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IN-PLANT SOURCE TERM MEASUREMENTS AT FORT CALHOUN STATION — UNIT 1

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IN-PLANT SOURCE TERM MEASUREMENTS AT FORT CALHOUN STATION — UNIT 1

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ABSTRACT

This report presents data obtained from an in-plant source term measurement program conducted for the Office of Nuclear Regulatory Research in support of the Effluent Treatment Systems Branch of the Office of Nuclear Reactor Regulation. The objective of this program is to provide operational data that can be used in the generic evaluation of plant system design in the licensing process and for updating of the calculational models used by the NRC staff in their evaluation of radioactive waste management systems for operating pressurized water reactors. A data base is provided for radioisotope inventory in plant systems, radioactive waste management system performance, and source terms for both liquid and gaseous systems.

Data presented were obtained at the Fort Calhoun Station - Unit 1, operated by the Omaha Public Power District (OPPD), located at Blair, Nebraska. In-plant measurements were conducted during the time period from August, 1976 through February 1977. This plant is the first of a planned series of six (6) operating PWR's to be studied.

ACKNOWLEDGEMENTS

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CHAPTER 1 - INTRODUCTION

1.1 In-Plant Measurement Program Objectives

The objective of the in-plant source term measurement study is to provide the Nuclear Regulatory Commission (NRC) with experimental data that can be used by the NRC in its evaluation of plant system design. The results of this study will provide a data base for radioisotope inventory in plant systems, radioactive waste treatment system performance, and source terms for both liquid and gaseous systems.

The in-plant measurement study identifies and characterizes the sources of radioisotopes in operating LWR's. Specific objectives are to provide NRC with:

1. Source term information so that the parameters used in NRC's calculational models can be updated as necessary.
2. Data on the inventory of the radioisotopes present (i.e., locations, concentrations, etc.) in operating reactor plant systems.
3. Radwaste equipment performance for use in NRC evaluations of radioactive waste management systems.

Measurements are made during the three stages of plant operation, i.e., (1) prior to refueling, (2) during refueling, and (3) following refueling.

1.2 Fort Calhoun Station

The NRC in-plant source term measurement program at operating PWR's was initiated during the summer of 1976. From August, 1976 through February, 1977, measurements were made at the Fort Calhoun Station - Unit 1, Omaha Public Power District (OPPD), Blair, Nebraska. Table 1.1 presents the operational status and the time span for measurement periods during the time between the reactor's first refueling in the spring of 1975 until the third refueling in the fall of 1977.

Relevant system parameters for the Fort Calhoun Station are listed below:

1. Reactor and Steam Generator Vendor: Combustion-Engineering
2. Turbine Vendor: General Electric
3. Power: 1420 MWt and 457 MWe, net
4. Reactor Coolant Volume: 6616 ft^3
5. Initial Criticality: August, 1973
6. Commercial Power Operations: June, 1974

TABLE 1.1

TIME PERIODS FOR IN-PLANT MEASUREMENTS

Plant Status	Time Period	Measurement Period
Power Operation	5/75 - 10/1/76	Liquid: 8/18 - 9/2/76
		Gaseous: 9/3 - 10/7/76
Second Refueling	10/1 - 12/15/76	Liquid: 10/11 - 12/3/76
		Gaseous: 10/7 - 12/15/76
Power Operation	12/15/76 - 9/30/77	Liquid: 2/9 - 2/22/77
		Gaseous: 12/15/76 - 2/17/77

CHAPTER 2 - MEASUREMENT METHODS AT FORT CALHOUN

2.1 Introduction

The measurement methods employed have been described in the report, "Procedures, Source Term Measurement Program" (1). Brief descriptions are given in the following three sections which refer to the liquid process streams, gaseous process streams and the plant operational information.

2.2 Liquid Streams

Liquid samples were collected using the installed plant sampling system. The types of liquid samples collected were:

1. Reactor coolant
2. Chemical and Volume Control System (CVCS) samples both upstream and downstream of the letdown demineralizers and filter
3. Steam generator blowdown water
4. Spent fuel pool water
5. Radwaste system:
   a. Collection tanks
   b. Evaporator feed, distillate and concentrate
   c. Monitor tanks

Figures 2.0, 2.1, 2.2, and 2.3 present simplified schematic diagrams for the liquid and solid pathways at Fort Calhoun Station. For more detailed diagrams, the Piping and Instrument Diagrams (P & ID) have been presented in the Appendix.

All liquid samples except those passed through and concentrated on ion exchange resins were collected in plastic volumetric measuring devices and promptly transferred to glass counting containers. No measurements were performed to quantify potential plateout of radionuclides on the plastic volumetric measuring devices. The size of the liquid samples collected depended upon the expected radionuclide activity level. Fifty ml samples were collected from reactor coolant systems while 450 ml samples were generally taken from other systems. Each glass counting container had been prefilled with enough concentrated HCl to provide a solution containing two percent concentrated HCl to minimize plate-out of radionuclides on the glass. Samples from very low activity streams were concentrated on an ion exchange column by passing 20-100 liters of sample through the column (1). Again, no measurements were made to quantify potential plateout on the aluminum resin holder or silicone tubing sampler feed lines.

FIGURE 2.0

Liquid and Solid Systems

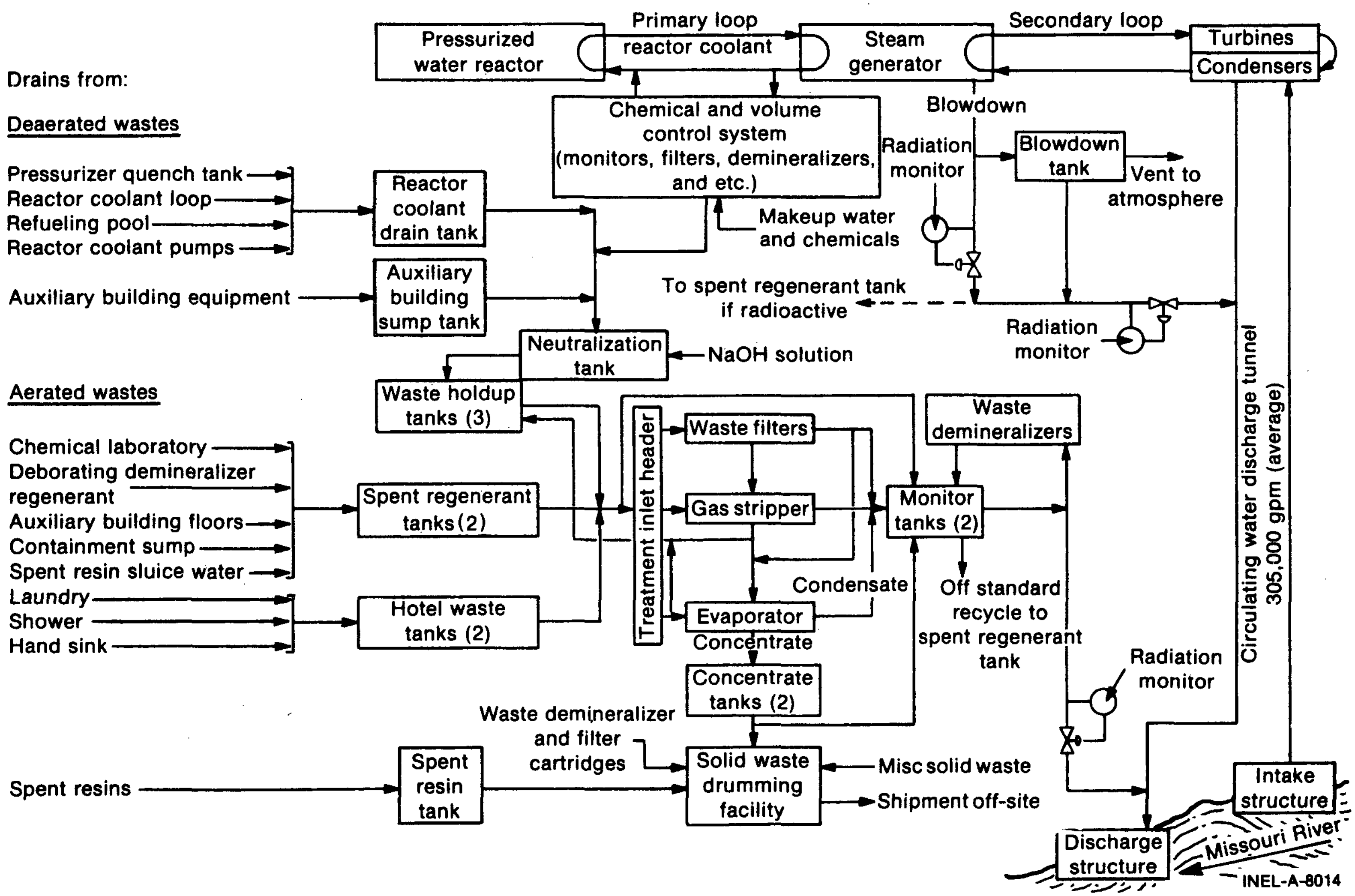


FIGURE 2.1

REACTOR COOLANT AND CVCS DIAGRAM

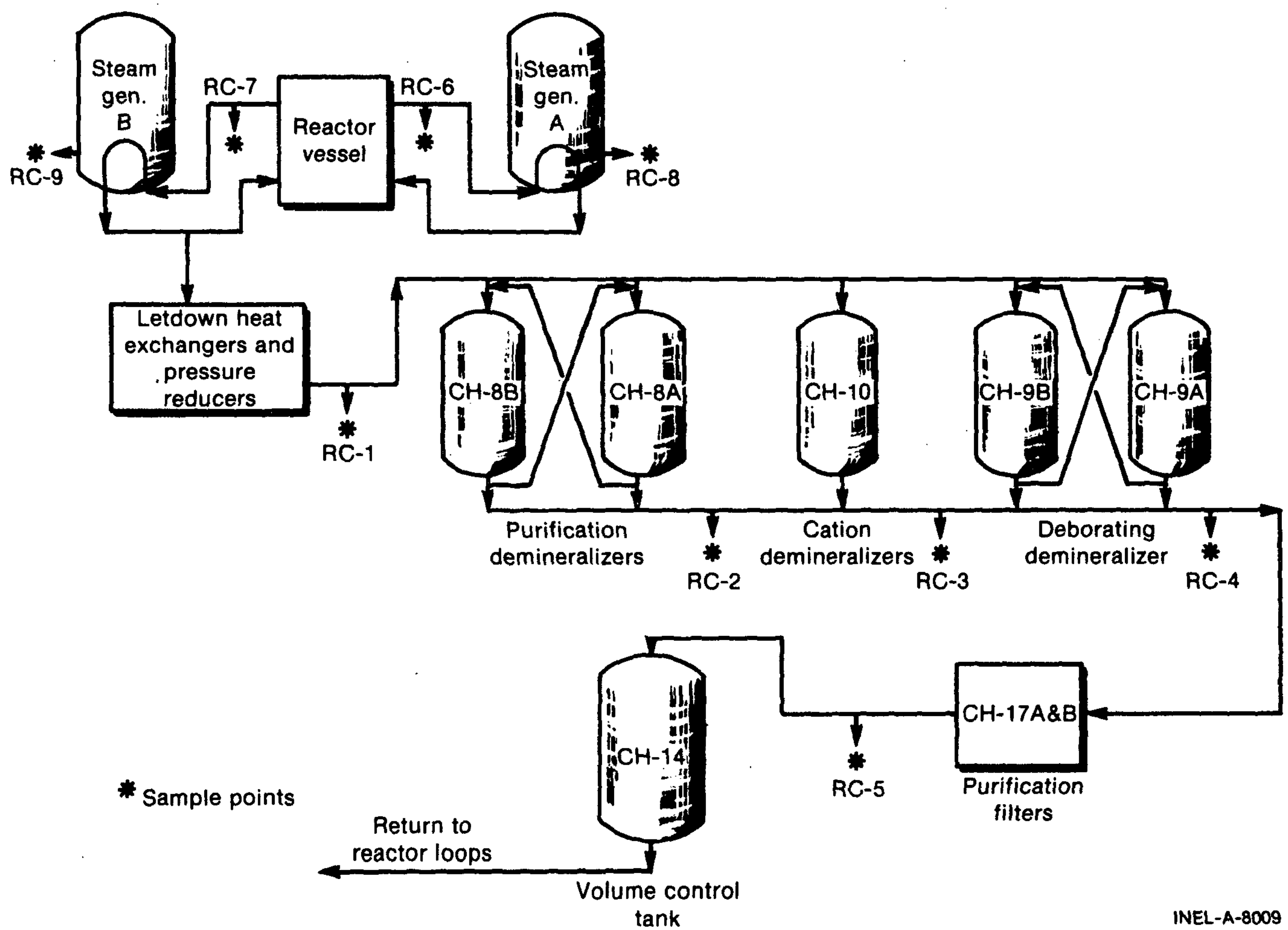


FIGURE 2.2

LIQUID RADWASTE SYSTEM COLLECTION TANKS

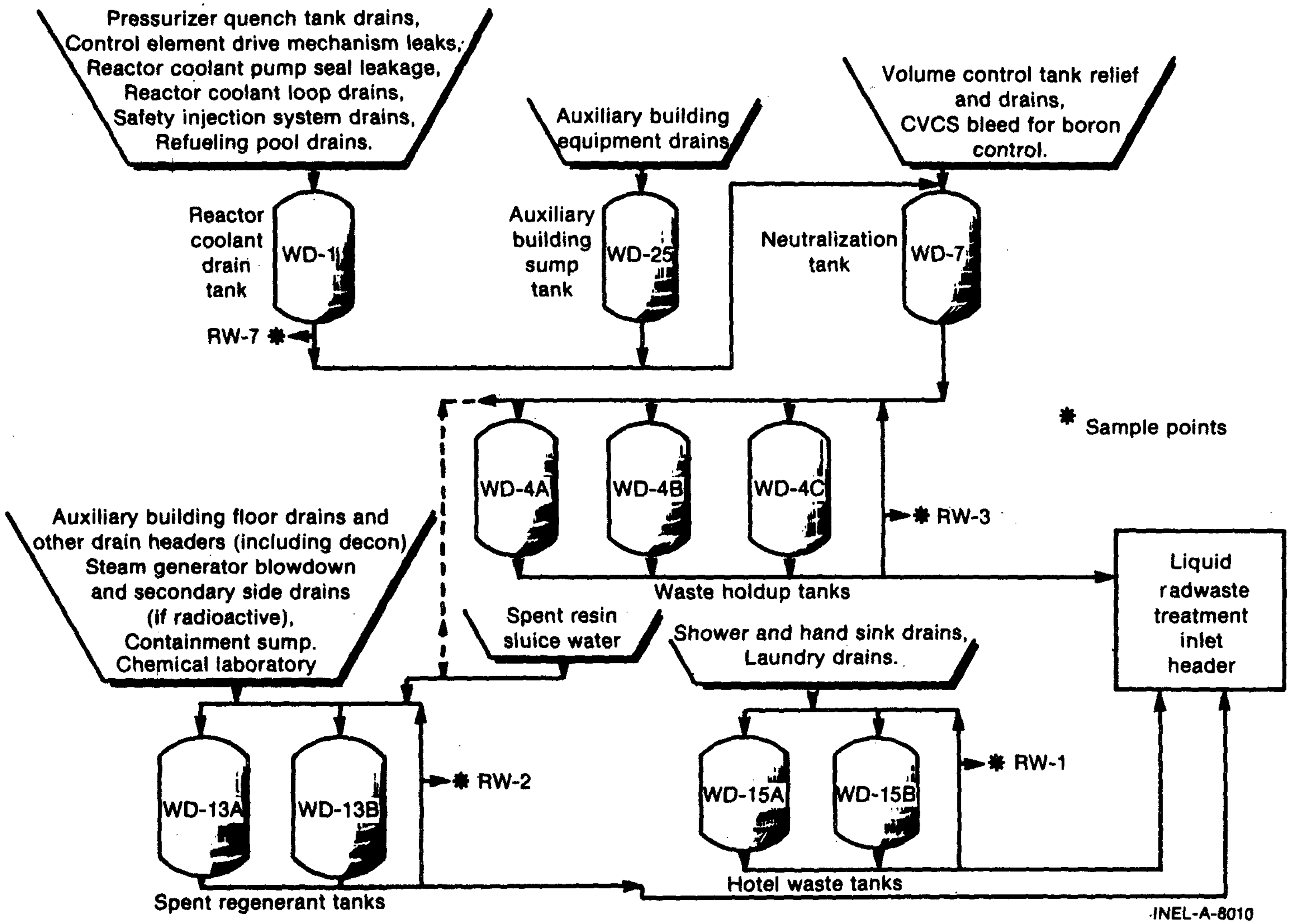
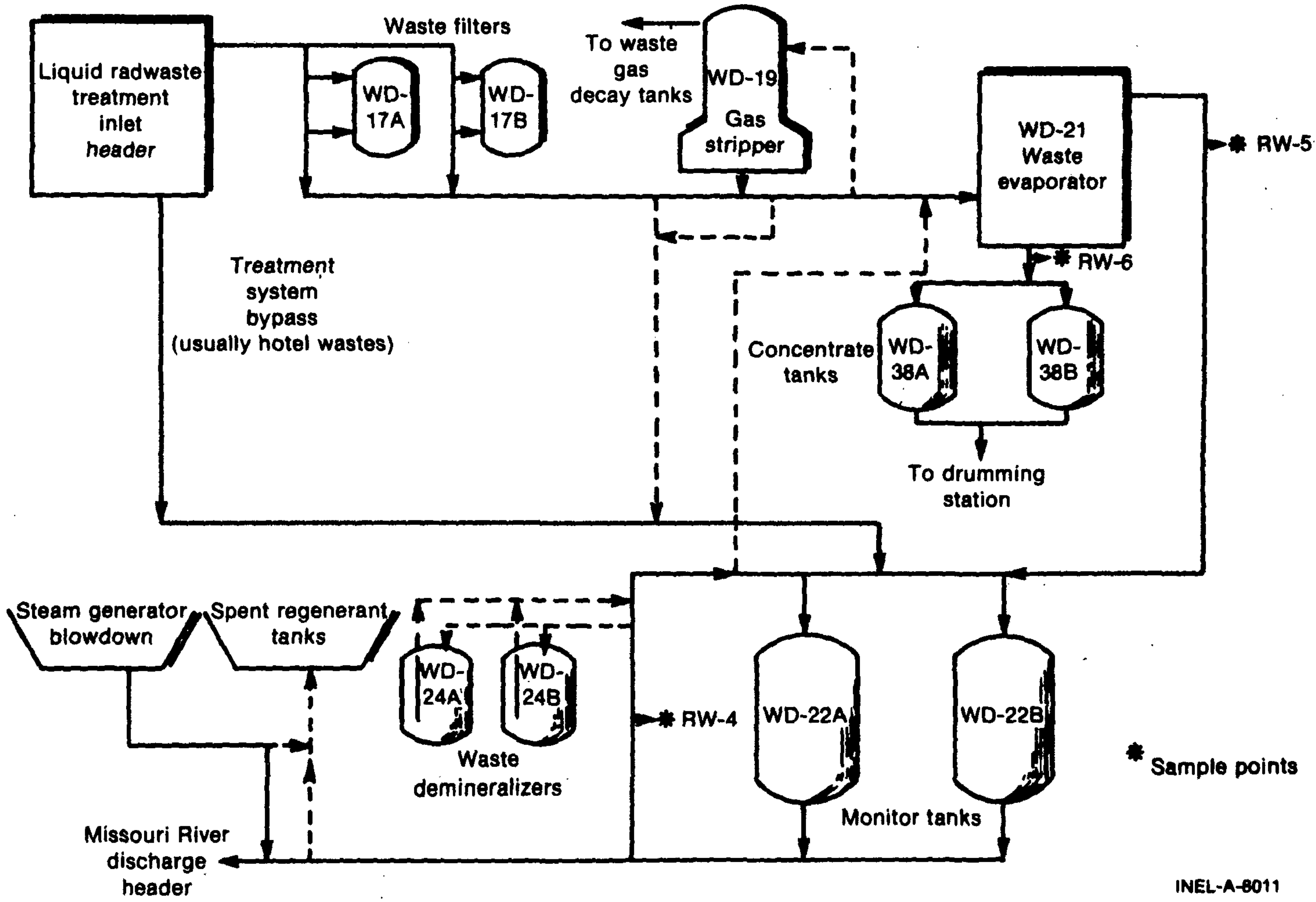


FIGURE 2.3

LIQUID RADWASTE PROCESS DIAGRAM



Gamma-ray spectra from the liquid samples were obtained using a Ge(Li) gamma-ray spectrometer installed in the NRC mobile laboratory (1). Samples were counted several times over an extended period after collection to optimize determination of both short- and long-lived radioisotopes. The detection efficiency for the gamma-ray detector was measured using NBS reference standards prior to moving the mobile laboratory to Fort Calhoun (1). During the initial set-up at Fort Calhoun, the calibration was re-checked for selected geometries using an NBS standard. Gamma-ray energy calibrations were made on a daily basis using a ^{228}Th source. The same reference standard was measured in a fixed geometry to verify overall performance of the spectrometer system.

The analysis of gamma-ray spectra was completed at INEL using the GAUSS VI computer program (2) on the IBM 360/75. This program searches each spectrum for gamma-ray peaks, provides isotopic identification and concentration and decay corrects to the sample collection time. A final output for the activity of each radionuclide found in the sample in μCi/ml is provided together with error estimates. Uncertainties quoted for radionuclide concentrations are errors due to counting statistics. An additional uncertainty of approximately 10 percent should be added to the quoted errors to account for calibration (~10%) and volume measurement (~5%) errors. Indeterminate sampling errors have not been treated.

Liquid samples were collected of reactor coolant, spent fuel pool, and the radwaste monitor tank waters to determine concentrations of pure beta-emitting radionuclides. Subsequent analyses of the samples were performed by the Department of Energy, Idaho Operations Office, Health Services Laboratory (DOE-HSL). Summaries of the analysis methods may be found in reference (1). Initially, samples (Tables A.2 and A.4) were analyzed for ^{3}H, ^{14}C, ^{32}P, ^{35}S, ^{45}Ca, ^{55}Fe, ^{63}Ni, ^{89}Sr, ^{90}Sr, ^{91}Y, and ^{147}Pm. The decision was made not to analyze the latter samples for ^{32}P, ^{35}S, ^{45}Ca, and ^{147}Pm. These data are presented in Tables A.2, A.3, and A.26.

2.2.1 Reactor Coolant and CVCS Samples

Figure 2.1 shows a simplified schematic diagram for the Reactor Coolant (RC) and the CVCS. Table 2.1 presents specifications for CVCS components obtained from the Final Safety Analysis Report (FSAR) (3). The RC samples were taken from points RC-6 and RC-7 which are in the two reactor coolant loops. There was no observed difference in the measured activity levels between the two loops. The input sample for the CVCS was taken at point RC-1. The output sample of the purification, cation, and deborating demineralizers were taken at points RC-2, RC-3, and RC-4, respectively, depending upon which CVCS demineralizer was in operation at the time of sampling. The output sample for the CVCS purification filters was taken at point RC-5. There was no sample point at the liquid output from the volume control tank. Results of measurements for the CVCS system are given in Tables A.8-A.12. Radioisotope data on reactor coolant (including pure beta-emitting nuclides) are presented in Tables A.1-A.7.

TABLE 2.1

CHEMICAL AND VOLUME CONTROL SYSTEM COMPONENTS

ION EXCHANGERS

Item No's.	CH-8A, 8B, 9A, 9B & 10
Quantity	5
Type	Flushable (CH-9A, and CH-9B renewable)
Design Pressure, psig	200
Design Temperature, F	250
Normal Operating Pressure, psig	25
Normal Operating Temperature, F	120
Resin Volume, ft^3	20
Resin Cross-sectional Area, ft^2	6.8
Normal Flow Rate, gpm	36
Maximum Flow Rate, gpm	116
Retention Screen	80 US Mesh
Code for Vessel	ASME III, Class C
Material	Austenitic Stainless Steel
Fluid	1 wt % Boric Acid, Maximum

PURIFICATION FILTERS

Item No's.	CH-17A & 17B
Quantity	2
Type of Elements	Wound Cartridge
Retention for 1 Micron Particle, %	95
Design Pressure, psig	200
Design Temperature, F	250
Design Flow, gpm	116
Normal Flow, gpm	36
Maximum Flow, gpm	160
Code for Vessel	ASME III, Class C
Material	Austenitic Stainless Steel
Fluid	1 wt % Boric Acid, Maximum

VOLUME CONTROL TANK

Item No.	CH-14
Quantity	1
Type	Vertical, Cylindrical
Design Pressure, Internal, psig	75
Design Pressure, External, psig	15
Design Temperature, F	250
Internal Volume, Minimum, ft^3	383
Operating Pressure Range, psig	0 to 65
Normal Operating Pressure, psig	50
Normal Operating Temperature, F	120
Normal Spray Flow, gpm	36
Blanket Gas	Hydrogen
Code	ASME III, Class C
Fluid	6-1/4 wt% Boric Acid, Maximum
Material	Austenitic Stainless Steel

2.2.2 Steam Generator Samples

Steam generator water for both steam generators (sample points RC-8 and RC-9 in Figure 2.1) were sampled both before (8/26/76) and after refueling (2/8-10/77). Based on the ^{131}I activity levels observed from nominal 200 liter ion exchange resin samples (Table A.13) and a blowdown rate of 12,500 lbs/hr, the primary to secondary leak rate during the February 1977 in-plant measurement program at the Fort Calhoun Station was less than 10^{-2} lbs/day. Also, the samples taken in August 1976 indicate no meaningful primary to secondary leak. Since there was no meaningful primary to secondary leak during either sampling period, the rest of the secondary system (both liquid and gaseous) was not sampled.

2.2.3 Spent Fuel Pool and Associated Samples

The spent fuel pool water was sampled using a sample point on the input line to the spent fuel pool heat removal and water clean-up system (sample point #1 in Figure A.12). During refueling the refueling cavity and fuel transfer canal were sampled using a container dipped into the waters. The safety injection and refueling water storage tank (SIRWT) water was sampled both before and after refueling using the sample point on the tank's recirculation line (sample point #3 in Figure A.12). Data for spent fuel pool and associated samples are presented in Tables A.11 and A.14-A.18.

2.2.4 Liquid Radwaste Samples

Figure 2.2 presents a diagram of the liquid radwaste collection tanks with information as to the sources of liquid radwaste for each collection tank. Tables 2.2 a, b, and c present the FSAR (3) component information for the liquid radwaste system with comments determined from discussions with plant personnel. The sample point for the reactor coolant drain tank (RW-7) was a local sample point on the tank. Validity of this sample in relationship to the tank contents is not known; however, before a sample was collected, the sample line was purged. There were no sample points for the auxiliary building sump tank or neutralization tank. The sample points for the waste holdup tanks (RW-3), spent regenerant tanks (RW-2), and hotel waste tanks (RW-1) were in the recirculation lines for these tanks. Before samples were taken from these tanks, as in all cases where the sample points were located in recirculation lines, the tank contents were recirculated for a minimum of two tank volumes.

Figure 2.3 is a diagram of the liquid radwaste treatment system. No sample point was available on the output stream of the waste filters. The feed samples for the waste evaporator were taken from the radwaste collection tanks being processed while the condensate and concentrate samples were taken from points RW-5 and RW-6, respectively. The samples for the monitor tanks were obtained from point RW-4 in the recirculation line. During measurements at Fort Calhoun the waste demineralizers

TABLE 2.2 a

COMPONENT DESIGN DATA, WASTE DISPOSAL SYSTEM TANKS

Tank	No. Installed/ Item No.	Capacity/Tank gallons/ft^3	Pressure, psig Design/ Operating	Tempera- ture, F Design/ Operating	Material*	Code
Reactor Coolant Drain Tank	1/WD-1	900/120	25/2	300/267	304 SS	ASME Section III, Class C
Waste Holdup Tanks	3/WD-4A, B&C	45,800/6,100	15/2	200/120	CS	ASME Section III, Class C
Neutralization Tank	1/WD-7	900/120	25/2	200/120	304 SS	ASME Section VIII
Caustic Dilution Tank	1/WD-11	750/100	Atmos/Atmos	200/ 70	CS	None
Spent Regenerant Tanks	2/WD-13A&B	4,800/650	5/Atmos	200/ 70	304 SS	ASME Section VIII
Hotel Waste Tanks	2/WD-15A&B	1,200/160	15/Atmos	200/140	CS	ASME Section VIII
Monitor Tanks	2/WD-22A&B	6,000/800	5/Atmos	200/140	304 SS	ASME Section VIII
Auxiliary Building Sump Tank	1/WD-25	700/95	25/2	200/120	304 SS	ASME Section VIII
Gas Decay Tanks	4/WD-29A, B,C&D	2,990/400	150/100	200/140	CS	ASME Section III, Class C
Spent Resin Storage Tank	1/WD-33	3,000/400	25/2	250/120	304 SS	ASME Section VIII
Concentrate Tanks	2/WD-38A&B	1,200/160	25/Atmos	300/140	CS	ASME Section VIII
Resin Measuring Tank	1/WD-43	60/8	50/30	100/70	304 SS	ASME Section VIII

* SS= Stainless Steel, CS= Carbon Steel

TABLE 2.2b

COMPONENT DESIGN DATA, WASTE DISPOSAL SYSTEM PUMPS

Pump	No. Installed/ Item No.	Type	Capacity	Fluid Side Material*
Reactor Coolant Drain Tank Pumps	2/WD-2A&B	Horizontal Centrifugal	2A, 250 gpm @ 75 ft. 2B, 50 gpm @ 75 ft.	316 SS 316 SS
Containment Sump Pumps	2/WD-3A&B	Vertical Centrifugal	20 gpm @ 34 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
Neutralization Transfer Pumps	2/WD-8A&B	Horizontal Centrifugal, Canned Rotor	200 gpm @ 82 ft.	316 SS
pH Sample Pump	1/WD-10	Horizontal Centrifugal, Canned Rotor	5 gpm @ 21 ft.	316 SS
Caustic Pumps	2/WD-12A&B	Positive Displacement, Diaphragm Type	3 gpm @ 160 ft.	CS & SS
Hotel Waste Pumps	2/WD-16A&B	Horizontal Centrifugal	50 gpm @ 130 ft.	AI
Gas Stripper Pumps	2/WD-20A&B	Horizontal Centrifugal Canned Rotor	16 gpm @ 75 ft.	316 SS
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. Sump Pumps	6/WD-27A&B, 40A&B, 41A &B	Vertical Centrifugal	20 gpm @ 36 ft.	AI

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

(continued)

TABLE 2.2 b (cont.)

Pump	No. Installed/ Item No.	Type	Capacity	Fluid Side Material*
Spent Resin Pump	1/WD-34	Horizontal Centrifugal	30 gpm @ 106 ft.	304 SS
Resin Dewatering Pump	1/WD-35	Centrifugal Displacement, Rotating Water Seal	35 gpm @ 44 ft.	CI bronze fitted
Concentrate Tank Pumps	2/WD-39A&B	Horizontal Centrifugal, Canned Rotor.	30 gpm @ 105 ft.	316 SS & CS

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

TABLE 2.2 c

COMPONENT DESIGN DATA, WASTE DISPOSAL SYSTEM PROCESS EQUIPMENT

Waste Filters, Item No's WD-17A&B	Description
Number	2
Type	Expendable element pressure type
Materials of Contruction	304 stainless steel vessel with synthetic fiber element
Vessel Design Pressure, psig	150
Vessel Design Temperature, F	250
Vessel Code	ASME Section III, Class C
Flow Rate, each, gpm	30
Retention, % (particles >25 microns)	98
Gas Stripper, Item No. WD-19	
Note A	
Number	1
Type	Packed column, heated liquid-vapor contact, non-condensable gas separation.
Design Flow Capacity lbs/hr	7500
Design Pressure, psig	50
Design Temperature, F	300
Operating Pressure, psig	2
Operating Temperature, F	220
Code	ASME Section III, Class C
Decontamination Factor, $\frac{\text{Gaseous activity in feed}}{\text{Gaseous activity in hotwell}}$	10^4

(continued)

TABLE 2.2 c (cont.)

Waste Evaporator, Item No. WD-21	Description
Number	1
Type	Steam heated horizontal tube, spray film type
Design Pressure (shell side), psig	15
Design Temperature, F	225
Operating Pressure, psig	2
Operating Temperature, F	217
Code	ASME Section III, Class C.
Flow Rate, gpm	17 (Note B)
Decontamination Factor, $\frac{\text{Activity in feed}}{\text{Activity in distillate}}$	10^4
Concentration of Bottoms, max, % $Na_2B_4O_7$	20
Note B	
Waste Demineralizers, Item No's. WD-24A&B	
Note C	
Number	2
Type	Non-regenerable, disposable vessel and resin
Materials of Construction	Carbon steel, phenolic lining
Design Pressure, psig	100
Design Temperature, F	150
Operating Pressure, psig	30
Operating Temperature, Max. F	120
Vessel Code	ASME Section VIII
Resin	Mixed bed, strong acid cation resin and intermediate base anion resin
Resin Volume, each, ft^3	3
Flow Rate, each, gpm	15
Waste Gas Compressors, Item No's. WD-28A&B	
Number	2
Type	Centrifugal displacement with rotating water seal.
Design Flow Rate, each, CFM @ 60 F & 1 atm.	40
Design Discharge Pressure, psig	100 (Note D)
Design Standard	American Society of Heating, Refrigerating and Air-Conditioning Engineers (ASHRAE)
Fluid Handled	Nitrogen, hydrogen, water vapor and traces of fission gases.
Controls	Bypass flow regulator for suction pressure control between 1.5 and 1.8 psig
Note D	

Note A: Gas stripper was under repair during measurement period.
Note B: Maximum about 15 gpm; usual flow rate, 10 gpm.
Note C: Waste Demineralizers not used during measurements period.
Note D: Normal discharge pressure; 50 to 86 psig.

or side-stream polishing demineralizers for the monitor tanks were not used. According to station personnel, the waste disposal problems associated with the spent resins (e.g., high personnel exposures) from the waste demineralizers outweigh the usefulness of processing lower activity liquid radwaste from the monitor tank through the waste demineralizers. With the exception noted below, following activity measurement, the monitor tanks were released to the discharge header without further treatment. At least once during the measurement period the total gamma activity in the monitor tank was above release limits due to dissolved noble gas activity (^{133}Xe). The noble gas activity was removed by sparging the monitor tanks. This released the noble gas to the station stack and reduced the total gamma activity so that the monitor tank could be released to the discharge header. During the Fort Calhoun study the gas stripper in the liquid radwaste treatment system was out of service. A project is underway by OPPD to make the gas stripper operational. Data obtained from analyses of radwaste samples are presented in Tables A.19-A.34.

2.3 Gaseous Streams

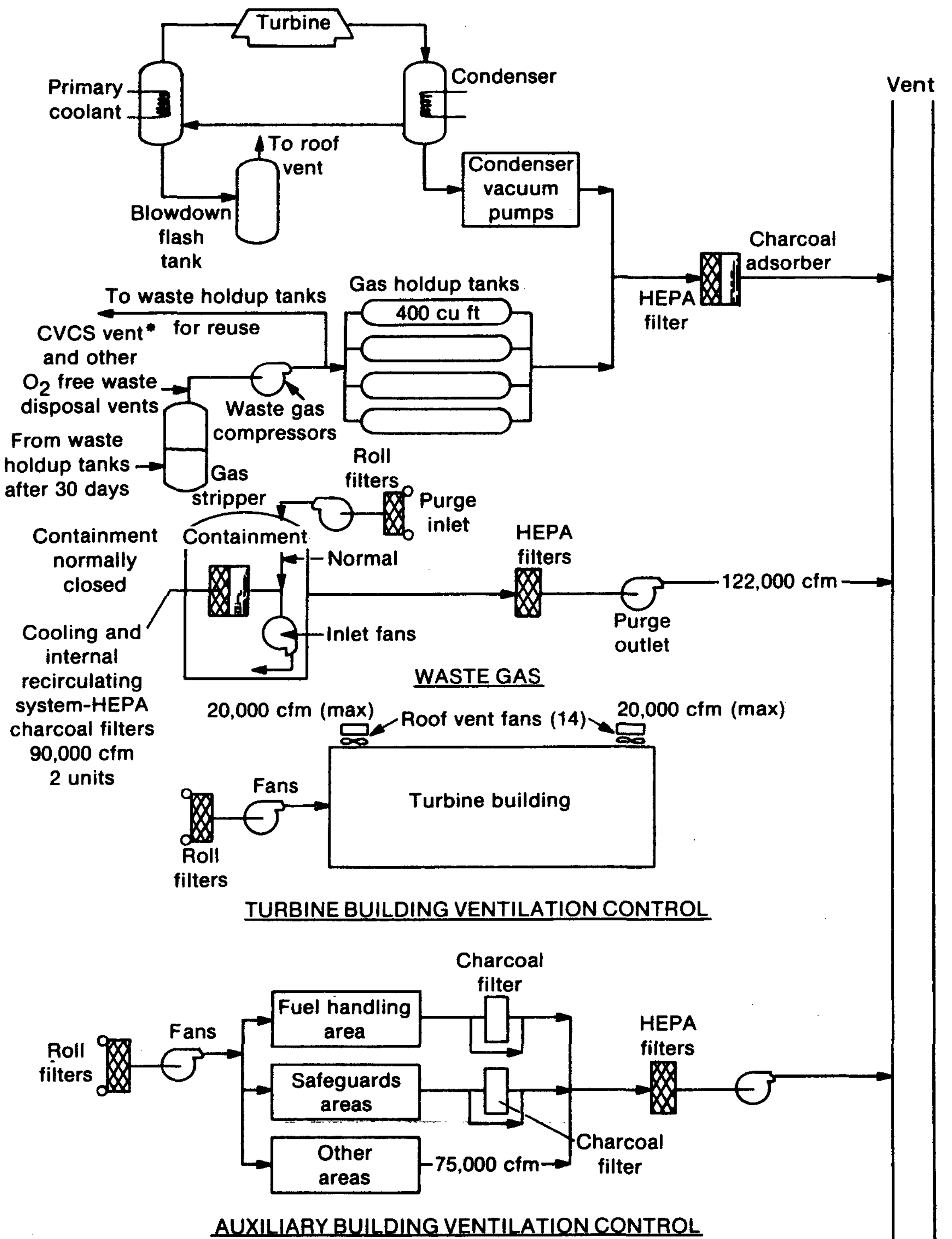
2.3.1 Ventilation Sampling

Figure 2.4a presents a schematic diagram of the gaseous waste disposal system. Figure 2.4b shows the auxiliary building ventilation system in more detail with the sampling locations noted. Prior to the installation of sampling devices in the auxiliary building, velocity profile measurements were made in the ducts in which sample probes were installed. The duct flows were calculated from velocity profiles and were compared to the design flows. Significant differences were observed. Discussions with OPPD staff resulted in changes in the operating status of the ventilation system. New velocity profiles were then measured and duct flows calculated. The results of these second measurements are shown in Table 2.3 along with the design flows taken from the P & ID, "Auxiliary Building Heating and Ventilating Flow Diagram" presented in the Appendix (Fig. A-28). One minor difference in the flow path was found during sample probe installation and is shown in Fig. A-29 of the Appendix. This area of the system is shown at coordinates F-4 in Fig. A-28. The new comparisons with design flows were deemed reasonable and sampling probes were installed.

In order to verify the validity of the samples being withdrawn from the ducts, helium was injected into the system upstream of each of the sampling probes and the helium concentration was measured in each of the sample streams. The duct flows were calculated from the helium concentration measurements using the ratio of the helium injection rate to the measured helium concentration (1). Results of these measurements are also shown in Table 2.3 The agreement between the flows measured by standard pitot tube traverse methods and those calculated from the helium dilution technique indicated that the samples taken from the ducts were representative of the gaseous species in the ducts.

FIGURE 2.4a

GASEOUS WASTE DISPOSAL SYSTEM



* CVCS - *chemical and volume control system*

FORT CALHOUN

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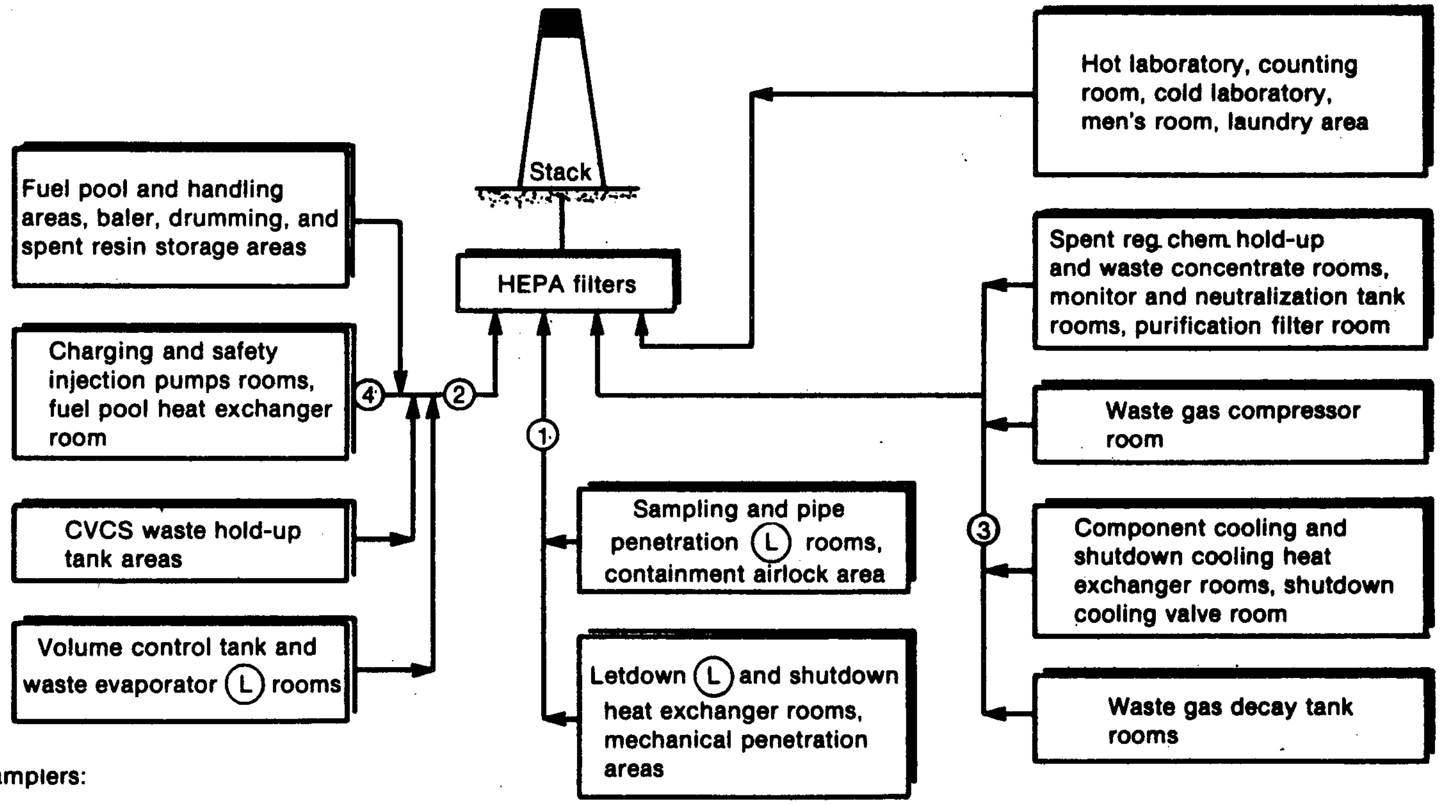
FIGURE 2.4b

FORT CALHOUN STATION
AUXILIARY BUILDING VENTILATION SYSTEM



Samplers:

Circled number shows sample station number with samplers for particulate, iodine, ^3H and ^{14}C.

Circled "L" shows local sample point with samplers for particulate and iodine.

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TABLE 2.3

AUXILIARY BUILDING VENTILATION DUCT FLOW MEASUREMENTS

Location	Design Flow, cfm	Measured Flow, cfm	
		Pitot Tube	Helium Dilution
Station #1	14,250	16,600	16,400
Station #2	39,400	28,900	28,400
Station #3	5,450	6,600	5,100
Station #4	19,200	24,500	23,800
Waste Evaporator Room	1,800	1,460	not measured

Table 2.4 tabulates the various areas of the auxiliary building feeding sampling stations 1 through 4. As indicated from the flow measurements (Table 2.3) station 2 sampled over half of the auxiliary building exhaust flow; consequently, sampling station 4 was installed in an attempt to more closely define specific sources of radioactivity. For the same purpose, local samplers were deployed in individual auxiliary building rooms. Local sampler locations are shown schematically in Figure 2.4.

The sampling systems installed in stations #1, #3, and #4 were modified commercial sampling systems which had been used in previous studies (4). The system consists of a sampling probe, particulate filter, iodine adsorber, water vapor adsorber, flow integrator, flow meter, and air mover. The sample inlet was a stainless steel nozzle sized so that isokinetic sampling conditions were met with a flow of approximately one cfm. The sample passed through a glass lined probe into a compartment which contained both the particulate filter and iodine adsorber. Both the probe and compartment were capable of being heated to prevent condensation. However, heating was not necessary at Fort Calhoun because all samplers were inside the plant. The particulate filter used was a 5" diameter fiberglass HEPA filter. Downstream of the filter was a 2" X 4" iodine adsorber equally divided in three sections. The first and last section were activated coconut shell charcoal impregnated with TEDA and the center section was silver zeolite. After passing through the iodine adsorber, the sample stream was split with a small sidestream (100 cc/min) passed through silica gel to remove water vapor. The two parts of the sample stream were recombined and passed through a dry gas integrating flow meter, and orifice meter to set flow rate and the air mover. The exhaust from the sampling system was vented into the sample duct. These as well as all samplers employed are described more fully in reference 1.

At station #2, an iodine species sampler was connected to the end of the probe. The species sampler consisted of, in order of flow, a fiberglass HEPA filter; an elemental iodine adsorber, cadmium iodide adsorbed on chromosorb-P; a hypoidous acid adsorber, 4-iodophenol adsorbed on alumina; and an organic iodide adsorber, TEDA charcoal. The charcoal bed was segmented to assure complete adsorption of all of the sample. The sample flow used in all measurements was nominally 2 cfm.

The local samplers employ the same sampling concept as the iodine species sampler described above. The difference being physical size and sample flow. The local samplers have a flow of 0.25 cfm.

A tritium and ^{14}C sampler (5) was also installed along with the particulate and iodine sampler at stations 1, 2, 3 and 4. This system consisted of a silica gel adsorber to remove water vapor which might contain HTO, a molecular sieve adsorber to remove carbon dioxide, a catalytic oxidizer to convert organic species to carbon dioxide and water plus any elemental hydrogen to water. Following the oxidizer was a second silica gel adsorber and molecular sieve adsorber. The sampling concept is to distinguish between oxidized and nonoxidized ^{14}C and ^{3}H species.

TABLE 2.4

AUXILIARY BUILDING SAMPLING STATION FEEDS

Station 1

1. Letdown heat exchanger room
2. Mechanical penetration area
3. Shutdown heat exchanger room
4. Valve room
5. Pipe penetration area
6. Personnel air lock area
7. Sampling room

Station 2

1. Cask decon room
2. Fuel arrival area
3. Fuel storage area
4. Drum storage area
5. Waste baler room
6. Spent resin storage room
7. Volume control tank room
8. Waste evaporator room
9. Waste holdup tank rooms
10. Spent fuel heat exchanger room
11. Safety injection pump rooms
12. Charging pump room
13. Charging pump valve room
14. Fuel Pool area

Station 3

1. Waste decay tank rooms
2. Shutdown cooling heat exchanger room
3. Shutdown cooling heat exchanger valve room
4. Component heat exchanger room

Station 4

1. Spent fuel heat exchanger room
2. Safety injection pump rooms
3. Charging pump room
4. Charging pump valve room

All of the ventilation sampling media were returned to INEL for analysis. The particulates on HEPA filter media were analyzed using a computer based Ge(Li) gamma spectrometer. For a selected list of nuclides, based on activities of long term counts of samples from other reactors, activity levels were determined. If no positive identification could be made for a particular isotope, a "less than" (or lower limit of detection) value was calculated. The "less than" value was calculated by fitting a baseline curve to the area of interest of the spectrum, integrating the area, and assigning an upper limit in the same manner as for a true photopeak.

The adsorbent media for iodine was also analyzed by a gamma spectrometer. In all cases the spectrometers were calibrated for iodine media counting employing secondary standards in the proper geometry (1). Samples analyzed using the secondary standard calibrations were inter-compared with DOE-HSL systems. The comparisons gave good agreement (to within 5 percent).

Tritium adsorbent aliquots were treated with low background water (contains nondetectable amounts of ^{3}H) and counted in situ using liquid scintillation counting techniques.

The carbon-14 adsorbents were degassed in a heated helium flow system and the released gas was collected at liquid nitrogen temperature. After making manometric measurements to establish sample size, the sample was allowed to react with ethanol amine. The sample was then counted by liquid scintillation techniques.

In all but one case, a disintegration rate for the sample or aliquot was determined. The disintegration rate was then converted to activity per unit volume of sample using the sample flow rate and sample time. The exception was the carbon-14 air samples, in this case the counting results were reduced to d/m/cc of CO_2 adsorbed. This value was used to multiply the abundance of CO_2 in air, 330 ppm, to convert to d/m or μCi per cc of sample. In all cases the nuclide concentrations were converted to nuclide release rates in activity per unit time.

The reported results for gaseous samples include an error associated with counting statistics and background fluctuations. In addition, an overall uncertainty of approximately 20 percent should be applied to account for errors in calibration and indeterminates such as duct flow variations and in a few cases uncertainty in sampling duration. The ventilation data are presented in Tables A.35-A.44 in the Appendix.

2.3.2 Waste Gas Processing System

Short term samples were taken from the pressurized storage tanks in the waste gas processing system. In all cases the samples were taken through the plant's Automatic Gas Analyzer System. The system is designed to do hydrogen-oxygen analysis of various cover gases.

Iodine and gas samples for noble gas analysis were taken from this system using the previously described iodine species sampler or a 250 cc gas cylinder (results are reported in Tables A.45-A.48 in the Appendix). Sample duration ranged from 10-60 minutes.

Additional samples for ^{14}C and tritium analysis were withdrawn from the pressurized storage tanks through the gas analyzer system. The analysis of these samples presented several unexpected problems. First, radiation fields from the silica gel adsorbents due to noble gases were 200-400 mr/hr measured at contact. Secondly, the samplers were designed to handle air samples, i.e., oxygen is required for the proper operation of the catalytic oxidizer which provided separation of the various tritium and ^{14}C species. With the exception of one sample (evidenced by an exothermic reaction in the catalytic oxidizer), the cover gas samples contained little or no oxygen. Consequently, special sample collection and handling techniques had to be developed.

For samples containing an appreciable amount of oxygen, an aliquot of the silica gel adsorbent was put in low background water and refluxed in the presence of a helium purge. This procedure excluded the noble gas activities. The purified solution was then counted by liquid scintillation techniques.

As an alternate sampling procedure for samples containing insufficient amounts of oxygen, 75 cc stainless steel gas cylinders were employed for collecting these samples. The sample cylinders were evacuated and pressurized with sample gas. Afterwards, the gas cylinders were returned to INEL where the contents were mixed with low activity air to provide oxygen and to reduce the potential of sample contamination. The samples were further processed by passing the mixture through the ^{3}H-^{14}C sampler and then handled in the previously described manner. The waste gas processing system data are presented in Tables A.45-A.50 in the Appendix.

2.3.3 Containment Atmosphere Sampling

Samples to determine iodine species and ^{3}H-^{14}C were obtained from the containment building atmosphere. In addition, a 250 cc gas cylinder sample of the containment atmosphere was obtained for noble gas analysis. Samples for the determination of iodine species and ^{3}H-^{14}C were taken employing the previously described samplers.

2.4 Plant Operational Data

The plant operational data collected to characterize samples included the reactor log (computerized form), the auxiliary building operator's log, plus information obtained in discussions with plant operating personnel. The information obtained from the reactor log included:

1. Reactor power level.
2. Letdown flow rate.
3. Steam generators steam flow rates and blowdown rates.
4. Reactor coolant boron concentration.

The above information was tabulated on an hourly basis throughout the day and is presented in Figures A.1, A.2, A.6, A.8, A.9, A.10, and A.11 in the Appendix. The information obtained from the auxiliary building operator's log included:

1. Levels in the liquid radwaste system tanks.

2. Pressure drop across the CVCS purification demineralizers and filters.

3. Evaporator operational parameters including liquid feed rate, bottoms boron concentration, and distillate conductivity.

4. Time during the day when specific liquid radwaste tanks were isolated, processed, filled and recirculated.

5. Times during the day when the evaporator was processing a specific liquid radwaste tank and when it was on recirculation.

Items 1 through 3 were on a two-hour interval throughout the day while the specific times when events occurred were noted by the operator in the log. These data are presented in Figures A.7, A.10, A.17, A.18, A.19, A.20, A.21, A.22, A.23, A.24, A.25, A.26, and A.27 in the Appendix.

Discussions with personnel at Fort Calhoun Station verified that the radwaste tank sizes, demineralizer types, and evaporator were as specified in the FSAR and presented in Tables 2.1 and 2.2. Fort Calhoun did not recycle any of the liquid or gas streams that reached the radwaste system.

CHAPTER 3 - DISCUSSION OF DATA

3.1 Introduction

In this chapter, data obtained from sample analysis are discussed on a component or plant system basis. Plant systems and components are related to the principal parameters used in the NRC calculational models (6). Each component is addressed in a separate section.

3.2 Thermal Power Level

The maximum designed and licensed power level for the Fort Calhoun Station is 1420 MWt. During the measurement period, the reactor was operating near or at this power level, except for hot shutdowns on 8/31/76, 9/24/76 and 1/14/77 and the refueling outage.

In the Appendix, Figure A.1 presents a plot of the power level for the period 8/15/76 through 10/3/76. Figure A.2 presents the power level for the period 12/15/76 through 2/24/77. From 10/1/76 through 12/15/76, Fort Calhoun was down for its second refueling.

3.3 Plant Capacity Factor

Information from OPPD (8) indicated that the capacity factors for the Fort Calhoun plant were: 60.4% in 1974, 52% in 1975 and 54.7% in 1976.

3.4 Radionuclide Concentrations in Pressurized Water Reactor Liquid System Components

Radionuclide concentrations were measured in samples obtained from liquid system components at Fort Calhoun. These included: reactor coolant, steam generator blowdown, spent fuel pool, reactor coolant drain tank, waste holdup tanks, spent regenerant tanks, hotel waste tanks and monitor tanks. Results will be discussed in detail later in this section.

3.4.1 Predicted Concentrations in Reactor Coolant and Secondary Water

Table 3.1 presents the expected radionuclide activity levels in the primary and secondary coolant water for the Fort Calhoun Station based on the N-237 standard (7). These values were derived by adjusting the parameters of the reference PWR to those of the Fort Calhoun plant. Techniques for the adjustments are presented in the American National Standard N-237/ANS-18.1 publication of the American Nuclear Society (7). The values used for adjustment of the reference PWR values to the Fort Calhoun values are presented in Table 3.2. The parameters are for U-tube

TABLE 3.1

PREDICTED CONCENTRATIONS IN REACTOR COOLANT AND SECONDARY WATER

Calculated for Fort Calhoun from N-237[7] Reference Values

Nuclide	Reactor Coolant (μCi/gm)	Secondary Water* (μCi/gm)
Noble Gases - Class 1:		
** ^{83m}Kr	1.7(-2)	Nil
^{85m}Kr	8.7(-2)	Nil
^{85}Kr	1.5(-1)	Nil
^{87}Kr	4.8(-2)	Nil
^{88}Kr	1.6(-1)	Nil
** ^{89}Kr	4.0(-3)	Nil
^{131m}Xe	9.3(-2)	Nil
^{133m}Xe	1.8(-1)	Nil
^{133}Xe	1.5(+1)	Nil
^{135m}Xe	1.0(-2)	Nil
^{135}Xe	2.8(-1)	Nil
** ^{137}Xe	7.1(-3)	Nil
^{138}Xe	3.5(-2)	Nil
Halogens - Class 2:		
** ^{83}Br	4.0(-3)	5.1(-7)
** ^{84}Br	2.1(-3)	5.1(-8)
** ^{85}Br	2.4(-4)	4.7(-10)
** ^{130}I	1.9(-3)	1.8(-6)
^{131}I	2.7(-1)	1.6(-3)
^{132}I	8.2(-2)	3.9(-5)
^{133}I	3.5(-1)	5.6(-4)
^{134}I	3.8(-2)	1.6(-7)
^{135}I	1.6(-1)	7.3(-5)
Cesiums, Rubidiums - Class 3:		
** ^{86}Rb	8.6(-5)	5.1(-7)
^{88}Rb	1.6(-1)	1.9(-6)
^{134}Cs	2.6(-2)	1.3(-4)
^{136}Cs	1.3(-2)	6.4(-5)
^{137}Cs	1.8(-2)	1.1(-4)
Water Activation Products - Class 4:		
** ^{16}N	4.0(+1)	2.9(-6)
Tritium - Class 5:		
^{3}H	1.0(0)	1.0(-3)

TABLE 3.1 (cont'd)

PREDICTED CONCENTRATIONS IN REACTOR COOLANT AND SECONDARY WATER

Calculated for Fort Calhoun from N-237(7) Reference Values

Nuclide	Reactor Coolant (μCi/gm)	Secondary Water* (μCi/gm)
Other Nuclides - Class 6:		
^{51}Cr	1.9(-3)	1.0(-5)
^{54}Mn	3.1(-4)	2.7(-6)
^{55}Fe	1.6(-3)	9.4(-6)
^{59}Fe	1.0(-3)	6.4(-6)
^{58}Co	1.6(-2)	9.0(-5)
^{60}Co	2.0(-3)	1.2(-5)
^{89}Sr	3.5(-4)	2.6(-6)
^{90}Sr	1.0(-5)	6.4(-8)
** ^{91}Sr	5.7(-4)	3.3(-7)
** ^{90}Y	1.2(-6)	2.1(-8)
^{91}Y	6.5(-5)	3.9(-7)
** ^{91m}Y	2.9(-4)	8.1(-8)
** ^{93}Y	3.0(-5)	2.3(-8)
^{95}Zr	6.0(-5)	3.9(-7)
^{95}Nb	5.0(-5)	3.9(-7)
^{99}Mo	8.2(-2)	3.2(-4)
^{99m}Tc	4.1(-2)	1.4(-4)
^{103}Ru	4.5(-5)	2.6(-7)
** ^{106}Ru	1.0(-5)	6.4(-8)
** ^{103m}Rh	3.6(-5)	5.6(-8)
** ^{106}Rh	7.9(-6)	1.2(-8)
** ^{125m}Te	2.9(-5)	1.2(-7)
** ^{127m}Te	2.8(-4)	1.2(-6)
** ^{127}Te	7.5(-4)	1.1(-6)
^{129m}Te	1.4(-3)	7.7(-6)
^{129}Te	1.3(-3)	1.7(-6)
^{131m}Te	2.3(-3)	4.3(-6)
** ^{131}Te	8.8(-4)	1.2(-6)
^{132}Te	2.6(-2)	8.6(-5)
** ^{137m}Ba	1.3(-2)	1.8(-5)
^{140}Ba	2.2(-4)	1.1(-6)
^{140}La	1.4(-4)	7.7(-7)
^{141}Ce	7.0(-5)	3.9(-7)
^{143}Ce	3.8(-5)	8.2(-8)
^{144}Ce	3.3(-5)	2.7(-7)
** ^{143}Pr	5.0(-5)	2.5(-7)
** ^{144}Pr	2.6(-5)	4.7(-8)
^{239}Np	1.2(-3)	2.5(-6)

* : Calculation assumed all volatile treatment chemistry for secondary water.
**: N-237 (7) radionuclides not directly measured at Fort Calhoun.

TABLE 3.2

PARAMETER VALUES USED TO MODIFY N-237 (7)
ACTIVITIES FOR FORT CALHOUN

Parameter	Symbol	Units	Fort Calhoun Value
Thermal power	P	MWt	1420 *
Steam flow rate (both generators)	FS	lbs/hr	6.2(6)*
Weight of water in reactor coolant system	WP	lbs	2.9(5)**
Weight of water in all steam generators	WS	lbs	1.56(5)***
Reactor coolant letdown flow (purification)	FD	lbs/hr	1.5(4)*
Reactor coolant letdown flow (yearly average for boron control)	FB	lbs/hr	200 *
Steam generator blowdown flow (total)	FBD	lbs/hr	25,000 *
Fraction of radioactivity in blowdown stream which is not returned to the secondary coolant system.	NBD	------	1.0 *
Flow through the purification system cation demineralizer	FA	lbs/hr	1600 *
Ratio of condensate demineralizer flow rate to total steam flow rate	NC	------	0.0 *
Ratio of the total amount of noble gases routed to gaseous radwaste from the purification system to the total amount routed from the primary coolant system to the purification system (not including the boron recovery system)	Y	------	0.0 *

* : Based on information obtained during measurement program.

**: Information from FSAR (3).

***: Information from (9).

steam generators assuming all volatile treatment (AVT) of the secondary coolant. In Tables 3.1 and 3.2, as in all other tables in this report, the number in parentheses following a number represents the power of ten multiplier.

The radionuclides presented in Table 3.1 include all the radioisotopes discussed in N-237. In the subsequent data tables, all of the radioisotopes discussed in N-237 are not included.

The techniques used in collecting the data would have detected and quantified all gamma-emitting radionuclides present in the collected samples. However, our experience with these measurements has indicated that certain radionuclides are either not present or should not be reported due to potential errors in sample collection.

An example of these potential errors is the measurement of the noble gas activities in liquid samples. Only with specialized sample collection techniques is it possible to quantitatively determine the dissolved noble gas concentrations in liquid samples. At almost all sampling points used, the only method available for sample collection was to transfer the liquids into open volumetric measuring devices. This procedure results in an unknown degasification factor for the dissolved noble gases (i.e., an indeterminate amount of noble gases is removed from the liquid sample). Consequently, noble gases have not been included in most of the data tables. Where noble gas concentrations are reported, more specialized sample techniques (i.e., plumbing the sampler in series with the sample line) were employed. This was done for selected reactor coolant samples and the data are presented in the reactor coolant section.

Certain radionuclides that are reported in N-237 have not been observed in data obtained at Fort Calhoun. For example, the radionuclides, ^{130}I and ^{86}Rb, were not detected because of their very low fission yield (about 2 X 10^{-3} percent for each isotope). The radionuclides ^{16}N and ^{93}Y were not detected due to their short half-lives or low gamma-ray branching ratios.

Several of the radioisotopes listed in N-237 are daughters of measured parent radioisotopes. For example, ^{90}Sr-^{90}Y, ^{91}Sr-^{91m}Y, ^{99}Mo-^{99m}Tc, ^{103}Ru-^{103m}Rh, ^{106}Ru-^{106m}Rh, ^{137}Cs-^{137m}Ba, and ^{144}Ce-^{144}Pr are such parent-daughter radioisotopic pairs. It is our experience in the radionuclide measurement of samples from nuclear plant systems that these parent-daughter pairs are in radioactive decay equilibrium because of production, half-life, and time considerations. Therefore, the parent activities are reported and the daughter activities can be easily calculated assuming radioactive decay equilibrium.

No bromine activities were detected in liquid samples from Fort Calhoun. Particular attention has been given to the detection of ^{84}Br due to its decay scheme and relatively large fission yield. However, it has not been observed. A potential explanation for this is that bromine is a very reactive element and could attach to the surfaces of reactor system components.

The tellurium isotopes have not been observed with any consistency in either reactor coolant or other plant process streams. The tellurium radioisotopes ^{129m}Te and ^{132}Te have large fission yields, relatively long half-lives and detectable gamma-ray emissions. Particular attention has been given to detection of these two tellurium isotopes, but they have not been observed with any consistency. A potential explanation is that these radionuclides are fixed on the reactor system components. The fixing of radionuclide activities to structural materials can be a time varying phenomena. That is, it can be a function of the chemical and physical properties (pH, temperature, flow rate, ions present, chemical form of radioelement, etc.). The tellurium isotopes are presented in the data tables when detected.

3.4.2 Measured Reactor Coolant Radionuclide Concentrations

Table 3.3 lists the number of samples, mean values, and ranges of radionuclide concentrations in reactor coolant. Data summarized in Table 3.3 are presented in Tables A.1, A.2, A.3, and A.7 of the Appendix. If only one of the samples had a measured value for a specific radionuclide, this value is presented as the mean along with its one sigma standard deviation error. Also, for the pure beta-emitting radionuclides, usually one sample was collected for analysis during a particular measurement period. If none of the samples collected during a particular measurement period resulted in a measurable value for a specific radionuclide, then the smallest lower detection limit is presented as the mean. The radionuclide concentrations presented in the first column are from Table 3.1. Values from Table 3.1 have been converted from μCi/gm to μCi/ml assuming a density of 1.0 gm/ml for water.

As indicated in Table 3.3, ^{131}I, ^{58}Co, and ^{3}H have significantly different concentrations for the pre-refueling and post-refueling periods. Iodine-131 and ^{58}Co were observed to decrease after refueling while the ^{3}H increased. There is no obvious explanation for the increase in tritium after refueling. However, weekly tritium analysis by OPPD of reactor coolant indicated the same increase. The iodine decrease is attributed to cleanup and decay. The ^{58}Co coolant level activity decreased by approximately 70 percent after refueling. Based on a forty-five day outage, the ^{58}Co should have decayed by only approximately 35 percent, indicating cleanup by the CVCS demineralizers during the refueling outage. A decrease in ^{60}Co levels was not observed. This may be attributed to the higher DF's for ^{58}Co than for ^{60}Co observed in both the CVCS purification and cation demineralizers (see Tables 3.23 and A.11). The higher observed DF's for ^{58}Co may be due to the fact that ^{58}Co and ^{60}Co exist as different chemical species in reactor liquids.

The rest of the radionuclides presented in Table 3.3 had similar radionuclide levels before and after refueling or were highly variable.

Table 3.4 presents radionuclide concentrations in reactor coolant during the refueling outage, before and after spent fuel movement. Since there was only one sample of reactor coolant taken after fuel

TABLE 3.3

RADIONUCLIDE CONCENTRATIONS IN REACTOR COOLANT (REACTOR POWER OPERATIONS)

Nuclide	Calculated Using N-237 (7) (μCi/ml)	Measured: Before Refueling — Samples: 14:52; 8/18/76, 09:17; 8/19/76, 09:58; 8/19/76, 11:19; 9/1/76, 11:54; 9/1/76 — Mean (μCi/ml)	Range (μCi/ml)	Measured: After Refueling — Samples: 08:40; 2/9/77, 08:44; 2/17/77, 13:01; 2/22/77 — Mean (μCi/ml)	Range (μCi/ml)
Noble Gases:					
^{85m}Kr	8.7(-2)	*		1.7(-1)	1.4-1.9(-1)
^{85}Kr	1.5(-1)	*		1.3(-2)	0.8-1.8(-2)
^{87}Kr	4.8(-2)	*		1.7(-1)	1.4-1.9(-1)
^{88}Kr	1.6(-1)	*		2.9(-1)	2.6-3.3(-1)
^{131m}Xe	9.3(-2)	*		4.5(-2)	3.1-5.2(-2)
^{133m}Xe	1.8(-1)	*		1.3(-1)	1.2-1.4(-1)
^{133}Xe	1.5(+1)	*		5.5(0)	5.0-6.0(0)
^{135m}Xe	1.0(-2)	*		8.6(-2)	7.0-9.5(-2)
^{135}Xe	2.8(-1)	*		9.8(-1)	0.9-1.1(0)
^{138}Xe	3.5(-2)	*		1.6(-1)	1.5-1.6(-1)
Iodines:					
^{131}I	2.7(-1)	2.7(-1)	1.5-3.9(-1)	8.5(-2)	7.4-9.5(-2)
^{132}I	8.2(-2)	9.7(-2)	0.82-1.1(-1)	5.2(-2)	4.6-6.3(-2)
^{133}I	3.5(-1)	1.4(-1)	0.34-2.4(-1)	1.1(-1)	1.0-1.3(-1)
^{134}I	3.8(-2)	3.3(-2)	2.8-3.8(-2)	3.5(-2)	2.8-4.2(-2)
^{135}I	1.6(-1)	6.1(-2)	2.8-7.3(-2)	7.3(-2)	6.8-8.2(-2)
Cs and Rb:					
^{88}Rb	1.6(-1)	4.5(-1)	3.8-5.3(-1)	5.0(-1)	4.6-5.3(-1)
^{89}Rb	***	5.9(-2)	4.5-7.1(-2)	1.9(-2)	1.3-2.3(-2)
^{134}Cs	2.6(-2)	1.5(-2)	1.4-1.6(-2)	2.2(-2)	0.1-6.2(-2)
^{136}Cs	1.3(-2)	4.0(-3)	3.6-4.7(-3)	6.3(-5)	3.5-9.7(-5)
^{137}Cs	1.8(-2)	1.4(-2)	1.4-1.6(-2)	2.6(-2)	0.2-7.1(-2)
^{138}Cs	***	3.3(-1)	3.1-3.7(-1)	1.7(-1)	1.5-2.0(-1)
^{139}Cs	***	1.6 ± 0.4(-1)	**	<4(-4)	
Tritium:					
^{3}H	1.0(0)	4.77 ± 0.02(-2)	**	2.15 ± 0.02(-1)	**
Other Nuclides:					
^{14}C	***	4.7 ± 0.4(-6)	**	7.0 ± 0.2(-6)	**
^{24}Na	***	1.9(-3)	1.3-2.8(-3)	1.1(-2)	0.75-1.4(-2)
^{32}P	***	4.5 ± 0.4(-6)	**	*	
^{35}S	***	5.6 ± 2.0(-6)	**	*	

See bottom of page, Table 3.3 (cont.) for asterisk notes.

TABLE 3.3 (cont'd)

RADIONUCLIDE CONCENTRATIONS IN REACTOR COOLANT (REACTOR POWER OPERATIONS)

		Measured: Before Refueling		Measured: After Refueling	
		Samples: 14:52; 8/18/76 09:17; 8/19/76 09:58; 8/19/76 11:19; 9/1/76 11:54; 9/1/76		Samples: 08:40; 2/9/77 08:44; 2/17/77 13:01; 2/22/77	
Nuclide	Calculated Using N-237 (7) (μCi/ml)	Mean (μCi/ml)	Range (μCi/ml)	Mean (μCi/ml)	Range (μCi/ml)
Other Nuclides: (cont'd)					
^{41}Ar	***	*		3.4(-3)	2.0-5.6(-3)
^{45}Ca	***	1.08 ± 0.01(-4)	**	*	
^{51}Cr	1.9(-3)	<1(-2)		7.3 ± 1.8(-4)	**
^{54}Mn	3.1(-4)	2.5(-4)	0.13-1.1(-3)	5.6(-4)	0.022-1.6(-3)
^{56}Mn	***	3.6(-2)	3.3-3.7(-2)	1.4 ± 0.4(-2)	**
^{55}Fe	1.6(-3)	1.06 ± 0.01(-3)	**	1.16 ± 0.02(-4)	**
^{59}Fe	1.0(-3)	7.5(-5)	0.56-1.1(-4)	1.1(-4)	0.47-1.8(-4)
^{57}Co	***	2.8(-3)	**	1.5(-3)	0.36-2.3(-3)
^{58}Co	1.6(-2)	7.1(-3)	0.60-1.2(-2)	2.2(-3)	1.1-3.6(-3)
^{60}Co	2.0(-3)	1.3(-4)	0.79-2.0(-4)	5.7(-5)	0.18-2.1(-4)
^{63}Ni	***	1.76 ± 0.02(-4)	**	3.7 ± 0.2(-5)	**
^{65}Zn	***	2.8(-3)	1.1-5.0(-3)	<2(-5)	
^{89}Sr	3.5(-4)	1.22 ± 0.05(-3)	**	8.6 ± 0.9(-6)	**
^{90}Sr	1.0(-5)	3.8 ± 0.5(-6)	**	<8(-7)	
^{91}Y	65.(-5)	7.7 ± 1.3(-6)	**	1.3 ± 0.2(-6)	**
^{95}Zr	6.0(-5)	4.4(-4)	0.8-8.1(-4)	4.3 ± 1.2(-5)	**
^{95}Nb	5.0(-5)	6.2(-5)	3.8-8.6(-5)	3.8 ± 0.3(-5)	**
^{99}Mo	8.2(-2)	1.3(-2)	0.25-1.5(-2)	1.4(-3)	0.064-1.4(-3)
^{103}Ru	4.5(-5)	6.9(-4)	0.27-1.1(-3)	1.7(-5)	1.7-1.7(-5)
^{110m}Ag	***	2.1(-4)	0.2-3.8(-4)	<3(-5)	
^{124}Sb	***	1.2 ± 0.7(-5)	**	3.6 ± 2.4(-6)	**
^{129m}Te	1.4(-3)	<2(-3)		1.9 ± 0.8(-4)	**
^{129}Te	1.3(-3)	<7(-3)		<2(-4)	
^{131m}Te	2.3(-3)	<3(-4)		<2(-4)	
^{132}Te	2.6(-2)	<1(-3)		<2(-3)	
^{139}Ba	***	6.0(-2)	4.2-7.4(-2)	5.6(-2)	4.9-6.3(-2)
^{140}Ba	2.2(-4)	5.0(-4)	3.2-5.9(-4)	<1(-4)	
^{140}La	1.4(-4)	4.0(-4)	3.0-4.7(-4)	2.6(-4)	0.87-4.4(-4)
^{141}Ce	7.0(-5)	3.0 ± 1.3(-4)	**	2.7 ± 1.5(-4)	**
^{143}Ce	3.8(-5)	1.2(-2)	0.083-3.0(-2)	<3(-3)	
^{144}Ce	3.3(-5)	<6(-4)		2.3 ± 0.3(-3)	**
^{147}Pm	***	9.0 ± 2.0(-7)	**	*	
^{187}W	***	1.9(-2)	0.26-3.8(-2)	7.0(-3)	5.4-8.6(-3)
^{239}Np	1.2(-3)	1.4(-2)	0.53-2.0(-2)	7.0(-3)	4.3-9.6(-3)

* : Analysis not performed for radionuclide.
** : Detected in one measurement, only, for this radionuclide.
***: Radionuclide not treated in N-237 (7).

TABLE 3.4

RADIONUCLIDE CONCENTRATIONS IN REACTOR COOLANT (REACTOR REFUELING)

Nuclide	Calculated Using N-237 (7) (μCi/ml)	Measured: Prior to Fuel Transfer, Sample: 12:36; 10/20/76, 11:00; 10/27/76 — Mean (μCi/ml)	Range (μCi/ml)	During Refueling: After Fuel Transfer, Sample: 13:15; 12/3/76 — Activity (μCi/ml)
Iodines:				
^{131}I	2.7(-1)	**1.2 ± 0.2(-2)**	**	<4(-4)
Cesiums:				
^{134}Cs	2.6(-2)	3.0(-2)	2.3 - 3.7(-2)	6.18 ± 0.09(-3)
^{136}Cs	1.3(-2)	1.5(-3)	0.9 - 2.1(-3)	1.9 ± 0.3(-4)
^{137}Cs	1.8(-2)	3.0(-2)	2.3 - 3.8(-2)	6.3 ± 0.1(-3)
Tritium:				
^{3}H	1.0(0)	3.5(-2)	3.4 - 3.5(-2)	1.06 ± 0.02(-2)
Other Nuclides:				
^{14}C	***	2.4(-5)	2.1 - 2.7(-5)	5.3 ± 0.2(-7)
^{32}P	***	1.1(-5)	0.5 - 1.6(-5)	*
^{35}S	***	**1.4 ± 0.9(-6)**	**	2.50 ± 0.06(-5)
^{45}Ca	***	<6(-7)		*
^{51}Cr	1.9(-3)	<2(-3)		2.9 ± 0.9(-3)
^{54}Mn	3.1(-4)	3.1(-4)	2.5 - 3.6(-4)	8.7 ± 0.2(-4)
^{55}Fe	1.6(-3)	1.3(-3)	0.39- 2.2(-3)	3.91 ± 0.02(-3)
^{59}Fe	1.0(-3)	1.4(-4)	1.2 - 1.6(-4)	3.2 ± 0.4(-4)
^{57}Co	***	3.2(-4)	2.4 - 4.1(-4)	2.3 ± 0.3(-4)
^{58}Co	1.6(-2)	1.6(-1)	1.3 - 1.8(-1)	7.0 ± 0.1(-2)
^{60}Co	2.0(-3)	6.7(-4)	6.4 - 6.9(-4)	1.6 ± 0.3(-3)
^{63}Ni	***	4.8(-3)	3.9 - 5.8(-3)	3.94 ± 0.02(-3)
^{65}Zn	***	**8.2 ± 2.5(-5)**	**	5.1 ± 1.0(-5)
^{89}Sr	3.5(-4)	2.7(-5)	0.53- 4.8(-5)	5.3 ± 0.3(-5)
^{90}Sr	1.0(-5)	1.2(-6)	0.4 - 2.0(-6)	1.08 ± 0.05(-5)
^{91}Y	6.5(-5)	9.0(-7)	0.3 - 1.5(-6)	2.13 ± 0.08(-5)
^{95}Zr	6.0(-5)	<1(-2)		1.1 ± 0.2(-4)
^{95}Nb	5.0(-5)	<9(-3)		2.6 ± 1.7(-4)
^{103}Ru	4.5(-5)	**1.5 ± 0.8(-4)**	**	<3(-4)
^{110m}Ag	***	<1(-3)		1.5 ± 0.1(-4)
^{124}Sb	***	9.6(-5)	0.61- 1.3(-4)	1.28 ± 0.09(-4)
^{129m}Te	1.4(-3)	<4(-1)		<2(-1)
^{140}Ba	2.2(-4)	<5(-4)		5.3 ± 2.1(-4)
^{141}Ce	7.0(-5)	<1(-4)		<4(-5)
^{144}Ce	3.3(-5)	**2.6 ± 2.3(-4)**	**	<2(-4)
^{147}Pm	***	1.4(-6)	1.1 - 1.7(-6)	*

* : Analysis not performed for radionuclide.
** : Detected in one measurement, only, for this radionuclide.
*** : Radionuclide not treated in N-237 (7).

movement, the measured activity level and the one sigma standard deviation based on counting statistics are presented instead of a mean and range. The data summarized in Table 3.4 are presented in Table A.4 of the Appendix. Radionuclides with half-lives of less than eight days are not presented in Table 3.4 because these was no production of radionuclides due to the reactor not operating. Between the collection of the two data groups of Table 3.4 (before and after fuel transfer), the reactor coolant was being processed through the cation demineralizer of the CVCS. This resulted in a significant reduction in the cesium concentrations as is shown in Table 3.4. The data for the cation demineralizer are presented in Table A.11 in the Appendix.

3.4.3 Radionuclide Concentrations in Steam Generator Water

Water samples were taken of the steam generator blowdown for both generators before and after refueling. Table 3.5 presents the results for blowdown samples. For comparison, the expected radionuclide concentrations calculated using the N-237 model are also listed. Data summarized in Table 3.5 are in Table A.13 in the Appendix. The blowdown activity levels were observed to be lower than predicted by N-237. At the time the samples were taken, the total steam flow rate was 6 million pounds per hour and the total blowdown rate was 20,000 lbs/hr. Samples obtained prior to refueling were 450 ml in volume, which did not provide adequate measurement sensitivity for many nuclides. After refueling, large volume samples (about 200 liters) were processed through ion exchange columns to improve sensitivity. As shown in Table 3.5, the presence of several radionuclides was confirmed using this technique. However, there was no indication of a primary to secondary leak (based on ^{131}I, levels $< 10^{-2}$ lbs/day). Fort Calhoun Station released all the steam generator blowdown water to the discharge pipe during the period of the in-plant measurements. That is, none of the blowdown waters were processed with demineralizers prior to release or recycled within the plant.

3.4.4 Concentrations of Radionuclides in Spent Fuel Pool and Refueling Associated Waters

Mean values and observed ranges of radionuclide concentrations in the spent fuel pool samples are presented in Table 3.6. All measurements are presented in Tables A.2, A.14, and A.18 of the Appendix. Before refueling, the spent fuel pool contained 32 expended fuel assemblies from the refueling in the spring of 1975. After refueling, the fuel pool contained 64 expended fuel assemblies. Table 3.7 presents the radionuclide concentrations observed in fuel pool water during refueling. The data are presented from measurements made before and after fuel movement to the pool. All data obtained are listed in Table A.17 of the Appendix. With the exception of ^{14}C, radionuclide concentrations in the spent fuel pool increased after fuel movement.

Using measured tritium concentrations in the reactor coolant, refueling cavity, spent fuel pool, and the safety injection and refueling

TABLE 3.5

RADIONUCLIDE CONCENTRATIONS IN STEAM GENERATOR WATER

Nuclide	Calculated Generator Water Using N-237(7) (μCi/ml)	Steam Generator Blowdown Activities: Measured: Before Refueling — Generator A: 09:40; 8/26/76; Generator B: 09:40; 8/26/76 — Mean of Both (μCi/ml)	Range (μCi/ml)	Measured: After Refueling — Generator A: 2/08 - 09/77; Generator B: 2/10 - 11/77 — Mean of Both (μCi/ml)	Range (μCi/ml)
Iodines:					
^{131}I	1.6(-3)	<4(-8)		5.5(-10)	1.7 - 9.3(-10)
^{133}I	5.6(-4)	*		<3(-10)	
Cesiums:					
^{134}Cs	1.3(-4)	<1(-7)		5.3(-9)	2.4 - 8.2(-9)
^{136}Cs	6.4(-5)	*		8.4 ± 6.7(-11)	**
^{137}Cs	1.1(-4)	<1(-6)		8.3(-9)	0.38 - 1.3(-8)
Tritium:					
^{3}H	1.0(-3)	1.7 ± 0.2(-5)	**	†	
Other Nuclides:					
^{14}C	***	<3(-8)		†	
^{24}Na	***	*		<1(-10)	
^{32}P	***	<4(-8)		†	
^{51}Cr	1.0(-5)	*		4.0(-10)	3.9 - 4.0(-10)
^{54}Mn	2.7(-6)	<1(-7)		1.0(-9)	0.70 - 1.3(-9)
^{55}Fe	9.4(-6)	5.2 ± 0.1(-6)	**	†	
^{59}Fe	6.4(-6)	*		<3(-10)	

TABLE 3.5 (cont'd)

RADIONUCLIDE CONCENTRATIONS IN STEAM GENERATOR WATER

		Steam Generator Blowndown Activities:			
		Measured: Before Refueling		Measured: After Refueling	
		Generator A: 09:40; 8/26/76		Generator A: 2/08 - 09/77	
		Generator B: 09:40; 8/26/76		Generator B: 2/10 - 11/77	
Nuclide	Calculated Generator Water Using N-237(7) (μCi/ml)	Mean of Both (μCi/ml)	Range (μCi/ml)	Mean of Both (μCi/ml)	Range (μCi/ml)
Other Nuclides: (cont)					
^{57}Co	***	*		<6(-11)	
^{58}Co	9.0(-5)	*		3.0(-9)	0.78 - 5.2(-9)
^{60}Co	1.2(-5)	<2(-7)		1.2(-9)	0.64 - 1.8(-9)
^{63}Ni	***	<8(-8)		†	
^{89}Sr	2.6(-6)	<4(-8)		†	
^{90}Sr	6.4(-8)	<2(-8)		†	
^{91}Y	3.9(-7)	<1(-8)		†	
^{95}Zr	3.9(-7)	*		<3(-10)	
^{95}Nb	3.9(-7)	*		<1(-10)	**
^{99}Mo	3.2(-4)	*		<6(-11)	
^{103}Ru	2.6(-7)	*		<3(-10)	**
^{124}Sb	***	*		2.1(-10)	0.9 - 3.3(-10)

* : Radionuclide not observed.
** : Detected in one measurement, only, for this radionuclide.
***: Radionuclide not treated in N-237 (7).
† : Analysis not done for this radionuclide.

TABLE 3.6

RADIONUCLIDE CONCENTRATIONS IN SPENT FUEL POOL (REACTOR POWER OPERATIONS)

	Before Refueling: Samples: 10:49; 08/23/76, 10:31; 08/31/76		After Refueling Sample: 08:45; 02/22/77
Nuclide	Mean (μCi/ml)	Range (μCi/ml)	Activity (μCi/ml)
Cesiums:			
^{134}Cs	9.4(-5)	9.2 - 9.6(-5)	7.6 ± 0.2(-5)
^{136}Cs	2.7 ± 1.9(-7)	**	<6(-5)
^{137}Cs	2.0(-4)	1.9 - 2.1(-4)	1.39 ± 0.02(-4)
Tritium:			
^{3}H	7.34 ± 0.04(-3)	**	8.94 ± 0.03(-3)
Other Nuclides:			
^{14}C	7.0 ± 2.0(-8)	**	1.8 ± 0.1(-7)
^{32}P	<2(-7)		*
^{35}S	<2(-7)		*
^{45}Ca	1.44 ± 0.02(-5)	**	*
^{54}Mn	1.7(-5)	1.7 - 1.7(-5)	1.38 ± 0.04(-5)
^{55}Fe	1.85 ± 0.02(-5)	**	1.12 ± 0.02(-4)
^{59}Fe	<2(-6)		2.0 ± 0.3(-6)
^{57}Co	6.0(-6)	5.6 - 6.4(-6)	7.3 ± 0.3(-6)
^{58}Co	8.7(-4)	8.6 - 8.7(-4)	1.39 ± 0.02(-3)
^{60}Co	4.1(-5)	4.0 - 4.2(-5)	4.77 ± 0.07(-5)
^{63}Ni	2.37 ± 0.04(-4)	**	3.42 ± 0.02(-4)
^{65}Zn	<1(-6)		<2(-6)
^{89}Sr	<4(-8)		5.5 ± 0.6(-7)
^{90}Sr	<2(-8)		1.7 ± 0.2(-7)
^{91}Y	<1(-7)		4.8 ± 0.2(-7)
^{95}Zr	7 ± 2(-7)	**	<1(-4)
^{95}Nb	<4(-7)		1.2 ± 0.4(-6)
^{99}Mo	9 ± 7(-7)	**	<4(-7)
^{110m}Ag	2.8(-6)	2.2 - 3.3(-6)	<2(-6)
^{124}Sb	4.9(-7)	3.9 - 5.9(-7)	<3(-6)
^{147}Pm	<2(-8)		*

* : Analysis not performed for radionuclide.

** : Detected in one measurement, only, for this radionuclide.

TABLE 3.7

RADIONUCLIDE CONCENTRATIONS IN SPENT FUEL POOL (REACTOR REFUELING)

Nuclide	Before Fuel Transfer: Samples: 10:05; 10/14/76, 11:05; 10/27/76 Mean (μCi/ml)	Range (μCi/ml)	After Fuel Transfer: Samples: 14:10; 11/08/76, 14:10; 12/02/76 Mean (μCi/ml)	Range (μCi/ml)
Iodines:				
^{131}I	<3(-7)	<6(-7)	<6(-6)	<8(-6)
Cesiums:				
^{134}Cs	1.0(-4)	0.82 - 1.2(-4)	2.5(-3)	2,5 - 2.6(-3)
^{136}Cs	2.3 ± 0.7(-6)	**	4.1(-5)	1.5 - 6.6(-5)
^{137}Cs	2.2(-4)	1.9 - 2.5(-4)	2.7(-3)	2.6 - 2.7(-3)
Tritium:				
^{3}H	7.3(-3)	7.2 - 7.3(-3)	9.2(-3)	9.1 - 9.2(-3)
Other Nuclides:				
^{14}C	9.2(-7)	0.67 - 1.2(-6)	4.6(-7)	0.3 - 8.8(-7)
^{32}P	<2(-7)		<4(-7)	
^{35}S	<2(-7)		1.1(-6)	0.5 - 1.7(-6)
^{45}Ca	<6(-8)		<6(-8)	
^{51}Cr	<3(-6)		7.6 ± 2.5(-5)	**
^{54}Mn	1.7(-5)	1.5 - 1.9(-5)	8.3(-5)	8.2 - 8.3(-5)
^{55}Fe	8.7(-6)	0.49 - 1.3(-5)	5.0(-5)	2.4 - 7.5(-5)
^{59}Fe	<4(-6)		9.5 ± 1.3(-6)	**
^{57}Co	5.2(-6)	4.5 - 5.9(-6)	4.0(-5)	3.7 - 4.2(-5)
^{58}Co	5.6(-4)	5.1 - 6.2(-4)	1.5(-2)	1.4 - 1.5(-2)
^{60}Co	4.8(-5)	4.5 - 5.0(-5)	1.4(-4)	1.2 - 1.5(-4)
^{63}Ni	2.7(-4)	2.6 - 2.8(-4)	1.0(-3)	0.97- 1.0(-3)
^{65}Zn	<4(-6)		4.0(-6)	2.2 - 5.7(-6)
^{89}Sr	<4(-8)		6.9(-6)	6.8 - 7.0(-6)
^{90}Sr	<2(-8)		7.8(-7)	7.2 - 8.4(-7)
^{91}Y	<8(-8)		3.1(-6)	2.9 - 3.4(-6)
^{95}Zr	<4(-5)		<2(-3)	
^{95}Nb	<2(-5)		<1(-3)	
^{103}Ru	<8(-6)		<1(-4)	
^{110m}Ag	2.0(-6)	1.5 - 2.5(-6)	6.2(-6)	4.0 - 8.4(-6)
^{124}Sb	4.0(-7)	2.9 - 5.1(-7)	6.1(-6)	0.11- 1.1(-5)
^{140}Ba	<3(-5)		1.3 ± 0.4(-4)	**
^{141}Ce	<3(-7)		<6(-6)	
^{144}Ce	<2(-6)		<3(-5)	
^{147}Pm	2.0(-7)	1.5 - 2.5(-7)	4.4(-7)	1.9 - 6.8(-7)

**: Detected in one measurement, only, for this radionuclide.

water storage tank (SIRWT), a tritium balance was estimated for these compartments during the refueling outage. The refueling cavity and SIRWT activity values are listed in Tables A.16 and A.15 in the Appendix. Table 3.8 gives these tritium balance estimates and the volumes of the respective compartments at the four different times during the refueling outage, i.e., before refilling the refueling cavity (10/12-10/20/76), after filling the refueling cavity but before spent fuel movement (10/27-11/1/76), after spent fuel movement but before draining the refueling cavity (11/8/76), and after completion of refueling and draining of the cavity (12/2-12/3/76). Note that during the refueling outage the reactor coolant total tritium content decreased by about 3 Ci while the total tritium in both the SIRWT and spent fuel pool increased by about 2 Ci. This discrepancy may be related to the sampling technique (dip samples) used in the reactor cavity and the fuel transfer canal. Sampling techniques and locations for spent fuel pool and refueling cavity are described and shown in Section 2.2.3 and Figure A.12, respectively.

3.4.5 Radionuclide Concentrations in Reactor Coolant Drain Tank

The sample point for the reactor coolant drain tank was on the tank and within containment. Because access to containment was limited, only one sample was taken. This sample was obtained after refueling. Table A.19 in the Appendix presents the radionuclides observed in this one sample. Figure 2.2 presents information on reactor component system feeding liquid radwastes to the reactor coolant drain tank. Section 3.7 presents information on the flow of liquids through the reactor coolant drain tank.

3.4.6 Radionuclide Concentrations in Waste Holdup Tanks

Table 3.9 presents a summary of the radionuclide concentrations measured in the waste holdup tanks before, during, and after refueling. Data summarized in Table 3.9 are presented in Tables A.20, A.21, and A.22 of the Appendix. Figure 2.2 shows which plant components feed liquid radwastes to the waste holdup tanks. The liquid flows through the waste holdup tanks during the measurement period are presented in Section 3.7.

Samples taken from the waste holdup tanks before refueling were very different in measured radionuclide concentration, so the data from these samples were not averaged but presented separately in Table 3.9. The sample taken on 9/2/76 was from waste holdup tank WD-4A. An inspection of the auxiliary building operator's log showed that Tank WD-4A had a large inflow of water between 22:00, 8/30/76 and 02:00,

TABLE 3.8

TRITIUM BALANCE DURING REFUELING

Compartment	Volume (gal.)	^3H Concentration ($\mu Ci/ml$)	Total ^3H (Ci)
	Before Filling Cavity		
Reactor Core	34,600	3.47 ± 0.04(-2)	4
SIRWT	314,000	9.25 ± 0.04(-3)	11
Fuel Pool	215,000	7.20 ± 0.03(-3)	6
Sum of All Components	563,600	---	21
	Filled Cavity, Before Fuel Movement		
Core plus Cavity	284,100	1.14 ± 0.02(-2)	12
SIRWT	64,500	9.25 ± 0.04(-3)	2
Fuel Pool	215,000	7.32 ± 0.05(-3)	6
Sum of All Components	563,600	---	20
	Filled Cavity, After Fuel Movement		
Core, Cavity, Fuel Pool	499,100	1.1 ± 0.2(-2)	21
SIRWT	64,500	9.25 ± 0.04(-3)	2
Sum of All Components	563,600	---	23
	Drained Cavity		
Core	34,600	1.06 ± 0.02(-2)	1
SIRWT	314,000	9.91 ± 0.04(-3)	12
Fuel Pool	215,000	9.12 ± 0.04(-3)	7
Sum of All Components	563,600	---	20

TABLE 3.9

RADIONUCLIDE CONCENTRATIONS IN WASTE HOLDUP TANKS

Nuclide	Before Refueling 13:34; 8/24/76 (μCi/ml)	Before Refueling 11:40; 9/2/76 (μCi/ml)	After Refueling 10:40; 2/11/77 (μCi/ml)	During Refueling, Samples: 16:00; 10/11/76, 20:30; 10/11/76, 23:55; 10/11/76 — Mean (μCi/ml)	During Refueling — Range (μCi/ml)
Iodines:					
^{131}I	5.06 ± 0.07(-3)	1.27 ± 0.01(0)	1.20 ± 0.01(-2)	1.4(-2)	0.77-2.4(-2)
^{133}I	6.9 ± 0.2(-5)	2.29 ± 0.01(-1)	2.9 ± 0.3(-4)	1.2 ± 0.5(-5)	**
^{135}I	<2(-6)	2.7 ± 0.4(-3)	<4(-5)	<4(-5)	
Cesiums:					
^{134}Cs	1.78 ± 0.08(-3)	1.56 ± 0.05(-1)	1.08 ± 0.01(-2)	7.0(-3)	0.058-1.1(-2)
^{136}Cs	2.34 ± 0.05(-4)	4.6 ± 0.2(-2)	9.6 ± 3.8(-6)	1.2(-3)	0.64-2.1(-3)
^{137}Cs	1.86 ± 0.06(-3)	1.47 ± 0.05(-1)	1.23 ± 0.01(-2)	2.7(-4)	0.95-5.9(-2)
Tritium:					
^{3}H	*	*	5.56 ± 0.02(-2)	*	
Other Nuclides:					
^{14}C	*	*	6.89 ± 0.03(-6)	*	
^{24}Na	4.9 ± 2.4(-7)	<3(-5)	<4(-6)	<8(-6)	
^{51}Cr	<7(-4)	<5(-2)	<1(-3)	1.3(-4)	1.1-1.4(-4)
^{54}Mn	8.0 ± 1.2(-6)	1.7 ± 0.8(-4)	1.36 ± 0.03(-4)	3.9(-4)	1.2-6.5(-4)
^{55}Fe	*	*	1.43 ± 0.02(-4)	*	
^{59}Fe	<6(-7)	<2(-4)	1.8 ± 1.5(-6)	5.3(-5)	3.7-6.9(-5)
^{57}Co	<8(-7)	1.1 ± 0.4(-3)	6.9 ± 1.7(-6)	1.6(-5)	1.3-1.8(-5)
^{58}Co	6.2 ± 0.1(-5)	1.4 ± 0.1(-3)	9.30 ± 0.08(-4)	4.9(-3)	0.55-8.2(-3)
^{60}Co	6.1 ± 0.4(-6)	2.6 ± 0.7(-4)	4.6 ± 0.1(-5)	1.7(-4)	1.1-2.5(-4)
^{63}Ni	*	*	1.51 ± 0.02(-4)	*	
^{65}Zn	<6(-7)	<1(-4)	<2(-5)	<2(-5)	
^{89}Sr	*	*	4.9 ± 0.2(-5)	*	
^{90}Sr	*	*	1.13 ± 0.06(-6)	*	
^{91}Y	*	*	1.7 ± 0.1(-7)	*	
^{95}Zr	<3(-6)	<3(-4)	<2(-6)	<6(-5)	
^{95}Nb	<1(-6)	<1(-4)	<9(-6)	<4(-5)	
^{99}Mo	9.2 ± 0.7(-6)	1.9 ± 1.0(-3)	4.7 ± 3.1(-5)	5.2(-5)	4.5-5.8(-5)
^{103}Ru	<6(-5)	<2(-3)	<1(-4)	<4(-4)	
^{110m}Ag	<6(-5)	<2(-3)	1.7 ± 1.1(-6)	<4(-4)	
^{124}Sb	<4(-7)	<6(-5)	1.4 ± 0.5(-6)	5.3 ± 1.9(-6)	**
^{140}Ba	<2(-4)	<6(-3)	<4(-4)	5.7 ± 1.7(-5)	**
^{140}La	1.1 ± 0.2(-6)	6.7 ± 2.9(-5)	1.9 ± 0.1(-5)	2.9(-5)	2.8-3.0(-5)
^{141}Ce	<2(-6)	<3(-4)	<7(-6)	<2(-5)	
^{143}Ce	2.6 ± 0.8(-6)	<1(-2)	<3(-4)	<4(-4)	
^{144}Ce	<6(-6)	<1(-3)	7.3 ± 2.0(-5)	<6(-5)	
^{187}W	<4(-6)	<4(-4)	<3(-5)	8.0 ± 2.0(-5)	**
^{239}Np	<5(-4)	3.5 ± 1.2(-3)	2.3 ± 1.8(-4)	<2(-3)	

* : Analysis not performed for radionuclide.
** : Detected in one measurement, only, for this radionuclide.

8/31/76 and then did not receive any additional liquids prior to sampling on 9/2/76. Also, according to the auxiliary building operator's log, the spent resin from the CVCS deborating demineralizer, CH-9B, was sluiced to the spent resin storage tank at 02:00, 8/31/76. Normal procedure was to send the resin sluice water to the spent regenerant tanks. However, the auxiliary building operator's log showed that the resin sluice water was put in waste holdup tank WD-4A. The high cesium and iodine concentrations were representative of the sluice water from a CVCS resin which had been in contact with reactor coolant. After refueling, only one of the waste holdup tanks (WD-4A) contained any radwaste liquid. This tank was sampled once (see Table 3.9).

3.4.7 Radionuclide Concentrations in Spent Regenerant Tanks

Table 3.10 presents mean values and ranges of radionuclide concentrations for the spent regenerant tanks for the three measurement periods. The data summarized in Table 3.10 are presented in Tables A.20, A.21, A.23, and A.32 of the Appendix. Figure 2.2 shows which plant components feed liquid radwastes to the spent regenerant tanks. The liquid flows through the spent regenerant tanks during the in-plant measurements are presented in Section 3.7.

The data indicate a large variance in spent regenerant tank activity before and after refueling and also within these operational periods. Higher levels for radionuclide concentration before refueling were due to the influx of reactor coolant wastes from the deborating demineralizers being used to remove the boron shim during coastdown to refueling. Also, resins from both of the CVCS deborating demineralizers were sluiced to the spent resin storage tank during the measurements made prior to refueling. (The spent regenerant tanks, except in the case noted in Section 3.4.5, collected the spent resin sluice water.)

3.4.8 Radionuclide Concentrations in Hotel Waste Tanks

Mean values and observed ranges for radionuclide concentrations measured in the hotel waste tanks are presented in Table 3.11. The data summarized in Table 3.11 are presented in Tables A.24 and A.25 of the Appendix. The hotel waste tanks receive liquid radwastes from the laundry drains plus the personnel decontamination shower and lavatory sink drains. Section 3.7 presents the liquid radwaste flow rates for the hotel waste tanks. No samples were collected from the hotel waste tanks during refueling.

3.4.9 Radionuclide Concentrations in Monitor Tanks

As indicated in Figure 2.3, the monitor tanks at Fort Calhoun receive liquid radwaste from the hotel waste tanks and the radwaste evaporator distillate. Water from the monitor tanks is released to the discharge pipe after activity measurement to assure that release limits are not exceeded. Liquid flow rates through the monitor tanks are discussed in Section 3.7.

TABLE 3.10

RADIONUCLIDE CONCENTRATIONS IN SPENT REGENERANT TANKS

	Before Refueling		After Refueling		During Refueling
	Samples: 10:48; 08/24/76 09:42; 08/26/76 11:50; 08/31/76		Samples: 10:20; 02/11/77 09:11; 02/14/77 07:30; 2/18/77		Sample: 12:35; 10/13/76
Nuclide	Mean (μCi/ml)	Range (μCi/ml)	Mean (μCi/ml)	Range (μCi/ml)	Activity (μCi/ml)
Iodines:					
^{131}I	8.0(-3)	0.078- 1.7(-2)	5.5(-4)	0.046- 1.1(-3)	5.46 ± 0.08(-4)
^{133}I	4.3(-3)	0.014- 1.3(-2)	1.8(-4)	0.024- 3.2(-4)	<4(-4)
^{135}I	1.6(-3)	0.065- 3.0(-3)	1.6 ± 0.4(-4)	**	<8(-2)
Cesiums:					
^{134}Cs	1.4(-3)	0.33 - 2.2(-3)	6.6(-4)	0.24 - 1.5(-3)	1.92 ± 0.03(-3)
^{136}Cs	2.2(-4)	0.53 - 4.0(-4)	<3(-6)		9.2 ± 0.2(-5)
^{137}Cs	1.5(-3)	0.35 - 2.6(-3)	7.7(-4)	0.30 - 1.7(-3)	2.18 ± 0.03(-3)
Tritium:					
^{3}H	*		4.14 ± 0.02(-3)	**	*
Other Nuclides					
^{14}C	*		2.5 ± 0.1(-7)	**	*
^{24}Na	2.0(-5)	0.25 - 3.4(-5)	2.1(-5)	1.6-2.6(-5)	<4(-4)
^{51}Cr	1.7 ± 0.2(-5)	**	3.8(-5)	0.32 - 7.2(-5)	4.0 ± 0.7(-5)
^{54}Mn	6.7(-5)	0.17 - 1.2(-4)	3.0(-5)	2.7 - 3.2(-5)	3.04 ± 0.04(-4)
^{55}Fe	*		1.72 ± 0.02(-4)	**	*
^{59}Fe	2.1(-5)	0.31 - 3.8(-5)	3.6(-6)	1.4 - 7.4(-6)	1.40 ± 0.04(-5)
^{57}Co	7.5(-6)	0.12 - 1.0(-5)	2.9(-6)	2.2 - 3.4(-6)	1.39 ± 0.04(-5)
^{58}Co	3.5(-4)	2.7 - 4.8(-4)	4.6(-4)	3.4 - 5.8(-4)	3.35 ± 0.07(-3)

TABLE 3.10 (cont'd)

RADIONUCLIDE CONCENTRATIONS IN SPENT REGENERANT TANKS

Nuclide	Before Refueling Samples: 10:48; 8/24/76 09:42; 8/26/76 11:50; 8/31/76		After Refueling Samples: 10:20; 2/11/77 09:11; 2/14/77 07:30; 2/18/77		During Refueling Sample: 12:35; 10/13/76
	Mean (μCi/ml)	Range (μCi/ml)	Mean (μCi/ml)	Range (μCi/ml)	Activity (μCi/ml)
Other Nuclides: (cont'd)					
^{60}Co	2.7(-4)	0.23-6.7(-4)	2.5(-5)	1.2-3.5(-5)	5.34 ± 0.09(-4)
^{63}Ni	*		8.57 ± 0.02(-5)	**	*
^{65}Zn	1.5(-5)	0.54-2.4(-5)	<5(-7)		8.5 ± 0.8(-6)
^{89}Sr	*		5.0 ± 0.7(-7)	**	*
^{90}Sr	*		1.5 ± 0.3(-7)	**	*
^{91}Y	*		4.0 ± 2.0(-8)	**	*
^{95}Zr	2.8 ±0.6(-6)	**	4.0 ± 1.0(-6)	**	5.3 ± 0.8(-6)
^{95}Nb	3.1 ±0.4(-6)	**	<6(-6)		6.3 ± 0.4(-6)
^{99}Mo	1.2(-5)	0.62-1.9(-5)	6.5 ± 4.1(-7)	**	4.9 ± 0.9(-6)
^{103}Ru	<7(-7)		<3(-6)		<5(-5)
^{110m}Ag	5.3(-6)	2.0-8.6(-6)	2.4(-6)	0.19-5.2(-6)	4.4 ± 0.8(-6)
^{124}Sb	<2(-7)		4.4(-7)	2.4-6.5(-7)	4.8 ± 1.1(-7)
^{140}Ba	<5(-6)		<1(-5)		<2(-4)
^{140}La	5.7(-7)	4.6-6.8(-7)	<2(-5)		1.80 ± 0.04(-5)
^{141}Ce	<1(-6)		<4(-7)		<4(-6)
^{143}Ce	4.7 ±1.0(-6)	**	7.7(-6)	0.47-1.1(-5)	<3(-5)
^{144}Ce	<6(-6)		<2(-6)		<2(-5)
^{187}W	2.0 ±0.4(-4)	**	2.0 ± 1.0(-5)	**	<5(-4)
^{239}Np	3.6 ±1.8(-4)	**	5.3 ± 1.9(-6)	**	<7(-5)

* : Analysis not performed for radionuclide.
** : Detected in one measurement, only, for this radionuclide.

TABLE 3.11

RADIONUCLIDE CONCENTRATIONS IN HOTEL WASTE TANKS

Nuclide	Before Refueling: Sample: 10:43; 8/24/76 Activity (µCi/ml)	After Refueling: Samples: 13:15; 2/15/77 12:04; 2/17/77 Mean (µCi/ml)	Range (µCi/ml)
Iodines:			
^{131}I	1.2 ± 0.5(-7)	2.7 ± 0.6(-7)	**
^{133}I	<1(-7)	<2(-7)	
Cesiums:			
^{134}Cs	1.1 ± 0.1(-6)	2.2(-5)	1.0-3.5(-5)
^{136}Cs	<8(-8)	<1(-7)	
^{137}Cs	1.69 ± 0.09(-6)	2.7(-5)	1.1-4.2(-5)
Tritium:			
^{3}H	*	6.0(-6)	4.0-8.0(-6)
Other Nuclides:			
^{14}C	*	6.5(-7)	0.16-1.1(-6)
^{51}Cr	<4(-7)	<5(-7)	
^{54}Mn	3.2 ± 0.5(-7)	3.0(-7)	1.8-4.1(-7)
^{55}Fe	*	2.5(-6)	2.0-3.0(-6)
^{59}Fe	<2(-7)	<1(-7)	
^{57}Co	<1(-7)	<5(-7)	
^{58}Co	1.42 ± 0.08(-6)	2.3(-6)	2.3-2.3(-6)
^{60}Co	5.9 ± 0.7(-7)	4.6(-7)	3.9-5.3(-7)
^{63}Ni	*	1.1(-6)	0.2-2.0(-6)
^{65}Zn	<2(-7)	<2(-7)	
^{89}Sr	*	2.2(-8)	1.3-3.0(-8)
^{90}Sr	*	<8(-9)	
^{91}Y	*	<2(-8)	
^{95}Zr	<2(-7)	<3(-7)	
^{95}Nb	<8(-8)	<1(-7)	
^{99}Mo	<8(-8)	<5(-7)	
^{103}Ru	<9(-8)	<2(-7)	
^{110m}Ag	<1(-7)	<2(-7)	
^{124}Sb	<1(-7)	<2(-7)	
^{140}Ba	<4(-7)	<8(-7)	
^{140}La	<2(-7)	<2(-7)	
^{141}Ce	<2(-7)	<7(-7)	
^{143}Ce	<1(-7)	<1(-7)	
^{144}Ce	<7(-7)	<7(-6)	
^{187}W	<3(-7)	<4(-7)	
^{239}Np	<3(-7)	<3(-7)	

* : Analysis not performed for radionuclide.
** : Detected in one measurement, only, for this radionuclide.

Table 3.12 presents the mean values and ranges for radionuclide concentrations measured in the monitor tanks during the three in-plant measurement periods. The data for these tanks are presented in Tables A.2, A.21, A.24, A.24A, A.26, and A.27 in the Appendix. Several times during the measurement period, OPPD collected samples of monitor tank water at the same time our samples were obtained. Comparisons of OPPD's and our results for these samples are presented in Tables A.24A, A.26 and A.27. The comparisons were good (within 10-20% in most cases).

3.5 Auxiliary Building Ventilation System Source Terms

Previously (Section 2.3.1), the sampling stations and locations for the auxiliary building ventilation system measurements are described. The radionuclide release rates measured at sample stations 1 through 4 are presented in Table A.35 through A.38 of the Appendix. Local area sampler data are given in Tables A.39 through A.41 in the Appendix. The local areas sampled were the waste evaporator room, pipe penetration room, and the letdown heat exchanger room. The waste evaporator room was a component of sampling system 2, while the pipe penetration room and the letdown heat exchanger room were components of sampling system 1. The waste evaporator room sampler was in operation throughout the measurements at Fort Calhoun. The pipe penetration room and letdown heat exchanger room samplers were operated only near the end of the measurement period. The latter two sampler stations (Fig. 2.4b) were installed in an attempt to better define the source term in system 1. Inspection of Tables A.35-A.38 show systems 1 and 2 to be the primary ventilation system sources for ^{131}I. System 2 was not divided into additional components other than the waste evaporator room and system 4 components (Fig. 2.4b).

Corresponding sampling interval data (Table 3.13) for the pipe penetration room, the letdown heat exchanger room, and sample station 1 show the component rooms sampled were not the main source in system 1. Consequently, the major source in system 1 would have to be one of the other components in system 1 (Table 2.4). Based on discussions with OPPD personnel, it is believed the plant sampling room was the major source in system 1.

As mentioned above, the waste evaporator room and sampling system 4 were components of sampling system 2. With the exception of two sampling intervals (9/3-9/9/76 and 9/23-9/30/76, Table 3.14) sampling station 4 was the major contributor to ^{131}I released via system 2. No individual components of system 4 (Table 2.4) were sampled. However, based on discussions with OPPD personnel, it is believed that due to the frequency of charging pump leaks, the charging pump room was the primary source in this system.

TABLE 3.12

RADIONUCLIDE CONCENTRATIONS IN MONITOR TANKS

Nuclide	Before Refueling: Samples: 13:25; 8/24/76, 15:15; 8/24/76, 08:17; 9/1/76 — Mean (μCi/ml)	Before Refueling — Range (μCi/ml)	After Refueling: Samples: 09:30; 2/10/77, 08:14; 2/14/77, 08:49; 2/19/77 — Mean (μCi/ml)	After Refueling — Range (μCi/ml)	During Refueling: Sample: 07:30; 10/14/76 — Activity (μCi/ml)
Iodines:					
^{131}I	1.6(-5)	1.2-2.2(-5)	4.2(-7)	2.5-5.2(-7)	6.4 ± 0.3(-7)
^{133}I	5.5(-7)	1.8-8.9(-7)	<2(-7)		<4(-7)
Cesiums:					
^{134}Cs	2.6(-6)	1.1-3.7(-6)	4.8(-6)	3.3-7.0(-6)	3.8 ± 0.2(-6)
^{136}Cs	3.9(-7)	1.0-7.1(-7)	<2(-7)		2.5 ± 0.6(-7)
^{137}Cs	3.0(-6)	1.4-4.1(-6)	5.9(-6)	4.1-8.5(-6)	4.59 ± 0.07(-6)
Tritium:					
^{3}H	3.75 ± 0.02(-3)	**	5.36 ± 0.02(-3)	**	*
Other Nuclides:					
^{14}C	2.6 ± 0.2(-7)	**	3.5 ± 0.1(-7)	**	*
^{32}P	<4(-8)		*		*
^{35}S	<2(-7)		*		*
^{45}Ca	1.79 ± 0.02(-5)	**	*		*
^{51}Cr	<2(-6)		<7(-7)		<6(-7)
^{54}Mn	5.6(-7)	2.7-8.5(-7)	2.4(-7)	2.0-2.7(-7)	4.2 ± 0.3(-7)
^{55}Fe	5.2 ± 0.1(-6)	**	5.0 ± 1.0(-7)	**	*
^{59}Fe	<1(-7)		<2(-7)		<2(-7)
^{57}Co	<2(-7)		<4(-8)		8.6 ± 1.9(-8)
^{58}Co	3.1(-6)	0.88-5.0(-6)	2.2(-6)	1.5-2.6(-6)	4.7 ± 0.1(-6)
^{60}Co	7.2(-7)	5.9-8.2(-7)	3.8(-7)	3.6-3.9(-7)	1.05 ± 0.04(-6)
^{63}Ni	3.6 ± 0.1(-7)	**	4.3 ± 0.1(-7)	**	*
^{65}Zn	<1(-7)		<2(-7)		<2(-7)

TABLE 3.12 (con't'd)

RADIONUCLIDE CONCENTRATIONS IN MONITOR TANKS

	Before Refueling Samples: 13:25; 8/24/76; 15:15; 8/24/76; 08:17; 9/1/76		After Refueling: Samples: 09:30; 2/10/77; 08:14; 2/14/77; 08:49; 2/19/77		During Refueling: Sample: 07:30; 10/14/76
Nuclide	Mean (μCi/ml)	Range (μCi/ml)	Mean (μCi/ml)	Range (μCi/ml)	Activity (μCi/ml)
Other Nuclides: (cont'd)					
^{89}Sr	3.0 ± 1.0(-8)	**	<2(-8)		*
^{90}Sr	<2(-8)		<1(-8)		*
^{91}Y	<1(-8)		<2(-8)		*
^{95}Zr	<1(-7)		<3(-7)		<6(-7)
^{95}Nb	<5(-8)		<2(-7)		<3(-7)
^{99}Mo	<2(-7)		1.7 ± 1.6(-8)	***	<3(-8)
^{103}Ru	<1(-7)		<2(-7)		<2(-7)
^{110m}Ag	<9(-8)		<2(-7)		<2(-7)
^{124}Sb	<9(-8)		<2(-7)		<2(-7)
^{140}Ba	<4(-7)		<6(-7)		<7(-7)
^{140}La	<8(-7)		9.2 ± 4.0(-7)	***	<3(-7)
^{141}Ce	<3(-7)		<6(-8)		<7(-8)
^{143}Ce	<4(-7)		<2(-7)		<2(-7)
^{144}Ce	1.2 ± 0.4(-6)	***	<3(-7)		<3(-7)
^{147}Pm	<2(-8)		*		*
^{187}W	<2(-7)		<4(-7)		<1(-6)
^{239}Np	<1(-6)		<5(-7)		<4(-7)

* : Analysis not done for this radionuclide.
** : One measurement, only, for this radionuclide.
***: Detected in one measurement, only, for this radionuclide.

TABLE 3.13

VENTILATION AIRBORNE ACTIVITIES (μCi/sec)

Sample Period	Sample Period One: Pipe Penetration Room 2/3 - 2/10/77	Sample Period One: Letdown Heat Exchanger Room 2/3 - 2/11/77	Sample Period One: Sample Station #1 2/3 - 2/10/77	Sample Period Two: Pipe Penetration Room 2/10 - 2/11/77	Sample Period Two: Letdown Heat Exchanger Room 2/11 - 2/17/77	Sample Period Two: Sample Station #1 2/10 - 2/17/77
^{131}I	7.9 ± 0.8(-6)	6.8 ± 0.4(-6)	1.76 ± 0.07(-4)	1.4 ± 0.6(-6)	3.0 ± 0.5(-6)	1.9 ± 0.1(-4)
^{134}Cs	<1.2(-7)	1.0 ± 0.4(-7)	<2.4(-6)	<1.3(-6)	<1.3(-7)	<8.3(-6)
^{136}Cs	<9.9(-8)	<1.4(-7)	<1.6(-6)	<2.6(-7)	<9.9(-8)	<1.5(-6)
^{137}Cs	<1.2(-7)	2.7 ± 0.5(-7)	2.6 ± 0.7(-6)	<1.2(-6)	<3.2(-7)	<1.7(-6)
^{3}H	*	*	[A]	*	*	[A]
^{51}Cr	<9.9(-7)	<1.4(-6)	<2.4(-5)	<4.6(-6)	<7.0(-7)	<5.2(-6)
^{54}Mn	<9.9(-8)	<1.4(-7)	<1.6(-6)	<4.2(-7)	<1.3(-7)	<6.8(-7)
^{59}Fe	<9.9(-7)	<3.5(-7)	5.0 ± 2.0(-6)	<7.5(-6)	<3.0(-7)	<3.0(-5)
^{57}Co	<1.6(-7)	<3.5(-8)	<7.2(-7)	<3.0(-7)	<6.4(-8)	<1.4(-7)
^{58}Co	<9.9(-8)	2.5 ± 0.6(-7)	<1.6(-6)	<3.2(-7)	<2.0(-7)	<2.4(-6)
^{60}Co	1.0 ± 0.2(-6)	2.7 ± 0.6(-7)	6.0 ± 2.0(-6)	<5.0(-7)	<1.3(-7)	<1.1(-6)
^{65}Zn	<7.9(-7)	<2.8(-7)	<8.0(-6)	<5.2(-6)	<2.0(-7)	<2.1(-5)
^{95}Zr	<2.0(-7)	<2.8(-7)	<3.2(-6)	<3.4(-7)	<1.3(-7)	<1.3(-5)
^{95}Nb	<9.9(-8)	<1.4(-7)	<1.6(-6)	<2.0(-7)	<1.3(-7)	<7.9(-7)
^{103}Ru	<1.2(-7)	<1.4(-7)	<1.6(-6)	<3.0(-7)	<1.2(-7)	<1.0(-6)
^{106}Ru	<1.2(-6)	<1.4(-6)	<1.6(-5)	<1.1(-6)	<1.3(-6)	<1.3(-5)
^{110m}Ag	<1.2(-7)	<1.4(-7)	<1.6(-6)	<2.8(-6)	<2.5(-7)	<1.2(-5)
^{124}Sb	<1.2(-7)	<1.4(-7)	<1.6(-6)	<7.3(-7)	<1.4(-7)	<1.4(-6)
^{125}Sb	<4.0(-7)	<2.8(-7)	<4.0(-6)	<5.2(-7)	<2.8(-7)	<2.4(-6)
^{140}Ba	<5.9(-7)	<3.5(-7)	<5.6(-6)	<5.7(-7)	<4.8(-7)	<2.6(-6)
^{140}La	<5.9(-7)	<5.7(-7)	<1.6(-5)	<5.0(-5)	<1.3(-5)	<1.7(-5)
^{141}Ce	<4.0(-7)	<6.4(-8)	<1.6(-6)	<8.7(-7)	<2.5(-7)	<7.9(-7)
^{152}Eu	<4.0(-7)	<4.2(-7)	<7.2(-6)	<1.5(-6)	<3.2(-7)	<4.9(-6)
^{154}Eu	<1.2(-6)	<4.2(-7)	<1.6(-5)	<8.3(-6)	<3.3(-7)	<3.3(-5)

* Not Measured

[A] SAMPLE STATION #1; ^{14}C-^{3}H Shutdown due to electrical circuit overload.

TABLE 3.14

VENTILATION ^{131}I AIRBORNE ACTIVITIES (μCi/sec)

Date	Waste Evaporator Room	Sampling Station #4	Sampling Station #2
9/3 - 9/9/76	1.63 ± 0.01(-4)	<4.5(-6)	6.13 ± 0.06(-4)
9/9 - 9/15/76	4.70 ± 0.05(-5)	1.6 ± 0.1(-4)	1.92 ± 0.06(-4)
9/15 - 9/23/76	4.80 ± 0.04(-5)	8.4 ± 0.7(-5)	6.0 ± 0.2(-4)[A]
9/23 - 9/30/76	1.92 ± 0.03(-5)	<3.2(-6)	8.3 ± 0.4(-5)
9/30 - 10/7/76	1.55 ± 0.03(-5)	2.1 ± 0.4(-3)	1.43 ± 0.01(-3)
10/7 - 10/14/76	1.91 ± 0.02(-5)	2.3 ± 0.1(-4)	3.80 ± 0.02(-4)
10/14 - 10/22/76	3.21 ± 0.06(-5)	7.3 ± 0.7(-5)	3.4 ± 0.3(-4)
10/22 - 10/28/76	4.37 ± 0.07(-5)[B]	7.6 ± 0.6(-5)	1.66 ± 0.01(-4)
10/28 - 11/4/76	[C]	7.8 ± 0.7(-5)	1.57 ± 0.01(-4)
11/4 - 11/10/76	1.51 ± 0.04(-5)	8.0 ± 1.0(-5)	5.27 ± 0.04(-5)
11/11 - 11/23/76	1.22 ± 0.01(-5)	8.0 ± 1.0(-5)	6.47 ± 0.04(-5)
11/23 - 11/30/76	6.0 ± 0.1(-6)	8.0 ± 0.6(-5)	1.75 ± 0.01(-3)
11/30 - 12/15/76	6.0 ± 2.0(-6)	1.8 ± 0.2(-4)	1.06 ± 0.09(-4)[D]
12/15/76 - 1/6/77	8.0 ± 2.0(-7)	7.8 ± 0.5(-5)	6.6 ± 0.4(-5)
1/6 - 1/20/77	<1.1(-7)	6.5 ± 0.7(-5)	5.0 ± 0.4(-5)
1/20 - 2/3/77	2.5 ± 0.4(-6)	2.3 ± 0.1(-4)	1.8 ± 0.2(-4)
2/3 - 2/16/77	1.03 ± 0.07(-6)	5.0 ± 2.0(-6)	2.1 ± 0.2(-5)

[A] Sample from 9/15 - 9/18/76.

[B] Sample from 10/22 - 10/27/76.

[C] No sample due to sampler motor malfunction.

[D] Sample from 12/1 - 12/15/76.

It should be pointed out that the total releases from the waste evaporator room and sample station 4 were at times greater than the release rate of station 2. This is attributed to the before mentioned indeterminate errors (estimated to be 20% overall) associated with varying duct flows.

3.6 Gaseous Iodine Species

3.6.1 Auxiliary Building Ventilation Iodine Species

The iodine species distribution in the Auxiliary Building ventilation air was variable (Tables A.42-A.44 in Appendix). The average percent values are shown in Table 3.15 along with their ranges. The usefullness of averaging these data is questionable when the wide range of values are considered. Furthermore, there is no obvious relationship between species and the environment in which the measurements were conducted, i.e., high temperature and humidity of the waste evaporator room versus lower humidity and temperature of sampling system 2 components.

3.6.2 Waste Gas Processing System-Chemical Species of Radioiodine

In the waste gas processing system (i.e., waste gas decay tanks and various cover gases), the predominate iodine species was found to be organic iodine. One explanation for this is that the radiation fields in these gas volumes were high and could cause the formation of free radicals. Any organic free radical, from hydrocarbon impurities in the gas, could form organic iodine. Secondly, elemental iodine has a much greater tendency to plate out on surfaces than does organic iodine. The remaining species in the gas phase would therefore be enriched in organic iodine. This latter phenomenon causes the observation that the older a sample containing gaseous iodine is, the more likely it is to be highly organic. As shown in Table A.46, when the short-lived iodine-135 nuclide was observed, iodine was less converted to organic iodine than when ^{135}I was not observed.

3.7 Liquid Waste Flow Rates

At Fort Calhoun, information concerning specific sources of liquid radwaste flow was not collected by plant personnel. However, there were data from the auxiliary building operator's log which presented liquid radwaste tank levels on a bi-hourly basis. From the log, average daily flow rates into the tanks were calculated for the three in-plant measurement periods. Table 3.16 presents the average flow rates into the liquid radwaste tanks together with the source of radwaste liquids for each tank. Graphs of liquid radwaste tank levels are presented in the Appendix (Figures A.17, A.18, A.19, A.20, A.21, and A.22).

3.8 Detergent Wastes

At Fort Calhoun, detergent wastes are collected in the hotel waste tanks. Table 3.16 presents the average daily flow rates for the hotel waste tanks. The radionuclide concentrations in the hotel waste tanks

TABLE 3.15

AVERAGE FRACTIONAL PERCENTAGE FOR ^{131}I SPECIES

Station	Number of Samples	Particulate Filter Average (%)	Particulate Filter Range (%)	I_2 Average (%)	I_2 Range (%)	HoI Average (%)	HoI Range (%)	Organic Average (%)	Organic Range (%)
#2	17	6.6	(0-25.0)	33.3	(6.6-61.3)	15.9	(1.1-41.9)	44.4	(23.3-68.2)
Waste Evaporator	15	12.1	(0-33.5)	38.7	(1.2-75.0)	17.5	(0-80.2)	31.6	(8.1-56.0)

TABLE 3.16

LIQUID WASTE FLOW RATES
Into Radwaste Tanks

Tank	Source of Liquid*	Flow Rates (Gal/day) Before Refueling 8/2/-9/3/76	During Refueling 10/1-16/76	After Refueling 1/31-2/25/77
Reactor Coolant Drain Tank	Reactor Coolant Leakage Fuel Pool Drains	**	250	500
Waste Holdup Tanks	Reactor Coolant Drain Tank Aux. Bld. Equip. Drains CVCS Bleed VCT Relief and Drains	2500	6800	690
Spent Regenerant Tanks	Aux. Bld. Floor Drains Secondary Side Drains Spent Resin Sluice Water Deborating Demins Containment Sump	3800	2400	2500
Hotel Waste Tanks	Laundry Drains Shower and Hand Sink Drains	1100	2500	570
Monitor Tanks	Evaporator Condensate Hotel Tanks	7000	4600	3600

* : See Liquid Radwaste Schematics, Figure 2.2 and 2.3.

** : Information not collected.

are presented in Table 3.11. Liquid radwastes from equipment, drum, and cask decontaminations fed into the drain headers in the auxiliary building and were collected by the spent regenerant tanks.

3.9 Chemical Wastes from Regeneration of Condensate Demineralizers

There are no demineralizers in the secondary system for processing the condensate. The steam generator blowdown waters are normally released to the discharge pipe without treatment. Also, the normal practice was to dispose of all demineralizer spent resins as radioactive solid waste.

3.10 Containment Purge Frequency

The containment purge frequency at Fort Calhoun via high volume purging and venting is 37 times annually; however, the approved plant procedures for containment purge releases do not differentiate between containment purging and containment venting. (8)

3.11 Containment Internal Cleanup System

Due to limited access to the containment building, measurements were not made for evaluation of the containment cleanup system.

In the Fort Calhoun containment building there are two banks of charcoal adsorbers with associated HEPA filters. The design flow through each unit is 90,000 cubic feet per minute which results in 5.4 containment volume changes per hour. These units were operated prior to entry into containment of maintenance personnel and/or when high volume purges were done.

3.12 Gaseous Leakage Rate to Containment Building

Two samples (Table A.51) of the containment building atmosphere were obtained. One was taken during the refueling outage (Oct. 1976) approximately 10 days after purging of the containment atmosphere. The second was collected during power operations in February, 1977. The former was taken from a plant installed containment penetration used for pressure testing of the containment building. The latter sample was taken inside the containment structure, consequently, it is the more representative sample. During October, the measurement utilized only an iodine species sampler, while during February an iodine species and ^{14}C and ^{3}H samplers were used. The additional tritium data available for the February measurement allowed the leakage rate to be calculated on the basis of both iodine and tritium.

The results were calculated using the following formulas (3):

$$(1) \quad C_g \times V_c \times K = T_c$$

$$(2) \quad T_c = \frac{T_{1/2} \times D'}{\ln 2}$$

$$(3)\quad D' \times 100\% = D \times C_\ell \times K \times V_\ell$$

where:

C_g = Airborne concentration of nuclide in containment - μCi/cc

V_c = Free volume of containment - ft^3

K = Conversion factor = 2.83×10^4 cc/ft^3

T_c = Total activity in containment atmosphere - μCi

$T_{1/2}$ = Half-life of nuclide - days

D' = Leakage - μCi/day

C_ℓ = Concentration of nuclide in primary coolant - μCi/cc

V_ℓ = Volume of primary coolant - ft^3

D = Percent of nuclide inventory in primary coolant release per day.

Iodine-131 data from the October measurement indicate a daily leakage rate of the iodine inventory in the reactor coolant to the containment atmosphere of 1.6×10^{-4}%/day. Leakage rates calculated using ^{131}I and ^{133}I data from the February measurement averaged 1.35×10^{-6}%/day. The corresponding leakage rate based on tritium (HTO) was 2.5×10^{-7}%/day. The higher leakage rate for the October sample was based on data taken after the reactor had been down for refueling for several days and the containment was secured and was in the process of being pressure tested. Therefore, the mechanism for release of iodine into the containment atmosphere could be drastically different than during the February measurement period when the reactor was at full power.

Data from two other pressurized water reactors (6) indicate a containment leakage rate of 0.001%/day. The best estimate of the present work for leakage into the containment building is on the order of 1×10^{-6} %/day.

3.13 Auxiliary Building Gaseous Leakage

Leakage of primary coolant contributing to the source term of the auxiliary building was calculated for both ^{131}I and tritium (both oxidized and nonoxidized species). One set of gaseous ^{131}I and tritium values used for the calculations was taken from ventilation sampling stations 1, 2 and 3 for the sampling interval 9/3-9/9/76 (Tables A.35-A.37).

A second set of gaseous ^{131}I and tritium values was taken from the same ventilation sampling stations for the period 1/6-1/20/77 (Tables A.35-A.37 in the Appendix). Ventilation stations 1, 2, and 3 account for 80 percent of the total auxiliary building ventilation exhaust flow and in excess of 80 percent of the total activity exhausting the auxiliary building. Consequently, the calculated numbers could underestimate the actual leakage by as much as 10 percent. The basis for the latter statement is that approximately one half of the balance of the flow comes from an area outside the personnel monitoring control point, and has a lower activity. This can be seen from Figure 2.4 and Table 2.3 plus Figures A.28 and A.29 in the Appendix.

Results calculated using ^{131}I and ^{3}H values from the first set of data indicate primary coolant leakage rates of 570 lbs/day and 36 lbs/day, respectively. The ^{131}I and ^{3}H data obtained in February yield leakage rates of 85 lbs/day and 77 lbs/day, respectively. In the iodine calculations an iodine partition factor of 0.0075 (5% volatile iodine in primary coolant with a partition coefficient of 0.15 (6)) was used. A partition factor of 1.0 was used for tritium. The formulas used in the calculations were:

(1) $C_g \times 60 \text{ sec/min} \times 1440 \text{ min/day} = M$

(2) $\dfrac{M}{C_\ell \times 454 \text{ g/lb} \times P} = L$

where:

C_g = airborne nuclide activity per time - μCi/sec

M = activity per day - μCi/day

C_ℓ = nuclide activity per unit mass - μCi/g

P = nuclide partition factor - dimensionless

L = primary coolant leakage rate - lbs/day.

The ^{131}I value from the first set of data is obviously inconsistent with the other three data results. An explanation of this inconsistency is the unpredictive chemical behavior of iodine, i.e., partition coefficient. Therefore, it is believed that tritium yields a more valid estimate of the primary coolant leakage rate into the auxiliary building.

3.14 Particulate Releases for Gaseous Effluents

3.14.1 Containment Building Radioactive Particulate Releases for Gaseous Effluents

Table 3.17 shows the extrapolated annual radioactive particulate releases for gaseous effluents from the containment structure.

The containment building calculations are based on information stated earlier, i.e., 37 high volume purges annually with ventings being treated as high volume purges. Furthermore, it was assumed that duration of a high volume purge is long enough to reduce the activity to an insignificant level. A DF of 70 (see Sect. 3.18) was used for the containment building exhaust filters. The containment free volume is 1.05 million cubic feet. The data (Table A.51) were handled as shown below:

$$(1)\quad C_c \times V_A \times \frac{1}{(DF)} \times \frac{10^{-6}\,Ci}{\mu Ci} = R_A$$

$$(2)\quad C_v \times P_f \times \frac{18.7\ psia}{14.7\ psia} \times 28320\ cc/ft^3 = V_A$$

where:

C_c = nuclide particulate concentration - μCi/cc

V_A = total volume of containment atmosphere released - cc/year

(DF) = decontamination factor

R_A = nuclide released - Ci/year

C_v = containment free volume - ft^3

P_f = purging frequency - $year^{-1}$

3.14.2 Auxiliary Building Particulate Releases for Gaseous Effluents

Table 3.18 presents the extrapolated annual radioactive particulate releases for gaseous effluents for the auxiliary building. The data used in the calculations were obtained from sampling stations 1, 2 and 3 (Tables A.35-A.37). For each station, only positive values, (i.e., no lower detection limits) were used to obtain an average μCi/sec release rate. The average release rate was multiplied by the number of seconds per year (3.15 (+7) seconds) to obtain an annual release rate before filtration. A DF of 70 (see Section 3.18) was used to obtain the values in Table 3.18.

TABLE 3.17

EXTRAPOLATED ANNUAL CONTAINMENT BUILDING PARTICULATE RELEASES FOR GASEOUS EFFLUENTS

Nuclide	Ci/year[A]
^{134}Cs	4.6(-8)
^{137}Cs	5.8(-8)
^{58}Co	8.0(-10)
^{60}Co	5.4(-10)
^{54}Mn	2.0(-10)
^{59}Fe	*

* - Not observed.

[A] A DF of 70 was assumed for HEPA filters.

TABLE 3.18

EXTRAPOLATED ANNUAL AUXILIARY BUILDING PARTICULATE RELEASES FOR GASEOUS EFFLUENTS

Nuclide	Average Release Rate Sum of Stations 1, 2, 3 (μCi/sec)	Extrapolated Annual Releases[A] (Ci/year)
^{134}Cs	5.1(-5)	2.3(-5)
^{137}Cs	5.8(-5)	2.6(-5)
^{58}Co	6.5(-5)	2.9(-5)
^{60}Co	8.6(-6)	3.9(-6)
^{54}Mn	*	
^{59}Fe	*	

* - Not observed.

[A] A DF of 70 was assumed for HEPA filters.

3.14.3 Gas Decay Tank Radioactive Particulate Releases for Gaseous Effluents

Table 3.19 shows the extrapolated annual radioactive releases via the waste gas processing system. The releases via this pathway were calculated for 36 waste gas decay tank releases per year at a maximum achievable discharge pressure of 86 psig and a minimum initial pressure of 10 psig in the gas decay tanks (8). The decay tank volumes are 400 cubic feet. The data in Table 3.19 do not assume a DF for any filter system. The calculations are based on a sample collected from waste gas decay tank "A" on 10/13/76. The tank had been isolated for decay on 10/4/76. The particulate radionuclides observed and their concentrations were $^{59}Fe(1.1 \times 10^{-9})$, $^{60}Co(2.3 \times 10^{-9})$, and $^{134}Cs(7.2 \times 10^{-10})$ microcuries per cubic centimeter. The formula used in the waste gas release calculations was:

$$R_{WG} = C_T \times T_v \times R_f \times \frac{100.7}{24.7} \times \frac{28320\ \text{cc}}{\text{ft}} \times 10^{-6}\ \text{Ci}/\mu\text{Ci}$$

where:

C_T = nuclide particulate concentration in tank - μCi/cc

T_v = tank volume - ft^3

R_f = release frequency - $year^{-1}$

R_{WG} = annual nuclide release - Ci/year

TABLE 3.19

EXTRAPOLATED ANNUAL WASTE GAS DECAY TANK PARTICULATE RELEASES FOR GASEOUS EFFLUENTS

Nuclide	Ci/year[A]
^{134}Cs	1.2(-6)
^{137}Cs	*
^{58}Co	3.8(-6)
^{60}Co	*
^{59}Fe	1.8(-6)

* - Not observed.

[A] Assumes no DF for any filters.

3.15 Tritium and Carbon-14 Releases

3.15.1 Liquid Tritium Releases

As indicated in Figures 2.2 and 2.3, except for the steam generator blowdown waters, all liquid radwastes including secondary side drains at Fort Calhoun were released via the monitor tanks. Using the data from Tables 3.12 and 3.16, the mean liquid tritium releases from the monitor tanks were estimated to be 9.8(-2) Ci/day before refueling and 7.2(-2) Ci/day after refueling which averages to 8.5(-2) Ci/day. Using the data from Table 3.5 plus Sections 3.3 and 3.4.2, the mean liquid tritium releases by the steam generator blowdowns were estimated to be 2.1(-3) Ci/day. This sums to 8.7(-2) Ci/day which translates to 3.2(1) Ci/year for the liquid tritium releases.

3.15.2 Liquid Carbon-14 Releases

At the Fort Calhoun Station, all liquid radwastes except for steam generator blowdown waters, were released via the monitor tanks. Using the data from Table 3.16 and the Appendix (Tables A.2 and A.26), the mean liquid ^{14}C releases were estimated to be 6.9(-6) Ci/day before refueling and 4.8(-6) Ci/day after refueling. This averages to 5.9(-6) Ci/day which translates to 2.2(-3) Ci/year of liquid carbon-14 releases. The method used for analyzing these samples measured only inorganic ^{14}C (CO_2), therefore, the results are potentially biased low by the degree that the sample contained any organic ^{14}C.

3.15.3 Containment Building Gaseous Tritium and Carbon-14 Releases

The tritium and ^{14}C release calculation was handled in the same manner as described for particulate releases, i.e., same volumes, purging frequencies, and pressures. Data were taken from Table A.51. Both oxidized and nonoxidized chemical species were used in the tritium analysis. However, due to a sample analysis problem, only the oxidized ^{14}C species was used. Consequently, the ^{14}C containment building releases could be as much as a factor of 10 higher. The results of the calculations are shown in Table 3.20.

3.15.4 Auxiliary Building Gaseous Tritium and Carbon-14 Releases

The annual tritium and ^{14}C release (both oxidized and non-oxidized species) via the vapor pathway from the auxiliary building ventilation system are given in Table 3.20. The data were obtained by sampling stations 1, 2, and 3 (Tables A.35-A.37). It should be noted that sampling station 2 includes the fuel pool area. Consequently, the calculated releases include those related to fuel movement during the refueling outage. The average total release from sampling stations 1, 2 and 3 were multiplied by the number of seconds in a year (3.15 (+7) seconds) to obtain an annual release rate.

TABLE 3.20

EXTRAPOLATED ANNUAL RELEASES OF GASEOUS TRITIUM AND CARBON-14

Ci/year

	Containment Building	Auxiliary Building	Waste Gas Processing System
^{3}H	5.9(-2)	0.9	0.65
^{14}C	7.8(-2)[A]	0.3	0.81

[A] Oxidized species only

3.15.5 Gas Decay Tank Tritium and Carbon-14 Gaseous Releases

Both oxidized and nonoxidized chemical forms of tritium and ^{14}C were included in the waste gas decay tank annual release calculations. The annual releases presented in Table 3.20) are based on average concentrations in the decay tanks (Tables A.49 and A.50 of Appendix). The "less than" (lower detection limit) value for tritium in Table A.50 was not included.

The results were obtained in the same manner as described in Section 3.14.3 above, i.e., the same release frequencies, volumes, and pressures.

3.16 Decontamination Factors for Demineralizers

At Fort Calhoun the demineralizers present in the liquid process systems were the purification, cation, and deborating demineralizers of the CVCS and the waste or polishing demineralizers which were a side stream in the recirculation line for the monitor tanks. Normal plant operating practice was not to use the waste demineralizers, which was continued during the in-plant measurements. According to station personnel, the waste disposal problems associated with the spent resins from the waste demineralizers (i.e., manual removal of the resins) override the usefulness of processing the liquid radwastes from the monitor tanks through the waste demineralizers. This was particularly true since the activity in the monitor tanks was well within the release limits for the station.

The purification demineralizers in the CVCS letdown stream are of the mixed bed type. Table 3.21 presents the input activities and the decontamination factors (DF's) measured for several radionuclides during the August, 1976 (before refueling) and February, 1977 (after refueling) measurement periods. In a case where the input activity contained two components (dissolved and suspended activities), the two components were added and the one standard deviation counting statistics errors were propagated. If one component was not measurable (i.e., only a lower limit of detection was obtained), the quoted input activity contains only the measurable concentration. The error was adjusted to reflect the fact that a second component (of unknown magnitude and possibly as large as the lower limit of detection) could exist by propagating the error in the measurable concentration together with an error equal to one-half the lower limit of detection (lower limit of detection values are at the two sigma level of counting statistics). The data used to calculate the DF's are presented in Tables A.8, A.9, and A.12. Also presented in Table 3.21 are when the demineralizer was placed in service, how long it was in use, and bed volumes passed through before measurement, together with the letdown flow rate, demineralizer pressure drop, and reactor coolant boron level at the time of measurement. These parameters were extracted from Figures A.5, A.6, A.7, A.8, A.9, and A.10.

For the purification demineralizers, the DF's are defined as the ratios of the input to the output radionuclide concentrations. The errors

TABLE 3.21

DECONTAMINATION FACTORS

CVCS PURIFICATION DEMINERALIZER

	Demineralizer CH-8B 10:57; 8/20/76 Placed in Service: 3/24/76 Used for: 93 days Bed Volumes Thru: 3.1(4) Letdown Flow Rate:28.6 gpm,4.2 gpm/ft^2 Demin Pressure: 1.8 psi Reactor Coolant Boron: 112 ppm		Demineralizer CH-8B 10:00; 8/30/76 Placed in Service: 3/24/76 Used for: 103 days Bed Volumes Thru: 3.5(4) Letdown Flow Rate:27.4 gpm,4.0 gpm/ft^2 Demin Pressure: 2.0 psi Reactor Coolant Boron: 58 ppm		Demineralizer CH-8A 10:20; 2/16/77 Placed in Service: 1/13/77 Used for: 34 days Bed Volumes Thru: 1.1(4) Letdown Flow Rate:30.6gpm,4.5 gpm/ft^2 Demin Pressure: 1.4 psi Reactor Coolant Boron: 449 ppm	
Nuclide	Input Activity (μCi/ml)	Decontamination Factor	Input Activity (μCi/ml)	Decontamination Factor	Input Activity (μCi/ml)	Decontamination Factor
Activation:						
^{24}Na	3.4 ± 0.2(-3)	13. ± 11.	2.2 ± 0.5(-3)	6.1 ± 2.6	1.84 ± 0.05(-2)	59. ± 12.
^{54}Mn	6.3 ± 0.3(-4)	1.4 ± 0.1	5.2 ± 0.4(-4)	22. ± 51.	1.8 ± 0.3(-4)	14. ± 4.
^{58}Co	3.44 ± 0.06(-3)	8.6 ± 0.3	2.05 ± 0.09(-3)	>49.	1.87 ± 0.04(-3)	10.1 ± 0.3
^{60}Co	6.3 ± 1.9(-5)	2.6 ± 0.9	8.7 ± 3.1(-5)	2.9 ± 2.4	2.6 ± 15.0(-5)	0.9 ± 5.1
Iodines:						
^{131}I	2.41 ± 0.02(-1)	13. ± 1.	2.40 ± 0.02(-1)	20. ± 1.	9.0 ± 0.1(-2)	1730. ± 1460
^{132}I	7.1 ± 0.1(-2)	14. ± 3.	1.01 ± 0.01(-1)	51. ± 11.	5.0 ± 0.2(-2)	>16.
^{133}I	1.20 ± 0.01(-1)	16. ± 1.	2.59 ± 0.02(-1)	25. ± 1.	1.04 ± 0.02(-1)	>170.
^{135}I	7.3 ± 0.2(-2)	7.4 ± 1.8	1.30 ± 0.04(-1)	>12.	6.9 ± 0.2(-2)	>23.
Other Fission Products:						
^{88}Rb	4.9 ± 0.2(-1)	2.7 ± 1.1	9.5 ± 0.6(-1)	2.6 ± 1.0	7.2 ± 0.4(-1)	2.3 ± 0.2
^{134}Cs	1.54 ± 0.01(-2)	13. ± 1.	1.37 ± 0.06(-3)	3.8 ± 0.3	1.91 ± 0.02(-1)	61.6 ± 0.9
^{136}Cs	4.10 ± 0.09(-3)	14. ± 1.	6.2 ± 1.5(-4)	5.2 ± 1.5	1.4 ± 0.5(-4)	10. ± 4.
^{137}Cs	1.56 ± 0.01(-2)	13. ± 1.	1.46 ± 0.04(-3)	3.8 ± 0.3	2.20 ± 0.02(-1)	58.0 0.6
^{138}Cs	2.9 ± 0.1(-1)	29. ± 41.	5.6 ± 0.2(-1)	11. ± 5.	2.2 ± 0.1(-1)	>110.
^{140}Ba	9.0 ± 1.7(-4)	5.3 ± 1.4	<3(-4)		<2(-2)	
^{140}La	2.6 ± 0.2(-3)	1.3 ± 0.2	1.3 ± 0.5(-3)	>8.	2.7 ± 2.2(-2)	>140.

on the DF's are the propagation of the one standard deviation counting statistics errors on the measured radionuclide concentrations for the input and output. An additional propagated error of 14 percent should be added to the stated errors in the DF's to account for other uncertainties (e.g., uncertainties in calibration, decay correction, volume measurement, random summing correction). Sampling uncertainties are unknown and therefore cannot be estimated. When a DF is expressed as a "greater than" number this means that the radionuclide was measurable in the demineralizer input but not in the output and the output lower limit of detection and input values have been used to calculate the DF.

Except for ^{88}Rb, the measured decontamination factors were higher for demineralizer CH-8A in February, 1977 than for demineralizer CH-8B in August, 1976. The fact that demineralizer CH-8A was newer (used for only 34 days and 1.1(4) bed volumes) than demineralizer CH-8B (used for 93-103 days and 3.1(4)-3.5(4) bed volumes) may account for the higher decontamination factor.

3.17 Decontamination Factors for Liquid Stream Filters

At Fort Calhoun there were liquid stream filters in the CVCS letdown stream, purification filters (CH-17A and B), and in the radwaste process stream, waste filters (WD-17A and B). Sample points were available for investigating the purification filters but not the waste filters. (Table 2.1 presents the physical information on the purification filters.) Table 3.22 presents the radionuclide decontamination factors (DF's) for the purification filters measured before refueling (August, 1976) and after refueling (February, 1977). The data used to calculate the DF's in Table 3.22 are found in Tables A.8, A.9, and A.12 of the Appendix. Also presented in Table 3.22 are the letdown flow rates, filter pressure drops, and reactor coolant boron levels at the time of measurement. These parameters were extracted from Figures A.6, A.7, A.8, A.9, and A.10 of the Appendix. For the purification filters, the DF's are defined as the ratios of the input to the output radionuclide concentrations. The errors shown were obtained in the same manner as discussed in Section 3.16.

As shown in Table 3.22, there is a large variability in the measured decontamination factors for all the radionuclides except ^{88}Rb which has a measured DF of about 1.0. For the measurement of 2/16/77, many of the radionuclides had a DF of less than one which means that radioactivity was being eluted from the filter by the letdown stream. The purification filters are designed to trap loose resin beads from the demineralizers and are not intended for radioactive clean-up, so the lack of consistent DF's is not surprising.

TABLE 3.22

DECONTAMINATION FACTORS
CVCS PURIFICATION FILTERS

	Filter CH-17B 10:57; 8/20/76 Letdown Flow Rate: 28.6 gpm Filter Pressure: 1.0 psi Reactor Coolant Boron: 112 ppm		Filter CH-17B 10:00; 8/30/76 Letdown Flow Rate: 27.4 gpm Filter Pressure: 0.5 psi Reactor Coolant Boron: 58 ppm		Filter CH-17A 10:20; 2/16/77 Letdown Flow Rate: 30.6 gpm Filter Pressure: 1.0 psi Reactor Coolant Boron: 449 ppm	
Nuclide	Input Activity (μCi/ml)	Decontamination Factor	Input Activity (μCi/ml)	Decontamination Factor	Input Activity (μCi/ml)	Decontamination Factor
Activation:						
^{24}Na	2.6 ± 2.1(-4)	>2.6	3.6 ± 1.3(-4)	>3.6	3.1 ± 0.6(-4)	0.22 ± 0.05
^{54}Mn	4.5 ± 0.1(-4)	14. ± 8.	2.4 ± 5.5(-5)	1.6 ± 5.4	1.3 ± 0.3(-5)	0.72 ± 0.17
^{58}Co	4.0 ± 0.1(-4)	>19.	<4(-5)		1.85 ± 0.04(-4)	3.4 ± 0.3
^{60}Co	2.4 ± 0.4(-5)	11. ± 4.	3.0 ± 1.5(-5)	0.75 ± 0.53	2.97 ± 0.07(-5)	7.4 ± 0.9
Iodine:						
^{131}I	1.92 ± 0.03(-2)	450. ± 270.	1.18 ± 0.02(-2)	59. ± 6.	5.2 ± 4.4(-5)	0.0080 ± 0.0068
Other Fission Products:						
^{88}Rb	1.8 ± 0.7(-1)	1.2 ± 0.6	3.7 ± 1.4(-1)	1.2 ± 1.5	3.2 ± 0.2(-1)	0.97 ± 0.08
^{134}Cs	1.21 ± 0.02(-3)	30. ± 8.	3.6 ± 0.2(-4)	3.8 ± 2.2	3.10 ± 0.03(-3)	0.190 ± 0.002
^{136}Cs	2.9 ± 0.2(-4)	>13.	1.2 ± 0.2(-4)	20 ± 40	1.4 ± 0.2(-5)	0.58 ± 0.13
^{137}Cs	1.21 ± 0.02(-3)	12. ± 5.	3.8 ± 0.3(-4)	4.1 ± 0.8	3.79 ± 0.02(-3)	0.201 ± 0.002
^{138}Cs	1.0 ± 1.4(-2)	0.5 ± 0.7	5.0 ± 2.2(-2)	1.7 ± 6.8	<2(-3)	

3.18 Decontamination Factor for Charcoal Adsorbers and HEPA Filters

It was not possible to install samples up and downstream of the fuel handling area charcoal adsorber because of the physical arrangement of the ventilation ducts. For this reason, no measurements were made of the DF of the charcoal adsorbers. A direct measurement of the DF of the auxiliary building HEPA exhaust filters was not possible because samplers could not be installed in valid sampling locations downstream of these filters.

A ^{137}Cs DF was calculated for the HEPA filters (installed new April, 1976) using the measurements of this work in conjunction with plant particulate release data for the fourth quarter of 1976 (8). The data are presented in Table 3.23. Decontamination factors for other radionuclides could not be calculated due to limited corresponding data. The data in Table 3.23 yield a DF of 70.

In addition, the plant numbers include releases from the containment and/or the waste gas releases. Consequently, a DF of 70 is somewhat conservative as the containment and waste gas systems do not enter the HEPA filters and are not measured as a part of the input value, but are included as a part of the output value.

3.19 Decontamination Factors for Evaporators

At Fort Calhoun Station, the radwaste evaporator is used for processing all liquid radwastes except detergent wastes or wastes from the hotel waste tanks. Table 2.2 presents the design information for the radwaste evaporator and Figure 2.3 presents a schematic diagram of the feed and output streams for this evaporator. Operating practice was to completely process a full radwaste collection tank with a constant feed rate (varying from 5 to 13 gpm, nominally 10 gpm). The concentrate bottoms were removed in a batch process mode when the bottoms boron concentration reached 8 to 10 percent. An anti-foaming agent (General Electric Co., Type AF-72) and sodium thiosulfate were added to the waste holdup and spent regenerant tanks just prior to processing the tank liquids with the radwaste evaporator. The sodium thiosulfate was added to keep the radioiodines in solution. For the waste holdup tanks, the anti-foam agent was added to a level of 110 ppm while the sodium thiosulfate was added to a level of 80 ppm. For the spent regenerant tanks, the anti-foam agent was added to a level of 84 ppm while the sodium thiosulfate was added to a level of 22 ppm.

In October, 1976, the radwaste evaporator tube bundle was replaced with a new bundle of the same design as the original one. The old tube bundle had developed leaks such that the distillate water quality could not be properly controlled. That is, the conductivity of the distillate became high and uncontrollable (see Figure A.23). The conductivity of the distillate is a measure of the carry-over of ions from the feed and/or bottoms of an evaporator. If an evaporator is working properly, only water vapor is transferred from the feed and/or bottoms to the distillate. A malfunction of the evaporator occurs when moisture droplets are carried over from the feed and/or bottoms to the

TABLE 3.23

FOURTH QUARTER 1976 ^{137}Cs AUXILIARY BUILDING RELEASE DATA

Date - (1976)	Total of Sampling Stations[A] 1, 2, and 3 (μCi/sec)
9/30 - 10/07	3.1(-5)
10/07 - 10/14	9.8(-5)
10/14 - 10/22	4.2(-5)
10/22 - 10/28	5.4(-5)
10/28 - 11/04	6.0(-5)
11/04 - 11/10	5.7(-5)
11/10 - 11/23	1.4(-5)
11/23 - 11/30	6.0(-5)
12/01 - 12/15	2.6(-5)
12/15 - 1/06	1.3(-4)
AVERAGE:	5.72(-5)

Plant Data (8):

^{137}Cs: 6.29(-6) Curies per quarter

or: 6.29(-6) Ci/7.88(+6) seconds[B] x 1(+6) μCi/Ci

= 7.98(-7) μCi/sec.

[A] Tables: A.35, A.36 and A.37 in Appendix.

[B] 365 days/year x 0.25 years x 1440 min/day x 60 sec/min = 7.88(+6) sec.

distillate. These moisture droplets contain ions or radioions and transport these ions to the distillate. This results in a high conductivity of the distillate and reduces the radionuclide clean-up properties of the evaporator. As can be seen by comparing the distillate conductivity data presented in Figures A.23 and A.26, the new tube bundle installed in October, 1976 caused the radwaste evaporator at Fort Calhoun Station to operate in a better manner as far as the distillate conductivity.

Measurements were made of the decontamination factor (DF) for the radwaste evaporator with both the old and new tube bundle for several different radionuclides. The evaporator DF was defined as the ratio of the feed radionuclide concentration to the radionuclide concentration of the distillate. The evaporator feed and distillate samples along with a sample of the evaporator bottoms were taken at the same time to provide a sample set for measurement of the DF. It is realized that there was a finite time delay between when a particular water molecule of the feed stream reached the distillate stream because of the evaporator operation. However, this time delay was not assessible and the measurements made at Fort Calhoun on the radwaste evaporator do provide the data on how well a radwaste evaporator operating in a nuclear plant environment will remove radionuclides from liquid radwastes.

Table 3.24 presents the radionuclide feed concentrations and decontamination factors (DF's) measured for three sample series taken on the "old" tube bundle in August, 1976. The data for Table 3.24 can be found in Tables A.28, A.29, and A.30. Also presented in Table 3.24 are the identity of the radwaste tank feeding the evaporator, radwaste tank level, the evaporator feed rate, the evaporator bottom's boron concentration, and the distillate conductivity at the time of the measurements. These parameters were obtained from Figures A.17, A.20, A.24 and A.25. No pH measurements were made on the feed liquid to the radwaste evaporator. The errors shown were calculated in the same way as discussed in Section 3.16.

From the measured DF's presented in Table 3.24, only the radionuclides ^{54}Mn, ^{60}Co, and ^{131}I have any indications of an increased DF with increased feed concentration. The other radionuclides do not show this. The observed DF for ^{58}Co was the highest for the sample series taken on 8/26/76. For this sample series, the spent regenerant tank WD-13A was at a fairly low level (800 gal of a capacity of 6000 gal or about 1/8 full). At Fort Calhoun, the activation or "crud" radionuclide (^{51}Cr, ^{54}Mn, ^{55}Fe, ^{59}Fe, ^{58}Co, and ^{60}Co) with the highest activity level was ^{58}Co. The larger DF for ^{58}Co from the 8/26/76 sample series was probably due to a higher fraction of suspended particulate ^{58}Co material in the feed because of a lower tank level than during the sample series done on 8/25/76 and 8/31/76. The distillate conductivity for the sample series taken on 8/31/76 was higher, indicating higher moisture and/or ion carry-over to the evaporator distillate. The measured DF's for ^{58}Co, ^{134}Cs, and ^{137}Cs were lower on 8/31/76 and reflect this higher moisture carry-over. However, the DF's for ^{54}Mn, ^{60}Co, and ^{131}I were not lower on 8/31/76. These apparent discrepancies in relating the DF's with distillate conductivity

TABLE 3.24

EVAPORATOR DECONTAMINATION FACTORS

"Old" Tube Bundle - August, 1976

	Feed: Waste Holdup WD-4B 13:43; 8/25/76 Tank Level: 34,400 gal Feed Rate: 7 gpm Bottoms Boron Conc: 4.6% Distillate Cond: 5.8 μmho		Feed: Spent Regenerant WD-13A 09:28; 8/26/76 Tank Level: 800 gal Feed Rate: 5 gpm Bottoms Boron Conc: 5.5% Distillate Cond: 3.4 μmho		Feed: Spent Regenerant WD-13B 13:45; 8/31/76 Tank Level: 3330 gal Feed Rate: 13 gpm Bottoms Boron Conc: 6.5% Distillate Cond: 8.2 μmho	
Nuclide	Feed Activity (μCi/ml)	Decontamination Factor	Feed Activity (μCi/ml)	Decontamination Factor	Feed Activity (μCi/ml)	Decontamination Factor
^{54}Mn	8.0 ± 1.2(-6)	26 ± 6	1.65 ± 0.07(-5)	80 ± 50	1.20 ± 0.06(-4)	67 ± 4
^{58}Co	6.2 ± 0.1(-5)	60 ± 4	2.96 ± 0.01(-4)	330 ± 60	2.74 ± 0.01(-4)	39 ± 1
^{60}Co	6.1 ± 0.4(-6)	12 ± 1	2.3 ± 0.1(-5)	40 ± 10	6.74 ± 0.06(-4)	630 ± 40
^{131}I	5.06 ± 0.07(-3)	179 ± 4	6.80 ± 0.07(-3)	320 ± 10	1.65 ± 0.01(-2)	880 ± 10
^{134}Cs	1.78 ± 0.08(-3)	1350 ± 90	2.23 ± 0.01(-3)	830 ± 120	1.49 ± 0.01(-3)	368 ± 5
^{137}Cs	1.86 ± 0.06(-3)	1110 ± 50	2.57 ± 0.02(-3)	950 ± 180	1.49 ± 0.01(-3)	313 ± 3

may be explained by the feed concentration levels, because higher feed concentrations result in higher DF's. That is, for ^{54}Mn, ^{60}Co, and ^{131}I, the feed concentrations on 8/31/76 were up by a factor of 5 to 10 times over the levels on 8/25/76 and 8/26/76. This increased feed activity level for ^{54}Mn, ^{60}Co, and ^{131}I caused the measured DF's to be higher even though the distillate conductivity was higher. The feed concentrations for ^{60}Co, ^{134}Cs, and ^{137}Cs were essentially the same for all three sample series. Therefore, for ^{58}Co, ^{134}Cs, and ^{137}Cs, the variance in measured DF's with distillate conductivity were not complicated by a simulataneous variance of feed concentration.

Table 3.25 presents the radionuclide feed concentrations and decontamination factors (DF's) measured for three sample series taken on the "new" tube bundle in February, 1977. The data for Table 3.25 can be found in Tables A.32, A.33, and A.34. Also presented in Table 3.25 are the identity of the radwaste tank feeding the evaporator, the radwaste tank level, the evaporator feed rate, the evaporator bottom's boron concentration, and the distillate conductivity at the time of the measurements. These parameters were extracted from Figures A.22 and A.27. No pH measurements on the feed to the radwaste evaporator were made. The errors on the DF's are the propagation of the one standard deviation errors on the measured radionuclide concentrations for the feed and distillate.

For the measurements on the "new" tube bundle, the distillate conductivities were significantly lower than for the August, 1976 data on the "old" tube bundle (1.2-1.3 μmho for "new" and 3.4-8.2 μmho for "old"). This lower distillate conductivity, which indicates less moisture and/or ion carry-over for the "new" tube bundle, is why the DF's were much higher for all radionuclides in February, 1977 than in August, 1976.

For the evaporator DF measurement series taken in February, 1977, a spent regenerant tank (WD-13A) was followed from a 75% level down to a 55% level while its contents were being processed through the radwaste evaporator at a constant feed rate of 5 gpm. Since the evaporator feed for this measurement series was from the same tank (WD-13A), the feed radionuclide concentrations for the six isotopes (^{54}Mn, ^{58}Co, ^{60}Co, ^{131}I, ^{134}Cs, and ^{137}Cs) did not vary significantly for the three sample sets (see Table A.32). However, as is also shown in Table 3.25, the measured DF's for the cobalt and cesium radionuclides increased with time that the tank WD-13A had been feeding the evaporator. This indicates that the longer a constant feed is fed to the radwaste evaporator, then an equilibrium condition is established between evaporator feed, bottoms, and distillate, resulting in higher radionuclide decontamination factors. Therefore, the evaporator DF measurements taken at Fort Calhoun Station in August, 1976 and February, 1977 indicate that radwaste evaporator DF is a function of the radionuclide feed concentration, distillate conductivity (i.e., moisture and/or ion carry-over), and the length of time the evaporator processes a given and constant feed stream. That is, the higher the feed radionuclide concentration, the higher is the radionuclide DF. Also, the higher the distillate conductivity, the lower is the DF while the longer a constant feed stream is processed, the higher the DF becomes.

TABLE 3.25

EVAPORATOR DECONTAMINATION FACTORS

"New" Tube Bundle - February, 1977

Feed: Spent Regenerant Tank WD-13A

	10:25; 2/18/77 Tank Level: 4530 gal Feed Rate: 5 gpm Bottoms Boron Conc: 6.4% Distillate Cond: 1.3 μmho		13:45; 2/18/77 Tank Level: 3800 gal Feed Rate: 5 gpm Bottoms Boron Conc: 6.8% Distillate Cond: 1.2 μmho		15:55; 2/18/77 Tank Level: 3330 gal Feed Rate: 5 gpm Bottoms Boron Conc: 6.9% Distillate Cond: 1.2 μmho	
Nuclide	Feed Activity (μCi/ml)	Decontamination Factor	Feed Activity (μCi/ml)	Decontamination Factor	Feed Activity (μCi/ml)	Decontamination Factor
^{54}Mn	3.37 ± 0.07(-5)	1080 ± 170	3.00 ± 0.06(-5)	2260 ± 440	2.97 ± 0.07(-5)	1800 ± 290
^{58}Co	5.92 ± 0.05(-4)	750 ± 10	5.56 ± 0.07(-4)	1030 ± 20	5.81 ± 0.09(-4)	1310 ± 30
^{60}Co	2.86 ± 0.08(-5)	330 ± 20	2.5 ± 0.2(-5)	460 ± 40	2.53 ± 0.08(-5)	490 ± 30
^{131}I	9.0 ± 0.2(-4)	6770 ± 340	1.13 ± 0.01(-3)	5890 ± 160	1.09 ± 0.02(-3)	6470 ± 330
^{134}Cs	1.22 ± 0.01(-3)	2980 ± 100	1.31 ± 0.02(-3)	3190 ± 470	1.38 ± 0.02(-3)	4440 ± 140
^{137}Cs	1.47 ± 0.02(-3)	2870 ± 180	1.54 ± 0.02(-3)	2750 ± 40	1.64 ± 0.02(-3)	4210 ± 180

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APPENDIX

A.1 Introduction

The data for the in-plant measurement program at the Fort Calhoun Station are presented in this Appendix. The data were collected over a period of about 6.5 months from 8/15/76 through 2/24/77. During this period Fort Calhoun was down for its second refueling from 10/1/76 through mid-December, 1976.

A.2 Reactor Power Level

The reactor power level for the period 8/15/76 through 10/3/76 is presented in Figure A.1. Figure A.2 presents the reactor power level for the period 12/15/76 through 2/24/77. Noted on both Figures A.1 and A.2 are the times when reactor coolant samples were taken at Fort Calhoun Station.

A.3 Liquid Samples

Liquid measurements were made from 8/18/76 through 9/2/76 for the end of an operating cycle prior to refueling, from 10/11/76 through 12/3/76 for the refueling period, and 2/9-22/77 for the beginning of an operating cycle following refueling.

A.3.1 Reactor Coolant

Figure A.3 is a reproduction of the FSAR diagram of the reactor coolant system. Samples were taken at sample points #6 and #7 shown on Figure A.3. Tables A.1-A.7 show the radionuclide concentrations measured in reactor coolant. The reactor loop number and time of collection are noted on the tables. Reactor coolant radionuclide concentrations for the periods of power operation prior to refueling and after refueling are shown in Tables A.1 and A.3, respectively. Concentrations of pure beta emitting radionuclides in reactor coolant, spent fuel pool water and a monitor tank are shown in Table A.2.

Three reactor coolant samples were taken during the refueling outage. The results for these samples are shown in Table A.4. The reactor vessel cover was removed on 10/22/76 and the refueling cavity and fuel transfer canal were filled on 10/26/76. The spent fuel was transferred from the reactor to the spent fuel pool between 11/2/76 and 11/7/76. The refueling cavity was drained and the reactor vessel cover replaced by 12/2/76 for testing prior to start-up.

Reactor coolant samples were degassed by Fort Calhoun operating personnel in the following manner. A 300 ml sample was collected in an in-line steel container on the same sample line as for the non-degassed samples. The reactor coolant gas was transferred to an evacuated 250 ml glass container by bubbling air through the steel container. The ^{133}Xe activities in both the liquid and gas fractions were used to determine the degassing efficiency.

For each of the before and after refueling measurement periods, reactor coolant samples were collected and degassed by Fort Calhoun personnel prior to our counting. Tables A.5 and A.6 present the analysis results for these degassed samples. By comparing the data in Tables A.1 and A.3 with Tables A.5 and A.6 it is apparent that the degassing did not significantly increase the sensitivity for the non-detected radionuclides. Also, the degassing procedure delayed the first count on these samples to 1 to 2 hours after collection. This precluded measurement of the short-lived radionuclides.

For each of the liquid reactor coolant samples presented in Table A.6, a sample of the reactor coolant gases was obtained from OPPD. The radionuclide analysis results of these samples are presented in Table A.7.

Shown in Tables A.1, A.5, A.6, and A.7 are the results from analysis of similar reactor coolant samples by OPPD.

A.3.2 CVCS Liquid

A diagram of the chemical and volume control system (CVCS) is presented in Figure A.4. The indicated local sample point on the volume control tank was not installed during construction. The information on when the mixed bed purification ion exchangers (CH-8A and CH-8B) were in service and recharged is presented in Figure A.5 for the period July, 1973 through February, 1977. For these demineralizers, the resin was replaced and the used resin disposed of as solid wastes.

During the before refueling in-plant measurement period at Fort Calhoun, the purification ion exchanger (CH-8B) was in service and the cation ion exchanger (CH-10) was not used. The samples for the CVCS were obtained from the points number 1, 2, and 5, in Figure A.4. The three samples (50 ml, each) were collected at essentially the same time and counted within 20 minutes of collection, with the input (#1) counted first.

Tables A.8 and A.9 present the radionuclide analysis data for the samples from the CVCS on 8/20/76 and 8/30/76, respectively. Figure A.6 presents a graph of the letdown flow rate for the CVCS during the period 8/15/76 through 9/9/76. Figure A.7 presents the graphs of the reactor coolant pH and conductivity plus the pressure drops across the purification demineralizers and filters for the period 8/9/76 through 9/2/76. Sample times are noted on both figures.

During the refueling measurement period, the purification ion exchangers were operated in series for clean-up of reactor coolant prior to reactor vessel head removal. Samples were taken on 10/11/76 when CH-8A and CH-8B were in series and Table A.10 presents the results. These samples were taken from points numbered 1, 2, and 5 on Figure A.4. Figure A.8 presents the letdown flow rate for the period 9/25/76 through 10/19/76.

After movement of spent fuel from the core to the fuel pool, the cation ion exchanger (CH-10) of the CVCS was used to clean up the refueling cavity water prior to placement in the safety injection and refueling water tank (SIRWT). On 11/11/76, samples were taken of the input and output of CH-10 (points #1 and #3, respectively in Figure A.4) during the processing of refueling cavity water. Table A.11 presents the analysis results.

During the post-refueling measurement period in February, 1977, a sample series was taken of the letdown flow being processed through the purification ion exchanger CH-8A. These samples were taken from points 1, 2, and 5 on Figure A.4. A filtration procedure (described in the procedures report (1)) was used on these samples to determine the suspended radioactive solids with sizes larger than 0.5 micron. The data from these samples is presented in Table A.12. In this and all subsequent tables where dissolved and suspended activities are reported, the suspended activities represent radionuclides observed on the filter. Dissolved activities represent activities found in the filtrate. The letdown flow rate during the period from 1/31/77 through 2/24/77 is presented in Figure A.9. Figure A.10 presents graphs of reactor coolant boron concentration and pressure drop across the purification demineralizers and filters for the same period.

A.3.3 Secondary Liquid

Steam generator blowdown samples were collected from both generators before refueling on 8/26/76 and after refueling on 2/8/77 and 2/10/77. The February samples were over 200 liters in size and were collected with an ion exchange column technique described in the procedures report (1). The radionuclide analysis results are presented in Table A.13. Because of other possible pathways for radioactivity to enter secondary liquid, the radionuclide concentrations measured in the steam generator blowdown were not necessarily indicative of a primary to secondary steam generator leak. Figure A.11 presents the steam generator blowdown and flow rates for the period 1/31/77 through 2/24/77.

A.3.4 Spent Fuel Pool and Fuel Transfer Associated Liquids

Figure A.12 presents a diagram of the spent fuel pool cooling and clean-up system. The local sample points are indicated. During the measurement period, the valve for sample point #2 was not operable. Therefore, only spent fuel pool water or the input to the fuel pool clean-up system could be sampled (#1). The volume of the spent fuel pool is 215,000 gallons. According to the auxiliary building operator's log, the spent fuel pool was full during the in-plant measurement periods. The temperature of the spent fuel pool water is presented on the tables with the radionuclide data. Table A.14 presents the radionuclide concentrations for two samples taken from the spent fuel pool during the before refueling. Table A.2 presents the activities from a beta analysis of a spent fuel pool sample taken before refueling. At this time, the pool contained spent fuel rods from one-third of the core removed during the first refueling in March, 1975.

During refueling, water is taken from the safety injection and refueling water tank (SIRWT) to fill the refueling cavity and fuel transfer canal. The volume of the SIRWT and refueling cavity are 314,000 and 249,500 gallons, respectively. The refueling cavity and fuel transfer canal were filled from the SIRWT on 10/26/76. The spent fuel movement occurred between 11/2/76 and 11/7/76. Water was transferred from the refueling cavity and fuel transfer canal after clean-up with the CVCS by 12/3/76.

Table A.15 presents the radionuclide concentrations in samples taken from the SIRWT before and after the reactor refueling (from #3 on Figure A.12). Table A.16 presents the radionuclide concentrations for samples taken from the fuel transfer canal and refueling cavity before and after movement of the spent fuel (dip samples). Table A.17 presents radionuclide concentrations for four spent fuel pool samples taken at different times during the refueling process.

During the post-refueling measurement period, a sample was taken from the spent fuel pool. This sample was filtered using the technique described in the procedures report (1). Table A.18 presents the radionuclide concentrations in the spent fuel pool sample taken in February, 1977.

A.3.5 Liquid Radwaste System

Figures A.13 through A.16 present diagrams of the liquid radwaste system. The radwaste tanks which were sampleable were the reactor coolant drain tank (WD-1), the spent regenerant tanks (WD-13A and WD-13B), the waste holdup tanks (WD-4A and WD-4B), the hotel waste tanks (WD-15A and WD-15B) and the monitor tanks (WD-22A and WD-22B). The sample point for the reactor coolant drain tank was a local valve on the tank, within containment. Sample points for the other tanks are noted by numbers one through four on Figures A.14 through A.16. Liquid radwaste samples obtained after refueling were filtered using the technique described in the procedures report (1).

Figures A.17, A.18, and A.19 present the tank levels as a function of time for the waste holdup tanks for the three measurement periods. Figures A.20, A.21 and A.22 present the tank level information for the reactor coolant drain tank, the spent regenerant tanks, the hotel tanks, and the monitor tanks for the three measurement periods.

A.3.5.1 Reactor Coolant Drain Tank

Table A.19 presents the radionuclide concentrations observed in the reactor coolant drain tank for a sample taken during the after refueling measurement period.

A.3.5.2 Waste Holdup Tanks

Table A.20 presents the radionuclide concentrations determined on two samples taken from the waste holdup tanks during the before refueling measurement period. A duplicate of the sample taken 11:40, 9/2/76 was analyzed by DOE-HSL and their results are also shown. Table A.21 presents the radionuclide concentrations in three waste holdup tank samples taken during refueling. Table A.22 presents the radionuclide concentrations measured in a sample from waste holdup tank WD-4A taken after refueling. A duplicate sample was also analyzed by DOE-HSL and their results are shown.

A.3.5.3 Spent Regenerant Tanks

Table A.20 presents radionuclide concentrations measured on three samples of the spent regenerant tanks taken during the before refueling measurement period. Table A.21 presents data on one sample taken during refueling. Radionuclide concentrations of two samples taken from the spent regenerant tanks during the after refueling measurement period are shown in Table A.23.

A.3.5.4 Hotel Waste Tanks

Table A.24 presents radionuclide concentrations observed in a sample taken from hotel tank WD-15A before refueling. Radionuclide data measured on two hotel tank samples taken after refueling are shown in Table A.25.

A.3.5.5 Monitor Tanks

Tables A.2 and A.24 present radionuclide data for three samples taken from the monitor tanks before refueling. One monitor tank sample was taken during refueling and the results are shown in Table A.21. Table A.26 presents data from two monitor tank samples taken after refueling. After refueling, a series of four samples were taken at different times in the release of monitor tank WD-22B to the discharge pipe. These results are presented in Table A.27. Several times during the measurement period, OPPD collected samples of monitor tank water at the same time our samples were obtained. Comparisons of OPPD's and our results for these samples are presented in Tables A.24A, A.26 and A.27. In order to make a valid comparison, our results reported in these tables are for unfiltered samples (i.e., same procedures used by OPPD).

A.3.5.6 Waste Evaporator

Evaporator feed samples were taken from the tanks being processed and the distillate and concentrate samples were taken from points 5 and 6 on Figure A.16, respectively. Tables A.28 through A.30 present the radionuclide data for the samples taken while the evaporator was processing feed from waste holdup tank WD-4B, spent regenerant tank WD-13A and spent regenerant tank WD-13B, respectively.

Figure A.23 presents the radwaste evaporator parameters, evaporator feed, bottoms boron concentration, and distillate conductivity for the period 8/9/76 through 9/2/76. Figures A.24 and A.25 present the same evaporator parameters with an expanded time scale to better show the variation of these parameters during the period when the samples of Tables A.28 through A.30 were taken.

During October, 1976 the evaporator tube bundle was replaced with a new bundle. Scraping samples were obtained from each end of the evaporator bundle on 10/13/76. Concentrations shown in Table A.31 are in μCi per sample and provide a relative indication of the radionuclide buildup on the evaporator tube bundle.

Figure A.26 presents the evaporator operating parameters, feed, bottoms boron concentrations and distillate conductivity for the period 1/31/77 through 2/21/77. The bottoms were removed in a batch mode. A series of measurements were made in February, 1977 to evaluate the waste evaporator performance with a constant feed. The operational parameters are shown on an expanded time scale in Figure A.27. The radionuclide concentrations measured for the feed, distillate, and concentrated bottoms are shown in Tables A.32, A.33, and A.34, respectively.

A.4 Gaseous Samples

In the text of this report (Section 2.3.1), the sampling stations and locations for the gaseous streams are described. Figure A.28 is the P. & I.D. for the ventilation system at Fort Calhoun Station with sample stations #1 through #4 noted on the figure. Figure A.29 presents an as-built modification to the P. & I.D. shown in Figure A.28. The radionuclide activities measured at sample stations #1 through #4 are presented in Tables A.35 through A.38, respectively. Additional samplers were placed in rooms to determine the airborne activities reaching the ventilation system from these areas. These rooms were the waste evaporator room, the pipe penetration room, and the letdown heat exchanger room. Tables A.39 through A.41 present the results of these measurements. Iodine-131 species data were collected at ventilation sample station #2 and the waste evaporator room during most of the in-plant measurement period at Fort Calhoun. Tables A.42 and A.43 present these data. During the post-refueling measurement period, ^{131}I species data were collected using the ventilation sample stations for station #1, letdown heat exchanger room and pipe penetration room. These data are presented in Table A.44.

In addition to the ventilation samples, grab samples were taken from portions of the process gas system. In all cases, the samples were taken through the plant's Automatic Gas Analyzer System. Iodine species samplers were used to obtain samples of several of the subsystems. Two samples of gas from different waste gas decay tanks were also taken for gamma analysis. The results of these measurements are shown in Tables A.45 through A.48.

Carbon-14 and tritium analyses were performed on the same process and cover gas subsystems as were the iodine species determinations. The sampler developed for air monitoring was not used because of the absence of oxygen in these systems. As an alternate sampling procedure, 75 cc stainless steel gas bombs were used to obtain these gas samples. The gas bombs were returned to INEL where the contents were mixed with low activity air and processed through the normal sampling system. The results of these measurements are shown in Tables A.49 and A.50.

Two samples of the containment atmosphere were obtained. The first sample, taken in October, 1976, was an iodine species sample taken through a plant installed containment penetration. The second sample, taken in January 1977, consisted of iodine species and ^{14}C and tritium measurements. These latter samplers were set up inside the containment structure and were operated for approximately two days. The results of these measurements are shown in Table A.51.

TABLE A.1

REACTOR COOLANT ACTIVITIES
Power Operations - Before Refueling

Nuclide	Loop #1 14:52; 8/18/76 (μCi/ml)	Loop #1 09:17; 8/19/76 (μCi/ml)	OPPD 12:58 8/19/76 (μCi/ml)	Loop #2 09:58; 8/19/76 (μCi/ml)	Loop #1 11:54; 9/1/76 (μCi/ml)	OPPD 10:05 9/1/76 (μCi/ml)	Loop #2 11:19; 9/1/76 (μCi/ml)
Activation Nuclides:							
^{24}Na	2.2 ± 0.4(-3)	1.9 ± 0.2(-3)		2.8 ± 0.3(-3)	1.3 ± 0.2(-3)		1.4 ± 0.1(-3)
^{51}Cr	<1(-2)	<2(-2)	<7(-3)	<2(-2)	<3(-2)	<6(-3)	<3(-2)
^{54}Mn	1.3 ± 0.1(-4)	2.1 ± 0.2(-4)	1.9(-2)	3.0 ± 0.3(-4)	1.1 ± 0.2(-3)	1.2(-2)	3.5 ± 1.3(-4)
^{56}Mn	<1(-3)	3.3 ± 0.3(-2)		3.7 ± 0.4(-2)	3.6 ± 0.3(-2)		3.6 ± 0.7(-2)
^{59}Fe	1.1 ± 0.2(-4)	5.6 ± 1.6(-5)	<1(-3)	5.8 ± 1.6(-5)	<2(-4)	<9(-4)	<8(-4)
^{57}Co	<1(-4)	2.8 ± 1.5(-3)	<5(-4)	<8(-5)	<2(-4)	<5(-4)	<4(-4)
^{58}Co	9.0 ± 0.1(-3)	6.03 ± 0.05(-3)	2.4(-1)	6.13 ± 0.05(-3)	7.1 ± 0.3(-3)	3.2(-2)	1.24 ± 0.08(-2)
^{60}Co	1.6 ± 0.7(-4)	7.9 ± 1.0(-5)	2.8(-3)	9.6 ± 0.9(-5)	2.0 ± 0.9(-4)	2.4(-3)	<4(-4)
^{65}Zn	<1(-4)	<2(-5)	<2(-3)	5.0 ± 3.8(-3)	1.1 ± 0.6(-3)	<1(-3)	2.2 ± 0.8(-3)
^{187}W	2.6 ± 0.8(-3)	8.3 ± 3.4(-3)		1.4 ± 0.2(-2)	3.0 ± 0.6(-2)		3.8 ± 0.7(-2)
^{239}Np	<8(-3)	1.8 ± 0.6(-2)		2.0 ± 1.5(-2)	5.3 ± 1.9(-3)		<2(-2)
Iodine Nuclides:							
^{131}I	1.48 ± 0.02(-1)	2.03 ± 0.02(-1)	2.4(-1)	2.05 ± 0.04(-1)	3.92 ± 0.01(-1)	4.3(-1)	3.93 ± 0.02(-1)
^{132}I	1.13 ± 0.01(-1)	8.2 ± 0.1(-2)	8.9(-2)	8.49 ± 0.09(-2)	1.05 ± 0.01(-1)	6.8(-2)	1.01 ± 0.01(-1)
^{133}I	3.36 ± 0.01(-2)	8.03 ± 0.05(-2)	9.5(-2)	8.57 ± 0.07(-2)	2.40 ± 0.01(-1)	2.4(-1)	2.40 ± 0.01(-1)
^{134}I	3.8 ± 0.9(-2)	3.2 ± 0.3(-2)	2.2(-2)	3.55 ± 0.67(-2)	2.8 ± 0.3(-2)	1.6(-2)	3.0 ± 0.3(-2)
^{135}I	2.8 ± 0.2(-2)	6.5 ± 0.2(-2)	6.1(-2)	6.7 ± 0.2(-2)	7.2 ± 0.2(-2)	6.1(-2)	7.3 ± 0.2(-2)

TABLE A.1 (cont.)

REACTOR COOLANT ACTIVITIES
Power Operations - Before Refueling

Nuclide	Loop #1 14:52; 8/18/76 (μCi/ml)	Loop #1 09:17; 8/19/76 (μCi/ml)	OPPD 12:58 8/19/76 (μCi/ml)	Loop #2 09:58; 8/19/76 (μCi/ml)	Loop #1 11:54; 9/1/76 (μCi/ml)	OPPD 10:05 9/1/76 (μCi/ml)	Loop #2 11:19; 9/1/76 (μCi/ml)
Fission Nuclides:							
^{88}Rb	5.3 ± 0.4(-1)	5.2 ± 0.2(-1)		4.4 ± 0.6(-1)	3.8 ± 0.2(-1)		4.0 ± 0.2(-1)
^{89}Rb	6.2 ± 0.3(-2)	4.8 ± 0.4(-2)		4.5 ± 0.4(-2)	7.0 ± 0.4(-2)		7.1 ± 0.3(-2)
^{95}Zr	8.0 ± 2.7(-5)	<3(-5)	<1(-3)	<1(-4)	8.1 ± 3.2(-4)	<8(-4)	<7(-4)
^{95}Nb	<9(-5)	8.6 ± 3.0(-5)	<7(-4)	3.8 ± 1.5(-5)	<3(-4)	<5(-4)	<3(-4)
^{99}Mo	1.4 ± 0.2(-2)	1.5 ± 0.2(-2)	8.4(-3)	1.32 ± 0.07(-2)	9.5 ± 0.6(-3)	1.3(-3)	2.5 ± 0.4(-3)
^{101}Mo	1.4 ± 0.6(-2)	1.1 ± 0.3(-2)		9.5 ± 6.2(-3)	2.1 ± 0.5(-2)		<2(-2)
^{104}Tc	<6(-3)	3.3 ± 1.3(-2)		<2(-3)	3.7 ± 1.1(-3)		2.4 ± 1.6(-3)
^{103}Ru	<2(-4)	<2(-4)	<8(-4)	<3(-4)	2.7 ± 1.1(-4)	<6(-4)	1.1 ± 0.6(-3)
^{110m}Ag	2.7 ± 2.1(-4)	3.5 ± 4.5(-4)		3.8 ± 5.0(-4)	0.2 ± 4.7(-4)		0.2 ± 4.7(-4)
^{124}Sb	1.2 ± 0.7(-5)	<1(-4)	<1(-3)	<2(-4)	<5(-4)	<1(-3)	<1(-4)
^{126}Sb	<1(-4)	<1(-4)		<3(-4)	<5(-4)		<1(-4)
^{127}Sb	1.1 ± 0.3(-3)	<2(-5)		<2(-4)	<7(-4)		<9(-4)
^{129}Te	<1(-2)	<9(-3)		<1(-2)	<7(-3)		<8(-3)
^{129m}Te	<3(-3)	<2(-3)		<2(-3)	<8(-3)		<9(-3)
^{131m}Te	<6(-4)	<3(-4)		<3(-4)	<6(-4)		<6(-4)
^{132}Te	<1(-3)	<2(-3)		<2(-3)	<8(-3)		<4(-3)
^{134}Cs	1.56 ± 0.01(-2)	1.36 ± 0.01(-2)	1.2(-2)	1.45 ± 0.01(-2)	1.47 ± 0.04(-2)	5.6(-3)	1.44 ± 0.01(-2)
^{136}Cs	3.8 ± 0.1(-3)	3.6 ± 0.1(-3)	<9(-4)	3.74 ± 0.06(-3)	4.7 ± 0.3(-3)	3.2(-3)	4.1 ± 0.2(-3)
^{137}Cs	1.56 ± 0.01(-2)	1.38 ± 0.01(-2)	1.7(-2)	1.38 ± 0.01(-2)	1.40 ± 0.02(-2)	1.6(-2)	1.42 ± 0.02(-2)
^{138}Cs	3.7 ± 0.1(-1)	3.1 ± 0.1(-1)	2.4(-1)	3.4 ± 0.1(-1)	3.22 ± 0.07(-1)	2.3(-1)	3.13 ± 0.10(-1)
^{139}Cs	<4(-4)	<7(-4)		<8(-4)	<3(-4)		1.6 ± 0.4(-1)
^{139}Ba	7.4 ± 0.7(-2)	6.4 ± 0.4(-2)		5.8 ± 0.6(-2)	6.4 ± 0.8(-2)		4.2 ± 1.1(-2)
^{140}Ba	3.2 ± 1.9(-4)	5.8 ± 1.3(-4)	<3(-3)	5.9 ± 1.5(-4)	<2(-3)	<3(-3)	<6(-4)
^{140}La	4.2 ± 1.0(-4)	3.0 ± 0.7(-4)	<7(-4)	4.7 ± 0.8(-4)	<4(-4)	<5(-4)	<2(-4)
^{141}Ce	<3(-4)	3.0 ± 1.3(-4)	<1(-3)	<1(-4)	<3(-4)	<1(-3)	<2(-4)
^{143}Ce	8.3 ± 7.5(-4)	3.0 ± 0.6(-2)		<4(-3)	4.2 ± 1.7(-3)		<8(-3)
^{144}Ce	<1(-3)	<2(-3)		<6(-4)	<1(-3)		<3(-3)

TABLE A.2

BETA, ONLY, EMITTING RADIONUCLIDE ACTIVITIES
Power Operations - Before Refueling

Nuclide	Loop #1 Reactor Coolant 12:00; 9/1/76 (µCi/ml)	Spent Fuel Pool 10:25; 8/31/76 (µCi/ml)	Monitor Tank 22B 08:17; 9/1/76 (µCi/ml)
^{3}H	4.77 ± 0.02(-2)	7.34 ± 0.04(-3)	3.75 ± 0.02(-3)
^{14}C [A]	4.7 ± 0.4(-6)	7 ± 2(-8)	2.6 ± 0.2(-7)
^{32}P	4.5 ± 0.4(-6)	<2(-7)	<4(-8)
^{35}S	5.6 ± 2.0(-6)	<2(-7)	<2(-7)
^{45}Ca	1.08 ± 0.01(-4)	1.44 ± 0.02(-5)	1.79 ± 0.02(-5)
^{55}Fe	1.06 ± 0.01(-3)	1.84 ± 0.02(-5)	5.2 ± 0.1 (-6)
^{63}Ni	1.76 ± 0.02(-4)	2.37 ± 0.04(-4)	3.6 ± 0.1(-7)
^{89}Sr	1.22 ± 0.05(-3)	<4(-8)	3 ± 1(-8)
^{90}Sr	3.8 ± 0.5(-6)	<2(-8)	<2(-8)
^{91}Y	7.7 ± 1.3(-6)	<1(-7)	<1(-8)
^{147}Pm	9 ± 2(-7)	<2(-8)	<2(-8)

[A] Analysis results are for inorganic ^{14}C only (CO_2)

TABLE A.3

REACTOR COOLANT ACTIVITIES

Power Operations - After Refueling

Nuclide	Loop #1 08:40; 2/9/77 (μCi/ml)	Loop #2 08:44; 2/17/77 (μCi/ml)	Loop #1 13:01; 2/22/77 (μCi/ml)
Activation Nuclides:			
^{24}Na	7.5 ± 1.2(-3)	1.41 ± 0.03(-2)	1.2 ± 0.2(-2)
^{51}Cr	7.3 ± 1.8(-4)	<2(-2)	<1(-2)
^{54}Mn	4.8 ± 0.3(-5)	1.6 ± 0.1(-3)	2.2 ± 0.8(-5)
^{56}Mn	1.4 ± 0.4(-2)	<4(-3)	<6(-3)
^{59}Fe	4.7 ± 0.6(-5)	1.8 ± 0.2(-4)	<3(-5)
^{57}Co	3.6 ± 0.8(-4)	2.3 ± 0.4(-3)	1.9 ± 0.2(-3)
^{58}Co	2.00 ± 0.02(-3)	3.57 ± 0.04(-3)	1.13 ± 0.05(-3)
^{60}Co	3.2 ± 0.4(-5)	1.2 ± 0.2(-4)	1.8 ± 0.7(-5)
^{65}Zn	<2(-5)	<6(-5)	<3(-5)
^{187}W	8.6 ± 0.6(-3)	<5(-4)	5.4 ± 0.6(-3)
^{239}Np	4.3 ± 1.8(-3)	<1(-2)	9.6 ± 5.7(-3)
Iodine Nuclides:			
^{131}I	9.54 ± 0.08(-2)	8.62 ± 0.03(-2)	7.37 ± 0.05(-2)
^{132}I	6.30 ± 0.08(-2)	4.56 ± 0.09(-2)	4.87 ± 0.06(-2)
^{133}I	1.27 ± 0.01(-1)	1.04 ± 0.01(-1)	1.02 ± 0.01(-1)
^{134}I	4.2 ± 0.3(-2)	2.8 ± 0.2(-2)	3.6 ± 0.3(-2)
^{135}I	8.2 ± 0.3(-2)	7.0 ± 0.2(-2)	6.8 ± 0.2(-2)
Beta Only, Nuclides:			
^{3}H	*	*	2.15 ± 0.02(-1)
^{14}C A	*	*	7.0 ± 0.2(-6)
^{55}Fe	*	*	1.16 ± 0.02(-4)
^{63}Ni	*	*	3.7 ± 0.2(-5)
^{89}Sr	*	*	8.6 ± 0.9(-6)
^{90}Sr	*	*	<8(-7)
^{91}Y	*	*	1.3 ± 0.2(-6)

* : Not analyzed for beta emitters.

A Analysis results are for inorganic ^{14}C only (CO_2).

TABLE A.3 (cont'd)

REACTOR COOLANT ACTIVITIES

Power Operations - After Refueling

Nuclide	Loop #1 08:40; 2/9/77 (μCi/ml)	Loop #2 08:44; 2/17/77 (μCi/ml)	Loop #1 13:01; 2/22/77 (μCi/ml)
Fission Products:			
^{88}Rb	4.6 ± 0.2(-1)	5.3 ± 0.3(-1)	5.0 ± 0.3(-1)
^{89}Rb	2.1 ± 0.2(-2)	1.3 ± 0.3(-2)	2.3 ± 0.2(-2)
^{95}Zr	4.3 ± 1.2(-5)	<3(-4)	<2(-4)
^{95}Nb	3.8 ± 0.3(-5)	<2(-4)	<1(-4)
^{99}Mo	1.4 ± 0.9(-3)	<8(-4)	6.4 ± 6.3(-5)
^{103}Ru	<3(-5)	1.7 ± 0.5(-5)	1.7 ± 0.6(-5)
^{110m}Ag	<3(-5)	<2(-3)	<8(-5)
^{124}Sb	3.6 ± 2.4(-6)	<2(-3)	<8(-5)
^{129}Te	<5(-4)	<2(-2)	<6(-3)
^{129m}Te	<3(-3)	<5(-3)	1.9 ± 0.8(-4)
^{131m}Te	<2(-4)	<3(-4)	<3(-4)
^{132}Te	<2(-3)	<2(-3)	<2(-3)
^{134}Cs	1.09 ± 0.02(-3)	6.23 ± 0.03(-2)	3.9 ± 0.1(-3)
^{136}Cs	3.5 ± 0.4(-5)	9.7 ± 1.3(-5)	5.6 ± 1.9(-5)
^{137}Cs	1.53 ± 0.02(-3)	7.13 ± 0.03(-2)	4.48 ± 0.06(-3)
^{138}Cs	1.73 ± 0.09(-1)	1.47 ± 0.07(-1)	2.0 ± 0.1(-1)
^{139}Cs	<2(-3)	<4(-4)	<3(-3)
^{139}Ba	6.3 ±1.4(-2)	4.9 ± 0.2(-2)	5.5 ± 0.3(-2)
^{140}Ba	<1(-4)	<5(-3)	<4(-4)
^{140}La	8.7 ± 2.3(-5)	4.4 ± 1.4(-4)	<6(-3)
^{141}Ce	2.7 ± 1.5(-4)	<2(-3)	<5(-4)
^{143}Ce	<3(-3)	<4(-3)	<3(-3)
^{144}Ce	<2(-3)	<7(-3)	2.3 ± 0.3(-3)

* : Not analyzed for beta emitters.

A : Analysis results are for inorganic ^{14}C only (CO_2).

TABLE A.4

REACTOR COOLANT ACTIVITIES

During Refueling

Nuclide	Prior to Head Removal 12:36; 10/20/76 (μCi/ml)	After Head Removal 11:00; 10/27/76 (μCi/ml)	After Head Replacement Prior to Start-up 13:15; 12/3/76 (μCi/ml)
Activation Nuclides:			
^{51}Cr	<5(-3)	<2(-3)	2.9 ± 0.9(-3)
^{54}Mn	3.60 ± 0.06(-4)	2.5 ± 0.1(-4)	8.7 ± 0.2(-4)
^{59}Fe	1.6 ± 0.1(-4)	1.2 ± 0.3(-4)	3.2 ± 0.4(-4)
^{57}Co	2.35 ± 0.08(-4)	4.1 ± 0.8(-4)	2.3 ± 0.3(-4)
^{58}Co	1.34 ± 0.03(-1)	1.78 ± 0.05(-1)	7.0 ± 0.1(-2)
^{60}Co	6.4 ± 0.1(-4)	6.9 ± 0.3(-4)	1.6 ± 0.3(-3)
^{65}Zn	<7(-5)	8.2 ± 2.5(-5)	5.1 ± 1.0(-5)
Iodine Nuclides:			
^{131}I	1.2 ± 0.2(-2)	<2(-3)	<4(-4)
Fission Products:			
^{95}Zr	<1(-2)	<3(-2)	1.1 ± 0.2(-4)
^{95}Nb	<9(-3)	<2(-2)	2.6 ± 1.7(-4)
^{103}Ru	<1(-3)	1.5 ± 0.8(-4)	<3(-4)
^{110m}Ag	<1(-3)	<2(-3)	1.5 ± 0.1(-4)
^{124}Sb	6.1 ± 0.5(-5)	1.3 ± 0.1(-4)	1.28 ± 0.09(-4)
^{129m}Te	<4(-1)	<5(-1)	<2(-1)
^{134}Cs	2.38 ± 0.02(-2)	3.65 ± 0.03(-2)	6.18 ± 0.09(-3)
^{136}Cs	2.12 ± 0.08(-3)	9.1 ± 1.0(-4)	1.9 ± 0.3(-4)
^{137}Cs	2.26 ± 0.05(-2)	3.83 ± 0.07(-2)	6.3 ± 0.1(-3)
^{140}Ba	<5(-4)	<9(-3)	5.3 ± 2.1(-4)
^{141}Ce	<4(-4)	<1(-4)	<4(-5)
^{144}Ce	<5(-1)	2.6 ± 2.3(-4)	<2(-4)
Beta Only, Nuclides:			
^{3}H	3.49 ± 0.04(-2)	3.44 ± 0.02(-2)	1.06 ± 0.02(-2)
^{14}C	2.74 ± 0.04(-5)	2.10 ± 0.02(-5)	5.3 ± 0.2(-7)
^{32}P	1.6 ± 0.7(-5)	5. ± 3.(-6)	*
^{35}S	<4(-6)	1.4 ± 0.9(-6)	2.50 ± 0.06(-5)
^{45}Ca	<2(-6)	<6(-7)	*
^{55}Fe	2.17 ± 0.04(-3)	3.89 ± 0.03(-4)	3.91 ± 0.02(-3)
^{63}Ni	5.75 ± 0.04(-3)	3.86 ± 0.02(-3)	3.94 ± 0.02(-3)
^{89}Sr	4.8 ± 0.4(-5)	5.3 ± 0.7(-6)	5.3 ± 0.3(-5)
^{90}Sr	2.0 ± 0.4(-6)	4. ± 2.(-7)	1.08 ± 0.05(-5)
^{91}Y	1.5 ± 0.4(-6)	3. ± 2.(-7)	2.13 ± 0.08(-5)
^{147}Pm	1.1 ± 0.2(-6)	1.7 ± 0.1(-6)	*

* : Sample not analyzed for these radionuclides.

TABLE A.5

REACTOR COOLANT ACTIVITIES (OPPD Degassed Sample)
Power Operations - Before Refueling

Nuclide	Loop #1 14:39; 8/23/76 (μCi/ml)	OPPD 09:04 8/26/76 (μCi/ml)	Loop #2 09:04; 8/26/74 (μCi/ml)	OPPD 08:33 8/27/76 (μCi/ml)	Loop #1 08:33; 8/27/76 (μCi/ml)
Activation Nuclides:					
^{24}Na	4.2 ± 0.2(-3)		2.8 ± 0.1(-3)		2.5 ± 0.2(-3)
^{51}Cr	<2(-2)	<7(-3)	<2(-2)	<7(-3)	<1(-2)
^{54}Mn	<1(-4)	3.2(-2)	1.1 ± 0.5(-2)	2.7(-2)	2.64 ± 0.06(-4)
^{56}Mn	2.7 ± 2.9(-3)		3.6 ± 3.1(-3)		<4(-3)
^{59}Fe	<5(-5)	<2(-3)	<2(-3)	<2(-3)	6.7 ± 2.7(-5)
^{57}Co	1.7 ± 0.4(-3)	<5(-4)	<1(-4)	<5(-4)	<7(-5)
^{58}Co	1.30 ± 0.05(-2)	2.4(-2)	1.22 ± 0.06(-2)	1.1(-2)	2.86 ± 0.04(-3)
^{60}Co	1.0 ± 0.2(-4)	<1(-3)	<1(-4)	<1(-3)	3.2 ± 0.8(-5)
^{65}Zn	<4(-5)	<2(-3)	<2(-3)	<2(-3)	<2(-5)
^{187}W	<4(-4)		<3(-3)		<2(-4)
^{239}Np	9.7 ± 4.9(-3)		<1(-2)		<1(-2)
Iodine Nuclides:					
^{131}I	2.18 ± 0.02(-1)	2.4(-1)	1.99 ± 0.02(-1)	1.9(-1)	1.71 ± 0.01(-1)
^{132}I	1.51 ± 0.01(-1)	8.7(-2)	1.06 ± 0.02(-1)	7.9(-2)	7.8 ± 0.8(-2)
^{133}I	3.20 ± 0.02(-1)	3.2(-1)	2.97 ± 0.06(-1)	2.5(-1)	2.41 ± 0.06(-1)
^{134}I	1.08 ± 0.07(-1)	5.0(-2)	5.8 ± 0.3(-2)	2.2(-2)	<5(-1)
^{135}I	2.11 ± 0.02(-1)	1.3(-1)	1.55 ± 0.03(-1)	9.7(-2)	1.24 ± 0.03(-1)

TABLE A.5 (cont.)

REACTOR COOLANT ACTIVITIES (OPPD Degassed Sample)
Power Operations - Before Refueling

Nuclide	Loop #1 14:39; 8/23/76 (μCi/ml)	OPPD 09:04 8/26/76 (μCi/ml)	Loop #2 09:04; 8/26/76 (μCi/ml)	OPPD 08:33 8/27/76 (μCi/ml)	Loop #1 08:33; 8/27/76 (μCi/ml)
Fission Products:					
^{95}Zr	6.3 ± 1.5(-4)	<1(-3)	<2(-3)	<1(-3)	<1(-4)
^{95}Nb	<1(-4)	<8(-4)	<8(-4)	<8(-4)	<6(-5)
^{99}Mo	2.45 ± 0.06(-3)	1.5(-3)	1.0 ± 0.1(-3)	1.2(-3)	3.1 ± 1.1(-4)
^{103}Ru	<1(-4)	<8(-4)	<9(-5)	<7(-4)	<2(-5)
^{110m}Ag	0.3 ± 2.6(-4)		1.5 ± 0.7(-4)		0.53 ± 0.54(-4)
^{124}Sb	2.2 ± 0.8(-4)	<2(-3)	<8(-5)	<2(-3)	<2(-5)
^{126}Sb	<1(-4)		<8(-5)		<2(-5)
^{127}Sb	2.5 ± 1.8(-4)		<2(-3)		<2(-4)
^{134}Cs	4.5 ± 0.1(-3)	<9(-4)	1.8 ± 0.5(-3)	<9(-4)	1.2 ± 0.2(-3)
^{136}Cs	1.2 ± 0.1(-3)	<1(-3)	4.5 ± 1.3(-4)	<1(-3)	3.06 ± 0.08(-4)
^{137}Cs	4.64 ± 0.07(-3)	<9(-4)	1.6 ± 0.1(-3)	<9(-4)	1.19 ± 0.01(-3)
^{140}Ba	<5(-4)	<3(-3)	<3(-4)	<3(-3)	1.40 ± 0.05(-3)
^{140}La	<4(-5)	<9(-4)	<1(-4)	<9(-4)	2.8 ± 1.6(-3)
^{141}Ce	<1(-4)	<1(-3)	<2(-4)	<1(-3)	<1(-4)
^{143}Ce	<4(-3)		<4(-3)		<3(-3)
^{144}Ce	<6(-4)		<1(-3)		<5(-4)

TABLE A.6

REACTOR COOLANT ACTIVITIES (OPPD Degassed Sample)
Power Operations - After Refueling

Nuclide	Loop #1 08:16; 2/9/77 INEL (μCi/ml)	Loop #1 08:16; 2/9/77 OPPD (μCi/ml)	Loop #2 08:21; 2/17/77 INEL (μCi/ml)	Loop #2 08:21; 2/17/77 OPPD (μCi/ml)
Activation Nuclides:				
^{24}Na	5.9 ± 0.1(-3)		1.49 ± 0.02(-2)	
^{51}Cr	1.0 ± 0.1(-3)	2.2 ± 0.5(-2)	8.9 ± 2.2(-4)	<5(-3)
^{54}Mn	6.6 ± 0.3(-5)	2.18 ± 0.07(-2)	2.3 ± 1.0(-4)	2.03 ± 0.07(-2)
^{56}Mn	<3(-3)		<7(-3)	
^{59}Fe	8.5 ± 0.6(-5)	2.7 ± 1.1(-3)	1.0 ± 0.3(-4)	5.5 ± 1.1(-3)
^{57}Co	2.3 ± 0.6(-5)	<4(-4)	9.7 ± 2.0(-4)	<4(-4)
^{58}Co	5.69 ± 0.07(-3)	9.7 ± 0.4(-3)	3.48 ± 0.04(-3)	6.3 ± 0.6(-3)
^{60}Co	7.5 ± 0.6(-5)	<6(-4)	9.3 ± 3.0(-5)	<6(-4)
^{65}Zn	<2(-5)	<1(-3)	1.4 ± 0.5(-4)	<1(-3)
^{187}W	9.9 ± 0.4(-3)		8.7 ± 0.4(-3)	
^{239}Np	4.1 ± 0.7(-3)		1.8 ± 1.3(-3)	
Iodine Nuclides:				
^{131}I	8.29 ± 0.03(-2)	1.02 ± 0.01(-1)	8.85 ± 0.08(-2)	9.25 ± 0.09(-2)
^{132}I	5.46 ± 0.07(-2)	4.5 ± 0.4(-2)	5.5 ± 0.1(-2)	4.5 ± 0.4(-2)
^{133}I	1.10 ± 0.01(-1)	1.23 ± 0.01(-1)	1.11 ± 0.02(-1)	9.9 ± 0.1(-2)
^{134}I	3.7 ± 0.3(-2)	2.2 ± 0.2(-2)	3.8 ± 0.4(-2)	1.6 ± 0.2(-2)
^{135}I	7.2 ± 0.1(-2)	5.7 ± 0.2(-2)	7.6 ± 0.2(-2)	5.3 ± 0.2(-2)
Fission Products:				
^{95}Zr	6.6 ± 0.5(-5)	<8(-4)	<3(-4)	<8(-4)
^{95}Nb	4.6 ± 0.3(-5)	<5(-4)	<2(-4)	<5(-4)
^{99}Mo	3.3 ± 1.6(-4)	4.3 ± 1.0(-4)	3.9 ± 1.2(-4)	<9(-5)
^{103}Ru	1.9 ± 0.6(-5)	<5(-4)	<2(-3)	<5(-4)
^{110m}Ag	1.3 ± 0.3(-5)		<2(-3)	
^{124}Sb	4.7 ± 1.2(-6)	4.4 ± 1.4(-3)	3.9 ± 3.9(-5)	<1(-3)
^{129}Te	<3(-3)		<2(-2)	
^{129m}Te	<4(-3)		<5(-3)	
^{131m}Te	<5(-4)		<3(-4)	
^{132}Te	<2(-3)		<2(-3)	
^{134}Cs	8.2 ± 0.1(-4)	<5(-4)	6.51 ± 0.05(-2)	5.08 ± 0.09(-2)
^{136}Cs	5.6 ± 1.3(-5)	<6(-4)	1.6 ± 0.3(-4)	<6(-4)
^{137}Cs	1.18 ± 0.02(-3)	<5(-4)	7.44 ± 0.04(-2)	7.14 ± 0.09(-2)
^{138}Cs	2.30 ± 0.08(-1)	1.12 ± 0.03(-1)	2.9 ± 0.1(-1)	1.39 ± 0.04(-1)
^{139}Ba	3.2 ± 0.1(-2)		4.3 ± 0.5(-2)	
^{140}Ba	<9(-5)	<2(-3)	<6(-3)	<2(-3)
^{140}La	1.4 ± 0.8(-3)	<6(-4)	<7(-3)	<5(-4)
^{141}Ce	<2(-5)	<7(-4)	<8(-4)	<7(-4)
^{143}Ce	<3(-3)		1.3 ± 0.3(-3)	
^{144}Ce	<1(-4)		<4(-3)	

TABLE A.6 (cont'd)

REACTOR COOLANT ACTIVITIES (OPPD Degassed Sample)
Power Operations - After Refueling

	Loop #1 13:38; 2/22/77	
Nuclide	INEL (μCi/ml)	OPPD (μCi/ml)
Activation Nuclides:		
^{24}Na	1.3 ± 0.3(-2)	
^{51}Cr	2.0 ± 0.4(-3)	<5(-3)
^{54}Mn	9.2 ± 1.2(-5)	2.08 ± 0.06(-2)
^{56}Mn	<4(-3)	
^{59}Fe	1.4 ± 0.4(-4)	2.3 ± 1.1(-3)
^{57}Co	<1(-3)	<4(-4)
^{58}Co	1.04 ± 0.05(-2)	1.07 ± 0.06(-2)
^{60}Co	1.5 ± 0.2(-4)	3.3 ± 0.7(-3)
^{65}Zn	<8(-5)	3.8 ± 1.3(-3)
^{187}W	7.5 ± 0.9(-3)	
^{239}Np	3.3 ± 1.9(-3)	
Iodine Nuclides:		
^{131}I	7.3 ± 0.1(-2)	7.58 ± 0.08(-2)
^{132}I	4.72 ± 0.08(-2)	4.7 ± 0.4(-2)
^{133}I	9.60 ± 0.05(-2)	8.84 ± 0.09(-2)
^{134}I	3.4 ± 0.3(-2)	1.2 ± 0.2(-2)
^{135}I	6.3 ± 0.1(-2)	4.6 ± 0.2(-2)
Fission Products:		
^{95}Zr	2.4 ± 0.2(-4)	<7(-4)
^{95}Nb	1.8 ± 0.4(-4)	<4(-4)
^{99}Mo	<1(-3)	<9(-5)
^{103}Ru	9.5 ± 2.4(-5)	4.2 ± 0.7(-3)
^{110m}Ag	<2(-4)	
^{124}Sb	<2(-4)	<1(-3)
^{129}Te	<6(-3)	
^{129m}Te	<6(-3)	
^{131m}Te	<4(-4)	
^{132}Te	<2(-3)	
^{134}Cs	3.9 ± 0.1(-3)	<5(-4)
^{136}Cs	1.1 ± 0.2(-4)	<6(-4)
^{137}Cs	~~4.53 ± 0.07(-3)~~	3.3 ± 0.6(-3)
^{138}Cs	2.64 ± 0.08(-1)	1.16 ± 0.03(-1)
^{139}Ba	4.3 ± 0.3(-2)	
^{140}Ba	<7(-4)	<2(-3)
^{140}La	1.1 ± 0.2(-4)	<5(-4)
^{141}Ce	<2(-3)	<6(-4)
^{143}Ce	<3(-3)	
^{144}Ce	<9(-3)	

TABLE A.7

REACTOR COOLANT GASEOUS ACTIVITIES
Power Operations - After Refueling

Nuclide	Loop #1 08:16; 2/9/77 INEL (μCi/ml)	Loop #1 08:16; 2/9/77 OPPD (μCi/ml)	Loop #2 08:21; 2/17/77 INEL (μCi/ml)	Loop #2 08:21; 2/17/77 OPPD (μCi/ml)	Loop #1 13:38; 2/22/77 INEL (μCi/ml)	Loop #1 13:38; 2/22/77 OPPD (μCi/ml)
^{41}Ar	5.6 ± 0.7(-3)	5.02(-3)	2.6 ± 0.5(-3)	5.05(-3)	2.0 ± 0.3(-3)	3.15(-3)
^{85}Kr	8.4 ± 1.2(-3)		1.8 ± 0.4(-2)		<8(-1)	
^{85m}Kr	1.40 ± 0.05(-1)	1.95(-1)	1.88 ± 0.02(-1)	2.30(-1)	1.7 ± 0.1(-1)	1.88(-1)
^{87}Kr	1.43 ± 0.02(-1)	1.67(-1)	1.92 ± 0.08(-1)	2.05(-1)	1.87 ± 0.06(-1)	1.51(-1)
^{88}Kr	2.6 ± 0.1(-1)	2.54(-1)	3.3 ± 0.2(-1)	3.00(-1)	2.93 ± 0.05(-1)	2.42(-1)
^{131m}Xe	3.1 ± 0.1(-2)		5.2 ± 0.3(-2)		5.2 ± 0.3(-2)	
^{133}Xe	5.0 ± 0.1(0)	1.32(+1)	6.1 ± 0.1(0)	1.43(+1)	5.54 ± 0.03(0)	1.11(+1)
^{133m}Xe	1.16 ± 0.01(-1)	1.27(-1)	1.38 ± 0.02(-1)	1.20(-1)	1.29 ± 0.01(-1)	8.91(-2)
^{135}Xe	8.5 ± 0.3(-1)	1.06(0)	1.05 ± 0.01(0)	1.24(0)	1.04 ± 0.02(0)	9.83(-1)
^{135m}Xe	7.0 ± 0.7(-2)	6.67(-2)	9.5 ± 0.4(-2)	9.32(-2)	9.2 ± 0.2(-2)	6.88(-2)
^{138}Xe	1.5 ± 0.2(-1)	3.22(-2)	1.58 ± 0.07(-1)	5.72(-2)	1.56 ± 0.05(-1)	4.28(-2)

TABLE A.8

CVCS LETDOWN ACTIVITIES - 8/20/76
Power Operations - Before Refueling
CH-8B: Placed in service, 3/24/76; use, 93 days by 8/20/76

Nuclide	Input to CVCS Ion Exch CH-8B 10:56; 8/20/76 (μCi/ml)	Output From Ion Exch CH-8B 10:57; 8/20/76 (μCi/ml)	Output From Filters 10:58; 8/20/76 (μCi/ml)
Activation Nuclides:			
^{24}Na	3.4 ± 0.2(-3)	2.6 ± 2.1(-4)	<1(-4)
^{54}Mn	6.3 ± 0.3(-4)	4.5 ± 0.1(-4)	3.2 ± 1.9(-5)
^{56}Mn	3.5 ± 0.2(-2)	<2(-3)	
^{58}Co	3.44 ± 0.06(-3)	4.0 ± 0.1(-4)	<2(-5)
^{60}Co	6.3 ± 1.9(-5)	2.4 ± 0.4(-5)	2.1 ± 0.7(-6)
^{187}W	<4(-3)		
Iodine Nuclides:			
^{131}I	2.41 ± 0.02(-1)	1.92 ± 0.03(-2)	4.2 ± 2.6(-5)
^{132}I	7.1 ± 0.1(-2)	5.0 ± 0.9(-3)	<8(-4)
^{133}I	1.20 ± 0.01(-1)	7.6 ± 0.3(-3)	<5(-4)
^{134}I	3.1 ± 0.3(-2)	<2(-3)	
^{135}I	7.3 ± 0.2(-2)	9.9 ± 2.4(-3)	<5(-3)
Fission Products:			
^{88}Rb	4.9 ± 0.2(-1)	1.8 ± 0.7(-1)	1.5 ± 0.4(-1)
^{89}Rb	5.0 ± 0.4(-3)	<1(-3)	
^{95}Zr	3.9 ± 1.7(-3)	<2(-5)	
^{95}Nb	7.5 ± 1.6(-5)	<8(-6)	
^{99}Mo	4.6 ± 0.2(-3)	3.1 ± 1.0(-5)	<2(-5)
^{101}Mo	1.8 ± 0.4(-2)	<1(-2)	
^{110m}Ag	1.6 ± 0.8(-5)	<1(-5)	3.2 ± 3.7(-4)
^{134}Cs	1.54 ± 0.01(-2)	1.21 ± 0.02(-3)	4.0 ± 1.0(-5)
^{136}Cs	4.10 ± 0.09(-3)	2.9 ± 0.2(-4)	<2(-5)
^{137}Cs	1.56 ± 0.01(-2)	1.21 ± 0.02(-3)	9.9 ± 4.0(-5)
^{138}Cs	2.9 ± 0.1(-1)	1.0 ± 1.4(-2)	1.9 ± 0.5(-2)
^{140}Ba	9.0 ± 1.7(-4)	1.7 ± 0.3(-4)	<1(-4)
^{140}La	2.6 ± 0.2(-3)	2.0 ± 0.3(-3)	<1(-4)

TABLE A.9

CVCS LETDOWN ACTIVITIES - 8/30/76
Power Operations - Before Refueling
CH-8B: Placed in service, 3/24/76; Use, 103 days by 8/30/76

Nuclide	Input to CVCS Ion Exch CH-8B 09:58; 8/30/76 (μCi/ml)	Output From Ion Exch CH-8B 10:02; 8/30/76 (μCi/ml)	Output From Filters 10:00; 8/30/76 (μCi/ml)
Activation Nuclides:			
^{24}Na	2.2 ± 0.5(-3)	3.6 ± 1.3(-4)	<1(-4)
^{54}Mn	5.2 ± 0.4(-4)	2.4 ± 5.5(-5)	1.5 ± 3.7(-5)
^{56}Mn	<1(-3)		
^{58}Co	2.05 ± 0.09(-3)	<4(-5)	9.2 ± 1.5(-5)
^{60}Co	8.7 ± 3.1(-5)	3.0 ± 1.5(-5)	4.0 ± 2.0(-5)
^{187}W	3.2 ± 0.4(-2)	<2(-4)	
Iodine Nuclides:			
^{131}I	2.40 ± 0.02(-1)	1.18 ± 0.02(-2)	2.0 ± 0.2(-4)
^{132}I	1.01 ± 0.01(-1)	2.0 ± 0.4(-3)	<8(-4)
^{133}I	2.59 ± 0.02(-1)	1.03 ± 0.01(-2)	<6(-4)
^{134}I	5.3 ± 0.7(-2)	<2(-3)	
^{135}I	1.30 ± 0.04(-1)	<1(-2)	
Fission Products:			
^{88}Rb	9.5 ± 0.6(-1)	3.7 ± 1.4(-1)	3.2 ± 3.9(-1)
^{89}Rb	8.5 ± 0.4(-2)	<1(-3)	
^{95}Zr	<2(-4)		
^{95}Nb	<8(-5)		
^{99}Mo	3.3 ± 0.4(-3)	<6(-5)	
^{101}Mo	4.2 ± 1.0(-2)	<2(-2)	
^{110m}Ag	2.6 ± 1.9(-4)	0.6 ± 1.3(-4)	2.1 ± 5.5(-3)
^{134}Cs	1.37 ± 0.06(-3)	3.6 ± 0.2(-4)	9.4 ± 5.5(-5)
^{136}Cs	6.2 ± 1.5(-4)	1.2 ± 0.2(-4)	6 ± 12(-6)
^{137}Cs	1.46 ± 0.04(-3)	3.8 ± 0.3(-4)	9.3 ± 1.6(-5)
^{138}Cs	5.6 ± 0.2(-1)	5.0 ± 2.2(-2)	0.3 ± 1.2(-1)
^{140}Ba	<3(-4)		
^{140}La	1.3 ± 0.5(-3)	<1(-4)	

TABLE A.10

CVCS LETDOWN ACTIVITIES

During Refueling (CH-8A and CH-8B in Series)

Nuclide	Input to CVCS Ion Exchangers 16:45; 10/11/76 (μCi/ml)	Output From Ion Exchangers 17:30; 10/11/76 (μCi/ml)	Output From Filters 16:45; 10/11/76 (μCi/ml)
Activation Nuclides:			
^{51}Cr	<3(-3)	<3(-4)	5.0 ± 1.4(-5)
^{54}Mn	3.4 ± 0.1(-4)	3.5 ± 0.2(-5)	<5(-7)
^{59}Fe	2.19 ± 0.09(-4)	2.6 ± 0.5(-5)	3.8 ± 0.2(-5)
^{57}Co	2.0 ± 0.2(-4)	1.6 ± 0.4(-5)	<3(-6)
^{58}Co	1.24 ± 0.02(-1)	9.17 ± 0.09(-3)	5.4 ± 0.2(-5)
^{60}Co	5.2 ± 0.2(-4)	4.5 ± 0.4(-5)	6.1 ± 0.6(-6)
Iodine Nuclides:			
^{131}I	1.64 ± 0.02(-2)	1.50 ± 0.02(-3)	3.0 ± 0.3(-5)
^{133}I	5.4 ± 2.3(-5)	<7(-5)	<4(-5)
Fission Products:			
^{95}Zr	<5(-3)	<4(-4)	<8(-6)
^{95}Nb	<3(-3)	<2(-4)	<5(-6)
^{99}Mo	9.1 ± 1.3(-4)	5.4 ± 0.3(-5)	1.8 ± 1.0(-5)
^{110m}Ag	<2(-4)	<6(-5)	<3(-5)
^{124}Sb	5.4 ± 0.6(-5)	9.0 ± 2.0(-6)	<3(-5)
^{134}Cs	6.82 ± 0.05(-3)	1.55 ± 0.02(-3)	7.24 ± 0.06(-4)
^{136}Cs	8.5 ± 0.2(-4)	7.3 ± 0.3(-5)	1.0 ± 0.1(-5)
^{137}Cs	6.07 ± 0.08(-3)	1.53 ± 0.02(-3)	7.8 ± 0.1(-4)
^{140}Ba	2.2 ± 0.8(-4)	<3(-4)	<2(-4)
^{140}La	2.7 ± 0.5(-5)	<2(-4)	<2(-5)

TABLE A.11

DEMINERALIZER FUNCTION: REFUELING CAVITY WATER

During Refueling

Nuclide	Input: Cation Demineralizer, CH-10 10:10; 11/8/76 (μCi/ml)	Output: Cation Demineralizer, CH-10 10:15; 11/8/76 (μCi/ml)
Activation Nuclides:		
^{51}Cr	4.4 ± 0.3(-4)	2.0 ± 0.3(-4)
^{54}Mn	1.3 ± 0.2(-4)	1.1 ± 0.2(-5)
^{59}Fe	5.0 ± 1.3(-5)	2.2 ± 0.3(-5)
^{57}Co	4.0 ± 0.1(-5)	<4(-7)
^{58}Co	1.80 ± 0.05(-2)	3.2 ± 0.6(-4)
^{60}Co	1.7 ± 0.4(-4)	2.2 ± 0.5(-5)
^{65}Zn	4.2 ± 0.9(-6)	1.1 ± 0.2(-6)
Iodine Nuclides:		
^{131}I	6.8 ± 1.3(-4)	5.29 ± 0.06(-4)
Fission Products:		
^{95}Zr	1.2 ± 0.3(-5)	4.2 ± 0.3(-6)
^{95}Nb	3.4 ± 1.0(-5)	7.0 ± 0.6(-6)
^{103}Ru	1.8 ± 0.3(-5)	5.3 ± 0.3(-6)
^{110m}Ag	2.7 ± 0.2(-5)	4.2 ± 0.2(-6)
^{124}Sb	1.4 ± 0.1(-4)	1.31 ± 0.08(-4)
^{134}Cs	1.85 ± 0.02(-3)	1.47 ± 0.02(-5)
^{136}Cs	8.5 ± 0.4(-5)	<3(-5)
^{137}Cs	1.96 ± 0.04(-3)	1.70 ± 0.02(-5)
^{140}Ba	5.0 ± 3.9(-4)	<3(-6)
^{141}Ce	<7(-6)	1.9 ± 0.3(-6)
^{144}Ce	<3(-5)	2.0 ± 0.6(-6)

TABLE A.12

CVCS LETDOWN ACTIVITIES - Power Operations - After Refueling (2/16/77)

Nuclide	Input to CVCS Ion Exch CH-8A Dissolved Activity (μCi/ml)	Input to CVCS Ion Exch CH-8A Suspended Solids (μCi/ml)	Output from CVCS Ion Exch CH-8A Dissolved Activity (μCi/ml)	Output from CVCS Ion Exch CH-8A Suspended Solids (μCi/ml)
Activation Nuclides:				
^{24}Na	1.83 ± 0.05(-2)	5.0 ± 0.5(-5)	3.1 ± 0.6(-4)[A]	<1(-6)
^{51}Cr	<2(-2)	3.78 ± 0.01(-4)	<8(-4)	3.58 ± 0.06(-5)
^{54}Mn	1.6 ± 0.3(-4)	2.08 ± 0.09(-5)	<6(-6)	1.29 ± 0.02(-5)
^{56}Mn	<8(-3)	2.8 ± 0.2(-4)	3.7 ± 2.6(-3)[A]	6.5 ± 0.4(-6)
^{59}Fe	<6(-4)	3.3 ± 0.3(-5)	<3(-6)	3.9 ± 0.1(-6)
^{57}Co	<2(-4)	9.4 ± 4.1(-7)	<4(-5)	5.9 ± 0.4(-7)
^{58}Co	1.47 ± 0.04(-3)	3.98 ± 0.07(-4)	9.0 ± 0.3(-5)	9.5 ± 0.2(-5)
^{60}Co	<3(-4)	2.6 ± 0.1(-5)	2.8 ± 0.6(-6)	2.69 ± 0.03(-5)
^{65}Zn	<5(-4)	<5(-6)	<3(-6)	1.6 ± 0.2(-6)
^{187}W	7.2 ± 1.0(-3)	7.5 ± 1.0(-5)	<8(-5)	1.4 ± 0.3(-6)
^{239}Np	<1(-2)	1.1 ± 0.2(-4)	<5(-4)	1.02 ± 0.06(-5)
Iodine Nuclides:				
^{131}I	8.5 ± 0.1(-2)	5.46 ± 0.04(-3)	5.0 ± 4.4(-5)	2.1 ± 0.1(-6)
^{132}I	4.7 ± 0.2(-2)	2.52 ± 0.03(-3)	<3(-3)[A]	1.8 ± 0.6(-6)
^{133}I	9.8 ± 0.2(-2)	6.2 ± 0.3(-3)	<6(-4)[A]	1.10 ± 0.08(-6)
^{134}I	3.4 ± 2.0(-2)	<2(-3)	<2(-3)[A]	<6(-5)
^{135}I	6.5 ± 0.2(-2)	4.35 ± 0.09(-3)	<3(-3)	6.2 ± 2.6(-7)
Fission Products:				
^{88}Rb	7.2 ± 0.4(-1)[A]	*	3.2 ± 0.2(-1)[A]	*
^{89}Rb	*	*	<2(-3)	*
^{95}Zr	<2(-4)	2.2 ± 0.1(-5)	<1(-5)	2.45 ± 0.07(-6)
^{95}Nb	<9(-5)	1.65 ± 0.07(-5)	<6(-6)	3.08 ± 0.08(-6)
^{99}Mo	5.7 ± 3.3(-3)	2.5 ± 0.1(-5)	<4(-5)	1.13 ± 0.02(-6)
^{103}Ru	<4(-3)	6.7 ± 0.8(-6)	<8(-5)	4.8 ± 0.5(-7)
^{110m}Ag	<4(-3)	<2(-5)	2.9 ± 1.2(-6)	1.8 ± 0.6(-6)
^{124}Sb	<4(-3)	2.4 ± 0.5(-6)	<7(-5)	<3(-7)
^{134}Cs	1.90 ± 0.02(-1)	5.5 ± 0.1(-4)	3.10 ± 0.03(-3)	2.5 ± 0.2(-6)
^{136}Cs	1.4 ± 0.5(-4)[A]	<2(-5)	1.4 ± 0.1(-5)	<4(-6)
^{137}Cs	2.19 ± 0.02(-1)	7.08 ± 0.07(-4)	3.79 ± 0.02(-3)	3.13 ± 0.09(-6)
^{138}Cs	2.2 ± 0.1(-1)[A]	*	<2(-3)[A]	*
^{139}Ba	3.6 ± 0.6(-2)	<2(-4)	7.2 ± 0.7(-2)[A]	<6(-6)
^{140}Ba	<2(-2)	<5(-5)	<3(-4)	<9(-7)
^{140}La	2.7 ± 2.2(-2)[A]	4.4 ± 1.2(-6)	< (-4)	2.4 ± 0.7(-7)
^{141}Ce	<4(-4)	<2(-6)	<8(-5)	4.3 ± 0.3(-7)
^{143}Ce	<4(-3)	<3(-4)	<2(-4)	4.0 ± 1.2(-7)
^{144}Ce	<2(-3)	1.1 ± 0.5(-5)	<4(-4)	8.4 ± 1.7(-7)

TABLE A.12 (cont'd)

CVCS LETDOWN ACTIVITIES - Power Operations - After Refueling (2/16/77)

Nuclide	Output from CVCS Filters Dissolved Activity (μCi/ml)	Suspended Solids (μCi/ml)
Activation Nuclides:		
^{24}Na	1.4 ± 0.1(-3) [A]	<1(-7)
^{51}Cr	<1(-3)	1.1 ± 0.2(-6)
^{54}Mn	1.8 ± 0.1(-5) [A]	8.3 ± 0.8(-7)
^{56}Mn	<3(-3) [A]	*
^{59}Fe	<2(-6)	<3(-7)
^{57}Co	<2(-5)	6.3 ± 2.1(-8)
^{58}Co	5.0 ± 0.4(-5)	4.1 ± 0.1(-6)
^{60}Co	3.0 ± 0.5(-6)	1.02 ± 0.03(-6)
^{65}Zn	<2(-6)	<2(-7)
^{187}W	<3(-5)	<1(-6)
^{239}Np	<8(-4)	2.3 ± 0.2(-6)
Iodine Nuclides:		
^{131}I	6.44 ± 0.07(-3)	7.4 ± 0.6(-5)
^{132}I	3.8 ± 0.4(-3) [A]	2.9 ± 0.2(-5)
^{133}I	7.7 ± 0.7(-3) [A]	8.2 ± 0.2(-5)
^{134}I	<4(-3) [A]	*
^{135}I	7.2 ± 0.6(-3) [A]	5.6 ± 0.1(-5)
Fission Products:		
^{88}Rb	3.3 ± 0.2(-1) [A]	*
^{89}Rb	5.2 ± 1.4(-3) [A]	*
^{95}Zr	<2(-5)	3.4 ± 0.4(-7)
^{95}Nb	<7(-6)	3.3 ± 0.3(-7)
^{99}Mo	<2(-5)	1.2 ± 0.3(-7)
^{103}Ru	<4(-4)	7.4 ± 3.2(-8)
^{110m}Ag	<4(-4)	1.5 ± 1.9(-7)
^{124}Sb	<4(-4)	3.6 ± 3.3(-8)
^{134}Cs	1.63 ± 0.01(-2)	7.1 ± 0.2(-6)
^{136}Cs	2.4 ± 0.4(-5)	<3(-7)
^{137}Cs	1.89 ± 0.02(-2)	9.7 ± 1.3(-6)
^{138}Cs	2.7 ± 0.3(-2) [A]	*
^{139}Ba	7.1 ± 0.4(-2) [A]	*
^{140}Ba	<2(-3)	<2(-6)
^{140}La	<6(-4)	1.3 ± 0.3(-7)
^{141}Ce	<3(-5)	9.1 ± 2.2(-8)
^{143}Ce	<3(-4)	<6(-6)
^{144}Ce	<2(-4)	<4(-7)

* : Sample not counted soon enough for these radionuclides.
[A] : Represents total activity (dissolved and suspended) in an unfiltered sample. Filtrate counted too late to observe nuclide.

TABLE A.13

STEAM GENERATOR BLOWDOWN ACTIVITIES

Nuclide	Before Refueling:		After Refueling: Generator A Start Sample: 14:30; 2/8/77 Time: 17.6 hours Volume: 211 Liters		After Refueling: Generator B Start Sample: 14:20; 2/10/77 Time: 17.7 hours Volume: 212 Liters	
	Generator A 09:40; 8/26/76 Volume: 450 ml Total Activity (μCi/ml)	Generator B 09:40; 8/26/76 Volume: 450 ml Total Activity (μCi/ml)	Dissolved Activity (μCi/ml)	Suspended Solids (μCi/ml)	Dissolved Activity (μCi/ml)	Suspended Solids (μCi/ml)
Activation Nuclides:						
^{24}Na	*	*	<9(-11)	<4(-11)	<6(-11)	<6(-11)
^{51}Cr	*	*	<8(-10)	4.0 ± 0.5(-10)	<6(-10)	3.9 ± 0.8(-10)
^{54}Mn	<1(-7)	<1(-7)	1.0 ± 0.5(-9)	3.0 ± 0.2(-10)	6.2 ± 0.6(-10)	7.5 ± 1.9(-11)
^{59}Fe	*	*	<3(-10)	7.7 ± 1.7(-11)	<2(-10)	<2(-10)
^{57}Co	*	*	<2(-10)	6.0 ± 4.2(-12)	<5(-11)	<2(-11)
^{58}Co	*	*	3.57 ± 0.09(-9)	1.63 ± 0.05(-9)	4.4 ± 0.3(-10)	3.4 ± 0.3(-10)
^{60}Co	<2(-7)	<2(-7)	1.09 ± 0.04(-9)	6.9 ± 0.2(-10)	3.3 ± 0.3(-10)	3.1 ± 0.3(-10)
Iodine Nuclides:						
^{131}I	<4(-8)	<2(-7)	8.9 ± 0.6(-10)	4.1 ± 0.6(-11)	1.7 ± 0.3(-10)	<2(-11)
^{133}I	*	*	<4(-10)	<4(-11)	<2(-10)	<8(-11)
Fission Nuclides:						
^{95}Zr	*	*	<4(-10)	<2(-10)	<2(-10)	<1(-10)
^{95}Nb	*	*	<3(-10)	1.1 ± 0.5(-11)	<7(-11)	<6(-11)
^{99}Mo	*	*	<2(-10)	<9(-12)	<5(-11)	<2(-11)
^{103}Ru	*	*	<3(-10)	6.2 ± 0.8(-11)	<2(-10)	<8(-11)
^{124}Sb	*	*	2.8 ± 1.1(-10)	4.5 ± 1.2(-11)	9.0 ± 10.1(-11)	<7(-11)
^{134}Cs	<1(-7)	<1(-7)	7.8 ± 0.1(-9)	3.5 ± 0.2(-10)	2.32 ± 0.08(-9)	6.3 ± 0.9(-11)
^{136}Cs	*	*	8.4 ± 4.4(-11)	<1(-10)	<7(-11)	<6(-11)
^{137}Cs	<1(-6)	<1(-6)	1.23 ± 0.02(-8)	5.1 ± 0.2(-10)	3.61 ± 0.08(-9)	1.4 ± 0.4(-10)

TABLE A.13 (cont'd)

STEAM GENERATOR BLOWDOWN ACTIVITIES

Nuclide	Before Refueling: Generator A 09:40; 8/26/76 Volume: 450 ml Total Activity (μCi/ml)	Before Refueling: Generator B 09:40; 8/26/76 Volume: 450 ml Total Activity (μCi/ml)	After Refueling: Generator A Start Sample: 14:30; 2/8/77 Time: 17.6 hours Volume: 211 Liters Dissolved Activity (μCi/ml)	After Refueling: Generator A Suspended Solids (μCi/ml)	After Refueling: Generator B Start Sample: 14:20; 2/10/77 Time: 17.7 hours Volume: 212 Liters Dissolved Activity (μCi/ml)	After Refueling: Generator B Suspended Solids (μCi/ml)
Beta Only, Nuclides:						
^{3}H	1.7 ± 0.2(-5)	*	*		*	
^{14}C	<3(-8)	*	*		*	
^{32}P	<4(-8)	*	*		*	
^{55}Fe	5.2 ± 0.1 (-6)	*	*		*	
^{63}Ni	<8(-8)	*	*		*	
^{89}Sr	<4(-8)	*	*		*	
^{90}Sr	<2(-8)	*	*		*	
^{91}Y	<1(-8)	*	*		*	

* : Sample not analyzed for these radionuclides.

TABLE A.14

SPENT FUEL POOL ACTIVITIES
Power Operations - Before Refueling

Nuclide	Spent Fuel Pool Temp: 78°F 10:49; 8/23/76 (μCi/ml)	Spent Fuel Pool Temp: 75°F 10:30; 8/31/76 (μCi/ml)
^{54}Mn	1.67 ± 0.03(-5)	1.72 ± 0.06(-5)
^{57}Co	6.4 ± 0.4(-6)	5.6 ± 0.2(-6)
^{58}Co	8.7 ± 0.1(-4)	8.61 ± 0.04(-4)
^{60}Co	4.03 ± 0.04(-5)	4.18 ± 0.06(-5)
^{65}Zn	<1(-6)	1.1 ± 0.2(-6)
^{95}Zr	7.4 ± 2.1(-7)	<9(-7)
^{95}Nb	<4(-7)	<4(-7)
^{99}Mo	9.2 ± 7.1(-7)	<3(-7)
^{110m}Ag	2.2 ± 0.1(-6)	3.3 ± 0.6(-6)
^{124}Sb	3.9 ± 1.0(-7)	5.9 ± 0.9(-7)
^{134}Cs	9.2 ± 0.1(-5)	9.6 ± 0.1(-5)
^{136}Cs	<5(-7)	2.7 ± 1.9(-7)
^{137}Cs	1.93 ± 0.02(-4)	2.06 ± 0.04(-4)

TABLE A.15

SAFETY INJECTION AND REFUELING WATER TANK ACTIVITIES

During Refueling

Nuclide	Before Refueling 10:04; 10/12/76 (μCi/ml)	After Refueling 14:25; 12/2/76 (μCi/ml)
Activation Nuclides:		
^{51}Cr	2.4 ± 0.5(-5)	6.4 ± 1.7(-4)
^{54}Mn	1.18 ± 0.01(-4)	2.19 ± 0.05(-4)
^{59}Fe	7.0 ± 0.4(-6)	4.3 ± 0.8(-5)
^{57}Co	2.9 ± 0.1(-5)	7.3 ± 0.8(-5)
^{58}Co	5.63 ± 0.06(-3)	2.52 ± 0.07(-2)
^{60}Co	1.41 ± 0.01(-4)	3.09 ± 0.05(-4)
^{65}Zn	2.5 ± 0.4(-6)	<4(-5)
Iodine Nuclides:		
^{131}I	7.5 ± 0.2(-4)	<8(-5)
Fission Products:		
^{95}Zr	<2(-4)	3.6 ± 1.4(-5)
^{95}Nb	<2(-4)	5.6 ± 1.5(-5)
^{103}Ru	<4(-5)	<2(-4)
^{110m}Ag	1.3 ± 0.1(-5)	2.6 ± 0.5(-5)
^{124}Sb	2.4 ± 0.2(-6)	8.4 ± 0.5(-5)
^{134}Cs	1.44 ± 0.02(-3)	3.97 ± 0.05(-3)
^{136}Cs	8.7 ± 0.2(-5)	4.4 ± 1.0(-5)
^{137}Cs	2.28 ± 0.02(-3)	4.10 ± 0.08(-3)
^{140}Ba	9.0 ± 2.1(-6)	<6(-4)
^{141}Ce	<2(-6)	<1(-5)
^{144}Ce	<9(-6)	<5(-5)
Beta Only, Nuclides:		
^{3}H	9.25 ± 0.04(-3)	9.91 ± 0.04(-3)
^{14}C	8.1 ± 0.2(-7)	3. ± 1.(-8)
^{32}P	6. ± 5.(-7)	*
^{35}S	4.9 ± 1.2(-7)	1.42 ± 0.04(-5)
^{45}Ca	<6(-8)	*
^{55}Fe	2.23 ± 0.02(-4)	1.27 ± 0.02(-3)
^{63}Ni	1.09 ± 0.02(-3)	3.46 ± 0.02(-3)
^{89}Sr	7.1 ± 0.7(-7)	1.55 ± 0.08(-5)
^{90}Sr	8. ± 2.(-8)	2.7 ± 0.1(-6)
^{91}Y	1.9 ± 0.2(-7)	6.5 ± 0.2(-6)
^{147}Pm	3.1 ± 0.2(-7)	4.4 ± 0.1(-6)

* : Sample not analyzed for these radionuclides.

TABLE A.16

FUEL TRANSFER CANAL AND REFUELING CAVITY ACTIVITIES

During Refueling

Nuclide	Before Fuel Movement Fuel Transfer Canal 18:30; 11/1/76 (μCi/ml)	Before Fuel Movement Refueling Cavity 18:00; 11/1/76 (μCi/ml)	After Fuel Movement Refueling Cavity 09:45; 11/8/76 (μCi/ml)
Activation Nuclides:			
^{51}Cr	1.5 ± 0.7(-4)	6.0 ± 1.2(-4)	6.4 ± 0.4(-4)
^{54}Mn	6.5 ± 0.7(-5)	1.62 ± 0.03(-4)	1.6 ± 0.3(-4)
^{59}Fe	1.1 ± 0.1(-5)	3.9 ± 0.4(-5)	7.3 ± 2.5(-5)
^{57}Co	3.54 ± 0.07(-5)	6.3 ± 0.4(-5)	3.94 ± 0.07(-5)
^{58}Co	1.44 ± 0.04(-2)	3.46 ± 0.08(-2)	1.84 ± 0.05(-2)
^{60}Co	1.12 ± 0.02(-4)	2.00 ± 0.07(-4)	1.7 ± 0.4(-4)
^{65}Zn	1.8 ± 0.7(-6)	<3(-5)	6.6 ± 0.9(-6)
Iodine Nuclides:			
^{131}I	2.2 ± 0.9(-4)	9.7 ± 1.7(-4)	6.2 ± 1.3(-4)
Fission Products:			
^{95}Zr	<2(-3)	<4(-3)	<3(-3)
^{95}Nb	<2(-3)	<2(-3)	1.5 ± 0.3(-5)
^{103}Ru	1.1 ± 0.5(-5)	<1(-4)	1.4 ± 0.3(-5)
^{110m}Ag	2.66 ± 0.09(-5)	1.30 ± 0.06(-4)	5.7 ± 0.2(-5)
^{124}Sb	1.19 ± 0.09(-5)	1.4 ± 0.1(-4)	1.6 ± 0.3(-4)
^{134}Cs	2.06 ± 0.02(-3)	2.59 ± 0.03(-3)	1.85 ± 0.03(-3)
^{136}Cs	1.04 ± 0.06(-4)	1.29 ± 0.08(-4)	6.2 ± 1.1(-5)
^{137}Cs	2.37 ± 0.04(-3)	2.76 ± 0.05(-3)	1.96 ± 0.04(-3)
^{140}Ba	<4(-4)	<4(-4)	<5(-4)
^{141}Ce	<4(-6)	<2(-5)	9.2 ± 3.1(-6)
^{144}Ce	<2(-5)	<7(-5)	2.2 ± 0.8(-5)
Beta, Only, Nuclides:			
^{3}H	9.29 ± 0.05(-3)	1.14 ± 0.02(-2)	1.19 ± 0.02(-2)
^{14}C	1.02 ± 0.02(-6)	1.02 ± 0.02(-6)	4.83 ± 0.04(-6)
^{32}P	5. ± 2.(-7)	2.0 ± 0.5(-5)	1.3 ± 0.3(-5)
^{35}S	7. ± 2.(-7)	1.02 ± 0.03(-5)	1.12 ± 0.03(-5)
^{45}Ca	*	*	<6(-8)
^{55}Fe	2.97 ± 0.02(-4)	3.99 ± 0.02(-4)	8.03 ± 0.04(-4)
^{63}Ni	9.71 ± 0.02(-4)	1.47 ± 0.02(-3)	9.25 ± 0.03(-4)
^{89}Sr	2.2 ± 0.2(-6)	1.15 ± 0.07(-5)	1.04 ± 0.05(-5)
^{90}Sr	2.2 ± 0.3(-7)	1.40 ± 0.07(-6)	1.07 ± 0.06(-6)
^{91}Y	2.4 ± 1.0(-6)	1.73 ± 0.07(-5)	5.2 ± 0.1(-6)
^{147}Pm	*	*	1.87 ± 0.05(-6)

* : Sample not analyzed for these radionuclides.

TABLE A.17

SPENT FUEL POOL ACTIVITIES

During Refueling

Nuclide	Before Filling: Refueling Cavity Temp: 67°F 10:05; 10/14/76 (μCi/ml)	Refueling Cavity Filled Before Fuel Transfer Temp: 58°F 11:05; 10/27/76 (μCi/ml)	Refueling Cavity Filled After Fuel Transfer Temp: 59°F 14:10;11/8/76 (μCi/ml)	After Draining: Refueling Cavity Temp: 57°F 14:10; 12/2/76 (μCi/ml)
Activation Nuclides:				
^{51}Cr	<3(-6)	<5(-6)	7.6 ± 2.5(-5)	<7(-5)
^{54}Mn	1.53 ± 0.02(-5)	1.87 ± 0.07(-5)	8.3 ± 0.3(-5)	8.2 ± 0.2(-5)
^{59}Fe	<4(-6)	<5(-6)	9.5 ± 1.3(-6)	<2(-5)
^{57}Co	4.5 ± 0.1(-6)	5.9 ± 0.4(-6)	3.7 ± 0.1(-5)	4.2 ± 0.2(-5)
^{58}Co	5.07 ± 0.06(-4)	6.2 ± 0.2(-4)	1.54 ± 0.04(-2)	1.38 ± 0.04(-2)
^{60}Co	4.53 ± 0.05(-5)	5.0 ± 0.1(-5)	1.22 ± 0.02(-4)	1.48 ± 0.03(-4)
^{65}Zn	<4(-6)	<4(-6)	2.2 ± 0.9(-6)	5.7 ± 1.9(-6)
Iodine Nuclides:				
^{131}I	<3(-7)	<6(-7)	<6(-6)	<8(-6)
Fission Products:				
^{95}Zr	<4(-5)	<7(-5)	<3(-3)	<2(-3)
^{95}Nb	<2(-5)	<4(-5)	<2(-4)	<1(-3)
^{103}Rh	<8(-6)	<1(-5)	<1(-4)	<1(-4)
^{110m}Ag	1.53 ± 0.09(-6)	2.5 ± 0.3(-6)	4.0 ± 0.6(-6)	8.4 ± 2.7(-6)
^{124}Sb	2.9 ± 0.3(-7)	5.1 ± 1.5(-7)	1.1 ± 0.3(-6)	1.1 ± 0.1(-5)
^{134}Cs	8.2 ± 0.1(-5)	1.21 ± 0.02(-4)	2.52 ± 0.02(-3)	2.55 ± 0.02(-3)
^{136}Cs	<2(-5)	2.3 ± 0.7(-6)	6.6 ± 0.6(-5)	1.5 ± 0.3(-5)
^{137}Cs	1.87 ± 0.02(-4)	2.53 ± 0.05(-4)	2.71 ± 0.05(-3)	2.63 ± 0.04(-3)
^{140}Ba	<3(-5)	<4(-5)	1.3 ± 0.4(-4)	<4(-4)
^{141}Ce	<3(-7)	<2(-6)	<6(-6)	<8(-6)
^{144}Ce	<2(-6)	<6(-6)	<3(-5)	<4(-5)
Beta Only, Nuclides:				
^{3}H	7.20 ± 0.03(-3)	7.32 ± 0.05(-3)	9.21 ± 0.05(-3)	9.12 ± 0.04(-3)
^{14}C	6.7 ± 0.2(-7)	1.16 ± 0.02(-6)	8.8 ± 0.2(-7)	3. ± 1.(-8)
^{32}P	<2(-7)	<1(-6)	<4(-7)	*
^{35}S	<2(-7)	<2(-7)	5. ± 2.(-7)	1.7 ± 0.2(-6)
^{45}Ca	<6(-8)	<6(-8)	<6(-8)	*
^{55}Fe	4.9 ± 0.2(-6)	1.25 ± 0.04(-5)	7.53 ± 0.05(-5)	2.41 ± 0.03(-5)
^{63}Ni	2.55 ± 0.02(-4)	2.79 ± 0.04(-4)	9.74 ± 0.03(-4)	1.02 ± 0.02(-3)
^{89}Sr	<4(-8)	<4(-8)	7.0 ± 0.4(-6)	6.8 ± 0.3(-6)
^{90}Sr	<2(-8)	<2(-8)	7.2 ± 0.5(-7)	8.4 ± 0.5(-7)
^{91}Y	<1(-7)	<8(-8)	2.92 ± 0.08(-6)	3.37 ± 0.09(-6)
^{147}Pm	2.5 ± 0.2(-7)	1.46 ± 0.08(-7)	1.9 ± 0.2(-7)	6.8 ± 0.2(-7)

* : Sample not analyzed for these radionuclides.

TABLE A.18

SPENT FUEL POOL ACTIVITIES

Power Operations - After Refueling

Nuclide	Temp: 55°F 08:45; 2/22/77 Dissolved Activity (μCi/ml)	Suspended Solids (μCi/ml)
Activation Nuclides:		
^{54}Mn	1.25 ± 0.04(-5)	1.26 ± 0.03(-6)
^{59}Fe	1.7 ± 0.3(-6)	3.4 ± 0.2(-7)
^{57}Co	7.3 ± 0.3(-6)	4.5 ± 0.4(-8)
^{58}Co	1.38 ± 0.02(-3)	7.0 ± 0.2(-6)
^{60}Co	4.54 ± 0.07(-5)	2.26 ± 0.04(-6)
^{65}Zn	<2(-6)	<2(-7)
Iodine Nuclides:		
^{131}I	3.31 ± 0.05(-5)	1.7 ± 0.1(-7)
Fission Products:		
^{95}Zr	<1(-4)	7.5 ± 1.5(-8)
^{95}Nb	1.1 ± 0.4(-6)	8.2 ± 0.9(-8)
^{99}Mo	<4(-7)	<8(-9)
^{110m}Ag	<2(-6)	8.2 ± 3.7(-8)
^{124}Sb	<3(-6)	1.8 ± 0.6(-8)
^{134}Cs	7.6 ± 0.2(-5)	4.5 ± 0.3(-7)
^{136}Cs	<6(-5)	4.9 ± 2.1(-8)
^{137}Cs	1.38 ± 0.02(-4)	6.1 ± 0.2(-7)
Beta Only, Nuclides:		
^{3}H	8.94 ± 0.03(-3) [A]	
^{14}C	1.8 ± 0.1(-7)	
^{55}Fe	1.12 ± 0.02(-4)	
^{63}Ni	3.42 ± 0.02(-4)	
^{89}Sr	5.5 ± 0.6(-7)	
^{90}Sr	1.7 ± 0.2(-7)	
^{91}Y	4.8 ± 0.2(-7)	

[A]: Total activity, suspended solids activity not determined in beta analysis.

TABLE A.19

REACTOR COOLANT DRAIN TANK ACTIVITIES (TANK WD-1)

Power Operations - After Refueling

Nuclide	09:15; 2/10/77 Dissolved Activity (μCi/ml)	Suspended Solids (μCi/ml)
Activation Nuclides:		
^{24}Na	1.18 ± 0.03(-3)	1.1 ± 0.2(-5)
^{51}Cr	<6(-3)	7.0 ± 0.7(-5)
^{54}Mn	1.56 ± 0.06(-4)	2.27 ± 0.05(-5)
^{56}Mn	<3(-4)	<1(-6)
^{59}Fe	<6(-6)	1.12 ± 0.06(-5)
^{57}Co	3.66 ± 0.34(-5)	3.1 ± 2.9(-6)
^{58}Co	3.46 ± 0.04(-3)	2.19 ± 0.02(-4)
^{60}Co	8.77 ± 0.29(-5)	3.54 ± 0.05(-5)
^{65}Zn	1.78 ± 0.60(-5)	<5(-6)
^{187}W	1.17 ± 0.11(-3)	2.7 ± 0.5(-5)
^{239}Np	<2(-3)	<1(-3)
Iodine Nuclides:		
^{131}I	3.49 ± 0.06(-2)	8.31 ± 0.08(-3)
^{132}I	9.60 ± 0.67(-4)	9.1 ± 0.8(-5)
^{133}I	1.99 ± 0.03(-2)	4.7 ± 0.2(-3)
^{134}I	<7(-5)	*
^{135}I	3.09 ± 0.09(-3)	8.9 ± 0.2(-4)
Fission Products:		
^{95}Zr	<1(-4)	4.2 ± 0.4(-6)
^{95}Nb	<7(-5)	1.32 ± 0.05(-5)
^{99}Mo	5.11 ± 0.57(-5)	2.2 ± 1.0(-6)
^{103}Ru	<6(-5)	1.5 ± 0.4(-6)
^{110m}Ag	1.67 ± 0.39(-5)	<2(-6)
^{124}Sb	8.34 ± 1.05(-6)	3.9 ± 2.3(-7)
^{134}Cs	2.33 ± 0.02(-3)	2.2 ± 0.1(-5)
^{136}Cs	2.70 ± 0.48(-5)	<5(-6)
^{137}Cs	2.98 ± 0.04(-3)	3.16 ± 0.05(-5)
^{139}Ba	<3(-5)	*
^{140}Ba	<2(-4)	<5(-6)
^{140}La	<3(-3)	<7(-4)
^{141}Ce	<1(-5)	<2(-6)
^{143}Ce	<1(-3)	<4(-4)
^{144}Ce	<6(-5)	<6(-6)
Beta Only Nuclides:		
^{3}H	1.13 ± 0.01(-1)	**
^{14}C	<1(-7)	
^{55}Fe	1.54 ± 0.02(-4)	
^{63}Ni	5.39 ± 0.02(-4)	
^{89}Sr	1.33 ± 0.07(-5)	
^{90}Sr	1.38 ± 0.09(-6)	
^{91}Y	8.3 ± 0.9(-7)	

* : Sample not counted soon enough for these radionuclides.
** : Suspended solids activity not determination in beta analysis.

TABLE A.20

SPENT REGENERANT AND WASTE HOLDUP TANKS
Power Operations - Before Refueling

Nuclide	Spent Regenerant Tank WD-13B 10:48; 8/24/76 (μCi/ml)	Spent Regenerant Tank WD-13A 09:42; 8/26/76 (μCi/ml)	Spent Regenerant Tank WD-13B 11:50; 8/31/76 (μCi/ml)	Waste Holdup Tank WD-4B 13:34; 8/24/76 (μCi/ml)	HSL Analysis: Duplicate Sample 11:40; 9/2/76 (μCi/ml)	Waste Holdup Tank WD-4A 11:40; 9/2/76 (μCi/ml)
Activation Nuclides:						
^{24}Na	2.5 ± 0.9(-6)	2.9 ± 0.3(-6)	3.4 ± 0.3(-5)	4.9 ± 2.4(-7)		<3(-5)
^{51}Cr	1.7 ± 0.2(-5)	<6(-4)	<5(-4)	<7(-4)		<5(-2)
^{54}Mn	6.53 ± 0.07(-5)	1.65 ± 0.07(-5)	1.20 ± 0.06(-4)	8.0 ± 1.2(-6)	4 ± 2(-4)	1.7 ± 0.8(-4)
^{59}Fe	3.1 ± 0.4(-6)	<1(-6)	3.8 ± 2.6(-5)	<6(-7)		<2(-4)
^{57}Co	3.7 ± 0.3(-6)	1.2 ± 0.6(-6)	1.0 ± 0.1(-5)	<8(-7)		1.1 ± 0.4(-3)
^{58}Co	4.81 ± 0.05(-4)	2.96 ± 0.01(-4)	2.74 ± 0.01(-4)	6.2 ± 0.1(-5)	2.60±0.08(-3)	1.4 ± 0.1(-3)
^{60}Co	1.08 ± 0.01(-4)	2.3 ± 0.1(-5)	6.74 ± 0.06(-4)	6.1 ± 0.4(-6)	1.8 ± 0.7(-4)	2.6 ± 0.7(-4)
^{65}Zn	5.4 ± 1.9(-6)	<1(-6)	2.4 ± 0.4(-5)	<6(-7)		<1(-4)
^{187}W	<2(-5)	<7(-6)	2.0 ± 0.4(-4)	<4(-6)		<4(-4)
^{239}Np	<2(-5)	<4(-4)	3.6 ± 1.8(-4)	<5(-4)		3.5 ± 1.2(-3)
Iodine Nuclides:						
^{131}I	7.77 ± 0.07(-4)	6.80 ± 0.07(-3)	1.65 ± 0.01(-2)	5.06 ± 0.07(-3)	1.21±0.04(0)	1.27 ± 0.01(0)
^{133}I	1.43 ± 0.03(-4)	2.72 ± 0.02(-4)	1.26 ± 0.01(-2)	6.9 ± 0.2(-5)		2.29 ± 0.01(-1)
^{135}I	6.5 ± 0.4(-5)	<2(-6)	3.04 ± 0.02(-3)	<2(-6)		2.7 ± 0.4(-3)

TABLE A.20 (cont.)

SPENT REGENERANT AND WASTE HOLDUP TANKS
Power Operations - Before Refueling

Nuclide	Spent Regenerant Tank WD-13B 10:48; 8/24/76 (μCi/ml)	Spent Regenerant Tank WD-13A 09:42; 8/26/76 (μCi/ml)	Spent Regenerant Tank WD-13B 11:50; 8/31/76 (μCi/ml)	Waste Holdup Tank WD-4B 13:34; 8/24/76 (μCi/ml)	Waste Holdup HSL Analysis: Duplicate Sample 11:40; 9/2/76 (μCi/ml)	Waste Holdup Tank WD-4A 11:40; 9/2/76 (μCi/ml)
Fission Products:						
^{95}Zr	2.8 ± 0.6(-6)	<3(-6)	<4(-6)	<3(-6)		<3(-4)
^{95}Nb	3.1 ± 0.4(-6)	<1(-6)	<2(-6)	<1(-6)		<1(-4)
^{99}Mo	6.2 ± 1.5(-6)	1.11 ± 0.08(-5)	1.9 ± 0.2(-5)	9.2 ± 0.7(-6)		1.9 ± 1.0(-3)
^{103}Ru	<7(-7)	<2(-5)	<2(-5)	<6(-5)		<2(-3)
^{110m}Ag	2.0 ± 0.3(-6)	<2(-5)	8.6 ± 1.5(-6)	<6(-5)		<2(-3)
^{124}Sb	<3(-7)	<2(-7)	<4(-7)	<4(-7)		<6(-5)
^{126}Sb	5.7 ± 2.7(-7)	<2(-5)	<5(-6)	<5(-6)		<2(-3)
^{127}Sb	<1(-5)	<4(-6)	<1(-5)	1.8 ± 0.6(-6)		<3(-4)
^{134}Cs	3.32 ± 0.02(-4)	2.23 ± 0.01(-3)	1.49 ± 0.01(-3)	1.78 ± 0.08(-3)	1.53±0.05(-1)	1.56 ± 0.05(-1)
^{136}Cs	5.32 ± 0.08(-5)	2.00 ± 0.04(-4)	3.95 ± 0.06(-4)	2.34 ± 0.05(-4)	4.5 ± 0.2(-2)	4.6 ± 0.2(-2)
^{137}Cs	3.48 ± 0.01(-4)	2.57 ± 0.02(-3)	1.49 ± 0.01(-3)	1.86 ± 0.06(-3)	1.62±0.05(-1)	1.47 ± 0.05(-1)
^{140}Ba	<5(-6)	<7(-5)	<2(-5)	<2(-4)		<6(-3)
^{140}La	6.8 ± 1.5(-7)	4.6 ± 1.2(-7)	<2(-7)	1.1 ± 0.2(-6)		6.7 ± 2.9(-5)
^{141}Ce	<1(-6)	<1(-6)	<5(-6)	<2(-6)		<3(-4)
^{143}Ce	4.7 ± 1.0(-6)	<1(-4)	<1(-4)	2.6 ± 0.8(-6)		<1(-2)
^{144}Ce	<6(-6)	<6(-6)	<2(-5)	<6(-6)		<1(-3)

TABLE A.21

LIQUID RADWASTE TANK ACTIVITIES
During Refueling

Nuclide	Waste Holdup Tank WD-4A 20:30; 10/11/76 (μCi/ml)	Waste Holdup Tank WD-4B 23:55; 10/11/76 (μCi/ml)	Waste Holdup Tank WD-4C 16:00; 10/11/76 (μCi/ml)	Spent Regenerant Tank WD-13B 12:35; 10/13/76 (μCi/ml)	Monitor Tank WD-22B 07:30; 10/14/76 (μCi/ml)
Activation Nuclides:					
^{24}Na	<3(-5)	<2(-5)	<8(-6)	<4(-4)	<4(-7)
^{51}Cr	<8(-3)	1.4 ± 0.3(-4)	1.1 ± 0.5(-4)	4.0 ± 0.7(-5)	<6(-7)
^{54}Mn	1.18 ± 0.07(-4)	4.04 ± 0.04(-4)	6.45 ± 0.08(-4)	3.04 ± 0.04(-4)	4.2 ± 0.3(-7)
^{59}Fe	<3(-5)	3.7 ± 0.3(-5)	6.9 ± 0.3(-5)	1.40 ± 0.04(-5)	<2(-7)
^{57}Co	<4(-5)	1.3 ± 0.4(-5)	1.8 ± 0.2(-5)	1.39 ± 0.04(-5)	8.6 ± 1.9(-8)
^{58}Co	5.5 ± 0.1(-4)	8.2 ± 0.1(-3)	5.82 ± 0.07(-3)	3.35 ± 0.07(-3)	4.7 ± 0.1(-6)
^{60}Co	1.1 ± 0.4(-4)	2.5 ± 0.2(-4)	1.4 ± 0.1(-4)	5.34 ± 0.09(-4)	1.05 ± 0.04(-6)
^{65}Zn	<3(-5)	<2(-5)	<2(-5)	8.5 ± 0.8(-6)	<2(-7)
^{187}W	<2(-4)	<2(-3)	8.0 ± 2.0(-5)	<5(-4)	<1(-6)
^{239}Np	<6(-3)	<2(-3)	<2(-3)	<7(-5)	<4(-7)
Iodine Nuclides:					
^{131}I	2.37 ± 0.03(-2)	7.69 ± 0.05(-3)	9.83 ± 0.07(-3)	5.46 ± 0.08(-4)	6.4 ± 0.3(-7)
^{133}I	<3(-3)	<5(-4)	1.2 ± 0.5(-5)	<4(-4)	<4(-7)
^{135}I	<2(-4)	<4(-5)	<6(-5)	<8(-2)	<1(-5)
Fission Products:					
^{95}Zr	<6(-5)	<6(-4)	<4(-4)	5.3 ± 0.8(-6)	<6(-7)
^{95}Nb	<4(-5)	<4(-4)	<3(-4)	6.3 ± 0.4(-6)	<3(-7)
^{99}Mo	<4(-5)	5.8 ± 0.4(-5)	4.5 ± 0.4(-5)	4.9 ± 0.9(-6)	<3(-8)
^{103}Ru	<3(-3)	<5(-4)	<4(-4)	<5(-5)	<2(-7)
^{110m}Ag	<3(-3)	<5(-4)	<4(-4)	4.4 ± 0.8(-6)	<2(-7)
^{124}Sb	<3(-3)	5.3 ± 1.9(-6)	<4(-4)	4.8 ± 1.1(-7)	<2(-7)
^{126}Sb	<3(-3)	<5(-4)	2.3 ± 1.1(-5)	<4(-5)	<2(-7)
^{127}Sb	<9(-5)	2.6 ± 1.2(-5)	<7(-4)	<4(-4)	<8(-7)
^{134}Cs	5.84 ± 0.05(-4)	1.12 ± 0.01(-2)	9.31 ± 0.08(-3)	1.92 ± 0.03(-3)	3.8 ± 0.2(-6)
^{136}Cs	2.13 ± 0.03(-3)	6.43 ± 0.08(-4)	9.48 ± 0.08(-4)	9.2 ± 0.2(-5)	2.5 ± 0.6(-7)
^{137}Cs	5.92 ± 0.08(-2)	1.21 ± 0.02(-2)	9.5 ± 0.1(-3)	2.18 ± 0.03(-3)	4.59 ± 0.07(-6)
^{140}Ba	<9(-3)	5.7 ± 1.7(-5)	<2(-3)	<2(-4)	<7(-7)
^{140}La	<4(-3)	2.8 ± 0.2(-5)	3.0 ± 0.2(-5)	1.80 ± 0.04(-5)	<3(-7)
^{141}Ce	<7(-5)	<2(-5)	<2(-5)	<4(-6)	<7(-8)
^{143}Ce	<2(-3)	<6(-4)	<4(-4)	<3(-5)	<2(-7)
^{144}Ce	<3(-4)	<6(-5)	<8(-5)	<2(-5)	<3(-7)

TABLE A.22

WASTE HOLDUP TANK ACTIVITIES (Tank WD-4A)

Power Operations - After Refueling

Nuclide	10:40; 2/11/77 Dissolved Activity (μCi/ml)	10:40; 2/11/77 Suspended Solids (μCi/ml)	Analyzed by HSL 10:40; 2/11/77 Total Activity (μCi/ml)
Activation Nuclides:			
^{24}Na	<1(-6)	<3(-6)	
^{51}Cr	<8(-4)	<5(-4)	
^{54}Mn	1.29 ± 0.03(-4)	6.6 ± 0.1(-6)	1.7 ± 0.2(-4)
^{59}Fe	<3(-6)	1.8 ± 0.2(-6)	
^{57}Co	6.42 ± 1.62(-6)	4.9 ± 3.9(-7)	
^{58}Co	8.67 ± 0.08(-4)	6.25 ± 0.07(-5)	1.17 ± 0.03(-3)
^{60}Co	3.26 ± 0.12(-5)	1.36 ± 0.02(-5)	4.4 ± 0.4(-5)
^{65}Zn	<2(-5)	3.1 ± 0.8(-7)	
^{187}W	<3(-5)	<5(-6)	
^{239}Np	2.33 ± 1.06(-4)	<3(-4)	
Iodine Nuclides:			
^{131}I	9.44 ± 0.05(-3)	2.54 ± 0.09(-3)	1.4 ± 0.2(-2)
^{133}I	2.20 ± 0.29(-4)	6.9 ± 0.2(-5)	
^{135}I	<5(-6)	<4(-5)	
Fission Products:			
^{95}Zr	<2(-6)	4.7 ± 0.7(-7)	
^{95}Nb	<9(-6)	7.2 ± 0.7(-7)	
^{99}Mo	4.42 ± 3.13(-5)	2.7 ± 0.5(-6)	
^{103}Ru	<1(-4)	2.8 ± 0.7(-7)	
^{110m}Ag	1.13 ± 1.11(-6)	5.6 ± 1.0(-7)	
^{124}Sb	1.34 ± 0.53(-6)	4.7 ± 3.4(-8)	
^{134}Cs	1.08 ± 0.01(-2)	2.11 ± 0.03(-5)	1.18 ± 0.02(-2)
^{136}Cs	9.63 ± 3.68(-6)	<2(-6)	
^{137}Cs	1.23 ± 0.01(-2)	2.81 ± 0.04(-5)	1.57 ± 0.03(-2)
^{140}Ba	<4(-4)	<4(-6)	
^{140}La	1.86 ± 0.11(-5)	3.4 ± 1.1(-7)	
^{141}Ce	<7(-6)	<3(-7)	
^{143}Ce	<2(-4)	<1(-4)	
^{144}Ce	7.31 ± 2.02(-5)	<2(-6)	
Beta Only, Nuclides:			
^{3}H	5.56 ± 0.02(-2)	**	
^{14}C	6.89 ± 0.03(-6)		
^{55}Fe	1.43 ± 0.02(-4)		
^{63}Ni	1.51 ± 0.02(-4)		
^{89}Sr	4.9 ± 0.2(-5)		
^{90}Sr	1.13 ± 0.06(-6)		
^{91}Y	1.7 ± 0.1(-7)		

** : Suspended solids activities not determined on these samples.

TABLE A.23

SPENT REGENERANT TANK ACTIVITIES

Power Operations - After Refueling

Nuclide	Tank WD-13A 10:20; 2/11/77 Dissolved Activity (μCi/ml)	Tank WD-13A 10:20; 2/11/77 Suspended Solids (μCi/ml)	Tank WD-13B 09:11; 2/14/77 Dissolved Activity (μCi/ml)	Tank WD-13B 09:11; 2/14/77 Suspended Solids (μCi/ml)
Activation Nuclides:				
^{24}Na	<6(-8)	<4(-6)	2.61 ± 0.09(-5)	<8(-6)
^{51}Cr	<5(-6)	3.2 ± 0.4(-6)	<4(-5)	5.9 ± 0.4(-6)
^{54}Mn	2.11 ± 0.04(-5)	5.7 ± 0.1(-6)	1.96 ± 0.04(-5)	1.09 ± 0.02(-5)
^{59}Fe	6.7 ± 1.3(-7)	7.4 ± 0.5(-7)	5.8 ± 1.3(-7)	1.3 ± 0.3(-6)
^{57}Co	2.8 ± 0.2(-6)	2.3 ± 0.3(-7)	2.6 ± 0.2(-6)	8.0 ± 0.3(-7)
^{58}Co	3.07 ± 0.07(-4)	3.09 ± 0.08(-5)	3.36 ± 0.06(-4)	1.12 ± 0.03(-4)
^{60}Co	<6(-8)	1.16 ± 0.02(-5)	1.29 ± 0.01(-5)	2.22 ± 0.04(-5)
^{65}Zn	<1(-7)	<5(-7)	<1(-6)	2.2 ± 0.7(-7)
^{187}W	<2(-5)	<7(-6)	2.0 ± 0.3(-5)	<2(-5)
^{239}Np	<4(-6)	<4(-6)	5.3 ± 1.9(-6)	<9(-7)
Iodine Nuclides:				
^{131}I	3.07 ± 0.06(-5)	1.56 ± 0.06(-5)	4.89 ± 0.06(-4)	1.09 ± 0.08(-6)
^{133}I	<3(-6)	2.4 ± 0.6(-6)	3.22 ± 0.06(-4)	<4(-6)
^{135}I	<2(-7)	*	1.59 ± 0.06(-4)	<1(-4)
Fission Products:				
^{95}Zr	<1(-5)	2.0 ± 0.4(-7)	<2(-5)	5.8 ± 0.8(-7)
^{95}Nb	<6(-6)	4.0 ± 0.5(-7)	<8(-6)	1.0 ± 0.1(-6)
^{99}Mo	<2(-7)	3.2 ± 2.0(-8)	5.6 ± 4.1(-7)	8.9 ± 2.0(-8)
^{103}Ru	<3(-6)	<2(-7)	<5(-6)	<5(-7)
^{110m}Ag	1.9 ± 1.1(-7)	<2(-7)	6.8 ± 1.2(-7)	1.1 ± 0.1(-6)
^{124}Sb	<3(-6)	<2(-7)	2.4 ± 0.8(-7)	<5(-7)
^{134}Cs	2.52 ± 0.02(-4)	1.5 ± 0.1(-6)	2.35 ± 0.02(-4)	6.2 ± 0.2(-6)
^{136}Cs	<1(-6)	<2(-6)	3.1 ± 1.1(-7)	<5(-6)
^{137}Cs	3.05 ± 0.04(-4)	2.14 ± 0.07(-6)	2.87 ± 0.03(-4)	8.5 ± 0.2(-6)
^{140}Ba	<1(-5)	<5(-7)	<2(-5)	<2(-6)
^{140}La	<2(-6)	<3(-6)	<2(-5)	<6(-7)
^{141}Ce	<4(-7)	5.0 ± 1.8(-8)	<5(-7)	<2(-7)
^{143}Ce	<1(-6)	4.7 ± 4.7(-6)	1.1 ± 1.1(-5)	<3(-7)
^{144}Ce	<2(-6)	<2(-7)	<2(-6)	2.6 ± 1.0(-7)
Beta Only, Nuclides:				
^{3}H	4.14 ± 0.02(-3)	**	***	
^{14}C	2.5 ± 0.1(-7)			
^{55}Fe	1.72 ± 0.02(-4)			
^{63}Ni	8.57 ± 0.02(-5)			
^{89}Sr	5.0 ± 0.7(-7)			
^{90}Sr	1.5 ± 0.3(-7)			
^{91}Y	4. ± 2.(-8)			

* : Sample not counted soon enough for these radionuclides.
** : Sample not analyzed for suspended solids.
*** : Beta analysis not done on this sample.

TABLE A.24

MONITOR AND HOTEL TANKS
Power Operations - Before Refueling

Nuclide	Monitor Tank WD-22A 15:15; 8/24/76 (μCi/ml)	Monitor Tank WD-22B 13:25; 8/24/76 (μCi/ml)	Monitor Tank WD-22B 08:17; 9/1/76 (μCi/ml)	Hotel Tank WD-15A 10:43; 8/24/76 (μCi/ml)
Activation Nuclides:				
^{51}Cr	<2(-6)	<2(-6)	<2(-6)	<4(-7)
^{54}Mn	5.5 ± 0.6(-7)	8.5 ± 0.7(-7)	2.7 ± 0.3(-7)	3.2 ± 0.5(-7)
^{59}Fe	<2(-7)	<2(-7)	<1(-7)	<2(-7)
^{57}Co	<2(-7)	<2(-7)	<9(-5)	<1(-7)
^{58}Co	3.4 ± 0.1(-6)	5.0 ± 0.1(-6)	8.8 ± 0.4(-7)	1.42 ± 0.08(-6)
^{60}Co	8.2 ± 2.6(-7)	7.4 ± 1.7(-7)	5.9 ± 0.4(-7)	5.9 ± 0.7(-7)
^{65}Zn	<2(-7)	<2(-7)	<1(-7)	<2(-7)
^{187}W	<4(-7)	<4(-7)	<2(-7)	<3(-7)
^{239}Np	<2(-6)	<1(-6)	<1(-6)	<3(-7)
Iodine Nuclides:				
^{131}I	1.42 ± 0.02(-5)	1.20 ± 0.04(-5)	2.2 ± 0.2(-5)	1.2 ± 0.5(-7)
^{133}I	5.9 ± 1.9(-7)	1.8 ± 1.2(-7)	8.9 ± 0.7(-7)	<1(-7)

TABLE A.24 (cont.)

MONITOR AND HOTEL TANKS

Power Operations - Before Refueling

Nuclide	Monitor Tank WD-22A 15:15; 8/24/76 (μCi/ml)	Monitor Tank WD-22B 13:25; 8/24/76 (μCi/ml)	Monitor Tank WD-22B 08:17; 9/1/76 (μCi/ml)	Hotel Tank WD-15A 10:43; 8/24/76 (μCi/ml)
Fission Products:				
^{95}Zr	<2(-7)	<3(-7)	<1(-7)	<2(-7)
^{95}Nb	<1(-7)	<1(-7)	<5(-8)	<8(-8)
^{99}Mo	<2(-7)	<2(-7)	<8(-5)	<8(-8)
^{103}Ru	<2(-7)	<2(-7)	<1(-7)	<9(-8)
^{110m}Ag	<2(-7)	<2(-7)	<9(-8)	<1(-7)
^{124}Sb	<2(-7)	<2(-7)	<9(-8)	<1(-7)
^{126}Sb	<2(-7)	<2(-7)	<9(-8)	<9(-8)
^{127}Sb	<3(-7)	<4(-7)	<1(-7)	<2(-7)
^{134}Cs	1.11 ± 0.05(-6)	3.7 ± 0.2(-6)	2.87 ± 0.08(-6)	1.1 ± 0.1(-6)
^{136}Cs	<1(-7)	7.1 ± 0.6(-7)	3.5 ± 0.3(-7)	<8(-8)
^{137}Cs	1.4 ± 0.2(-6)	4.1 ± 0.2(-6)	3.37 ± 0.07(-6)	1.69 ± 0.09(-6)
^{140}Ba	<8(-7)	<9(-7)	<4(-7)	<4(-7)
^{140}La	<1(-6)	<8(-7)	<9(-7)	<2(-7)
^{141}Ce	<4(-7)	<3(-7)	<2(-4)	<2(-7)
^{143}Ce	<6(-7)	<4(-7)	<4(-7)	<1(-7)
^{144}Ce	<2(-6)	1.2 ± 0.4(-6)	<7(-4)	<7(-7)

TABLE A.24A

MONITOR TANK ACTIVITIES

Comparison Between INEL and OPPD Measurements

Nuclide	Tank WD-22A INEL 15:15; 8/24/76 (μCi/ml)	Tank WD-22A OPPD 15:10; 8/24/76 (μCi/ml)	Tank WD-22B INEL 13:25; 8/24/76 (μCi/ml)	Tank WD-22B OPPD 13:20; 8/24/76 (μCi/ml)
Activation Nuclides:				
^{51}Cr	<2(-6)	1.7 ± 0.5(-6)	<2(-6)	<1(-6)
^{54}Mn	5.5 ± 0.6(-7)	8.3 ± 0.5(-7)	8.5 ± 0.7(-7)	7.9 ± 0.6(-7)
^{59}Fe	<2(-7)	<8(-8)	<2(-7)	<7(-8)
^{57}Co	<2(-7)	<9(-8)	<2(-7)	<1(-7)
^{58}Co	3.4 ± 0.1(-6)	4.6 ± 0.1(-6)	5.0 ± 0.1(-6)	5.6 ± 0.1(-6)
^{60}Co	8.2 ± 2.6(-7)	6.0 ± 0.5(-7)	7.4 ± 1.7(-7)	6.6 ± 0.6(-7)
^{65}Zn	<2(-7)	<8(-8)	<2(-7)	<7(-8)
Iodine Nuclides:				
^{131}I	1.42 ± 0.02(-5)	1.57 ± 0.02(-5)	1.20 ± 0.04(-5)	1.25 ± 0.02(-5)
^{133}I	5.9 ± 1.9(-7)	<9(-8)	1.8 ± 1.2(-7)	<2(-7)
Fission Products:				
^{95}Zr	<2(-7)	<8(-8)	<3(-7)	<1(-7)
^{95}Nb	<1(-7)	<4(-8)	<1(-7)	<2(-7)
^{99}Mo	<2(-7)	<3(-8)	<2(-7)	<3(-8)
^{103}Ru	<2(-7)	<8(-8)	<2(-7)	<1(-7)
^{124}Sb	<2(-7)	<6(-8)	<2(-7)	<6(-8)
^{134}Cs	1.11 ± 0.05(-6)	1.03 ± 0.07(-6)	3.7 ± 0.2(-6)	3.2 ± 0.1(-6)
^{136}Cs	<1(-7)	<6(-8)	7.1 ± 0.6(-7)	5.1 ± 0.6(-7)
^{137}Cs	1.4 ± 0.2(-6)	1.56 ± 0.07(-6)	4.1 ± 0.2(-6)	4.4 ± 0.1(-6)
^{140}Ba	<8(-7)	<2(-7)	<9(-7)	<3(-7)
^{140}La	<1(-6)	<3(-8)	<8(-7)	<3(-8)
^{141}Ce	<4(-7)	<2(-7)	<3(-7)	<2(-7)

TABLE A.24A (cont'd)

MONITOR TANK ACTIVITIES

Comparison Between INEL and OPPD Measurements

Nuclide	Tank WD-22B INEL 08:17; 9/1/76 (μCi/ml)	Tank WD-22B OPPD 08:17; 9/1/76 (μCi/ml)	Tank WD-22B INEL 07:30; 10/14/76 (μCi/ml)	Tank WD-22B OPPD 07:30; 10/14/76 (μCi/ml)
Activation Nuclides:				
^{51}Cr	<2(-6)	<2(-6)	<6(-7)	<8(-7)
^{54}Mn	2.7 ± 0.3(-7)	2.7 ± 0.4(-7)	4.2 ± 0.3(-7)	4.7 ± 0.5(-7)
^{59}Fe	<1(-7)	<6(-8)	<2(-7)	<8(-8)
^{57}Co	<9(-5)	1.4 ± 0.2(-6)	8.6 ± 1.9(-8)	3.2 ± 0.5(-7)
^{58}Co	8.8 ± 0.4(-7)	9.7 ± 0.6(-7)	4.7 ± 0.1(-6)	4.9 ± 0.1(-6)
^{60}Co	5.9 ± 0.4(-7)	2.9 ± 0.3(-7)	1.05 ± 0.04(-6)	8.5 ± 0.6(-7)
^{65}Zn	<1(-7)	<6(-8)	<2(-7)	<9(-8)
Iodine Nuclides:				
^{131}I	2.2 ± 0.2(-5)	2.34 ± 0.02(-5)	6.4 ± 0.3(-7)	7.3 ± 0.7(-7)
^{133}I	8.9 ± 0.7(-7)	7.8 ± 0.6(-7)	<4(-7)	<1(-7)
Fission Products:				
^{95}Zr	<1(-7)	<5(-8)	<6(-7)	<9(-8)
^{95}Nb	<5(-8)	<4(-8)	<3(-7)	<5(-8)
^{99}Mo	<8(-5)	<7(-8)	<3(-8)	<4(-8)
^{103}Ru	<1(-7)	<8(-8)	<2(-7)	<1(-7)
^{124}Sb	<9(-8)	<6(-8)	<2(-7)	<6(-8)
^{134}Cs	2.87 ± 0.08(-6)	2.60 ± 0.09(-6)	3.8 ± 0.2(-6)	3.8 ± 0.1(-6)
^{136}Cs	3.5 ± 0.3(-7)	2.8 ± 0.4(-7)	2.5 ± 0.6(-7)	2.1 ± 0.6(-7)
^{137}Cs	3.37 ± 0.07(-6)	2.9 ± 0.1(-6)	4.59 ± 0.07(-6)	4.6 ± 0.1(-6)
^{140}Ba	<4(-7)	<3(-7)	<7(-7)	<3(-7)
^{140}La	<9(-7)	<3(-8)	<3(-7)	<3(-8)
^{141}Ce	<2(-4)	<5(-7)	<7(-8)	<3(-7)

TABLE A.25

HOTEL TANK ACTIVITIES

Power Operations - After Refueling

	Tank WD-15A 12:04; 2/17/77		Tank WD-15B 13:15; 2/15/77	
Nuclide	Dissolved Activity (μCi/ml)	Suspended Solids (μCi/ml)	Dissolved Activity (μCi/ml)	Suspended Solids (μCi/ml)
Activation Nuclides:				
^{51}Cr	<5(-7)	<8(-8)	<1(-6)	<1(-7)
^{54}Mn	7.2 ± 6.1(-8)	1.07 ± 0.09(-7)	3.1 ± 0.4(-7)	1.0 ± 0.1(-7)
^{59}Fe	<1(-7)	<1(-7)	<1(-7)	<3(-8)
^{57}Co	<3(-7)	<4(-7)	<1(-6)	<5(-8)
^{58}Co	1.76 ± 0.08(-6)	5.3 ± 0.2(-7)	1.7 ± 0.1(-6)	5.9 ± 0.2(-7)
^{60}Co	2.1 ± 0.4(-7)	1.7 ± 0.1(-7)	3.1 ± 0.5(-7)	2.2 ± 0.1(-7)
^{65}Zn	<1(-7)	<9(-8)	<9(-8)	<8(-8)
^{187}W	<3(-7)	<2(-7)	<5(-7)	<2(-7)
^{239}Np	<3(-7)	<6(-8)	<8(-7)	<8(-8)
Iodine Nuclides:				
^{131}I	2.5 ± 0.6(-7)	1.7 ± 0.5(-8)	<1(-7)	3.5 ± 0.7(-8)
^{133}I	<2(-7)	<6(-8)	<4(-7)	<6(-8)
Fission Products:				
^{95}Zr	<2(-7)	<8(-8)	<3(-7)	<7(-8)
^{95}Nb	<9(-8)	<5(-8)	<1(-7)	<4(-8)
^{99}Mo	<3(-7)	<3(-7)	<1(-6)	4.4 ± 2.4(-8)
^{103}Ru	<2(-7)	<3(-8)	<4(-7)	<5(-8)
^{110m}Ag	<2(-7)	<3(-8)	<4(-7)	<5(-8)
^{124}Sb	<2(-7)	<3(-8)	<4(-7)	<5(-8)
^{134}Cs	9.5 ± 0.2(-6)	1.7 ± 0.1(-7)	3.41 ± 0.06(-5)	3.5 ± 0.1(-7)
^{136}Cs	<9(-8)	<5(-8)	<1(-7)	<4(-7)
^{137}Cs	1.12 ± 0.02(-5)	2.2 ± 0.1(-7)	4.16 ± 0.05(-5)	5.1 ± 0.3(-7)
^{140}Ba	<8(-7)	<1(-7)	<2(-6)	<2(-7)
^{140}La	<2(-7)	<4(-8)	<2(-7)	<5(-8)
^{141}Ce	<2(-7)	<6(-7)	<2(-6)	<9(-8)
^{143}Ce	<1(-7)	<2(-8)	<3(-7)	<9(-8)
^{144}Ce	<6(-6)	<3(-6)	<9(-6)	7.6 ± 3.8(-7)
Beta Only, Nuclides:				
^{3}H	4. ± 1.(-6)	**	8. ± 1.(-6)	**
^{14}C	1.6 ± 0.1(-7)		1.13 ± 0.02(-6)	
^{55}Fe	2.0 ± 0.1(-6)		3.0 ± 0.1(-6)	
^{63}Ni	2.0 ± 0.2(-7)		2.0 ± 0.1(-6)	
^{89}Sr	3. ± 1.(-8)		1.3 ± 0.9(-8)	
^{90}Sr	<1(-8)		<8(-9)	
^{91}Y	<2(-8)		<2(-8)	

** : Sample not analyzed for suspended solids.

TABLE A.26

MONITOR TANK ACTIVITIES

Power Operations - After Refueling

Nuclide	Tank WD-22A INEL 08:49; 2/19/77 (μCi/ml)	Tank WD-22A OPPD 08:50; 2/19/77 (μCi/ml)	Tank WD-22B INEL 09:30; 2/10/77 (μCi/ml)	Tank WD-22B OPPD 09:25; 2/10/77 (μCi/ml)
Activation Nuclides:				
^{51}Cr	<7(-7)	<7(-7)	<1(-6)	<2(-6)
^{54}Mn	<2(-7)	1.0 ± 0.2(-7)	2.0 ± 0.2(-7)	2.9 ± 0.3(-7)
^{59}Fe	<2(-7)	<4(-8)	<2(-7)	<6(-8)
^{57}Co	<5(-8)	<8(-8)	<6(-8)	<2(-7)
^{58}Co	1.45 ± 0.06(-6)	1.61 ± 0.07(-6)	2.44 ± 0.07(-6)	3.3 ± 0.1(-6)
^{60}Co	3.9 ± 0.4(-7)	1.2 ± 0.2(-7)	3.6 ± 0.4(-7)	2.2 ± 0.3(-7)
^{65}Zn	<2(-7)	<5(-8)	<2(-7)	<7(-8)
^{187}W	<4(-7)	*	<5(-7)	*
^{239}Np	<5(-7)	*	<7(-7)	*
Iodine Nuclides:				
^{131}I	5.2 ± 0.5(-7)	5.0 ± 0.6(-7)	2.5 ± 0.6(-7)	4.1 ± 0.8(-7)
^{133}I	<7(-7)	<8(-8)	<2(-6)	<2(-7)
Fission Products:				
^{95}Zr	<3(-7)	<6(-8)	<3(-7)	<1(-7)
^{95}Nb	<2(-7)	<3(-8)	<2(-7)	<5(-8)
^{99}Mo	<4(-8)	<2(-8)	<6(-8)	<3(-8)
^{103}Ru	<2(-7)	<8(-8)	<5(-7)	<2(-7)
^{110m}Ag	<2(-7)	*	<5(-7)	*
^{124}Sb	<2(-7)	<6(-8)	<5(-7)	<6(-8)
^{134}Cs	3.25 ± 0.07(-6)	3.4 ± 0.1(-6)	7.0 ± 0.1(-6)	8.7 ± 0.2(-6)
^{136}Cs	<2(-7)	<5(-8)	<2(-7)	<5(-8)
^{137}Cs	4.1 ± 0.1(-6)	4.3 ± 0.1(-6)	8.5 ± 0.3(-6)	1.04 ± 0.02(-5)
^{140}Ba	<6(-7)	<2(-7)	<2(-6)	<3(-7)
^{140}La	<4(-7)	<3(-8)	9.2 ± 4.0(-7)	<3(-8)
^{141}Ce	<8(-8)	<2(-7)	<1(-7)	<2(-7)
^{143}Ce	<2(-7)	*	<3(-7)	*
^{144}Ce	<4(-7)	*	<5(-7)	*
Beta Only, Nuclides:				
^{3}H	***	***	5.36 ± 0.02(-3)	***
^{14}C			3.5 ± 0.1(-7)	
^{55}Fe			5. ± 1.(-7)	
^{63}Ni			4.3 ± 0.1(-7)	
^{89}Sr			<2(-8)	
^{90}Sr			<1(-8)	
^{91}Y			<2(-8)	

* : Radionuclide not determined by OPPD.
***: Beta analysis not done on sample.

TABLE A.27

MONITOR TANK WD-22B ACTIVITIES WITH VARYING TANK LEVEL

Power Operations - After Refueling

Nuclide	Prior to Release Tank Level: 89 inches (5930 gal) INEL 08:41; 2/14/77 (μCi/ml)	Prior to Release OPPD 08:10; 2/14/77 (μCi/ml)	Start of Release Tank Level: 77 inches (5130 gal) 11:15; 2/14/77 (μCi/ml)	Middle of Release Tank Level: 37 inches (2470 gal) 12:40; 2/14/77 (μCi/ml)	End of Release Tank Level: 21 inches (1400 gal) 13:15; 2/14/77 (μCi/ml)
Activation Nuclides:					
^{51}Cr	<7(-7)	<2(-6)	<1(-6)	<2(-6)	5.5 ± 2.3(-7)
^{54}Mn	2.7 ± 0.3(-7)	2.3 ± 0.3(-7)	1.6 ± 0.4(-7)	2.7 ± 0.6(-7)	1.9 ± 0.3(-7)
^{59}Fe	<2(-7)	<5(-8)	<3(-7)	<3(-7)	1.3 ± 0.5(-7)
^{57}Co	<4(-8)	<2(-7)	<9(-8)	<6(-8)	<3(-6)
^{58}Co	2.62 ± 0.08(-6)	3.6 ± 0.1(-6)	2.5 ± 0.2(-6)	2.2 ± 0.1(-6)	1.94 ± 0.08(-6)
^{60}Co	3.9 ± 0.5(-7)	3.5 ± 0.4(-7)	4.5 ± 0.4(-7)	3.9 ± 0.9(-7)	3.3 ± 0.5(-7)
^{65}Zn	<2(-7)	<7(-8)	<3(-7)	<3(-7)	<2(-7)
^{187}W	<6(-7)	*	<1(-6)	<8(-7)	<6(-7)
^{239}Np	<5(-7)	*	<7(-7)	<1(-6)	<8(-7)
Iodine Nuclides:					
^{131}I	5.0 ± 0.7(-7)	<2(-7)	6.7 ± 0.6(-7)	4.7 ± 0.8(-7)	5.8 ± 0.7(-7)
^{133}I	<2(-7)	<2(-7)	<2(-6)	<7(-7)	<7(-7)
Fission Products:					
^{95}Zr	<3(-7)	<9(-8)	1.9 ± 0.6(-7)	<4(-7)	<3(-7)
^{95}Nb	<2(-7)	<6(-8)	<3(-7)	<3(-7)	<2(-7)
^{99}Mo	1.7 ± 1.6(-8)	<4(-8)	<8(-8)	<6(-8)	<2(-6)
^{103}Ru	<2(-7)	<2(-7)	<7(-7)	<3(-7)	<3(-7)
^{110m}Ag	<2(-7)	*	<7(-7)	<3(-7)	<3(-7)
^{124}Sb	<2(-7)	<6(-8)	<7(-7)	<3(-7)	<3(-7)
^{134}Cs	4.1 ± 0.1(-6)	8.6 ± 0.2(-6)	4.5 ± 0.2(-6)	4.5 ± 0.2(-6)	4.3 ± 0.1(-6)
^{136}Cs	<2(-7)	<4(-8)	<3(-7)	<3(-7)	<2(-7)
^{137}Cs	5.2 ± 0.1(-6)	1.10 ± 0.02(-5)	5.2 ± 0.4(-6)	5.3 ± 0.2(-6)	5.4 ± 0.2(-6)
^{140}Ba	<8(-7)	<4(-7)	<3(-6)	<2(-6)	<2(-6)
^{140}La	<3(-7)	<3(-8)	<5(-7)	<7(-7)	<5(-7)
^{141}Ce	<6(-8)	<3(-7)	<2(-7)	<1(-7)	<4(-6)
^{143}Ce	<2(-7)	*	<3(-7)	<4(-7)	<3(-7)
^{144}Ce	<3(-7)	*	<7(-7)	<5(-7)	<2(-5)

* : Radionuclide not determined by OPPD.

TABLE A.28

EVAPORATOR PROCESSING WASTE HOLDUP TANK WD-4B

Nuclide	Waste Holdup WD-4B 13:34; 8/24/76 (μCi/ml)	Evaporator Distillate 13:43; 8/25/76 (μCi/ml)	Evaporator Concentrate 13:43; 8/25/76 (μCi/ml)
Activation Nuclides:			
^{24}Na	4.9 ± 2.4(-7)	<6(-8)	5.9 ± 1.2(-5)
^{51}Cr	<7(-4)	<2(-6)	<3(-2)
^{54}Mn	8.0 ± 1.2(-6)	3.1 ± 0.5(-7)	7.7 ± 0.5(-4)
^{59}Fe	<6(-7)	<1(-7)	<1(-4)
^{57}Co	<8(-7)	<3(-4)	<1(-4)
^{58}Co	6.2 ± 0.1(-5)	1.04 ± 0.04(-6)	1.68 ± 0.01(-2)
^{60}Co	6.1 ± 0.4(-6)	5.2 ± 0.3(-7)	1.57 ± 0.07(-3)
^{65}Zn	<6(-7)	<1(-7)	<9(-5)
^{187}W	<4(-6)	<3(-7)	<7(-4)
^{239}Np	<5(-4)	5.8 ± 2.6(-7)	7.9 ± 5.6(-4)
Iodine Nuclides:			
^{131}I	5.06 ± 0.07(-3)	2.82 ± 0.04(-5)	8.07 ± 0.03(-1)
^{133}I	6.9 ± 0.2(-5)	<9(-8)	8.6 ± 0.5(-3)
^{135}I	<2(-6)	<2(-7)	4.6 ± 1.2(-4)
Fission Products:			
^{95}Zr	<3(-6)	<2(-7)	<4(-4)
^{95}Nb	<1(-6)	<8(-8)	<2(-4)
^{99}Mo	9.2 ± 0.7(-6)	<3(-4)	9.1 ± 1.1(-4)
^{103}Ru	<6(-5)	<8(-8)	<4(-3)
^{110m}Ag	<6(-5)	<8(-8)	<4(-3)
^{124}Sb	<4(-7)	<8(-8)	<2(-5)
^{126}Sb	<5(-6)	2.0 ± 0.5(-7)	<3(-3)
^{127}Sb	1.8 ± 0.6(-6)	<2(-7)	<5(-4)
^{134}Cs	1.78 ± 0.08(-3)	1.32 ± 0.06(-6)	3.37 ± 0.01(-1)
^{136}Cs	2.34 ± 0.05(-4)	<8(-8)	2.90 ± 0.02(-2)
^{137}Cs	1.86 ± 0.06(-3)	1.68 ± 0.06(-6)	3.81 ± 0.02(-3)
^{140}Ba	<2(-4)	<3(-7)	<1(-2)
^{140}La	1.1 ± 0.2(-6)	<1(-6)	<1(-5)
^{141}Ce	<2(-6)	<5(-4)	<2(-4)
^{143}Ce	2.6 ± 0.8(-6)	<6(-7)	1.5 ± 0.4(-3)
^{144}Ce	<6(-6)	<5(-6)	<9(-4)

TABLE A.29

EVAPORATOR PROCESSING SPENT REGENERANT TANK WD-13A

Nuclide	Spent Regen WD-13A 09:42; 8/26/76 (μCi/ml)	Evaporator Distillate 09:28; 8/26/76 (μCi/ml)	Evaporator Concentrate 09:28; 8/26/76 (μCi/ml)
Activation Nuclides:			
^{24}Na	2.9 ± 0.3(-6)	<2(-7)	9.9 ± 3.3(-5)
^{51}Cr	<6(-4)	<4(-6)	<6(-2)
^{54}Mn	1.65 ± 0.07(-5)	2.0 ± 1.3(-7)	2.1 ± 1.5(-4)
^{59}Fe	<1(-6)	<3(-7)	<2(-4)
^{57}Co	1.2 ± 0.06(-6)	<4(-5)	1.6 ± 1.6(-3)
^{58}Co	2.96 ± 0.01(-4)	8.9 ± 1.7(-7)	1.45 ± 0.02(-2)
^{60}Co	2.3 ± 0.1(-5)	5.8 ± 1.5(-7)	1.36 ± 0.06(-3)
^{65}Zn	<1(-6)	<3(-7)	<2(-4)
^{187}W	<7(-6)	<7(-7)	<7(-4)
^{239}Np	<4(-4)	<3(-6)	<4(-2)
Iodine Nuclides:			
^{131}I	6.80 ± 0.07(-3)	2.13 ± 0.08(-5)	8.32 ± 0.04(-1)
^{133}I	2.72 ± 0.02(-4)	<5(-7)	8.1 ± 0.5(-3)
^{135}I	<2(-6)	2.7 ± 4.0(-6)	1.2 ± 0.2(-3)
Fission Products:			
^{95}Zr	<3(-6)	<4(-7)	<5(-4)
^{95}Nb	<1(-6)	<2(-7)	<2(-4)
^{99}Mo	1.11 ± 0.08(-5)	<5(-5)	7.3 ± 1.5(-4)
^{103}Ru	<2(-5)	<5(-7)	<3(-3)
^{110m}Ag	<2(-5)	<5(-7)	<3(-3)
^{124}Sb	<2(-7)	<5(-7)	<7(-5)
^{126}Sb	<2(-5)	<5(-7)	<3(-3)
^{127}Sb	<4(-6)	<5(-7)	<6(-4)
^{134}Cs	2.23 ± 0.01(-3)	2.7 ± 0.4(-6)	3.48 ± 0.02(-1)
^{136}Cs	2.00 ± 0.04(-4)	<2(-7)	2.96 ± 0.06(-2)
^{137}Cs	2.57 ± 0.02(-3)	2.7 ± 0.5(-6)	3.86 ± 0.03(-1)
^{140}Ba	<7(-5)	<2(-6)	<1(-2)
^{140}La	4.6 ± 1.2(-7)	<2(-6)	<3(-5)
^{141}Ce	<1(-6)	<8(-5)	<3(-4)
^{143}Ce	<1(-4)	<1(-6)	<1(-2)
^{144}Ce	<6(-6)	<1(-5)	<1(-3)

TABLE A.30

EVAPORATOR PROCESSING SPENT REGENERANT TANK WD-13B

Nuclide	Spent Regen WD-13B 11:50; 8/31/76 (μCi/ml)	Evaporator Distillate 13:45; 8/31/76 (μCi/ml)	Evaporator Concentrate 13:40; 8/31/76 (μCi/ml)
Activation Nuclides:			
^{24}Na	3.4 ± 0.3(-5)	<1(-7)	<7(-5)
^{51}Cr	<5(-4)	<5(-6)	<2(-2)
^{54}Mn	1.20 ± 0.06(-4)	1.79 ± 0.03(-6)	6.3 ± 0.3(-4)
^{59}Fe	3.8 ± 2.6(-5)	<3(-7)	<2(-4)
^{57}Co	1.0 ± 0.1(-5)	<1(-4)	1.4 ± 0.5(-4)
^{58}Co	2.74 ± 0.01(-4)	6.99 ± 0.07(-6)	3.45 ± 0.04(-3)
^{60}Co	6.74 ± 0.06(-4)	1.07 ± 0.06(-6)	2.39 ± 0.03(-3)
^{65}Zn	2.4 ± 0.4(-5)	<2(-7)	<1(-4)
^{187}W	2.0 ± 0.4(-4)	2.3 ± 0.9(-6)	1.1 ± 0.3(-2)
^{239}Np	3.6 ± 1.8(-4)	5.7 ± 1.4(-6)	<2(-2)
Iodine Nuclides:			
^{131}I	1.65 ± 0.01(-2)	1.87 ± 0.02(-5)	2.98 ± 0.03(-1)
^{133}I	1.26 ± 0.01(-2)	1.1 ± 0.2(-6)	1.62 ± 0.02(-2)
^{135}I	3.04 ± 0.02(-3)	<4(-7)	3.3 ± 0.2(-3)
Fission Products:			
^{95}Zr	<4(-6)	<6(-7)	<2(-4)
^{95}Nb	<2(-6)	<3(-7)	<7(-5)
^{99}Mo	1.9 ± 0.2(-5)	<1(-4)	1.6 ± 0.6(-3)
^{103}Ru	<2(-5)	<1(-7)	<2(-3)
^{110m}Ag	8.6 ± 1.5(-6)	2.6 ± 1.8(-7)	7.5 ± 5.0(-5)
^{124}Sb	<4(-7)	<1(-7)	<2(-3)
^{126}Sb	<5(-6)	<1(-7)	<2(-3)
^{127}Sb	<1(-5)	<8(-7)	<2(-4)
^{134}Cs	1.49 ± 0.01(-3)	4.05 ± 0.05(-6)	1.87 ± 0.01(-1)
^{136}Cs	3.95 ± 0.06(-4)	4.5 ± 1.0(-7)	1.35 ± 0.01(-2)
^{137}Cs	1.49 ± 0.01(-3)	4.76 ± 0.04(-6)	2.09 ± 0.01(-1)
^{140}Ba	<2(-5)	<5(-7)	3.0 ± 0.5(-3)
^{140}La	<2(-7)	<2(-6)	3.9 ± 1.5(-3)
^{141}Ce	<5(-6)	<2(-4)	<2(-4)
^{143}Ce	<1(-4)	<1(-6)	5.2 ± 0.6(-2)
^{144}Ce	<2(-5)	<9(-4)	<8(-4)

TABLE A.31

SCRAPINGS FROM RADWASTE EVAPORATOR BUNDLE

During Refueling

Nuclide	Inlet - Outlet End 11:30; 10/13/76 (μCi)	U-Tube End 11:30; 10/13/76 (μCi)
Activation Nuclides:		
^{24}Na	<6(-2)	<3(-2)
^{51}Cr	5.5 ± 1.3(-2)	1.4 ± 0.7(-2)
^{54}Mn	1.33 ± 0.02(0)	1.64 ± 0.03(0)
^{59}Fe	1.4 ± 0.1(-2)	3.0 ± 0.6(-2)
^{57}Co	3.86 ± 0.06(-2)	3.98 ± 0.08(-2)
^{58}Co	4.93 ± 0.06(0)	5.2 ± 0.1(0)
^{60}Co	1.36 ± 0.01(0)	1.21 ± 0.02(0)
^{65}Zn	1.7 ± 0.2(-2)	2.3 ± 0.7(-2)
^{187}W	<4(-1)	<7(-1)
^{239}Np	<2(-2)	<6(-3)
Iodine Nuclides:		
^{131}I	2.3 ± 0.5(-2)	3.1 ± 0.5(-3)
^{133}I	<2(-1)	3.3 ± 0.9(-3)
^{135}I	<2(-1)	<9(-2)
Fission Products:		
^{95}Zr	5.6 ± 1.7(-3)	7.2 ± 2.2(-3)
^{95}Nb	1.0 ± 0.2(-2)	8.7 ± 1.6(-3)
^{99}Mo	<2(-3)	<2(-3)
^{103}Ru	<2(-1)	<4(-2)
^{110m}Ag	1.7 ± 0.3(-2)	2.3 ± 0.3(-2)
^{124}Sb	2.5 ± 0.6(-3)	3.5 ± 0.8(-3)
^{126}Sb	<2(-1)	<4(-2)
^{127}Sb	<3(-1)	<6(-1)
^{134}Cs	3.04 ± 0.09(0)	1.50 ± 0.03(0)
^{136}Cs	2.6 ± 0.2(-2)	7.0 ± 1.9(-3)
^{137}Cs	3.71 ± 0.05(0)	1.87 ± 0.02(0)
^{140}Ba	<6(-1)	<2(-1)
^{140}La	1.8 ± 0.4(-3)	2.1 ± 0.5(-3)
^{141}Ce	<3(-3)	<3(-3)
^{143}Ce	<7(-3)	6.8 ± 4.2(-3)
^{144}Ce	<2(-2)	<2(-2)

TABLE A.32

EVAPORATOR FUNCTION TEST

Feed Samples: Spent Regenerant Tank WD-13A

Nuclide	Sample 144** Prior to Test Tank Level: 80" (5330 gal) 07:30; 2/18/77 Total Activity (μCi/ml)	First Process Sample Tank Level: 70" (4670 gal) 10:30; 2/18/77 Dissolved Activity (μCi/ml)	Suspended Solids (μCi/ml)
Activation Nuclides:			
^{24}Na	1.6 ± 0.2(-5)	1.3 ± 0.1(-5)	*
^{51}Cr	7.2 ± 0.9(-5)	<2(-4)	7.68 ± 0.09(-5)
^{54}Mn	3.19 ± 0.07(-5)	2.08 ± 0.06(-5)	1.29 ± 0.03(-5)
^{59}Fe	7.4 ± 0.7(-6)	<2(-6)	9.2 ± 0.2(-6)
^{57}Co	2.2 ± 0.7(-6)	2.7 ± 1.1(-6)	6.9 ± 0.3(-7)
^{58}Co	5.82 ± 0.07(-4)	3.67 ± 0.03(-4)	2.25 ± 0.04(-4)
^{60}Co	2.8 ± 0.2(-5)	9.5 ± 0.8(-6)	1.91 ± 0.02(-5)
^{65}Zn	<3(-6)	<2(-6)	3.8 ± 0.7(-7)
^{187}W	<9(-5)	<3(-5)	<3(-5)
^{239}Np	<2(-4)	4.1 ± 1.5(-5)	<3(-6)
Iodine Nuclides:			
^{131}I	1.11 ± 0.02(-3)	8.89 ± 0.15(-4)	1.05 ± 0.03(-5)
^{133}I	2.19 ± 0.07(-4)	1.73 ± 0.16(-4)	*
^{135}I	<5(-5)	<3(-6)	*
Fission Products:			
^{95}Zr	4.0 ± 1.0(-6)	<2(-5)	3.3 ± 0.4(-6)
^{95}Nb	<3(-5)	<8(-6)	4.6 ± 0.5(-6)
^{99}Mo	<2(-6)	<2(-6)	2.9 ± 1.0(-6)
^{103}Ru	<7(-5)	<6(-5)	9.6 ± 1.0(-7)
^{110m}Ag	5.2 ± 0.7(-6)	<6(-5)	3.7 ± 0.1(-6)
^{124}Sb	6.5 ± 4.2(-7)	<5(-5)	2.0 ± 0.3(-7)
^{134}Cs	1.47 ± 0.02(-3)	1.20 ± 0.01(-3)	2.09 ± 0.06(-5)
^{136}Cs	<3(-5)	2.1 ± 1.6(-6)	3.6 ± 0.4(-7)
^{137}Cs	1.72 ± 0.02(-3)	1.44 ± 0.02(-3)	2.78 ± 0.05(-5)
^{140}Ba	<3(-4)	<2(-4)	1.9 ± 2.1(-6)
^{140}La	<9(-5)	<7(-5)	1.6 ± 0.3(-6)
^{141}Ce	<3(-6)	<4(-6)	5.1 ± 0.3(-7)
^{143}Ce	<5(-5)	<4(-5)	<9(-7)
^{144}Ce	<1(-5)	<2(-5)	8.2 ± 2.4(-7)

TABLE A.32 (cont'd)

EVAPORATOR FUNCTION TEST

Feed Samples: Spent Regenerant Tank WD-13A

	Second Process Sample Tank Level: 60" (4000 gal) 12:50; 2/18/77		Third Process Sample Tank Level: 50" (3330 gal) 16:00; 2/18/77	
Nuclide	Dissolved Activity (μCi/ml)	Suspended Solids (μCi/ml)	Dissolved Activity (μCi/ml)	Suspended Solids (μCi/ml)
Activation Nuclides:				
^{24}Na	1.3 ± 0.1(-5)	*	<3(-6)	<7(-6)
^{51}Cr	<2(-4)	6.01 ± 0.07(-5)	<2(-4)	6.3 ± 0.2(-5)
^{54}Mn	2.03 ± 0.05(-5)	9.7 ± 0.3(-6)	1.96 ± 0.07(-5)	1.01 ± 0.02(-5)
^{59}Fe	<3(-6)	7.47 ± 0.08(-6)	<4(-6)	6.8 ± 0.4(-6)
^{57}Co	2.7 ± 0.7(-6)	4.9 ± 0.2(-7)	<2(-6)	5.6 ± 0.4(-7)
^{58}Co	3.90 ± 0.06(-4)	1.66 ± 0.03(-4)	4.06 ± 0.08(-4)	1.75 ± 0.04(-4)
^{60}Co	1.1 ± 0.2(-5)	1.43 ± 0.02(-5)	1.06 ± 0.07(-5)	1.47 ± 0.03(-5)
^{65}Zn	<2(-6)	1.7 ± 0.6(-7)	<3(-6)	4.0 ± 1.3(-7)
^{187}W	1.8 ± 0.4(-5)	<6(-6)	<6(-5)	<3(-5)
^{239}Np	5.0 ± 4.3(-6)	1.4 ± 0.7(-6)	6.7 ± 3.3(-6)	<2(-6)
Iodine Nuclides:				
^{131}I	1.12 ± 0.01(-3)	8.7 ± 0.1(-6)	1.08 ± 0.02(-3)	1.00 ± 0.02(-6)
^{133}I	1.9 ± 0.1(-4)	<8(-7)	1.67 ± 0.04(-4)	1.9 ± 0.4(-6)
^{135}I	<1(-5)	*	<8(-5)	*
Fission Products:				
^{95}Zr	<3(-5)	2.3 ± 0.3(-6)	<3(-5)	2.5 ± 0.2(-6)
^{95}Nb	<2(-5)	2.7 ± 0.3(-6)	<2(-5)	2.8 ± 0.3(-6)
^{99}Mo	<1(-6)	2.3 ± 0.7(-7)	<2(-6)	2.6 ± 0.4(-7)
^{103}Ru	<3(-5)	7.6 ± 0.8(-7)	<3(-5)	7.6 ± 1.5(-7)
^{110m}Ag	2.0 ± 0.4(-6)	3.1 ± 0.2(-6)	6.1 ± 1.9(-6)	3.0 ± 0.1(-6)
^{124}Sb	<3(-5)	1.6 ± 0.2(-7)	<3(-5)	1.6 ± 0.4(-7)
^{134}Cs	1.30 ± 0.02(-3)	1.39 ± 0.03(-5)	1.36 ± 0.02(-3)	1.90 ± 0.05(-5)
^{136}Cs	4.2 ± 4.7(-7)	4.0 ± 0.9(-7)	<2(-5)	<7(-6)
^{137}Cs	1.52 ± 0.02(-3)	1.82 ± 0.02(-5)	1.62 ± 0.02(-3)	2.47 ± 0.03(-5)
^{140}Ba	<1(-4)	<2(-6)	<2(-4)	<2(-6)
^{140}La	<9(-5)	<6(-6)	<9(-5)	<8(-7)
^{141}Ce	<2(-6)	5.4 ± 1.7(-8)	<4(-6)	3.6 ± 0.5(-7)
^{143}Ce	<2(-5)	<4(-7)	<5(-5)	<4(-7)
^{144}Ce	<9(-6)	4.9 ± 1.4(-7)	<2(-5)	9.0 ± 2.6(-7)

* : Sample not counted soon enough for these radionuclides.
**: Sample not filtered.

TABLE A.33

EVAPORATOR FUNCTION TEST

Evaporator Distillate Samples (Feed: Spent Regenerant Tank WD-13A)

	First Process Sample 10:20; 2/18/77	
Nuclide	Dissolved Activity (μCi/ml)	Suspended Solids (μCi/ml)
Activation Nuclides:		
^{24}Na	<8(-9)	<1(-8)
^{51}Cr	<7(-8)	<2(-8)
^{54}Mn	2.2 ± 0.5(-8)	9.0 ± 1.1(-9)
^{59}Fe	<3(-8)	<1(-8)
^{57}Co	1.0 ± 0.2(-8)	<1(-9)
^{58}Co	7.1 ± 0.1(-7)	7.7 ± 0.3(-8)
^{60}Co	5.3 ± 0.4(-8)	3.4 ± 0.2(-8)
^{65}Zn	<3(-8)	<1(-8)
^{187}W	<1(-7)	<2(-8)
^{239}Np	1.3 ± 1.1(-8)	<1(-8)
Iodine Nuclides:		
^{131}I	1.28 ± 0.06(-7)	5.3 ± 0.9(-9)
^{133}I	1.7 ± 1.0(-8)	<6(-9)
^{135}I	<7(-8)	<7(-8)
Fission Products:		
^{95}Zr	<5(-8)	<2(-8)
^{95}Nb	<3(-8)	<6(-9)
^{99}Mo	<5(-9)	<1(-9)
^{103}Ru	<3(-8)	<6(-9)
^{110m}Ag	<3(-8)	<6(-9)
^{124}Sb	<3(-8)	<6(-9)
^{134}Cs	3.7 ± 0.1(-7)	4.3 ± 0.3(-8)
^{136}Cs	<3(-8)	<6(-9)
^{137}Cs	4.8 ± 0.1(-7)	3.1 ± 3.0(-8)
^{140}Ba	<8(-8)	1.1 ± 0.4(-7)
^{140}La	<4(-8)	<7(-9)
^{141}Ce	<8(-9)	<2(-9)
^{143}Ce	<3(-8)	<4(-9)
^{144}Ce	<4(-8)	<9(-9)

TABLE A.33 (cont'd)

EVAPORATOR FUNCTION TEST

Evaporator Distillate Samples (Feed: Spent Regenerant Tank WD-13A)

	Second Process Sample 13:40; 2/18/77		Third Process Sample 15:50; 2/18/77	
Nuclide	Dissolved Activity (μCi/ml)	Suspended Solids (μCi/ml)	Dissolved Activity (μCi/ml)	Suspended Solids (μCi/ml)
Activation Nuclides:				
^{24}Na	*	*	*	*
^{51}Cr	<1(-7)	<2(-8)	<1(-7)	<2(-8)
^{54}Mn	1.3 ± 0.2(-8)	<4(-9)	1.4 ± 0.3(-8)	2.4 ± 0.6(-9)
^{59}Fe	<2(-8)	<6(-9)	<2(-8)	<6(-9)
^{57}Co	6.5 ± 1.2(-9)	<9(-10)	3.2 ± 1.7(-9)	<5(-10)
^{58}Co	4.7 ± 0.1(-7)	6.6 ± 0.2(-8)	3.89 ± 0.06(-7)	5.4 ± 0.3(-8)
^{60}Co	3.9 ± 0.2(-8)	1.6 ± 0.1(-8)	3.6 ± 0.3(-8)	1.54 ± 0.09(-8)
^{65}Zn	<2(-8)	<6(-8)	<1(-8)	<5(-9)
^{187}W	<8(-8)	<2(-8)	<8(-8)	<3(-8)
^{239}Np	<9(-8)	1.0 ± 1.0(-8)	<9(-8)	<8(-9)
Iodine Nuclides:				
^{131}I	1.84 ± 0.05(-7)	7.8 ± 0.9(-9)	1.65 ± 0.08(-7)	2.2 ± 0.7(-9)
^{133}I	*	*	*	*
^{135}I	*	*	*	*
Fission Products:				
^{95}Zr	<4(-8)	<8(-9)	<4(-8)	2.2 ± 0.7(-9)
^{95}Nb	<2(-8)	<4(-9)	<2(-8)	<7(-9)
^{99}Mo	<3(-9)	<9(-10)	<5(-9)	<6(-10)
^{103}Ru	<1(-8)	<4(-9)	<2(-8)	<3(-9)
^{110m}Ag	<1(-8)	<4(-9)	<2(-8)	<2(-9)
^{124}Sb	<1(-8)	<4(-9)	<2(-8)	<2(-9)
^{134}Cs	3.70 ± 0.06(-7)	4.1 ± 0.2(-8)	2.93 ± 0.09(-7)	1.76 ± 0.09(-8)
^{136}Cs	<2(-8)	<4(-9)	<2(-8)	<7(-9)
^{137}Cs	5.00 ± 0.07(-7)	6.1 ± 0.2(-8)	3.8 ± 0.1(-7)	1.1 ± 1.1(-8)
^{140}Ba	<6(-8)	<2(-8)	<1(-7)	<9(-9)
^{140}La	<6(-8)	<7(-9)	<6(-8)	<6(-9)
^{141}Ce	<6(-9)	4.8 ± 5.4(-10)	<9(-9)	<1(-9)
^{143}Ce	<3(-8)	<4(-9)	<3(-8)	<3(-9)
^{144}Ce	<3(-8)	<7(-9)	<5(-8)	<5(-9)

* : Sample not counted soon enough for these radionuclides.

TABLE A.34

EVAPORATOR FUNCTION TEST

Evaporator Concentrate Samples

Nuclide	Before Test Start 07:30; 2/18/77 Dissolved Activity (μCi/ml)	Before Test Start 07:30; 2/18/77 Suspended Solids (μCi/ml)	First Process Sample 10:20; 2/18/77 Dissolved Activity (μCi/ml)	First Process Sample 10:20; 2/18/77 Suspended Solids (μCi/ml)
Activation Nuclides:				
^{24}Na	3.7 ± 1.5(-5)	<7(-5)	8.0 ± 2.2 (-5)	<5(-5)
^{51}Cr	<5(-3)	<1(-4)	<5(-3)	2.1 ± 0.6(-4)
^{54}Mn	1.4 ± 0.2(-5)	1.08 ± 0.01(-3)	1.4 ± 0.2 (-4)	1.61 ± 0.02(-3)
^{59}Fe	<8(-5)	4.6 ± 1.3(-5)	<1(-4)	7.3 ± 1.3(-5)
^{57}Co	<2(-4)	6.9 ± 0.5(-5)	<2(-4)	9.2 ± 0.6(-5)
^{58}Co	3.6 ± 0.1 (-3)	1.05 ± 0.02(-2)	3.87 ± 0.07(-3)	1.62 ± 0.04(-2)
^{60}Co	2.0 ± 0.4 (-4)	8.4 ± 0.2(-4)	1.5 ± 0.1 (-4)	1.27 ± 0.02(-3)
^{65}Zn	<7(-5)	5.1 ± 1.4(-5)	<1(-4)	<1(-4)
^{187}W	<8(-4)	<8(-4)	<5(-4)	<4(-3)
^{239}Np	<3(-3)	<7(-5)	2.5 ± 2.6 (-4)	<2(-4)
Iodine Nuclides:				
^{131}I	2.62 ± 0.03(-2)	2.9 ± 0.6(-6)	2.54 ± 0.05(-2)	6.1 ± 1.5(-5)
^{133}I	2.6 ± 0.1 (-3)	<7(-5)	2.5 ± 0.2 (-3)	<1(-4)
^{135}I	<2(-4)	<2(-4)	<3(-4)	<3(-3)
Fission Products:				
^{95}Zr	<4(-4)	2.1 ± 0.6(-5)	<3(-4)	4.4 ± 1.0(-5)
^{95}Nb	<3(-4)	3.5 ± 0.4(-5)	<2(-4)	<1(-3)
^{99}Mo	<2(-4)	<6(-6)	<2(-4)	<5(-6)
^{103}Ru	<3(-3)	<7(-5)	<3(-3)	<1(-4)
^{110m}Ag	<3(-3)	9.5 ± 0.7(-5)	<3(-3)	1.3 ± 0.1(-4)
^{124}Sb	<3(-3)	<6(-5)	<3(-3)	<1(-4)
^{134}Cs	6.6 ± 0.1 (-2)	1.32 ± 0.02(-5)	6.50 ± 0.05(-2)	1.8 ± 0.3(-3)
^{136}Cs	<3(-4)	<2(-4)	<2(-4)	<2(-4)
^{137}Cs	8.0 ± 0.2 (-2)	1.66 ± 0.03(-5)	7.7 ± 0.2 (-2)	2.45 ± 0.04(-3)
^{140}Ba	<2(-2)	<4(-4)	7.4 ± 2.3 (-4)	<4(-4)
^{140}La	<2(-3)	<5(-5)	<2(-3)	<2(-4)
^{141}Ce	<4(-4)	<2(-5)	<3(-4)	<1(-5)
^{143}Ce	<1(-3)	1.4 ± 0.8(-5)	<1(-3)	<1(-4)
^{144}Ce	<2(-3)	<7(-5)	<2(-3)	<5(-5)

TABLE A.34 (cont'd)

EVAPORATOR FUNCTION TEST

Evaporator Concentrate Samples

	Second Process Sample 13:36; 2/18/77		Third Process Sample 15:50; 2/18/77	
Nuclide	Dissolved Activity (μCi/ml)	Suspended Solids (μCi/ml)	Dissolved Activity (μCi/ml)	Suspended Solids (μCi/ml)
Activation Nuclides:				
^{24}Na	<1(-4)	<5(-5)	<2(-4)	<2(-4)
^{51}Cr	<5(-3)	2.8 ± 0.3(-4)	<1(-2)	3.2 ± 0.4(-4)
^{54}Mn	1.4 ± 0.2 (-4)	1.25 ± 0.03(-3)	2.6 ± 0.2 (-4)	1.61 ± 0.03(-3)
^{59}Fe	<1(-4)	8.7 ± 0.9(-5)	<2(-4)	1.20 ± 0.08(-4)
^{57}Co	5.7 ± 2.1 (-5)	7.3 ± 0.3(-5)	<3(-3)	9.8 ± 0.3(-5)
^{58}Co	4.0 ± 0.2 (-3)	1.25 ± 0.03(-2)	4.7 ± 0.1 (-3)	1.59 ± 0.04(-2)
^{60}Co	1.6 ± 0.4 (-4)	1.02 ± 0.02(-3)	2.3 ± 0.7 (-4)	1.25 ± 0.03(-3)
^{65}Zn	<1(-4)	<4(-5)	<1(-4)	<2(-4)
^{187}W	<2(-3)	4.6 ± 2.2(-4)	<1(-3)	<3(-3)
^{239}Np	<4(-3)	<1(-4)	<1(-2)	<1(-4)
Iodine Nuclides:				
^{131}I	2.80 ± 0.06(-2)	4.9 ± 0.5(-5)	2.97 ± 0.08(-2)	7.5 ± 0.5(-5)
^{133}I	2.9 ± 0.1 (-3)	<1(-4)	2.3 ± 0.2 (-3)	<1(-4)
^{135}I	<1(-3)	*	<3(-3)	*
Fission Products:				
^{95}Zr	<1(-3)	5.4 ± 0.9(-5)	<5(-4)	3.5 ± 0.6(-5)
^{95}Nb	<4(-4)	6.8 ± 0.7(-5)	<3(-4)	6.2 ± 0.9(-5)
^{99}Mo	<4(-5)	<5(-6)	<2(-3)	<4(-6)
^{103}Ru	<4(-3)	<4(-5)	<2(-3)	<5(-5)
^{110m}Ag	<3(-3)	8.7 ± 0.6(-5)	<2(-3)	1.1 ± 0.1(-4)
^{124}Sb	<3(-3)	<3(-5)	2.8 ± 1.4 (-5)	1.7 ± 0.3(-5)
^{134}Cs	7.05 ± 0.05(-2)	1.51 ± 0.02(-3)	7.2 ± 0.1 (-2)	1.94 ± 0.06(-3)
^{136}Cs	<4(-4)	<5(-4)	<3(-4)	<1(-3)
^{137}Cs	8.2 ± 0.2 (-2)	5.6 ± 7.0(-4)	8.4 ± 0.1 (-2)	6.8 ± 10.9(-4)
^{140}Ba	<2(-2)	<2(-4)	<1(-2)	<2(-4)
^{140}La	<3(-3)	2.5 ± 1.2(-5)	<5(-3)	1.5 ± 0.3(-5)
^{141}Ce	<1(-4)	<1(-5)	<4(-3)	8.0 ± 3.0(-6)
^{143}Ce	<2(-3)	<3(-5)	<3(-3)	<3(-5)
^{144}Ce	<4(-4)	<4(-5)	<2(-2)	<4(-5)

* : Sample not counted soon enough for these radionuclides.

TABLE A.35

VENTILATION AIRBORNE ACTIVITIES

SAMPLE STATION #1 (μCi/sec)

Sample Period	9/3 - 9/9/76	9/9 - 9/15/76	9/15 - 9/23/76	9/23 - 9/30/76[A]	9/30 - 10/7/76
^{131}I	8.06 ± 0.07(-3)	8.1 ± 0.2(-3)	2.19 ± 0.09(-4)	1.22 ± 0.08(-4)	1.72 ± 0.04(-3)
^{134}Cs	<2.2(-6)	<9.2(-6)	<8.6(-6)	8.6 ± 0.8(-7)	<7.9(-6)
^{136}Cs	<5.4(-6)	<2.6(-5)	<1.8(-6)	<3.4(-7)	<1.1(-5)
^{137}Cs	3.2 ± 0.7(-5)	2.6 ± 0.7(-5)	2.7 ± 0.6(-5)	1.1 ± 0.2(-6)	1.4 ± 0.5(-5)
^{3}H	3.0 ± 1.0(-3)	9.5 ± 0.9(-3)	7.2 ± 0.8(-3)	2.0 ± 1.0(-3)	5.0 ± 1.0(-3)
^{14}C	6.5 ± 0.8(-4)	2.4 ± 0.1(-3)[B]		7.8 ± 0.4(-4)[C]	
^{51}Cr	<1.1(-4)	<1.4(-5)	<4.1(-5)	<2.4(-6)	5.3 ± 2.9(-5)
^{54}Mn	<9.7(-6)	<3(-6)	<2.7(-6)	<7.5(-7)	<6.6(-6)
^{59}Fe	<3.9(-6)	<1.2(-5)	<4.7(-6)	<1.2(-4)	<7.9(-6)
^{57}Co	3.7 ± 2.3(-5)	1.2 ± 0.7(-5)	1.2 ± 0.7(-6)	<3.9(-7)	<2.2(-5)
^{58}Co	<3.4(-6)	<3.4(-6)	<6.0(-6)	2.0 ± 1.0(-6)	<4.9(-6)
^{60}Co	8.3 ± 3.0(-6)	<1.4(-5)	<4.2(-6)	9.0 ± 2.0(-7)	<1.1(-5)
^{65}Zn	<8.0(-6)	<8.9(-6)	<3.0(-6)	<1.9(-6)	<5.6(-6)
^{95}Zr	<7.1(-6)	<3.5(-6)	<5.6(-6)	<2.3(-5)	<7.3(-6)
^{95}Nb	<2.2(-6)	<1.5(-6)	<6.3(-6)	<2.2(-4)	<6.2(-6)
^{103}Ru	<1.8(-6)	<6.3(-6)	<1.8(-6)	<8.7(-5)	<4.1(-6)
^{106}Ru	<1.5(-5)	<4.6(-5)	<1.6(-5)	<1.3(-5)	<3.0(-5)
^{110m}Ag	<6.4(-6)	<2.6(-5)	<3.9(-6)	<1.4(-6)	<3.9(-6)
^{124}Sb	<7.4(-6)	<6.2(-6)	<7.3(-6)	3.0 ± 0.3(-5)	6.3 ± 3.5(-6)
^{125}Sb	<9.0(-6)	<1.8(-5)	<9.4(-6)	4.0 ± 2.0(-7)	<1.4(-5)
^{140}Ba	<9.6(-6)	6.2 ± 2.1(-5)	<9.4(-6)	<1.2(-6)	<1.4(-5)
^{140}La	4.6 ± 0.3(-4)	<1.6(-4)	<3.0(-5)	<4.0(-7)	<1.3(-3)
^{141}Ce	2.8 ± 1.7(-5)	<2.4(-5)	1.0 ± 0.7(-6)	<3.4(-7)	<1.5(-5)
^{152}Eu	<1.9(-5)	<1.4(-5)	<1.2(-5)	<1.1(-6)	<2.0(-5)
^{154}Eu	<1(-5)	<6.1(-6)	<8.4(-6)	<1.2(-6)	<1.3(-5)
^{155}Eu	<7.5(-4)	<1.0(-4)	1.1 ± 0.4(-4)	<3.1(-4)	<1.4(-4)

[A] 60,000 second count

[B] Sample from 9/9 - 9/23/76

[C] Sample from 9/23 - 10/7/76

TABLE A.35 (Cont.)

VENTILATION AIRBORNE ACTIVITIES

SAMPLE STATION #1 (μCi/sec)

Sample Period	10/7 - 10/14/76	10/14 - 10/22/76	10/22 - 10/28/76	10/28 - 11/4/76	11/4 - 11/10/76
^{131}I	5.6 ± 0.2(-4)	3.0 ± 0.1(-4)	4.6 ± 0.6(-4)	7.6 ± 0.7(-6)	3.2 ± 0.6(-5)
^{134}Cs	<9.1(-6)	<1.7(-5)	<1.1(-5)	9.2 ± 3.9(-6)	<6.6(-6)
^{136}Cs	<7.8(-6)	<7.8(-6)	<1.0(-5)	<1.4(-6)	<9.2(-6)
^{137}Cs	5.4 ± 1.8(-5)	2.0 ± 0.5(-5)	5.1 ± 1.3(-5)	6.0 ± 1.0(-5)	5.4 ± 1.5(-5)
^{3}H	5.0 ± 1.0(-3)	9.8 ± 0.9(-3)	3.8 ± 0.3(-2)	7.2 ± 0.8(-3)	6.4 ± 0.1(-2)
^{14}C	1.7 ± 0.1(-3)[D]		2.97 ± 0.02(-2)[E]		1.11 ± 0.04(-2)
^{51}Cr	3.0 ± 2.0(-5)	<3.5(-5)	<1.5(-5)	<7.6(-5)	<1.0(-5)
^{54}Mn	<3.9(-6)	<9.6(-7)	<2.5(-6)	<2.3(-6)	<5.4(-6)
^{59}Fe	<1.1(-5)	<6.2(-6)	<2.0(-5)	<1.1(-5)	<1.1(-5)
^{57}Co	<4.1(-5)	<1.7(-5)	<5.3(-5)	<4.6(-6)	5.0 ± 2.0(-5)
^{58}Co	<5.6(-6)	<5.8(-6)	<3.7(-6)	<1.1(-5)	<9.2(-6)
^{60}Co	<5.3(-6)	<5.5(-6)	<1.2(-5)	<7.3(-6)	<2.5(-5)
^{65}Zn	<2.8(-6)	<9.4(-6)	<2.0(-5)	<1.0(-5)	<9.0(-6)
^{95}Zr	<2.2(-6)	<2.0(-6)	<4.0(-6)	<1.6(-6)	<8.6(-6)
^{95}Nb	<1.2(-6)	<3.4(-6)	<4.5(-6)	<2.2(-4)	<5.2(-6)
^{103}Ru	<3.1(-6)	<1.4(-6)	9.1 ± 4.6(-6)	<3.7(-6)	<1.2(-5)
^{106}Ru	<1.7(-5)	<4.0(-5)	<3.6(-5)	<1.6(-5)	<2.8(-5)
^{110m}Ag	<3.5(-6)	<2.3(-5)	<9.5(-6)	<8.9(-6)	<1.3(-5)
^{124}Sb	<7.8(-6)	<6.4(-6)	<9.8(-6)	<1.2(-5)	<2.0(-5)
^{125}Sb	<6.9(-6)	<4.9(-6)	<2.5(-5)	1.9 ± 0.8(-5)	<4.2(-5)
^{140}Ba	<2.0(-5)	<4.3(-6)	<1.6(-5)	<1.6(-5)	<4.1(-5)
^{140}La	<9.9(-5)	<8.5(-4)	<6.6(-5)	<1.9(-4)	<2.5(-3)
^{141}Ce	<3.8(-5)	<1.2(-5)	<4.6(-5)	<6.3(-5)	<8.1(-5)
^{152}Eu	<1.0(-5)	<2.1(-5)	2.8 ± 1.4(-5)	1.3 ± 0.8(-5)	2.6 ± 1.7(-5)
^{154}Eu	<1.1(-5)	<6.8(-6)	<2.0(-5)	<9.7(-6)	<2.9(-5)
^{155}Eu	<3.2(-4)	<2.1(-4)	<6.9(-4)	<1.6(-4)	<1.2(-3)

[D] Sample from 10/7 - 10/22/76

[E] Sample from 10/22 - 11/4/76

TABLE A.35 (Cont.)

VENTILATION AIRBORNE ACTIVITIES

SAMPLE STATION #1 (μCi/sec)

Sample Period	11/10 - 11/23/76	11/23 - 11/30/76	12/1 - 12/15/76[F]	12/15 - 1/6/77	1/6 - 1/20/77
^{131}I	2.8 ± 0.5(-5)	1.1 ± 0.4(-5)	<2.8(-5)	1.33 ± 0.05(-6)	1.96 ± 0.07(-4)
^{134}Cs	<6.3(-6)	<1.8(-5)	<1.3(-5)	9.0 ± 0.6(-6)	9.0 ± 0.8(-6)
^{136}Cs	<7.0(-6)	<1.3(-5)	<4.7(-5)	<2.3(-7)	<3.9(-6)
^{137}Cs	6.0 ± 4.0(-6)	5.0 ± 1.0(-5)	<4.2(-5)	1.0 ± 0.1(-5)	1.1 ± 0.1(-5)
^{3}H	9.3 ± 0.2(-3)	1.8 ± 0.1(-3)	4.4 ± 0.1(-3)	1.6 ± 0.2(-2)	4.7 ± 0.2(-2)
^{14}C	2.0 ± 0.1(-2)	3.6 ± 0.3(-3)	6.8 ± 0.3(-3)	2.4 ± 0.1(-3)	7.1 ± 0.2(-3)
^{51}Cr	<6.2(-5)	<7.9(-5)	<6.1(-5)	<4.8(-6)	<5.2(-6)
^{54}Mn	<1.6(-6)	<4.4(-5)	<9.5(-6)	<4.4(-7)	<2.0(-7)
^{59}Fe	<8.1(-6)	<3.2(-5)	<2.6(-5)	<3.6(-6)	<6.7(-6)
^{57}Co	2.0 ± 1.0(-5)	<3.1(-5)	5.0 ± 3.0(-5)	<2.9(-7)	<1.1(-7)
^{58}Co	5.0 ± 1.0(-5)	<2.1(-5)	5.0 ± 2.0(-5)	2.3 ± 0.5(-6)	4.0 ± 3.0(-7)
^{60}Co	<3.4(-6)	<1.4(-5)	<9.7(-6)	<6.3(-7)	<2.2(-7)
^{65}Zn	<7.4(-6)	<2.1(-5)	<9.4(-6)	<2.5(-7)	<5.9(-7)
^{95}Zr	<2.6(-6)	<2.3(-5)	<4.1(-5)	<4.5(-7)	<5.6(-7)
^{95}Nb	<1.6(-6)	<3.3(-6)	<8.9(-5)	<1.8(-6)	<2.9(-7)
^{103}Ru	<6.5(-6)	<1.0(-5)	<1.4(-5)	<8.4(-7)	<1.2(-6)
^{106}Ru	<3.6(-5)	<4.8(-5)	<2.2(-5)	<5.8(-6)	<3.4(-6)
^{110m}Ag	<3.2(-5)	<9.8(-6)	<4.4(-5)	<1.1(-7)	<2.8(-7)
^{124}Sb	<1.2(-5)	<1.8(-5)	<4.2(-5)	<7.5(-6)	1.1 ± 0.1(-5)
^{125}Sb	<2.5(-5)	<4.4(-5)	<1.5(-5)	<2.3(-6)	<1.6(-6)
^{140}Ba	<1.8(-5)	<1.2(-4)	<1.6(-4)	<3.0(-6)	<2.3(-6)
^{140}La	<1.2(-3)	<4.8(-3)	<1.5(-5)	<2.3(-6)	<1.2(-5)
^{141}Ce	<5.1(-5)	<6.9(-5)	<5.6(-5)	<6.2(-7)	<3.1(-7)
^{152}Eu	<2.1(-5)	<3.9(-5)	<1.1(-4)	5.0 ± 4.0(-7)	<2.4(-6)
^{154}Eu	<8.6(-5)	<2.8(-5)	<2.3(-5)	<6.2(-7)	<6.0(-7)

[F] Total Iodine Sampler was only operative between 12/1 - 12/3/76. ^{14}C and ^{3}H sampler was operative 12/1 - 12/15/76.

TABLE A.35 (Cont.)

VENTILATION AIRBORNE ACTIVITIES

SAMPLE STATION #1 (μCi/sec)

Sample Period	1/20 - 2/3/77	2/3 - 2/10/77	2/10 - 2/17/77
^{131}I	2.75 ± 0.08(-4)	1.76 ± 0.07(-4)	1.9 ± 0.1(-4)
^{134}Cs	<1.4(-6)	<2.4(-6)	<8.3(-6)
^{136}Cs	<1.6(-7)	<1.6(-6)	<1.5(-6)
^{137}Cs	<1.4(-6)	2.6 ± 0.7(-6)	<1.7(-6)
^{3}H	1.8 ± 0.1(-2)	[G]	[G]
^{14}C	4.3 ± 0.2(-3)	[G]	[G]
^{51}Cr	<1.7(-6)	<2.4(-5)	<5.2(-6)
^{54}Mn	<4.0(-6)	<1.6(-6)	<6.8(-7)
^{59}Fe	<6.5(-6)	5.0 ± 2.0(-6)	<3.0(-5)
^{57}Co	<1.7(-7)	<7.2(-7)	<1.4(-7)
^{58}Co	<8.4(-7)	<1.6(-6)	<2.4(-6)
^{60}Co	<2.6(-7)	6.0 ± 2.0(-6)	<1.1(-6)
^{65}Zn	<4.5(-6)	<8.0(-6)	<2.1(-5)
^{95}Zr	<6.6(-7)	<3.2(-6)	<1.3(-5)
^{95}Nb	<1.8(-6)	<1.6(-6)	<7.9(-7)
^{103}Ru	<1.3(-7)	<1.6(-6)	<1.0(-6)
^{106}Ru	<2.4(-6)	<1.6(-5)	<1.3(-5)
^{110m}Ag	<3.0(-7)	<1.6(-6)	<1.2(-5)
^{124}Sb	9.0 ± 4.0(-7)	<1.6(-6)	<1.4(-6)
^{125}Sb	<7.4(-7)	<4.0(-6)	<2.4(-6)
^{140}Ba	<6.6(-7)	<5.6(-6)	<2.6(-6)
^{140}La	<6.2(-6)	<1.6(-5)	<1.7(-5)
^{141}Ce	<2.4(-7)	<1.6(-6)	<7.9(-7)
^{152}Eu	<5.7(-7)	<7.2(-6)	<4.9(-6)
^{154}Eu	<7.1(-7)	<1.6(-5)	<3.3(-5)

[G] SAMPLE STATION #1 ^{14}C-^{3}H Shutdown due to electrical circuit overload.

TABLE A.36

VENTILATION AIRBORNE ACTIVITIES

SAMPLE STATION #2 (μCi/sec)

Sample Period	9/3 - 9/9/76	9/9 - 9/15/76	9/15 - 9/18/76	9/23 - 9/30/76	9/30 - 10/7/76
^{131}I	6.13 ± 0.06(-4)	1.92 ± 0.06(-4)	6.0 ± 0.2(-4)	8.3 ± 0.4(-5)	1.43 ± 0.01(-3)
^{134}Cs	<2.7(-6)	<1.5(-5)	<2.6(-5)	<2.1(-6)	<7.1(-6)
^{136}Cs	<4.4(-4)	<2.8(-6)	<1.5(-5)	<2.7(-6)	<3.1(-6)
^{137}Cs	8.0 ± 3.0(-6)	4.8 ± 2.4(-6)	2.0 ± 1.0(-5)	<9.0(-6)	6.8 ± 3.9(-6)
^{3}H	6.0 ± 1.0(-3)	<1.0(-3)	<1.0(-3)	2.0 ± 0.4(-3)	2.0 ± 0.4(-3)
^{14}C	1.41 ± 0.01(-1)	1.9 ± 0.1(-3)[A]		1.44 ± 0.09(-3)[B]	
^{51}Cr	<6.6(-5)	<1.3(-5)	<4.6(-5)	2.3 ± 1.1(-5)	<2.8(-5)
^{54}Mn	<1.1(-5)	<2.8(-6)	<1.0(-5)	<7.7(-7)	<2.1(-6)
^{59}Fe	<3.7(-5)	<6.9(-6)	<7.5(-6)	<4.4(-6)	<3.6(-6)
^{57}Co	1.5 ± 0.7(-6)	<1.2(-6)	<2.0(-6)	<9.1(-7)	<4.6(-7)
^{58}Co	<1.4(-5)	<4.0(-6)	<5.5(-6)	<3.0(-6)	<3.3(-6)
^{60}Co	<6.6(-6)	<5.3(-5)	<2.0(-5)	<2.9(-6)	<5.0(-6)
^{65}Zn	<1.2(-5)	<2.5(-6)	<4.6(-6)	<4.0(-6)	<1.9(-6)
^{95}Zr	<1.0(-5)	6.0 ± 2.0(-6)	<3.7(-6)	<4.6(-6)	<7.2(-6)
^{95}Nb	<2.4(-5)	<2.0(-6)	<1.4(-5)	<3.7(-6)	<2.6(-6)
^{103}Ru	<1.7(-5)	<4.7(-6)	<4.0(-6)	<2.5(-6)	<3.3(-6)
^{106}Ru	<8.4(-6)	<2.1(-5)	<3.8(-5)	<2.7(-5)	<9.2(-6)
^{110m}Ag	<5.8(-6)	<7.2(-6)	<1.2(-5)	<3.8(-6)	<1.7(-6)
^{124}Sb	<1.2(-5)	<2.2(-6)	<1.4(-5)	<2.3(-6)	<6.2(-6)
^{125}Sb	<2.5(-5)	<1.6(-5)	<1.4(-5)	<5.0(-6)	<1.6(-6)
^{140}Ba	<6.2(-3)	<1.0(-5)	<2.3(-5)	<6.5(-6)	<1.7(-5)
^{140}La	<3.3(-6)	<4.4(-5)	<4.7(-3)	<4.3(-5)	<1.1(-3)
^{141}Ce	<9.0(-6)	<4.7(-6)	4.5 ± 2.6(-6)	2.8 ± 1.6(-6)	<2.6(-6)
^{152}Eu	<1.4(-5)	<1.3(-5)	<6.2(-6)	<8.7(-6)	<5.8(-6)
^{154}Eu	<7.2(-6)	<4.5(-6)	<2.3(-5)	<1.2(-5)	<1.1(-5)
^{155}Eu	4.0 ± 2.0(-6)	<4.8(-6)	<8.5(-6)	<3.4(-6)	<7.5(-7)

[A] Sample from 9/9 - 9/23/76

[B] Sample from 9/23 - 10/7/76

TABLE A.36 (Cont.)

VENTILATION AIRBORNE ACTIVITIES
SAMPLE STATION #2 (μCi/sec)

Sample Period	10/7 - 10/14/76	10/14 - 10/22/76	10/22 - 10/28/76	10/28 - 11/4/76	11/4 - 11/10 76
^{131}I	3.80 ± 0.02(-4)	3.4 ± 0.3(-4)	1.66 ± 0.01(-4)	1.57 ± 0.01(-4)	5.72 ± 0.04(-5)
^{134}Cs	<5.7(-6)	<1.7(-6)	<2.2(-6)	<8.6(-7)	<3.5(-7)
^{136}Cs	<1.7(-6)	<3.1(-6)	<9.6(-7)	<8.1(-7)	<1.1(-6)
^{137}Cs	3.1 ± 1.1(-5)	<1.5 ± 0.4(-5)	2.1 ± 1.0(-6)	1.3 ± 0.6(-6)	1.5 ± 0.8(-6)
^{3}H	2.0 ± 1.0(-3)	2.0 ± 1.0(-3)	4.7 ± 0.5(-3)	4.7 ± 0.5(-3)	2.0 ± 1.0(-3)
^{14}C	2.5 ± 0.1(-3)[C]		9.4 ± 0.3(-3)[D]		1.16 ± 0.03(-2)
^{51}Cr	<1.3(-5)	<2.9(-5)	5.2 ± 2.6(-6)	<3.8(-6)	<6.7(-6)
^{54}Mn	<2.5(-6)	<7.3(-6)	2.9 ± 0.8(-6)	<1.1(-6)	<4.7(-7)
^{59}Fe	<8.0(-6)	<1.2(-5)	<2.8(-6)	<1.9(-6)	<1.3(-6)
^{57}Co	<1.5(-6)	<2.0(-6)	6.5 ± 2.6(-7)	<4.3(-7)	<3.2(-7)
^{58}Co	5.4 ± 1.7(-6)	4.6 ± 2.7(-6)	8.2 ± 1.6(-6)	<1.5(-6)	2.1 ± 0.8(-6)
^{60}Co	<1.1(-5)	<4.8(-6)	9.8 ± 1.5(-6)	2.4 ± 1.0(-6)	<1.6(-6)
^{65}Zn	<3.1(-6)	<8.5(-6)	<5.3(-6)	<8.8(-7)	<1.1(-6)
^{95}Zr	<7.0(-6)	<5.1(-6)	<2.2(-6)	<1.1(-6)	<7.7(-7)
^{95}Nb	<2.8(-6)	<3.9(-6)	<7.9(-7)	<6.7(-7)	<6.0(-7)
^{103}Ru	2.4 ± 1.4(-6)	<1.9(-6)	<2.0(-6)	<1.3(-6)	<5.7(-7)
^{106}Ru	2.0 ± 0.9(-5)	<1.4(-5)	<9.2(-6)	<4.7(-6)	<4.8(-6)
^{110m}Ag	<2.6(-6)	<4.2(-6)	<1.3(-6)	<4.7(-7)	<8.7(-8)
^{124}Sb	<3.0(-6)	3.6 ± 2.2(-6)	1.7 ± 0.7(-6)	<1.1(-6)	<1.7(-6)
^{125}Sb	<1.9(-6)	<1.6(-6)	<4.2(-6)	<2.2(-6)	<2.8(-6)
^{140}Ba	<9.7(-6)	<1.2(-5)	<5.1(-6)	2.9 ± 1.6(-6)	<2.0(-6)
^{140}La	<1.2(-4)	<3.4(-6)	<5.8(-6)	<2.0(-5)	<1.3(-5)
^{141}Ce	<2.8(-6)	2.2 ± 1.3(-6)	5.2 ± 2.6(-6)	4.2 ± 2.2(-6)	<9.2(-7)
^{152}Eu	<1.7(-5)	<6.7(-6)	2.0 ± 0.7(-6)	<2.2(-6)	<1.8(-6)
^{154}Eu	<1.2(-5)	<5.0(-6)	<3.0(-6)	<1.1(-6)	<1.3(-6)
^{155}Eu	<3.2(-6)	<2.9(-6)	<1.2(-6)	<1.1(-6)	<1.4(-6)

[C] Sample from 10/7 - 10/22/76

[D] Sample from 10/22 - 11/4/76

TABLE A.36 (Cont.)

VENTILATION AIRBORNE ACTIVITIES
SAMPLE STATION #2 (μCi/sec)

Sample Period	11/10 - 11/23/76	11/23 - 11/30/76	12/1 - 12/15/76	12/15/76-1/6/77	1/6 - 1/20/77[E]
^{131}I	6.47 ± 0.04(-5)	1.75 ± 0.01(-3)	1.06 ± 0.09(-4)	6.6 ± 0.4(-5)	5.0 ± 0.4(-5)
^{134}Cs	2.5 ± 0.5(-6)	<1.2(-6)	2.0 ± 0.4(-6)	9.8 ± 0.2(-5)	4.96 ± 0.03(-5)
^{136}Cs	<7.6(-7)	<5.3(-7)	<2.1(-6)	<9.3(-7)	<7.6(-8)
^{137}Cs	3.1 ± 0.7(-6)	3.0 ± 0.9(-6)	4.6 ± 0.8(-6)	1.20 ± 0.03(-4)	5.85 ± 0.03(-5)
^{3}H	2.2 ± 0.1(-2)	1.9 ± 0.9(-2)	5.5 ± 0.1(-2)	1.0 ± 0.1(-2)	6.3 ± 0.1(-3)
^{14}C	1.88 ± 0.04(-2)	4.3 ± 0.2(-3)	1.8 ± 0.2(-2)	3.2 ± 0.2(-3)	1.12 ± 0.04(-2)
^{51}Cr	<8.2(-6)	<5.5(-6)	4.0 ± 3.0(-6)	<1.5(-5)	<3.8(-4)
^{54}Mn	9.0 ± 4.0(-7)	<2.5(-6)	2.0 ± 0.5(-6)	<7.3(-7)	6.3 ± 0.8(-7)
^{59}Fe	<1.0(-6)	<6.5(-7)	<1.1(-6)	<5.6(-7)	<1.9(-6)
^{57}Co	<3.0(-7)	1.4 ± 0.9(-7)	5.0 ± 2.0(-7)	<6.1(-7)	<1.4(-7)
^{58}Co	9.0 ± 1.0(-6)	2.6 ± 0.8(-6)	3.5 ± 0.2(-5)	2.0 ± 0.4(-6)	1.0 ± 0.1(-5)
^{60}Co	1.0 ± 0.3(-6)	1.9 ± 0.7(-6)	2.5 ± 0.5(-6)	<5.0(-7)	1.05 ± 0.04(-6)
^{65}Zn	<4.4(-7)	<7.3(-7)	<9.8(-7)	<1.4(-6)	<3.0(-7)
^{95}Zr	<3.4(-7)	<1.2(-6)	<2.2(-6)	<1.4(-6)	<1.6(-6)
^{95}Nb	<1.2(-6)	<1.1(-6)	<1.8(-6)	3.0 ± 2.0(-7)	<7.4(-6)
^{103}Ru	<8.4(-7)	5.0 ± 3.0(-7)	6.0 ± 4.0(-7)	5.0 ± 4.0(-7)	<8.6(-6)
^{106}Ru	<9.9(-6)	<4.0(-6)	<3.5(-5)	<9.2(-6)	<2.4(-6)
^{110m}Ag	<2.8(-7)	<1.2(-6)	<5.2(-7)	<1.3(-6)	<3.1(-7)
^{124}Sb	3.3 ± 0.7(-6)	<1.0(-6)	3.0 ± 1.0(-6)	<8.2(-5)	<2.6(-4)
^{125}Sb	<2.2(-6)	<2.5(-6)	1.3 ± 0.7(-6)	<5.7(-6)	<8.7(-7)
^{140}Ba	1.7 ± 0.9(-6)	<4.6(-6)	3.0 ± 2.0(-6)	<7.9(-6)	<5.5(-7)
^{140}La	<1.6(-5)	<1.9(-5)	<3.9(-7)	<5.1(-7)	<1.3(-4)
^{141}Ce	2.0 ± 1.0(-7)	6.0 ± 3.0(-7)	<1.3(-6)	<1.4(-6)	<1.5(-5)
^{152}Eu	1.0 ± 0.5(-6)	<2.3(-6)	<4.1(-6)	<5.0(-6)	<6.0(-7)
^{154}Eu	<1.1(-6)	<9.6(-7)	<4.8(-7)	<1.0(-7)	<1.9(-7)

[E] 60,000 second count

TABLE A.36 (Cont.)

VENTILATION AIRBORNE ACTIVITIES
SAMPLE STATION #2 (μCi/sec)

Sample Period	1/20 - 2/3/77	2/3 - 2/16/77
^{131}I	1.8 ± 0.2(-4)	2.1 ± 0.2(-5)
^{134}Cs	<4.0(-6)	3.0 ± 1.0(-5)
^{136}Cs	<9.3(-8)	<4.2(-7)
^{137}Cs	4.5 ± 0.6(-6)	3.4 ± 0.2(-5)
^{3}H	[F]	6.0 ± 1.0(-3)
^{14}C	[F]	5.0 ± 0.3(-3)
^{51}Cr	<2.9(-6)	<1.0(-5)
^{54}Mn	<1.9(-7)	<7.2(-7)
^{59}Fe	<6.1(-7)	<3.8(-6)
^{57}Co	<2.3(-7)	9.0 ± 7.0(-8)
^{58}Co	2.0 ± 1.0(-7)	2.5 ± 0.5(-6)
^{60}Co	<1.7(-7)	<9.8(-7)
^{65}Zn	<7.6(-7)	<4.9(-7)
^{95}Zr	<4.9(-7)	<6.0(-7)
^{95}Nb	<2.6(-7)	<5.0(-7)
^{103}Ru	<4.0(-7)	<1.5(-6)
^{106}Ru	<2.0(-6)	<6.1(-6)
^{110m}Ag	<3.0(-7)	<6.7(-7)
^{124}Sb	5.2 ± 0.3(-6)	3.0 ± 2.0(-5)
^{125}Sb	<5.5(-7)	<3.9(-6)
^{140}Ba	<2.5(-6)	<5.9(-6)
^{140}La	<5.9(-6)	<2.2(-5)
^{141}Ce	<4.4(-7)	1.1 ± 0.9(-6)
^{152}Eu	4.0 ± 3.0(-7)	<4.4(-6)
^{154}Eu	<3.2(-7)	<4.2(-6)

[F] Sample station #2 - ^{14}C-^{3}H Sampler inoperable due to motor malfunction.

TABLE A.37

VENTILATION AIRBORNE ACTIVITIES

SAMPLE STATION #3 (μCi/sec)

Sample Period	9/3 - 9/9/76	9/9 - 9/15/76	9/15 - 9/23/76	9/23 - 9/30/76	9/30 - 10/7/76
^{131}I	1.7 ± 0.1(-5)	2.9 ± 0.6(-6)	2.3 ± 0.6(-6)	4.5 ± 0.7(-6)	6.5 ± 0.4(-5)
^{134}Cs	<3.1(-6)	<1.5(-6)	<1.1(-6)	<9.9(-7)	<2.9(-6)
^{136}Cs	<7.4(-7)	<1.7(-6)	<1.7(-6)	<1.5(-6)	<1.8(-6)
^{137}Cs	9.4 ± 2.3(-6)	8.3 ± 3.1(-6)	4.6 ± 1.3(-6)	5.2 ± 1.6(-6)	1.0 ± 0.2(-5)
^{3}H	<3.0(-4)	<3.0(-4)	<3.0(-4)	<3.0(-4)	<3.0(-4)
^{14}C	1.6 ± 0.7(-4)	1.9 ± 0.2(-4)[A]		4.3 ± 0.3(-4)[B]	
^{51}Cr	<1.8(-6)	<1.0(-5)	<1.8(-5)	<1.1(-5)	<2.2(-5)
^{54}Mn	<1.7(-6)	<1.1(-6)	<1.1(-6)	<1.0(-6)	<4.1(-7)
^{59}Fe	<3.0(-6)	<4.1(-6)	<9.9(-7)	<1.2(-6)	<1.5(-6)
^{57}Co	9.2 ± 4.1(-6)	<6.0(-6)	3.3 ± 2.3(-6)	3.5 ± 1.6(-6)	<6.9(-6)
^{58}Co	<4.5(-7)	<1.3(-6)	<7.3(-7)	<8.8(-7)	<5.2(-6)
^{60}Co	<2.2(-6)	<4.8(-6)	<1.4(-6)	<1.2(-6)	<1.1(-6)
^{65}Zn	<1.9(-6)	<1.7(-6)	<1.1(-6)	<5.4(-7)	<1.3(-6)
^{95}Zr	<2.1(-6)	<7.1(-7)	<8.4(-7)	<9.6(-7)	<1.1(-6)
^{95}Nb	<1.3(-6)	<6.2(-6)	<2.7(-7)	<7.9(-7)	<2.3(-6)
^{103}Ru	<2.2(-6)	<1.4(-6)	<1.3(-6)	<2.0(-6)	<1.5(-6)
^{106}Ru	<1.0(-5)	<1.3(-5)	<9.6(-6)	<4.3(-6)	<1.1(-5)
^{110m}Ag	<8.8(-7)	<1.3(-6)	<6.4(-6)	<2.4(-6)	<7.2(-6)
^{124}Sb	<2.4(-6)	<2.0(-6)	<2.1(-6)	<1.3(-6)	<1.7(-6)
^{125}Sb	<3.0(-6)	<9.8(-7)	<2.9(-6)	<8.3(-7)	<6.7(-6)
^{140}Ba	<6.8(-6)	<5.7(-6)	<5.0(-6)	<1.7(-6)	<5.3(-6)
^{140}La	<1.2(-5)	<1.9(-5)	<1.8(-5)	<7.6(-6)	<2.5(-4)
^{141}Ce	<1.2(-5)	5.3 ± 2.4(-6)	5.9 ± 3.1(-6)	<8.8(-6)	6.1 ± 3.8(-6)
^{152}Eu	<5.3(-6)	<1.8(-6)	<4.7(-6)	<9.8(-6)	3.3 ± 1.7(-6)
^{154}Eu	<1.2(-6)	<2.0(-6)	<2.5(-6)	<2.2(-6)	<4.1(-6)
^{155}Eu	<3.6(-5)	<1.8(-4)	<9.2(-5)	<2.2(-5)	<5.2(-5)

[A] Sample from 9/9 - 9/23/76

[B] Sample from 9/23 - 10/7/76

TABLE A.37 (Cont.)

VENTILATION AIRBORNE ACTIVITIES

SAMPLE STATION #3 (μCi/sec)

Sample Period	10/7 - 10/14/76	10/14 - 10/22/76	10/22 - 10/28/76	10/28 - 11/4/76	11/4 - 11/10/76
^{131}I	1.6 ± 0.2(-5)	1.3 ± 0.2(-5)	5.0 ± 2.0(-6)	2.0 ± 0.4(-6)	4.0 ± 0.9(-6)
^{134}Cs	<1.9(-6)	<2.9(-6)	<2.8(-6)	<3.1(-6)	<1.1(-6)
^{136}Cs	<2.8(-6)	<4.7(-6)	<9.8(-6)	<2.2(-6)	<2.8(-6)
^{137}Cs	1.2 ± 0.2(-5)	6.8 ± 2.7(-6)	1.1 ± 0.5(-6)	7.0 ± 2.0(-6)	2.4 ± 1.3(-6)
^{3}H	1.0 ± 0.3(-3)	1.0 ± 0.3(-3)	<3.0(-4)	1.3 ± 0.2(-3)	8.0 ± 2.0(-4)
^{14}C	2.0 ± 0.2(-4)[C]		1.4 ± 0.1(-4)[D]		1.45 ± 0.06(-3)
^{51}Cr	<3.5(-6)	<2.0(-5)	<2.2(-5)	<3.1(-5)	<1.9(-5)
^{54}Mn	<1.3(-6)	<1.3(-6)	<1.8(-6)	<2.2(-6)	<1.4(-6)
^{59}Fe	<2.0(-6)	<3.6(-6)	<3.4(-6)	<2.4(-6)	<3.4(-6)
^{57}Co	<2.6(-6)	<1.1(-5)	<2.2(-5)	<6.7(-7)	<1.0(-5)
^{58}Co	<1.5(-6)	<1.2(-6)	<3.3(-6)	<4.0(-6)	<8.4(-7)
^{60}Co	<2.0(-6)	<5.0(-6)	<3.9(-6)	<2.8(-6)	<3.0(-6)
^{65}Zn	<3.9(-7)	<2.6(-6)	<7.0(-7)	<3.0(-7)	<2.6(-6)
^{95}Zr	<6.2(-6)	<1.0(-5)	<2.5(-6)	<8.5(-7)	<2.6(-6)
^{95}Nb	<1.8(-6)	<1.6(-6)	<1.0(-6)	<4.9(-7)	<1.2(-6)
^{103}Ru	<1.7(-6)	<1.4(-6)	<5.1(-6)	<5.0(-7)	1.7 ± 0.9(-7)
^{106}Ru	<2.4(-5)	<7.3(-6)	<2.8(-5)	<6.1(-6)	<9.8(-6)
^{110m}Ag	<6.6(-7)	<1.6(-6)	<9.3(-7)	<9.1(-6)	<1.9(-6)
^{124}Sb	<1.3(-6)	<1.4(-6)	<6.6(-6)	<1.4(-6)	<9.1(-7)
^{125}Sb	<4.4(-6)	<1.8(-6)	<7.2(-6)	<2.8(-6)	<5.8(-6)
^{140}Ba	<4.7(-6)	<8.0(-6)	<1.7(-6)	<7.2(-6)	<2.6(-5)
^{140}La	<4.2(-5)	<1.1(-4)	<3.2(-5)	<1.4(-4)	<2.0(-5)
^{141}Ce	4.5 ± 2.6(-6)	5.9 ± 3.2(-6)	<4.0(-6)	<9.9(-7)	1.0 ± 0.5(-6)
^{152}Eu	<3.5(-6)	<8.3(-7)	<4.9(-6)	<4.9(-6)	<1.2(-6)
^{154}Eu	<3.0(-6)	<2.2(-5)	<5.7(-6)	<1.6(-6)	<3.8(-6)
^{155}Eu	<4.1(-5)	<1.4(-4)	<5.5(-5)	<2.6(-6)	5.4 ± 2.6(-5)

[C] Sample from 10/7 - 10/22/76

[D] Sample from 10/22 - 11/4/76

TABLE A.37 (Cont.)

VENTILATION AIRBORNE ACTIVITIES

SAMPLE STATION #3 (μCi/sec)

Sample Period	11/10 - 11/23/76	11/23 - 11/30/76[E]	11/30 - 12/15/76	12/15 - 1/6/77	1/6 - 1/20/77[F]
^{131}I	3.9 ± 0.1(-5)	4.3 ± 0.9(-6)	3.8 ± 0.6(-5)	7.0 ± 2.0(-7)	2.0 ± 0.6(-6)
^{134}Cs	<3.2(-6)	<2.1(-7)	<6.6(-7)	<3.6(-7)	4.2 ± 0.1(-7)
^{136}Cs	<1.2(-6)	<1.3(-7)	<1.7(-6)	<1.4(-7)	<1.1(-8)
^{137}Cs	5.0 ± 1.0(-6)	<2.7(-7)	1.8 ± 0.7(-6)	<3.0(-7)	4.6 ± 0.2(-7)
^{3}H	7.0 ± 1.0(-4)	<3.0(-4)	9.3 ± 0.1(-3)	2.3 ± 0.1(-3)	3.6 ± 0.2(-2)
^{14}C	1.56 ± 0.06(-3)	2.8 ± 0.2(-4)	4.3 ± 0.4(-3)	6.2 ± 0.4(-4)	1.62 ± 0.08(-3)
^{51}Cr	6.0 ± 4.0(-6)	<8.6(-4)	9.0 ± 4.0(-6)	<7.9(-7)	<2.8(-5)
^{54}Mn	<1.8(-6)	<2.6(-7)	<6.1(-7)	<1.3(-7)	4.4 ± 0.7(-8)
^{59}Fe	<2.3(-6)	<2.3(-6)	<1.4(-6)	<2.3(-6)	1.0 ± 0.2(-6)
^{57}Co	2.0 ± 1.0(-6)	<1.4(-7)	<2.4(-6)	<7.6(-8)	<4.0(-8)
^{58}Co	1.2 ± 0.2	<1.8(-6)	6.0 ± 1.0(-6)	2.0 ± 1.0(-7)	1.9 ± 0.3(-7)
^{60}Co	<8.6(-7)	1.0 ± 0.4(-7)	<1.9(-6)	<8.6(-8)	5.5 ± 0.5(-8)
^{65}Zn	<1.4(-6)	<4.9(-7)	<3.5(-7)	<2.0(-6)	<4.6(-8)
^{95}Zr	<1.8(-6)	<4.9(-6)	<2.1(-6)	<1.1(-6)	<1.7(-7)
^{95}Nb	<9.0(-7)	<2.8(-5)	<4.0(-6)	<2.0(-7)	<1.2(-6)
^{103}Ru	<2.4(-7)	<1.5(-5)	<2.8(-7)	<1.0(-7)	<1.8(-7)
^{106}Ru	<1.4(-5)	<3.5(-6)	<1.1(-5)	<1.3(-6)	<2.8(-7)
^{110m}Ag	<7.5(-7)	<4.0(-7)	<3.1(-7)	<1.2(-7)	<1.5(-8)
^{124}Sb	1.8 ± 0.9(-6)	<3.1(-6)	<1.0(-6)	<1.6(-7)	<2.0(-6)
^{125}Sb	3.0 ± 1.0(-6)	<4.5(-7)	<2.0(-6)	<3.2(-7)	2.4 ± 0.7(-8)
^{140}Ba	<2.7(-6)	<3.9(-7)	<2.2(-5)	<2.0(-7)	<7.8(-8)
^{140}La	<2.3(-5)	<1.0(-7)	<3.5(-7)	<1.3(-6)	<1.3(-8)
^{141}Ce	<8.7(-6)	<3.9(-5)	<1.4(-6)	<5.8(-8)	<2.1(-6)
^{152}Eu	<1.1(-6)	<4.5(-7)	<1.8(-6)	<7.8(-8)	8.0 ± 0.8(-8)
^{154}Eu	<2.1(-6)	<3.5(-7)	<9.7(-7)	<2.6(-6)	<6.1(-8)

[E] 24,201.9 second count
[F] 229,501.1 second count

TABLE A.37 (Cont.)

VENTILATION AIRBORNE ACTIVITIES

SAMPLE STATION #3 (μCi/sec)

Sample Period	1/20 - 2/3/77	2/3 - 2/16/77
^{131}I	1.6 ± 0.5(-6)	7.0 ± 2.0(-7)
^{134}Cs	<1.3(-7)	<1.5(-7)
^{136}Cs	<2.4(-7)	<3.6(-8)
^{137}Cs	<4.4(-7)	<4.9(-8)
^{3}H	2.0 ± 0.1(-3)	1.9 ± 0.3(-3)
^{14}C	1.38 ± 0.06(-3)	1.26 ± 0.08(-3)
^{51}Cr	<2.0(-6)	<2.4(-7)
^{54}Mn	<1.6(-7)	<9.9(-8)
^{59}Fe	<2.9(-6)	<2.9(-7)
^{57}Co	<1.6(-7)	<3.3(-8)
^{58}Co	<4.3(-7)	<2.6(-7)
^{60}Co	<4.2(-7)	<1.7(-7)
^{65}Zn	<2.0(-6)	<2.9(-7)
^{95}Zr	<2.2(-7)	<1.2(-7)
^{95}Nb	<2.2(-7)	<1.6(-7)
^{103}Ru	<1.8(-7)	<8.4(-8)
^{106}Ru	<2.2(-6)	<1.2(-6)
^{110m}Ag	<1.8(-7)	<1.6(-7)
^{124}Sb	<3.3(-7)	<1.8(-7)
^{125}Sb	<4.1(-7)	<1.3(-7)
^{140}Ba	<1.2(-6)	<2.8(-6)
^{140}La	<2.8(-6)	<5.6(-6)
^{141}Ce	<2.7(-7)	<6.4(-8)
^{152}Eu	<4.5(-7)	<3.6(-7)
^{154}Eu	<3.1(-6)	<3.2(-6)

TABLE A.38

VENTILATION AIRBORNE ACTIVITIES

SAMPLE STATION #4 (μCi/sec)

Sample Period	9/3 - 9/9/76	9/9 - 9/15/76	9/15 9/23/76	9/23 - 9/30/76	9/30 - 10/7/76
^{131}I	<4.5(-6)	1.6 ± 0.1(-4)	8.4 ± 0.7(-5)	<3.2(-6)	2.1 ± 0.4(-3)
^{134}Cs	<9.7(-6)	<1.8(-5)	<8.8(-6)	<1.0(-5)	<5.7(-6)
^{136}Cs	<8.4(-6)	<4.4(-5)	<6.8(-6)	<2.8(-6)	<4.9(-5)
^{137}Cs	4.6 ± 1.1(-5)	4.7 ± 2.0(-5)	4.0 ± 1.0(-5)	2.9 ± 0.8(-6)	3.6 ± 0.9(-5)
^{3}H	<1.0(-3)	<1.0(-3)	<1.0(-3)	<1.0(-3)	<1.0(-3)
^{14}C	3.0 ± 0.6(-4)	2.1 ± 0.4(-4)[A]		3.2 ± 0.6(-4)[B]	
^{51}Cr	<4.3(-5)	<1.7(-4)	<4.5(-5)	<3.4(-5)	<7.0(-5)
^{54}Mn	<2.7(-6)	<2.1(-6)	<9.3(-6)	<3.6(-6)	<3.0(-6)
^{59}Fe	<2.1(-5)	<2.1(-5)	<4.9(-6)	<1.0(-5)	<1.1(-5)
^{57}Co	<3.7(-5)	<1.7(-5)	2.6 ± 1.2(-5)	<2.8(-5)	<2.3(-5)
^{58}Co	<5.0(-6)	<8.0(-6)	<4.3(-6)	<6.7(-6)	<3.5(-6)
^{60}Co	<1.5(-5)	<1.4(-5)	<9.2(-6)	<1.1(-5)	<1.6(-5)
^{65}Zn	<1.0(-5)	<1.8(-5)	<6.7(-6)	<7.4(-6)	<8.2(-6)
^{95}Zr	<2.1(-5)	<1.1(-5)	<6.3(-6)	<4.4(-6)	<4.1(-6)
^{95}Nb	<7.1(-6)	<1.3(-5)	<1.5(-6)	<8.1(-6)	<7.5(-6)
^{103}Ru	<1.7(-5)	<4.4(-6)	<1.0(-5)	<6.4(-6)	<3.8(-6)
^{106}Ru	<7.5(-5)	<1.6(-5)	<2.2(-5)	<2.0(-5)	<2.6(-5)
^{110m}Ag	<9.6(-6)	<6.6(-6)	<9.5(-6)	<3.5(-6)	<3.7(-5)
^{124}Sb	<2.1(-5)	<4.5(-6)	<4.2(-6)	<1.5(-6)	<1.2(-5)
^{125}Sb	<1.4(-5)	<3.4(-5)	<1.5(-5)	<1.6(-5)	<1.8(-5)
^{140}Ba	<2.7(-5)	<1.5(-5)	<2.0(-5)	<3.0(-5)	<1.6(-5)
^{140}La	<1.4(-4)	<2.0(-4)	<8.3(-5)	<8.8(-5)	<1.0(-3)
^{141}Ce	4.2 ± 2.2(-5)	<5.7(-6)	<8.1(-6)	<4.6(-5)	<5.0(-5)
^{152}Eu	2.0 ± 1.0(-5)	<3.3(-5)	<2.0(-5)	<2.8(-5)	<1.2(-5)
^{154}Eu	<1.9(-5)	<5.9(-6)	<3.6(-5)	<9.6(-5)	<9.2(-6)
^{155}Eu	<5.4(-4)	<3.2(-4)	<3.2(-4)	<2.8(-4)	<6.3(-4)

[A] Sample from 9/9 - 9/23/76

[B] Sample from 9/23 - 10/7/76

TABLE A.38 (Cont.)

VENTILATION AIRBORNE ACTIVITIES

SAMPLE STATION #4 (μCi/sec)

Sample Period	10/7 - 10/14/76	10/14 - 10/22/76	10/22 - 10/28/76	10/28 - 11/4/76	11/4 - 11/10/76
^{131}I	2.3 ± 0.1(-4)	7.3 ± 0.7(-5)	7.6 ± 0.6(-5)	7.8 ± 0.7(-5)	8.0 ± 1.0(-5)
^{134}Cs	7.1 ± 4.0(-6)	<6.5(-6)	<2.3(-5)	<5.6(-6)	<9.8(-6)
^{136}Cs	<5.2(-6)	<5.3(-6)	<5.3(-6)	<1.4(-5)	<5.2(-6)
^{137}Cs	6.5 ± 2.3(-5)	2.5 ± 1.0(-5)	5.4 ± 2.4(-5)	3.2 ± 0.9(-5)	1.2 ± 0.7(-5)
^{3}H	<1.0(-3)	<1.0(-3)	2.3 ± 0.2(-2)	1.5 ± 0.3(-3)	<1.0(-3)
^{14}C	2.74 ± 0.07(-3)[C]		1.00 ± 0.07(-3)[D]		2.7 ± 0.1(-3)
^{51}Cr	<1.0(-5)	5.1 ± 2.8(-5)	<6.2(-5)	<2.6(-5)	4.8 ± 3.5(-5)
^{54}Mn	<8.7(-6)	<2.6(-6)	<8.9(-6)	<3.7(-6)	<2.9(-5)
^{59}Fe	<8.8(-5)	<1.1(-5)	<1.0(-4)	<5.7(-6)	<1.1(-5)
^{57}Co	3.2 ± 1.3(-5)	<2.1(-5)	<4.2(-5)	<7.2(-6)	<4.2(-5)
^{58}Co	<1.3(-5)	<4.5(-6)	<2.4(-6)	<1.0(-5)	<2.0(-5)
^{60}Co	<1.7(-5)	<1.2(-5)	<5.0(-6)	<9.5(-6)	<1.0(-5)
^{65}Zn	<6.1(-5)	<9.1(-7)	<9.2(-6)	<6.7(-6)	<3.4(-6)
^{95}Zr	<1.5(-5)	<4.0(-5)	<8.2(-6)	<3.0(-6)	<7.0(-6)
^{95}Nb	<2.1(-6)	<2.2(-6)	<6.3(-6)	<4.3(-6)	<7.9(-6)
^{103}Ru	<4.5(-6)	<3.1(-6)	<3.9(-6)	<7.1(-6)	<2.5(-5)
^{106}Ru	3.4 ± 1.6(-5)	<5.2(-5)	<4.3(-5)	<5.0(-5)	<9.7(-6)
^{110m}Ag	<7.5(-6)	<1.2(-5)	<4.3(-5)	<4.6(-6)	<5.4(-6)
^{124}Sb	<1.6(-5)	<3.1(-6)	<1.3(-5)	<8.3(-6)	<1.1(-5)
^{125}Sb	<2.3(-5)	<1.2(-5)	<4.3(-6)	<1.6(-5)	<5.4(-6)
^{140}Ba	<4.2(-5)	<1.2(-5)	<1.4(-5)	<1.8(-5)	<4.5(-5)
^{140}La	<1.6(-4)	<1.2(-3)	<5.8(-5)	<1.7(-4)	<2.6(-4)
^{141}Ce	<1.7(-5)	2.4 ± 1.2(-5)	<7.0(-5)	<2.1(-5)	<6.2(-5)
^{152}Eu	1.5 ± 0.9(-5)	<2.0(-5)	<1.1(-5)	<9.8(-6)	3.6 ± 1.8(-5)
^{154}Eu	<1.0(-5)	<1.2(-5)	<1.1(-4)	<1.3(-5)	<2.2(-5)
^{155}Eu	<5.9(-4)	9.6 ± 6.8(-5)	<8.7(-4)	<2.5(-4)	<3.3(-4)

[C] Sample from 10/7 - 10/22/76

[D] Sample from 10/22 - 11/4/76

TABLE A.38 (Cont.)

VENTILATION AIRBORNE ACTIVITIES

SAMPLE STATION #4 (μCi/sec)

Sample Period	11/10 - 11/23/76	11/23 - 11/30/76	11/30 - 12/15/76	12/15 - 1/6/77	1/6 - 1/20/77
^{131}I	8.0 ± 1.0(-5)	8.0 ± 0.6(-5)	1.8 ± 0.2(-4)	7.8 ± 0.5(-5)	6.5 ± 0.7(-5)
^{134}Cs	<5.0(-6)	<5.6(-6)	<6.5(-6)	1.55 ± 0.04(-4)	<6.5(-7)
^{136}Cs	<7.4(-6)	<3.1(-6)	<1.0(-5)	<2.1(-6)	<4.0(-7)
^{137}Cs	1.4 ± 0.4(-5)	2.0 ± 0.6(-5)	9.0 ± 3.0(-6)	1.82 ± 0.06(-4)	<1.1(-6)
^{3}H	7.4 ± 0.1(-2)	2.2 ± 0.1(-2)	1.8 ± 0.1(-2)	1.3 ± 0.1(-2)	9.3 ± 0.4(-3)
^{14}C	1.00 ± 0.04(-2)	4.6 ± 0.3(-3)	1.73 ± 0.04(-2)	3.6 ± 0.2(-3)	4.8 ± 0.2(-3)
^{51}Cr	<8.5(-6)	<2.2(-5)	<3.5(-5)	<3.7(-5)	<2.6(-6)
^{54}Mn	<3.7(-6)	<5.3(-6)	<2.9(-6)	<1.4(-6)	<3.5(-7)
^{59}Fe	<1.3(-5)	<1.9(-5)	<1.9(-5)	<6.7(-7)	<1.1(-5)
^{57}Co	<2.5(-6)	<2.6(-5)	1.0 ± 0.6(-5)	<1.6(-6)	<5.3(-7)
^{58}Co	6 ± 3(-6)	<4.6(-6)	2.4 ± 0.5(-5)	1.7 ± 0.6(-6)	<2.1(-6)
^{60}Co	<4.2(-7)	<1.2(-5)	<6.2(-6)	<2.5(-7)	<3.7(-7)
^{65}Zn	<1.5(-6)	<9.1(-6)	<1.4(-6)	<1.7(-6)	<7.4(-6)
^{95}Zr	<2.5(-5)	<4.3(-5)	<3.9(-6)	<1.7(-6)	<9.6(-7)
^{95}Nb	<2.5(-6)	<1.8(-6)	<3.5(-6)	<1.9(-6)	<4.5(-7)
^{103}Ru	<6.4(-6)	<8.2(-6)	<3.8(-6)	<4.8(-6)	<4.5(-7)
^{106}Ru	<3.8(-5)	<3.4(-5)	<4.8(-5)	<1.9(-5)	<2.5(-6)
^{110m}Ag	<1.6(-6)	<3.4(-5)	<7.8(-6)	<4.3(-7)	<3.4(-7)
^{124}Sb	<5.3(-6)	<1.3(-5)	<5.3(-6)	<1.3(-4)	<6.2(-7)
^{125}Sb	7.0 ± 4.0(-6)	<2.0(-5)	<5.7(-6)	<1.3(-5)	<9.4(-7)
^{140}Ba	<1.0(-5)	<5.9(-6)	<4.3(-5)	<1.6(-5)	<7.2(-7)
^{140}La	<4.9(-5)	<3.1(-4)	<1.9(-5)	<2.7(-6)	<5.1(-4)
^{141}Ce	<1.3(-5)	2.0 ± 1.0(-5)	<5.2(-5)	<3.6(-6)	<5.5(-7)
^{152}Eu	<9.0(-6)	<3.2(-5)	<7.7(-6)	<1.4(-5)	<6.8(-7)
^{154}Eu	<5.3(-6)	<1.5(-5)	<2.4(-6)	<1.9(-6)	<1.7(-6)

TABLE A.38 (Cont.)

VENTILATION AIRBORNE ACTIVITIES

SAMPLE STATION #4 (μCi/sec)

Sample Period	1/20 - 2/3/77	2/3 - 2/16/77
^{131}I	2.3 ± 0.1(-4)	5.0 ± 2.0(-6)
^{134}Cs	9.0 ± 1.0(-6)	<1.6(-6)
^{136}Cs	<4.4(-7)	<5.2(-6)
^{137}Cs	1.1 ± 0.2(-5)	<9.7(-7)
^{3}H	2.7 ± 0.1(-2)	1.1 ± 0.2(-2)
^{14}C	4.8 ± 0.3(-3)	5.2 ± 0.6(-3)
^{51}Cr	<4.8(-6)	<4.7(-6)
^{54}Mn	<7.1(-7)	<4.1(-7)
^{59}Fe	<1.7(-6)	<1.2(-6)
^{57}Co	<6.0(-7)	<2.4(-7)
^{58}Co	<1.6(-6)	<6.4(-7)
^{60}Co	<6.2(-7)	<7.2(-7)
^{65}Zn	<1.6(-6)	<8.6(-6)
^{95}Zr	<5.5(-6)	<5.2(-7)
^{95}Nb	<4.8(-7)	<5.3(-7)
^{103}Ru	<7.2(-7)	<9.4(-8)
^{106}Ru	<5.2(-6)	<4.1(-6)
^{110m}Ag	<4.7(-6)	<6.9(-7)
^{124}Sb	1.2 ± 0.2(-5)	<1.1(-6)
^{125}Sb	<3.3(-6)	<5.6(-7)
^{140}Ba	<5.8(-6)	<7.9(-7)
^{140}La	<1.2(-4)	<2.2(-5)
^{141}Ce	<7.3(-7)	<1.4(-7)
^{152}Eu	<5.4(-6)	<5.3(-6)
^{154}Eu	<1.3(-5)	<1.4(-5)

TABLE A.39

VENTILATION AIRBORNE ACTIVITIES

TABLE A.39

VENTILATION AIRBORNE ACTIVITIES
SAMPLE STATION WASTE EVAPORATOR (μCi/sec)

Sample Period	9/3 - 9/9/76	9/9 - 9/15/76	9/15 - 9/23/76	9/23 - 9/30/76	9/30 - 10/7/76
^{131}I	1.63 ± 0.01(-4)	4.70 ± 0.05(-5)	4.80 ± 0.04(-5)	1.92 ± 0.03(-5)	1.55 ± 0.03(-5)
^{134}Cs	<2.4(-7)	<6.4(-7)	<5.0(-7)	<6.6(-7)	<3.7(-7)
^{136}Cs	<2.4(-7)	<1.1(-6)	<1.0(-7)	<1.4(-7)	<3.2(-7)
^{137}Cs	<6.0(-7)	7.0 ± 4.0(-6)	1.0 ± 0.3(-7)	3.8 ± 1.9(-7)	<5.9(-8)
^{51}Cr	2.0 ± 1.0(-6)	<2.4(-6)	<1.1(-6)	<2.1(-6)	<1.9(-6)
^{54}Mn	<4.0(-7)	<3.4(-7)	<2.6(-7)	<9.8(-8)	<1.3(-7)
^{59}Fe	<6.7(-7)	<1.1(-6)	<2.8(-7)	<4.4(-6)	<5.7(-7)
^{57}Co	3.0 ± 1.0(-7)	<5.7(-8)	5.0 ± 3.0(-8)	<1.9(-7)	<8.2(-8)
^{58}Co	<6.0(-7)	<6.8(-7)	<5.4(-7)	<2.4(-7)	<2.6(-7)
^{60}Co	<4.2(-7)	<5.5(-7)	<4.5(-8)	<1.5(-7)	<3.0(-7)
^{65}Zn	<8.8(-8)	<1.8(-7)	<1.7(-7)	<1.4(-7)	<7.2(-7)
^{95}Zr	<7.7(-7)	<1.4(-7)	<3(-7)	<1.5(-7)	<5.2(-7)
^{95}Nb	<1.8(-7)	<2.1(-7)	<3.8(-7)	<3.7(-7)	<1.4(-7)
^{103}Ru	<2.7(-7)	<2.2(-7)	<1.7(-7)	<3.6(-7)	<3.1(-7)
^{106}Ru	<2.5(-6)	<1.8(-6)	<1.9(-6)	<4.9(-7)	<1.9(-6)
^{110m}Ag	<3.0(-7)	<3.0(-7)	<3.1(-7)	<4.6(-7)	<5.2(-7)
^{124}Sb	<9.6(-7)	<2.1(-7)	<5.4(-7)	<3.2(-7)	<4.4(-7)
^{125}Sb	<4.0(-7)	<1.3(-6)	<1.8(-7)	<2.5(-7)	<4.6(-7)
^{140}Ba	<5.5(-7)	<4.9(-7)	<1.0(-6)	<1.4(-6)	<1.0(-6)
^{140}La	<4.0(-6)	<4.9(-7)	<6.5(-6)	<5.0(-6)	<4.1(-5)
^{141}Ce	3.8 ± 1.7(-7)	<3.1(-7)	2.0 ± 1.0(-7)	1.4 ± 0.9(-8)	<1.4(-7)
^{152}Eu	<1.9(-7)	<3.8(-5)	<5.5(-7)	4.9 ± 3.7(-7)	<7.5(-7)
^{154}Eu	<3.0(-7)	<6.8(-6)	<2.6(-7)	<4.8(-6)	<9.9(-7)
^{155}Eu	3.0 ± 2.0(-7)	<5.1(-7)	<5.0(-7)	<2.3(-7)	<3.2(-7)

TABLE A.39 (Cont.)

VENTILATION AIRBORNE ACTIVITIES

SAMPLE STATION WASTE EVAPORATOR (μCi/sec)

Sample Period	10/7 - 10/14/76	10/14 - 10/22/76	10/22 - 10/27/76	10/27 - 11/4/76[A]	11/5 - 11/10/76
^{131}I	1.91 ± 0.02(-5)	3.21 ± 0.06(-5)	4.37 ± 0.07(-5)		1.51 ± 0.04(-5)
^{134}Cs	<9.1(-7)	<8.2(-7)	5.1 ± 2.7(-7)		<9.3(-7)
^{136}Cs	<4.9(-7)	<4.1(-7)	<2.7(-7)		<4.5(-7)
^{137}Cs	3.4 ± 0.6(-6)	1.6 ± 0.5(-6)	3.2 ± 0.7(-6)		8.4 ± 3.2(-7)
^{51}Cr	1.7 ± 1.1(-6)	<6.9(-6)	<4.6(-6)		<5.6(-6)
^{54}Mn	1.0 ± 0.3(-6)	<8.8(-7)	<3.1(-7)		<1.6(-7)
^{59}Fe	<9.4(-7)	<9.1(-7)	<6.2(-7)		<7.1(-7)
^{57}Co	9.6 ± 5.2(-8)	1.2 ± 0.7(-7)	<1.3(-7)		<9.3(-8)
^{58}Co	1.5 ± 0.4(-6)	1.7 ± 0.6(-7)	<8.6(-7)		<3.7(-7)
^{60}Co	6.3 ± 2.2(-7)	<4.2(-7)	<2.9(-7)		<3.8(-7)
^{65}Zn	<6.6(-8)	<5.1(-7)	<4.8(-7)		<9.3(-7)
^{95}Zr	<3.5(-7)	<5.0(-7)	<6.6(-7)		<2.9(-7)
^{95}Nb	<2.4(-7)	<1.0(-7)	<4.3(-7)		<1.6(-7)
^{103}Ru	<8.4(-7)	<2.1(-7)	<6.3(-7)		<1.0(-7)
^{106}Ru	3.1 ± 1.5(-6)	<7.5(-6)	<1.8(-6)		<1.0(-6)
^{110m}Ag	<1.5(-8)	<3.6(-7)	<6.9(-7)		<7.0(-7)
^{124}Sb	9.1 ± 3.2(-7)	<9.7(-7)	<5.6(-7)		<2.5(-7)
^{125}Sb	<8.5(-7)	<1.4(-6)	<1.2(-6)		<5.3(-7)
^{140}Ba	<1.2(-6)	<3.3(-6)	<3.1(-6)		<1.1(-6)
^{140}La	<8.4(-6)	<3.5(-4)	<1.3(-4)		<1.2(-5)
^{141}Ce	<3.5(-7)	<2.1(-7)	<2.5(-7)		1.4 ± 1.0(-7)
^{152}Eu	<7.0(-7)	<8.5(-7)	7.7 ± 4.4(-7)		<9.7(-7)
^{154}Eu	<4.7(-6)	<8.0(-7)	<1.2(-6)		<1.7(-6)
^{155}Eu	<3.0(-7)	<2.2(-7)	<2.0(-7)		<6.5(-7)

[A] No sample due to sampler motor malfunction

TABLE A.39 (Cont.)

VENTILATION AIRBORNE ACTIVITIES
SAMPLE STATION WASTE EVAPORATOR (μCi/sec)

Sample Period	11/11 - 11/23/76	11/23 - 11/30/76	11/30 - 12/15/76	12/15 - 1/6/77	1/6 - 1/20/77
^{131}I	1.22 ± 0.01(-5)	6.0 ± 0.1(-6)	6.0 ± 2.0(-6)	8.0 ± 2.0(-7)	<1.1(-7)
^{134}Cs	<3.7(-7)	<4.7(-7)	5.0 ± 2.0(-7)	2.9 ± 0.1(-6)	<1.6(-7)
^{136}Cs	<9.9(-8)	<2.4(-7)	<2.8(-7)	<9.9(-8)	<5.1(-7)
^{137}Cs	5.0 ± 3.0(-7)	6.0 ± 3.0(-7)	1.1 ± 0.3(-6)	3.5 ± 0.3(-6)	1.0 ± 0.5(-7)
^{51}Cr	<2.0(-6)	<6.7(-7)	<2.8(-6)	8.0 ± 4.0(-7)	<4.7(-7)
^{54}Mn	<2.6(-7)	<2.3(-7)	<3.7(-7)	1.5 ± 0.6(-7)	<6.2(-8)
^{59}Fe	<1.9(-7)	<9.2(-7)	<2.8(-7)	<8.5(-7)	<1.1(-7)
^{57}Co	8.0 ± 4.0(-8)	1.0 ± 0.5(-8)	4.0 ± 2.0(-8)	<6.5(-8)	<4.8(-7)
^{58}Co	4.0 ± 2.0(-7)	<2.7(-7)	1.0 ± 0.3(-6)	6.0 ± 1.0(-7)	2.1 ± 0.7(-7)
^{60}Co	<5.8(-7)	<4.0(-7)	<4.2(-7)	<1.1(-7)	<9.9(-8)
^{65}Zn	<2.1(-7)	<2.2(-7)	<2.0(-7)	<8.5(-8)	<6.7(-8)
^{95}Zr	<3.2(-7)	<1.7(-7)	<8.5(-8)	<6.8(-8)	<6.2(-7)
^{95}Nb	<1.3(-7)	<1.7(-7)	<1.4(-7)	<8.5(-8)	<3.7(-7)
^{103}Ru	<4.7(-8)	<1.9(-7)	<2.5(-7)	<2.7(-7)	<5.7(-8)
^{106}Ru	<6.2(-7)	<1.4(-6)	<3.2(-6)	<1.3(-6)	<7.1(-7)
^{110m}Ag	<1.1(-6)	<1.6(-7)	<1.8(-7)	<7.8(-8)	<7.1(-8)
^{124}Sb	<3.2(-7)	<2.3(-7)	1.5 ± 0.4(-6)	<2.3(-6)	<1.8(-7)
^{125}Sb	<1.8(-7)	<8.5(-7)	<7.1(-7)	1.4 ± 0.7(-7)	<8.5(-8)
^{140}Ba	<4.4(-7)	<1.2(-6)	<4.0(-7)	<7.1(-7)	<2.8(-7)
^{140}La	<5.2(-6)	<1.4(-5)	<7.1(-8)	<7.8(-7)	<1.8(-5)
^{141}Ce	<8.0 ± 4.0(-8)	<2.1(-7)	<2.6(-7)	6.0 ± 3.0(-8)	<7.8(-9)
^{152}Eu	<6.2(-7)	<7.1(-7)	<4.4(-7)	<5.4(-7)	<5.5(-7)
^{154}Eu	<4.7(-7)	<5.3(-7)	<2.1(-7)	<9.2(-8)	<1.3(-7)

TABLE A.39 (Cont.)

VENTILATION AIRBORNE ACTIVITIES

SAMPLE STATION WASTE EVAPORATOR (μCi/sec)

Sample Period	1/20 - 2/3/77	2/3 - 2/15/77
^{131}I	2.5 ± 0.4(-6)	1.03 ± 0.07(-6)
^{134}Cs	3.5 ± 0.7(-7)	2.1 ± 0.4(-7)
^{136}Cs	<2.0(-8)	<5.7(-8)
^{137}Cs	4.0 ± 1.0(-7)	4.3 ± 0.4(-7)
^{51}Cr	<5.2(-7)	<3.5(-7)
^{54}Mn	<4.0(-8)	7.0 ± 2.0(-8)
^{59}Fe	<1.3(-6)	<1.4(-7)
^{57}Co	<1.1(-8)	<1.4(-8)
^{58}Co	4.0 ± 1.0(-7)	2.8 ± 0.3(-7)
^{60}Co	<5.4(-7)	1.6 ± 0.3(-7)
^{65}Zn	<7.8(-8)	<1.4(-7)
^{95}Zr	<1.3(-7)	<1.4(-7)
^{95}Nb	<2.3(-8)	<5.7(-7)
^{103}Ru	<9.2(-8)	<7.1(-8)
^{106}Ru	<1.2(-6)	<6.4(-7)
^{110m}Ag	<3.2(-8)	<6.4(-8)
^{124}Sb	<4.0(-7)	<6.4(-8)
^{125}Sb	<8.5(-8)	7.0 ± 2.0(-8)
^{140}Ba	<1.9(-7)	<2.8(-7)
^{140}La	<1.0(-6)	<2.1(-7)
^{141}Ce	<2.9(-8)	<2.8(-8)
^{152}Eu	<9.2(-8)	<1.4(-7)
^{154}Eu	<1.5(-6)	4.0 ± 3.0(-8)

TABLE A.40

VENTILATION AIRBORNE ACTIVITIES (μCi/sec)

PIPE PENETRATION ROOM

Sample Period	2/3 - 2/10/77	2/10 - 2/11/77
^{131}I	7.9 ± 0.8(-6)	1.4 ± 0.6(-6)
^{134}Cs	<1.2(-7)	<1.3(-6)
^{136}Cs	<9.9(-8)	<2.6(-7)
^{137}Cs	<1.2(-7)	<1.2(-6)
^{3}H	[A]	[A]
^{51}Cr	<9.9(-7)	<4.6(-6)
^{54}Mn	<9.9(-8)	<4.2(-7)
^{59}Fe	<9.9(-7)	<7.5(-6)
^{57}Co	<1.6(-7)	<3.0(-7)
^{58}Co	<9.9(-8)	<3.2(-7)
^{60}Co	1.0 ± 0.2(-6)	<5.0(-7)
^{65}Zn	<7.9(-7)	<5.2(-6)
^{95}Zr	<2.0(-7)	<3.4(-7)
^{95}Nb	<9.9(-8)	<2.0(-7)
^{103}Ru	<1.2(-7)	<3.0(-7)
^{106}Ru	<1.2(-6)	<1.1(-6)
^{110m}Ag	<1.2(-7)	<2.8(-6)
^{124}Sb	<1.2(-7)	<7.3(-7)
^{125}Sb	<4.0(-7)	<5.2(-7)
^{140}Ba	<5.9(-7)	<5.7(-7)
^{140}La	<5.9(-7)	<5.0(-5)
^{141}Ce	<4.0(-7)	<8.7(-7)
^{152}Eu	<4.0(-7)	<1.5(-6)
^{154}Eu	<1.2(-6)	<8.3(-6)

[A] Not Measured

TABLE A.41

VENTILATION AIRBORNE ACTIVITIES (μCi/sec)

LETDOWN HEAT EXCHANGER ROOM

Sample Period	2/3 - 2/11/77	2/11 - 2/17/77
^{131}I	6.8 ± 0.4(-6)	3.0 ± 0.5(-6)
^{134}Cs	1.0 ± 0.4(-7)	<1.3(-7)
^{136}Cs	<1.4(-7)	<9.9(-8)
^{137}Cs	2.7 ± 0.5(-7)	<3.2(-7)
^{3}H	[A]	[A]
^{51}Cr	<1.4(-6)	<7.0(-7)
^{54}Mn	<1.4(-7)	<1.3(-7)
^{59}Fe	<3.5(-7)	<3.0(-7)
^{57}Co	<3.5(-8)	<6.4(-8)
^{58}Co	2.5 ± 0.6(-7)	<2.0(-7)
^{60}Co	2.7 ± 0.6(-7)	<1.3(-7)
^{65}Zn	<2.8(-7)	<2.0(-7)
^{95}Zr	<2.8(-7)	<1.3(-7)
^{95}Nb	<1.4(-7)	<1.3(-7)
^{103}Ru	<1.4(-7)	<1.2(-7)
^{106}Ru	<1.4(-6)	<1.3(-6)
^{110m}Ag	<1.4(-7)	<2.5(-7)
^{124}Sb	<1.4(-7)	<1.4(-7)
^{125}Sb	<2.8(-7)	<2.8(-7)
^{140}Ba	<3.5(-7)	<4.8(-7)
^{140}La	<5.7(-7)	<1.3(-5)
^{141}Ce	<6.4(-8)	<2.5(-7)
^{152}Eu	<4.2(-7)	<3.2(-7)
^{154}Eu	<4.2(-7)	<3.3(-7)

[A] Not Measured

TABLE A.42

^{131}I SPECIES DATA

Sample Station #2

Sample Dates:	Fractional ^{131}I Distribution (%) Particulate Filter	I_2	HOI	Organic
9/3 - 9/9/76	2.3	30.1	29.5	38.0
9/9 - 9/15/76	0.8	44.6	13.1	41.5
9/15 - 9/18/76	25.4	42.0	6.4	26.2
9/23 - 9/30/76	8.2	44.1	3.4	44.3
9/30 - 10/7/76	1.6	24.4	41.9	32.2
10/7 - 10/14/76	8.9	32.4	11.9	40.8
10/14 - 10/15/76	7.5	48.6	20.6	23.3
10/22 - 10/28/76	9.2	29.2	15.6	46.0
10/28 - 11/4/76	20.2	25.5	12.1	42.2
11/4 - 11/10/76	7.2	30.2	22.2	40.3

TABLE A.42 (cont.)

^{131}I SPECIES DATA

Sample Station #2

Sample Dates			Fractional ^{131}I Distribution (%) Particulate Filter	I_2	HOI	Organic
11/10	-	11/23/76	11.3	32.0	23.9	32.9
11/23	-	11/30/76	0.7	41.3	1.1	56.9
12/1	-	12/15/76	4.4	6.6	14.0	75.0
12/15/76	-	1/6/77	4.5	61.3	8.6	25.6
1/6	-	1/20/77	ND	18.1	13.6	68.2
1/20	-	2/3/77	ND	33.5	66.5[A]	----
2/3	-	2/16/77	ND	23.1	ND	76.8

[A] represents HOI + organic

ND - Iodine activity not detected on this sample component.

TABLE A.43

^{131}I SPECIES DATA

Waste Evaporator Room

Sample Dates:	Fractional ^{131}I Distribution (%) Particulate Filter	I_2	HOI	Organic
8/27 - 9/3/76	5.3	75.0	11.5	8.1
9/3 - 9/9/76	7.0	69.7	10.2	13.1
9/9 - 9/15/76	2.4	1.2	80.2	16.1
9/15 - 9/23/76	20.1	45.6	18.9	15.3
9/23 - 9/30/76	23.4	52.4	12.4	11.7
9/30 - 10/7/76	6.3	40.5	11.3	41.9
10/7 - 10/14/76	33.5	20.7	14.0	31.8
10/14 - 10/22/76	10.3	44.0	18.9	26.8
10/22 - 10/27/76	21.0	23.7	15.8	39.5
11/4 - 11/10/76	21.5	11.6	10.8	56.0

TABLE A.43 (cont.)

^{131}I SPECIES DATA
Waste Evaporator Room

Sample Dates			Fractional ^{131}I Distribution (%) Particulate Filter	I_2	HOI	Organic
11/11	-	11/23/76	8.4	22.7	19.8	49.1
11/23	-	11/30/76	11.5	22.2	22.2	44.0
11/30	-	12/15/76	----	----	----	100 [A]
12/15/76	-	1/6/77	ND	46.6	ND	53.4
1/6	-	1/20/77	ND	ND	ND	ND
1/20	-	2/3/77	ND	59.6	ND	40.4
2/3	-	2/15/77	11.7	45.4	16.2	26.6

[A] Iodine activity only visible as organic due to excessive decay period prior to analysis.

ND - Iodine activity not detected on this sample component.

TABLE A.44

VENTILATION IODINE SPECIES COMPARISONS

Sample	Date	^{131}I conc.(μCi/sec)	Fractional ^{131}I Distribution (%)			
			part. filter	I_2	HOI	Organic
Station #1	2/3 - 2/10/77	1.76 ± 0.07(-4)	8.7	69.2	13.8	14.5
	2/10 - 2/17/77	1.9 ± 0.1(-4)	6.0	26.0	4.8	63.1
Letdown Heat Exchanger Room	2/3 - 2/11/77	6.8 ± 0.4(-6)	4.2	19.4	5.0	71.4
	2/11 - 2/17/77	3.0 ± 0.5(-6)	ND	ND	ND	100
Pipe Penetration Room	2/3 - 2/10/77	7.9 ± 0.8(-6)	ND	54.5	35.6	1.0
	2/10 - 2/17/77	1.4 ± 0.6(-6)	ND	100	ND	ND

ND - Iodine activity not detected on this sample component.

TABLE A.45

^{131}I SPECIES MEASUREMENTS OF PROCESS AND COVER GAS
Power Operations - Before Refueling

Decay Corrected To:	Total ^{131}I (μCi/cc)	Fractional ^{131}I Species: Particulate Filter (%)	I_2 (%)	HOI (%)	Organic (%)
Waste Gas Decay Tank "B" Isolated: 01:30; 8/19/76					
Sampled: 11:05; 8/26/76					
@11:00; 8/26/76	3.0 (-7)	0	1.2	15.2	83.5
Sampled: 10:52; 8/31/76					
@11:00; 8/26/76	3.7 (-7)	0.5	1.2	12.3	86.0
Cover Gases:					
Waste Holdup "B" Tank:					
Sampled: 15:17; 8/26/76					
@16:00; 8/26/76	4.0 (-6)	0	0.7	9.3	90.0
Neutralization Tank:					
Sampled: 15:54; 8/26/76					
@17:00; 8/26/76	1.5 (-5)	0	1.4	17.2	81.4
Spent Resin Storage Tank:					
Sampled: 09:05; 8/27/76					
@11:00; 8/27/76	2.4 (-7)	0.5	0	43.5	56.0
Aux. Bldg. Sump Tank:					
Sampled: 09:25; 9/1/76					
@11:00; 9/1/76	9.0 (-6)	0	0.2	11.0	88.8
System Header					
Sampled: 09:16; 8/27/76					
@11:00; 8/27/76	4.6 (-6)	1.1	0.6	30.0	68.2

TABLE A.46

^{131}I SPECIES MEASUREMENTS OF PROCESS AND COVER GAS During Refueling and Power Operations After Refueling

Sample location	Date	Corrected to			Fractional Distribution (%) Part. Filter	I_2	HOI	Organic	Total(μCi/cc)
Waste Gas Decay Tank "A" isolated 1240 10/2/76	10/13/76	1457	10/13/76	^{131}I	0.2	0.9	18.6	80.3	2.14 ± 0.03(-7)
Waste Gas Decay Tank "D" isolated 2104 10/11/76	10/12/76	1600	10/12/76	^{131}I	ND	0.3	17.6	82.1	1.01 ± 0.01(-6)
				^{134}I	73.9	25.8	ND	ND	1.0 ± 0.5(-9)
				^{135}I	ND	ND	ND	100	1.3 ± 0.9(-9)
Waste Holdup Tank A 30% full	2/10/77	1115	2/10/77	^{131}I	ND	0.1	13.6	86.2	1.84 ± 0.02(-7)
				^{133}I	ND	ND	11.7	88.3	5.6 ± 0.5(-9)
Neutralization Tank 50% full	2/10/77	1245	2/10/77	^{131}I	0.07	ND	8.5	91.7	1.72 ± 0.03(-6)
				^{133}I	ND	ND	2.9	97.1	8 ± 1(-8)
Spent Resin Storage Tank 80% full	2/10/77	1400	2/10/77	^{131}I	ND	24.1	ND	75.9	3.8 ± 0.5(-9)
				^{133}I	ND	100	ND	ND	3 ± 1(-9)
				^{135}I	ND	100	ND	ND	7 ± 5(-9)
Vent Collection Header	2/11/77	0850	2/11/77	^{131}I	ND	2.5	25.0	72.5	2.4 ± 0.1(-8)
				^{133}I	ND	ND	ND	100	2.5 ± 0.6(-9)
Auxiliary Bld. Sump Tank	2/11/77	1000	2/11/77	^{131}I	ND	0.5	8.9	90.7	7.1 ± 0.1(-7)
				^{133}I	ND	0.7	11.9	87.4	4.0 ± 0.1(-8)
Reactor Coolant Drain Tank	2/11/77	1057	2/11/77	^{131}I	0.1	2.5	28.3	69.0	8.11 ± 0.08(-7)
				^{133}I	0.1	3.0	27.7	69.1	1.78 ± 0.04(-7)
				^{135}I	ND	ND	39.6	60.3	1.2 ± 0.4(-8)

ND - iodine activity not detected on this sample component.

TABLE A.47

WASTE GAS DECAY TANK "B" ACTIVITIES

Tank Isolated 15:15 10/4/76 - Results decay corrected to 17:00 10/13/76

Iodine Species Sample Taken 16:12 10/13/76

	Fractional ^{131}I Distribution (%)				
	Particulate Filter	I_2	HOI	Organic	Total (μCi/cc)
^{131}I	0.08	1.9	26.3	71.7	3.86 ± 0.06(-7)

250 cc gas bomb taken 17:15 10/13/76

^{131m}Xe	3.8 ± 0.03(-2)
^{133}Xe	1.04 ± 0.05(0)
^{133m}Xe	1.57 ± 0.05(-3)
^{85}Kr	1.38 ± 0.03(-1)

TABLE A.48

WASTE GAS DECAY TANK "A" ACTIVITIES

Tank isolated 17:15 1/20/77 - Results decay corrected to 09:26 2/10/77

Iodine Species Sample Taken 08:41 2/10/77

	Fractional ^{131}I Distribution (%)				
	Particulate Filter	I_2	HOI	Organic	Total (μCi/cc)
^{131}I	ND	ND	1.8	98.2	3.98 ± 0.08(-8)

250 cc gas bomb taken 09:04 2/10/77

^{131m}Xe	1.34 ± 0.07(-3)
^{133}Xe	5.49 ± 0.05(-3)
^{85}Kr	1.34 ± 0.03(-2)

ND - Iodine activity not detected on this sample component.

TABLE A.49

PROCESS GAS ^{14}C AND ^{3}H TRITIUM MEASUREMENTS

During Refueling

Waste Gas Decay Tank "A" Isolated: 12:46; 10/2/76

Sample Time	Total ^{14}C ($\mu Ci/cc$)	Fraction Not Oxidized (%)	HTO ($\mu Ci/cc$)	HT or RT ($\mu Ci/cc$)
1506; 10/17/76	7.29 ± 0.01(-4)	89	9.63 ± 0.02(-6)	3.86 ± 0.01(-4)

TABLE A.50

PROCESS AND COVER GAS ^{14}C AND ^{3}H ACTIVITIES
Power Operations - After Refueling

Sample location	Date	Total ^{14}C (μCi/cc)	Fraction Not Oxidized (%)	HTO (μCi/cc)	HT or RT (μCi/cc)
Waste Holdup Tank A	2/10/77	7.40 ± 0.05(-6)	90	3 ± 1(-8)	<2(-8)
Neutralization Tank	2/10/77	2.68 ± 0.02(-5)	97	1.1 ± 0.2(-7)	<2(-8)
Spent Resin Storage	2/10/77	8.22 ± 0.08(-6)	65	<2(-8)	<2(-8)
Vent Collection Header	2/11/77	8.7 ± 0.4(-7)	89	<2(-8)	<2(-8)
Reactor Coolant Drain Tank	2/11/77	2.65 ± 0.06(-6)	91	<2(-8)	<2(-8)
Waste Gas Decay Tank A	2/11/77	2.54 ± 0.01(-4)	92	<2(-8)	<2(-8)

TABLE A.51

CONTAINMENT ACTIVITIES

Date		Fraction Distribution (%): Particulate Filter	I_2	HOI	Organic	Total ($\mu Ci/cc$)
10/12/76	^{131}I	[A]	23.5	22.4	54.1	4.68 ± 0.07(-8)[A]
2/16/77-2/18/77	^{131}I	13.9	41.0	4.8	40.3	5.46 ± 0.05(-11)
	^{133}I	35.6	56.0	8.4	ND	1.4 ± 0.5(-11)
	^{58}Co					4 ± 1(-14)
	^{60}Co					2.7 ± 0.6(-14)
	^{134}Cs					2.29 ± 0.06(-12)
	^{137}Cs					2.94 ± 0.09(-12)
	^{54}Mn					1.0 ± 0.5(-14)
	^{3}H as HTO					2.00 ± 0.01(-8)
	^{3}H as HT or RT					2.24 ± 0.01(-8)
	^{14}C oxidized					5.60 ± 0.01(-8)
	^{14}C unoxidized					[B]

[A] filter not analyzed, assumes no activity on filter.

[B] sample lost during analysis

ND - iodine activity not detected on this sample component.

Figure A.1

Reactor Power Level: Fort Calhoun Station

Period: 8/15 - 10/3/76

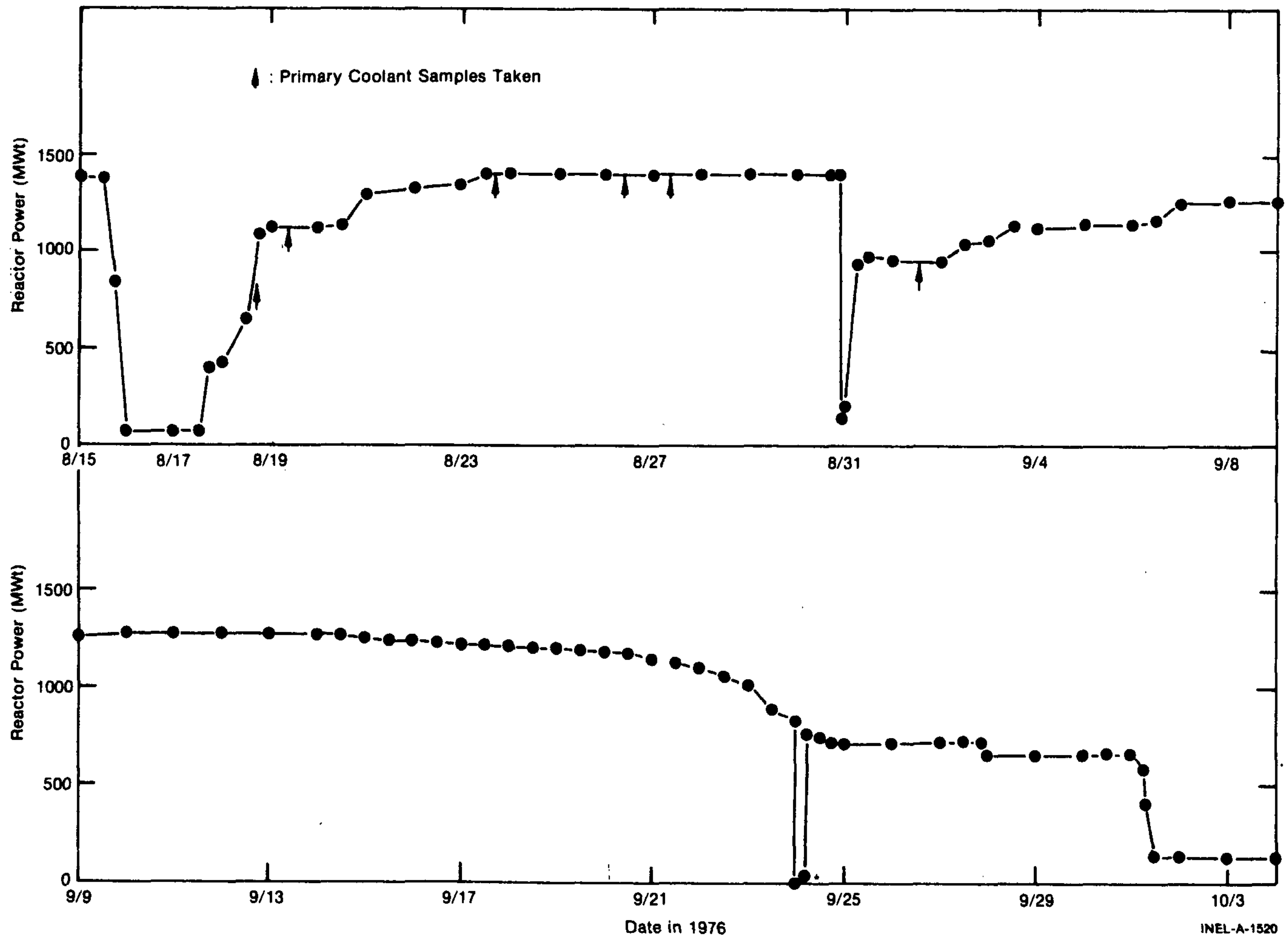


Figure A.2

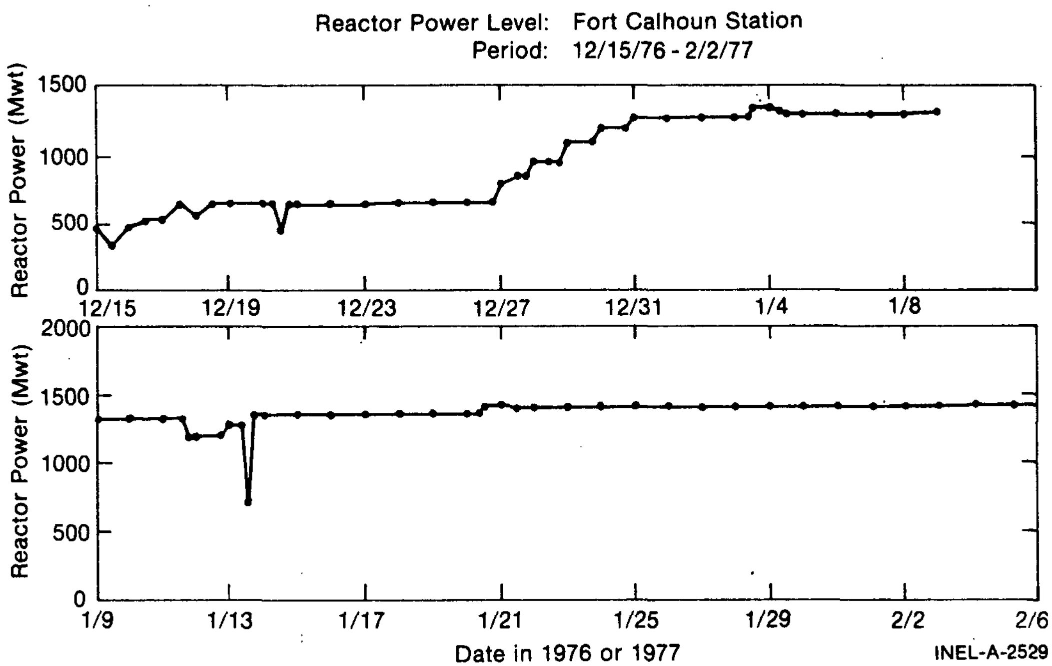

Reactor Power Level: Fort Calhoun Station
Period: 12/15/76 - 2/2/77
Reactor Power (Mwt)
1500
1000
500
0
12/15
12/19
12/23
12/27
12/31
1/4
1/8
Reactor Power (Mwt)
2000
1500
1000
500
0
1/9
1/13
1/17
1/21
1/25
1/29
2/2
2/6
Date in 1976 or 1977
INEL-A-2529

Figure A.2 (cont.)

Reactor Power Level: Fort Calhoun Station
Period: 2/3 - 24/77

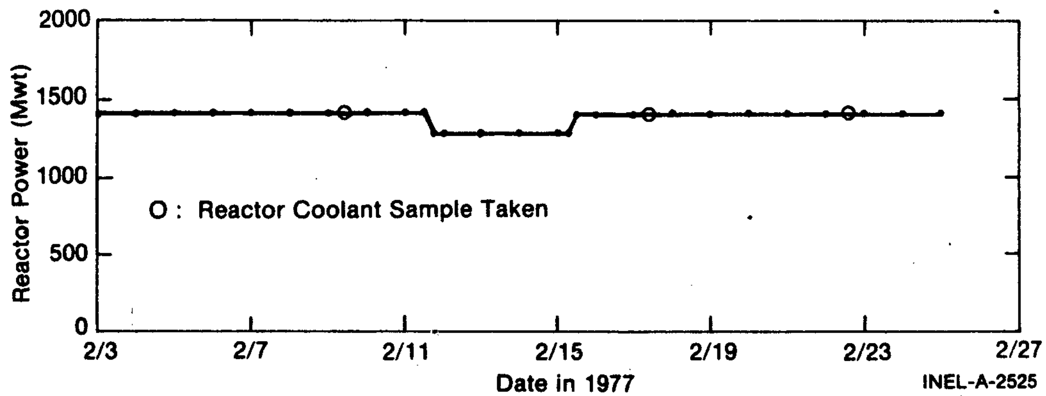
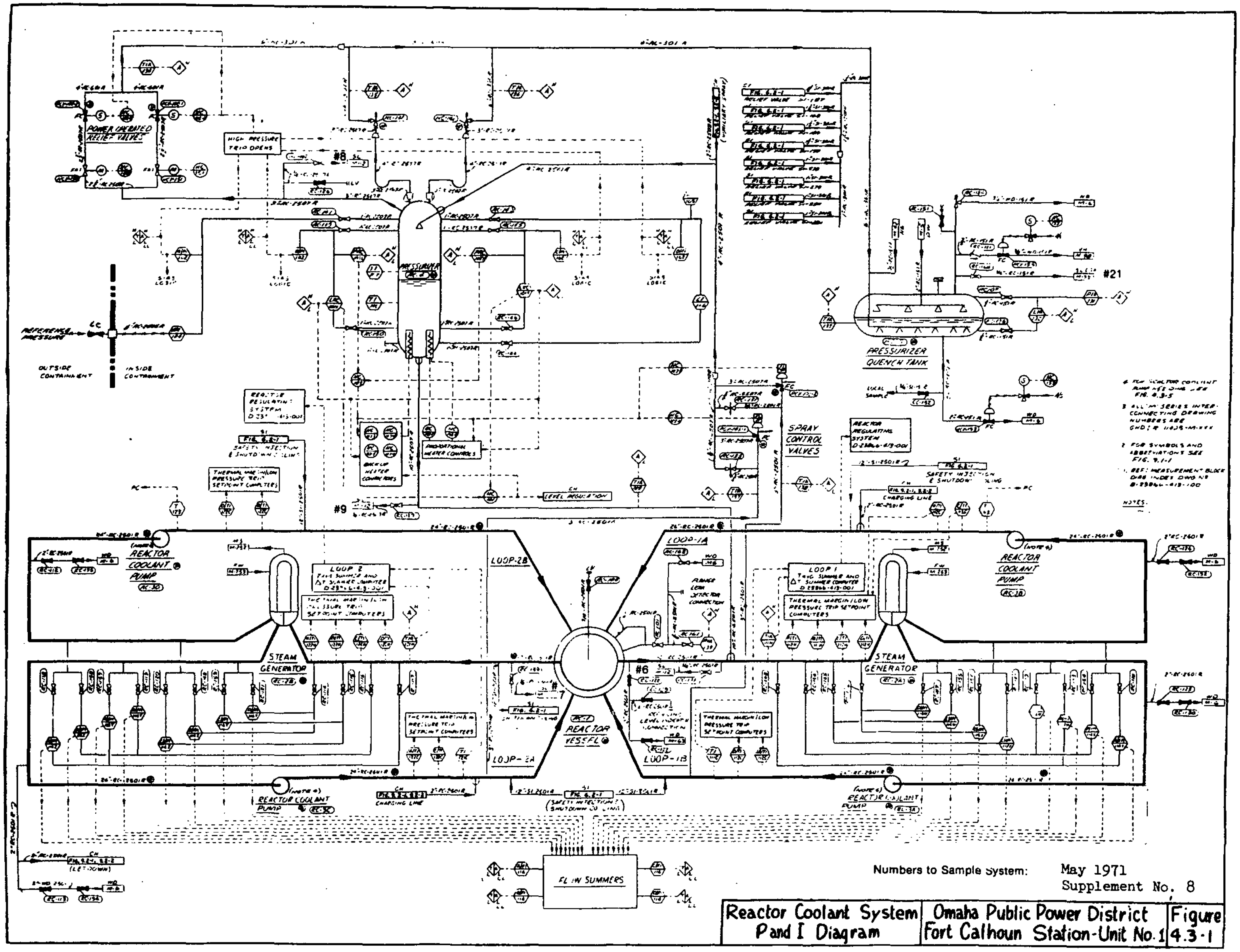
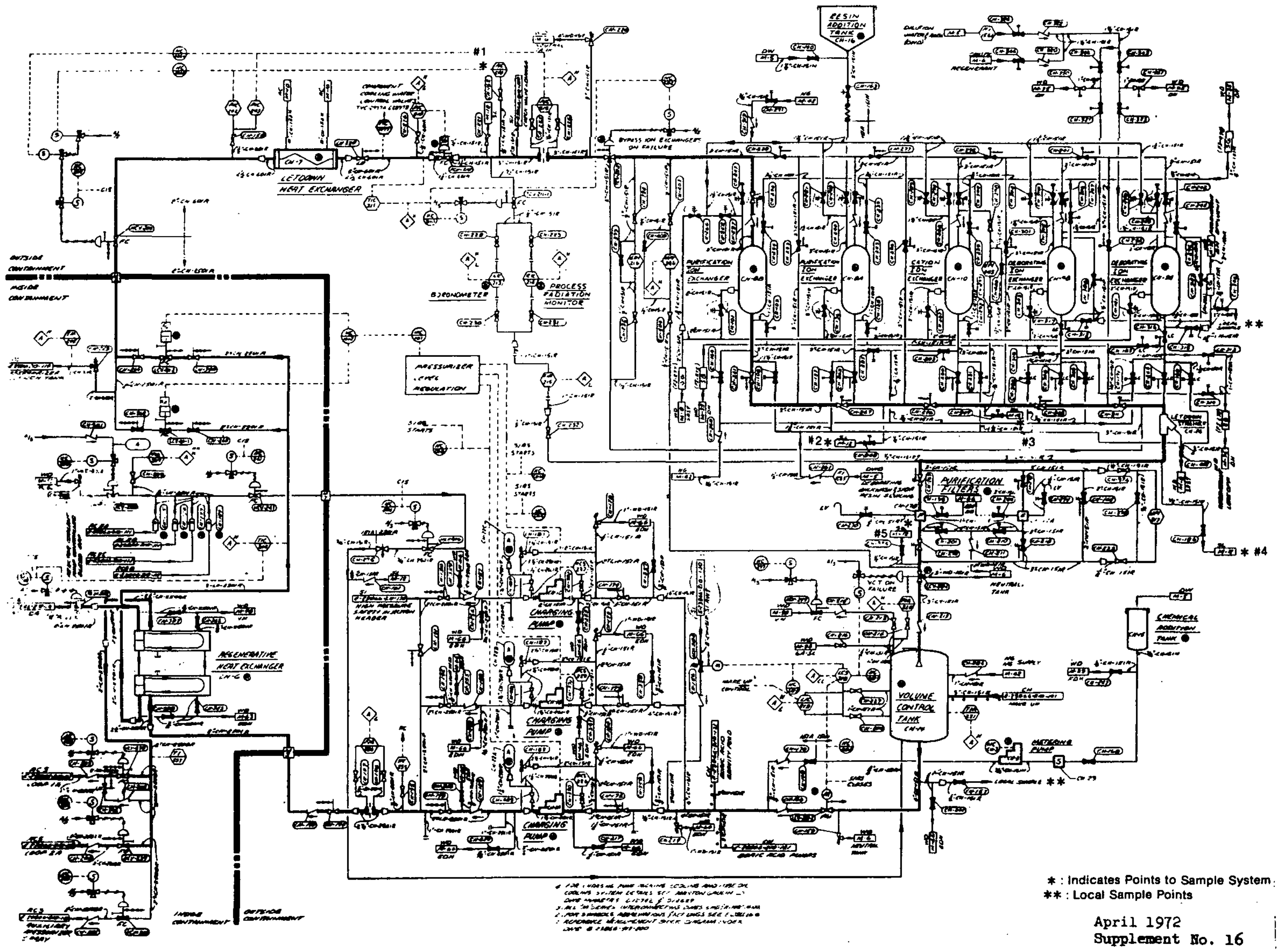


Figure A.3



POWER OPERATED RELIEF VALVE
HIGH PRESSURE TRIP OPENS
PRESSURIZER
REFERENCE PRESSURE
OUTSIDE CONTAINMENT
INSIDE CONTAINMENT
PROPORTIONAL HEATER CONTROLS
BACK-UP HEATER CONTROLS
LEVEL REGULATION
SPRAY CONTROL VALVES
PRESSURIZER QUENCH TANK
#21
#8
#9
#6
LOOP-2B
LOOP-1A
LOOP-2A
LOOP-1B
REACTOR COOLANT PUMP
STEAM GENERATOR
REACTOR VESSEL
FLUX SUMMERS
CHARGING LINE
LET-DOWN
Numbers to Sample System:
May 1971
Supplement No. 8
Reactor Coolant System
P and I Diagram
Omaha Public Power District
Fort Calhoun Station-Unit No. 1
Figure
4.3-1

Figure A.?



* : Indicates Points to Sample System
** : Local Sample Points

April 1972
Supplement No. 16

Chemical and Volume Control System P and I Diagram - Sheet 1	Omaha Public Power District Fort Calhoun Station-Unit No. 1	Figure 9.2-2

Figure A.5

CVCS Demineralizers

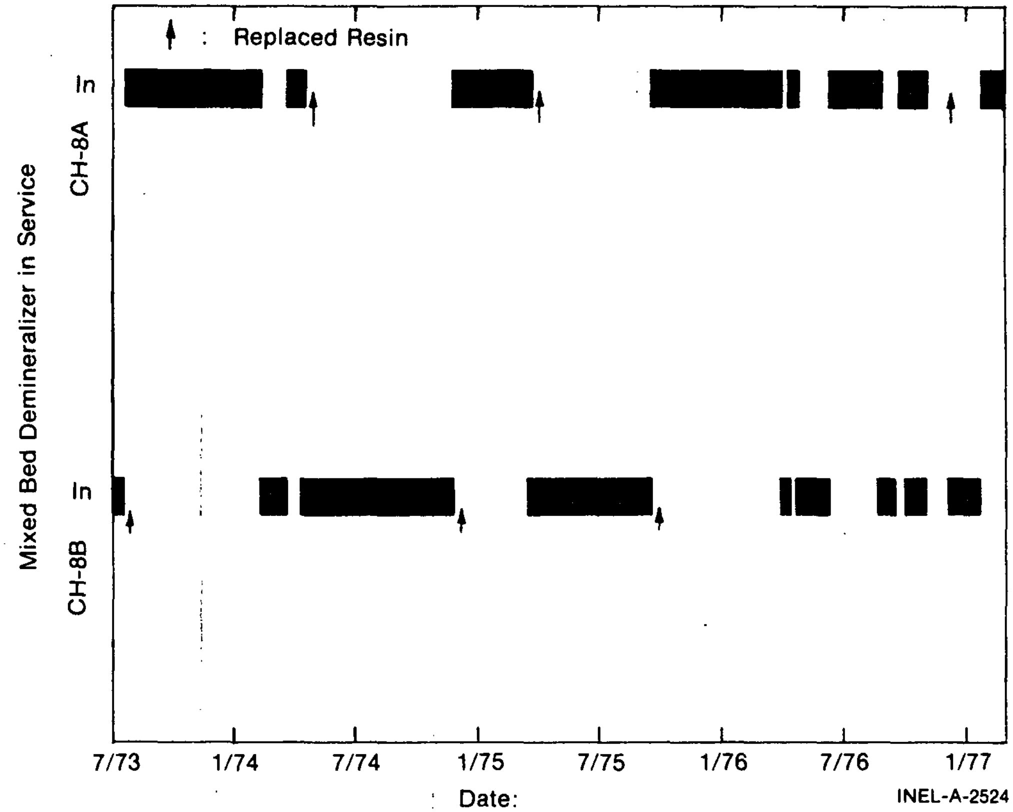


Figure A.6

Letdown Flow Rate: Power Operations - Before Refueling
Period: 8/15 - 9/9/76

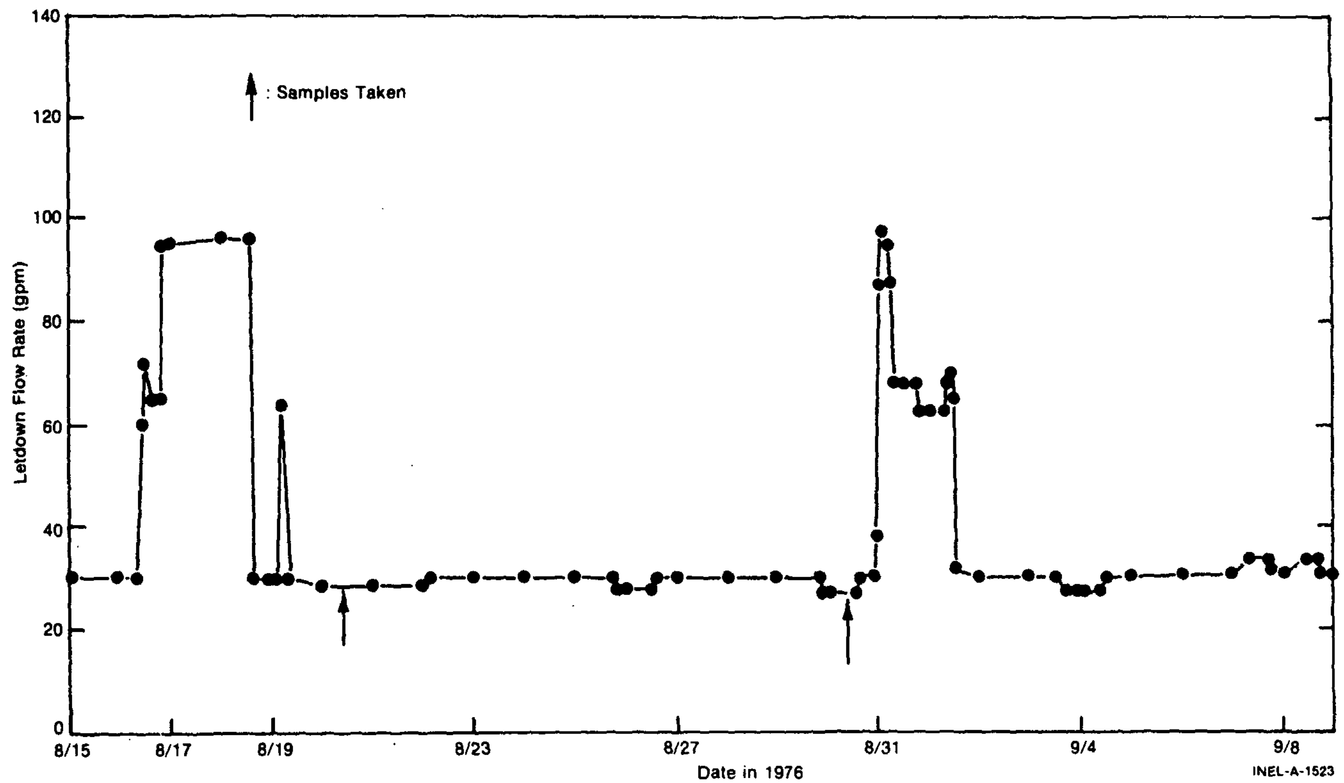


Figure A.7

CVCS LETDOWN PARAMETERS
Period: 8/9 - 9/3/76

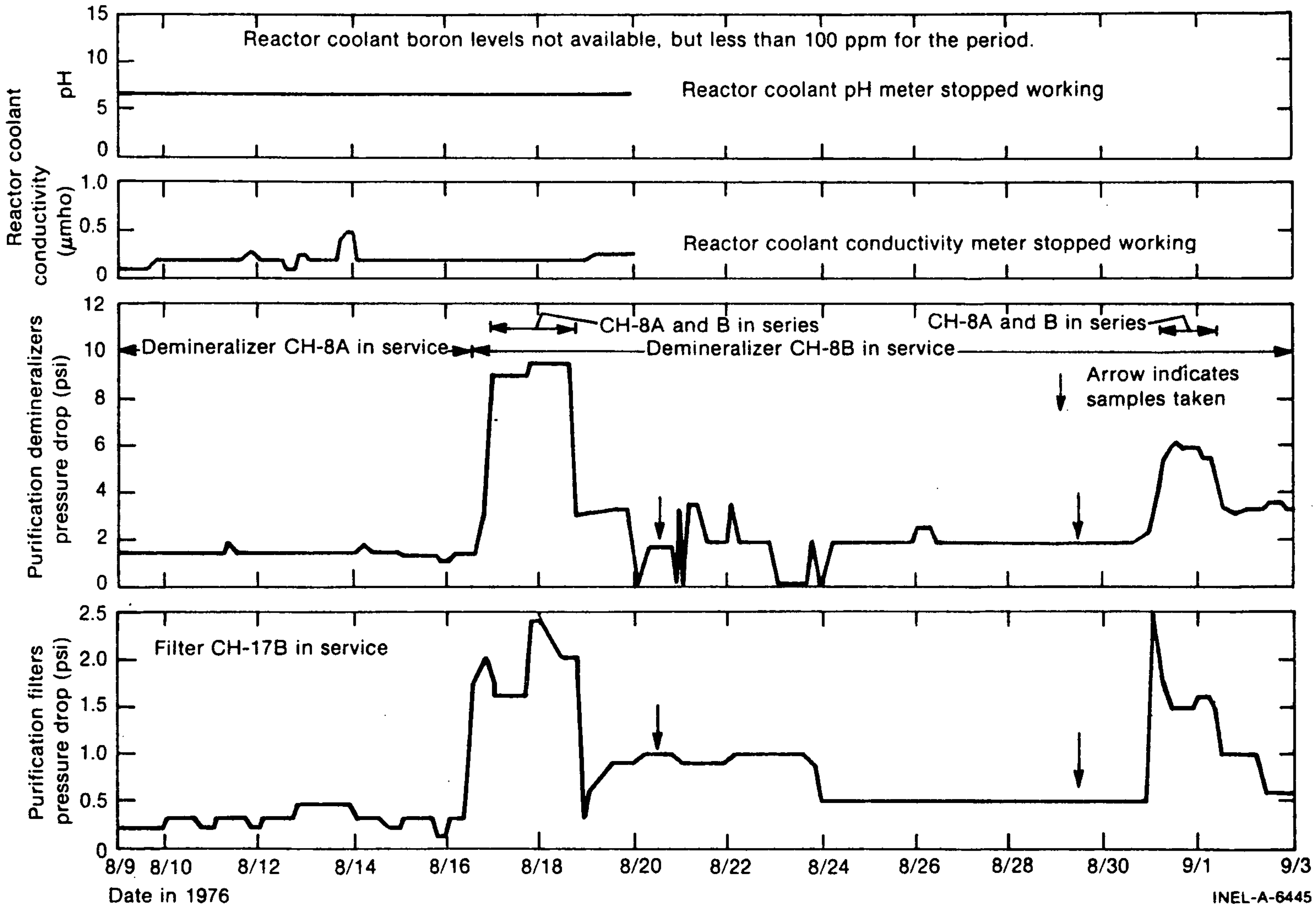


Figure A.8

Letdown Flow Rate: During Refueling
Period 9/25 - 10/19/76

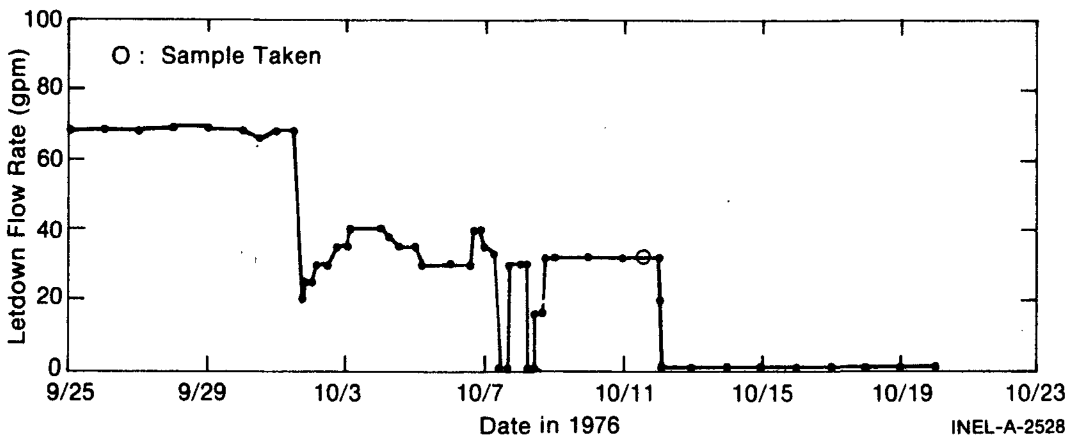


Figure A.9

Letdown Flow Rate: Power Operations - After Refueling
Period 1/31 - 2/24/77

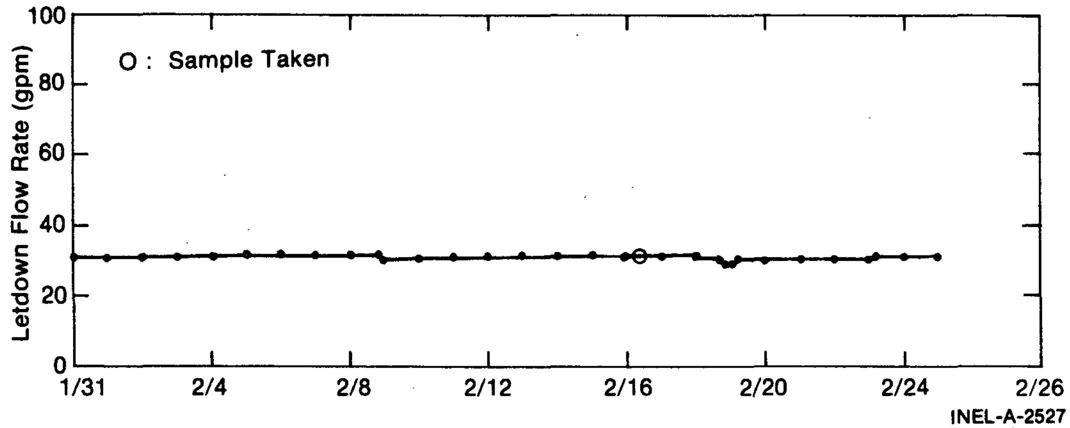


Figure A.10

CVCS LETDOWN PARAMETERS

Period: 1/31 - 2/24/77

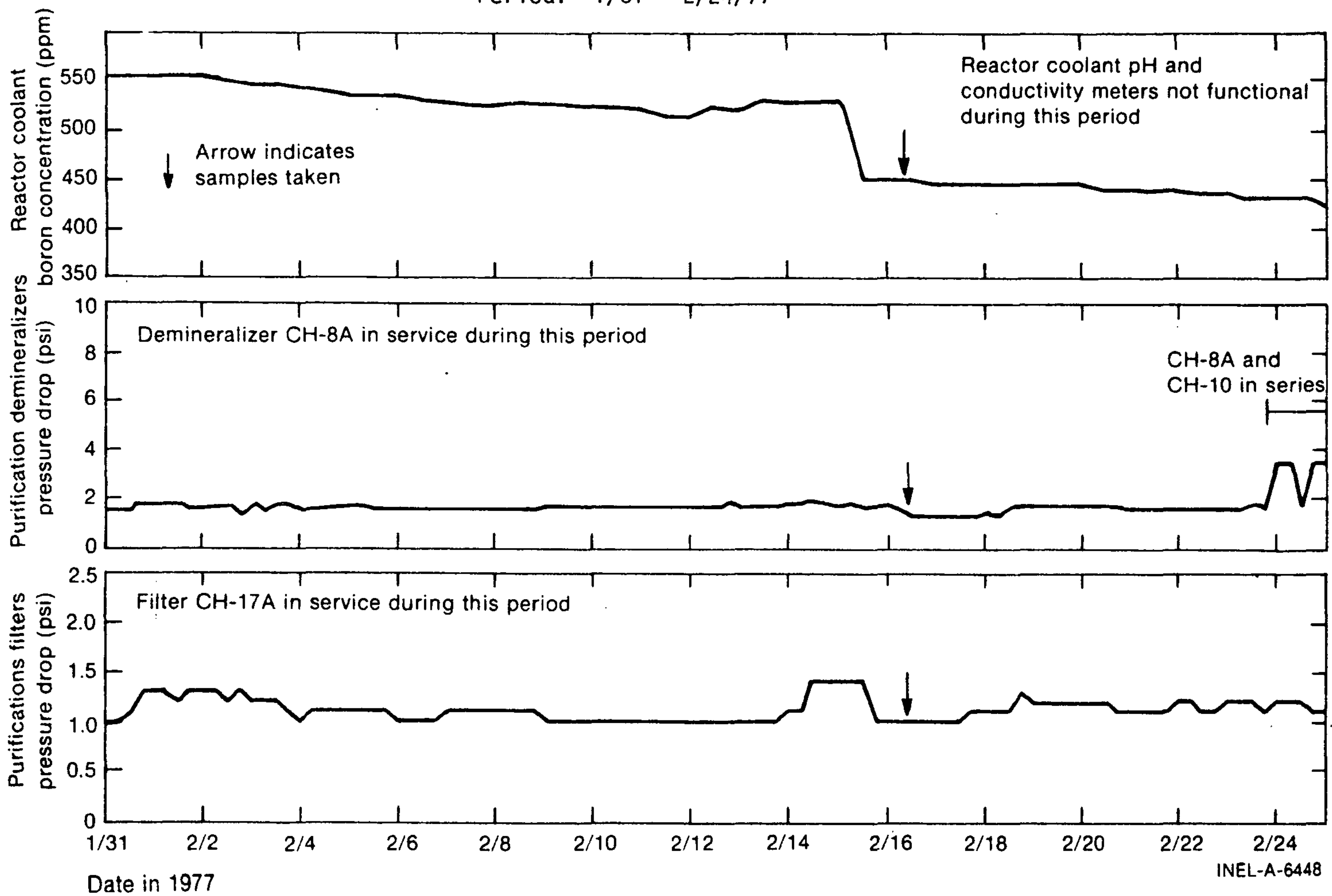


Figure A.11

Steam Generator Flows and Blowdown Rates: Power Operations - After Refueling
Period: 1/31 - 2/24/77

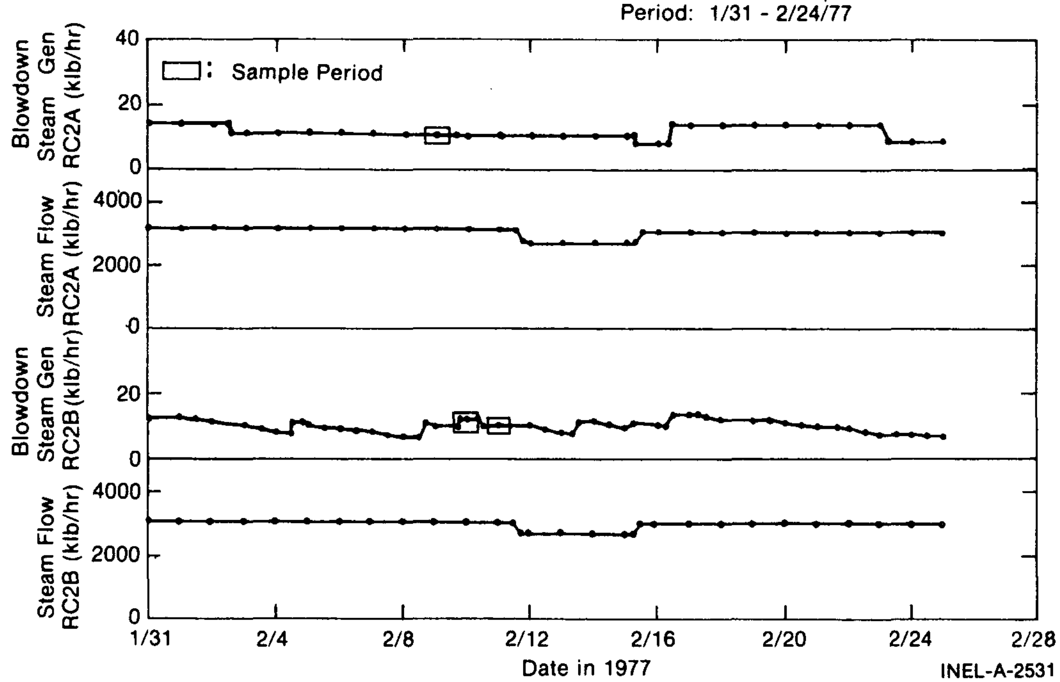


Figure A.12

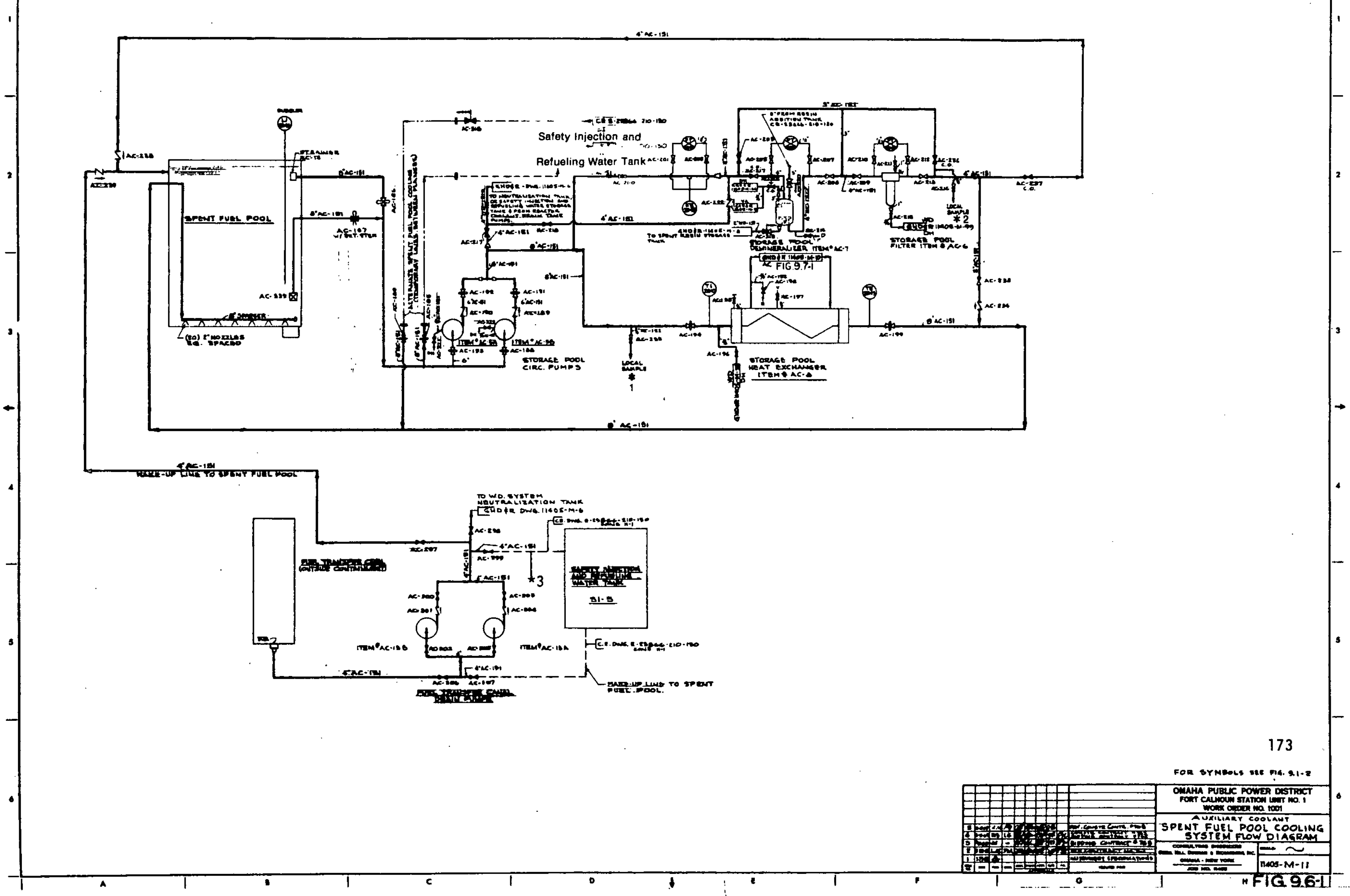

Safety Injection and
Refueling Water Tank
SPENT FUEL POOL
STORAGE POOL CIRC. PUMPS
STORAGE POOL HEAT EXCHANGER
FIG 9.7-1
MAKE-UP LINE TO SPENT FUEL POOL
SI-5
FOR SYMBOLS SEE FIG. 9.1-2
OMAHA PUBLIC POWER DISTRICT
FORT CALHOUN STATION UNIT NO. 1
WORK ORDER NO. 1001
AUXILIARY COOLANT
SPENT FUEL POOL COOLING SYSTEM FLOW DIAGRAM
11405-M-11
FIG 9.6-1

Figure A.13

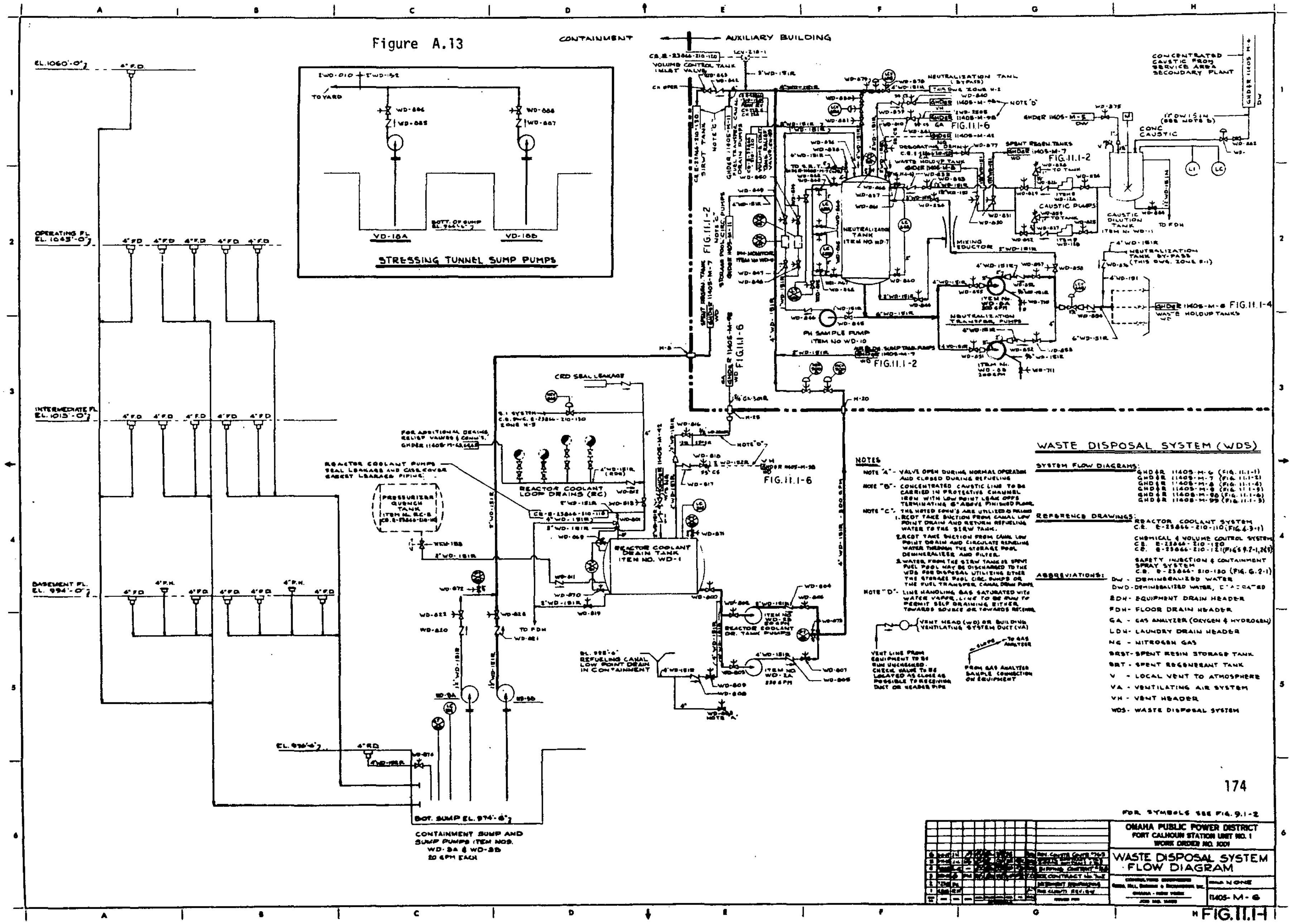

CONTAINMENT
AUXILIARY BUILDING
EL. 1060'-0"
OPERATING FL. EL. 1045'-0"
INTERMEDIATE FL. EL. 1013'-0"
BASEMENT FL. EL. 994'-0"
STRESSING TUNNEL SUMP PUMPS
TO YARD
BOTT. OF SUMP
NEUTRALIZATION TANK
CAUSTIC PUMPS
CAUSTIC DILUTION TANK
CONC CAUSTIC
CONCENTRATED CAUSTIC FROM SERVICE AREA SECONDARY PLANT
MIXING EDUCTOR
NEUTRALIZATION TRANSFER PUMPS
PH SAMPLE PUMP
SPENT REGEN TANKS
WASTE HOLDUP TANKS
CRD SEAL LEAKAGE
REACTOR COOLANT LOOP DRAINS (RC)
REACTOR COOLANT DRAIN TANK ITEM NO. WD-1
REACTOR COOLANT DR. TANK PUMPS
PRESSURIZER QUENCH TANK
REFUELING CANAL LOW POINT DRAIN IN CONTAINMENT
BOT. SUMP EL. 974'-6"
CONTAINMENT SUMP AND SUMP PUMPS ITEM NOS. WD-3A & WD-3B 20 GPM EACH
FIG. 11.1-2
FIG. 11.1-4
FIG. 11.1-6
WASTE DISPOSAL SYSTEM (WDS)
NOTES
SYSTEM FLOW DIAGRAMS
REFERENCE DRAWINGS
ABBREVIATIONS
DW - DEMINERALIZED WATER
EDH - EQUIPMENT DRAIN HEADER
FDH - FLOOR DRAIN HEADER
GA - GAS ANALYZER (OXYGEN & HYDROGEN)
LDH - LAUNDRY DRAIN HEADER
N2 - NITROGEN GAS
SRST - SPENT RESIN STORAGE TANK
SRT - SPENT REGENERANT TANK
V - LOCAL VENT TO ATMOSPHERE
VA - VENTILATING AIR SYSTEM
VH - VENT HEADER
WDS - WASTE DISPOSAL SYSTEM
FOR SYMBOLS SEE FIG. 9.1-2
OMAHA PUBLIC POWER DISTRICT
FORT CALHOUN STATION UNIT NO. 1
WORK ORDER NO. 1001
WASTE DISPOSAL SYSTEM FLOW DIAGRAM
11405-M-6
FIG. 11.1-1

Figure A.14

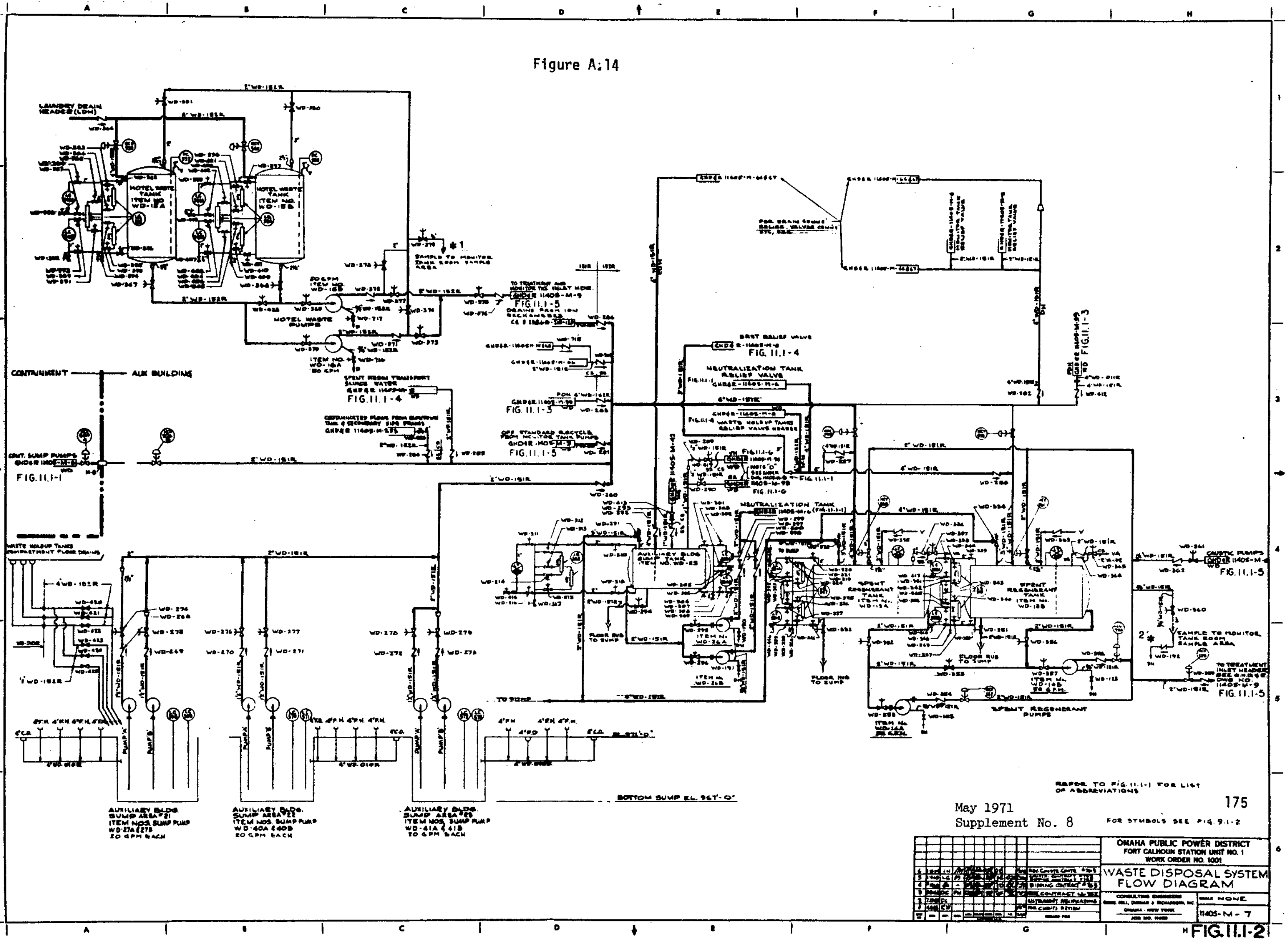


FIG. 11.1-2

May 1971
Supplement No. 8

Figure A.15

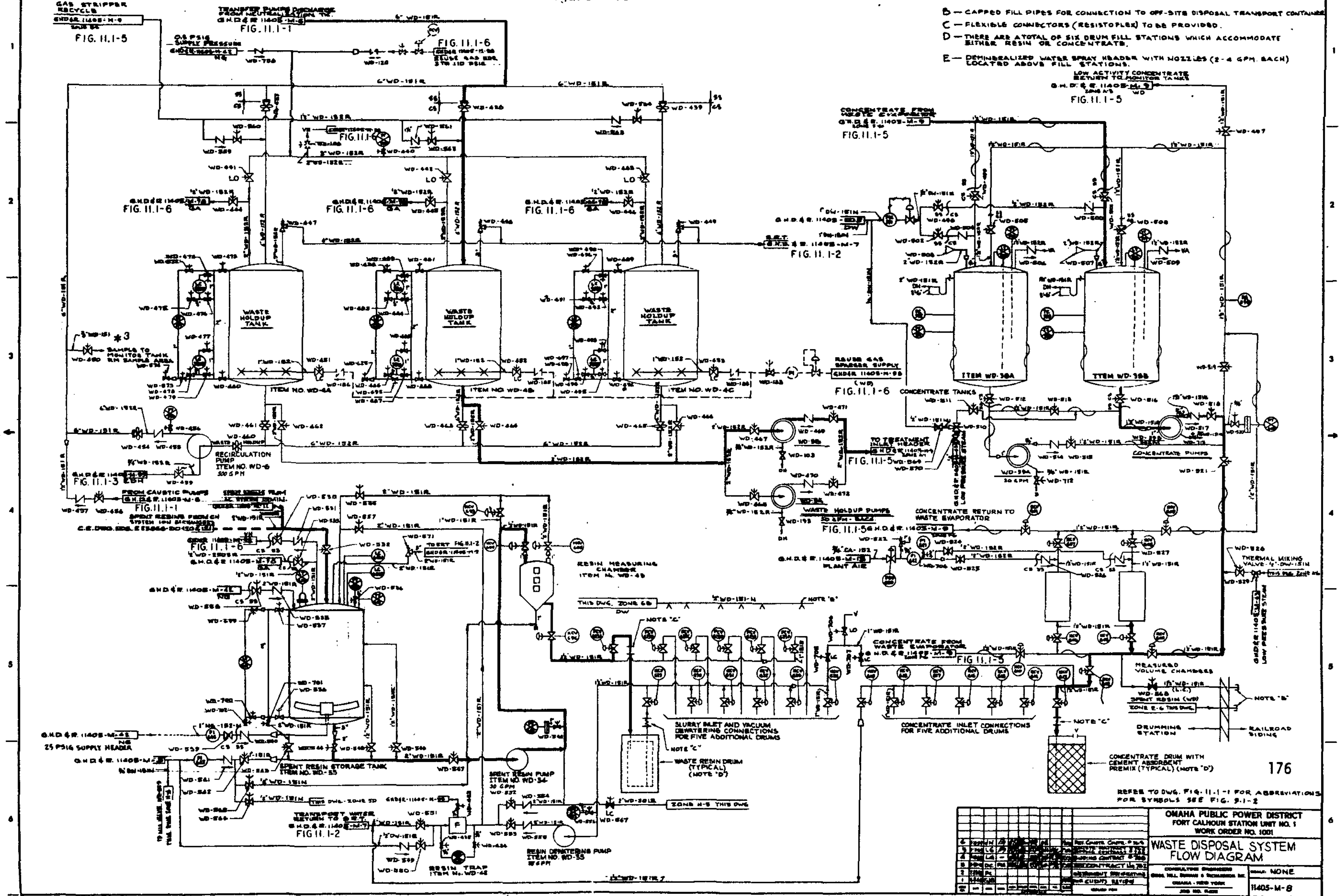
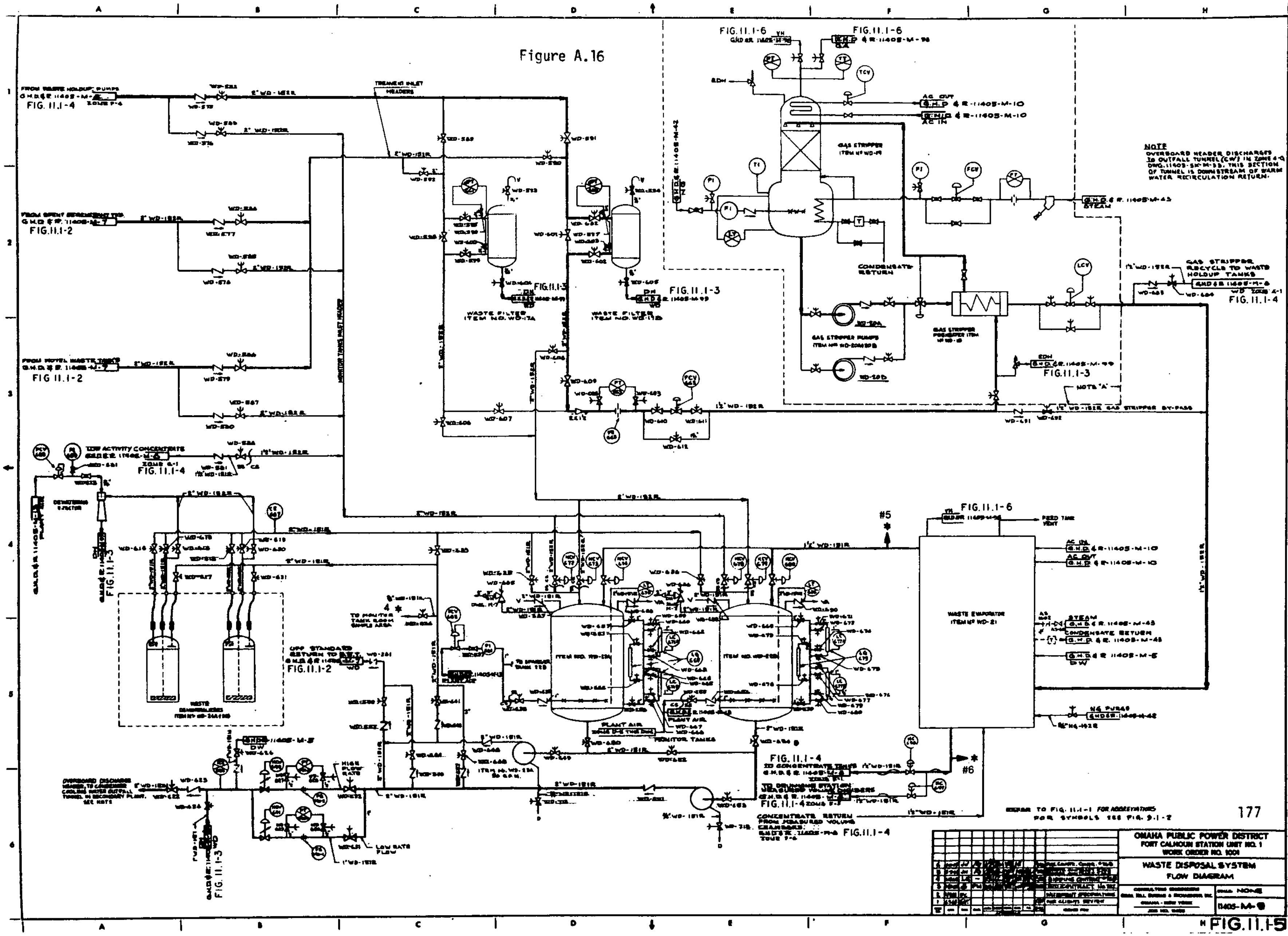


FIG.11.1-4

Figure A.16



177

FIG.11.1-5

Figure A.17

Level of Waste Holdup Tanks: Before Refueling

Tank Level in Inches of H_2O

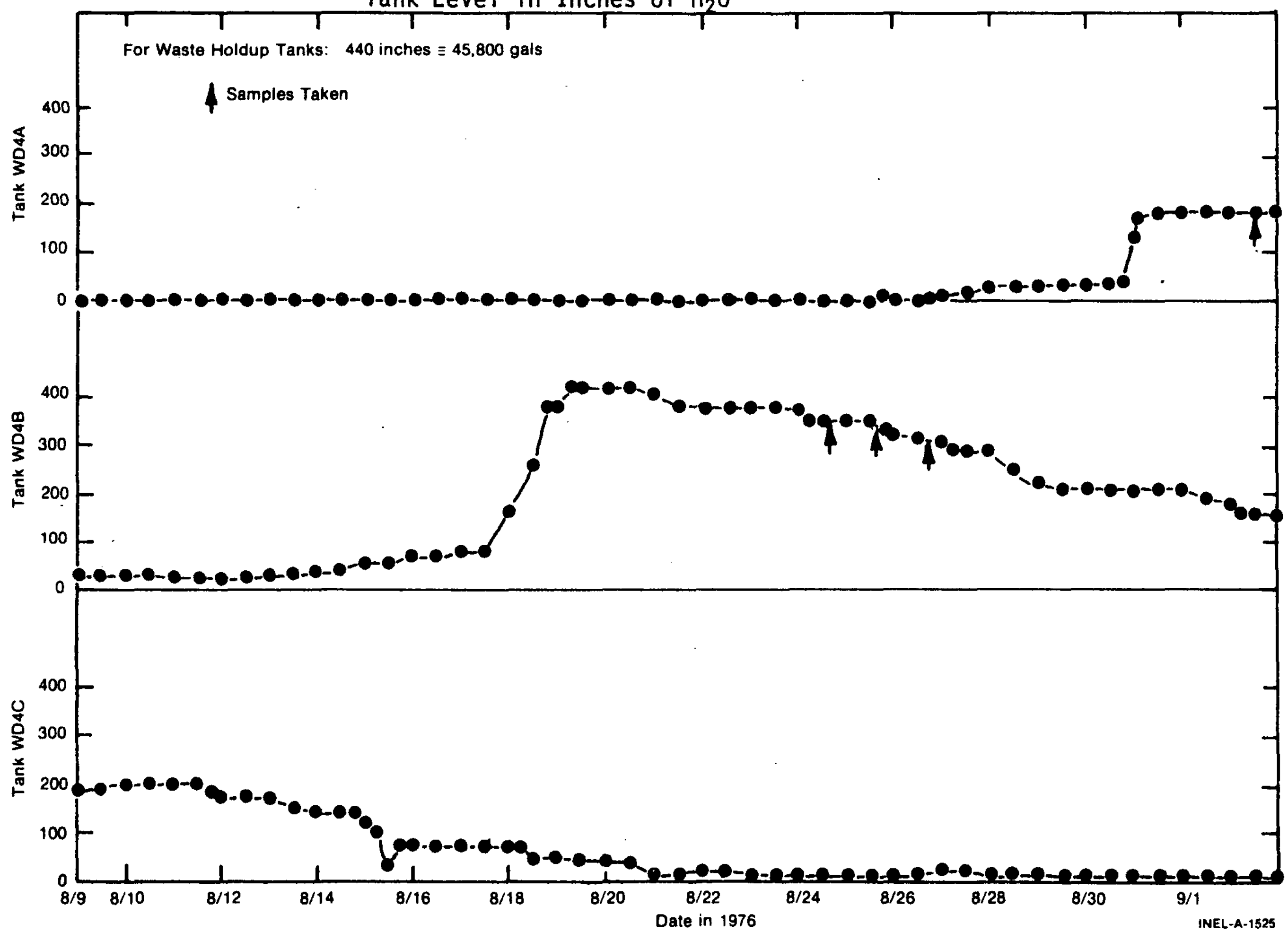


Figure A.18

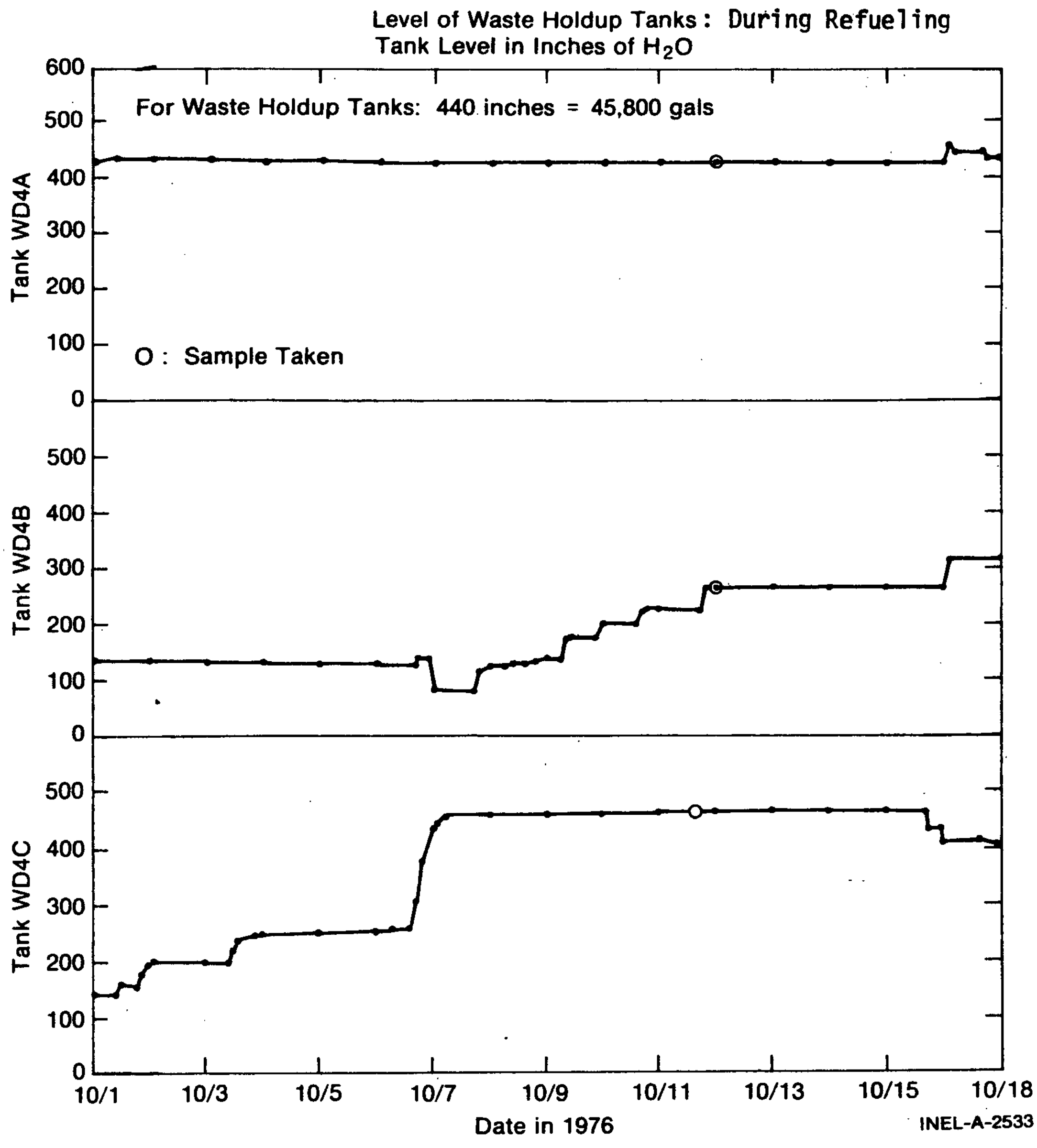

Level of Waste Holdup Tanks: During Refueling
Tank Level in Inches of H2O
For Waste Holdup Tanks: 440 inches = 45,800 gals
O: Sample Taken
Tank WD4A
Tank WD4B
Tank WD4C
600
500
400
300
200
100
0
10/1
10/3
10/5
10/7
10/9
10/11
10/13
10/15
10/18
Date in 1976
INEL-A-2533

Figure A.19

Level of Waste Holdup Tanks: After Refueling
Tank Level in Inches of H2O

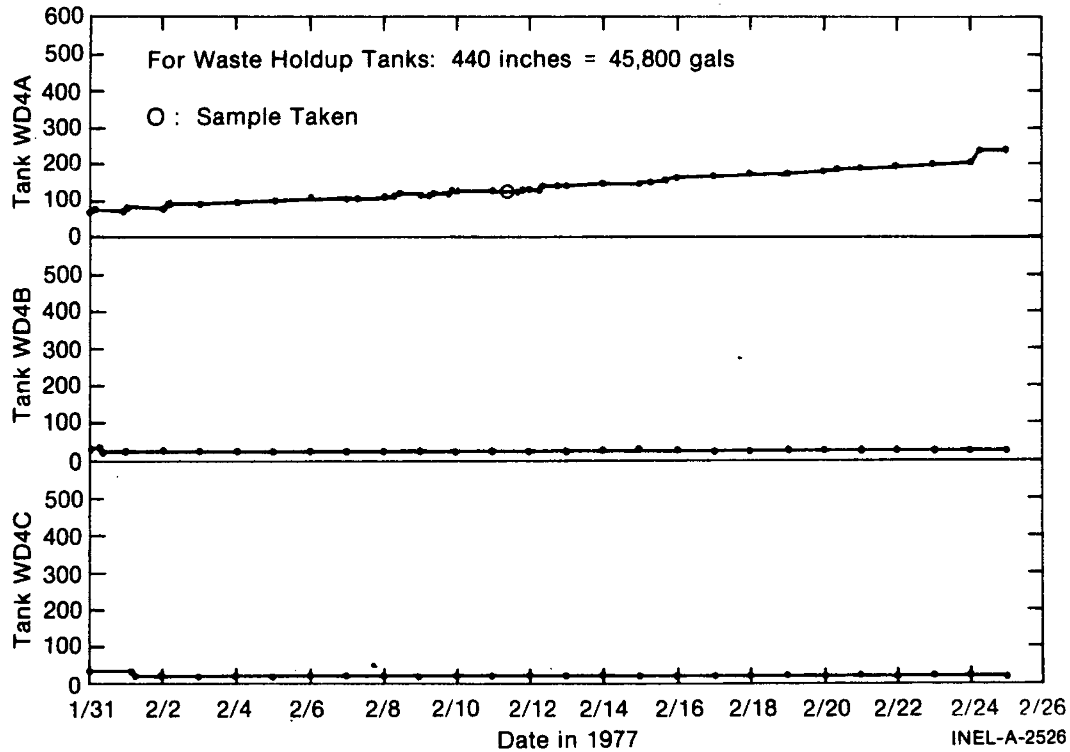


Figure A.20

Level of Radwaste Tanks: Before Refueling

Tank Level in Inches of H_2O

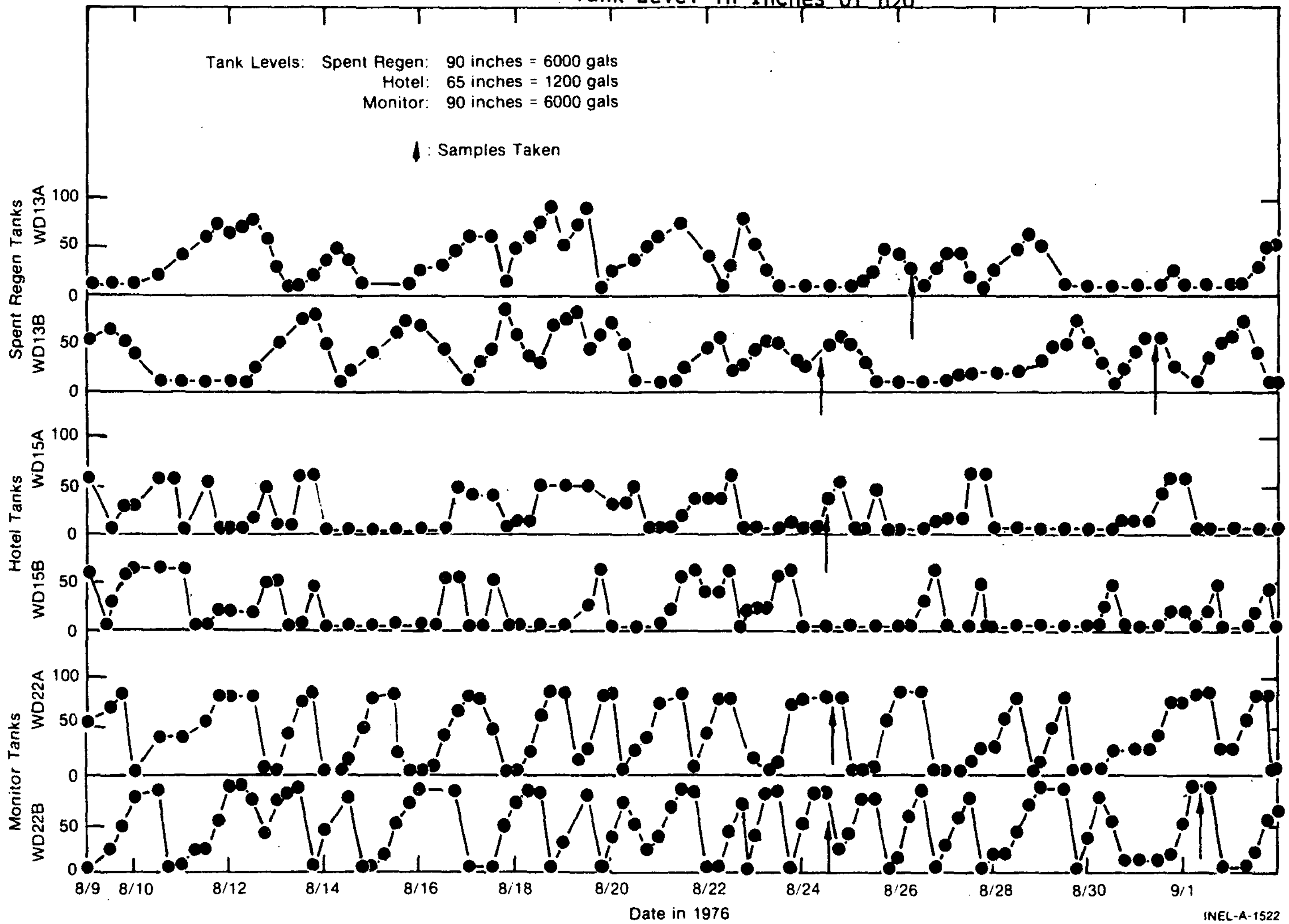
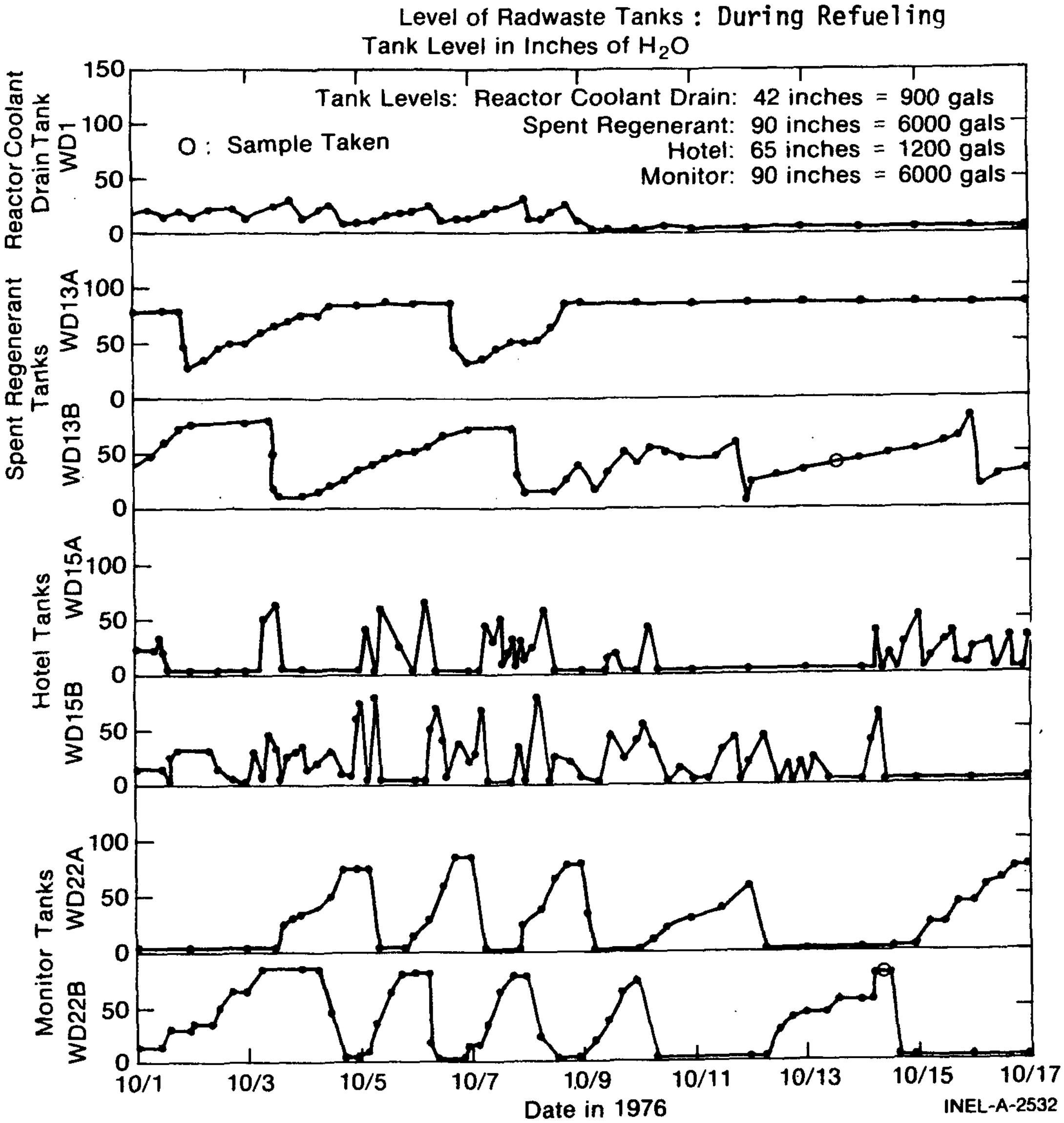


Figure A.21



Level of Radwaste Tanks : During Refueling
Tank Level in Inches of H_2O
Tank Levels: Reactor Coolant Drain: 42 inches = 900 gals
Spent Regenerant: 90 inches = 6000 gals
Hotel: 65 inches = 1200 gals
Monitor: 90 inches = 6000 gals
O : Sample Taken
Reactor Coolant Drain Tank WD1
Spent Regenerant Tanks
WD13A
WD13B
Hotel Tanks
WD15A
WD15B
Monitor Tanks
WD22A
WD22B
150
100
50
0
10/1
10/3
10/5
10/7
10/9
10/11
10/13
10/15
10/17
Date in 1976
INEL-A-2532

Figure A.22

Level of Radwaste Tanks: After Refueling
Tank Level in Inches of H_2O
Tank Levels: Reactor Coolant Drain: 42 inches = 900 gals
Spent Regenerant: 90 inches = 6000 gals
Hotel: 65 inches = 1200 gals
Monitor: 90 inches = 6000 gals

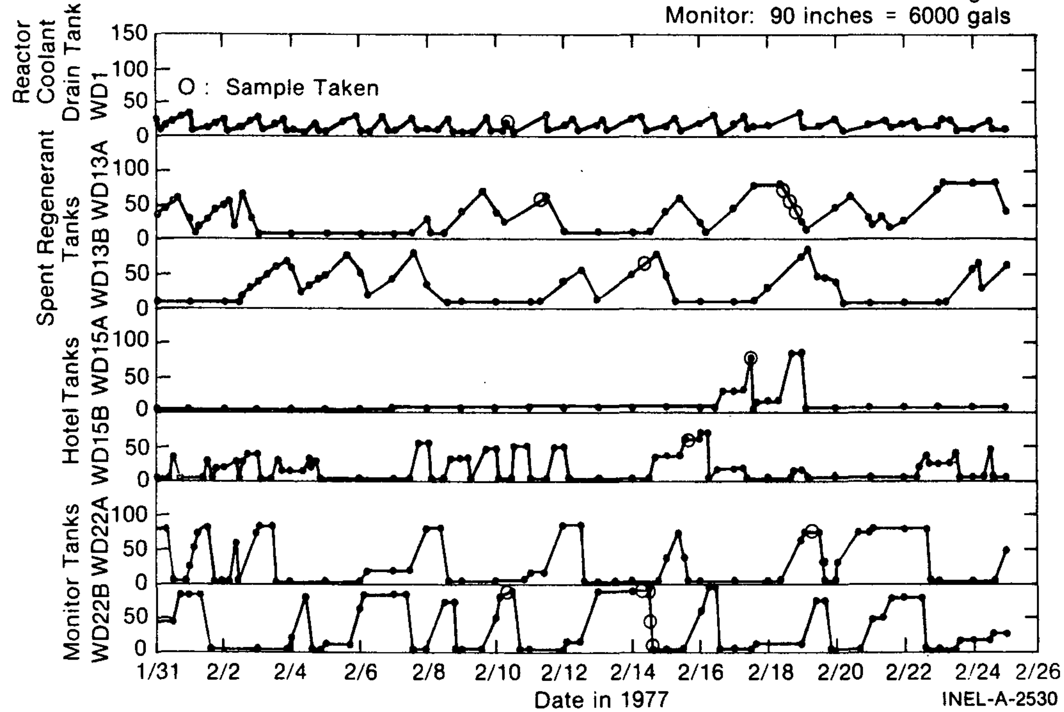


Figure A.23

RADWASTE EVAPORATOR PARAMETERS

Period: 8/9 - 9/2/76

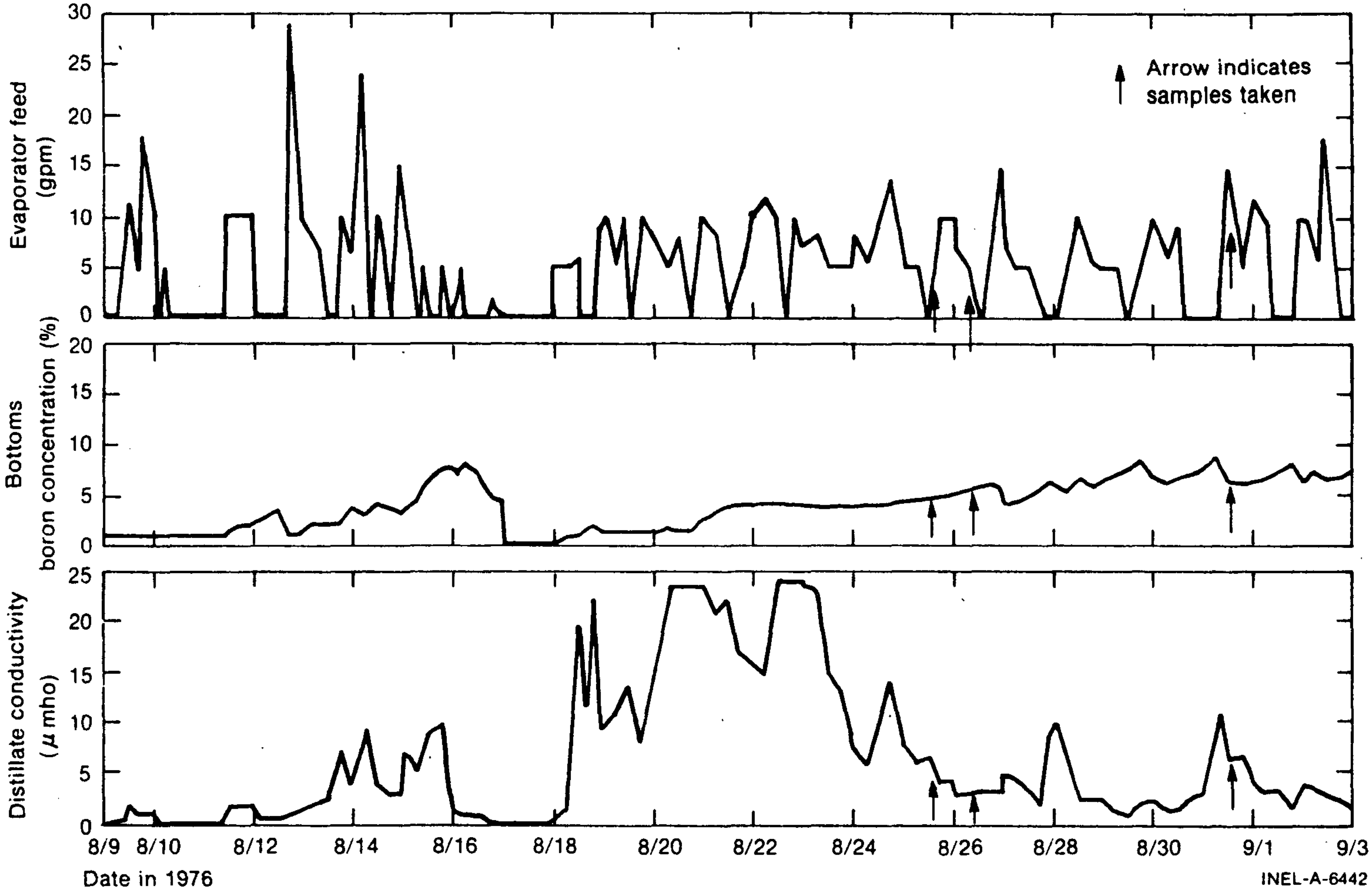


Figure A.24

RADWASTE EVAPORATOR PARAMETERS
Period: 00:00, 8/24/76 through 00:00, 8/28/76

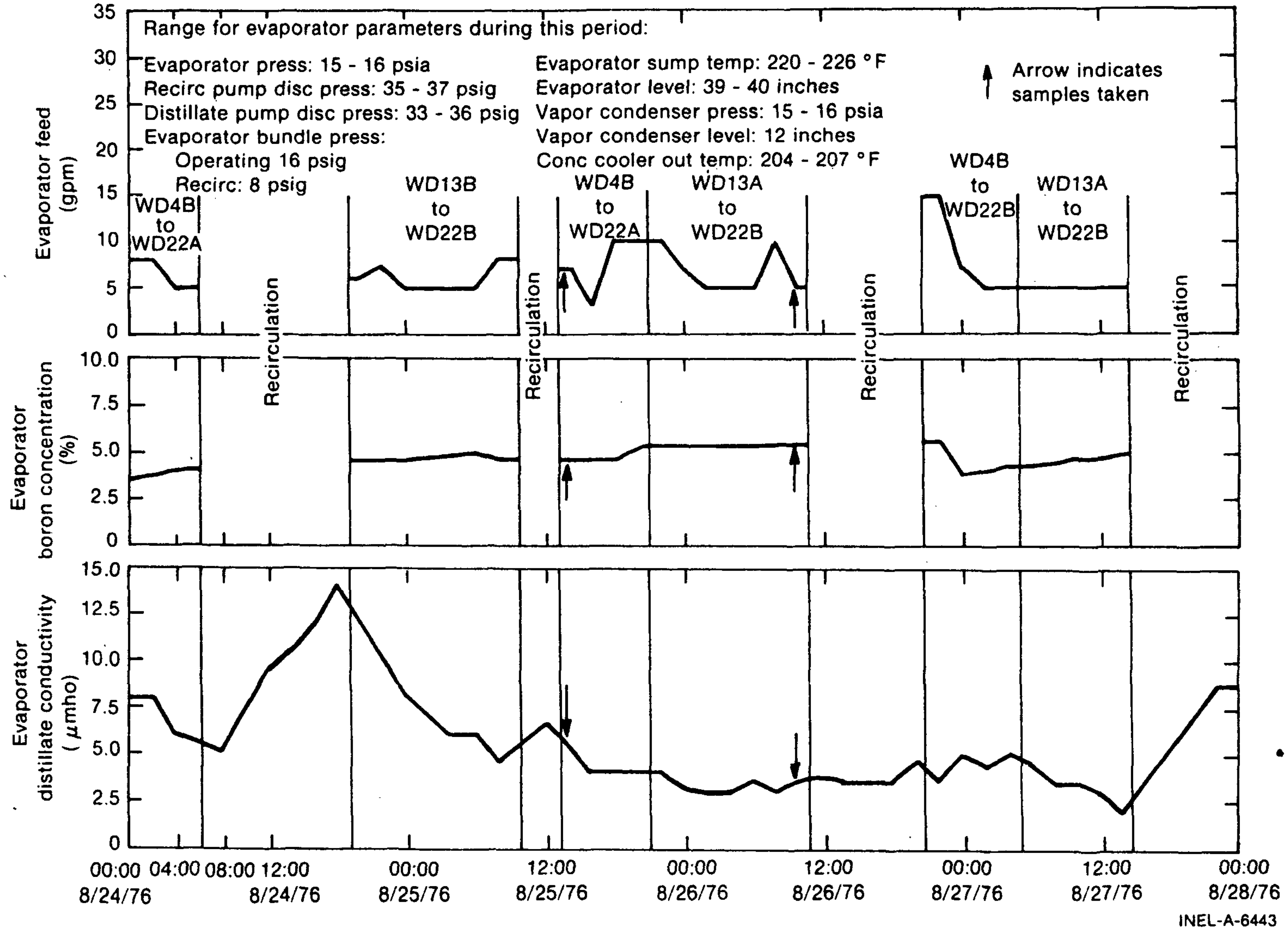
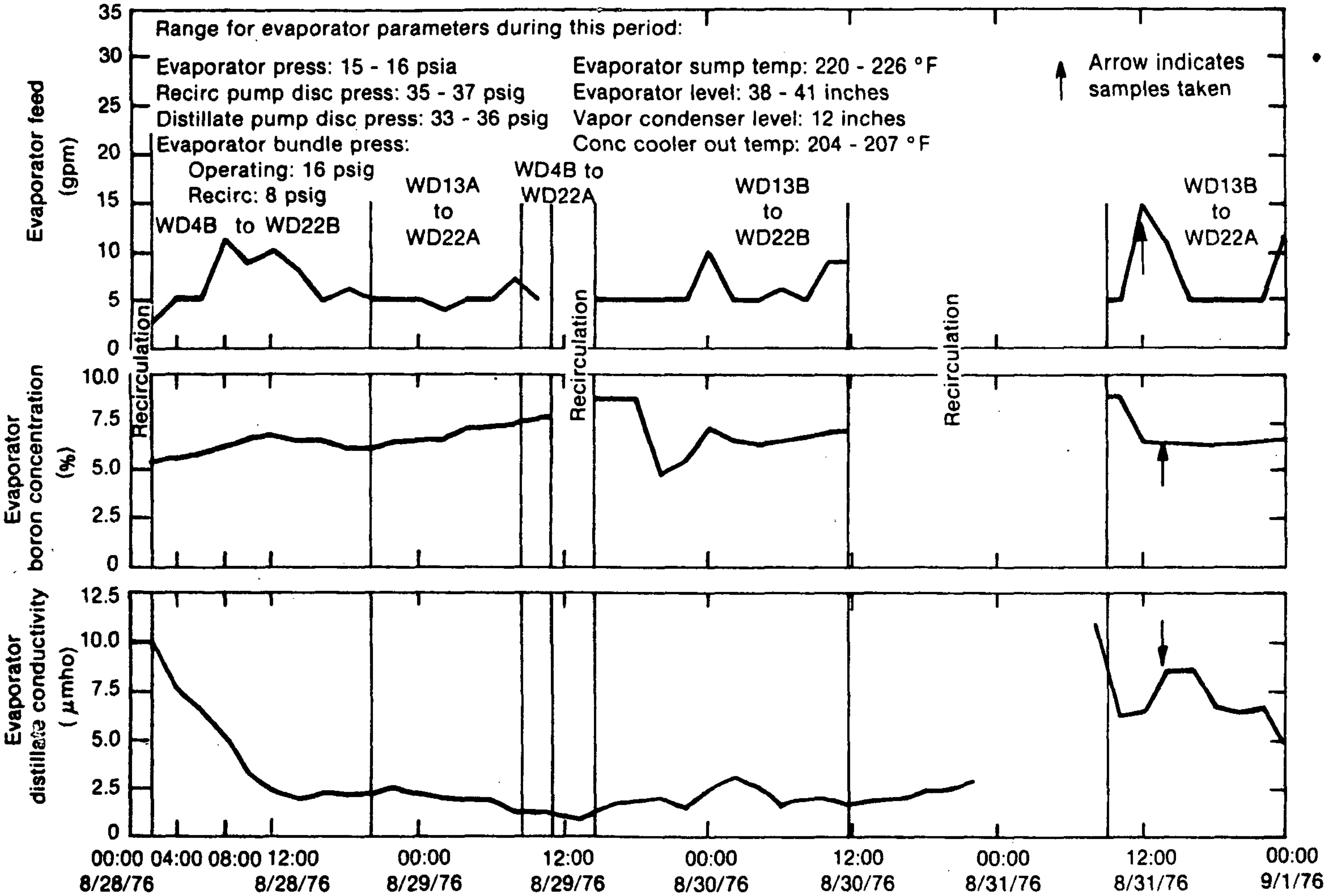


Figure A.25

RADWASTE EVAPORATOR PARAMETERS

Period: 00:00, 8/28/76 through 00:00, 9/1/76



INEL-A-6444

Figure A.26

RADWASTE EVAPORATOR PARAMETERS

Period: 1/31 - 2/21/77

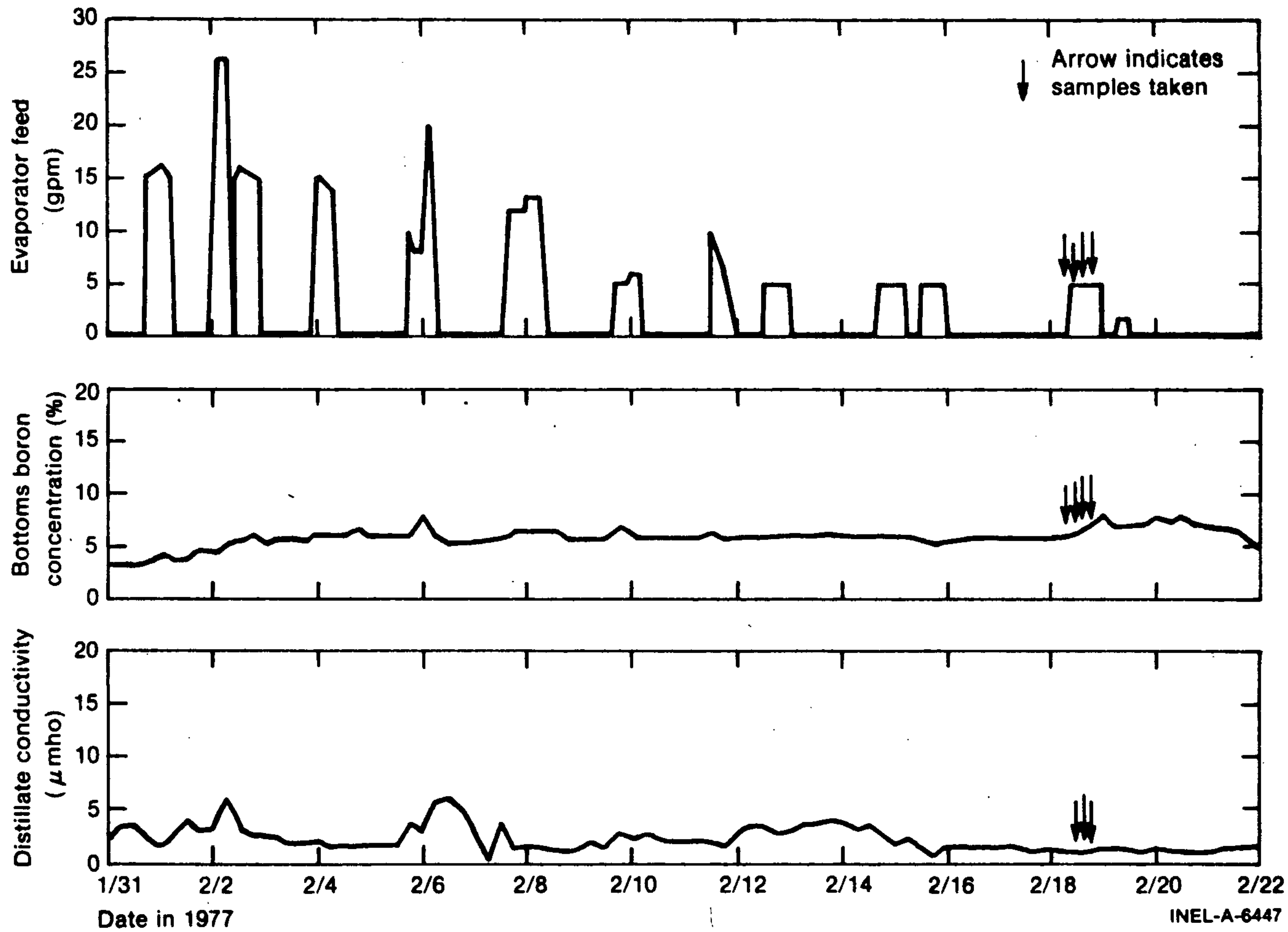
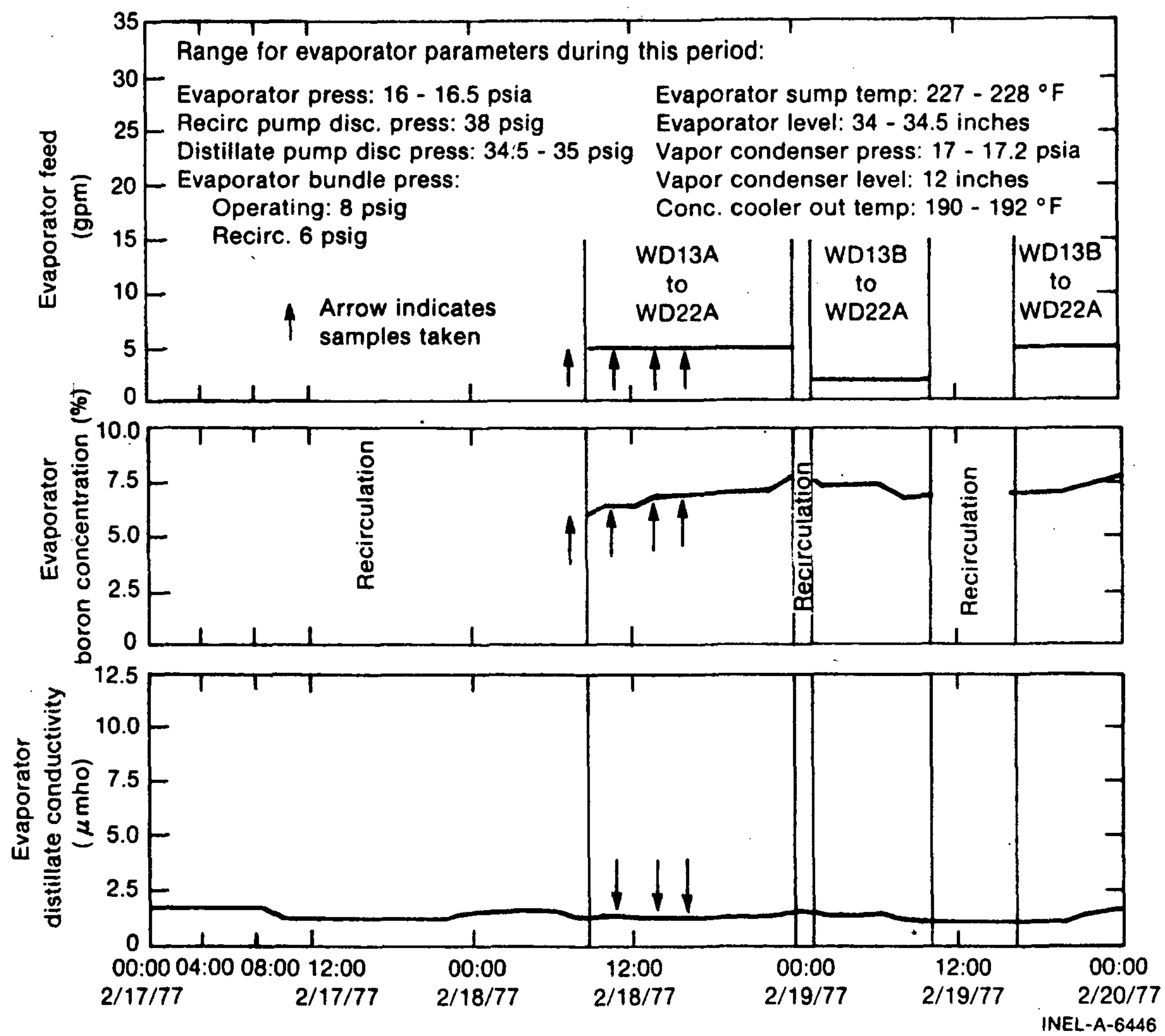


Figure A.27

RADWASTE EVAPORATOR PARAMETERS

Period: 00:00, 2/17/77 through 00:00, 2/20/77



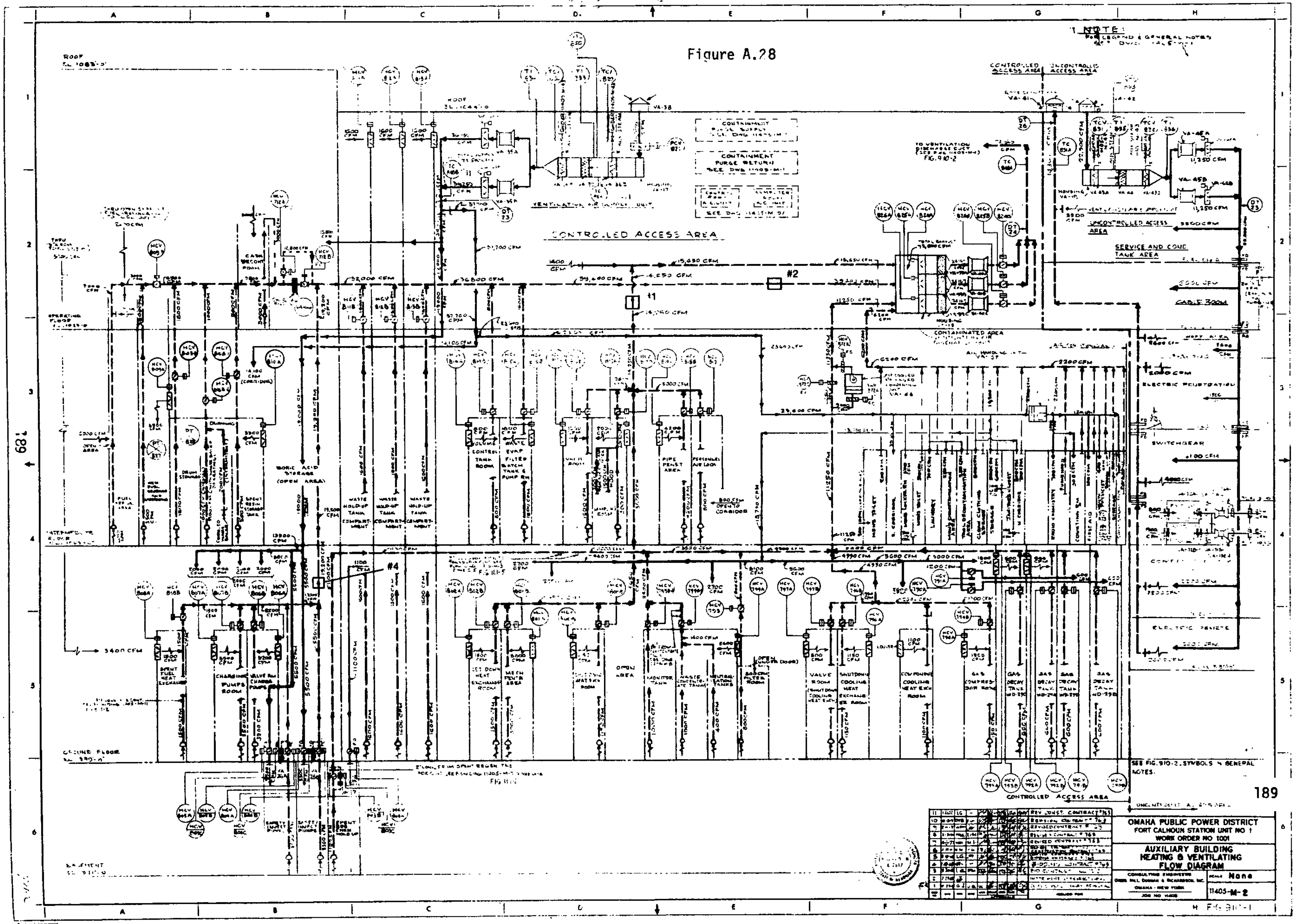

CONTROLLED ACCESS AREA
CONTAINMENT PURGE RETURN
SERVICE AND COND TANK AREA
CABLE ROOM
ELECTRIC PENETRATION
SWITCHGEAR
BORIC ACID STORAGE (OPEN AREA)
CONTROLLED ACCESS AREA
SEE FIG. 910-2, SYMBOLS & GENERAL NOTES.
OMAHA PUBLIC POWER DISTRICT
FORT CALHOUN STATION UNIT NO 1
WORK ORDER NO 1001
AUXILIARY BUILDING
HEATING & VENTILATING
FLOW DIAGRAM
None
11405-M-2

Figure A.28

Figure A.29

As Built: Ventilating Flow System

From Section F-4 of Fig. A.28