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Docket Nos. 50-424  
50-425

U. S. Nuclear Regulatory Commission  
ATTN: Document Control Desk  
Washington, D. C. 20555

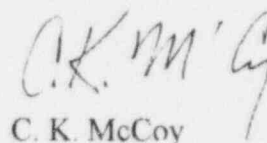
**VOGTLE ELECTRIC GENERATING PLANT**  
**1993 ANNUAL REPORT - PART 2**

Gentlemen:

In accordance with the applicable regulatory requirements, Georgia Power Company hereby submits Part 2 of the 1993 Annual Report of operating information.

The remainder of the 1993 reports not previously submitted are included.

Sincerely,

  
C. K. McCoy

CKM/JLL/gmb

Enclosure Annual Report - Part 2

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GEORGIA POWER COMPANY

VOGTLE ELECTRIC GENERATING PLANT UNITS 1 AND 2

NRC DOCKET NOS. 50-424 AND 50-425

FACILITY OPERATING LICENSE NOS. NPF-68 AND NPF-81

1993 ANNUAL REPORT - PART 2

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- II. ANNUAL RADIOLOGICAL ENVIRONMENTAL SURVEILLANCE REPORT
- III. ANNUAL ENVIRONMENTAL OPERATING REPORT
- IV. ANNUAL RADIOLOGICAL EFFLUENT RELEASE REPORT



I

GEORGIA POWER COMPANY

VOGTLE ELECTRIC GENERATING PLANT - UNITS 1 AND 2

NRC DOCKET NOS. 50-424 AND 50-425

FACILITY OPERATING LICENSE NOS. NPF-68 AND NPF-81

INTRODUCTION

The Vogtle Electric Generating Plant Units 1 and 2 are powered by pressurized water reactors, each rated at 3565 megawatts thermal. It is located on the Savannah River in Burke County Georgia, 34 miles southeast of Augusta. The Unit 1 operating license was received on January 16, 1987 and commercial operation started on May 31, 1987. Unit 1 is operating in its fifth fuel cycle. Unit 2 received its operating license on February 9, 1989, began commercial operation on May 20, 1989. Unit 2 is operating in its fourth fuel cycle.

II

GEORGIA POWER COMPANY

VOGTLE ELECTRIC GENERATING PLANT - UNITS 1 AND 2

NRC DOCKET NOS. 50-424 AND 50-425

FACILITY OPERATING LICENSE NOS. NPF-68 AND NPF-81

ANNUAL RADIOLOGICAL ENVIRONMENTAL SURVEILLANCE REPORT  
CALENDAR YEAR 1993

VOGTLE ELECTRIC GENERATING PLANT  
RADIOLOGICAL ENVIRONMENTAL SURVEILLANCE REPORT

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## ACRONYMS

A2LA	American Association of Laboratory Accreditation
ASTM	American Society for Testing and Materials
CL	Confidence Level
EL	Environmental Laboratory
EPA	Environmental Protection Agency
GPC	Georgia Power Company
LLD	Lower Limit of Detection
MDA	Minimum Detectable Activity
MDD	Minimum Detectable Difference
NA	Not Applicable
NDM	No Detectable Measurement(s)
NRC	Nuclear Regulatory Commission
ODCM	Offsite Dose Calculation Manual
REMP	Radiological Environmental Monitoring Program
RL	Reporting Level
RM	River Mile
SRS	Savannah River Site
TLD	Thermoluminescent Dosimeter
TS	Technical Specifications
VEGP	Alvin W. Vogtle Electric Generating Plant



# VOGTLE ELECTRIC GENERATING PLANT RADIOLOGICAL ENVIRONMENTAL SURVEILLANCE REPORT

## 1.0

### INTRODUCTION

The objectives of the Radiological Environmental Monitoring Program (REMP) are to ascertain the levels of radiation and concentrations of radioactivity in the environs of the Alvin W. Vogtle Electric Generating Plant (VEGP) and to assess any radiological impact upon the environment due to plant operations. A comparison between results obtained at control stations (locations where radiological levels are not expected to be significantly affected by plant operations) and at indicator stations (locations where it is anticipated that radiological levels are more likely to be affected by plant operations) provides some basis for such an assessment. A comparison of the results obtained during preoperation with those obtained during operation provides a further basis for this assessment.

The preoperational stage of the REMP started in August of 1981 when initial collections of the samples were made. There was a phase-in period of a few years before the preoperational program was fully implemented. The transition from the preoperational stage to the operational stage hinged about initial criticality for Unit 1 which occurred on March 9, 1987.

The REMP activities for 1993 are reported herein. All dates in this report are for 1993 unless otherwise indicated. The REMP was conducted in accordance with Technical Specifications (TS) Section 6.7.4.g and Offsite Dose Calculation Manual (ODCM) Section 3.

On March 4, 1992, VEGP proposed changes to their TS and ODCM in order to implement Nuclear Regulatory Commission (NRC) Generic Letter 89-01. These changes became effective on January 5, 1993 and were implemented within 30 days. The changes included the relocation of the procedural details of the REMP from the TS to the ODCM while the programmatic controls remained in the TS. Although there were no essential changes in the REMP, there were changes in the Section Numbers in both the TS and the ODCM and there were some minor word changes in the text. The Section Numbers of the TS and ODCM referenced throughout this report are those which became effective on January 5, 1993. On January 1, 1994, the ODCM was further amended and completely reorganized to accommodate implementation of the new 10CFR20.

A summary description of the REMP is provided in Section 2 of this report; maps showing the sampling stations are keyed to a table indicating the direction and distance of each station from a point midway between the two reactors. An annual summary of the main laboratory analysis results obtained from the samples utilized for environmental monitoring is presented in Section 3. A discussion of the results, including assessments of any radiological impacts upon the environment and the results of the land use census and the river survey, is provided in Section 4. The results of the Interlaboratory Comparison Program are presented in Section 5. Conclusions are stated in Section 6.

## 2.0

### SUMMARY DESCRIPTION

A summary description of the REMP is provided in Table 2-1. This table portrays the program in the manner by which it is being regularly carried out. Table 2-1 is essentially a copy of ODCM Table 3.1-1 which delineates the program's requirements. Sampling locations required by Table 2-1 are described in Table 2-2 and are shown on maps in Figures 2-1 through 2-4. This description of the sample locations closely follows the table and figures in ODCM 3.0.

In Footnote (1) of ODCM Table 3.1-1, it is stated that deviations are permitted from the required sampling schedule if specimens are unobtainable due to circumstances such as hazardous conditions, seasonal unavailability, and malfunction of equipment. Any deviations are accounted for in the discussions for each particular sample type in Section 4.

All laboratory analyses were performed by Georgia Power Company's (GPC) Environmental Laboratory (EL) in Smyrna, Georgia. Since 1987, the EL has been accredited by the American Association of Laboratory Accreditation (A2LA) for radiochemistry. The A2LA is a nonprofit, nongovernmental, public service, membership society dedicated to the formal recognition of competent laboratories and related activities. Accreditation is based upon internationally accepted criteria for laboratory competence.

TABLE 2-1 (SHEET 1 OF 5)

## SUMMARY DESCRIPTION OF RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

<u>Exposure Pathway and/or Sample</u>	<u>Number of Representative Samples and Sample Locations</u>	<u>Sampling and Collection Frequency</u>	<u>Type and Frequency of Analysis</u>
1. Direct Radiation	<p>Thirty nine routine monitoring stations with two or more dosimeters placed as follows:</p> <p>An inner ring of stations, one in each meteorological sector in the general area of the site boundary;</p> <p>An outer ring of stations, one in each meteorological sector in the 6 mile range from the site: and</p> <p>Special interest areas, such as population centers, nearby recreation areas, schools and control stations.</p>	Quarterly	Gamma dose quarterly

TABLE 2-1 (SHEET 2 OF 5)

## SUMMARY DESCRIPTION OF RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

<u>Exposure Pathway and/or Sample</u>	<u>Number of Representative Samples and Sample Locations</u>	<u>Sampling and Collection Frequency</u>	<u>Type and Frequency of Analysis</u>
2. Airborne			
Radioiodine and Particulates	<p>Samples from seven locations:</p> <p>Five locations close to the site boundary in different sectors;</p> <p>A community having the highest calculated annual average groundlevel D/Q; and</p> <p>A control location in the vicinity of a population center at a distance of about 14 miles.</p>	Continuous sampler operation with sample collection weekly, or more frequently if required by dust loading	<p>Radioiodine canister: I-131 analysis weekly</p> <p>Particulate sampler: Gross beta analysis (1) following filter change and gamma isotopic analysis (2) of composite (by location) quarterly</p>
3. Waterborne			
a. Surface (3)	<p>One sample upriver</p> <p>Two samples downriver</p>	Composite sample over one month period (4)	Gamma isotopic analysis (2) monthly. Composite for tritium analysis quarterly

TABLE 2-1 (SHEET 3 OF 5)

## SUMMARY DESCRIPTION OF RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

<u>Exposure Pathway and/or Sample</u>	<u>Number of Representative Samples and Sample Locations</u>	<u>Sampling and Collection Frequency</u>	<u>Type and Frequency of Analysis</u>
b. Drinking	Two samples at each of the two nearest water treatment plants that could be affected by plant discharges  Two samples at a control location	Composite sample of river water near the intake of each water treatment plant over two week period (4) when I-131 analysis is required to be performed on each sample; monthly composite otherwise; and grab sample of finished water at each water treatment plant every two weeks or monthly, as appropriate.	I-131 analysis on each sample when the dose calculated for the consumption of the water is greater than 1 mrem per year (5). Composite for gross beta and gamma isotopic analysis (2) on raw water monthly. Gross beta, gamma isotopic and I-131 analyses on grab sample of finished water monthly. Composite for tritium analysis on raw and finished water quarterly.
c. Sediment from Shoreline	One sample from downriver area with existing or potential recreational value  One sample from upriver area with existing or potential recreational value	Semiannually	Gamma isotopic analysis (2) semiannually
4. Ingestion			
a. Milk	Two samples from milking animals (6) at control locations at a distance of about 10 miles or more	Biweekly	Gamma isotopic analysis (2,7) biweekly



TABLE 2-1 (SHEET 4 OF 5)

## SUMMARY DESCRIPTION OF RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

<u>Exposure Pathway and/or Sample</u>	<u>Number of Representative Samples and Sample Locations</u>	<u>Sampling and Collection Frequency</u>	<u>Type and Frequency of Analysis</u>
b. Fish	At least one sample of any commercially or recreationally important species in vicinity of plant discharge area	Semiannually	Gamma isotopic analysis (2) on edible portions semiannually
	At least one sample of any commercially or recreationally important species in an area not influenced by plant discharges		
	At least one sample of any anadromous species in vicinity of plant discharge	During spring spawning season	Gamma isotopic analysis (2) on edible portions annually
c. Grass or Leafy Vegetation	One sample from two onsite locations near the site boundary in different sectors	Monthly during growing season	Gamma isotopic analysis (2,7) monthly
	One sample from a control location at a distance of about 17 miles		

TABLE 2-1 (SHEET 5 OF 5)

SUMMARY DESCRIPTION OF  
RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

TABLE NOTATIONS

- (1) Airborne particulate sample filters shall be analyzed for gross beta radioactivity 24 hours or more after sampling to allow for radon and thoron daughter decay. If gross beta activity in air particulate samples is greater than 10 times the yearly mean of control samples, gamma isotopic analysis shall be performed on the individual samples.
- (2) Gamma isotopic analysis means the identification and quantification of gamma-emitting radionuclides that may be attributable to the effluents from the facility.
- (3) The upriver sample is taken at a distance beyond significant influence of the discharge. The downriver samples are taken in areas beyond and near the mixing zone.
- (4) Composite sample aliquots shall be collected at time intervals that are very short (e.g., hourly) relative to the compositing period (e.g., monthly) to assure obtaining a representative sample.
- (5) The dose shall be calculated for the maximum organ and age group, using the methodology and parameters in the Offsite Dose Calculation Manual (ODCM).
- (6) A milking animal is a cow or goat producing milk for human consumption.
- (7) If gamma isotopic analysis is not sensitive enough to meet the Lower Limit of Detection (LLD) for I-131, a separate analysis for I-131 may be performed.

TABLE 2-2 (SHEET 1 OF 3)

## RADIOLOGICAL ENVIRONMENTAL SAMPLING LOCATIONS

Station Number	Station Type (1)	Descriptive Location	Direction (2)	Distance (miles) (2)	Sample Type (3)
1	I	Hancock Landing Road	N	1.1	D
2	I	River Bank	NNE	0.8	D
3	I	Discharge Area	NE	0.6	A
3	I	River Bank	NE	0.7	D
4	I	River Bank	ENE	0.8	D
5	I	River Bank	E	1.0	D
6	I	Plant Wilson	ESE	1.1	D
7	I	Simulator Building	SE	1.7	ADV
8	I	River Road	SSE	1.1	D
9	I	River Road	S	1.1	D
10	I	Meteorological Tower	SSW	0.9	A
10	I	River Road	SSW	1.1	D
11	I	River Road	SW	1.2	D
12	I	River Road	WSW	1.2	AD
13	I	River Road	W	1.3	D
14	I	River Road	WNW	1.8	D
15	I	Hancock Landing Road	NW	1.5	DV
16	I	Hancock Landing Road	NNW	1.4	AD
17	O	Savannah River Site (SRS), River Road	N	5.4	D
18	O	SRS, D Area	NNE	5.0	D
19	O	SRS, Road A.13	NE	4.6	D
20	O	SRS, Road A.13.1	ENE	4.8	D
21	O	SRS, Road A.17	E	5.3	D
22	O	River Bank	ESE	5.2	D
23	O	River Road	SE	4.6	D
24	O	Chance Road	SSE	4.9	D
25	O	Chance Road near Highway 23	S	5.2	D
26	O	Highway 23 and Ebenezer Church Road	SSW	4.6	D
27	O	Highway 23 opposite Boll Weevil Road	SW	4.7	D
28	O	Thomas Road	WSW	5.0	D
29	O	Claxton-Lively Road	W	5.1	D
30	O	Nathaniel Howard Road	WNW	5.0	D
31	O	River Road at Allen's Chapel Fork	NW	5.0	D
32	O	River Bank	NNW	4.7	D
33	O	Hunting Cabin	SE	3.3	D
35	O	Girard	SSE	6.6	AD

TABLE 2-2 (SHEET 2 OF 3)

## RADIOLOGICAL ENVIRONMENTAL SAMPLING LOCATIONS

Station Number	Station Type (1)	Descriptive Location	Direction (2)	Distance (miles) (2)	Sample Type (3)
36	C	GPC Waynesboro Op Hq	WSW	13.9	AD
37	C	Waynesboro Substation	WSW	16.7	DV
43	O	Employee's Rec Area	SW	2.2	D
47	C	Oak Grove Church	SE	10.4	D
48	C	McBean Cemetery	NW	10.2	D
80	C	Augusta Water Treatment Plant	NNW	29.0	W(4)
81	C	Savannah River	N	2.5	F(5)S(6)
82	C	Sav River (RM 151.2)	NNE	0.8	R
83	I	Sav River (RM 150.4)	ENE	0.8	RS(6)
84	O	Sav River (RM 149.5)	ESE	1.6	R
85	I	Savannah River	ESE	4.3	F(5)
87	I	Beaufort-Jasper County Water Treatment Plant, Beaufort, SC	SE	74	W(7)
88	I	Cherokee Hill Water Treatment Plant, Port Wentworth, GA	SSE	71	W(8)
98	C	W. C. Dixon Dairy	SE	9.8	M
99	C	Boyceland Dairy	W	20.9	M

## TABLE NOTATIONS

## (1) Station Types

- C - Control
- I - Indicator
- O - Other

## (2) Direction and distance are reckoned from a point midway between the two reactors.

## (3) Sample Types

- A - Airborne Radioactivity
- D - Direct Radiation
- F - Fish
- M - Milk
- R - River Water
- S - River Shoreline Sediment
- W - Drinking Water
- V - Vegetation

TABLE 2-2 (SHEET 3 OF 3)

RADIOLOGICAL ENVIRONMENTAL MONITORING LOCATIONS

TABLE NOTATIONS (CONTINUED)

- (4) The intake for the Augusta Water Treatment Plant is located on the Augusta Canal. The entrance to the canal is at River Mile (RM) 207 on the Savannah River. The canal effectively parallels the river. The intake to the pumping station is about 4 miles down the canal and only a tenth of a mile from the river (across land).
- (5) A 5 mile stretch of the river is generally needed to obtain adequate fish samples. Samples are normally gathered between RM 153 and 158 for upriver collections and between RM 144 and 149.4 for downriver collections.
- (6) Sediment is collected at locations with existing or potential recreational value. Because high water, shifting of the river bottom, or other reasons could cause a suitable location for sediment collections to become unavailable or unsuitable, a stretch of the river between RM 148.5 and 150.5 was designated for downriver collections while a stretch between RM 153 and 154 was designated for upriver collections. In practice, collections are normally made at RM 150.2 for downriver collections and RM 153.3 for upriver collections.
- (7) The intake for the Beaufort-Jasper County Water Treatment Plant is located at the end of a canal which begins at RM 39.3 on the Savannah River. This intake is about 16 miles by line of sight down the canal from its beginning on the Savannah River.
- (8) The intake for the Cherokee Hill Water Treatment Plant is located on Abercorn Creek which is about one and a quarter creek miles from its mouth on the Savannah River at RM 29.

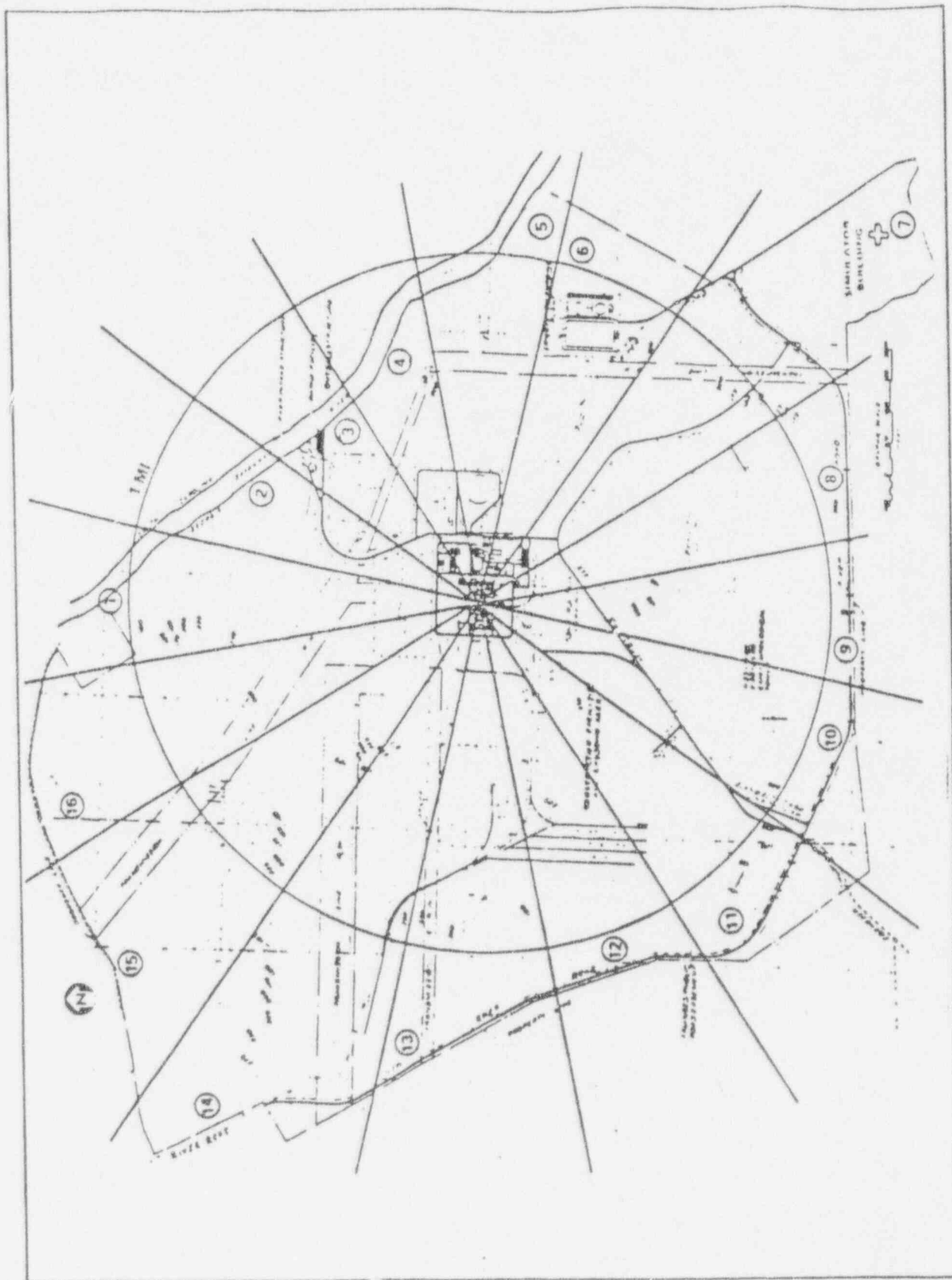


FIGURE 2-1 TERRESTRIAL STATIONS NEAR SITE BOUNDARY



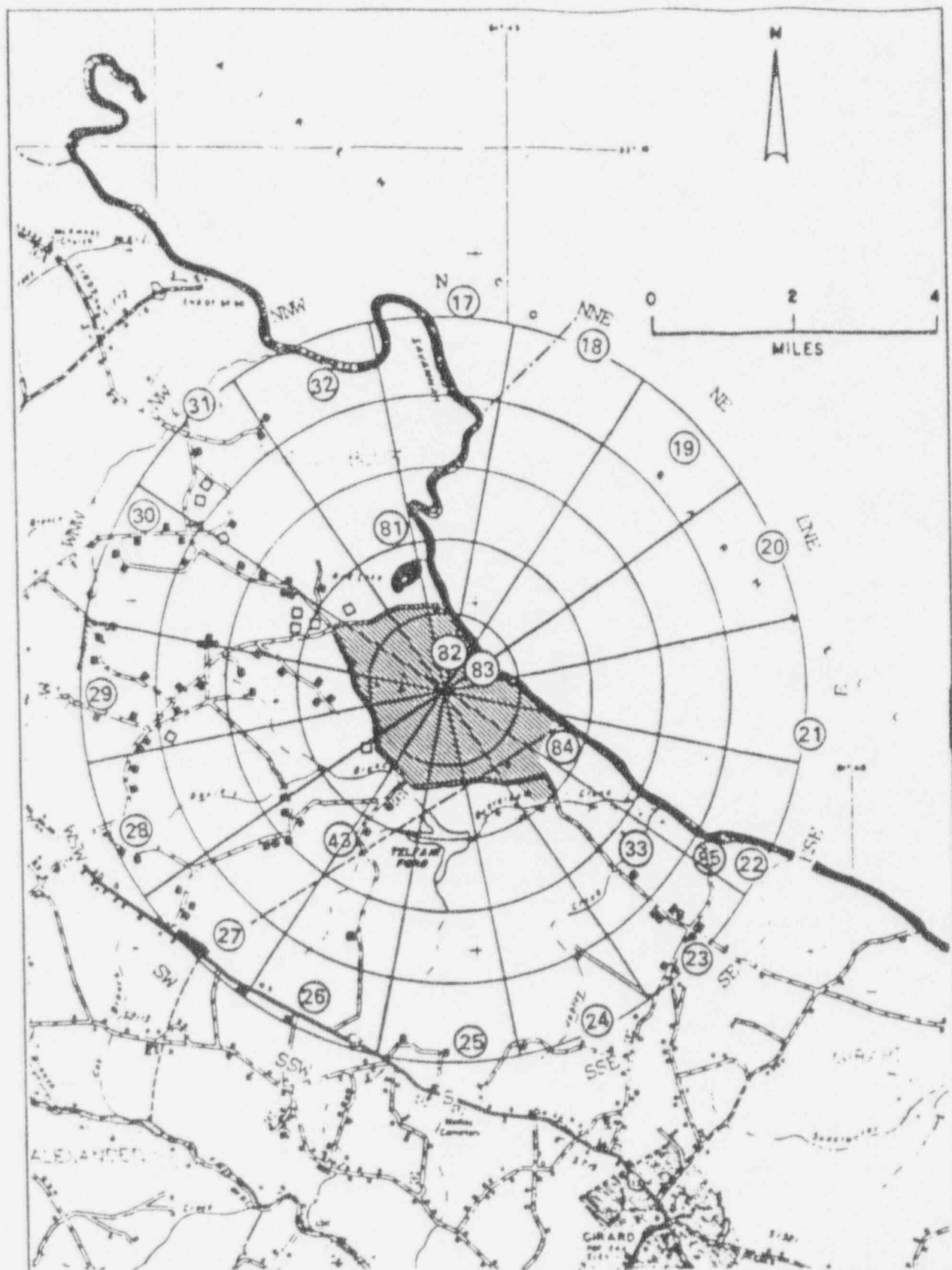


FIGURE 2-2 TERRESTRIAL AND AQUATIC STATIONS WITHIN SIX MILES

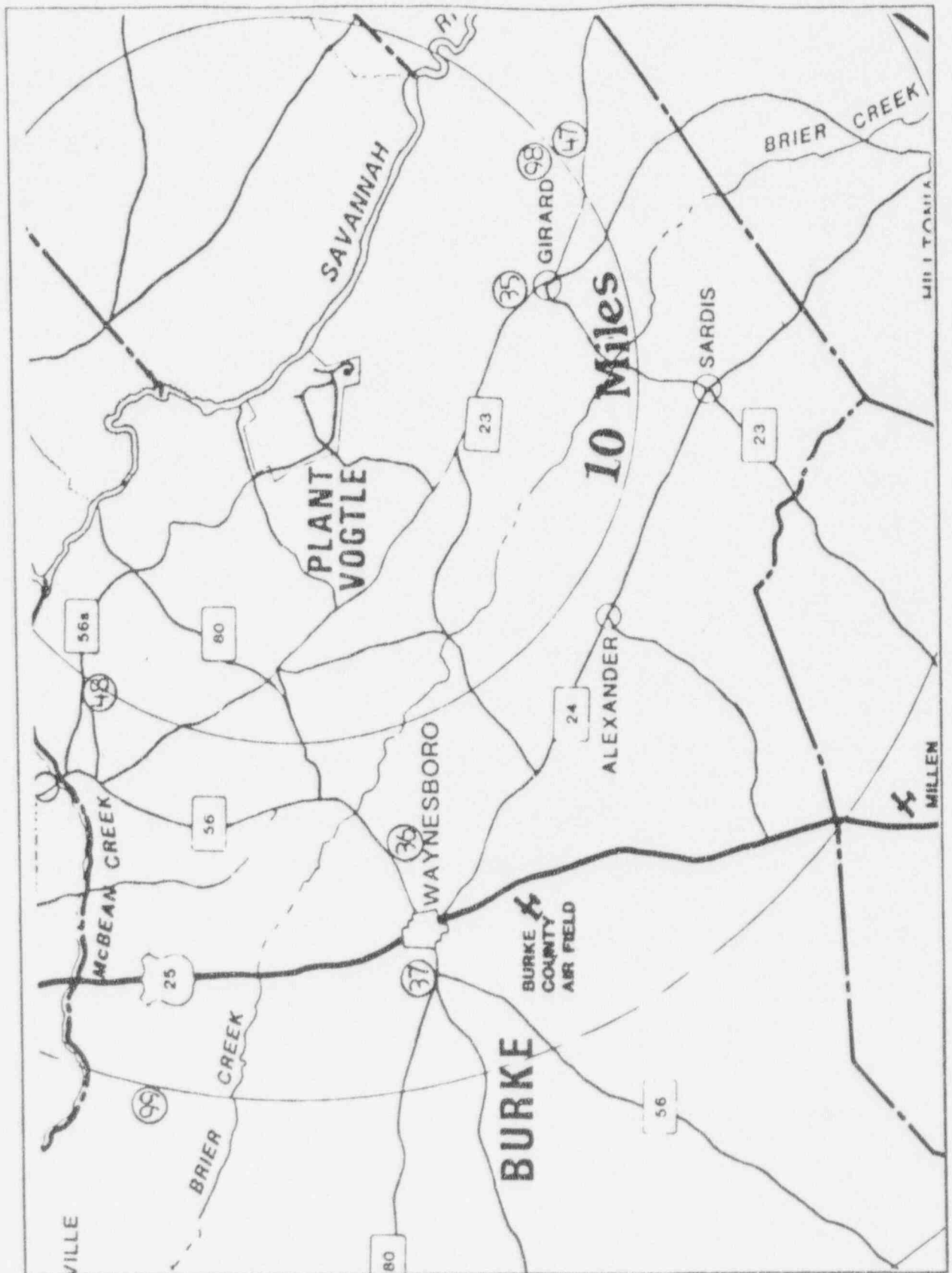


FIGURE 2-3 TERRESTRIAL STATIONS BEYOND SIX MILES

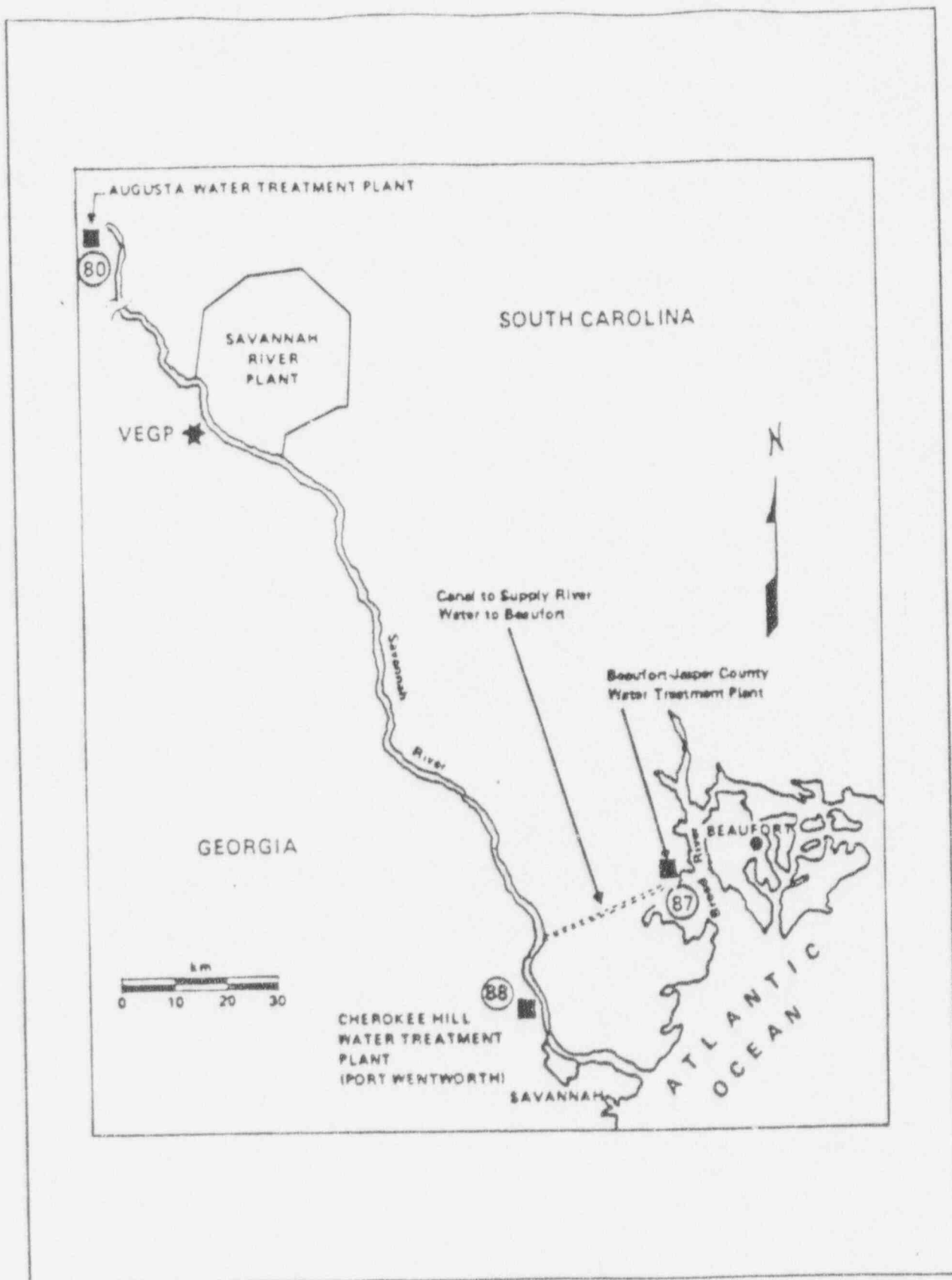


FIGURE 2-4 DRINKING WATER STATIONS

### 3.0 RESULTS SUMMARY

In accordance with ODCM 6.1, summarized and tabulated results for all of the regular samples collected for the year at the designated indicator and control stations are presented in Table 3-1 in a format similar to that found in Table 3 of the Nuclear Regulatory Commission (NRC) Radiological Assessment Branch Technical Position, Revision 1, November 1979. Results for samples collected at locations other than indicator or control stations or in addition to those stipulated by Table 2-1 are discussed in Section 4.

Naturally occurring radionuclides which are not found in the plant's effluent releases are not required to be reported. Be-7 which occurs abundantly in nature is also produced in the reactors. Miniscule quantities are found in the liquid releases. No other naturally occurring radionuclides have been found in the plant's effluent releases. Hence, the radionuclides of interest for the samples monitoring liquid releases (river water, drinking water, fish and river shoreline sediment) are man-made radionuclides plus Be-7, while only man-made radionuclides are of interest for the other samples.

TABLE 3-1 (SHEET 1 OF 10)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM SUMMARY FOR 1993  
Vogtle Electric Generating Plant, Docket Nos. 50-424 and 50-425  
Burke County, Georgia

Medium or Pathway Sampled (Unit of Measurement)	Type and Total Number of Analyses Performed	Lower Limit of Detection (a) (LLD)	Indicator Locations Mean (b) Range (Fraction)	Location with Highest Annual Mean Name Distance & Direction	Mean (b) Range (Fraction)	Control Locations Mean (b) Range (Fraction)	Number of Reportable Occurrences
3-2 Airborne Particulates (fCi/m <sup>3</sup> )	Gross Beta 312	10	21.2 3-34 (260/260)	No. 3 Discharge 0.6 miles NE	21.5 7-33 (52/52)	21.4 6-33 (52/52)	0
	Gamma Isotopic 28						
	Cs-134	50	NDM (c)		NDM	NDM	0
	Cs-137	60	NDM		NDM	NDM	0
Airborne Radioiodine (fCi/m <sup>3</sup> )	I-131 312	70	NDM		NDM	NDM	0
Direct Radiation (mR/91 days)	Gamma Dose 79	NA (d)	12.4 7-18 (63/63)	No. 36 GPC Op Hq 13.9 miles WSW	14.8 14-16 (4/4)	12.4 11-16 (16/16)	0

TABLE 3-1 (SHEET 2 OF 10)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM SUMMARY FOR 1993  
Vogtle Electric Generating Plant, Docket Nos. 50-424 and 50-425  
Burke County, Georgia

Medium or Pathway Sampled (Unit of Measurement)	Type and Total Number of Analyses Performed	Lower Limit of Detection (a) (LLD)	Indicator Locations Mean (b) Range (Fraction)	Location with Highest Annual Mean Name Distance & Direction	Mean (b) Range (Fraction)	Control Locations Mean (b) Range (Fraction)	Number of Reportable Occurrences
Milk (pCi/l)	Gamma Isotopic 54						
	Cs-134	15	NA		NDM	NDM	0
	Cs-137	18	NA		NDM	NDM	0
	Ba-140	60	NA		NDM	NDM	0
	La-140	15	NA		NDM	NDM	0
	I-131 54	1	NA		NDM	NDM	0
Vegetation (pCi/kg wet)	Gamma Isotopic 36						
	I-131	60	NDM		NDM	NDM	0
	Cs-134	60	NDM		NDM	NDM	0
	Cs-137	80	46.4 29-94 (8/24)	No. 7 Simulator 1.7 miles SE	49.0 29-94 (5/12)	34.1 34-34 (1/12)	0



TABLE 3-1 (SHEET 3 OF 10)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM SUMMARY FOR 1993  
Vogtle Electric Generating Plant, Docket Nos. 50-424 and 50-425  
Burke County, Georgia

Medium or Pathway Sampled (Unit of Measurement)	Type and Total Number of Analyses Performed	Lower Limit of Detection (a) (LLD)	Indicator Locations Mean (b) Range (Fraction)	Location with Highest Annual Mean (b) Mean (b) Range (Fraction) Name Distance & Direction	Control Locations Mean (b) Range (Fraction)	Number of Reportable Occurrences
River Water (pCi/l)	Gamma Isotopic 24					
	Be-7	124 (e)	NDM		NDM	0
	Mn-54	15	NDM		NDM	0
	Fe-59	30	NDM		NDM	0
	Co-58	15	NDM		NDM	0
	Co-60	15	NDM		NDM	0
	Zn-65	30	NDM		NDM	0
	Zr-95	30	NDM		NDM	0
	Nb-95	15	NDM		NDM	0
	I-131	15	NDM		NDM	0
	Cs-134	15	NDM		NDM	0
	Cs-137	19	NDM		NDM	0
	Ba-140	60	NDM		NDM	0
	La-140	15	NDM		NDM	0

TABLE 3-1 (SHEET 4 OF 10)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM SUMMARY FOR 1993  
Vogtle Electric Generating Plant, Docket Nos. 50-424 and 50-425  
Burke County, Georgia

Medium or Pathway Sampled (Unit of Measurement)	Type and Total Number of Analyses Performed	Lower Limit of Detection (a) (LLD)	Indicator Locations Mean (b) Range (Fraction)	Location with Highest Annual Mean		Control Locations Mean (b) Range (Fraction)	Number of Reportable Occurrences
				Name Distance & Direction	Mean (b) Range (Fraction)		
5-3 Water Near Intakes to Water Treatment Plants (pCi/l)	Tritium 8	3000	712 158-1410 (4/4)	No. 83 Downriver 0.4 miles	712 158-1410 (4/4)	238 213-263 (2/4)	0
	Gross Beta 36	4	3.17 1.4-5.6 (24/24)	No. 88 Port Went Downriver 122 miles	3.18 1.6-5.5 (12/12)	2.83 1.2-5.1 (11/12)	0
	Gamma Isotopic 36						
	Be-7	124 (e)	NDM		NDM	NDM	0
	Mn-54	15	NDM		NDM	NDM	0
	Fe-59	30	NDM		NDM	NDM	0
	Co-58	15	NDM		NDM	NDM	0
	Co-60	15	NDM		NDM	NDM	0
	Zn-65	30	NDM		NDM	NDM	0
	Zr-95	30	NDM		NDM	NDM	0
	I-131 (f)	15	NDM		NDM	NDM	0

TABLE 3-1 (SHEET 5 OF 10)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM SUMMARY FOR 1993  
Vogtle Electric Generating Plant, Docket Nos. 50-424 and 50-425  
Burke County, Georgia

Medium or Pathway Sampled (Unit of Measurement)	Type and Total Number of Analyses Performed	Lower Limit of Detection (a) (LLD)	Indicator Locations Mean (b) Range (Fraction)	Location with Highest Annual Mean Name Distance & Direction	Mean (b) Range (Fraction)	Control Locations Mean (b) Range (Fraction)	Number of Reportable Occurrences
	Cs-134	15	NDM		NDM	NDM	0
	Cs-137	18	NDM		NDM	NDM	0
	Ba-140	60	NDM		NDM	NDM	0
	La-140	15	NDM		NDM	NDM	0
	Tritium 12	3000	955 532-1170 (7/8)	No. 87 Beaufort Downriver 112 miles	968 575-1170 (3/4)	NDM	0
Finished Water at Water Treatment Plants (pCi/l)	Gross Beta 36	4	2.23 0.8-4.7 (24/24)	No. 80 Augusta Upriver 56 miles	2.30 1.4-5.4 (11/12)	2.30 1.4-5.4 (11/12)	0
	Gamma Isotopic 36						
	Be-7	124 (e)	NDM		NDM	NDM	0
	Mn-54	15	NDM		NDM	NDM	0
	Fe-59	30	NDM		NDM	NDM	0

TABLE 3-1 (SHEET 6 OF 10)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM SUMMARY FOR 1993  
Vogtle Electric Generating Plant, Docket Nos. 50-424 and 50-425  
Burke County, Georgia

Medium or Pathway Sampled (Unit of Measurement)	Type and Total Number of Analyses Performed	Lower Limit of Detection (a) (LLD)	Indicator Locations Mean (b) Range (Fraction)	Location with Highest Annual Mean Name Distance & Direction	Mean (b) Range (Fraction)	Control Locations Mean (b) Range (Fraction)	Number of Reportable Occurrences
	Co-58	15	NDM		NDM	NDM	0
	Co-60	15	NDM		NDM	NDM	0
	Zn-65	30	NDM		NDM	NDM	0
	Zr-95	30	NDM		NDM	NDM	0
	Nb-95	15	NDM		NDM	NDM	0
	Cs-134	15	NDM		NDM	NDM	0
	Cs-137	18	NDM		NDM	NDM	0
	Ba-140	60	NDM		NDM	NDM	0
	La-140	15	NDM		NDM	NDM	0
	I-131	1	NDM		NDM	NDM	0
	36						
	Tritium	2000	993	No. 88	1011	NDM	0
	12		589-1320 (7/8)	Port Went Downriver 122 miles	589-1320 (4/4)		

TABLE 3-1 (SHEET 7 OF 10)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM SUMMARY FOR 1993  
Vogtle Electric Generating Plant, Docket Nos. 50-424 and 50-425  
Burke County, Georgia

Medium or Pathway Sampled (Unit of Measurement)	Type and Total Number of Analyses Performed	Lower Limit of Detection (a) (LLD)	Indicator Locations Mean (b) Range (Fraction)	Location with Highest Annual Mean Name Distance & Direction	Mean (b) Range (Fraction)	Control Locations Mean (b) Range (Fraction)	Number of Reportable Occurrences
Anadromous Fish (pCi/kg wet)	Gamma Isotopic 1						
	Be-7	655 (e)	NDM		NDM	NDM	0
	Mn-54	130	NDM		NDM	NDM	0
	Fe-59	260	NDM		NDM	NDM	0
	Co-58	130	NDM		NDM	NDM	0
	Co-60	130	NDM		NDM	NDM	0
	Zn-65	260	NDM		NDM	NDM	0
	Cs-134	130	NDM		NDM	NDM	0
	Cs-137	150	NDM		NDM	NDM	0

TABLE 3-1 (SHEET 8 OF 10)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM FOR 1993  
 Vogtle Electric Generating Plant, Docket Nos. 50-424 and 50-425  
 Burke County, Georgia

Medium or Pathway Sampled (Unit of Measurement)	Type and Total Number of Analyses Performed	Lower Limit of Detection (a) (LLD)	Indicator Locations Mean (b) Range (Fraction)	Location with Highest Annual Mean		Control Locations Mean (b) Range (Fraction)	Number of Reportable Occurrences
				Name Distance & Direction	Mean (b) Range (Fraction)		
Fish (pCi/kg wet)	Gamma Isotopic 8						
	Be-7	655 (e)	NDM		NDM	NDM	0
	Mn-54	130	NDM		NDM	NDM	0
	Fe-59	260	NDM		NDM	NDM	0
	Co-58	130	NDM		NDM	NDM	0
	Co-60	130	NDM		NDM	NDM	0
	Zn-65	260	NDM		NDM	NDM	0
	Cs-134	130	NDM		NDM	NDM	0
	Cs-137	150	360.3 36-1050 (4/4)	No.85 Downriver 4.1 miles	360.3 36-1050 (4/4)	84.2 38-146 (4/4)	0

TABLE 3-1 (SHEET 9-10)  
 RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM SUMMARY FOR 1993  
 Vogtle Electric Generating Plant, Docket Nos. 50-424 and 50-425  
 Burke County, Georgia

Medium or Pathway Sampled (Unit of Measurement)	Type and Total Number of Analyses Performed	Lower Limit of Detection (a) (LLD)	Indicator Locations Mean (b) Range (Fraction)	Location with Highest Annual Mean		Control Locations Mean (b) Range (Fraction)	Number of Reportable Occurrences
				Name Distance & Direction	Mean (b) Range (Fraction)		
Sediment (pCi/kg dry)	Gamma Isotopic 4						
	Be-7	655 (e)	711 500-921 (2/2)	No. 81 Upriver 2.5 miles	902 883-921 (2/2)	902 883-921 (2/2)	0
	Co-60	70 (e)	65.9 43-89 (2/2)	No. 83 Downriver 0.6 miles	65.9 43-89 (2/2)	NDM	0
	Cs-134	150	NDM		NDM	NDM	0
	Cs-137	180	345 277-413 (2/2)	No. 83 Downriver 0.6 miles	345 277-413 (2/2)	115 106-123 (2/2)	0

TABLE 3-1 (SHEET 10 OF 10)

RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM SUMMARY FOR 1993  
Vogtle Electric Generating Plant, Docket Nos. 50-424 and 50-425  
Burke County, Georgia

NOTATIONS

- a. The LLD is defined by Notation 3 of ODCM Table 3.1-3. Except as noted otherwise, the values listed in this column are the detection capabilities required by that table. In practice, the LLDs attained were generally much lower than the values listed. Any attained LLDs greater than the values listed in ODCM Table 3.1-3 are discussed in Section 4.
- b. Mean and range are based upon detectable measurements only. The fraction of all measurements at specified locations which was detectable is placed in parenthesis.
- c. No Detectable Measurement(s).
- d. Not Applicable.
- e. The EL has determined that this value may be routinely attained under normal conditions. No value is provided in ODCM Table 3.1-3. Sample size, background count rate or chemical yield might make the Minimum Detectable Activity (MDA), an "a posteriori" or after the fact result, greater than the required LLD.
- f. Item 3b of ODCM Table 3.1-1 implies that an I-131 analysis is not required to be performed on these samples when the dose calculated from the consumption of water is less than 1 mrem per year.



#### 4.0 DISCUSSION OF RESULTS

An interpretation and evaluation, as appropriate, of the laboratory results for each type sample are included in this section. Relevant comparisons were made between the difference in average values for indicator and control stations and the calculated Minimum Detectable Difference (MDD) between these two groups at the 99 percent Confidence Level (CL). The MDD was determined using the standard Student's t-test. A difference in the average values which was less than the MDD was considered to be statistically indiscernable.

Pertinent results were also compared with past results including preoperation. The results were examined to perceive any trends. To provide perspective, a result might also be compared with its Lower Limit of Detection (LLD) or Reporting Level (RL) which are nominally provided by ODCM Tables 3.1-3 and 3.1-2, respectively. Attempts were made to explain any RLs or other high radiological levels found in the samples. During the year there were no failures in the laboratory analyses of any of the samples in attaining the LLDs required by ODCM Table 3.1-3.

Unless otherwise indicated, any reference made in this section to the results of a previous period are results which have been purged of any obvious extraneous short term impacts. During preoperation, these included the nuclear weapons test in the fall of 1980 (apparently the last in a series of atmospheric tests conducted on mainland China over a 9 year period), abnormal releases from the Savannah River Site (SRS), and the Chernobyl incident in the spring of 1986. After operation commenced, short term impacts included abnormal releases from SRS during 1987 and 1991. Unless otherwise indicated, any references to 1987 will be to the operations portion of 1987.

All results were tested for conformance to Chauvenet's Criterion<sup>1</sup> to flag any values which might differ from the others in its set by a relatively large amount. Identified outliers were investigated to determine reasons for deviation from the norm. If due to an equipment malfunction or other valid physical reason, the anomalous result was deemed non-representative and excluded from the data set. No datum was excluded for failing Chauvenet's Criterion only. Any exclusions are discussed in this section under the appropriate sample type.

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1. G. D. Chase and J. L. Rabinowetz, Principles of Radioisotope Methodology, (Burgess Publishing Company, 1962), pp 87-90.

The annual land use census as required by ODCM 3.1.2 was conducted on April 28. The locations of the nearest permanent residence, milk animal, and garden of greater than 500 square feet producing broad leaf vegetation in each of the 16 meteorological sectors found within a distance of 5 miles are tabulated in Table 4-1. Land within SRS was excluded from the census.

ODCM 3.1.2.1 requires a new controlling receptor for ODCM 2.5.3 if the land use census identifies a location that yields a greater calculated dose or dose commitment. An analysis of the survey's results showed that there was none.

ODCM 3.1.2.1 also requires that whenever the land use survey identifies a location which would yield a thyroid dose (via the same ingestion pathway) 20 percent greater than that from a current indicator station, the new location must become a REMP station (if samples are available). No milk animals or vegetable gardens were found in the land use census.

The results of the annual survey conducted downstream of the plant to determine whether water from the Savannah River is being used for drinking or irrigation purposes are presented in Section 4.5.

TABLE 4-1

## LAND USE SURVEY RESULTS

Distance in Miles to Nearest Location in Each Sector

<u>SECTOR</u>	<u>RESIDENCE</u>	<u>MILK ANIMAL</u>	<u>GARDEN</u>
N	1.2	*	*
NNE	*	*	*
NE	*	*	*
ENE	*	*	*
E	*	*	*
ESE	*	*	*
SE	4.3	*	*
SSE	4.0	*	*
S	4.3	*	*
SSW	4.7	*	*
SW	2.9	*	*
WSW	1.2	*	*
W	2.3	*	*
WNW	1.9	*	*
NW	1.6	*	*
NNW	1.4	*	*

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\* None within 5 miles and outside the SRS.

#### 4.1 Airborne

As indicated by Tables 2-1 and 2-2, airborne particulates and airborne radioiodine were collected at 5 indicator stations (Nos. 3, 7, 10, 12 and 16) which encircle the site and are on the site periphery, at a nearby community station (No. 35), and at a control station (No. 36) which is about 14 miles from the plant. At these locations air was continuously drawn in sequence through a Gelman Type A/E glass fiber filter and a SAI CP-200 charcoal canister in sequence to retain airborne particulates and to adsorb airborne radioiodine, respectively.

The filters and canisters were collected weekly. Each of the air particulate filters was counted for gross beta activity. A gamma isotopic analysis was performed quarterly on a composite of the air particulate filters for each station. Each charcoal canister was analyzed for I-131 by gamma spectroscopy.

There was one failure during 1993 to obtain an acceptable air particulate sample. There were 3 such failures during 1992, 2 in 1991, 3 in 1990 and 6 in 1989. In addition, on three occasions during 1993, a station was not operating for an extended period due to mechanical failures; however, the sample results in each of these cases were deemed to be acceptable since the results for the gross beta analysis were found to conform with Chauvenet's Criterion.

A hole was found in the air particulate filter for the March 16 collection at Station 35. The hole was caused by a wire protruding from the screen in the charcoal cartridge. The gross beta analysis result was deemed unacceptable since it failed Chauvenet's Criterion. The appropriate personnel were instructed to take care in assembling the filters and cartridges to prevent such an occurrence in the future.

For the March 23 collection at Station 7, the power was off 90.2 hours. The ground wire had come in contact with the power line to the motor causing the fuse to blow. A new ground wire was installed well away from the power line to preclude a reoccurrence.

The air pump was not running at Station 12 when the field team arrived on July 27; it had been off for 27 hours. The failure was thought to be due to a faulty outlet as the pump ran when plugged into a second outlet. The pump was not running again when the field team arrived one week later on August 3; it had run for only 23.6 hours. A new pump was installed.

As seen in Table 3-1, the average weekly beta activity during the year for the indicator stations was 0.2 fCi/m<sup>3</sup> less than that for the control station. However, this difference was not discernable, since it was less than the MDD, calculated as 1.9 fCi/m<sup>3</sup>. During the 6 year period from 1987 through 1992, the average weekly activity for the year at the indicator stations randomly varied from 1.0 fCi/m<sup>3</sup> greater than to 0.6 fCi/m<sup>3</sup> less than that for the control station. The average weekly activity for the indicator stations over this entire 6 year period was nearly 0.2 fCi/m<sup>3</sup> greater than that for the control station. The overall average weekly activity for the indicator stations during preoperation was 0.8 fCi/m<sup>3</sup> greater than that for the control station.

The average weekly gross beta activity in units of fCi/m<sup>3</sup> for the indicator, control and community stations during 1993 are compared with those during the previous years of operation, with the entire preoperation period (which began in September 1991 for the air monitoring stations) and with the range of annual averages during preoperation.

<u>Period</u>	<u>Indicator</u>	<u>Control</u>	<u>Community</u>
1993	21.2	21.4	20.3
1992	18.7	19.3	18.0
1991	19.3	19.2	18.6
1990	19.6	19.4	18.8
1989	19.1	18.2	18.8
1988	24.7	23.7	22.8
1987	23.0	23.5	22.3
Precp Overall	22.9	22.1	21.9
Preop Range	18.1-28.1	18.3-26.5	18.3-26.5

The average weekly readings for 1993 are seen to be within the range of values found previously. No trends were recognized in these data.

No positive results for man-made radionuclides were found during 1993 from the gamma isotopic analysis of the quarterly composites of the air particulate filters. During 1987, Cs-137 was found in one indicator composite at a level of 1.7 fCi/m<sup>3</sup>. During preoperation, Cs-137 was found in an eighth of the indicator composites and a seventh of the control composites with average levels of 1.7 and 1.0 fCi/m<sup>3</sup>, respectively; the required LLD is 60 fCi/m<sup>3</sup>. Also, during preoperation, Cs-134 was found in about 8 percent of the indicator composites; the average level was 1.2 fCi/m<sup>3</sup>; the required LLD is 50 fCi/m<sup>3</sup>.

I-131 was not detected in any samples during 1993. There were no positive results during previous years of operation. During preoperation, positive results were obtained only during the Chernobyl incident when levels as high as 182 fCi/m<sup>3</sup> were obtained. The maximum allowed LLD is 70 fCi/m<sup>3</sup>; however, the LLD generally attained is about a fifth of this value. The RL for airborne I-131 is 900 fCi/m<sup>3</sup>.

## 4.2 Direct Radiation

Direct (external) radiation was measured with TLDs. Two Panasonic UD-814 TLD badges were placed at each station. The phosphor for the badges consists of calcium sulfate (with thulium impurity) crystals on a polyimide substrate. Each badge has 3 phosphors with filters on each side of the phosphor to attenuate low energy photons. This filtration compensates for the overresponse of the calcium sulfate in this portion of the energy spectrum. The badges were nominally exposed for periods of a quarter of a year (91 days).

Two TLD stations were established in each of the 16 meteorological sectors about the plant forming two concentric rings. The stations comprising the inner ring (Nos. 1 through 16) are located near the site boundary, while those comprising the outer ring (Nos. 17 through 32) are located at distances of about 5 miles. The 16 stations forming the inner ring are designated as the indicator stations. Each of the 4 control stations (Nos. 36, 37, 47 and 48) is over 10 miles from the plant. Special interest areas consist of a hunting cabin (No. 33), the town of Girard (No. 35), and the employees' recreational area (No. 43).

As shown in Table 3-1, the average quarterly exposure of 12.4 mR acquired at the indicator stations (inner ring) was the same as that acquired at the control stations. If these average values had been rounded to 4 rather than 3 significant figures, the indicator stations would have been shown to be 0.06 mR greater than the control stations. However, this difference was not discernable since it was less than the MDD of 1.26 mR. During the 6 year period from 1987 (when operations began) through 1992, the average quarterly exposure for the year at the indicator stations randomly varied from 0.7 mR greater than to 0.5 mR less than that for the control stations. The average quarterly exposure for the indicator stations over this 6 year period was 0.03 mR less than that for the control stations. The overall average quarterly exposure for the indicator stations during preoperation was 1.2 mR less than that for the control stations.

The quarterly exposures acquired at outer ring stations ranged from 9.6 to 19.8 mR, with an average of 12.1 mR for the year, which was 0.3 mR less than that found for the inner ring. There was no discernable difference between the averages for the inner and the outer ring, since the difference was less than the MDD of 0.8 mR. For the 6 year period beginning in 1987, the average quarterly exposure for the year at the inner ring stations varied from 0.2 to 0.9 mR greater than that at the outer ring stations. The average quarterly exposure for the inner ring stations over this 6 year period was nearly 0.6 mR greater than that for the outer ring stations. The overall average quarterly exposure for the inner ring stations during preoperation was also 0.6 mR greater than that for the outer ring stations.

Listed below for the indicator, control and outer ring stations, are the average quarterly exposures in units of mR obtained during each year of operation, the entire period of preoperation (which began in August 1981 for TLD stations), and the range of annual averages obtained during the calendar years of preoperation.

<u>Period</u>	<u>Indicator</u>	<u>Control</u>	<u>Outer Ring</u>
1993	12.4	12.4	12.1
1992	12.3	12.5	12.1
1991	16.9	17.1	16.7
1990	16.9	16.6	16.3
1989	17.9	18.4	17.2
1988	16.8	16.1	16.0
1987	17.6	17.9	16.7
Preop Overall	15.3	16.5	14.7
Preop Range	15.1-16.9	14.1-18.2	12.5-16.2

The average quarterly exposures in units of mR at the special interest areas for the same periods as given above are listed below.

<u>Period</u>	<u>Station 33</u>	<u>Station 35</u>	<u>Station 43</u>
1993	12.9	13.3	12.1
1992	12.8	13.5	12.0
1991	17.3	19.6	17.0
1990	16.8	18.9	16.2
1989	21.2	18.7	17.4
1988	19.7	18.1	14.8
1987	21.3	18.5	15.2
Preop Overall	16.6	15.1	15.3
Preop Range	13.6-19.9	12.6-17.6	13.9-25.0



The average exposure for all of the TLDs during 1993 was 0.25 percent greater than that found in 1992. The average measured exposure in 1992 was about 27 percent less than that acquired in 1991; this anticipated reduction was due to the switch from Teledyne to Panasonic TLDs. The differences in exposures between various station groups continue to be on the same order.

There were 3 failures to obtain a reading for a station during the year. These occurred for the first quarter at Stations 5, 22 and 33. In addition for the third quarter: acceptable readings were not obtained for Badges 9B and 26B due to their high standard deviations; and Badge 25B was too wet to read. The readings for these stations for the third quarter were those obtained by the A badges.

During the first quarter, the river rose to an unusually high level. The badges at Station 5 were swept away and those at Station 22 were immersed in the high water and became too wet to read - the plastic bags in which Badges 22A and 22B were contained had ruptured. Both of these stations are located on the river bank. The replacement badges were placed at a higher elevation to preclude a repeat of these happenings.

The badges for Station 33 for the first quarter were lost in handling and were not recovered.

Badge 25B for the third quarter was too wet to read; there were holes in the plastic bag, caused by insects, which provided an entry for rain water.

The standard deviations for Badges 9B and 26B for the third quarter were 2.53 and 11.12, respectively. These are greater than the self imposed limit of 1.4 which was calculated using a method<sup>2</sup> developed by the American Society for Testing and Materials (ASTM). The calculation was based upon the standard deviations obtained with the UD-814 badges during 1992. This limit serves as a flag to evoke an investigation. To be conservative, readings with a greater standard deviation are deleted since the high standard deviation may indicate a defective TLD.

2. ASTM Special Technical Publication 15D, ASTM Manual on Presentation of Data and Control Chart Analysis, Fourth Revision, Philadelphia, PA, October 1976.

#### 4.3 Milk

As indicated by Tables 2-1 and 2-2, milk was collected biweekly from two control stations, the W. C. Dixon Dairy (Station 98) and the Boyceland Dairy (Station 99). Gamma isotopic and I-131 analyses were performed on each sample.

Milk has not been available from an indicator station (a location within 5 miles for the plant) since April 1986 when the cow from which milk was being obtained went dry and was subsequently removed from the area. The availability of milk within 5 miles of the plant was meager throughout preoperation and an adequate sample is yet to be obtained during operation. As shown by Table 4-1, no milk animals were found in the annual land use census which was conducted on April 28. A milk animal is a cow or goat producing milk for human consumption.

No man-made radionuclides were found from the gamma isotopic analysis of the milk samples during the year. During preoperation and each year of operation through 1991, Cs-137 was found in 2 to 6 percent of the samples at levels ranging from 5 to 27 pCi/l. The LLD and RL for Cs-137 in milk are 18 and 70 pCi/l, respectively. During preoperation, Cs-134 was detected in one sample and in the first year of operation, Zn-65 was detected in one sample.

I-131 was not detected in any of the milk samples during 1993. In 1990, I-131 was reported in two samples but its presence was questionable due to large counting uncertainties. I-131 was not detected during other years of operation. During preoperation, positive I-131 results were found only during the Chernobyl incident. The levels ranged from 0.53 to 5.07 pCi/l. The LLD and RL are 1 and 3 pCi/l, respectively.

#### 4.4 Vegetation

The ODCM calls for the gamma isotopic analysis of grass or leafy vegetation collected monthly from two indicator stations which are located onsite near the site boundary in different meteorological sectors (Stations 7 and 15) and one control station located about 17 miles from the plant (Station 37). Grass is collected at each of these stations.

The results presented in Table 3-1 show that Cs-137 was the only man-made radionuclide detected. The average value of 46.4 pCi/kg wet found at the indicator stations was 12.3 pCi/kg wet greater than that found at the control station. A standard MDD calculation to compare results between the indicator and control stations was not possible as only one positive result was found at the control station. A modified t-test which compared the single result for the control station with the average result for the indicator stations was performed; it showed that there was no statistical difference at the 99 percent CL between the results for the two station groups.

The average level of Cs-137 found in vegetation samples in units of pCi/kg wet along with the fraction of detectable measurements at the indicator and control stations is shown below for each year of operation and the period of preoperation.

<u>Period</u>	<u>Indicator Stations</u>		<u>Control Station</u>	
	<u>Average</u>	<u>Fraction</u>	<u>Average</u>	<u>Fraction</u>
1993	46.4	0.333	34.1	0.083
1992	38.1	0.250	144.0	0.083
1991	35.3	0.208	62.4	0.083
1990	30.0	0.083	102.0	0.167
1989	9.7	0.042	0.0	0.000
1988	38.7	0.280	0.0	0.000
1987	24.4	0.318	61.5	0.250
Preop	54.6	0.573	43.7	0.193

No trend was recognized in these data. The LLD and RL for Cs-137 in vegetation samples are 80 and 2000 pCi/kg wet, respectively.

To enhance the statistical base for the indicator stations, consideration is being given to adding an additional station. Beginning in August, grass samples were collected on a trial basis at a new onsite location at 1.2 miles in the SW sector; this location is adjacent to Gate 1. Positive results ranging from 27.4 to 61.2 pCi/kg wet with an average of 45.4 pCi/kg wet were found in 60 percent in these samples. These results were about the same as those found in samples from the other indicator locations.

Except for a short period following the Chernobyl incident, Cs-137 has been the only man-made radionuclide detected in vegetation samples by gamma isotopic analysis during both the preoperation and operation periods. As a consequence of the Chernobyl incident for a period of several weeks: I-131 was found in nearly all the samples collected (some at elevated levels), Cs-137 was also found in nearly all the samples collected, and Co-60 was found in one of the samples.

#### 4.5 River Water

Surface water was composited from the Savannah River at three locations using ISCO automatic samplers. Small quantities were collected at intervals not exceeding a few hours. River water samples collected by these machines were picked up monthly; quarterly composites were made from the monthly collections.

The collection points consist of a control station (No. 82) which is located about 0.4 miles upriver of the plant intake structure, an indicator station (No. 83) which is located about 0.4 miles downriver of the plant discharge structure, and a special station (No. 84) which is located approximately 1.3 miles downriver of the plant discharge structure. The increase in the radiological levels found in samples collected at the indicator station over those collected at the control station might be attributed to the plant radiological releases. The radiological levels found at the special station perhaps represent those for the river as a whole; the levels here might be attributed to plant releases as well as to other sources.

A gamma isotopic analysis was conducted on each monthly collection. As in all previous years of operation, there were no radionuclides of interest detected in the river water samples collected during 1993.

A tritium analysis was performed on each quarterly composite. As indicated in Table 3-1, the average level of 712 pCi/l found at the indicator station was 474 pCi/l greater than that found at the control station; however, this difference was not discernable since it was less than the calculated MDD of 1526 pCi/l. There was a discernable difference between these two stations in 1988 and 1989 (as indicated from the data presented below). At the special station, the results ranged from 245 to 1090 pCi/l with an average of 616 pCi/l. The required LLD is 3000 pCi/l and the RL is 10 times greater.

Listed below for each year of operation are the average tritium levels found at the special, indicator and control stations, along with the increase in the average level at the indicator station over that for the control station, the MDD between these two stations, and the total liquid releases of tritium from the plant. All of these values are in units of pCi/l except for the releases which are in units of Ci.

<u>Year</u>	<u>Special</u>	<u>Indicator</u>	<u>Control</u>	<u>Increase</u>	<u>MDD</u>	<u>Releases</u>
1993	616	712	238	474	1526	761
1992	929	1064	371	693	714	1481
1991	1298	1299	828	471	626	1094
1990	1081	1142	392	750	766	1172
1989	1268	1293	538	755	518	918
1988	1430	843	427	416	271	390
1987	1411	680	524	156	416	321

These data show a generally upward trend for plant releases through 1992, with nearly a 50 percent decrease in 1993 from the 1992 level. Although the releases are sufficient to account for increases in the levels at the indicator station over those at the control station, there is not a good correlation between the releases and the increases. The overall tritium level in the river has diminished in the past two years as indicated by the levels at each of the stations. In the early years of operation, the tritium level at the special station was much greater than that at the indicator station; whereas in recent years, it has become less. This indicates the contribution from other tritium sources has diminished and plant releases have become relatively more significant. Nevertheless, there has been a discernable difference between the indicator and control stations only during 1988 and 1989.

The annual whole body dose that the maximum individual (a child) would receive from drinking water with an average tritium concentration of 474 pCi/l (the assumed plant contribution) was conservatively calculated to be 0.049 mrem or 1.6 percent of the ODCM limit of 3 mrem.

On September 14, the annual survey of the Savannah River was conducted downstream of the plant for approximately 106 river miles to identify any users of river water for purposes of drinking or irrigation. As in all previous surveys, no intakes for drinking water or irrigation were observed over this stretch of the river. This was corroborated by information obtained from the Georgia Department of Natural Resources and the South Carolina Department of Health and Environmental Control; it was learned that no new surface or drinking water withdrawal permits had been issued in 1993 for the Savannah River downstream of the plant. The two water treatment plants used as indicator stations (Nos. 87 and 88) for drinking water samples are located further downstream.



#### 4.6 Drinking Water

Samples were collected at a control station (No. 80), the Augusta Water Treatment Plant in Augusta, Georgia, which is located about 56 miles upriver, and at two indicator stations (Nos. 87 and 88), the Beaufort-Jasper County Water Treatment Plant near Beaufort, South Carolina and the Cherokee Hill Water Treatment Plant near Port Wentworth, Georgia, which are respectively located approximately 112 and 122 miles downriver. These upriver and downriver distances in river miles are the distances from the plant to the point on the river where water is diverted to the intake for each of these water treatment plants.

At each of the water treatment plants, monthly collections were made of river water which was composited near the intake of the water treatment plant (raw drinking water) and of grab samples of finished drinking water; quarterly composites were made from the monthly collections. Gross beta and gamma isotopic analyses were conducted on each of the samples collected monthly. Tritium analysis was conducted on the quarterly composites. Although an I-131 analysis was not required to be conducted on these samples as the dose calculated from the consumption of water is less than 1 mrem per year (see Item 3b of ODCM Table 3.1-1), an I-131 analysis was conducted on each of the grab samples of the monthly collections of finished water since a drinking water pathway exists.

As indicated in Table 3-1, the average gross beta activity for raw drinking water of 3.17 pCi/l found at the indicator stations was 0.34 pCi/l greater than that found at the control station. However, this difference was not discernable since it was less than the calculated MDD of 1.09 pCi/l. For finished drinking water, the average gross beta activity of 2.23 pCi/l found at the indicator stations was 0.07 pCi/l less than that found at the control station. This difference was not discernable since it was less than the MDD of 0.91 pCi/l.

Listed below for each year of operation are the average gross beta levels in units of pCi/l found in the monthly collections for raw and finished drinking water at the indicator and control stations, along with the increases in the average levels for the indicator stations over those for the control station.

<u>Period</u>	<u>Indicator</u>	<u>Control</u>	<u>Increase</u>
RAW			
1993	3.17	2.83	0.34
1992	2.73	2.70	0.03
1991	2.83	3.08	-0.25
1990	2.53	2.55	-0.02
1989	2.93	3.05	-0.12
1988	2.67	3.04	-0.37
1987	2.20	5.50	-3.30
FINISHED			
1993	2.23	2.30	-0.07
1992	2.09	1.67	0.42
1991	1.90	1.53	0.37
1990	2.08	1.92	0.16
1989	2.36	2.38	-0.02
1988	2.28	2.35	-0.07
1987	2.10	1.80	0.30

No trend was recognized from these data. The overall gross beta level for all years of operation was 38 percent greater for the raw drinking water; this might be expected since the finished water has been filtered. There has not been a discernable difference between the average values at the indicator and control stations during any of the years of operation. The LLD for gross beta in water is 4 pCi/l.

As indicated in Table 3-1, there were no positive results for the radionuclides of interest from the gamma isotopic analysis of the monthly collections. Only one positive result has been found since operation began; Be-7 was found at a level of 68.2 pCi/l in the sample collected for September 1987 at Station 87. The LLD assigned for Be-7 in water is 124 pCi/l.

Listed below for each year of operation are the average tritium levels found in the quarterly composites of raw and finished drinking water in units of pCi/l collected at the indicator and control stations, along with the increases in the average levels at the indicator stations over those at the control station, and the calculated MDDs between these two stations. The average levels at the indicator stations for 1991 and 1992 have been purged of the impact of the inadvertent release at SRS of 7500 Ci of tritium to the Savannah River about 10 miles downriver of VEGP between December 22 and 25, 1991.



<u>Period</u>	<u>Indicator</u>	<u>Control</u>	<u>Increase</u>	<u>MDD</u>
RAW				
1993	955	NDM	NA	NA
1992	1131	179	952	353
1991	1471	165	1306	834
1990	1320	266	1054	572
1989	2508	259	2249	1000
1988	2630	240	2390	580
1987	2229	316	1913	793
FINISHED				
1993	993	NDM	NA	NA
1992	1162	211	951	427
1991	1240	225	1015	647
1990	1299	404	895	1131
1989	2236	259	1977	627
1988	2900	270	2630	830
1987	2406	305	2101	1007

The above tabulations show a steady decline of the tritium levels in the drinking water samples, both raw and finished, over a period of several years. No detectable measurements (NDM) were made in the samples collected at the control station during 1993; values for the average increased levels at the indicator stations and the calculated MDDs between the indicator and the control stations were thus not applicable (NA). In previous years of operation, there were detectable differences between the average levels at the indicator stations and those at the control station (the increased average level between the two exceeded the MDD) except for the finished drinking water during 1990.

As indicated in Table 3-1, there were no positive results from the I-131 analysis of the finished drinking water samples. Each result was below its Minimum Detectable Activity (MDA) which ranged from 0.18 to 0.61 pCi/l. Similar results were obtained in previous years of operation. The ODCM requires a LLD and a RL of 1 and 2 pCi/l, respectively.

#### 4.7 Fish

The ODCM calls for the collection of at least one sample of any anadromous species of fish in the vicinity of the plant discharge during the spring spawning season. The ODCM also calls for semiannual collections of any commercially or recreationally important species in the vicinity of the plant discharge area and in areas not influenced by plant discharges. Further, the ODCM calls for a gamma isotopic analysis on the edible portions of each sample collected.

A stretch of the river of about 5 miles is generally needed to obtain adequate fish samples. For the semiannual collections, the control station (No. 81) extends from approximately 2 to 7 miles upriver of the plant intake structure and the indicator station (No. 85) extends from about 1.4 to 7 miles downriver of the plant discharge structure. For anadromous species, all collection points can be considered as indicator stations.

On March 22, American shad, an anadromous species, was collected. As in all but two previous years of operation, no positive results for the radionuclides of interest were detected from the gamma isotopic analysis. In 1987 as well as in 1991, Cs-137 was found in a single sample of American shad at miniscule levels of 10 and 12 pCi/kg wet, respectively.

The dates and compositions of the semiannual catches at the indicator and control stations were as shown below.

<u>Date</u>	<u>Indicator</u>	<u>Control</u>
May 24	Channel Catfish Largemouth Bass	Largemouth Bass Redear Sunfish
October 25	Channel Catfish Largemouth Bass	Bullhead Catfish Largemouth Bass

As indicated in Table 3-1, Cs-137 was the only radionuclide of interest found in the semiannual collections. It has been found in all but one of 65 samples collected during operation. In Table 3-1, the average level at the indicator station is seen to be 276 pCi/kg wet greater than that at the control station. This difference was not discernable, however, since it was less than the calculated MDD of 1091 pCi/kg wet.

Listed below for each year of operation are the average levels of Cs-137 in units of pCi/kg wet found in fish samples at the indicator and control stations.

<u>Year</u>	<u>Indicator</u>	<u>Control</u>
1993	360	84
1992	178	80
1991	105	211
1990	103	249
1989	117	125
1988	66	116
1987	337	119

No trend was recognized in these data. The LLD and RL for Cs-137 in fish as specified by the ODCM are 150 and 2000 pCi/kg wet, respectively.

The only other radionuclide of interest found in fish samples during past years of operation was I-131. In 1990, it was found in one sample at both the indicator and control stations at levels of 13 and 12 pCi/kg wet, respectively; in 1989, it was found in one sample at the indicator station at a level of 18 pCi/kg wet. The LLD assigned for I-131 in fish is 53 pCi/kg wet.

#### 4.8 Sediment

Sediment was collected along the shoreline of the Savannah River on May 4 and October 5 at Stations 81 and 83. Station 81 is a control station located about 2.5 miles upriver of the plant intake structure while Station 83 is an indicator station located about 0.6 miles downriver of the plant discharge structure. A gamma isotopic analysis was performed on each sample.

Listed below for each year of operation are the average levels in units of pCi/kg dry for the radionuclides of interest found in the regular samples collected at the indicator and/or control stations along with the fractions of detectable measurements and the LLDs. Each of these radionuclides is found in the plant's liquid releases.

<u>Year</u>	<u>Indicator</u>	<u>Fraction</u>	<u>Control</u>	<u>Fraction</u>
Be-7, LLD=655				
1993	711	1.0	902	1.0
1992	2038	1.0	380	1.0
1991	826	1.0	427	1.0
1990	465	1.0	545	1.0
1989	1300	1.0	415	1.0
1988	970	1.0	810	1.0
1987	987	1.0	543	1.0
Mn-54, LLD=42				
1989	18	0.5		
1988	22	0.5		
Co-58, LLD=43				
1992	124	0.5		
1990	140	0.5		
1989	135	1.0		
1988	190	1.0		
Co-60, LLD=70				
1993	66	1.0		
1992	60	1.0		
1991	113	0.5		
1990	46	0.5		
1989	46	1.0		
1988	62	0.5		

<u>Year</u>	<u>Indicator</u>	<u>Fraction</u>	<u>Control</u>	<u>Fraction</u>
		I-131, LLD=53		
1992	194	0.5	51	0.5
		Cs-137, LLD=180		
1993	345	1.0	115	1.0
1992	259	1.0	111	1.0
1991	246	1.0	100	1.0
1990	155	1.0	140	1.0
1989	230	1.0	125	1.0
1988	175	1.0	175	1.0
1987	209	1.0	111	1.0

No trend was recognized in the above data. The levels for 1993 are seen to be on the same order as those found previously. As in all previous years of operation, positive results for Be-7 and Cs-137 were found in each sample. The only other radionuclide of interest found during 1993 was the activation product, Co-60.

For Be-7, the average level at the indicator station is seen to be 191 pCi/kg dry less than that at the control station; however, this difference is not discernable as it is less than the calculated MDD of 1472 pCi/kg dry. For Cs-137, the average level at the indicator station is 230 pCi/kg dry greater than that at the control station; this difference is less than the MDD of 477 pCi/kg dry, and therefore it too is not discernable. There continues to be no discernable difference between the indicator and control stations for both Be-7 and Cs-137 or for any other radionuclide of interest since operation began.

The continuing presence of Co-60 (although at low levels) and the presence in recent years of Mn-54 and Co-58 (also activation products) at the indicator station (only) is indicative of plant releases. Employing the methodology and parameters of NRC Regulatory Guide 1.109, Revision 1, October 1977, the annual whole body dose to an individual by direct radiation from sediment with Co-60 concentrations as found at the indicator station was estimated to be approximately 2.6 microrem or about 0.09 percent of the ODCM limit (3 mrem). This extremely low dose, although calculable, poses no measurable environmental or public health impact.

## 5.0 INTERLABORATORY COMPARISON PROGRAM

TS 3.12.3 requires that analyses be performed on radioactive materials supplied as part of an Interlaboratory Comparison Program approved by the NRC. The Intercomparison Studies (Crosscheck) Program conducted by the Environmental Protection Agency (EPA) at their Environmental Monitoring Systems Laboratory in Las Vegas, Nevada, provides such a program. Reported herein, as required by ODCM 3.2 is a summary of the results of the EL's participation in the EPA Crosscheck Program.

The Crosscheck Program was designed for laboratories involved with REMPs and includes environmental media and a variety of radionuclides with activities which might be as low as environmental levels. Participation in the program ensures that independent checks on the precision and accuracy of the measurements of radioactive materials in environmental sample matrices are performed.

Simulated environmental samples are distributed regularly to the participants who analyze the samples and return the results to the EPA for statistical analysis and comparison with known values and with results obtained from other participating laboratories. The EPA then provides each participant with documentation of its performance; this can be helpful in identifying an instrument or procedure problems.

The EL analyzed the EPA supplied crosscheck samples consistent with the requirements of Table 2-1. Analyses were performed in a normal manner. Each sample was analyzed in triplicate as required by the program. Results obtained for the gross beta and gamma isotopic analyses of air filters, the gamma isotopic analysis of milk samples, and the gross beta, tritium, gamma isotopic and I-131 analyses of water samples are summarized in Table 5-1.

Delineated in Table 5-1 for each of the environmental media are the type analyses performed, EPA's collection dates, the known values and expected precisions (one standard deviation) provided by the EPA, the average results obtained and reported by the EL along with the standard deviations of these results, and the normalized deviations and the normalized ranges from the known results. The normalized deviations and normalized ranges were also provided by the EPA.

TABLE 5-1 (SHEET 1 OF 2)  
CROSSCHECK PROGRAM RESULTS

<u>Radionuclide or Analysis</u>	<u>Date Collected</u>	<u>Known Value</u>	<u>Expected Precision</u>	<u>Reported Average</u>	<u>Standard Deviation</u>	<u>Normalized Deviation</u>	<u>Normalized Range</u>
Air Filters (pCi/filter)							
Gross Beta	08/27/93	47.0	5.0	48.33	0.58	0.46	0.12
Cs-137	08/27/93	9.0	5.0	10.67	0.58	0.58	0.12
Milk (pCi/l)							
I-131	09/24/93	120.0	12.0	121.33	8.33	0.19	0.79
Cs-137	09/24/93	49.0	5.0	50.67	4.73	0.58	1.12
Water (pCi/l)							
Gross Beta	01/29/93	44.0	5.0	39.33	2.52	-1.62	0.59
	04/20/93	177.0	27.0	171.33	2.31	-0.36	0.09
	07/23/93	43.0	6.9	44.00	1.73	0.25	0.26
	10/19/93	58.0	5.0	54.67	3.06	-1.15	0.71
	10/29/93	15.0	5.0	25.67	3.06	3.70	0.71
H-3	06/04/93	9844.0	984.0	9670.00	131.13	-0.31	0.16
	11/05/93	7398.0	740.0	7776.67	170.39	0.89	0.27
Co-60	04/20/93	39.0	5.0	39.00	3.61	0.00	0.83
	06/11/93	15.0	5.0	17.67	0.59	0.92	0.59
	10/19/93	10.0	5.0	14.33	1.15	1.50	0.24
	11/12/93	30.0	5.0	28.67	0.58	-0.46	0.12

TABLE 5-1 (SHEET 2 OF 2)

## CROSSCHECK PROGRAM RESULTS

<u>Radionuclide or Analysis</u>	<u>Date Collected</u>	<u>Known Value</u>	<u>Expected Precision</u>	<u>Reported Average</u>	<u>Standard Deviation</u>	<u>Normalized Deviation</u>	<u>Normalized Range</u>
Zn-65	06/11/93	103.0	10.0	101.33	5.03	-0.29	0.59
	11/12/93	150.0	15.0	141.33	2.52	-1.00	0.20
Ru-106	06/11/93	119.0	12.0	132.00	7.94	1.88	0.74
	11/12/93	201.0	20.0	183.00	15.62	-1.56	0.20
I-131	02/05/93	100.0	10.0	107.67	4.04	1.33	0.41
	10/08/93	117.0	12.0	166.00	6.08	7.07	0.54
Cs-134	04/20/93	27.0	5.0	24.33	1.15	-0.92	0.24
	06/11/93	5.0	5.0	8.00	1.73	1.04	0.35
	10/19/93	12.0	5.0	11.67	1.15	-0.12	0.24
	11/12/93	59.0	5.0	53.67	1.15	-1.84	0.24
Cs-137	04/20/93	32.0	5.0	33.00	1.00	0.35	0.24
	06/11/93	5.0	5.0	8.67	3.06	1.27	0.71
	10/19/93	12.0	5.0	13.33	0.58	0.46	0.12
	11/12/93	40.0	5.0	38.67	2.52	-0.46	0.59
Ba-133	06/11/93	99.0	10.0	98.33	2.52	-0.12	0.30
	11/12/93	79.0	8.0	75.33	1.53	-0.79	0.22



The normalized deviation from the known value provides a measure of the central tendency of the data (accuracy). The normalized range is a measure of the dispersion of the data (precision). An absolute value of 3 standard deviations for the normalized deviation and for the normalized range was established by the EPA as the control limit. An absolute value of 2 standard deviations was established as the warning limit. The EL considers any value greater than the control limit as unacceptable. Investigations were undertaken whenever any value exceeded the warning limit or whenever a plot of the values indicated a trend.

The following may be noted from Table 5-1 in regard to the normalized deviation:

1. The control limit was exceeded for the gross beta analysis of water on October 29 and the I-131 analysis of water on October 8;
2. A negative bias was found for the gross beta analysis of water; and
3. An upward trend was indicated from plots of Co-60, Ru-106 and Ba-133 found from the gamma isotopic analysis of water.

The water sample which was found to have a normalized deviation for the gross beta analysis in excess of the control limit was reanalyzed. The new value was 0.28.

The collection date for the water sample which was found to have a normalized deviation for the I-131 analysis in excess of the control limit was incorrectly recorded as October 4 while the correct collection date was October 8. The activities reported to the EPA were based upon the incorrect collection date. The activities were recalculated with the correct collection date; the results in units of pCi/l then became:

Average Activity	116.07
One Standard Deviation	4.05
Normalized Deviation	-0.14
Normalized Range	0.39

The negative trend with the normalized deviation values from the gross beta analysis of water samples has been extensively investigated. The Cs-137 self absorption curve standards were made using a new method. In 1993, the use of this curve corrected the negative bias when the water sample contained a single isotope. However, the negative trend persisted with blind performance samples which contain a variety of isotopes. Each of the isotopes in the blind samples have different beta energies and therefore different counting efficiencies on the gas proportional counter. The feasibility of a gross beta self absorption curve made using a mixture of beta emitting isotopes will be investigated in 1994.

The higher activity trend for Co-60, Ru-106 and Ba-133 found from the gamma isotopic analysis of water began after the relocation of the germanium detectors (used in the detection of these radionuclides) in the latter part of 1990. In 1991, new computer software was purchased to convert from a DEC based system to a PC based system. However, the PC based software did not calculate absolute activities below the detection limits; these activities would have been used to correct small biases due to background contributions. A program was developed to make these calculations. Usage of the new peak background correction values began on January 22, 1994. The future EPA and QC samples will be monitored to evaluate the peak background correction values.

## 6.0 CONCLUSIONS

This report confirms the licensee's conformance with TS 6.7.4.g during 1993. It shows that all data were carefully examined. A summary and discussion of the results of the laboratory analyses for each type sample collected were presented.

The increase in the level of tritium in river water and the presence of Co-60 in shoreline sediment at a short distance downriver from the discharge structure may possibly be related to plant releases. The consequent doses which were shown to be small fractions of the TS limits pose no measurable radiological impact to the environment or the public.

III

GEORGIA POWER COMPANY

VOGTLE ELECTRIC GENERATING PLANT - UNITS 1 AND 2

NRC DOCKET NOS. 50-424 AND 50-425

FACILITY OPERATING LICENSE NOS. NPF-68 AND NPF-81

ANNUAL ENVIRONMENTAL OPERATING REPORT  
(NONRADIOLOGICAL) FOR 1993

**VOGTLE ELECTRIC GENERATING PLANT - UNIT 1 AND UNIT 2  
ANNUAL ENVIRONMENTAL OPERATING REPORT (NONRADIOLOGICAL)  
1993**

**SPECIFICATION**

In accordance with Section 5.4.1 of the Vogtle Electric Generating Plant (VEGP) Environmental Protection Plan (Nonradiological), Appendix B to Facility Operating License Nos. NPF-68 and NPF-81, this report is submitted describing implementation of the Environmental Protection Plan for the calendar year 1993.

**REPORTING REQUIREMENTS**

**A. Summaries and Analyses of Results of the Environmental Monitoring Activities for the Reporting Period**

1. Aquatic Monitoring - Liquid effluent monitoring was performed in accordance with National Pollutant Discharge Elimination System (NPDES) Permit GA 0226786; there was no additional requirement for aquatic monitoring during 1993. Two minor NPDES Permit noncompliance events were reported to the State of Georgia during 1993.
2. Terrestrial Monitoring - Terrestrial monitoring is not required.
3. Maintenance of Transmission Line Corridors

- a. Corridor re-clearing was conducted on the VEGP-SCE&G portion and the VEGP-Wadley portion of the VEGP-Scherer 500 KV line in May 1993. Maintenance work was performed with rotary mowers equipped with low ground pressure tires. In cultural resource areas, clearing was conducted by hand utilizing chain saws and brush axes.

There was no herbicide usage associated with any corridor maintenance during 1993.

There were no other transmission corridor maintenance activities conducted on VEGP-related transmission lines during 1993.

- b. There were no clearing or maintenance activities conducted within the Ebenezer Creek or Francis Plantation areas during 1993.
    - c. Routine maintenance activities within the designated cultural properties along transmission line corridors were conducted in accordance with the Final Cultural Resources Management Plan.
4. Noise Monitoring - There were no complaints received by Georgia Power Company during 1993 regarding noise along the VEGP-related high voltage transmission lines.

**B. Comparison of the 1993 Monitoring Activities with Preoperational Studies, Operational Controls, and Previous Monitoring Reports**

These programs were not required because no nonradiological environmental monitoring programs were conducted during the reporting period beyond those performed in accordance with NPDES Permit No. GA 0026786 referenced in Section A above.

**C. An Assessment of Observed Impacts of Plant Operation on the Environment**

There were no significant adverse environmental impacts associated with plant operation in 1993.

**D. Environmental Protection Plan (EPP) Noncompliances and Corrective Actions**

There were no EPP noncompliances during 1993.

**E. Changes in Station Design or Operation, Tests, or Experiments Made in Accordance with EPP Subsection 3.1 which Involved a Potentially Significant Unreviewed Environmental Question**

In correspondence dated February 28, 1992, supplemented by additional correspondence dated August 28, 1992 and February 12, 18, 23, and 25, 1993, Vogtle Electric Generating Plant submitted an application for license amendments to increase rated core power level from 3411 megawatts thermal (MWT) to 3565 MWT. An environmental assessment provided relative to the application concluded that the conclusions of the Final Environmental Statement (FES) remain valid for operation at the uprated condition. The plant operating parameters impacted by the proposed uprate remain within the bounding conditions on which the conclusions of the FES are based. The above correspondence provided the bases for the determination that the power level increase does not involve an unreviewed environmental question or constitute a decrease in the effectiveness of the Vogtle Electric Generating Plant EPP.

By letter dated March 9, 1993, the NRC provided an "Environmental Assessment and Finding of No Significant Impact" relative to the referenced application. Copies of relevant information relating to the power uprate application are provided as Attachment 1.

**F. Nonroutine Reports Submitted in Accordance with EPP Subsection 5.4.2**

There were no nonroutine reports submitted in 1993.

ATTACHMENT 1

TO THE VOGTLE ELECTRIC GENERATING PLANT  
ANNUAL ENVIRONMENTAL OPERATING REPORT FOR 1993  
(NONRADIOLOGICAL)

ATTACHMENT 1



UNITED STATES  
NUCLEAR REGULATORY COMMISSION  
WASHINGTON, D.C. 20585

March 9, 1993

Docket Nos. 50-424  
and 50-425

Mr. W. G. Hairston, III  
Senior Vice President -  
Nuclear Operations  
Georgia Power Company  
P. O. Box 1295  
Birmingham, Alabama 35201

Dear Mr. Hairston:

SUBJECT: ENVIRONMENTAL ASSESSMENT AND FINDING OF NO SIGNIFICANT IMPACT  
UPGRADE IN RATED CORE POWER - VOGTLE ELECTRIC GENERATING PLANT,  
UNITS 1 AND 2 (TAC NOS. M82724 AND M82725)

Enclosed is a copy of an "Environmental Assessment and Finding of No Significant Impact" for your information. This environmental assessment relates to your application dated February 28, 1992, as supplemented June 26 and August 28, 1992, and February 12, 18, 23, and 25, 1993, regarding your application for license amendments to increase rated core power level from 3411 megawatts thermal (MWt) to 3565 MWt.

This environmental assessment is being forwarded to the Office of the Federal Register for publication.

Sincerely,

A handwritten signature in dark ink, appearing to read "Darl Hood", is written over the typed name.

Darl S. Hood, Project Manager  
Project Directorate II-3  
Division of Reactor Projects - I/II  
Office of Nuclear Reactor Regulation

Enclosure:  
Environmental Assessment

cc w/enclosure:  
See next page



ATTACHMENT 1 (CONTINUED)

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Georgia Power Company

Vogtle Electric Generating Plant

cc:

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UNITED STATES NUCLEAR REGULATORY COMMISSIONGEORGIA POWER COMPANY, ET AL.DOCKET NOS. 50-424 AND 50-425NOTICE OF ISSUANCE OF ENVIRONMENTAL ASSESSMENT AND  
FINDING OF NO SIGNIFICANT IMPACT

The U.S. Nuclear Regulatory Commission (the Commission) is considering issuance of amendments to Facility Operating License Nos. NPF-68 and NPF-81 issued to Georgia Power Company (the licensee), for operation of the Vogtle Electric Generating Plant, Units 1 and 2, (Vogtle or the facility) located in Burke County, Georgia.

ENVIRONMENTAL ASSESSMENTIdentification of Proposed Action:

This Environmental Assessment is written in connection with the proposed core uprate for Vogtle in response to the licensee's application for license amendments dated February 28, 1992, as supplemented June 26 and August 28, 1992, and February 12, 18, 23, and 25, 1993. The proposed action would increase the rated core power level for each of the two Vogtle units from the current level of 3411 Megawatts-thermal (MWt) to 3565 MWt, and upgrade the corresponding Nuclear Steam Supply System (NSSS) power from the current level of 3425 MWt to 3579 MWt. This uprate would represent an increase of approximately 4.5 percent over the current rated core power and NSSS thermal power.

The proposed action involves NRC issuance of license amendments to uprate the authorized power level by changing the operating licenses, including Appendix A of licenses (Technical Specifications). No changes are needed to Appendix B of the licenses (Environmental Protection Plan - Non-radiological).

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The Need for the Proposed Action:

The proposed action would increase the electrical output of each Vogtle unit by 50 Megawatts and, thus, would provide additional electric power to service commercial and domestic areas of The Southern Company grid (i.e., the Southern Electric System or SES). In its letter of February 18, 1993, the licensee explained that this additional power is needed to meet current and projected loads. Specifically, the licensee stated that:

The Southern Electric System (SES) experienced substantial load growth during the 1980s. By the late 1980s, the system recognized that new sources would be needed in the early- and mid-1990s.

Three of the five load-serving companies in the system filed for certification of new generating units in 1991 and 1992. The Alabama Public Service Commission certified the need for 720 MW of new capacity at the Greene County site for completion in 1995 and 1996. The Georgia Public Service Commission certified the need for 160 MW in 1994 for Savannah Electric and 480 MW of new capacity in 1994 and 1995. Georgia Power Company filed in early 1993 for an additional 160 MW of capacity in 1995 and released a request for proposals for up to 800 MW in 1996 and up to 800 MW in 1997 of independent power.

All of these certifications and requests for proposals assumed the VEGP [Vogtle Electric Generating Plant] uprates will be successful, or more capacity would have been needed.

Environmental Impacts of the Proposed Action:

In the "Environmental Statement Relating to Operation of Vogtle Electric Generating Plant, Units 1 and 2," (FES) dated March 1985, the NRC staff addressed radiological impacts for plant operation up to a maximum (or "stretch") core design output of 3565 Mwt and corresponding NSSS stretch power output of 3579 Mwt. Thus, the radiological impacts of the proposed action have already been addressed and determined not to present a significant risk nor to present a significant adverse impact on the quality of the human environment. The proposed increase in power involves no significant change in types or significant increase in the amount of any radiological effluents that may be released offsite which have not already been evaluated and found

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acceptable in the FES. Similarly, there would be no significant increase in individual or cumulative occupational radiation exposure.

The FES for Vogtle addressed the non-radiological environmental impacts based upon the current NSSS power level of 3425 MWt. To support the application for amendments to increase power level, the licensee re-evaluated relevant parameters from the Vogtle Environmental Report - Operating License Stage based upon operation at the proposed NSSS power level of 3579 MWt (see enclosure 4 of the licensee's submittal of February 28, 1992, as supplemented by the licensee's letter of February 18, 1993). The re-evaluation included environmental parameters associated with the following systems:

River Water Intake and Circulating Water Systems

In the re-evaluation, the licensee concluded that the proposed uprated power operation would not require an increase in the blowdown rate for the natural draft cooling towers because the circulating water system (CWS) design flow rate already envelopes uprated conditions. Changes in the temperature of the cooling tower blowdown would be negligible.

The CWS design flow rate is the primary basis for determining makeup water requirements for the natural draft cooling towers. Other factors affecting makeup requirements include tower performance and meteorological conditions such as dry bulb temperature and relative humidity.

The licensee found that the tower makeup water requirements for the uprate would be within the current design values since the existing CWS design flow rate of 509,600 gpm envelopes the proposed uprate conditions. The licensee's conclusion took into consideration the total

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heat loads that would be conveyed to the natural draft cooling tower due to the uprate. The existing cooling tower design heat loads enveloped the licensee's calculated condenser heat loads for the proposed uprate.

Based on the manufacturer's instructions, the licensee states that the maximum drift loss for the natural draft cooling towers is 0.03% of the circulating water flow. The licensee finds that the existing drift loss would envelope the proposed uprate since the CWS design flow would not change for the uprate. The licensee does not expect the natural draft cooling water chemistry to change for the proposed uprate since the design cycles of concentration are expected to be maintained.

Accordingly, the licensee concluded that the design makeup flow to the natural draft cooling towers would not increase due to the proposed uprating, and the intake canal velocity would not be significantly affected.

#### Groundwater Withdrawal System

The licensee's re-evaluation for the groundwater withdrawal system included withdrawal rates to supply the nuclear service cooling water (NSCW) cooling towers. The primary makeup to the NSCW tower basins is supplied from the makeup well water storage tank. Backup makeup water is also supplied from the river water intake. The current normal makeup requirements to each of the NSCW cooling towers varies from about 230 gpm to 400 gpm based on NSCW design flow rates and the cycles of concentration.

The NSCW makeup requirements are also a function of the design heat loads during normal plant operation. The licensee found that the NSCW design heat loads for normal operation after the power uprate would

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increase by about 7% due to an increase in the calculated heat load for the refueling spent fuel pool. The licensee based the evaluation of the design heat loads upon the Unit 2 spent fuel pool, which bounds the Unit 1 design, since the Unit 2 pool and plant fuel management scheme uses high-density spent fuel storage racks. The increase in spent fuel pool heat load is due to use of VANTAGE-5 fuel with its extended fuel cycle, the proposed increase in power level, and use of the Unit 2 high-density racks. The increase in makeup for the NSCW tower will be proportional to the increase in the NSCW heat load during normal operation. Therefore, the licensee concluded that the increase in makeup requirements for each of the two Vogtle units would be only about 7% (30 gpm) of the current design makeup requirements.

The licensee evaluated this projected increase of 30 gpm. The FES for Vogtle was based on a conservatively estimated total groundwater usage of 840 gpm for the two units. The licensee has found from operating experience that actual total groundwater usage based on current data is about 750 gpm. The Permit to Use Groundwater that was issued for the Vogtle facility by the Georgia Department of Natural Resources - Environmental Protection Division authorizes an average withdrawal rate of 5,500,000 gpd (3819 gpm). Thus, the expected 60 gpm withdrawal rate associated with the proposed power increase does not result in a total groundwater withdrawal rate which exceeds the 840 gpm used in the FES. Moreover, the previous conclusions in the FES remain valid for the proposed power increase.

The licensee's re-evaluation also included other systems that use groundwater. The licensee concluded that groundwater withdrawal to

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supply ~~the~~ water treatment plant and the fire protection system would not increase due to the proposed power uprate. The licensee also does not expect any significant increase in makeup requirements for the reactor coolant system, component cooling water system, condensate and feedwater system, turbine plant cooling water system, auxiliary steam system, or liquid radwaste system due to the proposed power uprate. Therefore, the licensee concluded that there would be no changes in groundwater withdrawal.

#### Other Systems

The licensee's re-evaluation considered the flow rate required by the radwaste dilution system due to the proposed power increase. The licensee does not expect increases in liquid radwaste quantities or activity levels that would increase the required radwaste dilution flows.

The licensee also re-evaluated the river water discharge system and concluded that there would be no significant changes to the discharge flow rate, velocity, temperature or thermal plume, or chemical composition due to the uprate. The licensee found that water discharge parameters subject to the NPDES Permit would not change from parameters evaluated in the FES. The licensee also found that the discharge characteristics on which the NPDES Permit was based would also not be affected by the power increase and, therefore, that no modification to the existing NPDES Permit was required.

From its evaluation summarized above, the licensee finds that the environmental effects of Vogtle operation as presented in the FES were based on conservative estimates of operating conditions such that the conclusions of



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the FES remain valid for operation at the proposed uprated conditions. The licensee finds that the plant operating parameters impacted by the proposed uprate would remain within the bounding conditions on which the conclusions of the FES are based.

The NRC staff has reviewed the licensee's re-evaluation of the potential non-radiological environmental impacts for the proposed action. On the basis of this review, the NRC staff finds that the non-radiological environmental impacts associated with the proposed small increase in power are essentially immeasurable and do not change the conclusion in the FES that the operation of Vogtle would cause no significant adverse impact upon the quality of the human environment.

Accordingly, the Commission concludes that this proposed action would result in no significant radiological or non-radiological environmental impact.

Alternative to the Proposed Action:

The principal alternative would be to deny the requested amendments. Denial would not significantly reduce the environmental impact of plant operations, but would restrict operation of the Vogtle facility to the currently licensed power level. Denial of the amendments would prevent the facility from generating the additional 50 MWe from each Vogtle unit that is needed for present and future system loads.

Alternative Use of Resources:

This action does not involve the use of resources not previously considered in the "Final Environmental Statement Relating to Operation of Vogtle Electric Generating Plant, Units 1 and 2," dated March 1985.



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Agencies and Persons Consulted:

The Commission's staff reviewed the licensee's request and consulted with the Environmental Protection Division, Department of Natural Resources, for the State of Georgia. The State Liaison Officer acknowledged that the State of Georgia has no outstanding actions with respect to the proposed uprate and has no comment regarding the NRC's proposed action.

FINDING OF NO SIGNIFICANT IMPACT

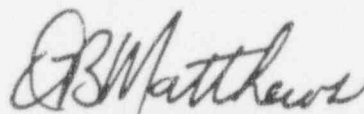
The Commission has determined not to prepare an environmental impact statement for the proposed license amendments.

Based on the foregoing environmental assessment, the NRC staff concludes that the proposed action will not have a significant adverse effect on the quality of the human environment.

For further details with respect to this action, see the application for amendments dated February 28, 1992, and supplemental letters from the licensee dated June 26 and August 28, 1992, and February 12, 18, 23, and 25, 1993. These documents are available for public inspection at the Commission's Public Document Room, 2120 L Street, NW., Washington, DC 20555, and at the Burke County Library, 412 Fourth Street, Waynesboro, Georgia 30830.

Dated at Rockville, Maryland, this **9th** day of **March** 1993.

FOR THE NUCLEAR REGULATORY COMMISSION

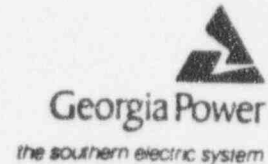


David B. Matthews, Director  
Project Directorate II-3  
Division of Reactor Projects - 1/11  
Office of Nuclear Reactor Regulation

ATTACHMENT 1 (CONTINUED)

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C. K. McCoy  
Vice President, Nuclear  
Vogtle Project



February 18, 1993

ELV-05251  
003066

Docket Nos. 50-424  
50-425

U. S. Nuclear Regulatory Commission  
ATTN: Document Control Desk  
Washington, D. C. 20555

VOGTLE ELECTRIC GENERATING PLANT  
ENVIRONMENTAL EFFECTS OF POWER UPDATING

Gentlemen:


By letter ELV-03375, dated February 28, 1992, Georgia Power Company (GPC) requested license amendments for the Vogtle Electric Generating Plant (VEGP) Units 1 and 2 which will allow operation at the engineered safety features design power level of 3565 Mwt. These changes will allow the electrical output of each unit to be increased by about 50 MWe. Included with the request was a summary of an evaluation of the effects of operating at the increased power level which concluded that the proposed uprate did not constitute an unreviewed environmental question.

The NRC Staff requested additional information concerning (1) the need for the additional power, (2) the effects of the power uprating on the National Pollutant Discharge Elimination System (NPDES) Permit, (3) the effects of additional use of groundwater, and (4) the basis for the conclusion that the environmental effects for the uprated conditions are within those previously evaluated.

- 1) The Southern Electric System (SES) experienced substantial load growth during the 1980s. By the late 1980s, the system recognized that new sources would be needed in the early- and mid-1990s.

Three of the five load-serving companies in the system filed for certification of new generating units in 1991 and 1992. The Alabama Public Service Commission certified the need for 720 MW of new capacity at the Greene County site for completion in 1995 and 1996. The Georgia Public Service Commission certified the need for 160 MW in 1994 for Savannah Electric and 480 MW of new capacity in 1994 and 1995. Georgia Power Company filed in early 1993 for an additional 160 MW of capacity in 1995 and released a request for proposals for up to 800 MW in 1996 and up to 800 MW in 1997 of independent power.

All of these certifications and requests for proposals assumed the VEGP uprates will be successful, or more capacity would have been needed.

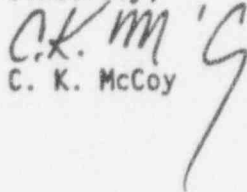
Georgia Power 

U. S. Nuclear Regulatory Commission  
ELV-05251  
Page 2

- 2) The evaluation of the effects of operation at the increased power level on water discharges subject to the NPDES Permit indicates that discharge parameters do not change from parameters evaluated in the Final Environmental Statement (FES). The discharge characteristics on which the NPDES Permit was based are also not affected by the proposed increase in power. Therefore, no modification to the existing NPDES Permit is required.
- 3) The projected increase of 30 gpm per unit in groundwater usage that was indicated in ELV-03375 has been evaluated. The FES for VEGP was based on a conservatively estimated total groundwater usage of 840 gpm for the two units. Operating experience has shown that actual total groundwater usage based on current data is approximately 750 gpm. The expected additional 60 gpm withdrawal associated with the proposed power increase does not result in a total groundwater withdrawal rate (810 gpm) that exceeds the 840 gpm rate defined in the FES. In addition, the 810 gpm (1,166,400 gpd) withdrawal rate does not exceed the withdrawal limit defined in the Permit to Use Groundwater issued to VEGP by the Georgia Department of Natural Resources - Environmental Protection Division. The permit authorizes an average withdrawal rate of 5,500,000 gpd. As such, no additional evaluation of groundwater withdrawal is necessary to support the proposed increase in power.
- 4) The environmental effects of VEGP operation were evaluated based on conservative estimates of operating conditions. The results were summarized in the FES. The effects of operation at the increased power level have been evaluated relative to the FES, and it has been determined that the conclusions of the FES remain valid for operation at uprated conditions. The plant operating parameters impacted by the proposed uprate remain within the bounding conditions on which the conclusions of the FES are based.

The additional information described above does not alter the conclusions or information provided in our letter ELV-03375.

Sincerely,

  
C. K. McCoy

CKM/HWM/gmb

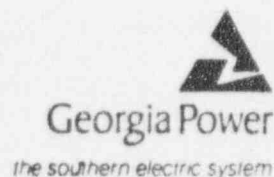
xc: Georgia Power Company  
W. B. Shipman  
M. Sheibani  
NORMS

U. S. Nuclear Regulatory Commission  
Mr. S. D. Ebnetter, Regional Administrator  
Mr. D. S. Hood, Licensing Project Manager  
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ATTACHMENT 1 (CONTINUED)

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February 28, 1992

ELV-03375  
001039

Docket Nos. 50-424  
50-425

TAC-M82724  
M82725

U. S. Nuclear Regulatory Commission  
ATTN: Document Control Desk  
Washington, D. C. 20555

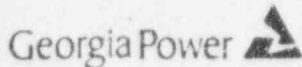
Gentlemen:

**VOGTLE ELECTRIC GENERATING PLANT  
LICENSE CHANGE POWER UPDATING**

In accordance with 10 CFR 50.90, Georgia Power Company (GPC) proposes to amend the Vogtle Electric Generating Plant (VEGP) Units 1 and 2 licenses NPF-68 and NPF-81. The proposed amendment changes the Technical Specifications by revising the definition of rated thermal power from 3411 megawatts thermal (Mwt) to 3565 Mwt. In order to operate at the higher power level it is also necessary to change the value of the statistical summation of errors assumed in the setpoint calculation for the overtemperature delta (OTDT) trip function and to revise the value of the power distribution reset function for OTDT.

The thermal hydraulic and nuclear aspects of the core were originally evaluated on the basis of a core thermal output of 3411 Mwt. Although the current licensed core power level is 3411 Mwt, all safety systems including the containment and engineering safety features were designed for operation at a higher core power level of 3565 Mwt. The proposed change to the licensed power level will allow VEGP Units 1 and 2 to operate at the engineering safety features design power rating without any modifications. This has been confirmed by performing detailed evaluations and reanalyses of VEGP systems at the uprated core power rating level. Georgia Power Company has previously submitted reanalyses of most accidents and transients using the higher power level (GPC letter ELV-02166 dated November 29, 1990). The results of the reanalyses for the remainder of the accidents and transients are discussed in the nuclear steam supply system (NSSS) and balance of plant (BOP) licensing reports. These reports are included with this letter.

The uprating efforts were performed in accordance with the licensing bases that currently exist for the Vogtle units. These bases assure the same level of protection for the public health and safety at the uprated conditions as at the presently licensed power level. The analyses show that operation at the uprated power level can be achieved without changing any of the criteria that have



U. S. Nuclear Regulatory Commission  
ELV-03375  
Page 2

previously been used as the bases for acceptable operation. Therefore, the margins of safety provided by the acceptance criteria used for the current licensed power level will not be changed by operating at the uprated power level.

The analyses that were necessary to support operation at uprated conditions were performed to allow the formation of as much operating margin as possible. The results of these analyses were used to support previous submittals to the NRC such as for the use of VANTAGE-5 fuel and the reduction in the required amount of reactor coolant system flow. The input assumptions used in the analyses include conservative values for other parameters such as response times and initial reactor coolant system temperatures. While these conservative assumptions are indicated in the descriptions of the analyses that are attached to this letter, only the changes necessary for operating at the increased core power level are being requested.

Enclosure 1 provides a description of the proposed changes to the Technical Specifications and the reasons for the changes. Enclosure 2 provides an evaluation of operation at the uprated power level in accordance with the requirements of 10 CFR 50.92. Enclosure 3 provides the revised pages for insertion into the Technical Specifications. Enclosure 4 provides an environmental evaluation, and enclosure 5 contains the NSSS licensing report, and enclosure 6 contains the BOP licensing report.

In accordance with the long-term fuel management and power planning of GPC, approval of operation at the uprated power level is requested by March 1, 1993. This schedule will allow orderly implementation of the power increase prior to peak load demands anticipated in the summer of 1993.

In accordance with 10 CFR 50.91, the designated state official will be sent a copy of this letter and all enclosures.

Mr. C. K. McCoy states that he is a vice president of Georgia Power Company and is authorized to execute this oath on behalf of Georgia Power Company and that, to the best of his knowledge and belief, the facts set forth in this letter and enclosures are true.

GEORGIA POWER COMPANY

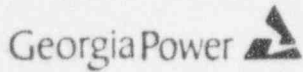
By:

*C. K. McCoy*  
C. K. McCoy

Sworn to and subscribed before me this 28<sup>th</sup> day of February, 1992.

*Mary Bentley*  
Notary Public

MY COMMISSION EXPIRES MAY 6, 1995



U. S. Nuclear Regulatory Commission  
ELV-03375  
Page 3

Enclosures:

1. Basis for Proposed Changes
2. 10 CFR 50.92 Evaluation
3. Instructions for Incorporation
4. Environmental Evaluation
5. NSSS Licensing Report
6. BOP Licensing Report

c(w): Georgia Power Company  
Mr. W. B. Shipman  
Mr. M. Sheibani  
NORMS

U. S. Nuclear Regulatory Commission  
Mr. S. D. Ebnetter, Regional Administrator  
Mr. D. S. Hood, Licensing Project Manager, NRR  
Mr. B. R. Bonser, Senior Resident Inspector, Vogtle

State of Georgia  
Mr. J. D. Tanner, Commissioner, Department of Natural Resources



IV

GEORGIA POWER COMPANY

VOGTLE ELECTRIC GENERATING PLANT - UNITS 1 AND 2

NRC DOCKET NOS. 50-424 AND 50-425

FACILITY OPERATING LICENSE NOS. NPF-68 AND NPF-81

ANNUAL RADIOLOGICAL EFFLUENT RELEASE REPORT  
CALENDAR YEAR 1993

*VOGTLE ELECTRIC GENERATING PLANT*  
*ANNUAL REPORT*  
*PLANT RADIOACTIVE EFFLUENT RELEASES*

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*VOGTLE ELECTRIC GENERATING PLANT*

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## **1.0 Liquid Effluents**

### **1.1 Regulatory Limits/Technical Specifications**

#### **1.1.1 Concentration Limits**

The concentration of radioactive material released in liquid effluents to UNRESTRICTED AREAS shall be limited to the concentrations specified in 10 CFR Part 20, Appendix B, Table II, Column 2 for radionuclides other than dissolved or entrained noble gases. For dissolved or entrained noble gases, the concentration shall be limited to  $2.0\text{E-}4$  microcuries/ml total activity.

#### **1.1.2 Dose Limits**

The dose or dose commitment to a MEMBER OF THE PUBLIC from radioactive materials in liquid effluents released, from each unit, to UNRESTRICTED AREAS shall be limited:

- a. During any calendar quarter to less than or equal to 1.5 mrem to the whole body and to less than or equal to 5 mrem to any organ, and
- b. During any calendar year to less than or equal to 3 mrem to the whole body and to less than or equal to 10 mrem to any organ.

### **1.2 Maximum Permissible Concentration (MPC)**

MPC values used in determining allowable liquid radwaste release rates and concentrations for principal gamma emitters, I-131, tritium, Sr-89, Sr-90 and Fe-55 are taken from 10 CFR Part 20, Appendix B, Table II, Column 2.

For dissolved or entrained noble gases in liquid radwaste, the MPC is  $2\text{E-}04$  uCi/ml total activity.

For gross alpha in liquid radwaste, the MPC is obtained from 10 CFR Part 20, Appendix B, Note 2.d as  $3.0\text{E-}08$  uCi/ml.

Further, for all the above radionuclides or categories of radioactivity, the overall MPC fraction is determined in accordance with 10 CFR Part 20, Appendix B, Note 1.

The method whereby the MPC fraction is used to determine release rates and liquid radwaste effluent radiation monitor setpoints is described in Subsection 1.3 of this report.

### 1.3 Measurements and Approximations of Total Radioactivity

Prior to release of any tank containing liquid radwaste, and following the required recirculations, samples are collected and analyzed in accordance with ODCM Section 1.5.1. A sample from each tank planned for release is analyzed for principal gamma emitters, I-131, and dissolved and entrained noble gases by gamma spectrometry. Monthly and quarterly composites are prepared for analysis by extracting aliquots from each sample taken from tanks which are released. Liquid radwaste sample analyses are performed as follows:

	MEASUREMENT	FREQUENCY	METHOD
1.	Gamma Isotopic	Each Batch	Gamma Spectroscopy with computerized data reduction.
2.	Dissolved or entrained noble gases	Each Batch	Gamma Spectroscopy with computerized data reduction
3.	Tritium	Monthly Composite	Distillation and liquid scintillation counting
4.	Gross Alpha	Monthly Composite	Gas flow proportional counting
5.	Sr-89 & Sr-90	Quarterly Composite	Chemical separation and gas flow proportional or scintillation counting
6.	Fe-55	Quarterly Composite	Chemical separation and liquid scintillation counting

Gamma isotopic measurements are performed in-house in the radiochemistry lab using germanium detectors with resolution of 1.80 keV or lower. The detectors are shielded by four inches of lead. A liquid radwaste sample is typically counted for 2000-4000 seconds. A peak search of the resulting gamma ray spectrum is performed by the computer system. Energy and net count data of all significant peaks are determined, and a quantitative reduction or MDA



calculation is performed. The procedure ensures that the LLD's are met for the nuclides specified in ODCM 1.5.1.2.1, Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, Ce-141 and Ce-144. The quantitative calculations, corrections for counting time, decay time, sample volume, sample geometry, detector efficiency, baseline counts, branching ratio and MDA calculations, are made based on the counts at the location on the spectrum where the peak for that radio-nuclide would be located, if present.

Tritium, Gross Alpha, Sr-89, Sr-90 and Fe-55 are, in some cases, analyzed offsite.

The radionuclide concentrations determined by gamma spectroscopic analysis of a sample taken from a tank planned for release and the most current sample analysis results available for tritium, gross alpha, Sr-89, Sr-90 and Fe-55 are used along with the corresponding MPC values to determine a MPC fraction for the tank planned for release. This MPC fraction is then used, with appropriate safety factors, along with the minimum assured dilution stream flow to calculate maximum permissible release rate and a liquid effluent monitor setpoint. The monitor setpoint is calculated to assure that the limits of Offsite Dose Calculation Manual (ODCM) are not exceeded.

A monitor reading in excess of the calculated setpoint results in an automatic termination of the liquid radwaste discharge. Liquid effluent discharge is also automatically terminated if the dilution stream flow rate falls below the minimum assured dilution flow rate used in the setpoint calculations and established as a setpoint on the dilution stream flow monitor.

Radionuclide concentrations, safety factors, dilution stream flow rate, and liquid effluent radiation monitor calibrations are entered into the computer and a pre-release printout is generated. If the release is not permissible, appropriate warnings will be included on the computer screen. If the release is permissible, it is approved by the Chemistry Foreman on duty and sent to the Operations Department for approval and processing. When the release is completed, the necessary data from the release (i.e., release volume, etc.) are transferred from the Operations Department to the Chemistry Department. These data are input to the computer and a post-release printout is generated. The post release printout contains actual release rates, actual release concentrations and quantities, actual dilution flow, and calculated doses to an individual.

## 1.4 Liquid Effluent Release Data

### 1.4.1 Tables

Regulatory Guide 1.21 Tables 2A, 2B, and 2-C are found in this report as Tables 1-2A, 1-2AA, 1-2B, 1-2BB, 2-C, and 2-CC. Data is presented on a quarterly basis as required by Regulatory Guide 1.21 for all four quarters.

### 1.4.2 Total Error Measurement

The total or maximum error associated with the effluent measurement will include the cumulative errors resulting from the total operation of sampling and measurement. Because it may be very difficult to assign error terms for each parameter affecting the final measurement, detailed statistical evaluation of error is not suggested. The objective should be to obtain an overall estimate of the error associated with measurements of radioactive materials released in liquid effluents.

#### 1.4.2.1 Fission and activation total release was calculated from sample analysis results and release point flow rates.

Sampling and statistical error	10%
Counting Equipment Calibration	10%
Tank Volumes and System Flow Rates	20%
TOTAL ERROR	40%

#### 1.4.2.2 Total Tritium release was calculated from sample analysis results and release point volumes.

Tank volumes and system flow rate	20%
Sampling and statistical errors	10%
Counting equipment calibration	10%
TOTAL ERROR	40%

#### 1.4.2.3 Dissolved and entrained gases were calculated from sample analysis results and release point volumes.

Tank volumes and system flow rate	20%
Sampling and statistical error	20%
Counting equipment calibration	10%
TOTAL ERROR	50%

**1.4.2.4** Gross alpha radioactivity was calculated from sample analysis results and release point volumes.

Tank volumes and system flowrates	20%
Sampling and statistical error	10%
Counting Equipment calibration	10%
Compositing sample error	5%
TOTAL ERROR	45%

**1.4.2.5** Volume of waste prior to dilution was calculated from level indicators on the tanks and pump discharge flow rates and times.

Level Indicator error	10%
Operator Interpretation of gauge	10%
TOTAL ERROR	20%

**1.4.2.6** Volume of dilution water used was calculated from flow rate indicators and pump discharge flow rates and times.

Flow rate indicator error	10%
Operator interpretation of gauge	10%
TOTAL ERROR	20%

## **1.5 Radiological Impact on Man Due to Liquid Releases**

Doses to an individual due to radioactivity in liquid effluent were calculated in accordance with the Offsite Dose Calculation Manual. Results are presented in Table 1-4A for Unit 1 and 1-4B for Unit 2, for all four quarters.

## **1.6 Abnormal Releases**

**1.6.1** There were no unplanned releases for this reporting period.

## **1.7 River Flow**

The average flow rate of the Savannah River for the Annual Effluent Report period was obtained from the Clark Hill Dam, Corporation of Engineers Office. The average flow rate is 10665 cubic feet/sec.

## TABLE1-2A

## GEORGIA POWER COMPANY

## VOGTLE ELECTRIC GENERATING PLANT U-1

## SEMIANNUAL SUMMATION OF ALL RELEASES BY QUARTER

## ALL LIQUID EFFLUENTS

UNIT : 1

STARTING : 1-JAN-1993

ENDING : 30-JUN-1993

TYPE OF EFFLUENT	UNITS	QUARTER1	QUARTER 2	EST.TOTAL ERROR %
A. FISSION & ACTIVATION PRODUCTS				
1. TOTAL RELEASE (NOT INCLUDING TRITIUM,GASES,ALPHA)	CURIES	4.07E-02	4.64E-02	40
2. AVERAGE DILUTED CONCENTRATION DURING PERIOD	uCi/ML	4.39E-08	9.19E-08	
3. PERCENTAGE OF APPLICABLE LIMIT	%	N/A	N/A	
B. TRITIUM				
1. TOTAL RELEASE	CURIES	1.78E+02	3.78E+01	40
2. AVERAGE DILUTED CONCENTRATION DURING PERIOD	uCi/ML	1.92E-04	7.49E-05	
3. PERCENTAGE OF APPLICABLE LIMIT	%	N/A	N/A	
C.DISSOLVED AND ENTRAINED GASES				
1. TOTAL RELEASE	CURIES	7.65E-04	0.00E+00	50
2. AVERAGE DILUTED CONCENTRATION DURING PERIOD	uCi/ML	8.24E-10	0.00E+00	
3. PERCENTAGE OF APPLICABLE LIMIT	%	N/A	N/A	
D.GROSS ALPHA RADIOACTIVITY				
1. TOTAL RELEASE	CURIES	6.07E-06	0.00E+00	45
E.WASTE VOL RELEASED(PRE-DILUTION)	LITRE	8.42E+05	5.97E+05	20
F.VOLUME OF DILUTION WATER USED	LITRE	9.27E+08	5.04E+08	20

Zeros in this table indicate that no radioactivity was present above detectable levels. See Table 1-5 for typical LLD for liquid sample analyses

TABLE1-2B

## GEORGIA POWER COMPANY

## VOGTLE ELECTRIC GENERATING PLANT U-2

## SEMIANNUAL SUMMATION OF ALL RELEASES BY QUARTER

## ALL LIQUID EFFLUENTS

UNIT : 2

STARTING : 1-JAN-1993

ENDING : 30-JUN-1993

TYPE OF EFFLUENT	UNITS	QUARTER1	QUARTER 2	EST.TOTAL ERROR %
A. FISSION & ACTIVATION PRODUCTS				
1. TOTAL RELEASE (NOT INCLUDING TRITIUM,GASES,ALPHA)	CURIES	1.55E-02	9.22E-02	40
2. AVERAGE DILUTED CONCENTRATION DURING PERIOD	uCi/ML	3.08E-08	1.42E-07	
3. PERCENTAGE OF APPLICABLE LIMIT	%	N/A	N/A	
B. TRITIUM				
1. TOTAL RELEASE	CURIES	6.71E+01	2.57E+01	40
2. AVERAGE DILUTED CONCENTRATION DURING PERIOD	uCi/ML	1.33E-04	3.96E-05	
3. PERCENTAGE OF APPLICABLE LIMIT	%	N/A	N/A	
C.DISSOLVED AND ENTRAINED GASES				
1. TOTAL RELEASE	CURIES	2.40E-05	6.91E-07	50
2. AVERAGE DILUTED CONCENTRATION DURING PERIOD	uCi/ML	4.77E-11	1.06E-12	
3. PERCENTAGE OF APPLICABLE LIMIT	%	N/A	N/A	
D.GROSS ALPHA RADIOACTIVITY				
1. TOTAL RELEASE	CURIES	0.00E+00	2.04E-04	45
E.WASTE VOL RELEASED(PRE-DILUTION)	LITRE	2.39E+05	1.03E+07	20
F.VOLUME OF DILUTION WATER USED	LITRE	5.03E+08	6.39E+08	20

Zeroes in this table indicate that no radioactivity was present above detectable levels. See Table 1-5 for typical LLD for liquid sample analyses

TABLE1-2C

## GEORGIA POWER COMPANY

## VOGTLE ELECTRIC GENERATING PLANT SITE

SEMIANNUAL SUMMATION OF ALL RELEASES BY QUARTER  
ALL LIQUID EFFLUENTS  
SITE

STARTING : 1-JAN-1993      ENDING : 30-JUN-1993

TYPE OF EFFLUENT	UNITS	QUARTER1	QUARTER 2	EST.TOTAL ERROR %
A. FISSION & ACTIVATION PRODUCTS				
1. TOTAL RELEASE (NOT INCLUDING TRITIUM,GASES,ALPHA)	CURIES	5.62E-02	1.39E-01	40
2. AVERAGE DILUTED CONCENTRATION DURING PERIOD	uCi/ML	3.93E-08	1.20E-07	
3. PERCENTAGE OF APPLICABLE LIMIT	%	N/A	N/A	
B. TRITIUM				
1. TOTAL RELEASE	CURIES	2.45E+02	6.35E+01	40
2. AVERAGE DILUTED CONCENTRATION DURING PERIOD	uCi/ML	1.71E-04	5.50E-05	
3. PERCENTAGE OF APPLICABLE LIMIT	%	N/A	N/A	
C.DISSOLVED AND ENTRAINED GASES				
1. TOTAL RELEASE	CURIES	7.89E-04	6.91E-07	50
2. AVERAGE DILUTED CONCENTRATION DURING PERIOD	uCi/ML	5.51E-10	5.99E-13	
3. PERCENTAGE OF APPLICABLE LIMIT	%	N/A	N/A	
D.GROSS ALPHA RADIOACTIVITY				
1. TOTAL RELEASE	CURIES	6.07E-06	2.04E-04	45
E.WASTE VOL RELEASED(PRE-DILUTION)	LITRE	1.08E+06	1.09E+07	20
F.VOLUME OF DILUTION WATER USED	LITRE	1.43E+09	1.14E+09	20

Zeroes in this table indicate that no radioactivity was present above detectable levels. See Table 1-5 for typical LLD for liquid sample analyses

## TABLE1-2AA

## GEORGIA POWER COMPANY

## VOGTLE ELECTRIC GENERATING PLANT U-1

## SEMIANNUAL SUMMATION OF ALL RELEASES BY QUARTER

## ALL LIQUID EFFLUENTS

UNIT : 1

STARTING : 1-JUL-1993      ENDING : 31-DEC-1993

TYPE OF EFFLUENT	UNITS	QUARTER3	QUARTER 4	EST.TOTAL ERROR %
A. FISSION & ACTIVATION PRODUCTS				
1. TOTAL RELEASE (NOT INCLUDING TRITIUM,GASES,ALPHA)	CURIES	1.27E-01	6.99E-02	40
2. AVERAGE DILUTED CONCENTRATION DURING PERIOD	uCi/ML	3.53E-07	1.90E-07	
3. PERCENTAGE OF APPLICABLE LIMIT	%	N/A	N/A	
B. TRITIUM				
1. TOTAL RELEASE	CURIES	1.11E+02	5.37E+01	40
2. AVERAGE DILUTED CONCENTRATION DURING PERIOD	uCi/ML	3.09E-04	1.46E-04	
3. PERCENTAGE OF APPLICABLE LIMIT	%	N/A	N/A	
C.DISSOLVED AND ENTRAINED GASES				
1. TOTAL RELEASE	CURIES	2.99E-05	1.48E-04	50
2. AVERAGE DILUTED CONCENTRATION DURING PERIOD	uCi/ML	8.31E-11	4.02E-10	
3. PERCENTAGE OF APPLICABLE LIMIT	%	N/A	N/A	
D.GROSS ALPHA RADIOACTIVITY				
1. TOTAL RELEASE	CURIES	0.00E+00	0.00E+00	45
E.WASTE VOL RELEASED(PRE-DILUTION)	LITRE	7.80E+05	9.83E+05	20
F.VOLUME OF DILUTION WATER USED	LITRE	3.59E+08	3.68E+08	20

Zeroes in this table indicate that no radioactivity was present above detectable levels. See Table 1-5 for typical LLD for liquid sample analyses

TABLE1-2BB

## GEORGIA POWER COMPANY

## VOGTLE ELECTRIC GENERATING PLANT U-2

## SEMIANNUAL SUMMATION OF ALL RELEASES BY QUARTER

## ALL LIQUID EFFLUENTS

UNIT : 2

STARTING : 1-JUL-1993      ENDING : 31-DEC-1993

TYPE OF EFFLUENT	UNITS	QUARTER3	QUARTER 4	EST.TOTAL ERROR %
A. FISSION & ACTIVATION PRODUCTS				
1. TOTAL RELEASE (NOT INCLUDING TRITIUM,GASES,ALPHA)	CURIES	1.46E-01	9.82E-01	40
2. AVERAGE DILUTED CONCENTRATION DURING PERIOD	uCi/ML	2.77E-07	1.14E-06	
3. PERCENTAGE OF APPLICABLE LIMIT	%	N/A	N/A	
B. TRITIUM				
1. TOTAL RELEASE	CURIES	2.45E+02	4.30E+01	40
2. AVERAGE DILUTED CONCENTRATION DURING PERIOD	uCi/ML	4.64E-04	4.98E-05	
3. PERCENTAGE OF APPLICABLE LIMIT	%	N/A	N/A	
C.DISSOLVED AND ENTRAINED GASES				
1. TOTAL RELEASE	CURIES	2.31E-03	1.16E-04	50
2. AVERAGE DILUTED CONCENTRATION DURING PERIOD	uCi/ML	4.39E-09	1.34E-10	
3. PERCENTAGE OF APPLICABLE LIMIT	%	N/A	N/A	
D.GROSS ALPHA RADIOACTIVITY				
1. TOTAL RELEASE	CURIES	1.16E-05	0.00E+00	45
E.WASTE VOL RELEASED(PRE-DILUTION)	LITRE	1.06E+06	3.64E+07	20
F.VOLUME OF DILUTION WATER USED	LITRE	5.26E+08	8.27E+08	20

Zeros in this table indicate that no radioactivity was present above detectable levels. See Table 1-5 for typical LLD for liquid sample analyses



TABLE1-2CC

## GEORGIA POWER COMPANY

## VOGTLE ELECTRIC GENERATING PLANT SITE

SEMIANNUAL SUMMATION OF ALL RELEASES BY QUARTER  
ALL LIQUID EFFLUENTS  
SITE

STARTING : 1-JUL-1993      ENDING : 31-DEC-1993

TYPE OF EFFLUENT	UNITS	QUARTER3	QUARTER 4	EST.TOTAL ERROR %
A. FISSION & ACTIVATION PRODUCTS				
1. TOTAL RELEASE (NOT INCLUDING TRITIUM,GASES,ALPHA)	CURIES	2.73E-01	1.05E+00	40
2. AVERAGE DILUTED CONCENTRATION DURING PERIOD	uCI/ML	3.08E-07	8.54E-07	
3. PERCENTAGE OF APPLICABLE LIMIT	%	N/A	N/A	
B. TRITIUM				
1. TOTAL RELEASE	CURIES	3.56E+02	9.67E+01	40
2. AVERAGE DILUTED CONCENTRATION DURING PERIOD	uCI/ML	4.01E-04	7.84E-05	
3. PERCENTAGE OF APPLICABLE LIMIT	%	N/A	N/A	
C.DISSOLVED AND ENTRAINED GASES				
1. TOTAL RELEASE	CURIES	2.34E-03	2.64E-04	50
2. AVERAGE DILUTED CONCENTRATION DURING PERIOD	uCI/ML	2.64E-09	2.14E-10	
3. PERCENTAGE OF APPLICABLE LIMIT	%	N/A	N/A	
D.GROSS ALPHA RADIOACTIVITY				
1. TOTAL RELEASE	CURIES	1.16E-05	0.00E+00	45
E.WASTE VOL RELEASED(PRF-DILUTION)	LITRE	1.84E+06	3.74E+07	20
F.VOLUME OF DILUTION WATER USED	LITRE	8.85E+08	1.20E+09	20

Zeroes in this table indicate that no radioactivity was present above detectable levels. See Table 1-5 for typical LLD for liquid sample analyses

TABLE 1-3A

REPORT CATEGORY : SEMIANNUAL LIQUID CONTINUOUS AND BATCH RELEASES

: TOTALS FOR EACH NUCLIDE RELEASED.

TYPE OF ACTIVITY : ALL RADIONUCLIDES

REPORTING PERIOD : QUARTER 1 AND QUARTER 2 YEAR 1993

: UNIT 1

ISOTOPE	UNIT	CONTINUOUS		BATCH	
		QUARTER1	QUARTER2	QUARTER1	QUARTER2
Co-57	CURIES	0.00E+00	0.00E+00	4.58E-06	4.41E-05
Co-58	CURIES	0.00E+00	2.47E-05	2.02E-03	1.10E-02
Co-60	CURIES	0.00E+00	3.91E-06	2.12E-03	3.72E-03
Cr-51	CURIES	0.00E+00	7.63E-06	3.93E-04	4.30E-03
Cs-134	CURIES	0.00E+00	3.41E-06	1.21E-04	1.07E-03
Cs-137	CURIES	0.00E+00	2.98E-06	1.89E-04	1.04E-03
Fe-55	CURIES	0.00E+00	1.76E-03	2.07E-02	1.25E-02
Fe-59	CURIES	0.00E+00	0.00E+00	5.75E-05	4.16E-04
G-ALPHA	CURIES	0.00E+00	0.00E+00	6.07E-06	0.00E+00
H-3	CURIES	0.00E+00	3.20E-04	1.78E+02	3.78E+01
I-131	CURIES	0.00E+00	0.00E+00	3.64E-03	4.63E-04
I-133	CURIES	0.00E+00	0.00E+00	4.41E-05	0.00E+00
Mn-54	CURIES	0.00E+00	1.75E-06	5.25E-04	1.41E-03
Nb-95	CURIES	0.00E+00	0.00E+00	9.65E-05	1.85E-03
Nb-97	CURIES	0.00E+00	0.00E+00	1.23E-04	2.15E-04
Sb-124	CURIES	0.00E+00	0.00E+00	1.21E-04	1.16E-05
Sb-125	CURIES	0.00E+00	0.00E+00	1.02E-02	3.29E-03
Sn-113	CURIES	0.00E+00	0.00E+00	0.00E+00	7.25E-05
Sr-89	CURIES	0.00E+00	1.41E-03	7.83E-05	2.52E-05
Sr-90	CURIES	0.00E+00	1.70E-04	0.00E+00	2.70E-06
I-132	CURIES	0.00E+00	0.00E+00	1.08E-04	0.00E+00
Sb-122	CURIES	0.00E+00	0.00E+00	1.15E-05	0.00E+00
Te-132	CURIES	0.00E+00	0.00E+00	1.02E-04	0.00E+00
Xe-133m	CURIES	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-133	CURIES	0.00E+00	0.00E+00	7.65E-04	0.00E+00
Xe-135	CURIES	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Zn-65	CURIES	0.00E+00	0.00E+00	8.44E-06	1.64E-04
Zr-95	CURIES	0.00E+00	0.00E+00	1.19E-05	1.13E-03
Ag-110m	CURIES	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Br-82	CURIES	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Na-24	CURIES	0.00E+00	0.00E+00	0.00E+00	0.00E+00
W-187	CURIES	0.00E+00	0.00E+00	0.00E+00	0.00E+00
La-140	CURIES	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Be-7	CURIES	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ce-144	CURIES	0.00E+00	0.00E+00	0.00E+00	1.17E-04
Hf-181	CURIES	0.00E+00	0.00E+00	0.00E+00	2.86E-05
Te-129m	CURIES	0.00E+00	0.00E+00	0.00E+00	9.67E-05
TOTAL CURIES		0.00E+00	3.70E-03	1.78E+02	3.78E+01

TABLE 1-3B

REPORT CATEGORY : SEMIANNUAL LIQUID CONTINUOUS AND BATCH RELEASES  
 :TOTALS FOR EACH RADIONUCLIDE  
 TYPE OF ACTIVITY :ALL RADIONUCLIDES  
 REPORTING PERIOD :QUARTER 1 AND QUARTER 2 YEAR 1993  
 :UNIT 2

ISOTOPE	UNIT	CONTINUOUS		BATCH	
		QUARTER1	QUARTER 2	QUARTER1	QUARTER 2
Co-57	CURIES	0.00E+00	0.00E+00	9.69E-06	1.96E-05
Co-58	CURIES	0.00E+00	5.90E-04	1.49E-03	1.74E-03
Co-60	CURIES	0.00E+00	2.91E-06	1.17E-03	6.76E-04
Cr-51	CURIES	0.00E+00	2.14E-06	9.30E-07	5.10E-04
Cs-134	CURIES	0.00E+00	2.14E-06	5.94E-05	1.71E-04
Cs-137	CURIES	0.00E+00	1.60E-05	7.89E-05	1.97E-04
Fe-55	CURIES	0.00E+00	7.65E-02	7.85E-03	1.06E-02
Fe-59	CURIES	0.00E+00	4.28E-06	0.00E+00	5.31E-05
G-ALPHA	CURIES	0.00E+00	2.04E-04	0.00E+00	0.00E+00
H-3	CURIES	0.00E+00	0.00E+00	6.71E+01	2.57E+01
I-131	CURIES	0.00E+00	2.14E-06	3.27E-06	0.00E+00
I-133	CURIES	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Mn-54	CURIES	0.00E+00	8.10E-07	2.30E-04	2.19E-04
Nb-95	CURIES	0.00E+00	1.07E-06	1.50E-05	1.67E-04
Nb-97	CURIES	0.00E+00	0.00E+00	6.91E-05	1.25E-05
Sb-124	CURIES	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sb-125	CURIES	0.00E+00	2.14E-06	4.41E-03	6.02E-04
Sn-113	CURIES	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-89	CURIES	0.00E+00	0.00E+00	6.34E-05	1.10E-05
Sr-90	CURIES	0.00E+00	0.00E+00	4.29E-06	0.00E+00
I-132	CURIES	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sb-122	CURIES	0.00E+00	0.00E+00	0.00E+00	2.85E-05
Te-132	CURIES	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-133m	CURIES	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-133	CURIES	0.00E+00	0.00E+00	2.33E-05	0.00E+00
Xe-135	CURIES	0.00E+00	0.00E+00	7.22E-07	8.91E-07
Zn-65	CURIES	0.00E+00	0.00E+00	8.91E-06	1.09E-05
Zr-95	CURIES	0.00E+00	1.07E-06	0.00E+00	1.04E-04
Ag-110m	CURIES	0.00E+00	0.00E+00	0.00E+00	7.72E-06
Br-82	CURIES	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Na-24	CURIES	0.00E+00	0.00E+00	0.00E+00	0.00E+00
W-187	CURIES	0.00E+00	0.00E+00	0.00E+00	0.00E+00
La-140	CURIES	0.00E+00	5.25E-08	0.00E+00	0.00E+00
Be-7	CURIES	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ce-144	CURIES	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Hf-181	CURIES	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Te-129m	CURIES	0.00E+00	0.00E+00	0.00E+00	0.00E+00
TOTAL CURIES		0.00E+00	7.73E-02	6.71E+01	2.57E+01

TABLE 1-3C

REPORT CATEGORY : SEMIANNUAL LIQUID CONTINUOUS AND BATCH RELEASES  
 :TOTALS FOR EACH RADIONUCLIDE  
 TYPE OF ACTIVITY : ALL RADIONUCLIDES  
 REPORTING PERIOD :QUARTER 1 AND QUARTER 2 YEAR 1993  
 :SITE

ISOTOPE	UNIT	CONTINUOUS		BATCH	
		QUARTER 1	QUARTER 2	QUARTER 1	QUARTER 2
Co-57	CURIES	0.00E+00	0.00E+00	1.43E-05	6.37E-05
Co-58	CURIES	0.00E+00	6.15E-04	3.51E-03	1.27E-02
Co-60	CURIES	0.00E+00	6.82E-06	3.29E-03	4.40E-03
Cr-51	CURIES	0.00E+00	9.77E-06	3.94E-04	4.81E-03
Cs-134	CURIES	0.00E+00	5.55E-06	1.80E-04	1.24E-03
Cs-137	CURIES	0.00E+00	1.90E-05	2.68E-04	1.24E-03
Fe-55	CURIES	0.00E+00	7.83E-02	2.86E-02	2.31E-02
Fe-59	CURIES	0.00E+00	4.28E-06	5.75E-05	4.69E-04
G-ALPHA	CURIES	0.00E+00	2.04E-04	6.07E-06	0.00E+00
H-3	CURIES	0.00E+00	3.20E-04	2.45E+02	6.35E+01
I-131	CURIES	0.00E+00	2.14E-06	3.64E-03	4.63E-04
I-133	CURIES	0.00E+00	0.00E+00	4.41E-05	0.00E+00
Mn-54	CURIES	0.00E+00	2.56E-06	7.55E-04	1.63E-03
Nb-95	CURIES	0.00E+00	1.07E-06	1.12E-04	2.02E-03
Nb-97	CURIES	0.00E+00	0.00E+00	1.92E-04	2.28E-04
Sb-124	CURIES	0.00E+00	0.00E+00	1.21E-04	1.16E-05
Sb-125	CURIES	0.00E+00	2.14E-06	1.46E-02	3.89E-03
Sn-113	CURIES	0.00E+00	0.00E+00	0.00E+00	7.25E-05
Sr-89	CURIES	0.00E+00	1.41E-03	1.42E-04	3.62E-05
Sr-90	CURIES	0.00E+00	1.70E-04	4.29E-06	2.70E-06
I-132	CURIES	0.00E+00	0.00E+00	1.08E-04	0.00E+00
Sb-122	CURIES	0.00E+00	0.00E+00	1.15E-05	2.85E-05
Te-132	CURIES	0.00E+00	0.00E+00	1.02E-04	0.00E+00
Xe-133m	CURIES	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-133	CURIES	0.00E+00	0.00E+00	7.88E-04	0.00E+00
Xe-135	CURIES	0.00E+00	0.00E+00	7.22E-07	6.91E-07
Zn-65	CURIES	0.00E+00	0.00E+00	1.74E-05	1.75E-04
Zr-95	CURIES	0.00E+00	1.07E-06	1.19E-05	1.23E-03
Ag-110m	CURIES	0.00E+00	0.00E+00	0.00E+00	7.72E-06
Br-82	CURIES	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Na-24	CURIES	0.00E+00	0.00E+00	0.00E+00	0.00E+00
W-187	CURIES	0.00E+00	0.00E+00	0.00E+00	0.00E+00
La-140	CURIES	0.00E+00	5.25E-08	0.00E+00	0.00E+00
Be-7	CURIES	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ce-144	CURIES	0.00E+00	0.00E+00	0.00E+00	1.17E-04
Hf-181	CURIES	0.00E+00	0.00E+00	0.00E+00	2.86E-05
Te-129m	CURIES	0.00E+00	0.00E+00	0.00E+00	9.67E-05
TOTAL CURIES		0.00E+00	8.10E-02	2.45E+02	6.36E+01

TABLE 1-3AA

REPORT CATEGORY : SEMIANNUAL LIQUID CONTINUOUS AND BATCH RELEASES  
 : TOTALS FOR EACH NUCLIDE RELEASED.  
 TYPE OF ACTIVITY : ALL RADIONUCLIDES  
 REPORTING PERIOD : QUARTER 3 AND QUARTER 4 YEAR 1993  
 : UNIT 1

ISOTOPE	UNIT	CONTINUOUS		BATCH	
		QUARTER3	QUARTER4	QUARTER3	QUARTER4
Co-57	CURIES	0.00E+00	0.00E+00	3.21E-04	1.68E-04
Co-58	CURIES	0.00E+00	0.00E+00	2.63E-02	1.57E-02
Co-60	CURIES	0.00E+00	0.00E+00	1.82E-02	1.03E-02
Cr-51	CURIES	0.00E+00	0.00E+00	7.01E-03	5.78E-03
Cs-134	CURIES	0.00E+00	0.00E+00	7.56E-04	1.86E-03
Cs-137	CURIES	0.00E+00	0.00E+00	1.14E-03	2.52E-03
Fe-55	CURIES	0.00E+00	0.00E+00	5.31E-02	1.85E-02
Fe-59	CURIES	0.00E+00	0.00E+00	4.49E-04	4.53E-04
G-ALPHA	CURIES	0.00E+00	0.00E+00	0.00E+00	0.00E+00
H-3	CURIES	0.00E+00	0.00E+00	1.11E+02	5.37E+01
I-131	CURIES	0.00E+00	0.00E+00	0.00E+00	6.45E-05
I-133	CURIES	0.00E+00	0.00E+00	0.00E+00	8.86E-06
Mn-54	CURIES	0.00E+00	0.00E+00	5.09E-03	3.20E-03
Nb-95	CURIES	0.00E+00	0.00E+00	2.90E-03	1.27E-03
Nb-97	CURIES	0.00E+00	0.00E+00	4.92E-04	0.00E+00
Sb-124	CURIES	0.00E+00	0.00E+00	0.00E+00	9.26E-06
Sb-125	CURIES	0.00E+00	0.00E+00	8.92E-03	8.00E-03
Sn-113	CURIES	0.00E+00	0.00E+00	1.74E-04	4.12E-05
Sr-89	CURIES	0.00E+00	0.00E+00	4.69E-05	0.00E+00
Sr-90	CURIES	0.00E+00	0.00E+00	0.00E+00	4.12E-06
I-132	CURIES	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sb122	CURIES	0.00E+00	0.00E+00	8.18E-05	5.42E-05
Te-132	CURIES	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-133m	CURIES	0.00E+00	0.00E+00	0.00E+00	4.52E-05
Xe-133	CURIES	0.00E+00	0.00E+00	2.99E-05	1.03E-04
Xe-135	CURIES	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Zn-65	CURIES	0.00E+00	0.00E+00	3.76E-04	1.30E-03
Zr-95	CURIES	0.00E+00	0.00E+00	1.49E-03	6.46E-04
Ag-110m	CURIES	0.00E+00	0.00E+00	3.05E-04	0.00E+00
Br-82	CURIES	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Na-24	CURIES	0.00E+00	0.00E+00	0.00E+00	0.00E+00
W-187	CURIES	0.00E+00	0.00E+00	0.00E+00	0.00E+00
La-140	CURIES	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Be-7	CURIES	0.00E+00	0.00E+00	0.00E+00	2.61E-05
Ce-144	CURIES	0.00E+00	0.00E+00	1.16E-04	0.00E+00
Hf-181	CURIES	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Te-129m	CURIES	0.00E+00	0.00E+00	0.00E+00	0.00E+00
TOTAL CURIES		0.00E+00	0.00E+00	1.11E+02	5.38E+01

TABLE 1-3BB

REPORT CATEGORY : SEMIANNUAL LIQUID CONTINUOUS AND BATCH RELEASES  
 :TOTALS FOR EACH RADIONUCLIDE  
 TYPE OF ACTIVITY :ALL RADIONUCLIDES  
 REPORTING PERIOD :QUARTER 3 AND QUARTER 4 YEAR 1993  
 :UNIT 2

ISOTOPE	UNIT	CONTINUOUS		BATCH	
		QUARTER3	QUARTER 4	QUARTER3	QUARTER 4
Co-57	CURIES	0.00E+00	0.00E+00	2.57E-04	7.35E-05
Co-58	CURIES	0.00E+00	3.54E-04	2.54E-02	9.15E-03
Co-60	CURIES	0.00E+00	9.36E-04	1.47E-02	3.90E-03
Cr-51	CURIES	0.00E+00	0.00E+00	8.23E-03	5.28E-03
Cs-134	CURIES	0.00E+00	0.00E+00	1.47E-03	5.65E-05
Cs-137	CURIES	0.00E+00	0.00E+00	1.97E-03	1.49E-04
Fe-55	CURIES	0.00E+00	9.40E-01	7.75E-02	1.20E-02
Fe-59	CURIES	0.00E+00	0.00E+00	6.28E-04	4.14E-04
G-ALPHA	CURIES	0.00E+00	0.00E+00	1.16E-05	0.00E+00
H-3	CURIES	0.00E+00	1.98E+00	2.45E+02	4.01E+01
I-131	CURIES	0.00E+00	0.00E+00	0.00E+00	3.35E-05
I-133	CURIES	0.00E+00	0.00E+00	0.00E+00	2.72E-05
Mn-54	CURIES	0.00E+00	3.91E-04	4.80E-03	1.14E-03
Nb-95	CURIES	0.00E+00	0.00E+00	2.61E-03	9.39E-04
Nb-97	CURIES	0.00E+00	0.00E+00	2.82E-04	0.00E+00
Sb-124	CURIES	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sb-125	CURIES	0.00E+00	0.00E+00	5.87E-03	6.53E-03
Sn-113	CURIES	0.00E+00	0.00E+00	1.82E-05	1.80E-05
Sr-89	CURIES	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-90	CURIES	0.00E+00	8.25E-06	0.00E+00	0.00E+00
I-132	CURIES	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sb-122	CURIES	0.00E+00	0.00E+00	5.01E-05	0.00E+00
Te-132	CURIES	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-133m	CURIES	0.00E+00	0.00E+00	9.33E-05	0.00E+00
Xe-133	CURIES	0.00E+00	0.00E+00	2.15E-03	1.16E-04
Xe-135	CURIES	0.00E+00	0.00E+00	7.15E-05	0.00E+00
Zn-65	CURIES	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Zr-95	CURIES	0.00E+00	0.00E+00	1.35E-03	5.21E-04
Ag-110m	CURIES	0.00E+00	0.00E+00	3.68E-04	0.00E+00
Br-82	CURIES	0.00E+00	0.00E+00	0.00E+00	1.54E-05
Na-24	CURIES	0.00E+00	0.00E+00	0.00E+00	4.58E-06
W-187	CURIES	0.00E+00	0.00E+00	0.00E+00	3.04E-05
La-140	CURIES	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Be-7	CURIES	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Ce-144	CURIES	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Hf-181	CURIES	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Te-129m	CURIES	0.00E+00	0.00E+00	0.00E+00	0.00E+00
TOTAL CURIES		0.00E+00	2.92E+00	2.45E+02	4.01E+01



TABLE 1-3CC

REPORT CATEGORY : SEMIANNUAL LIQUID CONTINUOUS AND BATCH RELEASES  
 : TOTALS FOR EACH RADIONUCLIDE  
 TYPE OF ACTIVITY : ALL RADIONUCLIDES  
 REPORTING PERIOD : QUARTER 3 AND QUARTER 4 YEAR 1993  
 : SITE

ISOTOPE	UNIT	CONTINUOUS		BATCH	
		QUARTER 3	QUARTER 4	QUARTER 3	QUARTER 4
Co-57	CURIES	0.00E+00	0.00E+00	5.78E-04	2.42E-04
Co-58	CURIES	0.00E+00	3.54E-04	5.17E-02	2.49E-02
Co-60	CURIES	0.00E+00	9.36E-04	3.29E-02	1.42E-02
Cr-51	CURIES	0.00E+00	0.00E+00	1.52E-02	1.11E-02
Cs-134	CURIES	0.00E+00	0.00E+00	2.23E-03	1.92E-03
Cs-137	CURIES	0.00E+00	0.00E+00	3.11E-03	2.67E-03
Fe-55	CURIES	0.00E+00	9.40E-01	1.31E-01	3.05E-02
Fe-59	CURIES	0.00E+00	0.00E+00	1.08E-03	8.67E-04
G-ALPHA	CURIES	0.00E+00	0.00E+00	1.16E-05	0.00E+00
H-3	CURIES	0.00E+00	1.98E+00	3.56E+02	9.38E+01
I-131	CURIES	0.00E+00	0.00E+00	0.00E+00	9.80E-05
I-133	CURIES	0.00E+00	0.00E+00	0.00E+00	3.61E-05
Mn-54	CURIES	0.00E+00	3.01E-04	9.89E-03	4.34E-03
Nb-95	CURIES	0.00E+00	0.00E+00	5.51E-03	2.21E-03
Nb-97	CURIES	0.00E+00	0.00E+00	7.74E-04	0.00E+00
Sb-124	CURIES	0.00E+00	0.00E+00	0.00E+00	9.26E-06
Sb-125	CURIES	0.00E+00	0.00E+00	1.48E-02	1.45E-02
Sn-113	CURIES	0.00E+00	0.00E+00	1.92E-04	5.92E-05
Sr-89	CURIES	0.00E+00	0.00E+00	4.69E-05	0.00E+00
Sr-90	CURIES	0.00E+00	8.25E-06	0.00E+00	4.12E-06
I-132	CURIES	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sb-122	CURIES	0.00E+00	0.00E+00	1.32E-04	5.42E-05
Te-132	CURIES	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-133m	CURIES	0.00E+00	0.00E+00	9.33E-05	4.52E-05
Xe-133	CURIES	0.00E+00	0.00E+00	2.18E-03	2.19E-04
Xe-135	CURIES	0.00E+00	0.00E+00	7.15E-05	0.00E+00
Zn-65	CURIES	0.00E+00	0.00E+00	3.76E-04	1.30E-03
Zr-95	CURIES	0.00E+00	0.00E+00	2.84E-03	1.17E-03
Ag-110m	CURIES	0.00E+00	0.00E+00	6.73E-04	0.00E+00
Br-82	CURIES	0.00E+00	0.00E+00	0.00E+00	1.54E-05
Na-24	CURIES	0.00E+00	0.00E+00	0.00E+00	4.58E-06
W-187	CURIES	0.00E+00	0.00E+00	0.00E+00	3.04E-05
La-140	CURIES	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Be-7	CURIES	0.00E+00	0.00E+00	0.00E+00	2.61E-05
Ce-144	CURIES	0.00E+00	0.00E+00	1.16E-04	0.00E+00
Hf-181	CURIES	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Te-129m	CURIES	0.00E+00	0.00E+00	0.00E+00	0.00E+00
TOTAL CURIES		0.00E+00	2.92E+00	3.56E+02	9.39E+01

**TABLE 1-4A**  
**VOGTLE ELECTRIC GENERATING PLANT**  
**ANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT**  
**INDIVIDUAL DOSES DUE TO LIQUID RELEASES**  
**JANUARY, 1993 THROUGH JUNE, 1993**

**UNIT 1**

**CUMULATIVE DOSE PER QUARTER**

ORGAN	ODCM LIMIT	UNITS	QUARTER 1	% OF ODCM LIMIT	QUARTER 2	% OF ODCM LIMIT
Bone	5.0	mrem	1.58E-03	3.16E-02	1.17E-02	2.34E-01
Liver	5.0	mrem	5.00E-03	1.00E-01	1.47E-02	2.94E-01
T. Body	1.5	mrem	4.35E-03	2.90E-01	1.19E-02	7.93E-01
Thyroid	5.0	mrem	5.61E-03	1.12E-01	1.31E-03	2.62E-02
Kidney	5.0	mrem	3.37E-03	6.74E-02	5.49E-03	1.10E-01
Lung	5.0	mrem	1.58E-02	3.16E-01	7.53E-03	1.51E-01
GI-LLI	5.0	mrem	4.16E-03	8.32E-02	4.19E-03	8.38E-02

**CUMULATIVE DOSE PER YEAR**

ORGAN	ODCM LIMIT	UNITS	YEAR TO DATE	% OF ODCM LIMIT
Bone	10.0	mrem	1.33E-02	1.33E-01
Liver	10.0	mrem	1.97E-02	1.97E-01
T. Body	3.0	mrem	1.63E-02	5.43E-01
Thyroid	10.0	mrem	6.92E-03	6.92E-02
Kidney	10.0	mrem	8.86E-03	8.86E-02
Lung	10.0	mrem	2.33E-02	2.33E-01
GI-LLI	10.0	mrem	8.35E-03	8.35E-02



**TABLE 1-4B**  
**VOGTLE ELECTRIC GENERATING PLANT**  
**ANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT**  
**INDIVIDUAL DOSES DUE TO LIQUID RELEASES**  
**JANUARY, 1993 THROUGH JUNE, 1993**

**UNIT 2**

**CUMULATIVE DOSE PER QUARTER**

ORGAN	ODCM LIMIT	UNITS	QUARTER 1	% OF ODCM LIMIT	QUARTER 2	% OF ODCM LIMIT
Bone	5.0	mrem	7.30E-04	1.46E-02	3.25E-03	6.50E-02
Liver	5.0	mrem	2.34E-03	4.68E-02	5.45E-03	1.09E-01
T. Body	1.5	mrem	2.06E-03	1.37E-01	3.91E-03	2.61E-01
Thyroid	5.0	mrem	1.27E-03	2.54E-02	5.96E-04	1.19E-02
Kidney	5.0	mrem	1.61E-03	3.22E-02	2.01E-03	4.02E-02
Lung	5.0	mrem	7.72E-03	1.54E-01	2.72E-03	5.44E-02
GI-LLI	5.0	mrem	2.06E-03	4.12E-02	1.43E-03	2.88E-02

**CUMULATIVE DOSE PER YEAR**

ORGAN	ODCM LIMIT	UNITS	YEAR TO DATE	% OF ODCM LIMIT
Bone	10.0	mrem	3.98E-03	3.98E-02
Liver	10.0	mrem	7.79E-03	7.79E-02
T. Body	3.0	mrem	5.97E-03	1.99E-01
Thyroid	10.0	mrem	1.87E-03	1.87E-02
Kidney	10.0	mrem	3.62E-03	3.62E-02
Lung	10.0	mrem	1.04E-02	1.04E-01
GI-LLI	10.0	mrem	3.49E-03	3.49E-02

**TABLE 1-4AA**  
**VOGTLE ELECTRIC GENERATING PLANT**  
**ANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT**  
**INDIVIDUAL DOSES DUE TO LIQUID RELEASES**  
*July, 1993 THROUGH December, 1993*

**UNIT 1**

**CUMULATIVE DOSE PER QUARTER**

ORGAN	ODCM LIMIT	UNITS	QUARTER 3	% OF ODCM LIMIT	QUARTER 4	% OF ODCM LIMIT
Bone	5.0	mrem	1.40E-02	2.80E-01	2.80E-02	5.60E-01
Liver	5.0	mrem	2.64E-02	5.28E-01	5.07E-02	1.01E+00
T. Body	1.5	mrem	1.97E-02	1.31E+00	3.74E-02	2.49E+00
Thyroid	5.0	mrem	2.88E-03	5.76E-02	1.85E-03	3.70E-02
Kidney	5.0	mrem	1.06E-02	2.12E-01	1.83E-02	3.66E-01
Lung	5.0	mrem	2.82E-02	5.64E-01	2.64E-02	5.28E-01
GI-LLI	5.0	mrem	1.18E-02	2.36E-01	8.24E-03	1.65E-01

**CUMULATIVE DOSE PER YEAR**

ORGAN	ODCM LIMIT	UNITS	YEAR TO DATE	% OF ODCM LIMIT
Bone	10.0	mrem	5.53E-02	5.53E-01
Liver	10.0	mrem	9.68E-02	9.68E-01
T. Body	3.0	mrem	7.34E-02	2.45E-02
Thyroid	10.0	mrem	1.17E-02	1.17E-01
Kidney	10.0	mrem	3.78E-02	3.78E-01
Lung	10.0	mrem	7.79E-02	7.79E-01
GI-LLI	10.0	mrem	2.84E-02	2.84E-01

**TABLE 1-4BB**  
**VOGTLE ELECTRIC GENERATING PLANT**  
**ANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT**  
**INDIVIDUAL DOSES DUE TO LIQUID RELEASES**  
**JULY, 1993 THROUGH DECEMBER, 1993**

**UNIT 2**

**CUMULATIVE DOSE PER QUARTER**

ORGAN	ODCM LIMIT	UNITS	QUARTER 3	% OF ODCM LIMIT	QUARTER 4	% OF ODCM LIMIT
Bone	5.0	mrem	2.54E-02	5.08E-01	2.49E-02	4.98E-01
Liver	5.0	mrem	5.05E-02	1.01E+00	1.95E-02	3.90E-01
T. Body	1.5	mrem	3.87E-02	2.58E+00	7.06E-03	4.71E-01
Thyroid	5.0	mrem	6.91E-03	1.38E-01	1.21E-03	2.42E-02
Kidney	5.0	mrem	2.10E-02	4.20E-01	2.09E-03	4.18E-02
Lung	5.0	mrem	2.55E-02	5.10E-01	2.35E-02	4.70E-01
GI-LLI	5.0	mrem	1.51E-02	3.-2E-01	1.45E-02	2.90E-01

**CUMULATIVE DOSE PER YEAR**

ORGAN	ODCM LIMIT	UNITS	YEAR TO DATE	% of ODCM LIMIT
Bone	10.0	mrem	5.43E-02	5.43E-01
Liver	10.0	mrem	7.78E-02	7.78E-01
T. Body	3.0	mrem	5.17E-02	1.72E+00
Thyroid	10.0	mrem	9.99E-03	9.99E-02
Kidney	10.0	mrem	2.67E-02	2.67E-01
Lung	10.0	mrem	5.94E-02	5.94E-01
GI-LLI	10.0	mrem	3.31E-02	3.31E-01

**TABLE 1-5****LOWER LIMITS OF DETECTION - LIQUID SAMPLE ANALYSES****VOGTLE ELECTRIC GENERATING PLANT  
JANUARY, 1993 - DECEMBER 31, 1993**

The values in this table represent a priori lower limits of detection (LLD) which are typically achieved in laboratory analyses of liquid radwaste samples.

<b>RADIONUCLIDE</b>	<b>LLD</b>	<b>UNITS</b>
Mn-54	2.73E-08	uCi/ml
Fe-59	8.33E-08	uCi/ml
Co-58	3.78E-08	uCi/ml
Co-60	6.76E-08	uCi/ml
Zn-65	1.32E-07	uCi/ml
Mo-99	4.31E-07	uCi/ml
Cs-134	3.06E-08	uCi/ml
Cs-137	4.51E-08	uCi/ml
Ce-141	6.99E-08	uCi/ml
Ce-144	2.95E-07	uCi/ml
I-131	5.97E-08	uCi/ml
Xe-133	9.11E-08	uCi/ml
Xe-135	4.27E-08	uCi/ml
Fe-55	1.00E-06	uCi/ml
Sr-89	5.00E-08	uCi/ml
Sr-90	7.00E-09	uCi/ml
H-3	2.00E-06	uCi/ml
Gross Alpha	7.00E-08	uCi/ml

**TABLE 1-6A**

**GEORGIA POWER COMPANY  
VOGTLE ELECTRIC GENERATING PLANT - UNIT 1**

**BATCH RELEASE SUMMARY OF ALL RELEASES**

**STARTING: JANUARY, 1993    ENDING: JUNE, 1993**

**LIQUID RELEASES**

Number of Releases	40	
Total Time For All Releases	15323.00	minutes
Maximum Time For a Release	1406.00	minutes
Average Time For a Release	383.08	minutes
Minimum Time For a Release	9.00	minutes
Average Stream Flow	23.52	GPM

**GASEOUS RELEASES**

Number of Releases:	109	
Total Time For All Releases	57618.00	minutes
Maximum Time For A Release	5820.00	minutes
Average Time For A Release	528.61	minutes
Minimum Time For A Release	4.00	minutes

**TABLE 1-6B**

**GEORGIA POWER COMPANY**

**VOGTLE ELECTRIC GENERATING PLANT - UNIT 2**

**BATCH RELEASE SUMMARY OF ALL RELEASES**

**STARTING:            JANUARY, 1993    ENDING:            JUNE, 1993**

**LIQUID RELEASES**

Number of Releases	17	
Total Time For All Releases	5107.00	minutes
Maximum Time For A Release	1019.00	minutes
Average Time For A Release	300.41	minutes
Minimum Time For A Release	32.00	minutes
Average Stream Flow	20.41	GPM

**GASEOUS RELEASES**

Number of Releases	37	
Total Time For All Releases	5793.00	minutes
Maximum Time For a Release	1725.00	minutes
Average Time For A Release	156.57	minutes
Minimum Time For A Release	32.00	minutes

**TABLE 1-6AA**

**GEORGIA POWER COMPANY  
VOGTLE ELECTRIC GENERATING PLANT - UNIT 1**

**BATCH RELEASE SUMMARY OF ALL RELEASES**

**STARTING: JULY, 1993    ENDING: DECEMBER, 1993**

**LIQUID RELEASES**

Number of Releases	61	
Total Time For All Releases	7908.00	minutes
Maximum Time For a Release	449.00	minutes
Average Time For a Release	129.64	minutes
Minimum Time For a Release	19.00	minutes
Average Stream Flow	58.33	GPM

**GASEOUS RELEASES**

Number of Releases:	87	
Total Time For All Releases	11368.00	minutes
Maximum Time For A Release	1184.00	minutes
Average Time For A Release	130.67	minutes
Minimum Time For A Release	6.00	minutes

**TABLE 1-6BB**

**GEORGIA POWER COMPANY**

**VOGTLE ELECTRIC GENERATING PLANT - UNIT 2**

**BATCH RELEASE SUMMARY OF ALL RELEASES**

**STARTING: JULY, 1993    ENDING: DECEMBER, 1993**

**LIQUID RELEASES**

Number of Releases	45	
Total Time For All Releases	8577.00	minutes
Maximum Time For A Release	482.00	minutes
Average Time For A Release	190.60	minutes
Minimum Time For A Release	56.00	minutes
Average Stream Flow	44.51	GPM

**GASEOUS RELEASES**

Number of Releases	60	
Total Time For All Releases	57679.00	minutes
Maximum Time For a Release	6880.00	minutes
Average Time For A Release	961.32	minutes
Minimum Time For A Release	7.00	minutes



## **2.0 Gaseous Effluents**

### **2.1 ODCM Specifications**

The ODCM Specifications presented in this section are for Unit 1 and Unit 2.

#### **2.1.1 Dose Rate Limit**

The dose rate due to radioactive materials released in gaseous effluents from the site to areas at and beyond the SITE BOUNDARY shall be limited to the following:

- a. For noble gases: Less than or equal to 500 mrem/yr to the whole body and less than or equal to 3000 mrem/yr to the skin and,
- b. For Iodine-131, for Iodine-133, for tritium and for all radionuclides in particulate form with half lives greater than 8 days: Less than or equal to 1500 mrem/yr to any organ.

#### **2.1.2 Air Dose Due To Noble Gas**

The air dose due to noble gases released in gaseous effluents, from each unit, to areas at and beyond the SITE BOUNDARY shall be limited to the following:

- a. During any calendar quarter: Less than or equal to 5 mrad for gamma radiation and less than or equal to 10 mrad for beta radiation, and
- b. During any calendar year: Less than or equal to 10 mrad for gamma radiation and less than or equal to 20 mrad for beta radiation.

#### **2.1.3 Dose To Any Organ**

The dose to a MEMBER OF THE PUBLIC from Iodine-131, Iodine-133, tritium and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released, from each unit, to areas at and beyond the SITE BOUNDARY shall be limited to the following.

- a. During any calendar quarter: Less than or equal to 7.5 mrem to any organ.
- b. During any calendar year: Less than or equal to 15 mrem to any organ.

#### **2.1.4 Total Fuel Cycle Dose Commitment (40CFR190)**

The annual (calendar year) dose or dose commitment to any MEMBER OF THE PUBLIC due to releases of radioactivity and to radiation from uranium fuel cycle sources shall be limited to less than or equal to 25 mrem to the whole body or to any organ, except the thyroid, which shall be limited to less than or equal to 75 mrem.

#### **2.2 Release Points of Gaseous Effluents**

Gaseous Effluents at Vogtle Electric Generating Plant are currently confined to four paths: plant vents (Unit 1 and Unit 2) the condenser air ejector and steam packing exhausters systems (Unit 1 and Unit 2).

Waste gas decay tanks are batch releases and the waste gas decay tanks are released through the Unit 1 plant vent. Containment purges are released through their respective plant vents.

#### **2.3 Sample Collection and Analysis**

All of the paths can be continuously monitored for gaseous radioactivity. Each is equipped with an integrated-type sample collection device for collecting particulates and iodines. During this release period, there were no radioactive releases through the condenser air ejector and steam packing exhausters system vents. Unless required more frequently under certain circumstances samples are collected as follows:

1. Noble gas samples are collected by grab sampling at least monthly.
2. Tritium samples are collected by grab sampling at least monthly. Tritium grab samples are taken at least once per 24 hours when the refueling cavity is flooded. Also tritium grab samples are taken at least once per 7 days from the Unit 1 Plant Vent, whenever spent fuel is in the Spent Fuel Pool (Unit 1 Plant Vent Only)
3. Radioiodine samples are collected from the sample stream through a charcoal cartridge over a 7 day period.
4. Particulates are collected from the sample stream through a particulate filter over a 7 day period.
5. The 7-day particulate filters above are analyzed for gross alpha activity.

6. Quarterly composite samples are prepared from the particulate filters collected over the previous quarter and the quarterly composite sample is analyzed for Sr-89 and Sr-90.

Batch Waste Gas Decay Tank releases are analyzed for iodines, particulates, and noble gases before each release. In addition, the containment atmosphere is analyzed for tritium on at least a monthly basis.

Sample analyses results and release flow rates from the release points form the basis for calculating released quantities of radionuclide specific radioactivity, dose rates associated with gaseous releases, and cumulative doses for the current quarter and year. This task is normally performed with computer assistance.

With each release period and batch release, radioactivity, dose rates, and cumulative doses are calculated. Cumulative dose results are tabulated, along with percent of ODCM limits for each release for the current quarter and year.

## **2.4 Total Quantities of Radioactivity, Dose Rates, and Cumulative Doses**

The methods for determining release quantities of radioactivity, dose rates, and cumulative doses follow:

### **2.4.1 Fission and Activation Gas**

The released radioactivity is determined from sample analyses results collected as described above and average release flow rates over the period represented by the collected sample.

Instantaneous dose rates due to noble gases and due to radioiodines, tritium, and particulates are calculated (with computer assistance). Calculated dose rates are compared to the dose rate limits specified in ODCM 2.5.1 for noble gases, radiiodine, tritium, and particulates. Dose rate calculation methodology is presented in the ODCM.

Beta and gamma air doses due to noble gases are calculated for the location in the unrestricted area with the potential for the highest exposure due to gaseous releases. Air doses are calculated for each release period and cumulative totals are kept for each unit for the calendar quarter and year. Cumulative air doses are compared to the dose limits specified in ODCM 2.5.2. Current percent of the ODCM limits are shown on the printout for each release period. Air dose calculation methodology is presented in the ODCM.

#### **2.4.2 Radioiodine, Tritium and Particulate Releases**

Released quantities of radioiodines are determined from the weekly samples and release flow rates for the two release points. Radioiodine concentrations are determined by gamma spectroscopy.

Release quantities of particulates are determined from the weekly (filter) samples and release flow rates for the two release points. Gamma spectroscopy is used to quantify concentrations of principal gamma emitters.

After each quarter, the particulate filters from each vent are combined, fused, and a strontium separation is performed. If Sr-89 or Sr-90 is not detected, LLD's are calculated. Strontium concentrations are input to the composite file of the computer to be used for release dose rate and individual dose calculations.

Tritium samples are obtained at least monthly from each vent by bubbling the sample stream through a water trap. The tritium concentration in water is converted to tritium concentration in air and this value is input into the composite file of the computer to be used in release, dose rate, and individual dose calculations.

Dose rates due to radioiodine, tritium, and particulates are calculated for a hypothetical child, exposed to the inhalation pathway, at the location in the unrestricted area where the potential dose rate is expected to be the highest. Dose rates are calculated for each release point, for each release period, and the total dose rate from both release points are compared to the dose rate limits specified in ODCM 2.5.1

Individual doses due to radioiodine, tritium and particulates are calculated for the controlling receptor, which for Vogtle Electric Generating Plant is a child exposed to the inhalation and ground-plane pathways. Individual doses are calculated for each release period, and cumulative totals are kept for each unit for the current calendar quarter and year. Cumulative individual doses are compared to the dose limits specified in ODCM 2.5.3.

Current percent of ODCM limits are shown on the printout for each release period.

#### **2.4.3 Gross Alpha Release**

The gross alpha release is calculated each month by counting the particulate filters for each week for gross alpha activity in a proportional counter by an offsite laboratory. The four or five weeks' numbers are then recorded on a data sheet and the activity is summed at the end of the month. This concentration is input to the composite file of the computer and is used for release calculations.

## **2.5 Gaseous Effluent Release Data**

### **2.5.1 Methodology**

Regulatory Guide 1.21 Tables 1A, 1B, and 1C are found in this report as Tables 2-2A, 2-2AA, 2-2B, 2-2BB, 2-2C, 2-2CC, 2-3A, 2-3AA, 2-3B, 2-3BB, 2-3C, 2-3CC, 2-4A, 2-4AA, 2-4B, 2-4BB, 2-4C, and 2-4CC. Data is presented on a quarterly basis as required by Regulatory Guide 1.21 for all quarters.

To complete table 2-2A and 2-2B, total release for each of the four categories (fission and activation gases, iodines, particulates, and tritium) was divided by the number of seconds in the quarter to obtain a release rate in uCi/second for each category for each quarter. However, the percent of the ODCM limits are not applicable because we have no curie limits for gaseous releases. Noble gases are limited as specified in ODCM 2.5.1. The other three categories (tritium, radioiodines, and particulates) are limited as a group as specified in ODCM 2.5.1.

Dose rates due to noble gas releases and due to radioiodine, tritium, and particulates were calculated as part of the pre-release and post-release permits on individual permits. No limits were exceeded for this reporting period.

Gross alpha radioactivity is reported in Table 2-2A and 2-2B as curies released in each quarter.

Limits for cumulative beta and gamma air doses due to noble gases are specified in ODCM 2.5.2. Cumulative air doses are presented in Table 2-6A and 2-6B along with percent of technical specification limits.

Limits for cumulative individual doses due to radioiodine, tritium and particulates, are specified in ODCM 2.5.3. Cumulative individual doses are presented in Table 2-7A and 2-7B along with percent of ODCM limits.

The total or maximum error associated with the effluent measurement will include the cumulative errors resulting from the total operation of sampling and measurement. Because it may be very difficult to assign error terms for each parameter affecting the final measurement, detailed statistical evaluation of error are not suggested. The objective should be to obtain an overall estimate of the error associated with measurements of radioactive materials released in liquid and gaseous effluents and solid waste.

Estimated errors are based on errors in counting equipment calibration, counting statistics, vent-flow rates, vent sample flow rates, non steady release rates, chemical yield factors and sample losses for such items as charcoal cartridges.

**2.5.1.1** Fission and activation total release was calculated from sample analysis results and release point flow rates.

Sampling and statistical error in counting	10%
Counting equipment calibration	10%
Vent flow Rates	10%
Non-steady release rates	20%
<b>TOTAL ERROR</b>	<b>50%</b>

**2.5.1.2** I-131 releases were calculated from each weekly sample:

Statistical error in counting	10%
Counting equipment calibration	10%
Vent Flow Rates	10%
Vent Sample Flow Rates	50%
Non-Steady release rates	10%
Losses from charcoal cartridges	10%
<b>TOTAL ERROR</b>	<b>100%</b>

**2.5.1.3** Particulates with half lives greater than 8 day releases were calculated from sample and analysis results and release point flow rates.

Statistical error at LLD concentration	10%
Counting equipment calibration	10%
Vent flow rates	10%
Vent sample flow rates	50%
Non steady release rates	10%
<b>TOTAL ERROR</b>	<b>90%</b>

**2.5.1.4** Total tritium releases were calculated from sample analysis results and release point flow rates.

Water vapor in sample stream determination	10%
Vent flow rates	10%
Counting calibration and statistics	10%
Non-steady release rates	10%
<b>TOTAL ERROR</b>	<b>40%</b>



**2.5.1.5** Gross Alpha radioactivity was calculated from sample analysis results and release point flow rates.

Statistical error at LLD concentration	10%
Counting equipment calibration	10%
Vent flow rates	10%
Vent sample flow rates	50%
Non Steady release rates	10%
TOTAL ERROR	90%

### **2.5.2 Gaseous Batch Data**

Other data pertinent to batch releases of radioactive gaseous effluent from Unit 1 and Unit 2 are listed in Table 1-6A and 1-6B.

## **2.6 Radiological Impact Due to Gaseous Releases**

Dose rates due to noble gas release were calculated for the site in accordance with ODCM 2.5.1. Dose rates due to radioiodine, tritium, and particulates in gaseous releases were calculated in accordance with ODCM 2.5.1.

As part of pre-release and post release on individual release permits, these dose rates were calculated. No limits were exceeded for this reporting period.

Cumulative air doses due to noble gas releases were calculated for each unit in accordance with ODCM 2.5.3. These results are presented in Tables 2-7A and 2-7B.

Dose rates and doses were calculated using the methodology presented in the Vogtle Electric Generating plant Offsite Dose Calculation Manual.

## **2.7 Unplanned Releases During 1993**

While GDT # 5 was being released on May 16, 1993, at 22:50 under Permit # 930110-G GDT # 2 had an unplanned release. The valves 1-1902-U4073 and 1-1902-U4077 were found to be leaking. These two valves will be replaced during the 1994 Fall Outage. During the release samples were pulled and analyzed to calculate the doses per ODCM requirements. No limits were exceeded. Procedure 13202-C is revised to check all the GDT pressures while a tank is being released

During the waste gas leak assessment test on 4-16-93, GDT # 5 lost greater than 60 psi of gas. The samples pulled during the leak assessment were used to calculate the dose and dose rate calculations per ODCM, to meet the requirements of ODCM. No ODCM limits were exceeded during this incident. Valve 1-1902-U4066 was leaking. The leaky valve is replaced under MWO # 1-93-01913 and Leak Assessment Procedure is revised to log the GDT pressure while the test is in progress. Also another leaky valve A1902U4109 was replaced under MWO # A-93-00672.



TABLE2-2A

## GEORGIA POWER COMPANY

## VOGTLE ELECTRIC GENERATING PLANT U-1

## SEMIANNUAL SUMMATION OF ALL RELEASES BY QUARTER

## ALL AIRBORNE EFFLUENTS

UNIT : 1

STARTING : 1-JAN-1993

ENDING : 30-JUN-1993

TYPE OF EFFLUENT	UNITS	QUARTER1	QUARTER 2	EST.TOTAL ERROR %
A. FISSION & ACTIVATION PRODUCTS				
1. TOTAL RELEASE	CURIES	2.20E+02	9.11E+00	50
2. AVERAGE RELEASE RATE FOR PERIOD	uCi/Sec	2.83E+01	1.16E+00	
3. PERCENTAGE OF APPLICABLE LIMIT	%	N/A	N/A	
B. RADIOIODINES				
1. TOTAL IODINE-131	CURIES	2.14E-04	2.26E-04	100
2.AVERAGE RELEASE RATE FOR PERIOD DURING PERIOD	uCi/Sec	2.75E-05	2.87E-05	
3. PERCENTAGE OF APPLICABLE LIMIT	%	N/A	N/A	
C. PARTICULATES				
1.PARTICULATES(HALF-LIVES >8DAYS)	CURIES	1.10E-06	9.35E-06	90
2.AVERAGE RELEASE RATE FOR PERIOD	uCi/Sec	1.42E-07	1.19E-06	
3. PERCENTAGE OF APPLICABLE LIMIT	%	N/A	N/A	
D.TRITIUM				
1.TOTAL RELEASE	CURIES	5.02E+01	4.95E+01	40
2.AVERAGE RELEASE RATE FOR PERIOD	uCi/Sec	6.46E+00	6.30E+00	
3.PERCENTAGE OF APPLICABLE LIMIT	%	N/A	N/A	
4.GROSS ALPHA RADIOACTIVITY	CURIES	1.90E-06	8.04E-06	90

Zeros in this table indicate that no radioactivity was present above detectable level. See Table 2-8 for typical LLD for gaseous sample analyses.

TABLE2-2B

## GEORGIA POWER COMPANY

## VOGTLE ELECTRIC GENERATING PLANT U-2

## SEMIANNUAL SUMMATION OF RELEASES BY QUARTER

ALL AIRBORNE RELEASES

UNIT : 2

STARTING : 1-JAN-1993      ENDING : 30-JUN-1993

TYPE OF EFFLUENT	UNITS	QUARTER1	QUARTER 2	EST.TOTAL ERROR %
A. FISSION & ACTIVATION PRODUCTS				
1. TOTAL RELEASE	CURIES	3.12E-01	1.46E+00	50
2. AVERAGE RELEASE RATE FOR PERIOD	uCi/Sec	4.01E-02	1.86E-01	
3. PERCENTAGE OF APPLICABLE LIMIT	%	N/A	N/A	
B. RADIOIODINES				
1. TOTAL IODINE-131	CURIES	9.8E-07	1.41E-05	100
2. AVERAGE RELEASE RATE FOR PERIOD DURING PERIOD	uCi/Sec	1.01E-07	1.79E-06	
3. PERCENTAGE OF APPLICABLE LIMIT	%	N/A	N/A	
C. PARTICULATES				
1. PARTICULATES (HALF-LIVES > 8 DAYS)	CURIES	1.61E-06	9.72E-07	90
2. AVERAGE RELEASE RATE FOR PERIOD	uCi/Sec	2.07E-07	1.24E-07	
3. PERCENTAGE OF APPLICABLE LIMIT	%	N/A	N/A	
D. TRITIUM				
1. TOTAL RELEASE	CURIES	1.87E-02	1.41E+01	40
2. AVERAGE RELEASE RATE FOR PERIOD	uCi/Sec	2.40E-03	1.79E+00	
3. PERCENTAGE OF APPLICABLE LIMIT	%	N/A	N/A	
4. GROSS ALPHA RADIOACTIVITY	CURIES	1.10E-05	5.54E-07	90

Zeros in this table indicate that no radioactivity was present above detectable level. See Table 2-8 for typical LLD for gaseous sample analyses.

TABLE2-2C

## GEORGIA POWER COMPANY

## VOGTLE ELECTRIC GENERATING PLANT SITE

SEMIANNUAL SUMMATION OF ALL RELEASES BY QUARTER  
ALL AIRBORNE EFFLUENTS  
SITE

STARTING : 1-JAN-1993      ENDING : 30-JUN-1993

TYPE OF EFFLUENT	UNITS	QUARTER1	QUARTER 2	EST.TOTAL ERROR %
A. FISSION & ACTIVATION PRODUCTS				
1. TOTAL RELEASE	CURIES	2.20E+02	1.06E+01	50
2. AVERAGE RELEASE RATE FOR PERIOD	uCi/Sec	2.83E+01	1.34E+00	
3. PERCENTAGE OF APPLICABLE LIMIT	%	N/A	N/A	
B. RADIOIODINES				
1. TOTAL IODINE-131	CURIES	2.15E-04	2.40E-04	100
2.AVERAGE RELEASE RATE FOR PERIOD DURING PERIOD	uCi/Sec	2.76E-05	3.05E-05	
3. PERCENTAGE OF APPLICABLE LIMIT	%	N/A	N/A	
C. PARTICULATES				
1.PARTICULATES(HALF-LIVES >8DAYS)	CURIES	2.71E-06	1.03E-05	90
2.AVERAGE RELEASE RATE FOR PERIOD	uCi/Sec	3.49E-07	9.20E-04	
3. PERCENTAGE OF APPLICABLE LIMIT	%	N/A	N/A	
D.TRITIUM				
1.TOTAL RELEASE	CURIES	5.02E+01	6.36E+01	40
2.AVERAGE RELEASE RATE FOR PERIOD	uCi/Sec	6.46E+00	8.09E+00	
3.PERCENTAGE OF APPLICABLE LIMIT	%	N/A	N/A	
4.GROSS ALPHA RADIOACTIVITY	CURIES	1.29E-05	8.59E-06	90

Zeros in this table indicate that no radioactivity was present above detectable level. See Table 2-8 for typical LLD for gaseous sample analyses.

## TABLE2-2AA

## GEORGIA POWER COMPANY

## VOGTLE ELECTRIC GENERATING PLANT U-1

## SEMIANNUAL SUMMATION OF ALL RELEASES BY QUARTER

## ALL AIRBORNE EFFLUENTS

UNIT : 1

STARTING : 1-JUL-1993

ENDING : 31-DEC-1993

TYPE OF EFFLUENT	UNITS	QUARTER3	QUARTER 4	EST.TOTAL ERROR %
A. FISSION & ACTIVATION PRODUCTS				
1. TOTAL RELEASE	CURIES	1.72E-01	3.43E-01	50
2. AVERAGE RELEASE RATE FOR PERIOD	uCi/Sec	2.16E-02	4.32E-02	
3. PERCENTAGE OF APPLICABLE LIMIT	%	N/A	N/A	
B. RADIOIODINES				
1. TOTAL IODINE-131	CURIES	4.42E-06	1.90E-06	100
2.AVERAGE RELEASE RATE FOR PERIOD DURING PERIOD	uCi/Sec	5.56E-07	2.39E-07	
3. PERCENTAGE OF APPLICABLE LIMIT	%	N/A	N/A	
C. PARTICULATES				
1.PARTICULATES(HALF-LIVES >8DAYS)	CURIES	0.00E+00	4.60E-07	90
2.AVERAGE RELEASE RATE FOR PERIOD	uCi/Sec	0.00E+00	5.79E-08	
3. PERCENTAGE OF APPLICABLE LIMIT	%	N/A	N/A	
D.TRITIUM				
1.TOTAL RELEASE	CURIES	3.06E+01	3.44E+01	40
2.AVERAGE RELEASE RATE FOR PERIOD	uCi/Sec	3.85E+00	4.33E+00	
3.PERCENTAGE OF APPLICABLE LIMIT	%	N/A	N/A	
4.GROSS ALPHA RADIOACTIVITY	CURIES	1.08E-06	1.38E-06	90

Zeroes in this table indicate that no radioactivity was present above detectable level. See Table 2-8 for typical LLD for gaseous sample analyses.

## TABLE2-2BB

## GEORGIA POWER COMPANY

## VOGTLE ELECTRIC GENERATING PLANT U-2

## SEMIANNUAL SUMMATION OF ALL RELEASES BY QUARTER

## ALL AIRBORNE EFFLUENTS

UNIT : 2

STARTING : 1-JUL-1993

ENDING : 31-DEC-1993

TYPE OF EFFLUENT	UNITS	QUARTER3	QUARTER 4	EST.TOTAL ERROR %
A. FISSION & ACTIVATION PRODUCTS				
1. TOTAL RELEASE	CURIES	1.07E-01	3.03E+00	50
2. AVERAGE RELEASE RATE FOR PERIOD	uCi/Sec	1.35E-02	3.81E-01	
3. PERCENTAGE OF APPLICABLE LIMIT	%	N/A	N/A	
B. RADIOIODINES				
1. TOTAL IODINE-131	CURIES	3.41E-07	5.43E-06	100
2.AVERAGE RELEASE RATE FOR PERIOD DURING PERIOD	uCi/Sec	4.29E-08	6.83E-07	
3. PERCENTAGE OF APPLICABLE LIMIT	%	N/A	N/A	
C. PARTICULATES				
1.PARTICULATES(HALF-LIVES >8DAYS)	CURIES	2.77E-06	4.02E-05	90
2.AVERAGE RELEASE RATE FOR PERIOD	uCi/Sec	3.49E-07	5.06E-06	
3. PERCENTAGE OF APPLICABLE LIMIT	%	N/A	N/A	
D.TRITIUM				
1.TOTAL RELEASE	CURIES	3.47E+01	9.87E+00	40
2.AVERAGE RELEASE RATE FOR PERIOD	uCi/Sec	4.37E+00	1.24E+00	
3.PERCENTAGE OF APPLICABLE LIMIT	%	N/A	N/A	
4.GROSS ALPHA RADIOACTIVITY	CURIES	4.30E-07	1.28E-06	90

Zeros in this table indicate that no radioactivity was present above detectable level. See Table 2-8 for typical LLD for gaseous sample analyses.

## TABLE2-2CC

## GEORGIA POWER COMPANY

## VOGTLE ELECTRIC GENERATING PLANT U-2

## SEMIANNUAL SUMMATION OF ALL RELEASES BY QUARTER

## ALL AIRBORNE EFFLUENTS

## SITE

50

STARTING : 1-JUL-1993

ENDING : 31-DEC-1993

TYPE OF EFFLUENT	UNITS	QUARTER3	QUARTER 4	
A. FISSION & ACTIVATION PRODUCTS				
1. TOTAL RELEASE	CURIES	2.79E-01	3.37E+00	100
2. AVERAGE RELEASE RATE FOR PERIOD	uCi/Sec	3.51E-02	4.24E-01	
3. PERCENTAGE OF APPLICABLE LIMIT	%	N/A	N/A	
B. RADIOIODINES				
1. TOTAL IODINE-131	CURIES	4.76E-06	7.33E-06	90
2.AVERAGE RELEASE RATE FOR PERIOD DURING PERIOD	uCi/Sec	5.99E-07	9.22E-07	
3. PERCENTAGE OF APPLICABLE LIMIT	%	N/A	N/A	
C. PARTICULATES				
1.PARTICULATES(HALF-LIVES >8DAYS)	CURIES	2.77E-06	4.07E-05	40
2.AVERAGE RELEASE RATE FOR PERIOD	uCi/Sec	3.49E-07	5.12E-06	
3. PERCENTAGE OF APPLICABLE LIMIT	%	N/A	N/A	
D.TRITIUM				
1.TOTAL RELEASE	CURIES	6.53E+01	4.43E+01	90
2.AVERAGE RELEASE RATE FOR PERIOD	uCi/Sec	8.22E+00	5.57E+00	
3.PERCENTAGE OF APPLICABLE LIMIT	%	N/A	N/A	
4.GROSS ALPHA RADIOACTIVITY	CURIES	1.51E-06	2.66E-06	

Zerones in this table indicate that no radioactivity was present above detectable level. See Table 2-8 for typical LLD for gaseous sample analyses.

**TABLE 2-3A (PAGE 1 OF 2)**

**VOGTLE ELECTRIC GENERATING PLANT**

**ANNUAL EFFLUENTS - MIXED MODE**

**JANUARY, 1993 THROUGH JUNE, 1993**

**UNIT 1**

<b>NUCLIDES RELEASED</b>	<b>UNIT</b>	<b>CONTINUOUS MODE</b>		<b>BATCH MODE</b>	
<b>1. Fission Gasses</b>		<b>QUARTER 1</b>	<b>QUARTER 2</b>	<b>QUARTER 1</b>	<b>QUARTER 2</b>
Ar-41	Ci	0.00E+00	0.00E+00	3.72E-01	1.30E-01
Kr-85	Ci	0.00E+00	0.00E+00	5.00E-01	1.17E+00
Kr-85M	Ci	0.00E+00	0.00E+00	3.84E-03	0.00E+00
Xe-131M	Ci	0.00E+00	0.00E+00	1.86E-01	1.00E-01
Xe-133	Ci	6.86E+01	5.84E-01	1.34E+02	7.12E+00
Xe-133M	Ci	0.00E+00	0.00E+00	6.12E-01	2.40E-03
Xe-135	Ci	2.74E-01	0.00E+00	8.32E-02	1.44E-03
Kr-87	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
<b>TOTAL FOR PERIOD</b>	Ci	6.89E+01	5.84E-01	1.36E+02	8.53E+00
<b>2. Iodines</b>					
I-131	Ci	2.14E-04	2.26E-04	0.00E+00	0.00E+00
I-133	Ci	1.09E-05	1.63E-06	0.00E+00	0.00E+00
<b>TOTAL FOR PERIOD</b>	Ci	2.25E-04	2.28E-04	0.00E+00	0.00E+00

\*Zeroes in this table indicate that no radioactivity was present above detectable levels. See Table 2-8 for typical lower limits of detection for gaseous sample analyses.

TABLE 2-3A (PAGE 2 OF 2)

VOGTLE ELECTRIC GENERATING PLANT

ANNUAL EFFLUENTS - MIXED MODE

JANUARY, 1993 THROUGH JUNE , 1993

UNIT 1

NUCLIDES RELEASED	UNIT	CONTINUOUS MODE		BATCH MODE	
		QUARTER 1	QUARTER 2	QUARTER 1	QUARTER 2
3. Particulates					
H-3	Ci	4.98E+01	3.56E+01	3.64E-01	1.39E+01
G-ALPHA	Ci	1.90E-06	8.04E-06	0.00E+00	0.00E+00
Na-24	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cr-51	Ci	0.00E+00	1.29E-06	0.00E+00	0.00E+00
Co-58	Ci	1.87E-07	7.24E-06	0.00E+00	0.00E+00
Fe-59	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Nb-95	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Mn-54	Ci	0.00E+00	4.32E-07	0.00E+00	0.00E+00
Sr-89	Ci	9.16E-07	0.00E+00	0.00E+00	0.00E+00
Sr-90	Ci	1.03E-07	0.00E+00	0.00E+00	0.00E+00
Zr-95	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Co-60	Ci	0.00E+00	3.90E-07	0.00E+00	0.00E+00
<b>TOTAL</b>	Ci	4.98E+01	3.56E+01	3.64E-01	1.39E+01

\*Zeroes in this table indicate that no radioactivity was present above detectable levels. See Table 2-8 for typical lower limits of detection for gaseous sample analyses.

\*\* Half lives greater than 8 days.



TABLE 2-3B (PAGE 1 OF 2)

VOGTLE ELECTRIC GENERATING PLANT

ANNUAL EFFLUENTS - MIXED MODE

JANUARY, 1993 THROUGH JUNE, 1993

UNIT 2

NUCLIDES RELEASED	UNIT	CONTINUOUS MODE		BATCH MODE	
1. Fission Gasses		QUARTER 1	QUARTER 2	QUARTER 1	QUARTER 2
Ar-41	Ci	0.00E+00	0.00E+00	7.50E-02	1.29E+00
Kr-85	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-85M	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-131M	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-133	Ci	2.18E-01	0.00E+00	1.87E-02	1.74E-01
Xe-133M	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135	Ci	0.00E+00	0.00E+00	6.80E-05	0.00E+00
Kr-87	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
TOTAL FOR PERIOD	Ci	2.18E-01	0.00E+00	9.37E-02	1.46E+00
2. Iodines					
I-131	Ci	7.88E-07	1.41E-05	0.00E+00	0.00E+00
I-133	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
TOTAL FOR PERIOD	Ci	7.88E-07	1.41E-05	0.00E+00	0.00E+00

\*Zeroes in this table indicate that no radioactivity was present above detectable levels. See Table 2-8 for typical lower limits of detection for gaseous sample analyses.

TABLE 2-3B (PAGE 2 OF 2)

VOGTLE ELECTRIC GENERATING PLANT

ANNUAL EFFLUENTS - MIXED MODE

JANUARY, 1993 THROUGH JUNE, 1993

UNIT 2

NUCLIDES RELEASED	UNIT	CONTINUOUS MODE		BATCH MODE	
		QUARTER 1	QUARTER 2	QUARTER 1	QUARTER 2
3. Particulates					
H-3	Ci	0.00E+00	1.34E+01	1.87E-02	7.48E-01
G-ALPHA	Ci	1.10E-05	5.54E-07	0.00E+00	0.00E+00
Na-24	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cr-51	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Co-58	Ci	1.54E-07	7.10E-07	0.00E+00	0.00E+00
Fe-59	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Nb-95	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Mn-54	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-89	Ci	1.43E-06	0.00E+00	0.00E+00	0.00E+00
Sr-90	Ci	1.50E-07	0.00E+00	0.00E+00	0.00E+00
Zr-95	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Co-60	Ci	2.44E-08	2.62E-07	0.00E+00	0.00E+00
<b>TOTAL</b>	Ci	1.28E-05	1.34E+01	1.87E-02	7.48E-01

\*Zeroes in this table indicate that no radioactivity was present above detectable levels. See Table 2-8 for typical lower limits of detection for gaseous sample analyses.

\*\* Half lives greater than 8 days.

**TABLE 2-3C (PAGE 1 OF 2)**

**VOGTLE ELECTRIC GENERATING PLANT**

**ANNUAL EFFLUENTS - MIXED MODE**

**JANUARY, 1993 THROUGH JUNE, 1993**

**SITE**

NUCLIDES RELEASED	UNIT	CONTINUOUS MODE		BATCH MODE	
		QUARTER 1	QUARTER 2	QUARTER 1	QUARTER 2
<b>1. Fission Gasses</b>					
Ar-41	Ci	0.00E+00	0.00E+00	4.47E-01	1.42E+00
Kr-85	Ci	0.00E+00	0.00E+00	5.00E-01	1.17E+00
Kr-85M	Ci	0.00E+00	0.00E+00	3.84E-03	0.00E+00
Xe-131M	Ci	0.00E+00	0.00E+00	1.86E-01	1.00E-01
Xe-133	Ci	6.88E+01	5.84E-01	1.34E+02	7.29E+00
Xe-133M	Ci	0.00E+00	0.00E+00	6.12E-01	2.40E-03
Xe-135	Ci	2.74E-01	0.00E+00	8.33E-02	1.44E-03
Kr-87	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
<b>TOTAL FOR PERIOD</b>	Ci	6.91E+01	5.84E-01	1.36E+02	9.99E+00
<b>2. Iodines</b>					
I-131	Ci	2.15E-04	2.40E-04	0.00E+00	0.00E+00
I-133	Ci	1.09E-05	1.63E-06	0.00E+00	0.00E+00
<b>TOTAL FOR PERIOD</b>	Ci	2.26E-04	2.42E-04	0.00E+00	0.00E+00

\*Zeroes in this table indicate that no radioactivity was present above detectable levels. See Table 2-8 for typical lower limits of detection for gaseous sample analyses.

**TABLE 2-3C (PAGE 2 OF 2)**  
**VOGTLE ELECTRIC GENERATING PLANT**  
**ANNUAL EFFLUENTS - MIXED MODE**  
**JANUARY, 1993 THROUGH JUNE, 1993**

**SITE**

NUCLIDES RELEASED	UNIT	CONTINUOUS MODE		BATCH MODE	
		QUARTER 1	QUARTER 2	QUARTER 1	QUARTER 2
3. Particulates					
H-3	Ci	4.98E+01	4.90E+01	3.83E-01	1.46E+01
G-ALPHA	Ci	1.29E-05	8.59E-06	0.00E+00	0.00E+00
Na-24	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cr-51	Ci	0.00E+00	1.29E-06	0.00E+00	0.00E+00
Co-58	Ci	3.41E-07	7.95E-06	0.00E+00	0.00E+00
Fe-59	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Nb-95	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Mn-54	Ci	0.00E+00	4.32E-07	0.00E+00	0.00E+00
Sr-89	Ci	2.35E-06	0.00E+00	0.00E+00	0.00E+00
Sr-90	Ci	2.52E-07	0.00E+00	0.00E+00	0.00E+00
Zr-95	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Co-60	Ci	2.44E-08	6.52E-07	0.00E+00	0.00E+00
Cs-134	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cs-137	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Co-57	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
<b>TOTAL</b>	Ci	4.98E+01	4.90E+01	3.83E-01	1.46E+01

\*Zeroes in this table indicate that no radioactivity was present above detectable levels. See Table 2-8 for typical lower limits of detection for gaseous sample analyses.

\*\* Half lives greater than 8 days.

TABLE 2-3AA (PAGE 1 OF 2)

VOGTLE ELECTRIC GENERATING PLANT

ANNUAL EFFLUENTS - MIXED MODE

JULY, 1993 THROUGH DECEMBER, 1993

UNIT 1

NUCLIDES RELEASED	UNIT	CONTINUOUS MODE		BATCH MODE	
1. Fission Gasses		QUARTER 3	QUARTER 4	QUARTER 3	QUARTER 4
Ar-41	Ci	0.00E+00	0.00E+00	1.57E-01	2.00E-01
Kr-85	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-85M	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-131M	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-133	Ci	0.00E+00	0.00E+00	1.54E-02	1.43E-01
Xe-133M	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135	Ci	0.00E+00	0.00E+00	0.00E+00	5.14E-04
Kr-87	Ci	0.00E+00	0.00E+00	0.00E+00	1.53E-04
TOTAL FOR PERIOD	Ci	0.00E+00	0.00E+00	1.72E-01	3.43E-01
2. Iodines					
I-131	Ci	4.42E-06	1.90E-06	0.00E+00	0.00E+00
I-133	Ci	1.64E-05	6.70E-06	0.00E+00	0.00E+00
TOTAL FOR PERIOD	Ci	2.08E-05	8.60E-06	0.00E+00	0.00E+00

\*Zeroes in this table indicate that no radioactivity was present above detectable levels. See Table 2-8 for typical lower limits of detection for gaseous sample analyses.

TABLE 2-3AA(PAGE 2 OF 2)

VOGTLE ELECTRIC GENERATING PLANT

ANNUAL EFFLUENTS - MIXED MODE

JULY, 1993 THROUGH DECEMBER , 1993

UNIT 1

NUCLIDES RELEASED	UNIT	CONTINUOUS MODE		BATCH MODE	
3. Particulates		QUARTER 3	QUARTER 4	QUARTER 3	QUARTER 4
I-131	Ci	3.06E+01	3.44E+01	6.54E-03	3.68E-02
U-235 ALPHA	Ci	1.08E-06	1.38E-06	0.00E+00	0.00E+00
Na-24	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cr-51	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Co-58	Ci	0.00E+00	4.60E-07	0.00E+00	0.00E+00
Fe-59	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Nb-95	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Mn-54	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-89	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-90	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Zr-95	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Po-210	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
TOTAL	Ci	3.06E+01	3.44E+01	6.54E-03	3.68E-02

\*Zeroes in this table indicate that no radioactivity was present above detectable levels. See Table 2-8 for typical lower limits of detection for gaseous sample analyses.

\*\* Half lives greater than 8 days.



TABLE 2-3BB (PAGE 1 OF 2)

## VOGTLE ELECTRIC GENERATING PLANT

## ANNUAL EFFLUENTS - MIXED MODE

JULY, 1993 THROUGH DECEMBER, 1993

## UNIT 2

NUCLIDES RELEASED	UNIT	CONTINUOUS MODE		BATCH MODE	
1. Fission Gasses		QUARTER 3	QUARTER 4	QUARTER 3	QUARTER 4
Ar-41	Ci	0.00E+00	0.00E+00	1.03E-01	2.10E-01
Kr-85	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-85M	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-131M	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-133	Ci	0.00E+00	2.60E+00	3.90E-03	9.64E-02
Xe-133M	Ci	0.00E+00	0.00E+00	0.00E+00	1.07E-03
Xe-135	Ci	0.00E+00	1.24E-01	3.56E-05	9.56E-05
Kr-87	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
TOTAL FOR PERIOD	Ci	0.00E+00	2.72E+00	1.07E-01	3.08E-01
2. Iodines					
-131	Ci	3.40E-07	5.42E-06	0.00E+00	0.00E+00
-133	Ci	0.00E+00	1.30E-05	0.00E+00	0.00E+00
TOTAL FOR PERIOD	Ci	3.40E-07	1.84E-05	0.00E+00	0.00E+00

\*Zeroes in this table indicate that no radioactivity was present above detectable levels. See Table 2-8 for typical lower limits of detection for gaseous sample analyses.

**TABLE 2-3BB (PAGE 2 OF 2)**  
**VOGTLE ELECTRIC GENERATING PLANT**  
**ANNUAL EFFLUENTS - MIXED MODE**  
**JULY, 1993 THROUGH DECEMBER, 1993**  
**UNIT 2**

NUCLIDES RELEASED	UNIT	CONTINUOUS MODE		BATCH MODE	
		QUARTER 3	QUARTER 4	QUARTER 3	QUARTER 4
3. Particulates					
H-3	Ci	3.44E+01	9.56E+00	3.48E-01	2.78E-01
G-ALPHA	Ci	4.30E-07	1.28E-06	0.00E+00	0.00E+00
Na-24	Ci	0.00E+00	2.92E-06	0.00E+00	0.00E+00
Cr-51	Ci	7.38E-07	1.11E-05	0.00E+00	0.00E+00
Co-58	Ci	1.13E-06	1.80E-05	0.00E+00	0.00E+00
Fe-59	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Nb-95	Ci	0.00E+00	1.08E-06	0.00E+00	0.00E+00
Mn-54	Ci	0.00E+00	1.43E-06	0.00E+00	0.00E+00
Sr-89	Ci	0.00E+00	1.39E-07	0.00E+00	0.00E+00
Sr-90	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Zr-95	Ci	0.00E+00	7.02E-07	0.00E+00	0.00E+00
Co-60	Ci	9.06E-07	4.84E-06	0.00E+00	0.00E+00
<b>TOTAL</b>	Ci	3.44E+01	9.56E+00	3.48E-01	2.78E-01

\*Zeroes in this table indicate that no radioactivity was present above detectable levels. See Table 2-8 for typical lower limits of detection for gaseous sample analyses.

\*\* Half lives greater than 8 days.



**TABLE 2-3CC (PAGE 1 OF 2)**

**VOGTLE ELECTRIC GENERATING PLANT**

**ANNUAL EFFLUENTS - MIXED MODE**

**JULY, 1993 THROUGH DECEMBER, 1993**

**SITE**

<b>NUCLIDES RELEASED</b>	<b>UNIT</b>	<b>CONTINUOUS MODE</b>		<b>BATCH MODE</b>	
<b>1. Fission Gasses</b>		<b>QUARTER 3</b>	<b>QUARTER 4</b>	<b>QUARTER 3</b>	<b>QUARTER 4</b>
Ar-41	Ci	0.00E+00	0.00E+00	2.60E-01	4.10E-01
Kr-85	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-85M	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-131M	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-133	Ci	0.00E+00	2.60E+00	1.93E-02	2.39E-01
Xe-133M	Ci	0.00E+00	0.00E+00	0.00E+00	1.07E-03
Xe-135	Ci	0.00E+00	1.24E-01	3.56E-05	6.10E-04
Kr-87	Ci	0.00E+00	0.00E+00	0.00E+00	1.53E-04
<b>TOTAL FOR PERIOD</b>	Ci	0.00E+00	2.72E+00	2.79E-01	6.50E-01
<b>2. Iodines</b>					
I-131	Ci	4.76E-06	7.32E-06	0.00E+00	0.00E+00
I-133	Ci	1.64E-05	1.97E-05	0.00E+00	0.00E+00
<b>TOTAL FOR PERIOD</b>	Ci	2.11E-05	2.70E-05	0.00E+00	0.00E+00

\*Zeroes in this table indicate that no radioactivity was present above detectable levels. See Table 2-8 for typical lower limits of detection for gaseous sample analyses.

**TABLE 2-3CC (PAGE 2 OF 2)**

**VOGTLE ELECTRIC GENERATING PLANT**

**ANNUAL EFFLUENTS - MIXED MODE**

**JULY, 1993 THROUGH DECEMBER, 1993**

**SITE**

NUCLIDES RELEASED	UNIT	CONTINUOUS MODE		BATCH MODE	
		QUARTER 3	QUARTER 4	QUARTER 3	QUARTER 4
3. Particulates					
H-3	Ci	6.50E+01	4.40E+01	3.55E-01	3.15E-01
G-ALPHA	Ci	1.51E-06	2.67E-06	0.00E+00	0.00E+00
Na-24	Ci	0.00E+00	2.92E-06	0.00E+00	0.00E+00
Cr-51	Ci	7.38E-07	1.11E-05	0.00E+00	0.00E+00
Co-58	Ci	1.13E-06	1.84E-05	0.00E+00	0.00E+00
Fe-59	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Nb-95	Ci	0.00E+00	1.08E-06	0.00E+00	0.00E+00
Mn-54	Ci	0.00E+00	1.43E-06	0.00E+00	0.00E+00
Sr-89	Ci	0.00E+00	1.39E-07	0.00E+00	0.00E+00
Sr-90	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Zr-95	Ci	0.00E+00	7.02E-07	0.00E+00	0.00E+00
Co-60	Ci	9.06E-07	4.84E-06	0.00E+00	0.00E+00
Cs-134	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cs-137	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Co-57	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
<b>TOTAL</b>	Ci	6.50E+01	4.40E+01	3.55E-01	3.15E-01

\*Zeroes in this table indicate that no radioactivity was present above detectable levels. See Table 2-8 for typical lower limits of detection for gaseous sample analyses.

\*\* Half lives greater than 8 days.

TABLE 2-4A (PAGE 1 OF 2)

VOGTLE ELECTRIC GENERATING PLANT

ANNUAL EFFLUENTS - GROUND MODE

JANUARY, 1993 THROUGH JUNE, 1993

UNIT 1

NUCLIDES RELEASED	UNIT	CONTINUOUS MODE		BATCH MODE	
1. Fission Gasses		QUARTER 1	QUARTER 2	QUARTER 1	QUARTER 2
Ar-41	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-85	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-85M	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-131M	Ci	0.00E+00	0.00E+00	2.82E-01	0.00E+00
Xe-133	Ci	0.00E+00	0.00E+00	1.48E+01	0.00E+00
Xe-133M	Ci	0.00E+00	0.00E+00	2.12E-01	0.00E+00
Xe-135	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-87	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
TOTAL FOR PERIOD	Ci	0.00E+00	0.00E+00	1.53E+01	0.00E+00
2. Iodines					
I-131	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-133	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
TOTAL FOR PERIOD	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00

\*Zeroes in this table indicate that no radioactivity was present above detectable levels. See Table 2-8 for typical lower limits of detection for gaseous sample analyses.

TABLE 2-4A (PAGE 2 OF 2)

VOGTLE ELECTRIC GENERATING PLANT

ANNUAL EFFLUENTS - GROUND MODE

JANUARY, 1993 THROUGH JUNE, 1993

UNIT 1

NUCLIDES BASED on Half-Lives	UNIT	CONTINUOUS MODE		BATCH MODE	
		QUARTER 1	QUARTER 2	QUARTER 1	QUARTER 2
H-3	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
G-ALPHA	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Na-24	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cr-51	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Co-58	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Co-59	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Nb-95	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Mn-54	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-89	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-90	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Zr-95	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Co-60	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
<b>TOTAL</b>	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00

\*Zeroes in this table indicate that no radioactivity was present above detectable levels. See Table 2-8 for typical lower limits of detection for gaseous sample analyses.

\*\* Half lives greater than 8 days.

TABLE 2-4B (PAGE 1 OF 2)

VOGTLE ELECTRIC GENERATING PLANT

ANNUAL EFFLUENTS - GROUND MODE

JANUARY, 1993 THROUGH JUNE, 1993

UNIT 2

NUCLIDES RELEASED	UNIT	CONTINUOUS MODE		BATCH MODE	
1. Fission Gasses		QUARTER 1	QUARTER 2	QUARTER 1	QUARTER 2
Ar-41	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-85	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-85M	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-131M	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-133	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-133M	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-87	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
TOTAL FOR PERIOD	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
2. Iodines					
I-131	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-133	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
TOTAL FOR PERIOD	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00

\*Zeroes in this table indicate that no radioactivity was present above detectable levels. See Table 2-8 for typical lower limits of detection for gaseous sample analyses.

TABLE 2-4B (PAGE 2 OF 2)

VOGTLE ELECTRIC GENERATING PLANT

ANNUAL EFFLUENTS - GROUND MODE

JANUARY, 1993 THROUGH JUNE, 1993

UNIT 2

NUCLIDES RELEASED	UNIT	CONTINUOUS MODE		BATCH MODE	
		QUARTER 1	QUARTER 2	QUARTER 1	QUARTER 2
3. Particulates					
H-3	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
G-ALPHA	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Na-24	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cr-51	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Co-58	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Fe-59	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Nb-95	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Mn-54	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-89	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-90	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Zr-95	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Co-60	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
<b>TOTAL</b>	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00

\*Zeroes in this table indicate that no radioactivity was present above detectable levels. See Table 2-8 for typical lower limits of detection for gaseous sample analyses.

\*\* Half lives greater than 8 days.



TABLE 2-4C (PAGE 1 OF 2)

## VOGTLE ELECTRIC GENERATING PLANT

## ANNUAL EFFLUENTS - GROUND MODE

JANUARY, 1993 THROUGH JUNE, 1993

## SITE

NUCLIDES RELEASED	UNIT	CONTINUOUS MODE		BATCH MODE	
		QUARTER 1	QUARTER 2	QUARTER 1	QUARTER 2
<b>1. Fission Gasses</b>					
Ar-41	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-85	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-85M	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-131M	Ci	0.00E+00	0.00E+00	2.82E-01	0.00E+00
Xe-133	Ci	0.00E+00	0.00E+00	1.48E+01	0.00E+00
Xe-133M	Ci	0.00E+00	0.00E+00	2.12E-01	0.00E+00
Xe-135	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-87	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
<b>TOTAL FOR PERIOD</b>	Ci	0.00E+00	0.00E+00	1.53E+01	0.00E+00
<b>2. Iodines</b>					
I-131	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-133	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
<b>TOTAL FOR PERIOD</b>	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00

\*Zeroes in this table indicate that no radioactivity was present above detectable levels. See Table 2-8 for typical lower limits of detection for gaseous sample analyses.

**TABLE 2-4C(PAGE 2 OF 2)**

**VOGTLE ELECTRIC GENERATING PLANT**

**ANNUAL EFFLUENTS - GROUND MODE**

**JANUARY, 1993 THROUGH JUNE , 1993**

**SITE**

NUCLIDES RELEASED	UNIT	CONTINUOUS MODE		BATCH MODE	
		QUARTER 1	QUARTER 2	QUARTER 1	QUARTER 2
3. Particulates					
H-3	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
G-ALPHA	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Na-24	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cr-51	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Co-58	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Fe-59	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Nb-95	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Mn-54	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-89	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-90	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Zr-95	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Co-60	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cs-134	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cs-137	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Co-57	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
<b>TOTAL</b>	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00

\*Zeroes in this table indicate that no radioactivity was present above detectable levels. See Table 2-8 for typical lower limits of detection for gaseous sample analyses.

\*\* Half lives greater than 8 days.



TABLE 2-4AA (PAGE 1 OF 2)

VOGTLE ELECTRIC GENERATING PLANT

ANNUAL EFFLUENTS - GROUND MODE

JULY, 1993 THROUGH DECEMBER, 1993

UNIT 1

NUCLIDES RELEASED	UNIT	CONTINUOUS MODE		BATCH MODE	
		QUARTER 3	QUARTER 4	QUARTER 3	QUARTER 4
<b>1. Fission Gasses</b>					
Ar-41	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-85	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-85M	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-131M	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-133	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-133M	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-87	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
<b>TOTAL FOR PERIOD</b>	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
<b>2. Iodines</b>					
I-131	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
I-133	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
<b>TOTAL FOR PERIOD</b>	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00

\*Zeroes in this table indicate that no radioactivity was present above detectable levels. See Table 2-8 for typical lower limits of detection for gaseous sample analyses.

TABLE 2-4AA (PAGE 2 OF 2)

VOGTLE ELECTRIC GENERATING PLANT

ANNUAL EFFLUENTS - GROUND MODE

JULY, 1993 THROUGH DECEMBER, 1993

UNIT 1

NUCLIDES RELEASED	UNIT	CONTINUOUS MODE		BATCH MODE	
		QUARTER 3	QUARTER 4	QUARTER 3	QUARTER 4
3. Particulates					
H-3	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
G-ALPHA	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Na-24	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cr-51	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Co-58	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Fe-59	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Nb-95	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Mn-54	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-89	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-90	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Zr-95	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Co-60	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
TOTAL	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00

\*Zeroes in this table indicate that no radioactivity was present above detectable levels. See Table 2-8 for typical lower limits of detection for gaseous sample analyses.

\*\* Half lives greater than 8 days.

**TABLE 2-4BB (PAGE 1 OF 2)**

**VOGTLE ELECTRIC GENERATING PLANT**

**ANNUAL EFFLUENTS - GROUND MODE**

**JULY, 1993 THROUGH DECEMBER, 1993**

**UNIT 2**

<b>NUCLIDES RELEASED</b>	<b>UNIT</b>	<b>CONTINUOUS MODE</b>		<b>BATCH MODE</b>	
<b>1. Fission Gasses</b>		<b>QUARTER 3</b>	<b>QUARTER 4</b>	<b>QUARTER 3</b>	<b>QUARTER 4</b>
Ar-41	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-85	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-85M	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-131M	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-133	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-133M	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Xe-135	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Kr-87	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
<b>TOTAL FOR PERIOD</b>	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
<b>2. Iodines</b>					
I-131	Ci	0.00E+00	0.00E+00	9.73E-10	6.54E-09
I-133	Ci	0.00E+00	0.00E+00	3.30E-07	0.00E+00
<b>TOTAL FOR PERIOD</b>	Ci	0.00E+00	0.00E+00	3.31E-07	6.54E-09

\*Zeroes in this table indicate that no radioactivity was present above detectable levels. See Table 2-8 for typical lower limits of detection for gaseous sample analyses.

**TABLE 2-4BB(PAGE 2 OF 2)**

**VOGTLE ELECTRIC GENERATING PLANT**

**ANNUAL EFFLUENTS - GROUND MODE**

**JULY, 1993 THROUGH DECEMBER, 1993**

**UNIT 2**

NUCLIDES RELEASED	UNIT	CONTINUOUS MODE		BATCH MODE	
		QUARTER 3	QUARTER 4	QUARTER 3	QUARTER 4
3. Particulates					
H-3	Ci	0.00E+00	0.00E+00	6.00E-03	3.30E-02
G-ALPHA	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Na-24	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cr-51	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Co-58	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Fe-59	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Nb-95	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Mn-54	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-89	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-90	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Zr-95	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Co-60	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
<b>TOTAL</b>	Ci	0.00E+00	0.00E+00	6.00E-03	3.30E-02

\*Zeroes in this table indicate that no radioactivity was present above detectable levels. See Table 2-8 for typical lower limits of detection for gaseous sample analyses.

\*\* Half lives greater than 8 days.

TABLE 2-4CC (PAGE 1 OF 2)

## VOGTLE ELECTRIC GENERATING PLANT

## ANNUAL EFFLUENTS - GROUND MODE

JULY, 1993 THROUGH DECEMBER, 1993

## SITE

NUCLIDES RELEASED	UNIT	CONTINUOUS MODE		BATCH MODE	
		QUARTER 3	QUARTER 4	QUARTER 3	QUARTER 4
<i>Fission Gasses</i>					
-41	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
-85	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
-85M	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
-131M	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
-133	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
-133M	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
-135	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
-87	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
TOTAL FOR PERIOD	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
<i>Iodines</i>					
31	Ci	0.00E+00	0.00E+00	9.73E-10	6.54E-09
33	Ci	0.00E+00	0.00E+00	3.30E-07	0.00E+00
TOTAL FOR PERIOD	Ci	0.00E+00	0.00E+00	3.31E-07	6.54E-09

Zeroes in this table indicate that no radioactivity was present above detectable levels. See Table 2-8 for typical lower limits of detection for gaseous sample analyses.

**TABLE 2-4CC (PAGE 2 OF 2)**

**VOGTLE ELECTRIC GENERATING PLANT**

**ANNUAL EFFLUENTS - GROUND MODE**

**JULY, 1993 THROUGH DECEMBER, 1993**

**SITE**

<b>NUCLIDES RELEASED</b>	<b>UNIT</b>	<b>CONTINUOUS MODE</b>		<b>BATCH MODE</b>	
<b>3. Particulates</b>		<b>QUARTER 3</b>	<b>QUARTER 4</b>	<b>QUARTER 3</b>	<b>QUARTER 4</b>
H-3	Ci	0.00E+00	0.00E+00	6.00E-03	3.30E-02
G-ALPHA	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Na-24	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cr-51	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Co-58	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Fe-59	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Nb-95	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Mn-54	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-89	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-90	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Zr-95	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Co-60	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cs-134	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Cs-137	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Co-57	Ci	0.00E+00	0.00E+00	0.00E+00	0.00E+00
<b>TOTAL</b>	Ci	0.00E+00	0.00E+00	6.00E-03	3.30E-02

\*Zeroes in this table indicate that no radioactivity was present above detectable levels. See Table 2-8 for typical lower limits of detection for gaseous sample analyses.

\*\* Half lives greater than 8 days.



**TABLE 2-6A**  
**VOGTLE ELECTRIC GENERATING PLANT**  
**SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT**  
**AIR DOSE DUE TO NOBLE GAS RELEASES**  
**JANUARY, 1993 THROUGH JUNE, 1993**

**UNIT 1**

**CUMMULATIVE DOSE PER QUARTER**

TYPE OF RADIATION	ODCM LIMIT	UNITS	QUARTER 1	% OF ODCM LIMIT	QUARTER 2	% OF ODCM LIMIT
GAMMA	5.0	mrads	1.54E-03	3.08E-02	5.81E-05	1.16E-03
BETA	10.0	mrads	4.48E-03	4.48E-02	1.60E-04	1.60E-03

**CUMULATIVE DOSES PER YEAR (YEAR TO DATE)**

TYPE OF RADIATION	ODCM LIMIT	UNITS	YEAR TO DATE	% OF ODCM LIMIT
GAMMA	10.0	mrads	1.60E-03	1.60E-02
BETA	20.0	mrads	4.64E-03	2.32E-02

**TABLE 2-6B**  
**VOGTLE ELECTRIC GENERATING PLANT**  
**SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT**  
**AIR DOSE DUE TO NOBLE GAS RELEASES**  
**JANUARY, 1993 THROUGH JUNE, 1993**

**UNIT 2**

**CUMMULATIVE DOSE PER QUARTER**

TYPE OF RADIATION	ODCM LIMIT	UNITS	QUARTER 1	% OF ODCM LIMIT	QUARTER 2	% OF ODCM LIMIT
GAMMA	5.0	mrads	1.15E-05	2.30E-04	1.77E-04	3.54E-03
BETA	10.0	mrads	7.27E-06	7.27E-05	6.46E-05	6.46E-04

**CUMULATIVE DOSES PER YEAR (YEAR TO DATE)**

TYPE OF RADIATION	ODCM LIMIT	UNITS	YEAR TO DATE	% OF ODCM LIMIT
GAMMA	10.0	mrads	1.89E-04	1.89E-03
BETA	20.0	mrads	7.19E-05	3.60E-04



**TABLE 2-6AA**  
**VOGTLE ELECTRIC GENERATING PLANT**  
**SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT**  
**AIR DOSE DUE TO NOBLE GAS RELEASES**  
**JULY, 1993 THROUGH DECEMBER, 1993**

**UNIT 1**

**CUMMULATIVE DOSE PER QUARTER**

TYPE OF RADIATION	ODCM LIMIT	UNITS	QUARTER 3	% OF ODCM LIMIT	QUARTER 4	% OF ODCM LIMIT
GAMMA	5.0	mrads	2.15E-05	4.30E-04	2.80E-05	5.60E-04
BETA	10.0	mrads	7.78E-06	7.78E-05	1.18E-05	1.18E-04

**CUMULATIVE DOSES PER YEAR (YEAR TO DATE)**

TYPE OF RADIATION	ODCM LIMIT	UNITS	YEAR TO DATE	% OF ODCM LIMIT
GAMMA	10.0	mrads	1.65E-03	1.65E-02
BETA	20.0	mrads	4.66E-03	2.33E-02

**TABLE 2-6BB**  
**VOGTLE ELECTRIC GENERATING PLANT**  
**SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT**  
**AIR DOSE DUE TO NOBLE GAS RELEASES**  
**JULY, 1993 THROUGH DECEMBER, 1993**

**UNIT 2**

**CUMMULATIVE DOSE PER QUARTER**

TYPE OF RADIATION	ODCM LIMIT	UNITS	QUARTER 3	% OF ODCM LIMIT	QUARTER 4	% OF ODCM LIMIT
GAMMA	5.0	mrads	1.40E-05	2.30E-04	5.01E-06	1.00E-04
BETA	10.0	mrads	4.62E-05	4.62E-04	5.64E-05	5.64E-04

**CUMULATIVE DOSES PER YEAR (YEAR TO DATE)**

TYPE OF RADIATION	ODCM LIMIT	UNITS	YEAR TO DATE	% OF ODCM LIMIT
GAMMA	10.0	mrads	2.08E-04	2.08E-03
BETA	20.0	mrads	1.75E-04	8.75E-04

TABLE 2-7A  
VOGTLE ELECTRIC GENERATING PLANT  
ANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT  
INDIVIDUAL DOSES DUE TO RADIOIODINE, TRITIUM  
JANUARY, 1993 THROUGH JUNE, 1993

UNIT 1

CUMULATIVE DOSE PER QUARTER

ORGAN	ODCM LIMIT	UNITS	QUARTER 1	% OF ODCM LIMIT	QUARTER 2	% OF ODCM LIMIT
Bone	7.5	mrem	2.05E-07	2.73E-06	5.77E-07	7.69E-06
Liver	7.5	mrem	3.03E-04	4.04E-03	2.25E-04	3.00E-03
T. Body	7.5	mrem	3.03E-04	4.04E-03	2.25E-04	3.00E-03
Thyroid	7.5	mrem	3.03E-04	4.04E-03	2.25E-04	3.00E-03
Kidney	7.5	mrem	3.03E-04	4.04E-03	2.25E-04	3.00E-03
Lung	7.5	mrem	3.03E-04	4.04E-03	2.25E-04	3.00E-03
GI-LLI	7.5	mrem	3.03E-04	4.04E-03	2.25E-04	3.00E-03

CUMULATIVE DOSE PER YEAR

ORGAN	ODCM LIMIT	UNITS	YEAR TO DATE	% of ODCM LIMIT
Bone	15.0	mrem	7.82E-07	5.21E-06
Liver	15.0	mrem	5.28E-04	3.52E-03
T. Body	15.0	mrem	5.28E-04	3.52E-03
Thyroid	15.0	mrem	5.28E-04	3.52E-03
Kidney	15.0	mrem	5.28E-04	3.52E-03
Lung	15.0	mrem	5.28E-04	3.52E-03
GI-LLI	15.0	mrem	5.28E-04	3.52E-03

**TABLE 2-7B**  
**VOGTLE ELECTRIC GENERATING PLANT**  
**ANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT**  
**INDIVIDUAL DOSES DUE TO RADIOIODINE, TRITIUM**  
**JANUARY, 1993 THROUGH JUNE, 1993**

**UNIT 2**

**CUMULATIVE DOSE PER QUARTER**

ORGAN	ODCM LIMIT	UNITS	QUARTER 1	% OF ODCM LIMIT	QUARTER 2	% OF ODCM LIMIT
Bone	7.5	mrem	6.69E-08	8.92E-07	2.25E-07	3.00E-06
Liver	7.5	mrem	8.71E-08	1.16E-06	6.40E-05	8.53E-04
T. Body	7.5	mrem	9.10E-08	1.21E-06	6.40E-05	8.53E-04
Thyroid	7.5	mrem	1.39E-07	1.85E-06	6.40E-05	8.53E-04
Kidney	7.5	mrem	9.96E-08	1.33E-06	6.40E-05	8.53E-04
Lung	7.5	mrem	9.75E-08	1.30E-06	6.40E-05	8.53E-04
GI-LLI	7.5	mrem	8.78E-08	1.17E-06	6.40E-05	8.53E-04

**CUMULATIVE DOSE PER YEAR**

ORGAN	ODCM LIMIT	UNITS	YEAR TO DATE	% of ODCM LIMIT
Bone	15.0	mrem	2.92E-07	1.95E-06
Liver	15.0	mrem	6.41E-05	4.27E-04
T. Body	15.0	mrem	6.41E-05	4.27E-04
Thyroid	15.0	mrem	6.41E-05	4.27E-04
Kidney	15.0	mrem	6.41E-05	4.27E-04
Lung	15.0	mrem	6.41E-05	4.27E-04
GI-LLI	15.0	mrem	6.41E-05	4.27E-04

**TABLE 2-7AA**  
**VOGTLE ELECTRIC GENERATING PLANT**  
**ANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT**  
**INDIVIDUAL DOSES DUE TO RADIOIODINE, TRITIUM**  
**JULY, 1993 THROUGH DECEMBER, 1993**

**UNIT 1**

**CUMULATIVE DOSE PER QUARTER**

ORGAN	ODCM LIMIT	UNITS	QUARTER 3	% OF ODCM LIMIT	QUARTER 4	% OF ODCM LIMIT
Bone	7.5	mrem	5.60E-09	7.47E-08	7.82E-09	1.04E-07
Liver	7.5	mrem	1.39E-04	1.85E-03	1.56E-04	2.08E-03
T. Body	7.5	mrem	1.39E-04	1.85E-03	1.56E-04	2.08E-03
Thyroid	7.5	mrem	1.40E-04	1.87E-03	1.56E-04	2.08E-03
Kidney	7.5	mrem	1.39E-04	1.85E-03	1.56E-04	2.08E-03
Lung	7.5	mrem	1.39E-04	1.85E-03	1.56E-04	2.08E-03
GI-LLI	7.5	mrem	1.39E-04	1.85E-03	1.56E-04	2.08E-03

**CUMULATIVE DOSE PER YEAR**

ORGAN	ODCM LIMIT	UNITS	YEAR TO DATE	% of ODCM LIMIT
Bone	15.0	mrem	7.95E-07	5.30E-06
Liver	15.0	mrem	8.23E-04	5.49E-03
T. Body	15.0	mrem	8.23E-04	5.49E-03
Thyroid	15.0	mrem	8.24E-04	5.49E-03
Kidney	15.0	mrem	8.23E-04	5.49E-03
Lung	15.0	mrem	8.23E-04	5.49E-03
GI-LLI	15.0	mrem	8.23E-04	5.49E-03

**TABLE 2-7BB**  
**VOGTLE ELECTRIC GENERATING PLANT**  
**ANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT**  
**INDIVIDUAL DOSES DUE TO RADIOIODINE, TRITIUM**  
**JULY, 1993 THROUGH DECEMBER, 1993**

**UNIT 2**

**CUMULATIVE DOSE PER QUARTER**

ORGAN	ODCM LIMIT	UNITS	QUARTER 3	% OF ODCM LIMIT	QUARTER 4	% OF ODCM LIMIT
Bone	7.5	mrem	7.25E-07	9.67E-06	4.10E-06	5.47E-05
Liver	7.5	mrem	1.58E-04	2.11E-03	4.94E-05	6.59E-04
T. Body	7.5	mrem	1.58E-04	2.11E-03	4.94E-05	6.59E-04
Thyroid	7.5	mrem	1.58E-04	2.11E-03	4.96E-05	6.61E-04
Kidney	7.5	mrem	1.58E-04	2.11E-03	4.94E-05	6.59E-04
Lung	7.5	mrem	1.58E-04	2.11E-03	4.97E-05	6.63E-04
GI-LLI	7.5	mrem	1.58E-04	2.11E-03	4.94E-05	6.59E-04

**CUMULATIVE DOSE PER YEAR**

ORGAN	ODCM LIMIT	UNITS	YEAR TO DATE	% of ODCM LIMIT
Bone	15.0	mrem	5.12E-06	3.41E-05
Liver	15.0	mrem	2.71E-04	1.81E-03
T. Body	15.0	mrem	2.71E-04	1.81E-03
Thyroid	15.0	mrem	2.73E-04	1.82E-03
Kidney	15.0	mrem	2.71E-04	1.81E-03
Lung	15.0	mrem	2.72E-04	1.81E-03
GI-LLI	15.0	mrem	2.71E-04	1.81E-03

**TABLE 2-8**  
**LOWER LIMITS OF DETECTION - GASEOUS SAMPLE ANALYSES**  
**VOGTLE ELECTRIC GENERATING PLANT**  
**JANUARY, 1993 THROUGH DECEMBER, 1993**

The values in this table represent a priori lower limits of detection (LLD) which are typically achieved in laboratory analyses of gaseous radwaste samples.

<b>RADIONUCLIDE</b>	<b>LLD</b>	<b>UNITS</b>
kR-87	1.82E-08	uCi/cc
Kr-88	2.53E-08	uCi/cc
Xe-133	2.05E-08	uCi/cc
Xe-133m	8.63E-08	uCi/cc
Xe-135	7.12E-08	uCi/cc
Xe-138	1.05E-07	uCi/cc
I-131	7.93E-15*	uCi/cc
Mn-54	3.94E-14*	uCi/cc
Fe-59	2.45E-14*	uCi/cc
Co-58	1.39E-14*	uCi/cc
Co-60	1.75E-14*	uCi/cc
Zn-65	2.82E-14*	uCi/cc
Mo-99	9.57E-14*	uCi/cc
Cs-134	1.12E-14*	uCi/cc
Cs-137	8.71E-15*	uCi/cc
Ce-141	8.62E-15*	uCi/cc
Ce-144	2.77E-14*	uCi/cc
Sr-89	1.00E-13	uCi/cc
Sr-90	1.00E-13	uCi/cc
H-3	9.00E-08	uCi/cc
Gross Alpha	1.00E-13	uCi/cc

\* Based on an estimated sample volume of 5.7E-08 cc's.



### **3.0 Solid Waste**

#### **3.1 Regulatory Limits/ODCM**

The ODCM Limits presented in this section are for Unit 1 and Unit 2 and are stated in part.

##### **3.1.1 Use of Solid Radioactive Waste System**

###### **10.2.1 Process Control Program**

Radioactive wastes shall be solidified or dewatered in accordance with the Process Control Program to meet shipping and transportation requirements during the transit, and disposal site requirements when received at the disposal site.

##### **3.1.2 Reporting Requirements**

###### **6.8.1.4**

The Annual Radioactive Effluent Release Report covering the operation of the units during the previous calendar year shall be submitted before May 1 of each year. The report shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from the unit. The material provided shall be (1) consistent with the objectives outlined in the ODCM and PCP and (2) in conformance with 10 CFR 50.36a and Section IV.B.1 of Appendix I to 10 CFR, Part 50.

##### **3.1.3 Process Control Program (PCP)**

###### **6.12.2**

Licensee - initiated changes to the PCP

Shall be submitted to the Commission in the Radioactive Effluent Release Report for the period in which the change(s) was made.

PCP was revised as part of Generic Letter 89-01 conversion. Now the PCP is a stand alone document. The revised copy of PCP is attached as part of reporting requirements.



### **3.2 Solid Waste Data**

Regulatory Guide 1.21, Table 3 is found in this report as Table 3-1.

### **4.0 Changes to the Vogtle Electric Generating Plant ODCM**

#### **6.13.2**

There were changes to the Vogtle Electric Generating Plant ODCM for the period January 1, 1993 through December 31, 1993.

As part of the generic letter 89-01 requirements Radiological Effluent Technical Specifications (RETS) have been taken out of the Vogtle Technical Specifications and put in the current ODCM. All the changes were done in accordance with the requirements of 89-01. The revised copy of the ODCM was reviewed by the commission and approved by the commission before the document was put in place. A copy of the same ODCM is attached as part of the reporting requirements.

#### **3.0 of ODCM**

The Radiological Environmental Monitoring Program shall be conducted as specified in ODCM 3.1.1 and 3.1.2.

Table Notation (1) states in part:

It is recognized that, at times, it may not be possible or practicable to continue to obtain samples of the media of choice at the most desired location or time. In these instances, suitable alternative media and locations may be chosen for the particular pathway in questions and appropriate substitutions, if available, will be made within 30 days in the Radiological Environmental Monitoring Program given in the ODCM.

Pursuant to specification 6.13, submit in the next Radioactive Effluent Release Report documentation for a change in the ODCM including a revised figure(s) and Table for the ODCM reflecting the new location(s), if any, with supporting information identifying the cause of the unavailability of samples for the pathway and justifying the selection of the new location(s) for obtaining samples, or the unavailability of suitable new locations.

3.1.2 of ODCM states in part:

A land Use Census shall be conducted . . . . .

The Action Statement for this requirement states in part:

- a. With a Land Use Census identifying a location(s) that yields a calculated dose or dose commitment greater than the value currently being calculated in ODCM 3.12.1 pursuant to specification 6.13. Identify the new location(s) in the next Annual Radioactive Effluent Release Report.

#### **4.1 Changes in the Radiological Environmental Monitoring Program**

For this reporting period, there has been no changes to the Radiological Environmental Monitoring Program.

#### **5.0 Doses to Members of the Public Inside the Site Boundary**

6.8.1.4 states in part:

This same report shall also include assessment of the radiation doses from radioactive liquid and gaseous effluents to MEMBERS OF THE PUBLIC due to their activities inside the Site Boundary during the report period. All assumptions used in making these assessments, i.e., specific activity, exposure time, and location shall be included in these reports.

The location of concern within the site boundary are the Visitors Center and Plant Wilson. The activities at the Visitor Center consists of the occasional attendance at meetings and/or short visits for informational purposes. The activity at Plant Wilson consists of regular employment. There will be no radiation dose at these locations due to radioactive liquid effluents. Delineated in Table 5-1 for each of these locations are the values of the basic data assumed in the dose assessment due to radioactive gaseous effluents. Listed in this table are: The distance and directions from a point midway between the center of Unit 1 and the Unit 2 reactors, the dispersion and deposition factors for any releases from the plant vent (mixed mode) and from the turbine building (ground level); and the estimated maximum occupancy factor for an individual and the assumed age group of this individual.

The source term is not listed in Table 5-1. The source term is listed in Tables 2-3A, 2-3AA, 2-3B, and 2-3BB for the mixed mode releases. Similarly, it is listed in Tables 2-4A, 2-4AA, 2-4B, and 2-4BB for the ground level releases. The tritium releases in units of curies were as follows:

ARTER	1	2	3	4
ed Mode	5.02E+01	6.36E+01	6.54E+01	4.43E+01
ound Mode	0.00E+00	0.00E+00	6.00E-03	3.30E-02

e maximum doses in units of mrem accumulated by an individual MEMBER  
 THE PUBLIC due to their activities inside the site boundary during  
 reporting period were assessed to be as follows:

VISITOR CENTER

Quarter 1 and 2      Quarter 3 and 4

tal Body  
 irect Radiation from Plume)

1.21E-06

7.25E-08

PLANT WILSON

Quarter 1 and 2      Quarter 3 and 4

otal Body  
 irect Radiation from Plume)

1.31E-04

8.89E-06

VISITOR CENTER

Quarter 1      Quarter 2      Quarter 3      Quarter 4

aximum Organ  
 hyroid)  
 on Ground Plane)

5.91E-07

7.45E-07

7.29E-07

5.08E-07

PLANT WILSON

Quarter 1      Quarter 2      Quarter 3      Quarter 4

aximum Organ  
 hyroid)  
 on Ground Plane)

7.59E-05

9.61E-05

9.68E-05  
(Liver)

6.67E-05

Inoperable Tech Spec monitors are tracked on Limiting Condition of Operation (LCO) Forms. The operators declare equipment operable and inoperable and monitors are considered inoperable if there are open LCO's for that monitor.

1 The LCO's initiated do not have to be reported for this report period since the LCO's initiated were closed before 30 day period.

## 0 Tanks Exceeding Curie Content Limits

The Annual Radioactive Effluent Release Reports shall also include the following "and description of the events leading to liquid holdup tanks or gas storage tanks exceeding the limits of specifications 3.11.1.4 or 3.11.2.6, respectively" . . .

There were no outside temporary liquid tanks for radioactive liquids during this reporting period. The radioactive material contained in each waste gas decay tank did not exceed 2E5 curies of noble gases (considered as Xe-133 equivalent).

VOGTLE ELECTRIC GENERATING PLANT  
OFFSITE DOSE CALCULATION MANUAL

REVISION LOG

<u>Revision No.</u>	<u>Date</u>
0	8/85
1	8/86
2	11/86
3	3/87
4	7/87
5	10/87
6	9/88
7	3/91
8	2/93

## INTRODUCTION

The Offsite Dose Calculation Manual is a supporting document of the Technical Specifications. As such, the ODCM describes the methodology and parameters to be used in the calculation of offsite doses due to radioactive liquid and gaseous effluents and in the calculation of liquid and gaseous effluent monitoring instrumentation alarm/trip setpoints. The ODCM contains schematics of liquid and gaseous radwaste effluent treatment systems, which include release points to unrestricted areas. It includes a list and maps indicating specific sample locations for the radiological environmental monitoring program. It also includes the radioactive effluent controls and radiological environmental monitoring programs required by section 6.7.4 of the Technical Specifications and descriptions of the information that should be included in the Annual Radiological Environmental Surveillance and Semiannual Radioactive Effluent Release Reports required by specifications 6.8.1.3 and 6.8.1.4.

The ODCM will be maintained at the plant for use as a reference guide and training document of accepted methodologies and calculations. Changes in the calculational methods or parameters will be incorporated into the ODCM in order to assure that the ODCM represents current methodology in all applicable areas. Computer software to perform the described calculations will be maintained current with the ODCM.

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## 1.0 LIQUID EFFLUENTS

The Vogtle Electric Generating Plant is located on the west bank of the Savannah River approximately 151 river miles from the Atlantic Ocean. There are two pressurized water reactors on the site. Each unit is served by a separate liquid waste processing system; however, certain components are shared between the two systems. All liquid radwastes treated by the liquid waste processing system are collected in waste monitor tanks for sampling and analysis prior to release. The 5000-gallon waste monitor tanks are recirculated for a minimum of 30 minutes, and the 20,000-gallon waste monitor tanks are recirculated for a minimum of 45 minutes. This mixing method assures that a representative sample can be taken from the tank. Releases from the waste monitor tanks are to the discharge line from the blowdown sump to the Savannah River. The blowdown sump receives input from the waste water retention basins, turbine plant cooling tower blowdown, and nuclear service cooling tower blowdown. Additional dilution water is available from the cooling tower makeup water bypass line.

Although no significant quantities of radioactivity are expected in the nuclear service cooling water, the steam generator blowdown processing system or the turbine building drain system, these effluent pathways are monitored as a precautionary measure. The monitors serving the latter two effluent pathways provide for automatic termination of release from these systems in the event radioactivity is detected above predetermined levels. These two systems discharge to the waste water-retention basin. Sampling and analysis of releases via these effluent pathways must be sufficient to assure the dose limits specified in subsection 1.5.2 are not exceeded.

Section 1.0 of the ODCM describes the methodology for calculating monitor setpoints and individual doses due to liquid effluents released from Plant Vogtle to the Savannah River. Schematics of the liquid waste processing systems are presented in figures 1.6-1 and 1.6-2. Liquid discharge pathways are shown in figure 1.6-3.



## 1.1 LIQUID EFFLUENT MONITOR SETPOINTS

Liquid monitor setpoint values calculated in accordance with the methodology presented in this section will be regarded as upper bounds for the actual monitor setpoints for high alarms. However, a lower setpoint may be established on the monitor if desired. Intermediate level setpoints should be established at an appropriate level to give sufficient warning prior to reaching the high alarm setpoint. The basic calculated monitor setpoint value is in terms of concentration,  $\mu\text{Ci/ml}$ . Monitor calibration data may include operational data obtained from monitor response to concentrations determined by liquid sample analyses. In addition, monitor background must be controlled so that the monitor is capable of responding to concentrations in the range of the setpoint value.

For planned releases from the liquid waste processing system's monitor tanks, monitor setpoints are determined to assure that the limits of 10 CFR 20 are not exceeded. For the steam generator processing system effluent line, and the turbine building drain effluent line, the purpose of the monitor setpoints is to minimize releases of radioactivity from these systems by terminating releases upon detection of low levels of radioactivity.

### 1.1.1 Liquid Waste Processing System Effluent Monitor (1(2)RE0018) (one monitor per unit)

The liquid waste processing system effluent line radioactivity monitors provide alarm and automatic termination of release prior to exceeding the concentration limits specified in 10 CFR 20, Appendix B, Table II, Column 2,



at the release point to the unrestricted area. Concentration limits are specified in subsection 1.5.1; setpoint requirements are specified in subsection 1.5.4. To meet these specifications, the alarm/trip setpoint for this liquid effluent monitor is set to assure that the following equation is satisfied:

$$\frac{cf}{F + f} \leq C_{MPC} \quad (1)$$

where:

- $C_{MPC}$  = the effluent concentration limit corresponding to the specific mix of radionuclides in the waste monitor tank being considered for discharge, in  $\mu\text{Ci/ml}$ .
- $c$  = the setpoint, in  $\mu\text{Ci/ml}$ , of the radioactivity monitor measuring the concentration of radioactivity in the effluent line prior to dilution and subsequent release. (Note that the monitor setpoint is inversely proportional to the effluent flowrate,  $f$ , and directly proportional to the dilution stream flowrate,  $F + f$ .) The setpoint represents a concentration value which, if exceeded, could result in concentrations exceeding the limits of 10 CFR 20 in the unrestricted area.
- $f$  = the effluent flowrate at the location of the radioactivity monitor, in volume per unit time, and in the same units as  $F$ , below.
- $F$  = the dilution stream flowrate which can be assured prior to the release point to the river, in volume per unit time.

At Plant Vogtle, the liquid waste processing system collects liquid wastes in monitor tanks prior to release. There are two waste monitor tanks for each unit. The discharge lines from the two tanks join to form a common line, on which the radioactivity monitor is installed. The lines from each unit then join to form a common line which releases to the blowdown sump discharge line to the Savannah River. (See figure 1.6-3.)

Dilution flow comes from the blowdown sump which receives water from nuclear service cooling tower blowdown, turbine plant cooling tower blowdown, waste water retention basin discharge, and the cooling tower makeup line. The two major sources for dilution are the turbine plant cooling tower blowdown and the cooling tower makeup bypass line. A predetermined dilution flowrate must be assured for use in the calculation of the radioactivity monitor setpoint.

While equation (1) shows the relationships between the limiting concentration,  $C_{MPC}$ , the effluent flowrate,  $f$ , the dilution flowrate,  $F$ , and the radioactivity monitor setpoint, it cannot practically be applied to a mixture of radionuclides with different limiting concentrations, i.e., different MPC values.

For a mixture of radionuclides, equation (1) is satisfied in a practicable manner, based on measured radionuclide concentrations and a dilution stream flowrate which can be assured for the duration of the release, designated as  $F_d$ , by calculating the MPC fraction for the radionuclide mixture, the maximum permissible effluent flowrate,  $f_m$ , and the radioactivity monitor setpoint,  $c$ .

In order to facilitate effluent release control and accountability, liquid releases normally should be controlled administratively such that only one waste monitor tank per unit is released at a time. Paragraph 1.1.1.1 presents the methodology for calculating the monitor setpoint for this situation. In the event it becomes necessary to release both waste monitor tanks, of the same unit, at the same time, the methodology for calculating the monitor setpoint is more complex. This increased complexity is due to the fact that the two waste monitor tanks discharge through a common line served by a single monitor. Therefore, the radioactivity concentration at the monitor is a function of the concentrations measured in each tank and the flowrates at which the tanks are released. The setpoint methodology for this situation is presented in paragraph 1.1.1.2.

1.1.1.1 Monitor Setpoint Calculation Methodology When One Waste Monitor Tank per Unit Is To Be Released at a Time

Step 1

The radionuclide concentrations for the waste monitor tank planned for release are determined in accordance with subsection 1.5.1. The relationship of the various required sample analyses is shown as follows:

$$\sum_i C_i = \sum_g C_g + (C_a + C_s + C_f + C_t) \quad (2)$$

where:

- $C_g$  = the concentration of each measured gamma emitter observed by gamma ray spectroscopy of the particular waste sample.
- $C_a$  = the concentration of alpha emitters in liquid waste as measured in the MONTHLY composite sample. (NOTE: Sample is analyzed for gross alpha.)
- $C_s$  = the measured concentrations of Sr-89 and Sr-90 in liquid waste as observed in the QUARTERLY composite sample.
- $C_f$  = the measured concentrations of Fe-55 in liquid waste as observed in the QUARTERLY composite sample.
- $C_t$  = the measured concentration of H-3 in liquid waste as determined from analysis of the MONTHLY composite sample.

The  $C_g$  term will include the analysis of each batch; terms for alpha, strontiums, iron, and tritium will be included in accordance with subsection 1.5.1 as appropriate.

To assure sample analyses are based on samples that are representative of the volume from which the samples are taken, liquid volumes must be thoroughly mixed prior to sampling. Mixing may be accomplished by any method of mixing which has been demonstrated to achieve mixing sufficient to allow representative sampling.

## Step 2

Measured radionuclide concentrations are used to calculate maximum permissible concentration (MPC) fractions. The MPC fractions are used along with a safety factor to calculate a required dilution factor, which is the ratio of dilution flowrate to monitor tank discharge flowrate which are required to assure the limiting concentrations given in 10 CFR 20, Appendix B, Table II, Column 2, are not exceeded at the point of release to the river. The required dilution factor, RDF, is calculated as follows:

$$RDF = \sum_i \frac{C_i}{MPC_i} + SF$$

$$RDF = \left[ \sum_g \frac{C_g}{MPC_g} + \frac{C_a}{MPC_a} + \frac{C_s}{MPC_s} + \frac{C_f}{MPC_f} + \frac{C_t}{MPC_t} \right] + SF \quad (3)$$

where:

$C_i$  = the measured concentrations of  $C_g$ ,  $C_a$ ,  $C_s$ ,  $C_f$ , and  $C_t$  as defined in step 1. Terms  $C_a$ ,  $C_s$ ,  $C_f$ , and  $C_t$  will be included in the calculation as appropriate.

$MPC_i$  =  $MPC_g$ ,  $MPC_a$ ,  $MPC_s$ ,  $MPC_f$ , and  $MPC_t$  are limiting concentrations of the appropriate radionuclide from 10 CFR 20, Appendix B, Table II, Column 2. For dissolved or entrained noble gases, the concentration shall be limited to  $2 \times 10^{-4} \mu\text{Ci/ml}$  total activity. For gross alpha the maximum permissible concentration shall be  $3 \times 10^{-8} \mu\text{Ci/ml}$ . If specific alpha-emitting radionuclides are measured, the MPC for the specific radionuclide(s) shall be used.

SF = the safety factor, which is a conservative factor selected to compensate for statistical fluctuations and errors of measurements. The value for the safety factor must be between 0 and 1; a value of 0.5 is a reasonable value for liquid releases. A more precise value may be developed if desired.

### Step 3

Determine the dilution stream flowrate which will be assured during the period of the release, which is designated as  $F_d$ . For Plant Vogtle the flowrate which can be assured is the value selected as the setpoint for the dilution stream flowrate measurement device. Since the value selected as the setpoint for the dilution stream flowrate measurement device is the dilution stream flowrate which can be assured during the release, this value must be used as the basis for calculating the maximum permissible effluent release rate,  $f_m$ , and the radioactivity monitor setpoint,  $c$ .

If simultaneous releases are planned from the liquid waste processing systems of Unit 1 and Unit 2, the dilution stream must be allocated between the two units. This is accomplished by multiplying the assured dilution stream flowrate,  $F_d$ , by an allocation factor, AF, to obtain a unit-specific assured dilution stream flowrate,  $F_{du}$ :

$$F_{du} = F_d (AF) \quad (4)$$

where:

AF = an allocation factor selected to apportion the diluting capacity of the dilution stream between the two units when simultaneous releases from the liquid waste processing systems are planned. AF may be assigned any value between 0 and 1 for each unit under the condition that the sum of the allocation factors does not exceed 1. For convenience AF may be assigned the value of 0.5 for each unit. Also, if it is desired to make liquid waste processing system releases from each unit without regard to releases from the other unit, AF should be assigned the value of 0.5 for each unit.

If more precise allocation values are desired, they may be determined based on the relative radiological impact of each unit's liquid waste processing system effluent stream on the dilution stream, which may be approximated by multiplying the MPC fraction of each effluent stream by its associated planned release flowrate and comparing these values for the two units.

If no simultaneous liquid waste processing system releases are being made, AF may be assigned the value of 1 and then  $F_{du}$  is equal to  $F_d$ .

For the case  $RDF < 1$ , the waste monitor tank meets the limits of 10 CFR 20 without dilution and could be released at any desired flowrate. However, in order to maintain individual doses due to radioactivity in liquids released to unrestricted areas as low as reasonably achievable (ALARA), no releases from the liquid waste processing system should be made if assured dilution stream flowrate,  $F_d$ , is less than 5000 gpm.

#### Step 4

For the case  $RDF > 1$ , calculate the maximum permissible waste monitor tank discharge flowrate,  $f_m$ , as follows:

$$f_m = \frac{F_{du}}{RDF-1} \quad (5)$$

For the case  $RDF \leq 1$ , equation (5) is not valid. However, as discussed above, for the case  $RDF \leq 1$ , the release may be made at full pump discharge capacity and the monitor setpoint calculated in accordance with Step 5.

NOTE 1: Waste monitor tank discharge flowrate is actually limited by pump design discharge capacity which is 100 gpm (maximum). When calculated maximum permissible release flowrates are  $\geq 100$  gpm, the release may be made at full pump capacity. Release rates  $< 100$  gpm may be achieved by throttling.

NOTE 2: If radioactivity due to plant operations is detected in any of the effluent streams discharging to the blowdown sump (waste water retention basin, nuclear service water cooling tower blowdown, or turbine plant cooling tower blowdown), the diluting capacity of the dilution stream would be diminished. (Further, sampling and analysis of these effluent streams must be sufficient to assure that the dose limits specified in subsection 1.5.2 are not exceeded.) Under these conditions, equation (5) must be modified to include a term to account for radioactivity present in the dilution stream prior to the introduction of the liquid waste processing system effluent:

$$f_m = \frac{F_{du}}{RDF-1} \left[ 1 - \sum_r \sum_i \left( \frac{C_i}{MPC_i} \right) \frac{f_r}{F_d} \right] \quad (6)$$

where:

$\sum_i (C_i/MPC_i)_r$  = the MPC fraction of the effluent stream(s) containing the detectable radioactivity.

$f_r$  = the flowrate of the effluent stream(s) containing the radioactivity.

If  $RDF \leq 1$ , NOTE 2 does not apply.

#### Step 5

Based on the values determined in the previous steps, a liquid waste-processing system effluent radioactivity monitor base setpoint is calculated to provide assurance that the limits of 10 CFR 20, Appendix B, Table II, Column 2, will not be exceeded. The radioactivity monitor response is to gamma radiation primarily; therefore, the monitor setpoint calculation is based on  $\sum_g C_g$ , in units of  $\mu\text{Ci/ml}$ , as follows:

$$c = A \sum_g C_g \quad (7)$$



where:

- A = the adjustment factor which will allow the setpoint to be established in a practical manner to prevent spurious alarms and to allow for the margin between measured concentrations and concentrations which would approach 10 CFR 20 limits:

$$A = \frac{ADF}{RDF} \quad (8)$$

NOTE: ADF is the assured dilution factor:

$$ADF = \frac{F_{du} + f_a}{f_a} \quad (8a)$$

and  $f_a$  is the anticipated release flowrate from the waste monitor tank to be discharged.

If  $A \geq 1$ , calculate the monitor setpoint,  $c$ . However, if the calculated setpoint value is within 10 percent of the actual concentration planned for release, it may be impractical to set the monitor setpoint based on this value. If this situation should arise, it indicates that measured concentrations are approaching values which could cause 10 CFR 20 limits to be exceeded. Therefore, steps should be taken to reduce potential release concentrations. These steps may include decreasing the planned waste monitor tank release rate, increasing the dilution stream flowrate, postponing simultaneous releases, and/or by decreasing concentrations by further processing of the liquid waste planned for release. Following these actions, repeat the previous steps and calculate a new monitor setpoint.

If  $A < 1$ , no release may be made under planned conditions. Consider the alternatives discussed above to reduce potential release concentrations, and calculate a new monitor setpoint based on the results of the alternatives selected.

The setpoint thus calculated is in the units  $\mu\text{Ci/ml}$ . The monitor actually measures counts per minute, subtracts a predetermined background count rate, and then multiplies by a calibration factor to convert the counts per minute to  $\mu\text{Ci/ml}$ .

Calibration of the monitors by the manufacturer and Georgia Power Company utilized NBS traceable liquid solutions in the exact geometry of each production monitor over a gamma-ray energy range of 0.08 to 1.33 MeV. The calibration factor is a function of the radionuclide mix in the liquid to be released and will be calculated for the monitor based on the results of the predischARGE sample results from the laboratory gamma-ray spectrometer system.

The mix-dependent calibration factor will be used as the gain factor in the PERMS monitor or used to modify the calculated base monitor setpoint so that the default calibration factor can be left in the PERMS monitor.

The monitor setpoints determined in accordance with the methodology described above establish the upper bound for a particular monitor setpoint. Monitor setpoints may be established at lower values, if desired.

#### 1.1.1.2 Monitor Setpoint Calculation Methodology When Two Waste Monitor Tanks per Unit Are To Be Released at a Time

##### Step 1

Determine radionuclide concentrations for each waste monitor tank as described in step 1 of paragraph 1.1.1.1.

From the  $\sum_g C_g$  terms determined for each tank, determine an effective  $(\sum_g C_g)_e$  for the two tanks considered together as follows:

$$(\sum_g C_g)_e = \frac{V_1(\sum_g C_g)_1 + V_2(\sum_g C_g)_2}{V_1 + V_2} \quad (9)$$

where:

$V_1$  = the volume of liquid in tank containing greater quantity  
(referred to throughout this subsection as first tank.)

$V_2$  = the volume of liquid in tank containing lesser quantity  
(referred to throughout this subsection as second tank).

$(\sum C_g)_1$  = the measured concentrations of gamma-emitting radionuclides  
in first tank.

$(\sum C_g)_2$  = the measured concentration of gamma-emitting radionuclides in  
second tank.

#### Step 2

Determine a required dilution factor, RDF, for each tank in accordance with step 2 of paragraph 1.1.1.1. Using these values calculate an effective required dilution factor, (RDF)<sub>e</sub>, for the two tanks considered together as follows:

$$(RDF)_e = \frac{V_1 (RDF)_1 + V_2 (RDF)_2}{V_1 + V_2} \quad (10)$$

where:

$V_1$  = the volume of first tank.

$V_2$  = the volume of second tank.

$(RDF)_1$  = the required dilution factor for first tank.

$(RDF)_2$  = the required dilution factor for second tank.

### Step 3

Determine the dilution stream flowrate in accordance with step 3 of paragraph 1.1.1.1.

### Step 4

To facilitate calculation of the monitor setpoint, determine release flowrates for each tank so that the durations of the releases from the two tanks are equal. First, select a release flowrate for the first tank,  $f_1$ . Then, determine the release flowrate for the second tank,  $f_2$ , as follows:

$$f_2 = \frac{f_1 V_2}{V_1} \quad (11)$$

Next, determine a combined flowrate,  $f_c$ , for releases from both tanks, as follows:

$$f_c = f_1 + f_2 \quad (12)$$

Next, calculate a maximum permissible flowrate,  $f_m$ , for the combined release in accordance with step 4 of paragraph 1.1.1.1 using the effective (RDF), determined in step 2 above.

Then, compare the combined release flowrate,  $f_c$ , with the maximum permissible combined release flowrate,  $f_m$ . If  $f_m > f_c$ , the release may be made under the assumed conditions. If  $f_m < f_c$ , the two release flowrates may be throttled, maintaining the same ratio of  $f_2$  to  $f_1$ , as determined earlier. If it is impractical to throttle the release flowrates to the necessary degree to achieve  $f_c < f_m$ , steps must be taken to reduce potential release concentrations prior to making the release, and a new  $f_c$  determined following the necessary actions. (Steps which may be undertaken to reduce potential release concentrations were discussed in step 5 of paragraph 1.1.1.1.)

### Step 5

Calculate the monitor setpoint in accordance with step 5 of paragraph 1.1.1.1 with the following substitutions:

$$\text{Let } \sum_g C_g = (\sum_g C_g)_e;$$

$$\text{RDF} = (\text{RDF})_e;$$

$$\text{and } f_s = f_e$$

Observe the same limiting conditions discussed in step 5 of paragraph 1.1.1.1.

- 1.1.2 Steam Generator Blowdown Effluent Radioactivity Monitor (1(2)RE-0021);  
Turbine Building Drain Effluent Radioactivity Monitor (1(2)RE-0848);  
(one of each monitor per unit)

According to Plant Vogtle design and operating philosophy, the purpose of these radioactivity monitors is to minimize release of radioactivity via these effluent streams by automatically isolating or diverting effluent flow upon radioactivity in either of the effluent streams reaching certain low levels. To achieve the desired objective, setpoints for these monitors should be established as close to background radiation levels as practical to prevent spurious alarms and yet alarm should an inadvertent release occur. The actual setpoint for each monitor should be established under operating conditions and within the stated objective of preventing releases of radioactivity via these pathways to the extent practicable. All three of these effluent streams discharge to the waste water retention basin, which in turn releases to the blowdown sump; the blowdown sump discharges to the Savannah River.

Should it become necessary to make releases from any of these two sources containing levels of radioactivity above that which would normally be isolated or diverted, radioactivity monitor setpoints should be determined in the same manner as described in subsection 1.1.1. However, special consideration must be given to step 3. An allocation factor must be assigned to the release pathway under consideration here and allocation factors for other pathways, which may be releasing simultaneously, adjusted if necessary so that for simultaneous liquid releases from the site, the sum of the allocation factors does not exceed 1.

As stated earlier, both of these effluent streams discharge to the waste water retention basin. Composite samples are collected from the discharge line from the waste water retention basin to the blowdown sump. Sample collection and analysis must be sufficient to assure that the dose limits specified in subsection 1.5.2 are not exceeded.

1.1.3 Nuclear Service Cooling Water System Effluent Radioactivity Monitor  
(1(2)RE-0020 A and B) (Two monitors per unit)

Radioactivity in these effluent streams normally is expected to be below detectable levels. Therefore, the radioactivity monitor setpoints should be established as close to background as practical to prevent spurious alarms and yet alarm should an inadvertent release occur. If any one of these effluent streams should become contaminated with radioactivity, radionuclide concentrations must be determined and a radioactivity monitor setpoint determined in the same manner as described in subsection 1.1.1. However, special consideration must be given to step 3. An allocation factor must be assigned to the release pathway under consideration here and allocation factors for other pathways, which may be releasing simultaneously, adjusted if necessary so that for simultaneous liquid releases from the site the sum of the allocation factors does not exceed 1. Determination of concentrations of radioactivity in these streams must be adequate to assure that the dose limits specified in subsection 1.5.2 are not exceeded.

## 1.2 DOSE CALCULATION FOR LIQUID EFFLUENTS

For liquid releases from Plant Vogtle to the Savannah River, two human exposure pathways exist: consumption of drinking water and fish taken from the river. Fish are considered to be taken from the vicinity of the plant discharge; drinking water is taken from the river for potable use at Beaufort, South Carolina, which is approximately 112 river miles downstream from the plant site. The methodology for calculating doses to an individual due to exposure to these two pathways is presented in this subsection.

The dose limits specified in subsection 1.5.2 are on a per reactor basis. Therefore, the doses calculated in accordance with this subsection must be determined and recorded on a per reactor basis.

The dose to the maximum exposed individual due to radionuclides identified in liquid effluents released from each unit to unrestricted areas will be calculated for the purpose of implementation of subsection 1.5.2 as follows:

$$D_r = \sum_i A_{ir} \sum_{\ell=1}^m \Delta t_{\ell} C_{i\ell} F_{\ell} \quad (13)$$

where:

$D_r$  = the cumulative dose commitment to the total body or any organ,  $r$ , due to radioactivity in liquid effluents for the total time period:

$$\sum_{\ell=1}^m \Delta t_{\ell} \text{ in mrem (Reference 1).}$$

$\Delta t_{\ell}$  = the length of the  $\ell$ th time period over which  $C_{i\ell}$  and  $F_{\ell}$  are averaged for any liquid release, in hours.

$C_{i\ell}$  = the average concentration of radionuclide  $i$ , in undiluted liquid effluent during time period  $\Delta t_{\ell}$  from any liquid release, in  $\mu\text{Ci/ml}$ .

$F_d$  = the near-field average dilution factor in the Savannah River during any liquid effluent release, defined as the ratio of the undiluted liquid waste flow during release to the product of the average dilution stream flowrate into the river times Z.

NOTE: If simultaneous releases from both units occur, the dilution stream flowrate must be apportioned between the two units as discussed in subsection 1.1.1, step 3. In such cases,  $F_d$  is unit-specific.

$$F_d = \frac{(\text{average undiluted liquid waste flow}) + ((\text{average dilution stream flow during the period of release of radioactivity}) \times Z)}{\quad} \quad (14)$$

NOTE: The denominator of equation (14) is limited to 1000 cfs (448,800 gpm) or less. (Reference 1, section 4.3.)

Z = the applicable dilution factor for the Savannah River. For the months May through December, Z=10; for the months January through April, Z=20. (Reference 5, paragraph 11.2.3.4; Reference 11.)

$A_{ir}$  = the site-related adult ingestion dose commitment factor to the total body or any organ for each identified radionuclide. Site-related  $A_{ir}$  values for Plant Vogtle are listed in table 1.2-3, in mrem-ml/h- $\mu$ Ci.

$$A_{ir} = K_o \left( (U_w/D_w) e^{-\lambda_i t_w} + U_f B F_i e^{-\lambda_i t_f} \right) D F_{ir} \quad (15)$$

$K_o$  = the units conversion factor  $1.14 \times 10^5$ , determined by:

$$10^6 \frac{\text{pCi}}{\mu\text{Ci}} \times 10^3 \frac{\text{ml}}{\text{l}} + 8760 \frac{\text{h}}{\text{year}}$$

$U_w$  = the adult drinking-water consumption (730 liters/year; Reference 3, table E-5).



- $D_w$  = the dilution factor from the vicinity of the liquid release point for the plant site to the potable water intake location (8; Reference 11).
- $\lambda_i$  = the decay constant for radionuclide  $i$  ( $h^{-1}$ ).
- $t_w$  = the transit time from release to receptor for water consumption (48 h; Reference 3, section A.2; Reference 10).
- $U_r$  = the adult fish consumption (21 kg/year; Reference 3, table E-5).
- $BF_i$  = the bioaccumulation factor for radionuclide  $i$ , in fresh water fish, in pCi/kg/pCi/l. (See table 1.2-1; Reference 3, table A-1; Reference 2 for Ag; Reference 14 for P.)
- $t_f$  = the transit time from release to receptor for fish consumption (24 h; Reference 3, section A.2).
- $DF_{i,\tau}$  = the dose conversion factor for radionuclide  $i$ , for adults in organ,  $\tau$  in mrem/pCi, from table 1.2-2 (Reference 3, table E-11).

TABLE 1.2-1

BIOACCUMULATION FACTORS  
(pCi/kg per pCi/liter)\*

<u>Element</u>	<u>Freshwater Fish</u>
H	9.0E-01
C	4.6E 03
Na	1.0E 02
P	3.0E 03
Cr	2.0E 02
Mn	4.0E 02
Fe	1.0E 02
Co	5.0E 01
Ni	1.0E 02
Cu	5.0E 01
Zn	2.0E 03
Br	4.2E 02
Rb	2.0E 03
Sr	3.0E 01
Y	2.5E 01
Zr	3.3E 00
Nb	5.5E 02
Mo	1.0E 01
Tc	1.5E 01
Ru	1.0E 01
Rh	1.0E 01
Ag	2.3E 00
Sb	2.0E 02
Te	4.0E 02
I	1.5E 01
Cs	2.0E 03
Ba	4.0E 00
La	2.5E 01
Ce	1.0E 00
Pr	2.5E 01
Nd	2.5E 01
W	1.2E 03
Np	1.0E 01

\*Reference 3, Table A-1; Reference 2 for Ag; Reference 14 for P; and  
Reference 17 for Nb and Sb.

Table 1.2-2 (SHEET 1 OF 3)

ADULT INGESTION DOSE FACTORS\*  
(mrem per pCi ingested)

Nuclide	Bone	Liver	T Body	Thyroid	Kidney	Lung	GI-LLI
H-3	No Data	1.05E-07	1.05E-07	1.05E-07	1.05E-07	1.05E-07	1.05E-07
C-14	2.84E-06	5.68E-07	5.68E-07	5.68E-07	5.68E-07	5.68E-07	5.68E-07
Na-24	1.70E-06	1.70E-06	1.70E-06	1.70E-06	1.70E-06	1.70E-06	1.70E-06
P-32	1.93E-04	1.20E-05	7.46E-06	No Data	No Data	No Data	2.17E-05
Cr-51	No Data	No Data	2.66E-09	1.59E-09	5.86E-10	3.53E-09	6.69E-07
Mn-54	No Data	4.57E-06	8.72E-07	No Data	1.36E-06	No Data	1.40E-05
Mn-56	No Data	1.15E-07	2.04E-08	No Data	1.46E-07	No Data	3.67E-06
Fe-55	2.75E-06	1.90E-06	4.43E-07	No Data	No Data	1.06E-06	1.09E-06
Fe-59	4.34E-06	1.02E-05	3.91E-06	No Data	No Data	2.85E-06	3.40E-05
Co-58	No Data	7.45E-07	1.67E-06	No Data	No Data	No Data	1.51E-05
Co-60	No Data	2.14E-06	4.72E-06	No Data	No Data	No Data	4.02E-05
Ni-63	1.30E-04	9.01E-06	4.36E-06	No Data	No Data	No Data	1.88E-06
Ni-65	5.28E-07	6.86E-08	3.13E-08	No Data	No Data	No Data	1.74E-06
Cu-64	No Data	8.33E-08	3.91E-08	No Data	2.10E-07	No Data	7.10E-06
Zn-65	4.84E-06	1.54E-05	6.96E-06	No Data	1.03E-05	No Data	9.70E-06
Zn-69	1.03E-08	1.97E-08	1.37E-09	No Data	1.28E-08	No Data	2.96E-09
Br-83	No Data	No Data	4.02E-08	No Data	No Data	No Data	5.79E-08
Br-84	No Data	No Data	5.21E-08	No Data	No Data	No Data	4.09E-13
Br-85	No Data	No Data	2.14E-09	No Data	No Data	No Data	LT E-24
Rb-86	No Data	2.11E-05	9.83E-06	No Data	No Data	No Data	4.16E-06
Rb-88	No Data	6.05E-08	3.21E-08	No Data	No Data	No Data	8.36E-19
Rb-89	No Data	4.01E-08	2.82E-08	No Data	No Data	No Data	2.33E-21
Sr-89	3.08E-04	No Data	8.84E-06	No Data	No Data	No Data	4.94E-05
Sr-90	7.58E-03	No Data	1.86E-03	No Data	No Data	No Data	2.19E-04
Sr-91	5.67E-06	No Data	2.29E-07	No Data	No Data	No Data	2.70E-05
Sr-92	2.15E-06	No Data	9.30E-08	No Data	No Data	No Data	4.26E-05
Y-90	9.62E-09	No Data	2.58E-10	No Data	No Data	No Data	1.02E-04
Y-91m	9.09E-11	No Data	3.52E-12	No Data	No Data	No Data	2.67E-10
Y-91	1.41E-07	No Data	3.77E-09	No Data	No Data	No Data	7.76E-05
Y-92	8.45E-10	No Data	2.47E-11	No Data	No Data	No Data	1.48E-05

\*Reference 3, Table E-11.

Table 1.2-2 (SHEET 2 OF 3)

ADULT INGESTION DOSE FACTORS\*  
(mrem per pCi ingested)

Nuclide	Bone	Liver	T Body	Thyroid	Kidney	Lung	GI-LLI
Y-93	2.68E-09	No Data	7.40E-11	No Data	No Data	No Data	8.50E-05
Zr-95	3.04E-08	9.75E-09	6.60E-09	No Data	1.53E-08	No Data	3.09E-05
Zr-97	1.68E-09	3.39E-10	1.55E-10	No Data	5.12E-10	No Data	1.05E-04
Nb-95	6.22E-09	3.46E-09	1.86E-09	No Data	3.42E-09	No Data	2.10E-05
Mo-99	No Data	4.31E-06	8.20E-07	No Data	9.76E-06	No Data	9.99E-06
Tc-99m	2.47E-10	6.98E-10	8.89E-09	No Data	1.06E-08	3.42E-10	4.13E-07
Tc-101	2.54E-10	3.66E-10	3.59E-09	No Data	6.59E-09	1.87E-10	1.10E-21
Ru-103	1.85E-07	No Data	7.97E-08	No Data	7.06E-07	No Data	2.16E-05
Ru-105	1.54E-08	No Data	6.08E-09	No Data	1.99E-07	No Data	9.42E-06
Ru-106	2.75E-06	No Data	3.48E-07	No Data	5.31E-06	No Data	1.78E-04
Ag-110m	1.60E-07	1.48E-07	8.79E-08	No Data	2.91E-07	No Data	6.04E-05
Sb-124	2.81E-06	5.3E-08	1.11E-06	6.79E-09	No Data	2.18E-06	7.95E-05**
Sb-125	2.23E-06	2.4E-08	4.48E-07	1.98E-09	No Data	2.33E-04	1.97E-05**
Te-125m	2.68E-06	9.71E-07	3.59E-07	8.06E-07	1.09E-05	No Data	1.07E-05
Te-127m	6.77E-06	2.42E-06	8.25E-07	1.73E-06	2.75E-05	No Data	2.27E-05
Te-127	1.10E-07	3.95E-08	2.38E-08	8.15E-08	4.48E-07	No Data	8.68E-06
Te-129m	1.15E-05	4.29E-06	1.82E-06	3.95E-06	4.80E-05	No Data	5.79E-05
Te-129	3.14E-08	1.18E-08	7.65E-09	2.41E-08	1.32E-07	No Data	2.37E-08
Te-131m	1.73E-06	8.46E-07	7.05E-07	1.34E-06	8.57E-06	No Data	8.40E-05
Te-131	1.97E-08	8.23E-09	6.22E-09	1.62E-08	8.63E-08	No Data	2.79E-09
Te-132	2.52E-06	1.63E-06	1.53E-06	1.80E-06	1.57E-05	No Data	7.71E-05
I-130	7.56E-07	2.23E-06	8.80E-07	1.89E-04	3.48E-06	No Data	1.92E-06
I-131	4.16E-06	5.95E-06	3.41E-06	1.95E-03	1.02E-05	No Data	1.57E-06
I-132	2.03E-07	5.43E-07	1.90E-07	1.90E-05	8.65E-07	No Data	1.02E-07
I-133	1.42E-06	2.47E-06	7.53E-07	3.63E-04	4.31E-06	No Data	2.22E-06
I-134	1.06E-07	2.88E-07	1.03E-07	4.99E-06	4.58E-07	No Data	2.51E-10
I-135	4.43E-07	1.16E-06	4.28E-07	7.65E-05	1.86E-06	No Data	1.31E-06
Cs-134	6.22E-05	1.48E-04	1.21E-04	No Data	4.79E-05	1.59E-05	2.59E-06
Cs-136	6.51E-06	2.57E-05	1.85E-05	No Data	1.43E-05	1.96E-06	2.92E-06
Cs-137	7.97E-05	1.09E-04	7.14E-05	No Data	3.70E-05	1.23E-05	2.11E-06
Cs-138	5.52E-08	1.09E-07	5.40E-08	No Data	8.01E-08	7.91E-09	4.65E-13
Ba-139	9.70E-08	6.91E-11	2.84E-09	No Data	6.46E-11	3.92E-11	1.72E-07
Ba-140	2.03E-05	2.55E-08	1.33E-06	0.00E+00	8.67E-09	1.46E-08	4.18E-05
Ba-141	4.71E-08	3.56E-11	1.59E-09	0.00E+00	3.31E-11	2.02E-11	2.22E-17
Ba-142	2.13E-08	2.19E-11	1.34E-09	0.00E+00	1.85E-11	1.24E-11	3.00E-26

\* Reference 3, Table E-11.

\*\*Reference 17.

Table 1.2-2 (SHEET 3 OF 3)

ADULT INGESTION DOSE FACTORS\*  
(mrem per pCi ingested)

Nuclide	Bone	Liver	T Body	Thyroid	Kidney	Lung	GI-LLI
Ba-140	2.03E-05	2.55E-08	1.33E-06	No Data	8.67E-09	1.46E-08	4.18E-05
Ba-141	4.71E-08	3.56E-11	1.59E-09	No Data	3.31E-11	2.02E-11	2.22E-17
Ba-142	2.13E-08	2.19E-11	1.34E-09	No Data	1.85E-11	1.24E-11	3.00E-26
La-140	2.50E-09	1.26E-09	3.33E-10	No Data	No Data	No Data	9.25E-05
La-142	1.28E-10	5.82E-11	1.45E-11	No Data	No Data	No Data	4.25E-07
Ce-141	9.36E-09	6.33E-09	7.18E-10	No Data	2.94E-09	No Data	2.42E-05
Ce-143	1.65E-09	1.22E-06	1.35E-10	No Data	5.37E-10	No Data	4.56E-05
Ce-144	4.88E-07	2.04E-07	2.62E-08	No Data	1.21E-07	No Data	1.65E-04
Pr-143	9.20E-09	3.69E-09	4.56E-10	No Data	2.13E-09	No Data	4.03E-05
Pr-144	3.01E-11	1.25E-11	1.53E-12	No Data	7.05E-12	No Data	4.33E-18
Nd-147	6.29E-09	7.27E-09	4.35E-10	No Data	4.25E-09	No Data	3.49E-05
W-187	1.03E-07	8.61E-08	3.01E-08	No Data	No Data	No Data	2.82E-05
Np-239	1.19E-09	1.17E-10	6.45E-11	No Data	3.65E-10	No Data	2.40E-05

\*Reference 3, Table E-11.

Table 1.2-3 (SHEET 1 OF 2)

SITE-RELATED INGESTION DOSE FACTORS,  $A_{if}$   
 (FOR FRESHWATER FISH AND DRINKING WATER CONSUMPTION)\*  
 (mrem/h per  $\mu\text{Ci/ml}$ )

Nuclide	Bone	Liver	T Body	Thyroid	Kidney	Lung	GI-LLI
H-3	0.00E+00	1.32E+00	1.32E+00	1.32E+00	1.32E+00	1.32E+00	1.32E+00
C-14	3.13E+04	6.26E+03	6.26E+03	6.26E+03	6.26E+03	6.26E+03	6.26E+03
Na-24	1.36E+02	1.36E+02	1.36E+02	1.36E+02	1.36E+02	1.36E+02	1.36E+02
P-32	1.32E+06	8.22E+04	5.11E+04	0.00E+00	0.00E+00	0.00E+00	1.49E+05
Cr-51	0.00E+00	0.00E+00	1.27E+00	7.58E-01	2.79E-01	1.68E+00	3.19E+02
Mn-54	0.00E+00	4.41E+03	8.42E+02	0.00E+00	1.31E+03	0.00E+00	1.35E+04
Mn-56	0.00E+00	1.74E-01	3.09E-02	0.00E+00	2.21E-01	0.00E+00	5.55E+00
Fe-55	6.86E+02	4.74E+02	1.11E+02	0.00E+00	0.00E+00	2.65E+02	2.72E+02
Fe-59	1.07E+03	2.51E+03	9.61E+02	0.00E+00	0.00E+00	7.01E+02	8.36E+03
Co-58	0.00E+00	9.59E+01	2.15E+02	0.00E+00	0.00E+00	0.00E+00	1.94E+03
Co-60	0.00E+00	2.78E+02	6.14E+02	0.00E+00	0.00E+00	0.00E+00	5.23E+03
Ni-63	2.25E+04	2.25E+03	1.09E+03	0.00E+00	0.00E+00	0.00E+00	4.70E+02
Ni-65	1.72E-01	2.23E-02	1.02E-02	0.00E+00	0.00E+00	0.00E+00	5.67E-01
Cu-64	0.00E+00	2.76E+00	1.29E+00	0.00E+00	6.95E+00	0.00E+00	2.35E+02
Zn-65	2.32E+04	7.37E+04	3.33E+04	0.00E+00	4.93E+04	0.00E+00	4.64E+04
Zn-69	7.91E-07	1.51E-06	1.05E-07	0.00E+00	9.83E-07	0.00E+00	2.27E-07
Br-83	0.00E+00	0.00E+00	3.84E-02	0.00E+00	0.00E+00	0.00E+00	5.53E-02
Br-84	0.00E+00	0.00E+00	1.23E-12	0.00E+00	0.00E+00	0.00E+00	9.67E-18
Br-85	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Rb-86	0.00E+00	9.75E+04	4.54E+04	0.00E+00	0.00E+00	0.00E+00	1.92E+04
Rb-88	0.00E+00	1.30E-22	6.90E-23	0.00E+00	0.00E+00	0.00E+00	1.80E-33
Rb-89	0.00E+00	1.64E-26	1.38E-26	0.00E+00	0.00E+00	0.00E+00	0.00E+00
Sr-89	2.49E+04	0.00E+00	7.16E+02	0.00E+00	0.00E+00	0.00E+00	4.00E+03
Sr-90	6.23E+05	0.00E+00	1.53E+05	0.00E+00	0.00E+00	0.00E+00	1.80E+04
Sr-91	7.25E+01	0.00E+00	2.93E+00	0.00E+00	0.00E+00	0.00E+00	3.45E+02
Sr-92	3.34E-01	0.00E+00	1.14E-02	0.00E+00	0.00E+00	0.00E+00	6.61E+00
Y-90	5.04E-01	0.00E+00	1.35E-02	0.00E+00	0.00E+00	0.00E+00	5.34E+03
Y-91M	1.04E-11	0.00E+00	4.03E-13	0.00E+00	0.00E+00	0.00E+00	3.06E-11
Y-91	9.77E+00	0.00E+00	2.61E-01	0.00E+00	0.00E+00	0.00E+00	5.38E+03
Y-92	4.61E-04	0.00E+00	1.35E-05	0.00E+00	0.00E+00	0.00E+00	8.08E+00
Y-93	3.19E-02	0.00E+00	8.82E-04	0.00E+00	0.00E+00	0.00E+00	1.01E+03
Zr-95	5.47E-01	1.75E-01	1.19E-01	0.00E+00	2.75E-01	0.00E+00	5.56E+02
Zr-97	7.40E-03	1.49E-03	6.83E-04	0.00E+00	2.26E-03	0.00E+00	4.63E+02
Nb-95	8.16E+00	4.54E+00	2.44E+00	0.00E+00	4.48E+00	0.00E+00	2.76E+04
Sb-124	1.36E+03	2.56E+01	5.43E+02	3.28E+00	0.00E+00	1.05E+03	3.88E+04
Sb-125	1.09E+03	1.16E+01	2.19E+02	9.57E-01	0.00E+00	1.13E+05	9.63E+03
Mo-99	0.00E+00	1.07E+02	2.04E+01	0.00E+00	2.43E+02	0.00E+00	2.49E+02
Tc-99M	5.70E-04	1.61E-03	2.05E-02	0.00E+00	2.45E-02	7.89E-04	9.53E-01
Tc-101	2.75E-33	3.96E-33	3.89E-32	0.00E+00	7.14E-32	2.03E-33	0.00E+00

\*Calculated using Equation (15). When "no data" is reported for dose factors for specific radionuclide-organ combinations in Reference 3, site-related dose factors are presented as zero in this table.



Table 1.2-3 (SHEET 2 OF 2)

SITE-RELATED INGESTION DOSE FACTORS,  $A_{ir}$   
 (FOR FRESHWATER FISH AND DRINKING WATER CONSUMPTION)\*  
 (mrem/h per  $\mu\text{Ci/ml}$ )

Nuclide	Bone	Liver	T Body	Thyroid	Kidney	Lung	GI-LLI
Ru-103	6.21E+00	0.00E+00	2.68E+00	0.00E+00	2.37E+01	0.00E+00	7.25E+02
Ru-105	5.79E-03	0.00E+00	3.47E-03	0.00E+00	1.14E-01	0.00E+00	5.38E+00
Ru-106	9.42E+01	0.00E+00	1.19E+01	0.00E+00	1.82E+02	0.00E+00	6.10E+03
Ag-110M	2.53E+00	2.34E+00	1.39E+00	0.00E+00	4.61E+00	0.00E+00	9.56E+02
Te-125M	2.56E+03	9.29E+02	3.43E+02	7.71E+02	1.04E+04	0.00E+00	1.02E+04
Te-127M	6.51E+03	2.33E+03	7.93E+02	1.66E+03	2.64E+04	0.00E+00	2.18E+04
Te-127	1.78E+01	6.40E+00	3.85E+00	1.32E+01	7.26E+01	0.00E+00	1.41E+03
Te-129M	1.09E+04	4.07E+03	1.73E+03	3.75E+03	4.55E+04	0.00E+00	5.49E+04
Te-129	1.78E-05	6.70E-06	4.34E-06	1.37E-05	7.50E-05	0.00E+00	1.35E-05
Te-131M	9.58E+02	4.68E+02	3.90E+02	7.42E+02	4.74E+03	0.00E+00	4.65E+04
Te-131	8.71E-17	3.64E-17	2.75E-17	7.16E-17	3.82E-16	0.00E+00	1.23E-17
Te-132	1.97E+03	1.27E+03	1.19E+03	1.41E+03	1.23E+04	0.00E+00	6.02E+04
I-130	7.60E+00	2.24E+01	8.85E+00	1.90E+03	3.50E+01	0.00E+00	1.93E+01
I-131	1.73E+02	2.48E+02	1.42E+02	8.13E+04	4.25E+02	0.00E+00	6.55E+01
I-132	5.28E-03	1.41E-02	4.94E-03	4.94E-01	2.25E-02	0.00E+00	2.65E-03
I-133	2.59E+01	4.51E+01	1.37E+01	6.62E+03	7.86E+01	0.00E+00	4.05E+01
I-134	2.19E-08	5.96E-08	2.13E-08	1.03E-06	9.48E-08	0.00E+00	5.19E-11
I-135	1.31E+00	3.44E+00	1.27E+00	2.27E+02	5.52E+00	0.00E+00	3.89E+00
Cs-134	2.98E+05	7.10E+05	5.80E+05	0.00E+00	2.30E+05	7.62E+04	1.24E+04
Cs-136	2.96E+04	1.17E+05	8.42E+04	0.00E+00	6.51E+04	8.92E+03	1.33E+04
Cs-137	3.82E+05	5.23E+05	3.43E+05	0.00E+00	1.78E+05	5.90E+04	1.01E+04
Cs-138	9.18E-12	1.81E-11	8.98E-12	0.00E+00	1.33E-11	1.32E-12	7.73E-17
Ba-139	5.66E-06	4.03E-09	1.66E-07	0.00E+00	3.77E-09	2.29E-09	1.00E-05
Ba-140	3.74E+02	4.69E-01	2.45E+01	0.00E+00	1.60E-01	2.69E-01	7.69E+02
Ba-141	8.57E-25	6.47E-28	2.89E-26	0.00E+00	6.02E-28	3.67E-28	4.04E-34
Ba-142	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00	0.00E+00
La-140	1.10E-01	5.56E-02	1.47E-02	0.00E+00	0.00E+00	0.00E+00	4.08E+03
La-142	2.19E-07	9.98E-08	2.49E-08	0.00E+00	0.00E+00	0.00E+00	7.29E-04
Ce-141	1.15E-01	7.79E-02	8.84E-03	0.00E+00	3.62E-02	0.00E+00	2.98E+02
Ce-143	8.65E-03	6.40E+00	7.08E-04	0.00E+00	2.82E-03	0.00E+00	2.39E+02
Ce-144	6.22E+00	2.60E+00	3.34E-01	0.00E+00	1.54E+00	0.00E+00	2.10E+03
Pr-143	6.10E-01	2.45E-01	3.02E-02	0.00E+00	1.41E-01	0.00E+00	2.67E+03
Pr-144	1.50E-28	6.22E-29	7.61E-30	0.00E+00	3.51E-29	0.00E+00	2.15E-35
Nd-147	4.11E-01	4.75E-01	2.84E-02	0.00E+00	2.78E-01	0.00E+00	2.28E+03
W-187	1.48E+02	1.23E+02	4.31E+01	0.00E+00	0.00E+00	0.00E+00	4.04E+04
Np-239	2.81E-02	2.76E-03	1.52E-03	0.00E+00	8.61E-03	0.00E+00	5.67E+02

\*Calculated Using Equation (15). When "no data" is reported for dose factors for specific radionuclide-organ combinations in Reference 3, site-related dose factors are presented as zero in this table.

### 1.3 DOSE PROJECTIONS FOR LIQUID EFFLUENTS

#### 1.3.1 Thirty-One-Day Dose Projections

In order to meet the requirements of subsection 1.5.3, which pertains to operation of the liquid radwaste treatment systems, dose projections must be made at least once per 31 days, during periods in which discharge of liquid effluents containing radioactive materials to unrestricted areas occurs or is expected.

Projected 31-day doses to individuals due to liquid effluents may be determined as follows:

$$D_{tb(prj)} = \left( \frac{D_{tb(c)}}{t} \right) \times 31 \quad (16)$$

$$D_{o(prj)} = \left( \frac{D_{o(c)}}{t} \right) \times 31 \quad (17)$$

where:

- $D_{tb(c)}$  = the cumulative total body dose for the elapsed portion of the current quarter plus the release under consideration.
- $t$  = the number of days into the current quarter, including the period of the release under consideration.
- $D_{o(c)}$  = the cumulative organ doses for specific organs, for the elapsed portion of the current quarter plus the release under consideration.

If operational activities planned during the ensuing 31-day period are expected to result in liquid releases which will contribute a dose in addition to the dose due to routine liquid effluents, this additional dose contribution should be included in the equations as follows:



$$D_{tb(prj)} = \left[ \frac{D_{tb(c)}}{t} \times 31 \right] + D_{PA} \quad (18)$$

$$D_{o(prj)} = \left[ \frac{D_{o(c)}}{t} \times 31 \right] + D_{PA} \quad (19)$$

where  $D_{PA}$  is the expected dose due to the particular planned activity.

### 1.3.2 Dose Projections for Specific Releases

Dose projections may be performed for a particular release by performing a pre-release dose calculation assuming that the planned release will proceed as anticipated. For individual dose projections due to liquid releases, follow the methodology presented in section 1.2 using sample analyses values for the source to be released and parametric values expected to exist for the release period.

## 1.4 DEFINITIONS OF LIQUID EFFLUENT TERMS

<u>Term</u>	<u>Definition</u>	<u>Section of Initial Use</u>
A	= the adjustment factor used in calculating monitor setpoints, which is the ratio of the assured dilution factor to the required dilution factor (unitless).	1.1.1
A <sub>ir</sub>	= the site-related ingestion dose commitment factor for the total body or any organ for each identified principal radionuclide listed in table 1.2-3, in mrem-ml per h- $\mu$ Ci.	1.2
<u>ANALOG CHANNEL OPERATIONAL TEST(ACOT)</u>		
	An Analog Channel Operational Test shall be the injection of a simulated signal into the channel as close to the sensor as practicable to verify operability of alarm, interlock, and/or trip functions. The Analog Channel Operational Test shall include adjustments, as necessary, of the alarm, interlock, and/or trip setpoints, such that the setpoints are within the required range and accuracy.	1.5
ADF	= the assured dilution factor, which is the ratio of unit-specific assured dilution flowrate to anticipated effluent release rate (unitless).	1.1.1.1
BF <sub>i</sub>	= the bioaccumulation factor for nuclide i, in freshwater fish, pCi/kg per pCi/l, from table 1.2-1.	1.2

<u>Term</u>	<u>Definition</u>	<u>Section of Initial Use</u>
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#### BATCH RELEASE

	A batch release is the discharge of liquid wastes of a discrete volume. Prior to sampling for analyses, each batch shall be isolated and then thoroughly mixed by a method described in the ODCM to assure representative sampling.	1.5
c	= the base setpoint of the radioactivity monitor which measures the radioactivity concentration in the effluent line prior to dilution and subsequent release.	1.1.1
C <sub>α</sub>	= the effluent concentration of alpha-emitting nuclides observed by gross alpha analysis of the monthly composite sample, in $\mu\text{Ci/ml}$ .	1.1.1
C <sub>r</sub>	= the concentration of Fe-55 in liquid wastes as observed in the quarterly composite sample, in $\mu\text{Ci/ml}$ .	1.1.1
C <sub>g</sub>	= the effluent concentration of a gamma-emitting nuclide, g, observed by gamma ray spectroscopy of the waste sample, in $\mu\text{Ci/ml}$ .	1.1.1
C <sub>i</sub>	= the concentration of nuclide i as determined by the analysis of the waste sample, in $\mu\text{Ci/ml}$ .	1.1.1
C <sub>ie</sub>	= the average concentration of radionuclide i, in undiluted liquid effluent during time period $\Delta t$ , for a release, in $\mu\text{Ci/ml}$ .	1.2
C <sub>MPC</sub>	= the effluent concentration limit (Technical Specification 3.11.1.1) implementing 10 CFR 20 for the site, in $\mu\text{Ci/ml}$ .	1.1.1

<u>Term</u>	<u>Definition</u>	<u>Section of Initial Use</u>
C <sub>q</sub>	= the concentration of Sr-89 and Sr-90 in liquid wastes as determined by analysis of the QUARTERLY composite sample, in $\mu\text{Ci/ml}$ .	1.1.1
C <sub>i</sub>	= the measured concentration of H-3 in liquid waste as determined by analysis of the MONTHLY composite sample, in $\mu\text{Ci/ml}$ .	1.1.1

#### CHANNEL CALIBRATION

A channel calibration shall be the adjustment, as necessary, of the channel, such that it responds within the required range and accuracy to known values of input. The channel calibration shall encompass the entire channel including the sensors and alarm, interlock, and/or trip functions and may be performed by any series of sequential, overlapping, or total channel steps, such that the entire channel is calibrated.

1.5

#### CHANNEL CHECK

A channel check shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrument channels measuring the same parameter.

1.5

#### COMPOSITE SAMPLE

A composite sample is one in which the quantity of liquid sampled is proportional to the quantity of liquid waste discharged and in which the method of sampling employed results in a specimen that is representative of the liquids released.

1.5

<u>Term</u>	<u>Definition</u>	<u>Section of Initial Use</u>
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#### CONTINUOUS RELEASE

	A continuous release is the discharge of liquid wastes of a nondiscrete volume, e.g., from a volume of a system that has an input flow during the continuous release.	1.5
$D_{\tau}$	= the cumulative dose commitment to the total body or any organ, $\tau$ , from the liquid effluents for the total time period, in mrem.	1.2
$D_{\omega}$	= the additional dilution factor between vicinity of release point and drinking water location (unitless).	1.2
$DF_{i,\tau}$	= a dose conversion factor for nuclide, $i$ , for adults in organ, $\tau$ , in mrem/pCi found in table 1.2-2.	1.2
$f$	= the flow as determined for the radiation monitor location in gpm. (General expression for equation 1.)	1.1.1
$F$	= the dilution water flowrate as determined prior to the point at which the dilution stream discharges to the river, in gpm. (General expression for equation 1.)	1.1.1

<u>Term</u>	<u>Definition</u>	<u>Section of Initial Use</u>
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#### FREQUENCY NOTATION

The frequency notation specified for the performance of surveillance requirements shall correspond to the intervals defined below. 1.5

<u>NOTATION</u>	<u>FREQUENCY</u>
S	At least once per 12 hours.
D	At least once per 24 hours.
W	At least once per 7 days.
M	At least once per 31 days.
Q	At least once per 92 days.
SA	At least once per 184 days.
R	At least once per 18 months.
S/U	Prior to each reactor startup.
N.A.	Not applicable.
P	Completed prior to each release.

$F_d$  = the flowrate of the dilution stream which can be assured during the time of release in gpm. This is also the setpoint for the dilution stream flowrate measurement device. 1.1.1

$F_{du}$  = the unit-specific assured flowrate of the dilution stream used as the basis for setpoint calculations, in gpm. 1.1.1

$F_\ell$  = the near-field average dilution factor for  $C_{i\ell}$  during any liquid effluent release (unitless). 1.2

$f_a$  = the anticipated effluent flowrate in gpm. 1.1.1

<u>Term</u>	<u>Definition</u>	<u>Section Of Initial Use</u>
$f_m$	= the maximum permissible effluent flowrate in gpm.	1.1.1
$K_o$	= $1.14 \times 10^6$ , units conversion factor, which converts $\mu\text{Ci}$ to $\text{pCi}$ , liters to ml, and hours to year.	1.2

#### LOWER LIMIT OF DETECTION

The principal gamma emitters for which the lower limit of detection (LLD) specification applies include the following radionuclides: Mn-54, Fe-59, Co-58, Co-60, Zn-65, Mo-99, Cs-134, Cs-137, and Ce-141. Ce-144 shall also be measured, but with an LLD of  $5 \times 10^{-6}$ . This list does not mean that only these nuclides are to be considered. Other gamma peaks that are identifiable, together with those of the above nuclides, shall also be analyzed and reported in the semiannual Radioactive Effluent Release Report pursuant to Technical Specification 6.8.1.4.

The LLD is defined, for purposes of these specifications, as the smallest concentration of radioactive material in a sample that will yield a net count above system background and that will be detected with 95-percent probability, with only 5-percent probability of falsely concluding that a blank observation represents a "real" signal.

<u>Term</u>	<u>Definition</u>	<u>Section Of Initial Use</u>
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For a particular measurement system, which may include radiochemical separation:

$$LLD = \frac{4.66 s_b}{E \cdot V \cdot 2.22 \times 10^6 \cdot Y \cdot \exp(-\lambda \Delta t)}$$

where:

LLD	= the <i>a priori</i> lower limit of detection (microCurie per unit mass or volume).
$s_b$	= the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (counts per minute).
E	= the counting efficiency (counts per disintegration).
V	= the sample size (units of mass or volume).
$2.22 \times 10^6$	= the number of disintegrations per minute per microCurie.
Y	= the fractional radiochemical yield, when applicable.
$\lambda$	= the radioactive decay constant for the particular radionuclide ( $\text{sec}^{-1}$ ).
$\Delta t$	= the elapsed time between the midpoint of sample collection and the time of counting (sec).

Typical values of E, V, Y, and  $\Delta t$  should be used in the calculation.

It should be recognized that the LLD is defined as an *a priori* (before the fact) limit representing the capability of a measurement system and not as an *a posteriori* (after the fact) limit for a particular measurement.

m = the number of liquid releases. 1.2

MPC<sub>i</sub> = MPC<sub>g</sub>, MPC<sub>a</sub>, MPC<sub>st</sub>, MPC<sub>ir</sub>, and MPC<sub>tr</sub>, which are the limiting concentrations of the appropriate gamma-emitting radionuclides, alpha-emitting radionuclides, strontium, iron, and tritium, respectively, from 10 CFR 20, Appendix B, table II, column 2. 1.1.1



<u>Term</u>	<u>Definition</u>	<u>Section Of Initial Use</u>
RDF	= the required dilution factor, which is the ratio of the dilution flowrate to the effluent stream flowrate(s) which would be required to assure that the limiting concentration of 10 CFR 20, Appendix B, table II, column 2 are met at the point of discharge to the unrestricted area (unitless).	1.1.1

#### REPRESENTATIVE SAMPLE FOR CONTINUOUS RELEASES

To be representative of the quantities and concentrations of radioactive materials in liquid effluents, samples shall be collected continuously in proportion to the rate of flow of the effluent stream. Prior to analyses, all samples taken for the composite shall be thoroughly mixed in order for the composite sample to be representative of the effluent release.

1.5

SF	= the safety factor, which is a conservative factor used to compensate for statistical fluctuations and errors of measurement. The value for the safety factor must be between 0 and 1.	1.1.1
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#### SOURCE CHECK

A source check shall be the qualitative assessment of channel response when the channel sensor is exposed to a source of increased radioactivity.

1.5

t,	= the transit time from release to receptor (fish consumption), in hours.	1.2
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<u>Term</u>	<u>Definition</u>	<u>Section Of Initial Use</u>
$t_w$	= the transit time from release to receptor (drinking-water consumption), in hours.	1.2
$\Delta t_e$	= the duration of release under consideration, in hours.	1.2
$U_f$	= 21 kg/year, fish consumption (adult).	1.2
$U_w$	= 730 liters/year, water consumption (adult).	1.2
$Z$	= the applicable factor when additional receiving water body dilution is considered (unitless).	1.2
$A_i$	= the decay constant for radionuclide $i$ ( $h^{-1}$ ).	1.2

## 1.5 LIMITS OF OPERATION

### 1.5.1 Concentration in Liquid Effluents

The concentration of radioactive material released in liquid effluents to unrestricted areas (see figures 3.0-1 and 3.0-2) shall be limited to the concentrations specified in 10 CFR 20, Appendix B, table II, column 2, for radionuclides other than dissolved or entrained noble gases. For dissolved or entrained noble gases, the concentration shall be limited to  $2.00\text{E-}4$  microCurie/ml total activity. This limit applies at all times for all modes of operation.

#### 1.5.1.1 Exceeding the Limits

If the concentration of radioactive material released in liquid effluents to unrestricted areas exceeds the above limits, immediately restore the concentration to within the above limits. These limits do not affect mode changes.

#### 1.5.1.2 Sampling and Analysis

To assure these limits are not exceeded, the following sampling and analysis program must be implemented:

1.5.1.2.1 Batch Releases. For batch waste releases from any of the six waste monitor tanks or from the drainage of a system, each batch must be sampled prior to its release and analyzed for (1) principal gamma emitters to an LLD of  $5.00\text{E-}7$  microCurie/ml and (2) I-131 to an LLD of  $1.00\text{E-}6$  microCurie/ml. These results shall be used to calculate setpoints and flowrates, per section 1.1, to assure that the concentrations at the point of release are maintained within the limits specified above. In addition, this same sample taken prior to release, shall be used to measure for dissolved and entrained gases (gamma

emitters) at least monthly. This analysis must meet an LLD of  $1.00\text{E-}5$  microCurie/ml. Normally, this can be combined into the same analysis for principal gamma emitters.

Also included in this sample taken prior to release, enough of the composite sample shall be set aside to prepare a monthly composite for monthly tritium analysis to an LLD of  $1.00\text{E-}5$   $\mu\text{Ci/ml}$  and monthly gross alpha analysis to an LLD of  $1.00\text{E-}7$  microCurie/ml. Finally, a quarterly composite shall be analyzed quarterly for Sr-89 and Sr-90 to an LLD of  $5.00\text{E-}8$  microCurie/ml and for Fe-55 to an LLD of  $1.00\text{E-}6$  microCurie/ml. These composites are prepared for each unit.

1.5.1.2.2 Continuous Releases. All liquid effluents that do not normally contain contamination but could become contaminated in the event of steam generator tube leak are discharged from the site through the waste water retention basins (WWRB). Because of the potential for contamination, a representative sample of each WWRB is collected continuously with the composite sample container being changed weekly. The WWRB will not be considered to be a release point until there is a confirmed primary to secondary release. Once a primary to secondary leak has been confirmed to occur, this composite shall be analyzed weekly for (1) principal gamma emitters to an LLD of  $5.00\text{E-}7$   $\mu\text{Ci/ml}$  and (2) I-131 to an LLD of  $1.00\text{E-}6$   $\mu\text{Ci/ml}$ . These results can be used to calculate setpoints and flowrates, per section 1.1, to assure that the concentrations at the point of release are maintained within the limits specified above. In addition, a monthly grab sample shall be taken and analyzed monthly for dissolved and entrained gases (gamma emitters) to an LLD of  $1.00\text{E-}5$   $\mu\text{Ci/ml}$ . Normally, this can be combined into the same analysis for principal gamma emitters.

Also included in this sample taken continuously, a monthly composite shall be analyzed monthly for tritium to an LLD of  $1.00\text{E-}5$   $\mu\text{Ci/ml}$  and for gross alpha to an LLD of  $1.00\text{E-}7$   $\mu\text{Ci/ml}$ . Finally, a quarterly composite shall be analyzed

quarterly for Sr-89 and Sr-90 to an LLD of  $5.00\text{E-}8 \mu\text{Ci/ml}$  and for Fe-55 to an LLD of  $1.00\text{E-}6 \mu\text{Ci/ml}$ . These composites are prepared for each unit. Once these composites begin, they shall continue until three consecutive weekly composite samples show no activity above LLD.

1.5.1.2.3 The low levels specified in 10 CFR 20 provide additional assurance that the levels of radioactive materials in bodies of water in unrestricted areas will result in exposures within (1) the section II.A, design objectives, Appendix I, 10 CFR 50, to a member of the public and (2) the limits of 10 CFR 20.106(e) to the population. The concentration limit for dissolved or entrained noble gases is based upon the assumption that Xe-135 is the controlling radioisotope, and its MPC in air (submersion) was converted to an equivalent concentration in water using the methods described in International Commission on Radiological Protection (ICRP) Publication 2.

The specification applies to the release of radioactive materials in liquid effluents from all units at the site.

The required detection capabilities for radioactive materials in liquid waste samples are tabulated in terms of the LLD's. Detailed discussion of the LLD and other detection limits can be found in HASL Procedures Manual, HASL-300 (revised annually), L. A. Currie, "Limits for Qualitative Detection and Quantitative Determination - Application to Radiochemistry," Anal. Chem. 40, 586-93, 1968, and J. K. Hartwell, "Detection Limits for Radioanalytical Counting Techniques," Atlantic Richfield Hanford Company Report ARH-SA-215, June 1975.

Composite samples are sent to off-site labs at the plant's discretion; in this case, the date the sample is mailed via the warehouse is the date used to sign off the surveillance.

#### 1.5.2 Dose Due to Liquid Effluents

The dose or dose commitment to a member of the public from radioactive materials in liquid effluents released from each unit to unrestricted areas (see figure 3.0-1 and 3.0-2) shall be limited (1) during any calendar quarter

to less than or equal to 1.5 mrem to the whole body and to less than or equal to 5 mrem to any organ and (2) during any calendar year to less than or equal to 3 mrem to the whole body and to less than or equal to 10 mrem to any organ. This limit applies at all times for all modes of operation.

#### 1.5.2.1 Calculated Dose That Exceeds Limits

If the calculated dose from the release of radioactive materials in liquid effluents exceeds any of the above limits, prepare and submit to the NRC within 30 days, pursuant to Technical Specification 6.8.2, a special report that (1) identifies the cause(s) for exceeding the limit(s) and (2) defines the corrective action(s) that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits. These limits do not affect mode changes.

#### 1.5.2.2 Requirements for Calculations

To assure that these limits are not exceeded, the cumulative dose contributions from liquid effluents for the current calendar quarter and the current calendar year shall be determined in accordance with subsection 1.3.1 at least once per 31 days.

#### 1.5.2.3 Bases for The Limits

These limits are provided to implement the requirements of sections II.A, III.A, and IV.A, Appendix I, 10 CFR 50. Also for fresh water sites with drinking water supplies that can be potentially affected by plant operations, there is reasonable assurance that the operation of the facility will not result in radionuclide concentrations that are in excess of the requirements of 40 CFR 141, in the finished drinking water. The dose calculation methodology and parameters in this ODCM implement the Appendix I, section III.A requirements that conformance with the Appendix I guides be shown by

calculational procedures based on models and data, such that the actual exposure to a member of the public through appropriate pathways is unlikely to be substantially underestimated. The equations specified in this ODCM for calculating the doses that are due to the actual release rates of radioactive materials in liquid effluents are consistent with the methodology provided in Regulatory Guide 1.109 and 1.113.

This specification applies to the release of radioactive materials in liquid effluents from each unit at the site. When shared radwaste treatment systems are used by more than one unit at the site, the wastes from all units are mixed for shared treatment. By such mixing, the effluent releases cannot accurately be ascribed to a specific unit. An estimate should be made of the contributions from each unit based on input conditions, e.g., flowrates and radioactivity concentrations, or, if not practicable, the treated effluent releases may be allocated equally to each of the radioactive waste-producing units sharing the radwaste treatment system.

The laundry and hot shower tank liquids are common to both units. The recycle holdup tanks receive liquids from both the units. During normal operations the volume is proportioned equally between each processing unit. Whenever there is an outage, consideration is given to allocating more of the waste to the unit that is in the outage. The chemical drain tank liquid is charged to Unit 1 only.

### 1.5.3 Liquid Radwaste Treatment System

#### 1.5.3.1 Dose Limits for Operation of the Liquid Radwaste Treatment System

The liquid radwaste treatment system shall be operable, and appropriate portions of the system shall be used to reduce radioactivity releases when the projected doses that are due to the liquid effluent, from each unit to unrestricted areas (see figure 3.0-1 and 3.0-2), would exceed 0.06 mrem to the whole body or 0.2 mrem to any organ in a 31-day period. This limit applies at all times for all modes of operation.



#### 1.5.3.2 Projected Doses That Exceed Limits

If radioactive liquid waste is being discharged in excess of the above limits without treatment and if any portion of the liquid radwaste treatment system is not in operation, prepare and submit to the NRC within 30 days, pursuant to Technical Specification 6.8.2, a special report that includes the following information:

- 1) An explanation of why liquid radwaste was being discharged without treatment. Identification of any inoperable equipment or subsystems and the reason for inoperability.
- 2) What actions will be taken to restore the inoperable equipment to operable status.
- 3) A summary description of action(s) that will be taken to prevent a recurrence.

These limits do not affect mode changes.

#### 1.5.3.3 Bases for Projecting Doses

To assure these limits are not exceeded, doses that are due to liquid releases, from each unit to unrestricted areas, shall be projected at least once per 31 days, in accordance with the methodology and parameters in this ODCM, when liquid radwaste treatment systems are not being fully utilized.

The operability of the liquid radwaste treatment system ensures that this system will be available for use whenever liquid effluents require treatment prior to release to the environment. The requirement that the appropriate portions of this system be used when specified provides assurance that the releases of radioactive materials in liquid effluents will be kept "as low as is reasonably achievable." This specification implements the requirements of 10 CFR 50.36a, General Design Criterion 60, Appendix A, 10 CFR 50 and the design objective given in section II.D, Appendix I, 10 CFR 50. The specified limits governing the use of appropriate portions of the liquid radwaste treatment system were specified as a suitable fraction of the dose design objectives that are set forth in section II.A, Appendix I, 10 CFR 50, for liquid effluents.



#### 1.5.4 Liquid Effluent Monitoring Instrumentation

The radioactive liquid effluent monitoring instrumentation channels specified below shall be operable with their alarm/trip setpoints set to ensure that the limits in subsection 1.5.1 of this ODCM are not exceeded. This limit applies at all times for all modes of operation.

##### 1.5.4.1 Nonconservative Liquid Monitor Setpoint and ACTIONS

If a radioactive liquid effluent monitoring instrumentation channel alarm/trip setpoint is less conservative than required, immediately suspend the release of radioactive liquid effluents monitored by the affected channel or declare the channel inoperable.

If less than the minimum number of radioactive liquid effluent monitoring instrumentation channels are operable, take the ACTION shown in table 1.5-1. Restore the inoperable instrumentation to operable status within 30 days or, if unsuccessful, explain in the next Semiannual Radioactive Effluent Release Report, per Technical Specification 6.8.1.4, why this inoperability was not corrected in a timely manner. These limits do not affect modes changes.

##### 1.5.4.2 Liquid Monitor Operability

Each radioactive liquid effluent monitoring instrumentation channel shall be demonstrated operable by performance of the channel check, source check, channel calibration, and analog channel operational test at the frequencies shown in table 1.5-1. Specific instrument numbers are provided in parentheses for information only. The numbers apply to each unit. These numbers will help to identify associated channels or loops and are not intended to limit the requirements to the specific instruments associated with the number.

TABLE 1.5-1 (SHEET 1 OF 4)

## LIQUID EFFLUENT MONITORING INSTRUMENTATION

	<u>Minimum Channels Operable</u>	<u>ACTION</u>	<u>Channel Check</u>	<u>Source Check</u>	<u>Channel Calibration</u>	<u>Analog Channel Operational Test</u>
1. Radwaste monitors providing alarm and auto termination of release						
a) Liquid radwaste effluent (RE-0018)	1	37	D	P	R(3)	Q(1)
b) SG blowdown effluent (RE-0021)	1	38	D	M	R(3)	Q(1)
c) Turbine building effluent (RE-0848)	1	38	D	M	R(3)	Q(1)
2. Radwaste monitors providing alarm but not providing auto termination of release						
a) NSCW effluent line (RE-0020 A&B)	1	39	D	M	R(3)	Q(2)
3. Flowrate measurement devices						
a) Liquid radwaste effluent line (FT-0018) or (FT-1035) or (FT-1036)	1	40	D(4)	NA	R	NA
b) Steam generator blowdown effluent (FT-0021)	1	40	D(4)	NA	R	NA
c) Flow to blowdown sump (AFQI-7620, FR7620 pen 1)	1	40	D(4)	NA	R	Q

TABLE 1.5-1 (SHEET 2 OF 4)

TABLE NOTATIONS

- (1) The analog channel operational test shall also demonstrate that automatic isolation of this pathway (for item a. below only) and control room alarm annunciation occur if any of the following conditions exists:
  - a. Instrument indicates measured levels above the alarm/trip setpoint.
  - b. Circuit failure.
  - c. Instrument indicates a downscale failure.
  - d. Instrument controls not set in operate mode. (Annunciation via computer printout.)
- (2) The analog channel operational test shall also demonstrate that control room alarm annunciation occurs if any of the following conditions exists:
  - a. Instrument indicates measured levels above the alarm setpoint.
  - b. Circuit failure.
  - c. Instrument indicates a downscale failure.
  - d. Instrument controls not set in operate mode. (Annunciation via computer printout).
- (3) The initial channel calibration shall be performed using one or more of the reference standards certified by the National Bureau of Standards (NBS) or using standards that have been obtained from suppliers that participate in measurement assurance activities with NBS. These standards shall permit calibrating the system over its intended range of energy and measurement range. For subsequent channel calibration, sources that have been related to the initial calibration shall be used.
- (4) Channel check shall consist of verifying indication of flow during periods of release. Channel check shall be made at least once per 24 hours on days during which continuous, periodic, or batch releases are made.

TABLE 1.5-1 (SHEET 3 OF 4)

ACTION STATEMENTS

ACTION 37 - With the number of channels operable less than required by the minimum channels operable requirement, effluent releases via this pathway may continue provided that prior to initiating a release:

- a. at least two independent samples are analyzed in accordance with subsection 1.5.1 of this ODCM
- b. at least two technically qualified members of the facility staff independently verify the release-rate calculations and discharge line valving.

Otherwise, suspend release of radioactive effluents via this pathway.

ACTION 38 - With the number of channels operable less than required by the minimum channels operable requirement, effluent releases via this pathway may continue provided grab samples are analyzed for radioactivity at a lower limit of detection no more than  $1 \times 10^{-7} \mu\text{Ci/ml}$ :

- a. at least once per 12 hours when the specific activity of the secondary coolant is greater than  $0.01 \mu\text{Ci/gram}$  dose equivalent I-131
- b. at least once per 24 hours when the specific activity of the secondary coolant is less than or equal to  $0.01 \mu\text{Ci/gram}$  dose equivalent I-131.

ACTION 39 - With the number of channels operable less than required by the minimum channels operable requirement, effluent releases via this pathway may continue provided that, at least once per 12 hours, grab samples are collected and analyzed for radioactivity at a lower level of detection no more than  $1 \times 10^{-7} \mu\text{Ci/ml}$ .

TABLE 1.5-1 (SHEET 4 OF 4)

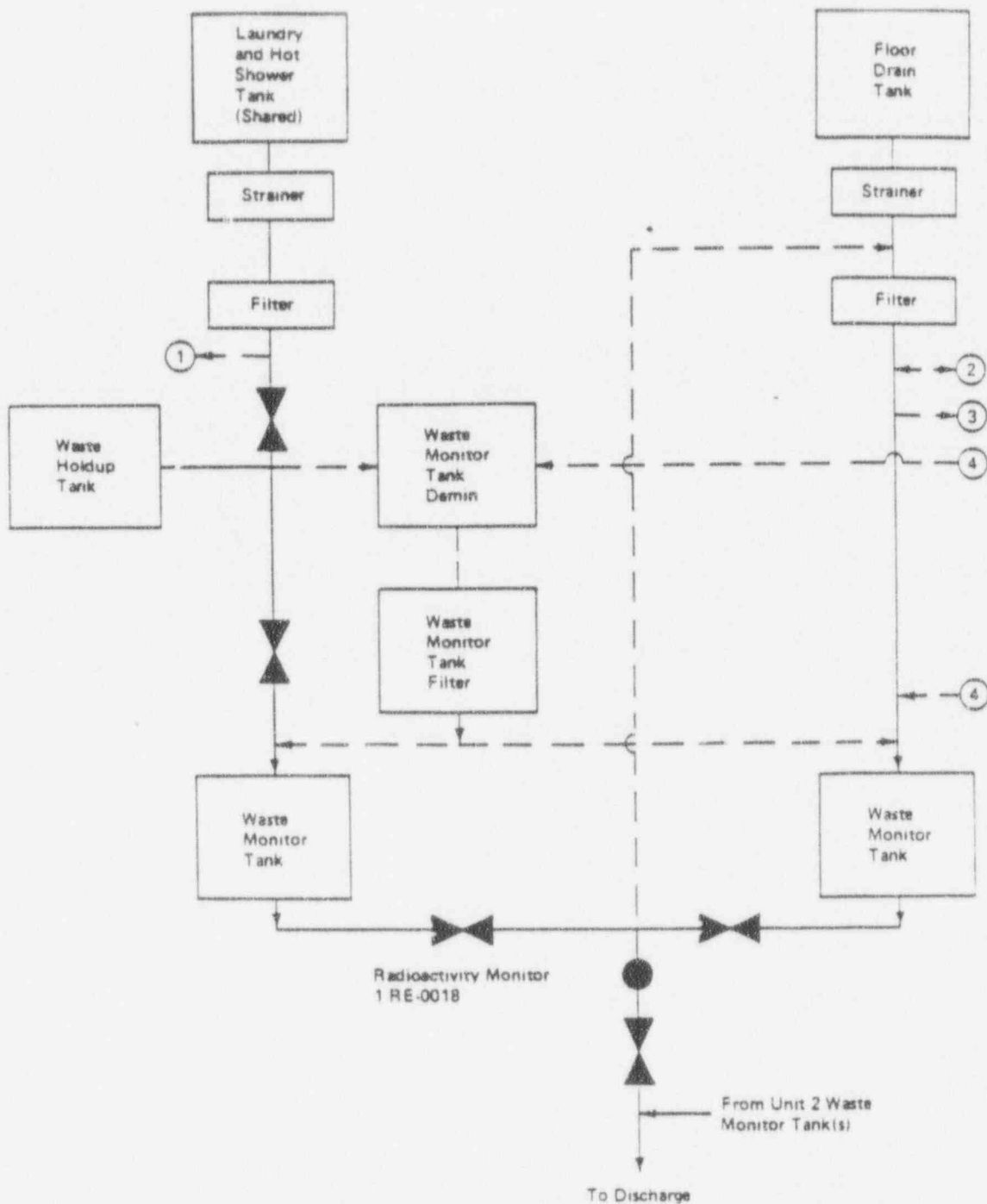
ACTION STATEMENTS

ACTION 40 - With the number of channels operable less than required by the minimum channels operable requirement, effluent releases via this pathway may continue provided the flowrate is estimated at least once per 4 hours during actual releases. Pump performance curves generated in place may be used to estimate flow.

## 1.6 LIQUID WASTE PROCESSING SYSTEM AND LIQUID DISCHARGE PATHWAYS

Figures 1.6-1 and 1.6-2 are schematics of the liquid waste processing systems for Unit 1 and Unit 2, respectively. The dotted lines indicate alternate pathways through which liquid wastes may be routed as appropriate. These alternate routes increase the operational flexibility of the liquid waste processing systems.

Figure 1.6-3 is a schematic of plant discharge pathways for liquids.



\*Dotted lines indicate alternate routes for increased operational flexibility and/or additional processing.

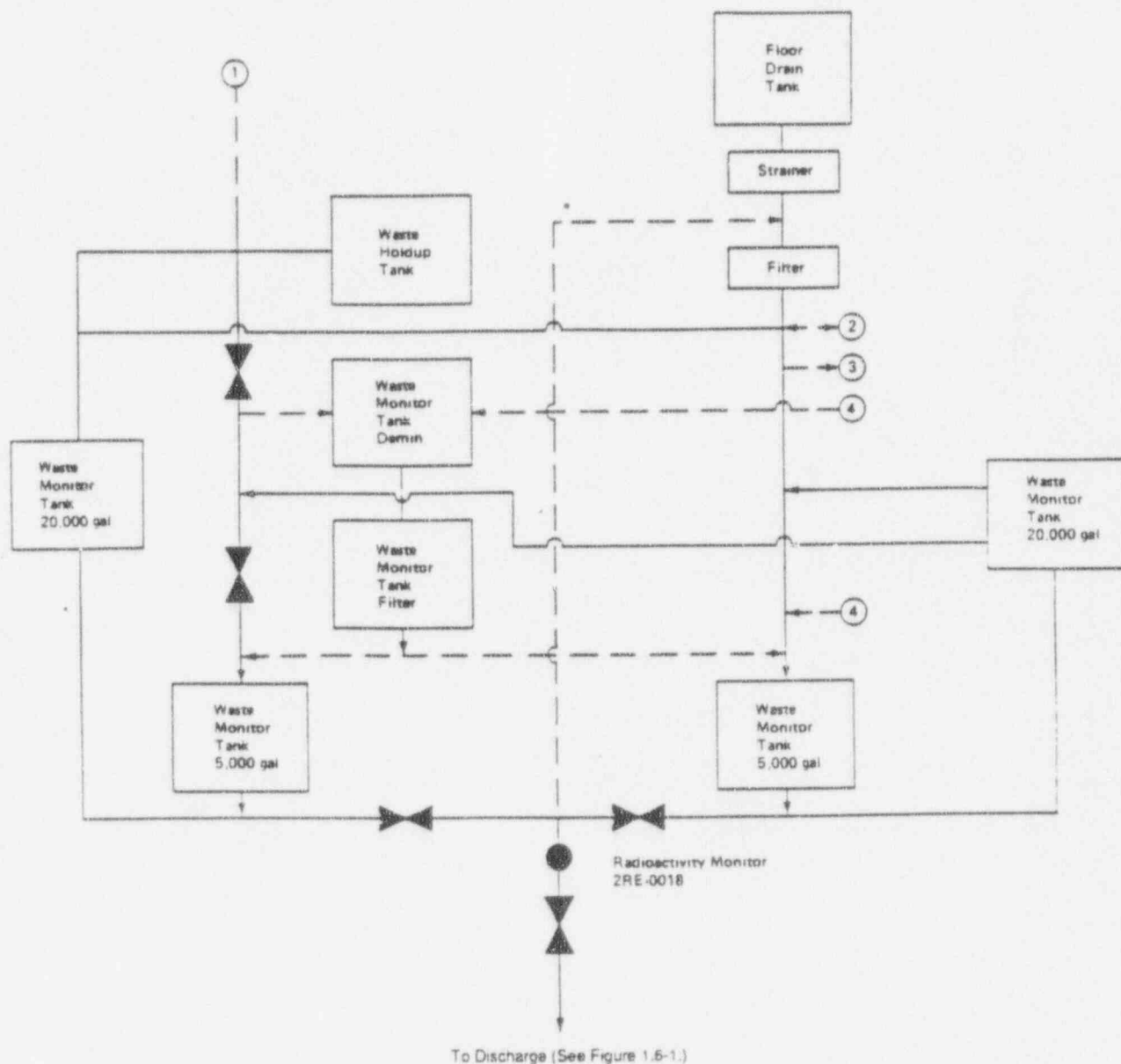
- ① To Unit 2
- ② From/To Unit 2
- ③ To Drain Channel A
- ④ From Drain Channel A

Georgia Power

VOGTLE  
ELECTRIC GENERATING PLANT  
UNIT 1 AND UNIT 2

LIQUID WASTE PROCESSING SYSTEM  
(Unit 1)

FIGURE 1.6-1



\*Dotted lines indicate alternate routes for increased operational flexibility and/or additional processing.

- ① From Unit 1
- ② From/To Unit 1
- ③ To Drain Channel A
- ④ From Drain Channel A

Georgia Power

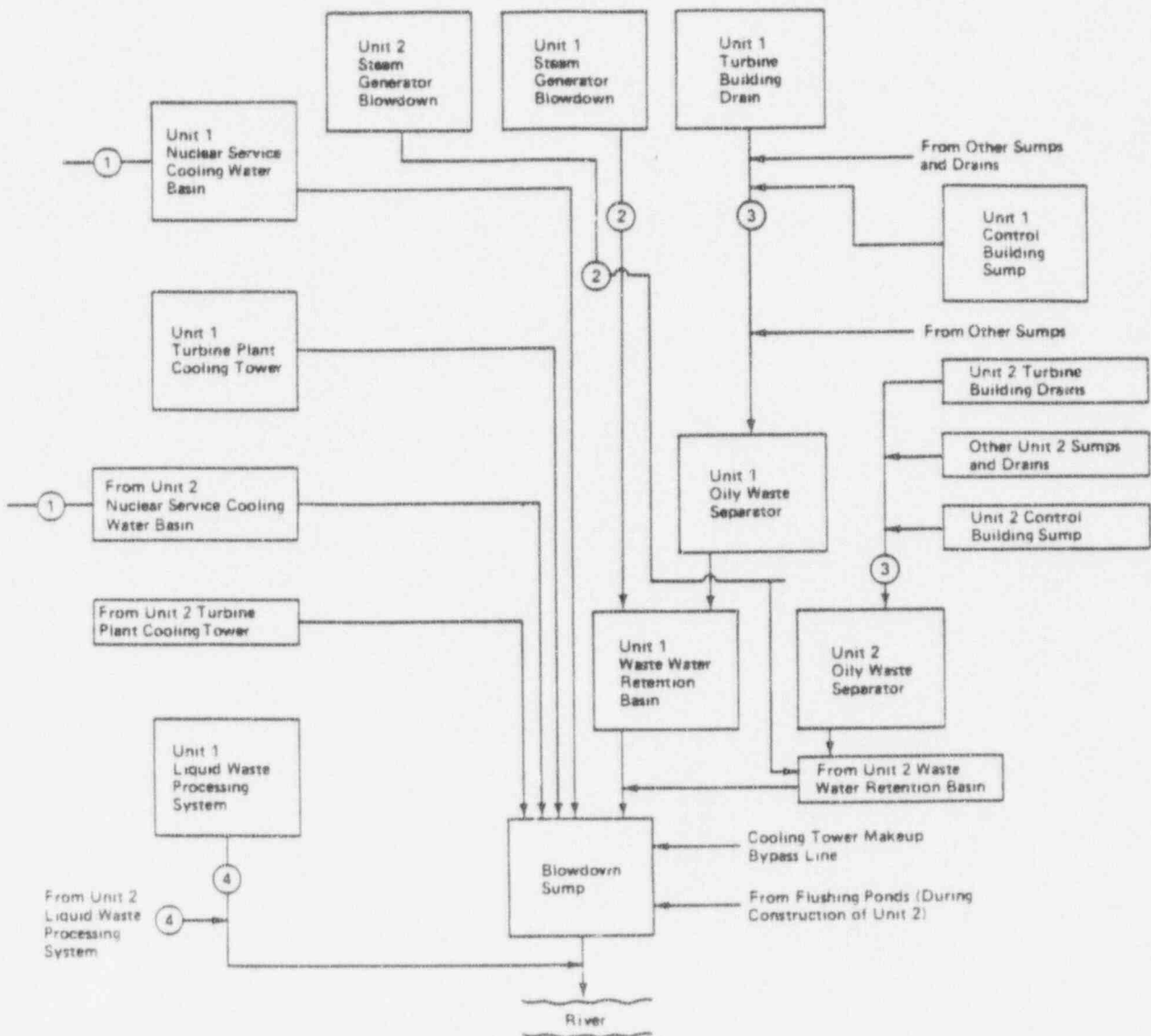


VOGTLE  
ELECTRIC GENERATING PLANT  
UNIT 1 AND UNIT 2

LIQUID WASTE PROCESSING SYSTEM  
(Unit 2)

FIGURE 1.6-2





Radioactivity Monitors:

- ① 1(2) RE0020
- ② 1(2) RE0021
- ③ 1(2) RE0848
- ④ 1(2) RE0018

\*The blowdown sump is common to both units.

Georgia Power

VOGTLE  
ELECTRIC GENERATING PLANT  
UNIT 1 AND UNIT 2

LIQUID DISCHARGE PATHWAYS

FIGURE 1.6-3

## 2.0 GASEOUS EFFLUENTS

At Plant Vogtle, there are five potential points where radioactivity is released to the atmosphere in gaseous discharges. These five potential release points are: Unit 1 plant vent; Unit 2 plant vent; Unit 1 and Unit 2 turbine building vents, which are not normal release pathways until a primary to secondary leak exists; and the dry active waste processing building vent.

The turbine building vent serves as the discharge point for the condenser air ejector and steam packing exhaust system. The fuel handling building is common to both units; however, ventilation from this area is through the Unit 1 plant vent. Certain components of the Gaseous Waste Processing System are shared between the two units and releases from this system are through the Unit 1 plant vent. Containment building releases are through the respective plant vents.

Gaseous releases from the turbine building vents, the radwaste solidification building vent, and the dry active waste processing building vent are considered to be ground-level releases. Gaseous releases from the plant vents are considered to be mixed-mode releases, as determined by the wake-split model. (See NOTE in subsection 2.1.2.) All six release points are considered to be continuous releases. In the absence of primary to secondary leak(s), the turbine building vents are not release points.

Gaseous effluent monitor setpoints are required only for noble gas monitors serving the five release points excluding the dry active waste processing building. Methodology for calculating noble gas monitor setpoints is presented in section 2.1. Although setpoint calculations are not required for radioiodine and particulate monitors, the methodology for assuring the potential organ dose rates due to radioiodines, tritium, and particulates in gaseous releases from the site do not exceed the limits of subsection 2.5.1 is presented in the NOTE in paragraph 2.2.1.2.

## 2.1 GASEOUS EFFLUENT MONITOR SETPOINTS

The setpoint for a particular noble gas monitor is determined by calculating a basic setpoint value,  $c_b$ , which will assure the limits of subsection 2.5.1 are not exceeded. |

Subsections 2.1.1, 2.1.2, and 2.1.3 present the methodology for calculating the basic setpoint value,  $c_b$ , for the respective noble gas monitors. Monitor response and background considerations are discussed in subsection 2.1.4.

Monitor setpoints determined in accordance with this subsection will be regarded as upper bounds for the high alarm setpoint. However, a lower setpoint may be established on the monitor, if desired. Also, intermediate alarm setpoints should be established at a level below the high alarm setpoint to give appropriate warning prior to reaching the high alarm setpoint.

If no release is planned for a particular pathway or if there is no detectable activity in the planned release, the setpoint must be calculated in accordance with subsection 2.1.1 or 2.1.2, assuming Kr-88 as the radionuclide being released.

If a calculated setpoint is less than the current monitor reading associated with the particular release pathway, no release may be made under current conditions. Steps must be taken to reduce contributing source terms and/or reassign allocation factors (discussed in subsection 2.1.5) and the setpoint recalculated, if releases via the pathway under consideration are to continue.

### 2.1.1 Unit 1 Turbine Building Vent and Unit 2 Turbine Building Vent

Monitors: 1RE-12839C (Unit 1) and 2RE-12839C (Unit 2)

NOTE: Turbine building vent serves as the release point for the condenser air ejector and the steam packing exhauster.

For the purpose of implementation of subsection 2.5.4, the setpoint for these noble gas monitors will be calculated as follows:

$c_b$  = the calculated basic setpoint value.

$$(AG) (SF) (R_s) (D_{T_s}) \quad (1)$$

$c_b$  = the lesser of \_\_\_\_\_ or \_\_\_\_\_

$$(AG) (SF) (R_s) (D_{s_s}) \quad (2)$$

SF = the safety factor, which is a conservative factor applied to each noble gas monitor to compensate for statistical fluctuations and errors of measurement. The value of the safety factor must be between 0 and 1; a value of 0.4 to 0.6 is a reasonable range of values for gaseous releases. A more precise value may be developed, if desired.

AG = an administrative allocation factor applied to apportion the release setpoints among all gaseous release discharge pathways to assure that release limits will not be exceeded by simultaneous releases. The allocation factor for a particular discharge pathway may be assigned any desired value between 0 and 1 under the condition that the sum of the allocation factors for all simultaneous release pathways does not exceed 1. For ease of implementation, AG may be set equal to  $1/n$ ,

where  $n$  is the number of simultaneous final gaseous release points. For a more exact determination of allocation factors, see subsection 2.1.5.

$D_{TB}$  = the dose rate limit to the total body of an individual which is 500 mrem/year.

$R_t$  = the relationship between the noble gas concentration and the dose rate to the total body for the conditions of the release under consideration.

$$R_t = C_m + ((\overline{X/Q})_G \sum_i K_i Q_{ig}) \quad (3)$$

where:

$C_m$  = the noble gas grab sample radionuclide concentration taken in accordance with subsection 2.5.2, the release point under consideration.

$(\overline{X/Q})_G$  = the highest annual average relative concentration at the site boundary. The release points addressed in this subsection are ground-level releases.

$(\overline{X/Q})_G = 2.55 \times 10^{-6} \text{ s/m}^3$  in the NE sector.

$K_i$  = the total-body dose factor due to gamma emissions from radionuclide  $i$  (mrem/year/ $\mu\text{Ci/m}^3$ ) from table 2.1-1.

$Q_{ig}$  = the rate of release of noble gas radionuclide  $i$  ( $\mu\text{Ci/s}$ ) from the vent release pathway under consideration (ground level),

which is the product of  $X_{i,v}$  and  $F_v$ , where  $X_{i,v}$  is the concentration of radionuclide  $i$  for the particular release and  $F_v$  is the maximum expected release flowrate for this release point ( $X_{i,v}$  in  $\mu\text{Ci/ml}$  and  $F_v$  in  $\text{ml/s}$ ).

$D_{ss}$  = the dose rate limit to the skin of the body of an individual in an unrestricted area which is 3000 mrem/year.

$R_s$  = the relationship between the noble gas concentration and the dose rate to the skin for the conditions of the release under consideration.

$$R_s = C_m + ((\overline{X/Q})_G \sum_i (L + 1.1 M_i) Q_{ig}) \quad (4)$$

where:

$L_i$  = the skin dose factor due to beta emissions from radionuclide  $i$  (mrem/year/ $\mu\text{Ci/m}^3$ ) from table 2.1-1.

1.1 = the mrem skin dose per mrad air dose.

$M_i$  = the air dose factor due to gamma emissions from radionuclide  $i$  (mrad/year/ $\mu\text{Ci/m}^3$ ) from table 2.1-1.

### 2.1.2 Unit 1 Plant Vent and Unit 2 Plant Vent

Monitors: 1RE-12442C (Unit 1), 2RE-12442C (Unit 2)  
1RE-12444C (Unit 1), 2RE-12444C (Unit 2)

Gaseous releases from the plant vent(s) are regarded as mixed-mode releases in that under certain conditions of vent exit velocity and meteorological conditions, the plume will behave as an elevated release. Under other conditions of vent exit velocity and meteorological conditions, the plume will behave as a ground-level release. Using the wake-split model, dispersion values have been calculated utilizing expected annual average conditions.

The setpoint calculation methodology presented in subsection 2.1.1 applies to these monitors with the exception that  $Q_{i0}$  must be replaced with  $Q_{im}$  and  $(\overline{X/Q})_0$  must be replaced with  $(\overline{X/Q})_m$ , where  $Q_{im}$  and  $(\overline{X/Q})_m$  are defined as follows:

$Q_{im}$  = the rate of release of noble gas radionuclide  $i$  ( $\mu\text{Ci/s}$ ) from the plant vent release pathway under consideration (mixed mode), which is the product of  $X_{iv}$  and  $F_v$ , where  $X_{iv}$  is the concentration of radionuclide  $i$  for the particular release and  $F_v$  is the maximum expected release flowrate for this release point ( $X_{iv}$  in  $\mu\text{i/ml}$  and  $F_v$  in  $\text{ml/s.}$ ).

$(\overline{X/Q})_m$  = the highest annual average relative concentration for a mixed-mode release release type ( $\text{s/m}^3$ ). Currently  $4.62 \times 10^{-7} \text{ s/m}^3$  in the NE sector.

### 2.1.3 Gaseous Waste Processing System Discharge and Reactor Containment Purge

Monitors: ARE0014, 1RE-2565C (Unit 1) and 2RE-2565C (Unit 2)

The Gaseous Waste Processing System discharges to the Unit 1 plant vent, Unit 1 containment purge discharges to the Unit 1 plant vent, and Unit 2 containment purge discharges to the Unit 2 plant vent. The plant vents are equipped with continuous final effluent monitors as discussed in subsection 2.1.2. However, due to the potential significance of releases from these sources, the setpoint methodology is also presented for the system effluent monitors. The system monitors have the control logic to terminate the release at alarm trip point. The final monitors have no trip logic.

Sampling and analyses are completed, and monitor setpoints determined prior to release. These setpoints must take into account simultaneous release pathways; the combined allocation factors for contributing pathway monitors must not exceed the allocation factor for the final release pathway monitor to which they contribute.

Downstream monitors must also take into consideration the combinations of source terms for the particular release pathway.

#### 2.1.3.1 Gaseous Waste Processing System

Monitor: ARE-0014

The Gaseous Waste Processing System discharges through the Unit 1 plant vent; therefore, the Gaseous Waste Processing System effluent monitor is not the final monitor for releases from this system. However, because of the significance of this release pathway and because the Unit 1 plant vent monitor setpoint has to accommodate releases from the Gaseous Waste Processing System, and the trip logic is associated with this monitor, the setpoint methodology for this monitor is presented.

The methodology presented in subsection 2.1.2 applies to this monitor with the following five exceptions:

Exception 1:

$$R_t = r_t = C_m + ((\overline{X/Q})_M \sum_i K_i q_i) \quad (5)$$

Exception 2:

$C_m$  = the Gaseous Waste Processing System noble gas concentration to be discharged (sample taken and analyzed prior to discharge).

Exception 3:

$q_i$  = the rate of release of noble gas radionuclide  $i$  ( $\mu\text{Ci/s}$ ) from the Gaseous Waste Processing System, determined by multiplying the expected release rate by the concentration of radionuclide  $i$ .



Exception 4:

$$R_s = r_s = C_m + ((\bar{X}/Q)_M \sum_i (L_i + 1.1M_i)q_i) \quad (6)$$

Exception 5:

AG = a selected allocation factor value which must be less than the allocation factor for the monitor serving the final release point, Unit 1 plant vent.

When releases are to be made from the Gaseous Waste Processing System, it will be necessary to redetermine the setpoint for the Unit 1 plant vent monitor (1RE-12442C or 1RE-12444C), which is downstream from the Gaseous Waste Processing System effluent monitor (ARE-0014).

Redetermination of this setpoint is accomplished by applying the methodology of subsection 2.1.2 with the following two exceptions:

Exception 1:

A new source term,  $(Q_{im})_{GP}$ , which includes the routine Unit 1 plant vent source term  $Q_{im}$  (based on sample results from subsection 2.5.2) combined with the Gaseous Waste Processing System source term for the tank planned for release,  $q_i$ , as follows:

$$(Q_{im})_{GP} = Q_{im} + q_i \quad (7)$$

Exception 2:

$C_m$  = the noble gas concentration determined for the combined release; this concentration value is obtained by dividing  $\sum_i (Q_{im})_{GP}$  by the combined flowrate (ml/s) through the plant vent.

### 2.1.3.2 Reactor Containment Purge

Monitors: 1(2)RE-2565C (one for each unit)

Unit 1 containment purge discharges through the Unit 1 plant vent; Unit 2 containment purge discharges through the Unit 2 plant vent. Therefore, the containment purge monitor is not the final monitor for containment purge releases. However, because of the significance of these releases and because the respective plant vent monitor setpoint has to accommodate containment purge releases, the setpoint methodology for these monitors is presented.

The methodology presented in subsection 2.1.2 applies to this monitor with the following five exceptions:

Exception 1:

$$R_t = r_t = C_m + ((\overline{X/Q})_M \sum_i K_i q_i) \quad (8)$$

Exception 2:

$C_m$  = the containment purge noble gas concentration to be discharged (sample taken and analyzed prior to discharge).

Exception 3:

$q_i$  = the rate of release of noble gas radionuclide  $i$  ( $\mu\text{Ci/s}$ ) from containment purge, determined by multiplying the expected release rate by the concentration of radionuclide  $i$ .

Exception 4:

$$R_s = r_s = C_m + ((\overline{X/Q})_M \sum_i (L_i + 1.1M_i) q_i) \quad (9)$$

Exception 5:

AG = a selected allocation factor value which must be less than the allocation factor for the monitor serving the final release point; the respective plant vent.

When containment purge releases are to be made, it will be necessary to redetermine the setpoint for the respective plant vent monitor (1(2)RE-12442C) which is downstream from the containment purge monitor (1(2)RE-2565C).

Redetermination of this setpoint is accomplished by applying the methodology of subsection 2.1.2 with the following two exceptions:

Exception 1:

A new source term,  $(Q_{im})_{CP}$ , is determined which includes the routine respective plant vent source term  $Q_{im}$  (based on sample results from subsection 2.5.2) combined with the containment purge source term for the unit containment planned for release,  $q_i$ , as follows:

$$(Q_{im})_{CP} = Q_{im} + q_i \quad (10)$$

Exception 2:

$C_m$  = the noble gas concentration determined for the combined release; this concentration value is obtained by dividing  $\sum_i (Q_{im})_{CP}$  by the flowrate (ml/s) through the plant vent.

2.1.4 Consideration of Monitor Response and Background in Establishing Gaseous Effluent Monitor Setpoints

The calculated monitor setpoint,  $c_s$ , establishes the base value for the monitor setpoint. The monitor setpoint thus calculated is in the units

$\mu\text{Ci/ml}$ . The monitor actually measures counts per minute, subtracts a predetermined background count rate, and then multiplies by a calibration factor to convert the counts per minute to  $\mu\text{Ci/ml}$ .

Calibration of the gas monitors by the manufacturer and Georgia Power Company utilized at least one NBS traceable gaseous radionuclide source, except for the main steam lines in the exact geometry of each production monitor and point sources covering the Beta end point energy range for 0.293 to at least 1.488 MeV. The calibration factor is a function of the radionuclide mix in the gas to be released and will be calculated for the monitor based on the results of the predischARGE sample results from the laboratory gamma-ray spectrometer system.

The mix-dependent calibration factor will be used as the gain factor in the PERMS monitor or used to modify the calculated base monitor setpoint so the default calibration factor can be left in the PERMS monitor.

Contributions to background radiation levels may include ambient background, plant environmental background at monitor locations when plant is in shutdown status, plant environmental background at monitor location when plant is at power, and internal background of monitor due to contamination of sample chamber. Background levels must be controlled such that radioactivity levels in the effluent stream being monitored can be accurately assessed at or below the calculated setpoint value.

#### 2.1.5 Determination of Allocation Factor, AG

When simultaneous gaseous releases are made to the environment, an (administrative) allocation factor must be applied to each discharge pathway. This is to ensure that simultaneous gaseous releases from the site to unrestricted areas will not exceed the dose rate limits specified in Technical Specification 3.11.2.1. For Plant Vogtle, final discharge pathways which may be released simultaneously are the Unit 1 plant vent, Unit 2 plant

vent, Unit 1 turbine building vent, Unit 2 turbine building vent, and the DAW processing building vent. The allocation factor for each gaseous discharge pathway must be between 0 and 1 and the sum of the allocation factors for the simultaneous releases must not exceed 1.

There are three methods by which allocation factors may be determined:

1. The allocation factor for a particular release pathway may be administratively selected based on an estimate of the fraction of the total dose rate (from all simultaneous releases) which is contributed by the particular release pathway. If the building is not in service for its intended function an allocation factor of zero can be used for that building.
2. The allocation factor may be calculated using the expression

$$AG = 1/n \quad (11)$$

where: n = the number of release pathways to be released simultaneously.

3. The allocation factor may be determined for a particular discharge pathway by calculating the ratio of the total-body dose rate due to noble gases released from the particular discharge pathway under consideration to the total-body dose rate due to noble gases in all simultaneous releases, as follows:

For Unit 1 turbine building vent and Unit 2 turbine building vent (ground-level releases):

$$AG = \frac{(\bar{X}/Q)_G \sum_i K_i Q_{ig}(r)}{(\bar{X}/Q)_G \sum_{ng} \sum_i K_i (Q_{ig})_{ng} + (\bar{X}/Q)_M \sum_{nm} \sum_i K_i (Q_{im})_{nm}} \quad (12)$$

For Unit 1 plant vent and Unit 2 plant vent (mixed mode releases),

$$AG = \frac{(\overline{X/Q})_M \sum_i K_i Q_{im}(r)}{(\overline{X/Q})_G \sum_{ng} \sum_i K_i (Q_{ig})_{ng} + (\overline{X/Q})_M \sum_{nm} \sum_i K_i (Q_{im})_{nm}} \quad (13)$$

where ng is the number of simultaneous vent releases (ground level); nm is the number of simultaneous vent releases (mixed mode); and (r) is the particular discharge pathway for which an allocation factor is being determined. When using equations 12 and 13, be sure to include the DAW processing building vent for the particulate releases.

#### 2.1.6 Particulate Effluent Monitor Setpoints

Particulate effluent monitor setpoints for RE-12442A, RE-2565A, and ARE-13256 are determined using the method outlined for gaseous effluent monitor setpoints. Co-60 and lung organ were used as the main isotope and organ. Reference 18 details the way the initial setpoint calculations were made for certain monitors.

TABLE 2.1-1

## DOSE FACTORS FOR EXPOSURE TO A SEMI-INFINITE CLOUD OF NOBLE GASES\*

Nuclide	Gamma	Beta	Gamma	Beta
	-Body*** (K)	-Skin*** (L)	-Air** (M)	-Air** (N)
Kr-83m	7.56E-02		1.93E+01	2.88E+02
Kr-85m	1.17E+03	1.46E+03	1.23E+03	1.97E+03
Kr-85	1.61E+01	1.34E+03	1.72E+01	1.95E+03
Kr-87	5.92E+03	9.73E+03	6.17E+03	1.03E+04
Kr-88	1.47E+04	2.37E+03	1.52E+04	2.93E+03
Kr-89	1.66E+04	1.01E+04	1.73E+04	1.06E+04
Kr-90	1.56E+04	7.29E+03	1.63E+04	7.83E+03
Xe-131m	9.15E+01	4.76E+02	1.56E+02	1.11E+03
Xe-133m	2.51E+02	9.94E+02	3.27E+02	1.48E+03
Xe-133	2.94E+02	3.06E+02	3.53E+02	1.05E+03
Xe-135m	3.12E+03	7.11E+02	3.36E+03	7.39E+02
Xe-135	1.81E+03	1.86E+03	1.92E+03	2.46E+03
Xe-137	1.42E+03	1.22E+04	1.51E+03	1.27E+04
Xe-138	8.83E+03	4.13E+03	9.21E+03	4.75E+03
Ar-41	8.84E+03	2.69E+03	9.30E+03	3.28E+03

\* Values taken from Reference 3, Table B-1.

\*\*  $\frac{\text{mrad-m}^3}{\mu\text{Ci-year}}$ \*\*\*  $\frac{\text{mrem-m}^3}{\mu\text{Ci-year}}$

TABLE 2.1-2  
GASEOUS RELEASE POINTS FLOWRATES

<u>Release Point</u>	<u>Flowrates</u>	
	<u>ft<sup>3</sup>/min</u>	<u>ml/s</u>
Plant Vent (Unit 1)	1.5E05	7.08E07
Plant Vent (Unit 2)	9.0E04	4.25E07
Turbine Building Vent (Unit 1; Condenser Air Ejector and Steam Packing Exhaust)	9.0E02	4.25E05
Turbine Building Vent (Unit 2; Condenser Air Ejector and Steam Packing Exhaust)	9.0E02	4.25E05
Dry Active Waste Processing Building Vent	2.20E3	1.04E6

Reference 5, Tables 11.5.2-1 and 11.5.5-1.



## 2.2 GASEOUS EFFLUENT DOSE RATE AND DOSE CALCULATIONS

### 2.2.1 Dose Rates at or Beyond Site Boundary

#### 2.2.1.1 Dose Rates Due to Noble Gases

For the purpose of implementing subsection 2.5.1, the dose rate in areas at or beyond the site boundary due to noble gases shall be calculated as follows:

$D_t$  = the total-body dose rate at time of release (mrem/year).

$$= \left[ (\overline{X/Q})_G \sum_{ng} \sum_i K_i (Q_{ig})_{ng} \right] + \left[ (\overline{X/Q})_M \sum_{nm} \sum_i K_i (Q_{im})_{nm} \right] \quad (14)$$

$D_s$  = the skin dose rate at time of release (mrem/year).

$$= \left[ (\overline{X/Q})_G \sum_{ng} \sum_i (L_i + 1.1M_i)(Q_{ig})_{ng} \right] + \left[ (\overline{X/Q})_M \sum_{nm} \sum_i (L_i + 1.1M_i)(Q_{im})_{nm} \right] \quad (15)$$

Where:  $ng$  = the number of simultaneous ground-level vent releases, and  $nm$  = the number of simultaneous mixed-mode vent releases. Other terms were defined previously in subsection 2.1.

The dose rate limits are site limits at any point in time; therefore, dose rates are summed over all gaseous releases occurring simultaneously. For Plant Vogtle, Unit 1 turbine building vent and Unit 2 turbine building vent are ground-level releases. Unit 1 plant vent and Unit 2 plant vent are mixed-mode releases.

### 2.2.1.2 Dose Rates Due to I-131, I-133, Tritium, and Particulates

For the purpose of implementing subsection 2.5.1, organ dose rates due to I-131, I-133, tritium, and all radioactive materials in particulate form with half-lives greater than eight days, are required to be calculated for the inhalation pathway for the child age group. The child age group would experience the highest potential dose rate via the inhalation pathway. These dose rates are calculated as follows:

$$D_o = \text{the organ dose rate at time of release (mrem/year).}$$

$$= (\overline{X/Q})_G \sum_{ng} \sum_i P_{io} (Q'_{ig})_{ng} + (\overline{X/Q})_M \sum_{nm} \sum_i P_{io} (Q'_{im})_{nm} \quad (16)$$

where:

nm and ng = defined previously in paragraph 2.2.1.1.

$(\overline{X/Q})_G$  = defined previously in subsection 2.1.1.

$(\overline{X/Q})_M$  = defined previously in subsection 2.1.2.

$Q'_{ig}$  = the release rate ( $\mu\text{Ci/s}$ ) of I-131, I-133, tritium, and particulates from Unit 1 turbine building vent; Unit 2 turbine building vent; and dry active waste processing building vent which are ground-level releases. NOTE: DAW is particulates only.

$Q'_{im}$  = the release rate ( $\mu\text{Ci/s}$ ) of I-131, I-133, tritium, and particulates from Unit 1 plant vent and Unit 2 plant vent, which are mixed-mode releases.

$P_{io}$  = the organ dose parameter for organ o and radionuclide i, (mrem/year per  $\mu\text{Ci/m}^3$ ) for inhalation determined as follows:

$$P_{io} = K'(BR) DF_{io} \quad (17)$$

where:

$K'$  = the constant of unit conversion,  $10^6 \text{pCi}/\mu\text{Ci}$ .

BR = the breathing rate for child age group;  $3700 \text{ m}^3/\text{year}$  from table 2.2-10.

$DF_{io}$  = the inhalation pathway dose factor for child age group for organ o and radionuclide i, from table 2.2-2.

NOTE: To ensure that potential dose rates (pre-release) to an organ due to I-131, I-133, tritium, and particulates in simultaneous gaseous releases from the site do not exceed  $1500 \text{ mrem}/\text{year}$  as specified in subsection 2.5.1, the potential organ dose rate  $D_o$  must be limited as follows:

$$D_o + (AG)(SF) \leq 1500 \text{ mrem}/\text{year} \quad (18)$$

where AG and SF are assigned the same values as were used in subsection 2.1 for the gaseous discharge pathway under consideration. To further ensure that dose rate limits were not exceeded (post-release), dose rates from simultaneous releases should be summed, as shown in equation (16) above.

## 2.2.2 Air Doses and Dose to a Member of the Public at or Beyond the Site Boundary

### 2.2.2.1 Air Doses at or Beyond the Site Boundary

For the purpose of implementing subsection 2.5.2, air doses in areas at or beyond the site boundary shall be determined as follows:

$D_{\text{gamma}}$  = the air dose due to gamma emissions from noble gas radionuclides (mrad)

$$= 3.17 \times 10^{-8} (\overline{X/Q})_G \sum_i M_i \overline{Q}_{ig} + (\overline{X/Q})_M \sum_i M_i \overline{Q}_{im} \quad (19)$$

where:

$3.17 \times 10^{-8}$  = the fraction of one year/one second.

$\overline{Q}_{ig}$  = the cumulative release of noble gas radionuclide  $i$  over the period of interest ( $\mu\text{Ci}$ ) from the vent release (ground level) under consideration.

$\overline{Q}_{im}$  = cumulative release of noble gas radionuclide  $i$  over the period of interest ( $\mu\text{Ci}$ ) from the vent release (mixed-mode) under consideration.

$M_i$  = defined previously in subsection 2.1.1.

$(\overline{X/Q})_G$  =  $2.55 \times 10^{-6} \text{ s/m}^3$  in the NE sector.

$(\overline{X/Q})_M$  =  $4.62 \times 10^{-7} \text{ s/m}^3$  in the NE sector.

$D_{\text{beta}}$  = the air dose due to beta emissions from noble gas radionuclides (mrad).

$$= 3.17 \times 10^{-8} (\overline{X/Q})_G \sum_i N_i \overline{Q}_{ig} + (\overline{X/Q})_M \sum_i N_i \overline{Q}_{im} \quad (20)$$

where:

$N_i$  = the air dose factor due to beta emissions from noble gas radionuclide  $i$  (mrad/year/ $\mu\text{Ci/m}^3$ ), from table 2.1-1.

#### 2.2.2.2 Dose to a Member of the Public at or Beyond the Site Boundary

Doses to a member of the public due to I-131, I-133, tritium, and radioactive materials in particulate form, in gaseous releases, will be calculated for the purpose of implementing subsection 2.5.3 as follows:

(NOTE: The member of the public expected to receive the highest dose in the plant vicinity is referred to as the controlling (or critical) receptor. The dose received depends on the location, age group, and exposure pathways present. For Plant Vogtle, the controlling receptor(s) for which doses must be calculated, and the applicable exposure pathways, are presented in table 2.2-12.)

$$D_j = \text{the dose to an organ } j \text{ of an individual in age group } a \text{ from radioiodines, tritium, and radionuclides in particulate form with half-lives greater than 8 days (mrem).}$$
$$= 3.17 \times 10^{-8} \sum_{pi} R_{aipj} (W'_{GP} \bar{Q}'_{ig} + W'_{MP} \bar{Q}'_{im}) \quad (21)$$

where:

$3.17 \times 10^{-8}$  = the fraction of one year/one second.

$pi$  = for all pathways and all isotopes.

$W'_{GP}$  = the pathway-dependent relative dispersion or deposition at the location of the controlling receptor, associated with ground-level plant releases as follows:

$(\bar{X}/Q')_{GP}$  = the annual average relative dispersion parameter for location of controlling (critical) receptor for ground-level plant releases.  $(\bar{X}/Q')_{GP}$  applies only to inhalation and all tritium pathways. (For all tritium pathways the  $Q'_i$  source term is limited to tritium.) See table 2.2-12 for value.

$(\overline{D/Q'})_{GP}$  = the annual average deposition parameter for the location of controlling (critical) receptor for ground-level vent releases.  $(\overline{D/Q'})_{GP}$  applies to all other pathways. See table 2.2-12 for value.

$W'_{MP}$  = the pathway-dependent relative dispersion or deposition at the location of the controlling receptor, associated with plant vent releases, which are mixed mode as follows:

$(\overline{X/Q'})_{MP}$  = the annual average relative dispersion parameter for location of controlling (critical) receptor for mixed-mode releases.  $(\overline{X/Q'})_{MP}$  applies only to inhalation and all tritium pathways. (For all tritium pathways, the  $Q'i$  source term is limited to tritium.) See table 2.2-12 for values.

$(\overline{D/Q'})_{MP}$  = the annual average deposition parameter for the location of controlling (critical) receptor for mixed-mode releases.  $(\overline{D/Q'})_{MP}$  applies to all other pathways. See table 2.2-12 for values.

The selection of the dispersion or deposition parameter,  $X/Q$  or  $D/Q$ , is dependent upon the pathway being considered. The dispersion parameter,  $X/Q$ , is required for the inhalation pathway and the tritium contribution to ingestion pathways, since tritium is taken up by vegetation directly from the surrounding air. The deposition parameter,  $D/Q$ , is required for the ground plane pathway and I-131, I-133, and particulate contributions to ingestion pathways.

$\bar{Q}'_i$  = the cumulative release ( $\mu\text{Ci}$ ), from ground-level plant releases, of radionuclide  $i$ , as required by subsection 2.5.3 over the period of interest. Dose determinations required by

subsection 2.5.3 are on a per reactor basis; therefore, cumulative release quantities must also be unit-specific. Since the dry active waste processing building serves both units, release quantities must be apportioned between the two units. In absence of evidence that one unit contributes a greater quantity of radioactivity than the other over the period of interest, release quantities may be apportioned equally between the two units. (For dose contributions due to tritium from the ingestion pathways, the  $\bar{Q}_{i_0}$  term is limited to tritium.)

- $Q'_{i_m}$  = the cumulative release ( $\mu\text{Ci}$ ), from the mixed-mode plant vent releases, of radionuclide  $i$  as required by subsection 2.5.3 over the period of interest. Dose determinations required by subsection 2.5.3 are on a per reactor basis; therefore, cumulative release quantities must also be unit-specific. (For dose contributions due to tritium from the ingestion pathways, the term  $\bar{Q}_{i_m}$  is limited to tritium.)
- $R_{a,p,j}$  = the pathway-specific, individual age-specific, organ dose factor for radionuclide  $i$ , pathway  $p$ , organ  $j$ , and individual age group,  $a$ . Routine individual dose calculations address the inhalation, ground plane, grass-cow-milk, grass-goat-milk, grass-cow-meat, and garden vegetation pathways. However, the dose pathways actually present at the controlling location, as well as the controlling individual age group, are determined through the Land Use Census for the site vicinity and are presented in table 2.2-12. Pathway factors  $R_{a,p,j}$  are determined as shown in the following subsections.

Plant Vogtle site-specific values, or appropriate default values, required in the pathway factor determinations are presented in table 2.2-13.

### Inhalation Pathway Factor

$$R_{aipj} = K'(BR)_a (DFA_{ij})_a \text{ mrem/year}/\mu\text{Ci}/\text{m}^3 \quad (22)$$

where:

$K'$  = the constant of unit conversion  $10^6 \text{ pCi}/\mu\text{Ci}$ .

$(BR)_a$  = the breathing rate for a particular age group in  $\text{m}^3/\text{year}$  from table 2.2-10.

$DFA_{ij}$  = the inhalation dose factor for receptor age group a, organ j, and for radionuclide i, in  $\text{mrem}/\text{pCi}$  from tables 2.2-1 through 2.2-4.

### Ground-Plane Pathway Factor

$$R_{aipj} = K'K'' (\text{SHF})(\text{DFG}_{ij}) ((1-e^{-\lambda_i t})/\lambda_i) \quad (23)$$

$(\text{m}^2\text{mrem}/\text{year}/\mu\text{Ci}/\text{s}).$

where:

$K'$  = the constant of unit conversion,  $10^6 \text{ pCi}/\text{mCi}$ .

$K''$  = the constant of unit conversion, 8760 h/year.

$\text{SHF}$  = the shielding factor, 0.7 (dimensionless).

$\text{DFG}_{ij}$  = the ground-plane dose conversion factor for radionuclide i (same for all age groups and specific organs are assumed to receive the same dose as the total body) ( $\text{mrem}/\text{h}/\text{pCi}/\text{m}^2$ ) table 2.2-9.

$\lambda_i$  = the decay constant for radionuclide i, in  $\text{s}^{-1}$ .

$t$  = the exposure time,  $4.73 \times 10^6 \text{ s}$  (15 years).



### Vegetation Pathway Factor

$$R_{aipj} = K' \frac{r}{Y_V(\lambda_i + \lambda_w)} (DFL_{ij})_a (U_{al} f_l e^{-\lambda_i t_l} + U_{as} f_g e^{-\lambda_i t_{hv}}) \quad (24)$$

(m<sup>2</sup>mrem/year per  $\mu$ Ci/s)

where:

- $K'$  = a constant of unit conversion,  $10^6$  pCi/ $\mu$ Ci.
- $r$  = the fraction of deposited activity retained on vegetation (1.0 for radioiodines; 0.2 for particulates).
- $U_{al}$  = the consumption rate of fresh leafy vegetation by the receptor in age group a, in kg/year. (See table 2.2-10.)
- $U_{as}$  = the consumption rate of stored vegetation by the receptor in age group a, in kg/year. (See table 2.2-10.)
- $f_l$  = the fraction of the annual intake of fresh leafy vegetation grown locally.
- $f_g$  = the fraction of the annual intake of stored vegetation grown locally.
- $t_l$  = the average time between harvest of leafy vegetation and its consumption in s, ( $8.6 \times 10^4$ ).
- $t_{hv}$  = the average time between harvest of stored vegetation and its consumption in s, ( $5.18 \times 10^6$ ).
- $Y_V$  = the vegetation areal density, in kg/m<sup>2</sup>.
- $(DFL_{ij})_a$  = the organ ingestion dose factor for the  $i$ th radionuclide for the receptor in age group a, in mrem/pCi from tables 2.2-5 through 2.2-8.

- $\lambda_i$  = the decay constant for the  $i$ th radionuclide, in  $s^{-1}$ .
- $\lambda_w$  = the decay constant for removal of activity on leaf and plant surfaces by weathering,  $5.73 \times 10^{-7} s^{-1}$  (corresponding to a 14-day half-life).

For tritium in vegetation, the vegetation pathway factor is a special case due to the fact that the concentration of tritium in vegetation is based on airborne concentration rather than deposition:

$$R_{a,ipj} = K'K''(U_{a,i}f_i + U_{a,s}f_s)(DFL_{ij})_a(0.75(0.5/H)) \quad (25)$$

(mrem/year/ $\mu$ Ci/ $m^3$ )

where:

- $K''$  = a constant of unit conversion,  $10^3$  gm/kg.
- $H$  = the absolute humidity of the atmosphere, in gm/ $m^3$ .
- 0.75 = the fraction of total vegetation that is water.
- 0.5 = the ratio of the specific activity of the vegetation water to the atmospheric water.

Other parameters and values are given above.

#### Grass-Cow-Milk Pathway Factor

$$R_{a,ipj} = K' \frac{Q_F(U_{ap})}{\lambda_i + \lambda_w} F(r)(DFL_{ij})_a \left[ \frac{f_p f_s}{Y_p} + \frac{(1-f_p f_s)e^{-\lambda_i t_{hm}}}{Y_s} \right] e^{-\lambda_i t_f} \quad (26)$$

( $m^2$ mrem/year/ $\mu$ Ci/s)

where:

- $K'$  = a constant of unit conversion,  $10^6$  pCi/ $\mu$ Ci.
- $Q_r$  = the cow's consumption rate, in kg/day (wet weight).
- $U_{a,p}$  = the receptor's milk consumption rate for age group a, in liters/year from table 2.2-10.
- $Y_p$  = the agricultural productivity by unit area of pasture feed grass, in kg/m<sup>2</sup>.
- $Y_s$  = the agricultural productivity by unit area of stored feed, in kg/m<sup>2</sup>.
- $F_m$  = the stable element transfer coefficients, in days/liter. (See table 2.2-11.)
- $r$  = the fraction of deposited activity retained on feed grass (1.0 for radioiodines; 0.2 for particulates).
- $(DFL_{ij})_a$  = the organ ingestion dose factor for the ith radionuclide for the receptor in age group a, in mrem/pCi from tables 2.2-5 through 2.2-8.
- $\lambda_i$  = the decay constant for the ith radionuclide, in s<sup>-1</sup>.
- $\lambda_w$  = the decay constant for removal of activity on leaf and plant surfaces by weathering,  $5.73 \times 10^{-7}$  s<sup>-1</sup> (corresponding to a 14-day half-life).
- $t_r$  = the transport time from pasture to cow, to milk, to receptor, in s ( $1.73 \times 10^5$ ).
- $t_{rm}$  = the transport time from pasture, to harvest, to cow, to milk, to receptor, in s ( $7.78 \times 10^6$ ).

- $f_p$  = the fraction of the year that the cow is on pasture (dimensionless).
- $f_s$  = the fraction of the cow feed that is pasture grass while the cow is on pasture (dimensionless).

For tritium in milk, the grass-cow-milk pathway factor is a special case due to the fact that the concentration of tritium in milk is based on airborne concentration rather than deposition:

$$R_{aipj} = K'K''F_m Q_F U_{ap} (DFL_{ij})_a (0.75(0.5/H)) \quad (27)$$

(mrem/year/ $\mu$ Ci/ $m^3$ )

where:

- $K''$  = a constant of unit conversion,  $10^3$  gm/kg.
- $H$  = the absolute humidity of the atmosphere, in gm/ $m^3$ .
- 0.75 = the fraction of total feed that is water.
- 0.5 = the ratio of the specific activity of the feed grass water to the atmospheric water.

Other parameters and values as previously defined.

#### Grass-Goat-Milk Pathway Factor

$$R_{aipj} = K' \frac{Q_F(U_{ap})}{\lambda_i + \lambda_w} F_m(r) (DFL_{ij})_a \left( \frac{f_p f_s}{Y_p} + \frac{(1-f_p f_s)e^{-\lambda_i t_{hm}}}{Y_s} \right) e^{-\lambda_i t_f} \quad (28)$$

( $m^2$ mrem/year/ $\mu$ Ci/s)

where:

- $K'$  = a constant of unit conversion,  $10^6$  pCi/ $\mu$ Ci.

- $Q_r$  = the goat's consumption rate, in kg/day (wet weight).
- $U_{ap}$  = the receptor's milk consumption rate for age group a, in liters/year from table 2.2-10.
- $Y_p$  = the agricultural productivity by unit area of pasture feed grass, in kg/m<sup>2</sup>.
- $Y_s$  = the agricultural productivity by unit area of stored feed, in kg/m<sup>2</sup>.
- $F_m$  = the stable element transfer coefficients, in days/liter. (See table 2.2-11.)
- $r$  = the fraction of deposited activity retained on feed grass (1.0 for radioiodines; 0.2 for particulates).
- $(DFL_{ij})_a$  = the organ ingestion dose factor for the ith radionuclide for the receptor in age group a, in mrem/pCi from tables 2.2-5 through 2.2-8.
- $\lambda_i$  = the decay constant for the ith radionuclide, in s<sup>-1</sup>.
- $\lambda_w$  = the decay constant for removal of activity on leaf and plant surfaces by weathering,  $5.73 \times 10^{-7} \text{ s}^{-1}$  (corresponding to a 14-day half-life).
- $t_r$  = the transport time from pasture to goat, to milk, to receptor, in s ( $1.73 \times 10^5$ ).
- $t_{hm}$  = the transport time from pasture, to harvest, to goat, to milk, to receptor, in s ( $7.78 \times 10^6$ ).
- $f_p$  = the fraction of the year that the goat is on pasture (dimensionless).

- $f_s$  = the fraction of the goat feed that is pasture grass while the goat is on pasture (dimensionless).

For tritium in milk, the grass-goat-milk pathway factor is a special case due to the fact that the concentration of tritium in milk is based on airborne concentration rather than desposition:

$$R_{aipj} = K'K''F_m Q_F U_{ap} (DFL_{ij})_a (0.75(0.5/H)) \quad (29)$$

(mrem/year/ $\mu$ Ci/ $m^3$ )

where:

- $K''$  = a constant of unit conversion,  $10^3$  gm/kg.
- $H$  = the absolute humidity of the atmosphere, in gm/ $m^3$ .
- 0.75 = the fraction of total feed that is water.
- 0.5 = the ratio of the specific activity of the feed grass water to the atmospheric water.

Other parameters and values are given above.

#### Grass-Cow-Meat-Pathway-Factor

$$R_{aipj} = K' \frac{Q_F (U_{ap})}{\lambda_i + \lambda_w} F_f(r) (DFL_{ij})_a \left[ \frac{f_p f_s}{Y_p} + \frac{(1-f_p f_s) e^{-\lambda_i t_{hm}}}{Y_s} e^{-\lambda_i t_f} \right] \quad (30)$$

( $m^2$ mrem/year/ $\mu$ Ci/s)

where:

- $K'$  = a constant of unit conversion,  $10^6$  pCi/ $\mu$ Ci.
- $Q_F$  = the cow's consumption rate, in kg/day (wet weight).

- $U_{a,p}$  = the receptor's meat consumption rate for age group a, in kg/year from table 2.2-10.
- $Y_p$  = the agricultural productivity by unit area of pasture feed grass, in kg/m<sup>2</sup>.
- $Y_s$  = the agricultural productivity by unit area of stored feed, in kg/m<sup>2</sup>.
- $F_r$  = the stable element transfer coefficients, in days/kg. (See table 2.2-11.)
- $r$  = the fraction of deposited activity retained on feed grass (1.0 for radioiodines; 0.2 for particulates).
- $(DFL_{i,j})_a$  = the organ ingestion dose factor for the ith radionuclide for the receptor in age group a, in mrem/pCi from tables 2.2-5 through 2.2-8.
- $\lambda_i$  = the decay constant for the ith radionuclide, in s<sup>-1</sup>.
- $\lambda_w$  = the decay constant for removal of activity on leaf and plant surfaces by weathering,  $5.73 \times 10^{-7}$  s<sup>-1</sup> (corresponding to a 14-day half-life).
- $t_r$  = the transport time from pasture to cow, to meat, to receptor, in s ( $1.73 \times 10^6$ ).
- $t_{rm}$  = the transport time from pasture, to harvest, to cow, to meat, to receptor, in s ( $7.78 \times 10^6$ ).
- $f_p$  = the fraction of the year that the cow is on pasture (dimensionless).
- $f_s$  = the fraction of the cow feed that is pasture grass while the cow is on pasture (dimensionless).

For tritium in meat, the grass-cow-meat pathway factor is a special case due to the fact that the concentration of tritium in meat is based on airborne concentration rather than deposition:

$$R_{a1pj} = K'K''F_pQ_pU_{ap}(DFL_{1j})_a(0.75(0.5/H)) \quad (31)$$

(mrem/year/ $\mu$ Ci/ $m^3$ )

where:

$K''$  = a constant of unit conversion,  $10^3$  gm/kg.

$H$  = the absolute humidity of the atmosphere, in gm/ $m^3$ .

0.75 = the fraction of total feed that is water.

0.5 = the ratio of the specific activity of the feed grass water to the atmospheric water.

Other parameters and values are given above.

#### 2.2.2.3 Dose Calculations to Support Other Technical Specifications

In the event radiological impact assessment becomes necessary to support Technical Specification 6.6.1, which pertains to reportable events, dose calculations may be performed using the equations in subsection 2.2.2.2 with the substitution of average meteorological (dispersion and deposition) parameters for the period covered by the report, and the appropriate pathway dose factors ( $R_{a1pj}$ ) for the receptor of interest.

For the purpose of supporting subsection 3.1.2, which pertains to the Annual Land Use Survey, it may become necessary to perform dose calculations in addition to those required by subsection 2.5.3. In the event that the Land Use Survey reveals that exposure pathways have changed at previously identified locations, or if new locations are identified, it may become



necessary to perform dose calculations at two or more locations to either confirm the previously identified controlling receptor or identify the new receptor which should be designated as the controlling receptor. The necessary dose calculations may be performed using the equations presented in paragraph 2.2.2.2, substituting the appropriate pathway dose factors ( $R_{a,pj}$ ) and the appropriate meteorological (dispersion and deposition) parameters for the receptor(s) and location(s) of interest. Annual average meteorological parameters may be used for these calculations.

TABLE 2.2-1 (SHEET 1 OF 3)

INHALATION DOSE FACTORS FOR INFANT\*  
(mrem per pci inhaled)

Nuclide	Bone	Liver	T Body	Thyroid	Kidney	Lung	GI-LLI
H-3	No Data	4.62E-07	4.62E-07	4.62E-07	4.62E-07	4.62E-07	4.62E-07
C-14	1.89E-05	3.79E-06	3.79E-06	3.79E-06	3.79E-06	3.79E-06	3.79E-06
Na-24	7.54E-06	7.54E-06	7.54E-06	7.54E-06	7.54E-06	7.54E-06	7.54E-06
P-32	1.45E-03	8.03E-05	5.53E-05	No Data	No Data	No Data	1.15E-05
Cr-51	No Data	No Data	6.39E-08	4.11E-08	9.45E-09	9.17E-06	2.55E-07
Mn-54	No Data	1.81E-05	3.56E-06	No Data	3.56E-06	7.14E-04	5.04E-06
Mn-56	No Data	1.10E-09	1.58E-10	No Data	7.86E-10	8.95E-06	5.12E-05
Fe-55	1.41E-05	8.39E-06	2.38E-06	No Data	No Data	6.21E-05	7.82E-07
Fe-59	9.69E-06	1.68E-05	6.77E-06	No Data	No Data	7.25E-04	1.77E-05
Co-58	No Data	8.71E-07	1.30E-06	No Data	No Data	5.55E-04	7.95E-06
Co-60	No Data	5.73E-06	8.41E-06	No Data	No Data	3.22E-03	2.28E-05
Ni-63	2.42E-04	1.46E-05	8.29E-06	No Data	No Data	1.49E-04	1.73E-06
Ni-65	1.71E-09	2.03E-10	8.79E-11	No Data	No Data	5.80E-06	3.58E-05
Cu-64	No Data	1.34E-09	5.53E-10	No Data	2.84E-09	6.64E-06	1.07E-05
Zn-65	1.38E-05	4.47E-05	2.22E-05	No Data	2.32E-05	4.62E-04	3.67E-05
Zn-69	3.85E-11	6.91E-11	5.13E-12	No Data	2.87E-11	1.05E-06	9.44E-06
Br-83	No Data	No Data	2.72E-07	No Data	No Data	No Data	LT E-24
Br-84	No Data	No Data	2.86E-07	No Data	No Data	No Data	LT E-24
Br-85	No Data	No Data	1.46E-08	No Data	No Data	No Data	LT E-24
Rb-86	No Data	1.36E-04	6.30E-05	No Data	No Data	No Data	2.17E-06
Rb-88	No Data	3.98E-07	2.05E-07	No Data	No Data	No Data	2.42E-07
Rb-89	No Data	2.29E-07	1.47E-07	No Data	No Data	No Data	4.87E-08
Sr-89	2.84E-04	No Data	8.15E-06	No Data	No Data	1.45E-03	4.57E-05
Sr-90	2.92E-02	No Data	1.85E-03	No Data	No Data	8.03E-03	9.36E-05
Sr-91	6.83E-08	No Data	2.47E-09	No Data	No Data	3.76E-05	5.24E-05
Sr-92	7.50E-09	No Data	2.79E-10	No Data	No Data	1.70E-05	1.00E-04
Y-90	2.35E-06	No Data	6.30E-08	No Data	No Data	1.92E-04	7.43E-05
Y-91M	2.91E-10	No Data	9.90E-12	No Data	No Data	1.99E-06	1.68E-06
Y-91	4.20E-04	No Data	1.12E-05	No Data	No Data	1.75E-03	5.02E-05
Y-92	1.17E-08	No Data	3.29E-10	No Data	No Data	1.75E-05	9.04E-05

\* Reference 3, Table E-10.

TABLE 2.2-1 (SHEET 2 OF 3)

INHALATION DOSE FACTORS FOR INFANT\*  
(mrem per pci inhaled)

Nuclide	Bone	Liver	T Body	Thyroid	Kidney	Lung	GI-LLI
Y-93	1.07E-07	No Data	2.91E-09	No Data	No Data	5.46E-05	1.19E-04
Zr-95	8.24E-05	1.99E-05	1.45E-05	No Data	2.22E-05	1.25E-03	1.55E-05
Zr-97	1.07E-07	1.83E-08	8.36E-09	No Data	1.85E-08	7.88E-05	1.00E-04
Nb-95	1.12E-05	4.59E-06	2.70E-06	No Data	3.37E-06	3.42E-04	9.05E-06
Mo-99	No Data	1.18E-07	2.31E-08	No Data	1.89E-07	9.63E-05	3.48E-05
Tc-99M	9.98E-13	2.06E-12	2.66E-11	No Data	2.22E-11	5.79E-07	1.45E-06
Tc-101	4.65E-14	5.88E-14	5.80E-13	No Data	6.99E-13	4.17E-07	6.03E-07
Ru-103	1.44E-06	No Data	4.85E-07	No Data	3.03E-06	3.94E-04	1.15E-05
Ru-105	8.74E-10	No Data	2.93E-10	No Data	6.42E-10	1.12E-05	3.46E-05
Ru-106	6.20E-05	No Data	7.77E-06	No Data	7.61E-05	8.26E-03	1.17E-04
Ag-110M	7.13E-06	5.16E-06	3.57E-06	No Data	7.80E-06	2.62E-03	2.36E-05
Te-125M	3.40E-06	1.42E-06	4.70E-07	1.16E-06	No Data	3.19E-04	9.22E-06
Te-127M	1.19E-05	4.93E-06	1.48E-06	3.48E-06	2.68E-05	9.37E-04	1.95E-05
Te-127	1.59E-09	6.81E-10	3.49E-10	1.32E-09	3.47E-09	7.39E-06	1.74E-05
Te-129M	1.01E-05	4.35E-06	1.59E-06	3.91E-06	2.27E-05	1.20E-03	4.93E-05
Te-129	5.63E-11	2.48E-11	1.34E-11	4.82E-11	1.25E-10	2.14E-06	1.88E-05
Te-131M	7.62E-08	3.93E-08	2.59E-08	6.38E-08	1.89E-07	1.42E-04	8.51E-05
Te-131	1.24E-11	5.87E-12	3.57E-12	1.13E-11	2.85E-11	1.47E-06	5.87E-06
Te-132	2.66E-07	1.69E-07	1.26E-07	1.99E-07	7.39E-07	2.43E-04	3.15E-05
I-130	4.54E-06	9.91E-06	3.98E-06	1.14E-03	1.09E-05	No Data	1.42E-06
I-131	2.71E-05	3.17E-05	1.40E-05	1.06E-02	3.70E-05	No Data	7.56E-07
I-132	1.21E-06	2.53E-06	8.99E-07	1.21E-04	2.82E-06	No Data	1.36E-06
I-133	9.46E-06	1.37E-05	4.00E-06	2.54E-03	1.60E-05	No Data	1.54E-06
I-134	6.58E-07	1.34E-06	4.75E-07	3.18E-05	1.49E-06	No Data	9.21E-07
I-135	2.76E-06	5.43E-06	1.98E-06	4.97E-04	6.05E-06	No Data	1.31E-06
Cs-134	2.83E-04	5.02E-04	5.32E-05	No Data	1.36E-04	5.69E-05	9.53E-07
Cs-136	3.45E-05	9.61E-05	3.78E-05	No Data	4.03E-05	8.40E-06	1.02E-06
Cs-137	3.92E-04	4.37E-04	3.25E-05	No Data	1.23E-04	5.09E-05	9.53E-07
Cs-138	3.61E-07	5.58E-07	2.84E-07	No Data	2.93E-07	4.67E-08	6.26E-07
Ba-139	1.06E-09	7.03E-13	3.07E-11	No Data	4.23E-13	4.25E-06	3.64E-05
Ba-140	4.00E-05	4.00E-08	2.07E-06	No Data	9.59E-09	1.14E-03	2.74E-05
Ba-141	1.12E-10	7.70E-14	3.55E-12	No Data	4.64E-14	2.12E-06	3.39E-06
Ba-142	2.84E-11	2.36E-14	1.40E-12	No Data	1.36E-14	1.11E-06	4.95E-07

\* Reference 3, Table E-10.

TABLE 2.2-1 (SHEET 3 OF 3)

INHALATION DOSE FACTORS FOR INFANT\*  
(mrem per pci inhaled)

<u>Nuclide</u>	<u>Bone</u>	<u>Liver</u>	<u>T Body</u>	<u>Thyroid</u>	<u>Kidney</u>	<u>Lung</u>	<u>GI-LLI</u>
La-140	3.61E-07	1.43E-07	3.68E-08	No Data	No Data	1.20E-04	6.06E-05
La-142	7.36E-10	2.69E-10	6.46E-11	No Data	No Data	5.87E-06	4.25E-05
Ce-141	1.98E-05	1.19E-05	1.42E-06	No Data	3.75E-06	3.69E-04	1.54E-05
Ce-143	2.09E-07	1.38E-07	1.58E-08	No Data	4.03E-08	8.30E-05	3.55E-05
Ce-144	2.28E-03	8.65E-04	1.26E-04	No Data	3.84E-04	7.03E-03	1.06E-04
Pr-143	1.00E-05	3.74E-06	4.99E-07	No Data	1.41E-06	3.09E-04	2.66E-05
Pr-144	3.42E-11	1.32E-11	1.72E-12	No Data	4.80E-12	1.15E-06	3.06E-06
Nd-147	5.67E-06	5.81E-06	3.57E-07	No Data	2.25E-06	2.30E-04	2.23E-05
W-187	9.26E-09	6.44E-09	2.23E-09	No Data	No Data	2.83E-05	2.54E-05
Np-239	2.65E-07	2.37E-08	1.34E-08	No Data	4.73E-08	4.25E-05	1.78E-05

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\* Reference 3, Table E-10.

TABLE 2.2-2 (SHEET 1 OF 3)  
 INHALATION DOSE FACTORS FOR CHILD\*  
 (mrem per pci inhaled)

Nuclide	Bone	Liver	T Body	Thyroid	Kidney	Lung	GI-LLI
H-3	No Data	3.04E-07	3.04E-07	3.04E-07	3.04E-07	3.04E-07	3.04E-07
C-14	9.70E-06	1.82E-06	1.82E-06	1.82E-06	1.82E-06	1.82E-06	1.82E-06
Na-24	4.35E-06	4.35E-06	4.35E-06	4.35E-06	4.35E-06	4.35E-06	4.35E-06
P-32	7.04E-04	3.09E-05	2.67E-05	No Data	No Data	No Data	1.14E-05
Cr-51	No Data	No Data	4.17E-08	2.31E-08	6.57E-09	4.59E-06	2.93E-07
Mn-54	No Data	1.16E-05	2.57E-06	No Data	2.71E-06	4.26E-04	6.19E-06
Mn-56	No Data	4.48E-10	8.43E-11	No Data	4.52E-10	3.55E-06	3.33E-05
Fe-55	1.28E-05	6.80E-06	2.10E-06	No Data	No Data	3.00E-05	7.75E-07
Fe-59	5.59E-06	9.04E-06	4.51E-06	No Data	No Data	3.43E-04	1.91E-05
Co-58	No Data	4.79E-07	8.55E-07	No Data	No Data	2.99E-04	9.29E-06
Co-60	No Data	3.55E-06	6.12E-06	No Data	No Data	1.91E-03	2.60E-05
Ni-63	2.22E-04	1.25E-05	7.56E-06	No Data	No Data	7.43E-05	1.71E-06
Ni-65	8.08E-10	7.99E-11	4.44E-11	No Data	No Data	2.21E-06	2.27E-05
Cu-64	No Data	5.39E-10	2.90E-10	No Data	1.63E-09	2.59E-06	9.92E-06
Zn-65	1.15E-05	3.06E-05	1.90E-05	No Data	1.93E-05	2.69E-04	4.41E-06
Zn-69	1.81E-11	2.61E-11	2.41E-12	No Data	1.58E-11	3.84E-07	2.75E-06
Br-83	No Data	No Data	1.28E-07	No Data	No Data	No Data	LT E-24
Br-84	No Data	No Data	1.48E-07	No Data	No Data	No Data	LT E-24
Br-85	No Data	No Data	6.84E-09	No Data	No Data	No Data	LT E-24
Rb-86	No Data	5.36E-05	3.09E-05	No Data	No Data	No Data	2.16E-06
Rb-88	No Data	1.52E-07	9.90E-08	No Data	No Data	No Data	4.66E-09
Rb-89	No Data	9.33E-08	7.83E-08	No Data	No Data	No Data	5.11E-10
Sr-89	1.62E-04	No Data	4.66E-06	No Data	No Data	5.83E-04	4.52E-05
Sr-90	2.73E-02	No Data	1.74E-03	No Data	No Data	3.99E-03	9.28E-05
Sr-91	3.28E-08	No Data	1.24E-09	No Data	No Data	1.44E-05	4.70E-05
Sr-92	3.54E-09	No Data	1.42E-10	No Data	No Data	6.49E-06	6.55E-05
Y-90	1.11E-06	No Data	2.99E-08	No Data	No Data	7.07E-1	7.24E-05
Y-91M	1.37E-10	No Data	4.98E-12	No Data	No Data	7.60E-07	4.64E-07
Y-91	2.47E-04	No Data	6.59E-06	No Data	No Data	7.10E-04	4.97E-05
Y-92	5.50E-09	No Data	1.57E-10	No Data	No Data	6.46E-06	6.46E-05

\* Reference 3, Table E-9.

TABLE 2.2-2 (SHEET 2 OF 3)

INHALATION DOSE FACTORS FOR CHILD\*  
(mrem per pci inhaled)

Nuclide	Bone	Liver	T Body	Thyroid	Kidney	Lung	GI-LLI
Y-93	5.04E-08	No Data	1.38E-09	No Data	No Data	2.01E-05	1.05E-04
Zr-95	5.13E-05	1.13E-05	1.00E-05	No Data	1.61E-05	6.03E-04	1.65E-05
Zr-97	5.07E-08	7.34E-09	4.32E-09	No Data	1.05E-08	3.06E-05	9.49E-05
Nb-95	6.35E-06	2.48E-06	1.77E-06	No Data	2.33E-06	1.66E-04	1.00E-05
Mo-99	No Data	4.66E-08	1.15E-08	No Data	1.06E-07	3.66E-05	3.42E-05
Tc-99M	4.81E-13	9.41E-13	1.56E-11	No Data	1.37E-11	2.57E-07	1.30E-06
Tc-101	2.19E-14	2.30E-14	2.91E-13	No Data	3.92E-13	1.58E-07	4.41E-09
Ru-103	7.55E-07	No Data	2.90E-07	No Data	1.90E-06	1.79E-04	1.21E-05
Ru-105	4.13E-10	No Data	1.50E-10	No Data	3.63E-10	4.30E-06	2.69E-05
Ru-106	3.68E-05	No Data	4.57E-06	No Data	4.97E-05	3.87E-03	1.16E-04
Ag-110M	4.56E-06	3.08E-06	2.47E-06	No Data	5.74E-06	1.48E-03	2.71E-05
Te-125M	1.82E-06	6.29E-07	2.47E-07	5.20E-07	No Data	1.29E-04	9.13E-06
Te-127M	6.72E-06	2.31E-06	8.16E-07	1.64E-06	1.72E-05	4.00E-04	1.93E-05
Te-127	7.49E-10	2.57E-10	1.65E-10	5.30E-10	1.91E-09	2.71E-06	1.52E-05
Te-129M	5.19E-06	1.85E-06	8.22E-07	1.71E-06	1.36E-05	4.76E-04	4.91E-05
Te-129	2.64E-11	9.45E-12	6.44E-12	1.93E-11	6.94E-11	7.93E-07	6.89E-06
Te-131M	3.63E-08	1.60E-08	1.37E-08	2.64E-08	1.08E-07	5.56E-05	8.32E-05
Te-131	5.87E-12	2.28E-12	1.78E-12	4.59E-12	1.59E-11	5.55E-07	3.60E-07
Te-132	1.30E-07	7.36E-08	7.12E-08	8.58E-08	4.79E-07	1.02E-04	3.72E-05
I-130	2.21E-06	4.43E-06	2.28E-06	4.99E-04	6.61E-06	No Data	1.38E-06
I-131	1.30E-05	1.30E-05	7.37E-06	4.39E-03	2.13E-05	No Data	7.68E-07
I-132	5.72E-07	1.10E-06	5.07E-07	5.23E-05	1.69E-06	No Data	8.65E-07
I-133	4.48E-06	5.49E-06	2.08E-06	1.04E-03	9.13E-06	No Data	1.48E-06
I-134	3.17E-07	5.84E-07	2.69E-07	1.37E-05	8.92E-07	No Data	2.58E-07
I-135	1.33E-06	2.36E-06	1.12E-06	2.14E-04	3.62E-06	No Data	1.20E-06
Cs-134	1.76E-04	2.74E-04	6.07E-05	No Data	8.93E-05	3.27E-05	1.04E-06
Cs-136	1.76E-05	4.62E-05	3.14E-05	No Data	2.58E-05	3.93E-06	1.13E-06
Cs-137	2.45E-04	2.23E-04	3.47E-05	No Data	7.63E-05	2.81E-05	9.78E-07
Cs-138	1.71E-07	2.27E-07	1.50E-07	No Data	1.68E-07	1.84E-08	7.29E-08
Ba-139	4.98E-10	2.66E-13	1.45E-11	No Data	2.33E-13	1.56E-06	1.56E-05
Ba-140	2.00E-05	1.75E-08	1.17E-06	No Data	5.71E-09	4.71E-04	2.75E-05
Ba-141	5.29E-11	2.95E-14	1.72E-12	No Data	2.56E-14	7.89E-07	7.44E-08
Ba-142	1.35E-11	9.73E-15	7.54E-13	No Data	7.87E-15	4.44E-07	7.41E-10

\* Reference 3, Table E-9.

TABLE 2.2-2 (SHEET 3 OF 3)  
 INHALATION DOSE FACTORS FOR CHILD\*  
 (mrem per pci inhaled)

<u>Nuclide</u>	<u>Bone</u>	<u>Liver</u>	<u>T Body</u>	<u>Thyroid</u>	<u>Kidney</u>	<u>Lung</u>	<u>GI-LLI</u>
La-140	1.74E-07	6.08E-08	2.04E-08	No Data	No Data	4.94E-05	6.10E-05
La-142	3.50E-10	1.11E-10	3.49E-11	No Data	No Data	2.35E-06	2.05E-05
Ce-141	1.06E-05	5.28E-06	7.83E-07	No Data	2.31E-06	1.47E-04	1.53E-05
Ce-143	9.89E-08	5.37E-08	7.77E-09	No Data	2.26E-08	3.12E-05	3.44E-05
Ce-144	1.83E-03	5.72E-04	9.77E-05	No Data	3.17E-04	3.23E-03	1.05E-04
Pr-143	4.99E-06	1.50E-06	2.47E-07	No Data	8.11E-07	1.17E-04	2.63E-05
Pr-144	1.61E-11	4.99E-12	8.10E-13	No Data	2.64E-12	4.23E-07	5.32E-08
Nd-147	2.92E-06	2.36E-06	1.84E-07	No Data	1.30E-06	8.87E-05	2.22E-05
W-187	4.41E-09	2.61E-09	1.17E-09	No Data	No Data	1.11E-05	2.46E-05
Np-239	1.26E-07	9.04E-09	6.35E-09	No Data	2.63E-08	1.57E-05	1.73E-05

\* Reference 3, Table E-9.



TABLE 2.2-3 (SHEET 1 OF 3)

INHALATION DOSE FACTORS FOR TEENAGER\*  
(mrem per pci inhaled)

Nuclide	Bone	Liver	T Body	Thyroid	Kidney	Lung	GI-LLI
H-3	No Data	1.59E-07	1.59E-07	1.59E-07	1.59E-07	1.59E-07	1.59E-07
C-14	3.25E-06	6.09E-06	6.09E-07	6.09E-07	6.09E-07	6.09E-07	6.09E-07
Na-24	1.72E-06	1.72E-06	1.72E-06	1.72E-06	1.72E-06	1.72E-06	1.72E-06
P-32	2.36E-04	1.37E-05	8.95E-06	No Data	No Data	No Data	1.16E-05
Cr-51	No Data	No Data	1.69E-08	9.37E-09	3.84E-09	2.62E-06	3.75E-07
Mn-54	No Data	6.39E-06	1.05E-06	No Data	1.59E-06	2.48E-04	8.35E-06
Mn-56	No Data	2.12E-10	3.15E-11	No Data	2.24E-10	1.90E-06	7.18E-06
Fe-55	4.18E-06	2.98E-06	6.93E-07	No Data	No Data	1.55E-05	7.99E-07
Fe-59	1.99E-06	4.62E-06	1.79E-06	No Data	No Data	1.91E-04	2.23E-05
Co-58	No Data	2.59E-07	3.47E-07	No Data	No Data	1.68E-04	1.19E-05
Co-60	No Data	1.89E-06	2.48E-06	No Data	No Data	1.09E-03	3.24E-05
Ni-63	7.25E-05	5.43E-06	2.47E-06	No Data	No Data	3.84E-05	1.77E-06
Ni-65	2.73E-10	3.66E-11	1.59E-11	No Data	No Data	1.17E-06	4.59E-06
Cu-64	No Data	2.54E-10	1.06E-10	No Data	8.01E-10	1.39E-06	7.68E-06
Zn-65	4.82E-06	1.67E-05	7.80E-06	No Data	1.08E-05	1.55E-04	5.83E-06
Zn-69	6.04E-12	1.15E-11	8.07E-13	No Data	7.53E-12	1.98E-07	3.56E-08
Br-83	No Data	No Data	4.30E-08	No Data	No Data	No Data	LT E-24
Br-84	No Data	No Data	5.41E-08	No Data	No Data	No Data	LT E-24
Br-85	No Data	No Data	2.29E-09	No Data	No Data	No Data	LT E-24
Rb-86	No Data	2.38E-05	1.05E-05	No Data	No Data	No Data	2.21E-06
Rb-88	No Data	6.82E-08	3.40E-08	No Data	No Data	No Data	3.65E-15
Rb-89	No Data	4.40E-08	2.91E-08	No Data	No Data	No Data	4.22E-17
Sr-89	5.43E-05	No Data	1.56E-06	No Data	No Data	3.02E-04	4.64E-05
Sr-90	1.35E-02	No Data	8.35E-04	No Data	No Data	2.06E-03	9.56E-05
Sr-91	1.10E-08	No Data	4.39E-10	No Data	No Data	7.59E-06	3.24E-05
Sr-92	1.19E-09	No Data	5.08E-11	No Data	No Data	3.43E-06	1.49E-05
Y-90	3.73E-07	No Data	1.00E-08	No Data	No Data	3.66E-05	6.99E-05
Y-91M	4.63E-11	No Data	1.77E-12	No Data	No Data	4.00E-07	3.77E-09
Y-91	8.26E-05	No Data	2.21E-06	No Data	No Data	3.67E-04	5.11E-05
Y-92	1.84E-09	No Data	5.36E-11	No Data	No Data	3.35E-06	2.06E-05

\* Reference 3, Table E-8.



TABLE 2.2-3 (SHEET 2 OF 3)

INHALATION DOSE FACTORS FOR TEENAGER\*  
(mrem per pci inhaled)

Nuclide	Bone	Liver	T Body	Thyroid	Kidney	Lung	GI-LLI
Y-93	1.69E-08	No Data	4.65E-10	No Data	No Data	1.04E-05	7.24E-05
Zr-95	1.82E-05	5.73E-06	3.94E-06	No Data	8.42E-06	3.36E-04	1.86E-05
Zr-97	1.72E-08	3.40E-09	1.57E-09	No Data	5.15E-09	1.62E-05	7.88E-05
Nb-95	2.32E-06	1.29E-06	7.08E-07	No Data	1.25E-06	9.39E-05	1.21E-05
Mo-99	No Data	2.11E-08	4.03E-09	No Data	5.14E-08	1.92E-05	3.36E-05
Tc-99M	1.73E-13	4.83E-13	6.24E-12	No Data	7.20E-12	1.44E-07	7.66E-07
Tc-101	7.40E-15	1.05E-14	1.03E-13	No Data	1.90E-13	8.34E-08	1.09E-16
Ru-103	2.63E-07	No Data	1.12E-07	No Data	9.29E-07	9.79E-05	1.36E-05
Ru-105	1.40E-10	No Data	5.42E-11	No Data	1.76E-10	2.27E-06	1.13E-05
Ru-106	1.23E-05	No Data	1.55E-06	No Data	2.38E-05	2.01E-03	1.20E-04
Ag-110M	1.73E-06	1.64E-06	9.99E-07	No Data	3.13E-06	8.44E-04	3.41E-05
Te-125M	6.10E-07	2.80E-07	8.34E-08	1.75E-07	No Data	6.70E-05	9.38E-06
Te-127M	2.25E-06	1.02E-06	2.73E-07	5.48E-07	8.17E-06	2.07E-04	1.99E-05
Te-127	2.51E-10	1.14E-10	5.52E-11	1.77E-10	9.10E-10	1.40E-06	1.01E-05
Te-129M	1.74E-06	8.23E-07	2.81E-07	5.72E-07	6.49E-06	2.47E-04	5.06E-05
Te-129	8.87E-12	4.22E-12	2.20E-12	6.48E-12	3.32E-11	4.12E-07	2.02E-07
Te-131M	1.23E-08	7.51E-09	5.03E-09	9.06E-09	5.49E-08	2.97E-05	7.76E-05
Te-131	1.97E-12	1.04E-12	6.30E-13	1.55E-12	7.72E-12	2.92E-07	1.89E-09
Te-132	4.50E-08	3.63E-08	2.74E-08	3.07E-08	2.44E-07	5.61E-05	5.79E-05
I-130	7.80E-07	2.24E-06	8.96E-07	1.86E-04	3.44E-06	No Data	1.14E-06
I-131	4.43E-06	6.14E-06	3.30E-06	1.83E-03	1.05E-05	No Data	8.11E-07
I-132	1.99E-07	5.47E-07	1.97E-07	1.89E-05	8.65E-07	No Data	1.59E-07
I-133	1.52E-06	2.56E-06	7.78E-07	3.65E-04	4.49E-06	No Data	1.29E-06
I-134	1.11E-07	2.90E-07	1.05E-07	4.94E-06	4.58E-07	No Data	2.55E-09
I-135	4.62E-07	1.18E-06	4.36E-07	7.76E-05	1.86E-06	No Data	8.69E-07
Cs-134	6.28E-05	1.41E-04	6.86E-05	No Data	4.69E-05	1.83E-05	1.22E-06
Cs-136	6.44E-06	2.42E-05	1.71E-05	No Data	1.38E-05	2.22E-06	1.36E-06
Cs-137	8.38E-05	1.06E-04	3.89E-05	No Data	3.80E-05	1.51E-05	1.06E-06
Cs-138	5.82E-08	1.07E-07	5.58E-08	No Data	8.28E-08	9.84E-09	3.38E-11
Ba-139	1.67E-10	1.18E-13	4.87E-12	No Data	1.11E-13	8.08E-07	8.06E-07
Ba-140	6.84E-06	8.38E-09	4.40E-07	No Data	2.85E-09	2.54E-04	2.86E-05
Ba-141	1.78E-11	1.32E-14	5.93E-13	No Data	1.23E-14	4.11E-07	9.33E-14
Ba-142	4.62E-12	4.63E-15	2.84E-13	No Data	3.92E-15	2.39E-07	5.99E-20

\* Reference 3, Table E-8.

TABLE 2.2-3 (SHEET 3 OF 3)

INHALATION DOSE FACTORS FOR TEENAGER\*  
(mrem per pci inhaled)

<u>Nuclide</u>	<u>Bone</u>	<u>Liver</u>	<u>T Body</u>	<u>Thyroid</u>	<u>Kidney</u>	<u>Lung</u>	<u>GI-LLI</u>
La-140	5.99E-08	2.95E-08	7.82E-09	No Data	No Data	2.68E-05	6.09E-05
La-142	1.20E-10	5.31E-11	1.32E-11	No Data	No Data	1.27E-06	1.50E-06
Ce-141	3.55E-06	2.37E-06	2.71E-07	No Data	1.11E-06	7.67E-05	1.58E-05
Ce-143	3.32E-08	2.42E-08	2.70E-09	No Data	1.08E-08	1.63E-05	3.19E-05
Ce-144	6.11E-04	2.53E-04	3.28E-05	No Data	1.51E-04	1.67E-03	1.08E-04
Pr-143	1.67E-06	6.64E-07	8.28E-08	No Data	3.86E-07	6.04E-05	2.67E-05
Pr-144	5.37E-12	2.20E-12	2.72E-13	No Data	1.26E-12	2.19E-07	2.94E-14
Nd-147	9.83E-07	1.07E-06	6.41E-08	No Data	6.28E-07	4.65E-05	2.28E-05
W-187	1.50E-09	1.22E-09	4.29E-10	No Data	No Data	5.92E-06	2.21E-05
Np-239	4.23E-08	3.99E-09	2.21E-09	No Data	1.25E-08	8.11E-06	1.65E-05

\* Reference 3, Table E-8.

TABLE 2.2-4 (SHEET 1 OF 3)  
 INHALATION DOSE FACTORS FOR ADULTS\*  
 (mrem per pci inhaled)

Nuclide	Bone	Liver	T Body	Thyroid	Kidney	Lung	GI-LLI
H-3	No Data	1.58E-07	1.58E-07	1.58E-07	1.58E-07	1.58E-07	1.58E-07
C-14	2.27E-06	4.26E-07	4.26E-07	4.26E-07	4.26E-07	4.26E-07	4.26E-07
Na-24	1.28E-06	1.28E-06	1.28E-06	1.28E-06	1.28E-06	1.28E-06	1.28E-06
P-32	1.65E-04	9.64E-06	6.26E-06	No Data	No Data	No Data	1.08E-05
Cr-51	No Data	No Data	1.25E-08	7.44E-09	2.85E-09	1.80E-06	4.15E-07
Mn-54	No Data	4.95E-06	7.87E-07	No Data	1.23E-06	1.75E-04	9.67E-06
Mn-56	No Data	1.55E-10	2.29E-11	No Data	1.63E-10	1.18E-06	2.53E-06
Fe-55	3.07E-06	2.12E-06	4.93E-07	No Data	No Data	9.01E-06	7.54E-07
Fe-59	1.47E-06	3.47E-06	1.32E-06	No Data	No Data	1.27E-04	2.35E-05
Co-58	No Data	1.98E-07	2.59E-07	No Data	No Data	1.16E-04	1.33E-05
Co-60	No Data	1.44E-06	1.85E-06	No Data	No Data	7.46E-04	3.56E-05
Ni-63	5.40E-05	3.93E-06	1.81E-06	No Data	No Data	2.23E-05	1.67E-06
Ni-65	1.92E-10	2.62E-11	1.14E-11	No Data	No Data	7.00E-07	1.54E-06
Cu-64	No Data	1.83E-10	7.69E-11	No Data	5.78E-10	8.48E-07	6.12E-06
Zn-65	4.05E-06	1.29E-05	5.82E-06	No Data	8.62E-06	1.08E-04	6.68E-06
Zn-69	4.23E-12	8.14E-12	5.65E-13	No Data	5.27E-12	1.15E-07	2.04E-09
Br-83	No Data	No Data	3.01E-08	No Data	No Data	No Data	2.90E-08
Br-84	No Data	No Data	3.91E-08	No Data	No Data	No Data	2.05E-13
Br-85	No Data	No Data	1.60E-09	No Data	No Data	No Data	LT E-24
Rb-86	No Data	1.69E-05	7.37E-06	No Data	No Data	No Data	2.08E-06
Rb-88	No Data	4.84E-08	2.41E-08	No Data	No Data	No Data	4.18E-19
Rb-89	No Data	3.20E-08	2.12E-08	No Data	No Data	No Data	1.16E-21
Sr-89	3.80E-05	No Data	1.09E-06	No Data	No Data	1.75E-04	4.37E-05
Sr-90	1.24E-02	No Data	7.62E-04	No Data	No Data	1.20E-03	9.02E-05
Sr-91	7.74E-09	No Data	3.13E-10	No Data	No Data	4.56E-06	2.39E-05
Sr-92	8.43E-10	No Data	3.64E-11	No Data	No Data	2.06E-06	5.38E-06
Y-90	2.61E-07	No Data	7.01E-09	No Data	No Data	2.12E-05	6.32E-05
Y-91M	3.26E-11	No Data	1.27E-12	No Data	No Data	2.40E-07	1.66E-10
Y-91	5.78E-05	No Data	1.55E-06	No Data	No Data	2.13E-04	4.81E-05
Y-92	1.29E-09	No Data	3.77E-11	No Data	No Data	1.96E-06	9.19E-06

\* Reference 3, Table E-7.

TABLE 2.2-4 (SHEET 2 OF 3)

INHALATION DOSE FACTORS FOR ADULTS\*  
(mrem per pci inhaled)

Nuclide	Bone	Liver	T Body	Thyroid	Kidney	Lung	GI-LLI
Y-93	1.18E-08	No Data	3.26E-10	No Data	No Data	6.06E-06	5.27E-05
Zr-95	1.34E-05	4.30E-06	2.91E-06	No Data	6.77E-06	2.21E-04	1.88E-05
Zr-97	1.21E-08	2.45E-09	1.13E-09	No Data	3.71E-09	9.84E-06	6.54E-05
Nb-95	1.76E-06	9.77E-07	5.26E-07	No Data	9.67E-07	6.31E-05	1.30E-05
Mo-99	No Data	1.51E-08	2.87E-09	No Data	3.64E-08	1.14E-05	3.10E-05
Tc-99M	1.29E-13	3.64E-13	4.63E-12	No Data	5.52E-12	9.55E-08	5.20E-07
Tc-101	5.22E-15	7.52E-15	7.38E-14	No Data	1.35E-13	4.99E-08	1.36E-21
Ru-103	1.91E-07	No Data	8.23E-08	No Data	7.29E-07	6.31E-05	1.38E-05
Ru-105	9.88E-11	No Data	3.89E-11	No Data	1.27E-10	1.37E-06	6.02E-06
Ru-106	8.64E-06	No Data	1.09E-06	No Data	1.67E-05	1.17E-03	1.14E-04
Ag-110M	1.35E-06	1.25E-06	7.43E-07	No Data	2.46E-06	5.79E-04	3.78E-05
Te-125M	4.27E-07	1.98E-07	5.84E-08	1.31E-07	1.55E-06	3.92E-05	8.83E-06
Te-127M	1.58E-06	7.21E-07	1.96E-07	4.11E-07	5.72E-06	1.20E-04	1.87E-05
Te-127	1.75E-10	8.03E-11	3.87E-11	1.32E-10	6.37E-10	8.14E-07	7.17E-06
Te-129M	1.22E-06	5.84E-07	1.98E-07	4.30E-07	4.57E-06	1.45E-04	4.79E-05
Te-129	6.22E-12	2.99E-12	1.55E-12	4.87E-12	2.34E-11	2.42E-07	1.96E-08
Te-131M	8.74E-09	5.45E-09	3.63E-09	6.88E-09	3.86E-08	1.82E-05	6.95E-05
Te-131	1.39E-12	7.44E-13	4.49E-13	1.17E-12	5.46E-12	1.74E-07	2.30E-09
Te-132	3.25E-08	2.69E-08	2.02E-08	2.37E-08	1.82E-07	3.60E-05	6.37E-05
I-130	5.72E-07	1.68E-06	6.60E-07	1.42E-04	2.61E-06	No Data	9.61E-07
I-131	3.15E-06	4.47E-06	2.56E-06	1.49E-03	7.66E-06	No Data	7.85E-07
I-132	1.45E-07	4.07E-07	1.45E-07	1.43E-05	6.48E-07	No Data	5.08E-08
I-133	1.08E-06	1.85E-06	5.65E-07	2.69E-04	3.23E-06	No Data	1.11E-06
I-134	8.05E-08	2.16E-07	7.69E-08	3.73E-06	3.44E-07	No Data	1.26E-10
I-135	3.35E-07	8.73E-07	3.21E-07	5.60E-05	1.39E-06	No Data	6.56E-07
Cs-134	4.66E-05	1.06E-04	9.10E-05	No Data	3.59E-05	1.22E-05	1.30E-06
Cs-136	4.88E-06	1.83E-05	1.38E-05	No Data	1.07E-05	1.50E-06	1.46E-06
Cs-137	5.98E-05	7.76E-05	5.35E-05	No Data	2.78E-05	9.40E-06	1.05E-06
Cs-138	4.14E-08	7.76E-08	4.05E-08	No Data	6.00E-08	6.07E-09	2.33E-13
Ba-139	1.17E-10	8.32E-14	3.42E-12	No Data	7.78E-14	4.70E-07	1.12E-07
Ba-140	4.88E-06	6.13E-09	3.21E-07	No Data	2.09E-09	1.59E-04	2.73E-05
Ba-141	1.25E-11	9.41E-15	4.20E-13	No Data	8.75E-15	2.42E-07	1.45E-17
Ba-142	3.29E-12	3.38E-15	2.07E-13	No Data	2.86E-15	1.49E-07	1.96E-26

\* Reference 3, Table E-7.

TABLE 2.2-4 (SHEET 3 OF 3)

INHALATION DOSE FACTORS FOR ADULTS\*  
(mrem per pci inhaled)

<u>Nuclide</u>	<u>Bone</u>	<u>Liver</u>	<u>T Body</u>	<u>Thyroid</u>	<u>Kidney</u>	<u>Lung</u>	<u>GI-LLI</u>
La-140	4.30E-08	2.17E-08	5.73E-09	No Data	No Data	1.70E-05	5.73E-05
La-142	8.54E-11	3.88E-11	9.65E-12	No Data	No Data	7.91E-07	2.64E-07
Ce-141	2.49E-06	1.69E-06	1.91E-07	No Data	7.83E-07	4.52E-05	1.50E-05
Ce-143	2.33E-08	1.72E-08	1.91E-09	No Data	7.60E-09	9.97E-06	2.83E-05
Ce-144	4.29E-04	1.79E-04	2.30E-05	No Data	1.06E-04	9.72E-04	1.02E-04
Pr-143	1.17E-06	4.69E-07	5.80E-08	No Data	2.70E-07	3.51E-05	2.50E-05
Pr-144	3.76E-12	1.56E-12	1.91E-13	No Data	8.81E-13	1.27E-07	2.69E-18
Nd-147	6.59E-07	7.62E-07	4.56E-08	No Data	4.45E-07	2.76E-05	2.16E-05
W-187	1.06E-09	8.85E-10	3.10E-10	No Data	No Data	3.63E-06	1.94E-05
Np-239	2.87E-08	2.82E-09	1.55E-09	No Data	8.75E-09	4.70E-06	1.49E-05

\* Reference 3, Table E-7.

TABLE 2.2-5 (SHEET 1 OF 3)  
 INGESTION DOSE FACTORS FOR INFANT\*  
 (mrem per pci ingested)

Nuclide	Bone	Liver	T Body	Thyroid	Kidney	Lung	GI-LLI
H-3	No Data	3.08E-07	3.08E-07	3.08E-07	3.08E-07	3.08E-07	3.08E-07
C-14	2.37E-05	5.06E-06	5.06E-06	5.06E-06	5.06E-06	5.06E-06	5.06E-06
Na-24	1.01E-05	1.01E-05	1.01E-05	1.01E-05	1.01E-05	1.01E-05	1.01E-05
P-32	1.70E-03	1.00E-04	6.59E-05	No Data	No Data	No Data	2.30E-05
Cr-51	No Data	No Data	1.41E-08	9.20E-09	2.01E-09	1.79E-08	4.11E-07
Mn-54	No Data	1.99E-05	4.51E-05	No Data	4.41E-06	No Data	7.31E-06
Mn-56	No Data	8.18E-07	1.41E-07	No Data	7.03E-07	No Data	7.43E-05
Fe-55	1.39E-05	8.98E-06	2.40E-06	No Data	No Data	4.39E-06	1.14E-06
Fe-59	3.08E-05	5.38E-05	2.12E-05	No Data	No Data	1.59E-05	2.57E-05
Co-58	No Data	3.60E-06	8.98E-06	No Data	No Data	No Data	8.97E-06
Co-60	No Data	1.08E-05	2.55E-05	No Data	No Data	No Data	2.57E-05
Ni-63	6.34E-04	3.92E-05	2.20E-05	No Data	No Data	No Data	1.95E-06
Ni-65	4.70E-06	5.32E-07	2.42E-07	No Data	No Data	No Data	4.05E-05
Cu-64	No Data	6.09E-07	2.82E-07	No Data	1.03E-06	No Data	1.25E-05
Zn-65	1.84E-05	6.31E-05	2.91E-05	No Data	3.06E-05	No Data	5.33E-05
Zn-69	9.33E-08	1.68E-07	1.25E-08	No Data	6.98E-08	No Data	1.37E-05
Br-83	No Data	No Data	3.63E-07	No Data	No Data	No Data	LT E-24
Br-84	No Data	No Data	3.82E-07	No Data	No Data	No Data	LT E-24
Br-85	No Data	No Data	1.94E-08	No Data	No Data	No Data	LT E-24
Rb-86	No Data	1.70E-04	8.40E-05	No Data	No Data	No Data	4.35E-06
Rb-88	No Data	4.98E-07	2.73E-07	No Data	No Data	No Data	4.85E-07
Rb-89	No Data	2.86E-07	1.97E-07	No Data	No Data	No Data	9.74E-08
Sr-89	2.51E-03	No Data	7.20E-05	No Data	No Data	No Data	5.16E-05
Sr-90	1.85E-02	No Data	4.71E-03	No Data	No Data	No Data	2.31E-04
Sr-91	5.00E-05	No Data	1.81E-06	No Data	No Data	No Data	5.92E-05
Sr-92	1.92E-05	No Data	7.13E-07	No Data	No Data	No Data	2.07E-04
Y-90	8.69E-08	No Data	2.33E-09	No Data	No Data	No Data	1.20E-04
Y-91M	8.10E-10	No Data	2.76E-11	No Data	No Data	No Data	2.70E-06
Y-91	1.13E-06	No Data	3.01E-08	No Data	No Data	No Data	8.10E-05
Y-92	7.65E-09	No Data	2.15E-10	No Data	No Data	No Data	1.46E-04

\* Reference 3, Table E-14.



TABLE 2.2-5 (SHEET 2 OF 3)

INGESTION DOSE FACTORS FOR INFANT\*  
(mrem per pci ingested)

Nuclide	Bone	Liver	T Body	Thyroid	Kidney	Lung	GI-LLI
Y-93	2.43E-08	No Data	6.62E-10	No Data	No Data	No Data	1.92E-04
Zr-95	2.06E-07	5.02E-08	3.56E-08	No Data	5.41E-08	No Data	2.50E-05
Zr-97	1.48E-08	2.54E-09	1.16E-09	No Data	2.56E-09	No Data	1.62E-04
Nb-95	4.20E-08	1.73E-08	1.00E-08	No Data	1.24E-08	No Data	1.46E-05
Mo-99	No Data	3.40E-05	6.63E-06	No Data	5.08E-05	No Data	1.12E-05
Tc-99M	1.92E-09	3.96E-09	5.10E-08	No Data	4.26E-08	2.07E-09	1.15E-06
Tc-101	2.27E-09	2.86E-09	2.83E-08	No Data	3.40E-08	1.56E-09	4.86E-07
Ru-103	1.48E-06	No Data	4.95E-07	No Data	3.08E-06	No Data	1.80E-05
Ru-105	1.36E-07	No Data	4.58E-08	No Data	1.00E-06	No Data	5.41E-05
Ru-106	2.41E-05	No Data	3.01E-06	No Data	2.85E-05	No Data	1.83E-04
Ag-110M	9.96E-07	7.27E-07	4.81E-07	No Data	1.04E-06	No Data	3.77E-05
Te-125M	2.33E-05	7.79E-06	3.15E-06	7.84E-06	No Data	No Data	1.11E-05
Te-127M	5.85E-05	1.94E-05	7.08E-06	1.69E-05	1.44E-04	No Data	2.36E-05
Te-127	1.00E-06	3.35E-07	2.15E-07	8.14E-07	2.44E-06	No Data	2.10E-05
Te-129M	1.00E-04	3.43E-05	1.54E-05	3.84E-05	2.50E-04	No Data	5.97E-05
Te-129	2.84E-07	9.79E-08	6.63E-08	2.38E-07	7.07E-07	No Data	2.27E-05
Te-131M	1.52E-05	6.12E-06	5.05E-06	1.24E-05	4.21E-05	No Data	1.03E-04
Te-131	1.76E-07	6.50E-08	4.94E-08	1.57E-07	4.50E-07	No Data	7.11E-06
Te-132	2.08E-05	1.03E-05	9.61E-06	1.52E-05	6.44E-05	No Data	3.81E-05
I-130	6.00E-06	1.32E-05	5.30E-06	1.48E-03	1.45E-05	No Data	2.83E-06
I-131	3.59E-05	4.23E-05	1.86E-05	1.39E-02	4.94E-05	No Data	1.51E-06
I-132	1.66E-06	3.37E-06	1.20E-06	1.58E-04	3.76E-06	No Data	2.73E-06
I-133	1.25E-05	1.82E-05	5.33E-06	3.31E-03	2.14E-05	No Data	3.08E-06
I-134	8.69E-07	1.78E-06	6.33E-07	4.15E-05	1.99E-06	No Data	1.84E-06
I-135	3.64E-06	7.24E-06	2.64E-06	6.49E-04	8.07E-06	No Data	2.62E-06
Cs-134	3.77E-04	7.03E-04	7.10E-05	No Data	1.81E-04	7.42E-05	1.91E-06
Cs-136	4.59E-05	1.35E-04	5.04E-05	No Data	5.38E-05	1.10E-05	2.05E-06
Cs-137	5.22E-04	6.11E-04	4.33E-05	No Data	1.64E-04	6.64E-05	1.91E-06
Cs-138	4.81E-07	7.82E-07	3.79E-07	No Data	3.90E-07	6.09E-08	1.25E-06
Ba-139	8.81E-07	5.84E-10	2.55E-08	No Data	3.51E-10	3.54E-10	5.58E-05

\* Reference 3, Table E-14.

TABLE 2.2-5 (SHEET 3 OF 3)  
 INGESTION DOSE FACTORS FOR INFANT\*  
 (mrem per pci ingested)

<u>Nuclide</u>	<u>Bone</u>	<u>Liver</u>	<u>T Body</u>	<u>Thyroid</u>	<u>Kidney</u>	<u>Lung</u>	<u>GI-LLI</u>
Ba-140	1.71E-04	1.71E-07	8.81E-06	No Data	4.06E-08	1.05E-07	4.20E-05
Ba-141	4.25E-07	2.91E-10	1.34E-08	No Data	1.75E-10	1.77E-10	5.19E-06
Ba-142	1.84E-07	1.53E-10	9.06E-09	No Data	8.81E-11	9.26E-11	7.59E-07
La-140	2.11E-08	8.32E-09	2.14E-09	No Data	No Data	No Data	9.77E-05
La-142	1.10E-09	4.04E-10	9.67E-11	No Data	No Data	No Data	6.86E-05
Ce-141	7.87E-08	4.80E-08	5.65E-09	No Data	1.48E-08	No Data	2.48E-05
Ce-143	1.48E-08	9.82E-06	1.12E-09	No Data	2.86E-09	No Data	5.73E-05
Ce-144	2.98E-06	1.22E-06	1.67E-07	No Data	4.93E-07	No Data	1.71E-04
Pr-143	8.13E-08	3.04E-08	4.03E-09	No Data	1.13E-08	No Data	4.29E-05
Pr-144	2.74E-10	1.06E-10	1.38E-11	No Data	5.84E-11	No Data	4.93E-06
Nd-147	5.53E-08	5.68E-08	3.48E-09	No Data	2.19E-08	No Data	3.60E-05
W-187	9.03E-07	6.28E-07	2.17E-07	No Data	No Data	No Data	3.69E-05
Np-239	1.11E-08	9.93E-10	5.61E-10	No Data	1.98E-09	No Data	2.87E-05

\* Reference 3, Table E-14.



TABLE 2.2-6 (SHEET 2 OF 3)

INGESTION DOSE FACTORS FOR CHILD\*  
(mrem per pci ingested)

Nuclide	Bone	Liver	T Body	Thyroid	Kidney	Lung	GI-LLI
Y-93	1.14E-08	No Data	3.13E-10	No Data	No Data	No Data	1.70E-04
Zr-95	1.16E-07	2.55E-08	2.27E-08	No Data	3.65E-08	No Data	2.66E-05
Zr-97	6.99E-09	1.01E-09	5.96E-10	No Data	1.45E-09	No Data	1.53E-04
Nb-95	2.25E-08	8.76E-09	6.26E-09	No Data	8.23E-09	No Data	1.62E-05
Mo-99	No Data	1.33E-05	3.29E-06	No Data	2.84E-05	No Data	1.10E-05
Tc-99M	9.23E-10	1.81E-09	3.00E-08	No Data	2.63E-08	9.19E-10	1.03E-06
Tc-101	1.07E-09	1.12E-09	1.42E-08	No Data	1.91E-08	5.92E-10	3.56E-09
Ru-103	7.31E-07	No Data	2.81E-07	No Data	1.84E-06	No Data	1.89E-05
Ru-105	6.45E-08	No Data	2.34E-08	No Data	5.67E-07	No Data	4.21E-05
Ru-106	1.17E-05	No Data	1.46E-06	No Data	1.58E-05	No Data	1.82E-04
Ag-110M	5.39E-07	3.64E-07	2.91E-07	No Data	6.78E-07	No Data	4.33E-05
Te-125M	1.14E-05	3.09E-06	1.52E-06	3.20E-06	No Data	No Data	1.10E-05
Te-127M	2.89E-05	7.78E-06	3.43E-06	6.91E-06	8.24E-05	No Data	2.34E-05
Te-127	4.71E-07	1.27E-07	1.01E-07	3.26E-07	1.34E-06	No Data	1.84E-05
Te-129M	4.87E-05	1.36E-05	7.56E-06	1.57E-05	1.43E-04	No Data	5.94E-05
Te-129	1.34E-07	3.74E-08	3.18E-08	9.56E-08	3.92E-07	No Data	8.34E-06
Te-131M	7.20E-06	2.49E-06	2.65E-06	5.12E-06	2.41E-05	No Data	1.01E-04
Te-131	8.30E-08	2.53E-08	2.47E-08	6.35E-08	2.51E-07	No Data	4.36E-07
Te-132	1.01E-05	4.47E-06	5.40E-06	6.51E-06	4.15E-05	No Data	4.50E-05
I-130	2.92E-06	5.90E-06	3.04E-06	6.50E-04	8.82E-06	No Data	2.76E-06
I-131	1.72E-05	1.73E-05	9.83E-06	5.72E-03	2.84E-05	No Data	1.54E-06
I-132	8.00E-07	1.47E-06	6.76E-07	6.82E-05	2.25E-06	No Data	1.73E-06
I-133	5.92E-06	7.32E-06	2.77E-06	1.36E-03	1.22E-05	No Data	2.95E-06
I-134	4.19E-07	7.78E-07	3.58E-07	1.79E-05	1.19E-06	No Data	5.16E-07
I-135	1.75E-06	3.15E-06	1.49E-06	2.79E-04	4.83E-06	No Data	2.40E-06
Cs-134	2.34E-04	3.84E-04	8.10E-05	No Data	1.19E-04	4.27E-05	2.07E-06
Cs-136	2.35E-05	6.46E-05	4.18E-05	No Data	3.44E-05	5.13E-06	2.27E-06
Cs-137	3.27E-04	3.13E-04	4.62E-05	No Data	1.02E-04	3.67E-05	1.96E-06
Cs-138	2.28E-07	3.17E-07	2.01E-07	No Data	2.23E-07	2.40E-08	1.46E-07
Ba-139	4.14E-07	2.21E-10	1.20E-08	No Data	1.93E-10	1.30E-10	2.39E-05
Ba-140	8.31E-05	7.28E-08	4.85E-06	No Data	2.37E-08	4.34E-08	4.21E-05
Ba-141	2.00E-07	1.12E-10	6.51E-09	No Data	9.69E-11	6.58E-10	1.14E-07
Ba-142	8.74E-08	6.29E-11	4.88E-09	No Data	5.09E-11	3.70E-11	1.14E-09

\* Reference 3, Table E-13.

TABLE 2.2-6 (SHEET 1 OF 3)  
 INGESTION DOSE FACTORS FOR CHILD\*  
 (mrem per pci ingested)

Nuclide	Bone	Liver	T Body	Thyroid	Kidney	Lung	GI-LLI
H-3	No Data	2.03E-07	2.03E-07	2.03E-07	2.03E-07	2.03E-07	2.03E-07
C-14	1.21E-05	2.42E-06	2.42E-06	2.42E-06	2.42E-06	2.42E-06	2.42E-06
Na-24	5.80E-06	5.80E-06	5.80E-06	5.80E-06	5.80E-06	5.80E-06	5.80E-06
P-32	8.25E-04	3.86E-05	3.18E-05	No Data	No Data	No Data	2.28E-05
Cr-51	No Data	No Data	8.90E-09	4.94E-09	1.35E-09	9.02E-09	4.72E-07
Mn-54	No Data	1.07E-05	2.85E-06	No Data	3.00E-06	No Data	8.98E-06
Mn-56	No Data	3.34E-07	7.54E-08	No Data	4.04E-07	No Data	4.84E-05
Fe-55	1.15E-05	6.10E-06	1.89E-06	No Data	No Data	3.45E-06	1.13E-06
Fe-59	1.65E-05	2.67E-05	1.33E-05	No Data	No Data	7.74E-06	2.78E-05
Co-58	No Data	1.80E-06	5.51E-06	No Data	No Data	No Data	1.05E-05
Co-60	No Data	5.29E-06	1.56E-05	No Data	No Data	No Data	2.93E-05
Ni-63	5.38E-04	2.88E-05	1.83E-05	No Data	No Data	No Data	1.94E-06
Ni-65	2.22E-06	2.09E-07	1.22E-07	No Data	No Data	No Data	2.56E-05
Cu-64	No Data	2.45E-07	1.48E-07	No Data	5.92E-07	No Data	1.15E-05
Zn-65	1.37E-05	3.65E-05	2.27E-05	No Data	2.30E-05	No Data	6.41E-06
Zn-69	4.38E-08	6.33E-08	5.85E-09	No Data	3.84E-08	No Data	3.99E-06
Br-83	No Data	No Data	1.71E-07	No Data	No Data	No Data	LT E-24
Br-84	No Data	No Data	1.98E-07	No Data	No Data	No Data	LT E-24
Br-85	No Data	No Data	9.12E-09	No Data	No Data	No Data	LT E-24
Rb-86	No Data	6.70E-05	4.12E-05	No Data	No Data	No Data	4.31E-06
Rb-88	No Data	1.90E-07	1.32E-07	No Data	No Data	No Data	9.32E-09
Rb-89	No Data	1.17E-07	1.04E-07	No Data	No Data	No Data	1.02E-09
Sr-89	1.32E-03	No Data	3.77E-05	No Data	No Data	No Data	5.11E-05
Sr-90	1.70E-02	No Data	4.31E-03	No Data	No Data	No Data	2.29E-04
Sr-91	2.40E-05	No Data	9.06E-07	No Data	No Data	No Data	5.30E-05
Sr-92	9.03E-06	No Data	3.62E-07	No Data	No Data	No Data	1.71E-04
Y-90	4.11E-08	No Data	1.10E-09	No Data	No Data	No Data	1.17E-04
Y-91M	3.82E-10	No Data	1.39E-11	No Data	No Data	No Data	7.48E-07
Y-91	6.02E-07	No Data	1.61E-08	No Data	No Data	No Data	8.02E-05
Y-92	3.60E-09	No Data	1.03E-10	No Data	No Data	No Data	1.04E-04

\* Reference 3, Table E-13.

TABLE 2.2-7 (SHEET 1 OF 3)

INGESTION DOSE FACTORS FOR TEENAGER\*  
(mrem per pci ingested)

Nuclide	Bone	Liver	T Body	Thyroid	Kidney	Lung	GI-LLI
H-3	No Data	1.06E-07	1.06E-07	1.06E-07	1.06E-07	1.06E-07	1.06E-07
C-14	4.06E-06	8.12E-07	8.12E-07	8.12E-07	8.12E-07	8.12E-07	8.12E-07
Na-24	2.30E-06	2.30E-06	2.30E-06	2.30E-06	2.30E-06	2.30E-06	2.30E-06
P-32	2.76E-04	1.71E-05	1.07E-05	No Data	No Data	No Data	2.32E-05
Cr-51	No Data	No Data	3.60E-09	2.00E-09	7.89E-10	5.14E-09	6.05E-07
Mn-54	No Data	5.90E-06	1.17E-06	No Data	1.76E-06	No Data	1.21E-05
Mn-56	No Data	1.58E-07	2.81E-08	No Data	2.00E-07	No Data	1.04E-05
Fe-55	3.78E-06	2.68E-06	6.25E-07	No Data	No Data	1.70E-06	1.16E-06
Fe-59	5.87E-06	1.37E-05	5.29E-06	No Data	No Data	4.32E-06	3.24E-05
Co-58	No Data	9.72E-07	2.24E-06	No Data	No Data	No Data	1.34E-05
Co-60	No Data	2.81E-06	6.33E-06	No Data	No Data	No Data	3.66E-05
Ni-63	1.77E-04	1.25E-05	6.00E-06	No Data	No Data	No Data	1.99E-06
Ni-65	7.49E-07	9.57E-08	4.36E-08	No Data	No Data	No Data	5.19E-06
Cu-64	No Data	1.15E-07	5.41E-08	No Data	2.91E-07	No Data	8.92E-06
Zn-65	5.75E-06	2.00E-05	9.33E-06	No Data	1.28E-05	No Data	8.47E-06
Zn-69	1.47E-08	2.80E-08	1.96E-09	No Data	1.83E-08	No Data	5.16E-08
Br-83	No Data	No Data	5.74E-08	No Data	No Data	No Data	LT E-24
Br-84	No Data	No Data	7.22E-08	No Data	No Data	No Data	LT E-24
Br-85	No Data	No Data	3.05E-09	No Data	No Data	No Data	LT E-24
Rb-86	No Data	2.98E-05	1.40E-05	No Data	No Data	No Data	4.41E-06
Rb-88	No Data	8.52E-08	4.54E-08	No Data	No Data	No Data	7.30E-15
Rb-89	No Data	5.50E-08	3.69E-08	No Data	No Data	No Data	8.43E-17
Sr-89	4.40E-04	No Data	1.26E-05	No Data	No Data	No Data	5.24E-05
Sr-90	8.30E-03	No Data	2.05E-03	No Data	No Data	No Data	2.33E-04
Sr-91	8.07E-06	No Data	3.21E-07	No Data	No Data	No Data	3.66E-05
Sr-92	3.05E-06	No Data	1.30E-07	No Data	No Data	No Data	7.77E-05
Y-90	1.37E-08	No Data	3.69E-10	No Data	No Data	No Data	1.13E-04
Y-91M	1.29E-10	No Data	4.93E-12	No Data	No Data	No Data	6.09E-09
Y-91	2.01E-07	No Data	5.39E-09	No Data	No Data	No Data	8.24E-05
Y-92	1.21E-09	No Data	3.50E-11	No Data	No Data	No Data	3.32E-05

\* Reference 3, Table E-12.

TABLE 2.2-6 (SHEET 3 OF 3)

INGESTION DOSE FACTORS FOR CHILD\*  
(mrem per pci ingested)

<u>Nuclide</u>	<u>Bone</u>	<u>Liver</u>	<u>T. Body</u>	<u>Thyroid</u>	<u>Kidney</u>	<u>Lung</u>	<u>GI-LLI</u>
La-140	1.01E-08	3.53E-09	1.19E-09	No Data	No Data	No Data	9.84E-05
La-142	5.24E-10	1.67E-10	5.23E-11	No Data	No Data	No Data	3.31E-05
Ce-141	3.97E-08	1.98E-08	2.94E-09	No Data	8.68E-09	No Data	2.47E-05
Ce-143	6.99E-09	3.79E-06	5.49E-10	No Data	1.59E-09	No Data	5.55E-05
Ce-144	2.08E-06	6.52E-07	1.11E-07	No Data	3.61E-07	No Data	1.70E-04
Pr-143	3.93E-08	1.18E-08	1.95E-09	No Data	6.39E-09	No Data	4.24E-05
Pr-144	1.29E-10	3.99E-11	6.49E-12	No Data	2.11E-11	No Data	8.59E-08
Nd-147	2.79E-08	2.26E-08	1.75E-09	No Data	1.24E-08	No Data	3.58E-05
W-187	4.29E-07	2.54E-07	1.14E-07	No Data	No Data	No Data	3.57E-05
Np-239	5.25E-09	3.77E-10	2.65E-10	No Data	1.09E-09	No Data	2.79E-05

\* Reference 3, Table E-13.

TABLE 2.2-7 (SHEET 3 OF 3)

INGESTION DOSE FACTORS FOR TEENAGER\*  
(mrem per pci ingested)

Nuclide	Bone	Liver	T Body	Thyroid	Kidney	Lung	GI-LLI
Ba-140	2.84E-05	3.48E-08	1.83E-06	No Data	1.18E-08	2.34E-08	4.38E-05
Ba-141	6.71E-08	5.01E-11	2.24E-09	No Data	4.65E-11	3.43E-11	1.43E-13
Ba-142	2.99E-08	2.99E-11	1.84E-09	No Data	2.53E-11	1.99E-11	9.18E-20
La-140	3.48E-09	1.71E-09	4.55E-10	No Data	No Data	No Data	9.82E-05
La-142	1.79E-10	7.95E-11	1.98E-11	No Data	No Data	No Data	2.42E-06
Ce-141	1.33E-08	8.88E-09	1.02E-09	No Data	4.18E-09	No Data	2.54E-05
Ce-143	2.35E-09	1.71E-06	1.91E-10	No Data	7.67E-10	No Data	5.14E-05
Ce-144	6.96E-07	2.88E-07	3.74E-08	No Data	1.72E-07	No Data	1.75E-04
Pr-143	1.31E-08	5.23E-09	6.52E-10	No Data	3.04E-09	No Data	4.31E-05
Pr-144	4.30E-11	1.75E-11	2.18E-12	No Data	1.01E-11	No Data	4.74E-14
Nd-147	9.38E-09	1.02E-08	6.11E-10	No Data	5.99E-09	No Data	3.68E-05
W-187	1.46E-07	1.19E-07	4.17E-08	No Data	No Data	No Data	3.22E-05
Np-239	1.76E-09	1.66E-10	9.22E-11	No Data	5.21E-10	No Data	2.67E-05

\* Reference 3, Table E-12.

TABLE 2.2-7 (SHEET 2 OF 3)

INGESTION DOSE FACTORS FOR TEENAGER\*  
(mrem per pci ingested)

Nuclide	Bone	Liver	T Body	Thyroid	Kidney	Lung	GI-LLI
Y-93	3.83E-09	No Data	1.05E-10	No Data	No Data	No Data	1.17E-04
Zr-95	4.12E-08	1.30E-08	8.94E-09	No Data	1.91E-08	No Data	3.00E-05
Zr-97	2.37E-09	4.69E-10	2.16E-10	No Data	7.11E-10	No Data	1.27E-04
Nb-95	8.22E-09	4.56E-09	2.51E-09	No Data	4.42E-09	No Data	1.95E-05
Mo-99	No Data	6.03E-06	1.15E-06	No Data	1.38E-05	No Data	1.08E-05
Tc-99M	3.32E-10	9.26E-10	1.20E-08	No Data	1.38E-08	5.14E-10	6.08E-07
Tc-101	3.60E-10	5.12E-10	5.03E-09	No Data	9.26E-09	3.12E-10	8.75E-17
Ru-103	2.55E-07	No Data	1.09E-07	No Data	8.99E-07	No Data	2.13E-05
Ru-105	2.18E-08	No Data	8.46E-09	No Data	2.75E-07	No Data	1.76E-05
Ru-106	3.92E-06	No Data	4.94E-07	No Data	7.56E-06	No Data	1.88E-04
Ag-110M	2.05E-07	1.94E-07	1.18E-07	No Data	3.70E-07	No Data	5.45E-05
Te-125M	3.83E-06	1.38E-06	5.12E-07	1.07E-06	No Data	No Data	1.13E-05
Te-127M	9.67E-06	3.43E-06	1.15E-06	2.30E-06	3.92E-05	No Data	2.41E-05
Te-127	1.58E-07	5.60E-08	3.40E-08	1.09E-07	6.40E-07	No Data	1.22E-05
Te-129M	1.63E-05	6.05E-06	2.58E-06	5.26E-06	6.82E-05	No Data	6.12E-05
Te-129	4.48E-08	1.67E-08	1.09E-08	3.20E-08	1.88E-07	No Data	2.45E-07
Te-131M	2.44E-06	1.17E-06	9.76E-07	1.76E-06	1.22E-05	No Data	9.39E-05
Te-131	2.79E-08	1.15E-08	8.72E-09	2.15E-08	1.22E-07	No Data	2.29E-09
Te-132	3.49E-06	2.21E-06	2.08E-06	2.33E-06	2.12E-05	No Data	7.00E-05
I-130	1.03E-06	2.98E-06	1.19E-06	2.43E-04	4.59E-06	No Data	2.29E-06
I-131	5.85E-06	8.19E-06	4.40E-06	2.39E-03	1.41E-05	No Data	1.62E-06
I-132	2.79E-07	7.30E-07	2.62E-07	2.46E-05	1.15E-06	No Data	3.18E-07
I-133	2.01E-06	3.41E-06	1.04E-06	4.76E-04	5.98E-06	No Data	2.58E-06
I-134	1.46E-07	3.87E-07	1.39E-07	6.45E-06	6.10E-07	No Data	5.10E-09
I-135	6.10E-07	1.57E-06	5.82E-07	1.01E-04	2.48E-06	No Data	1.74E-06
Cs-134	8.37E-05	1.97E-04	9.14E-05	No Data	6.26E-05	2.39E-05	2.45E-06
Cs-136	8.59E-06	3.38E-05	2.27E-05	No Data	1.84E-05	2.90E-06	2.72E-06
Cs-137	1.12E-04	1.49E-04	5.19E-05	No Data	5.07E-05	1.97E-05	2.12E-06
Cs-138	7.76E-08	1.49E-07	7.45E-08	No Data	1.10E-07	1.28E-08	6.76E-11
Ba-139	1.39E-07	9.78E-11	4.05E-09	No Data	9.22E-11	6.74E-11	1.24E-06

\* Reference 3, Table E-12.



TABLE 2.2-8 (SHEET 2 OF 3)

INGESTION DOSE FACTORS FOR ADULTS\*  
(mrem per pci ingested)

Nuclide	Bone	Liver	T Body	Thyroid	Kidney	Lung	GI-LLI
-Y-93	2.68E-09	No Data	7.40E-11	No Data	No Data	No Data	8.50E-05
Zr-95	3.04E-08	9.75E-09	6.60E-09	No Data	1.53E-08	No Data	3.09E-05
Zr-97	1.68E-09	3.39E-10	1.55E-10	No Data	5.12E-10	No Data	1.05E-04
Nb-95	6.22E-09	3.46E-09	1.86E-09	No Data	3.42E-09	No Data	2.10E-05
Mo-99	No Data	4.31E-06	8.20E-07	No Data	9.76E-06	No Data	9.99E-06
Tc-99M	2.47E-10	6.98E-10	8.89E-09	No Data	1.06E-08	3.42E-10	4.13E-07
Tc-101	2.54E-10	3.66E-10	3.59E-09	No Data	6.59E-09	1.87E-10	1.10E-21
Ru-103	1.85E-07	No Data	7.97E-08	No Data	7.06E-07	No Data	2.16E-05
Ru-105	1.54E-08	No Data	6.08E-09	No Data	1.99E-07	No Data	9.42E-06
Ru-106	2.75E-06	No Data	3.48E-07	No Data	5.31E-06	No Data	1.78E-04
Ag-110M	1.60E-07	1.48E-07	8.79E-08	No Data	2.91E-07	No Data	6.04E-05
Sb-124	2.81E-06	5.3E-08	1.11E-06	6.79E-09	No Data	2.18E-06	7.95E-05
Sb-125	2.23E-06	2.4E-08	4.48E-07	1.98E-09	No Data	2.33E-04	1.97E-05
Te-125M	2.68E-06	9.71E-07	3.59E-07	8.06E-07	1.09E-05	No Data	1.07E-05
Te-127M	6.77E-06	2.42E-06	8.25E-07	1.73E-06	2.75E-05	No Data	2.27E-05
Te-127	1.10E-07	3.95E-08	2.38E-08	8.15E-08	4.48E-07	No Data	8.68E-06
Te-129M	1.15E-05	4.29E-06	1.82E-06	3.95E-06	4.80E-05	No Data	5.79E-05
Te-129	3.14E-08	1.18E-08	7.65E-09	2.41E-08	1.3E-07	No Data	2.37E-08
Te-131M	1.73E-06	8.46E-07	7.05E-07	1.34E-06	8.57E-06	No Data	8.40E-05
Te-131	1.97E-08	8.23E-09	6.22E-09	1.62E-08	8.63E-08	No Data	2.79E-09
Te-132	2.52E-06	1.63E-06	1.53E-06	1.80E-06	1.57E-05	No Data	7.71E-05
I-130	7.56E-07	2.23E-06	8.80E-07	1.89E-04	3.48E-06	No Data	1.92E-06
I-131	4.16E-06	5.95E-06	3.41E-06	1.95E-03	1.02E-05	No Data	1.57E-06
I-132	2.03E-07	5.43E-07	1.90E-07	1.90E-05	8.65E-07	No Data	1.02E-07
I-133	1.42E-06	2.47E-06	7.53E-07	3.63E-04	4.31E-06	No Data	2.22E-06
I-134	1.06E-07	2.88E-07	1.03E-07	4.99E-06	4.58E-07	No Data	2.51E-10
I-135	4.43E-07	1.16E-06	4.28E-07	7.65E-05	1.86E-06	No Data	1.31E-06
Cs-134	6.22E-05	1.48E-04	1.21E-04	No Data	4.79E-05	1.59E-05	2.59E-06
Cs-136	6.51E-06	2.57E-05	1.85E-05	No Data	1.43E-05	1.96E-06	2.92E-06
Cs-137	7.97E-05	1.09E-04	7.14E-05	No Data	3.70E-05	1.23E-05	2.11E-06
Cs-138	5.52E-08	1.09E-07	5.40E-08	No Data	8.01E-08	7.91E-09	4.65E-13
-Ba-139	9.70E-08	6.91E-11	2.84E-09	No Data	6.46E-11	3.92E-11	1.72E-07
Ba-140	2.03E-05	2.55E-08	1.33E-06	No Data	8.67E-09	1.46E-08	4.18E-05
Ba-141	4.71E-08	3.56E-11	1.59E-09	No Data	3.31E-11	2.02E-11	2.22E-17
Ba-142	2.13E-08	2.19E-11	1.34E-09	No Data	1.85E-11	1.24E-11	3.00E-26

\* Reference 3, Table E-11.

TABLE 2.2-8 (SHEET 1 OF 3)

INGESTION DOSE FACTORS FOR ADULTS\*  
(mrem per pci ingested)

Nuclide	Bone	Liver	T Body	Thyroid	Kidney	Lung	GI-LLI
H-3	No Data	1.05E-07	1.05E-07	1.05E-07	1.05E-07	1.05E-07	1.05E-07
C-14	2.84E-06	5.68E-07	5.68E-07	5.68E-07	5.68E-07	5.68E-07	5.68E-07
Na-24	1.70E-06	1.70E-06	1.70E-06	1.70E-06	1.70E-06	1.70E-06	1.70E-06
P-32	1.93E-04	1.20E-05	7.46E-06	No Data	No Data	No Data	2.17E-05
Cr-51	No Data	No Data	2.66E-09	1.59E-09	5.86E-10	3.53E-09	6.69E-07
Mn-54	No Data	4.57E-06	8.72E-07	No Data	1.36E-06	No Data	1.40E-05
Mn-56	No Data	1.15E-07	2.04E-08	No Data	1.46E-07	No Data	3.67E-06
Fe-55	2.75E-06	1.90E-06	4.43E-07	No Data	No Data	1.06E-06	1.09E-06
Fe-59	4.34E-06	1.02E-05	3.91E-06	No Data	No Data	2.85E-06	3.40E-05
Co-58	No Data	7.45E-07	1.67E-06	No Data	No Data	No Data	1.51E-05
Co-60	No Data	2.14E-06	4.72E-06	No Data	No Data	No Data	4.02E-05
Ni-63	1.30E-04	9.01E-06	4.36E-06	No Data	No Data	No Data	1.88E-06
Ni-65	5.28E-07	6.86E-08	3.13E-08	No Data	No Data	No Data	1.74E-06
Cu-64	No Data	8.33E-08	3.91E-08	No Data	2.10E-07	No Data	7.10E-06
Zn-65	4.84E-06	1.54E-05	6.96E-06	No Data	1.03E-05	No Data	9.70E-06
Zn-69	1.03E-08	1.97E-08	1.37E-09	No Data	1.28E-08	No Data	2.96E-09
Br-83	No Data	No Data	4.02E-08	No Data	No Data	No Data	5.79E-08
Br-84	No Data	No Data	5.21E-08	No Data	No Data	No Data	4.09E-13
Br-85	No Data	No Data	2.14E-09	No Data	No Data	No Data	LT E-24
Rb-86	No Data	2.11E-05	9.83E-06	No Data	No Data	No Data	4.16E-06
Rb-88	No Data	6.05E-08	3.21E-08	No Data	No Data	No Data	8.36E-19
Rb-89	No Data	4.01E-08	2.82E-08	No Data	No Data	No Data	2.33E-21
Sr-89	3.08E-04	No Data	8.84E-06	No Data	No Data	No Data	4.94E-05
Sr-90	7.58E-03	No Data	1.86E-03	No Data	No Data	No Data	2.19E-04
Sr-91	5.67E-06	No Data	2.29E-07	No Data	No Data	No Data	2.70E-05
Sr-92	2.15E-06	No Data	9.30E-08	No Data	No Data	No Data	4.26E-05
Y-90	9.62E-09	No Data	2.58E-10	No Data	No Data	No Data	1.02E-04
Y-91M	9.09E-11	No Data	3.52E-12	No Data	No Data	No Data	2.67E-10
Y-91	1.41E-07	No Data	3.77E-09	No Data	No Data	No Data	7.76E-05
Y-92	8.45E-10	No Data	2.47E-11	No Data	No Data	No Data	1.48E-05

\* Reference 3, Table E-11.



TABLE 2.2-9 (SHEET 1 OF 2)

EXTERNAL DOSE FACTORS FOR STANDING ON CONTAMINATED GROUND\*  
(mrem/h per pCi/m<sup>2</sup>)

<u>RADIONUCLIDE</u>	<u>TOTAL BODY</u>	<u>SKIN</u>
H-3	0.0	0.0
C-14	0.0	0.0
Na-24	2.50E-08	2.90E-08
P-32	0.0	0.0
Cr-51	2.20E-10	2.60E-10
Mn-54	5.80E-09	6.80E-09
Mn-56	1.10E-08	1.30E-08
Fe-55	0.0	0.0
Fe-59	8.00E-09	9.40E-09
Co-58	7.00E-09	8.20E-09
Co-60	1.70E-08	2.00E-08
Ni-63	0.0	0.0
Ni-65	3.70E-09	4.30E-09
Cu-64	1.50E-09	1.70E-09
Zn-65	4.00E-09	4.60E-09
Zn-69	0.0	0.0
Br-83	6.40E-11	9.30E-11
Br-84	1.20E-08	1.40E-08
Br-85	0.0	0.0
Rb-86	6.30E-10	7.20E-10
Rb-88	3.50E-09	4.00E-09
Rb-89	1.50E-08	1.80E-08
Sr-89	5.60E-13	6.50E-13
Sr-91	7.10E-09	8.30E-09
Sr-92	9.00E-09	1.00E-08
Y-90	2.20E-12	2.60E-12
Y-91M	3.80E-09	4.40E-09
Y-91	2.40E-11	2.70E-11
Y-92	1.60E-09	1.90E-09
Y-93	5.70E-10	7.80E-10
Zr-95	5.00E-09	5.80E-09
Zr-97	5.50E-09	6.40E-09
Nb-95	5.10E-09	6.00E-09
Mo-99	1.90E-09	2.20E-09
Tc-99M	9.60E-10	1.10E-09
Tc-101	2.70E-09	3.00E-09
Ru-103	3.60E-09	4.20E-09
Ru-105	4.50E-09	5.10E-09
Ru-106	1.50E-09	1.80E-09
Ag-110M	1.80E-08	2.10E-08
Te-125M	3.50E-11	4.80E-11

\* Reference 3, Table E-6.

TABLE 2.2-8 (SHEET 3 OF 3)

INGESTION DOSE FACTORS FOR ADULTS\*  
(mrem per pci ingested)

<u>Nuclide</u>	<u>Bone</u>	<u>Liver</u>	<u>T Body</u>	<u>Thyroid</u>	<u>Kidney</u>	<u>Lung</u>	<u>GI-LLI</u>
La-140	2.50E-09	1.26E-09	3.33E-10	No Data	No Data	No Data	9.25E-05
La-142	1.28E-10	5.82E-11	1.45E-11	No Data	No Data	No Data	4.25E-07
Ce-141	9.36E-09	6.33E-09	7.18E-10	No Data	2.94E-09	No Data	2.42E-05
Ce-143	1.65E-09	1.22E-06	1.35E-10	No Data	5.37E-10	No Data	4.56E-05
Ce-144	4.88E-07	2.04E-07	2.62E-08	No Data	1.21E-07	No Data	1.65E-04
Pr-143	9.20E-09	3.69E-09	4.56E-10	No Data	2.13E-09	No Data	4.03E-05
Pr-144	3.01E-11	1.25E-11	1.53E-12	No Data	7.05E-12	No Data	4.33E-18
Nd-147	6.29E-09	7.27E-09	4.35E-10	No Data	4.25E-09	No Data	3.49E-05
W-187	1.03E-07	8.61E-08	3.01E-08	No Data	No Data	No Data	2.82E-05
Np-239	1.19E-09	1.17E-10	6.45E-11	No Data	3.65E-10	No Data	2.40E-05

\* Reference 3, Table E-11.

TABLE 2.2-10  
INDIVIDUAL USAGE FACTORS\*

	<u>INFANT</u>	<u>CHILD</u>	<u>TEENAGER</u>	<u>ADULT</u>
Milk Consumption Rate, $U_{m,p}$ (liters/year)	330	330	400	310
Meat Consumption Rate, $U_{m,p}$ (kg/year)	0	41	65	110
Fresh Leafy Vegetation Consumption Rate, $U_{v,l}$ (kg/year)	0	26	42	64
Stored Vegetation Consumption Rate, $U_{v,s}$ (kg/year)	0	520	630	520
Breathing Rate ( $m^3$ /year)	1400	3700	8000	8000

\* Reference 3, Table E-5.

TABLE 2.2-9 (SHEET 2 OF 2)

EXTERNAL DOSE FACTORS FOR STANDING ON CONTAMINATED GROUND\*  
(mrem/h per pCi/m<sup>2</sup>)

<u>RADIONUCLIDE</u>	<u>TOTAL BODY</u>	<u>SKIN</u>
Te-127M	1.10E-12	1.30E-12
Te-127	1.00E-11	1.10E-11
Te-129M	7.70E-10	9.00E-10
Te-129	7.10E-10	8.40E-10
Te-131M	8.40E-09	9.90E-09
Te-131	2.20E-09	2.60E-06
Te-132	1.70E-09	2.00E-09
I-130	1.40E-08	1.70E-08
I-131	2.80E-09	3.40E-09
I-132	1.70E-08	2.00E-08
I-133	3.70E-09	4.50E-09
I-134	1.60E-08	1.90E-08
I-135	1.20E-08	1.40E-08
Cs-134	1.20E-08	1.40E-08
Cs-136	1.50E-08	1.70E-08
Cs-137	4.20E-09	4.90E-09
Cs-138	2.10E-08	2.40E-08
Ba-139	2.40E-09	2.70E-09
Ba-140	2.10E-09	2.40E-09
Ba-141	4.30E-09	4.90E-09
Ba-142	7.90E-09	9.00E-09
La-140	1.50E-08	1.70E-08
La-142	1.50E-08	1.80E-08
Ce-141	5.50E-10	6.20E-10
Ce-143	2.20E-09	2.50E-09
Ce-144	3.20E-10	3.70E-10
Pr-143	0.0	0.0
Pr-144	2.00E-10	2.30E-10
Nd-147	1.00E-09	1.20E-09
W-187	3.10E-09	3.60E-09
Np-239	9.50E-10	1.10E-09

\* Reference 3, Table E-6.

TABLE 2.2-12  
CONTROLLING RECEPTOR  
(To support subsection 2.2)

The location and exposure pathways associated with the controlling receptors are determined during the Annual Land Use Census. Dispersion and deposition values were calculated based on VEGP site meteorological data collected for the period January 1, 1985, through December 31, 1987 (Reference 13).

Sector: WSW                      Distance: 1.2 miles                      Age Group: Child

Dispersion:  $(\overline{X/Q'})_{GP} = 6.2E-7 \text{ s/m}^3$   $(\overline{X/Q'})_{MP} = 1.27E-7 \text{ s/m}^3$

Deposition:  $(\overline{D/Q'})_{GP} = 2.8E-9 \text{ m}^{-2}$   $(\overline{D/Q'})_{MP} = 9.9E-10 \text{ m}^{-2}$

Exposure pathways: Inhalation and ground plane.

\* References 12, 13, 15 and 16.

TABLE 2.2-11

STABLE ELEMENT TRANSFER DATA\*  
(Milk-days/liter; Meat-days/kg)

ELEMENT	F <sub>m</sub> - MILK	F <sub>m</sub> - MILK	F <sub>f</sub> - MEAT
	(COW)	(GOAT)	
H	1.0E-02	1.7E-01	1.2E-02
C	1.2E-02	1.0E-01	3.1E-02
Na	4.0E-02	4.0E-02	3.0E-02
P	2.5E-02	2.5E-01	4.6E-02
Cr	2.2E-03	2.2E-03	2.4E-03
Mn	2.5E-04	2.5E-04	8.0E-04
Fe	1.2E-03	1.3E-04	4.0E-02
Co	1.0E-03	1.0E-03	1.3E-02
Ni	6.7E-03	6.7E-03	5.3E-02
Cu	1.4E-02	1.3E-02	8.0E-03
Zn	3.9E-02	3.9E-02	3.0E-02
Rb	3.0E-02	3.0E-02	3.1E-02
Sr	8.0E-04	1.4E-02	6.0E-04
Y	1.0E-05	1.0E-05	4.6E-03
Zr	5.0E-06	5.0E-06	3.4E-02
Nb	2.5E-03	2.5E-03	2.8E-01
Mo	7.5E-03	7.5E-03	8.0E-03
Tc	2.5E-02	2.5E-02	4.0E-01
Ru	1.0E-06	1.0E-06	4.0E-01
Rh	1.0E-02	1.0E-02	1.5E-03
Ag	5.0E-02	5.0E-02	1.7E-02
Te	1.0E-03	1.0E-03	7.7E-02
I	6.0E-03	6.0E-02	2.9E-03
Cs	1.2E-02	3.0E-01	4.0E-03
Ba	4.0E-04	4.0E-04	3.2E-03
La	5.0E-06	5.0E-06	2.0E-04
Ce	1.0E-04	1.0E-04	1.2E-03
Pr	5.0E-06	5.0E-06	4.7E-03
Nd	5.0E-06	5.0E-06	3.3E-03
W	5.0E-04	5.0E-04	1.3E-03
Np	5.0E-06	5.0E-06	2.0E-04

\* References 3, Table E-1; Reference 3, Table E-2 for H, C, P, Fe, Cu, Sr, I, and Cs in goat's milk; the remainder of elements in goat's milk are taken from Table E-1 as presented for cow's milk.

TABLE 2.2-13 (SHEET 2 OF 4)

SITE-SPECIFIC (OR DEFAULT) VALUES TO  
BE USED IN PATHWAY FACTOR CALCULATIONS

<u>Parameter</u>	<u>Description</u>	<u>Value</u>
Grass-Cow-Meat		
$Q_f$	Feed consumption rate for cow	50 kg/day (Reference 3, table E-3)
$U_{a,i}$	Meat consumption rate for age group	table 2.2-10
$(DFL)_{i,j,k}$	Ingestion dose factor for age group	tables 2.2-5 - 2.2-8
$Y_p$	Pasture grass areal density	0.7 kg/m <sup>2</sup> (Reference 3, table E-15)
$Y_s$	Stored feed areal density	2.0 kg/m <sup>2</sup> (Reference 3, table E-15)
$f_p$	Fraction of year that cow grazes on pasture	1.0 (Reference 1, page 33)
$f_s$	Fraction of total feed that is pasture grass while cow is on pasture	1.0 (Reference 1, page 33)
$H$	Absolute humidity of the atmosphere	8.0 gm/m <sup>3</sup> (Reference 1, page 34)
Grass-Cow-Milk		
$Q_f$	Feed consumption rate for cow	50 kg/day (Reference 3, table E-3)
$U_{a,i}$	Milk consumption rate for age group	table 2.2-10
$(DFL)_{i,j,k}$	Ingestion dose factor for age group	tables 2.2-5 - 2.2-8

TABLE 2.2-13 (SHEET 1 OF 4)

SITE-SPECIFIC (OR DEFAULT) VALUES TO  
BE USED IN PATHWAY FACTOR CALCULATIONS  
(Supports subsections 2.2.2.2 and 2.2.2.3)

<u>Parameter</u>	<u>Description</u>	<u>Value</u>
Inhalation		
$(BR)_a$	Breathing rate for age group	table 2.2-10
$(DFA)_{i,j,a}$	Inhalation dose factor for age group	tables 2.2-1, 2.2-4
Ground plane		
SHF	Shielding factor due to structure	0.7 (Reference 3, table E-15)
$(DFG)_{i,j}$	Ground plane dose factor	table 2.2-9 (Same for all age groups)
Garden Vegetation		
$Y_v$	Garden vegetation areal density	2.0 kg/m <sup>2</sup> (Reference 3, table E-15)
$U_{a,l}$	Leafy vegetation consumption rate for age group	table 2.2-10
$U_{a,s}$	Stored vegetation consumption rate for age group	table 2.2-10
$f_l$	Fraction of annual intake of leafy vegetation grown locally	1.0 (Reference 1, page 36)
$f_s$	Fraction of annual intake of stored vegetation grown locally	0.76 (Reference 1, page 36)
$H$	Absolute humidity of the atmosphere	8.0 gm/m <sup>3</sup> (Reference 1, page 34)



TABLE 2.2-13 (SHEET 4 OF 4)

SITE-SPECIFIC (OR DEFAULT) VALUES TO  
BE USED IN PATHWAY FACTOR CALCULATIONS

<u>Parameter</u>	<u>Description</u>	<u>Value</u>
f <sub>s</sub>	Fraction of total feed that is pasture grass while goat is on pasture	1.0 (Reference 1, page 33)
H	Absolute humidity of the atmosphere	8.0 gm/m <sup>3</sup> (Reference 1, page 34)

TABLE 2.2-13 (SHEET 3 OF 4)

SITE-SPECIFIC (OR DEFAULT) VALUES TO  
BE USED IN PATHWAY FACTOR CALCULATIONS

<u>Parameter</u>	<u>Description</u>	<u>Value</u>
$Y_p$	Pasture grass areal density	0.7 kg/m <sup>2</sup> (Reference 3, table E-15)
$Y_s$	Stored feed areal density	2.0 kg/m <sup>2</sup> (Reference 3, table E-15)
$f_p$	Fraction of year that cow grazes	1.0 (Reference 1, page 33)
$f_s$	Fraction of total feed that is pasture grass while cow is on pasture	1.0 (Reference 1, page 33)
$H$	Absolute humidity of the atmosphere	8.0 gm/m <sup>3</sup> (Reference 1, page 34)
Grass-Goat-Milk		
$Q_r$	Feed consumption rate for goat	6.0 kg/day (Reference 3, table E-3)
$U_{..}$	Milk consumption rate for age group	table 2.2-10
$(DFL)_{i,j}$	Ingestion dose factor for age group	tables 2.2-5 - 2.2-8
$Y_p$	Pasture grass areal density	0.7 kg/m <sup>2</sup> (Reference 3, table E-15)
$Y_s$	Stored feed areal density	2.0 kg/m <sup>2</sup> (Reference 3, table E-15)
$f_p$	Fraction of year that goat is on pasture	1.0 (Reference 1, page 33)

TABLE 2.2-15  
DISPERSION AND DEPOSITION PARAMETERS  
(To Support Subsection 2.2.2.3)

(This table has been deleted.)

TABLE 2.2-14

POTENTIAL RECEPTOR LOCATIONS AND PATHWAYS  
(To support subsection 2.2.2.3)

(This table has been deleted.)

$D_{o(c)}$  = the cumulative organ dose to an individual due to I-131, I-133, tritium and particulates, for the elapsed portion of the current quarter plus the release under consideration.

$t$  = the number of days into the current quarter, including the period of the release under consideration.

If operational activities planned during the ensuing 31-day period are expected to result in gaseous releases which will contribute a dose in addition to the dose due to routine gaseous effluents, this additional dose contribution should be included in the projected dose as follows:

Air Doses:

$$D_{\text{beta}(prj)} = \left[ \frac{D_{\text{beta}(c)}}{t} \times 31 \right] + D_{PA}$$

$$D_{\text{gamma}(prj)} = \left[ \frac{D_{\text{gamma}(c)}}{t} \times 31 \right] + D_{PA}$$

$$D_{o(prj)} = \left[ \frac{D_{o(c)}}{t} \times 31 \right] + D_{PA}$$

Where  $D_{PA}$  is the expected dose due to the particular planned activity.

#### 2.2.3.2 Dose Projections for Specific Releases

Dose projections may be performed for a particular release by performing a pre-release dose calculation assuming that the planned release will proceed as anticipated. For air dose projections due to noble gases, follow the methodology presented in paragraph 2.2.2.1 using sample analyses results for the particular release point and parametric values expected to exist for the release period. For individual organ dose projections, due to I-131, I-133, tritium, and particulates, follow the methodology presented in paragraph 2.2.2.2 using sample analyses results for the particular release point and parametric values expected to exist for the release period.

### 2.2.3 Dose Projections for Gaseous Effluents

#### 2.2.3.1 Thirty-One-Day Dose Projections

In order to meet the requirements of Technical Specification 3.11.2.4, which pertains to operation of the Ventilation Exhaust Treatment System and the Gaseous Waste Processing System, dose projections must be made at least once per 31 days, during periods in which discharge of gaseous effluents containing radioactive materials to unrestricted areas occurs or is expected.

Projected 31-day air doses and doses to individuals due to gaseous effluents may be determined as follows:

Air Doses:

$$D_{\text{beta}(\text{prj})} = \frac{D_{\text{beta}(\text{c})}}{t} \times 31 \quad (32)$$

$$D_{\text{gamma}(\text{prj})} = \frac{D_{\text{gamma}(\text{c})}}{t} \times 31 \quad (33)$$

Individual:

$$D_{\text{o}(\text{prj})} = \frac{D_{\text{o}(\text{c})}}{t} \times 31 \quad (34)$$

where:

$D_{\text{beta}(\text{c})}$  = the cumulative air dose, due to beta emissions from noble gases, for the elapsed portion of the current quarter plus the release under consideration.

$D_{\text{gamma}(\text{c})}$  = the cumulative air dose, due to gamma emissions from noble gases, for the elapsed portion of the current quarter plus the release under consideration.

- RCF = the open terrain recirculation factor. Values for specific distances are obtained from Appendix A of Reference 13.
- $n_{jk}$  = the number of hours meteorological conditions are observed to be in a given wind direction, windspeed class j, and stability class k. NOTE: If periodic data (hourly) are used instead of the joint frequency data, the summation over j and k is deleted and the summation is accomplished for all hours at all distances for each direction.
- N = the total hours of valid meteorological data throughout the period of interest.
- $u_{jk}$  = the wind speed (mid-point of windspeed class j) at ground level (m/s), during stability class k.
- x = the distance from release point to location of interest (m).
- $\Sigma_{z_k}$  = the vertical standard deviation of the plume concentration distribution considering the initial dispersion within the building wake.
- = the lesser of  $(\sigma_z^2 + (b^2/2\pi))^{1/2}$  or  $\sqrt{3}(\sigma_z)$
- $\sigma_{z_k}$  = the vertical standard deviation of the plume concentration distribution (m) for a given distance and stability category k as shown in figure 2.3-1. The stability category is determined by the vertical temperature gradient  $\Delta T/\Delta Z$  ( $^{\circ}\text{C}/100 \text{ m}$ ).
- $\pi$  = 3.1416
- b = the maximum height of adjacent plant structure (55 m).

## 2.3 METEOROLOGICAL MODEL

(References 7 and 13, and subsection 2.3.5 of Reference 5)

### 2.3.1 Atmospheric Dispersion

Atmospheric dispersion (long-term) may be calculated using the appropriate form of the sector-averaged straight line flow Gaussian model. Gaseous releases are considered to be either ground-level or mixed-mode. Considered as ground level are releases from the turbine building(s) vents and the dry active waste processing building vent. Releases from reactor building(s) (plant) vent(s) are considered to be mixed mode. (See NOTE in subsection 2.1.2).

#### 2.3.1.1 Ground-Level Releases

$(X/Q)_g$  = the ground-level sector-averaged relative concentration for a given wind direction (sector) and distance (s/m<sup>3</sup>).

$$= (\text{RCF}) 2.032 d_p \sum_{jk} \frac{n_{jk}}{N u_{jk} \times \sum_{zk}} \quad (38)$$

where:

2.032 =  $(2/\pi)^{1/2}$  divided by the number of radians in a 22.5° sector (0.3927 radians).

$d_p$  = the plume depletion factor for all radionuclides other than noble gases at a distance x shown in figure 2.3-2 for ground-level releases; for noble gases the depletion factor is unity. If an undepleted relative concentration is desired, the depletion factor is unity. Only depletion by deposition is considered since depletion by decay would be of little significance at the distances considered.



$W_o$  = the vent exit velocity (m/s).  
 $h$  = the effective release height (m).  
 $= h_v + h_{pr} - h_t - c_v$  (40)  
 $h_v$  = the height of release point (m).  
 $h_t$  = the maximum terrain height between the release point and the point of interest (m). (See table 2.3-1.)

$h_{pr}$  = the additional height due to plume rise (m).  
 $= 1.44d (W_o/u)^{2/3} (x/d)^{1/3}$  (41)  
 = limited by the lesser of the following two equations:

$h_{pr(max)}$  =  $3(W_o/u)d$  or  $h_{pr(max)} = 1.5(F'_m/u)^{1/3} S^{-1/6}$

$d$  = the inside diameter of vent.

$c_v$  = the correction for low vent exit velocity (m).

$$\begin{aligned}
 & 3(1.5 - \frac{W_o}{u}) d \quad \text{for } \frac{W_o}{u} \leq 1.5 \\
 & = 0 \quad \text{for } \frac{W_o}{u} > 1.5
 \end{aligned}$$

$F'_m$  = the momentum flux parameter ( $m^4/s^2$ ).  
 $= (W_o)^2 (d/2)^2$

$S$  = the stability parameter.

$$\begin{aligned}
 & 8.75 \times 10^{-4} s^{-2} \text{ for } -0.5 < \Delta T \leq 1.5 \\
 & = 1.75 \times 10^{-3} s^{-2} \text{ for } 1.5 < \Delta T \leq 4.0 \\
 & 2.45 \times 10^{-3} s^{-2} \text{ for } \Delta T > 4.0
 \end{aligned}$$

Other terms were defined in subsection 2.3.1.1.

### 2.3.1.2 Mixed-Mode Releases

$(X/Q)_m$  = the mixed-mode sector-averaged relative concentration for a given wind direction (sector) and distance ( $s/m^3$ ).

$$= 2.032(RCF)d_p \sum_{jk} \frac{n_{jk}}{Nx} \frac{E}{u_{jk} \sigma_{zk}} \quad (39)$$

$$+ \frac{(1-E)}{u_{jk} \sigma_{zk}} \exp(-h^2/2\sigma_{zk}^2)$$

where:

$d_p$  = the plume depletion factor for all radionuclides other than noble gases at a distance  $x$  shown in figures 2.3-3 through 2.3-5 for elevated releases; for noble gases the depletion factor is unity. If an undepleted relative concentration is desired, the depletion factor is unity. Only depletion by deposition is considered since depletion by decay would be of little significance at the distances considered.

$u_{jk}$  = the wind speed extrapolated to the effective release height. Extrapolation is accomplished by raising the ratio of the two heights to the  $n$  power where  $n = 0.25, 0.33$ , and  $0.5$  for unstable, neutral, and stable conditions, respectively. (Reference 5, subsection 2.3.5.)

$E$  = the fraction considered as ground-level releases.

$$1.0 \quad \text{for} \quad \frac{W_0}{u} \leq 1.0$$

$$2.58 - 1.58 \left( \frac{W_0}{u} \right) \quad \text{for} \quad 1.0 < \frac{W_0}{u} \leq 1.5$$

$$0.3 - 0.06 \left( \frac{W_0}{u} \right) \quad \text{for} \quad 1.5 < \frac{W_0}{u} \leq 5.0$$

$$0 \quad \text{for} \quad \frac{W_0}{u} > 5.0$$

where:

- $D_g$  = the relative deposition rate for the ground-level portion of mixed-mode releases from figure 2.3-6.
- $D_e$  = the relative deposition rate for the elevated portion of mixed-mode releases from figures 2.3-7 through 2.3-9.
- $E$  = the fraction of releases considered as ground level.

Other terms were defined in previous subsections.

## 2.3.2 Relative Deposition

### 2.3.2.1 Ground-Level Releases

$(D/Q)_g$  = the ground-level sector-averaged relative deposition at a given distance and for a given sector ( $1/m^2$ ).

$$= (RCF) \sum_k \frac{2.55 D_g n_k}{N x} \quad (42)$$

where:

2.55 = the inverse of the number of radians in a  $22.5^\circ$  sector  $(2 \pi/16)^{-1}$ .

$D_g$  = the deposition rate at a given distance, taken from figure 2.3-6 for ground-level releases.

$n_k$  = the number of hours the wind is directed into the sector of interest, during which time stability category  $k$  exists.

$N$  = the total number of hours of valid meteorological data.

RCF = the open terrain recirculation factor. Values for specific distances are obtained from Appendix A of Reference 13.

### 2.3.2.2 Mixed-Mode Releases

$(D/Q)_m$  = the mixed-mode sector-averaged relative deposition at a given distance and for a given sector ( $1/m^2$ ).

$$\frac{2.55}{x} (RCF) ((E)(D_g) + (1 - E) D_e) \quad (43)$$

TABLE 2.3-1 (SHEET 2 OF 2)

TERRAIN ELEVATION ABOVE PLANT GRADE  
WIND DIRECTION FROM PLANT TO RECEPTOR

<u>Distance</u> <u>(m)</u>	<u>S</u>	<u>SSW</u>	<u>SW</u>	<u>WSW</u>	<u>W</u>	<u>WNW</u>	<u>NW</u>	<u>NNW</u>
500	0.0	4.7	8.7	5.7	1.4	5.8	5.7	3.5
1,000	0.0	4.7	16.7	13.4	3.3	10.4	11.8	6.8
1,500	0.0	4.7	21.7	18.6	7.3	12.2	14.3	7.3
2,000	0.0	4.7	21.7	18.6	7.3	12.2	14.3	7.3
2,500	0.0	4.7	21.7	18.6	7.3	12.2	14.3	7.3
3,000	0.0	4.7	23.7	18.6	7.3	12.2	14.3	7.3
3,500	0.0	4.7	24.4	18.6	7.3	12.2	16.9	7.3
4,000	0.0	4.7	24.4	18.6	7.3	12.2	16.9	7.3
5,000	0.0	4.7	24.7	18.6	7.3	12.2	16.9	7.3
6,000	0.0	4.7	26.8	18.6	7.3	12.2	16.9	7.3
7,000	3.6	4.7	26.8	18.6	7.3	12.2	16.9	7.3
8,000	14.6	4.7	26.8	18.6	7.3	12.2	16.9	7.3
9,000	14.6	5.1	26.8	18.6	7.3	12.2	16.9	7.3
10,000	14.6	6.8	26.8	18.6	7.3	12.2	16.9	7.3
12,000	14.6	6.8	34.1	28.9	13.4	12.2	16.9	7.3
14,000	14.6	6.8	34.1	28.9	13.4	16.5	19.7	7.3
16,000	14.6	6.8	34.1	28.9	13.4	16.5	25.7	7.3

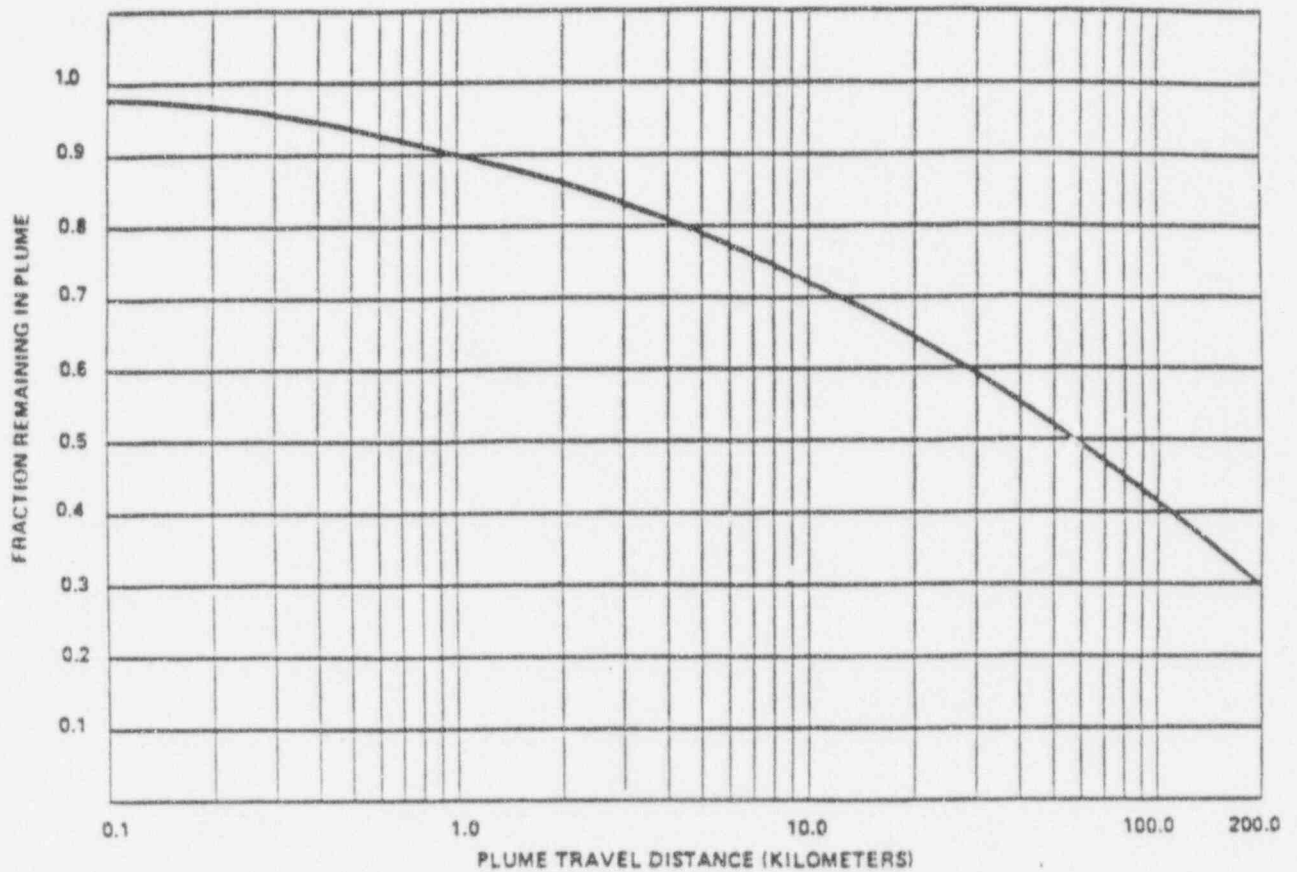
TABLE 2.3-1 (SHEET 1 OF 2)

TERRAIN ELEVATION ABOVE PLANT GRADE  
WIND DIRECTION FROM PLANT TO RECEPTOR

<u>Distance</u> <u>(m)</u>	<u>N</u>	<u>NNE</u>	<u>NE</u>	<u>ENE</u>	<u>E</u>	<u>ESE</u>	<u>SE</u>	<u>SSE</u>
500	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
1,000	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
1,500	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
2,000	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
2,500	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
3,000	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
3,500	0.0	0.0	0.0	0.0	0.0	0.0	0.0	0.0
4,000	0.0	0.0	0.0	0.0	0.0	0.0	4.5	0.0
5,000	0.0	0.0	0.0	0.0	0.0	0.0	11.1	0.0
6,000	0.0	0.0	0.0	0.0	0.0	0.0	11.1	0.0
7,000	0.0	0.0	0.0	7.8	0.0	0.0	11.1	0.0
8,000	0.0	0.0	21.1	13.9	0.0	0.0	11.8	0.0
9,000	0.0	0.0	24.4	14.6	0.0	0.0	12.7	7.1
10,000	0.0	10.2	24.4	20.2	0.0	0.0	17.1	17.0
12,000	0.0	15.9	26.8	20.2	0.0	0.0	17.1	19.5
14,000	0.0	15.9	26.8	20.2	0.0	0.0	17.1	19.5
16,000	0.0	15.9	26.8	21.7	13.2	0.0	17.1	19.5

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Reference 5.



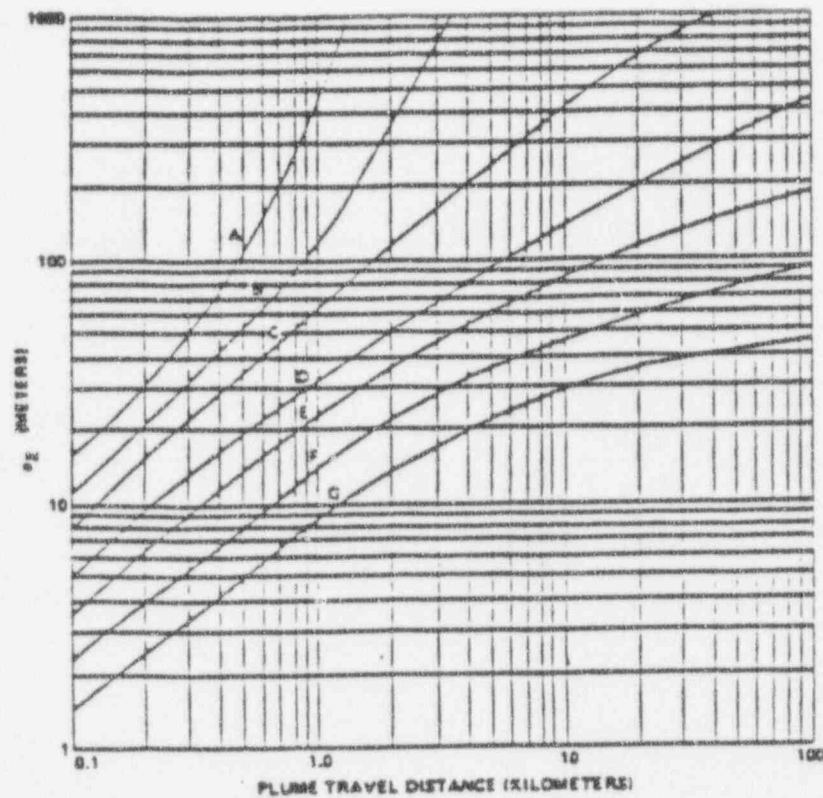
\*Reference 7.

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ELECTRIC GENERATING PLANT  
UNIT 1 AND UNIT 2

PLUME DEPLETION EFFECT FOR GROUND-  
LEVEL RELEASES  
(All Atmospheric Stability Classes)

FIGURE 2.3-2



Category	Range of Vertical Temperature Gradient ( $^{\circ}\text{C}/100\text{m}$ )	Range of Vertical Temperature Gradient ( $^{\circ}\text{F}/100\text{ft}$ )
A	$\Delta T/\Delta Z < -1.9$	$\Delta T < -1.0$
B	$-1.9 \leq \Delta T/\Delta Z < -1.7$	$-1.0 \leq \Delta T < -0.9$
C	$-1.7 \leq \Delta T/\Delta Z < -1.5$	$-0.9 \leq \Delta T < -0.8$
D	$-1.5 \leq \Delta T/\Delta Z < -0.5$	$-0.8 \leq \Delta T < -0.3$
E	$-0.5 \leq \Delta T/\Delta Z < 1.5$	$-0.3 \leq \Delta T < 0.8$
F	$1.5 \leq \Delta T/\Delta Z < 4.0$	$0.8 \leq \Delta T < 2.2$
G	$4.0 \leq \Delta T/\Delta Z$	$2.2 \leq \Delta T$

\* Reference 7.

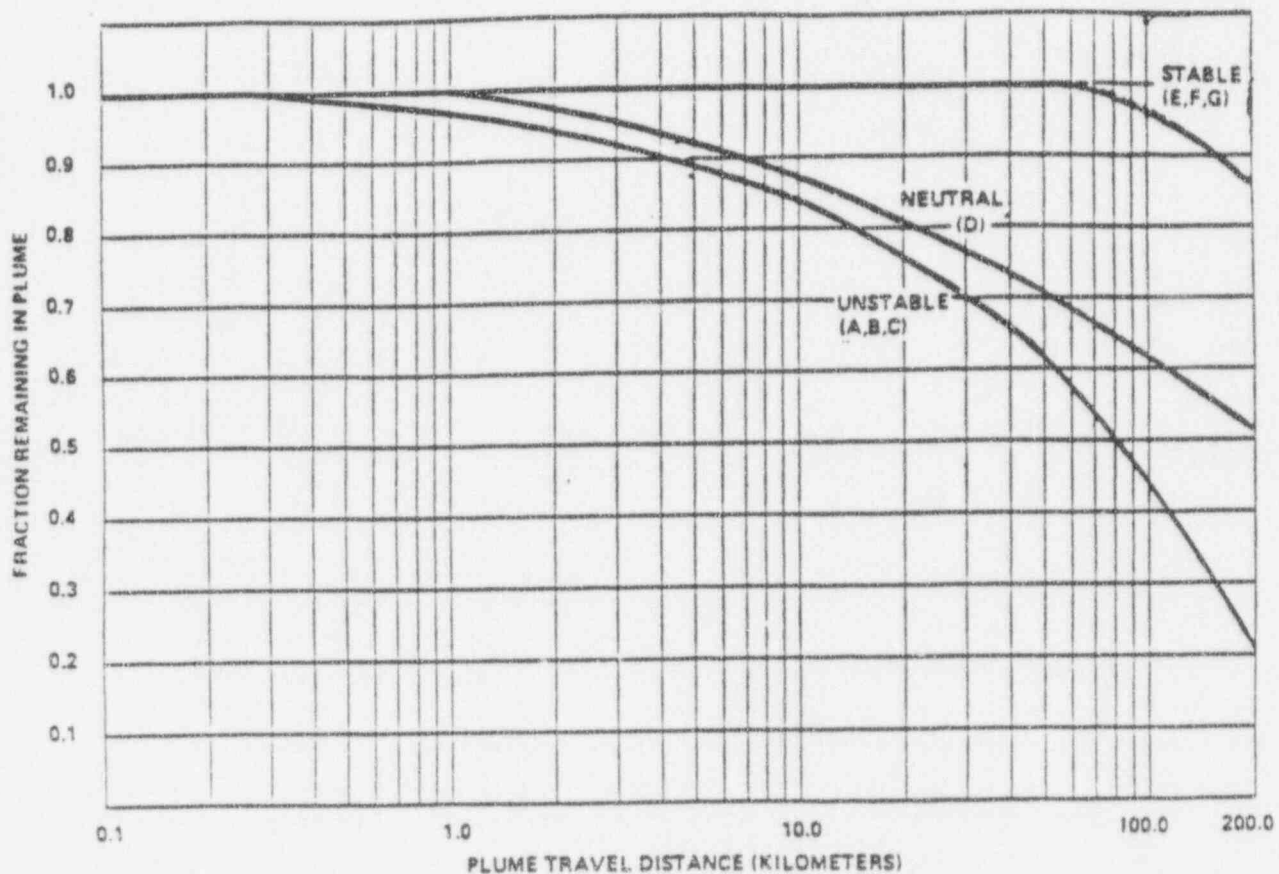
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VERTICAL STANDARD DEVIATION OF  
MATERIAL IN A PLUME ( $\sigma_z$ )\*  
(Letters Denote Pasquill Stability Class)

FIGURE 2.3-1





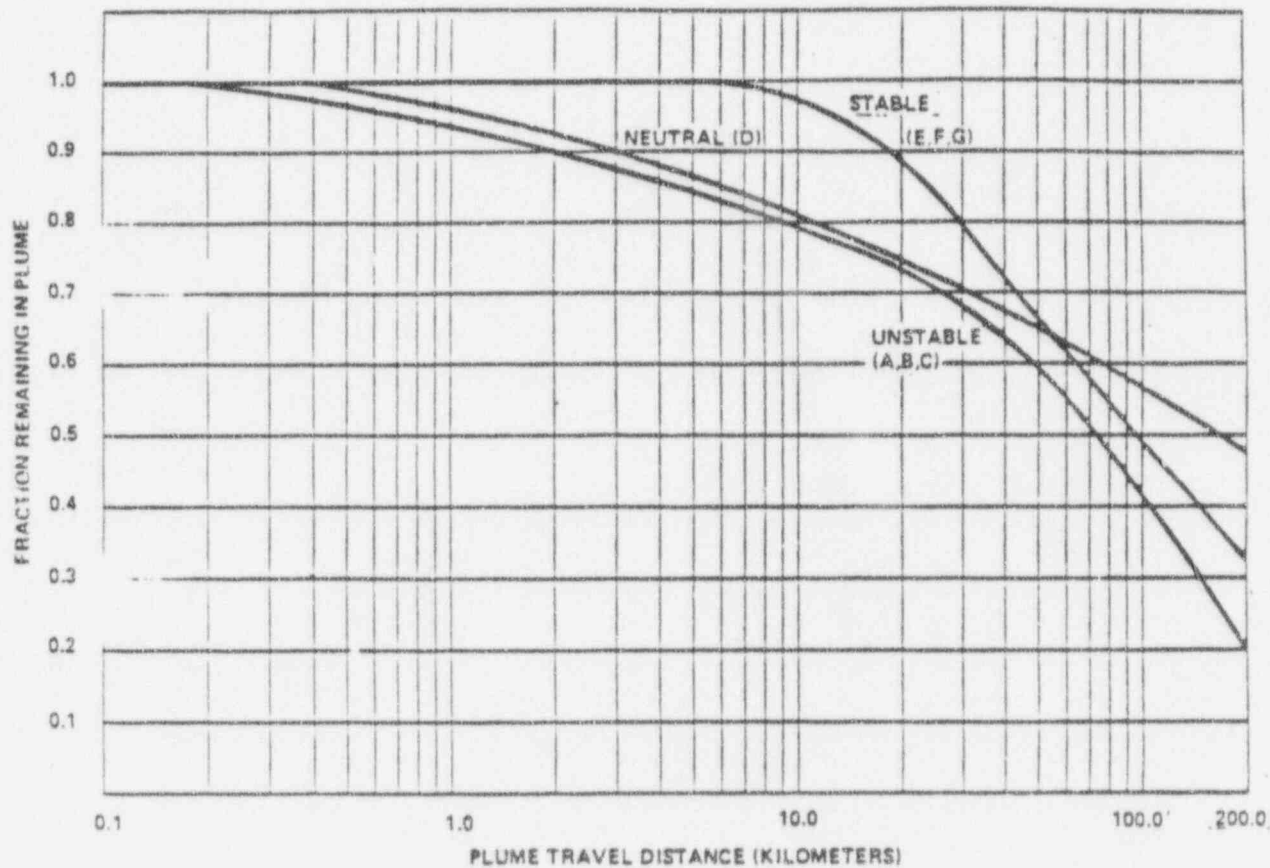
\*Reference 7.

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PLUME DEPLETION EFFECT FOR 60-METER RELEASES\*  
(Letters Denote Pasquill Stability Class)

FIGURE 2.3-4



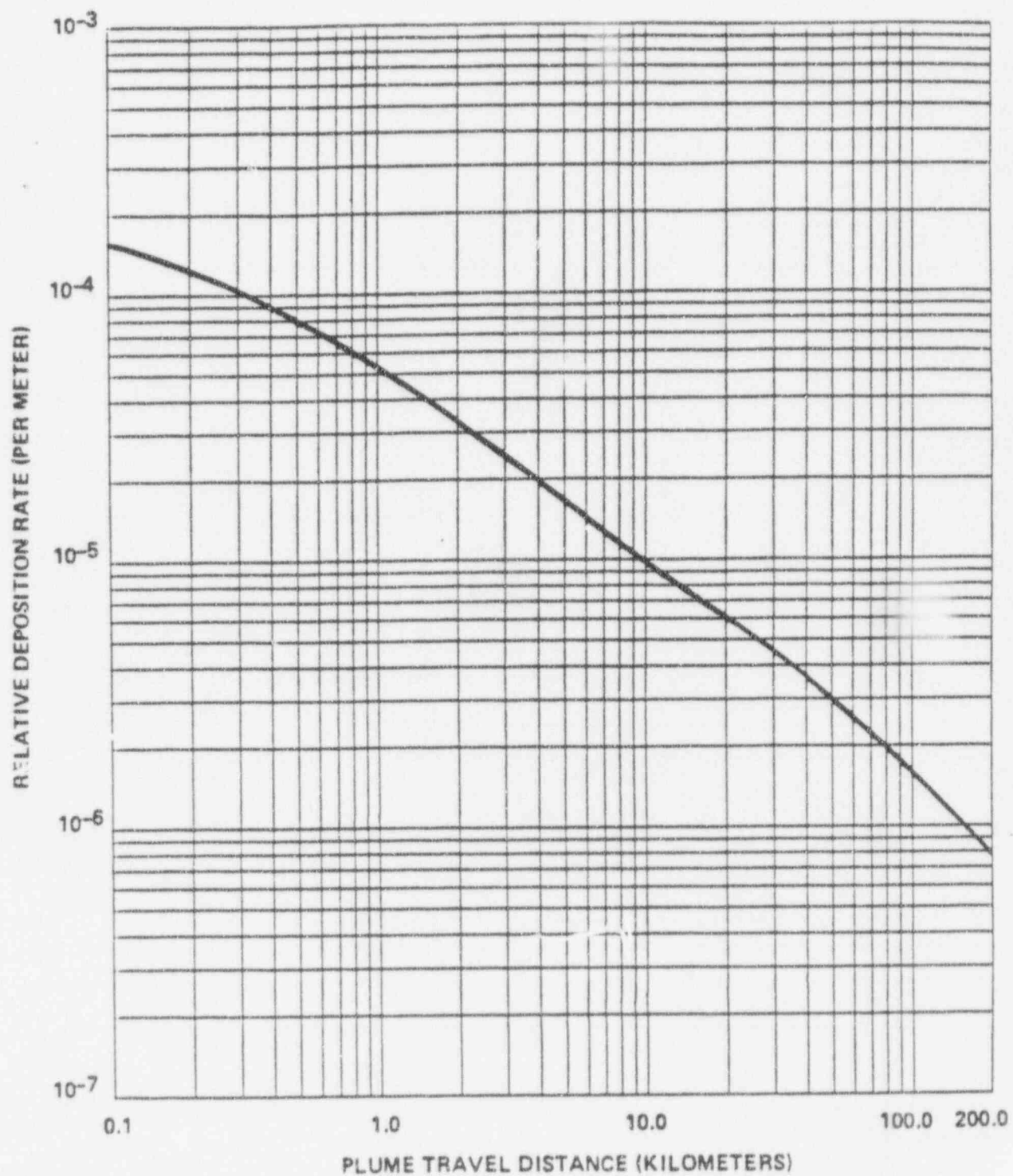
Reference 7.


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UNIT 1 AND UNIT 2

PLUME DEPLETION EFFECT FOR 30-METER RELEASES\*  
(Letters Denote Pasquill Stability Class)

FIGURE 2.3-3

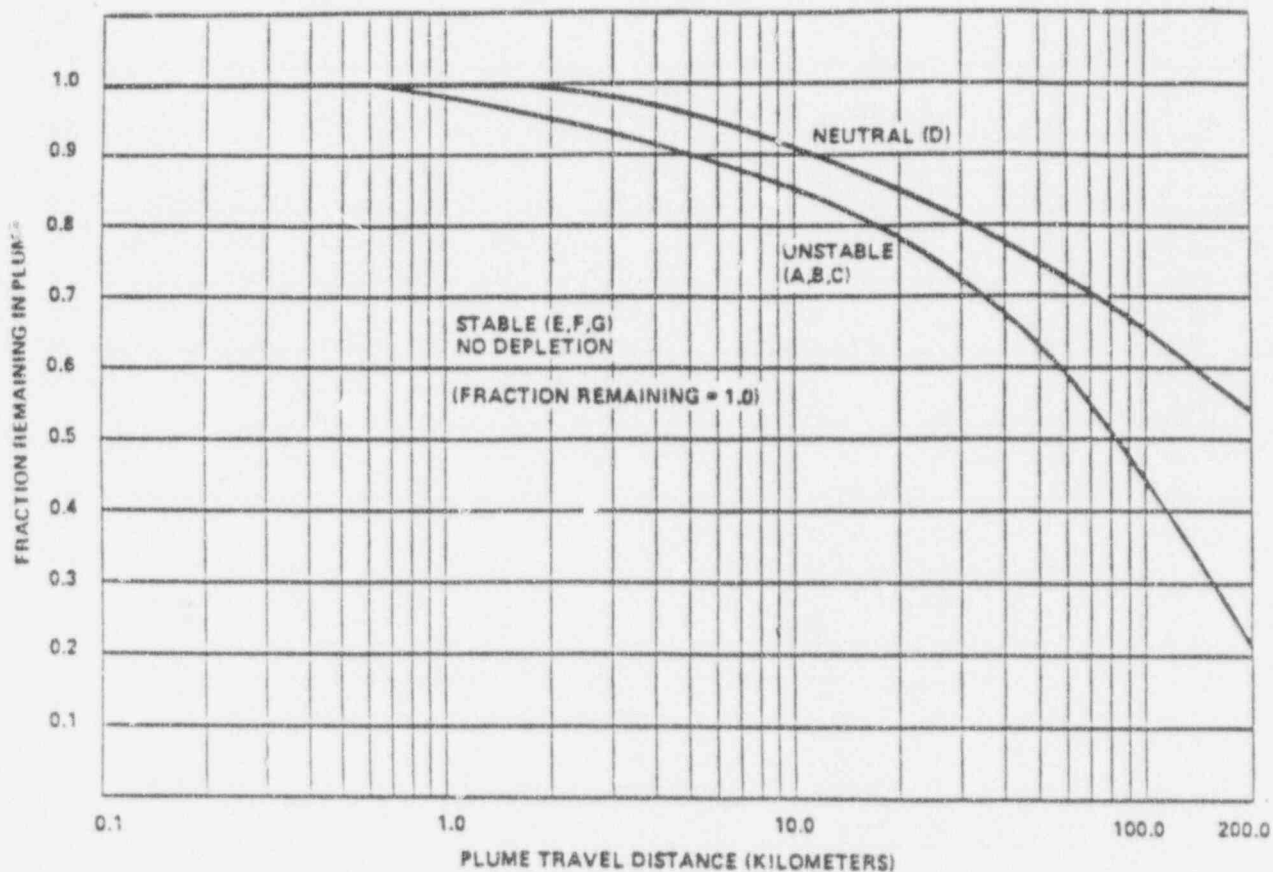


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RELATIVE DEPOSITION FOR GROUND-LEVEL  
RELEASES  
(All Atmospheric Stability Classes)

FIGURE 2.3-6



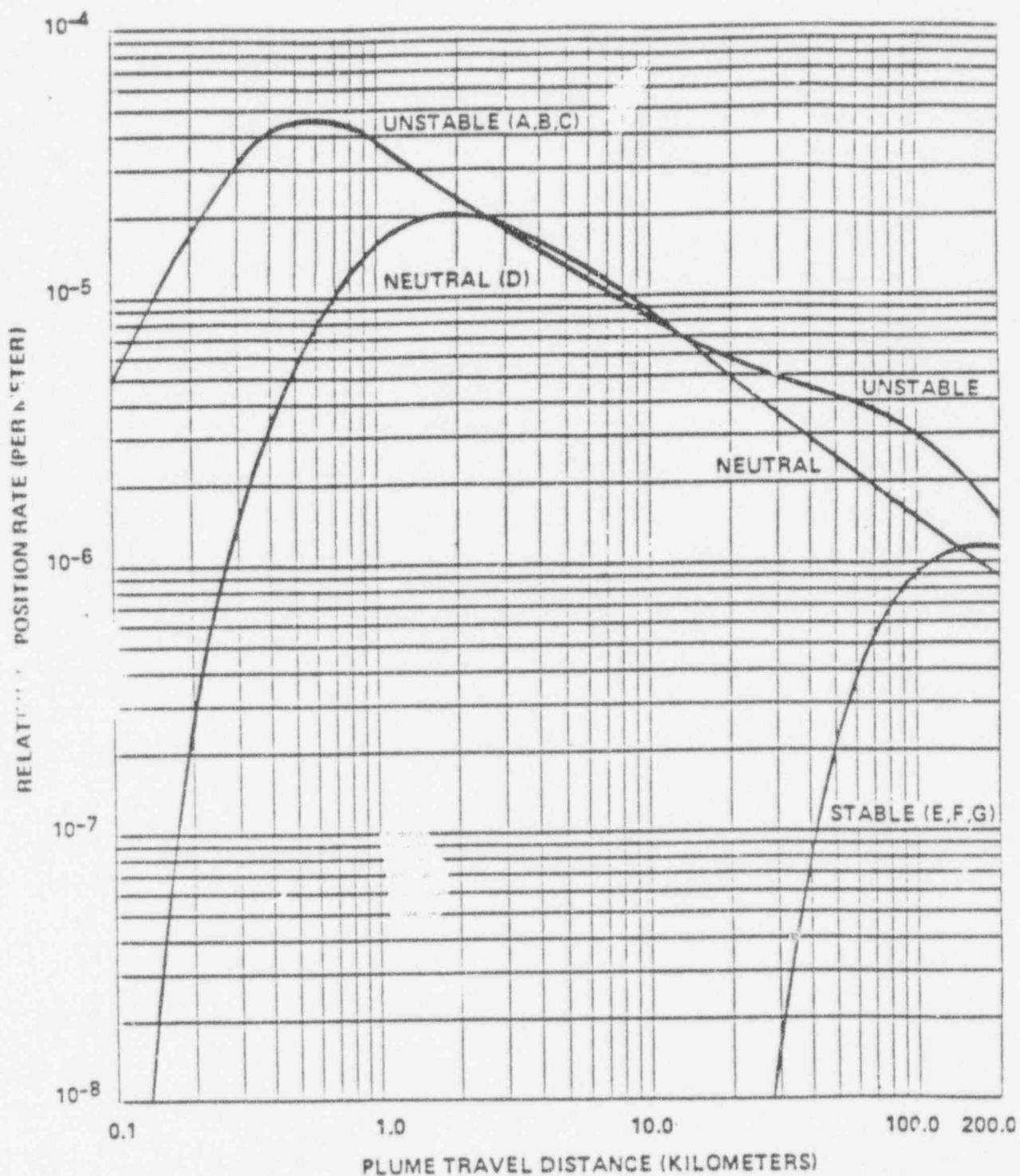
\*Reference 7.

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UNIT 1 AND UNIT 2

PLUME DEPLETION EFFECT FOR 100-METER RELEASES\*  
(Letters Denote Pasquill Stability Class)

FIGURE 2.3-5



\* Reference 7.

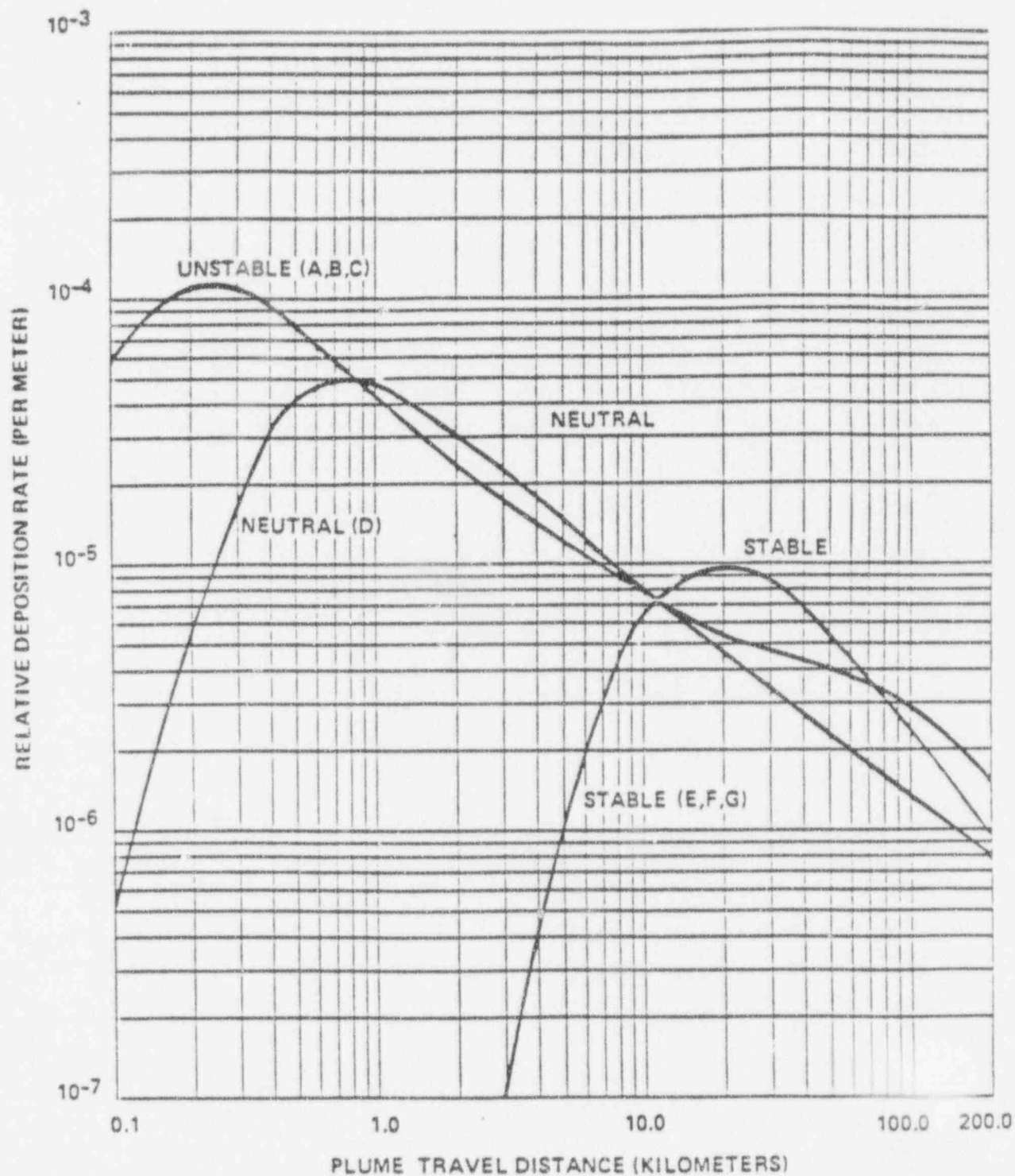
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UNIT 1 AND UNIT 2

RELATIVE DEPOSITION FOR 60-METER RELEASES\*  
(Letters Denote Pasquill Stability Class)

FIGURE 2.3-8



\*Reference 7.

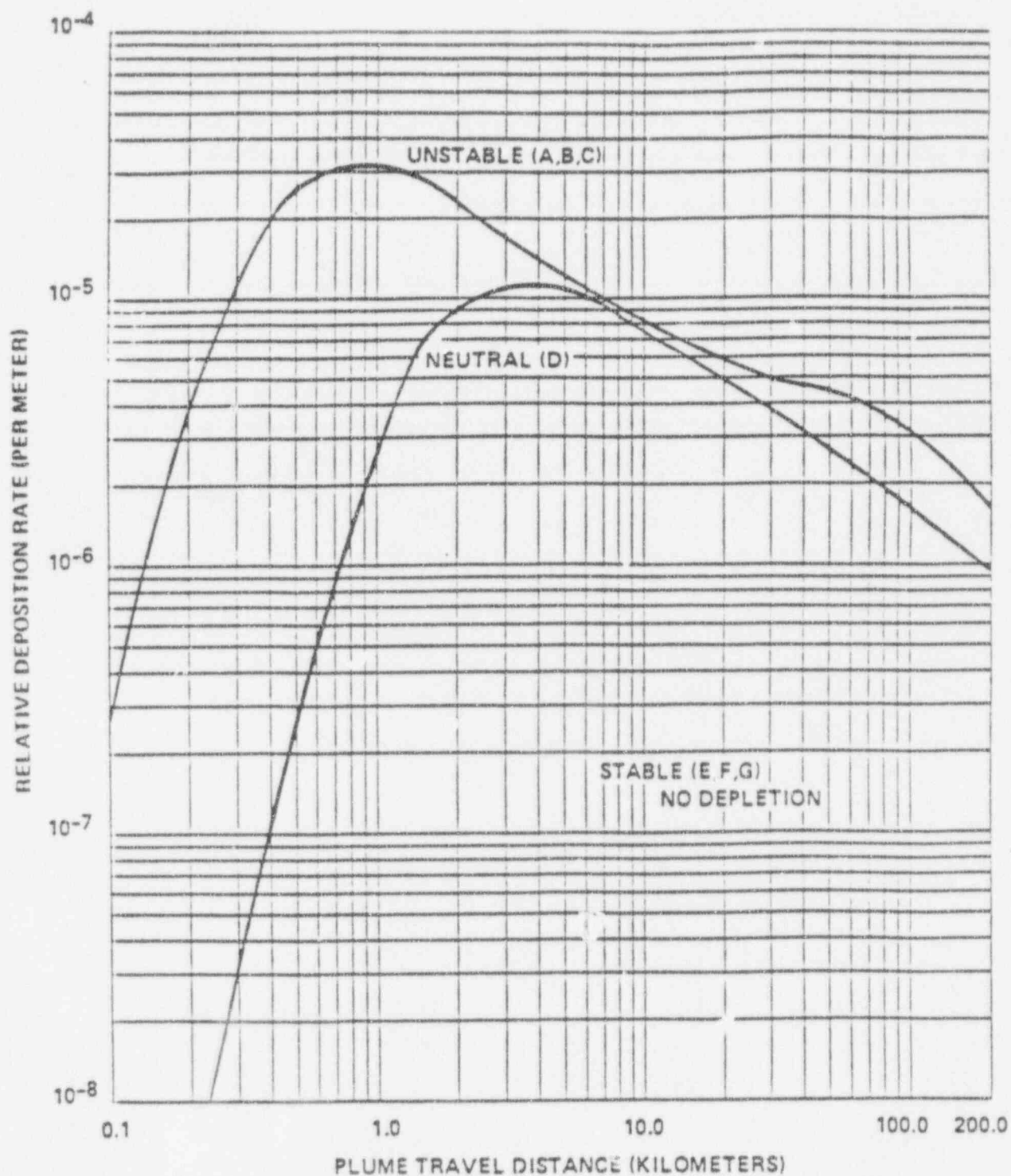
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ELECTRIC GENERATING PLANT  
UNIT 1 AND UNIT 2

RELATIVE DEPOSITION FOR 30-METER RELEASES\*  
(Letters Denote Pasquill Stability Class)

FIGURE 2.3-7





\*Reference 7.

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ELECTRIC GENERATING PLANT  
UNIT 1 AND UNIT 2

RELATIVE DEPOSITION FOR 100-METER RELEASES\*  
(Letters Denote Pasquill Stability Class)

FIGURE 2.3-9

## 2.4 DEFINITIONS OF GASEOUS EFFLUENT TERMS

<u>Term</u>	<u>Definition</u>	<u>Subsection of Initial Use</u>
AG	= the administrative allocation factor for gaseous effluent pathways (unitless).	2.1.1
BR	= the breathing rate for individual from table 2.2-10 (m <sup>3</sup> /year).	2.2.1.2
b	= the maximum height of the adjacent building (m).	2.3.1.1
C <sub>g</sub>	= Noble gas grab sample concentrations, taken in accordance with subsection 2.5.2, for the release under consideration.	2.1.1
c <sub>b</sub>	= the calculated basic monitor setpoint value. (μCi/cc)	2.1.1
c <sub>v</sub>	= the correction to effective release height due to low vent exit velocity (m).	2.3.1.2
D <sub>TB</sub>	= the limiting dose rate to the total body of an individual in an unrestricted area which is 500 mrem/year.	2.1.1
D <sub>t</sub>	= the total body dose rate at time of release (mrem/year).	2.2.1.1
D <sub>sk</sub>	= the limiting dose rate to the skin of an individual in an unrestricted area which is 3000 mrem/year.	2.1.1



<u>Term</u>	<u>Definition</u>	<u>Subsection of Initial Use</u>
$D_s$	= the skin dose rate at time of release (mrem/year).	2.2.1.1
$D_o$	= the organ dose rate at time of release (mrem/year).	2.2.1.2
$DF_{io}$	= the inhalation pathway dose factor for child age group for organ o and radionuclide i (mrem/pCi inhaled) from table 2.2-2.	2.2.1.2
$D_{\text{beta}}$	= the air dose due to beta emissions from noble gases (mrad).	2.2.2.1
$D_{\text{gamma}}$	= the air dose due to gamma emissions from noble gases (mrad).	2.2.2.1
$D_j$	= the dose to an organ of individual from radioiodines, tritium, and radionuclides in particulate form with half-lives greater than eight days (mrem).	2.2.2.2
$(DFA_{ij})_a$	= the inhalation dose factor for the ith radionuclide for the receptor in age group a (mrem/pCi) from tables 2.2-1 through 2.2-4.	2.2.2.2
$DFG_{ij}$	= the ground plane dose conversion factor for radionuclide i (same for all age groups) (mrem/h per pCi/m <sup>2</sup> ) from table 2.2-9.	2.2.2.2
$(DFL_{ij})_a$	= the organ ingestion dose factor for the ith radionuclide for the receptor in age group a (mrem/pCi) from tables 2.2-5 through 2.2-8.	2.2.2.2

<u>Term</u>	<u>Definition</u>	<u>Subsection of Initial Use</u>
$d_p$	= the plume depletion factor for all radionuclides other than noble gases at distance $x$ (unitless).	2.3.1.1
$d$	= the inside diameter of plant vent (m).	2.3.1.2
$D_g$	= the deposition rate for ground-level releases ( $m^{-1}$ ).	2.3.2.1
$D_e$	= the deposition rate for elevated releases ( $m^{-1}$ ).	2.3.2.1
$E$	= the fraction of release considered to be ground level (unitless).	2.3.2.1
$f_t$	= the fraction of the annual intake of fresh leafy vegetation grown locally (dimensionless).	2.2.2.2
$f_s$	= the fraction of annual intake of stored vegetation grown locally (dimensionless).	2.2.2.2
$f_p$	= the fraction of the year that the cow (or goat) is on pasture (dimensionless).	2.2.2.2
$f_a$	= the fraction of the cow (or goat) feed that is pasture grass while the cow or goat is on pasture (dimensionless).	2.2.2.2
$F_r$	= the stable element transfer coefficient for meat (days/kg) from table 2.2-11.	2.2.2.2
$F_m$	= the stable element transfer coefficient for milk (days/liter) from table 2.2-11.	2.2.2.2

<u>Term</u>	<u>Definition</u>	<u>Subsection of Initial Use</u>
$F_m'$	= the momentum flux parameter ( $m^4/s^2$ ).	2.3.1.2
$F_v$	= the maximum expected release flowrate through a particular release point (ml/s) from table 2.1-2.	2.1.1
$H$	= the absolute humidity of the atmosphere ( $gm/m^3$ ).	2.2.2.2
$h$	= the effective release height (m).	2.3.1.2
$h_v$	= the height of release point (m).	2.3.1.2
$h_t$	= the maximum terrain height between the release point and the point of interest (m).	2.3.1.2
$h_{pr}$	= the additional height due to plume rise (m).	2.3.1.2
$K_i$	= the total body dose factor due to gamma emissions from radionuclide i (mrem/year per $\mu Ci/m^3$ ) from table 2.1-1.	2.1.1
$K'$	= the constant of unit conversion ( $10^6 pCi/mCi$ ).	2.2.1.2
$K''$	= the constant of unit conversion (8760 h/year)	2.2.2.2
$K'''$	= the constant of unit conversion ( $10^3 gm/kg$ ).	2.2.2.2
$L_i$	= the skin dose factor due to beta emissions from radionuclide i (mrem/year per $\mu Ci/m^3$ ) from table 2.1-1.	2.1.1
$M_i$	= the air dose factor due to gamma emissions from radionuclide i (mrad/year per $\mu Ci/m^3$ ) from table 2.1-1.	2.1.1

<u>Term</u>	<u>Definition</u>	<u>Subsection of Initial Use</u>
$N_i$	= the air dose factor due to beta emissions from noble gas radionuclide $i$ (mrad/year per $\mu\text{Ci}/\text{m}^3$ ) from table 2.1-1.	2.2.2.1
$n$	= the number of simultaneous gaseous release pathways.	2.1.5
$n_{j,k}$	= the number of hours meteorological conditions are observed to be in a given wind direction, windspeed class $j$ , and atmospheric stability class $k$ .	2.3.1.1
$N$	= the total hours of valid meteorological data.	2.3.1.1
$P_{i0}$	= the dose parameter for radionuclide $i$ , (mrem/year per $\mu\text{Ci}/\text{m}^3$ ) for the inhalation pathway.	2.2.1.2
$Q_{i0}$	= the source term for ground-level release noble gas radionuclide $i$ ( $\mu\text{Ci}/\text{s}$ ).	2.1.1
$Q_{im}$	= the source term for mixed-mode release noble gas radionuclide $i$ ( $\mu\text{Ci}/\text{s}$ ).	2.1.1
$Q_{i0(r)}$	= the source term for ground-level release noble gas radionuclide $i$ from a specific release point ( $\mu\text{Ci}/\text{s}$ ).	2.1.5
$Q_{im(r)}$	= the source term for mixed-mode release noble gas radionuclide $i$ from a specific release point ( $\mu\text{Ci}/\text{s}$ ).	2.1.5

<u>Term</u>	<u>Definition</u>	<u>Subsection of Initial Use</u>
$Q'_{ig}$	= the source term for ground-level release radioiodine, tritium, and particulate radionuclide $i$ ( $\mu\text{Ci/s}$ ).	2.2.1.2
$Q'_{im}$	= the source term for mixed-mode release radioiodine, tritium, and particulate radionuclide $i$ ( $\mu\text{Ci/s}$ ).	2.2.1.2
$\bar{Q}_{ig}$	= the cumulative ground-level release of noble gas radionuclide $i$ ( $\mu\text{Ci}$ ).	2.2.2.1
$\bar{Q}_{im}$	= the cumulative mixed-mode release of noble gas radionuclide $i$ ( $\mu\text{Ci}$ ).	2.2.2.1
$\bar{Q}'_{ig}$	= the cumulative ground-level release of radioiodines, tritium, and particulate radionuclide $i$ ( $\mu\text{Ci}$ ).	2.2.2.2
$\bar{Q}'_{im}$	= the cumulative mixed-mode release of radioiodines, tritium, and particulate radionuclide $i$ ( $\mu\text{Ci}$ ).	2.2.2.2
$Q_f$	= the feed consumption rate for cow or goat (kg/day).	2.2.2.3
$q_i$	= the noble gas source term for the Gaseous Waste Processing System or Containment purge ( $\mu\text{Ci/s}$ ).	2.1.3.1 and 2.1.3.2
$R_i$	= the relationship between noble gas concentration and the dose rate to the total body for the conditions of the release under consideration (mrem/year/ $\mu\text{Ci/cc}$ ).	2.1.1

<u>Term</u>	<u>Definition</u>	<u>Subsection of Initial Use</u>
$R_s$	= the relationship between noble gas concentration and the dose rate to the skin for the conditions of the release under consideration (mrem/year/ $\mu$ Ci/cc).	2.1.1
$r_t$	= the relationship between noble gas concentration and the dose rate to the total body for Gaseous Waste Processing System or containment purge release for the conditions of the release under consideration (mrem/year/ $\mu$ Ci/cc).	2.1.3.1
$r_s$	= the relationship between noble gas concentration and the dose rate to the skin for Gaseous Waste Processing System or containment purge release for the conditions of the release under consideration.	2.1.3.1
$R_{a,p,j}$	= the pathway-specific, individual age-specific, organ dose factor for radionuclide i, pathway p, organ j, and age group a, (mrem/year per $\mu$ Ci/m <sup>3</sup> ) or (m <sup>2</sup> -mrem/year/ $\mu$ Ci/s).	2.2.2.2
$r$	= the fraction of deposited radionuclide retained on vegetation (unitless).	2.2.2.2
SF	= the safety factor used to introduce a margin of conservatism into setpoint calculations.	2.1.1
SHF	= the shielding factor afforded by structure (unitless).	2.2.2.2
S	= the stability parameter (s <sup>-2</sup> ).	2.3.1.2

<u>Term</u>	<u>Definition</u>	<u>Subsection of Initial Use</u>
$t$	= the exposure time for radioactivity deposited on ground (s).	2.2.2.2
$t_{\ell}$	= the time between harvest of leafy vegetation and consumption (s).	2.2.2.2
$t_{hv}$	= the time between harvest of stored vegetation and consumption (s).	2.2.2.2
$t_{fm}$	= the transport time from feed to receptor for stored feed (s).	2.2.2.3
$t_r$	= the transport time from feed to receptor for pasture grass (s).	2.2.2.3
$U_{a,n}$	= the receptor's milk (liters/year) or meat (kg/year) consumption rate for age group a from table 2.2-10.	2.2.2.3
$U_{a,s}$	= the consumption rate of stored vegetation by the receptor in age group a (kg/year) from table 2.2-10.	2.2.2.2
$U_{a,\ell}$	= the consumption rate of fresh leafy vegetation by the receptor in age group a (kg/year) from table 2.2-10.	2.2.2.2
$u_{j,k}$	= the wind speed (midpoint of wind speed class j) at ground level during atmospheric stability class k (m/s).	2.3.1.1
$U_{j,k}$	= the wind speed (midpoint of wind speed class j) at the height of release, h, of an elevated release during atmospheric stability class k (m/s).	2.3.1.2

<u>Term</u>	<u>Definition</u>	<u>Subsection of Initial Use</u>
$W'_{GP}$	= the pathway-dependent relative dispersion or deposition for ground-level releases at the location of the critical receptor.	2.2.2.2
$W'_{MF}$	= the pathway-dependent relative dispersion or deposition for mixed-mode releases at the location of the critical receptor.	2.2.2.2
$x$	= the distance from release point-to-point of interest (meters).	2.3.1.1
$Y_v$	= the vegetation areal density ( $\text{kg}/\text{m}^2$ ).	2.2.2.2
$Y_p$	= the agricultural productivity by unit area of pasture feed grass ( $\text{kg}/\text{m}^2$ ).	2.2.2.3
$Y_s$	= the agricultural productivity by unit area of stored feed ( $\text{kg}/\text{m}^2$ ).	2.2.2.3
$(\overline{X/Q})_G$	= the highest annual average relative concentration for a ground-level release type ( $\text{s}/\text{m}^3$ ).	2.1.1
$(\overline{X/Q})_M$	= the highest annual average relative concentration for a mixed-mode release type ( $\text{s}/\text{m}^3$ ).	2.1.2
$(\overline{X/Q'})_{GP}$	= the annual average relative concentration for location of controlling (critical) receptor for inhalation and all tritium pathways for a ground-level release type ( $\text{s}/\text{m}^3$ ).	2.2.2.2



<u>Term</u>	<u>Definition</u>	<u>Subsection of Initial Use</u>
$(\overline{X/Q'})_{\text{ref}}$	= the annual average relative concentration for location of controlling (critical) receptor for inhalation and all tritium pathways for a mixed-mode release type ( $\text{s/m}^3$ ).	2.2.2.2
$\lambda_i$	= the decay constant for the $i$ th radionuclide ( $\text{s}^{-1}$ ).	2.2.2.2
$\lambda_w$	= the decay constant for removal of activity on leaf and plant surfaces by weathering ( $\text{s}^{-1}$ ).	2.2.2.2
$\Sigma_{zk}$	= the vertical standard deviation of the plume concentration distribution considering the initial dispersion within the building wake (m).	2.3.1.1
$\sigma_{zk}$	= the vertical standard deviation of the plume (m), for a given distance under the stability category $k$ , indicated by $\Delta T/\Delta Z$ from figure 2.3-1.	2.3.1.1
$\Delta T/\Delta Z$	= the vertical temperature gradient used to determine the atmospheric stability category ( $^{\circ}\text{C}/100 \text{ m}$ or $^{\circ}\text{F}/100 \text{ ft}$ ).	2.3.1.1
$X_{i,v}$	= the concentration of radionuclide $i$ for the particular vent release pathway under consideration ( $\mu\text{Ci/ml}$ )	2.1.1

## 2.5 LIMITS OF OPERATION

### 2.5.1 Gaseous Effluents Dose Rate

The dose rate due to radioactive materials released in gaseous effluents, from the site to areas at and beyond the site boundary (see figures 3.0-1 and 3.0-2) shall be limited to less than or equal to 500 mrem/yr to the whole body for noble gases and less than or equal to 3000 mrem/yr to the skin for noble gases. The dose rate shall also be limited to 1500 mrem/yr to any organ for Iodine-131, for Iodine-133, for tritium, and for all radionuclides in particulate form with half-lives greater than 8 days. This limit applies at all times for all modes of operation.

#### 2.5.1.1 Exceeding Gaseous Effluent Dose Rate Limits

If the dose rates exceed the above limits, immediately restore the release rate to within the above limits. These limits do not affect mode changes.

#### 2.5.1.2 Sampling and Analysis of Gaseous Effluents

The dose rate due to Iodine-131, Iodine-133, tritium, noble gases, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents shall be determined to be within the above limits by obtaining representative samples and performing analyses in accordance with the following sampling and analysis program.

##### 2.5.1.2.1 Waste Gas Decay Tank Releases

For waste gas decay tanks, each tank must be sampled prior to its release and analyzed for noble gas principal gamma emitters to a LLD of  $1.00\text{E-}4 \mu\text{Ci/ml}$ .

#### 2.5.1.2.2 Containment Purge (Venting)

For containment purges (14 in. mini or 24 in. maxi) the containment atmosphere must be sampled prior to each purge and analyzed for noble gas principal gamma emitters to a LLD of  $1.00\text{E-}4 \mu\text{Ci/ml}$ . In addition, a monthly tritium from a containment purge must be sampled and analyzed to an LLD of  $1.00\text{E-}6 \mu\text{Ci/ml}$ .

If a purge is in progress and a shutdown, startup, or a thermal power change exceeding 15 percent of rated thermal power within a one-hour period occurs, then a grab sample of the containment atmosphere must be taken and analyzed for noble gas principal gamma emitters to an LLD of  $1.00\text{E-}4 \mu\text{Ci/ml}$ . However, this requirement does not apply if (1) analysis shows that the dose equivalent I-131 concentration in the primary coolant has not increased more than a factor of 3 and (2) the noble gas monitor shows that effluent activity has not increased more than a factor of 3.

#### 2.5.1.2.3 Plant Vent Continuous Release

For the plant vent a grab sample must be taken monthly and analyzed monthly for (1) noble gas principal gamma emitters to an LLD of  $1.00\text{E-}4 \mu\text{Ci/ml}$  and (2) tritium to an LLD of  $1.00\text{E-}6 \mu\text{Ci/ml}$ . Tritium samples are also required at least once per 24 hours for the Unit 1 plant vent, when the refueling canal is flooded. This does not apply to the Unit 2 plant vent, because the refueling floor ventilation exits the Unit 1 plant vent only. Also for Unit 1 plant vent only, tritium samples are required at least once per 7 days whenever spent fuel is in the spent fuel pool.

If a shutdown, startup, or a thermal power change exceeding 15 percent of rated thermal power within a 1-hour period occurs, then a grab sample of the plant vent for the affected unit must be taken and analyzed for noble gas principal gamma emitters to an LLD of  $1.00\text{E-}4 \mu\text{Ci/ml}$  and for tritium to an LLD of  $1.00\text{E-}6 \mu\text{Ci/ml}$ . However, this requirement does not apply if (1) analysis shows that the dose equivalent I-131 concentration in the primary coolant has not increased more than a factor of 3 and (2) the plant vent noble gas monitor shows that effluent activity has not increased more than a factor of 3.

In addition to the monthly sampling, a continuous composite charcoal sample must be taken and analyzed weekly for I-131 to an LLD of  $1.00\text{E-}12 \mu\text{Ci/ml}$ . A continuous composite particulate sample must be taken and analyzed (1) weekly for principal gamma emitters to an LLD of  $1.00\text{E-}11 \mu\text{Ci/ml}$ , (2) monthly for gross alpha contamination to an LLD of  $1.00\text{E-}11 \mu\text{Ci/ml}$ , and (3) quarterly for Sr-89 and SR-90 to an LLD of  $1.00\text{E-}11 \mu\text{Ci/ml}$ .

For both the charcoal and the particulate composite, the ratio of the sample flowrate to the sampled stream flowrate shall be known for the time period covered by each dose or dose rate calculation made in accordance with subsections 2.5.1, 2.5.2, and 2.5.3 of this ODCM. In addition, the charcoal and particulate samples shall be changed at least once per 7 days, and analyses shall be completed within 48 hours after removal from sampler. Sampling shall also be performed at least once per 24 hours for at least 7 days following each shutdown, startup, or thermal power change exceeding 15 percent of rated thermal power within a 1-hour period, and analyses shall be completed within 48 hours of changing. When samples collected for 24 hours are analyzed, the corresponding LLDs may be increased by a factor of 10. The requirement does not apply if (1) analysis shows that the dose equivalent I-131 concentration in the reactor coolant has not increased more than a factor of 3 and (2) the plant vent noble gas monitor shows that effluent activity has not increased more than a factor of 3.

#### 2.5.1.2.4 Turbine Building Vent Exhaust

The turbine building vent is the release point for the condenser air ejector and steam packing exhaust. All sampling and analyses may be omitted for this vent, provided the absence of a primary to secondary leak has been demonstrated. that is, the gamma activity in the secondary water does not exceed background by more than 20 percent.

If required, a grab sample must be taken monthly and analyzed monthly for (1) noble gas principal gamma emitters to an LLD of  $1.00\text{E-}4 \mu\text{Ci/ml}$  and (2) tritium to an LLD of  $1.00\text{E-}6 \mu\text{Ci/ml}$ .

In addition to the monthly sampling, a continuous composite charcoal sample must be taken and analyzed weekly for I-131 to an LLD of  $1.00\text{E-}12 \mu\text{Ci/ml}$ . A continuous composite particulate sample must be taken and analyzed (1) weekly for principal gamma emitters to an LLD of  $1.00\text{E-}11 \mu\text{Ci/ml}$ , (2) monthly for gross alpha contamination to an LLD of  $1.00\text{E-}11 \mu\text{Ci/ml}$ , and (3) quarterly for Sr-89 and Sr-90 to an LLD of  $1.00\text{E-}11 \mu\text{Ci/ml}$ .

For both the charcoal and the particulate composite, the ratio of the sample flowrate to the sampled stream flowrate shall be known for the time period covered by each dose or dose rate calculation made in accordance with subsections 2.5.1, 2.5.2, and 2.5.3 of this ODCM. In addition, the charcoal and particulate samples shall be changed at least once per 7 days, and analyses shall be completed within 48 hours after changing. Sampling shall also be performed at least once per 24 hours for at least 7 days following each shutdown, startup, or thermal power change exceeding 15 percent of rated thermal power within a 1-hour period and analyses shall be completed within 48 hours of changing. When samples collected for 24 hours are analyzed, the corresponding LLD's may be increased by a factor of 10. This requirement does not apply if (1) analysis shows that the dose equivalent I-131 concentration in the reactor coolant has not increased more than a factor of 3 and (2) the turbine building vent noble gas monitor shows that effluent activity has not increased more than a factor of 3.

#### 2.5.1.2.5 Basis for the Gaseous Effluent Dose Rate

This specification is provided to ensure that the dose at any time at and beyond the site boundary from gaseous effluents from all units on the site will be within the annual dose limits of 10 CFR 20 to unrestricted areas. The annual dose limits are the doses associated with the concentrations of 10 CFR 20, Appendix B, table II, column I. These limits provide reasonable assurance that radioactive material discharged in gaseous effluents will not result in the exposure of a member of the public in an unrestricted area, either within or outside the site boundary, to annual average concentrations exceeding the limits specified in Appendix B, table II, of 10 CFR 20 (10 CFR 20.106(b)). For members of the public who may at times be within the site boundary, the

occupancy of that member of the public will usually be sufficiently low to compensate for any increase in the atmospheric diffusion factor above that for the site boundary. Examples of calculations for such members of the public, with the appropriate occupancy factors, shall be given in the ODCM. The specified release rate limits restrict, at all times, the corresponding gamma and beta dose rates above background to a member of the public at or beyond the site boundary to less than or equal to 500 mrem/yr to the whole body or to less than or equal to 3000 mrem/yr to the skin. These release-rate limits also restrict, at all times, the corresponding thyroid dose rate above background to a child, via the inhalation pathway, to less than or equal to 1500 mrem/yr.

This specification applies to the release of radioactive materials in gaseous effluents from all units at the site.

The required detection capabilities for radioactive materials in gaseous waste samples are tabulated in terms of the LLDs. Detailed discussion of the LLD and other detection limits can be found in HASL Procedures Manual, HASL-300 (revised annually), L. A., Currie, "Limits for Qualitative Detection and Quantitative Determination - Application to Radio-chemistry," Anal. Chem. 40, 586-93, 1968, and J. K. Hartwell, "Detection Limits for Radioanalytical Counting Techniques," Atlantic Richfield Hanford Company Report ARH-SA-215, June 1975.

#### 2.5.2 Noble Gas Air Dose

The air dose due to noble gases released in gaseous effluents, from each unit to areas at and beyond the site boundary (figures 3.0-1 and 3.0-2), shall be limited, during any calendar quarter, to less than or equal to 5 mrad for gamma radiation and to less than or equal to 10 mrad for beta radiation. In addition, during any calendar year, it shall be limited to less than or equal to 10 mrad for gamma radiation and to less than or equal to 20 mrad for beta radiation. This limit applies at all times for all modes of operation.



#### 2.5.2.1 Exceeding the Noble Gas Air Doses

If the calculated air dose from radioactive noble gases in gaseous effluents exceeds any of the above limits, prepare and submit to the NRC within 30 days, pursuant to Technical Specification 6.8.2, a special report that identifies the cause(s) for exceeding the limit(s) and defines the corrective action(s) that have been taken to reduce the releases and the proposed corrective actions to be taken to assure that subsequent releases will be in compliance with the above limits. These limits do not affect mode changes.

#### 2.5.2.2 Cumulative Dose Calculation Requirements

To assure these limits are not exceeded, the cumulative dose contributions for the current calendar quarter and current calendar year for noble gases shall be determined in accordance with the methodology and parameters of subsection 2.2.2 at least once per 31 days.

#### 2.5.2.3 Basis for Noble Gas Air Doses

This specification is provided to implement the requirements of sections II.B, III.A, and IV.A of Appendix I, 10 CFR 50. The limiting condition for operation implements the guides set forth in section I.B of Appendix I. The action statements provide the required operating flexibility and at the same time implement the guides set forth in section IV.A of Appendix I to assure that the releases of radioactive material in gaseous effluents to unrestricted areas will be kept "as low as is reasonably achievable." The surveillance requirements implement the requirements in section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data such that the actual exposure of a member of the public through appropriate pathways is unlikely to be substantially underestimated. The dose calculation methodology and parameters established in the ODCM for calculating the doses due to the actual release rates of radioactive noble gases in gaseous effluents are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses

to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water Cooled Reactors," Revision 1, July 1977. The ODCM equations provided for determining the air doses at and beyond the site boundary are based upon the historical average atmospheric conditions.

### 2.5.3 Dose to A Member of the Public from Tritium, Radioiodines, and Particulates

The dose to a member of the public from I-131, I-133, tritium, and all radionuclides in particulate form with half-lives greater than 8 days in gaseous effluents released, from each unit to areas at and beyond the site boundary (figures 3.0-1 and 3.0-2), shall be limited during any calendar quarter to less than or equal to 7.5 mrem to any organ and during any calendar year to less than or equal to 15 mrem to any organ. This limit applies at all times for all modes of operation.

#### 2.5.3.1 Exceeding the Dose Limits

If the calculated dose from the release of I-131, I-133, tritium, and radionuclides in particulate form with half-lives greater than 8 days, in gaseous effluents exceeding any of the above limits, prepare and submit to the NRC within 30 days, pursuant to Technical Specification 6.8.2, a special report that identifies the cause(s) for exceeding the limit(s) and defines the corrective action(s) that have to be taken to assure that subsequent releases will be in compliance with the above limits. These limits do not affect modes changes.



#### 2.5.3.2 Cumulative Dose Calculation Requirements

To assure these limits are not exceeded, the cumulative dose contributions for the current calendar quarter and current calendar year for I-131, I-133, tritium, and radionuclides in particulate form with half-lives greater than 8 days shall be determined in accordance with the methodology and parameters in subsection 2.2.2 at least once per 31 days.

#### 2.5.3.3 Basis for Tritium, Radioiodine, and Particulate Organ Dose Limits

This specification is provided to implement the requirements of sections II.C, III.A, and IV.A of Appendix I, 10 CFR 50. The limiting conditions for operation are the guides set forth in section II.C of Appendix I. The action statements provide the required operating flexibility and at the same time implement the guides set forth in section IV.A of Appendix I to assure that the releases of radioactive materials in gaseous effluents to unrestricted areas will be kept "as low as is reasonably achievable." The ODCM calculational methods specified in the surveillance requirements implement the requirements in section III.A of Appendix I that conformance with the guides of Appendix I be shown by calculational procedures based on models and data, such that the actual exposure of a member of the public through appropriate pathways is unlikely to be substantially underestimated. The ODCM calculational methodology and parameters for calculating the doses due to the actual release rates of the subject materials are consistent with the methodology provided in Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR 50, Appendix I," Revision 1, October 1977 and Regulatory Guide 1.111, "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors," Revision 1, July 1977. These equations also provide for determining the actual doses based upon the historical average atmospheric conditions. The release rate specifications for Iodine-131 Iodine-133, tritium, and radionuclides in particulate form with half-lives greater than 8 days are dependent upon the existing radionuclide pathways to man in the areas

at and beyond the site boundary. The pathways that were examined in the development of the calculations were (1) individual inhalation of airborne radionuclides, (2) deposition of radionuclides onto green leafy vegetation with subsequent consumption by man, (3) deposition onto grassy areas where milk animals and meat-producing animals graze, with consumption of the milk and meat by man, and (4) deposition on the ground with subsequent exposure of man.

#### 2.5.4 Gaseous Radwaste Treatment System Operation

The ventilation exhaust treatment system and the gaseous waste processing system shall be operable, and appropriate portions of these systems shall be used to reduce releases of radioactivity when the projected doses in 31 days due to gaseous effluent releases, from each unit to areas at and beyond the site boundary (figures 3.0-1 and 3.0-2), would exceed  $2.00E-1$  mrad to air from gamma radiation,  $4.00E-1$  mrad to air from beta radiation, or  $3.00E-1$  mrem to any organ of a member of the public. This limit applies at all times for all modes of operation.

##### 2.5.4.1 Exceeding the Dose Limits for Operation of the Gaseous Waste Processing System

If radioactive gaseous waste is being discharged without treatment and is in excess of the above limits, prepare and submit to the NRC within 30 days, pursuant to Technical Specification 6.8.2, a special report that includes the following information (1) identification of any inoperable equipment or subsystems and the reason for the inoperability, (2) action(s) taken to restore the inoperable equipment to operable status, and (3) summary description of action(s) taken to prevent a recurrence. These limits do not affect mode changes.

#### 2.5.4.2 Requirement for Projecting Doses Due to Gaseous Releases

To assure these limits are not exceeded, doses due to gaseous releases, from each unit to areas at and beyond the site boundary, shall be projected at least once per 31 days in accordance with the methodology and parameters in subsection 2.2.2, when gaseous radwaste treatment systems are not being fully utilized.

#### 2.5.4.3 Basis for the Operability of the Gaseous Waste Processing System

The operability of the gaseous waste processing system and the ventilation exhaust treatment system ensures that the systems will be available for use whenever gaseous effluents require treatment prior to release to the environment. The requirement that the appropriate portions of these systems be used, when specified, provides reasonable assurance that the releases of radioactive materials in gaseous effluents will be kept "as low as is reasonably achievable." This specification implements the requirements of 10 CFR 50.36a, General Design Criterion 60, Appendix A, 10 CFR 50, and the design objectives given in section II.D, Appendix I, 10 CFR 50. The specified limits governing the use of appropriate portions of the systems were specified as a suitable fraction of the dose design objectives set forth in sections II.B and II.C, Appendix I, 10 CFR 50, for gaseous effluents.

#### 2.5.5 Dose to the Public from the Uranium Fuel Cycle

The annual (calendar year) dose or dose commitment to any member of the public due to releases of radioactivity and to radiation from uranium fuel cycle sources shall be limited to less than or equal to 25 mrem to the whole body or any organ, except the thyroid, which shall be limited to less than or equal to 75 mrem. This limit applies at all times for all modes of operation.

#### 2.5.5.1 Cumulative Dose Calculation Requirements

To assure these limits are not exceeded, cumulative dose contributions from direct radiation from each unit (including outside storage tanks, etc.) shall be determined in accordance with the methodology and parameters in section 4.0. This requirement is applicable only if the conditions in paragraph 2.5.5.2 are met.

#### 2.5.5.2 Conditions Required for Cumulative Dose Calculations

If the calculated doses from the release of radioactive materials in liquid or gaseous effluents exceeds twice the limits of subsections 1.5.2, 2.5.2, or 2.5.3, calculations shall be made including direct radiation contributions from the unit (including outside storage tanks etc.) to determine whether the above limits have been exceeded. If such is the case, prepare and submit to the NRC within 30 days, pursuant to Technical Specification 6.8.2, a special report that (1) defines the corrective action(s) to be taken to reduce subsequent releases to prevent recurrence of exceeding the above limits and (2) includes the schedule for achieving conformance with the above limits. This special report, as defined in 10 CFR 20.405(c), shall include an analysis that estimates the radiation exposure (dose) to a member of the public from uranium fuel cycle sources, including all effluent pathways and direct radiation, for the calendar year that includes the release(s) covered by this report. It shall also describe levels of radiation and concentrations of radioactive material involved, and the cause of the exposure levels or concentrations. If the estimated dose(s) exceeds the above limits and if the release condition resulting in violation of 40 CFR 190 has not already been corrected, the special report shall include a request for a variance in accordance with the provisions of 40 CFR 190. Submittal of the report is considered a timely request, and a variance is granted until staff action on the request is complete. These limits do not affect mode changes.

### 2.5.5.3 Exceeding the Dose Limitations

This specification is provided to meet the dose limitations of 40 CFR 190 that have been incorporated into 10 CFR 20 by 46 FR 18525. The specification requires the preparation and submittal of a special report whenever the calculated doses due to releases of radioactivity and to radiation from uranium fuel cycle sources that directly support the production of electrical power for public use exceed 25 mrem to the whole body or any organ, except the thyroid, which shall be limited to less than or equal to 75 mrem. For sites containing up to four reactors, it is highly unlikely that the resultant dose to a member of the public will exceed the dose limits of 40 CFR 190 if the individual reactors remain within twice the dose design objectives of Appendix I and if direct radiation doses from the units including outside storage tanks, etc. are kept small. The special report will describe a course of action that should result in the limitation of the annual dose to a member of the public to within the 40 CFR 190 limits. For the purposes of the special report, it may be assumed that the dose commitment to the member of the public from other uranium fuel cycle sources is negligible, with the exception that dose contributions from other nuclear fuel cycle facilities at the same site or within a radius of 8 km must be considered. If the dose to any member of the public is estimated to exceed the requirements of 40 CFR 190, the special report with a request for a variance (provided the release conditions resulting in violation of 40 CFR 190 have not already been corrected), in accordance with the provisions of 40 CFR 190.11 and 10 CFR 20.405c, is considered to be a timely request and fulfills the requirements of 40 CFR 190 until NRC staff action is completed. The variance only relates to the limits of 40 CFR 190 and does not apply in any way to the other requirements of dose limitation of 10 CFR 20, as addressed in subsections 1.5.1 and 2.5.1. An individual is not considered a member of the public during any period in which he/she is engaged in carrying out any operation that is part of the nuclear fuel cycle.

#### 2.5.6 Gaseous Effluent Monitoring Instrumentation

The radioactive gaseous effluent monitoring instrumentation channels specified below shall be operable with their alarm/trip setpoints set to ensure that the limits of subsection 2.5.1 of this ODCM are not exceeded. This limit applies at all times for all modes of operation.

##### 2.5.6.1 Nonconservative Gaseous Effluent Monitor Setpoint and ACTIONS

If a radioactive gaseous effluent monitoring instrumentation channel alarm/trip setpoint is less conservative than required by the above specification, immediately suspend the release of radioactive gaseous effluents monitored by the affected channel, or declare the channel inoperable.

If less than the minimum number of radioactive gaseous effluent monitoring instrumentation channels are operable, take the ACTION shown in table 2.5-1. Restore the inoperable instrumentation to operable status within 30 days, or, if unsuccessful, explain in the next Semiannual Radioactive Effluent Release Report, pursuant to Technical Specification 6.8.1.4, why this inoperability was not corrected in a timely manner. These limits do not affect mode changes.

##### 2.5.6.2 Gaseous Effluent Monitor Operability

Each radioactive gaseous effluent monitoring instrumentation channel shall be demonstrated operable by performance of the channel check, source check, channel calibration, and analog channel operational test at the frequencies shown in table 2.5-1. Specific instrument numbers are provided in parentheses for information only. The numbers apply to each unit. These numbers will help to identify associated channels or loops and are not intended to limit the requirements to the specific instruments associated with the number.



TABLE 2.5-1 (SHEET 1 OF 4)

## GASEOUS EFFLUENT MONITORING INSTRUMENTATION

	<u>Minimum Channels Operable</u>	<u>ACTION</u>	<u>Channel Check</u>	<u>Source Check</u>	<u>Channel Calibration</u>	<u>Analog Channel Operational Test</u>	<u>Modes Required</u>
1. Gas waste processing system							
a) Noble gas activity monitor - providing alarm and automatic termination of release (ARE-0014) (common)	1	45	P	P	R(3)	Q(1)	*
b) Effluent system flowrate measuring Device (AFT-0014) (common)	1	45	P	NA	R	NA	*
2. Turbine building vent							
a) Noble gas activity monitor (RE-12839C)	1	47	D	M	R(3)	Q(2)	*
b) Iodine and particulate samplers (RE-12839A&B)	1	51	W(6)	NA	NA	NA	*
c) Flowrate monitor (FT-12839 or FIS-12862)**	1	46	D	NA	R	NA	*
d) Sampler flowrate monitor (FI-13211)	1	46	D	NA	R	Q	*
3. Plant vent							
a) Noble gas activity monitor (RE-12442C or RE-12444C)	1	47,48	D	M	R(3)	Q(2)	at all times
b) Iodine sampler/monitor (RE-12442B or RE-12444B)	1	51	W(6)	NA	NA/R	NA	at all times
c) Particulate sampler/monitor (RE-12442A or RE-12444A)	1	51	D	NA	NA/R	Q	at all times
d) Flowrate monitor (FT-12442 or 12835)	1	46	D	NA	R	NA	at all times
e) Sampler flowrate monitor (FI-12442 or FI-12444)	1	46	D	NA	R	Q	at all times

TABLE 2.5-1 (SHEET 2 OF 4)

TABLE NOTATIONS

- 1) The analog channel operational test shall also demonstrate that automatic isolation of this pathway (for item a. below only) and control room alarm annunciation occurs if any of the following conditions exists:
  - a. Instrument indicates measured levels above the alarm/trip setpoint.
  - b. Circuit failure.
  - c. Instrument indicates a downscale failure.
  - d. Instrument controls not set in operate mode. (Annunciation via computer printout.)
- (2) The analog channel operational test shall also demonstrate that control room alarm annunciation occurs if any of the following conditions exists:
  - a. Instrument indicates measured levels above the alarm setpoint.
  - b. Circuit failure.
  - c. Instrument indicates a / e failure.
  - d. Instrument controls not e perate mode. (Annunciation via computer printout.)
- (3) The initial channel calibration shall be performed using one or more of the reference standards certified by the National Bureau of Standards (NBS) or using standards that have been obtained from suppliers that participate in measurement assurance activities with NBS. These standards shall permit calibrating the system over its intended range of energy and measurement range. For subsequent channel calibration, sources that have been related to the initial calibration shall be used.
- (4) Not used.
- (5) Not used.
- (6) The channel check shall consist of visually verifying that the collection device (i.e., particulate filter or charcoal cartridge, etc.) is in place for sampling.



TABLE 2.5-1 (SHEET 3 OF 4)

ACTION STATEMENTS

ACTION 41-44 (Not Used)

ACTION 45 - With the number of channels operable less than required by the minimum channels operable requirement, the contents of the tank(s) may be released to the environment provided that prior to initiating the release

- a. at least two independent samples of the tank's contents are analyzed
- b. at least two technically qualified members of the facility staff independently verify the release rate calculations and discharge valve lineup.

Otherwise, suspend release of radioactive effluents via this pathway.

ACTION 46 - With the number of channels operable less than required by the minimum channels operable requirement, effluent releases via this pathway may continue provided the flowrate is estimated at least once per 4 hours.

ACTION 47 - With the number of channels operable less than required by the minimum channels operable requirement, effluent releases via this pathway may continue provided grab samples are taken at least once per 12 hours and these samples are analyzed for radioactivity within 24 hours.

ACTION 48 - With the number of channels operable less than required by the minimum channels operable requirement, immediately suspend containment purging of radioactive effluents via this pathway.

ACTION 49 - Not Used.

TABLE 2.5-1 (SHEET 4 OF 4)

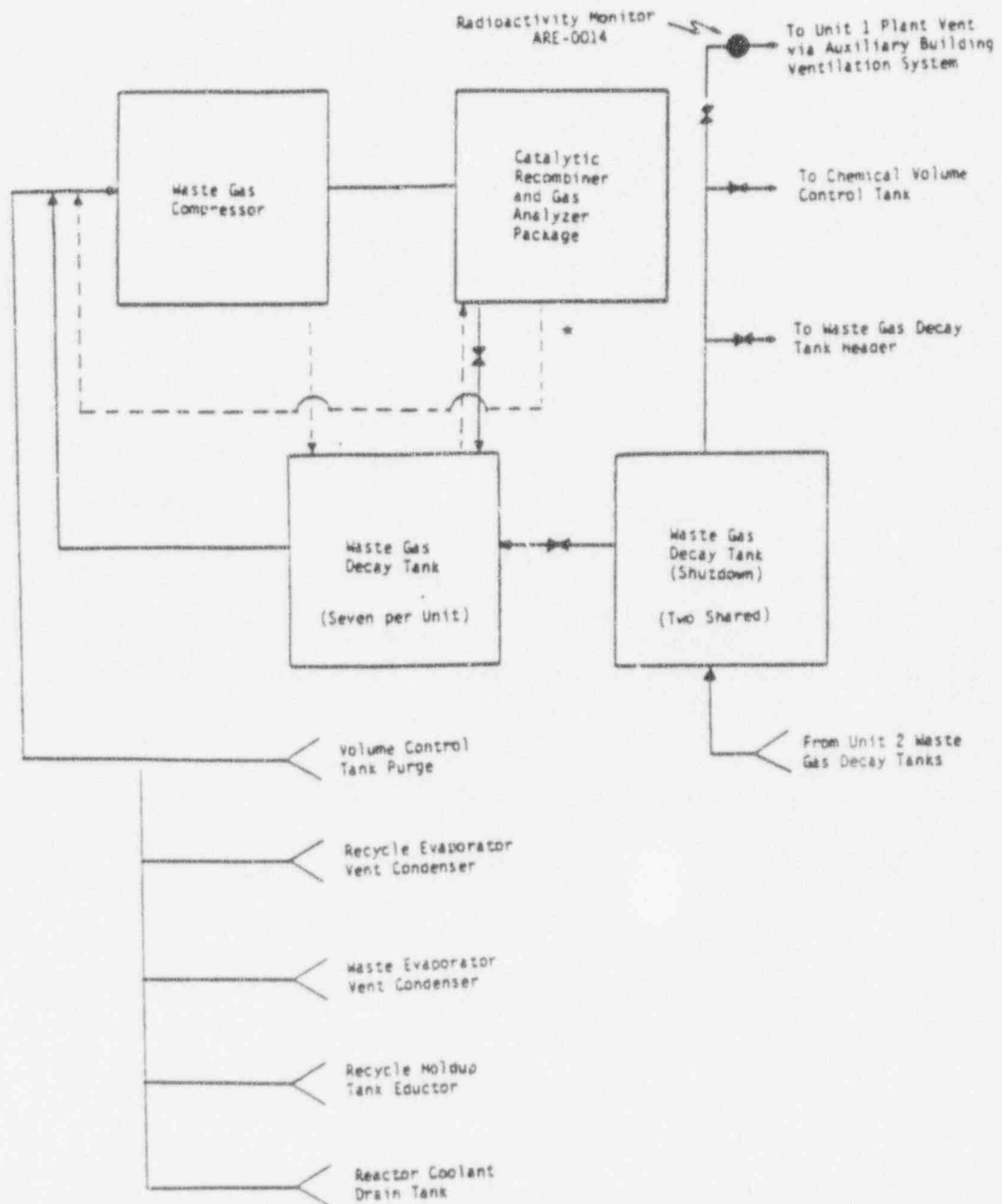
- ACTION 50 - Not Used.
- ACTION 51 - With the number of channels operable less than required by the minimum channels operable requirement, effluent releases via the affected pathway may continue provided samples are continuously collected with auxiliary sampling equipment.

\*During radioactive releases via this pathway.

\*\*During emergency filtration.

## 2.6 GASEOUS RADWASTE TREATMENT SYSTEMS

Figures 2.6-1, 2.6-2, 2.6-3, and 2.6-4 present schematics of the Gaseous Waste Processing System and Ventilation Exhaust Treatment Systems (Reference 5).



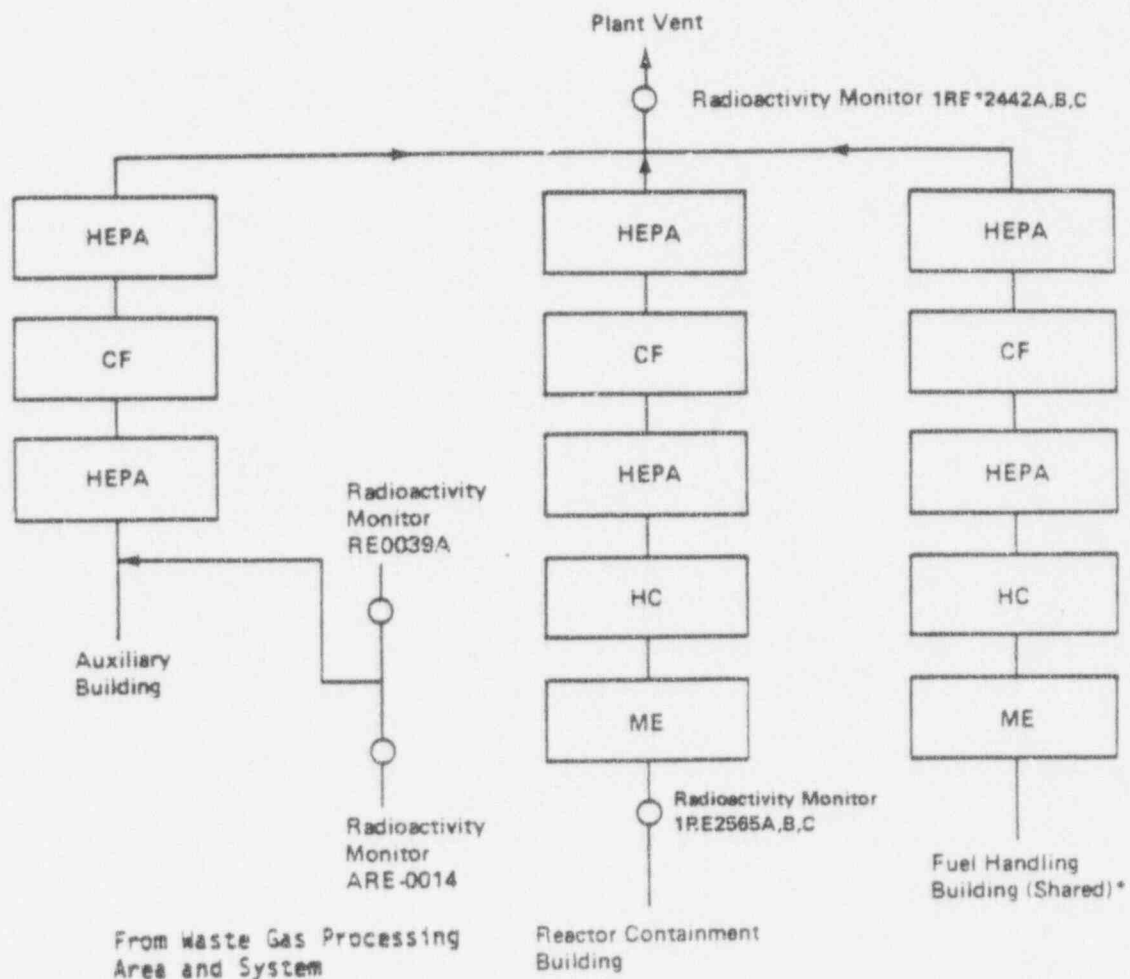
NOTE: This is typical of both units. However, Unit 2 GWPS releases via Unit 1 plant vent.

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ELECTRIC GENERATING PLANT  
UNIT 1 AND UNIT 2

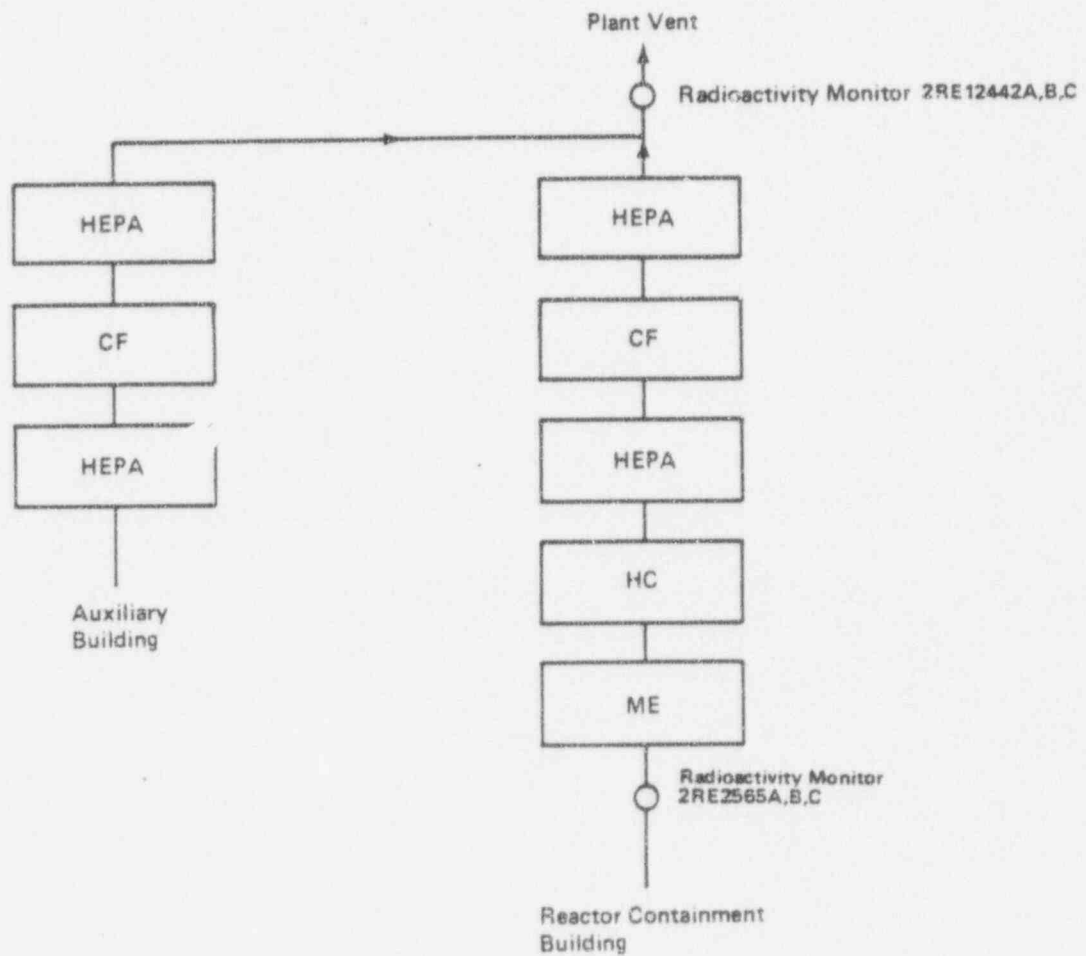
GASEOUS WASTE PROCESSING SYSTEM

FIGURE 2.6-1

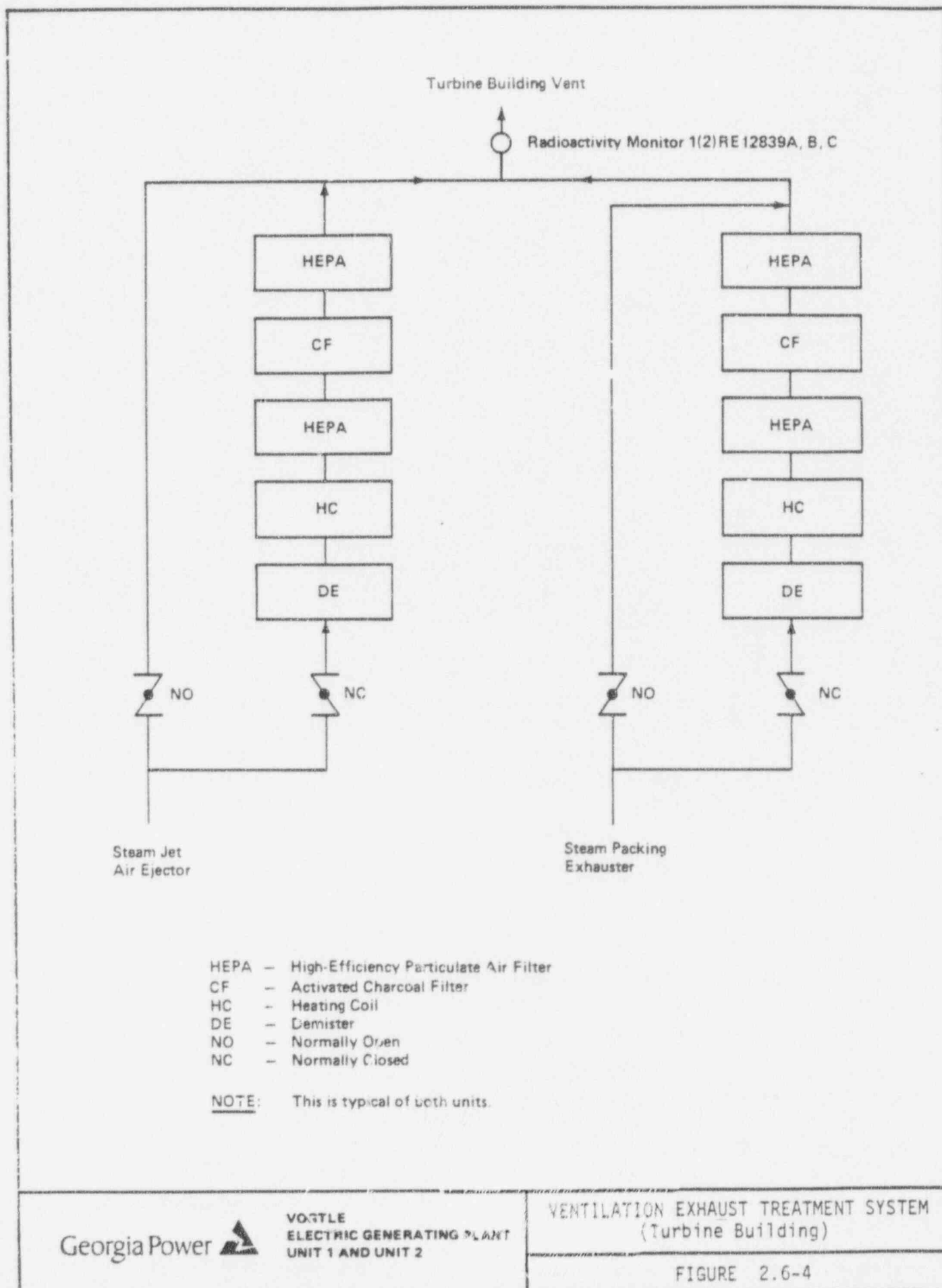


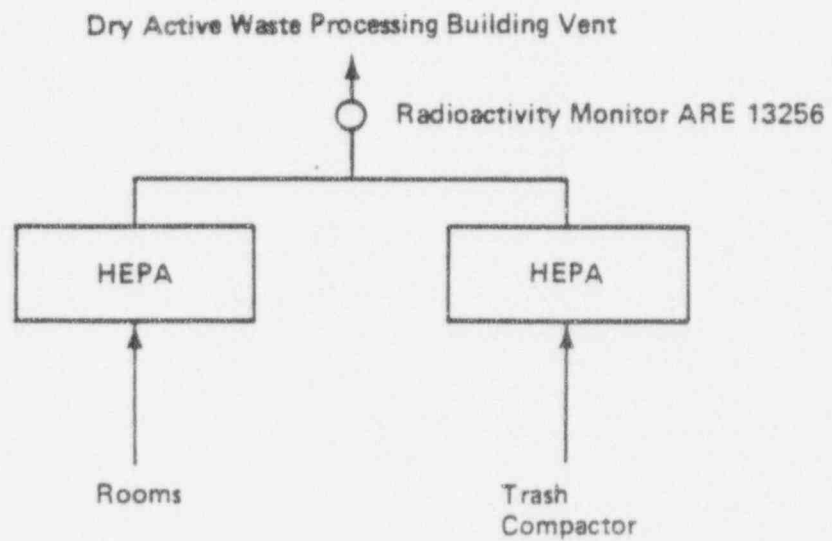
HEPA - High-Efficiency Particulate Air Filter  
 CF - Activated Charcoal Filter  
 HC - Heating Coil  
 ME - Moisture Eliminator

\*Prior to treatment by the Fuel Handling Building Ventilation Exhaust Treatment System, Exhaust from Unit 1 Spent Fuel Pool Area is monitored by ARE2532B and ARE2533B; exhaust from Unit 2 Spent Fuel Pool Area is monitored by ARE2532A and ARE2533A.



HEPA — High-Efficiency Particulate Air Filter  
 CF — Activated Charcoal Filter  
 HC — Heating Coil  
 ME — Moisture Eliminator







### 3.0 RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

Radiological environmental sampling and monitoring locations are described in table 3.0-1 and shown on maps in figures 3.0-1, 3.0-2, 3.0-3, and 3.0-4 as required by subsections 3.1.1 and 3.1.2.

TABLE 3.0-1 (SHEET 1 OF 3)

## RADIOLOGICAL ENVIRONMENTAL SAMPLING LOCATIONS

<u>LOCATION NUMBER</u>	<u>DESCRIPTIVE LOCATION</u>	<u>DIRECTION</u>	<u>DISTANCE (miles)</u>	<u>SAMPLE TYPE<sup>(1)</sup></u>
1	Hancock Landing Road	N	1.1	D
2	River Bank	NNE	0.8	D
3	Discharge Area	NE	0.6	A
3	River Bank	NE	0.7	D
4	River Bank	ENE	0.8	D
5	River Bank	E	1.0	D
6	Plant Wilson	ESE	1.1	D
7	Simulator Building	SE	1.7	D, V, A
8	River Road	SSE	1.1	D
9	River Road	S	1.1	D
10	Met Tower	SSW	0.8	A
10	River Road	SSW	1.1	D
11	River Road	SW	1.2	D
12	River Road	WSW	1.1	D, A
13	River Road	W	1.3	D, V
14	River Road	WNW	1.8	D
15	Hancock Landing Road	NW	1.5	D, V
16	Hancock Landing Road	NNW	1.4	D, A
17	Savannah River Plant River Road	N	5.4	D
18	Savannah River Plant D Area	NNE	5.0	D
19	Savannah River Plant Road A.13	NE	4.6	D
20	Savannah River Plant Road A.13.1	ENE	4.8	D
21	Savannah River Plant Road A.17	E	5.3	D
22	River Bank Downstream of Buxton Landing	ESE	5.2	D
23	River Road	SE	4.7	D
24	Chance Road	SSE	4.9	D
25	Chance Road and Highway 23	S	5.2	D
26	Highway 23, mile 15.5	SSW	4.6	D
27	Highway 23, mile 17	SW	4.8	D
28	Clayton Road	WSW	5.0	D
29	Claxton-Lively Road	W	5.0	D
30	Nathaniel Howard Road	WNW	5.0	D
31	River Road at Allen's Church Fork	NW	5.0	D
32	River Bank	NNW	4.8	D
33	Nearby Residence	SE	3.3	D
34	Girard Elementary School	SSE	6.3	D
35	Girard	SSE	6.6	D, A
36	Waynesboro	WSW	14.9	D, A
37	Substation (Waynesboro)	WSW	17.5	D, V

TABLE 3.0-1 (SHEET 2 OF 3)

## RADIOLOGICAL ENVIRONMENTAL SAMPLING LOCATIONS

<u>LOCATION NUMBER</u>	<u>DESCRIPTIVE LOCATION</u>	<u>DIRECTION</u>	<u>DISTANCE (miles)</u>	<u>SAMPLE TYPE<sup>(1)</sup></u>
43	Employees Recreation Area	SW	2.2	D
80	Augusta Water Treatment Plant	NNW	24.5	W <sup>(2)</sup>
81	Savannah River (RM 153.1)	N	2.2	F <sup>(3)</sup>
82	Savannah River (RM 151.2)	NNE	0.8	R
83	Savannah River (RM 150.4)	ENE	0.8	R
84	Savannah River (RM 149.5)	ESE	1.6	R, S <sup>(4)</sup>
85	Savannah River (RM 146.7)	ESE	5.0	F <sup>(3)</sup>
87	Beaufort-Jasper Water Treat- ment Plant; Beaufort, S.C.	SE	76	W <sup>(5)</sup>
88	Cherokee Hill Water Treatment Plant; Port Wentworth, Ga.	SSE	72	W <sup>(6)</sup>
98	W. C. Dixon Dairy	SE	9.8	M
99	Boyceland Dairy	W	24.5	M

TABLE NOTATIONS:

## (1) Sample Types:

- A - Airborne Radioactivity
- D - Direct Radiation
- F - Fish
- M - Milk
- R - River Water
- S - River Shoreline Sediment
- W - Drinking Water (at water treatment plant)
- V - Vegetation

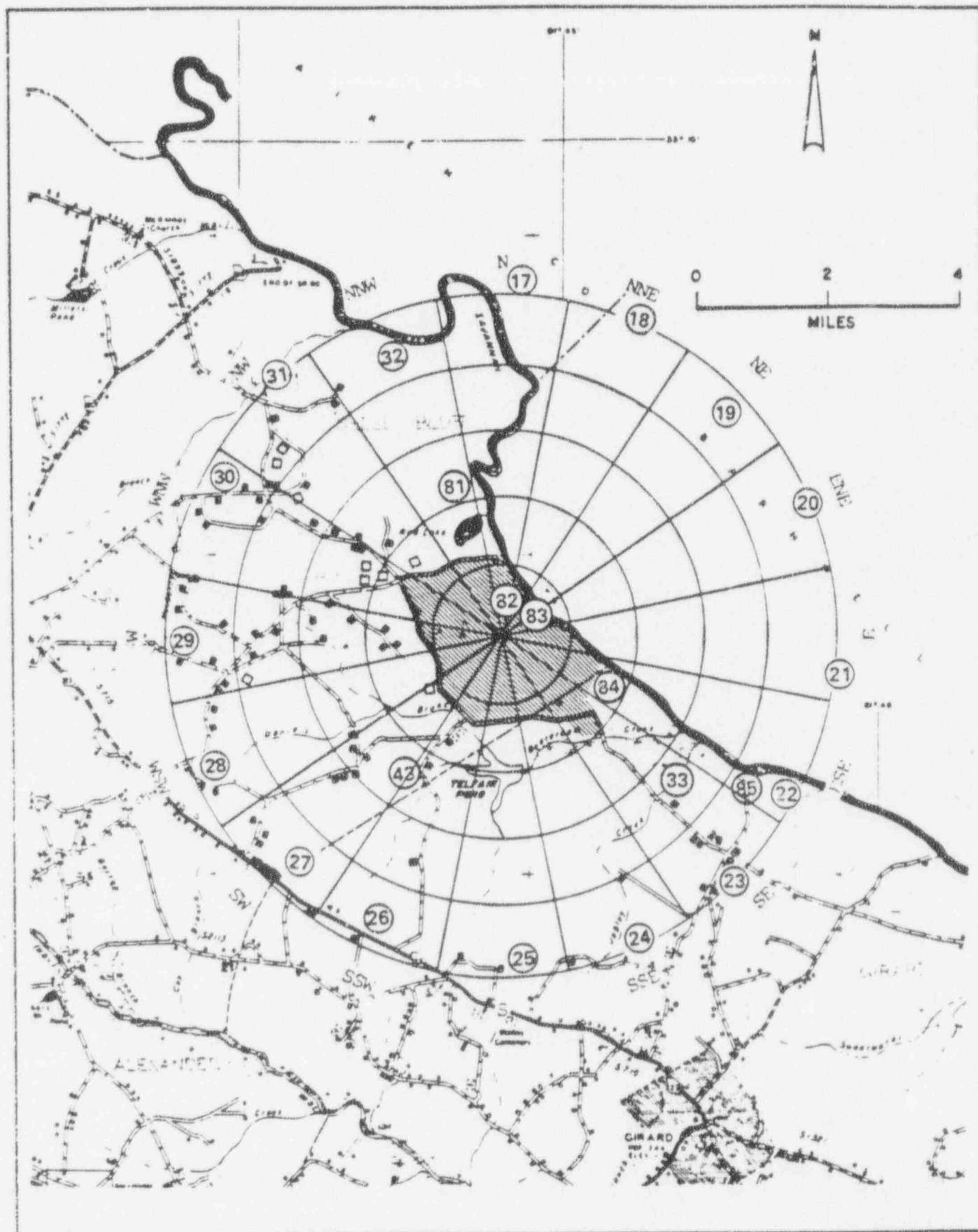
- (2) The intake for the Augusta Water Treatment Plant is located on the Augusta Canal. The entrance to this canal is at river mile (RM) 207 on the Savannah River. The canal effectively parallels the river. The pumping station is 3.5 miles down the canal and only a few tenths of a mile from the river (across land).
- (3) About a 5-mile stretch of the river is generally needed to obtain adequate fish samples. Samples are normally gathered between RM 153 and 158 for upriver collections and between RMs 144 and 149.4 for downriver collections.

TABLE 3.0-1 (SHEET 3 OF 3)

RADIOLOGICAL ENVIRONMENTAL SAMPLING LOCATIONS

Table Notations (Continued)

- (4) Sediment is collected at locations with existing or potential recreational value. Because high water shifting of the river bottom or other reasons could cause a suitable location for sediment collection to become unavailable or unsuitable, a stretch of the river between RM 148.5 and 150.5 which is downriver of the discharge is assigned for sediment collections. In practice, collections are normally made at RM 150.2.
- (5) The intake for the Beaufort-Jasper Water Treatment Plant is located at the end of a canal which begins at RM 59.3 on the Savannah River. This intake is about 16 miles by line of sight down the canal from its beginning on the Savannah River.
- (6) The intake for the Cherokee Hill Water Treatment Plant is located on Abercorn Creek which is about one and a quarter creek miles from its mouth on the Savannah River at RM 29.

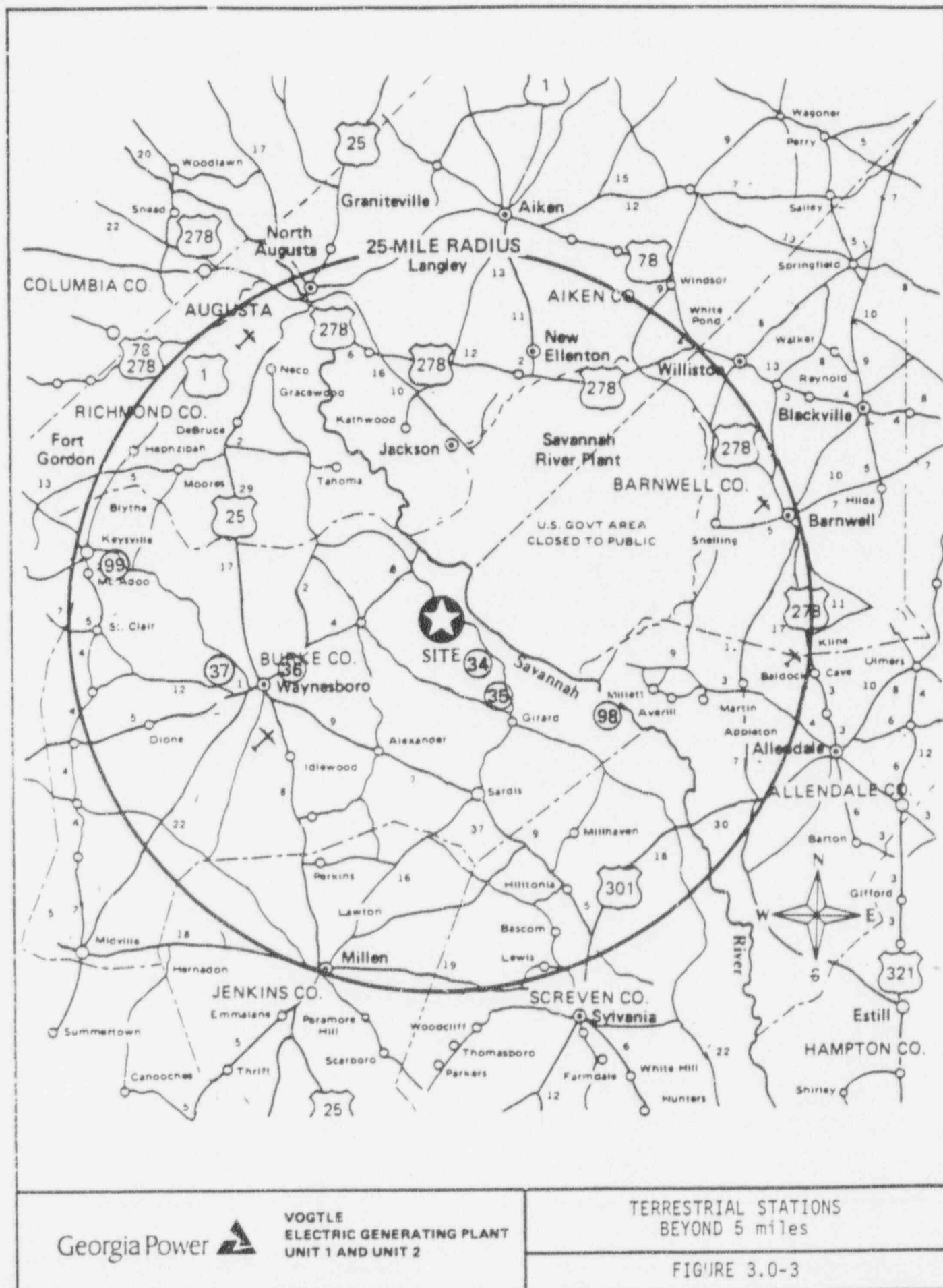


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UNIT 1 AND UNIT 2

TERRESTRIAL STATIONS BEYOND SITE  
BOUNDARY OUT TO APPROXIMATELY 5 miles  
AND AQUATIC STATIONS

FIGURE 3.0-2



### 3.1 Limits of Operation

#### 3.1.1 Conduct of the Radiological Environmental Monitoring Program

The Radiological Environmental Monitoring Program shall be conducted as specified in table 3.1-1. This limit applies at all times for all modes of operation.

##### 3.1.1.1 Reporting of Abnormal Conditions

If the Radiological Environmental Monitoring Program is not being conducted as specified in table 3.1-1, prepare and submit to the NRC, in the Annual Radiological Environmental Surveillance Report required by Technical Specification 6.8.1.3, a description of the reasons for not conducting the program as required and the plans for preventing a recurrence.

If the confirmed\* level of radioactivity as the result of plant effluents in an environmental sampling medium at a specified location exceeds the reporting levels of table 3.1-2 when averaged over any calendar quarter, prepare and submit to the NRC within 30 days, pursuant to Technical Specification 6.8.2, a special report that identifies the cause(s) for exceeding the limit(s) and defines the corrective action(s) to be taken to reduce radioactive effluents so that the potential annual dose\*\* to a member of the public is less than the calendar year limits of subsections 1.5.2 , 2.5.2, or 2.5.3. When more than one of the radionuclides in table 3.1-2 are detected in the sampling medium, this report shall be submitted if:

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\*A confirmatory reanalysis of the original, a duplicate, or a new sample may be desirable or appropriate. The result of the confirmatory analysis shall be completed at the earliest time consistent with the analysis but in any case within 30 days.

\*\*The methodology and parameters used to estimate the potential annual dose to a member of the public shall be indicated in this report.



$$\frac{\text{concentration (1)}}{\text{reporting level (1)}} + \frac{\text{concentration (2)}}{\text{reporting level (2)}} + \dots \geq 1.0$$

When radionuclides other than those in table 3.1-2 are detected and are the result of plant effluents, this report shall be submitted if the potential annual dose\* to a member of the public from all radionuclides is equal to or greater than the calendar-year limits of subsections 1.5.2, 2.5.2, or 2.5.3. This report is not required if the measured level of radioactivity was not the result of plant effluents. However, in such an event, the condition shall be reported and described in the Annual Radiological Environmental Surveillance Report required by Technical Specification 6.8.1.3.

If milk or vegetation samples are unavailable from one or more of the sample locations required by table 3.1-1, identify specific locations for obtaining replacement samples, and add them within 30 days to the Radiological Environmental Monitoring Program given in section 3.0. The specific locations from which samples were unavailable may then be deleted from the monitoring program. Pursuant to Technical Specification 6.13, submit in the next Semiannual Radioactive Effluent Release Report documentation for a change in the ODCM including a revised figure(s) and table for this ODCM reflecting the new location(s), with supporting information identifying the cause of the unavailability of samples and justifying the selection of the new location(s) for obtaining samples. These limits do not affect mode changes.

#### 3.1.1.2 Collection and Analysis of Samples

To assure that these limits are not exceeded, the radiological environmental monitoring samples shall be collected pursuant to table 3.1-1 from the specific locations given in the table and figure(s) in section 3.0 and shall be analyzed pursuant to the requirements of table 3.1-1 and the detection capabilities required by table 3.1-3.

\*The methodology and parameters used to estimate the potential annual dose to a member of the public shall be indicated in this report.



### 3.1.1.3 Basis for the Radiological Environmental Monitoring Program

The Radiological Environment Monitoring Program required by this specification provides representative measurements of radiation and of radioactive materials in those exposure pathways and for those radionuclides that lead to the highest potential radiation exposure of members of the public resulting from the plant operation. This monitoring program implements section IV.B.2, Appendix I, 10 CFR 50, and thereby supplements the Radiological Effluent Monitoring Program by measuring concentrations of radioactive materials and levels of radiation that may be compared with those expected on the basis of the effluent measurements and the modeling of the environmental exposure pathways. Guidance for this monitoring program is provided by the "Radiological Assessment Branch Technical Position on Environmental Monitoring," Revision 1, November 1979. The initially specified monitoring program will be effective for at least the first 3 years of commercial operation. Following this period, program changes may be initiated based on operational experience.

The required detection capabilities for environmental sample analyses are tabulated in terms of the LLDs. The LLDs required by table 3.1-3 are considered optimum for routine environmental measurements in industrial laboratories. It should be recognized that the LLD is defined as an *a priori* (before the fact) limit representing the capability of a measurement system and not as an *a posteriori* (after the fact) limit for a particular measurement.

Detailed discussion of the LLD, and other detection limits, can be found in L. A. Currie, "Limits for Qualitative Detection and Quantitative Determination - Application to Radiochemistry," Anal. Chem. 40, 586-93, 1968, and J. K. Hartwell, "Detection Limits for Radioanalytical Counting Techniques," Atlantic Richfield Hanford Company Report ARH-SA-215, June 1975.

### 3.1.2 Land Use Census

A land use census shall be conducted and shall identify within a distance of 5 miles the location in each of the 16 meteorological sectors of the nearest milk animal, the nearest residence, and the nearest garden of greater than 500 ft<sup>2</sup> producing broad leaf vegetation. Land within the Savannah River plant may be excluded from this survey. This limit applies at all times for all modes of operation.

#### 3.1.2.1 Identification of New Critical Locations

If a land use census identifies a location(s) that yields a calculated dose or dose commitment greater than the values currently being calculated in subsection 2.5.3 pursuant to Technical Specification 6.8.1.4, identify the new location(s) in the next Semiannual Radioactive Effluent Release Report.

If a land use census identifies a location(s) that yields a calculated dose or dose commitment (via the same exposure pathway) 20 percent greater than at a location from which samples are currently being obtained in accordance with subsection 3.1.1, add the new location(s) within 30 days to the Radiological Environmental Monitoring Program given in section 3.0, if samples are available. The sampling location(s) (excluding the control-station location) having the lowest calculated dose or dose commitment(s), via the same exposure pathway, may then be deleted from this monitoring program. Pursuant to Technical Specification 6.13, submit in the next Semiannual Radioactive Effluent Release Report documentation for a change in this ODCM including a revised figure(s) and table(s) reflecting the new location(s) with information supporting the change in sampling locations. These limits do not affect modes changes.

#### 3.1.2.2 Conduct of Census

To assure that these limits are not exceeded, the land use census shall be conducted during the growing season at least once per 12 months using that information that will provide good results, such as by a door-to-door survey, by visual survey from automobile or aircraft, by consulting local agriculture authorities, or by some combination of these methods as feasible. The results of the land use census shall be included in the Annual Radiological Environmental Surveillance Report pursuant to Technical Specification 6.8.1.3.

#### 3.1.2.3 Identification of Changes

This specification is provided to ensure that changes in the use of areas at and beyond the site boundary are identified and that modifications to the Radiological Environmental Monitoring Program are made if required by the results of this census. The best information from the door-to-door survey, from aerial survey, or from consulting with local agricultural authorities shall be used. This census satisfies the requirements of section IV.B.3, Appendix I, 10 CFR 50. Restricting the census to gardens of greater than 500 ft<sup>2</sup> provides assurance that significant exposure pathways via leafy vegetables will be identified and monitored since a garden of this size is the minimum required to produce the quantity (26 kg/yr) of leafy vegetables assumed in Regulatory Guide 1.109 for consumption by a child. To determine this minimum garden size, the following assumptions were made: (1) 20 percent of the garden was used for growing broad leaf vegetation (i.e., similar to lettuce and cabbage) and (2) a vegetation yield of 2 kg/m<sup>2</sup> was obtained.

TABLE 3.1-1 (SHEET 1 OF 6)

## RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

<u>EXPOSURE PATHWAY AND/OR SAMPLE</u>	<u>NUMBER OF REPRESENTATIVE SAMPLES AND SAMPLE LOCATIONS<sup>(1)</sup></u>	<u>SAMPLING AND COLLECTION FREQUENCY</u>	<u>TYPE AND FREQUENCY OF ANALYSIS</u>
1. Direct Radiation <sup>(2)</sup>	<p>Thirty-six routine monitoring stations, either with two or more dosimeters or with one instrument for measuring and recording dose rate continuously, placed as follows:</p> <p>An inner ring of stations, one in each meteorological sector in the general area of the site boundary.</p> <p>An outer ring of stations, one in each meteorological sector in the 6-mile range from the site.</p> <p>The balance of the stations to be placed in special interest areas such as population centers, nearby residences, schools, and in one or two areas to serve as control stations.</p>	Quarterly.	Gamma dose quarterly.

TABLE 3.1-1 (SHEET 2 OF 6)

## RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

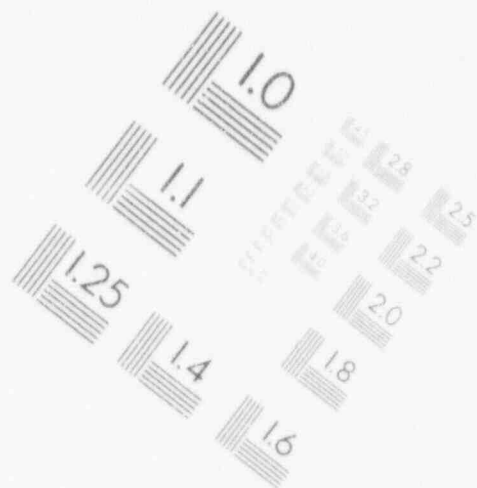
<u>EXPOSURE PATHWAY AND/OR SAMPLE</u>	<u>NUMBER OF REPRESENTATIVE SAMPLES AND SAMPLE LOCATIONS<sup>(1)</sup></u>	<u>SAMPLING AND COLLECTION FREQUENCY</u>	<u>TYPE AND FREQUENCY OF ANALYSIS</u>
2. Airborne			
Radioiodine and Particulates	<p>Samples from five locations.</p> <p>Three samples from close to the three site boundary locations, in different sectors.</p> <p>One sample from the vicinity of a community having the highest calculated annual average ground-level D/Q.</p> <p>One sample from a control location, as, for example, a population center 10 to 20 miles distant and in the least prevalent wind direction.</p>	Continuous sampler operation with sample collection weekly, or more frequently if required by dust loading.	<p><u>Radioiodine Canister:</u> I-131 analysis weekly.</p> <p><u>Particulate Sampler:</u> Gross beta radioactivity analysis following filter change,<sup>(3)</sup> and gamma isotopic analysis<sup>(4)</sup> of composite (by location) quarterly.</p>
3. Waterborne			
a. Surface <sup>(5)</sup>	<p>One sample upstream.</p> <p>One sample downstream.</p>	Composite sample over 1-month period. <sup>(6)</sup>	Gamma isotopic analysis <sup>(4)</sup> monthly; composite for tritium analysis quarterly.

TABLE 3.1-1 (SHEET 3 OF 6)

## RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

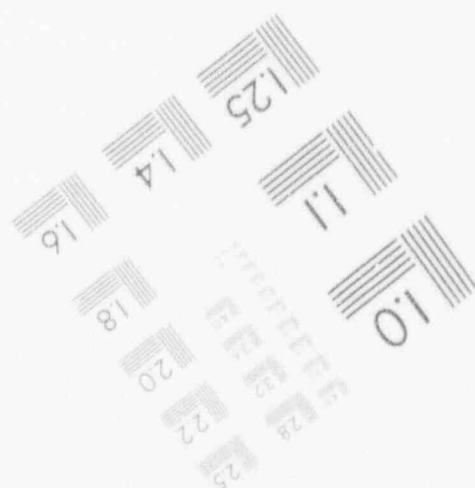
<u>EXPOSURE PATHWAY AND/OR SAMPLE</u>	<u>NUMBER OF REPRESENTATIVE SAMPLES AND SAMPLE LOCATIONS<sup>(1)</sup></u>	<u>SAMPLING AND COLLECTION FREQUENCY</u>	<u>TYPE AND FREQUENCY OF ANALYSIS</u>
3. Waterborne (Continued)			
b. Drinking	Two samples at each of one to three of the nearest water treatment plants that could be affected by its discharge.  Two samples at a control location.	Composite sample of river water near intake at each water treatment plant over 2-week period <sup>(6)</sup> when I-131 analysis is performed, monthly composite otherwise; and grab sample of finished water at each water treatment plant every 2 weeks or monthly, as appropriate.	I-131 analysis on each sample when the dose calculated for the consumption of the water is greater than 1 mrem per year <sup>(7)</sup> . Composite for gross beta and gamma isotopic analyses <sup>(4)</sup> monthly. Composite for tritium analysis quarterly.
c. Sediment from Shoreline	One sample from downstream area with existing or potential recreational value.	Semiannually.	Gamma isotopic analysis <sup>(4)</sup> semiannually.
4. Ingestion			
a. Milk	Samples from milking animals in three locations within 3 miles distance having the highest dose potential; if there are none, then one sample from milking animals <sup>(8)</sup> in each of three areas between 3 and 5 miles distance where doses are calculated to be greater than 1 mrem per yr. <sup>(7)</sup>	Semimonthly.	Gamma isotopic analysis <sup>(4)(9)</sup> semimonthly.

IMAGE EVALUATION  
TEST TARGET (MT-3)



150mm

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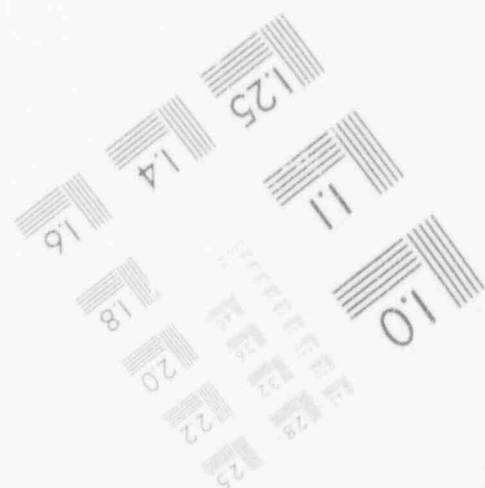
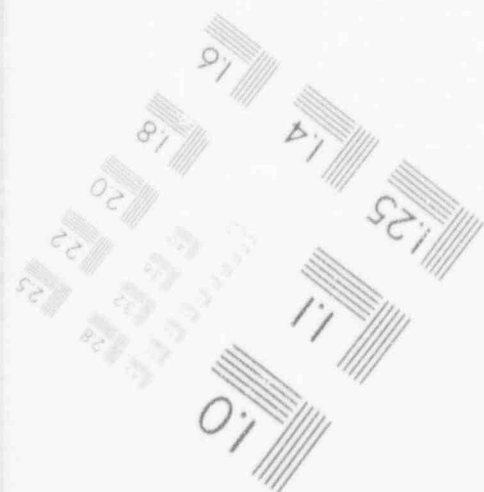
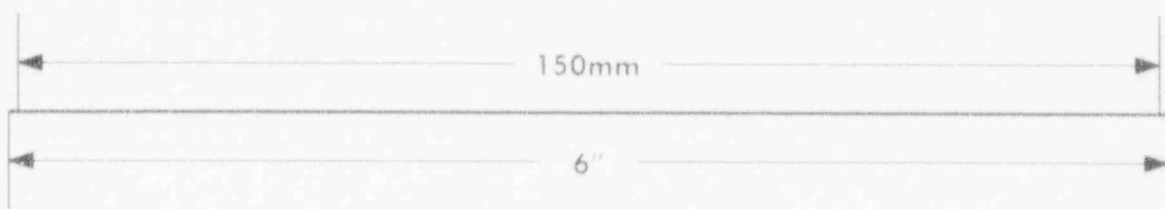
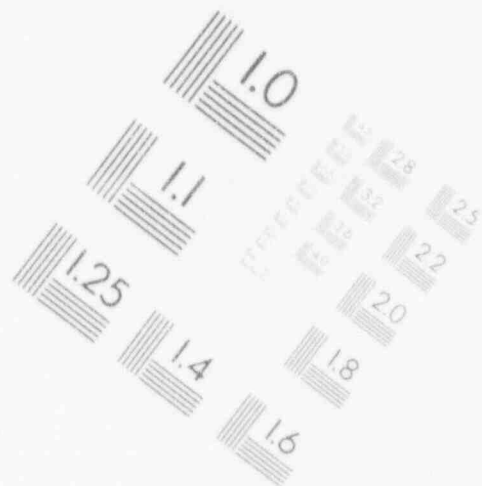
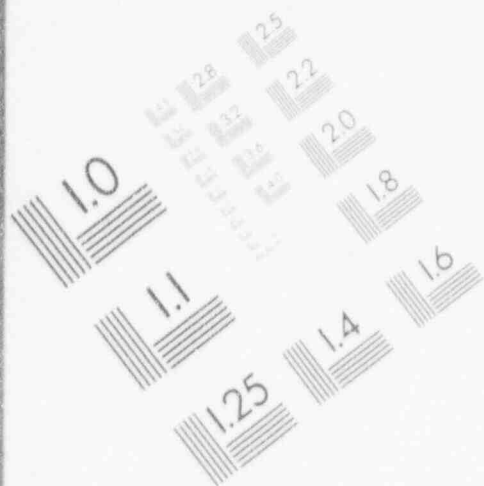
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## IMAGE EVALUATION TEST TARGET (MT-3)



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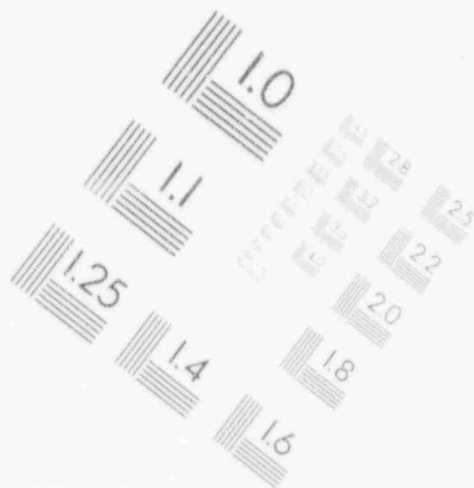
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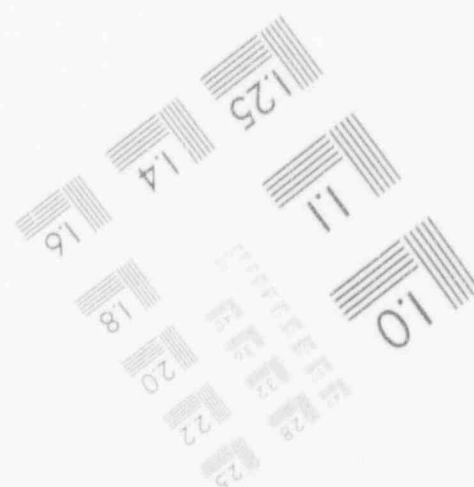


IMAGE EVALUATION  
TEST TARGET (MT-3)



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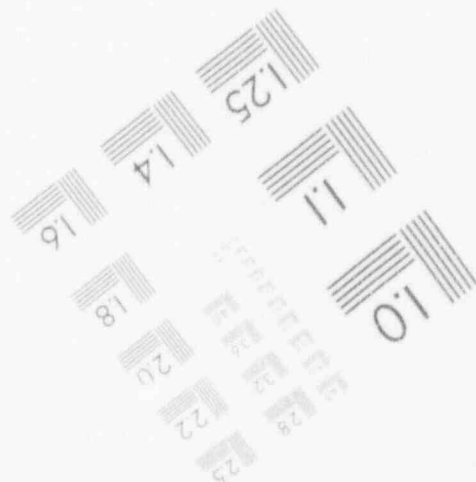
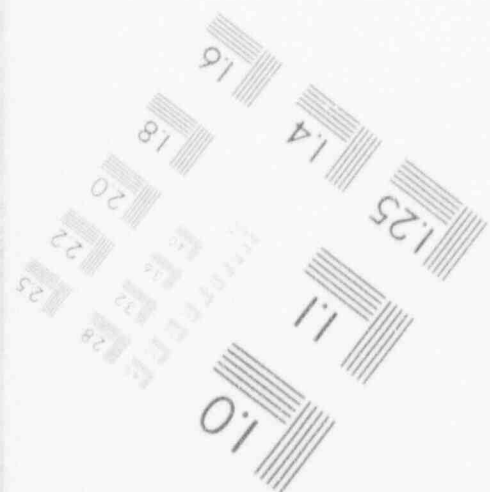
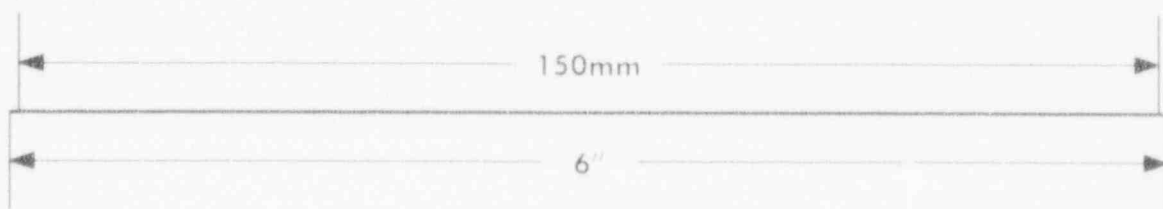
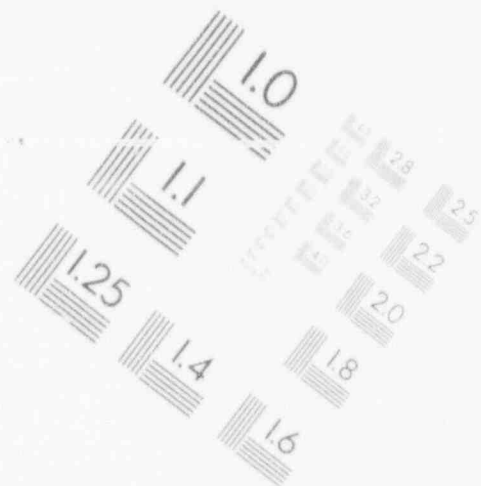
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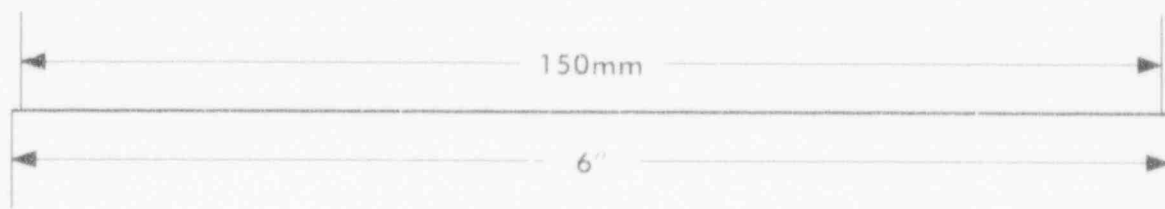
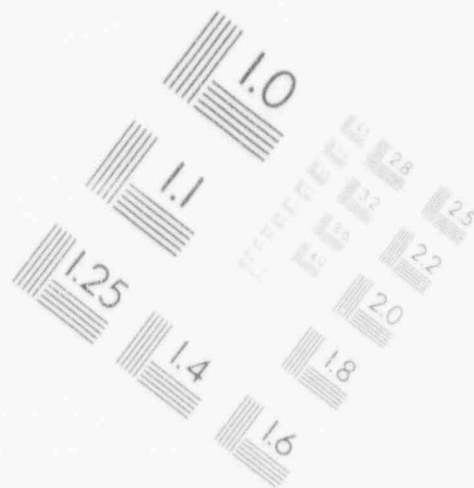
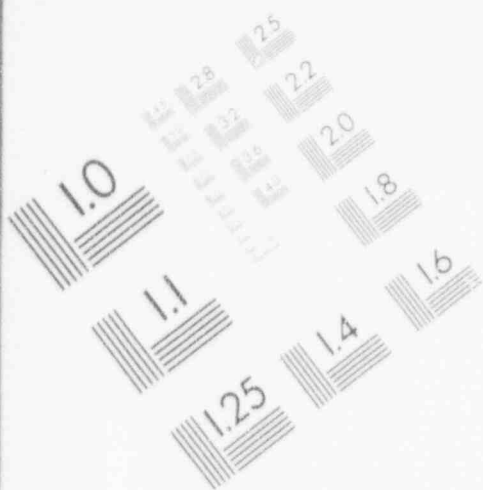
## IMAGE EVALUATION TEST TARGET (MT-3)



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## IMAGE EVALUATION TEST TARGET (MT-3)



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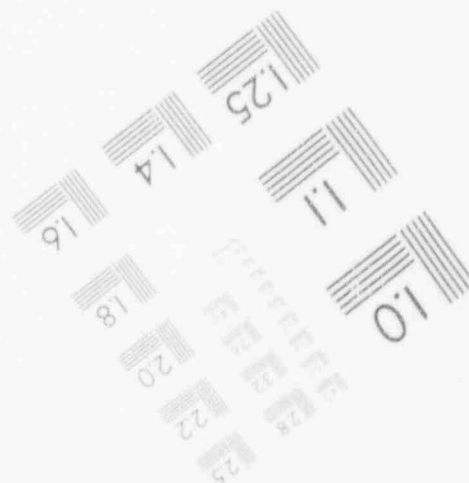
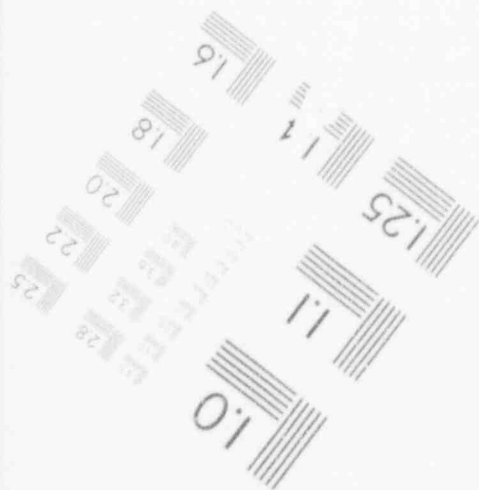


TABLE 3.1-1 (SHEET 4 OF 6)

## RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM

<u>EXPOSURE PATHWAY AND/OR SAMPLE</u>	<u>NUMBER OF REPRESENTATIVE SAMPLES AND SAMPLE LOCATIONS<sup>(1)</sup></u>	<u>SAMPLING AND COLLECTION FREQUENCY</u>	<u>TYPE AND FREQUENCY OF ANALYSIS</u>
4. Ingestion (Continued)			
a. Milk	One sample from milking animals <sup>(8)</sup> at a control location about 10 miles distant or beyond and preferably in a wind direction of lower prevalence.		
b. Fish	At least one sample of any commercially and recreationally important species in vicinity of plant discharge area.  At least one sample of any species in areas not influenced by plant discharge.  At least one sample of any anadromous species in vicinity of plant discharge.	Semiannually.	Gamma isotopic analysis <sup>(4)</sup> on edible portions.
c. Grass or Leafy Vegetation	One sample from two onsite locations near the site boundary in different sectors.  One sample from a control location about 15 miles distant.	During spring spawning season.  Monthly during growing season.  Monthly during growing season.	Gamma isotopic analyses <sup>(4)</sup> on edible portion.  Gamma isotopic. <sup>(4) (9)</sup>  Gamma isotopic. <sup>(4) (9)</sup>

TABLE 3.1-1 (SHEET 5 OF 6)

## TABLE NOTATIONS

- (1) Specific parameters of distance and direction sector from a point midway between the center of the two reactors, and additional description where pertinent, shall be provided for each and every sample location in table 3.1-1 in a table and figure(s) in this ODCM. Each sample location will be designated by a number, name, or some other label. Refer to NUREG-0133, "Preparation of Radiological Effluent Technical Specifications for Nuclear Power Plants," October 1978 and to "Radiological Assessment Branch Technical Position," Revision 1, November 1979.

Deviations are permitted from the required sampling schedule if specimens are unobtainable due to circumstances, such as hazardous conditions, seasonal unavailability, and malfunction of sampling equipment. If specimens are unobtainable due to sampling equipment malfunction, effort shall be made to complete corrective action prior to the end of the next sampling period. All deviations from the sampling schedule shall be documented in the Annual Radiological Environmental Surveillance Report pursuant to Technical Specification 6.8.1.3.

It is recognized that, at times, it may not be possible or practicable to continue to obtain samples of the media of choice at the most desired location or time. In these instances, suitable alternative media and locations may be chosen for the particular pathway in question and appropriate substitutions, if available, will be made within 30 days in the Radiological Environmental Monitoring Program given in this ODCM. Pursuant to Technical Specification 6.13, submit in the next Semiannual Radioactive Effluent Release Report documentation for a change in this ODCM, including a revised figure(s) and table reflecting the new location(s), if any, with supporting information identifying the cause of the unavailability of samples for the pathway and justifying the selection of the new location(s) for obtaining samples or the unavailability of suitable new locations.

- (2) One or more instruments, such as a pressurized ion chamber, for measuring and recording dose rate continuously, may be used in place of or in addition to integrating dosimeters. For the purpose of this table, a thermoluminescent dosimeter (TLD) is considered to be one phosphor; two or more phosphors in a packet are considered as two or more dosimeters. Film badges shall not be used as dosimeters for measuring direct radiation.
- (3) Airborne particulate sample filters shall be analyzed for gross beta radioactivity 24 hours or more after sampling to allow for radon and thoron daughter decay. If gross beta activity in air particulate samples is greater than 10 times the yearly mean of control samples, gamma isotopic analysis shall be performed on the individual samples.

TABLE 3.1-1 (SHEET 6 OF 6)

## TABLE NOTATIONS

- (4) Gamma isotopic analysis means the identification and quantification of gamma-emitting radionuclides that may be attributable to the effluents from the facility.
- (5) The upstream sample shall be taken at a distance beyond significant influence of the discharge. The downstream sample shall be taken in an area beyond but near the mixing zone.
- (6) Composite sample aliquots shall be collected at time intervals that are very short (e.g., hourly) relative to the compositing period (e.g., monthly) in order to assure obtaining a representative sample.
- (7) The dose shall be calculated for the maximum organ and age group, using the methodology and parameter in the ODCM.
- (8) A milking animal is a cow or goat producing milk for human consumption.
- (9) If gamma isotopic analysis is not sensitive enough to meet the lower limit of detection for I-131, a separate analysis for I-131 will be performed.



TABLE 3.1-2  
REPORTING LEVELS FOR RADIOACTIVITY CONCENTRATIONS IN ENVIRONMENTAL SAMPLES  
REPORTING LEVELS

ANALYSIS	WATER (pCi/l)	AIRBORNE PARTICULATE OR GASES (pCi/m <sup>3</sup> )	FISH (pCi/kg, wet)	MILK (pCi/l)	GRASS OR LEAFY VEGETATION (pCi/kg, wet)
H-3	20,000*				
Mn-54	1,000		30,000		
Fe-59	400		10,000		
Co-58	1,000		30,000		
Co-60	300		10,000		
Zn-65	300		20,000		
Zr-95	400				
Nb-95	400				
I-131	2	0.9		3	100
Cs-134	30	10	1,000	60	1,000
Cs-137	50	20	2,000	70	2,000
Ba-140	200			200	
La-140	100			300	

\*For drinking water samples. This is 40 CFR 141 value. If no drinking water pathway exists, a value of 30,000 pCi/l may be used.

TABLE 3.1-3 (SHEET 1 OF 3)  
 DETECTION CAPABILITIES FOR ENVIRONMENTAL SAMPLE ANALYSIS<sup>(1)(2)</sup>  
 LOWER LIMIT OF DETECTION (LLD)<sup>(3)</sup>

ANALYSIS	WATER (pCi/l)	AIRBORNE PARTICULATE OR GASES (pCi/m <sup>3</sup> )	FISH (pCi/kg, wet)	MILK (pCi/l)	GRASS OR LEAFY VEGETATION (pCi/kg, wet)	SEDIMENT (pCi/kg, dry)
Gross Beta	4	0.01				
H-3	2000 <sup>(4)</sup>					
Mn-54	15		130			
Fe-59	30		260			
Co-58	15		130			
Co-60	15		130			
Zn-65	30		260			
Zr-95	30					
Nb-95	15					
I-131	1 <sup>(5)</sup>	0.07		1	60	
Cs-134	15	0.05	130	15	60	150
Cs-137	18	0.06	150	18	80	180
Ba-140	60			60		
La-140	15			15		



TABLE 3.1-3 (SHEET 2 OF 3)

## TABLE NOTATIONS

- (1) This list does not mean that only these nuclides are to be considered. Other peaks that are identifiable as plant effluents, together with those of the above nuclides, shall also be analyzed and reported in the Annual Radiological Environmental Surveillance Report pursuant to Technical Specification 6.8.1.3.
- (2) Required detection capabilities for thermoluminescent dosimeters used for environmental measurements shall be in accordance with the recommendations of Regulatory Guide 4.13.
- (3) The LLD is defined, for purposes of these specifications, as the smallest concentration of radioactive material in a sample that will yield a net count, above system background, that will be detected with 95-percent probability, with only 5-percent probability of falsely concluding that a blank observation represents a "real" signal.

For a particular measurement system, which may include radiochemical separation:

$$LLD = \frac{4.66 s_b}{E \cdot V \cdot 2.22 \cdot Y \cdot \exp(-\lambda \Delta t)}$$

where:

- LLD = the *a priori* lower limit of detection (picoCuries per unit mass or volume)
- $s_b$  = the standard deviation of the background counting rate or of the counting rate of a blank sample as appropriate (counts per minute)
- E = the counting efficiency (counts per disintegration)
- V = the sample size (units of mass or volume)

TABLE 3.1-3 (SHEET 3 OF 3)

TABLE NOTATIONS (Continued)

- 2.22 = the number of disintegrations per minute per picoCurie
- $Y$  = the fractional radiochemical yield, when applicable
- $\lambda$  = the radioactive decay constant for the particular radionuclide ( $\text{sec}^{-1}$ )
- $\Delta t$  = the elapsed time between sample collection, or end of the sample collection period, and time of counting (sec)

Typical values of  $E$ ,  $V$ ,  $Y$ , and  $\Delta T$  should be used in the calculation.

It should be recognized that the LLD is defined as an *a priori* (before the fact) limit representing the capability of a measurement system and not as an *a posteriori* (after the fact) limit for a particular measurement. Analyses shall be performed in such a manner that the stated LLDs will be achieved under routine conditions. Occasionally background fluctuations, unavoidable small sample sizes, the presence of interfering nuclides, or other uncontrollable circumstances may render these LLDs unachievable. In such cases, the contributing factors shall be identified and described in the Annual Radiological Environmental Surveillance Report pursuant to Technical Specification 6.8.1.3.

- (4) If no drinking water pathway exists, a value of 3000 pCi/l may be used.
- (5) If no drinking water pathway exists, a value of 15 pCi/l may be used.

## 3.2 INTERLABORATORY COMPARISON PROGRAM

### 3.2.1 Requirements

Analysis shall be performed on all radioactive materials, supplied as part of an interlaboratory comparison program that has been approved by the NRC, that correspond to samples and analysis required by table 3.1-1. This limit applies at all times for all modes of operation.

### 3.2.2 Deviations

If analyses are not being performed as required above, report the corrective actions taken to prevent a recurrence to the NRC in the Annual Radiological Environmental Surveillance Report pursuant to Technical Specification 6.8.1.3.

### 3.2.3 Summary of Results

To assure that these limits are not exceeded, a summary of the results obtained as part of the above required interlaboratory comparison program shall be included in the Annual Radiological Environmental Surveillance Report pursuant to Technical Specification 6.8.1.3.

### 3.2.4 Basis

The requirement for participation in an approved interlaboratory comparison program is provided to ensure that independent checks on the precision and accuracy of the measurements of radioactive materials in environmental sample matrices are performed as part of the quality assurance program for environmental monitoring in order to demonstrate that the results are valid for the purposes of section IV.B.2, Appendix I, 10 CFR 50.

#### 4.0 TOTAL DOSE DETERMINATIONS

Subsection 2.5.5 addresses the requirements of 40 CFR 190 and 10 CFR 20.105(c), which pertain to limitation of annual doses to a member of the public from nuclear fuel cycle facilities. No other nuclear fuel cycle facility is located within five miles of Plant Vogtle. Therefore, it is only necessary to include doses from the two Plant Vogtle units in the total dose determinations.

For the purpose of implementing subsection 2.5.5, total dose determinations will be made by calculating doses due to liquid effluents in accordance with subsection 1.5.2 by calculating doses due to gaseous effluents in accordance with subsection 2.5.3 and by combining direct radiation doses based on direct radiation measurements, or calculations, with these effluent doses to determine total dose to a real individual. Methodology for calculating individual doses due to liquid effluents was presented in section 1.2. Methodology for calculating individual doses due to gaseous effluents was presented in paragraph 2.2.2.2.

## 5.0 POTENTIAL DOSES TO MEMBERS OF THE PUBLIC DUE TO THEIR ACTIVITIES INSIDE THE SITE BOUNDARY

For the purpose of implementing Technical Specification 6.8.1.4, an assessment of potential doses to members of the public due to their activities within the site boundary will be performed by calculating total-body doses due to noble gas releases and organ doses due to radioiodine, tritium, and particulates in gaseous releases. The locations of interest within the site boundary at Plant Vogtle are the Visitors Center and Plant Wilson. (Plant Wilson is owned and operated by Georgia Power Company, but individuals working at Plant Wilson are not directly associated with Plant Vogtle. Therefore, those individuals are considered in this dose determination as a precautionary measure.)

Annual average atmospheric dispersion and deposition values for these two locations, expected occupancy factors, (for an individual during the year), and applicable age groups are presented in table 5.0-1.

Total-body doses due to noble gases are determined using the following equation:

$$D_{tg} = 3.17E-8 \left[ \left( \overline{X/Q}_g \sum_i K_i \tilde{Q}_{i,g} + \overline{X/Q}_m \sum_i K_i \tilde{Q}_{i,m} \right) \right] \quad (OF)$$

where  $\overline{X/Q}_g$  and  $\overline{X/Q}_m$  are ground-level and mixed-mode dispersion terms for the location of interest;  $K_i$  is the total-body dose factor from table 2.1-1; and OF is the occupancy factor. Other terms are described in subsection 2.2.2.

Organ doses due to radioiodine, tritium, and particulates in gaseous releases are determined in accordance with the methodology presented in paragraph 2.2.2.2 (equation (21)). Only the inhalation pathway (equation (22)) and the ground-plane pathway (equation (23)) are applicable for locations inside the site boundary. After doses are calculated for the locations of interest inside the site boundary, using parameters from

table 5.0-1, the results are multiplied by the appropriate occupancy factors to determine doses to individuals at the locations of interest inside the site boundary.

TABLE 5.0-1 (SHEET 1 OF 2)

LOCATION-SPECIFIC PARAMETERS FOR LOCATIONS  
INSIDE THE SITE BOUNDARY

Location: Visitors Center; SE at 447 meters  
Age group: Child  
Estimated occupancy factor: 4.6E-4 (4 hours)  
Ground-level dispersion and deposition parameters:

$$(\overline{X/Q}) = 5.93E-6 \text{ s/m}^3$$

$$\text{Depleted: } (\overline{X/Q}) = 5.58E-6 \text{ s/m}^3$$

$$(\overline{D/Q}) = 2.28E-8 \text{ 1/m}^2$$

Mixed-mode dispersion and deposition parameters:

$$(\overline{X/Q}) = 7.12E-7 \text{ s/m}^3$$

$$\text{Depleted: } (\overline{X/Q}) = 6.74E-7 \text{ s/m}^3$$

$$(\overline{D/Q}) = 5.77E-9 \text{ 1/m}^2$$

Location: Plant Wilson; ESE at 1420 meters  
Age group: Adult  
Estimated occupancy factor: 2.28E-1 (2000 hours)\*  
Ground-level dispersion and deposition parameters:

$$(\overline{X/Q}) = 9.45E-7 \text{ s/m}^3$$

$$\text{Depleted: } (\overline{X/Q}) = 8.34E-7 \text{ s/m}^3$$

$$(\overline{D/Q}) = 4.20E-9 \text{ 1/m}^2$$

TABLE 5.0-1 (SHEET 2 OF 2)

Mixed-mode dispersion and deposition parameters:

$$(\overline{X/Q}) = 1.76E-7 \text{ s/m}^3$$

Depleted:  $(\overline{X/Q}) = 1.59E-7 \text{ s/m}^3$

$$(\overline{D/Q}) = 2.07E-9 \text{ 1/m}^2$$

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\*This value is based on a 40-hour work week, assuming an individual is assigned to the facility for the entire year.



## 6.0 REPORTS

### 6.1 ANNUAL RADIOLOGICAL ENVIRONMENTAL SURVEILLANCE REPORT\*\*\*

#### 6.1.1 Due Date

Routine Annual Radiological Environmental Surveillance Reports covering activities of the radiological environmental monitoring program during the previous calendar year shall be submitted prior to May 1 of each year. The initial report shall be submitted prior to May 1 of the year following initial criticality and shall include copies of reports of the preoperational radiological environmental monitoring program of the plant for at least two years prior to initial criticality.

#### 6.1.2 Evaluation Content Requirement

The Annual Radiological Environmental Surveillance Report shall include summaries, interpretations, and an analysis of trends of the results of the radiological environmental surveillance activities for the report period, including, as appropriate, a comparison with preoperational studies, with operational controls, and with previous environmental surveillance reports, and an assessment of any observed impacts of plant operations on the environment. The report shall also include the results of the land use census required by section 3.1.2.

#### 6.1.3 Data Content Requirement

The Annual Radiological Environmental Surveillance Report shall include the results of analysis of all radiological environmental samples and of all environmental radiation measurements taken during the period pursuant to the

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\*\*\*A single submittal may be made for Units 1 and 2.

locations specified in the table and figures in section 3.0, as well as summarized and tabulated results of these analyses and measurements in the format of the table in the Radiological Assessment Branch Technical Position, Revision 1, November 1979. The radiological level of radionuclides, which are naturally occurring, not included in the plant effluents, need not be reported. In the event that some individual results are not available for inclusion with the report, the report shall be submitted noting and explaining the reasons for the missing results. The missing data shall be submitted as soon as practicable in a supplementary report.

#### 6.1.4 Program Content Requirement

The report shall also include the following: a summary description of the radiological environmental monitoring program, at least two legible maps covering all sampling locations keyed to a table giving distances and directions from a point midway between the two reactors, the results of licensee participation in the interlaboratory comparison program and the corrective action taken if the specified program is not being performed, reasons for not conducting the radiological environmental monitoring program and discussion of all deviations from the sampling schedule, discussion of environmental sample measurements that exceed the reporting levels but are not the result of plant effluents, and discussion of all analyses in which the LLD required was not achieved.

## 6.2 SEMIANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT\*

Routine Semiannual Radioactive Effluent Release Reports covering the operation of the unit during the previous 6 months of operation shall be submitted within 60 days after January 1 and July 1 of each year.

### 6.2.1 Gaseous and Liquid Effluent Summaries

The Semiannual Radioactive Effluent Release Reports shall include a summary of the quantities of radioactive liquid and gaseous effluents and solid waste released from both units as outlined in Regulatory Guide 1.21, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants," Revision 1, June 1974, with data summarized on a quarterly basis following the format of Appendix B thereof. For solid wastes, the format for table 3 in Appendix B shall be supplemented with three additional categories: class of solid wastes (as defined by 10 CFR 61), type of container (e.g., LSA, type A, type B, large quantity), and solidification agent or absorbent (e.g., cement, urea formaldehyde).

### 6.2.2 Meteorological Data Summary

The Semiannual Radioactive Effluent Release Report to be submitted within 60 days after January 1 of each year shall include an annual summary of hourly meteorological data collected over the previous year. This annual summary may be either in the form of an hour-by-hour listing on magnetic tape of wind speed, wind direction, atmospheric stability, and precipitation (if measured), or in the form of joint frequency distributions of windspeed, wind direction,

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\*A single submittal may be made for Units 1 and 2. The submittal should combine those sections that are common to both units at the plant; however, the submittal shall specify the releases of gaseous and liquid radioactive material from each unit.

and atmospheric stability.\* This same report shall include an assessment of released from each unit during the previous calendar year. This same report the radiation doses due to the radioactive liquid and gaseous effluents shall also include an assessment of the radiation doses from radioactive liquid and gaseous effluents to members of the public due to their activities inside the site boundary during the report period. All assumptions used in making these assessments (i.e., specific activity, exposure time, and location) shall be included in these reports. Historical annual average meteorological conditions or the meteorological conditions concurrent with the time of release of radioactive materials in gaseous effluents, as determined by sampling frequency and measurement, shall be used for determining the gaseous pathway doses. The assessment of radiation doses shall be performed in accordance with the methodology and parameters in this manual.

#### 6.2.3 Radiation Doses to the Public

The Semiannual Radioactive Effluent Release Report to be submitted within 60 days after January 1 of each year shall also include an assessment of radiation doses to the likely most exposed member of the public from reactor releases and other uranium fuel cycle sources within 8 km, including doses from primary effluent pathways and direct radiation, for the previous calendar year to show conformance with 40 CFR 190, "Environmental Radiation Protection Standards for Nuclear Power Operation." Acceptable methods for calculating the dose contribution from liquid and gaseous effluents are given in Regulatory Guide 1.109, Revision 1, October 1977.

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\*In lieu of submission with the Semiannual Radioactive Effluent Release Report, the licensee has the option of retaining this summary of required meteorological data on site in a file that shall be provided to the NRC upon request.

#### 6.2.4 Unplanned Releases

The Semiannual Radioactive Effluent Release Reports shall include a list and description of unplanned releases from the site to unrestricted areas of radioactive materials in gaseous and liquid effluents made during the reporting period.

#### 6.2.5 Changes to the ODCM

The Semiannual Radioactive Effluent Release Reports shall include any changes made during the reporting period to the Offsite Dose Calculation Manual (ODCM), pursuant to Technical Specification 6.13, as well as any major change to liquid, gaseous, or solid radwaste treatment systems pursuant to subsection 6.2.7. It shall also include a listing of new locations for dose calculations and/or environmental monitoring identified by the land use census pursuant to subsection 3.1.2.

#### 6.2.6 Inoperable Liquid or Gaseous Effluent Monitoring Instrumentation

The Semiannual Radioactive Effluent Release Reports shall also include the following: an explanation as to why the inoperability of liquid or gaseous effluent monitoring instrumentation was not corrected within the time specified in subsection 1.5.4 or 2.5.6, respectively, and description of the events leading to liquid holdup tanks or gas storage tanks exceeding the limits of Technical Specification 3.11.1.4 or 3.11.2.6, respectively.

#### 6.2.7 Major Changes to Liquid, Gaseous, and Solid Radwaste Treatment Systems\*

Licensee-initiated major changes to the radwaste treatment systems (liquid, gaseous, and solid)

- a. Shall be reported to the NRC in the Semiannual Radioactive Effluent Release Report for the period in which the evaluation was reviewed by the PRB. The discussion of each change shall contain the following:
  - 1) A summary of the evaluation that led to the determination that the change could be made in accordance with 10 CFR 50.59.
  - 2) Sufficient detailed information to totally support the reason for the change without benefit of additional or supplemental information.
  - 3) A detailed description of the equipment, components, and processes involved and the interfaces with other plant systems.
  - 4) An evaluation of the change, which shows the predicted releases of radioactive materials in liquid and gaseous effluents and/or quantity of solid waste that differ from those previously predicted in the license application and amendments thereto.
  - 5) An evaluation of the change, which shows the expected maximum exposures to a member of the public in the unrestricted area and to the general population that differ from those previously estimated in the license application and amendments thereto.

\*Licensees may choose to submit the information called for in this specification as part of the annual FSAR update.

- 6) A comparison of the predicted releases of radioactive materials, in liquid and gaseous effluents and in solid waste, to the actual releases for the period prior to when the change is to be made.
  - 7) An estimate of the exposure to plant operating personnel as a result of the change.
  - 8) Documentation of the fact that the change was reviewed and found acceptable by the PRB.
- b. Shall become effective upon approval by the General Manager - Nuclear Plant.



## REFERENCES

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4. Vogtle Electric Generating Plant Units 1 and 2 Environmental Report - Operating License Stage, Georgia Power Company.
5. Vogtle Electric Generating Plant Units 1 and 2 Final Safety Analysis Report, Georgia Power Company.
6. "Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors," U.S. NRC Regulatory Guide 1.111, March 1976.
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8. "Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I," U. S. NRC Regulatory Guide 1.113, Revision 1, April 1977.
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12. Vogtle Electric Generating Plant Land Use Survey - 1988, Georgia Power Company, April 1988.
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17. Memo from S. E. Ewald, Georgia Power Company, to C. C. Eckert, Georgia Power Company, May 9, 1988.
18. Memo from A. C. Stalker, Georgia Power Company, to D. F. Hallman, Georgia Power Company, February 5, 1987.

VOGTLE ELECTRIC GENERATING PLANT  
PROCESS CONTROL PROGRAM  
Revision 3

APPROVED: W. Shyman 1/22/93  
General Manager - Nuclear Plant

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1.0

PURPOSE

1.1

The purpose of the Process Control Program (PCP) is to define the necessary program guidance used at Vogtle Electric Generating Plant to ensure that solid radioactive waste management activities result in the production of a waste form that meets or exceeds the requirements of 10 CFR 20, 10 CFR 50, 10 CFR 61, 10 CFR 71, Radiological Effluent Technical Specifications, as well as other Federal, State, and burial site requirements.

1.1.1

The PCP is a description of the techniques and policies associated with the disposal of solid radioactive waste. It describes the steps used by the plant staff to characterize the waste prior to stabilization, assure the quality of materials used during processing, and verify and document the integrity of the final waste form.

1.1.2

This PCP meets the requirements of Technical Specification 6.12.

2.0

SCOPE

2.1

The PCP explains the methods by which the radwaste disposal technology is monitored to assure production of a certifiable waste product. Applicable procedures, administrative controls, and quality control techniques are presented.

2.1.1

Written procedures have been established, implemented, and maintained covering the implementation of the Process Control Program. The PCP and implementing procedures are approved by Vogtle management prior to use. Changes to the PCP will be reviewed by the Plant Review Board and in accordance with the Technical Specifications.

2.1.2

The implementing procedures reference and incorporate, where appropriate, the process information found in the vendor's topical report. Vendor topical reports provide information such as:

- Binder used.
- Additive used.
- Process control parameters.
- Waste form properties.
- Description of solidification process.
- Mixing times.
- Boundary conditions.

3.0

## REFERENCES

3.1

### Regulatory Guidance

3.1.1.

49 CFR Parts 171-178

Department of Transportation  
Hazardous Materials Regulations.

3.1.2

10 CFR 20.311

Transfer For Disposal and  
Manifests.

3.1.3

10 CFR 50

Domestic Licensing of Production  
and Utilization Facilities.

3.1.4

10 CFR 61

Licensing Requirements for Land  
Disposal of Radioactive Wastes.

3.1.5

10 CFR 71

Packaging of Radioactive Material  
for Transport and Transportation  
of Radioactive Material Under  
Certain Conditions.

3.1.6

NUREG 0800

Standard Review Plan 11.4, Solid  
Waste Management Systems.

3.1.7

Branch Technical Position ETSB 11-3, Design Guidance for Solid  
Radioactive Waste Management Systems Installed in Light-Water  
Cooled Nuclear Power Reactor Plants.

3.1.8

Low-Level Waste Management Branch Technical Position on  
Radioactive Waste Classification - May 1983, Revision 0.

3.1.9

Low-Level Waste Management Branch Technical Position on Waste  
Form - January 1991, Revision 1.

3.2

### Licensing Documents

3.2.1

Vogtle FSAR sections 11.4, 13.0.

3.2.2

Vogtle Technical Specifications 6.7, 6.8, 6.9, and 6.12.

3.2.3

Vogtle Safety Evaluation Report section 11.4.

3.3

### State and Local Guidance

3.3.1

South Carolina Department of Health And Environmental Control  
Radioactive Material License No. 097, (Barnwell Facility).

3.3.2

Barnwell Site Disposal Criteria, (Chem Nuclear).

3.4        Vendor Topicals

- 3.4.1      Topical Report Covering Nuclear Packaging, Inc., Dewatering System, Topical Report No. TP-02, Acceptance Date September 1985.

3.5        Vendor Process Control Program

- 3.5.1      Nuclear Packaging, Inc., Waste Sampling Procedure, PT-10.  
3.5.2      Nuclear Packaging, Inc., Sieve Analysis Procedure, PT-11.  
3.5.3      Nuclear Packaging, Inc., Procedure for Visual Operations, PT-12.

3.6        Vendor Operating and Maintenance Manuals

- 3.6.1      Operating Procedure Pacific Nuclear Systems, Inc./Nuclear Packaging, Inc., Resin Drying (Dewatering) System, OM-43.  
3.6.2      Nuclear Packaging, Inc., Operation and Maintenance Manual Resin Drying System, OM-42.

3.7        Vogtle Operating Procedures

- 3.7.1      Spent Resin Processing Procedure for Nuclear Packaging, Inc., Dewatering System.  
3.7.2      Chemistry Analysis Procedure  
3.7.3      Health Physics Procedures for Solid Radwaste Processing, Waste Classification, and Shipping.  
3.7.4      ALARA Program

4.0        OPERATING RESPONSIBILITIES

- 4.1        The operating organization for radwaste is in accordance with FSAR chapter 13.  
4.2        The responsibilities for processing and disposing of radwaste are divided between the Operations and Health Physics/Chemistry Departments.  
4.3        Processing of liquid and gaseous radwaste is performed by the Operations Department. Radwaste operators operate the equipment used for processing, such as demineralizers and filters and their associated pumps and valves.

- 4.4 Health Physics/Chemistry Waste and Decon personnel process and dispose of solid radwaste. They sort dry active waste (DAW), load disposal containers, classify, and ship the radwaste offsite for burial.
- 4.5 The Chemistry Department is responsible for analyzing the waste stream and reagent samples generated during radwaste processing. Samples are analyzed using VEGP approved procedures.
- 5.0 DEFINITIONS
- 5.1 Additive
- 5.1.1 Material which is introduced into the waste container for the purpose of promoting even, thorough solidification of the waste.
- 5.2 Batch
- 5.2.1 All waste held in a storage vessel for representative sampling prior to processing.
- 5.3 Batch Test Sample
- 5.3.1 A quantity of waste physically removed from the waste batch and subsequently solidified according to the batch test sample procedure, the test sample is subject to verification testing which, after having successfully met test requirements, certifies the entire batch.
- 5.4 Binder
- 5.4.1 The actual material which when intimately mixed with the waste according to vendor approved formulas, physically encapsulates the waste into a free standing monolith following an acceptable cure time.
- 5.5 Cure Time
- 5.5.1 The time interval elapsing from the initial mixing of the water, binder, and additives until the onset of solidification as verified by either the exothermic heat generation or the characteristic time experienced during qualification testing of a successful batch test sample which yielded a certifiable waste form.

5.6 Free Standing Monolith

5.6.1 The resulting homogeneous mixture of waste, binder, and additives which, when combined in proper ratios according to the PCP procedures, yielded a nonflowing, self-supporting mass.

5.7 Free Standing Liquid

5.7.1 The quantity of water or liquid that can be drained from a solidification container following the accepted cure time, or in the case of dewatering, following the accepted dewatering time and process parameter acceptance criteria.

5.8 Onset of Solidification

5.8.1 The time at which the waste form is free standing and meets minimum compressive strength criteria.

5.9 Solidification

5.9.1 Solidification shall be the conversion of wet wastes into a form that meets shipping and burial ground requirements. When the specific amounts of waste, binder, and additive are mixed in accordance with vendor PCP formulas, resulting in a free standing monolith and whose process parameters are within the PCP stated boundary conditions and the batch test sample was certified.

5.10 Waste Form

5.10.1 Waste in a final packaged form acceptable for shipment to a licensed disposal facility.

5.11 Waste

5.11.1 Those low-level radioactive wastes containing source, special nuclear, or byproduct material that are acceptable for disposal in a licensed disposal facility. For the purposes of this definition, low-level waste has the same meaning as in the Low-Level Waste Policy Act, that is, radioactive waste not classified as high-level radioactive waste, transuranic waste, spent nuclear fuel, or byproduct material as defined in section 11.(2) of the Atomic Energy Act (uranium or thorium tailings or waste).



6.0 WASTE DESCRIPTION

6.1 Portable Vendor Supplied Liquid Processing Equipment

6.1.1 The liquid waste processing system (LWPS) is described in detail in Vogtle FSAR section 11.2. Potentially radioactive liquid wastes are collected in waste holdup tanks, floor drain tanks, boron recycle holdup tanks and laundry and hot shower drain tank, and chemical drain tank. These liquids are processed through portable vendor supplied liquid processing equipment located in the alternate radwaste building (ARB). This equipment consists of mechanical cartridge filters, ion exchange demineralizers, and a microfiltration system. The microfiltration system is a precoatable, backflushable filter designed to remove small, submicron (0.1 to 0.3 micron) particulate from the waste stream.

6.1.2 The portable vendor supplied liquid processing equipment can be aligned into numerous processing logics which are dependent upon the radioactivity and chemical makeup of the liquid being processed. Solid radioactive waste from the portable vendor supplied liquid processing system consists of mechanical cartridge filters and ion exchange resins. These solid wastes are processed using portable vendor supplied solid radwaste processing equipment, which is also located in the ARB.

6.2 Evaporator Concentrates

6.2.1 Reactor coolant system liquids may be recycled using the boron recycle system evaporator. Waste system liquids may be processed using the LWPS waste evaporators. Evaporator concentrates are transferred to the ARB for solidification using portable vendor supplied solid radwaste processing equipment.

6.3 Contaminated Oil

6.3.1 Contaminated oil is collected at its source, transported, and stored in leak-tight containers. Contaminated oil may be filtered and sampled for unconditional release or shipped offsite to a licensed waste processor for volume reduction and disposal.

#### 6.4

##### Dry Active Waste

Dry active wastes are roughly segregated, collected, and bagged in the radiation controlled areas. The bagged waste is then transported to a staging area where bags are sorted to eliminate undesirable materials. The waste is then typically prepared for bulk shipment to an offsite licensed waste processor for volume reduction and disposal. The waste may also be processed in the dry active waste processing facility where compactible waste is compacted into 92-cubic-foot metal boxes, 55-gallon metal drums, or other suitable containers. Non-compactible wastes are packaged manually in metal boxes, 55-gallon drums, or other suitable containers. Waste processed in the dry active waste processing facility is shipped to a licensed radioactive waste disposal facility or to an offsite licensed waste processor for further volume reduction prior to disposal.

#### 6.5

##### Spent Resin and Filter Crud

##### 6.5.1

Condensate polisher resins are normally not radioactive. However, in the event of a primary-to-secondary steam generator tube leak, condensate polisher resins may be transferred to the alternate radwaste building for processing using portable vendor supplied solid radwaste processing equipment. Connections are also available in the turbine building for portable vendor supplied solid radwaste processing equipment should the processing demand became too great on the ARB.

##### 6.5.2

Steam generator blowdown demineralizer (SGBD) resins and liquid waste processing system (LWPS) resins are transferred to the SGBD and LWPS spent resin storage tanks. The resins are transferred to the ARB for processing utilizing portable vendor supplied solid radwaste processing equipment.

##### 6.5.3

Filter crud from the backflushable filter system is processed via portable vendor supplied liquid processing equipment located in the ARB. Provisions are also available to transfer filter crud directly to portable vendor supplied solid radwaste processing equipment located in the ARB.

#### 6.6

##### Cartridge Filters

Cartridge filters are transported to the ARB for disposal in 55-gallon drums, high integrity containers, or other suitable containers.

## 7.0

### PROCESS DESCRIPTION

Vogtle's radwaste processing system, which this PCP addresses, is composed of portable vendor supplied solid radwaste processing equipment located in the alternate radwaste building. This section contains a brief description of VEGP systems designed to handle radioactive waste. For further details, see FSAR chapter 11. Solid radioactive waste streams and their respective processing systems are also outlined in table 1.

## 7.1 Portable Vendor Supplied Solid Radwaste Processing Equipment

7.1.1 The portable vendor supplied solid radwaste processing equipment consists of the Nuclear Packaging Inc., dewatering system, located in the alternate radwaste building. This equipment has an approved topical report (Report No. TP-02-P) dated September 6, 1985. The equipment is operated in accordance with the procedures and the Process Control Program contained in the topical report in order to ensure that all established free standing liquid requirements are met or exceeded for shipment and disposal of dewatered ion exchange and filter media.

7.1.2 Connections for portable vendor supplied solidification equipment are also located in the ARB. Should solidification be required, vendor processes will be required to have NRC approved topical reports and will be required to be accepted for disposal by the sited state. Specific plant procedures, consistent with the vendors, shall be written and approved by the Plant Review Board to ensure that the waste processed is within the boundary conditions established in the qualification test programs that are addressed in the vendor's topical report.

## 7.2 High Integrity Containers

High integrity containers are used in conjunction with the Nuclear Packaging, Inc., dewatering system in order to meet the requirements of 10 CFR 20.311(d) and disposal site requirements. These containers are high density polyethylene (HDPE), high integrity containers. Assurance and documentation has been provided by the disposal site operator that structural stability consistent with 10 CFR 61 requirements will be provided at the disposal site. The State of South Carolina has authorized the Barnwell disposal site operator to receive class B & C waste in HDPE/HICS for disposal in concrete overpacks.

7.3

### Licensed Waste Processor

All waste transferred to a licensed waste processor is prepared in accordance with 10 CFR 20.311 (d) and other applicable regulations. This waste consists principally of dry active waste and contaminated oil. Volume reduction techniques used by the licensed waste processor consist of compaction, incineration, decontamination, and other techniques. Following volume reduction, the licensed waste processor repackages and ships the waste to the disposal facility in accordance with 10 CFR 20.311 (f) and other applicable regulations governing the transportation and disposal of radioactive waste.

7.4

### Dry Active Waste Processing And Storage Facility

Should the services of the licensed waste processor be unavailable, or at VEGP's option, dry active waste may be processed for disposal at the dry active waste processing and storage facility. Equipment is available for sorting and segregating undesirable items and for compacting waste into 55-gallon drums or 92-cubic-foot metal boxes. Disposal containers are designed to meet the requirements of 10 CFR 61 and Department of Transportation packaging regulations. Temporary storage for processing waste containers is available in the dry active waste storage facility.

8.0

## ADMINISTRATIVE CONTROLS

8.1

### Procurement

8.1.1

Vendors are chosen using the VEGP procurement process. The bid specification requires that vendors have approved NRC topical reports and that the process and final waste form are acceptable for disposal by the sited state. Vendor processes are required to meet the Federal, State, and burial ground regulations identified in section 3.0 of this Process Control Program. In addition to meeting the regulations, vendors are required to provide VEGP with their own NRC approved topical report, which is subject to VEGP approval and policies. Plant procedures will be approved consistent with vendor process controls, as detailed in the topical report.

8.2            Independent Audit

8.2.1        The Radwaste Department is subject to independent audits of its activities. Audits are conducted according to sections 13.4 and 17.2 of the FSAR.

8.2.2        Audits include, but are not limited to, evaluating ALARA techniques, sampling techniques, shipment forms, and procedures. Audits are done annually, as a minimum. Audit results are documented and forwarded to the appropriate department and plant managers.

8.3            Procedure Control

8.3.1        This Process Control Program and the procedures referenced herein are based on documented test data and vendor documents. These documents are controlled as described in FSAR section 13.5. A detailed listing of these procedures is contained in table 2 of this document.

9.0            WASTE CLASSIFICATION

9.1            This section addresses the analytical techniques utilized to comply with the applicable State, Federal, and burial site regulations. These techniques are subject to modification in the event of regulatory change or in the interest of worker exposure and reporting accuracy. Any changes are reflected in approved operating procedures.

9.1.1        Prior to shipping wastes to the burial site, the waste is classified according to 10 CFR 61.55. In order to classify the waste as to the type of radioactive material to be buried, selected radionuclides, their radiation level, and any other chemical species which may be present in appreciable quantities are measured.

9.2            Methods used for sampling, analyzing, and determining the radionuclides and their concentrations are consistent with the guidance set in the Low-Level Waste Licensing Branch Technical Position on Radioactive Waste Classification - May 1983, Revision 0 and are contained in approved plant procedures.

- 9.2.1 Complete detailed analyses for all 10 CFR 61 radionuclides are performed periodically to confirm the correlation of measurements made from gross activity measurements or whenever there is reason to believe that facility or process changes may have altered previously determined correlations of gross radioactivity measurements.

10. SOLIDIFICATION/DEWATERING VERIFICATION

10. Bases - Radioactive Effluents - Solid Radioactive Wastes

This specification implements the requirements of General Design Criterion 60 of Appendix A to 10 CFR Part 50. The process parameters included in establishing the Process Control Program may include, but are not limited to, waste type, waste pH, waste/liquid/solidification agent/catalyst ratios, waste oil content, waste principal chemical constituents, and mixing and curing times.

10.2 Limiting Condition for Operation

- 10.2.1 Radioactive wastes shall be solidified or dewatered in accordance with the Process Control Program to meet shipping and transportation requirements during transit and disposal site requirements when received at the disposal site.

- 10.2.2 Applicability - At all times

10.2.3 Action

- a. With solidification or dewatering not meeting disposal site and shipping and transportation requirements, suspend shipment of the inadequately processed wastes and correct the Process Control Program, the procedures, and/or the solid waste system as necessary to prevent recurrence.
- b. With solidification or dewatering not performed in accordance with the Process Control Program, test the improperly processed waste in each container to ensure that it meets burial ground and shipping requirements and take appropriate administrative action to prevent recurrence.



10.3

SURVEILLANCE REQUIREMENTS

10.3.1

Solidification of at least one representative test specimen from at least every tenth batch of each type of wet radioactive wastes (e.g., filter sludges, spent resins, evaporator bottoms, boric acid solutions, and sodium sulfate solutions) shall be verified in accordance with the Process Control Program:

- a. 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, alternative 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.
- b. If the initial test specimen from a batch of waste fails to verify solidification, the Process Control Program shall provide for the collection and testing of representative test specimens from each consecutive batch of the same type of wet waste until at least three consecutive initial test specimens demonstrate solidification. The Process Control Program shall be modified as required, as provided in Technical Specifications section 6.12, to assure solidification of subsequent batches of waste.
- c. With the installed equipment incapable of meeting section 10.2 or declared inoperable, restore the equipment to operable status or provide for contract capability to process wastes as necessary to satisfy all applicable transportation and disposal requirements.

11.0

RECORDS

11.1

Materials Receipt Records

11.1.1

Materials are inspected upon receipt, sampled as necessary, and logged. The receipt date, condition of material, shelf life, quantities, and other pertinent information are recorded.

11.2        Processing, Shipping, Disposal, and Waste Classification Records

11.2.1      Records regarding the processing, classification, shipping, and disposal of radioactive waste are developed, completed, reviewed, and maintained in accordance with Technical Specifications section 6.9, FSAR subsection 17.2.6, and approved plant procedures.

12.0        REPORTS

12.1        Semiannual Radioactive Effluent Release Report

The Semiannual Radioactive Effluent Release Report, submitted in accordance with Technical Specifications 6.8.1.4, shall include a summary of the quantities of solid radwaste released from the units, as outlined in Regulatory Guide 1.21, "Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants," Revision 1, June 1974, with data summarized on a 6-month basis following the format of Appendix B thereof. For each type of solid radwaste shipped offsite during the report period, the report shall include the following information:

- a. Container volume.
- b. Total curie quantity (specify whether determined by measurement or estimate).
- c. Principal radionuclides (specify whether determined by measurement or estimate).
- d. Type of waste (e.g., spent resin, compacted dry waste, evaporator bottoms).
- e. Type of container (e.g., LSA, type A, type B, large quantity).
- f. Solidification agent (e.g., cement, urea formaldehyde).



Licensee initiated major changes to the solid radioactive waste treatment system shall be reported to the Nuclear Regulatory Commission in the Semiannual Radioactive Effluent Release Report for the period in which the change was implemented. The discussion of each change shall include the following:

- a. A summary of the evaluation that led to the determination that the change could be made in accordance with 10 CFR 50.59.
- b. Sufficient detailed information to totally support the reason for the change without benefit of additional or supplemental information.
- c. A detailed description of the equipment, components, and processes involved and the interfaces with other plant systems.
- d. An evaluation of the change which shows the predicted quantity of solid waste that differs from those previously predicted in the license application and amendments thereto.
- e. An evaluation of the change which shows the expected maximum exposures to individuals in the UNRESTRICTED AREA and to the general population that differ from those previously estimated in the license application and amendments thereto.
- f. A comparison of the predicted releases of radioactive materials in solid waste to the actual releases for the period prior to when the changes are to be made.
- g. An estimate of the exposure to plant operating personnel as a result of the change.
- h. Documentation of the fact that the change was reviewed and found acceptable by the PRB.

TABLE 1  
VOGTLE ELECTRIC GENERATING PLANT  
RADIOACTIVE WASTE STREAMS

WASTE STREAM	SOURCE	PROCESSING SYSTEM
SPENT RESINS		
Mixed bed bead	CVCS mixed bed demineralizer Recycle evaporator feed demineralizers Recycle evaporator condensate demineralizers Steam generator blowdown mixed bed demineralizers Waste evaporator condensate demineralizer Waste monitor tank demineralizer Spent fuel pool demineralizer Turbine building drain system demineralizer	Vendor dewatering
Cation Bead Resin (Borated)	CVCS cation bed demineralizer	Vendor dewatering
Anion Bead Resin (Borated)	Boron thermal regeneration demineralizer	Vendor dewatering
Powdex Resin	Condensate polisher demineralizers	Vendor dewatering
Ecodex Resin	Vendor supplied micro-filtration unit	Vendor dewatering
Mixed Bed/Cation, Anion Bead Resin	Vendor supplied liquid waste processing system	Vendor dewatering
EVAPORATOR CONCENTRATES		
Boric Acid (6-12 wt percent)	Waste evaporator bores, recycle evaporator	Vendor solidification
CONTAMINATED OIL		
Oil (various weights and grades)	Maintenance, oil spills, etc.	Filtration or licensed waste processor

TABLE 1 (CONTINUED)

DRY ACTIVE WASTE

Compactible	Paper, rags, trash, etc.	Compactor, or licensed waste processor
Noncompactible	Piping, valves, tubing, etc.	Packaged in DOT package, or licensed waste processor

FILTER CRUD

Filter Particulates	Vasco backflushable filters	Vendor supplied liquid waste processing equipment, vendor solidification
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CARTRIDGE FILTERS

Mechanical Cartridge Filters	Vendor supplied liquid waste processing system, spent fuel pool purifi- cation, reactor cavity filtration, etc.	High integrity container
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TABLE 2

VOGTLE ELECTRIC GENERATING PLANT  
PROCEDURES GOVERNING THE  
PROCESSING, CLASSIFICATION, SHIPPING,  
AND DISPOSAL OF RADIOACTIVE WASTE

46023-C	Dry Active Waste Sorting, Segregation, Monitoring and Processing
46100-C	10 CFR 61 Waste Classification Program
46101-C	Dry Waste Processing
46102-C	Operation of Support Systems in the Dry Active Waste Processing and Storage Facilities
46104-C	Shipment of Radwaste to a Licensed Waste Processor
46105-C	Radwaste Disposal and Notification Requirements
46106-C	Waste Classification Resin Shipment
46107-C	Waste Classification DAW Shipments
46108-C	Waste Classification Filter Shipments
46109-C	Waste Classification Miscellaneous Waste Shipments
46110-C	Shipment of Radioactive Waste
46111-C	Contaminated Oil Processing
46112-C	Spent Filter Processing
13275-C	Mobile Solidification
13285-C	Spent Resin Processing
31000-C	Chemistry Quality Assurance and Control Program
33035-C	Gamma Spectroscopy for Radiochemistry
36025-C	Semi Annual Effluent Release Report Preparation
00100-C	Quality Assurance Records Administrative

TABLE 2 (CONTINUED)

00161-C	Permits and Licenses
00256-C	Radioactive Waste Minimization Program

III

GEORGIA POWER COMPANY

VOGTLE ELECTRIC GENERATING PLANT - UNITS 1 & 2

NRC DOCKET NOS. 50-424 AND 50-425

FACILITY OPERATING LICENSE NOS. NPF-68 AND NPF-81

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IV

GEORGIA POWER COMPANY

VOGTLE ELECTRIC GENERATING PLANT - UNITS 1 & 2

NRC DOCKET NOS. 50-424 AND 50-425

FACILITY OPERATING LICENSE NOS. NPF-68 AND NPF-81

1993 ANNUAL REPORT - PART 2

ANNUAL ENVIRONMENTAL (NON-RADIOLOGICAL) OPERATING REPORT

for

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