-136	3.95E-02	6.50E-11	2.32E-09	6.00000000E-06	0.001084%
Cs-137	4.60E-01	7.58E-10	2.71E-08	1.00000000E-06	0.075806%
Cs-138	0.00E+00	0.00E+00	0.00E+00	4.00000000E-04	0.000000%

Table 11.1-16 - Anticipated Quantities and Concentrations of Principle Radionuclides in the Discharge Tunnel Including Hypothetical Bounding Case and Comparison to 10 CFR 20 Limits

NUCLIDE	Total Activity Released Annually	Average Annual Concentration	Bounding Case Annual Concentration	10 CFR 20 Limits Appendix B Table 2, Col 2	Fraction of Limits Annual Concentration
	Ci	µCi/cc	µCi/cc	µCi/cc	Limit
Cs-139	0.00E+00	0.00E+00	0.00E+00		NA
Cs-140	0.00E+00	0.00E+00	0.00E+00		NA
Rb-86	5.42E-03	8.93E-12	3.19E-10	7.00000000E-06	0.000128%
Rb-88	0.00E+00	0.00E+00	0.00E+00	4.00000000E-04	0.000000%
Rb-89	0.00E+00	0.00E+00	0.00E+00	9.00000000E-04	0.000000%
Rb-90	0.00E+00	0.00E+00	0.00E+00		NA
Rb-90m	0.00E+00	0.00E+00	0.00E+00		NA
Ag-110	0.00E+00	0.00E+00	0.00E+00		NA
Ag-110m	1.38E-02	2.27E-11	8.11E-10	6.00000000E-06	0.000378%
Ag-111	1.35E-02	2.22E-11	7.94E-10	2.00000000E-05	0.000111%
Ag-112	9.80E-71	1.61E-79	5.77E-78	4.00000000E-05	0.000000%
As-76	4.26E-13	7.03E-22	2.51E-20	1.00000000E-05	0.000000%
Cd-115	2.64E-06	4.35E-15	1.55E-13	1.00000000E-05	0.000000%
Cd-115m	8.67E-04	1.43E-12	5.11E-11	4.00000000E-06	0.000036%
Ga-72	2.55E-20	4.20E-29	1.50E-27	2.00000000E-05	0.000000%
Ge-77	1.68E-22	2.77E-31	9.92E-30	1.00000000E-04	0.000000%
In-115m	1.48E-50	2.44E-59	8.73E-58	2.00000000E-04	0.000000%
Sb-122	1.60E-06	2.63E-15	9.40E-14	1.00000000E-05	0.000000%
Sb-124	1.94E-03	3.19E-12	1.14E-10	7.00000000E-06	0.000046%
Sb-125	3.10E-02	5.11E-11	1.83E-09	3.00000000E-05	0.000170%
Sb-127	1.41E-03	2.33E-12	8.33E-11	1.00000000E-05	0.000023%
Sb-129	6.51E-50	1.07E-58	3.83E-57	4.00000000E-05	0.000000%
Sb-130	0.00E+00	0.00E+00	0.00E+00	3.00000000E-04	0.000000%
Sb-130m	0.00E+00	0.00E+00	0.00E+00		NA
Sb-131	0.00E+00	0.00E+00	0.00E+00	2.00000000E-04	0.000000%
Sb-132	0.00E+00	0.00E+00	0.00E+00		NA
Sb-132m	0.00E+00	0.00E+00	0.00E+00		NA
Sb-133	0.00E+00	0.00E+00	0.00E+00		NA
Se-83	0.00E+00	0.00E+00	0.00E+00	4.00000000E-04	0.000000%
Sn-121	2.87E-10	4.73E-19	1.69E-17	8.00000000E-05	0.000000%
Sn-123	1.96E-03	3.23E-12	1.15E-10	9.00000000E-06	0.000036%
Sn-125	2.07E-03	3.42E-12	1.22E-10	6.00000000E-06	0.000057%
Sn-127	8.16E-105	1.34E-113	4.81E-112	9.00000000E-05	0.000000%
Te-127	2.06E-24	3.39E-33	1.21E-31	1.00000000E-04	0.000000%
Te-127m	4.24E-02	6.99E-11	2.50E-09	9.00000000E-06	0.000777%
Te-129	1.72E-187	2.84E-196	1.01E-194	4.00000000E-04	0.000000%
Te-129m	1.20E-01	1.97E-10	7.05E-09	7.00000000E-06	0.002817%
Te-131	0.00E+00	0.00E+00	0.00E+00	8.00000000E-05	0.000000%

Table 11.1-16 - Anticipated Quantities and Concentrations of Principle Radionuclides in the Discharge Tunnel Including Hypothetical Bounding Case and Comparison to 10 CFR 20 Limits

NUCLIDE	Total Activity Released Annually	Average Annual Concentration	Bounding Case Annual Concentration	10 CFR 20 Limits Appendix B Table 2, Col 2	Fraction of Limits Annual Concentration
	Ci	µCi/cc	µCi/cc	µCi/cc	Limit
Te-131m	4.28E-08	7.05E-17	2.52E-15	8.00000000E-06	0.000000%
Te-132	8.70E-03	1.43E-11	5.12E-10	9.00000000E-06	0.000159%
Te-133	0.00E+00	0.00E+00	0.00E+00	4.00000000E-04	0.000000%
Te-133m	6.83E-235	1.13E-243	4.02E-242	9.00000000E-05	0.000000%
Te-134	0.00E+00	0.00E+00	0.00E+00	3.00000000E-04	0.000000%
Ba-137m	0.00E+00	0.00E+00	0.00E+00		NA
Ba-139	1.55E-153	2.55E-162	9.11E-161	2.00000000E-04	0.000000%
Ba-140	1.30E+00	2.13E-09	7.63E-08	8.00000000E-06	0.026676%
Ba-142	0.00E+00	0.00E+00	0.00E+00	7.00000000E-04	0.000000%
Sr-89	2.56E+00	4.22E-09	1.51E-07	8.00000000E-06	0.052776%
Sr-90	3.60E-01	5.93E-10	2.12E-08	5.00000000E-07	0.118514%
Sr-91	8.29E-23	1.37E-31	4.88E-30	2.00000000E-05	0.000000%
Sr-92	5.49E-80	9.04E-89	3.23E-87	4.00000000E-05	0.000000%
Sr-93	0.00E+00	0.00E+00	0.00E+00		NA
Sr-94	0.00E+00	0.00E+00	0.00E+00		NA
Mo-99	9.49E-03	1.56E-11	5.59E-10	2.00000000E-05	0.000078%
Mo-101	0.00E+00	0.00E+00	0.00E+00	7.00000000E-04	0.000000%
Pd-109	2.09E-16	3.45E-25	1.23E-23	3.00000000E-05	0.000000%
Rh-103m	1.20E-231	1.98E-240	7.06E-239	6.00000000E-03	0.000000%
Rh-105	2.73E-06	4.50E-15	1.61E-13	5.00000000E-05	0.000000%
Rh-105m	0.00E+00	0.00E+00	0.00E+00		NA
Rh-106	0.00E+00	0.00E+00	0.00E+00		NA
Ru-103	3.36E+00	5.54E-09	1.98E-07	3.00000000E-05	0.018473%
Ru-106	1.95E+00	3.22E-09	1.15E-07	3.00000000E-06	0.107204%
Tc-99m	5.23E-36	8.62E-45	3.08E-43		NA
Tc-101	0.00E+00	0.00E+00	0.00E+00	2.00000000E-03	0.000000%
Tc-104	0.00E+00	0.00E+00	0.00E+00	4.00000000E-04	0.000000%
Tc-105	0.00E+00	0.00E+00	0.00E+00		NA
Ce-141	3.22E+00	5.30E-09	1.89E-07	3.00000000E-05	0.017661%
Ce-143	1.56E-06	2.57E-15	9.18E-14	2.00000000E-05	0.000000%
Ce-144	4.37E+00	7.21E-09	2.58E-07	3.00000000E-06	0.240264%
Np-239	1.11E-02	1.82E-11	6.52E-10	2.00000000E-05	0.000091%
Pu-238	1.22E-02	2.02E-11	7.21E-10	2.00000000E-08	0.100818%
Pu-239	1.57E-03	2.59E-12	9.27E-11	2.00000000E-08	0.012971%
Pu-240	2.06E-03	3.39E-12	1.21E-10	2.00000000E-08	0.016934%
Pu-241	5.28E-01	8.70E-10	3.11E-08	1.00000000E-06	0.086990%
Pu-242	8.73E-06	1.44E-14	5.14E-13	2.00000000E-08	0.000072%
Th-228	1.33E-08	2.20E-17	7.85E-16	2.00000000E-07	0.000000%

Table 11.1-16 - Anticipated Quantities and Concentrations of Principle Radionuclides in the Discharge Tunnel Including Hypothetical Bounding Case and Comparison to 10 CFR 20 Limits

NUCLIDE	Total Activity Released Annually	Average Annual Concentration	Bounding Case Annual Concentration	10 CFR 20 Limits Appendix B Table 2, Col 2	Fraction of Limits Annual Concentration
	Ci	µCi/cc	µCi/cc	µCi/cc	Limit
Am-241	6.37E-04	1.05E-12	3.75E-11	2.00000000E-08	0.005245%
Cm-242	1.59E-01	2.61E-10	9.34E-09	7.00000000E-07	0.037332%
Cm-244	1.74E-02	2.86E-11	1.02E-09	3.00000000E-08	0.095298%
Eu-154	2.66E-02	4.38E-11	1.57E-09	7.00000000E-06	0.000626%
Eu-155	1.15E-02	1.89E-11	6.77E-10	5.00000000E-05	0.000038%
Eu-156	2.14E-01	3.52E-10	1.26E-08	8.00000000E-06	0.004400%
Eu-157	4.74E-16	7.80E-25	2.79E-23	3.00000000E-05	0.000000%
Eu-158	1.71E-285	2.82E-294	1.01E-292	3.00000000E-04	0.000000%
Eu-159	0.00E+00	0.00E+00	0.00E+00		NA
Gd-159	4.52E-14	7.46E-23	2.67E-21	4.00000000E-05	0.000000%
Ho-166	2.44E-12	4.02E-21	1.44E-19	1.00000000E-05	0.000000%
La-140	2.82E-05	4.65E-14	1.66E-12	9.00000000E-06	0.000001%
La-141	3.18E-55	5.23E-64	1.87E-62	5.00000000E-05	0.000000%
La-142	1.13E-142	1.86E-151	6.67E-150	1.00000000E-04	0.000000%
La-143	0.00E+00	0.00E+00	0.00E+00	5.00000000E-04	0.000000%
Nb-95	3.51E+00	5.79E-09	2.07E-07	3.00000000E-05	0.019299%
Nb-95m	2.29E-04	3.77E-13	1.35E-11	3.00000000E-05	0.000001%
Nb-97	2.78E-180	4.59E-189	1.64E-187	3.00000000E-04	0.000000%
Nb-97m	0.00E+00	0.00E+00	0.00E+00		NA
Nd-147	3.66E-01	6.03E-10	2.16E-08	2.00000000E-05	0.003016%
Pm-147	7.46E-01	1.23E-09	4.40E-08	7.00000000E-05	0.001757%
Pm-148	1.27E-02	2.09E-11	7.46E-10	7.00000000E-06	0.000298%
Pm-148m	7.19E-02	1.18E-10	4.24E-09	1.00000000E-05	0.001185%
Pm-149	1.70E-04	2.80E-13	1.00E-11	2.00000000E-05	0.000001%
Pm-151	1.68E-08	2.78E-17	9.93E-16	2.00000000E-05	0.000000%
Pr-142	1.01E-12	1.66E-21	5.93E-20	1.00000000E-05	0.000000%
Pr-143	1.22E+00	2.00E-09	7.16E-08	2.00000000E-05	0.010012%
Pr-144	0.00E+00	0.00E+00	0.00E+00	6.00000000E-04	0.000000%
Sm-153	3.34E-05	5.50E-14	1.97E-12	3.00000000E-05	0.000000%
Tb-160	2.70E-03	4.45E-12	1.59E-10	1.00000000E-05	0.000045%
Y-90	4.25E-04	7.00E-13	2.50E-11	7.00000000E-06	0.000010%
Y-91	9.51E+00	1.57E-08	5.60E-07	8.00000000E-06	0.195969%
Y-91m	2.15E-261	3.54E-270	1.27E-268	2.00000000E-03	0.000000%
Y-92	8.51E-61	1.40E-69	5.01E-68	4.00000000E-05	0.000000%
Y-93	3.66E-21	6.02E-30	2.15E-28	2.00000000E-05	0.000000%
Y-94	0.00E+00	0.00E+00	0.00E+00	4.00000000E-04	0.000000%
Y-95	0.00E+00	0.00E+00	0.00E+00	7.00000000E-04	0.000000%
Zr-95	4.58E+00	7.55E-09	2.70E-07	2.00000000E-05	0.037757%

Table 11.1-16 - Anticipated Quantities and Concentrations of Principle Radionuclides in the Discharge Tunnel Including Hypothetical Bounding Case and Comparison to 10 CFR 20 Limits					
NUCLIDE	Total Activity Released Annually Ci	Average Annual Concentration μCi/cc	Bounding Case Annual Concentration μCi/cc	10 CFR 20 Limits Appendix B Table 2, Col 2 μCi/cc	Fraction of Limits Annual Concentration Limit
Zr-97	8.88E-13	1.46E-21	5.23E-20	9.00000000E-06	0.000000%
H-3	1.33E+01	2.19E-08	7.82E-07	1.00000000E-03	0.002188%
Rn-220	0.00E+00	0.00E+00	0.00E+00		NA
Total	6.46E+01	1.06E-07	3.80E-06		1.62E-02

Annual Average Concentration (excluded Tritium) = 8.2E-08 μCi/cc

11.1.2.8 Availability and Reliability

The liquid waste system will function properly with wide variations in processed volume.

The liquid waste process equipment is dependent on the electrical systems, and on the demineralized water system. Collection of waste is chiefly by gravity and is therefore, almost wholly independent of auxiliary systems.

The liquid waste system has a duplicate sampling and analyzing capability. Liquid waste is analyzed at the waste hold-up tanks or at the Filtration/Ion exchange influent and then again at the monitor tanks, thus ensuring that effluent to the overboard header has always had two independent analyses. In addition, the radiation monitor at the overboard header automatically stops this flow if it exceeds a pre-determined concentration of radioactivity.

The transport pumping sets in the liquid waste system have redundancies, with one of the two pumps being a spare for the other.

Redundant volume is provided in the waste holdup tanks, spent regenerant tanks, and the hotel waste tanks; two tanks are furnished for spent regenerant and two for hotel wastes whereas three tanks are furnished for waste holdup. In the case of two tanks, the second is normally a complete spare of the first in volume capacity. In the case of three tanks, the capacity of 1-1/2 tanks is spare volume. The usual mode of operation is for one tank to be collecting while another tank is being discharged to treatment.

11.1.2.9 Operation

The liquid waste processing system is operated to minimize the amount of radioactivity contained in liquid effluents from the plant. A program of equipment operation and maintenance will be in effect to provide maximum system availability. Only under unusual circumstances of severe need would a system component be bypassed if it could, within detectable limits, significantly reduce the activity of the waste liquid. Waste liquids are segregated as to radioactivity level and point of origin. Liquid wastes are held for sufficient duration to allow decay of short-lived radioactive nuclides prior to processing and release.

Hotel waste tanks are normally diverted for processing if the activity level is above the limits established for release. All liquids are sampled and analyzed prior to release.

System flexibility ensures that proper treatment brings waste quantities and activities well within the limits of 10 CFR Part 20 and 40 CFR 190. In addition to this flexibility, it is possible to reprocess any volume of liquid if this need should occur.

The radiation monitors may be inoperable and liquid releases may continue provided the requirements of the ODCM are complied with. All liquid radioactive wastes originating within the containment, CARP and Radioactive Waste Processing Building are pumped to the auxiliary building. All radioactive liquids in the auxiliary building are collected in the RWDS. The radiation monitors utilized for monitoring RWDS are described in [DSAR Section 11.2.3](#).

11.1.2.10 Tests and Inspections

The purpose of the testing and inspection program was to ensure that the liquid waste system components meet design objectives and specifications.

All equipment in the system was subject to two types of test and inspections: manufacturer's shop tests and on-site tests.

Shop Tests

All equipment was tested and inspected in the manufacturer's shop in accordance with the then applicable codes and standards. In addition, some equipment was given performance type tests in the manufacturer's shop.

After preliminary operation to demonstrate the mechanical integrity and suitability of all components, a short term test program to demonstrate specific modes and methods of operation was undertaken. Chemical tests, such as boron concentration, and operating parameters, such as flow rates, were recorded during the test program. After the successful completion of the above tests, the equipment was partially disassembled and shipped to the plant site.

On-Site Tests

On-site tests of the performance type to ensure that the overall liquid waste system functions in a safe and efficient manner were conducted prior to actual plant startup. Provisions were made to test the full operational sequence of the system. Pumps were started, valves operated, and instruments put into service. Flow paths, flow capacity, and mechanical operability were thoroughly checked. Pressure, temperature, flow and level indicating instruments were calibrated and checked for performance. All safety equipment, including alarms were thoroughly tested. Special emphasis was placed on the proper functioning of the liquid waste instrumentation and controls on the waste control panel.

11.1.3 Gaseous Wastes

11.1.3.1 General

The waste gas header is vented to the Auxiliary Building atmosphere. All waste gas release will be monitored in the Auxiliary Building stack (See DSAR section 11.2.3).

The calculated annual air dose at any location which could be occupied by individuals in unrestricted areas shall not exceed 10 millirads for gamma radiation, 20 millirads for beta radiation and 15 millirems to any organ for iodine-131, tritium, and other particulates with half-lives greater than eight days as required by 10 CFR Part 50, Appendix I.

The methods of dose calculation are defined in the Offsite Dose Calculation Manual.

There may be small amounts of radioactive gas in the Radioactive Waste Processing and CARP buildings. The amount of gas will be extremely low and releases will be measured and recorded.

The annual average dispersion factor (χ/Q) for gaseous releases used to determine exposures in the unrestricted area is calculated using data obtained from the meteorological program. This program is described in detail in [DSAR Section 2.5](#). The annual average value of χ/Q is specified in the ODCM. A revision of this value, either due to subsequent data or revised criteria, would affect the gaseous release concentration in direct ratio to the change. The ODCM ensures that all releases are within applicable criteria.

11.1.3.2 Sources of Waste Gas

The waste gas header is vented to the Auxiliary Building atmosphere. All waste gas release will be monitored in the Auxiliary Building stack (See DSAR section 11.2.3).

Table 11.1-18 - Deleted

11.1.3.3 Processing of Waste Gases

Waste gases from all of the sources mentioned above are collected in the vent header as shown in the process flow diagram P&ID 11405-M-98. The gas is then vented to the Auxiliary Building Ventilation system for discharge.

11.1.3.4 Deleted

11.1.3.5 Deleted

11.1.3.6 Deleted

Table 11.1-21 - Deleted

11.1.3.7 Design Evaluation

Radiological Gases Released from Auxiliary Building

It is expected that small amounts of radioactive gases, halogens and particulates may leak into the auxiliary building atmosphere. Potential sources include the following:

a. Venting of Spent Regenerant Tanks.

The vapor spaces of the Spent Regenerant Tanks (SRT) in the RWDS are vented to the Auxiliary Building Ventilating System.

The only liquids collected in the SRT are those which have been depressurized and aerated in the process of becoming a waste. WDS design in addition to reducing activity releases to the extent practicable, must also be inherently safe.

Liquids collected in the SRT along with their design activities are listed under "Auxiliary Systems Process Wastes", Section 11.1.2.1. Waste volumes for these sources as listed in Table 11.1-23 indicate that a total of 11,000 cu. ft. of liquid per cycle is discharged to the SRT. Design activities for liquids entering the SRT are expected to be variable over a range of 10^{-7} to $6.0 \mu\text{Ci/cc}$ as shown. The maximum amount of gaseous activity that may be present in the Auxiliary Building Ventilating System from the SRT has been calculated and is summarized along with applicable design parameters in Table 11.1-23.

It is expected that small amounts of radioactive gases, halogens and particulates may be released to the CARP and Radioactive Waste Processing Building HVAC systems. The HVAC systems in these two buildings are designed to capture such releases and maintain personnel exposure ALARA. The sources for airborne radioactivity in the CARP and Radioactive Waste Processing Buildings were previously located in the existing Auxiliary Building. Therefore they do not constitute a new source of airborne radioactive releases and the releases tabulated in Table 11.1-23 remain unchanged.

Potential sources include the following:

- a. Radioactive Waste Processing Building
 - 1. DAW sorting.
 - 2. DAW compaction
 - 3. DAW Decontamination
 - 4. Radwaste Filtration and Ion Exchange System
 - 5. Radwaste Solidification System

- b. CARP Building
 - 1. Laboratory

Table 11.1-23 - Maximum Gaseous Release, Spent Regenerant Tanks

Design

Liquid volume cycle [1] to SRT = 11,000 cu. ft.
Maximum annual average activity, liquid mixture = 2.18 $\mu\text{Ci/cc}$
Fraction volatiles present in liquid [2] = 0.5
Fraction volatiles immediately released = 1.0
Auxiliary Building ventilation rate = 7.25×10^4 SCFM

Maximum Average Activity [3]

Concentration in Aux. Bldg. Vent. Sys. = 3.15×10^{-7} $\mu\text{Ci/cc}$
Maximum SRT release to Auxiliary Building Ventilation System is approximately 0.9 Ci/day of waste gases. The majority of the activity is from noble gases.

[1] One cycle is equivalent to 530 full power days.

[2] Estimate is conservative since liquid has been previously aerated.

[3] Volatile composition as shown in Table 11.1-21.

c. Discussion of RWDS Vent Connections.

It is concluded that under design conditions for failed fuel the liquids contained in Spent Regenerant are not a significant source of gaseous activity release.

d. Relief Valve Discharges.

The RWDS is designed, by making maximum use of connected tankage, to contain relief discharges with the system. Referring to P&ID 11405-M-98, the RWDS waste gas circuit flow diagram, the Vent Header is connected through unchecked piping to the vapor spaces of all three Waste Holdup Tanks. Locked open valves WD-441, 442 and 443 and the tank vent lines as shown in P&ID 11405-M-8 provide an interconnecting manifold between the vapor spaces of the three tanks.

e. Total Releases from Auxiliary Building

Of the sources discussed above, the major one is projected to be released from venting of the concentrate tanks. Total gaseous releases from the auxiliary building over a year's time have been assumed to be 150% of the releases from the concentrate tank vents, based on the total quantity of liquid wastes to be processed (see Table 11.1-2). It is further assumed that the decontamination factors given in item c above apply and that the HEPA filters in the auxiliary building discharge have a 90% efficiency for removal of particulates. The resulting releases are as given in Table 11.1-25.

Radioactive Gases Released from Containment

Gases from Containment will be minimal following plant shutdown and no longer having Reactor Coolant. Containment gas will be processed through normal Auxiliary Building ventilation. Table 11.1-25 will not be updated. New release data would be conservative.

Total Radioactive Gaseous Releases

NOTE: Containment gas contributors have been removed from this section. Table 11.1-25 will not be updated. New release data would be conservative.

The total expected annual activity release to the atmosphere from the (1) waste gas system, (2) containment purges, (3) auxiliary building ventilation and (4) primary-to-secondary leakage and (5) Radioactive Waste Processing and CARP buildings are (accounted for in the Auxiliary Building Ventilation Release) listed in Table 11.1-25. Also given are the average concentration at the boundary of the unrestricted area. An average-annual dispersion factor of 5.0×10^{-6} sec/m³ has been used to determine the isotopic activities at the boundary (Amendment 113^(11.4.12, 11.4.13)). The maximum whole body dose at the boundary of the restricted area, consistent with the average concentrations at the boundary in Table 11.1-25, is approximately 1.08 millirad/year, based on continuous occupancy.

Table 11.1-25 - Annual Releases of Radioactive Gases and Particulates

NUCLIDE	Annual Release Gas 12 month basis Ci	SG BLOWDOWN Secondary Activity 45 day release Ci	GASEOUS RELEASE Auxiliary Building Annually Ci	GASEOUS RELEASE Containment Building Annually Ci	TOTAL CURIES Released Annually Ci	CONCENTRATION Site Boundry µCi/cc	10 CFR 20 LIMITS Appendix B Table 2 Column 1 µCi/cc	FRACTION OF 10 CFR 20 Limits
Kr-83m	7.65E-117	9.01E+00	6.910E+00	2.76E-01	1.62E+01	2.5685E-12	NA	
Kr-85	1.73E+01	7.92E-01	6.069E-01	6.58E+00	2.53E+01	4.0058E-12	NA	
Kr-85m	1.79E-46	1.91E+01	1.462E+01	1.43E+00	3.51E+01	5.5690E-12	NA	
Kr-87	3.50E-168	3.83E+01	2.937E+01	8.14E-01	6.85E+01	1.0862E-11	NA	
Kr-88	5.87E-74	5.35E+01	4.102E+01	2.54E+00	9.71E+01	1.5390E-11	NA	
Kr-89	0.00E+00	6.72E+01	5.150E+01	5.93E-02	1.19E+02	1.8827E-11	NA	
Kr-90	0.00E+00	7.23E+01	5.538E+01	1.08E-02	1.28E+02	2.0238E-11	NA	
Xe-131m	2.56E+01	6.68E+00	5.123E+00	2.63E+01	6.37E+01	1.0100E-11	2E-6	5.05E-06
Xe-133	1.92E+02	4.60E+02	3.527E+02	9.49E+02	1.95E+03	3.0968E-10	5E-7	0.0006194
Xe-133m	7.40E-03	4.47E+00	3.429E+00	3.93E+00	1.18E+01	1.8767E-12	6E-7	3.128E-06
Xe-135	2.28E-21	5.32E+01	4.076E+01	8.12E+00	1.02E+02	1.6180E-11	7E-8	0.0002311
Xe-135m	0.00E+00	2.97E+01	2.277E+01	1.26E-01	5.26E+01	8.3414E-12	4E-8	0.0002085
Xe-137	0.00E+00	1.30E+02	9.951E+01	1.38E-01	2.29E+02	3.6381E-11	NA	
Xe-138	0.00E+00	1.24E+02	9.485E+01	4.85E-01	2.19E+02	3.4733E-11	2E-8	0.0017367
Br-82	4.89E-09	3.06E-07	2.348E-07	1.81E-08	5.64E-07	8.9423E-20	5E-9	1.788E-11
Br-83	1.45E-91	1.29E-05	9.894E-06	5.17E-08	2.29E-05	3.6235E-18	9E-8	4.026E-11
Br-85	0.00E+00	2.72E-05	2.087E-05	2.17E-09	4.81E-05	7.6257E-18	NA	
I-129	8.22E-08	0.00E+00	2.870E-12	3.12E-12	8.22E-08	1.3029E-20	4E-11	3.257E-10
I-130	1.45E-19	2.26E-06	1.729E-06	4.66E-08	4.03E-06	6.3920E-19	3E-9	2.131E-10
I-131	8.02E-01	4.85E-04	3.717E-04	1.44E-04	8.03E-01	1.2729E-13	2E-10	0.0006364
I-132	4.24E-95	1.45E-04	1.112E-04	5.54E-07	2.57E-04	4.0734E-17	2E-8	2.037E-09
I-133	1.72E-10	2.05E-04	1.575E-04	7.14E-06	3.70E-04	5.8662E-17	1E-9	5.866E-08
I-134	3.32E-247	2.29E-04	1.759E-04	3.36E-07	4.06E-04	6.4324E-17	6E-8	1.072E-09
I-135	4.45E-33	1.95E-04	1.494E-04	2.14E-06	3.47E-04	5.4942E-17	6E-9	9.157E-09
I-136	0.00E+00	9.14E-05	7.006E-05	3.54E-09	1.61E-04	2.5600E-17	NA	
Cs-132	3.36E-06	3.79E-09	2.907E-09	9.45E-10	3.36E-06	5.3340E-19	6E-9	8.89E-11

Table 11.1-25 - Annual Releases of Radioactive Gases and Particulates

NUCLIDE	Annual Release Gas 12 month basis	SG BLOWDOWN Secondary Activity 45 day release	GASEOUS RELEASE Auxiliary Building Annually	GASEOUS RELEASE Containment Building Annually	TOTAL CURIES Released Annually	CONCENTRATION Site Boundry	10 CFR 20 LIMITS Appendix B Table 2 Column 1	FRACTION OF 10 CFR 20 Limits
	Ci	Ci	Ci	Ci	Ci	µCi/cc	µCi/cc	
Cs-134	3.64E-01	1.71E-05	1.308E-05	1.40E-05	3.64E-01	5.7780E-14	2E-10	0.0002889
Cs-134m	2.90E-76	3.87E-06	2.963E-06	1.88E-08	6.85E-06	1.0855E-18	2E-7	5.428E-12
Cs-135m	4.23E-247	3.99E-06	3.056E-06	5.88E-09	7.05E-06	1.1175E-18	3E-7	3.725E-12
Cs-136	2.41E-02	5.32E-06	4.081E-06	2.23E-06	2.41E-02	3.8188E-15	9E-10	4.243E-06
Cs-137	2.81E-01	1.28E-05	9.820E-06	1.07E-05	2.81E-01	4.4505E-14	2E-10	0.0002225
Cs-138	0.00E+00	1.92E-04	1.472E-04	1.72E-07	3.39E-04	5.3814E-17	8E-8	6.727E-10
Cs-139	0.00E+00	1.80E-04	1.377E-04	4.64E-08	3.17E-04	5.0321E-17	NA	
Cs-140	0.00E+00	1.62E-04	1.239E-04	4.78E-09	2.86E-04	4.5277E-17	NA	
Rb-86	1.21E-02	1.68E-06	1.291E-06	8.46E-07	1.21E-02	1.9214E-15	1E-9	1.921E-06
Rb-88	0.00E+00	7.90E-04	6.056E-04	3.91E-07	1.40E-03	2.2133E-16	9E-8	2.459E-09
Rb-89	0.00E+00	1.04E-03	7.938E-04	4.38E-07	1.83E-03	2.9011E-16	2E-7	1.451E-09
Rb-90	0.00E+00	9.60E-04	7.360E-04	6.82E-08	1.70E-03	2.6894E-16	NA	
Rb-90m	0.00E+00	2.94E-04	2.255E-04	3.52E-08	5.20E-04	8.2387E-17	NA	
Ag-110	0.00E+00	1.66E-05	1.275E-05	1.90E-10	2.94E-05	4.6570E-18	NA	
Ag-110m	8.40E-03	4.16E-07	3.186E-07	3.32E-07	8.40E-03	1.3313E-15	1E-10	1.331E-05
Ag-111	8.22E-03	6.10E-06	4.677E-06	1.71E-06	8.23E-03	1.3055E-15	1E-9	1.305E-06
Ag-112	5.98E-71	2.80E-06	2.143E-06	1.47E-08	4.95E-06	7.8529E-19	1E-8	7.853E-11
As-76	2.60E-13	2.03E-09	1.554E-09	8.91E-11	3.67E-09	5.8199E-22	2E-9	2.91E-13
Cd-115	1.61E-06	8.29E-07	6.354E-07	7.40E-08	3.15E-06	4.9905E-19	2E-9	2.495E-10
Cd-115m	5.29E-04	3.84E-08	2.944E-08	2.56E-08	5.29E-04	8.3881E-17	1E-10	8.388E-07
Ga-72	1.56E-20	1.66E-09	1.269E-09	3.90E-11	2.96E-09	4.6983E-22	4E-9	1.175E-13
Ge-77	1.03E-22	7.03E-08	5.385E-08	1.33E-09	1.25E-07	1.9886E-20	8E-9	2.486E-12
In-115m	9.04E-51	8.29E-07	6.354E-07	6.21E-09	1.47E-06	2.3315E-19	6E-8	3.886E-12
Sb-122	9.74E-07	9.80E-08	7.509E-08	1.06E-08	1.16E-06	1.8351E-19	2E-9	9.175E-11
Sb-124	1.18E-03	7.61E-08	5.832E-08	5.36E-08	1.18E-03	1.8750E-16	3E-10	6.25E-07
Sb-125	1.89E-02	8.80E-07	6.745E-07	7.26E-07	1.89E-02	2.9997E-15	7E-10	4.285E-06

Table 11.1-25 - Annual Releases of Radioactive Gases and Particulates

NUCLIDE	Annual Release Gas 12 month basis Ci	SG BLOWDOWN Secondary Activity 45 day release Ci	GASEOUS RELEASE Auxiliary Building Annually Ci	GASEOUS RELEASE Containment Building Annually Ci	TOTAL CURIES Released Annually Ci	CONCENTRATION Site Boundry µCi/cc	10 CFR 20 LIMITS Appendix B Table 2 Column 1 µCi/cc	FRACTION OF 10 CFR 20 Limits
Sb-127	8.63E-04	8.70E-06	6.671E-06	1.34E-06	8.79E-04	1.3944E-16	1E-9	1.394E-07
Sb-129	3.97E-50	3.21E-05	2.460E-05	2.36E-07	5.69E-05	9.0245E-18	1E-8	9.024E-10
Sb-130	0.00E+00	1.07E-05	8.180E-06	1.17E-08	1.89E-05	2.9907E-18	9E-8	3.323E-11
Sb-130m	0.00E+00	4.47E-05	3.429E-05	7.84E-09	7.90E-05	1.2529E-17	NA	
Sb-131	0.00E+00	7.88E-05	6.037E-05	5.04E-08	1.39E-04	2.2067E-17	6E-8	3.678E-10
Sb-132	0.00E+00	4.67E-05	3.578E-05	5.46E-09	8.25E-05	1.3073E-17	NA	
Sb-132m	0.00E+00	4.55E-05	3.484E-05	3.54E-09	8.03E-05	1.2732E-17	NA	
Sb-133	0.00E+00	6.66E-05	5.106E-05	4.63E-09	1.18E-04	1.8656E-17	NA	
Se-83	0.00E+00	6.00E-06	4.602E-06	3.73E-09	1.06E-05	1.6823E-18	2E-7	8.411E-12
Sn-121	1.75E-10	8.12E-07	6.224E-07	3.67E-08	1.47E-06	2.3325E-19	2E-8	1.166E-11
Sn-123	1.19E-03	6.39E-08	4.901E-08	4.92E-08	1.19E-03	1.8946E-16	2E-10	9.473E-07
Sn-125	1.27E-03	4.98E-07	3.820E-07	1.70E-07	1.27E-03	2.0085E-16	5E-10	4.017E-07
Sn-127	4.98E-105	3.50E-06	2.683E-06	1.23E-08	6.20E-06	9.8235E-19	3E-8	3.275E-11
Te-127	1.26E-24	8.58E-06	6.578E-06	1.34E-07	1.53E-05	2.4246E-18	2E-8	1.212E-10
Te-127m	2.59E-02	1.43E-06	1.094E-06	1.08E-06	2.59E-02	4.1043E-15	4E-10	1.026E-05
Te-129	1.05E-187	3.06E-05	2.348E-05	5.93E-08	5.42E-05	8.5880E-18	9E-8	9.542E-11
Te-129m	7.30E-02	6.17E-06	4.733E-06	3.84E-06	7.30E-02	1.1576E-14	3E-10	3.859E-05
Te-131	0.00E+00	8.36E-05	6.410E-05	5.82E-08	1.48E-04	2.3430E-17	2E-8	1.172E-09
Te-131m	2.61E-08	1.99E-05	1.524E-05	9.96E-07	3.61E-05	5.7314E-18	1E-9	5.731E-09
Te-132	5.31E-03	1.43E-04	1.094E-04	1.86E-05	5.58E-03	8.8425E-16	9E-10	9.825E-07
Te-133	0.00E+00	1.13E-04	8.627E-05	3.92E-08	1.99E-04	3.1529E-17	8E-8	3.941E-10
Te-133m	4.17E-235	9.26E-05	7.099E-05	1.43E-07	1.64E-04	2.5963E-17	2E-8	1.298E-09
Te-134	0.00E+00	1.87E-04	1.433E-04	2.18E-07	3.30E-04	5.2391E-17	7E-8	7.484E-10
Ba-137m	0.00E+00	1.22E-05	9.335E-06	8.65E-10	2.15E-05	3.4111E-18	NA	
Ba-139	9.44E-154	1.84E-04	1.407E-04	4.32E-07	3.25E-04	5.1472E-17	4E-8	1.287E-09
Ba-140	7.90E-01	1.84E-04	1.409E-04	7.55E-05	7.91E-01	1.2534E-13	2E-9	6.267E-05

Table 11.1-25 - Annual Releases of Radioactive Gases and Particulates

NUCLIDE	Annual Release Gas 12 month basis	SG BLOWDOWN Secondary Activity 45 day release	GASEOUS RELEASE Auxiliary Building Annually	GASEOUS RELEASE Containment Building Annually	TOTAL CURIES Released Annually	CONCENTRATION Site Boundry	10 CFR 20 LIMITS Appendix B Table 2 Column 1	FRACTION OF 10 CFR 20 Limits
	Ci	Ci	Ci	Ci	Ci	µCi/cc	µCi/cc	
Ba-142	0.00E+00	1.60E-04	1.224E-04	4.71E-08	2.82E-04	4.4739E-17	2E-7	2.237E-10
Sr-89	1.56E+00	1.07E-04	8.236E-05	7.34E-05	1.56E+00	2.4789E-13	2E-10	0.0012394
Sr-90	2.19E-01	1.00E-05	7.677E-06	8.34E-06	2.19E-01	3.4789E-14	6E-12	0.0057982
Sr-91	5.06E-23	1.33E-04	1.021E-04	2.12E-06	2.37E-04	3.7646E-17	5E-9	7.529E-09
Sr-92	3.35E-80	1.40E-04	1.070E-04	6.32E-07	2.47E-04	3.9180E-17	9E-9	4.353E-09
Sr-93	0.00E+00	1.55E-04	1.189E-04	3.20E-08	2.74E-04	4.3443E-17	NA	
Sr-94	0.00E+00	1.53E-04	1.176E-04	5.35E-09	2.71E-04	4.2962E-17	NA	
Mo-99	2.12E-02	1.87E-03	1.433E-03	2.06E-04	2.47E-02	3.9212E-15	2E-9	1.961E-06
Mo-101	0.00E+00	1.69E-03	1.295E-03	6.87E-07	2.99E-03	4.7329E-16	2E-7	2.366E-09
Pd-109	1.28E-16	3.82E-05	2.925E-05	8.73E-07	6.83E-05	1.0828E-17	6E-9	1.805E-09
Rh-103m	7.31E-232	1.58E-04	1.215E-04	2.48E-07	2.80E-04	4.4430E-17	2E-6	2.221E-11
Rh-105	1.66E-06	1.02E-04	7.807E-05	6.02E-06	1.88E-04	2.9745E-17	8E-9	3.718E-09
Rh-105m	0.00E+00	3.11E-05	2.385E-05	6.50E-10	5.50E-05	8.7148E-18	NA	
Rh-106	0.00E+00	6.30E-05	4.826E-05	8.70E-10	1.11E-04	1.7634E-17	NA	
Ru-103	2.05E+00	1.59E-04	1.217E-04	1.03E-04	2.05E+00	3.2538E-13	9E-10	0.0003615
Ru-106	1.19E+00	5.74E-05	4.397E-05	4.65E-05	1.19E+00	1.8882E-13	2E-11	0.0094409
Tc-99m	3.19E-36	1.65E-04	1.267E-04	1.66E-06	2.94E-04	4.6560E-17	NA	
Tc-101	0.00E+00	1.69E-04	1.295E-04	6.68E-08	2.99E-04	4.7329E-17	5E-7	9.466E-11
Tc-104	0.00E+00	1.31E-04	1.002E-04	6.66E-08	2.31E-04	3.6640E-17	1E-7	3.664E-10
Tc-105	0.00E+00	1.08E-04	8.310E-05	2.29E-08	1.92E-04	3.0369E-17	NA	
Ce-141	1.96E+00	1.69E-04	1.299E-04	1.04E-04	1.96E+00	3.1109E-13	8E-10	0.0003889
Ce-143	9.51E-07	1.60E-04	1.224E-04	8.80E-06	2.92E-04	4.6277E-17	2E-9	2.314E-08
Ce-144	2.67E+00	1.31E-04	1.002E-04	1.05E-04	2.67E+00	4.2317E-13	2E-11	0.0211587
Np-239	6.76E-03	2.10E-03	1.610E-03	1.98E-04	1.07E-02	1.6909E-15	3E-9	5.636E-07
Pu-238	7.47E-03	3.40E-07	2.609E-07	2.84E-07	7.47E-03	1.1838E-15	2E-14	0.0591892
Pu-239	9.60E-04	4.38E-08	3.354E-08	3.65E-08	9.61E-04	1.5230E-16	2E-14	0.007615

Table 11.1-25 - Annual Releases of Radioactive Gases and Particulates

NUCLIDE	Annual Release Gas 12 month basis	SG BLOWDOWN Secondary Activity 45 day release	GASEOUS RELEASE Auxiliary Building Annually	GASEOUS RELEASE Containment Building Annually	TOTAL CURIES Released Annually	CONCENTRATION Site Boundry	10 CFR 20 LIMITS Appendix B Table 2 Column 1	FRACTION OF 10 CFR 20 Limits
	Ci	Ci	Ci	Ci	Ci	µCi/cc	µCi/cc	
Pu-240	1.25E-03	5.71E-08	4.379E-08	4.76E-08	1.25E-03	1.9883E-16	2E-14	0.0099417
Pu-241	3.22E-01	1.47E-05	1.129E-05	1.23E-05	3.22E-01	5.1071E-14	8E-13	0.0638391
Pu-242	5.33E-06	2.43E-10	1.860E-10	2.02E-10	5.33E-06	8.4442E-19	2E-14	4.222E-05
Th-228	8.13E-09	0.00E+00	2.925E-13	3.13E-13	8.13E-09	1.2894E-21	2E-14	6.447E-08
Am-241	3.88E-04	1.77E-08	1.357E-08	1.47E-08	3.88E-04	6.1589E-17	2E-14	0.0030794
Cm-242	9.68E-02	5.01E-06	3.838E-06	3.92E-06	9.68E-02	1.5342E-14	4E-13	0.0383554
Cm-244	1.06E-02	4.84E-07	3.708E-07	4.02E-07	1.06E-02	1.6785E-15	3E-14	0.055949
Eu-154	1.62E-02	7.44E-07	5.702E-07	6.18E-07	1.62E-02	2.5720E-15	3E-11	8.573E-05
Eu-155	7.01E-03	3.23E-07	2.478E-07	2.68E-07	7.01E-03	1.1117E-15	2E-10	5.559E-06
Eu-156	1.30E-01	2.33E-05	1.789E-05	1.06E-05	1.30E-01	2.0673E-14	6E-10	3.446E-05
Eu-157	2.89E-16	2.48E-06	1.901E-06	6.29E-08	4.44E-06	7.0442E-19	7E-9	1.006E-10
Eu-158	1.04E-285	8.70E-07	6.671E-07	1.11E-09	1.54E-06	2.4392E-19	8E-8	3.049E-12
Eu-159	0.00E+00	4.40E-07	3.373E-07	2.22E-10	7.77E-07	1.2327E-19	NA	
Gd-159	2.76E-14	5.96E-07	4.565E-07	1.85E-08	1.07E-06	1.6973E-19	8E-9	2.122E-11
Ho-166	1.49E-12	8.26E-09	6.335E-09	3.70E-10	1.50E-08	2.3737E-21	2E-9	1.187E-12
La-140	1.72E-05	1.88E-04	1.444E-04	1.27E-05	3.63E-04	5.7505E-17	2E-9	2.875E-08
La-141	1.94E-55	1.68E-04	1.288E-04	1.10E-06	2.98E-04	4.7220E-17	1E-8	4.722E-09
La-142	6.90E-143	1.65E-04	1.263E-04	4.18E-07	2.92E-04	4.6227E-17	3E-8	1.541E-09
La-143	0.00E+00	1.58E-04	1.215E-04	6.24E-08	2.80E-04	4.4401E-17	1E-7	4.44E-10
Nb-95	2.14E+00	1.77E-04	1.357E-04	1.11E-04	2.14E+00	3.3994E-13	2E-9	0.00017
Nb-95m	1.40E-04	2.02E-06	1.545E-06	2.91E-07	1.43E-04	2.2731E-17	3E-9	7.577E-09
Nb-97	1.70E-180	1.65E-04	1.265E-04	3.31E-07	2.92E-04	4.6281E-17	1E-7	4.628E-10
Nb-97m	0.00E+00	1.56E-04	1.193E-04	4.33E-09	2.75E-04	4.3574E-17	NA	
Nd-147	2.23E-01	6.76E-05	5.180E-05	2.53E-05	2.23E-01	3.5431E-14	1E-9	3.543E-05
Pm-147	4.55E-01	2.12E-05	1.625E-05	1.75E-05	4.55E-01	7.2197E-14	2E-10	0.000361
Pm-148	7.72E-03	1.69E-05	1.295E-05	3.56E-06	7.76E-03	1.2299E-15	7E-10	1.757E-06

Table 11.1-25 - Annual Releases of Radioactive Gases and Particulates

NUCLIDE	Annual Release Gas 12 month basis Ci	SG BLOWDOWN Secondary Activity 45 day release Ci	GASEOUS RELEASE Auxiliary Building Annually Ci	GASEOUS RELEASE Containment Building Annually Ci	TOTAL CURIES Released Annually Ci	CONCENTRATION Site Boundry µCi/cc	10 CFR 20 LIMITS Appendix B Table 2 Column 1 µCi/cc	FRACTION OF 10 CFR 20 Limits
Pm-148m	4.39E-02	3.31E-06	2.534E-06	2.16E-06	4.39E-02	6.9554E-15	4E-10	1.739E-05
Pm-149	1.04E-04	5.71E-05	4.379E-05	5.06E-06	2.10E-04	3.3248E-17	2E-9	1.662E-08
Pm-151	1.03E-08	2.00E-05	1.532E-05	9.48E-07	3.63E-05	5.7484E-18	4E-9	1.437E-09
Pr-142	6.14E-13	6.03E-06	4.621E-06	1.93E-07	1.08E-05	1.7190E-18	3E-9	5.73E-10
Pr-143	7.41E-01	1.56E-04	1.198E-04	6.66E-05	7.42E-01	1.1760E-13	9E-10	0.0001307
Pr-144	0.00E+00	1.31E-04	1.006E-04	6.31E-08	2.32E-04	3.6775E-17	2E-7	1.839E-10
Sm-153	2.04E-05	4.47E-05	3.429E-05	3.46E-06	1.03E-04	1.6303E-17	4E-9	4.076E-09
Tb-160	1.65E-03	1.00E-07	7.677E-08	7.25E-08	1.65E-03	2.6149E-16	3E-10	8.716E-07
Y-90	9.42E-04	1.03E-04	7.901E-05	1.10E-05	1.13E-03	1.7995E-16	9E-10	1.999E-07
Y-91	2.11E+01	1.37E-03	1.051E-03	9.62E-04	2.11E+01	3.3451E-12	2E-10	0.0167255
Y-91m	4.76E-261	7.73E-04	5.925E-04	1.07E-06	1.37E-03	2.1668E-16	2E-7	1.083E-09
Y-92	1.89E-60	1.41E-03	1.077E-03	8.31E-06	2.49E-03	3.9484E-16	1E-8	3.948E-08
Y-93	8.10E-21	1.05E-03	8.068E-04	1.78E-05	1.88E-03	2.9762E-16	3E-9	9.921E-08
Y-94	0.00E+00	1.65E-03	1.265E-03	8.59E-07	2.92E-03	4.6243E-16	1E-7	4.624E-09
Y-95	0.00E+00	1.70E-03	1.306E-03	4.98E-07	3.01E-03	4.7735E-16	2E-7	2.387E-09
Zr-95	2.80E+00	1.76E-04	1.351E-04	1.25E-04	2.80E+00	4.4335E-13	4E-10	0.0011084
Zr-97	5.42E-13	1.64E-04	1.256E-04	4.62E-06	2.94E-04	4.6622E-17	2E-9	2.331E-08
H-3	2.97E+02	1.36E-03	1.042E-03	1.13E-03	2.97E+02	4.7075E-11	1E-7	0.0004708
Rn-220	0.00E+00	0.00E+00	2.944E-13	9.91E-18	2.94E-13	4.6680E-26	3E-11	1.556E-15
Total	5.72E+02	1.07E+03	8.19E+02	1.00E+03	3.46E+03	5.48E-10		

11.1.3.8 Availability and Reliability

The waste gas header is vented to the Auxiliary Building atmosphere. All waste gas release will be monitored in the Auxiliary Building stack (See DSAR section 11.2.3). The general Waste Gas Disposal system description below pertains to how the system was operated prior to being permanently defueled.

Radioactive gaseous effluents can be released from the plant without being so indicated on an installed radiation monitor if the requirements of the ODCM are complied with. The monitoring system is described in [DSAR Section 11.2.3](#).

The radiation monitor at the discharge duct provides a check on radioactivity, and if the activity exceeds a predetermined limit, it provides an alarm.

Interlocks and other design features have been incorporated in the RWDS to preclude in so far as practical any gaseous release except under fully controlled conditions. Typical among these features are:

- a. Maximum use of available RWDS tankage by an unchecked, interconnected vapor space arrangement as previously described in part "e" of this section (Relief Valve Discharges).
- b. Vents and drains arrangements as described in part "a" of this section (Venting of Spent Regenerant Tanks) provides three separate liquid drain circuits and a closed vent circuit arranged to retain activity within the auxiliary building.
- c. Interlocks on RWDS components provide equipment shutdown in the event of malfunction.

11.1.3.9 Operation

The operation of the gaseous waste system is such that values for radioactive effluent release are maintained as low as reasonably achievable (ALARA). The normal operation for waste gas systems is the release of waste gas to the Auxiliary Building ventilation header. The release rates for radioactive materials, other than noble gases, in gaseous effluents is controlled such that concentrations of radionuclides do not exceed ten times 10 CFR 20 Appendix B, Table 2, Column 1 limits. For noble gases, the concentration shall be limited to five times 10 CFR 20, Appendix B, Table 2, Column 1 limits. Concentrations shall be calculated based upon the annual average $\text{Chi}/\text{Q}^{(11.4.12, 11.4.13)}$. Cumulative dose contributions must be determined in accordance with the Offsite Dose Calculation manual (ODCM) on a quarterly basis. Prior to discharge of radioactive materials in gaseous effluents, the equipment used in processing gaseous effluents is operated in accordance with the requirements of the ODCM. The setpoints for the effluent radiation monitors are calculated in accordance with the ODCM. The requirements for equipment functionality are defined in the ODCM. The requirements for sampling and activity analyses for radioactive gaseous waste and the requirements for verification of equipment operability are given in the ODCM.

11.1.3.10 Tests and Inspections

Shop Tests

Some equipment was tested and inspected in the manufacturer's shop in accordance with then applicable codes and standards. In addition, some equipment was given performance type tests in the manufacturer's shop.

On-Site Tests

These tests were primarily of the performance type and were designed to ensure that the overall gaseous waste system functions in a safe and efficient manner and were conducted prior to actual plant startup.

Provision was made to test the full operational sequence of the system. Flow paths, flow capacity, and mechanical operability were thoroughly checked. Pressure, temperature, flow and level indicating instruments were calibrated and checked for performance. All safety equipment, including alarms, were thoroughly tested.

Tracer gases, such as P-10 (10% Methane - 90% Argon), Helium and Sulfur Hexafluoride can be used as a leak detection medium in conjunction with a suitable detector, to locate leaks in the waste gas system outside containment. P-10 is non-flammable, non-toxic and does not become radioactive unless subjected to a neutron radiation field.

11.1.4 Solid Wastes

11.1.4.1 General

The general types of radioactive solid wastes produced at the station include process resins, used waste and process filters, dewatered ion exchange and filtration media, and miscellaneous solid wastes.

Spent resin from the filtration/ion exchange system is sluiced to a high integrity container which is dewatered and eventually shipped for disposal. Used filters are placed in a shielded container, stored in the cask decontamination area and eventually shipped from the plant. Miscellaneous solid wastes, such as equipment parts and laboratory glassware, are stored prior to off-site shipment.

The flow diagram, P&ID 11405-M-8, shows the process portion of the solid waste disposal system.

11.1.4.2 Sources of Solid Waste

- a. Radioactive liquid waste is processed either through a filtration/ion exchange system with the processed water being directed to the monitor tanks.

- b. Process wastes containing spent resins are obtained from the spent fuel storage pool demineralizer.

The resins from other sources and their sluice water are collected in the spent resin storage tank. The contents of this tank are mixed and solids are kept in suspension by nitrogen gas sparging. The contents of the tank are forced by pressurized demineralized water into a shielded resin cask after which the contents are dewatered and shipped from the plant. At this point it is considered to be a solid waste.

- c. Used filter baskets originate from other, the waste filters and the spent fuel pool cooling system filter. Solids removed from the liquid are retained on the filter elements which form the basket.
- d. Miscellaneous solid waste consist of contaminated articles such as equipment parts, laboratory glassware, clothing, gloves, cleaning tools, rags, towels, and plastic covers originating in the controlled access areas of the plant.

Table 11.1-26 shows the anticipated waste volumes on cycle basis.

Note: Since Fort Calhoun Station is permanently shutdown and defueled, the radioactive waste generation and removal needs are reduced. As a result, radioactive waste estimates for a running plant are considered to bound the needs of the shutdown plant.

Table 11.1-26 - Solid Waste Volumes

<u>Sources</u>	<u>Volume (ft³/cycle)</u>	<u>Basis</u>
Spent Resins		
Spent Fuel Pool Demineralizer	10	1/5 vessel/cycle
Filter Elements		
Waste Filters	5	each filter assembly per cycle
Spent Fuel Pool Filter	5	
Miscellaneous Solids	2,000	Assumed value for low activity solids.
Total		<u>2,020</u>

* With an 18 month cycle operation, the amount of solid waste volumes generated is assumed to remain the same based on the number shutdowns and waste inventory.

11.1.4.3 System Components

The major components of the solid wastes system of the RWDS are as follows; the referenced tables summarize pertinent data:

- a. Spent resin storage tank (see Table 11.1-12);
- b. Spent resin pump (see Table 11.1-13);
- c. Mobile Radwaste Processing System/Filtration/Ion Exchanger (FIX)

11.1.4.4 System Operation

Radioactive Liquid and Spent Resins

The following operation is followed for the processing of liquid and resin.

- a. If the filtration/ion exchange system is in operation, the radioactive liquid is transferred from the waste holdup tanks using the waste holdup transfer pumps. The water that has been processed is directed to the monitor tanks to be analyzed and discharged to the Missouri River through the overboard discharge piping. Depleted filtration ion exchange media is sluiced to a high integrity container and then dewatered using vendor supplied system prior to being shipped offsite for disposal.
- b. The resin is flushed from the spent resin storage tank by demineralized water to a shielded resin cask with liner located in the Radioactive Waste Processing Building through shielded piping. The resin is then dewatered/solidified. The liner with resin is placed in the cask which is shipped offsite.

Miscellaneous Solid Waste

Non-compactable waste are placed in large steel boxes for disposal. The activity of this material is normally low and special shielding is not necessary.

11.1.4.5 Design Evaluation

The spent resin storage tank has a volume of 400 cu. ft. and is designed to hold at least two to three years production of spent resins. Excess transport water used to convey resins to the tank is removed by pumping from a screened lateral connection in the tank. Transport water returns to the spent regenerant tanks. Nitrogen is admitted through the bottom lateral at a sufficient rate to mix the resin slurry.

Spent resin can have high activity; therefore the resin casks are equipped with internal shields designed to reduce the external dose rate to a level permitting in-plant handling.

11.1.4.6 Availability and Reliability

The solid waste system is normally operated on a batch basis, and is available to perform abnormal or emergency functions.

The solid waste system is dependent on the operation of the filtration/ion exchange system. These systems are also dependent on the electrical systems, the demineralized water system, the plant compressed air system, and nitrogen gas.

The Process Control Program (PCP) is used to verify satisfactory solidification of waste prior to shipment offsite.

The Radioactive Waste Processing Building is sized to accumulate a number of containers (e.g., liners, drums, high integrity containers) to permit scheduling of off-site shipments.

11.1.4.7 Tests and Inspections

All equipment in the solid waste system was subject to both shop and on-site tests.

Shop Tests

All equipment was tested and inspected in the manufacturer's shop in accordance with the then applicable codes.

In addition, some equipment was given performance type tests in the manufacturer's shop.

On-Site Tests

These tests were primarily of the performance type and were designed to ensure that the overall solid waste system functions in a safe and efficient manner. These tests were conducted prior to actual plant startup.

Provision was made to test the full operational sequence of the system. Pumps were started, valves operated, instruments put into service.

Inspection of Containers in Storage

Provisions are included for inspection of containers while in storage by using TV cameras or boroscope for high radiation level conditions, and by direct observation when radiation levels are low.