SEMIANNUAL RADIOACTIVE

EFFLUENT RELEASE

REPORT

1988

JANUARY - JUNE

FLORIDA POWER CORPORATION

CRYSTAL RIVER - UNIT 3

FACILITY OPERATING LICENSE NO. DPR-72

DOCKET NO. 50-302

AUGUST, 1988

Approved by: David Glenn Johnson
Manager, Site Nuclear Services

Date: 9-22-88

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CONTENTS

INTRODUCTION

This report is submitted as required by Technical Specification 6.9.1.5.d to Crystal River Facility Operating License No. DPR-72. In accordance with Technical Specifications, the following information must be included in this report:

A summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the plant as outlined in Regulatory Guide 1.21 (Rev. 1, 1974) with data summarized on a quarterly basis following the format of Appendix B thereof.

For each type of solid waste shipped off site:

- Container Volume
- Total Curie Quantity (specified as measured or estimated)
- Principal Radionuclides (specified as measured or estimated)
- Type of Waste (e.g., spent resin, compacted dry waste)
- Type of container (e.g., LSA, Type A, Type B)
- Solidification Agent (e.g., cement)

A list and description of unplanned releases to unrestricted areas.

A description of any changes to the:

- Process Control Program (PCP)
- Off-Site Dose Calculation Manual (ODCM)
- Radioactive Waste Treatment Systems

A listing of new Environmental Radiological Monitoring Program dose calculation location changes identified by the land-use census.

Information relating to effluent monitors being inoperable for thirty or more days.

Information regarding meteorological data and environmental dose assessments will be included in the year-end semiannual report as required by Technical Specifications.

In addition to the required data, trend graphs of curies released per six month period have been included in order to put the current data into historical perspective.

With regard to the 1987 year-end Semiannual report an error has been noted in Table IV-2 (Total Curies): 1.69E+03 should be 1.69E+02.

TABLE 1

EFFLUENT AND WASTE DISPOSAL SEMIANNUAL REPORT - 1988

GASEOUS EFFLUENTS - SUMMATION OF ALL RELEASES

	Unit	Quarter 1	Quarter 2	Est.Total Error %

A. Fission and Activation Gases

1. Total Release	Ci	2.22E+02	1.26E+02	110
2. Average Release Rate for Period	uCi/sec	2.82E+01	1.60E+01	
3. Percent of Technical Specification Limit	%	2.02E-01	1.14E-01	

B. Iodines

1. Total Iodine - 131	Ci	5.12E-05	1.73E-05	110
2. Average Release Rate for Period	uCi/sec	6.52E-06	2.20E-06	
3. Percent of Technical Specification Limit	%	4.55E-01	1.49E-01	

C. Particulates

1. Particulates with half-lives > 8 days	Ci	1.35E-04	8.13E-05	100
2. Average Release Rate for Period	uCi/sec	1.72E-05	1.03E-05	
3. Percent of Technical Specification Limit	%	4.55E-01	1.49E-01	
4. Gross Alpha Radioactivity	Ci	7.38E-11	<LLD	

D. Tritium

1. Total Release	Ci	4.06E+00	5.32E-01	90
2. Average Release Rate for Period	uCi/sec	5.16E-01	6.77E-02	
3. Percent of Technical Specification Limit	%	4.55E-00	1.49E-01	

TABLE 2

EFFLUENT AND WASTE DISPOSAL SEMIANNUAL REPORT - 1988

GASEOUS EFFLUENTS - GROUND LEVEL RELEASES

		CONTINUOUS MODE		BATCH MODE	
Nuclides Released	Unit	Quarter 1	Quarter 2	Quarter 1	Quarter 2
1. Fission gases					
argon-41	Ci				
krypton-85	Ci			8.02E-01	2.88E+00
krypton-85m	Ci				
krypton-87	Ci				
krypton-88	Ci				
xenon-131m	Ci			3.93E-01	7.92E-01
xenon-133	Ci	1.88E+02	8.79E+01	2.16E+01	2.65E+01
xenon-133m	Ci	6.72E-02		1.35E-01	1.59E-01
xenon-135	Ci	1.11E+01	7.52E+00	2.11E-02	1.71E-02
xenon-135m	Ci				
xenon-138	Ci				
unidentified	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Total for Period	Ci	1.99E+02	9.54E+01	2.30E+01	3.04E+01
Iodines					
iodine-131	Ci	5.12E-05	1.73E-05		
iodine-133	Ci	1.54E-05	1.16E-05		
iodine-135	Ci				
Total for Period	Ci	6.66E-05	2.89E-05	0.00E+00	0.00E+00
3. Particulates					
manganese-54	Ci				
cobalt-58	Ci				
iron-59	Ci				
cobalt-60	Ci				
zinc-65	Ci				
strontium-89	Ci	7.02E-08		2.46E-11	
strontium-90	Ci				
molybdenum-99	Ci				
tellurium-132	Ci				
cesium-134	Ci	1.98E-05	5.75E-06		
cesium-137	Ci	4.02E-05	1.06E-05		
cesium-138	Ci		2.96E-07		
cerium-141	Ci				
cerium-144	Ci				
unidentified	Ci	7.48E-05	6.49E-05	0.00E+00	0.00E+00
Total	Ci	1.35E-04	8.15E-05	2.46E-11	0.00E+00

TABLE 3

EFFLUENT AND WASTE DISPOSAL SEMIANNAUL REPORT - 1988

LIQUID EFrLUENTS - SUMMATION OF ALL RELEASES

	Unit	Quarter 1	Quarter 2	Est.Total Error %
A. Fission and Activation Products				
1. Total Release (not including tritium, gases, alpha)	Ci	9.36E-02	7.02E-02	100
2. Average diluted concentration during period	uCi/ml	6.08E-09	4.28E-09	
3. Percent of applicable limit	%	2.3E+00	1.12E+00	
B. Tritium				
1. Total Release	Ci	6.96E+01	1.48E+02	40
2. Average diluted concentration during period	uCi/ml	4.52E-06	9.02E-06	
3. Percent of applicable limit	%	1.51E-01	3.01E-01	
C. Dissolved and entrained gases				
1. Total release	Ci	1.22E+00	1.62E+00	100
2. Average diluted concentration during period	uCi/ml	7.92E-08	9.88E-08	
3. Percent of applicable limit	%	3.96E-02	4.94E-02	
D. Gross alpha radioactivity				
1. Total release	Ci	<LLD	<LLD	100
E. Volume of Waste released (prior to dilution)				
1. Batch and Continuous Modes	Liters	1.62E+07	9.20E+06	10
F. Volume of dilution water used during period				
1. Batch and Continuous Modes	Liters	1.54E+10	1.64E+10	10

TABLE 4

EFFLUENT AND WASTE DISPOSAL SEMIANNUAL REPORT - 1988

LIQUID EFFLUENTS

		CONTINUOUS MODE		BATCH MODE	
Nuclides Released	Unit	Quarter 1	Quarter 2	Quarter 1	Quarter 2
sodium-24	Ci				
chromium-51	Ci			5.79E-04	
manganese-54	Ci			1.76E-03	7.89E-04
iron-55	Ci	4.57E-03		3.12E-03	2.29E-03
cobalt-58	Ci			2.52E-02	4.43E-03
iron-59	Ci				
cobalt-60	Ci			1.95E-02	9.95E-03
zinc-65	Ci				
rubidium-88	Ci				
strontium-89	Ci	1.15E-04		6.61E-04	1.91E-03
strontium-90	Ci			2.78E-05	3.45E-05
strontium-92	Ci			1.78E-03	4.65E-03
niobium-95	Ci			6.45E-04	1.27E-04
zirconium-95	Ci			1.71E-04	
zirconium-97	Ci			1.94E-04	5.94E-04
molybdenum-99	Ci				6.94E-05
technetium-99m	Ci			1.76E-04	4.37E-04
ruthenium-103	Ci				
ruthenium-106	Ci				3.00E-03
silver-110m	Ci			5.10E-03	1.27E-02
iodine-131	Ci			1.05E-04	3.28E-05
iodine-132	Ci				
tellurium-132	Ci				
iodine-133	Ci			6.24E-06	3.01E-05
cesium-134	Ci			5.02E-05	1.06E-05
cesium-136	Ci				
cesium-137	Ci			2.30E-04	1.79E-04
barium-139	Ci			2.76E-04	
lanthanum-140	Ci				5.85E-05
cerium-141	Ci				
cerium-144	Ci				
unidentified	Ci	9.92E-03	0.00E+00	1.94E-02	2.83E-02

Total for period (above)	Ci	1.46E-02	0.00E+00	7.90E-02	7.02E-02

TABLE 4 (CONTINUED)

EFFLUENT AND WASTE DISPOSAL SEMIANNUAL REPORT - 1988

LIQUID EFFLUENTS

		CONTINUOUS MODE		BATCH MODE	
Dissolved & Entrained Gases	Unit	Quarter 1	Quarter 2	Quarter 1	Quarter 2
argon-41	Ci				
krypton-85	Ci			2.94E-03	1.66E-02
krypton-85m	Ci			2.17E-05	
krypton-88	Ci				
xenon-131m	Ci			1.85E-02	2.68E-02
xenon-133	Ci			1.18E+00	1.56E+00
xenon-133m	Ci			7.70E-03	1.15E-02
xenon-135	Ci			4.65E-03	8.99E-03
xenon-135m	Ci				
tritium	Ci	1.31 E-02	0.00E+00	6.96E+01	1.48E+02

TABLE 5

EFFLUENT AND WASTE DISPOSAL SEMIANNUAL REPORT - 1988

SOLID WASTE AND IRRADIATED FUEL SHIPMENTS

A. SOLID WASTE SHIPPED OFFSITE FOR BURIAL OR DISPOSAL (Non irradiated fuel)

1. Type of waste	Unit	First 6-month period	Est. Total Error, %
a. Spent resins, filter sludges, evaporator bottoms, etc.	m^3 Ci	2.92E+01 8.33E+02	20
b. Dry compressible waste, contaminated equip, etc.	m^3 Ci	1.04E+02 9.68E+00	50
c. Irradiated components, control rods, etc.			
d. Other (describe)			

2. Estimate* of major nuclide composition (by type of waste)

a.	Co-58	5.05E+01	Ni-63	6.89E+00	
	Cs-137	1.33E+01	Mn-54	4.29E+00	
	Co-60	1.01E+01	Cr-51	2.22E+00	
	Cs-134	8.81E+00			
b.	Cs-137	2.67E+01	Cs-134	1.19E+01	
	Fe-55	2.02E+01	Ni-63	4.08E+00	
	Co-60	1.78E+01			
	Co-58	1.24E+01			
c.					
d.					

3. Solid Waste Disposition

Number of Shipments	Mode of Transporation	Destination
11	Transport Truck - Exclusive Use Vehicle	Chem-Nuclear Systems, Inc. Barnwell, SC

B. IRRADIATED FUEL SHIPMENTS (Disposition)

Number of Shipments	Mode of Transporation	Destination
None	NA	NA

* With the exception of Secondary Resin shipments all curie values and principle radionuclides are determined by indirect methods and are therefore estimates. Secondary Resin shipments are evaluated by direct (gamma spectroscopy) and indirect (applying scaling factors) methods, with the results being a combination of measured and estimated value.

TABLE 6

EFFLUENT AND WASTE DISPOSAL SEMIANNUAL REPORT - 1988

SHIPMENT SUMMARY

DATE AND SHIPMENT #	CONTAINER VOLUME*	TOTAL CURIES	PRINCIPLE RADIONUCLIDES	WASTE TYPE	CONTAINER TYPE	SOLIDIF. AGENT
1-22 88-1	402.2	1.45E-04	H-3, C-14, Fe-55, Cs-134, Cs-137	SC	ST**	N.A.
1-28 88-6	86 @ 7.5	2.37E-01	Mn-54, Fe-55, Co-58, Co-60, Ni-63, Nb-95, Zr-95	CW	ST	N.A.
2-11 88-7	121	5.37E+02	Mn-54, Co-58 Co-60, Ni-63, Cs-134, Cs-137	SR	Type B	N.A.
2-16 88-8	120.3	2.91E+02	Cr-51, Mn-54, Co-58, Co-60, Nb-95, Cs-137	F	Type B	N.A.
2-25 88-9	85 @ 7.5 1 @ 11.6	2.73E-01	Mn-54, Fe-55, Co-58, Co-60, Ni-63, Nb-95, Cs-137	CW	ST	N.A.
3-10 88-11	85 @ 7.5 1 @ 11.6	2.92E-01	Mn-54, Fe-55, Co-58, Co-60, Ni-63, Nb-95, Cs-137	CW	ST	N.A.
3-17 88-12	193	1.61	H-3, Mn-54, Co-58, Co-60, Ni-63, Cs-134, Cs-137	SR	ST	N.A.
4-15 88-14	86 @ 7.5	4.38E-01	Fe-55, Co-58, Co-60, Ni-63, Cs-134, Cs-137, Pu-241	CW	ST	N.A.
4-21 88-15	194.1	2.97	H-3, Cr-51, Mn-54, Co-58, Co-60 Ni-63, Nb-95, Ag-110m, Cs-134, Cs-137	SR	ST	N.A.
5-10 88-17	328.5	8.36	Fe-55, Co-58, Co-60, Ni-63 Cs-134, Cs-137, Pu-241	NW	ST	N.A.
6-29 88-18	4 @ 96 3 @ 120	2.82E-01	Mn-54, Fe-55, Co-58, Co-60, Ni-63, Sr-89, Nb-95, Cs-134, Cs-137, Pu-241	NW	ST	N.A.

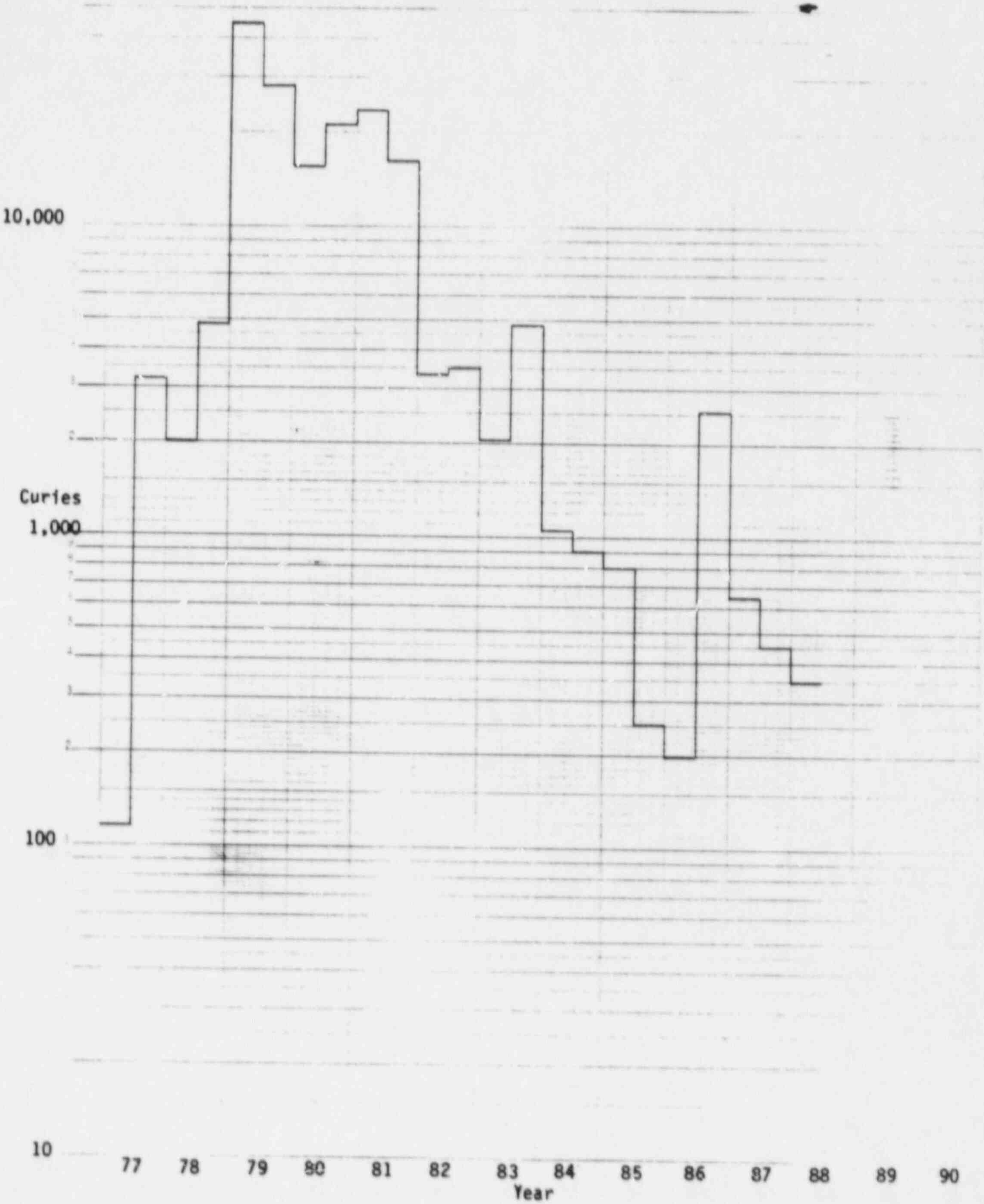
WASTE TYPE:
SR - Spent Resin
SC - Secondary Resin
F - Filters
NW - Non-Compacted Waste
CW - Compacted Waste
EB - Evaporator Bottoms
CE - Contaminated Equipment
IC - Irradiated Components

* Container volume in cubic feet

** ST - Strong Tight

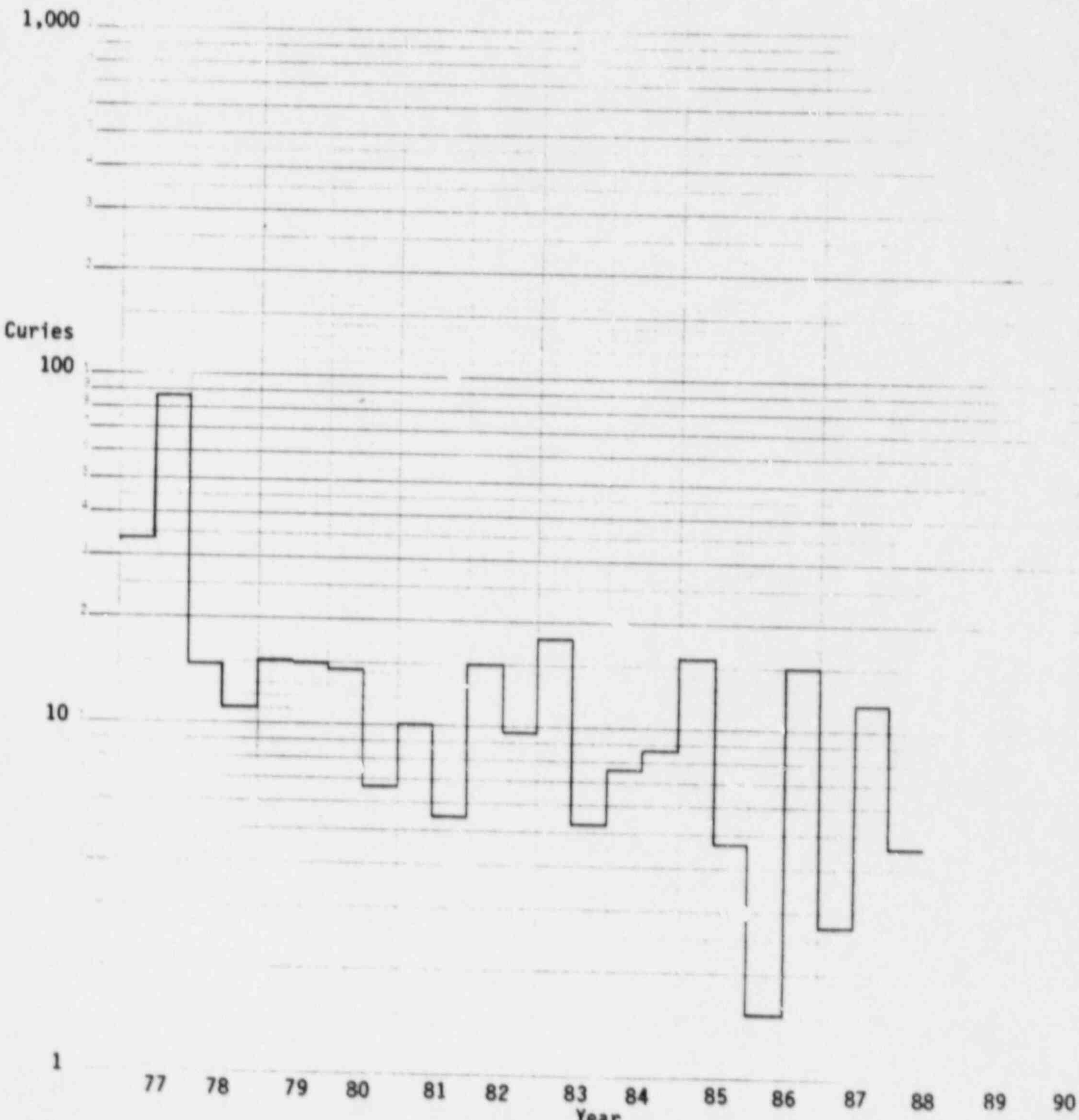
- FIGURE 1 -

GASEOUS RELEASES
FISSION AND ACTIVATION PRODUCTS



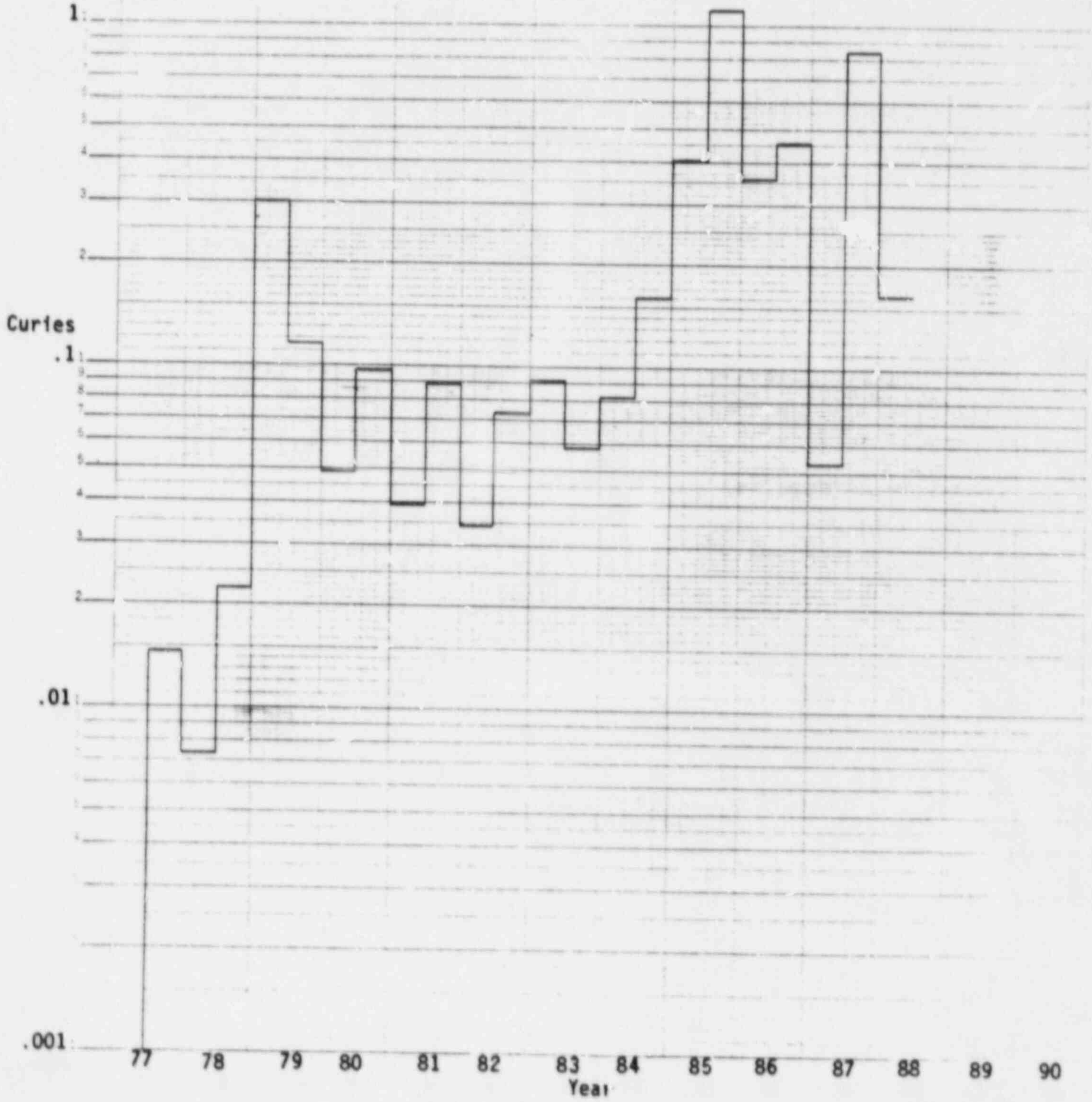
- FIGURE 2 -

GASEOUS RELEASES
TRITIUM



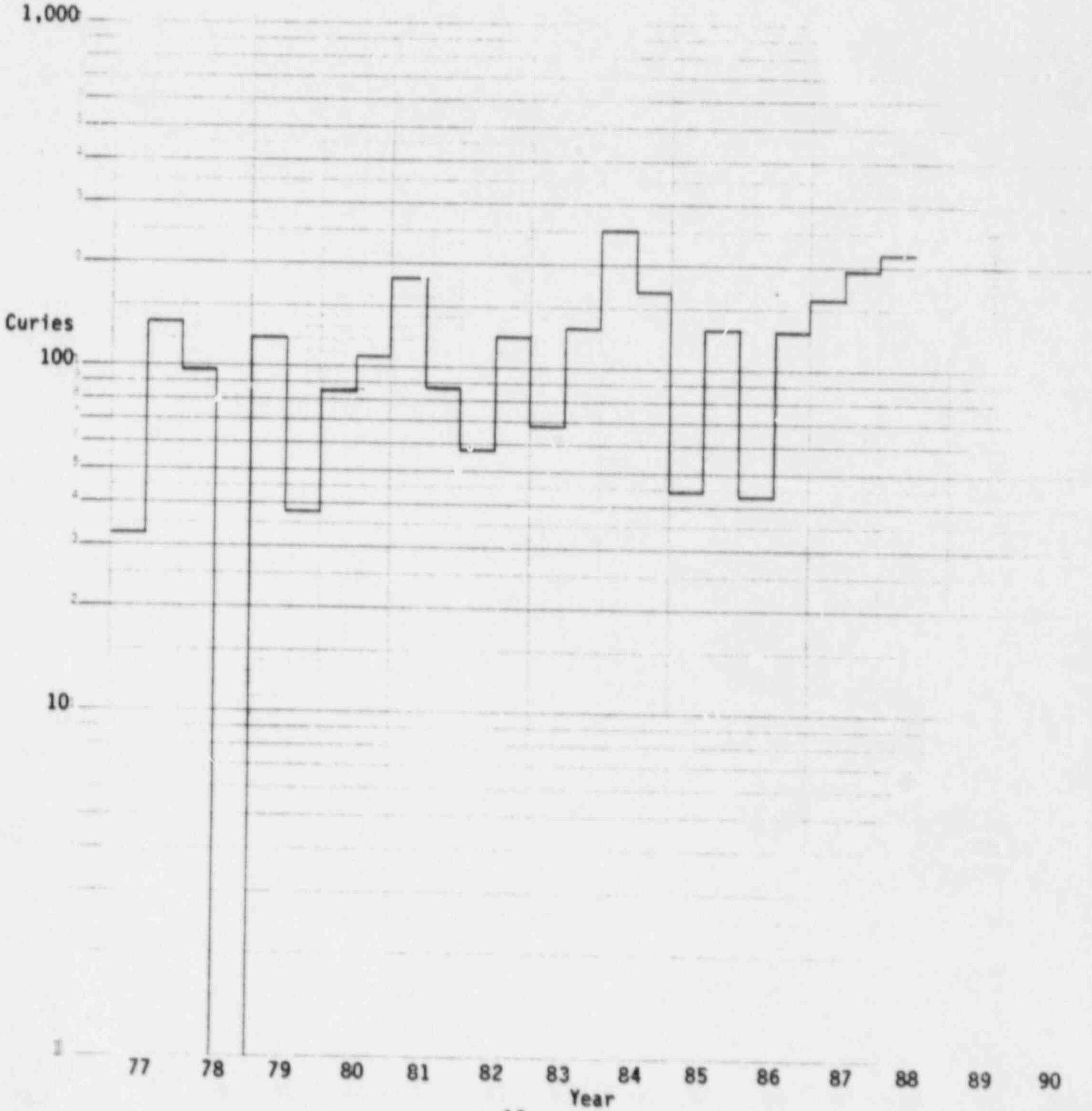
- FIGURE 3 -

LIQUID RELEASES
FISSION AND ACTIVATION PRODUCTS



- FIGURE 4 -

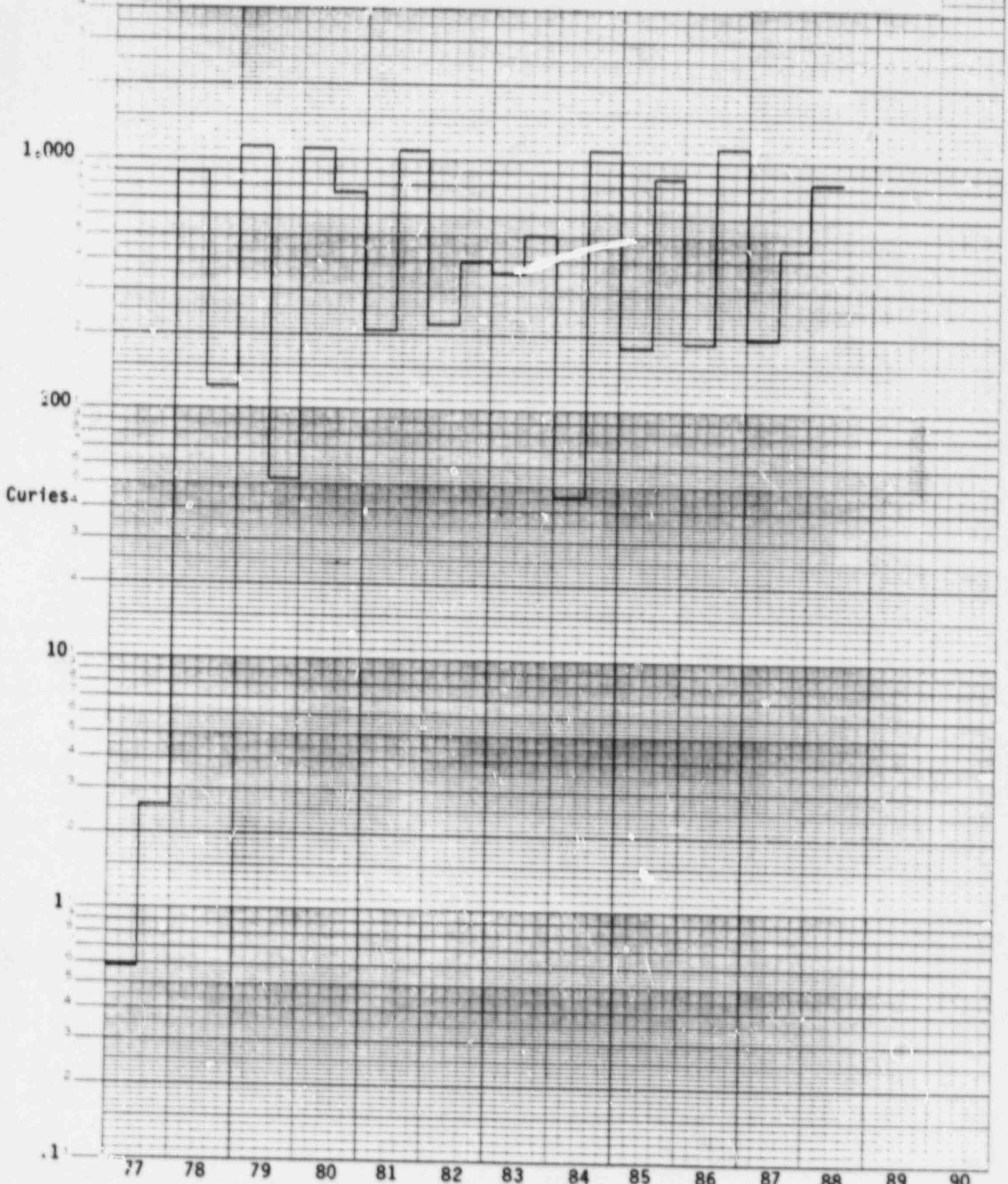
LIQUID RELEASES
TRITIUM



- FIGURE 5 -

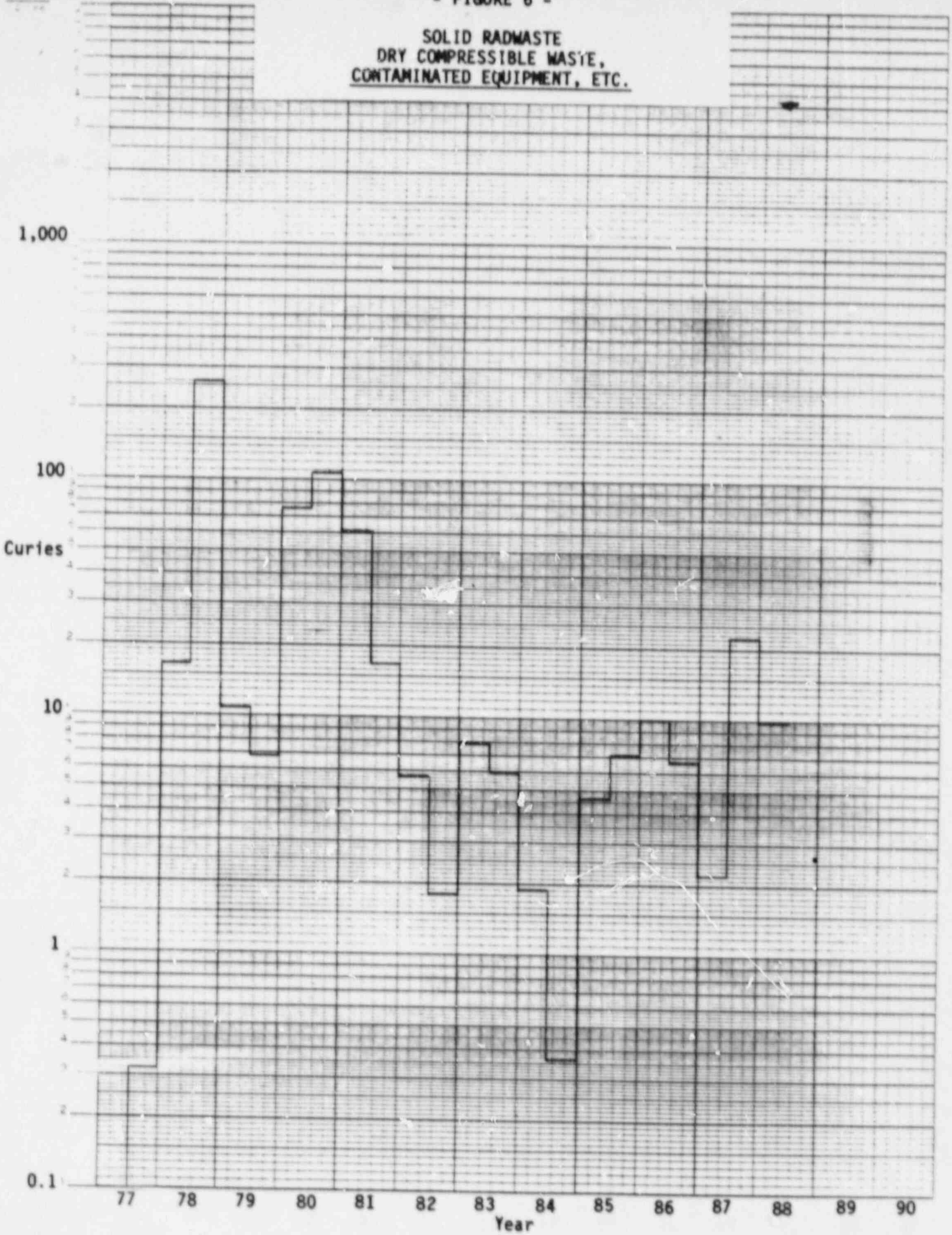
SOLID RADWASTE

SPENT RESINS, FILTER SLUDGES, EVAPORATOR BOTTOMS, ETC.



- FIGURE 6 -

SOLID RADWASTE
DRY COMPRESSIBLE WASTE,
CONTAMINATED EQUIPMENT, ETC.



UNPLANNED RELEASES

There were no unplanned liquid releases for the period of this report. There were two unplanned gaseous releases as described below:

DATE	DESCRIPTION
2-28-88	Steam relief due to reactor trip. Estimated release amount: 5.6 E-4 curies
4-1-88	Steam relief due to failure of limit switch during valve (MSIVS) stroking. Estimated release amount: 9.6 E-5 curies

RADIOACTIVE WASTE TREATMENT SYSTEMS

There were no significant changes to the radioactive waste treatment systems for the period of this report.

ENVIRONMENTAL RADIOLOGICAL MONITORING PROGRAM

The June 1988 land-use census did not identify any new dose calculation locations.

EFFLUENT MONITOR OPERABILITY

No effluent monitor was inoperable for a period of thirty days or more.

ODCM AND PCP

There were several changes to the ODCM and the PCP, the descriptions of which are provided below. The revised pages have also been included.

ODCM CHANGES

PAGE	DESCRIPTION
Introduction	Changed to indicate annual dose calculations.
6, 7, 8, 46, 47, 48	Deleted reference to noble gas monitor LLD.
12, 13, 17, 18, 18a, 18b, 18c, 19, 20, 21, 22, 23, 24	Added guidance for establishing trip setpoints when effluent concentrations are below monitor LLD.
14	Corrected grammar in introductory sentence.
17, 18a, 18c, 19, 21, 22, 23, 24	Changed introductory statement to indicate that effluent monitor trips on concentration only.
80, 82	Corrected environmental station locations.

PCP CHANGES

PAGE	DESCRIPTION
Title	Deleted non-required signatures and added position for Manager, Site Nuclear Services.
i	Added Appendix I to Table of Contents
2	Included Bartlett PCP Report Number
9	Clarified Testing Frequency in Section 5.3.2.2.
14, 15	Added drawings of solidification and dewatering systems.

These PCP changes became effective in January of this year and were submitted in the 1987 year-end report in February. Technical Specifications, however, require submission of PCP changes in the Semiannual Report for the period in which the changes were made. Consequently, the changes are being submitted again.

INTRODUCTION

The Off-Site Dose Calculation Manual (ODCM) is provided to support implementation of the Crystal River Unit 3 Radiological Effluent Technical Specifications. The ODCM contains calculational methods to be used in determining the dose to members of the public resulting from routine radioactive effluents released from Crystal River Unit 3. More accurate estimation of doses is performed annually in preparation of the year-end Semiannual Radioactive Effluent Release Report. The ODCM also contains the methodology used to determine effluent monitor alarm/trip setpoints which assure that releases of radioactive materials remain within specified concentrations.

The ODCM will be controlled by the Site Nuclear Services Department and revisions shall be made with the approval of the Manager, Site Nuclear Services. In addition all revisions must be approved by the Plant Review Committee prior to implementation at Crystal River Unit 3. Historical documentation and distribution of the ODCM shall be the responsibility of the Nuclear Operations Records Manager per NOD-05, Document Control Program.

NUCLIDE ANALYSIS 1.2-1
REACTOR BUILDING PURGE EXHAUST

NUCLIDE	SAMPLE SOURCE	LLD[b] (uCi/cc)
A. Principal Gamma Emitters [a]		
Mn-54	Pre-release grab sample for Batch Type release. Weekly Particulate Filter Analysis for continuous(c) type release.	$1x10^{-4}/1x10^{-11}$
Fe-59		$1x10^{-4}/1x10^{-11}$
Co-58		$1x10^{-4}/1x10^{-11}$
Co-60		$1x10^{-4}/1x10^{-11}$
Zn-65		$1x10^{-4}/1x10^{-11}$
Mo-99		$1x10^{-4}/1x10^{-11}$
Cs-134		$1x10^{-4}/1x10^{-11}$
Cs-137		$1x10^{-4}/1x10^{-11}$
Ce-141		$1x10^{-4}/1x10^{-11}$
Ce-144		$1x10^{-4}/1x10^{-11}$
Kr-87	Pre-release grab sample for Batch type release. Noble Gas monitor during batch and continuous releases Grab sample within 2-6 hr. following startup, shutdown or > 15% RTP change in 1 hr.	$1x10^{-4}$
Kr-88		$1x10^{-4}$
Xe-133		$1x10^{-4}$
Xe-133m		$1x10^{-4}$
Xe-135		$1x10^{-4}$
Xe-138		$1x10^{-4}$
B. Iodine 131	Pre-release grab sample for Batch type release. Weekly charcoal filter and once per 24 hr for 7 days following startup shutdown or > 15% RTP change in 1 hr unless I-131 concentration at site boundary < 10% 10 CFR 20 limit.	NA/1×10^{-12}
C. Tritium	Pre-release Grab Sample and within 12-24 hr following flooding of refueling canal and once per 7 days while canal is flooded.	$1x10^{-6}$
D. Gross Alpha	Monthly Particulate Filter Composite	$1x10^{-11}$
E. Sr-89	Quarterly Particulate Filter Composite	$1x10^{-11}$
F. Sr-90	Quarterly Particulate Filter Composite	$1x10^{-11}$

(a) Other identified Gamma Emitters not listed in this table shall be included in dose and setpoint calculations.

(b) The first value refers to the LLD for pre-release grab sample; the second value refers to the LLD for weekly Particulate Filter Analysis.

(c) Reactor Building Purge is considered continuous after a minimum of one Reactor Building volume has been released on a continuous basis (i.e., first volume is a batch type).

NUCLIDE ANALYSIS 1.2-2
AUXILIARY BUILDING AND FUEL HANDLING AREA EXHAUST

NUCLIDE	SAMPLE SOURCE	LLD (b) (uCi/ml)
A. Principal Gamma Emitters (a)		
Mn-54	Weekly Particulate Filter Analysis.	$1x10^{-4}/1x10^{-11}$
Fe-59		$1x10^{-4}/1x10^{-11}$
Co-58		$1x10^{-4}/1x10^{-11}$
Co-60		$1x10^{-4}/1x10^{-11}$
Zn-65		$1x10^{-4}/1x10^{-11}$
Mo-99		$1x10^{-4}/1x10^{-11}$
Cs-134		$1x10^{-4}/1x10^{-11}$
Cs-137		$1x10^{-4}/1x10^{-11}$
Ce-141		$1x10^{-4}/1x10^{-11}$
Ce-144		$1x10^{-4}/1x10^{-11}$
Kr-87	Monthly Grab Sample and Continuous Noble Gas monitor. Grab sample within 2-6 hr following startup, shutdown or > 15% RTP change in 1 hr.	$1x10^{-4}$
Kr-88		$1x10^{-4}$
Xe-133		$1x10^{-4}$
Xe-133m		$1x10^{-4}$
Xe-135		$1x10^{-4}$
Xe-138		$1x10^{-4}$
B. Iodine 131	Weekly Charcoal Filter analysis and once per 24 hr for 7 days following startup shutdown or > 15% RTP change in 1 hr unless I-131 concentration at site boundary < 10% 10 CFR 20 limit.	$1x10^{-12}$
C. Tritium	Monthly Grab Sample and within 12-24 hr following flooding of refueling canal and once per 7 days while canal is flooded.	$1x10^{-6}$
D. Gross Alpha	Monthly Particulate Filter Composite	$1x10^{-11}$
E. Sr-89	Quarterly Particulate Filter Composite	$1x10^{-11}$
F. Sr-90	Quarterly Particulate Filter Composite	$1x10^{-11}$

(a) Other identified Gamma Emitters not listed in this table shall be included in dose and setpoint calculations.

(b) The first value refers to the LLD for pre-release grab sample; the second value refers to the LLD for weekly Particulate Filter Analysis.

NUCLIDE ANALYSIS 1.2-3
WASTE GAS DECAY TANKS

NUCLIDE	SAMPLE SOURCE	LLD(b) (µCi/ml)
A. Principal Gamma Emitters (a)		
Mn-54	Pre-release Grab sample and Weekly Particulate Filter Sample from RM-A2.	$1x10^{-4}/1x10^{-11}$
Fe-59		$1x10^{-4}/1x10^{-11}$
Co-58		$1x10^{-4}/1x10^{-11}$
Co-60		$1x10^{-4}/1x10^{-11}$
Zn-65		$1x10^{-4}/1x10^{-11}$
Mo-99		$1x10^{-4}/1x10^{-11}$
Cs-134		$1x10^{-4}/1x10^{-11}$
Cs-137		$1x10^{-4}/1x10^{-11}$
Ce-141		$1x10^{-4}/1x10^{-11}$
Ce-144		$1x10^{-4}/1x10^{-11}$
Kr-87	Pre-release Grab sample.	$1x10^{-4}$
Kr-88		$1x10^{-4}$
Xe-133		$1x10^{-4}$
Xe-133m		$1x10^{-4}$
Xe-135		$1x10^{-4}$
Xe-138		$1x10^{-4}$
B. Iodine 131	Weekly Charcoal Filter from RM-A2.	$1x10^{-12}$

(a) Other identified Gamma Emitters not listed in this table shall be included in dose and setpoint calculations.

(b) The first value refers to the LLD for pre-release grab sample; the second value refers to the LLD for weekly Particulate Filter Analysis.

Flow Rates (Variable - based on setpoint needs, nominal or maximum values listed below).

1) Reactor Building Purge Exhaust Duct = 50,000 cfm = 2.4×10^7 cc/sec

2) Auxiliary Building and Fuel Handling Area Exhaust Duct = 156,000 cfm = 7.4×10^7 cc/sec

3) Waste Gas Decay Tank Release Line = 50 cfm max = 2.4×10^4 cc/sec

X/Q = 2.5×10^{-6} sec/m^3 for all vent releases. This value is the highest calculated annual average relative concentration for any area at or beyond the unrestricted area boundary.

In order for a gaseous release to be within the limits of specification 1.1-1, the Projected Dose Rate Ratio (PDRR) must not exceed 1. The PDRR for each limit is calculated as follows:

$$PDRR_{TB} = PDR_{TB} / 500 \quad (1.4)$$

$$PDRR_{SK} = PDR_{SK} / 3000 \quad (1.5)$$

$$PDRR_{ORG} = PDR_{ORG} / 1500 \quad (1.6)$$

PDR_{TB} = Projected Dose Rate to the TOTAL BODY due to noble gas emmissions.

PDR_{SK} = Projected Dose Rate to the SKIN due to noble gas emmissions.

PDR_{ORG} = Projected Dose Rate to any organ due to inhalation of iodine, tritium and particulates with half-lives greater than 8 days.

500 = The allowable total body dose rate due to noble gas gamma emissions in mrem/yr.

3000 = The allowable skin dose rate due to noble gas beta emissions in mrem/yr.

1500 = The allowable organ dose rate in mrem/yr.

If the concentration of radionuclides to be released is less than the effluent monitor LLD set PDRR equal to 1.

Equations 1.1, 1.2, and 1.3 are solved for each release type and release point currently releasing or awaiting release. If relationships 1.4, 1.5, and 1.6 are satisfied, the release can be made under the assumed flow rates. If one or more of the relationships 1.4, 1.5 and 1.6 are not satisfied, action must be taken to reduce the radionuclide release rate prior to initiating a release (or to reduce the radionuclide release rate already in progress).

The following actions are available to reduce the release rates at the three release points.

1) Waste Gas Decay Tanks

a) Release Valve may be throttled
b) Tank contents may be diluted
c) Release may be delayed for longer decay time.

2) Reactor Building Purge Exhaust Duct

a) Dilution flow may be opened to reduce purge rate while maintaining the same flow rate.

3) Auxiliary Building and Fuel Handling Area Exhaust

a) Reduce inlet air supply to areas in Auxiliary Building to reduce radioactivity source rate to vent.
b) Identify and isolate the sources of radioactive releases into the Auxiliary Building.

Effluent Monitor LLD Determination

The Technical Specification LLD equation or the relationship given below may be used to calculate a monitor LLD.

$$LLD = \frac{4.66\sqrt{B}}{Slope}$$

B = Average monitor background count rate in cpm.

Slope = Slope of monitor calibration curve in cpm/μCi/ml

I. **INTRODUCTION**

Prior to initiating a release of liquid radwaste, it must be determined that the concentration of radionuclides to be released and the flow rates at which they will be released will not lead to a release concentration greater than the limits of specification 1.1-2 at the point of discharge.

II. **INFORMATION REQUIRED**

Results of appropriate Nuclide Analysis from Section 1.2

III. **CALCULATION**

$$\text{Discharge Concentration} = \left[\frac{\sum \frac{C_{\gamma i}}{MPC_{\gamma i}} + \frac{C_I}{MPC_I} + \frac{C_G}{MPC_G} + \frac{C_\alpha}{MPC_\alpha} + \frac{C_T}{MPC_T} + \frac{C_s}{MPC_s} + \frac{C_{Fe}}{MPC_{Fe}}}{(D + E) / E} \right]$$

where:

$C_{\gamma i}$ = The concentration of isotope i, in the gamma spectrum excluding I-131 and dissolved or entrained noble gases.

C_I = Iodine 131 concentration.

C_G = Dissolved or entrained noble gas concentration.

C_T = Tritium Concentration from most recent analysis.

C = Gross alpha concentration from most recent analysis.

C_s = Sr-89, 90 concentration from most recent analysis.

C_{Fe} = Fe-55 concentration from most recent analysis.

E = Effluent Stream Flow Rate

D = Dilution Stream Flow Rate (Nuclear Services seawater flow only)

MPC = 10CFR20 Appendix B, Table II, Column 2 Maximum Permissible Concentration by isotope. MPC = 2E-4 uCi/ml total activity for all dissolved or entrained noble gases.

If the Calculated Discharge Concentration is less than or equal to 1, the discharge may be initiated. If the calculated discharge concentration is greater than 1, action must be taken to reduce the effluent concentration or effluent stream flow rate prior to initiating discharge.

Setpoint Calculation 1.4-1
Reactor Building Purge Exhaust Duct Monitor (RM-A1)
(Batch Type Releases)

INTRODUCTION

Following completion of the analyses required by Section 1.2-1 and determination of release rates and concentration limits in accordance with Section 1.3-1, the monitor setpoint requires adjustment to ensure that alarm and pathway isolation occur if nuclide concentration limits are exceeded.

METHODOLOGY

Reactor Building atmosphere is circulated through radiation monitor RM-A6 (containment atmosphere noble gas monitor) and the count rate is observed. The observed count rate is correlated to a corresponding count rate for RM-A1 (Reactor Building purge exhaust duct monitor), and factors are applied to account for background radiation and statistical counting variations, and the pressure difference between the detector chambers and exhaust vent. The obtained value establishes the maximum allowable setpoint. The alarm/trip setpoint is adjusted to this or a more conservative value prior to intitiating the release. If the concentration of radionuclides to be released is less than the effluent monitor LLD "Net CPM" is obtained from the calibration curve by determining the CPM which corresponds to 2.5E-2 μCi/ml.

CALCULATION

$$\text{RM-A1 Setpoint (CPM)} = \left[\frac{\text{Net CPM} \times \text{VF}}{\text{PDRR} \times \text{AL}}\left(\frac{29.9 - V1}{29.9 - V6}\right)\frac{(\mu Ci/cc/CPM)_{A6}}{(\mu Ci/cc/CPM)_{A1}}\right] + Bkg + 3.3\sqrt{Bkg}$$

where:

Net CPM = The observed RM-A6 count rate, in cpm, less background, or obtained from the calibration curve.

= The vent fraction; that portion of the total plant gaseous release associated with this vent and discharge type. Value can be set to a number between 0 and 1. The summation of the vent fractions of RM-A1 and RM-A2 cannot exceed 1.

PDRR = The noble gas gamma emission Projected Dose Rate Ratio calculated in accordance with Section 1.3. This ratio is the actual projected dose rate divided by the allowable dose rate referenced in Section 1.3-1, relationship 1.4.

AL = Administrative Limit to reduce setpoint to 10% of the allowable limit. AL = 10.

V6 = The actual gauge vacuum reading at RM-A6 at the time of sampling.

V1 = The actual or average gauge vacuum reading at RM-A1 during normal operation.

$(\mu Ci/cc/CPM)_{A6}$ = μCi/cc per cpm for RM-A6. This is based on an actual sample or derived from the calibration curve.

$(\mu Ci/cc/CPM)_{A1}$ = $\mu Ci/cc$ per cpm for RM-A1. This is based on an actual sample or derived from the calibration curve.

Bkg = RM-A1 background count rate in cpm.

$3.3\sqrt{Bkg}$ = A statistical spread on the background count rate which represents a 99.95% confidence level on monitor counting. This factor is included to prevent inadvertent high/trip alarms due to random counts on the monitor.

SETPOINT CALCULATION 1.4-1A
REACTOR BUILDING PURGE EXHAUST DUCT MONITOR (RM-A1)
(SPECIAL RELEASE FOR FUNCTIONAL TESTING OF THE REACTOR BUILDING PURGE SYSTEM)

INTRODUCTION

Following completion of the analyses required by Section 1.2-1 and determination of release rates and concentration limits in accordance with Section 1.3-1, the monitor setpoint requires adjustment to ensure that alarm and pathway isolation occur if nuclide concentration limits are exceeded.

METHODOLOGY

Auxiliary Building and Fuel Handling Area atmosphere is continuously passed through radiation monitor RM-A2 and the count rate is observed. The observed count rate is correlated to a corresponding count rate for RM-A1, and factors are applied to account for background radiation and statistical counting variations, and the pressure difference between the detector chambers and exhaust vent. The obtained value establishes the maximum allowable setpoint. The alarm/trip setpoint is adjusted to this or a more conservative value prior to intitiating the release. If the concentration of radionuclides to be released is less than the effluent monitor LLD "Net CPM" is obtained from the calibration curve by determining the CPM which corresponds to 2.5E-2 μCi/ml.

CALCULATION

$$\text{RM-A1 Setpoint (CPM)} = \left[\frac{\text{Net CPM} \times \text{VF}}{\text{PDRR} \times \text{AL}}\left(\frac{29.9 - \text{V1}}{29.9 - \text{V2}}\right)\frac{(\mu\text{Ci/cc/CPM})_{A2}}{(\mu\text{Ci/cc/CPM})_{A1}}\right] + \text{Bkg} + 3.3\sqrt{\text{Bkg}}$$

Where:

Net CPM = The observed RM-A2 count rate, in cpm, less background or obtained from the calibration curve.

VF = The vent fraction; that portion of the total plant gaseous release associated with this vent and discharge type. VF can be set to a value from 0 to 1. The sum of RM-A1 and RM-A2 vent fractions can not exceed 1.

PDRR = The noble gas gamma emission Project Dose Rate Ratio calculated in accordance with Section 1.3. This ratio is the actual projected dose rate divided by the allowable dose rate referenced in Section 1.3-1 relationship 1.4.

AL = Administrative Limit to reduce setpoint to 10% of allowable limit. AL = 10.

V2 = The actual gauge vacuum reading at RM-A2 at the time of sampling.

V1 = The actual or average gauge vacuum reading at RM-A1 during normal operation.

$(\mu Ci/cc/CPM)_{A2}$ = $\mu Ci/cc$ per cpm for RM-A2. This is based on an actual sample or derived from the calibration curve.

$(\mu Ci/cc/CPM)_{A1}$ = $\mu Ci/cc$ per cpm for RM-A1. This is based on an actual sample or derived from the calibration curve.

Bkg = RM-A1 background count rate in cpm.

$3.3\sqrt{Bkg}$ = A statistical spread on the background count rate which represents a 99.95% confidence level on monitoring counting. This factor is included to prevent inadvertent high/trip alarms due to random counts on the monitor.

SETPOINT CALCULATION 1.4-1B
REACTOR BUILDING PURGE EXHAUST DUCT MONITOR (RM-A1)
(SPECIAL RELEASE FOLLOWING ILRT OF REACTOR BUILDING)

INTRODUCTION

Following completion of the analyses required by Section 1.2-1 and determination of release rates and concentration limits in accordance with Section 1.3-1, the monitor setpoint should be adjusted to ensure that alarm and pathway isolation occur if nuclide concentration limits are exceeded.

METHODOLOGY

Reactor Building atmosphere is circulated through a sampling apparatus. The Noble gas sample is analyzed to determine the projected dose rate ratio (PDRR). Net CPM is obtained from the calibratyion curve by determining the CPM which corresponds to 2.5E-2 μCi/ml. These values are combined with the monitor background, vent fraction, and administrative limit to arrive at the monitor setpoint. The obtained value establishes the maximum allowable setpoint. The alarm/trip setpoint is adjusted to this or a more conservative value prior to intitiating the release.

Shortly, after beginning the purge, new RM-A1 alarm/trip setpoints are determined using the methodology of Setpoint Calculation 1.4-2.

CALCULATION

$$\text{RM-A1 Setpoint (CPM)} = \left[\frac{\text{Net CPM} \times \text{VF}}{\text{PDRR} \times \text{AL}}\right] + \text{Bkg} + 3.3\sqrt{\text{Bkg}}$$

Where:

Net CPM	=	A value derived from RM-A1 calibration curve.
VF	=	The vent fraction; that portion of the total plant gaseous release associated with this vent and discharge type. VF can be set to a value from 0 to 1. The sum of RM-A1 and RM-A2 vent fractions can not exceed 1.
PDRR	=	1
AL	=	Administrative Limit to reduce setpoint to 10% of allowable limit. AL = 10.
Bkg	=	RM-A1 background count rate in cpm.
$3.3\sqrt{\text{Bkg}}$	=	A statistical spread on the background count rate which represents a 99.95% confidence level on monitoring counting. This factor is included to prevent inadvertent high/trip alarms due to random counts on the monitor.

Setpoint Calculation 1.4-2
Reactor Building Purge Exhaust Duct Monitor (RM-A1)
(Continuous Type Releases)

INTRODUCTION

Following completion of the analyses required by Section 1.2-1 and determination of release rates and concentration limits in accordance with Section 1.3-1, the monitor setpoint requires adjustment to ensure that alarm and pathway isolation occur if nuclide concentration limits are exceeded.

METHODOLOGY

Reactor Building atmosphere is passing through radiation monitor RM-A1 during a continuous type release. Factors are applied to the observed count rate to account for background radiation and statistical counting variations. The obtained value establishes the maximum allowable setpoint. The alarm/trip setpoint is adjusted to this or a more conservative value weekly during continuous releases. If the concentration of radionuclides to be released is less than the effluent monitor LLD "Net CPM" is obtained from the calibration curve by determining the CPM which corresponds to 2.5E-2 μCi/ml.

CALCULATION

$$\text{RM-A1 Setpoint (CPM)} = \left[\frac{\text{Net CPM} \times \text{VF}}{\text{PDRR} \times \text{AL}}\right] + \text{Bkg} + 3.3\sqrt{\text{Bkg}}$$

where:

Net CPM = The observed RM-A1 count rate, in cpm, less background, or obtained from the calibration curve.

VF = The vent fraction; that portion of the total plant gaseous release associated with this vent and discharge type. Value can be set to a number between 0 and 1. The summation of the vent fractions of RM-A1 and RM-A2 cannot exceed 1.

PDRR = The noble gas gamma emission Projected Dose Rate Ratio calculated in accordance with Section 1.3. This ratio is the actual projected dose rate divided by the allowable dose rate referenced in Section 1.3-1, relationship 1.4.

AL = Administrative Limit to reduce setpoint to 10% of the allowable limit. Admin. Limit = 10.

Bkg = RM-A1 background count rate in cpm.

$3.3\sqrt{\text{Bkg}}$ = A statistical spread on the background count rate which represents a 99.95% confidence level on monitor counting. This factor is included to prevent inadvertent high/trip alarms due to random counts on the monitor.

Setpoint Calculation 1.4-3
Auxiliary Building & Fuel Handling Area Exhaust Monitor (RM-A2)
(Continuous Type Releases)

INTRODUCTION

Following completion of the analyses required by Section 1.2-2 and determination of release rates and concentration limits in accordance with Section 1.3-1, the monitor setpoint requires adjustment to assure that alarm and pathway isolation occur if the nuclide concentration exceeds the determined limits.

METHODOLOGY

Auxiliary Building and Fuel Handling Area atmosphere is continuously passing through radiation monitor RM-A2. Factors are applied to the observed count rate to account for background radiation and statistical counting variations. The obtained value establishes the maximum allowable setpoint. The alarm/trip setpoint is adjusted to this or a more conservative value weekly during continuous releases. If the concentration of radionuclides to be released is less than the effluent monitor LLD "Net CPM" is obtained from the calibration curve by determining the CPM which corresponds to 8E-3 μCi/ml.

CALCULATION

$$\text{RM-A2 Setpoints (CPM)} = \left[\frac{\text{Net CPM} \times \text{VF}}{\text{PDRR} \times \text{AL}}\right] + \text{Bkg} + 3.3\sqrt{\text{Bkg}}$$

where:

Net CPM = The observed RM-A2 count rate, in cpm, less background or obtained from the calibration curve.

VF = The vent fraction; that portion of the total plant gaseous release associated with this vent and discharge type. Value can be set to a number between 0 and 1. The summation of the vent fractions of RM-A1 and RM-A2 cannot exceed 1.

PDRR = The noble gas gamma emission Projected Dose Rate Ratio calculated in accordance with Section 1.3. This ratio is the actual projected dose rate divided by the allowable dose rate referenced in Section 1.3-1, relationship 1.4.

AL = Administrative Limit to reduce setpoint to 10% of the allowable limit. AL = 10.

Bkg = RM-A2 background count rate in cpm.

$3.3\sqrt{\text{Bkg}}$ = A statistical spread on the background count rate which represents a 99.95% confidence level on monitor counting. This factor is included to prevent inadvertent high/trip alarms due to random counts on the monitor.

INTRODUCTION

Following completion of the analyses required by Section 1.2-3 and determination of release rates and concentration limits in accordance with Section 1.3-1, the monitor setpoint requires adjustment to assure that alarm and pathway isolation occur if nuclide concentration limits are exceeded.

METHODOLOGY

Prior to initiating a Waste Gas Decay Tank release, its contents are drawn through radiation monitor RM-A11 and returned to the waste gas header. Factors are applied to the observed count rate to account for background radiation and statistical counting variations. The obtained value establishes the maximum allowable setpoint. The alarm/trip setpoint is adjusted to this or a more conservative value prior to intitiating the release. If the concentration of radionuclides to be released is less than the effluent monitor LLD "Net CPM" is obtained from the calibration curve by determining the CPM which corresponds to 20 μCi/ml.

CALCULATION

$$\text{RM-A11 Setpoint (CPM)} = \left[\frac{\text{Net CPM} \times \text{VF} \times 10}{\text{PDRR} \times \text{AL} \times \text{P}}\right] + \text{Bkg} + 3.3\sqrt{\text{Bkg}}$$

where:

Net CPM	=	The observed RM-A11 count rate, in cpm, less background, or obtained from the calibration curve.
VF	=	The vent fraction; that portion of the total plant gaseous release associated with this vent and discharge type. Value is equal to 0.5.
PDRR	=	The noble gas gamma emission Projected Dose Rate Ratio calculated in accordance with Section 1.3. This ratio is the actual projected dose rate divided by the allowable dose rate referenced in Section 1.3-1, relationship 1.4.
AL	=	Administrative Limit to reduce setpoint to 10% of allowable limit. AL = 10.
10	=	The maximum pressure (psig) which RM-A11 detector chamber should be subjected to. This corresponds to a flow of 15 CFM from the release line to the vent.
P	=	Pressure (psig) in RM-A11 at time of obtaining net CPM.
Bkg	=	RM-A11 background count rate in cpm.
$3.3\sqrt{\text{Bkg}}$	=	A statistical spread on the background count rate which represents a 99.95% confidence level on monitor counting. This factor is included to prevent inadvertent high/trip alarms due to random counts on the monitor.

INTRODUCTION

Following completion of the analyses required by Section 1.2-4 and determination of release rates and concentration limits in accordance with Section 1.3-2, the monitor setpoint requires adjustment to ensure that alarm and pathway isolation occur if nuclide concentration limits are exceeded.

METHODOLOGY

Evaporator Condensate Storage Tank or Laundry and Shower Sump Tank contents are circulated through radiation monitor RM-L2 and returned to the auxiliary building sump to obtain the actual count rate at RM-L2 for the concentration contained in the tank for release. The observed count rate is adjusted for release flow, background and statistical counting variations, particular to this release flow path. The resulting value is used as the alarm/trip setpoint and RM-L2 is adjusted to this or a more conservative value prior to initiating the release. If the concentration of radionuclides to be released is less than the effluent monitor LLD set "$\Sigma Ci/MPCi$" equal to 1 and determine "Net CPM" from the calibration curve by locating the CPM which corresponds to 3E-7 µCi/ml.

CALCULATION

$$\text{RM-L2 Setpoint (CPM)} = \left[\frac{\text{Net CPM} \times AF \times (E + D)}{(\Sigma\, C_i/MPC_i) \times E}\right] + Bkg + 3.3\sqrt{Bkg}$$

where:

Net CPM	=	The observed RM-L2 count rate, in cpm, less background, or obtained from the calibration curve.
AF	=	Administration Factor to account for error in setpoint determination. AF = 0.8.
$\Sigma C_i/MPC_i$	=	The ratio of the actual gamma emitting concentrations (excluding dissolved and entrained gases) of the tank contents to be released to the Maximum Permissible Concentration (MPC) as listed in 10 CFR 20, Table II, Column 2 for unrestricted areas.
E	=	The release flow rate of waste to be discharged in gallons per minute. A maximum flow rate of 100 gpm will be used for the Evaporator Condensate Storage Tanks and 40 gpm for the Laundry and Shower Sump Tanks.
D	=	The dilution flow from the Nuclear Services Sea Water system in gallons per minute.
Bkg	=	RM-L2 background count rate in cpm.
$3.3\sqrt{Bkg}$	=	A statistical spread on the background count rate which represents a 99.95% confidence level on monitor counting. This factor is included to prevent inadvertent high/trip alarms due to random counts on the monitor.

Setpoint Calculation 1.4-6
Turbine Building Basement Discharge Line Monitor (RM-L7)
(Continuous Type Releases)

INTRODUCTION

The activity released through the Turbine Building Basement Discharge Line Monitor RM-L7 is analyzed in accordance with Section 1.2-5. The setpoint is a fixed concentration based on worst case nuclide released at the worst case rate as described in the Methodology Section below. The monitor setpoint is adjusted to ensure isolation of the release pathway if nuclide concentration limits are exceeded.

METHODOLOGY

The alarm/trip setpoint determination is based on the worst case assumption that I-131 is the only nuclide being discharged. This assumption equates all counts on RM-L7 to I-131 with an MPC of 3×10^{-7} uci/ml. I-131 has the most conservative MPC of the nuclides available to this release path and 'visible' to RM-L7. The setpoint is based on assuring 1 MPC or less of I-131 in the discharge canal and is determined by obtaining the cpm on RM-L7 calibration curve which corresponds to a concentration of 3×10^{-7} uci/ml and applying the flow dilution factor, background counts, and statistical counting variations. The resulting value is used as the alarm/trip setpoint and RM-L7 is adjusted to this or a more conservative value to maintain control on release conditions.

CALCULATION

$$\text{RM-L7 Setpoint (CPM)} = \left[\frac{\text{CPM} \times (E + D)}{E}\right] + \text{Bkg} + 3.3\sqrt{\text{Bkg}}$$

where:

- CPM = The counts per minute corresponding to 3×10^{-7} uci/ml (1 MPC I-131) from the current RM-L7 calibration curve.
- E = The maximum release flow rate of water able to be discharged in gallons per minute.
- D = The dilution flow from the Nuclear Services Sea Water system in gallons per minute.
- Bkg = The background count rate at RM-L7 in cpm.
- $3.3\sqrt{\text{Bkg}}$ = A statistical spread on the background count rate which represents a 99.95% confidence level on monitor counting. This factor is included to prevent inadvertent high/trip alarms due to random counts on the monitor.

Turbine Building Basement Discharge Line Monitor (RM-L7)
(Batch Type Releases)

INTRODUCTION

Following completion of the analyses required by Section 1.2-4 and determination of release rates and concentration limits in accordance with Section 1.3-2, the monitor setpoint requires adjustment to ensure that alarm and pathway isolation occur if nuclide concentration limits are exceeded.

METHODOLOGY

Station Drain Tank (SDT-1) contents are circulated through radiation monitor RM-L7 and returned to the sump to obtain the actual count rate at RM-L7 for the concentration contained in the tank for release. The observed count rate is adjusted for release flow, background and statistical counting variations, particular to this release flow path. The resulting value is used as the alarm/trip setpoint and RM-L7 is adjusted to this or a more conservative value prior to initiating the release. If the concentration of radionuclides to be released is less than the effluent monitor LLD set "$\Sigma Ci/MPCi$" equal to 1 and determine "Net CPM" from the calibration curve by locating the CPM which corresponds to 3E-7 µCi/ml.

CALCULATION

$$\text{RM-L7 Setpoint (CPM)} = \left[\frac{\text{Net CPM} \times AF \times E + D}{(\Sigma\, C_i/MPC_i) \times E}\right] + Bkg + 3.3\sqrt{Bkg}$$

where:

Net CPM	=	The observed RM-L7 count rate, in cpm, less background.
AF	=	Administration Factor to account for error in setpoint determination. AF = 0.8
$\Sigma C_i/MPC_i$	=	The ratio of the actual gamma emitting concentrations (excluding dissolved and entrained gases) of the tank contents to be released to the Maximum Permissible Concentration (MPC) as listed in 10 CFR 20, Table II, Column 2 for unrestricted areas.
E	=	The release flow rate of waste to be discharged in gallons per minute. A maximum flow rate of 600 gpm will be used.
D	=	The dilution flow from the Nuclear Services Sea Water system in gallons per minute.
Bkg	=	RM-L7 background count rate in cpm.
$3.3\sqrt{Bkg}$	=	A statistical spread on the backgound count rate which represents a 99.95% confidence level on monitor counting. This factor is included to prevent inadvertent high/trip alarms due to random counts on the monitor.

NUCLIDE ANALYSIS 4.2-1
REACTOR BUILDING PURGE EXHAUST

NUCLIDE	SAMPLE SOURCE	LLD(b) (uCi/ml)
A. Principal Gamma Emitters (a)		
Mn-54	Batch release particulate filter for Batch Releases. Weekly Particulate Filter Analysis for continuous(c) type release.	$1x10^{-4}/1x10^{-11}$
Fe-59		$1x10^{-4}/1x10^{-11}$
Co-58		$1x10^{-4}/1x10^{-11}$
Co-60		$1x10^{-4}/1x10^{-11}$
Zn-65		$1x10^{-4}/1x10^{-11}$
Mo-99		$1x10^{-4}/1x10^{-11}$
Cs-134		$1x10^{-4}/1x10^{-11}$
Cs-137		$1x10^{-4}/1x10^{-11}$
Ce-141		$1x10^{-4}/1x10^{-11}$
Ce-144		$1x10^{-4}/1x10^{-11}$
Kr-87	Pre-release grab sample for Batch type release. Weekly grab sample for continuous type release.	$1x10^{-4}$
Kr-88		$1x10^{-4}$
Xe-133		$1x10^{-4}$
Xe-133m		$1x10^{-4}$
Xe-135		$1x10^{-4}$
Xe-138		$1x10^{-4}$
B. Iodine 131	Batch release charcoal filter for Batch Releases. Weekly charcoal filter for continuous releases.	$NA/1 \times 10^{-12}$
C. Tritium	Pre-release Grab Sample.	$1x10^{-6}$
D. Gross Alpha	Monthly Particulate Filter Composite	$1x10^{-11}$
E. Sr-89	Quarterly Particulate Filter Composite	$1x10^{-11}$
F. Sr-90	Quarterly Particulate Filter Composite	$1x10^{-11}$

(a) Other identified Gamma Emitters not listed in this table shall be included in dose calculations.
(b) The first value refers to the LLD for pre-release grab sample; the second value refers to the LLD for weekly Particulate Filter Analysis.
(c) Reactor Building Purge is considered continuous after minimum of one Reactor Building volumes have been released on a continuous basis (i.e., first one volume is a batch type).

NUCLIDE ANALYSIS 4.2-2
AUXILIARY BUILDING AND FUEL HANDLING AREA EXHAUST

NUCLIDE		SAMPLE SOURCE	LLD(b) (uCi/ml)
A.	Principal Gamma Emitters (a)		
	Mn-54	Weekly Particulate Filter Analysis.	$1x10^{-4}/1x10^{-11}$
	Fe-59		$1x10^{-4}/1x10^{-11}$
	Co-58		$1x10^{-4}/1x10^{-11}$
	Co-60		$1x10^{-4}/1x10^{-11}$
	Zn-65		$1x10^{-4}/1x10^{-11}$
	Mo-99		$1x10^{-4}/1x10^{-11}$
	Cs-134		$1x10^{-4}/1x10^{-11}$
	Cs-137		$1x10^{-4}/1x10^{-11}$
	Ce-141		$1x10^{-4}/1x10^{-11}$
	Ce-144		$1x10^{-4}/1x10^{-11}$
	Kr-87	Monthly Grab Sample.	$1x10^{-4}$
	Kr-88		$1x10^{-4}$
	Xe-133		$1x10^{-4}$
	Xe-133m		$1x10^{-4}$
	Xe-135		$1x10^{-4}$
	Xe-138		$1x10^{-4}$
B.	Iodine 131	Weekly Charcoal Filter Analysis.	$1x10^{-12}$
C.	Tritium	Monthly Grab Sample.	$1x10^{-6}$
D.	Gross Alpha	Monthly Particulate Filter Composite	$1x10^{-11}$
E.	Sr-89	Quarterly Particulate Filter Composite	$1x10^{-11}$
F.	Sr-90	Quarterly Particulate Filter Composite	$1x10^{-11}$

(a) Other identified Gamma Emitters not listed in this table shall be included in dose calculations.

(b) The first value refers to the LLD for pre-release grab sample; the second value refers to the LLD for weekly Particulate Filter Analysis.

NUCLIDE ANALYSIS 4.3-3
WASTE GAS DECAY TANKS

NUCLIDE	SAMPLE SOURCE	LLD(b) (uCi/ml)
A. Principal Gamma Emitters (a)		
Mn-54	Weekly Particulate Filter sample (from RM-A2)	$1x10^{-4}/1x10^{-11}$
Fe-59		$1x10^{-4}/1x10^{-11}$
Co-58		$1x10^{-4}/1x10^{-11}$
Co-60		$1x10^{-4}/1x10^{-11}$
Zn-65		$1x10^{-4}/1x10^{-11}$
Mo-99		$1x10^{-4}/1x10^{-11}$
Cs-134		$1x10^{-4}/1x10^{-11}$
Cs-137		$1x10^{-4}/1x10^{-11}$
Ce-141		$1x10^{-4}/1x10^{-11}$
Ce-144	Pre-release Grab sample	$1x10^{-4}$
Kr-87		$1x10^{-4}$
Kr-88		$1x10^{-4}$
Xe-133		$1x10^{-4}$
Xe-133m		$1x10^{-4}$
Xe-135		$1x10^{-4}$
Xe-138		$1x10^{-4}$
B. Iodine 131	Weekly Charcoal Filter (from RM-A2)	$1x10^{-12}$

(a) Other identified Gamma Emitters not listed in this table shall be included in dose and setpoint calculations.

(b) The first value refers to the LLD for pre-release grab sample; the second value refers to the LLD for weekly Particulate Filter Analysis.

Table 5.1-1

Environmental Radiological Monitoring
Stations Locations

STATION	LOCATION	DIRECTION FROM PLANT	DISTANCE FROM PLANT (mi)
C04	State Park Old Dam on River near road intersection	ENE	6.3
C07	Crystal River Public Water Plant	ESE	7.5
C09	Fort Island Gulf Beach	S	3.2
C10	Indian Waters Public Water Supply	ESE	5.9
C13	Mouth of Intake Canal	WSW	3.4
C14H	Head of Discharge Canal	NW	0.1
C14M	Midpoint of Discharge Canal	W	1.2
C14G	Discharge Canal at Gulf of Mexico	W	2.8
C18	Yankeetown City Well	N	5.2
C19	NW Corner State Roads 488 & 495	ENE	8.5
C29	Discharge Area	N	2.0
C30	Intake Area	WSW	3.6
C40	Near N.E. Site Boundary near excavated pond & pump station	E	3.5
C41	Onsite meteorological tower	SW	0.4
C46	North Pump Station	N	0.4
C47	University of Florida, Gainesville	NNE	52
C48A	Onsite North of CR 4 & 5	N	0.8
C48B	Onsite NNE of CR 4 & 5	NNE	0.8

REVISION 11

TABLE 5.1-3

RING TLDs
(5 MILE RING)

LOCATION	DIRECTION	DISTANCE (Mi.)
C18	N	5.2
C03	NNE	5.3
C04	NE	6.3
C74	ENE	5.5
C75	E	4.2
C76	ESE	5.4
C08	SE	3.5
C77	SSE	3.2
C09	S	3.2
C78	WSW	4.1
C14G	W	2.8
C01	NW	4.9
C79	NNW	5.0

REVISION 11

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NUCLEAR OPERATIONS

CRYSTAL RIVER UNIT 3

PROCESS CONTROL PROGRAM

Approved by: G L Robinson
Chem/Rad Supt.

Date: 2/2/88

Approved by: Sarah Glenn Johnson
Manager, Site Nuclear Services

Date: 2/5/88

TABLE OF CONTENTS

2.0 REFERENCES

2.1. NUREG 0472, "Radiological Effluent Technical Specifications for PWR's"

2.2 WP 101, "Packaging, Storing, and Shipping of Radioactive Materials"

2.3 WP 102, "Radioactive Shipments Certificates of Compliance"

2.4 WP 201, "Processing with the Nuclear Waste Demineralizer System"

2.5 WP 202, "Liner Dewatering Leg Construction"

2.6 WP 301, "Radioactive Waste Solidification"

2.7 OP 413, "Waste Drumming System

2.8 OP 601B, "Secondary Resin Liner Dewatering"

2.9 SP 743, "Solidification Test Batch Verification Program"

2.10 Bartlett PCP, Report #A5889

2.11 CNSI Bead Resin Dewatering Procedure

2.12 10 CFR 61, "Licensing Requirements for Land Disposal of Radioactive Waste"

2.13 Crystal River Unit 3 Technical Specifications

2.14 South Carolina Department of Health and Environmental Control, Radioactive Materials License #097, Amendment #34

5.3.2 Test Frequency

5.3.2.1 Process Test Frequency

A process test solidification shall be made prior to full scale solidification to determine ratios and additives as per Section 5.3.1.1.

5.3.2.2 Solidification Test Frequency

The PCP shall be used to verify the solidification of at least one (1) representative test specimen from at least every tenth batch of each type of wet radioactive waste.

5.3.3 Acceptance Criteria

SP 743, "Solidification Test Batch Verification Program," stipulates the activities and documentation necessary to verify acceptance of solidified waste.

The solidified waste acceptance criteria is verified by:

a. Visually inspecting for defects in the structure.
b. Uniformity in color and density.
c. No free-standing liquid (<0.5% of total waste volume).
d. Free-standing monolith.
e. After 24 hours from solidification, the final cured product shall resist penetration when probed by hand with a spatula or firm object (>50 psi).

If any portion of the specimen fails to pass the Acceptance Criteria, the applicable actions of Section 5.3.4 must be met.

5.3.4 Corrective Action

a. If the initial test specimen from a batch of waste fails to verify solidification, representative test specimens from each consecutive batch of the same type of wet waste shall be collected and tested until at least 3 consecutive initial test specimens demonstrate solidification. The process and/or additives shall be modified as required, as provided in Section 4.1, to assure solidification of subsequent batches of waste.

b. If any test specimen fails to verify solidification, the solidification of the batch under test shall be suspended until such time as additional test specimens can be obtained, alternate solidification parameters can be determined in accordance with the Process Control Program, and a subsequent test verifies solidification. Solidification of the batch may then be resumed using the alternative solidification parameters determined by the Process Control Program.

c. With installed equipment incapable of meeting Technical Specification 3.7.13.4 or declared inoperable, restore the equipment to operate status or provide for contract capability to process wastes as necessary to satisfy all applicable transportation and disposal requirements.

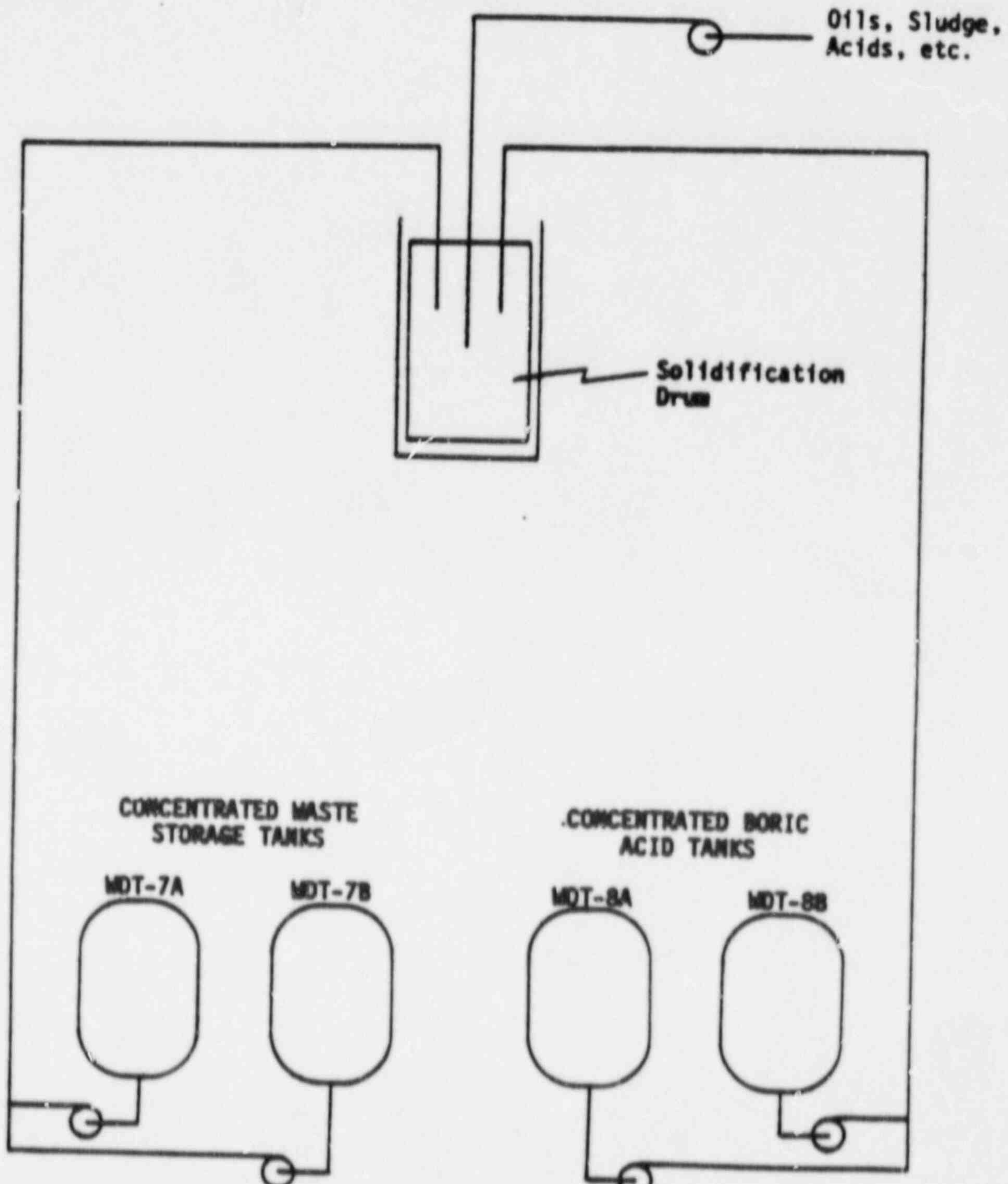

FIGURE I-1: SOLIDIFICATION SYSTEM

REVISION 02

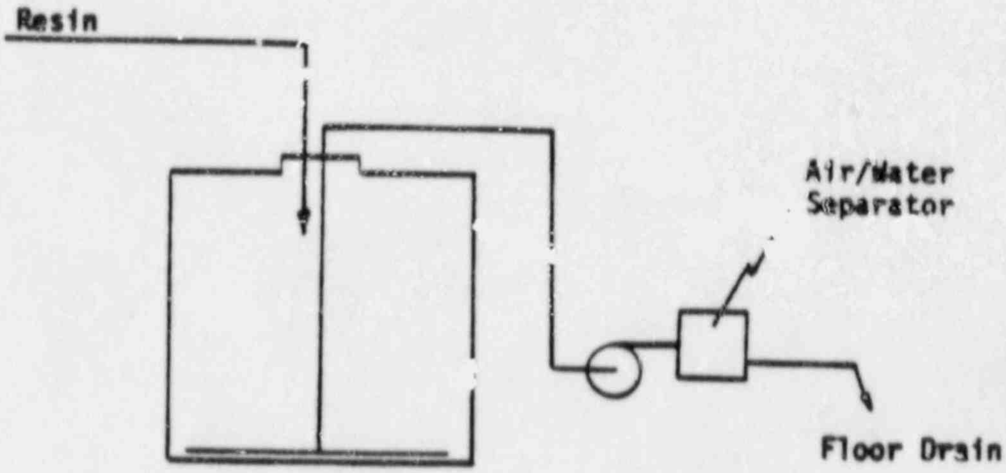

FIGURE I-11: SECONDARY RESIN DEWATERING SYSTEM

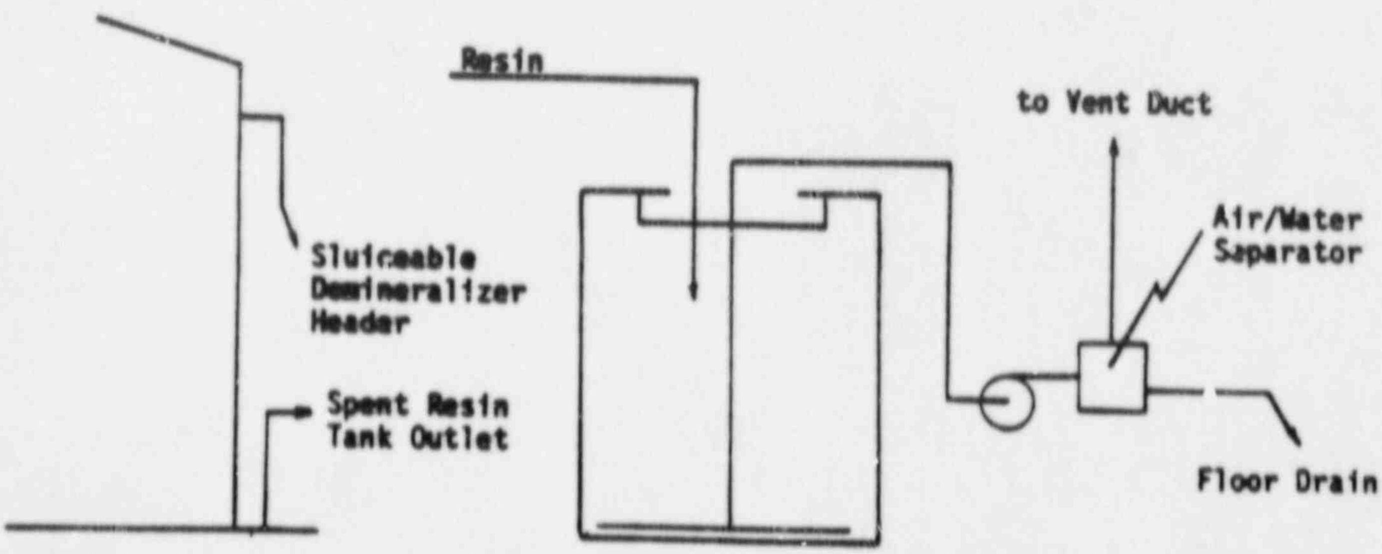

FIGURE I-111: PRIMARY RESIN DEWATERING SYSTEM

Florida Power
CORPORATION

August 26, 1988
3F0888-16

Document Control Desk
U.S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: Crystal River Unit 3
Docket No. 50-302
Operating License No. DPR-72
Semiannual Radioactive Effluent Release Report

Dear Sir:

Pursuant to Title 10, Code of Federal Regulations, Part 50.36(a)(2) and Crystal River Unit 3 Technical Specification 6.9.1.5(d), Florida Power Corporation hereby submits the Crystal River Unit 3 Semiannual Radioactive Effluent Release Report for the period January 1, 1988 through June 30, 1988.

If you have any questions concerning this matter please contact this office.

Sincerely,

Rolf C. Widell, Director
Nuclear Operations Site Support

RCW:DLH
xc: Regional Administrator, Region II
Senior Resident Inspector

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1/1

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