TABLE OF CONTENTS

Section	Title	Page
12.0	RADIATION PROTECTION	12.1-1
12.1	RADIATION SHIELDING	12.1-4
12.1.1	DESIGN OBJECTIVES	12.1-4
12.1.2	DESIGN DESCRIPTION	12.1-5
12.1.2.1	Plant Shielding Description	12.1-6
12.1.2.2	Plant Shielding Design Criteria	12.1-12
12.1.2.3	Accident Shielding	12.1-14
12.1.2.4	Shielding Calculational Methods	12.1-14
12.1.3	SOURCE TERMS	12.1-14
12.1.3.1	Sources for Normal Full Power Operation	12.1-15
12.1.3.2	Sources for Shutdown	12.1-20
12.1.3.3	Sources Under Accident Conditions	12.1-22
12.1.4	AREA MONITORING	12.1-22
12.1.4.1	Design Basis	12.1-22
12.1.4.2	System Description	12.1-22
12.1.5	OPERATING PROCEDURES	12.1-24
12.1.6	ESTIMATES OF EXPOSURE	12.1-25
12.1.6.1	Dose Rates at Selected Inplant Locations	12.1-26
12.1.6.2	Estimates of Personnel Occupancy Requirements	12.1-26
12.1.6.3	Estimates of Annual Man-Rem Doses	12.1-26
12.1.6.4	Doses at the Radiation Controlled Area Boundary	12.1-28
12.1.7	REFERENCES	12.1-29
12.2	VENTILATION	12.2-1
12.2.1	DESIGN OBJECTIVES	12.2-1
12.2.2	DESIGN DESCRIPTION	12.2-1
12.2.3	SOURCE TERMS	12.2-2
12.2.3.1	Assumptions Used in Determining Airborne Activity in the Reactor Building	12.2-3
12.2.3.2	Assumptions Used in Determining Airborne Activity in the Auxiliary Building	12.2-3
12.2.3.3	Assumptions Used in Determining Airborne Activity in the Turbine Building	12.2-3
12.2.3.4	Assumptions Used in Determining Airborne Activity in Other Buildings	12.2-4
12.2.4	AIRBORNE RADIOACTIVITY MONITORING	12.2-4
12.2.4.1	Design Basis	12.2-4
12.2.4.2	Airborne Radioactivity Monitors System Description	12.2-4
12.2.4.3	Detection of MPC_a Levels of Airborne Radioiodine and Particulates	12.2-10
12.2.4.4	Reactor Coolant Leak Detection	12.2-12
12.2.5	OPERATING PROCEDURES	12.2-13
12.2.6	ESTIMATES OF INHALATION DOSES	12.2-15
12.2.6.1	Peak Airborne Concentrations and Inhalation Dose Rates	12.2-15
12.2.6.2	Methods Used to Estimate Airborne Concentrations and Inhalation Doses	12.2-17
12.2.7	REFERENCES	12.2-18

TABLE OF CONTENTS (Continued)

Section	Title	Page
12.3	HEALTH PHYSICS PROGRAM	12.3-1
12.3.1	OBJECTIVES AND ORGANIZATION	12.3-1
12.3.1.1	Program Objectives	12.3-1
12.3.1.2	Program Organization	12.3-3
12.3.2	FACILITIES, EQUIPMENT AND PROCEDURES	12.3-5
12.3.2.1	Facilities	12.3-5
12.3.2.2	Equipment	12.3-6
12.3.2.3	Health Physics Program: Procedures	12.3-9
12.3.3	PERSONNEL DOSIMETRY	12.3-16
12.3.3.1	External Radiation Exposure	12.3-16
12.3.3.2	Internal Radiation Exposure	12.3-16
Appendix 12A	DESIGN REVIEW OF PLANT SHIELDING AND ENVIRONMENTAL QUALIFICATION OF EQUIPMENT FOR SPACES/SYSTEMS WHICH MAY BE USED IN POST ACCIDENT OPERATIONS OUTSIDE THE REACTOR BUILDING AT THE VIRGIL C. SUMMER NUCLEAR STATION	12A-1
12A.1	INTRODUCTION	12A-1
12A.2	SUMMARY	12A-2
12A.3	SOURCE TERMS AND CALCULATIONAL METHODOLOGY	12A-3
12A.3.1	SOURCE TERMS	12A-3
12A.3.1.1	Basis for the Source Terms	12A-3
12A.3.1.2	Liquid Source Terms	12A-4
12A.3.1.3	Gas Source Terms	12A-4
12A.3.2	METHODOLOGY	12A-6
12A.3.2.1	Calculation of Dose Rates	12A-6
12A.3.2.2	Calculation of Doses to Personnel During Post Accident Access to Vital Areas	12A-6
12A.3.2.3	Calculation of Integrated Doses to Safety Equipment	12A-6
12A.4	REVIEW OF AREAS REQUIRING ACCESS FOR POST ACCIDENT OPERATIONS	12A-11
12A.4.1	AREA A: RHR/SPRAY PUMP ROOMS AND COOLING UNITS	12A-11
12A.4.2	AREA B: CHARGING/SI PUMP ROOMS	12A-11
12A.4.3	AREA C: CHARGING/SI PUMP C AUXILIARIES LOCAL CONTROL PANEL	12A-12
12A.4.4	AREA D: SPENT FUEL PIT HEAT EXCHANGER	12A-13
12A.4.5	AREA E: RADWASTE GAS HANDLING CONTROL PANELS	12A-13
12A.4.6	AREA F: SPENT FUEL PIT COOLING PUMPS AND LOCAL CONTROL PANELS	12A-13
12A.4.7	AREA G: ENGINEERED SAFETY FEATURE MOTOR CONTROL CENTER	12A-14
12A.4.8	AREA H: WASTE PROCESSING SYSTEM CONTROL PANEL	12A-14
12A.4.9	AREA I: CHARGING/SAFETY INJECTION PUMP COOLING UNITS	12A-15
12A.4.10	AREA J: ENGINEERED SAFETY FEATURE SUBSTATION BUS AND MOTOR CONTROL CENTER	12A-16
12A.4.11	AREA K: HYDROGEN RECOMBINER CONTROL AND POWER SUPPLY PANELS	12A-16
12A.4.12	AREA L: HYDROGEN RECOMBINER CONTROL AND POWER SUPPLY PANELS	12A-17

TABLE OF CONTENTS (Continued)

12A.4.13-1	AREA M1: REACTOR BUILDING ATMOSPHERE SAMPLE COLLECTION POINT AND HYDROGEN GAS ANALYZER	12A-17
12A.4.13-2	AREA M2: REACTOR BUILDING ATMOSPHERE SAMPLE COLLECTION POINT AND HYDROGEN GAS ANALYZER	12A-18
12A.4.14	AREA N: TECHNICAL SUPPORT CENTER	12A-18
12A.4.15	AREA O: CONTROL ROOM	12A-18
12A.4.16	AREA P: SAMPLE ROOM	12A-19
12A.4.17	AREA Q: RADIOCHEMISTRY LAB	12A-19
12A.4.18	AREA R: COUNT ROOM	12A-19
12A.4.19	INTERMEDIATE BUILDING	12A-20
12A.4.20	EMERGENCY OPERATIONS FACILITY	12A-21
12A.5	REVIEW OF RADIATION QUALIFICATION OF SAFETY RELATED EQUIPMENT	12A-32
12A.6	REFERENCES	12A-32

LIST OF TABLES

Table	Title	Page	
12.1-1	Dose Limits for Occupational Workers	12.1-31	
12.1-1a	Radiation Shielding Thicknesses	12.1-32	
12.1-2	Radiation Source Terms for Demineralizers	12.1-34	
12.1-3	Radiation Source Terms for Tanks	12.1-35	
12.1-4	Radiation Source Terms for Filters	12.1-37	
12.1-5	Radiation Source Terms for Heat Exchangers	12.1-39	
12.1-6	Neutron Source Term for Primary Concrete Shield	12.1-40	
12.1-7	Gamma Source Term for Primary Concrete Shield	12.1-41	
12.1-8	Pressurizer Deposited Activity	12.1-42	
12.1-9	Reactor Coolant N^{16} Activity	12.1-43	
12.1-10	Radiation Source Term - Letdown Coolant	12.1-44	
12.1-11	Solid Waste Source for Drumming Station	12.1-45	
12.1-12	Core Average Gamma Ray Source Strengths at Various Times After Shutdown	12.1-46	
12.1-13	Irradiated Ag-In-Cd Control Rod Source Strengths	12.1-47	RN 02-025
12.1-14	Irradiated Incore Detector Drive Cable Sources Strengths	12.1-48	
12.1-15	Deposited Corrosion Product Activity on Steam Generator Surfaces	12.1-49	
12.1-16	Residual Heat Removal Loop Shutdown Source	12.1-50	
12.1-17	Deleted RN 02-025		
12.1-17a	Deleted RN 02-025		RN 02-025
12.1-18	Deleted RN 02-025		
12.1-19	Area Radiation Monitors	12.1-51	
12.1-20	Radiation Dose Rates in Selected Inplant Locations	12.1-54	
12.1-21	Estimates of Personnel Occupancy Requirements in Specific Radiation Zones	12.1-55	
12.1-22	Man-Rem Estimates for Normal Plant Operation, Anticipated Operational Occurrences and Routine Maintenance	12.1-56	
12.1-22a	Man-Rem Dose Per Task - Normal Operation	12.1-57	
12.1-22b	Man-Rem Dose Per Task - Refueling Outage	12.1-58	
12.1-22c	Percent Time of Worker Effort Per Task - Normal Operation	12.1-59	
12.1-22d	Percent Time of Worker Effort Per Task - Refueling Outage	12.1-60	
12.1-22e	Calculated Man-Rem/year Per Identified Task	12.1-62	
12.1-22f	Estimated Radiation Dose Rates From Airborne Contamination in Specific Areas of the Plant	12.1-64	
12.1-22g	Estimated Man-Rem Doses From Airborne Contamination	12.1-65	

LIST OF TABLES (Continued)

Table	Title	Page
12.1-23	Deleted by Amendment 3	
12.1-24	Estimated Post LOCA Response of Containment Monitor RM-G7	12.1-66
12.2-1	Calculated Airborne Concentrations	12.2-19
12.2-2	Air Flow Rates and Approximate Net Free Volumes Served by Ventilation Systems	12.2-20
12.2-3	Airborne Radioactivity Monitors	12.2-21
12.2-4	Estimated Detection Time for I-131 and Cs-137 in the Fuel Handling Building	12.2-24
12.3-1	Contamination Control Monitors	12.3-17
12.3-2	Fixed Laboratory Instrumentation	12.3-18
12.3-3	Portable Health Physics Instruments	12.3-20
12.3-4	Respiratory Protection Equipment	12.3-22
12A.3-1	Containment Sump Minimum Liquid Inventory	12A-7
12A.3-2	V. C. Summer Shielding Source Terms (T = 0)	12A-8
12A.4-1	Exposure Rates For Vital Areas As a Function of Time Exposure Rate (mr/hr)	12A-22
12A.4-2	Vital Area Radiation Dose Summary	12A-25
12A.4-3	Access Routes to Vital Areas	12A-30
12A.4-4	Deleted Per RN 05-025	

RN
05-025

LIST OF FIGURES

Figure	Title	
12.1-0	Reference Legend For Section 12.1 Figures	02-01
12.1-1	Auxiliary and Reactor Buildings - Radiation Exposure Zones for Normal Operation Plan at Elevation 374'-0"	
12.1-2	Auxiliary and Reactor Buildings - Radiation Exposure Zones for Normal Operation Plan at Elevations 388'-0" and 397'-0"	
12.1-2a	Auxiliary Building - Radiation Exposure Zones for Piping Shield Slabs at Elevations 400'-0", 426'-6", 446'-0" and 448'-0" Normal Operation	
12.1-3	Auxiliary, Reactor and Fuel Handling Buildings - Radiation Exposure Zones for Normal Operation Plan at Elevation 412'-0"	
12.1-4	Intermediate Building - Radiation Exposure Zones for Normal Operation Plan at Elevation 412'-0"	
12.1-5 (1 of 2)	Auxiliary, Reactor and Fuel Handling Buildings - Radiation Exposure Zones for Normal Operation Plan at Elevation 436'-0"	
12.1-5 (2 of 2)	Drumming Station and Hot Machine Shop - Radiation Exposure Zones Plan at Elevations 436'-0" and 447'-0"	
12.1-6	Intermediate Building - Radiation Exposure Zones for Normal Operation Plan at Elevation 436'-0"	
12.1-7	Auxiliary, Reactor and Fuel Handling Buildings - Radiation Exposure Zones for Normal Operation Plan at Elevation 463'-0"	
12.1-8	Reactor and Fuel Handling Buildings, Section Looking West - Radiation Exposure Zones for Normal Operation	
12.1-9	Auxiliary and Reactor Buildings, Section Looking North - Radiation Exposure Zones for Normal Operation	
12.1-10	Auxiliary and Reactor Buildings - Radiation Exposure Zones for Shutdown/Refueling Plan at Elevation 374'-0"	
12.1-11	Auxiliary and Reactor Buildings - Radiation Exposure Zones for Shutdown/Refueling Plan at Elevations 388'-0" and 397'-0"	
12.1-11a	Auxiliary Building - Radiation Exposure Zones for Piping Shielding Slabs at Elevations 400'-0", 426'-6", 446'-0", and 448'-0" Shutdown/Refueling	
12.1-12	Auxiliary, Reactor and Fuel Handling Buildings - Radiation Exposure Zones for Shutdown/Refueling Plan at Elevation 412'-0"	
12.1-13	Intermediate Building - Radiation Exposure Zones for Shutdown/Refueling Plan at Elevation 412'-0"	
12.1-14	Auxiliary, Reactor and Fuel Handling Buildings - Radiation Exposure Zones for Shutdown/Refueling Plan at Elevation 436'-0"	
12.1-15	Intermediate Building - Radiation Exposure Zones for Shutdown/Refueling Plan at Elevation 436'-0"	
12.1-16	Auxiliary, Reactor and Fuel Handling Buildings - Radiation Exposure Zones for Shutdown/Refueling Plan at Elevation 463'-0"	

LIST OF FIGURES (Continued)

Figure	Title
12.1-17	Reactor and Fuel Handling Buildings, Section Looking West - Radiation Exposure Zones for Shutdown/Refueling
12.1-18	Auxiliary and Reactor Buildings, Section looking North - Radiation Exposure Zones for Shutdown/Refueling
12.1-19	Control Complex Plan at Elevations 412'-0", 436'-0", 463'-0" and 482'-0"
12.1-19a	Control Building - Radiation Exposure Zones for Normal Operation, Plan Cable Spreading Floor
12.1-20	Control Complex Sections Radiation Exposure Zones
12.1-21	Primary Shield Streaming Paths
12.1-22	Fuel Transfer Canal Shielding
12.1-23	Deleted
12.1-24	Deleted
12.2-1	Control Room Emergency Charcoal Filter Plenum
12.2-2	Airborne Activity Monitoring
12.2-3	Buildup of Predominate Particulates in Reactor Building From 1 gpm Reactor Coolant Leak as a Function of Time After Leakage Begins RN 13-011
12.2-4	Buildup of XE-133 in Reactor Building From 1 gpm Reactor Coolant Leak as a Function of Time After Leakage Begins
12.3-1	Health Physics Organization Chart
12A.4-1	Auxiliary and Reactor Buildings, Plan Above Elevation 374'-0"
12A.4-2	Auxiliary and Reactor Buildings, Plan Above Elevations 388'-0" and 397'-0"
12A.4-3	Auxiliary, Reactor and Fuel Handling Buildings, Plan Above Elevation 412'-0"
12A.4-4	Auxiliary Building Miscellaneous Plans, Piping Shields Slabs
12A.4-5	Auxiliary, Reactor and Fuel Handling Buildings, Plan Above Elevation 436'-0"
12A.4-6	Auxiliary, Reactor and Fuel Handling Buildings, Plan Above Elevation 463'-0"
12A.4-7	Control Complex, Plan at Elevations 412'-0", 436'-0", 463'-0" and 482'-0"
12A.4-8	Intermediate Building and Diesel Generator Building, Plan Above Basement Floor Elevation 412'-0"

LIST OF EFFECTIVE PAGES (LEP)

The following list delineates pages to Chapter 12 of the Virgil C. Summer Nuclear Station Final Safety Analysis Report which are current through March 2015. The latest changes to pages and figures are indicated below by Revision Number (RN) in the Amendment column along with the Revision Number and date for each page and figure included in the Final Safety Analysis Report.

Page/Fig.No.		Amend. No.	Date	Page/Fig.No.		Amend. No.	Date
Page	12-i	Reset	March 2015	Page	12.1-24	00-01	December 2000
	12-ii	Reset	March 2015		12.1-25	RN01-119	June 2006
	12-iii	Reset	March 2015		12.1-26	00-01	December 2000
	12-iv	Reset	March 2015		12.1-27	00-01	December 2000
	12-v	Reset	March 2015		12.1-28	00-01	December 2000
	12-vi	Reset	March 2015		12.1-29	00-01	December 2000
	12-vii	Reset	March 2015		12.1-30	00-01	December 2000
	12-viii	Reset	March 2015		12.1-31	RN06-017	December 2008
	12-ix	Reset	March 2015		12.1-32	RN03-038	November 2011
	12-x	Reset	March 2015		12.1-33	RN03-038	November 2011
Page	12.1-1	02-01	May 2002		12.1-34	RN03-038	November 2011
	12.1-2	00-01	December 2000		12.1-35	02-01	May 2002
	12.1-3	02-01	May 2002		12.1-36	RN03-038	November 2011
	12.1-4	RN00-058	December 2000		12.1-37	RN03-038	November 2011
	12.1-5	RN12-034	July 2014		12.1-38	02-01	May 2002
	12.1-6	00-01	December 2000		12.1-39	02-01	May 2002
	12.1-7	00-01	December 2000		12.1-40	02-01	May 2002
	12.1-8	00-01	December 2000		12.1-41	02-01	May 2002
	12.1-9	RN12-034	July 2014		12.1-42	02-01	May 2002
	12.1-10	RN12-034	July 2014		12.1-43	02-01	May 2002
	12.1-11	00-01	December 2000		12.1-44	02-01	May 2002
	12.1-12	00-01	December 2000		12.1-45	RN08-003	January 2011
	12.1-13	00-01	December 2000		12.1-46	RN02-025	July 2002
	12.1-14	00-01	December 2000		12.1-47	RN02-025	July 2002
	12.1-15	00-01	December 2000		12.1-48	RN02-025	July 2002
	12.1-16	00-01	December 2000		12.1-49	02-01	May 2002
	12.1-17	00-01	December 2000		12.1-50	02-01	May 2002
	12.1-18	RN07-037	November 2011		12.1-51	RN02-025	July 2002
	12.1-19	RN03-038	November 2011		12.1-52	02-01	May 2002
	12.1-20	00-01	December 2000		12.1-53	RN02-025	July 2002
	12.1-21	00-01	December 2000		12.1-54	02-01	May 2002
	12.1-22	RN02-025	July 2002		12.1-55	02-01	May 2002
	12.1-23	00-01	December 2000		12.1-56	02-01	May 2002

Page/Fig.No.		Amend. No.	Date	Page/Fig.No.		Amend. No.	Date
Page	12.1-57	02-01	May 2002	Fig.	12.1-22	02-01	May 2002
	12.1-58	02-01	May 2002	Page	12.2-1	RN02-002	October 2002
	12.1-59	02-01	May 2002		12.2-2	RN01-119	June 2006
	12.1-60	02-01	May 2002		12.2-3	RN02-002	October 2002
	12.1-61	02-01	May 2002		12.2-4	RN01-119	June 2006
	12.1-62	02-01	May 2002		12.2-5	RN04-023	January 2005
	12.1-63	02-01	May 2002		12.2-6	RN13-022	April 2014
	12.1-64	02-01	May 2002		12.2-7	RN13-022	April 2014
	12.1-65	02-01	May 2002		12.2-8	RN01-119	June 2006
	12.1-66	02-01	May 2002		12.2-9	RN03-011	April 2006
Fig.	12.1-0	02-01	May 2002		12.2-10	RN01-119	June 2006
	12.1-1	2	August 1986		12.2-11	RN13-022	April 2014
	12.1-2	99-01	June 1999		12.2-12	RN13-022	April 2014
	12.1-2a	02-01	May 2002		12.2-13	00-01	December 2000
	12.1-3	99-01	June 1999		12.2-14	00-01	December 2000
	12.1-4	99-01	June 1999		12.2-15	RN01-119	June 2006
	12.1-5 (pg 1)	02-01	May 2002		12.2-16	RN01-111	November 2001
	12.1-5 (pg 2)	RN03-041	April 2004		12.2-17	00-01	December 2000
	12.1-6	93-10	December 1993		12.2-18	00-01	December 2000
	12.1-7	00-01	December 2000		12.2-19	RN01-111	November 2011
	12.1-8	2	August 1986		12.2-20	RN01-119	June 2006
	12.1-9	5	August 1989		12.2-21	RN03-011	April 2006
	12.1-10	2	August 1986		12.2-22	RN03-011	April 2006
	12.1-11	99-01	June 1999		12.2-23	RN03-011	April 2006
	12.1-11a	02-01	May 2002		12.2-24	RN01-119	June 2006
	12.1-12	99-01	June 1999	Fig.	12.2-1	0	August 1984
	12.1-13	99-01	June 1999		12.2-2	RN01-119	June 2006
	12.1-14	02-01	May 2002		12.2-3	RN13-022	April 2014
	12.1-15	99-01	June 1999		12.2-4	RN13-022	April 2014
	12.1-16	00-01	December 2000	Page	12.3-1	RN13-005	April 2013
	12.1-17	2	August 1986		12.3-2	RN13-005	April 2013
	12.1-18	02-01	May 2002		12.3-3	RN99-129	March 2000
	12.1-19	00-01	December 2000		12.3-4	RN99-129	March 2000
	12.1-19a	RN02-011	May 2002		12.3-5	RN14-034	March 2015
	12.1-20	0	August 1984		12.3-6	RN14-034	March 2015
	12.1-21	0	August 1984		12.3-7	RN14-034	March 2015

LIST OF EFFECTIVE PAGES (Continued)

	Page/Fig.No.	Amend. No.	Date
Page	12.3-8	RN14-034	March 2015
	12.3-9	RN99-162	June 2000
	12.3-10	RN99-164	January 2001
	12.3-11	RN14-034	March 2015
	12.3-12	RN99-129	March 2000
	12.3-13	RN99-129	March 2000
	12.3-14	RN99-165	July 2000
	12.3-15	RN99-129	March 2000
	12.3-16	RN06-017	August 2008
	12.3-17	RN99-164	January 2001
	12.3-18	RN14-034	March 2015
	12.3-19	RN14-034	March 2015
	12.3-20	RN14-034	March 2015
	12.3-21	RN14-034	March 2015
	12.3-22	02-01	May 2002
Fig.	12.3-1	02-01	May 2002
Page	12A-1	99-01	June 1999
	12A-2	99-01	June 1999
	12A-3	99-01	June 1999
	12A-4	99-01	June 1999
	12A-5	99-01	June 1999
	12A-6	RN05-011	June 2007
	12A-7	RN03-008	June 2003
	12A-8	RN05-011	June 2007
	12A-9	RN05-011	June 2007
	12A-10	RN05-011	June 2007
	12A-11	RN05-011	June 2007
	12A-12	RN05-011	June 2007
	12A-13	00-01	December 2000
	12A-14	RN05-011	June 2007
	12A-15	00-01	December 2000
	12A-16	RN05-011	June 2007
	12A-17	00-01	December 2000
	12A-18	RN05-011	June 2007
	12A-19	RN05-025	January 2006
	12A-20	RN05-025	January 2006
Page	12A-21	RN10-029	November 2011
	12A-22	RN05-011	June 2007
	12A-23	RN05-011	June 2007
	12A-24	02-01	May 2002
	12A-25	RN05-011	June 2007
	12A-26	RN05-011	June 2007
	12A-27	RN05-011	June 2007
	12A-28	RN05-011	June 2007
	12A-29	RN05-011	June 2007
	12A-30	99-01	June 1999
	12A-31	99-01	June 1999
	12A-32	99-01	June 1999
Fig.	12A.4-1	02-01	May 2002
	12A.4-2	02-01	May 2002
	12A.4-3	02-01	May 2002
	12A.4-4	02-01	May 2002
	12A.4-5	02-01	May 2002
	12A.4-6	00-01	December 2000
	12A.4-7	00-01	December 2000
	12A.4-8	RN11-027	May 2013

NOTE 12.0

Section 12.0 is being retained for historical purposes only. 02-01

12.0 RADIATION PROTECTION

Paragraph 20.1(c) of 10 CFR 20, states that licensees should, in addition to complying with the limits set forth in that part, make every reasonable effort to maintain radiation exposures as far below the limits as practicable. The as low as practicable philosophy was followed in designing the Virgil C. Summer Nuclear Station.

This philosophy is reflected in the design by layout recommendations for pipes carrying radioactive fluids, equipment design, handling, and maintenance features. For example, pipe runs carrying radioactive fluid are arranged to minimize potential crud traps. Lines which transport spent resin are arranged with five diameter bends and continuous sloping. Piping vent and drain locations are minimized to reduce possible leakage paths and crud traps. Where possible, vents and drains are consolidated so that one vent and one drain serve several lines. Components that may require periodic maintenance are shielded from components most likely to be high radiation sources. Shielding is provided for fixed sample vessels associated with equipment stream pumps. Canned pumps are used in the liquid waste processing system (LWPS) to minimize potential leakage of radioactive fluids. Pumps requiring local start-stop or jog switches are placed in shielded cubicles. Control switches for such pumps are located outside the cubicles. Filters have lifting bails located on the heads to facilitate cartridge assembly removal and disposable cartridges for ease of changing. Demineralizers in the LWPS are considered to be potentially radioactive and are located inside individually shielded cubicles. Tanks containing radioactive waste are located in watertight compartments to contain the tank contents in the event of leakage. Heat exchangers incorporate design features that minimize exposure time during maintenance. Valves are located so that operation and maintenance can be performed from standard service equipment. Valves which normally carry radioactive fluid have leakoffs.

Equipment and/or components that may require servicing are located or designed to be movable to the lowest practicable radiation field. Penetrations through shielding walls are designed to minimize exposure. Radiation sources and normally occupied areas are separated. Pipes or ducts containing potentially highly radioactive fluids either do not pass through occupied areas or are shielded by shield slabs. Provisions are made for flushing equipment prior to maintenance where practicable. Ventilation systems are designed for easy access and service to minimize doses during maintenance, decontamination, filter changes, etc. Remote handling equipment is provided wherever practicable. Movable shields and means for their utilization are available for use as practicable. Where practicable, shielding is provided between radiation sources and areas requiring normal or routine access. As a result of operating experience, permanent removable shielding has been incorporated to minimize radiation exposure for operations and maintenance personnel in local areas such as for parts of the Residual Heat Removal and Safety Injection piping.

General guidance on the radiation protection aspects of the balance of plant design (BOP) is provided to engineers and designers by the appropriate discipline project engineer. Specific guidelines concerning BOP design are provided to the individual engineers whose specialties are radiation protection. These guidelines are based upon accumulated indepth design experience available to the architect-engineer from previously completed nuclear projects, as well as the experience acquired from frequent visits to operating nuclear plants. In addition to the design experience of the architect-engineer, conceptual shielding design guidelines are also provided from References [3], [8], and [9] (see Section 12.1.7).

BOP design is reviewed by both the architect-engineer and SCE&G personnel who are competent in radiation protection.

The review by the architect-engineer is accomplished through internal circulation of drawings and other design documents. When these documents contain design information that affects the radiation exposures to which plant workers will be subjected, they are reviewed by a group of specialists competent in radiation protection. This review results in the elimination or mitigation, to the extent practicable, of design features that would result in unnecessary radiation exposure. The individuals who have performed such reviews for the Virgil C. Summer Nuclear Station have the following position titles: Supervising Engineer, Project Engineer, Nuclear Engineer, Staff Health Physicist. Their qualifications include either a B. S. or M. S. degree in nuclear engineering, health physics or a related science, and six to ten years of experience related to radiation protection.

Westinghouse is responsive to the ALARA requirements of Regulatory Guide 8.8 and has devoted considerable effort via design and technology to minimize occupational radiation exposures due to equipment/systems/components within Westinghouse scope. Although Westinghouse does not charge the design engineers whose expertise lies in fluid dynamics, thermodynamics, etc. with radiation protection functions, Westinghouse does employ system analysis engineers, competent in the area of health physics and radiation protection, to work with those engineers in their disciplines of component or system design. Together, these personnel participate in the design review process in a systematic manner.

Following the review of design by the architect-engineer, and NSSS vendor, the design documents are similarly reviewed through internal circulation by the Applicant's Engineering and Operations groups. Comments or changes required as a result of these reviews are incorporated or resolved prior to the issuance of the documents for construction. This review is accomplished by various engineers cognizant of radiation protection considerations assigned to the Nuclear Engineering Group, by the Staff Health Physicist, and by Station Health Physics. Qualifications of these reviewing personnel include appropriate education/degrees and experience ranging from 5 to 10 years.

Specific examples of BOP areas reviewed during the design review process and changes made in the BOP as a result of the design review include:

1. Radiation Shielding

   a. A permali shield for the push-pull rod for the out-of-core detector at the 180° reactor vessel position was added to prevent neutron streaming to an area outside the secondary shield.

   b. The gap between the reactor building and the fuel handling building was shielded to prevent streaming when spent fuel elements are transferred from the reactor building to the fuel handling building.

   c. The wall separating the mixed bed demineralizers in the chemical and volume control system (CVCS) was increased in thickness to lower the potential dose from the adjacent demineralizer.

   d. The sample sink was relocated to make it possible to put the sample lines and the sample vessels in a shielded chase.

2. Facility and Equipment Design

   a. The original design of the solid radwaste area, which did not include provisions for separation of high and low level waste, was modified to provide separate storage areas.

   b. The decontamination area in the hot machine shop was increased and additional equipment (a turbolator and additional ultrasonic sinks) was added to provide more capability for equipment and tool decontamination.

   c. The design previously included a provision for processing blowdown with minimal radioactivity through the cycle makeup demineralizers. This provision was deleted to preclude the possibility of a significant radiation source occurring in an unlimited access area.

3. Location of Instrumentation

   Two (2) monitors were relocated to lower radiation areas. Radiation monitor RM-L1 was moved from a CVCS piping area to a lower radiation area. Radiation Monitor RM-L6 was moved from a valve gallery to an operator access corridor. 02-01

Examples of design features developed by the NSSS vendor and other features in Westinghouse PWR’s that reduce occupational radiation exposure are given in WCAP-8872, "Design, Inspection, Operation and Maintenance Aspects of the Westinghouse NSSS to Maintain Occupational Exposures as Low as Reasonably Achievable, edited by R. J. Lutz, April, 1977."

These radiation protection design considerations have resulted in satisfying 10 CFR 20. Thus, the basic intent of Regulatory Guide 8.8 concerning as low as reasonably achievable (ALARA) is satisfied. It is the policy of the management of SCE&G that future design and construction also be accomplished in such a manner as to maintain occupational radiation exposures ALARA.

12.1 RADIATION SHIELDING

Radiation protection of plant operating personnel is accomplished by the use of adequate shielding against radiation. This radiation shielding is provided principally by concrete walls, floors, and ceilings, the thickness of which was determined by the need for access to the area, the type of radiation sources present, and structural requirements. Therefore, where structural strength is the controlling criteria, the radiation shielding provided is in excess of requirements.

NOTE 12.1.1
Section 12.1.1 is being retained for historical purposes only.

RN 00-058

12.1.1 DESIGN OBJECTIVES

The primary objective of the radiation protection systems, shielding design, and administrative controls is protection of operating personnel and the general public from potential radiation sources in the reactor, radwaste systems, and other auxiliary systems, including associated equipment and piping.

During normal operation, including anticipated operational occurrences, the design objectives for the radiation shielding are as follows:

1. To restrict the quarterly and annual doses to plant operating personnel and visiting radiation workers to within the limits set forth in 10 CFR 20. These limits are given in Table 12.1-1. Maximum whole body exposure rates are generally less than 100 mrem/hr, except in the case of certain maintenance, inspection, and refueling functions. The maximum dose rate in each instance is determined by the 10 CFR 20 quarterly restriction, by the required exposure time and by the previous and anticipated subsequent doses during the quarter.

2. To limit onsite whole body doses to nonoccupational workers and site visitors to 500 mrem/yr, with a maximum whole body dose during any given seven consecutive day period of 100 mrem. These personnel are not permitted in areas where whole body exposure rates are greater than 2.0 mrem/hr.

3. To ensure that the integrated offsite dose is within the limits specified by 10 CFR 20.

4. To protect certain components from excessive radiation damage or activation.

In the unlikely event of an accident, the radiation shielding design objectives are as follows:

1. To satisfy the requirements of 10 CFR 50, Appendix A, Criterion 19, i.e., to provide adequate radiation protection for operating personnel following a reactor accident such that the accident may be terminated without excessive radiation exposure to the operators or the general public.

2. To limit offsite doses from both contained and released radioactivity to within the limitations set forth in 10 CFR 50.67. RN 12-034

The above shielding objectives served as an upper limit during design of the radiation shielding. The philosophy followed during design and construction of the plant has been to give extensive consideration to the shielding design to assure that radiation exposures to personnel are kept as low as is reasonably achievable (ALARA). This philosophy has resulted in a shielding design that results in occupational and general public exposures being well within the design objectives previously listed. During the operational phase of plant life, this policy of maintaining radiation exposures at a minimum will be continued.

12.1.2 DESIGN DESCRIPTION

Scaled layouts and cross sections of the Reactor Building and surrounding shield buildings and of the Auxiliary, Control, and Fuel Handling Buildings are provided by Figures 12.1-1 through 12.1-20. While these figures do show specific layout of equipment, the purpose of presenting these figures is to show specific areas in the plant which have been zoned in accordance with maximum design radiation levels and occupancy requirements. These figures will be revised only when the boundaries or designations of the radiation zones change and will not be revised to show equipment relocations, deletions, additions, etc. Figure 1.2-1 shows the total plant layout within the site boundary, as well as identifying any outside storage areas and railroad spur or siding locations.

12.1.2.1 Plant Shielding Description

Specific areas in the plant have been zoned in accordance with maximum design radiation levels and occupancy requirements. The occupancy requirements have been based on the general access requirements (as defined by an analysis of the operation of the plant and from information obtained from operating facilities) and access requirements for other operations such as maintenance, refueling, instrument calibration, and similar reoccurring activities. An explanation of each of the five zones and the maximum dose rates anticipated are given below:

Zone	Occupancy	Dose Rate (mrem/hr)
I	Uncontrolled. No restrictions on occupancy expected.	< 1.0
II	Controlled, Unlimited access, 40 hrs/week	< 2.5
III	Controlled. Limited access, 6 to 40 hrs/week.	< 15.0
IV	Limited access for short periods, 1 to 6 hrs/week.	< 100.0
V	Controlled, high radiation area, occupancy averages less than one hr/week.	> 100.0

The radiation shielding is designed to attenuate direct and scattered radiation according to the dose limits required by this zoning.

The locations of these radiation zones throughout the plant are depicted in the layouts shown by Figures 12.1-1 through 12.1-20. An equipment list is presented as Figure 1.2-28a. Figures 12.1-1 through 12.1-9 show this radiation zoning for the plant during normal operation, while Figures 12.1-10 through 12.1-18 depict this zoning for zoning of the Control Building during normal operating and shutdown periods. The radiation sources used in determining these radiation levels are defined in Section 12.1.3 and were calculated assuming fuel cladding failures of one percent. The thickness of radiation shielding in various parts of the plant is presented in Table 12.1-1a.

Plant radiation shielding is divided into six categories: primary, secondary, Reactor Building, Control Room, fuel transfer, and Auxiliary Building shielding. Each of these categories is discussed below:

1. Primary Shield

The primary shield is a concrete structure surrounding the reactor pressure vessel. It is of varied thickness and has inspection openings (above the reactor-to-coolant pipe nozzles) and penetration ports adjacent to the core for excore neutron detectors.

The excore neutron detector push rods penetrate the primary shield at the height of the core. Where these penetrations are not inside the secondary shield, one foot thick permali (densified beechwood laminate) shield covers are provided to preclude any neutron streaming from these penetrations.

The inspection openings above the reactor-to-coolant pipe nozzles remain open during operation. They are closed during shutdown for refueling. The design of the reactor cavity and control rod missile shield is such that a direct exposure to plant personnel from these inspection openings or from the annular gap between the reactor vessel and primary shield is not possible. However, the scattering of neutron and gamma radiation from the Reactor Building dome will result in dose rates that range from 1 mrem/hr to 100 mrem/hr at the operating floor level. Measured dose rates in operating plants support this conclusion. Exposure of workers to these dose rates will be minimized since access to the operating floor during power operation is under strict administrative control.

The three basic paths by which neutron and gamma radiation may stream out of the primary shield are indicated by Figure 12.1-21, with the basic locations involved as follows:

a. Primary piping penetrations.

b. Excore neutron detector instrumentation push rods.

c. Vertical streaming of neutrons along the pressure vessel wall.

Primary piping penetrations do not contribute an appreciable amount to the dose rate in any accessible area since the penetrations are only open to the inside of the secondary compartments which are inaccessible at all times during power operation.

Neutron detector push rods penetrate the primary shield in an area which is not circumscribed by the secondary shield and neutrons of energy less than 1.0 MEV stream in iron; therefore, the COHORT-II Monte Carlo computer code with a mono-directional point source directed into the inner end of the push rod was utilized to calculate a neutron removal cross Section for this particular geometry. Neutron shield covers were designed to cover the areas where the push rods exit the primary shield. The neutron dose rate outside these shield covers is estimated to be less than 50 mrem/hr.

It is well known that neutrons emanating from the core scatter up and out of the gap between the pressure vessel and the concrete primary shield, whereupon they are free to scatter to the operating elevation inside containment. The magnitude of the neutron dose rate on the operating elevation from this neutron source is strongly dependent upon the size of the gap between the pressure vessel flange and the primary shield. The gap width between the pressure vessel flange and the primary shield is only 7.25 inches. Therefore, it may be anticipated that the neutron dose rate on the operating elevation will be rather low.

Additionally, Monte Carlo calculations have been performed utilizing the COHORT-II computer code. The main feature of the geometry of the pressure vessel, primary shield, gap, nozzles, reactor cavity, and containment were incorporated. The source was conservatively assumed to be isotropic and located in the gap on the surface of the pressure vessel over the core height. Neutrons were biased in direction (going up the gap) and in energy (toward the high energy end of the spectrum). Point fluxes and resultant dose rates were obtained on the operating level using this code. These calculation estimates indicate that the neutron dose rate will be less than 100 mrem/hr on the operating level.

Primary shielding includes the following:

a. The elements inside the reactor pressure vessel, including the core baffle, core barrel, thermal shield, and water annuli.

b. The reactor vessel wall.

c. The concrete structure surrounding the reactor vessel. This structure extends from below the floor at the 412' elevation to an elevation of 437'-2-3/4" and, at a reduced thickness except across the refueling canal, from the 437'-2-3/4" elevation up to the operating floor (elevation 463'). The shield thickness in the lower region of the primary shield ranges in thickness from 8'-8-1/4" to 13'-8-1/4", while the upper, reduced thickness portion of the shield has a thickness of 4 feet. The upper part of the primary concrete shielding is completed by the walls of the refueling canal which also extend upward to the 463' elevation. These walls have a minimum thickness of 3 feet.

   The primary shield as a whole or in part serves to:

   (1) Attenuate the neutron flux sufficiently to prevent excessive radiation damage to the reactor vessel.

   (2) Reduce the heat flux from neutron and gamma radiation at the reactor vessel outer surface such that any cooling necessary to avoid high temperatures and possible dehydration in the surrounding concrete is an easier task.

   (3) Attenuate neutron and gamma fluxes to prevent excessive radiation damage and activation of plant components and structures.

   (4) Reduce, in conjunction with the secondary shield, the radiation escaping the reactor vessel sufficiently to allow limited access to certain areas of the Reactor Building during normal full power operation.

(5) Reduce the radiation escaping the reactor vessel sufficiently to permit access inside the secondary shield, at a reasonable time after shutdown, for routine maintenance and inspection.

2. Secondary Shield

The secondary shield is a series of concrete structures which enclose the reactor coolant loops and pumps, the pressurizer and pressurizer relief tank, the steam generators, and portions of the primary shield. These structures extend from the floor at the 412' elevation to varying elevations, the highest of which is elevation 513'-6". The secondary shield is 3'-8" thick.

The main function of the secondary shield is to reduce the dose rates from the primary coolant loops such that most areas within the Reactor Building outside the secondary shield are classified as radiation Zone IV during normal full power operation and Zone III when the plant is shut down.

3. Reactor Building

The Reactor Building is a steel reinforced concrete structure consisting of a 4 foot thick cylindrical wall and a 3 foot thick shallow dome roof. This structure encloses the Nuclear Steam Supply System (NSSS) and serves to attenuate any radiation escaping the primary-secondary shield complex such that the radiation level in occupied areas outside the Reactor Building permits unlimited occupancy. In addition, this structure shields operating personnel and the general public from radiation sources resulting from activity released to the Reactor Building from postulated accidents. The general shielding considerations employed in the arrangement of the Reactor Building penetrations are discussed in Section 12.1.2.2.

4. Control Room Shielding

Shielding for the Control Room is designed to satisfy the requirements of General Design Criterion 19, i.e., adequate radiation shielding is provided to permit continuous occupancy of the Control Room under accident conditions described in Regulatory Guide 1.183 (see Appendix 3A) without personnel receiving radiation exposures in excess of 5 rem TEDE for the duration of the accident. The shielding that is provided for the control room is as follows: RN 12-034

a. Control Building walls - each 2 feet thick.

b. Control Building roof - 2 feet thick.

c. Control Room floor - 8 inches thick.

d. Control Room ceiling - 9 inches thick between column lines G and H and 2 feet thick between column lines H and H.6 (see Figure 1.2-19).

This shielding is of ordinary concrete and is adequate to attenuate radiation from external sources (e.g., activity inside the Reactor Building, airborne activity external to the Control Room, and activity in areas surrounding the Control Building) to a small fraction of the 5 rem TEDE limit in accordance with Regulatory Guide 1.183. A more significant portion of the radiation dose to Control Room personnel during a design basis accident is due to airborne activity within the Control Room. This subject is discussed in Chapter 15. RN 12-034

Figures 12.1-19 and 12.1-20 present a layout and section of the Control Room.

5. Fuel Transfer Shielding

During fuel transfer operations, the refueling canal and the region above the reactor vessel are filled with borated water to an elevation of 461.5'. The water level in the fuel transfer canal and spent fuel pool in the Fuel Handling Building is also at elevation 461.5'. This water, along with the surrounding concrete, serves as radiation shielding for the spent fuel as it is removed from the core, transferred through the refueling canal and transfer tubes, stored in the spent fuel pool, and, after a decay period, loaded into a shipping cask. This water and concrete also provide shielding from activated control rod clusters and reactor internals which are removed during refueling periods. The concentration of radioactivity in the refueling water is controlled by administrative procedures and a sufficient depth of water above the fuel assembly is maintained so that the direct exposure rate at the surface of the water does not normally exceed 2.5 mrem/hr. However, certain manipulations of fuel assemblies, rod clusters, or reactor internals may produce short term exposures in excess of 2.5 mrem/hr. The fuel transfer tube is shielded to limit doses due to radiation streaming through penetrations and gaps. Lead shielding is provided in the fuel transfer tube inspection accessway inside the Reactor Building, in the gap between the refueling cavity wall and the Reactor Building liner, and in the gap between the Reactor Building and Fuel Handling Building (see Figure 12.1-22). This shielding ensures that radiation doses do not exceed 100 mrem/hr in areas adjacent to the tube during transfer of fuel inside the Reactor Building. 02-01 02-01

Lead shielding was not provided on the underside of the 3 inch gap between the Reactor Building and the cantilevered top of the retaining wall adjacent to the Reactor Building. Access to this area will be limited during refueling operations by administrative control. Radiation levels are closely monitored during refueling operations to ensure that plant personnel doses are maintained well within the limits specified by 10 CFR 20.

6. Auxiliary Building Shielding

Shielding in the Auxiliary Building is designed to protect plant personnel from radiation from components and piping of the following systems during normal operation, including anticipated operational occurrences:

a. Chemical and Volume Control System.

b. Waste Processing Systems.

c. Solid Waste Disposal System.

d. Residual Heat Removal System.

e. Spent Fuel Cooling System.

f. Boron Recycle System.

g. Sampling Systems.

h. Nuclear Blowdown Processing System.

Auxiliary Building shielding includes concrete floors, walls, doors, covers and/or removable blocks, and local permanent shielding such as lead plates supported by steel framing. The shielding is designed to satisfy the radiation zoning requirements set forth previously.

Where practical, shielding is also provided between components to permit plant personnel to enter the equipment compartments under controlled conditions and perform required maintenance without shutdown or decontamination of adjacent compartments.

Additionally, shielding design considered the location of certain manways to components to provide optimum accessibility and the best possible work area to reduce exposure time during required maintenance.

For example, in the case of the recycle holdup tanks, entrance to the cubicle is from the 412’ elevation (bottom of tank is at elevation 388’-6") since the compartment is watertight up to a level equivalent to the maximum volume of the tank. For normal maintenance, a platform is provided around the east side of the tank. For major maintenance, such as removal of the tank diaphragm or tank cleaning, the tank will be emptied and flushed prior to the commencement of maintenance. The manway, approximately three feet above the floor (elevation 388’), is located to maximize freedom of movement into and out of the tank. The minimal increase in exposure during major maintenance to a worker walking to the manway is more than offset by the provision of the best possible work area at the manway.

12.1.2.2 Plant Shielding Design Criteria

To ensure that doses to plant personnel are kept within the limits specified in Section 12.1.1, shield walls are erected around plant components and piping which are anticipated to contain potentially significant amounts of radioactive materials. These shield walls are constructed of ordinary concrete (density of 145 lb/ft^3) and are sized such that radiation levels are reduced to satisfy the radiation zoning requirements defined in Section 12.1.2.1. The calculational methods used to determine the thickness and other dimensions of the shield walls are discussed in Section 12.1.2.4. In cases where access to equipment enclosures is through these shield walls, shielding effectiveness is maintained through the use of labyrinth entrances. Separation of shielded components and equipment compartments, where practicable, has been utilized in efforts to limit potential exposure during plant operation. The design criterion for piping penetrations, ducts, voids, and other irregularities in the shield walls is to situate them to minimize radiation streaming from a high radiation area to low radiation areas. If the penetrations, etc., cannot be so situated, compensatory shielding (e.g., steel or lead wool, steel plates, shadow shields, etc.) are provided. Examples of use of compensatory shielding are discussed below:

1. Most of the Reactor Building penetrations are in compartmentalized penetration access areas which provide further shielding for those areas of the plant more frequently occupied. Location of equipment anticipated to require frequent attention has been minimized in these access areas.

2. A concrete shadow shield for the Reactor Building equipment hatch above the 463’ elevation is provided.

3. A shadow shield is provided for the out of core neutron detector push rod penetrations of the primary shield.

4. Most of the Auxiliary Building shield wall penetrations are located close to the ceiling of the shielded penetration access areas.

When practical, piping containing radioactive material is routed through high radiation areas where accessibility is precluded. When this is not possible, such piping is routed through shielded pipe chases which serve to reduce radiation exposure rates to plant personnel when in the vicinity of this piping. Based on operating experience additional permanent, removable local shielding has been added to local areas such as around parts of the Residual Heat Removal and Safety Injection piping to reduce radiation exposure to operation and maintenance personnel.

Piping containing highly radioactive fluid is routed by engineering as stated in Section 12.0. Changes in the routing of such piping due to field conditions must be approved by engineering. In this manner, the same controls and considerations applied to the original pipe routing are applied to the field changes. If the changes are not acceptable from the standpoint of radiation exposure, other measures, such as additional shielding, etc., are employed.

Routing of very small piping is accomplished in the same manner as is the routing of large piping. Small, safety class piping is designed and routed by engineering, not by the field.

Equipment which contains or processes radioactive fluids is so located as to cause minimal occupational dose to plant personnel. Demineralized Water Nuclear Services supplies hose fittings located approximately every 50 feet along Reactor Building, Auxiliary Building, and Fuel Handling Building access ways for decontamination.

The components of the Excess Liquid Waste System, such as filters, demineralizers, tanks, and pumps, provide illustrative examples of equipment location. These components are located in individual shielded cubicles with labyrinthine entrances if personnel entry is required (see Figure 12.1-3).

Figures 12.1-3 through 12.1-20 illustrate the location of equipment associated with systems, such as Spent Fuel Cooling, Reactor Makeup Water, Sampling, and Solid Waste Disposal, and individual items, such as the miscellaneous waste drain tank. Color codes on these figures indicate expected dose rates in various areas and the length of time such areas may be occupied (on a weekly basis) by plant personnel.

Systems processing radioactive liquids are equipped with diaphragm type, leak proof valves to minimize spillage of radioactive materials.

Elevated door stops, concrete curbs, and floor drain systems which route a spill to the proper waste processing system are employed to minimize the spread of contamination. An effort has been made to locate instruments requiring inplace calibration in the lowest practicable radiation zones.

Provisions have been made to allow for flushing of systems that may become contaminated.

Remote handling equipment and portable shielding is available on site for use when needed.

12.1.2.3 Accident Shielding

The accident shielding objective is to attenuate radiation exposures due to activity in the Reactor Building to a small fraction of 10 CFR 100 limits and to a level at which required access to onsite locations can be achieved. The shielding design review performed for NUREG-0578 addresses the limitations imposed by the post-accident radiation fields on the timely recovery from an accident due to limited personnel access capability and/or safety equipment operation degradation. The results of the shielding review are presented in Appendix 12A.

12.1.2.4 Shielding Calculational Methods

The sizing of the radiation shielding required to reduce the exposure rates from the sources defined in Section 12.1.3 to satisfy the radiation zoning defined in Section 12.1.2.1 is done using the computer codes SDC [1]or QAD6G [2] and/or hand calculations. Both SDC and the hand calculations are utilized in performing basic shielding calculations involving relatively simple geometric source configurations, such as point, line, cylinder, sphere, slab, or disk. These methods employ many of the integrations and techniques found in Reference [3]. QAD6G is a point kernel code utilized in performing shielding calculations with more generalized, complex geometric configurations.

Radiation dose rates in and around shielded labyrinth entranceways to cubicles were estimated using a simplified albedo hand calculation. The wall that is directly exposed to the radiation source was divided into scattering areas. Then, using the geometrical configuration to determine the appropriate angles, the incident dose rates on each of the scattering areas and tabulated values of concrete gamma ray albedos, an estimate of radiation dose rates at areas in and around the labyrinths can be determined. This methodology has been checked using the G3 Code [10].

Source strengths and geometric models used for equipment during the performance of shielding calculations are presented in Tables 12.1-2 through 12.1-5.

12.1.3 SOURCE TERMS

The shielding design source items are based upon the three general plant conditions of normal full power operation, shutdown, and design basis events.

12.1.3.1 Sources for Normal Full Power Operation

Design consideration has been given to the reduction of activation product formation and buildup. For example, materials with low cobalt content are specified for primary coolant system alloys. However, the availability, durability, and economics of component materials must always be considered during material procurement. High cobalt, hard facing wear materials, such as Stellite, which come in contact with the primary coolant are used only where substitute materials cannot meet performance requirements.

The process of high flow rate/high temperature filtration is not a standard practice in Westinghouse NSSS and is not planned for Virgil C. Summer Nuclear Station.

These methods of cobalt reduction along with valve design and quantities of nickel alloys in the primary system are discussed in detail in WCAP-8872, "Design, Inspection, Operation and Maintenance Aspects of the Westinghouse NSSS to Maintain Occupational Radiation Exposures as Low as Reasonably Achievable."

To minimize exposure levels associated with valve stem leakage, packless valves are provided for those systems that normally contain radioactive fluid. These valves are designed for zero stem leakage. A metal bellows seal is used in high pressure applications; an elastomer diaphragm, in low pressure applications. For safety related diaphragm valves, a new valve design is used. This design substantially reduces maintenance requirements since diaphragm life has been increased by a factor of five.

Several features are incorporated into the design of packed valves to reduce crud buildup and maintenance requirements. These features include positive backseats to permit inline replacement of packing, lantern ring seal leakoffs, and special, close tolerance graphoil packing in lieu of conventional packing.

The main sources of radiation during normal full power operation are the reactor core, the reactor coolant, and auxiliary systems associated with the processing or handling of reactor coolant.

12.1.3.1.1 Reactor Core

The neutron fluxes at the inside surface of the primary shield concrete at the core midplane are listed in Table 12.1-6. These fluxes represent average values on the core center plane and are based upon a core axial peaking factor of 1.30. Axial variation factors for off midplane variation of neutron flux in the primary shield concrete are also listed in Table 12.1-6.

The gamma fluxes at the inside surface of the primary shield concrete at the core midplane are listed in Table 12.1-7. Values of gamma fluxes above and below the core midplane can be obtained by applying the axial variation factors which are also presented in Table 12.1-7.

12.1.3.1.2 Reactor Coolant

The main sources of activity in the reactor coolant during normal full power operation are N16 from coolant activation processes, fission products from fuel clad defects, and corrosion and activation products. All shielding has as its design basis the maximum case of clad defects in fuel rods producing 1.0 percent of core thermal power. Activity concentrations in the reactor coolant are listed in Table 11.1-6. The activities in the pressurizer steam and liquid phases are given in Table 11.1-4 and the pressurizer deposited sources in Table 12.1-8. The N16 activity of the coolant is given in Table 12.1-9 as a function of transport time in a reactor coolant loop.

The N-16 sources in Table 12.1-9 include the effects of irradiation of the coolant both in the core and reflector regions outside the core. Transit times around a coolant loop and within the reactor vessel are based on design flow rates and active flow volumes at 100% power operation. The transit times are included in Table 12.1-9.

12.1.3.1.3 Sources for Auxiliary Systems

The auxiliary system equipment for which a determination of shielding sources is required consists of pumps, heat exchangers, demineralizers, filters, units of evaporator packages, liquid tanks, gas tanks, and pipes. Reactor coolant activities assumed in the development of all auxiliary system source terms are given in Table 11.1-2 and are based upon 1.0 percent failed fuel.

12.1.3.1.3.1 Chemical and Volume Control System Sources

The purpose of the Chemical and Volume Control System (CVCS) is to provide continuous purification and volume control of the reactor coolant water. The major equipment items include the regenerative and letdown heat exchangers, mixed bed and cation bed demineralizers, reactor coolant filter, volume control tank, and charging pumps. The Boron Thermal Regeneration (BTR) subsystem contains the three BTR heat exchangers and the BTR demineralizers. The Seal Water subsystem for the reactor coolant pumps includes the injection and return filters and the seal water heat exchanger. The spectral source strengths in the CVCS purification letdown flow are tabulated in Table 12.1-10. The sources assume sufficient delay time from the reactor coolant loop for decay of the N16 isotope.

The radiation sources in the ion exchangers of the CVCS system are listed in Table 12.1-2. The mixed bed retains the fission product activity, both cations and anions, and the corrosion product (crud) metals. The cation bed can be used intermittently to remove lithium for pH control and to supplement the mixed bed in removing yttrium, cesium, molybdenum, and the crud metals. The Boron Thermal Regeneration beds are used to regulate the boron concentration in the reactor coolant water. They are utilized during load following operations and for removal of boron from the coolant as the nuclear fuel is depleted. These demineralizers also collect radioactive anions, such as iodine, which may have passed through the mixed bed.

The sources in the volume control tank are listed in Table 12.1-3. These sources correspond to a nominal tank operating level of 125 ft^3 in the liquid phase and 175 ft^3 in the vapor phase.

Before transferring the contents of the evaporator batch tank to the boric acid tank, a sample is taken which should satisfy two criteria:

1. A boron content of ~ 7000 ppm

2. An activity concentration of < 0.01 μc/cc

If both criteria are not satisfied, the contents are recycled through the evaporator feed demineralizers. An activity concentration of < 0.01 μc/cc in the feed to the boric acid storage tanks is the basis for not shielding the tanks and associated piping.

For filters in the CVCS system, the design criteria for handling and shielding are primarily based upon operating experience and are as follows:

1. Capability should be provided for changing filters with radiation levels of 100 R/hr at contact with the filter housing. This criterion is applicable to the following filters: seal water injection and seal water return.

2. Capability should be provided for changing the reactor coolant filter with radiation levels of 500 R/hr at contact with the filter housing.

3. The total combined exposure to personnel changing any one of the above filters should not exceed 100 mr.

The exposure rates listed in Items 1 and 2, above, are considered to be maximums, with limits controlled by periodic monitoring.

The specific source strengths of the subject filters are included in Table 12.1-4. The sources for the reactor coolant filter correspond to an exposure rate of 500 R/hr at contact. The sources for the remaining filters correspond to an exposure rate of 100 R/hr at contact. The filters are assumed to be drained of process fluid and are considered to be homogeneous sources.

The radiation sources in the heat exchangers of the CVCS system are listed in Table 12.1-5. The regenerative and excess letdown heat exchangers are located in the Reactor Building. They provide the initial cooling for the reactor coolant letdown and their sources include N16 activity. The remaining CVCS heat exchangers are located in the Auxiliary Building where N16 activity is not a significant factor. The letdown heat exchanger provides second stage cooling for the reactor coolant prior to entering the demineralizers. The seal water heat exchanger cools water from several sources, including the reactor coolant discharged from the excess letdown heat exchangers. The

activity of these heat exchangers is identical to that listed in Table 12.1-10. The thermal regeneration heat exchangers include the moderating, chiller, and letdown reheat units. The radiation sources in this equipment are modified to account for activity removed by the demineralizers upstream of the units.

12.1.3.1.3.2 Boron Recycle System Sources

The major equipment items included in the Boron Recycle System (BRS) are the recycle holdup tank and the Reactor Water Grade System Demineralizers and RN 07-037 associated equipment, i.e., feed demineralizers and filter, condensate demineralizers and filter, and concentrates filter. Radiation sources in the various pumps are assumed to be identical to the liquid sources in the tank from which the pump takes suction.

The radiation sources in the BRS ion exchangers are listed in Table 12.1-2. The evaporator feed demineralizers are located upstream of the holdup tanks and contain resins which remove nongaseous activity from the reactor coolant directed to the holdup tanks. A decontamination factor (DF) of 10 across those beds is taken for all particulate activity. RN 07-037

The recycle holdup tanks are each equipped with a diaphragm. Gases which flash from the reactor coolant letdown to the holdup tanks are retained under the diaphragm until approximately 500 ft^3 of gas has accumulated. The gases are then removed to the Waste Gas System. The radiation sources in the holdup tanks are based upon an assumed letdown rate of 120 gpm to a single holdup tank with 50 percent of the volatile activity flashing into the vapor phase. These sources are listed in Table 12.1-3.

The recycle evaporator feed filter and condensate filter are located downstream of their respective demineralizers and serve to retain particulates and any resin fines which may escape from the demineralizers. The maximum radiation sources in these filters are listed in Table 12.1-4. The sources for the feed filter correspond to a radiation level of 100 R/hr at contact. The condensate filter sources result in levels of less than 1 R/hr at contact. The radiation sources in the concentrates filter correspond to an exposure rate RN 02-025 of approximately 10 R/hr. RN 07-037

12.1.3.1.3.3 Waste Processing System Sources

The radiation sources in the Waste Processing System (WPS) are tabulated in Tables 12.1-2 through 12.1-5 and 12.1-11. The major equipment items in the waste gas portion are the waste gas compressors, hydrogen recombiners, and gas decay tanks. The radiation sources in this equipment are based upon cold shutdown procedures during which the radioactive gases are stripped from the reactor coolant system. Since the gases are continuously recirculated, the radiation sources in the waste gas equipment are identical.

The Liquid Waste Processing System is considered as several subsystems, based upon intended use during normal operation. RN 03-038

Low activity, non-reactor grade water is directed to the Floor Drain Laundry and Hot shower, or Excess Waste/Decon Pit Collection (ELWS) tanks. Normally this water is analyzed, then discharged. If activity levels prevent this, the water can be processed by a demineralizer/filter. The equipment items included in the subsystems are the floor drain tank and filter, laundry and hot shower tank and filter, excess waste holdup tank demineralizer and filter, decon pit collection tank demineralizer and filter, waste monitor tank demineralizer and filter, and two waste monitor tanks. The floor drain, waste RN 03-038 monitor, and excess waste tanks provide surge capacity for the waste holdup tank during periods when abnormal volumes of liquid waste are encountered. Hence, for shielding purposes, the radiation sources in these tanks are assumed to be the same, i.e., degassed reactor coolant. Similarly, the sources in the floor drain tank filter are based on degassed reactor coolant. The sources in the waste monitor tank demineralizer and filter are based upon circulating reactor coolant through these components.

Maximum radiation sources in demineralizers of the WPS and ELWS are presented in Table 12.1-2. Radioactive spent resins, discharged from the various demineralizers, are retained in the spent resin storage tank. Mixed bed demineralizers contain the most radioactive resin discharged to the spent resin storage tank. These sources determine the required tank shielding. The short lived activity is allowed to decay (approximately 30 days) and the resin is then directed to the drumming station for packaging. The associated equipment includes the spent resin storage tank and the resin sluice pump and filter. The resin sluice filter is shielded for radiation levels of 100 R/hr at contact. RN 03-038

The waste monitor tank demineralizer can be used to remove activity admitted to the waste monitor tanks.

The maximum radiation sources in the various filters of the WPS and ELWS are listed in Table 12.1-4. The waste monitor tank filter is located downstream of the demineralizer. The sources in the condensate filter result in radiation levels of approximately 1 R/hr at RN 03-038 contact. The monitor tank filter sources produce approximately 85 R/hr at contact. The maximum radiation levels in the spent resin sluice and floor drain tank filters are 100 R/hr at contact.

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Radiation sources in the various pumps of the WPS and ELWS are assumed to be identical to the liquid sources in the tank from which the pump takes suction.

Sources in the laundry and hot shower tank and filter and in the waste condensate tank are negligible. Therefore, these items do not require shielding.

Sources of solid waste for the drumming station are given in Table 12.1-11. A description of the solid waste system is given in Section 11.5.

12.1.3.1.3.4 Sources for Blowdown Systems

Radiation sources for equipment in the Steam Generator Blowdown and Nuclear Blowdown Processing Systems are presented in Tables 12.1-2 through 12.1-5.

The major equipment items in these systems are the steam generator blowdown heat exchangers, nuclear blowdown holdup and monitor tanks, blowdown filters, blowdown demineralizers, and the blowdown spent resin storage tank. Radiation sources are based upon one percent failed fuel and a maximum steam generator tube leakage rate of 0.1 gpm.

12.1.3.1.3.5 Sources for Spent Fuel Cooling Systems

The radiation sources in the spent fuel demineralizer and filter are listed in Tables 12.1-2 and 12.1-4, respectively. This equipment is used to maintain water clarity and remove activity released during refueling operations and the subsequent fuel cooling period. The filter sources correspond to an exposure rate of 100 R/hr at contact. The spent fuel pool skimmer is used only for cleaning the surface of the spent fuel pool water and does not require shielding.

12.1.3.2 <u>Sources for Shutdown</u>

12.1.3.2.1 Primary System Sources at Shutdown

The core gamma sources after shutdown are used to establish radiation shielding requirements during refueling operations and during shipment of spent fuel. The sources associated with the spent fuel are based upon an average power assembly with an irradiation time of 108 seconds (3.1 years). These source strengths per unit volume of homogenized core are listed in Table 12.1-12 for various times after shutdown.

The irradiated control rod sources are used in establishing radiation shielding requirements during refueling operations and during shipment of irradiated control rods. The absorber material used in the control rods is silver-indium-cadmium (Ag-In-Cd). The source strengths associated with the control rods are listed in Table 12.1-13 for various times after shutdown. The values are in terms of centimeter of height of a single control rod for an irradiation period of 100,000 hours.

The Ag-In-Cd control rod sources included in Table 12.1-13 are based on an irradiation period of 100,000 hours. This design basis leads to conservative source term values since it is not expected that any control rod will spend a majority of its design life (15 years) in the inserted position.

The incore detector drive wire sources are used in establishing the radiation shielding requirements for the wires when the detectors are not in use and during shipment when the detectors have failed.

The incore drive wire sources are based on the assumption that the drive wire has been lodged in the core for one full year. These sources are tabulated in Table 12.1-14 and are conservative in that the drive wires will normally be inserted in the core for short periods of time during flux mapping.

The activities of steam generator primary side surfaces are used in determining access limitations in and around the steam generators at plant shutdown.

Estimated corrosion product deposited sources are given in Table 12.1-8 and 12.1-15. The corrosion product sources are based on the methods and equations presented in the following reference:

> S. Yerazunis, E. H. Alkire and R. L. Seidel, "Mechanisms of Reactor System Activation," KAPL-M-SMS-98, May, 1959.

The transport and corrosion parameters used in the calculations are selected so that the results predicted by this method reflect operating plant data. The buildup of corrosion products deposits is treated as follows:

a) Transient crud layers are assumed to exist at a nominal thickness of 50 mg/dm^2.

b) Permanent crud films are assumed to build up to 90% of their maximum thickness (nominally assumed to be 50 mg/dm^2) early in the initial cycle of operation. The remaining 10% of the deposit is then assumed to build up linearly thereafter, over the operating period.

12.1.3.2.2 Auxiliary Cooling System Sources

The maximum specific source strengths in the Residual Heat Removal System are listed in Table 12.1-16. The Residual Heat Removal System is placed in operation approximately 4 hours after the start of reactor shutdown and reduces the reactor coolant temperature to approximately 120°F within about 20 hours after the start of shutdown. The sources are maximum values with credit taken for 4 hours of activity decay and purification. Accident sources in the Residual Heat Removal System are considered separately in Section 12.1.3.3.

12.1.3.3 Sources Under Accident Conditions

The sources for accident conditions are provided in Chapter 15. RN 02-025

12.1.4 AREA MONITORING

12.1.4.1 Design Basis

The area radiation monitors are designed to detect, indicate, annunciate, and record the radiation levels monitored at selected locations inside the plant. These locations were selected on the basis of monitoring normally occupied zones critical to plant operation, monitoring occupied areas, access corridors and fuel handling areas with potential for airborne radioactivity, and having the capability to monitor the trend of Reactor Building activity following a loss of coolant accident. These monitors provide supporting data to the surveillance of plant radiation levels as recommended by ANSI N13.2 [5] and Regulatory Guide 8.2, and as required by 10 CFR 20. Area monitor instrumentation is provided to supply information and to aid in maintaining dose rates ALARA and is not considered safety-related.

Permanently installed area radiation monitors are not provided for the containment equipment access hatch area, low level waste storage area, or high level waste storage area because of the following considerations:

1. Portable radiation monitoring equipment will be used to survey these areas or any other potentially radioactive areas prior to access by personnel.

2. Access to these areas is required infrequently and will be permitted only under the supervision of health physics personnel after issuing a radiation work permit (RWP) as described in Section 12.3.2.3.7.

3. Although Figure 12.1-5 (sheet 2 of 2) indicates that the dose rates in the low and high level waste storage areas result in classification of these areas as Zone V, the dose rates will normally be reduced to a level less than required for Zone V classification by use, as required, of shielding.

12.1.4.2 System Description

The area radiation monitors consist of eighteen fixed and one movable channel, as identified in Table 12.1-19. The area radiation monitoring instruments provide indication of proper plant operation by measurement of the gamma dose rate levels in selected areas of the plant. Reliable power for the instrumentation is obtained from the diesel backed, 120 volt instrument buses. The local audible and visual alarm are powered from the 120 volt buses. The movable channel is powered from a 120 volt outlet.

Except for the movable unit, which is equipped for local recording and indication, the measured dose rate levels are indicated and recorded on the Area Radiation Monitoring System control panel located in the Control Room. Local indication is provided for each monitor except RM-G18. Each area radiation monitor is equipped with two adjustable alarm levels (alert and high) and a channel failure and/or loss of power alarm except RM-G7. Channel RM-G7 provides the following alarms: alert, high radiation, loss of power to the alarm module. These alarms (from the fixed monitors) are connected to an annunciator panel on the Radiation Monitoring System control panel in the control room. Each area radiation monitor, except for RM-G17A&B, RM-G7, and RM-G18, is provided with a local audible and visual alarm located near the detector or local readout. The manipulator crane area monitors RM-G17A&B, each have an audible alarm in the fuel handling area. The alarm setpoints are adjustable and are dependent upon the location of the individual monitors. In general, the alert setpoint is adjusted to a level below the high setpoint to provide warning of a change in normal plant operation conditions. The high setpoint is generally established by the zone limitations established for each area. The gamma dose rate resulting from a postulated loss of coolant accident (LOCA) is monitored by the high range Reactor Building monitors, RM-G7 and RM-G18. RM-G7 and RM-G18 are located inside the containment and are designed to meet IEEE-323-1974, IEEE-344-1975, ANSI N320-1978.

Channels RM-G7 and RM-G18 are composed of a Victoreen model 877-1 stainless steel ion chamber detector and a Victoreen model 876A-1, 7-decade ratemeter. Both of these channels of instrumentation meet seismic separation and 1E power source criteria and provide the appropriate display indication, record, and alarms in the Control Room. They also meet the IEEE-323 and LOCA environmental qualification requirements. Both channels have a clear view of the operating floor area of the containment as shown in Figure 1.2-6. The response of these channels to gamma (photon) radiation at 1 mev is approximately 100% and approximately 85% at 80 kev. Based on the average gamma energy anticipated after a LOCA as shown in report NUREG CR-1237, SAND 79-2143 (January 1980), RM-G7 and RM-G18 will respond to within +8%.

Radiation Monitors RM-G7 and RM-G18 were added to the Radiation Monitoring System to meet the requirements of NUREG-0578, NUREG-0737, and Regulatory Guide 1.97, Rev. 2. These provide a diverse means of measuring the containment for high level gamma radiation. The detectors are stainless steel gamma sensitive ion-chamber that are wall mounted inside the containment above the 463 foot floor elevation, reference Figure 1.2-6. Continuous 7 - decade analog readout is provided in the Control Room Radiation Monitoring System panel with an indication range of 1 to 107 R/hr. The detector energy response is sensitive to 60 KEV, -15% at 80 KEV and within ± 10% from 100 KEV to 3 Mev.

The analog signals are recorded on multipoint recorders which are seismically mounted on the Radiation Monitoring System panel. High gamma dose rate or channel failure is annunciated in the Control Room.

1E instrument power is obtained from the A and B-train diesel backed 120 Vac. The detector and Control Room readout have been type tested for seismic qualification to meet the requirements of IEEE-323(1974). The detector and its interface cable-connector assembly have been type tested for LOCA environment to meet Regulatory Guide 1.89. Calibration of the high range containment gamma monitors is performed during refueling and consists of verification of the readouts by a calibrated current source and by a point source verification of response of the ion-chamber detectors in the low range (approximately 10R).

The gamma dose rate in the Reactor Building manipulator crane area, during refueling, is monitored by RM-G17A and RM-G17B. Either one, upon detection of high activity or loss of signal, interlock to close the purge discharge valves (Figure 11.4-1). These monitors are not required during normal operation.

Area radiation monitor electronics have a five decade logarithmic scale with a nominal measurement accuracy of ± 25 percent of the reading. The precision is ± 15 percent at all levels. Area radiation monitors are calibrated on a routine basis and after any maintenance work is performed on the detector by exposure to a standard radioactive source with its calibration traceable, directly or indirectly, to the National Bureau of Standards. Calibration of the high range area monitor, RM-G7 and RM-G18 uses a calibrated electrical current source to verify the performance of the readouts. Functional verification of all area radiation monitors, except RM-G7 and RM-G18 is achieved by use of remotely actuated check sources (isotopic or LED signal input). Functional verification of RM-G7 and RM-G18 is achieved through use of a built-in electronic test signal.

12.1.5 OPERATING PROCEDURES

The Manager of Health Physics Services is responsible for developing the radiation protection training program, the radiation protection program and health physics procedures to ensure that exposures of all personnel are kept within the limits of 10 CFR 20 and ALARA. These procedures and programs are developed incorporating guidance contained in Regulatory Guides 8.2, 8.8, and 8.10. Plant personnel will receive radiation protection training, the depth of which will depend upon the work assignments, individual responsibilities, and the degree of radiation hazard anticipated. 00-01

Personnel whose duties entail entering controlled areas, or directing the activities of others who enter controlled areas, are required to have radiation protection training. This training qualifies personnel to implement plans and procedures, each in his area of responsibility, to maintain doses ALARA.

Administrative controls are established to assure that procedures and requirements relating to radiation protection are followed by plant personnel.

Radiation control procedures are established for systems that contain, collect, store, or transport radioactive liquids, gases, and solids. These procedures ensure that occupational radiation exposures are ALARA. The procedures that control radiation exposure are subject to review and approval as outlined in Section 13.5.
The basic principles of time, distance, and shielding are applied during operation and maintenance to ensure that personnel exposure is within limits. The following techniques are employed:

1. During initial startup, neutron and gamma dose rate surveys are performed to verify the adequacy of shielding.

2. During normal operations dose rate, surface contamination, and airborne radioactivity surveys are performed periodically throughout the plant and areas are zoned accordingly. These surveys ensure that data is available for planning operation and maintenance activities.

3. Areas are conspicuously posted in accordance with 10 CFR 20.1902 as appropriate. RN 01-119

4. A radiation work permit (RWP) system is employed to ensure proper administrative control over work in restricted areas. The RWP is designed to ensure that the radiation conditions are defined and that appropriate measures are taken to minimize the dose received by personnel. Section 12.3.1.3 explains the use of the RWP.

5. Extension tools are used when practicable to increase the distance from the radiation source to the worker.

6. Equipment is moved to areas of lower radiation fields for maintenance when practicable.

7. Portable shielding in the form of lead bricks, lead sheets, lead shot, high density concrete block, or steel plates is used as practicable.

8. A personnel dosimetry program, as described in Section 12.3.3, is administered by the Health Physics group to ensure compliance with 10CFR20.

Experience gained from the operation and maintenance of several nuclear plants with whom SCE&G has contact is used to provide a basis for further evaluation.

12.1.6 ESTIMATES OF EXPOSURE

The estimates of exposure presented in this Section address both the peak radiation exposure rates and expected annual doses. Personnel doses are based upon the radiation zone and the anticipated occupancy requirements for that area.

12.1.6.1 Dose Rates at Selected Inplant Locations

Figures 12.1-1 through 12.1-20 illustrate the location of the various zones throughout the plant. These zones have been established using design parameters and estimated occupancy requirements. Expected radiation levels should be only a fraction of the design values. Table 12.1-20 lists selected inplant locations and the associates design dose rate for each area.

12.1.6.2 Estimates of Personnel Occupancy Requirements

Occupancy requirements throughout the plant, based upon operating experience and estimated SCE&G personnel requirements, are considered in the establishment of radiation zones described in Section 12.1.2.1. To estimate occupancy requirements, plant personnel were categorized into five groups according to work function. Table 12.1-21 lists the estimated size of each group and the estimated occupancy requirements for each group in Zones I, II, and III. Estimates were not made for Zones IV and V since routine occupancy is not anticipated in these zones during normal operation or during anticipated operational occurrences.

12.1.6.3 Estimates of Annual Man-Rem Doses

Annual doses to plant personnel are estimated based upon the assumption of 2000 hours per year work time for each employee. For each zone, the lower and upper ends of the range have been evaluated in determining man-Rem doses. It is anticipated that the general radiation levels for each zone will be equal to or less than the lower values stated for the zone although isolated higher levels will exist in certain areas within the zone.

The estimated man-Rem doses for the various categories of plant personnel as discussed in Section 12.1.6.2 are listed in Table 12.1-21. The total annual dose for plant operation is conservatively estimated to range from 72 to 364 man-Rems. From the years 1969 to 1974, operational data indicate that for all light water reactors the average dose was approximately 404 man-Rem/unit/year [6]. At Virgil C. Summer Nuclear Station the man-Rem doses are maintained ALARA by utilizing applicable operational data and guidance available in the industry and from the government in conjunction with the Health Physics ALARA oriented program (see Section 12.3).

Numerous dose assessment techniques are utilized in optimizing shielding design and in maintaining radiation doses ALARA. Regulatory guidance, such as that found in Regulatory Guide 8.8, is followed in plant design and review. Some of the specific dose assessment techniques are as follows:

1. Utilization of operational data:

    a. Operational radiation levels.

    b. Trends in radiation levels associated with years of operation, plant type, plant size, power levels, plant design, etc.

    c. Radiation zones as described by occupancy requirements and actual radiation levels.

    d. Location of components with respect to plant operability.

    e. Reliability of components.

    f. Adequacy of plant layout in terms of traffic patterns; space allocation, such as around radioactive components requiring maintenance and inspection; pipe routing, etc.

    g. Number of plant employees associated with different tasks and the resulting man-Rem doses.

2. Review equipment and instrumentation and locate in zones I and II whenever possible to minimize occupancy requirements in higher radiation zones.

3. Review of the plant design by a competent professional in radiation protection and by the utility radiation protection manager.

As indicated by Table 12.1-22 no man-Rem estimates were made for Zones IV and V. It is anticipated that entry to these zones will be primarily during plant shutdown when special maintenance will be performed. Every effort will be made to decrease the radiation dose to the maintenance personnel by utilizing temporary shielding, component decontamination, or other acceptable techniques for assuring that radiation doses are ALARA. An analysis of operational data indicates that special maintenance has contributed approximately 20 percent to the annual plant dose [6,7]. Other operations have contributed to the annual plant man-Rem dose as follows [6]:

1. Routine reactor operation and surveillance, 14 percent.

2. Routine maintenance, 45 percent.

3. Inservice inspection, 2.7 percent.

4. Waste processing, 2.5 percent.

5. Refueling, 14 percent.

A further estimate of the man-Rem doses has been made by identifying specific tasks anticipated to occur at the plant. Various data from operating plants and in current publications (see References [11] through[19]) were used in identifying these tasks, manpower effort required to complete each task, and the radiation levels associated with doing the work. Tables 12.1-22a and 12.1-22b list the tasks anticipated for normal operation and those expected during an outage. Tables 12.1-22c and 12.1-22d provide a breakdown of personnel that will be involved with each of the tasks and the extent of their involvement. Special maintenance tasks have been analyzed although it is anticipated that these tasks will occur infrequently. When these tasks, such as steam generator maintenance, are performed the man-Rem dose varies directly with the conditions surrounding the maintenance and the extent of the work to be performed at the time. Therefore, because of the variability of the man-Rem dose for special maintenance, special maintenance has been excluded from the yearly personnel dose commitment (see Table 12.1-22e).

Radiation doses associated with airborne radioactivity have not been analyzed in terms of tasks due to the lack of sufficient data. Conservative estimates were made using the occupancies given in Table 12.1-21. Table 12.1-22f presents the dose rates in various parts of the plant using calculated airborne contamination levels listed in Table 12.2-1. To determine the man-Rem dose commitment from airborne activity (see Table 12.1-22g), it was assumed that the airborne radiation levels to which plant personnel will be exposed are represented by those levels for the Turbine Building (Zone I), Auxiliary Building (Zone II) and the letdown heat exchangers (Zones III through V) listed in Table 12.1-22f. No man-Rem doses were calculated using the specific concentrations for the Reactor Building since routine occupancy of this area during normal operation is not anticipated.

12.1.6.4 Doses at the Radiation Controlled Area Boundary

The Radiation Controlled Area is located within the site boundary as defined in Figure 2.1-3. Additionally, except for unusual and unanticipated situations, the Radiation Controlled Area will not extend beyond the Protected Area fence. The only normal exceptions to this are: 1) the Radiography Vault which is operated under a separate license and 2) the storage of very low level materials such as exempt sources and other materials which present no significant radiological hazards. The Radiation Controlled Area consists of the Reactor Building, Auxiliary Building, Fuel Handling Building, the west penetration access areas, the east penetration access area (412' elevation), portions of the Intermediate Building and tendon access area, radioactive waste areas, hot maintenance areas, hot warehouse, and other areas designated by Health Physics for the purposes of radiation protection. Areas may be removed from the Radiation Controlled Area at the direction of Health Physics when operating conditions and radiation levels are such that control is not necessary for the purposes of radiation protection. The area outside the Radiation Controlled Area has been zoned less than 1 mrem/hr. It is expected that, under normal operating conditions and during anticipated operational occurrences, the radiation level outside the above mentioned Radiation Controlled Area will be only a fraction of this design level.

12.1.7 REFERENCES

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NOTE 12.1-1

Table 12.1-1 is being retained for historical purposes.

RN 06-017

TABLE 12.1-1

DOSE LIMITS FOR OCCUPATIONAL WORKERS

	Maximum Dose (Rem)		
	Quarterly	Annually	Accumulated Dose
Whole body; head and trunk; active blood forming organs; lens of eyes; or gonads	3	12	5 (N-18) [(1)]
Hands and forearms; feet and ankles	18.75	75	
Skin of whole body	7.5	30	

(1) N equals the individual's age at last birthday.

TABLE 12.1-1a

RADIATION SHIELDING THICKNESSES

Equipment	Minimum Shielding Provided Between Equipment and Areas of Radiation Zone I or II (ft)	
Demineralizers:		
Mixed Bed	3.50	
Cation Bed	3.50	
Boron Thermal Regeneration	2.50	
Spent Fuel Pit	2.50	RN 03-038
Waste Monitor Tank	3.50	
NBS Primary	2.00	
NBS Polishing	2.00	
Excess Liquid Waste	3.00	
Tanks:		
Volume Control Tank	3.50	
Waste Holdup	2.50	
Floor Drain	2.50	
Waste Monitor	2.00	
Spent Resin Storage	4.00	
Catalytic Hydrogen Recombiner	2.00	
Waste Gas Compressor	2.00	
Gas Decay	3.50	
Recycle Holdup	3.50	
Reactor Coolant Drain (1)	- (1)	RN 03-038
NBS Holdup	2.00	
NBS Monitor	2.00	
NBS Spent Resin Storage	4.00	
Excess Waste Holdup	2.50	
Decontamination Pit Collection	2.50	
Pressurizer Relief	3.67 (1)	

TABLE 12.1-1a (Continued)

RADIATION SHIELDING THICKNESSES

Equipment	Minimum Shielding Provided Between Equipment and Areas of Radiation Zone I or II (ft)
Filters	
Reactor Coolant	2.00
Seal Water Return	2.00
Seal Water Injection	2.00
Recycle Evaporator Condensate	2.00
Recycle Evaporator Concentrates	2.00
Spent Fuel Purification	2.00
Floor Drain Tank	2.00
Spent Resin Sluice	2.00
Waste Monitor Tank	2.00
Waste Evaporator Condensate	2.00
Nuclear Blowdown Demineralizer Inlet	2.00
Nuclear Blowdown System Spent Resin Sluice	2.00
Nuclear Blowdown Demineralizer Outlet	2.00
Waste Gas Drain	2.00
Excess Liquid Waste	2.00
Heat Exchangers:	
Regenerative	2.00
Letdown	2.00
Seal Water	2.00
Excess Letdown (1)	1.00 (1)
Moderating	2.00
Letdown Chiller	2.00
Reactor Coolant Drain Tank (1)	- (1)
Letdown Reheat	2.00
Spent Fuel Pit	0.00
Residual Heat Removal	3.00

RN 03-038

NOTE: (1) This equipment is inside containment and the shielding provided is not intended to lower radiation exposure rates during normal operation to values appropriate for Zone I or II designation.

TABLE 12.1-2

RADIATION SOURCE TERMS FOR DEMINERALIZERS

Source	Source Location	Source Geometry	Source Volume (Ft^3)	Activity (Mev/cc-sec) for Energy (Mev) of 0.40	0.80	1.30	1.70	2.20	2.50	3.50	
Mixed Bed Demineralizer (2)	AB-M	Cylinder	30.02	$1.3x10^8$	$2.6x10^8$	$3.1x10^7$	$1.3x10^7$	$3.8x10^6$	$8.7x10^4$	$1.7 x 10^5$	02-01
Cation Bed Demineralizer (1)	AB-M	Cylinder	20.00	$1.5x10^6$	$2.9x10^8$	$4.0x10^6$	$3.0x10^6$	-	-	-	
Boron Thermal Regeneration Demineralizer (4)	AB-M	Cylinder	69.02	$1.0x10^6$	$6.0x10^5$	$2.3x10^5$	$1.0x10^5$	$3.8x10^4$	-	-	
Evaporator Feed Demineralizer (2)	AB-M	Cylinder	30.02	$1.2x10^7$	$3.4x10^7$	$3.0x10^6$	$1.5x10^6$	$3.9x10^5$	-	-	
Recycle Evaporator Condensate Demineralizer	AB-M	Cylinder	20.00	$4.6x10^4$	$2.6x10^4$	$6.3x10^3$	$2.8x10^3$	$1.2x10^3$	-	-	
Spent Fuel Pool Demineralizer (1)	AB-M	Cylinder	43.08	$2.1x10^6$	$7.4x10^5$	$4.1x10^3$	$4.7x10^3$	-	-	-	
Waste Evaporate Condensate Demineralizer (1)	AB-M	Cylinder	30.02	$2.8x10^4$	$1.8x10^4$	$4.3x10^3$	$1.7x10^3$	$6.5x10^2$	-	-	
Waste Monitor Tank Demineralizer (1)	AB-M	Cylinder	30.02	$8.8x10^5$	$3.3x10^6$	$8.3x10^5$	$4.1x10^5$	$1.6x10^5$	-	-	02-01
NBS Primary Demineralizer (2)	AB-M	Cylinder	150.00	$3.3x10^4$	$1.2x10^5$	$1.5x10^4$	$2.4x10^2$	$6.7x10^2$	-	-	
NBS Polishing Demineralizer (2)	AB-M	Cylinder	90.00	$3.3x10^4$	$1.2x10^5$	$1.5x10^4$	$2.4x10^2$	$6.7x10^2$	-	-	
Excess Liquid Waste Demineralizer (2)	FB-B	Cylinder	30.00	$2.8x10^4$	$1.8x10^4$	$4.3x10^3$	$1.7x10^3$	$6.5x10^2$	-	-	
Duratek Demineralizer (5)	AB (447')	Cylinder	(1)		(2)						RN 03-038

Source Location Legend

Location	Elevation
AB - Auxiliary Building	B - Basement
RB - Reactor Building	M - Mezzanine
FB - Fuel Handling Building	O - Operating Floor
Y - Yard	S - Sub-Basement

(1) For Source Volumes see Figure 11.2-5.

(2) Source Terms in Curies Provided in Table 11.2-1.

TABLE 12.1-3

RADIATION SOURCE TERMS FOR TANKS

Source	Source Location	Source Geometry	Source Volume (Ft^3)	Activity (Mev/cc-sec) for Energy (Mev) of 0.40	0.80	1.30	1.70	2.20	2.50	3.50	
Volume Control Tank (1) -VAPOR	AB-O	Cylinder	180.00	7.3×10^5	3.0×10^5	-	1.5×10^5	3.4×10^5	6.9×10^5	-	02-01
Volume Control Tank (1) -LIQUID	AB-0	Cylinder	120.00	9.3×10^4	7.5×10^4	2.3×10^4	2.7×10^4	4.2×10^4	8.7×10^4	2.0×10^3	
Waste Holdup Tank (1)	AB-S	Cylinder	1,337.00	3.9×10^4	2.2×10^5	1.1×10^5	4.9×10^4	1.7×10^4	-	-	
Floor Drain Tank (1)	AB-S	Cylinder	1,337.00	3.9×10^4	2.2×10^5	1.1×10^5	4.9×10^4	1.7×10^4	-	-	
Waste Monitor Tank (2)	AB-0	Cylinder	668.00	3.9×10^4	2.2×10^5	1.1×10^5	4.9×10^4	1.7×10^4	-	-	
Spent Resin Storage Tank (1)	AB-B	Cylinder	350.00	1.3×10^8	3.1×10^8	3.3×10^7	1.4×10^7	3.9×10^6	8.6×10^4	1.7×10^5	02-01
Catalytic Hydrogen Recombiner (2)	AB-S	Cylinder	2.02	3.6×10^5	9.1×10^4	-	7.1×10^4	1.3×10^5	3.1×10^5	-	
Waste Gas Compressor (2)	AB-S	Cylinder	2.09	3.6×10^5	9.1×10^4	-	7.1×10^4	1.3×10^5	3.1×10^5	-	
Gas Decay Tank (8)	AB-S	Cylinder	600.00	3.6×10^5	9.1×10^4	-	7.1×10^4	1.3×10^5	3.1×10^5	-	
Recycle Holdup Tank (2) -VAPOR	AB-S	Sphere	494.00	2.8×10^5	7.8×10^4	-	5.7×10^4	1.2×10^5	2.7×10^5	-	
Recycle Holdup Tank (2) -LIQUID	AB-S	Cylinder	5,120.00	4.7×10^5	5.7×10^4	1.7×10^3	3.2×10^4	1.2×10^5	1.7×10^5	-	
Reactor Coolant Drain Tank (1)	RB-B	Cylinder	46.8	4.3×10^5	7.6×10^4	2.3×10^4	2.7×10^4	4.2×10^4	8.7×10^4	2.0×10^3	

TABLE 12.1-3 (Continued)

RADIATION SOURCE TERMS FOR TANKS

Source	Source Location	Source Geometry	Source Volume (Ft^3)	Activity (Mev/cc-sec) for Energy (Mev) of 0.40	0.80	1.30	1.70	2.20	2.50	3.50	
											02-01
											RN 03-038
Nuclear Blowdown System Holdup Tank (1)	AB-M	Cylinder	1,740.00	8.4×10^1	2.9×10^2	1.0×10^2	1.7×10^1	5.2×10^1	-	-	02-01
Nuclear Blowdown System Monitor Tank (1)	AB-M	Cylinder	670.00	8.4×10^1	2.9×10^2	1.0×10^2	1.7×10^1	5.2×10^1	-	-	02-01
Nuclear Blowdown Spent Resin Storage Tank (1)	AB-B	Cylinder	600.00	1.3×10^8	3.1×10^8	3.3×10^7	1.4×10^7	3.9×10^6	-	-	
Excess Waste Holdup Tank (1)	FB-B	Cylinder	1,340.00	3.9×10^4	2.2×10^5	1.1×10^5	4.9×10^4	1.7×10^4	-	-	
Decontamination Pit Collection Tank (1)	FB-B	Cylinder	1,340.00	3.4×10^4	2.2×10^5	1.1×10^5	4.9×10^4	1.7×10^4	-	-	
Pressurizer Relief Tank (1) (See Note 1)	RB-B	Cylinder	1,300.00	1.9×10^4	6.8×10^4	1.1×10^4	4.2×10^3	1.6×10^3	3.2×10^0	9.9×10^1	

(1) In addition to the activity given in table, the other activities are 1.8×10^5 (mev/cc-sec) and 1.6×10^4 (mev/cc-sec) for 6.13 (mev) and 7.11 (mev), respectively.

TABLE 12.1-4

RADIATION SOURCE TERMS FOR FILTERS

Source	Source Location	Source Geometry	Source Volume (Ft^3)	Activity (Mev/cc-sec) for Energy (Mev) of						
				0.40	0.80	1.30	1.70	2.20	2.50	3.50
Reactor Coolant Filter (1)	AB-M	Cylinder	0.40	-	5.7×10^{7}	1.5×10^{7}	-	-	-	-
Seal Water Return Filter (1)	AB-M	Cylinder	0.40	-	1.1×10^{7}	3.0×10^{6}	-	-	-	-
Seal Water Injection Filter (2)	AB-M	Cylinder	0.06	-	4.8×10^{7}	1.2×10^{7}	-	-	-	-
Recycle Evaporator Feed Filter (1)	AB-M	Cylinder	0.40	-	1.1×10^{7}	3.0×10^{6}	-	-	-	-
Recycle Evaporator Condensate Filter (1)	AB-M	Cylinder	0.04	1.6×10^{5}	8.0×10^{4}	3.3×10^{4}	-	-	-	-
Recycle Evaporator Concentrates Filter (2)	AB-M	Cylinder	0.04	-	1.2×10^{6}	-	-	-	-	-
Spent Fuel Purification Filter (2)	AB-M	Cylinder	0.40	-	5.7×10^{7}	1.5×10^{7}	-	-	-	-
Floor Drain Tank Filter (1)	AB-M	Cylinder	0.04	-	3.4×10^{7}	8.9×10^{6}	-	-	-	-

02-01

RN 03-038

TABLE 12.1-4 (Continued)

RADIATION SOURCE TERMS FOR FILTERS

Source	Source Location	Source Geometry	Source Volume (Ft^3)	Activity (Mev/cc-sec) for Energy (Mev) of 0.40	0.80	1.30	1.70	2.20	2.50	3.50	02-01
Spent Resin Sluice Filter (1)	AB-M	Cylinder	0.40	-	1.1×10^{7}	3.0×10^{6}	-	-	-	-	
Waste Monitor Tank Filter (1)	AB-M	Cylinder	0.04	-	2.8×10^{7}	5.7×10^{6}	2.8×10^{6}	1.1×10^{6}	-	-	
Waste Evaporator Condensate Filter (1)	AB-M	Cylinder	0.04	1.2×10^{5}	2.2×10^{5}	2.5×10^{4}	-	-	-	-	
Laundry and Hot Shower Filter (1)	AB-S	Cylinder	0.04	-	1.1×10^{7}	3.0×10^{6}	-	-	-	-	
Nuclear Blowdown Demineralizer Inlet Filter (1)	AB-M	Cylinder	0.50	3.3×10^{4}	1.2×10^{5}	1.5×10^{4}	2.4×10^{3}	6.7×10^{2}	-	-	02-01
Nuclear Blowdown Spent Resin Sluice Filter (1)	AB-M	Cylinder	0.50	3.3×10^{4}	1.2×10^{5}	1.5×10^{4}	2.4×10^{3}	6.7×10^{2}	-	-	
Nuclear Blowdown Demineralizer Outlet Filter (1)	AB-M	Cylinder	0.50	3.3×10^{4}	1.2×10^{5}	1.5×10^{4}	2.4×10^{3}	6.7×10^{2}	-	-	
Waste Gas Drain Filter (1)	AB-M	Cylinder	0.04	1.9×10^{6}	3.6×10^{5}	9.1×10^{4}	7.1×10^{4}	1.3×10^{5}	1.3×10^{5}	-	
Excess Liquid Waste Filter (1)	FB-B	Cylinder	0.24	-	3.4×10^{7}	8.9×10^{6}	-	-	-	-	02-01

TABLE 12.1-5

RADIATION SOURCE TERMS FOR HEAT EXCHANGERS

Source	Source Location	Source Geometry	Source Volume (Ft^3)	Activity (Mev/cc-sec) For Energy (Mev) of 0.40	0.80	1.30	1.70	2.20	2.50	3.50	
Regenerative Heat Exchanger (1)	RB-B	Cylinder	9.75	4.96×10^5	2.57×10^5	1.56×10^5	1.14×10^5	1.84×10^5	1.57×10^5	1.72×10^4	02-01
Letdown Heat Exchanger (1)	AB-B	Cylinder	48.05	1.48×10^5	8.42×10^4	5.23×10^4	3.77×10^4	6.39×10^4	4.94×10^4	5.81×10^3	
Seal Water Heat Exchanger (1)	AB-B	Cylinder	27.3	1.72×10^5	9.78×10^4	6.07×10^4	4.38×10^4	7.08×10^4	5.73×10^4	6.74×10^3	
Excess Letdown Heat Exchanger	RB-M	Cylinder	5.21	1.13×10^5	6.40×10^4	3.97×10^4	2.87×10^4	4.64×10^4	3.75×10^4	4.42×10^3	
Moderating Heat Exchanger (1)	AB-B	Cylinder	30.90	4.80×10^5	1.10×10^5	2.30×10^4	4.10×10^4	1.30×10^5	1.70×10^5	2.00×10^3	
Letdown Chiller Heat Exchanger (1)	AB-B	Cylinder	38.92	1.21×10^5	2.76×10^4	5.78×10^3	1.03×10^4	3.27×10^4	4.27×10^4	5.03×10^2	
Reactor Coolant Drain Tank Heat Exchanger (1)	RB-B	Cylinder	9.97	1.66×10^5	9.45×10^4	5.87×10^4	4.24×10^4	7.17×10^4	5.54×10^4	6.52×10^3	
Steam Generator Blowdown Heat Exchanger (3)	TB-B	Cylinder	75.3	1.42×10^1	5.26×10^1	2.93×10^1	6.55×10^0	1.73×10^1	6.89×10^{-1}	1.62×10^0	02-01
Letdown Reheat Heat Exchanger (1)	AB-B	Cylinder	1.92	4.80×10^5	1.10×10^5	2.30×10^4	4.10×10^4	1.30×10^5	1.70×10^5	2.0×10^3	
Spent Fuel Pit Heat Exchanger (2)	AB-S	Cylinder	154.0	-	9.85×10^0	5.24×10^1	6.4×10^{-1}	-	-	-	
Residual Heat Removal Heat Exchanger (2)	AB-B	Cylinder	208.3	1.32×10^5	4.90×10^4	1.66×10^4	1.09×10^4	9.42×10^3	1.58×10^4	-	

TABLE 12.1-6

NEUTRON SOURCE TERM FOR PRIMARY CONCRETE SHIELD

Neutron Fluxes on Inside Surface of Primary Concrete

Energy Group	Neutron Flux (Neutrons/cm^2-sec)
ϕ_1 ($E > 1.0$ Mev)	1.6×10^9
ϕ_1 (5.53 kev $< E \leq 1.0$ Mev)	2.2×10^{10}
ϕ_3 (0.625 ev $< E \leq 5.53$ Kev)	1.2×10^{10}
ϕ_4 ($E < 0.625$ ev)	3.3×10^9

Axial Variation of Neutron Fluxes

Distance Above/Below Core Centerline (ft)	Variation Factor
0	1.0
1	1.0
2	0.99
3	0.93
4	0.76
5	0.51
6	0.30

TABLE 12.1-7

GAMMA SOURCE TERM FOR PRIMARY CONCRETE SHIELD

Gamma Fluxes on Inside Surface of Primary Concrete

Group	Flux (Mev/cm^2-sec)	Group Energy (Mev/Y)
1	6.9×10^9	7.5
2	6.2×10^9	4.0
3	3.2×10^9	2.5
4	2.0×10^9	0.8

Axial Variation of Gamma Fluxes

Distance Above/Below Core Centerline (ft)	F_a
0	1.0
1	1.0
2	0.99
3	0.93
4	0.76
5	0.51
6	0.30

TABLE 12.1-8

PRESSURIZER DEPOSITED ACTIVITY

Isotope	Activity (μci/cm^2)
Cr-51	9.8×10^{-2}
Mn-54	1.5×10^{-1}
Mn-56	2.2×10^{-2}
Co-58	3.8
Co-60	1.6×10^{-1}
Fe-59	1.4×10^{-1}

TABLE 12.1-9

REACTOR COOLANT N^{16} ACTIVITY

Position in Loop	Loop Transit Time (sec)	N^{16} Activity μCi/gram	02-01
Leaving core	0	132	
Leaving reactor vessel	1.0	119	
Entering steam generator	1.4	110	
Leaving steam generator	6.6	74	
Entering reactor coolant pump	7.2	70	
Entering reactor vessel	7.9	65	
Enter Core	10.4	52	
Leaving core	11.3	132	

TABLE 12.1-10

RADIATION SOURCE TERM - LETDOWN COOLANT

Gamma Energy (Mev/Y)	Specific Source Strength (Mev/gm-sec)
0.4	5.1×10^5
0.8	2.7×10^5
1.3	1.8×10^5
1.7	1.3×10^5
2.2	2.2×10^5
2.5	1.7×10^5
3.5	2.0×10^4

TABLE 12.1-11

SOLID WASTE SOURCE FOR DRUMMING STATION

Gamma Energy (Mev/Y)	Spent Resin Source Strength (Mev/cc-sec)
0.8	1.0×10^8
1.3	1.0×10^7

RN 08-003

TABLE 12.1-12

CORE AVERAGE GAMMA RAY SOURCE STRENGTHS AT VARIOUS TIMES AFTER SHUTDOWN

Energy Group	Source Strengths at Time After Shutdown (MeV / Watt-sec)								
MeV / gamma	12 Hours	24 Hours	100 Hours	1 Week	1 Month	3 Months	6 Months	1 Year	5 Years
0.20 - 0.40	1.8e+09	1.5e+09	8.3e+08	5.9e+08	1.5e+08	5.2e+07	3.1e+07	1.8e+07	2.3e+06
0.40 - 0.90	1.1e+10	9.7e+09	6.6e+09	5.7e+09	3.6e+09	2.1e+09	1.2e+09	5.4e+08	1.5e+08
0.90 - 1.35	2.0e+09	1.4e+09	7.3e+08	5.6e+08	1.7e+08	4.8e+07	3.2e+07	2.4e+07	9.3e+06
1.35 - 1.80	3.7e+09	3.3e+09	2.7e+09	2.3e+09	6.5e+08	4.8e+07	2.0e+07	1.5e+07	2.8e+06
1.80 - 2.20	3.4e+08	2.6e+08	1.8e+08	1.5e+08	6.0e+07	2.0e+07	1.4e+07	8.6e+06	2.5e+05
2.20 - 2.60	2.4e+08	1.8e+08	1.5e+08	1.3e+08	3.9e+07	1.5e+06	1.1e+04	0	0
2.60 - 3.00	6.2e+06	3.2e+06	2.7e+06	2.3e+06	6.7e+05	2.6e+04	2.0e+02	0	0
3.00 - 4.00	4.3e+06	1.3e+06	1.1e+06	9.1e+05	2.6e+05	1.0e+04	0	0	0
Total	2.0e+10	1.6+10	1.1e+10	9.4e+09	4.7e+09	2.3e+09	1.2e+09	6.1e+08	1.7e+08

RN
02-025

TABLE 12.1-13

IRRADIATED Ag-In-Cd CONTROL ROD SOURCE STRENGTHS

Photon Energy (MeV / gamma)	Source Strength at Time After Shutdown (MeV / cm^3-sec)				
	1 Day	1 Week	1 Month	6 Months	1 Year
0.20 - 0.40	2.3e+08	2.3e+08	2.2e+08	1.4e+08	8.7e+07
0.40 - 0.90	1.1e+12	1.1e+12	1.0e+12	6.7e+11	4.1e+11
0.90 - 1.35	2.0e+11	1.9e+11	1.8e+11	1.2e+11	7.3e+10
1.35 - 1.80	3.8e+11	3.8e+11	3.5e+11	2.3e+11	1.4e+11

RN 02-025

The absorber cross-sectional area is 0.589 square centimeters per rod.
The absorber material density is 10.17 grams per cubic centimeter.

TABLE 12.1-14

IRRADIATED INCORE DETECTOR DRIVE CABLE SOURCE STRENGTHS

Energy Group MeV / gamma	Source Strength Time After Shutdown (MeV / cm^3-sec)						
	8 Hours	1 Day	1 Week	1 Month	6 Months	1 Year	5 Years
0.20 - 0.40	5.9e+08	5.8e+08	5.0e+08	2.9e+08	1.0e+07	2.9e+05	0
0.40 - 0.90	1.7e+10	1.2e+10	1.2e+10	1.1e+10	8.0e+09	5.1e+09	2.0e+08
0.90 - 1.35	1.6e+10	1.6e+10	1.5e+10	1.2e+10	5.7e+09	4.3e+09	2.6e+09
1.35 - 1.80	2.1e+07	1.1e+07	1.1e+07	8.7e+06	2.0e+06	3.4e+05	0
1.80 - 2.20	4.6e+09	6.2e+07	0	0	0	0	0
2.20 - 2.60	1.5e+08	2.0e+06	0	0	0	0	0
2.60 - 3.00	1.6e+08	2.1e+06	0	0	0	0	0
3.00 - 4.00	3.7e+07	5.1e+05	0	0	0	0	0

The drive cable effective cross-sectional area is 0.095 square centimeters.

RN
02-025

TABLE 12.1-15

DEPOSITED CORROSION PRODUCT ACTIVITY
ON STEAM GENERATOR SURFACES

(μCi/cm^2)

	Operating Time (Months)				
Isotope	0	6	12	24	36
Mn-54	0.0	0.15	0.60	1.5	2.0
Mn-56	0.0	3.3	3.3	3.3	3.3
Co-58	0.0	4.5	10.2	11.0	11.0
Fe-59	0.0	1.4	3.0	3.0	3.0
Co-60	0.0	0.20	0.80	2.0	3.5

TABLE 12.1-16

RESIDUAL HEAT REMOVAL LOOP SHUTDOWN SOURCE

Gamma Energy (MeV/Y)	Specific Source Strength (Sec/gm-sec)
0.4	3.5×10^5
0.8	1.3×10^5
1.3	4.4×10^4
1.7	2.9×10^4
2.2	2.5×10^4
2.5	4.2×10^4

TABLE 12.1-19

AREA RADIATION MONITORS

Monitor	Function	Sensitivity	Power Source	
RM-G1 Control Room	Monitor control room operational area	10^{-1} to 10^{4} mR/hr	Bus A	02-01
RM-G2 Radiochemical Lab	Monitor radiochemical lab area	10^{-1} to 10^{4} mR/hr	Bus A	
RM-G3 Sampling Room	Monitor sampling room near entrance	10^{-1} to 10^{4} mR/hr	Bus B	
RM-G4 Hot Machine Shop	Monitor area in hot machine shop	10^{-1} to 10^{4} mR/hr	Bus B	
RM-G5 Reactor Building Personnel Access	Monitor area inside containment near access, alarm inside & outside of access	10^{-1} to 10^{4} mR/hr	Bus A	
RM-G6 Reactor Building Refueling Bridge	Monitor pool area and fuel handling	10^{-1} to 10^{4} mR/hr	Bus A	
RM-G7 Reactor Building High Range	Provide indication of magnitude of LOCA	1.0 to 10^{7} R/hr	Bus B	RN 02-025
RM-G8 Fuel Handling Building Fuel Handling Bridge	Monitor pool area and fuel handling	10^{-1} to 10^{4} mR/hr	Bus B	RN 03-032

TABLE 12.1-19 (Continued)

AREA RADIATION MONITORS

Monitor	Function	Sensitivity	Power Source	
RM-G9 Auxiliary Building Demineralizer Area	Monitor access area near demineralizers	10^{-1} to 10^{4} mR/hr	Bus A	02-01
RM-G10 Auxiliary Building Waste Gas Decay Tank Area	Monitor corridor near tank area	10^{-1} to 10^{4} mR/hr	Bus A	02-01
RM-G11 Auxiliary Building Drumming Area	Monitor operations area	10^{-1} to 10^{4} mR/hr	Bus B	
RM-G12 Auxiliary Building Waste Holdup Tank Area	Monitor corridor near tank area	10^{-1} to 10^{4} mR/hr	Bus B	
RM-G13 Auxiliary Building Charging Pump Area	Monitor access corridor to pump cubicles	10^{-1} to 10^{4} mR/hr	Bus A	
RM-G14 Reactor Incore Instrument Area	Monitor near seal table of incore instrumentation	10^{-1} to 10^{4} mR/hr	Bus A	
RM-G15 Movable Unit	Recording backup unit	10^{-1} to 10^{4} mR/hr	120 volt outlets	02-01
RM-G16 Turbine Building	Monitor area on main operating floor	10^{-1} to 10^{4} mR/hr	Bus A	02-01

TABLE 12.1-19 (Continued)

AREA RADIATION MONITORS

Monitor	Function	Sensitivity	Power Source	
RM-G17A Reactor Building Manipulator Crane	Monitor fuel handling area during refueling & close purge supply & exhaust isolation valves on high rad.	1 to 10^5 mR/hr	Bus A	RN 02-025
RM-G17B Reactor Building Manipulator Crane	Monitor fuel handling area during refueling and close purge supply and exhaust isolation valves on high rad.	1 to 10^5 mR/hr	Bus B	02-01
RM-G18 Reactor Building High Range	Provide indication of magnitude of LOCA	1 to 10^7 R/hr	Bus A	

TABLE 12.1-20

RADIATION DOSE RATES IN SELECTED INPLANT LOCATIONS

Location Normal Operation and Anticipated Operational Occurrences	Design Dose Rate (mrem/hr)	
Control Room	≤ 1	
Reactor Building Operating Floor	≤ 100	02-01
Auxiliary Building Corridors	≤ 2.5	
Valve Galleries	≤ 100	
Fuel Handling Building	≤ 2.5	
Radwaste Processing Area (control panel)	≤ 2.5	
Health Physics Control Complex	≤ 1	
Hot Machine Shop (general area)	≤ 2.5	
Turbine Building	≤ 1	
Service Building	≤ 1	

02-01

TABLE 12.1-21

ESTIMATES OF PERSONNEL OCCUPANCY
REQUIREMENTS IN SPECIFIC RADIATION ZONES

Personnel Type	Estimated Number Required	Occupancy in Zones I	II	III
Administrative	19	0.90	0.10	0
Operations	31	0.75	0.23	0.02
Maintenance	34	0.60	0.38	0.02
Technical Support	25	0.60	0.38	0.02

\

TABLE 12.1-22

02-01

MAN-REM ESTIMATES FOR NORMAL PLANT OPERATION, ANTICIPATED OPERATIONAL OCCURRENCES AND ROUTINE MAINTENANCE

Personnel Type	Zone I (mrem/hr)		Zone II (mrem/hr)		Zone III (mrem/hr)	
	0	1.0	1.0	2.5	2.5	15
Administrative	0	34	4	10	0	0
Operations	0	47	14	35	3	19
Maintenance	0	41	26	65	3	20
Technical Support	0	30	19	48	3	15

02-01

TOTALS:

1. Estimated Lower Limit 72 mrem
2. Estimated Upper Limit 364 mrem

TABLE 12.1-22a

MAN-REM DOSE PER TASK - NORMAL OPERATION

Task	Est. Avg Dose Rate (mRem/hr)	Duration (manhours/ task-year)	Calculated Man-Rem/yr	
Routine Patrol	2.5	1,100	3	02-01
Radiation Surveys	2.5	2,220	6	
Radiation Surveys	15	725	11	
Radiation Surveys	< 100	25	9	
Periodic Tests, Calibrations and Inspections	2.5	625	2	
Periodic Tests, Calibrations and Inspections	15	300	5	
Routine Maintenance:				
Pumps	50	100	5	
Valves	100	100	10	
Misc. Items	40	150	6	
Waste Disposal, Drumming, and Shipping	15	2,200	33	02-01
Filter Changes	25	100	3	
Control Room Operation	0.1	17,500	2	

TABLE 12.1-22b

MAN-REM DOSE PER TASK - REFUELING OUTAGE

Task	Est. Avg. Dose Rate (mRem/hr)	Duration (manhours/ task-year)	Calculated Man-Rem/yr	
Health Physics Coverage	25	750	19	02-01
Head Removal and Installation	27 (1)	1,500	40	
Fuel Handling	20	400	8	
Incore Instrumentation Removal and Replacement	50	200	10	02-01
Integrated Leak Rate Test	5	400	2	
Inservice Inspection	75	520	39	
Insulation Removal	50	200	10	
Routine Maintenance:				
Pumps	50	150	8	
Valves	100	150	15	
Misc. Items	40	300	12	
Special Maintenance:				
Steam Generator Testing and Repair	100 (1)	1,000 (2)	100 (3)	
Reactor Coolant Pump Seal Inspection and Repair	80	400 (2)	32 (3)	
Snubber Inspection	2.5	2,000 (2)	5 (3)	
Evaporator Maintenance	60	100 (2)	6 (3)	

NOTES:

1. See Vance, J., Weaver, C. L., and Lepper, E. M., "A Preliminary Assessment of the Potential Impacts on Operating Nuclear Power Plants of a 500 mrem/yr Occupational Exposure Limit," report to the Nuclear Regulatory Staff of the Atomic Industrial Forum, April, 1978, for a detailed breakdown of task.

2. Manhours/task

3. Man-Rem

TABLE 12.1-22c

PERCENT TIME OF WORKER EFFORT PER TASK - NORMAL OPERATION

Task	Man-Rem/yr	Percent C.R. Ops.	Plt. Ops.	Eng.	QA	Mech.	Elec.	I&C	Chem./ H.P.
Routine Patrols	3	-	70	-	-	10	10	10	-
Routine Surveys	26	-	-	-	-	-	-	-	100
Periodic Tests, Calibration and Inspections	7	-	-	5	5	25	25	25	15
Routine Maintenance	21	-	-	-	-	40	40	20	-
Waste Disposal, Drumming and Shipping	33	-	70	-	-	-	-	-	30
Filter Changes	3	-	20	-	-	40	-	-	40
Control Room Operation	2	100	-	-	-	-	-	-	-

TABLE 12.1-22d

PERCENT TIME OF WORKER EFFORT PER TASK - REFUELING OUTAGE

02-01

Task	Man-Rem/yr	Percent C.R. Ops.	Plt. Ops.	Eng.	QA	Mech.	Elec.	I&C	Chem./ H.P.
Health Physics Coverage	20	-	-	-	-	-	-	-	100
Head Removal and Installations	40	-	-	-	10	60	15	15	-
Fuel Handling	8	-	90	10	-	-	-	-	-
Incore Instrumentation Removal and Replacement	10	-	-	-	-	20	40	40	-
Integrated Leak Rate Test	2	-	50	20	-	-	-	20	10
Inservice Inspection	39	-	100	-	-	-	-	-	-
Insulation Removal	10	-	25	-	-	75	-	-	-
Routine Maintenance	35	-	-	-	-	40	40	20	-

TABLE 12.1-22d (Continued)

PERCENT TIME OF WORKER EFFORT PER TASK - REFUELING OUTAGE

Task	Man-Rem/yr	Percent C.R. Ops.	Plt. Ops.	Eng.	QA	Mech.	Elec.	I&C	Chem./ H.P.
Special Maintenance:									
Steam Generator Testing and Repair	100	-	80	-	5	15	-	-	-
Reactor Coolant Pump Seal Inspection and Repair	32	-	10	-	-	90	-	-	-
Snubber Inspection	5	-	40	-	-	60	-	-	-
Evaporator Maintenance	6	-	20	-	-	50	15	15	-

TABLE 12.1 22e

CALCULATED MAN-REM/YEAR PER IDENTIFIED TASK

	Personnel Group							
	C.R. Ops.	Plt. Ops.	Eng.	QA	Mech.	Elec.	I&C	Chem/ H.P.
No. of Personnel	(12)	(19)	(5)	(2)	(18)	(8)	(8)	(18)
Normal Operation	2 (1)	25.5	2	1	12	11	7	38
Refueling Outage	-	49.7	2	4	48	24	18	21
Total (Man-Rem)	2	75.2	4	5	60	35	25	59
Dose/Individual	0.17	4.0	2	2.5	3.3	4.4	3.1	3.2

02-01

1. Man-Rem/year; special maintenance has been excluded.

Sample Calculation: Group – Plant Operators

$$\text{Man-rem}_{T_i} = (\text{Dose Rate})_{\text{Task}_i}\,(\text{Manhours})_{\text{Task}_i}\,(\text{Effort})_{\text{Task}_i}$$ 02-01

Normal Operation:

Routine Patrols:	(2.5 mRem/hr) (1100 manhours) (0.7) = 1.9
Waste Disposal:	(15 mRem/hr) (2200 manhours) (0.7) = 23.1
Filter Changes:	(25 mRem/hr) (100 manhours) (0.2) = 0.5

Refueling Outage:

Fuel Handling:	(20 mRem/hr) (400 manhours) (0.9) = 7.2
Integrated Leak Rate Test:	(5 mRem/hr) (400 manhours) (0.5) = 1
In-Service Inspection:	(75 mRem/hr) (520 manhours) (1) = 39
Insulation Removal:	(50 mRem/hr) (200 manhours) (0.25) = 2.5

02-01

TABLE 12.1 22e (Continued)

Sample Calculation: Group - Plant Operators

TOTALS:

Normal Operation:	25.5 man-Rem/yr
Refueling Outage:	49.7 man-Rem/yr
Summation:	75.2 man-Rem/yr
Avg. Dose/Individual:	$\frac{75.2 \text{ man-Rem/yr}}{19 \text{ men}}$ = 4.0 rem/yr

02-01

TABLE 12.1-22f

ESTIMATED RADIATION DOSE RATES FROM AIRBORNE CONTAMINATION IN SPECIFIC AREAS OF THE PLANT

Plant Area	Whole Body Dose Rate, Gamma (mRem/hr)	Thyroid (mRem/hr)	Tritium Inhalation (mRad/hr)	
Turbine Building	2.94-7	1.0-4	3.98-5	02-01
Auxiliary Building	8.42-3	5.7-2	1.94-3	
Letdown Heat Exchanger	1.20+0	7.8	3.96-2	
Reactor Building	2.77+1	1.10+2	7.13+1	

TABLE 12.1-22g

ESTIMATED MAN-REM DOSES FROM AIRBORNE CONTAMINATION

Whole Body Gamma Dose

Personnel Type	Zone 1	Zone 2	Zone 3
Administrative	1.01-5	3.20-2	0
Operations	1.37-5	1.20-1	1.49+0
Maintenance	1.20-5	2.18-1	1.63+0
Technical Support	8.82-6	1.60-1	1.2+0

Thyroid Inhalation Dose

Personnel Type	Zone 1	Zone 2	Zone 3
Administrative	3.42-3	2.17-1	0
Operations	4.65-3	8.13-1	9.67+0
Maintenance	4.08-3	1.47+0	1.06+1
Technical Support	3.00-3	1.71+0	7.8+0

Tritium Inhalation Dose

Personnel Type	Zone 1	Zone 2	Zone 3
Administrative	1.36-3	7.37-3	0
Operations	1.85-3	2.77-2	4.91-2
Maintenance	1.62-3	5.01-2	5.39-2
Technical Support	1.19-3	3.70-2	3.96-2

Sample Calculation:

$$(2.94-7 \text{ mRem/hr})\left(\frac{\text{Rem}}{1000 \text{ mRem}}\right)\left(\frac{2000 \text{ hrs}}{\text{yr}}\right)(19 \text{ people})(0.9 \text{ fraction of time in particular zone})$$
$$= 1.05-5 \text{ man}-\text{Rem/yr}$$

TABLE 12.1-24

ESTIMATED POST LOCA RESPONSE OF CONTAINMENT MONITOR RM-G7

Post LOCA Time	Estimated Radiation Inside Containment (M-Rad/hr)	Average Gamma Energy (Mev)	Anticipated Response RM-G7 (M-Rad/hr)
1-min.	.060	1.075	.049
1-hr.	3.70	0.648	3.09
12-hr.	1.50	0.466	1.22
1-day	1.20	0.458	0.96
4-day	0.84	0.488	0.66
10-day	0.46	0.507	0.36
30-day	0.31	0.543	0.25
60-day	0.26	0.577	0.22

02-01

REFERENCE LEGEND FOR SECTION 12.1 FIGURES
PLANT RADIATION ZONE DESIGNATIONS

Zone	Color Code	Description	Dose Rate (mRem/hr)
I		Uncontrolled No restrictions on occupancy expected	< 1.0 mRem/hr
II		Controlled Unlimited access, 40 hrs/week	< 2.5 mRem/hr
III		Controlled Limited access, 6 to 40 hrs/week	< 15.0 mRem/hr
IV		Limited access for short periods, 1 to 6 hrs/week	< 100 mRem/hr
V		Controlled, High Radiation Area Occupancy averages less than one hour per week	> 100 mRem/hr

Notes For Layout Drawings
Showing Radiation Color Code

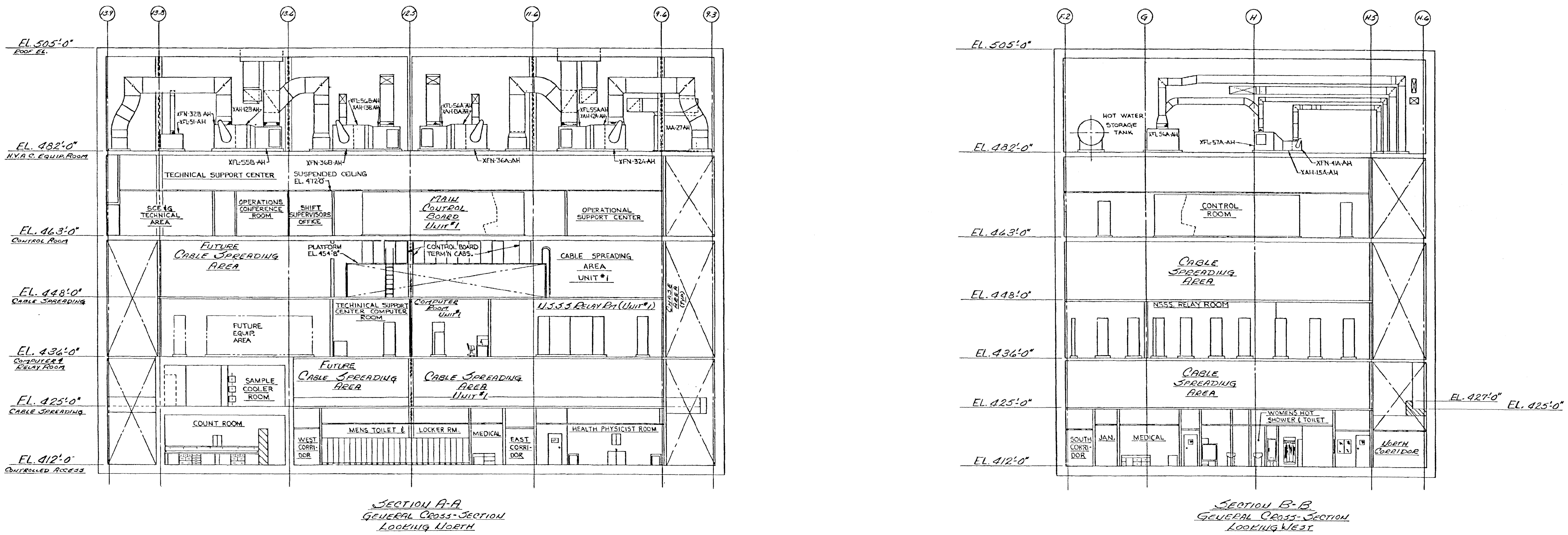
Figures 12.1-12, 12.1-13, 12.1-17 and 12.1-18

The solid color of the cross-hatched areas represents the radiation zone designations during a shutdown/refueling period while the RHR system is not in operation. The cross-hatched color represents the radiation zone designations during a shutdown/refueling period while the RHR system is in operation.

Figure 12.1-16

The solid color of the cross-hatched areas represents the radiation zone designations during a shutdown/refueling period while the refueling canal and the region above the reactor vessel are flooded. The cross-hatched color represents the radiation zone designations during a shutdown/refueling period while the refueling canal and the region above the reactor vessel are not flooded.

Figure 12.1-0



NOT FOR CONSTRUCTION

Amendment 02-01
May 2002

NUCLEAR SAFETY RELATED

FSAR Figure 12.1-20
CONTROL COMPLEX SECTIONS
RADIATION EXPOSURE ZONES

FSAR Figure 1.2-17
ALSO ASSOC. WITH FSAR FIG. 12.1-20

SOUTH CAROLINA ELECTRIC & GAS COMPANY
VIRGIL C. SUMMER NUCLEAR STATION

PLANT LAYOUT — FINAL
MISC. SECTION THRU CONTROL BUILDING

DESIGN ENGINEERING
SCE&G A SCANA COMPANY — V. C. SUMMER NUCLEAR STATION JENKINSVILLE, S. C.

MADE	CHECKED	LE APPROVAL
DDJ	MGR	DDJ

NO.	DATE	BY	REVISION	CKD. BY	LE APPROVAL
10	9/17/01	DDJ	REVISED PER ECR-70028	MGR	DDJ

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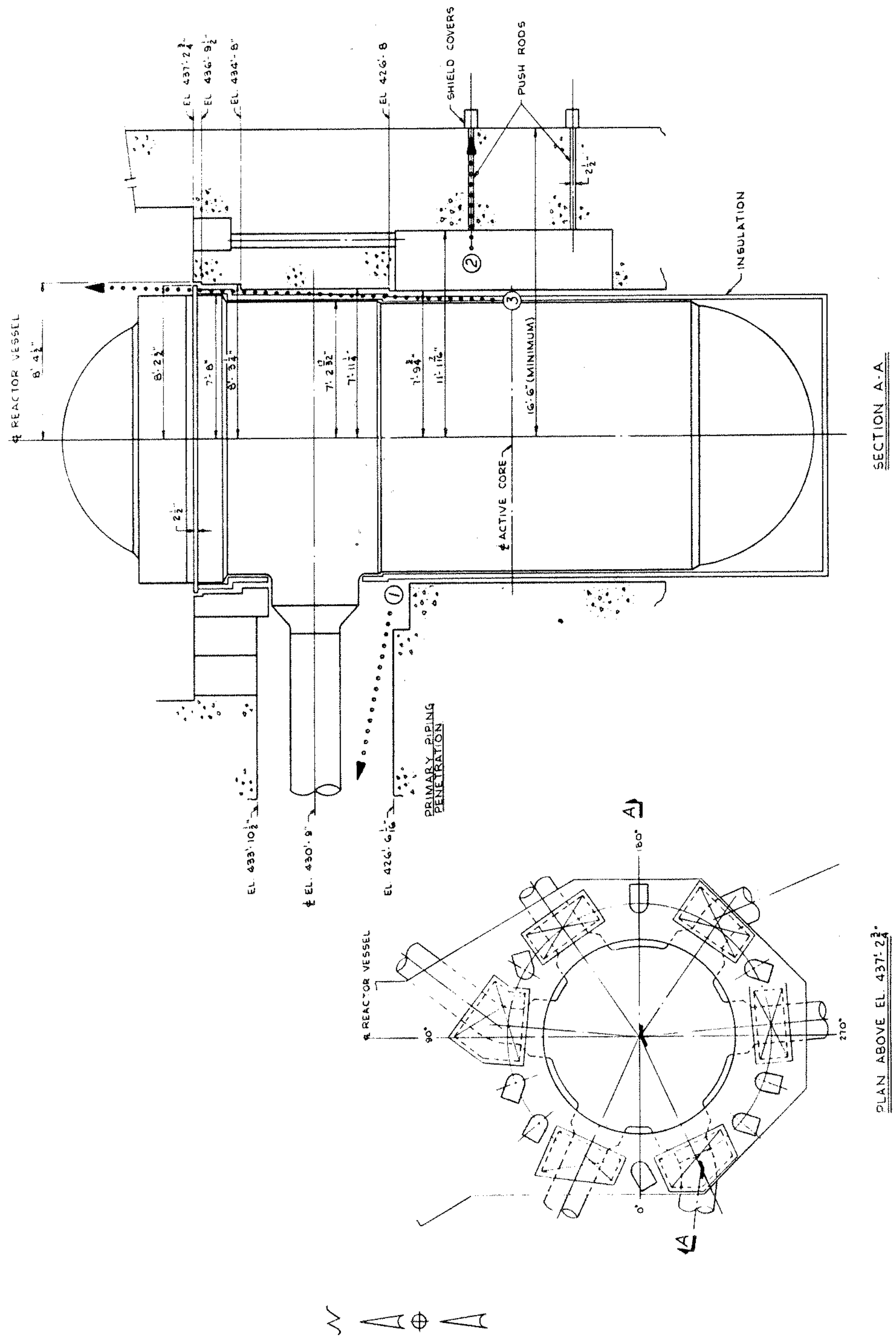
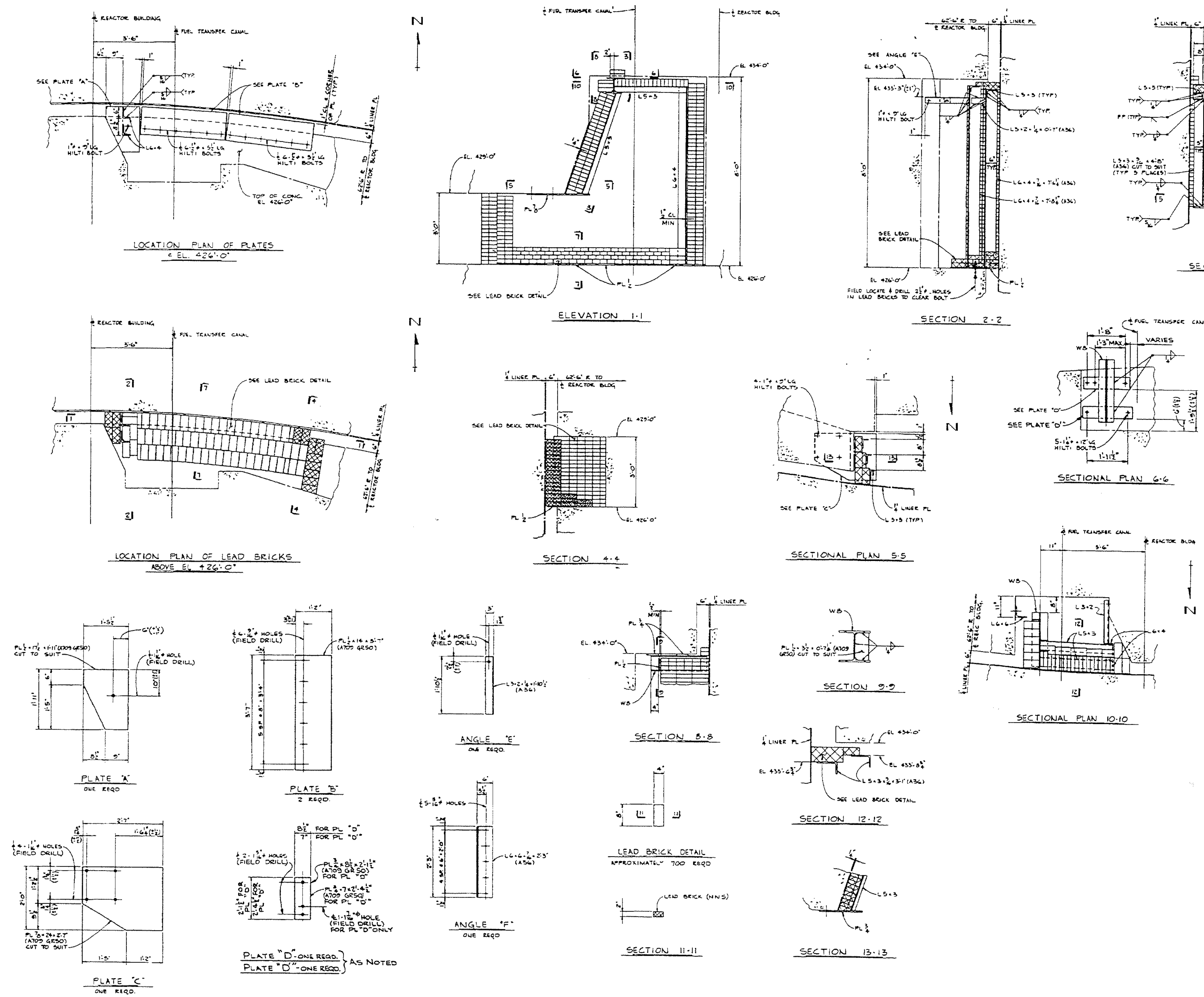


SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Primary Shield Streaming Paths

Figure 12.1-21

Amendment 0
August 1984



NOTES:-

1. NUCLEAR SAFETY RELATED (NSR). FOR ITEMS SPECIFICALLY MARKED NON-NUCLEAR SAFETY (NNS), SPECIFICATION SP-701-4461-00 IS NOT APPLICABLE.
2. QUALITY ASSURANCE PROGRAM REQUIREMENTS QA-1. SEE GAI SPECIFICATION QUALITY ASSURANCE PROGRAM REQUIREMENTS SP-701-4461-00.
3. FABRICATION OF MISCELLANEOUS STEEL TO BE IN ACCORDANCE WITH GAI SPECIFICATION MISCELLANEOUS STEEL SP-627-044461-000.
4. ERECTION OF STRUCTURAL STEEL TO BE IN ACCORDANCE WITH GAI SPECIFICATION ERECTION OF STRUCTURAL STEEL SPECIFICATION SP-208-044461-000.
5. ALL EXPOSED CARBON STEEL SURFACES TO BE PAINTED IN ACCORDANCE WITH GAI SPECIFICATION PAINTING STEEL INSIDE REACTOR BUILDING SP-713-4461-00.

REFERENCES:-

E-411-234 REACTOR BUILDING INTERIOR WALLS - PRIMARY SHIELD WALL AND FUEL TRANSFER CANAL WALLS - EL. 423'-0" AND EL. 426'-8-1/4" - CONCRETE OUTLINE

AMENDMENT 02-01
MAY 2002

THIS IS A NUCLEAR SAFETY RELATED DOCUMENT. NO DEVIATION SHALL BE INITIATED OR PERFORMED WITHOUT PRIOR DOCUMENTATION AND WRITTEN APPROVAL

FSAR Figure 12.1-22

SOUTH CAROLINA ELECTRIC & GAS COMPANY
VIRGIL C. SUMMER NUCLEAR STATION
REACTOR BUILDING - MISCELLANEOUS STEEL
FUEL TRANSFER CANAL SHIELDING

DESIGN ENGINEERING
V. C. SUMMER NUCLEAR STATION JENKINSVILLE, S.C.

MADE	CHECKED	LE APPROVAL
DDJ	MGR	DDJ

E-511-208 | 3

NO	DATE	BY	REVISION	CKD. BY	LE APPROVAL
3	9/26/01	DDJ	REVISED PER ECR-70028	MGR	DDJ

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RN 01-045

12.2 VENTILATION

12.2.1 DESIGN OBJECTIVES

The ventilation systems for various buildings and areas are designed to maintain ambient air temperatures suitable for personnel and equipment and to aid in minimizing radioactivity levels so that personnel are not exposed to airborne concentrations which exceed the limits of 10 CFR 20.

Calculated airborne radioactivity levels for various buildings during normal operation, including anticipated operational occurrences, are presented in Table 12.2-1, based upon the assumptions discussed in Sections 12.2.3 and 12.2.6. Calculated inhalation dose rates to personnel in various buildings are discussed in Sections 12.2.6.1.1, 12.2.6.1.2, and 12.2.6.1.3.

12.2.2 DESIGN DESCRIPTION

Ventilation systems serving potentially radioactive areas and the Sections where detailed descriptions are presented are as follows:

1. Control Room Ventilation System, Section 9.4.1.2.1.

2. Controlled Access Area Exhaust System, Section 9.4.1.2.5.

3. Auxiliary and Radwaste Area Ventilation Systems, Section 9.4.2.

4. Spend Fuel Pool Area Ventilation System, Section 9.4.3.

5. Turbine Building Ventilation Systems, Section 9.4.4.

6. Reactor Building Charcoal Cleanup System, Section 9.4.8.2.4. RN 02-002

Air flow rates and approximate free volumes of buildings served by these ventilation systems are presented in Table 12.2-2.

The ventilation system protective features concerning airborne and effluent radioactivity and shielding are described in Sections 11.4, 12.1, 12.1.4, and 12.2.4.

Figures 1.2-15 and 12.2-1 indicate the arrangement of the control room emergency charcoal filter plenum components and available service space. Figure 12.2-1 is representative of all charcoal-HEPA filters in the plant. Test connections are provided in the duct system or housings after installation is completed. RN 02-002

Criteria for change of filters is dependent upon results of laboratory and field tests of filter system components. Test requirements are listed in Sections 6.5.1.4, 9.4.1.4, 9.4.2.4, 9.4.3.4, and 9.4.8.4.

Regulatory Guide 1.52 is discussed in Appendix 3A while design guidance contained in Regulatory Guide 8.8 is further described in Section 12.0.

12.2.3 SOURCE TERMS

The airborne radioactivity which workers might be exposed to during normal operations have been considered as follows:

The concentration of radioactive gases in the coolant prior to reactor vessel head
removal is established based on DAC considerations and containment ventilation RN
system capabilities of the plant. The major isotope of concern is Xe-133, since the 01-119
other noble gas isotopes have either decayed away or been removed during the system
degasification period. With conservative calculations, RCS activity which would result in
the 10CFR20 occupational DAC in the containment atmosphere is approximately RN
0.1 μc/cc of Xe-133. However, operating plant experience has indicated that no 01-119
problem with fission gas release to the containment atmosphere was incurred when the
vessel head was removed, with the coolant Xe-133 concentration reduced to .05 μc/gm.
Also, the exposure period is short so long as the containment purge is in operation
during head removal.

The discharge of relief valves on lines which contain reactor coolant is piped to tanks, such as the pressurizer relief tank, volume control tank, recycle holdup tanks, reactor coolant drain tank, or waste holdup tank. Consequently, exposure of workers to airborne radioactivity from this source is precluded.

The gaseous activity released to the refueling water during the movement of spent fuel is expected to be insignificant. Fission product release from defective fuel cladding during refueling operations is a relatively slow process as compared to release during normal operation. Shutdown escape rate coefficients are several orders of magnitude below those during operation. Also, the noble gas activity inventories available for release from the defective fuel are reduced by radioactive decay during the fuel transfer operations.

Personnel entry into the Reactor Building at power will be limited to that absolutely necessary for essential operations. Appropriate radiological safety precautions will be implemented if entry into the reactor building at power is required.

The Reactor Building is normally purged prior to entry under shutdown conditions. If the shutdown is for maintenance, airborne activity is expected to be minimal. If the shutdown is for refueling, evaporation of tritiated water from the refueling cavity is expected to be the only source of airborne activity. As discussed in Section 11.1.4.3, airborne tritium concentrations during refueling will be low enough to allow 40 hour per week access to the reactor building if the tritium concentration in the refueling water is limited to 2.5 μCi/gm.

Airborne radioactivity is introduced into plant buildings when equipment processing or holding radioactive materials develops a leak. The assumed leakage rates into various buildings and the assumptions concerning the amount of radioactive material released from the leaked materials are presented in Sections 12.2.3.1 through 12.2.3.4. The values presented are based upon data presented in NUREG-0017[1]. RN 02-002

12.2.3.1 Assumptions Used in Determining Airborne Activity in the Reactor Building

1. Leakage to the Reactor Building is 1 percent per day of the primary coolant noble gas inventory and 0.001 percent per day of the primary coolant halogen inventory. It is conservatively assumed that the tritium leakage rate is equivalent to the noble gas leakage rate.

2. Primary coolant inventory is based upon values provided in Table 11.1-5. RN 01-111

3. A Reactor Building purge frequency of 24 per year is assumed.

4. Radioactive nuclides are removed between purges by radioactive decay only. RN 01-111

12.2.3.2 Assumptions Used in Determining Airborne Activity in the Auxiliary Building

1. Leakage to the Auxiliary Building is 160 lb/day at primary coolant activity.

2. Primary coolant inventory is based upon values provided in Table 11.1-5. RN 01-111

3. A partition factor of 1 is assumed for noble gases and 0.0075 for halogens.

12.2.3.3 Assumptions Used in Determining Airborne Activity in the Turbine Building

1. Leakage to the Turbine Building atmosphere is 1700 lb/hr of steam at main steam activity.

2. Main steam activity is based upon values provided in Table 11.1-5. RN 01-111

3. A partition factor of 1 is assumed for both noble gases and halogens.

12.2.3.4 Assumptions Used in Determining Airborne Activity in Other Buildings

All other buildings are expected to have negligible noble gas and halogen airborne activity.

12.2.4 AIRBORNE RADIOACTIVITY MONITORING

12.2.4.1 Design Basis

The airborne radioactivity monitors are designed to detect, indicate, annunciate and record the radioactivity levels monitored at selected locations inside the plant. These monitors provide supporting data for the surveillance of plant radioactivity levels as recommended by ANSI N13.2[2] and, Regulatory Guides 8.2 and 8.8, and as required by 10 CFR 20. Fixed continuous airborne radioactivity monitors taking particulate samples from ventilation ducts utilize isokinetic nozzles for air sampling designed according to ANSI N13.1-1969. Lengths of sample lines are minimized to reduce particulate sample loss. Sample point locations are selected to obtain an optimum mixed air sample. Areas not covered by airborne radioactivity monitors are periodically checked with movable air samplers. The Reactor Building and normally occupied areas which are considered sources of airborne activity such as the spent fuel area and the sampling room are monitored with equipment sensitive to DAC levels. Controlled access areas which contain sources of airborne activity require Health Physics air sampling prior to work in these areas. Ventilation air flow design provides clean air for normal access corridors. These corridors are periodically monitored by portable air samplers. This monitoring instrumentation aids in maintaining airborne exposure as low as reasonably achievable (ALARA). RN 01-119

12.2.4.2 Airborne Radioactivity Monitors System Description

The airborne monitors consist of ten fixed and three movable channels. Some of these channels are used for effluent monitoring, as discussed in Section 11.4.

Ventilation ducts, local areas, and effluent paths are monitored in selected locations, as listed in Table 12.2-3, to provide assurance of proper plant operation.

Ventilation air flow in the Auxiliary Building is so arranged that the air flow path is in the direction of progressively greater potential contamination. This is accomplished by supplying clean air to the normally accessible areas, which are considered to be relatively free of airborne activity, and removing air from areas that could potentially have high levels of airborne activity. These areas are considered to be controlled areas, access to which is under strict administrative control. The air in the Auxiliary Building discharge path contains air from both clean and potentially contaminated areas. Thus, this air would not be representative of air breathed by personnel in the normally accessible areas regardless of whether samples are obtained upstream or downstream of the filters.

In the Fuel Handling Building, the spent fuel area could be considered as a source of airborne activity. Since this is a normally accessible area, the local air and the exhaust air prior to filtration are monitored for airborne activity as discussed in Sections 12.2.4.2.5 and 12.2.4.2.7. RN 02-002

In the control complex, the sampling room could be considered as a source of airborne activity. This air is monitored for airborne activity as discussed in Section 12.2.4.2.6. RN 02-002

Periodic surveillance of accessible areas, in accordance with Sections 12.1.5 and 12.3.2, will be performed.

Reliable power for the fixed instrumentation is obtained from the diesel backed, 120 volt instrument buses. Associated sample pumps obtain power from the 480 volt diesel backed buses. This assures continuity of operation in the event of a loss of offsite power. Measured activity levels are indicated and recorded (except RM-A11) on the Radiation Monitoring System control panel located in the control room. Local indication is provided for each channel. A differential pressure switch is provided on the particulate and iodine collection filter holders to cause an alarm on filter blockage. Another differential pressure switch is provided across the two filter holders (except on RMA-4) to cause an alarm on loss of flow. The loss of flow alarm for RMA-4 originates from the flow indication device (photohelic). The movable monitors have local indication, recording, and alarms. Detectors have remotely actuated check sources to provide functional verification. In addition, each channel is calibrated routinely by exposure to a calibrated source traceable either directly or indirectly to NIST for verification against its initial calibration. Calibration of the monitors is performed following any required maintenance of the detectors. Measurements have an accuracy of ± 25 percent of the true value. Precision is ± 15 percent at all levels. Each ratemeter is equipped with two adjustable alarm levels (alert and high) and a channel failure/or loss of power alarm. These alarms, associated with the fixed monitors, are annunciated on the Radiation Monitoring System control panel in the control room. Channels which have interlock functions with other systems (see Figure 11.4-1) are provided with a bypass switch for use during maintenance or testing. Use of this switch is annunciated.

RN 04-023

RN 02-002

12.2.4.2.1 Control Room Supply Air, Channel RM-A1

This channel monitors the particulate, iodine, and gaseous activity of air supplied to the control room. A sample of air is taken from the air supply duct through an isokinetic sampler nozzle and is drawn successively through a particulate sampler, an iodine sampler, and a gas sampler. The particulate sampler is equipped with a fixed filter which is continuously monitored by a lead shielded scintillator detector. The iodine sampler is similar to the particulate sampler except that an activated charcoal cartridge is used instead of a fixed filter. The fixed filter and the charcoal filter are designed to be removable for laboratory analysis. The sensitive volume of the gas sampler is shielded with lead and monitored by a scintillation detector. The approximate sensitivity and range of this channel are as follows:

1. Particulate, 4.7×10^{-11} to 10^{-7} µCi/cc based upon Cs-137. RN 03-011

2. Gas, 2×10^{-6} to 2×10^{-2} µCi/cc based upon Kr-85.

3. Iodine, 2×10^{-11} to 2×10^{-7} µCi/cc based upon I-131. RN 03-011

A high activity alarm from the gas channel automatically places the Control Room, Computer Room, Relay Room, and Instrument Repair Room Ventilation Systems in the recirculation mode, starts the Control Room Emergency Ventilation System, and closes the outside air dampers. The iodine and particulate channels provide high activity alarms to alert operating personnel. The high alarm setpoints are established on the basis of the requirements of the plant Technical Specifications and the sensitivity of the detection channels. RN 02-004

12.2.4.2.2 Reactor Building Air Sample Line, Channel RM-A2

This channel monitors the particulate, iodine, and gaseous activity level of the air inside the Reactor Building and is located inside the Auxiliary Building. Reactor Building air drawn from and returned through Reactor Building penetrations is monitored by RM-A2. The readout of the monitor is used to detect leaks in systems containing primary coolant. Channel ranges are similar to those of Channel RM-A1. To meet initial licensing requirements, the sensitivity of this monitor provides the capability to detect 10-MPC-hours of particulate and iodine radioactivity. RN 01-119

The monitor air sample and return lines are isolated (closed) upon occurrence of a containment isolation signal (Phase A). Post accident, this monitor can be used as a Reactor Building air sampling station, provided that Reactor Building pressure and temperature have been reduced sufficiently to allow opening of the sample line isolation valves. This monitor is also designed to withstand seismic conditions as recommended by Regulatory Guide 1.45.

The monitor design is similar to that of RM-A1, except that a moving particulate filter is used. An additional sample pump is also provided to allow operation when one pump is undergoing maintenance. The pumps and valves are located in a separate adjoining panel (XPN7321). RN 99-067

A high activity alarm from the gas channel initiates closure of the Reactor Building purge valves (see Figure 11.4-1).

High alarm setpoints are based upon plant operating requirements, sensitivity, and measured normal background. RN 13-011

A fixed alarm setpoint is not applicable to radiation monitoring for reactor coolant leak detection. The activity of the reactor coolant and background radiation must be taken into consideration.

The particulate and gas channels of radiation monitor RM-A2 will be operated with an initial setpoint of not less than twice background and may subsequently be readjusted based upon the changes in reactor coolant leakage and reactor building activity. Typically, the adjustments are made to provide the most sensitive response without causing an excessive number of spurious alarms. The maximum setpoint will be limited to no more than twice the operational equilibrium or operability limit defined in design calculations for the monitor, whichever is less. RN 13-011

12.2.4.2.3 Main Plant Vent Exhaust, Channel RM-A3

This channel is discussed in Section 11.4.2. It is used for detection of activity in the prefiltered discharge effluent from the Auxiliary Building main plant vent.

12.2.4.2.4 Reactor Building Purge Exhaust, Channel RM-A4

This channel is discussed in Section 11.4.2. It is used for detection of activity in the prefiltered purge discharge effluent from the Reactor Building.

12.2.4.2.5 Fuel Handling Building Exhaust, Channel RM-A6

This channel is used to monitor Fuel Handling Building exhaust air activity. Its function is to provide early detection of activity released. To meet initial licensing requirements, the sensitivity of this monitor provides for detection of 10-MPC-hours of particulate, iodine, and gaseous activity. A sample of the exhaust air is taken by an isokinetic nozzle located in the exhaust duct upstream of the exhaust filters. A local audible alarm is provided. The design, range, and sensitivity of this channel are similar to those of RM-A1. The high level alarm setpoints are based upon the sensitivity of the detectors and are set to alarm per the plant Technical Specifications. RN 01-119 RN 02-004

12.2.4.2.6 Sampling Rooms, channel RM-A7

Channel RM-A7 is a movable unit normally used to monitor a non-isokinetic sample of the sampling room air for particulate, iodine, and gaseous activity. Its design, range, and sensitivity are similar to those of RM-A1. The monitor may be used in other areas during refueling or as required by Health Physics personnel. The high level alarm setpoints are based upon the sensitivity of the detectors and are normally set to provide a local high alarm upon the occurrence of activity in excess of 10 CFR 20 limitations.

12.2.4.2.7 Spent Fuel Area, Channel RM-A8

Channel RM-A8 is a movable unit similar to RM-A7. It is normally used for ambient air monitoring in the Fuel Handling Building for detection of particulate, iodine, and gaseous activity. The unit may be used in other areas at the discretion of Health Physics personnel. The alarm setpoints are based upon the sensitivity of the detectors and are normally set to provide a local high alarm upon occurrence of activity in excess of 10 CFR 20 limitations.

12.2.4.2.8 Condenser Air Removal, Channel RM-A9

This channel is discussed in Section 11.4.2. It is used for detection of activity in the effluent discharge of the Condenser Air Removal System and for detection of primary to secondary leakage.

12.2.4.2.9 Waste Gas Discharge, Channel RM-A10

This channel is discussed in Section 11.4.2. It is used to provide backup control of the batch release of waste gas from the waste gas decay tanks.

12.2.4.2.10 Auxiliary Building Ventilation Monitor, Channel RM-A11

Auxiliary Building ventilation duct air samples are monitored (see Figures 9.4-7 and 9.4-8) by a shielded scintillation detector for beta activity which is indicative of the presence of noble gases in the ventilation exhaust ducts. A pumping station is provided to sequentially obtain up to 12 different air samples from the Auxiliary Building ventilation ducts. Each of the 12 areas normally will be sequentially monitored at least once per hour. The activity of each sample is recorded locally. Indication and alarms are provided on the Radiation Monitoring System control panel in the control room. Should the sequential monitor alarm, operators in the control room can alert both individuals in the area and appropriate Health Physics personnel through the use of the Plant PA system. Should a high level alarm condition exist, appropriate Health Physics personnel can be dispatched to the specific area with portable sampling equipment. These samples will provide, on a timely basis, the information necessary to assess the airborne particulate and iodine radioactivity of the area. The approximate sensitivity and range of this channel are as follows: gas, 2×10^{-6} to 2×10^{-2} μCi/cc based upon Kr-85. Setpoints are adjusted as required by Health Physics personnel.

RN 04-023

RN 03-011

12.2.4.2.11 Movable Particulate Unit, Channel RM-A12

Channel RM-A12 is a movable, cart mounted unit provided to monitor and record particulate airborne activity in areas selected by Health Physics personnel. The unit provides a removable fixed particulate filter and a removable charcoal filter for laboratory analysis. The alarm setpoints are set as required by Health Physics personnel for the zones being monitored. Sensitivity and range of the particulate monitor are approximately 10^{-11} to 10^{-7}, referenced to Cs-137.

12.2.4.3 Detection of DAC Levels of Airborne Radioiodine and Particulates RN 01-119

An analysis of the capability of the Airborne Radioactivity Monitoring System to detect DAC levels of airborne radioiodine and particulates in areas having potential for airborne contamination and which are normally accessed by personnel is presented in Sections 12.2.4.3.1 through 12.2.4.3.3. RN 01-119

The assumptions upon which this analysis is based are as follows:

1. DAC for airborne iodine is 2×10^{-8} µCi/cc. RN 01-119

2. DAC for airborne particulates (as cesium) is 6×10^{-8} µCi/cc.

3. Sensitivity of radiation monitor, RM-A7, is as follows:

   a. For Cs-137, 8 hour sampling, 4.7×10^{-11} µCi/cc. RN 01-119

   b. For I-131, 8 hour sampling, 2×10^{-11} µCi/cc.

12.2.4.3.1 Control Building

The sampling room is the area in the Control Building complex considered to have potential as a source of airborne activity. This room is sampled by a three channel radiation monitor, RM-A7, which measures airborne particulate, iodine, and gaseous activity.

It is assumed that if DAC levels of iodine and cesium activity are present in the sampling room air, the local radiation monitor, RM-A7, will detect this activity in a sampling time which is inversely proportional to the concentration: RN 01-119

$$\frac{\text{Monitor sensitivity}}{\text{Air concentration}} : \frac{\text{Sample time}}{\text{8 hours}}$$

Therefore, the sample time to detect one DAC of cesium is determined as follows: RN 01-119

$$\text{Sample time} = \frac{4.7 \times 10^{-11} \mu\text{Ci/cc}}{6 \times 10^{-8} \mu\text{Ci/cc}} \times 8\,\text{hrs} \times \frac{60\,\text{min}}{1\text{hr}}$$

$$= 0.38\ \text{min}$$

RN 01-119

Assuming a high alarm setpoint at 3 times background, the time for response to one DAC of cesium would be approximately 1.1 minutes.

Similarly, the sample time to detect one DAC of iodine is determined as follows:

$$\text{Sample time} = \frac{2 \times 10^{-11} \mu Ci/cc}{2 \times 10^{-8} \mu Ci/cc} \times 8\,hrs \times \frac{60\,min}{1hr}$$

$$= 0.48 \text{ min}$$

RN 01-119

Assuming a high alarm setpoint at 3 times background, the time for response to one DAC of iodine would be approximately 1.4 minutes.

12.2.4.3.2 Fuel Handling Building

1. Spent Fuel Pool Floor Area

   Use of the local radiation monitor, RM-A8, would provide response times similar to the estimates used for the sampling room radiation monitor, RM-A7 (see Section 12.2.4.3.1):

Concentration	Detection Time (min)	Alarm at 3 Times Background (Min)
I-131, 2 x 10^{-8} µCi/cc	0.48	1.4
Cs-137, 6 x 10^{-8} µCi/cc	0.38	1.1

RN 01-119

2. Fuel Handling Building Exhaust

   Use of radiation monitor, RM-A6, in the Fuel Handling Building exhaust would provide the following detection capability under the previously stated assumed conditions, based upon the discharge flow rates shown by Figure 12.2-2.

   The exhaust air flow from the Fuel Handling Building is approximately 27,000 cfm. RN 01-119
   This flow is monitored by RM-A6, a three channel monitor, which samples the air in the discharge duct upstream of the filter plenum through an isokinetic nozzle. By use of a dilution factor, based upon the ratio of sub-area exhaust air flow to total air flow, an estimate of detection times can be determined as follows:

$$\text{Detection time for one DAC in sub-area} = \frac{\text{Detection time for one DAC}}{\text{Dilution ratio}}$$

RN 01-119

   Estimates of detection times for I-131 and Cs-137 obtained from the preceding expression are presented in Table 12.2-4.

12.2.4.3.3 Auxiliary Building

Normally accessible areas of the Auxiliary Building are supplied with clean air which then passes through such areas to restricted access areas with potentially high airborne activity levels. Use of a movable surveillance monitor, such as RM-A12, with an assumed sensitivity of 1.0×10^{-11} µCi/cc for cesium would provide the following detection capability:

$$\frac{1.0 \times 10^{-11} \mu Ci/cc}{6 \times 10^{-8} \mu Ci/cc} : \frac{x \min}{480 \min}$$

Where: RN 01-119

6×10^{-8} µCi/cc is DAC for cesium

x is the time to detect one DAC, local activity (i.e., 0.1 min).

With the alarm set at 3 times background, the response to one DAC of Cs-137 in the area under surveillance would be approximately 0.3 minutes.

Periodic surveillance sampling by Health Physics personnel will provide samples for laboratory analysis of particulates and radioiodine to verify operational levels of airborne activity in the plant.

12.2.4.4 <u>Reactor Coolant Leak Detection</u>

Assumptions relative to reactor coolant leak detection are as follows:

1. Free containment volume, 1.8×10^6 ft^3.

2. Continuous purge rate, approximately 1000 cfm.

3. Reactor Building Charcoal Cleanup System, not operating.

4. Reactor coolant activity, as listed below for the predominate isotopes. RN 13-011

5. Sensitivity of radiation monitor RM-A2, satisfies requirements of Regulatory Guide 1.45.

The major isotopes of interest and corresponding design basis activities for both the particulate and gas channels are as follows:

1. Cs-134, 2.6×10^{-2} µCi/gm.

2. Cs-137, 1.9×10^{-2} µCi/gm.

3. Co-58, 1.7×10^{-2} µCi/gm.

4. Co-60, 2.1×10^{-3} µCi/gm.

5. Rb-88, 2.3×10^{-1} µCi/gm.

6. Xe-133, 5.1 µCi/gm (gas).

Assuming a steady reactor coolant leak rate of 0.1 gpm, the equilibrium concentrations of particulate isotopes in the reactor building atmosphere are as follows:

1. Cs-134, 7.38×10^{-8} µCi/cc. RN 13-011

2. Cs-137, 5.40×10^{-8} µCi/cc.

3. Co-58, 4.77×10^{-8} µCi/cc.

4. Co-60, 5.96×10^{-9} µCi/cc.

5. Rb-88, 9.19×10^{-9} µCi/cc.

A step increase in leakage to 1 gpm for a period of 1 hour would increase total particulate activity by 75 percent which is readily detectable by the particulate channel of radiation monitor RM-A2.

Figure 12.2-3 shows the increase in the RB particulate concentration following a 1-gpm leak.

The major isotope of interest for gas channel detection is Xe-133. A reactor coolant leak of 0.1 gpm would result in an equilibrium Xe-133 concentration of 4.14×10^{-5} µCi/cc. A subsequent leak of 1 gpm over 1 hour would result in an increase of 1.55×10^{-5} µCi/cc. This is equivalent to a 37 percent increase which is detectable by the gas channel of radiation monitor RM-A2.

Figure 12.2-4 shows the increase in Xe-133 concentration.

12.2.5 OPERATING PROCEDURES

The Health Physics group is responsible for developing procedures for evaluating potential airborne radioactivity concentrations and providing guidance for minimizing personnel exposures. Occupational exposure is minimized during operations and maintenance by using the following methods and procedures to determine and cope with potential hazards:

1. Monitoring

   High volume air samples of various flow rates are used to collect particulates on high efficiency filter media, at regularly scheduled intervals, for subsequent counting. For tritium sampling, freezeout methods, bubblers, or desiccants are used to obtain samples for counting by the use of liquid scintillation techniques. Assay of noble gases is performed on a regular schedule by drawing an air sample into an evacuated chamber for appropriate analysis.

2. Respiratory Protection

   It is the responsibility of the Health Physics group to monitor and post areas of airborne radioactivity, to establish the requirements for respiratory equipment, to control the use of respiratory equipment, and to control access to areas through the radiation work permit program. Training programs for respiratory protection are the responsibility of the Health Physics Supervisor.

   Experience gained during operation and maintenance of several nuclear plants with whom SCE&G has contact is used to provide one basis for further evaluation and development of respiratory protection programs. Guidance contained in Regulatory Guides 8.2, 8.8, and 8.10 is utilized in the development and review of these programs.

   In relation to TMI Action Plan Item II.F.1, "Additional Accident Monitoring Instrumentation," station procedures will contain the following information for converting instrument readings to concentrations or/and release rates:

   a) Graphs relating cpm readings to concentration (μCi/ml) for effluent radiation monitors based on the manufacturer's recommendation or measurements from onsite analyses.

   b) Graph or formula for determining the release rate (Ci/sec) based on effluent radiation monitor readings and measured flow rates of the main vent.

c) Graph or formula for determining the volume of steam discharged after opening of main steam safety valves. A concentration based on the average reading of the radiation monitors can be used to quickly approximate the total amount of radioactivity released. Subsequent laboratory analysis more accurately determines the quantity and spectrum of radionuclides released.

Computer capabilities can also be used to assist with these calculations.

Also, as required by TMI Action Plan Item II.F.1, the Station Radiation Monitor Calibration Program is as follows:

Initial vendor calibration included the determination of response to various radioactive isotopes where they were calibrated to provide traceability to the National Bureau of Standards. The Liquid monitors were checked against Cs-137 and Mn-54; Particulate monitors were checked against Cs-137 and Ce-144; Iodine monitors were checked against Cs-137 and Ba-133; Gas monitors were checked against Kr-85 and Cs-137. The Condenser Offgas monitor was checked against Xe-133 and Cs-137. Area monitors were checked against Ra-226.

Inplant calibration will normally use calibrated sources for determination of monitor response. These sources will typically be Cs-137 or Co-60 for Area Gamma monitors; Cs-137 for Particulate monitors; Ba-133 for Iodine monitors; and Cs-137 for Liquid monitors. Frequency of periodic calibration is normally on a yearly basis except for those channels identified in the Technical Specification where frequency has been fixed. The High Range Gamma Area monitors require electronic verification as described in Section 12.1 as the use of high level isotopes is not practical.

12.2.6 ESTIMATES OF INHALATION DOSES

The annual inhalation doses to plant personnel from airborne activity in the various plant buildings depend upon the occupancy of the various plant areas in which airborne activity can occur. The dose to plant personnel is controlled by limiting personnel occupancy in contaminated areas, using engineering controls when available, and by using respiratory protection equipment when warranted by TEDE-ALARA evaluation of the task being performed. The doses to plant personnel are therefore limited to the values specified in 10 CFR 20 for occupationally exposed individuals. RN 01-119

Sections 12.2.6.1.1 through 12.2.6.1.3 discuss peak airborne concentrations and inhalation dose rates for various plant buildings based upon the leak rates discussed in Section 12.2.3. The methods used to calculate concentrations and inhalation dose rates are discussed in Section 12.2.6.2.

12.2.6.1 Peak Airborne Concentrations and Inhalation Dose Rates

12.2.6.1.1 Reactor Building

Peak airborne concentrations for the Reactor Building during operation at power are presented in Table 12.2-1. The concentrations presented in Table 12.2-1 are those that would occur at the end of the interval between purges with the assumptions given in Section 12.2.3.1.

Exposure to the reactor building concentrations given in Table 12.2-1 would result in an upper limit inhalation dose rate of 21.9 mRem/hr to the thyroid. Exposure of workers to this dose rate is unlikely since operation of the Reactor Building Charcoal Cleanup System and/or partial purging of the Reactor Building normally occur prior to any lengthy access. Respiratory protection equipment is used for short term access when required by plant procedures. RN 01-111

12.2.6.1.2 Auxiliary Building

Average airborne concentrations for the Auxiliary Building were calculated using the assumptions given in Section 12.2.3.2 and the building volume and flow rate discussed in Section 12.2.2. These concentrations are given in Table 12.2-1. The corresponding inhalation dose rate to the thyroid is 1.8×10^{-2} mRem/hr. RN 01-111

As described in Section 9.4.2, the Auxiliary Building Ventilation System supplies clean air to areas expected to be relatively free of airborne activity and removes air from areas that could potentially have high levels of airborne radioactivity. Thus, airborne activity levels in normally occupied areas should be significantly lower than the calculated average concentrations presented in Table 12.2-1.

The peak airborne concentrations in the Auxiliary Building are expected to be in the individual rooms which house auxiliary reactor equipment, such as the letdown heat exchanger, volume control tank and connecting instrumentation, piping, and valves. These rooms are within controlled areas and access to these areas is maintained under strict administrative control. Respiratory protection equipment or other protective measures are used in accordance with plant procedures.

For calculational purposes in determining peak concentrations, it has been assumed that one half of the total leakage in the Auxiliary Building occurs around the letdown heat exchanger. The resultant calculated concentrations for the letdown heat exchanger are presented in Table 12.2-1. The corresponding inhalation dose rate to the thyroid is 2.5 mRem/hr. RN 01-111

12.2.6.1.3 Turbine Building

Average airborne concentrations for the Turbine Building were calculated using assumptions given in Section 12.2.3.3 and the building volume and flow rate discussed in Section 12.2.2. These concentrations are given in Table 12.2-1. The corresponding inhalation dose rate to the thyroid is 2.8 x 10^{-5} mRem/hr. The Turbine Building concentrations and inhalation dose rate are sufficiently small to allow normal occupancy without the use of protective measures. RN 01-111

12.2.6.2 Methods Used to Estimate Airborne Concentrations and Inhalation Doses

The concentration of radionuclides in a given volume, V, with a steady-state leakage rate is determined from an activity balance. For the i^{th} isotope, the activity balance is as follows:

$$\frac{d}{dt}C_i(t) = \frac{S_i}{V} - \Lambda_i C_i(t) \qquad (12.2\text{-}1)$$

Where:

$C_i(t)$ = Concentration of i^{th} isotope at time t (µCi/cc).

S_i = Leakage rate of i^{th} isotope (µCi/sec).

Λ_i = Removal constant of i^{th} isotope (sec^{-1}).

In Equation (12.2-1), ingrowth from radioactive parent nuclides has been neglected and a homogeneous mixing model has been assumed. The solution is as follows:

$$C_i(t) = \frac{S_i(1-e^{-\Lambda_i t})}{V\Lambda_i} + C_i(0)e - \Lambda_i t \qquad (12.2\text{-}2)$$

If the system is purged at a flow rate, R, then, for a volume, V, the removal constant is:

$$\Lambda = \lambda_i + \frac{R}{V} \qquad (12.2\text{-}3)$$

Where:

λ_i = The decay constant of the i^{th} isotope.

When an intermittent purging scheme is used, there is a period of buildup followed by a period of removal. During the building interval, the removal constant is:

$$\Lambda = \lambda_i \qquad (12.2\text{-}4)$$

Once the airborne concentrations have been determined, inhalation dose rates are calculated using the following formula:

Inhalation Dose Rate = $(C_i)(BR)(DCF_i)$

Where:

Ci = Equilibrium concentration of isotope i (µCi/cc).

BR = Breathing rate (cc/hr).

DCF_i = Dose conversion factor for isotope i (mRem/µCi).

12.2.7 REFERENCES

1. U.S. Nuclear Regulatory Commission, "Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors," NUREG-0017, April 1976.

2. American National Standards Institute, "Guide for Administrative Practices in Radiation Monitoring," ANSI N13.2, 1969.

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TABLE 12.2-1

CALCULATED AIRBORNE CONCENTRATIONS [(1)]

Isotope	Reactor Building Concentrations[(2)] (μCi/cc)	Auxiliary Building Concentrations (μCi/cc)	Letdown Heat Exchanger Room Concentrations [(3)] (μCi/cc)	Turbine Building Concentrations (μCi/cc)
H-3	8.7×10^{-4}	1.4×10^{-8}	2.0×10^{-7}	3.0×10^{-10}
Kr-85m	1.3×10^{-6}	2.1×10^{-9}	2.9×10^{-7}	1.1×10^{-14}
Kr-85	9.6×10^{-6}	1.6×10^{-10}	2.2×10^{-8}	8.2×10^{-16}
Kr-87	4.3×10^{-7}	2.1×10^{-9}	3.2×10^{-7}	1.2×10^{-14}
Kr-88	1.6×10^{-6}	3.8×10^{-9}	5.4×10^{-7}	2.1×10^{-14}
Xe-131m	3.0×10^{-5}	1.1×10^{-9}	1.5×10^{-7}	5.4×10^{-15}
Xe-133m	2.5×10^{-6}	3.4×10^{-10}	4.7×10^{-8}	1.8×10^{-15}
Xe-133	1.1×10^{-4}	6.9×10^{-9}	9.5×10^{-7}	3.6×10^{-14}
Xe-135m	8.2×10^{-8}	$1.4.x 10^{-9}$	2.5×10^{-7}	1.0×10^{-14}
Xe-135	1.2×10^{-5}	9.6×10^{-9}	1.3×10^{-6}	$5.1.x 10^{-14}$
Xe-137	5.3×10^{-9}	1.6×10^{-10}	3.9×10^{-8}	2.1×10^{-15}
Xe-138	7.0×10^{-8}	1.2×10^{-9}	2.3×10^{-7}	9.5×10^{-15}
Br-84	2.1×10^{-11}	1.6×10^{-12}	2.6×10^{-10}	3.4×10^{-16}
I-131	1.6×10^{-8}	5.4×10^{-12}	7.4×10^{-10}	1.1×10^{-14}
I-132	1.2×10^{-9}	2.6×10^{-11}	3.7×10^{-9}	1.6×10^{-14}
I-133	6.6×10^{-9}	1.7×10^{-11}	2.4×10^{-9}	2.8×10^{-14}
I-134	7.3×10^{-10}	3.8×10^{-11}	5.8×10^{-9}	1.2×10^{-14}
I-135	4.1×10^{-9}	3.3×10^{-11}	4.5×10^{-9}	3.6×10^{-14}

02-01

(1) Based upon the assumptions given in Sections 12.2.3 and 12.2.6.

(2) Concentrations at the end of interval between purges and without operation of the reactor building charcoal cleanup system.

(3) Concentrations are based upon the assumption that one-half of the auxiliary building leakage occurs in the letdown heat exchanger room.

TABLE 12.2-2

AIR FLOW RATES AND APPROXIMATE NET FREE VOLUMES SERVED BY VENTILATION SYSTEMS

System	System Exhaust Flow Rate	Approximate Net Free Volume Served (ft^3)	
Control Room Ventilation System	21,270 cfm/emergency system fan	226,040	RN 02-048
Controlled Access Area Exhaust System	16,000 cfm/fan	73,890 (1)	
Spent Fuel Pool Area Ventilation System	27,000 cfm/fan	629,260 (1)	RN 01-119
Auxiliary and Radwaste Area Ventilation Systems	123,800 cfm, total	1,890,000	RN 02-048
Turbine Building Ventilation Systems	1,500,000 cfm, total	3,700,000	
Reactor Building Charcoal Cleanup System	24,000 cfm, total	1,800,000	

(1) Original Licensing / Construction Estimates RN 02-048

TABLE 12.2-3

AIRBORNE RADIOACTIVITY MONITORS

MONITOR	FUNCTION	DETECTORS	POWER SOURCE
RM-A1 Control Room Supply Air	Monitors ventilation air supplied to control room	1. Particulate-Beta, 4.7 x 10^{-11} to 10^{-7} μCi/cc, Cs-137 2. Iodine-Gamma, 2 x 10^{-11} to 2 x 10^{-7} μCi/cc, I-131 3. Gas-Beta, 2 x 10^{-6} to 2 x 10^{-2} μCi/cc, Kr-85	Bus B (1)
RM-A2 Reactor Building Air Sample Line	Monitors sample taken from containment	1. Particulate-Beta, 5.5 x 10^{-11} to 10^{-7} μCi/cc, Cs-137 2. Iodine-Gamma, 2 x 10^{-11} to 2 x 10^{-7} μCi/cc, I-131 3. Gas-Beta, 2.6 x 10^{-6} to 2 x 10^{-2} μCi/cc, Kr-85	Bus B (1)
RM-A3 Main Plant Vent Exhaust	Monitors effluent from auxiliary building	1. Particulate-Beta, 4.7 x 10^{-11} to 10^{-7} μCi/cc, Cs-137 2. Iodine-Gamma, 2 x 10^{-11} to 2 x 10^{-7} μCi/cc, I-131 3. Gas-Beta, 2.6 x 10^{-6} to 2 x 10^{-2} μCi/cc, Kr-85	Bus A (1)
RM-A4 Reactor Building Purge Exhaust	Monitors effluent from containment during purge	1. Particulate-Beta, 4.7 x 10^{-11} to 10^{-7} μCi/cc, Cs-137 2. Iodine-Gamma, 2 x 10^{-11} to 2 x 10^{-7} μCi/cc, I-131 3. Gas-Beta, 2 x 10^{-6} to 2 x 10^{-2} μCi/cc, Kr-85	Bus A (1)
RM-A6 Fuel Handling Building Exhaust	Monitors ventilation discharge from fuel handling building	1. Particulate-Beta, 4.7 x 10^{-11} to 10^{-7} μCi/cc, Cs-137 2. Iodine-Gamma, 2 x 10^{-11} to 2 x 10^{-7} μCi/cc, I-131 3. Gas-Beta, 2 x 10^{-6} to 2 x 10^{-2} μCi/cc, Kr-85	Bus B (1)

RN
03-011

TABLE 12.2-3 (Continued)

AIRBORNE RADIOACTIVITY MONITORS

MONITOR	FUNCTION	DETECTORS	POWER SOURCE	
RM-A7 Sampling Rooms	Monitors air sample from sampling room. Movable unit - may be used for other areas as required.	1. Particulate-Beta, 4.7 x 10^{-11} to 10^{-7} μCi/cc, Cs-137 2. Iodine-Gamma, 2 x 10^{-11} to 2 x 10^{-7} μCi/cc, I-131 3. Gas-Beta, 2 x 10^{-6} to 2 x 10^{-2} μCi/cc, Kr-85	480 volt outlets	
RM-A8 Spent Fuel Area	Monitors local air sample. Movable unit - may be used for other areas as required.	1. Particulate-Beta, 4.7 x 10^{-11} to 10^{-7} μCi/cc, Cs-137 2. Iodine-Gamma, 2 x 10^{-11} to 2 x 10^{-7} μCi/cc, I-131 3. Gas-Beta, 2 x 10^{-6} to 2 x 10^{-2} μCi/cc, Kr-85	480 volt outlets	RN 03-011
RM-A9 Condenser Air Removal	Monitors effluent of condenser off-gas discharge	Gas-Gamma, 2 x 10^{-6} to 2 x 10^{-2} μCi/cc, Xe-133	Bus B	
RM-A10 Waste Gas Discharge	Monitors discharge of waste gas-provides backup to decay tank analysis	Gas-Gamma, 2 x 10^{-4} to 2 μCi/cc, Xe-133	Bus B	
RM-A11 Auxiliary Building Ventilation	Sequentially monitors various auxiliary building ventilation exhaust ducts	Gas-Beta, 2 x 10^{-6} to 2 x 10^{-2} μCi/cc, Kr-85	Bus B (1)	02-01

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Amendment 02-01

TABLE 12.2-3 (Continued)

AIRBORNE RADIOACTIVITY MONITORS

MONITOR	FUNCTION	DETECTORS	POWER SOURCE
RM-A12 Movable Particulate Unit	Provides local particulate monitoring and iodine sampling	Particulate-Beta, 10^{-11} to 10^{-7} μCi/cc, Cs-137	120 volt outlets
RM-A13 Main Plant Vent Exhaust, High Range Gas Discharge	Effluent High Range Gas Monitor	Ion Chamber - Gamma 0.1 to 10^{7} mr/hr (10^{-2} to 10^{5} μCi/cc, Xe-133)	Bus A
RM-A14 Purge Exhaust Effluent, High Range	Effluent High Range Gas Monitor	Ion Chamber - Gamma 0.1 to 10^{7} mr/hr (10^{-2} to 10^{5} μCi/cc, Xe-133)	Bus A
RM-G19 A, B, C Steam Line	Steam Line High Range Gamma Monitor	Ion Chamber - Gamma 0.1 to 10^{7} mr/hr	Bus B

RN 03-011

RN 02-002

(1) Associated sample pump powered from diesel backed 480 volt bus.

TABLE 12.2-4

ESTIMATED DETECTION TIME FOR
I-131 AND Cs-137 IN THE FUEL HANDLING BUILDING

Fuel Handling Building Area (1)	Exhaust Flow (cfm)	Dilution Factor	Detection Time (Minutes)	
			I-131	Cs-137
Spent Fuel Area	15,320	0.567	0.8	0.7
General Flow Area	1,840	0.068	7.0	5.5
Railroad Area	3,020	0.112	4.3	3.4
Personnel Cleanup (2)	242	0.009	53.6	42.0
Emergency Personnel Hatch	1,750	0.065	7.4	5.8
412 South West	1,470	0.054	8.8	6.9
Decontamination	1,120	0.041	11.6	9.1
Excess Waste Holdup Tank	950	0.035	13.6	10.7
Decontamination Pit Collection Tank	686	0.025	18.9	14.8

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01-119

Notes:

1. Small areas, access to which is controlled due to potentially high activity levels, are not included since access is under administrative control.

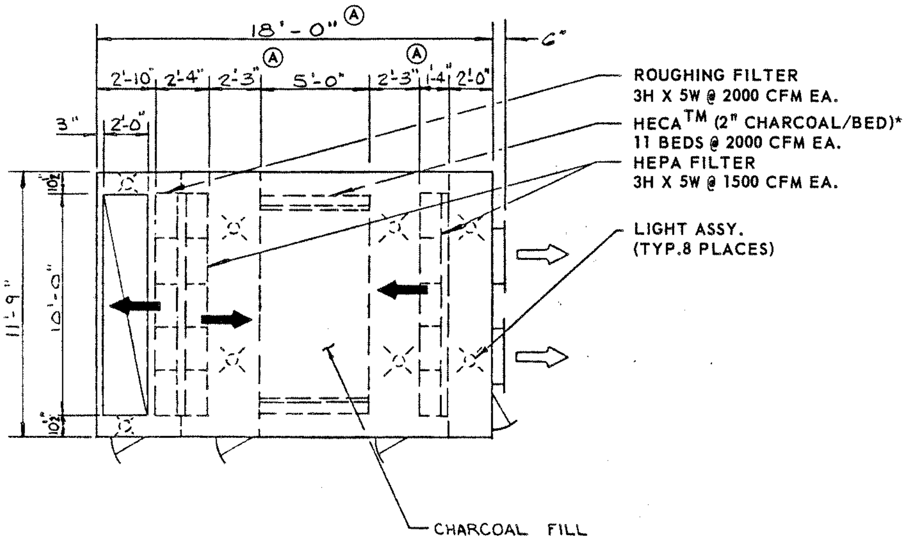
2. The personnel cleanup area is listed although it is not considered as a source of activity.

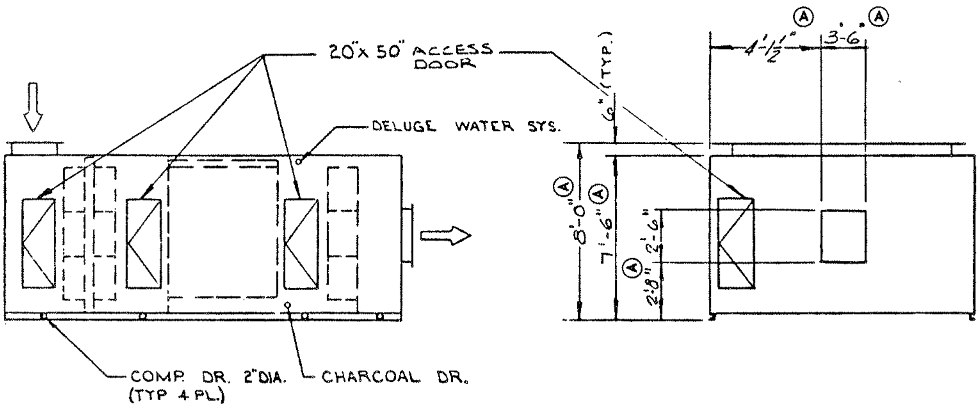
GENERAL NOTES:

1. ⇦ DENOTES AIR FLOW.
2. ⬅ DENOTES FILTER REMOVAL.
3. TAG NO. XAA–29A–AH.
4. TAG NO. XAA–29B–AH.
5. SAFETY CLASS–2b.
6. MAXIMUM UNIT CAPACITY 21,000 CFM.
7. OPERATING WT. 28,149 LB.
8. SHIPPING WT. 21,714 LB.
9. THIS DESIGN IS PRELIMINARY ONLY. THE FINAL DESIGN MAY DEVIATE FROM THIS ARRANGEMENT.

*10. ALLOW 42" CLEARANCE ABOVE FOR CHARCOAL FILLING.

Ⓐ 11. DUE TO HEIGHT LIMITATIONS FLOOR SLOPE WILL BE APPROX. 1/2" IN 11'–0".



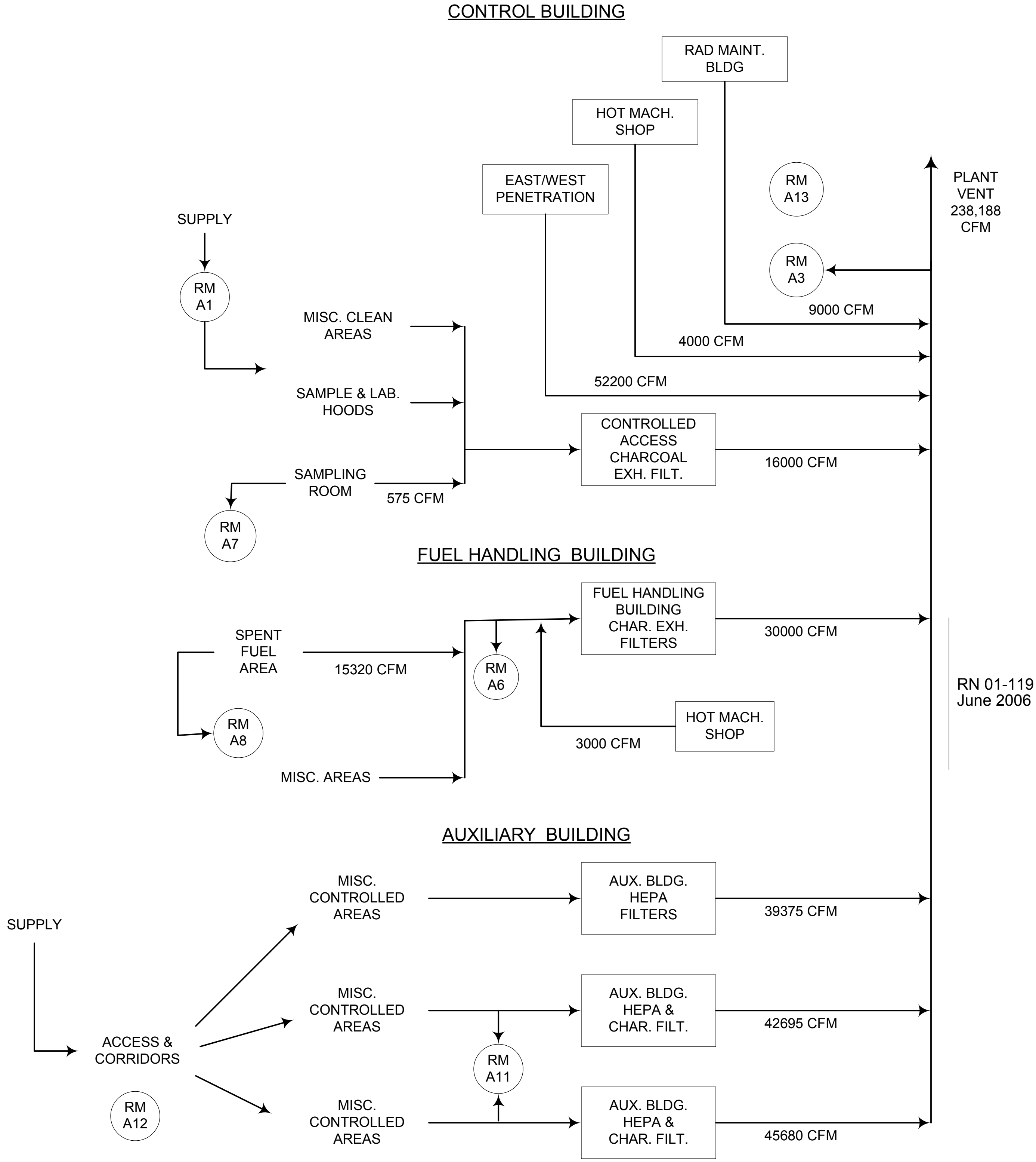


Amendment 0
August 1984

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

Control Room Emergency Charcoal Filter Plenum

Figure 12.2-1

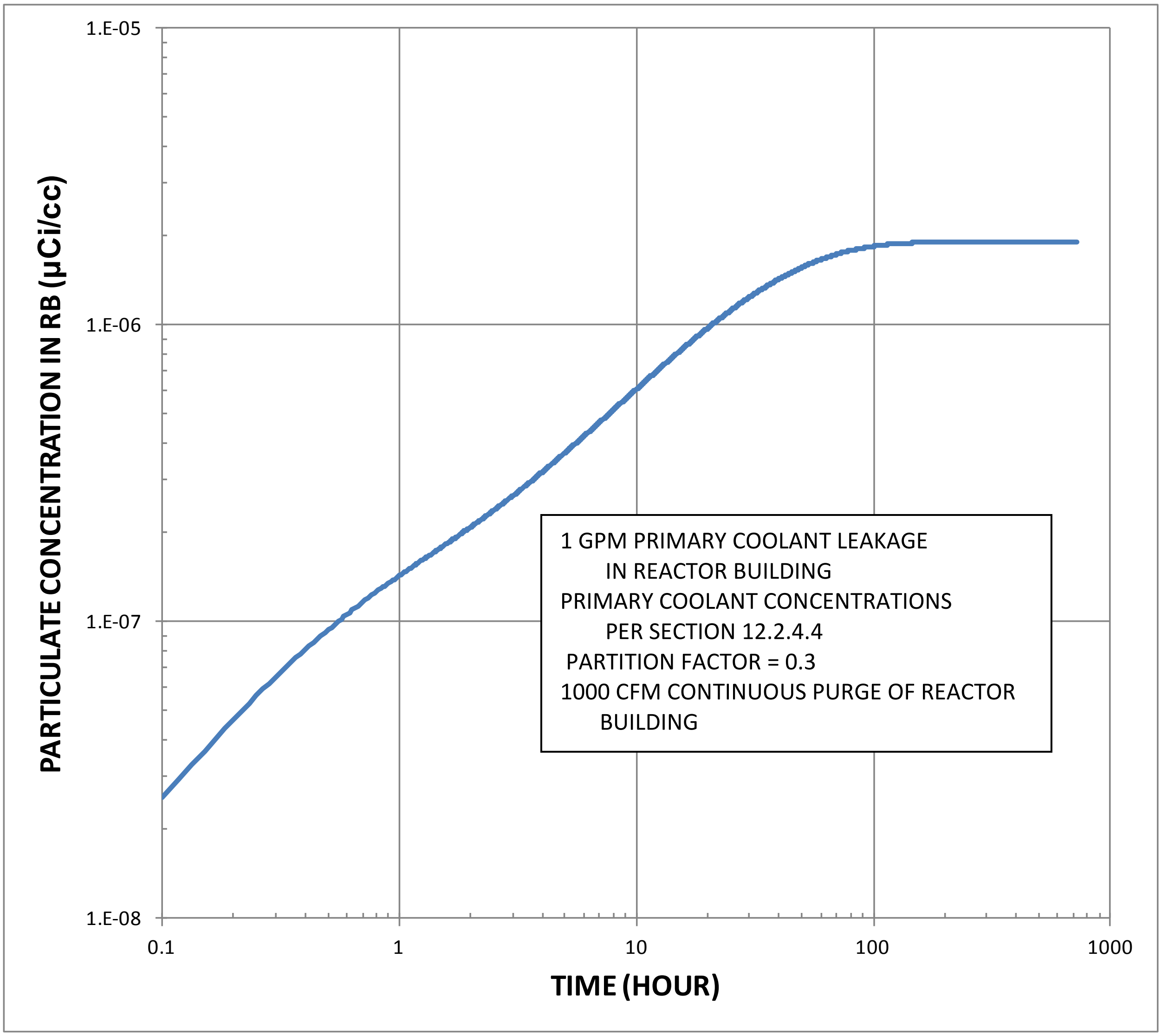


RN 01-119
June 2006

SOUTH CAROLINA ELECTRIC & GAS CO.
VIRGIL C. SUMMER NUCLEAR STATION

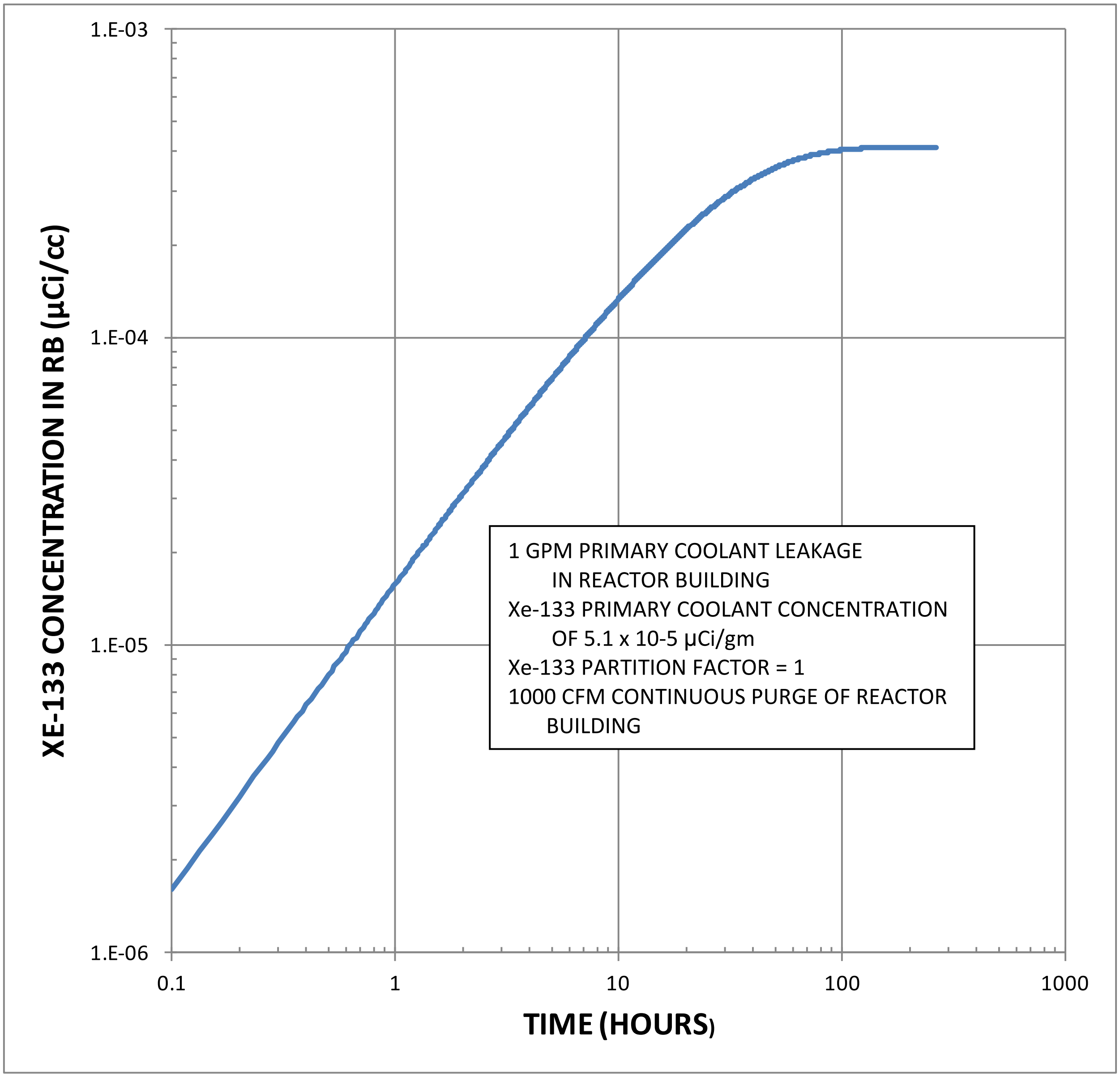
Airborne Activity Monitoring
ASSOCIATED WITH FSAR FIGS
9.4-4, 9.4-7, 9.4-8, 9.4-9,
9.4-10a, 9.4-11, 9.4-26a

Figure 12.2-2 Rev. 2



RN 13-011

SOUTH CAROLINA ELECTRIC & GAS CO. VIRGIL C. SUMMER NUCLEAR STATION
Buildup of Predominate Particulates in Reactor Building From 1 gpm Reactor Coolant Leak as a Function of Time After Leakage Begins Figure 12.2-3



RN 13-011

SOUTH CAROLINA ELECTRIC & GAS CO. VIRGIL C. SUMMER NUCLEAR STATION
Buildup of Xe-133 in Reactor Building From 1 gpm Reactor Coolant Leak as a Function of Time After Leakage Begins **Figure 12.2-4**

12.3 HEALTH PHYSICS PROGRAM

12.3.1 OBJECTIVES AND ORGANIZATION

12.3.1.1 Program Objectives

12.3.1.1.1 As Low as Reasonably Achievable (ALARA): Policy Considerations

It is the policy of the management of South Carolina Electric and Gas Company (SCE&G) that the Virgil C. Summer Nuclear Station is operated in such a manner that occupational radiation exposures of plant personnel are as low as reasonably achievable (ALARA). This policy is implemented by the Manager, Health Physics Services, who is directly responsible for the ALARA aspects of plant operation, and who also has the authority to prevent operations which are not consistent with the ALARA policy. The Manager, Health Physics Services, is responsible to the General Manager, Nuclear Plant Operations, who has overall responsibility for the implementation of the ALARA policy. The General Manager, Organizational Development and Effectiveness through the Manager, Quality Services is responsible for conducting periodic audits of plant operations to determine how exposures might be lowered. The Nuclear Safety Review Committee is designated as upper management's representative body to assure implementation of and adherence to the operational ALARA program. (see Section 13.4.2.2).

RN 99-129

RN 99-129 99-161

RN 13-005

The General Manager, Nuclear Plant Operations, has familiarized plant personnel with the commitment to keep occupational exposures ALARA by providing appropriate instructions, policy statements, and training. The General Manager, Nuclear Plant Operations, determines if modifications to operating and maintenance procedures and to plant equipment and facilities should be made, where practicable to minimize exposures and where unresolved safety problems are not involved.

12.3.1.1.2 ALARA: Operational Considerations

Procedural guidance in exposure reduction has been established for all systems that contain, collect, store, or transport radioactive liquids, gases, and solids, for all operating conditions. The following are examples of specific exposure-reduction techniques for operations such as maintenance, inservice inspection, radwaste handling, and refueling:

1. During initial startup, neutron and gamma-ray dose rate surveys are performed to verify the adequacy of shielding.

2. During normal operations, dose rate, surface contamination, and airborne radioactivity surveys are performed periodically throughout the plant and areas are zoned accordingly. These surveys ensure that data is available for planning operation and maintenance activities in accordance with the ALARA objective of minimizing exposures.

3. Areas are conspicuously posted in accordance with 10 CFR 20, as appropriate.

4. A radiation work permit (RWP) system is employed to ensure proper administrative control over work in restricted areas. The RWP is designed to ensure that the radiological conditions are defined and that appropriate measures are taken to minimize the dose received by personnel. Section 12.3.2.3.7 explains the use of the RWP.

5. Extension tools are used when practicable to increase the distance between the radiation source and the worker.

6. Equipment is moved to areas with lower radiation fields for maintenance when practicable.

7. Portable shielding such as lead bricks, lead sheets, lead shot, high density concrete block, or steel plates are used as practicable for assuring ALARA doses.

8. A personnel dosimetry program as described in Section 12.3.3 is administered by the Health Physics group to ensure compliance with 10 CFR 20.

Administrative controls have been established to assure that all procedures and requirements relating to radiation protection are followed by all plant personnel. The procedures that control radiation exposure are subject to review and approval as outlined in Section 13.4.

In general, the plant procedures are designed to keep occupational radiation exposures ALARA and follow the recommendations of Regulatory Guides 8.8 and 8.10. These procedures are organized and revised by the Manager, Health Physics Services, in cooperation with the Maintenance, Technical Services, or Operations Manager, such that operations proceed within the framework of the ALARA policy. RN 99-129

Independent technical and quality oversight is provided by the Manager, Quality Systems. This organization, through audits, surveillances, and reviews as described in Chapter 17, provides input to the Nuclear Safety Review Committee. Since Quality Systems maintains a reporting chain independent of plant operations to the Vice President, Nuclear Operations, continuing implementation of and adherence to the operational ALARA program is assured. RNs 97-068 13-005

12.3.1.2 Program Organization

The Manager, Health Physics Services, has the authority for developing and implementing company policy for radiation protection and contamination control. The administration of the Health Physics program is also the responsibility of the Manager, Health Physics Services, whose minimum qualifications are as presented in Section 13.1.3. The Manager, Health Physics Services, reports directly to the General Manager, Nuclear Plant Operations, on radiological protection issues. RN 99-129

The Manager, Health Physics Services is responsible for keeping plant staff informed of radiation hazards and conditions related to potential exposure, contamination of plant equipment or contamination of site and environs (see Figure 12.3-1). As the administrator of the health physics program, the responsibilities of the Manager, Health Physics Services include: RN 99-129

1. Training and supervision of the Health Physics staff.

2. Planning and scheduling Health Physics activities.

3. Establishing and maintaining data on plant, environmental and laboratory radiation and contamination levels, personnel exposures, and work or effluent release restrictions.

4. Maintaining a current Health Physics manual which contains the standards and procedures to implement the ALARA policy for radiation exposures and effluent releases.

5. Ensuring that plant operations comply with applicable regulations pertaining to radiation and environmental radiological protection.

6. Assessing radiological conditions/issues for normal operations and emergencies and advising plant management or the emergency director.

7. Developing, implementing, directing, controlling, and supervising all aspects of the RWP, ALARA, Effluent Control, Radiological Environmental, Radwaste, Dosimetry, Respiratory Protection, Equipment CAL/Contamination Control, Surveillance, RCA Access, and Radiological Data Base Programs.

8. Controlling the receipt, shipment, and storage of all radioactive materials.

9. Providing radiological engineering expertise and assessing events, occurrences, program performance, and data while developing and implementing solutions and enhancements.

10. Reviewing radiation safety-related operating procedures.

The Manager of Health Physics Services is responsible for developing a radiation protection training program, a radiation surveillance program, and an effluent control program to ensure that radiation exposures of all personnel are kept as far below the limits of 10 CFR 20 as is reasonable achievable. He also develops policy and specific procedural guidance as necessary to implement the ALARA policy. RN 99-129

In order to carry out his responsibilities, the Manager of Health Physics Services is in charge of the Health Physics groups. These groups are made up of technicians, supervisors, and professional Health Physicists who meet the qualifications presented in Section 13.1.3. RN 99-129

The Health Physics groups are organized to provide the following services:

1. Establish and maintain a radiological surveillance program to routinely collect, post, and record data concerning radiation and contamination levels, both surface and airborne, throughout the plant.

2. Perform radiation monitoring for special plant operations and specific maintenance activities as required and maintain records of the surveys performed.

3. Establish and implement the ALARA and RWP/planning programs for RCA work/exposure control.

4. Provide for effective access control by establishing zones throughout the plant, based on radiation and contamination levels.

5. Provide, maintain, and instruct plant personnel in the proper use of protective clothing and respiratory equipment for plant operation and maintenance.

6. Provide procedures for dealing with radiation hazards in performing day to day operations and maintenance and verify the effectiveness of such procedures.

7. Develop and implement the personnel dosimetry and respiratory protection programs.

8. Assist in the plant personnel training and emergency preparedness programs by providing specialized training in radiation protection and expertise/manpower for emergency response and scenario development.

9. Make recommendations for performing equipment, area, and personnel decontamination.

10. Receive and ship radioactive materials, including special nuclear, source, and by-product material to ensure compliance with Federal and State regulations.

11. Calibrate and maintain Health Physics, laboratory, and dosimetry equipment and facilities.

12. Sample, analyze, and perform calculations necessary to ensure that radioactivity discharged to the environment in effluents is kept as far below established limits as is reasonably achievable.

13. Assess internal exposures and maintain official dose records.

14. Provide oversight and direction to the chosen dosimetry processor in processing Radiological Environmental and site personnel TLDs. RN 99-163

15. Develop, direct, and implement the Radwaste Process and Minimization program.

16. Develop and maintain the ODCM, serve as member of PSRC, and as Radiation Safety Officer for various DHEC rad material licenses.

The Health Physics section supervises the implementation of the procedures specified by the Manager, Health Physics Services, for personnel and environment exposure reduction during normal operations, maintenance, inservice inspection, radwaste handling, and refueling in accordance with ALARA policy. RN 99-129

12.3.2 FACILITIES, EQUIPMENT, AND PROCEDURES

12.3.2.1 <u>Facilities</u>

12.3.2.1.1 Health Physics Facilities

The Health Physics facilities consist of:

1. A Health Physics room and instrument storage area, sampling room, counting room, personnel decontamination area, locker room, and protective clothing laundry area on the 412’ elevation of the Control Building.

2. Health Physics station(s), containing protection clothing, portable survey equipment, and other Health Physics materials at locations around the plant specified by the Manager, Health Physics Services to be of strategic importance for contamination and exposure control.

3. A Health Physics facility for Dosimetry, Whole Body Counting, and Respiratory Protection.

4. An offsite Radiological Support facility located at the Environmental Lab which provides laboratory and equipment resources necessary for the radiological, environmental, and emergency response programs. RN 14-034 RN 99-163

The relevant areas of the Control Building at elevation 412' are shown in Figures 12.1-19 and 12.1-20 to illustrate the layout of the access control area Health Physics facilities.

12.3.2.1.2 Instrument Storage, Calibration, and Maintenance Facilities

Portable instruments for routine Health Physics surveys are stored in the Health Physics areas around the CB412 HP Lab, RCA Access Control Point, and AB412 HP Instrument Calibration Facility. Other portable instruments are located at Health Physics stations around the plant (see Section 12.3.2.2.2) and at the Environmental Lab. Portable instruments for emergency use are stored at strategic locations around the plant, Environmental Lab, and other approved offsite locations. RN 99-164 RN 14-034

Portable instruments and laboratory equipment are routinely maintained and calibrated by Health Physics personnel or by returning the instruments to the manufacturer for repair.

12.3.2.1.2.1 Personnel Contamination Control Facilities

The primary contamination control facility for personnel is the access control area, which is illustrated in Figure 12.1-19. Additional contamination control points are established in the immediate vicinity of known contaminated areas, as necessary, to prevent the spread of contamination. These points provide a controlled interface between the clean and contaminated areas. The personnel decontamination showers are shown in Figure 12.1-19. Additionally, emergency showers are located around the plant in areas with a high potential for personnel contamination. Personnel are checked for contamination with the equipment described in Table 12.3-1.

12.3.2.1.2.2 Equipment Contamination Control Facilities

An equipment decontamination area is located adjacent to the radwaste processing area of the Auxiliary Building. This facility contains installed and portable decontamination equipment. A decontamination area for spent fuel shipping casks is also provided, and is located in the Fuel Handling Building.

12.3.2.2 <u>Equipment</u>

12.3.2.2.1 Selection Criteria

The equipment and instrumentation used in the Health Physics program for the assessment of radioactivity and personnel exposures are selected according to the following criteria:

1. Ability to measure the quantity of interest to an acceptable degree of precision and accuracy.

2. Ease of operation, maintenance, and calibration.

3. Appropriate sensitivity and range for various operational situations, including normal operations, anticipated operational occurrences, and accident conditions, as determined by the requirements of applicable regulations pertaining to each measurement situation.

These criteria apply to portable instrumentation for radiation and contamination surveys, laboratory technical equipment, inplant airborne radioactivity monitoring and sampling equipment, area monitors, and personnel monitoring equipment.

12.3.2.2.2 Protective Clothing and Equipment

The following respiratory protection devices are made available at the plant:

1. Full face particulate filter masks.

2. Full face masks with air line respirator.

3. Hoods or suits with air line respirator.

4. Full-face masks with self contained breathing apparatus of the bottle air or chemox type.

Protective clothing is required in contaminated areas or in areas where the potential for radioactive material contamination exists. Protective clothing available at the plant includes the following:

1. Coveralls.

2. Laboratory coats.

3. Plastic suits.

4. Disposable caps.

5. Hoods.

6. Plastic and rubber shoe covers.

7. Plastic, rubber, and cotton gloves.

8. Disposable protective clothing. RN 14-034

This protective clothing and equipment is located at the access control area clothing storage and issue room, and around the plant, at locations maintained by the Health Physics group. The protective clothing and equipment is readily available at key location(s) throughout the plant, without the need to return to the access control area to obtain this equipment. Also, protective clothing and equipment is provided in the emergency kits located around, and outside, the plant.

12.3.2.2.3 Technical Equipment and Instrumentation

The equipment and instrumentation necessary to measure radioactivity, radiation fields, and exposures falls into two major categories:

1. Fixed or installed.

2. Portable.

The fixed equipment includes the Radiation Monitoring System, access-control contamination monitors, radiochemistry laboratory equipment, personnel dosimetry readout equipment, and the counting room radioactivity analysis instrumentation. The portable equipment includes radiation and contamination survey instruments, airborne radioactivity samplers, movable airborne radioactivity monitor, and personnel dosimeters. RN 02-049

The Radiation Monitoring System is described in Sections 11.4, 12.1.4, and 12.2.4. The access-control contamination monitors are described in Table 12.3-1. The sensitivity of these monitors is a function of the background level at the location of the monitor; the monitors are shielded to the extent practicable in order to increase their sensitivity. The radiochemistry laboratory equipment includes fume hoods, analytical balances, glassware, reagents, and other materials and equipment as required to perform analyses for personnel protection, radioactivity surveys, and related health physics functions. The counting room instrumentation is described in Table 12.3-2.

The portable Health Physics equipment is described in Table 12.3-3. These devices are calibrated at least annually and also after any major maintenance is performed on an instrument. These calibrations are performed using a source of the appropriate type directly or indirectly traceable to the National Institute of Standards and Technology. RN 14-034 RN 00-038

12.3.2.2.4 Accident Radiation Monitoring

In-plant accident radiation monitoring combines the use of fixed radiation monitoring equipment and the use of portable monitoring instrumentation.

Area gamma radiation monitors and atmospheric radiation monitors are described in Sections 12.1.4 and 12.2.4. Portable monitoring instrumentation is tabulated in Table 12.3-3.

The fixed radiation monitors (area and airborne) are supplied with diesel backed power and are designed to operate in the normal and anticipated operational environment (see Tables 12.1-20 and 3.11-3), except that the area monitors in the Reactor Building are not required to withstand LOCA environment. This function is performed by area monitor, RM-G7, as described in Section 12.1.4.

In the event of an unplanned gas release involving high concentrations of noble gases, silver zeolite filter cartridges will be used for sampling atmospheres as conditions dictate. Silver zeolite cartridges do not significantly absorb noble gases, enabling a more accurate determination of the radioiodine collected. Fixed process radiation monitors, portable continuous air monitors, and grab samplers can utilize silver zeolite cartridges. Table 12.3-3 outlines the types and quantities of portable air samplers available. Table 11.4-1 lists the fixed process radiation monitors which detect radioiodine.

Normal laboratory instrumentation, i.e., Intrinsic Gemanium detectors and a gas flow proportional counter, will be utilized to determine the quantity of radioiodine collected on a filter cartridge. Portable single channel analyzers and NaI detectors are available if normal laboratory instrumentation is inoperable. This equipment is available in the Control Building. RN 99-162

Health Physics procedures include provisions for determining iodine concentrations in noble gas environments. Health Physics personnel are trained on action levels requiring the use of silver zeolite cartridges and the use of portable single channel analyzers and NaI detectors. This meets the requirements of NUREG-0578 Section 2.1.8.C.

12.3.2.3 Health Physics Program: Procedures

It is the intent of SCE&G that the Health Physics Procedures discussed in this section will conform with 10 CFR 20 and follow the recommendations contained in Regulatory Guides 8.7, 8.8, 8.9, and 8.10.

12.3.2.3.1 Radiation Surveys

12.3.2.3.1.1 Radiation Field Surveys

The Health Physics group conducts routine measurements of radiation field intensities around the plant, using portable instrumentation appropriate for the type(s) of radiation present, on a schedule which is determined by:

1. The level of radiation exposure rate.

2. The variability of radiation level.

3. The occupancy factor of the location.

Routine radiation field survey frequencies could increase or decrease due to plant conditions. Locations of strategic importance for ALARA (high exposure rate and occupancy) are routinely surveyed and the latest measured values are documented and available for review prior to work in these areas. Trending of specific areas, such as those containing RHR, and letdown system components is done on a routine basis to provide job planning information in such areas. Records are maintained of the results of these surveys as required by 10CFR20. RN 99-164

Prior to the initiation of any operation for which a RWP is required (see Section 12.3.2.3.7) a survey is made of the radiation field(s) in the vicinity where the operation is to be performed, using instrumentation appropriate for the type(s) of radiation present. The results of this survey are recorded on the RWP.

Further aspects of radiation-field surveys are discussed in Section 12.3.2.3.2, and the portable instruments are discussed in Section 12.3.2.2.3.

12.3.2.3.1.2 Surface Contamination Surveys

The Health Physics Services group conducts routine surface contamination surveys at locations around the plant, including the access control areas, administrative area, lunchrooms, control room, and main entrance security gate. All locations of importance for controlling the potential spread of contamination are surveyed at least weekly, using the "smear" technique, or an appropriate portable instrument, and the remaining locations are surveyed at least monthly. These survey frequencies could increase or decrease due to plant conditions. The results of these surveys are recorded and tabulated by locations so that trends in the data may be readily observed. RN 99-129

Contamination surveys are also made on personnel, equipment, and materials from time to time as necessary to ensure complete control over the levels and spread of removable contamination. Further aspects of contamination control are discussed in Sections 12.3.2.3.3 and 12.3.2.1.

12.3.2.3.1.3 Airborne Radioactivity Surveys

Levels of airborne radioactivity are assessed on a routine basis to ensure compliance with 10 CFR 20. These assessments are performed using equipment and techniques appropriate for the type(s) of radioactivity present at the sampling locations.

Any operation requiring respiratory protection will have the results of airborne radioactivity in μci/cc placed on the radiation work permit for that operation. The airborne radioactivity may be assessed prior to the operation but must be assessed during the operation.

The results of routine airborne radioactivity surveys are recorded by location so that trends in the radioactivity level can be readily identified. Air sampling surveys are performed at least daily in areas with both high occupancy factor and a high potential for the existence of airborne radioactivity. Other locations are surveyed weekly or monthly, as a function of the significance of the location as a source of occupational radiation exposure (ORE) from airborne radioactivity. These survey frequencies could increase or decrease due to the plant conditions. Section 12.3.2.3.4 contains further discussion of airborne radioactivity surveys.

12.3.2.3.2 Radiation Area Access Control

All areas of the plant are subject to access control restrictions to an extent proportional to the potential for ORE in each area. The particular access control requirements for each area are specified by the Manager of Health Physics Services. RN 99-129

Specifically, any area of the plant in which the radiation level is at least 100 mrem/hr at 12 inches (30 cm) from a radiation source or from any surface that the radiation penetrates will be declared a "high radiation area", and access to such areas is restricted. These areas are posted and secured in accordance with 10 CFR 20. Physical barriers to control access to high radiation areas include: locked and/or annunciated doors or gates, fences, or rope barriers. These barriers are arranged in such a manner as to allow exit from the area without a key or other special device. RN 14-034

Administrative measures for access control to high radiation areas include the use of an RWP for operations taking place in such areas. The estimated exposure for individuals entering such areas is based on radiation surveys made prior to, or at the start of work, and the nature of the operation to be performed. Again depending upon the operation, a health physics technician may be assigned to supervise stay times and make appropriate surveys while the operation is in progress. RN 99-159

12.3.2.3.3 Contamination Control

Surveys for contamination control are performed by the Health Physics Services group on a routine basis at various locations around the plant (see Section 12.3.2.3.1.2). Nonroutine surveys are also made whenever there is a possibility of the spread of contamination by personnel, equipment, or materials. RN 99-129

Since the complete removal of surface contamination from parts of the plant is a practical impossibility, certain plant areas may be designated as "contamination areas". The level of contamination and the number of these areas are reduced to the minimum practicable level, and control points are established to prevent the spread of contamination to other plant areas. Entrance to such an area requires compliance with the protective clothing and equipment requirements specified by Health Physics. These areas are conspicuously posted and roped off from normal traffic. Such areas are under the supervision of the Health Physics Services group. RN 99-159 RN 99-129

Personnel shall be checked for contamination immediately after exiting a contaminated area at the nearest personnel monitor, or by standing on a "hand-and-foot" or "portal" monitor for a predetermined length of time or by portable Health Physics instrumentation. Permanent personnel contamination monitors are located so that an intercom station is close enough that help can be summoned if necessary.

Any equipment or materials leaving a potentially contaminated area, and all equipment or materials leaving the plant radiation controlled area, will be surveyed for surface contamination and nonremovable radioactivity using smears and/or portable instruments of appropriate sensitivity.

12.3.2.3.4 Airborne Radioactivity Assessment and Control

Airborne radioactivity is routinely assessed using local sampling, moveable air monitors, and the installed Radiation Monitoring System. Airborne radioactive materials (particulates, noble gases, halogens, tritium) are sampled and analyzed using appropriate techniques. Since local sampling will provide better estimates of airborne contamination levels existing in a work area than will a monitor reading, such special air sampling is used in the RWP program to keep ORE due to airborne radioactivity ALARA. Moveable air monitors and the installed airborne radioactivity monitors are used to provide alarm indications and additional information which is used with local sampling for the assessment of airborne radioactivity.

Airborne radioactivity is controlled by minimizing to the extent practical the sources of this radioactivity, such as surface contamination and leaking valves; and confining unavoidable or accidental releases of airborne radioactivity in plant areas through the use of the ventilation system.

12.3.2.3.5 Respiratory Protection

The respiratory protection equipment available at the plant is listed in Section 12.3.2.2.2. The issuance of this equipment is specified, when necessary, as part of the RWP program, on the basis of special air sampling conducted in areas where work is being performed. This equipment is selected and maintained by the Health Physics group. The fitting and training programs for respiratory protection equipment is the responsibility of the Manager of Health Physics Services. Training and fitting is required for all personnel who use this equipment. RN 99-129

12.3.2.3.6 Radioactive Materials Handling

Radioactive materials, sealed or unsealed, are handled and stored in such a manner that ORE will be kept ALARA.

Examples of specific methods of handling and storage of radioactive materials to meet this objective include:

1. Minimizing the distance that radioactive samples are transported by personnel.

2. Use of shielded sample transporters as appropriate.

3. Storage of calibration sources in appropriately shielded, labeled, and secured containers.

4. Use of remote actuators or similar devices to maximize the distance of personnel from high level sources.

5. Maintenance of accurate source activity inventories on all calibration sources.

Procedures are written and implemented by the Manager of Health Physics Services for the handling of radioactive materials, including procedures for the safe recovery from radioactive spills or source leakage. RN 99-129

12.3.2.3.7 Radiation Work Permit Program

A RWP program is established as an integral part of the ALARA policy implementation, and is the responsibility of the Manager of Health Physics Services. A RWP may, at the discretion of the Manager of Health Physics Services, be required for any operation (including maintenance, adjustments, calibrations, inspections, or normal operations) which: RN 99-129

1. Takes place in a radiation, high radiation, or airborne radiation area, as defined in 10 CFR 20 or in a contaminated area as specified in the plant administrative procedures.

2. Involves maintenance or other adjustments to any system or component which contains, stores, transports, or collects radioactive materials.

3. In the judgement of the Manager of Health Physics Services warrants the issuance of a RWP prior to initiating the operations. RN 99-129

The primary objectives of the RWP program are:

1. To ensure that the radiological conditions associated with operations involving radioactive materials, directly or indirectly, are known as accurately as possible.

2. To ensure that appropriate and adequate protective measures are taken against these conditions.

3. To ensure that the personnel involved in these operations are aware of the radiological conditions and required protective measures associated with the operation.

4. To ensure that the Manager of Health Physics Services, Health Physicist, Field Operations, or the HP Shift Leader are aware of such operations, and approves of the required protective measures. This approval will be indicated by the signature of any of the previously mentioned individuals on the RWP form. RN 99-165

5. Provides a means for tracking radiation dose by group, job, location, and component.

The assessment of the radiological conditions associated with operations requiring the issuance of a RWP is performed by the Health Physics Services group. The protective measures taken to mitigate these conditions will be subject to the approval of the Manager of Health Physics Services. RN 99-129

Examples of such measures include:

1. Requiring the use of personnel protective clothing and equipment, as appropriate, for the conditions involved.

2. Limitation of stay time in radiation/radioactivity areas.

3. The use of temporary shielding, temporary containments, remote handling devices, or other techniques to minimize the conditions in the operation area.

The RWP form is used to record the total estimated exposure received as derived from the personnel dosimetry readings in order to acquire a total estimated man-rem figure. The actual stay time and estimated exposure of each individual in the RWP-controlled area is recorded electronically via computer or on a dose card. At the discretion of a Health Physicist, Health Physics group personnel may be assigned to supervise the implementation of RWP's and/or to make special surveys during the course of the operation. 98-01 Personnel involved in the operation acknowledge (by their signature on the RWP form or other appropriate documents) their commitment to the ALARA policy and their obligation to abide by the radiological protection decisions of the Health Physics group personnel.

12.3.2.3.8 Radiation Protection Program Review

The Manager of Health Physics Services is responsible for conducting an ongoing review of the plant radiation protection program, with the primary objective of finding means to reduce ORE as far below the limits of 10 CFR 20 as reasonably achievable. This review includes an analysis of the following aspects of the program: RN 99-129

1. Personnel exposures, as indicated by RWP data and periodic dosimetry readings, to identify trends and identify groups or individuals that are receiving large doses.

2. Radiation survey data, to identify trends in exposure or contamination levels for various locations around the plant.

3. Methods and procedures, to determine means of further reducing ORE's, when practical.

4. Equipment and facilities, to determine where improvements are practical.

In addition to these reviews, management provides independent audit functions as described in Section 12.3.1.1.1.

12.3.2.3.9 Radiation Protection Training

Plant personnel, both permanently assigned and temporary, receive training in the principles of radiation protection commensurate with the degree of hazard associated with their respective work assignments. Such training encompass at least the following topics:

1. Properties of radiation and radioactivity.

2. Biological effects of radiation on humans.

3. Measurement of radiation and radioactivity.

4. Principles and techniques of radiation protection.

5. General regulatory and specific facility license radiation protection requirements.

6. Specific procedures appropriate for particular work assignments.

7. ALARA concept and management commitment thereto.

Personnel who are reassigned to positions involving a greater degree of radiation hazard are retrained and tested at the higher level of radiation protection proficiency required for that assignment. The Manager of Health Physics Services approves the content of the radiation protection training program. RN 99-129

12.3.3 PERSONNEL DOSIMETRY

12.3.3.1 External Radiation Exposure

Plant employees, visitors, and support personnel who enter the radiation controlled area are required to wear personnel dosimetry. The type of dosimetry issued is based upon expected radiological conditions. Neutron exposure is tracked using staytime and exposure rate calculations and is measured (for record) using neutron sensitive dosimetry or calculations, as appropriate. Beta and gamma exposure is tracked/controlled using pocket dosimeters, electronic dosimeters, calculations, or secondary thermoluminescent dosimeters (TLDs) as appropriate. Beta and gamma exposure is measured (for record) by TLD or by calculation/investigation in unusual circumstances. TLDs are processed periodically, not to exceed annually, by a NVLAP accredited processor. TLDs may be processed more frequently during outages, refueling, or when an individual's exposure status is in question. Records of exposure are kept for each individual in accordance with the recommendations of Regulatory Guide 8.7. The results of the personnel dosimetry measurements are periodically analyzed to find trends in exposures, identify groups or individuals whose exposures are consistently high, and to find methods of reducing these exposures. RN 06-017

12.3.3.2 Internal Radiation Exposure

Internal radiation exposure is assessed using whole body counting and bioassay techniques and estimated where appropriate by calculational methods. The concepts, models, equations, and assumptions used for the assessment of internal radiation exposure follow the recommendations of Regulatory Guide 8.9 or those regulations/regulatory guides in effect at the time. All personnel who regularly enter areas where the potential exists for inhalation, ingestion, or absorption of radioactive materials are routinely assessed for internal contamination. Nonroutine internal radioactivity assessments are made whenever there is reasonable chance that personnel have inhaled, ingested, or absorbed radioactive materials. RN 00-017

TABLE 12.3-1

CONTAMINATION CONTROL MONITORS

INSTRUMENT TYPE	REMARKS	
Portal Monitor	Alarm sounds on high activity or insufficient length of count. Controls egress from potentially contaminated areas.	RN 99-164
"Frisker"	"Pancake" type detector for quick contamination checks of personnel or equipment.	RN 99-162

TABLE 12.3-2

FIXED LABORATORY INSTRUMENTATION

INSTRUMENT	DETECTOR	SENSITIVITY[1]	NUMBER	LOCATION	REMARKS	
Low Background Gas Flow Proportional Counter	Proportional Counter	200 dpm	2	412' CB	Used for counting smears, residue from evaporated water and radiochemistry samples; equipped with automatic sample chamber.	
Spectroscopy System	Intrinsic Germanium	As per the Technical Specifications	2	412' CB	Spectroscopy system with low-level shield; used primarily for effluent sample analysis.	RN 99-162
Liquid Scintillation	--	As per the Technical Specifications	1	412' CB	Used for tritium determination in water, air (freeze out or gel), and urine.	
Shielded GM Counting System	GM	200 dpm	2	412' CB	High-level sample counting; backup for low background system.	RN 14-034
Laboratory Monitor	GM	N/A	1	412' CB	Check potentially high-level samples before placing in low-level system; monitor background in counting room, warn of changes.	

TABLE 12.3-2 (Continued)

FIXED LABORATORY INSTRUMENTATION

INSTRUMENT	DETECTOR	SENSITIVITY [1]	NUMBER	LOCATION	REMARKS	
Self-Reading Pocket Dosimeter Chargers	N/A	N/A	2	412' CB	Used for charging and/or zeroing pocket chamber-type dosimeters.	RN 99-163
Electronic Dosimeter Readers	N/A	N/A	2	412' CB	Used for programming and turning electronic dosimeters on/off.	RN 14-034

(1) Instruments will be calibrated as part of the counting laboratory quality control program, using NBS-traceable sources appropriate for each counting system.

TABLE 12.3-3

PORTABLE HEALTH PHYSICS INSTRUMENTS

INSTRUMENT	RADIATION DETECTED	RANGE	ACCURACY	NUMBER[1]	LOCATION	REMARKS	RN 14-034
GM Survey Meter	beta gamma	0-200 mr/hr	± 10% full Scale	6	Refer to Section 12.3.2.1.2	Equipped with end window, side window, or pancake probe.	RNs 99-164 12-033
GM Survey Meter	gamma	0-1000 R/hr	± 10%	14	Refer to Section 12.3.2.1.2	Fishing pole type with extendable probe; 20 in to 13 ft.	
GM Survey Meter-Friskers	gamma	0-1M cpm	± 10%	50	Refer to Section 12.3.2.1.2	Primarily personnel monitoring usage.	RN 12-033
GM Survey Meter-Remote Monitoring Gamma	gamma	0-1000 mr/hr	± 10% full scale	13	Refer to Section 12.3.2.1.2	Provide remote area monitoring capability.	RN 12-033
Ionization Survey Meter	beta gamma	0-5000 mr/hr	± 10% full Scale	33	Refer to Section 12.3.2.1.2	Air ionization chamber.	RNs 99-164 12-033
High Range Survey Meter	beta gamma	0-1000 R/hr	± 10% full Scale	2	Refer to Section 12.3.2.1.2	Air ionization chamber.	
Neutron Rem Counter	thermal through fast neutrons	0-5 Rem/hr	± 10% full Scale	3	Refer to Section 12.3.2.1.2	A BF_3 tube inside a moderator for detection of neutrons up to 10 MeV.	RN 12-033
Proportional Alpha Counter	alpha	0-150,000 cpm	± 10% full Scale	1	Refer to Section 12.3.2.1.2	Air or gas proportional chamber.	RN 12-033
Scintillation Counter	alpha	0-2M cpm	± 10% full Scale	1	Refer to Section 12.3.2.1.2	An alpha scintillation crystal covered with a thin mylar window.	RNs 99-164 12-033
Pocket Dosimeter	gamma x-ray	0-500 mr	± 10%	360	Various	Used for personnel exposure tracking. Also placed in Emergency Kits. The number is based on having sufficient in stock for E-kit changes.	RNs 99-164 14-034
Pocket Dosimeter	gamma x-ray	1500 mr	± 10%	120	Various	Used for personnel exposure tracking. Also placed in Emergency Kits. The number is based on having sufficient in stock for E-kit changes.	

TABLE 12.3-3 (Continued)

PORTABLE HEALTH PHYSICS INSTRUMENTS

INSTRUMENT	RADIATION DETECTED	RANGE	ACCURACY	NUMBER[1]	LOCATION	REMARKS	RN 14-034
Electronic Dosimeters	gamma	-	± 20%	400	Various	Used for personnel exposure tracking. They are rate and integrating. Number is based on having sufficient in stock to support operational and E-kit needs.	RNs 99-164 14-034
Thermoluminescent Dosimeters	gamma, beta, neutron	-	± 20%	variable	Vendor Supplied	Used personnel exposure monitoring as dose of legal record. Integrating.	RN 14-034
Job Site Air Samplers	-	-	-	14	Various	Particulate and iodine sampler; may have extendable sample head (tygon); normally used at job sites for longer or continuous sampling.	RNs 99-164 12-033 14-034
Hand Held Air "Grab" Samplers	-	-	-	17	Various	Particulate and iodine sampler; normally used for quick grab samples or short duration jobs.	
Tritium Samplers	-	-	-	2	Various		
Noble-Gas Samplers	-	-	-	5	Various		

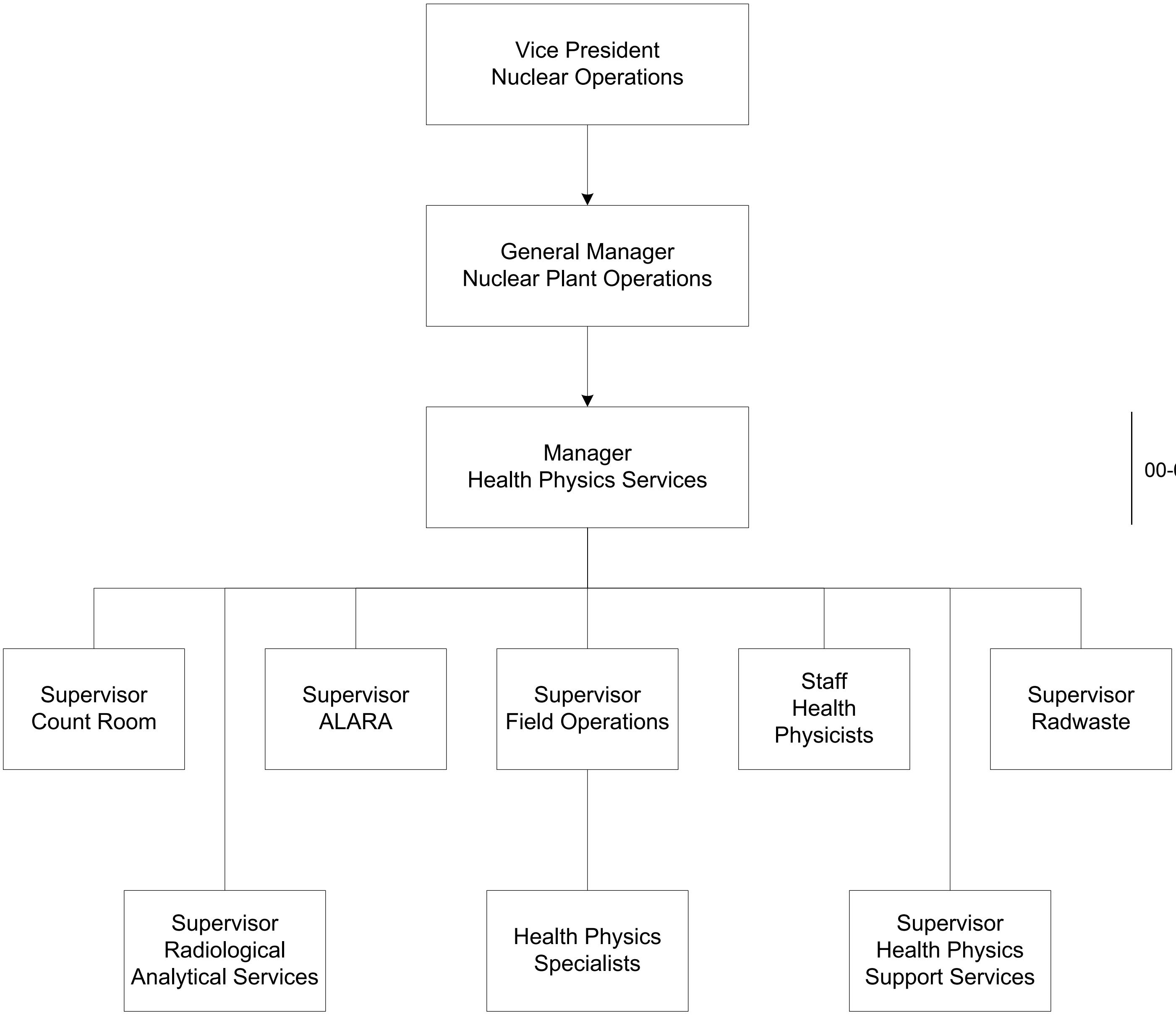
(1) The number of instruments is based on the minimum number that should be in inventory (not necessarily in calibration or in service) to support normal operational needs and in some cases, emergency kits; exceptions are noted in the remarks.

TABLE 12.3-4

RESPIRATORY PROTECTION EQUIPMENT

DEDICATED TO:	TYPE	QUANTITY
Normal Operations and Major Outages	Full Face Particulate Respirator.	100
	Full Face Continuous Air Flow Respirator.	40
	Continuous Air Flow Hood.	6
	Self-Contained Breathing Apparatus (30 Min. Air Supply).	10
Emergency Use	Full Face Particulate Respirator.	20
	Self-Contained Breathing Apparatus (30 Min. Air Supply).	14
	Air Supplied Suit or Hood	4

NOTE FIGURE 12.3-1
Figure 12.3-1 is being retained for historical purposes per NEI 98-03, Revision 1. 00-01



00-01

SOUTH CAROLINA ELECTRIC & GAS CO,
VIRGIL C. SUMMER NUCLEAR STATION

Health Physics
Organization Chart

Figure 12.3-1

Amendment 00-01
Decmeber 2000

12A DESIGN REVIEW OF PLANT SHIELDING AND ENVIRONMENTAL QUALIFICATION OF EQUIPMENT FOR SPACES/SYSTEMS WHICH MAY BE USED IN POST ACCIDENT OPERATIONS OUTSIDE THE REACTOR BUILDING AT THE VIRGIL C. SUMMER NUCLEAR STATION

12A.1 INTRODUCTION

This report has been prepared in response to the September 13, 1979, letter from Darrel G. Eisenhut of the NRC to all operating nuclear power plants. This letter presented an implementation schedule for the recommendations presented in "TMI-2 Lessons Learned Task Force Status Report and Short-Term Recommendations" (NUREG-0578). The requirements defined in the September 13th letter were subsequently clarified in a letter from Harold R. Denton to all operating nuclear power plants dated October 30, 1979.

Among the requirements defined in the two NRC letters is a review to determine whether post-accident radiation fields unduly limit personnel access to areas necessary for mitigation of or recovery from an accident or unduly degrade the proper operation of safety equipment. Corrective actions for problems identified as a result of the review are also to be determined. This report presents the results of such a review for the Virgil C. Summer Nuclear Station.

The review was based on the following guidelines:

a) The post-accident dose rate in areas requiring continuous occupancy should not exceed 15 mr/hr.

b) The post-accident dose rate in areas which do not require continuous occupancy should be such that the dose to an individual during a required access period is less than 5 Rem whole body or its equivalent, in order to maintain compliance with General Design Criterion 19.

c) The integrated dose to safety equipment as a result of the accident should be less than the dose for which the equipment has been qualified to ensure that the capability of the equipment to perform its safety function has not been degraded, in order to maintain compliance with General Design Criterion 4.

d) The minimum radioactive source term used in the evaluation should be equivalent to the source terms recommended in Regulatory Guides 1.3 and 1.4 [4].

12A.2 SUMMARY

This report provides a review of the Virgil C. Summer Nuclear Power Plant to determine whether post-accident radiation fields unduly limit personnel access to areas necessary for mitigation of or recovery from an accident, or unduly degrade the proper operation of safety equipment. This review was accomplished using the recommendations and guidelines of NUREG-0578 and subsequent clarifications as provided by the NRC.

A detailed discussion of the evaluation of plant vital areas and equipment qualification for post-accident conditions is presented in Sections 4.0 and 5.0. Results indicated that certain areas of the plant required additional radiation shielding to reduce the post-accident radiation fields. These areas and the shielding modifications utilized are described in the following paragraphs.

Area "A" is the location of the residual heat removal (RHR) pumps, reactor building spray pumps, and the pump room cooling units. The RHR/Spray piping is adequately shielded and does not restrict post-accident access to this area. However, a significant potential source of exposure to plant personnel can result from the RHR/Spray pump room sump line. This line was unshielded and located in the hallway between pump room B and the waste holdup tank, and resulted in a potentially high radiation area during both normal and accident operating conditions. The resultant dose levels in the subject area have been reduced to acceptable values by the re-routing and venting of the line to permit the line to drain during sump pump shutdown periods, and the placement of lead shielding to prevent streaming to the operating floor 374’-0". The new pipe configuration and radiation shielding allows personnel access in accordance with the occupancy requirements of Table 12A.4-2, and the resultant integrated dose meets the dose requirements of 10CFR50, Appendix A, General Design Criterion 19 as outlined in NUREG-0737.

Following an accident, personnel access to the Auxiliary Building is controlled at the 412’-0" elevation from the Control Building. As a result, the use of the stairway and elevator located in cubicle 12-09 as an access route to vital areas above and below the 412’-0" elevation is important. A 10-inch RHR line penetrates the south and east shield walls of cubicle 12-09 at elevation 429’-6" resulting in significant doses to workers in this area. Shortly after the accident, the integrated dose to a worker transversing this area would be of the order of 1 rem. The resultant dose levels in the subject area have been reduced to acceptable values by the addition of lead shielding. The new configuration allows personnel access in accordance with the occupancy requirements of Table 12A.4-2, and the resultant integrated dose meets the dose requirements of 10CFR50, Appendix A, General Design Criterion 19 as outlined in NUREG-0737.

Area "I" is the location of the charging/safety injection pump room cooling units. This area was found to contain high radiation levels due to unshielded CVCS piping located at several places within the area. In order to permit the required occupancy time in the area for manual valve alignment or maintenance, the resultant dose levels in the area have been reduced to acceptable values by the addition of lead and concrete shielding. The new configuration allows personnel access in accordance with the occupancy requirements of Table 12A.4-2, and the resultant integrated dose meets the dose requirements of 10CFR50, Appendix A, General Design Criterion 19 as outlined in NUREG-0737.

In the area of the Intermediate Building between the east and west penetration access areas there were several RHR, safety injection, and CVCS lines which were routed above the sides of an open pipe chase at elevation 422'-3". This permitted direct streaming from the piping to reach the floor area of Intermediate Building floor elevation 412'-0" which contains safety related equipment. The resultant dose levels in this area of the Intermediate Building have been reduced to acceptable values by the modification of the pipe chase as follows: (1) extension of the pipe chase side above the 12" RHR line, (2) addition of 6" concrete shielding to the existing 1 foot concrete shielding, (3) re-location of an access ladder to prevent streaming through a concrete cutout. The new configuration allows personnel access in accordance with the occupancy requirements of Table 12A.4-2, and the resultant integrated dose meets the dose requirements of 10CFR50, Appendix A, General Design Criterion 19 as outlined in NUREG-0737.

12A.3 SOURCE TERMS AND CALCULATIONAL METHODOLOGY

12A.3.1 SOURCE TERMS

12A.3.1.1 Basis for the Source Terms

The activity releases assumed in this review are based on the assumptions and regulatory positions contained in the Regulatory Guides 1.4 [4] and 1.7 [5]. The activity assumed for liquid source term calculation is based on 100 percent of the noble gas inventory, 50 percent of the halogen core inventory, and 1 percent of all other nuclides in the core inventory. The activity assumed for gaseous source term calculation is based on 100 percent of the noble gas core inventory and 25 percent of the halogen core inventory.

12A.3.1.2 Liquid Source Terms

Two liquid sources are considered in the design review: (1) the undiluted fluid as found within the reactor coolant system, and (2) the diluted fluid as found within the reactor building recirculation sump. The first source term, undiluted reactor coolant, is required in the examination of those systems whose flow originates from either the Reactor Coolant System or an auxiliary system containing the undiluted primary fluid. The source terms are based on the dilution of the liquid activity inventory discussed in the first paragraph with the fluid volume of the Reactor Coolant System. This source is used for the examination of the reactor coolant fluid sampling section of the Nuclear Sampling System, and those portions of the Chemical and Volume System associated with the degasification of the reactor coolant fluid.

In the second liquid source term, consideration is given for the dilution of the liquid activity inventory discussed in the first paragraph with the fluid volume contained in the reactor building recirculation sump. The minimum fluid volume expected in the sump, and the individual contributors to that volume, are given in Table 12A.3-1. These source terms are utilized in the examination of those systems which receive their fluid supply from the reactor building recirculation sump. Those systems which are considered in the review are:

- Residual Heat Removal System
- Reactor Building Spray System
- Safety Injection System
- Nuclear Sampling System (RHR process fluid sample)

12A.3.1.3 Gas Source Terms

The gaseous source terms were determined for containment and the waste gas system using the activity releases described in Section 12A.3.1.1.

The containment airborne source term was based on the dilution of the gaseous activity inventory by the containment free volume atmosphere.

Although not designed for reactor coolant degasification under post-accident conditions, the waste gas system was evaluated for the exposures emitted and received under the post-accident conditions. The waste gas system is designed to remove the fission product gases from the reactor coolant contained in the Volume Control Tank (VCT). The amount of fission gases removed from the reactor coolant in the VCT and collected by the waste gas system can be related to the amount entering the VCT as follows:

(1) Stripping Efficiency (SE):

$$SE = \frac{C_R - C_L}{C_R - C_{L_{eq}}}$$

2) Stripping Fraction (SF):

$$SF = \frac{C_R - C_L}{C_R}$$

where C_R = the gas concentration in the reactor coolant liquid entering the Volume Control Tank,

C_L = the gas concentration in the reactor coolant liquid leaving the Volume Control Tank,

$C_{L_{eq}}$ = the gas concentration in the reactor coolant liquid leaving the Volume Control Tank, assuming the ratio of the gas concentration in the liquid and gas phases in the Volume Control Tank follows Henry's Law.

The waste gas system source terms were determined for the degasification of the reactor coolant liquid by calculation of the quantity of activity entering the Volume Control Tank via the normal letdown path. For the sake of conservatism, the stripping efficiency of this process is assumed to be 100 percent. Therefore, in the previous equation of the stripping fraction, $C_L = C_{L_{eq}}$, and the stripping fraction is:

$$SF = \frac{C_R - C_{L_{eq}}}{C_R}$$

Thus, the separation of the fission gases from the reactor coolant liquid in the VCT will follow Henry's Law. This results in the maximum theoretical gas concentration in the vapor phase of the VCT and, hence, the maximum quantity of gas enters the waste gas system.

12A.3.2 METHODOLOGY

12A.3.2.1 Calculation of Dose Rates

Dose rates for the areas of interest in this review were calculated by determining the potential contributing sources at a representative location and using the appropriate source term from Table 12A.3-2 adjusted for decay as required. The dose rate at the representative location was used as the general area dose rate for the area. The SDC computer code (Reference [1]) was used in performing the dose rate calculations. Energy groups required as input to the computer code were determined using the gamma ray energy and intensity data in References. [2] and [3] for the nuclides in Table 12A.3-2.

MicroShield (Reference [7]) was used to determine gamma energy, intensities, and dose rates for the waste gas sources and the containment sump sources. RN 05-011

12A.3.2.2 Calculation of Doses to Personnel During Post Accident Access to Vital Areas

Personnel doses received in performing a specific task in a given vital area are calculated as the sum of the doses received during travel to and from the vital area and the dose received while performing the given operation in the vital area.

The doses received during travel are determined by calculating dose rates at selected locations (or at a single location if the dose rate along the travel route is relatively uniform) along the travel route using the methodology discussed in Section 12A.3.2.1 and multiplying the dose rates by the appropriate travel time for each selected location along the travel route.

Doses received while performing a given operation are determined by multiplying the dose rate for the given area by the time required to perform the operation. Dose rates for the given vital area are determined using the methodology discussed in Section 12A.3.2.1.

12A.3.2.3 Calculation of Integrated Doses to Safety Equipment

The integrated dose to a given item of safety equipment is determined by integrating the dose rate appropriate for the given item over the time period that it is required to be available to perform its safety function. Dose rates are calculated using the methodology discussed in Section 12A.3.2.1.

TABLE 12A.3-1

CONTAINMENT SUMP MINIMUM LIQUID INVENTORY

Liquid Source	Liquid Volume (ft^3)	
Refueling Water Storage Tank	44,483 [a]	
Safety Injection Accumulators	3,052 [b]	
Sodium Hydroxide Storage Tank	406 [c]	RN 03-008
Reactor Coolant System	6,623	
Containment Sump Extension [d]	1,159	
Minimum Containment Sump Volume =	55,723 ft^3	

NOTES:

(a) Refueling Water Storage Tank at minimum operating water level at start of drawdown of the tank. The tank drawdown will be terminated at LO-LO level upon the automatic initiation of recirculation via the RHR system.

(b) The total volume attributed to the accumulators includes the minimum tank volumes plus the volume of the injection lines. RN 03-008

(c) The minimum usable volume of liquid in the Sodium Hydroxide Storage Tank.

(d) Containment Sump Extension refers to the fluid volume contained in systems that recirculate sump fluid during the recirculation phase of LOCA. RN 03-008

TABLE 12A.3-2

V. C. SUMMER
SHIELDING SOURCE TERMS (T = 0)

Isotope	Liquid [1] Source Activity (Ci)	Gaseous [2] Source Activity (Ci)	Containment Sump Concentration (μCi/cc)	Reactor Coolant Concentration (μCi/cc)	Containment Airborne Concentration (μCi/cc)	Waste Gas [3] Concentration (μCi/cc)	
							99-01
Br-84	7.5 + 6	3.75 + 6	4.80 + 3	2.90 + 4	7.20 + 1	1.40 + 4	
Kr-87	3.70 + 7	3.70 + 7	3.10 + 4	1.43 + 5	7.10 + 2	2.79 + 6	
Te-133	4.40 + 5	-	2.80 + 2	1.70 + 3	-	-	
Cs-134	1.70 + 5	-	1.10 + 2	6.56 + 2	-	-	
Cs-136	4.70 + 4	-	3.00 + 1	1.81 + 2	-	-	
Cs-137	7.03 + 4	-	4.40 + 1	2.71 + 2	-	-	
Ba-139	1.50 + 6	-	9.50 + 2	5.79 + 3	-	-	
Br-83	3.20 + 6	1.60 + 6	2.00 + 3	1.24 + 4	3.07 + 1	8.33 + 3	
Kr-83m	6.30 + 6	6.30 + 6	6.00 + 3	2.43 + 4	1.21 + 2	6.31 + 5	
Kr-85m	1.90 + 7	1.90 + 7	1.70 + 4	7.34 + 4	3.65 + 2	1.30 + 6	
Kr-85	7.42 + 5	7.42 + 5	5.30 + 2	2.86 + 3	1.42 + 1	4.20 + 4	
Kr-88	5.50 + 7	5.50 + 7	4.30 + 4	2.12 + 5	1.06 + 3	3.59 + 6	
Rb-88	5.41 + 5	-	3.40 + 2	2.09 + 3	-	-	RN
Rb-89	6.80 + 5	-	4.30 + 2	2.63 + 3	-	-	05-011
Sr-89	7.40 + 5	-	5.30 + 2	2.86 + 3	-	-	
Sr-90	5.00 + 4	-	3.20 + 1	1.93 + 2	-	-	
Y-90	5.00 + 4	-	3.20 + 1	1.93 + 2	-	-	
Sr-92	9.10 + 5	-	7.00 + 2	3.51 + 3	-	-	
Y-92	1.00 + 6	-	7.20 + 2	3.86 + 3	-	-	
Sr-93	1.10 + 6	-	7.00 + 2	4.25 + 3	-	-	
Y-93	1.10 + 6	-	8.10 + 2	4.25 + 3	-	-	
Mo-99	1.50 + 6	-	9.50 + 2	5.79 + 3	-	-	
Tc-99m	1.30 + 6	-	8.20 + 2	5.02 + 3	-	-	
Ru-103	1.10 + 6	-	7.00 + 2	4.25 + 3	-	-	
Rh-103m	1.10 + 6	-	7.00 + 2	4.25 + 3	-	-	
Ru-106	4.10 + 5	-	2.60 + 2	1.58 + 3	-	-	
Rh-106	4.10 + 5	-	2.60 + 2	1.58 + 3	-	-	

TABLE 12A.3-2 (Continued)

V. C. SUMMER
SHIELDING SOURCE TERMS (T = 0)

Isotope	Liquid [(1)] Source Activity (Ci)	Gaseous [(2)] Source Activity (Ci)	Containment Sump Concentration (μCi/cc)	Reactor Coolant Concentration (μCi/cc)	Containment Airborne Concentration (μCi/cc)	Waste Gas [(3)] Concentration (μCi/cc)	99-01
Te-132	1.10 + 6	-	7.00 + 2	4.25 + 3	-	-	
I-132	5.90 + 7	2.95 + 7	3.80 + 4	2.28 + 5	5.66 + 2	1.06 + 5	
Te-134	1.60 + 6	-	1.00 +3	6.18 + 3	-	-	
I-134	8.90 + 7	4.45 + 7	5.70 + 4	3.44 + 5	8.54 + 2	1.59 + 5	
Xe-138	1.50 + 8	1.50 + 8	9.50 + 4	5.79 + 5	2.88 + 3	1.08 + 7	
Cs-138	1.50 + 6	-	9.50 + 2	5.79 + 3	-	-	
Ba-140	1.40 + 6	-	9.50 + 2	5.41 + 3	-	-	
La-140	1.50 + 6	-	9.50 + 2	5.79 + 3	-	-	
Ce-143	1.20 + 6	-	8.20 + 2	4.63 + 3	-	-	
Pr-143	1.20 + 6	-	8.20 + 2	4.63 + 3	-	-	
Ce-144	9.20 + 5	-	5.80 + 2	3.55 + 3	-	-	
Pr-144	1.20 + 6	-	7.60 + 2	4.63 + 3	-	-	
Sr-91	9.20 + 5	-	7.00 + 2	3.55 + 3	-	-	
Y-91	9.60 + 5	-	6.50 + 2	3.71 + 3	-	-	RN 05-011
Zr-95	1.30 + 6	-	8.20 + 2	5.02 + 3	-	-	
Nb-95m	-	-	-	-	-	-	
Nb-95	1.30 + 6	-	8.20 + 2	5.02 + 3	-	-	
Zr-97	1.30 + 6	-	8.60 + 2	5.02 + 3	-	-	
Ru-105	8.60 + 5	-	5.40 + 2	3.32 + 3	-	-	
Rh-105m	8.60 + 5	-	5.40 + 2	3.32 + 3	-	-	
Rh-105	5.50 + 5	-	3.50 + 2	2.12 + 3	-	-	
Te-131	7.20 + 5	-	4.60 + 2	2.78 + 3	-	-	
I-131	4.10 + 7	2.05 + 7	2.60 + 4	1.58 + 3	3.93 + 2	7.26 + 4	
Xe-131m	6.51 + 5	6.51 + 5	4.10 + 2	2.51 + 5	1.25 + 1	2.33 + 4	
I-133	7.80 + 7	3.90 + 7	5.40 + 4	3.01 + 5	7.49 + 2	1.49 + 5	
Xe-133m	3.80 + 6	3.80 + 6	1.50 + 4	1.47 + 4	7.29 + 1	1.04 + 6	
Xe-133	1.50 + 8	1.50 + 8	1.10 + 5	5.79 + 5	2.88 + 3	7.23 + 6	

TABLE 12A.3-2 (Continued)

V. C. SUMMER
SHIELDING SOURCE TERMS (T = 0)

Isotope	Liquid [(1)] Source Activity (Ci)	Gaseous [(2)] Source Activity (Ci)	Containment Sump Concentration (μCi/cc)	Reactor Coolant Concentration (μCi/cc)	Containment Airborne Concentration (μCi/cc)	Waste Gas [(3)] Concentration (μCi/cc)	99-01
I-135	6.90 + 7	3.45 + 7	4.90 + 4	2.66 + 5	6.62 + 2	1.36 + 5	
Xe-135m	4.20 + 7	4.20 + 7	2.70 + 4	1.62 + 5	8.06 + 2	2.80 + 6	
Xe-135	2.90 + 7	2.90 + 7	2.30 + 4	1.12 + 5	5.57 + 2	1.80 + 6	RN 05-011
Ba-141	1.30 + 6	-	8.20 + 2	5.02 + 3	-	-	
La-141	1.30 + 6	-	8.90 + 2	5.02 + 3	-	-	
Ce-141	1.40 + 6	-	8.90 + 2	5.41 + 3	-	-	

NOTES:

(1) Based on 100% noble gas core inventory, 50% halogen core inventory, and 1% of all others core inventory.

(2) Based on 100% noble gas core inventory and 25% halogen core inventory.

(3) Source concentrations are for the VCT gas space and inlet line up to the waste gas compressor. 99-01

12A.4 REVIEW OF AREAS REQUIRING ACCESS FOR POST ACCIDENT OPERATIONS

A vital area is defined as any area which will or may require occupancy to permit an operator to aid in the mitigation of or recover from an accident. Vital areas for the Virgil C. Summer Nuclear Station are shown in Figures 12A.4-1 through 12A.4-7. Calculated exposure rates as a function of time after the accident are given in Table 12A.4-1. Vital area occupancy and access dose are summarized in Table 12A.4-2. Access routes to the vital areas are described in Table 12A.4-3.

12A.4.1 AREA A: RHR/SPRAY PUMP ROOMS AND COOLING UNITS

Area "A" as identified on Figure 12A.4-1 is the location of the residual heat removal (RHR) pumps, reactor building spray pumps and the common pump room cooling units. Failure of an RHR or Spray pump following an accident will result in a minimum safeguard condition and access to this area may be required to ensure that more than one pump is available for long term cooling of the reactor core. Under these circumstances, plant personnel may require access to this area for operations such as pump seal replacement or minor maintenance. Tasks such as these require that three individuals spend approximately 4 hours in the pump rooms or at the cooling units. RN 05-011

Sources of exposure in this area are the RHR and spray system piping in the system train that is operating. This piping is located in the adjacent pump room and in the valve and piping area located above at elevation 397’- 0". Piping in the train requiring maintenance or repair is assumed to be flushed and drained. Access to this area is not required immediately since only long term operation need be considered. One day after the accident, the exposure rate is 383 mr/hr in the pump rooms and 614 mr/hr at the cooling units. The calculated dose to an individual for 4 hours of occupancy is 1.41 rem and 2.25 rem, respectively. Table 12A.4-2 shows that 4 hours occupancy in this area is possible prior to one day following the accident. RN 05-011

The access route to area "A" is route "1" described in Table 12A.4-3. This access route will add a total of 0.024 rem to the worker dose if the task is started one day after the accident. RN 05-011

12A.4.2 AREA B: CHARGING/SI PUMP ROOMS

Area "B" is the location of the charging/safety injection (SI) pumps and is identified on Figure 12A.4-2. Failure of one of these pumps following an accident will result in a minimum safeguards condition and access to this area may be required to ensure that all pumps are available for long term cooling of the reactor core. Under these circumstances, plant personnel may require access to this area for operations such as pump seal replacement or minor maintenance. Tasks such as these require that three individuals spend approximately 4 hours in the pump rooms.

Sources of exposure in this area are from the charging/SI pump piping for the pumps that are operational and are located in the adjacent pump rooms. Piping in the pump room where maintenance or repair is required is assumed to be flushed and drained. Access to this area is not required immediately since only long term operation need be considered. Two days after the accident, the exposure rate is 630 mr/hr. The calculated dose of occupancy is 2.39 rem. Table 12A.4-2 shows that 4 hours occupancy in this area is possible prior to two days following the accident.

The access routes to area "B" are routes "3" and "8" and are described in Table 12A.4-3. These access routes will add 0.02 rem and 0.05 rem to the dose if the task is started two days after the accident. 99-01

12A.4.3 AREA C: CHARGING/SI PUMP C AUXILIARIES LOCAL CONTROL PANEL

Area "C" as identified on Figure 12A.4-2 is the location of charging/safety injection pump C auxiliaries local control panel. Failure of one of the charging/SI pumps will result in minimum safeguards condition and access to this area may be required for manual switching of the charging/SI pumps. This task would require that two workers spend approximately 2 minutes at the control panels.

The major source of exposure in this area is a 3 inch safety injection line located inside the shield labyrinth at the entrance to cubicles 88-23 and 88-24. A direct line of sight is possible from this source to the immediate vicinity of the pump C transfer switch. This line will not become a source of exposure until recirculation of containment sump liquid is initiated. Additional exposure in this area results from charging/SI piping in pump room 88-23.

Access to this area could be required immediately after the accident. Access at 30 minutes is assumed in order to evaluate the worst case of containment sump liquid in the 3 inch safety injection line at the start of recirculation. Assuming that the workers are in a direct line of sight with the unshielded safety injection line, the exposure rate in the area is 82.6 rem/hr which results in an occupancy dose of 2.75 rem for the 2 minute time period. RN 05-011

The access route to area "C" is route "7" described in Table 12A.4-3. This access route will add a total of 1.40 rem to the worker dose if the task is started 30 minutes after the accident. RN 05-011

12A.4.4 AREA D: SPENT FUEL PIT HEAT EXCHANGER

Area "D" is the location of the spent fuel pit heat exchangers (see Figure 12A.4-2). Access to this area is not required to aid in the mitigation of or recovery from an accident, but should conditions dictate, access may be needed to service the heat exchanger to maintain cooling of the spent fuel. Under these circumstances, post-accident operations in this area require only long-term periodic surveillance.

The exposure rate in this area following an accident is negligible since there are no post-accident sources of activity in the vicinity of the heat exchangers. This permits unlimited access (40 hours/week) in this area.

The access route to area "D" is route "9" described in Table 12A.4-3. This access route will add a total of 0.02 rem to the worker dose if the task is started one day after the accident.

12A.4.5 AREA E: RADWASTE GAS HANDLING CONTROL PANELS

Area "E" as identified in Figure 12A.4-2 is the location of the radwaste gas recombiner analyzer and control panels. Access to this area is not required following an accident since it is not planned to start-up the radwaste gas handling system. If degassing of the primary coolant system becomes necessary following an accident, the operation will be accomplished by utilizing the reactor head vent system.

12A.4.6 AREA F: SPENT FUEL PIT COOLING PUMPS AND LOCAL CONTROL PANELS

Area "F" as identified on Figure 12A.4-3 is the location of the spent fuel pit cooling pumps and local control panels. Access to this area is not required to aid in the mitigation of or recovery from an accident, but should conditions dictate, access to this area may be needed in order to maintain cooling of the spent fuel. Under these circumstances, plant personnel may require access to this area for operations such as pump seal replacement or minor maintenance. Tasks such as these require that three individuals spend approximately 4 hours at the pumps.

Dose contributors in this area are the containment building and a 3 inch safety injection line located approximately 20 feet away above the shield slab at elev. 426’-6". The time at which access to this area is required is based on the minimum time it takes to reach maximum spent fuel pool temperature with loss of both cooling pumps. This time is 12.5 hours. Eight hours after the accident, the exposure rate is 1.16 rem/hr. The calculated dose to an individual for 4 hours of occupancy is 3.03 rem.

The access route to area "F" is route "6" described in Table 12A.4-3. This access route will add a total of 0.11 rem to the worker dose if the task is started 8 hours after the accident.

12A.4.7 AREA G: ENGINEERED SAFETY FEATURE MOTOR CONTROL CENTER

Area "G" is the location of the Engineered Safety Feature (ESF) Motor Control Center 1DA2Y and is identified on Figure 12A.4-3. It is not expected that access to this area would be required since all ESF control functions are performed from the control room. However, access may be required in the unlikely event that a safety-related component loses remote control power. Should such a component lose remote control power and fail in the unsafe position, in conjunction with a simultaneous failure of its redundant counterpart, it may be deemed necessary to gain access to this area for manual control of the component. This task would require that two workers spend approximately 2 to 5 minutes in this area.

Dose contributors to this area are spray system, safety injection system, and chemical and volume control system piping located below and adjacent to this area. Access to this area could be required immediately after the accident. Access at 30 minutes is assumed in order to evaluate the worst case which is the initiation of the recirculation of containment sump liquid in the spray and safety injection systems. The exposure rate in this area is 24.1 rem/hr resulting in estimated exposures to an individual of 0.80 rem and 2.01 rem for occupancy times of 2 minutes and 5 minutes, respectively. RN 05-011

The access route to area "G" is route "4" described in Table 12A.4-3. This access route will add a total of 0.28 rem to the worker dose if access is required 30 minutes after the accident. RN 05-011

12A.4.8 AREA H: WASTE PROCESSING SYSTEM CONTROL PANEL

Area "H" is the location of the waste processing system panel (see Figure 12A.4-3). The waste gas system is not intended to be used for degassing the primary coolant system in post-accident conditions. Access to this area may be needed since the waste gas compressor are controlled from this panel. This task will require two persons to occupy this area for one minute.

Sources of exposure in this area are the containment building, and a 2 inch CVCS line located approximately 10 feet away and above the 2 foot thick concrete shield slab at elevation 426’-0". Immediately after the accident, the dose rate from these sources in area "H" is 3.74 rem/hr. The calculated dose for the one minute occupancy time is 0.06 rem.

The access route to area "H" is route "5" described in Table 12A.4-3. This route will add a total of 0.98 rem to the worker dose in this area for access immediately after the accident.

12A.4.9 AREA I: CHARGING/SAFETY INJECTION PUMP COOLING UNITS

Area "I" is the location of the charging/safety injection pump room cooling units and is identified on Figure 12A.4-4. Although not a vital access area required for the mitigation of or recovery from an accident, the access may be established so as to ensure that more than one of the charging/safety injection pumps are available for the long term cooling of the reactor core. By this action, undesirable minimum safeguards condition that would result upon failure of one of these cooling units can be avoided. Under these circumstances, plant personnel may require access to one of the areas for operations such as manual switchover, maintenance, or repair. Tasks such as these require that two persons spend five minutes at the cooling units for manual switchover, or three persons spend four hours for the maintenance or repair of a cooling unit.

The major sources of exposure in this area are the charging/safety injection pump cubicle piping, the piping area immediately west of the cooling units on elevation 400'-0", and a 3-inch chemical and volume control system (CVCS) pipe located inside the room containing cooling units 1 and 2.

The access route to Area "I" is route "10" described in Table 12A.4-3. For the manual valve switchover procedure, the access route will add a total of 1.03 rem to the worker dose at the time of the accident (i.e., 0 hr.). For the maintenance and repair activities after the accident, the access route will contribute 0.02 rem one day after the accident to the worker dose, and becomes negligible five days after the accident. 99-01

The resultant dose levels in this area have been reduced to the acceptable values shown in Table 12A.4-2 by the addition of concrete and lead shielding around the charging/safety injection and CVCS piping located within the charging/safety injection pump cooling unit cubicle. The new configuration allows personnel access in accordance with the occupancy requirements of Table 12A.4-2 and the occupancy guidelines for vital access areas requiring infrequent occupancy as contained in NUREG-0737.

12A.4.10 AREA J: ENGINEERED SAFETY FEATURE SUBSTATION BUS AND MOTOR CONTROL CENTER

Area "J" is the location of the engineered safety feature 480 volt unit substation bus 1DB1 and motor control center, 1DB2Y (see Figure 12A.4-6). Occupancy in this area following an accident will be the same as explained for area "G".

Dose contributors to this area are the volume control tank (cubicle 63-02), and valve gallery located in cubicle 63-03.

Access to this area could be required immediately after the accident. With the assumption that zero hour entry occurs, the exposure rate in this area will result in calculated exposures of 3.85 and 9.62 rem to an individual for the occupancy periods of 2 and 5 minutes, respectively. However, the resultant dose levels in this area are reduced to acceptable levels quickly as shown in Table 12A.4-2, and the resultant integrated doses meet the dose requirements of 10CFR50, Appendix A, General Design Criteria 19 as outlined in NUREG-0737. 99-01

The access route to area "J" is route "14" described in Table 12A.4-3. This access route will add 2.80 rem to a worker's dose immediately after the accident. RN 05-011

12A.4.11 AREA K: HYDROGEN RECOMBINER CONTROL AND POWER SUPPLY PANELS

Area "K" as identified on Figure 12A.4-6 is the general location of the hydrogen recombiner control and power supply panels. Post-accident hydrogen formation in the Reactor Building may require operation of the hydrogen recombiner system. The system is started and controlled by operator action at the recombiner control panel. This task requires that two workers spend 10 minutes in this area per hour over a 3 hour period.

Dose contributors in this area are the personnel access hatch and a 3 inch chemical and volume control system pipe located below at elevation 450'6". Access to the control panels could be required shortly after an accident. The exposure rate 2 hours after the accident in this area is 16 rem/hr. The calculated dose to an individual in this area for 10 minutes of occupancy is 2.67 rem. If occupancy is required at one day after the accident, the exposure rate is calculated to be 1.14 rem/hr which results in a 10 minute dose of 0.19 rem. 99-01

The access route to area "K" is route "13" described in Table 12A.4-3. The calculated dose to an individual traveling this route is 0.92 rem 2 hours after the accident and 0.03 rem 1 day after the accident. RN 05-011

12A.4.12 AREA L: HYDROGEN RECOMBINER CONTROL AND POWER SUPPLY PANELS

Area "L", as identified on Figure 12A.4-6, is the general location of the hydrogen recombiner control and power supply panels. Occupancy in this area following an accident will be the same as explained for area "K".

Dose contributors in this area are the Reactor Building located approximately 40 feet away and a 10 inch spray line located below the 2 ft. concrete floor at elevation 436'-0". Access to the control panels could be required shortly after an accident. The exposure rate 2 hours after the accident in this area is 0.74 rem/hr. The calculated dose to an individual in this area for 10 minutes of occupancy is 0.12 rem. If occupancy is required at one day after the accident, the exposure rate is calculated to be 16.0 mrem/hr which results in a 10 minutes dose of 0.003 rem. 99-01

The access route to area "L" is route "11" described in Table 12A.4-3. This access route will add a total of 0.52 rem to the worker dose 2 hours after the accident and 0.02 rem 1 day after the accident. 99-01

12A.4.13-1 AREA M1: REACTOR BUILDING ATMOSPHERE SAMPLE COLLECTION POINT AND HYDROGEN GAS ANALYZER

Area "M1" as identified in Figure 12A.4-6 is the location where a reactor building air sample will be collected following an accident. It is also the location of the reactor building atmosphere hydrogen analyzer. The hydrogen analyzers are designed to continuously monitor the reactor building atmosphere following an accident. Access to the analyzer is not required since they are redundant (the other analyzer is located at elevation 463'-0" of the Auxiliary Building in cubicle 63-17) and all necessary functions are controlled remotely from the Control Room. However, access to this area is required to collect a reactor building air sample. The time required to collect the air sample has been estimated to be 10 minutes.

Dose contributors to this area are the Reactor Building located approximately 10 ft away, 3/8" gas sample lines for the sample cylinder and the hydrogen analyzer, and the gas sample cylinder itself. Access to this area could be required shortly after the accident. Access at 30 minutes is assumed. The exposure rate in this area 30 minutes after the accident is 12.9 rem/hr. This results in a calculated dose of 2.15 rem for 10 minutes of occupancy.

The access route to area "M" is route "12" described in Table 12A.4-3. This access route will add a total of 1.14 rem to the worker dose for access to and from this area 30 minutes after the accident.

12A.4.13-2 AREA M2: REACTOR BUILDING ATMOSPHERE SAMPLE COLLECTION POINT AND HYDROGEN GAS ANALYZER

Area "M2" as identified in Figure 12A.4-6 is the location where a reactor building air sample will be collected following an accident. It is also the location of the reactor building atmosphere hydrogen analyzer. The hydrogen analyzers are designed to continuously monitor the reactor building atmosphere following an accident. Access to the analyzer is not required since they are redundant (the other analyzer is located at elevation 463'-0" of the Fuel Handling Building) and all necessary functions are controlled remotely from the Control Room. However, access to this area is required to collect a reactor building air sample. The time required to collect the air sample has been estimated to be 10 minutes.

Dose contributors to this area are the Reactor Building located approximately 30 ft. away, 3/8-inch gas sample lines for the sample cylinder and the hydrogen analyzer, and the gas sample itself. Access to this area could be required shortly after the accident. Access at 30 minutes is assumed. The exposure rate in this area 30 minutes after the accident is 13.0 rem/hr. This results in a calculated dose of 2.17 rem for 10 minutes of occupancy.

The access route to area "M2" is route "13" described in Table 12A.4-3. This access route will add a total of 1.96 rem to the worker dose for access to and from this area 30 minutes after the accident. RN 05-011

12A.4.14 AREA N: TECHNICAL SUPPORT CENTER

Area "N" as identified in Figure 12A.4-7 is the location of the technical support center. The technical support center is located adjacent to the control room on elevation 463'-0" of the Control Building. Continuous occupancy is required in this area following an accident. 02-01

A complete analysis of post-accident radiation doses in the control room and technical support center is presented in Section 15.4.1. 02-01

12A.4.15 AREA O: CONTROL ROOM

Area "O" as identified in Figure 12A.4-7 is the location of the control room. Continuous occupancy is required in this area following an accident. 02-01

A complete analysis of post-accident radiation doses in the control room is presented in Section 15.4.1. 02-01

12A.4.16 AREA P: SAMPLE ROOM

Area "P" as identified in Figure 12A.4-7 is the location of the sample room. Access to this area is required to place a primary system liquid sample on recirculation using the nuclear sampling system following an accident for assessing the extent of fuel damage using the measured recirculation radiation dose rate. The sample shall consist of fluid taken from the primary coolant hot leg, the pressurizer liquid, or residual heat removal loop. RN 05-025

The nuclear sampling system provides the means for the collection of post-accident samples in the sample room. The system has been designed to reduce the activity concentration of the samples to a level that will result in dose rates within the range of values normally associated with the 1% failed fuel design criteria.

In addition to the use of minimum activity concentrations, the resultant dose levels in this area have been reduced by the addition of 4-inch lead shielding between the sample rack in the chase area and the sink area of the sample room. RN 05-025

The access route to area "P" is route "15" described in Table 12A.4-3.

12A.4.17 AREA Q: RADIOCHEMISTRY LAB

Area "Q" as identified in Figure 12A.4-7 is the location of the radiochemistry laboratory. Access to this area provides for the radiological and chemical analyses of the post-accident samples following their transfer to this location. RN 05-025

Since the existing radiation shield design and sample analysis procedures in this area were designed for 1% failed fuel sources, the nuclear sampling system has been designed to reduce the activity concentration of the samples to a level that will result in dose rates in the radiochemistry lab within the range of values normally associated with the 1% failed fuel design criteria (< 2.5 mrem/hr). RN 05-025 RN 05-025

The access route to area "Q" is route "15" as described in Table 12A.4-3.

12A.4.18 AREA R: COUNT ROOM

Area "R" as identified in Figure 12A.4-7 is the location of the count room. Access to this area provides for the radiological analysis of the post-accident samples prepared in the radiochemistry laboratory. RN 05-025

The nuclear sampling system design criteria to provide post-accident sources with a radioisotope activity level comparable to that of the 1% failed fuel design criteria results in insignificant increases in personnel exposure above those received in the count room under normal operating conditions (<1 mrem/hr). The need for additional shielding has been reduced to make provisions for the availability of temporary personnel. RN 05-025 RN 05-025

The access route to area "R" is route "15" as described in Table 12A.4-3.

12A.4.19 INTERMEDIATE BUILDING

The Intermediate Building was reviewed to identify post-accident radiation sources, access requirements, and the location of safety related equipment. It was found that the only area of the Intermediate Building where exposure to post-accident radiation sources is possible is at the 412'-0" elevation adjacent to the east and west penetration access areas as shown in Figure 12A.4-8. At this elevation, residual heat removal (RHR), safety injection (SI), and chemical and volume control system (CVCS) piping containing post-accident sources is located in the east and west penetration access areas. The areas adjacent to the penetration access areas contain component cooling pumps, motor driven emergency feedwater pumps, and miscellaneous safety related valves and instrumentation which may require access. All other areas of the Intermediate Building are not exposed to post-accident sources and receive a negligible increase in dose.

The Intermediate Building at elevation 412'-0" is shielded from the radioactive piping located in the penetration access areas by a 2 foot concrete wall. This shield wall reduces the dose rates in this general area of the Intermediate Building to a level which permits limited access approximately 15 minutes immediately after the accident, 3 hours of access 8 hours following the accident, and 20 hours of access 1 day after the accident.

In the area between the east and west penetration access areas, several pipes containing post-accident sources were found to be located above the sides of a pipe chase located at elevation 422'-3". This allowed direct streaming of a 12-inch RHR pipe to the Intermediate Building floor elevation 412'-0". The resultant dose levels in this area of the Intermediate Building have been reduced to acceptable values by the modification of the pipe chase as follows: (1) extension of the pipe chase side above the 12" RHR line, (2) addition of 6" concrete shielding to the existing 1 foot concrete shielding, (3) relocation of access ladder to prevent streaming through concrete cutout. The new configuration allows personnel access in accordance with the occupancy requirements of Table 12A.4-2, and the resultant integrated doses meet the dose requirements of 10CFR50, Appendix A, General Design Criteria 19 as outlined in NUREG-0737.

12A.4.20 EMERGENCY OPERATIONS FACILITY

The EOF is located beyond the 10-mile EPZ. Therefore there are no regulatory shielding requirements or ventilation isolation/HEPA filter requirements for the facility. The EOF is required to be “well engineered for design life of the plant” per NUREG-0696. The EOF is designed to the standards of the International Building Code, and as such resists the applicable loadings (wind, snow, earthquake, etc.) defined in ASCE 7, Minimum Design Loads for Buildings and Other Structures. RN 10-029

TABLE 12A.4-1
EXPOSURE RATES FOR VITAL AREAS AS A FUNCTION OF TIME
EXPOSURE RATE (mr/hr)

	Vital Area	TIME AFTER ACCIDENT 0	30 minutes	1 hour	8 hours	1 day	5 days	10 days	30 days	90 days	6 months	
A	RHR/Spray Pump Rooms	(1)	2.15 + 4	1.70 + 4	2.79 + 3	3.83 + 2	8.31 + 1	6.19 + 1	2.30 + 1	3.84 + 0	1.94 + 0	RN 05-011
	RHR/Spray Pump Room Cooling Units	1.46 + 4	1.67 + 4	1.33 + 4	2.33 + 3	6.14 + 2	7.82 + 1	5.80 + 1	2.26 + 1	4.89 + 0	2.46 + 0	
B	Charging/SI Pump Rooms	5.57 + 4	4.46 + 4	3.46 + 4	6.33 + 3	1.54 + 3	1.92 + 2	1.44 + 2	5.56 + 1	1.27 + 1	5.95 + 0	
C	Auxiliaries Local Control Panel											
	Source Shielded	(1)	3.10 + 2	2.47 + 2	4.57 + 1	7.50 + 0	1.62 + 0	1.20 + 0	4.85 - 1	1.24 - 1	6.30 - 2	RN 05-011
	Source Unshielded	(1)	8.26 + 4	6.70 + 4	1.83 + 4	7.35 + 3	2.30 + 3	1.50 + 3	5.19 + 2	1.80 + 2	9.11 + 1	
D	Spent Fuel Pit Heat Exchanger	(1)	(1)	(1)	(1)	(1)	(1)	(1)	(1)	(1)	(1)	
E	Waste Gas System Control Panels	Access Not Required										
F	Spent Fuel Pit Cooling Pumps and Local Control Panels	5.63 + 3	1.05 + 4	6.42 + 3	1.16 + 3	1.15 + 2	2.65 + 1	1.97 + 1	7.62 + 0	2.68 + 0	7.91 - 1	02-01
G	Engineered Safety Feature Motor Control Center	1.46 + 4	2.41 + 4	1.92 + 4	3.37 + 3	7.76 + 2	1.13 + 2	8.36 + 1	3.27 + 1	7.19 + 0	3.63 + 0	RN 05-011
H	Waste Processing System Control Panel	3.74 + 3	2.30 + 3	1.37 + 3	1.89 + 2	1.88 + 1	4.27 + 0	3.20 + 0	1.20 + 0	3.10 - 1	1.02 - 1	

TABLE 12A.4-1 (Continued)
EXPOSURE RATES FOR VITAL AREAS AS A FUNCTION OF TIME
EXPOSURE RATE (mr/hr)

		TIME AFTER ACCIDENT										
	Vital Area	0	30 minutes	1 hour	8 hours	1 day	5 days	10 days	30 days	90 days	6 months	
I	Charging/SI Pump Room Cooling Units											
	Cooling Unit 1	3.19 + 5	2.48 + 5	1.87 + 5	3.16 + 4	7.62 + 3	1.61 + 3	1.15 + 3	4.81 + 2	1.61 + 2	8.05 + 1	RN 05-011
	Cooling Unit 2	1.06 + 5	8.34 + 4	6.31 + 4	1.16 + 4	2.57 + 3	4.52 + 2	3.27 + 2	1.32 + 2	4.04 + 1	1.94 + 1	
	Cooling Unit 3	8.07 + 4	6.32 + 4	4.96 + 4	8.82 + 3	2.36 + 3	2.76 + 2	2.05 + 2	7.94 + 1	1.71 + 1	8.31 + 0	
J	Engineered Safety Feature Substation Bus and Motor Control Center	1.15 + 5	5.39 + 4	4.05 + 4	3.91 + 3	1.01 + 2	4.39 + 0	3.12 + 0	1.14 + 0	1.93 - 1	(1)	99-01
K	Hydrogen Recombiner Control and Power Supply Panels	6.32 + 4	3.66 + 4	2.49 + 4	4.10 + 3	1.14 + 3	2.50 + 2	1.73 + 2	7.31 + 1	2.82 + 1	1.29 + 1	
L	Hydrogen Recombiner Control and Power Supply Panels	5.63 + 3	2.85 + 3	1.57 + 3	1.12 + 2	1.67 + 1	3.71 + 0	2.77 + 0	9.77 - 1	1.59 - 1	(1)	RN 05-011
M1	Reactor Building Air Sample Collection Point "B"	2.43 + 4	1.29 + 4	7.35 + 3	6.80 + 2	1.60 + 2	3.67 + 1	2.36 + 1	5.77 + 0	5.70 - 1	2.93 - 1	
M2	Reaction Building Air Sample Collection Point "A"	1.90 + 4	1.30 + 4	9.65 + 3	6.59 + 3	1.12 + 3	3.79 + 2	2.37 + 2	6.70 + 1	2.25 + 1	1.20 + 1	

TABLE 12A.4-1 (Continued)
EXPOSURE RATES FOR VITAL AREAS AS A FUNCTION OF TIME
EXPOSURE RATE (mr/hr)

		TIME AFTER ACCIDENT									
	Vital Area	0	30 minutes	1 hour	8 hours	1 day	5 days	10 days	30 days	90 days	6 months
N	Technical Support Center	Refer to Section 15.4.1									
O	Control Room	Refer to Section 15.4.1									
P	Sample Room										
	At Sample Sink	1.70 + 2	1.10 + 2	8.30 + 1	1.10 + 1	1.30 + 0	3.40 - 1	2.20 - 1	7.70 - 2	(1)	(1)
	At Control Panel	2.30 + 1	1.50 + 1	1.10 + 1	1.50 + 0	1.70 - 1	4.60 - 2	(1)	(1)	(1)	(1)
Q	Radiochemistry Lab	(3)									
R	Count Room	(3)									
	Intermediate Building	4.45 + 4	3.28 + 4	2.54 + 4	4.58 + 3	5.23 + 2	1.73 + 2	1.27 + 2	5.95 + 1	1.40 + 1	7.03 + 0

02-01

NOTES:

(1) Negligible

(2) Not Calculated

(3) Increase of Personnel Total Dose Insignificant From Current Considerations.

TABLE 12A.4-2

VITAL AREA RADIATION DOSE SUMMARY

	Vital Area	Occupancy Requirements	Time After Accident	Occupancy Dose (Rem/Person)	Access Dose To and From Area (Rem/Person)	Total Dose (Rem/Person)	Total Dose Man-Rem	
A	RHR/Spray Pump Rooms	3 persons for 4 hours	1 day	1.41	0.024	1.43	4.29	RN 05-011
A	RHR/Spray Pump Room Cooling Units	3 persons for 4 hours	1 day	2.25	0.024	2.27	6.81	
B	Charging/SI Pump Rooms	3 persons for 4 hours	2 days	2.39	0.05	2.44	7.32	99-01
C	Charging/SI Pump C Auxiliaries Local Control Panel	2 persons for 2 minutes	1/2 hour	2.75	1.40	4.15	8.30	RN 05-011
			1 hour	2.23	1.04	3.27	6.54	
D	Spent Fuel Pit Heat Exchanger	Long term periodic surveillance	1 hour	(1)	0.72	0.72	1.44	
			8 hours	(1)	0.14	0.14	0.28	
			24 hours	(1)	0.02	0.02	0.04	99-01
E	Waste Gas System Control Panels	Not Required						

TABLE 12A.4-2 (Continued)

VITAL AREA RADIATION DOSE SUMMARY

	Vital Area	Occupancy Requirements	Time After Accident	Occupancy Dose (Rem/Person)	Access Dose To and From Area (Rem/Person)	Total Dose (Rem/Person)	Total Dose Man-Rem	
F	Spent Fuel Pit Cooling Pumps and Local Control Panels	3 persons for 4 hours	8 hours	3.03	0.11	3.14	9.42	99-01
			12 1/2 hours	1.36	0.06	1.42	4.26	
			1 day	0.42	0.02	0.44	1.32	
G	Engineered Safety Feature Motor Control Center	2 persons for 2-5 minutes	0	0.49 - 1.22	0.20	0.69 - 1.42	1.38 - 2.84	RN 05-011
			1/2 hour	0.80 - 2.01	0.28	1.08 - 2.28	2.16 - 4.57	
			1 hour	0.64 - 1.60	0.22	0.86 - 1.82	1.72 - 3.64	
			8 hours	0.11 - 0.28	0.04	0.15 - 0.32	0.30 - 0.64	
			1 day	0.03 - 0.06	0.01	0.03 - 0.07	0.07 - 0.14	
H	Waste Processing System Control Panel	2 persons for 1 minute	0	0.06	0.98	1.04	2.08	99-01
			1/2 hour	0.04	0.86	0.90	1.80	
			1 hour	0.02	0.66	0.68	1.36	
			8 hours	0.003	0.12	0.12	0.24	
I	Charging/SI Pump Room Cooling Units							
	Cooling Unit I (Manual Switch Over)	2 persons for 5 minutes	8 hours	2.62	0.14	2.77	5.54	RN 05-011
			1 day	0.63	0.02	0.66	1.32	
	Cooling Unit 2 (Manual Switch Over)	2 persons for 5 minutes	8 hours	0.97	0.14	1.10	2.20	99-01
			1 day	0.21	0.02	0.23	0.46	

TABLE 12A.4-2 (Continued)

VITAL AREA RADIATION DOSE SUMMARY

Vital Area	Occupancy Requirements	Time After Accident	Occupancy Dose (Rem/Person)	Access Dose To and From Area (Rem/Person)	Total Dose (Rem/Person)	Total Dose Man-Rem	
Cooling Unit 3	2 persons for 5	1 hour	4.13	0.72	4.85	9.70	
(Manual Switch Over)	minutes	8 hours	0.73	0.14	0.87	1.74	99-01
		1 day	0.20	0.02	0.22	0.44	
							RN
Cooling Unit 1	3 persons for 4	10 days	4.59	(1)	4.59	13.77	05-011
(Maintenance or Repair)	hours	30 days	1.92	(1)	1.92	5.76	
		90 days	0.64	(1)	0.65	1.95	
Cooling Unit 2	3 persons for 4	5 days	1.80	(1)	1.80	5.40	
(Maintenance or Repair)	hours	10 days	1.30	(1)	1.30	3.90	
		30 days	0.53	(1)	0.53	1.59	
		90 days	0.16	(1)	0.17	0.51	99-01
Cooling Unit 3	3 persons for 4	5 days	1.10	(1)	1.10	3.30	
(Maintenance or Repair)	hours	10 days	0.81	(1)	0.82	2.46	
		30 days	0.32	(1)	0.32	0.96	
		90 days	0.07	(1)	0.07	0.21	

TABLE 12A.4-2 (Continued)

VITAL AREA RADIATION DOSE SUMMARY

	Vital Area	Occupancy Requirements	Time After Accident	Occupancy Dose (Rem/Person)	Access Dose To and From Area (Rem/Person)	Total Dose (Rem/Person)	Total Dose Man-Rem	
J	Engineered Safety Feature Substation Bus and Motor Control Center	2 persons for 2-5 minutes	1 hour	1.35 - 3.38	1.47	2.82 - 4.85	5.64 - 9.90	
			8 hours	0.13 - 0.33	0.17	0.30 - 0.50	0.60 - 1.00	
			1 day	0.003 - 0.008	0.023	0.030 - 0.033	0.06 - 0.08	
K	Hydrogen Recombiner Control and Power Supply Panels	2 persons for 10 minutes per hour over a 3 hour period	2 hours	2.67	0.92	3.59	7.18	RN 05-011
			3 hours	1.67	0.67	2.34	4.68	
			4 hours	1.33	0.52	1.85	3.70	
			5 hours	1.02	0.43	1.44	2.88	
			24 hours	0.19	0.03	0.22	0.44	
			25 hours	0.18	0.03	0.21	0.42	
			26 hours	0.18	0.02	0.20	0.40	
			27 hours	0.16	0.02	0.18	0.36	
L	Hydrogen Recombiner Control and Power Supply Panels	2 persons for 10 minutes per hour over a 3 hour period	2 hours	0.12	0.52	0.64	1.28	99-01
			3 hours	0.07	0.38	0.45	0.90	
			4 hours	0.05	0.30	0.35	0.70	
			5 hours	0.04	0.24	0.28	0.56	
			24 hours	0.0027	0.02	0.0227	0.0454	
			25 hours	0.0026	0.0196	0.0222	0.0444	
			26 hours	0.0025	0.018	0.0205	0.041	
			27 hours	0.0023	0.017	0.0193	0.0386	

TABLE 12A.4-2 (Continued)

VITAL AREA RADIATION DOSE SUMMARY

	Vital Area	Occupancy Requirements	Time After Accident	Occupancy Dose (Rem/Person)	Access Dose To and From Area (Rem/Person)	Total Dose (Rem/Person)	Total Dose Man-Rem	
M1	Reactor Building Air Sample Collection Point "B"	2 persons for 10 minutes	1/2 hour	2.15	1.14	3.29	0.58	99-01
			1 hour	1.23	0.82	2.05	4.10	
			8 hours	0.11	0.14	0.25	0.50	
			1 day	0.03	0.02	0.05	0.10	
M2	Reactor Building Air Sample Collection Point "A"	2 persons for 10 minutes	1/2 hour	2.17	1.96	4.13	8.16	RN 05-011
			1 hour	1.60	1.47	3.07	6.14	
			8 hours	1.08	0.17	1.25	2.50	
			1 day	0.19	0.03	0.22	0.44	
N	Technical Support Center	Continuous	30 days	-	-	Refer to Section 5.4.1.		02-01
O	Control Room	Continuous	30 days	-	-	Refer to Section 5.4.1.		
P	Sample Room		0	(3)	(1)	(3)	(3)	RN 05-025
Q	Radiochemistry Lab		0	(3)	(1)	(3)	(3)	99-01
R	Count Room		0	(3)	(1)	(3)	(3)	

Notes: (1) Negligible

(2) Not Calculated

(3) Increase of Personnel Total Dose Insignificant from Current Considerations.

TABLE 12A.4-3

ACCESS ROUTES TO VITAL AREAS

Access Routes	Description
1.	From Control Building (C.B.), 463’ elev., down either stairway to 412’ elev. through 412’ elev. to north corridor. From C.B., 412’ elevation, enter Auxiliary Building (A.B.) at stairway tower 1, down stairway to 374’ elev., through hallways 74-01 and 74-09 W (formerly 74-09) to rooms 74-16 and 74-17. For cooling units, ladder up from hallway 74-09 to 385’ elev. to rooms 85-01 and 85-02.
2.	From C.B., 463’ elev., down either stairway to 412’ elev., through 412’ elev. to north corridor. From C.B., 412’ elev. enter A.B. at hallway 12-11, through hallways 12-11 and 12-09 to stairway tower 2, down stairway to 397’ elev. and through room 97-01 to stairs at column lines M and 9.5. Down stairs to 388’ elev. to room 88-13 S.
3.	Same as "2" but through room 88-13S to room 88-25.
4.	From C.B., 463’ elev., down either stairway to 412’ elev., through 412’ elev. to north corridor. From C.B., 412’ elev., enter A.B. at hallway 12-11, north through hallway 12-11 to room 12-28.
5.	Same as "4" but continues through hallway 12-11 to room 12-15.
6.	Same as "4" but north through hallway 12-11 and east through hallway 12-11 N to room 12-18.
7.	From C.B., 463’ elev., down either stairway to 412’ elev., through 412’ elev. to north corridor. From C.B., 412’ elev., enter A.B. at hallway 12-11, north through hallway 12-11 and east through hallway 12-11 N to stairway tower 3. Down stairway to 388’ elev. to room 88-13 NE.
8.	Same as "7" but through room 88-13 NE to rooms 88-23 and 88-24
9.	Same as "7" but through room 88-13 NE to room 88-13 N.
10	Same as "7" but continues through room 88-13 NE to room 88-13 N. Ladder up to 400’ elev.

TABLE 12A.4-3 (Continued)

ACCESS ROUTES TO VITAL AREAS

Access Routes	Description
11.	From C.B., 463’ elev., down either stairway to 412’ elev., through 412’ elev. to north corridor. From C.B., 412’ elev., enter A.B. at hallway 12-11, north through hallway 12-11 and east through hallway 12-11 N to stairway tower 3. Up stairway to 463’ elev. to room 63-09 at column line 7.7.
12.	Same as "11" but from room 63-09 enter Fuel Building (F.B.) 463’ elev., through room 63-01 S to column lines P.4-P.9 and 3.3-4.0.
13.	From C.B., 463’ elev., down either stairway to 412’ elev. through 412’ elev. to north corridor. From C.B., 412’ elev. enter A.B. at hallway 12-11, through hallways 12-11 and 12-09 to stairway tower 2. Up stairway to 463’ elev., through hallway 63-04 and south through hallway 63-16 to room 63-19.
14.	Same as "13" but through room 63-09 to room 63-01.
15.	From C.B., 463’ elev., down either stairway to 412’ elev. to rooms 12-09, 12-10 and 12-11.

12A.5 REVIEW OF RADIATION QUALIFICATION OF SAFETY RELATED EQUIPMENT

Radiation qualification of safety-related equipment is provided in the applicable equipment qualification documentation packages. Refer to Section 3.11 for further discussion.

12A.6 REFERENCES

1. E. D. Arnold and B. F. Maskervitz, "SDC - A Shield Design Calculation Code for Fuel Handling Facilities," ORNL-3041, March 1966.

2. C. M. Lederer, J. M. Hollander, and J. Perlman, "Table of Isotopes," Sixth Edition, John Wiley and Sons, Inc., 1967.

3. A. Tobias, "Data for the Calculation of Gamma Radiation Spectra and Beta Heating from Fission Products (Revision 3), RD/B/M2669, June 1973.

4. Regulatory Guide 1.4, Revision 2 (6/74).

5. Regulatory Guide 1.7, Revision 2 (11/78).

6. Source Term Data for Westinghouse Pressurized Water Reactors, WCAP-8253 (May 1974).

7. Grove Engineering, "MicroShield", Version 5, 1998. 99-01