Millstone Power Station Unit 3
Safety Analysis Report

Chapter 12

CHAPTER 12 - RADIATION PROTECTION

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12.0 INTRODUCTION

Prior to the licensing and operation of a nuclear power reactor, the applicant must include, in Chapter 12 of the FSAR, an estimate of the radiation dose expected to be received by station personnel. This includes an estimate for both whole body dose from direct radiation and internal dose from airborne activity. This is provided to ensure the proposed station design related to occupational radiation exposure control (e.g., shielding and airborne activity control) will be sufficient to ensure compliance with 10 CFR 20. The assessments presented in Chapter 12 are based on nominal assumptions and generic models and criteria that were appropriate at the time the original FSAR was written. They represent estimates chosen for the purpose of projecting occupational dose consequences. They do not represent design or operational requirements. It was fully expected that actual operational data would not match the chosen assumptions and criteria presented in Chapter 12, but in general, the estimates were expected to be conservative.

Once the plant is operational, compliance with occupational exposure limits and controls is ensured and controlled by compliance with the Technical Specifications and 10 CFR 20. These documents require a Radiation Protection Program and an ALARA Program. These are dynamic programs that change to meet changing regulations, industry initiatives and state-of-the-art practices. These programs provide detailed controls on occupational exposure. The Radiation Protection Program ensures that the requirements of 10 CFR 20 are met. The ALARA Program ensures that controls are imposed and assessments are performed to reduce occupational exposure to levels that meet current standards and are not the original estimates of Chapter 12. These programs are routinely audited by licensee and NRC staff for compliance and effectiveness. Annual occupational exposure reports are provided to the NRC to provide a real time measure of the effectiveness of occupational exposure controls.

Therefore, compliance with occupational exposure regulations is controlled by the Radiation Protection and ALARA programs, which are described in Chapter 12 of the FSAR. Current program measures are contained in Chapter 12, however, the design parameters or quantities provided here are not updated from original values. The original bases for station design and radiation control are relegated to historical perspective, as these bases were never intended to describe conditions of operation. More accurate information on radiological quantities or conditions should be determined by referencing current radiation protection data available in the radiation protection department.

12.1 ENSURING THAT OCCUPATIONAL RADIATION EXPOSURES ARE AS LOW AS IS REASONABLY ACHIEVABLE (ALARA)

It is Millstone policy to implement a program that meets the intent of 10 CFR 20 and ensure that occupational radiation exposures at its nuclear facilities are kept “as low as reasonably achievable” (ALARA).

The ALARA Program criteria shall be in accordance with 10 CFR 20, Regulatory Guide 8.8, Rev. 4 and Regulatory Guide 8.10 (Rev. 1-R); and, it should meet the intent of INPO 91-014, Rev 1.

The program shall ensure that:

Annual and lifetime doses to individuals are ALARA. External and internal exposure is optimized by keeping TEDE ALARA.

Annual collective doses, (person rem) are ALARA.

Individual doses within work groups are balanced to be consistent with:

a. experience

b. manpower availability

c. existing agreements

Annual and three year goals are developed for collective doses at each unit.

Outage goals are developed thirty to sixty days prior to the start of an outage.

An ALARA job review process exists for jobs with the potential for significant exposure.

ALARA economic evaluations are performed in support of backfits, modifications, decommissioning, etc.

Personnel are aware of ALARA program philosophy and trained in ALARA concepts.

Millstone management provides the necessary policy, resources and commitment for ALARA program.

An ALARA feedback system exists for workers to identify ALARA concerns or suggest ALARA improvements.

Corrective actions are considered when the attainment of specific ALARA goals are jeopardized.

12.2 RADIATION SOURCES

12.2.1 CONTAINED SOURCES

The radioactivity values provided in this section are the design basis values used for the design of plant shielding. As such they are considered historical and not subject to future updating. This information is retained to avoid loss of original licensing bases. As discussed in Chapter 12.0, compliance with occupational exposure limits and controls is ensured and controlled by compliance with the Technical Specifications and 10 CFR 20 which was implemented at MPS-3 via the Radiation Protection Program and the ALARA Program.

The source of radioactivity contained in the streams of the various radioactive waste management systems are the nuclides generated in the reactor core and activation of nuclides in the reactor coolant system and the air surrounding the reactor vessel. These sources are described in Chapter 11. Table 12.2-1 presents the principal parameters which are used to establish design radiation source inventories. The design basis for the shielding source terms for fission products in this section is cladding defects in fuel rods producing 1 percent of the core thermal power. The design basis for activation and corrosion product activities are derived from measurements at operating plants and are independent of fuel defect level. The radionuclide activity levels in the reactor coolant at the design basis level are given in Section 11.1. The models and assumptions used in determining these sources are also given in Section 11.1.

The reactor core source description is similar in that given in Topical Report RP-8A (Stone & Webster Engineering Corporation), Section 4.1.1, with appropriate adjustment to account for power level difference.

The activity of a spent fuel assembly is calculated using appropriate fission yields, decay constants, and thermal neutron cross sections. Isotopic inventories are based on full power operation for 650 days. The inventory of an average fuel assembly at shutdown and 100 hours after shutdown is given in Table 12.2-6. The source strengths in MeV/sec for the most radioactive fuel assembly at several decay times (assuming a radial peaking factor of 1.65) is given in Table 12.2-7. The isotopic activities (expected and design) in the fuel pool water are given in Table 12.2-8 based on expected and design primary coolant activities homogeneously mixed with refueling cavity water and spent fuel pool water from refueling operations 100 hours after reactor shutdown.

The location and geometry of significant sources of radiation in the containment building, auxiliary building, fuel building, waste disposal building, condensate polishing building, ESF building, and the tank yard are presented in Tables 12.2-2 through 12.2-5A and on Figures 12.2-1 through 12.2-3. The method used to arrive at the source terms presented in Tables 12.2-2 through 12.2-5A considers the operating parameters described in Table 11.1-3. The evaporator bottoms activity is based on liquid particulate concentrations at the start of plant operations, thereby minimizing the decay time to produce a batch of concentrate. Normal operating flow rates are also used to develop demineralizer nuclide concentrations consistent with the decontamination factors used to determine radioactive concentrations of waste liquid streams. The inventory of radioactive nuclides on filters downstream of the demineralizer assumes a fraction of the resin fines are transported via the liquid flow and deposited on the filter cartridge. Significant sources are listed by name in the tables and are numbered to correspond to numbered locations on the figures. Radiation source terms are also presented in these tables in terms of seven discrete energy levels (MeV) used in plant shielding design.

The eighth energy level corresponds to the N-16 activity in the reactor coolant for various components, listed in Table 12.2-2. N-16 activity is the controlling source in the design of the secondary shield and is tabulated in Table 12.2-9, in Ci per gram of coolant, as a function of transport time through the reactor coolant loop.

The method used to calculate the source strength of N-16 in each component is described in detail in the Stone & Webster Topical Report RP-8A (Stone & Webster Engineering Corporation). The required parameters for this calculation are transit time to the component, transit time through the component. N-16 is not a factor in the radiation sources for the systems and components located outside the containment due to its short (7.11 seconds) half life and the greater than one minute transport time before the letdown flow exits the containment.

12.2.1.1 Sources for Design Basis Loss-of-Coolant Accident

The radiation sources of importance for the design basis accident are the sources within the containment and sources transported via the emergency safeguards features (ESF) cooling system.

The fission product radiation sources considered to be released from the fuel to the containment following a maximum credible accident are based on the assumptions given in Regulatory Guide 1.4 and NUREG-0737. This source term was used to evaluate the original plant shielding design, and is not used in Chapter 15 analyses.

The sources in the ESF system are based on the nongaseous activity, i.e., 50 percent halogens and 1 percent remainder, being retained in the coolant water. Noble gases formed by the decay of halogens in the sump water are assumed to be released to the containment and not retained in the water. Credit has been taken for dilution by the reactor coolant system volume plus the contents of the refueling water storage tank and other ESF system component volumes.

Isotopic fission product sources in the Fuel are given in Section 11.1.

12.2.2 AIRBORNE RADIOACTIVE SOURCES

The principal sources of airborne radionuclides are the reactor coolant system and the air surrounding the reactor vessel. Reactor coolant leaking into plant buildings results in the release of airborne contamination. The radioactivity sources which contribute to the radioactive airborne releases from the plant waste management system and the plant ventilation system are described in Chapter 11.

Concentrations of airborne activity for the expected and design conditions in the containment structure, turbine building, and fuel building are listed in Table 12.2-11. The bases used to derive these concentrations are presented in Table 12.2-10 and in NUREG-0017. Airborne radioactivity concentrations in aisleways and manned spaces in the auxiliary building and at other locations is considered negligible.

Airborne levels in general access areas of the auxiliary, turbine, and fuel buildings are expected to be lower than equipment cubicles of these buildings, since ventilation flow paths are normally directed from areas with less potential contamination to areas with greater potential for contamination.

Containment Structure

The containment structure is not normally occupied during power operation. Radiation protection procedures control access. Two recirculating charcoal filters can be operated to ensure that airborne iodine in the containment is as low as is reasonably achievable for work in that area (Section 9.4.7).

Figure 12.2-4 presents expected iodine-131 concentrations in the containment structure after the two charcoal filters have been placed in operation.

After shutdown, the containment purge air system can be used to reduce the airborne activity within the containment structure. The filtered purge is rated at 30,000 cfm.

Radioactivity associated with primary coolant leakage is mixed in the containment atmosphere by the containment atmosphere recirculation system (Section 9.4.7.1). Removal occurs through radioactive decay. At appropriate times, the radioactive inventory may be reduced by means of recirculating the containment air through the charcoal filters or containment purging.

The containment atmosphere tritium assumes the same relative concentration (Ci of tritium per gram of water) as exists in the reactor coolant leakage.

Turbine Building

Radioactivity associated with steam leakage is assumed to be uniformly distributed by the turbine building ventilation system (Section 9.4.4).

Removal occurs through decay and ventilation exhaust. The tritium concentration in the turbine building is calculated assuming that all the steam leakage into the turbine building remains gaseous.

Auxiliary Building

Airborne radioactivity associated with primary coolant leakage is assumed to be limited to process equipment cubicles and is removed by auxiliary building ventilation system (Section 9.4.3). Removal occurs through decay and ventilation exhaust. The ventilation system is configured to preclude mixing of the atmosphere in process equipment cubicles with the atmosphere in the general access areas as described above.

The tritium concentration in the auxiliary building atmosphere is conservatively calculated assuming all the primary coolant leakage into the auxiliary building evaporates when in fact the leakage is collected in sumps and drains and is not generally available for evaporation.

Fuel Building

The fuel building ventilation system is described in Section 9.4.2.

The tritium concentrations in the fuel building atmosphere assumes that the atmosphere above spent fuel pool has the same relative tritium concentration, Ci of tritium per gram of water, as the fuel water. Airborne concentrations are presented in Table 12.2-11.

12.2.3 REFERENCES FOR SECTION 12.2

12.2-1 NUREG-0017 United States Nuclear Regulatory Commission. Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Pressurized Water Reactors (PWR-GALE CODE). 1976 Office of Standards Development.

12.2-2 Regulatory Guide 1.4.

12.2-3 Stone & Webster Engineering Corporation (SWEC) 1975. Radiation Shielding Design and Analysis Approach for Light Water Reactor Power Plants RP 8A. Topical Report. Cambridge, Mass.

12.2-4 Westinghouse Letter NEU-3492, Dated July 31, 1980.

TABLE 12.2–1 PARAMETERS USED IN CALCULATION OF DESIGN RADIATION SOURCE INVENTORIES (HISTORICAL)

Parameter	
1. Power level (MWt)	3,636
2. Failed Fuel Fraction	0.01
3. Primary-to-Secondary Leak Rate (lb/day)	1,370
4. Reactor Operating Time (days)	650
5. Escape Rate Coefficients (sec^{-1}):	
1. Noble Gases	6.5×10^{-8}
2. Br, Rb, I, and Cs nuclides	1.3×10^{-8}
3. Te nuclides	1.0×10^{-9}
4. Mo nuclides	2.0×10^{-9}
5. Sr and Ba nuclides	1.0×10^{-11}
6. Y, La, Ce, Pr nuclides	1.6×10^{-12}
6. Purification Letdown Flow Rate (gpm)	75
7. Degasification Charcoal Delay Bed Holdup Time (days):	
1. Kr	6.1
2. Xe	142.1

Historical, not subject to future updating. This table has been retained to preserve original design basis.

TABLE 12.2–2 RADIOACTIVE SOURCES IN CONTAINMENT BUILDING (HISTORICAL)

		Activity (MeV/cc-Sec) for Energy (MeV)										
Source No.	**Source**	**0.40**	**0.80**	**1.30**	**1.70**	**2.20**	**2.50**	**3.50**	**6.10**	**Height (cm)**	**Outside Diameter (cm)**	**Volume (cc)**
1	Reactor Core	1.05+12 (1)	4.63+12	4.68+12	3.83+12	7.98+11	1.27+12	3.29+12	2.23+07	1303	439	1.97+08
2	Regenerative Heat Exchanger											
	Shell	1.73+05	2.91+05	1.26+05	1.22+05	9.74+04	1.36+05	3.03+04	2.01+07	427.0	30.48	3.12+05
	Tube	7.23+03	6.11+04	2.67+04	3.99+04	1.92+03	1.20+04	9.42+03	-			
3	Steam Generator											
	Shell	1.44+05	2.43+05	1.05+05	1.02+05	8.14+04	1.14+05	2.53+04	1.94+07		335	2.11+08
	Tube	2.97+01	1.38+02	3.19+01	1.62+01	1.79+00	9.84+01	1.21+00	-	1970	427	
4	Pressurizer Liquid	2.83+04	1.62+05	8.61+04	6.30+04	1.13+04	1.01+04	1.85+04	4.93+05	1499	213	3.06+07
	Gas	8.79+04	6.22+04	4.92+03	3.52+04	9.16+04	1.30+05	4.20+03	-			2.04+07
5	Containment Drains Transfer Tank	1.98+05	3.33+05	1.43+05	1.40+05	1.11+05	1.55+05	3.46+04	-	259	145.0	3.75+06
6	Reactor Coolant Piping	1.34+05	2.26+05	9.74+04	9.49+04	7.57+04	1.06+05	2.35+04	(See Table 12.2-9)	-	-	-
9	Excess Letdown Heat Exchanger	1.81+05	3.05+05	1.31+05	1.28+05	1.02+05	1.43+05	3.17+04	1.32+07	366	30.48	2.67+05
10	Pressurizer Relief Tank	3.87+04	2.74+04	2.16+03	1.55+04	4.03+04	5.72+04	1.85+03	-	826	290	5.10+07

TABLE 12.2–2 RADIOACTIVE SOURCES IN CONTAINMENT BUILDING (HISTORICAL)

		Activity (MeV/cc-Sec) for Energy (MeV)										
Source No.	**Source**	**0.40**	**0.80**	**1.30**	**1.70**	**2.20**	**2.50**	**3.50**	**6.10**	**Height (cm)**	**Outside Diameter (cm)**	**Volume (cc)**
11	Pressurizer Relief Tank Transfer Pump	3.87+04	2.74+04	2.16+03	1.55+04	4.03+04	5.72+04	1.85+03	-	29.2	38.1	3.02+04
12	Containment Drains Transfer Pump	1.98+05	3.33+05	1.43+05	1.40+05	1.11+05	1.55+05	3.46+04	-	29.2	38.1	3.02+04
13	Reactor Coolant Pump	1.51+05	2.54+05	1.09+05	1.07+05	8.52+04	1.19+05	2.64+04	1.25+07	427.0	122.0	5.0+06

(1) 1.05+12 = 1.05×10^{12}

Historical, not subject to future updating. This table has been retained to preserve original design basis.

TABLE 12.2–3 RADIOACTIVE SOURCES IN THE AUXILIARY BUILDING (HISTORICAL)

		Activity (MeV/cc-Sec) for Energy (MeV)									
Source No.	**Source**	**0.40**	**0.80**	**1.30**	**1.70**	**2.20**	**2.50**	**3.50**	**Height (cm)**	**Outside Diameter (cm)**	**Volume (cc)**
14	Cesium Removal Ion Exchanger	1.87+07 [1]	5.02+08	2.67+07	1.48+06	3.34+05	1.46+05	6.02+04	206.4	106.7	1.71+06
15	Boron Evaporator Reboiler	4.68+05	1.35+06	1.16+05	2.73+04	8.74+03	1.70+02	2.09+01	441.9	53.3	9.21+05
16	Boron Evaporator	4.68+05	1.35+06	1.16+05	2.73+04	8.74+03	1.70+02	2.09+01	983.0	229	2.14+07
17	Boron Evaporator Reboiler Pump	4.68+05	1.35+06	1.16+05	2.73+04	8.74+03	1.70+02	2.09+01	104.1	43.2	5.29+04
18	Boron Evaporator Bottoms Pump	4.68+05	1.35+06	1.16+05	2.73+04	8.74+03	1.70+02	2.09+01	35.6	18.4	3.82+03
19	Boron Recovery Filters	9.00+06	2.45+08	6.06+07	9.57+05	1.60+05	6.97+04	2.88+04	83.3	17.7	2.08+04
20	Boron Evaporator Bottoms Filter	4.68+05	1.35+06	1.16+05	2.73+04	8.74+03	1.70+02	2.09+01	78.4	17.7	1.95+04
21	Reactor Coolant Filter	1.72+08	4.11+08	6.33+07	1.32+07	3.08+06	2.88+05	1.74+05	48.2	16.8	1.07+04
22	Sealwater Injection Filter	4.02+05	4.72+07	4.68+08	2.44+06	--	--	--	55.8	7.0	2.14+03
23	Sealwater Heat Exchanger	1.09+04	5.96+04	2.62+04	4.45+04	1.48+03	1.28+04	1.14+04	419.1	35.5	4.17+05
24	Thermal Regeneration Demineralizer	5.05+06	4.28+06	2.73+05	1.37+05	3.94+04	3.87+01	1.30+02	253.6	122.0	2.10+06
25	Cation Bed Demineralizer	2.38+05	4.34+08	1.39+07	0.00+00	0.00+00	0.00+00	0.00+00	274.9	66.0	9.42+05
26	Mixed Bed Demineralizer	2.25+08	5.37+08	8.27+07	1.73+07	4.02+06	3.76+05	2.27+05	270.8	81.3	1.40+06
27	Letdown Heat Exchanger	1.67+05	2.75+05	1.27+05	1.32+05	1.05+05	1.43+05	4.14+04	541.0	53.3	1.21+06
28	Letdown Reheat Heat Exchanger	1.67+05	2.75+05	1.27+05	1.32+05	1.05+05	1.44+05	4.15+04	221	22.9	9.10+04

TABLE 12.2–3 RADIOACTIVE SOURCES IN THE AUXILIARY BUILDING (HISTORICAL)

		Activity (MeV/cc-Sec) for Energy (MeV)									
Source No.	**Source**	**0.40**	**0.80**	**1.30**	**1.70**	**2.20**	**2.50**	**3.50**	**Height (cm)**	**Outside Diameter (cm)**	**Volume (cc)**
29	Letdown Chiller Heat Exchanger	1.37+05	1.24+05	3.07+04	8.16+04	9.96+04	1.51+05	1.59+04	518.2	53.3	1.16+06
30	Moderating Heat Exchanger	1.37+05	1.24+05	3.07+04	8.16+04	9.96+04	1.51+05	1.59+04	556.3	45.7	9.12+05
31	Volume Control Tank Liquid	8.39+03	8.87+04	3.24+04	4.30+04	3.09+03	1.30+04	1.25+04	125.5	228.6	4.53+06
	Volume Control Tank Gases	6.67+04	4.24+04	3.00+03	2.16+04	6.55+04	9.67+04	1.29+03	188.3	228.6	6.80+06
32	Degasifier Recirculation Pump	1.13+04	9.46+04	2.64+04	3.87+04	1.40+03	1.16+04	8.19+03	35.1	27.0	1.57+04
33	Sealwater Return Filter	7.51+04	8.82+06	8.74+07	4.56+05	--	--	--	48.3	16.8	1.07+04
34	Letdown Filter	7.56+05	8.88+07	8.80+08	4.59+06	--	--	--	48.3	16.8	1.07+04
35	Boric Acid Tanks	4.68+05	1.35+06	1.16+05	2.73+04	8.74+03	1.70+02	2.09+01	579	488	1.08+08
36	Boric Acid Filters	4.68+05	1.35+06	1.16+05	2.73+04	8.74+03	1.70+02	2.09+01	48.3	16.8	1.07+04
37	Boric Acid Transfer Pumps	4.68+05	1.35+06	1.16+05	2.73+04	8.74+03	1.70+02	2.09+01	34.3	28.6	2.19+04
38	Primary Drains Transfer Tanks	1.98+05	3.33+05	1.48+05	1.40+05	1.11+05	1.55+05	3.46+04	356.9	152.4	6.51+06
39	Primary Drains Transfer Pump	1.98+05	3.33+05	1.48+05	1.40+05	1.11+05	1.55+05	3.46+04	29.8	30.5	2.17+04
40	Process Gas Charcoal Bed Adsorbers	2.56+07	2.67+05	1.46+05	1.90+05	4.73+05	7.48+05	7.91+04	492.7	182.8	1.29+07
41	Degasifier Recovery Heat Exchanger	1.55+05	1.45+05	3.32+04	8.30+04	1.02+05	1.58+05	1.28+04	709.0	38.1	7.11+05
42	Degasifier Trim Cooler	1.55+05	1.45+05	3.32+04	8.30+04	1.02+05	1.58+05	1.28+04	652.8	27.3	3.86+05
43	Degasifier Liquid	1.13+04	9.46+04	2.64+04	3.87+04	1.40+03	1.16+04	8.19+03	538.8	167.6	9.78+06

TABLE 12.2–3 RADIOACTIVE SOURCES IN THE AUXILIARY BUILDING (HISTORICAL)

		Activity (MeV/cc-Sec) for Energy (MeV)									
Source No.	**Source**	**0.40**	**0.80**	**1.30**	**1.70**	**2.20**	**2.50**	**3.50**	**Height (cm)**	**Outside Diameter (cm)**	**Volume (cc)**
	Degasifier Vapor	1.51+03	7.78+02	4.89+01	3.91+02	1.05+03	1.50+03	3.68+01	152.4	167.6	1.12+06
44	Degasifier Feed Preheater	1.51+05	1.40+05	3.22+04	8.05+04	9.88+04	1.53+05	1.24+04	373.4	30.48	2.72+05
45	Fuel Pool Post Filter	2.01+05	3.77+05	4.31+04	5.77+03	2.50+03	1.18+03	3.03-02	114	45.7	1.87+05
46	Fuel Pool Demineralizer	2.01+07	3.75+07	4.27+06	5.76+05	2.49+05	1.18+05	3.03+00	259.1	60.9	7.34+05
47	Process Gas Receiver	0.0+00	8.55+02	0.0+00	0.0+00	0.0+00	0.0+00	0.0+00	183	61.0	5.34+05
48	Charging Pumps	4.76+05	1.44+06	1.48+05	7.03+04	1.18+04	1.32+04	1.25+04	145	91.4	9.50+05
49	Boron Evaporator Bottoms Cooler	4.68+05	1.35+06	1.16+05	2.73+04	8.74+03	1.70+02	2.09+01	336.5	21.9	1.26+05
50	Boron Evaporator Distillate Cooler	4.89-01	1.35+00	3.43-01	1.78-01	3.40-02	3.75-02	3.40-03	340.3	21.9	1.28+05
53	Degasifier Condenser	1.23+06	6.32+05	3.97+04	3.17+05	8.54+05	1.21+06	2.99+04	152.4	76.2	6.95+04
54	Effluent Filters	7.58-01	6.79+00	3.02+00	1.79+00	3.74-01	9.46-04	1.56-04	147.3	32.4	1.21+05

(1) 1.87+07 = 1.87×10^{7}

Historical, not subject to future updating. This table has been retained to preserve original design basis.

TABLE 12.2–4 RADIOACTIVE SOURCES IN THE WASTE DISPOSAL BUILDING (HISTORICAL)

		Activity (MeV/cc-Sec) for Energy (MeV)									
Source No.	Source	0.40	0.80	1.30	1.70	2.20	2.50	3.50	Height (cm)	Outside Diameter (cm)	Volume (cc)
51	Boron Demineralizer	1.31+03 (1)	3.26+04	9.59+02	5.12+01	1.38+01	9.80-01	2.31-02	206.4	106.7	1.7+06
52	Boron Demineralizer Filter	5.97+02	1.48+04	4.36+02	2.33+01	6.27+00	4.46-01	1.05-02	87.3	17.8	2.17+04
55	Waste Evaporator	3.80+04	1.31+05	1.38+04	1.80+03	5.57+02	2.18+02	6.70-03	973.5	182.9	2.10+07
56	Waste Evaporator Reboiler	3.80+04	1.31+05	1.38+04	1.80+03	5.57+02	2.18+02	6.70-03	621.3	55.9	1.52+06
57	Waste Evaporator Reboiler Pump	3.80+04	1.31+05	1.38+04	1.80+03	5.57+02	2.18+02	6.70-03	109.2	43.2	6.43+04
58	Boron Evaporator Feed Pumps	1.61+02	1.34+04	6.38+02	7.98+00	1.37+00	1.69+00	1.98+00	43.2	18.4	5.06+03
59	Waste Evaporator Feed Pumps	1.61+03	5.51+03	6.70+02	1.46+02	3.59+01	1.21+01	2.48+00	35.6	18.4	3.82+03
60	Waste Evaporator Bottoms Pump	3.80+04	1.31+05	1.38+04	1.80+03	5.57+02	2.18+02	6.70-03	35.6	18.4	3.82+03
61	Spent Resin Hold Tank	2.34+07	8.36+07	9.07+06	1.75+06	3.88+05	6.64+04	4.39+04	360.0	182.9	8.90+06
62	Spent Resin Transfer Pump Filter	2.25+06	5.37+06	8.27+05	1.73+05	4.02+04	3.76+03	2.27+03	83.8	17.8	2.08+04
63	Spent Resin Transfer Pump	2.25+06	5.37+06	8.27+05	1.73+05	4.02+04	3.76+03	2.27+03	30.5	30.5	2.22+04
64	High-Level Waste Drain Tank	1.61+03	5.51+03	6.70+02	1.46+02	3.59+01	1.21+01	2.48+00	1295.4	320.0	1.01+08
65	Low-Level Waste Drain Tank	7.64-01	7.11+00	3.13+00	1.85+00	3.74-01	9.46-04	1.56-04	340.4	274.3	1.80+07
66	Low Level Waste Drain Pump	7.64-01	7.11+00	3.13+00	1.85+00	3.74-01	9.46-04	1.56-04	35.6	18.4	3.82+03
67	Waste Evaporator Bottoms	3.80+04	1.31+04	1.38+04	1.80+03	5.57+02	2.18+02	6.70-03	335.3	21.9	1.27+05
68	Waste Distillate Cooler	3.80+00	1.31+01	1.38+00	1.80-01	5.57-02	2.18-02	6.70-07	365.8	27.3	2.17+05

TABLE 12.2–4 RADIOACTIVE SOURCES IN THE WASTE DISPOSAL BUILDING (HISTORICAL)

		Activity (MeV/cc-Sec) for Energy (MeV)									
Source No.	**Source**	**0.40**	**0.80**	**1.30**	**1.70**	**2.20**	**2.50**	**3.50**	**Height (cm)**	**Outside Diameter (cm)**	**Volume (cc)**
72	Waste Demineralizer	1.25+03	1.37+05	9.10+03	4.91+01	1.13+01	1.38+00	4.79-01	206.4	106.7	1.70+06
73	Waste Demineralizer Filter	5.67+02	6.22+04	4.14+03	2.23+01	5.13+00	6.27-01	2.18-01	87.3	17.8	2.17+04

(1) 1.31+03 = 1.31 x 10^3

Historical, not subject to future updating. This table has been retained to preserve original design basis.

TABLE 12.2–5 RADIOACTIVE SOURCES IN THE FUEL BUILDING (HISTORICAL)

		Activity (MeV/cc-Sec) for Energy (MeV)									
	Source	**0.40**	**0.80**	**1.30**	**1.70**	**2.20**	**2.50**	**3.50**	**Height (cm)**	**Diameter (cm)**	**Volume (cc)**
69	Most Radioactive Fuel Assembly 100 Hours after Sutdown	1.67+11(1)	1.02+12	7.21+10	5.10+11	1.49+10	3.68+10	1.01+09			
70	Fuel Pool Purification Filter	3.09+02	4.02+05	6.22+04	--	1.66+03	7.20+01	1.06+01	1.68+02	45.7	2.75+05
74	Fuel Pool (Filled to Capacity 100 Hours after Shutdown)	5.58+3	9.02+3	1.05+3	1.14+2	5.16+1	2.42+1	4.48-2			

NOTE:

(1) 1.67+11 = 1.67×10^{11}

Historical, not subject to future updating. This table has been retained to preserve original design basis.

TABLE 12.2–5A OTHER RADIOACTIVE SOURCES (HISTORICAL)

		Activity (MeV/cc-Sec) for Energy (MeV)									
Source No.	**Source**	**0.40**	**0.80**	**1.30**	**1.70**	**2.20**	**2.50**	**3.50**	**Height (cm)**	**Outside Diameter (cm)**	**Volume (cc)**
ESF Building											
7	RHR Pump	1.66+05 (1)	2.13+05	6.84+04	4.31+04	3.12+04	3.76+04	4.65+03	43.2	99.1	3.32+05
8	RHR Exchanger	1.66+05	2.13+05	6.84+04	4.31+04	3.12+04	3.76+04	4.65+03	1,370	100.0	1.09+07
Yard Tanks											
71	Boron Recovery Tank	1.61+02	1.34+04	6.38+02	7.98+00	1.37+00	1.69+00	1.98+00	9.14+02	9.14+02	5.70+08
75	RWST	1.68-02	1.17+00	8.11-02	1.09-05	1.89-06	1.19-06	2.27-10	1.80+03	1.80+03	4.41+09
Condensate Polishing Building											
76	Condensate Polishing Demineralizer	8.12+03	3.63+04	5.15+03	1.13+03	3.46+02	1.19+02	1.61+00	3.35+02	2.44+02	5.67+06
77	Cation Regeneration Tank	3.38+03	7.16+04	8.96+03	4.26+02	1.22+02	6.80+01	3.58+00	4.27+02	2.08+02	2.21+06
78	Anion Regeneration Tank	1.15+04	1.43+04	2.84+03	1.64+03	5.06+02	1.56+02	3.59-01	4.57+02	1.83+02	3.34+06

NOTE:

(1) 1.66+05=1.66×10^5

Historical, not subject to future updating. This table has been retained to preserve original design basis.

TABLE 12.2–6 INVENTORY OF AN AVERAGE FUEL ASSEMBLY AFTER 650 DAYS OF OPERATION AT 3,636 MWt AT SHUTDOWN AND 100 HOURS AFTER SHUTDOWN (μCi) (HISTORICAL)

Isotope	0 Hours	100 Hours
Kr-83m	8.19+10 *	1.07-01
Kr-85m	2.05+11	2.91+04
Kr-85	4.58+09	4.58+09
Kr-87	3.99+11	**
Kr-88	5.60+11	9.17+00
Kr-89	7.25+11	**
Xe-131m	4.15+08	8.55+10
Xe-133m	2.53+10	1.09+10
Xe-133	1.05+12	7.25+11
Xe-135m	2.85+11	9.53+06
Xe-135	2.79+11	1.45+09
Xe-137	9.48+11	**
Xe-138	9.33+11	**
Br-83	8.19+10	2.37-02
Br-84	1.47+11	**
Br-85	2.05+11	**
Br-87	3.95+11	**
I-129	1.12+04	1.12+04
I-131	4.72+11	3.38+11
I-132	6.74+11	2.84+11
I-133	1.06+12	3.77+10
I-134	1.23+12	**
I-135	9.74+11	3.06+07
I-136	4.89+11	**
Se-81	2.84+10	**
Se-83	3.47+10	**
Se-84	1.47+11	**
Rb-88	5.65+11	1.03+01

TABLE 12.2–6 INVENTORY OF AN AVERAGE FUEL ASSEMBLY AFTER 650 DAYS OF OPERATION AT 3,636 MWt AT SHUTDOWN AND 100 HOURS AFTER SHUTDOWN (μCi) (HISTORICAL)

Isotope	0 Hours	100 Hours
Rb-89	7.51+11	**
Rb-90	9.17+11	**
Rb-91	8.55+11	**
Rb-92	7.10+11	**
Sr-89	7.51+11	7.10+11
Sr-90	3.73+10	3.73+10
Sr-91	9.17+11	7.10+08
Sr-92	8.34+11	5.34+00
Sr-93	9.33+11	**
Sr-94	7.25+11	**
Y-90	3.70+10	3.70+10
Y-91m	5.39+11	4.66+08
Y-91	9.33+11	8.91+11
Y-92	9.33+11	1.08+04
Y-93	9.64+11	1.07+09
Y-94	8.50+11	**
Y-95	9.64+11	**
Zr-95	9.79+11	9.38+11
Zr-97	9.33+11	1.48+10
Nb-95m	1.95+10	1.93+10
Nb-95	1.01+12	1.01+12
Nb-97m	8.91+11	1.42+10
Nb-97	9.79+11	1.60+10
Mo-99	9.74+11	3.44+11
Mo-101	7.88+11	**
Mo-102	6.58+11	**
Mo-105	1.42+11	**
Tc-99m	8.55+11	3.28+11

TABLE 12.2–6 INVENTORY OF AN AVERAGE FUEL ASSEMBLY AFTER 650 DAYS OF OPERATION AT 3,636 MWt AT SHUTDOWN AND 100 HOURS AFTER SHUTDOWN (μCi) (HISTORICAL)

Isotope	0 Hours	100 Hours
Tc-101	1.58+12	**
Tc-102	6.58+11	**
Tc-105	1.90+11	**
Ru-103	4.74+11	4.40+11
Ru-105	1.42+11	2.33+04
Ru-106	4.46+10	4.42+10
Ru-107	3.00+10	**
Rh-103m	4.74+11	4.41+11
Rh-105m	1.42+11	2.34+04
Rh-105	1.42+11	2.06+10
Rh-106	4.47+10	4.42+10
Rh-107	3.00+10	**
Sn-127	1.74+10	1.10-04
Sn-128	5.85+10	**
Sn-130	1.74+11	**
Sb-127	2.16+10	1.03+10
Sb-128	7.88+09	**
Sb-129	1.58+11	1.80+04
Sb-130	3.16+11	**
Sb-131	4.26+11	**
Sb-132	5.28+11	**
Sb-133	5.39+11	**
Te-127m	4.67+09	4.95+09
Te-127	4.24+09	1.34+10
Te-129m	8.65+10	7.98+10
Te-129	1.69+11	8.03+10
Te-131m	6.94+10	6.89+09
Te-131	4.15+11	1.39+09

TABLE 12.2–6 INVENTORY OF AN AVERAGE FUEL ASSEMBLY AFTER 650 DAYS OF OPERATION AT 3,636 MWt AT SHUTDOWN AND 100 HOURS AFTER SHUTDOWN (μCi) (HISTORICAL)

Isotope	0 Hours	100 Hours
Te-132	6.74+11	2.76+11
Te-133m	7.41+11	**
Te-133	4.74+11	**
Te-134	1.09+12	**
Cs-137	3.91+10	3.91+10
Cs-138	1.06+12	**
Cs-139	1.02+12	**
Cs-140	9.33+11	**
Cs-142	5.18+11	**
Ba-137m	3.65+10	3.59+10
Ba-139	1.02+12	**
Ba-140	9.95+11	7.93+11
Ba-141	9.64+11	**
Ba-142	9.12+11	**
La-140	1.00+12	8.86+11
La-141	9.64+11	1.77+04
La-142	9.17+09	**
La-143	9.27+11	**
Ce-141	9.59+11	8.86+11
Ce-143	9.33+11	1.15+11
Ce-144	6.79+11	6.68+11
Ce-145	6.11+11	**
Ce-146	4.54+11	**
Pr-143	9.27+11	8.24+11
Pr-144	6.79+11	6.68+11
Pr-145	6.17+11	5.75+06
Pr-146	4.66+11	**
Nd-147	3.45+11	2.66+11

TABLE 12.2–6 INVENTORY OF AN AVERAGE FUEL ASSEMBLY AFTER 650 DAYS OF OPERATION AT 3,636 MWt AT SHUTDOWN AND 100 HOURS AFTER SHUTDOWN (μCi) (HISTORICAL)

Isotope	0 Hours	100 Hours
Nd-149	1.64+11	**
Nd-151	6.63+10	**
Pm-147	1.13+11	1.14+11
Pm-149	1.64+11	4.59+10
Pm-151	6.63+10	5.80+09
Sm-151	5.18+07	5.44+07
Sm-153	2.42+10	5.54+09

NOTES:

* 8.19+10 = 8.19×10^{10}

** Less than 1.00×10^{-6} μCi

Historical, not subject to future updating. This table has been retained to preserve original design basis.

TABLE 12.2–7 SOURCE INTENSITY IN THE MOST RADIOACTIVE FUEL ASSEMBLY * AFTER 650 DAYS OF OPERATION AT 3636 MWt (HISTORICAL)

Decay Time (Hrs)	Activity (MeV/Sec) for Energy Group (MeV)						
	0.4	**0.8**	**1.3**	**1.7**	**2.2**	**2.5**	**3.5**
0	1.77+17 **	7.83+17	7.91+17	6.47+17	1.35+17	2.15+17	5.57+17
2	7.45+16	4.86+17	1.38+17	2.23+17	2.68+16	3.66+16	5.75+15
4	6.69+16	4.11+17	9.75+16	1.74+17	1.88+16	2.14+16	2.24+15
8	6.11+16	3.56+17	6.76+16	1.38+17	1.11+16	1.21+16	8.81+14
16	5.47+16	3.03+17	4.70+16	1.13+17	6.24+15	8.33+15	2.84+14
20	5.21+16	2.84+17	3.51+16	1.08+17	5.38+15	7.91+15	2.25+14
24	4.97+16	2.69+17	3.03+16	1.04+17	4.86+15	7.70+15	2.02+14
48	3.93+16	2.15+17	1.86+16	9.58+16	3.62+15	7.17+15	1.85+14
100	2.82+16	1.72+17	1.22+16	8.63+16	2.52+15	6.23+15	1.71+14
168	2.06+16	1.49+17	8.55+15	7.42+16	1.69+15	5.19+15	1.50+14
720	5.81+15	9.49+16	1.97+15	2.13+16	6.81+14	1.40+15	4.68+13

NOTES:

* Includes a radial peaking factor of 1.65

** 1.77+17 = 1.77 x 10^{17}

Historical, not subject to future updating. This table has been retained to preserve original design basis.

TABLE 12.2–8 RADIONUCLIDE CONCENTRATIONS IN THE SPENT FUEL POOL FROM REFUELING 100 HOURS AFTER SHUTDOWN* (HISTORICAL)

Nuclide	Expected Concentration (μCi/cc)	Design Concentration (μCi/cc)
I-131	2.4E-02	2.2E-01
I-132	1.4E-03	1.3E-02
I-133	1.8E-03	1.7E-02
I-135	**	8.2E-06
Sr-89	4.1E-05	4.8E-04
Sr-90	1.2E-06	2.0E-05
Y-90	**	2.2E-05
Y-91	8.1E-06	7.9E-05
Zr-95	7.1E-06	8.0E-05
Nb-95m	**	1.7E-06
Nb-95	6.3E-06	8.7E-05
Mo-99	3.7E-03	1.4E-01
Tc-99m	3.6E-03	1.3E-01
Ru-103	5.1E-06	3.8E-05
Ru-106	1.2E-06	3.8E-06
Rh-103m	5.1E-06	3.8E-05
Rh-105	**	1.5E-06
Rh-106	1.2E-06	3.8E-06
Te-127m	3.3E-05	2.4E-04
Te-127	3.3E-05	2.4E-04
Te-129m	1.6E-04	4.2E-03
Te-129	1.6E-04	4.2E-03
Te-131m	3.2E-05	2.7E-04
Te-131	6.4E-06	5.5E-05
Te-132	1.4E-03	1.3E-02
Cs-134	3.2E-03	3.9E-02
Cs-136	1.3E-03	1.6E-02
Cs-137	2.2E-03	1.9E-01

TABLE 12.2–8 RADIONUCLIDE CONCENTRATIONS IN THE SPENT FUEL POOL FROM REFUELING 100 HOURS AFTER SHUTDOWN* (HISTORICAL)

Nuclide	Expected Concentration (µCi/cc)	Design Concentration (µCi/cc)
Ba-137m	2.1E-03	1.8E-01
Ba-140	2.2E-05	4.1E-04
La-140	2.3E-05	4.0E-04
Ce-141	7.9E-06	7.5E-05
Ce-143	**	7.5E-06
Ce-144	4.1E-06	5.8E-05
Pr-143	5.4E-06	6.9E-05
Pr-144	4.1E-06	5.8E-05
Nd-147	**	2.2E-05
Pm-147	**	9.9E-06
Pm-149	**	3.3E-06
Cr-51	2.1E-04	2.1E-04
Mn-54	3.8E-05	3.8E-05
Fe-55	2.0E-04	2.0E-04
Fe-59	1.2E-04	1.2E-04
Co-58	1.9E-04	1.9E-03
Co-60	2.5E-04	2.5E-04
H-3	1.2E-01	4.1E-01

NOTES:

* The expected and design concentrations assume complete mixing of reactor coolant with refueling cavity water and spent fuel pool water 100 hours after reactor shutdown.

** Concentrations less than 1.0E-06 µCi/cc are considered negligible and are not tabulated.

1.0E-06 = 1.0×10^{-6}

Historical, not subject to future updating. This table has been retained to preserve original design basis.

TABLE 12.2–9 RADIATION SOURCES * REACTOR COOLANT NITROGEN-16 ACTIVITY (HISTORICAL)

Position in Loop	Loop Transit Time (sec)	Nitrogen-16 Activity (μCi/gm)
Leaving Core	0.0	189
Leaving Reactor Vessel	1.1	170
Entering Steam Generator	1.4	164
Leaving Steam Generator	5.4	112
Entering Reactor Coolant Pump	6.0	106
Entering Reactor Vessel	6.8	98
Entering Core	9.0	86
Leaving Core	9.7	189

Nitrogen-16 Energy Emission	
Energy (MeV/gamma)	**Intensity (percent)**
1.75	0.13
2.74	9.76
6.13	60.0
7.12	5.0

* Source: Westinghouse letter NEU-3492, dated July 31, 1980.

Historical, not subject to future updating. This table has been retained to preserve original design basis.

TABLE 12.2–10 ASSUMPTIONS USED IN THE CALCULATION OF AIRBORNE CONCENTRATIONS (HISTORICAL)

			Containment Building	Turbine Building	Fuel Building
1.	Reactor coolant equilibrium concentrations		Table 11.1-2		
2.	Secondary side equilibrium concentrations		-	Table 11.1-6	-
3.	Iodine and noble gas core inventory		-	-	Table 11.1-1
4.	Leak rate into buildings				
	A.	Equivalent hot reactor coolant (lb/day)	$4.7x10^3$	-	-
	B.	Equivalent main steam leakage (lb/hr)	-	$1.7x10^3$	-
5.	Normal moisture in atmosphere (%)		60	-	-
6.	Fraction of primary coolant activities released (%/day)				
	A.	Noble gases	1.0	-	-
	B.	Iodines	0.001	-	-
7.	Mixing in building atmosphere (%)		70	100	100
8.	Building ventilation rate (cfm)		$3.0x10^4$	$1.55x10^5$	$3.0x10^4$
9.	Building free volume (ft^3)		$2.32x10^6$	$4.06x10^6$	$2.30x10^5$ *
10.	Recirculation - filters		Yes	No	No
11.	Filter efficiency		99%	-	-
12.	Fuel pool evaporation rate ($lb/hr\text{-}ft^2$)		-	-	1.74
13.	Recirculation rate (cfm)		$2.4x10^4$	-	-
14.	Fuel pool average volume (ft^3)		-	-	$4.88x10^4$

NOTE:

* Only the area above the fuel pool.

Historical, not subject to future updating. This table has been retained to preserve original design basis.

TABLE 12.2–11 AIRBORNE CONCENTRATIONS INSIDE MAJOR BUILDINGS (μCI/CC) (HISTORICAL)

	Containment Building Prior to Recirculation		Containment Building After 16-Hour Recirculation [(1)]		Turbine Building		Fuel Building	
Isotope	**Design**	**Expected**	**Design**	**Expected**	**Design**	**Expected**	**Design**	**Expected**
H-3	1.6E-4[(2)]	4.6E-5	1.6E-4	4.6E-5	1.2E-8	2.9E-9	3.8E-6	1.1E-6
I-131	9.8E-7	1.1E-7	9.1E-9	1.0E-9	3.8E-11	2.9E-13	2.1E-10	2.3E-11
I-132	4.1E-9	2.3E-10	1.7E-9	2.2E-10	9.8E-12	7.7E-14	5.0E-12	5.4E-13
I-133	1.7E-7	1.7E-8	1.2E-8	1.3E-9	5.5E-11	4.0E-13	1.8E-12	1.9E-18
I-134	9.8E-10	9.8E-11	6.5E-10	6.5E-11	1.8E-12	1.3E-14	-	-
I-135	2.9E-8	2.9E-9	5.6E-9	5.6E-10	2.32E-11	1.7E-13	2.7E-16	-
Kr-83m	1.6E-6	8.0E-8	1.6E-6	8.0E-8	4.1E-12	1.5E-14	-	-
Kr-85m	1.5E-5	7.9E-7	1.5E-5	7.9E-7	1.7E-11	6.6E-14	-	-
Kr-85	1.0E-4	6.1E-6	1.0E-4	6.1E-6	3.5E-13	1.7E-15	-	-
Kr-87	3.0E-6	1.6E-7	3.0E-6	1.6E-7	1.0E-11	4.0E-14	-	-
Kr-88	1.9E-5	1.0E-6	1.9E-5	1.0E-6	3.2E-11	1.3E-13	-	-
Kr-89	1.1E-8	6.4E-10	1.1E-8	6.4E-10	1.7E-13	7.0E-16	-	-
Xe-131m	6.2E-2	3.1E-6	6.2E-2	3.1E-6	1.2E-13	4.4E-15	2.0E-8	2.2E-9
Xe-133m	6.5E-5	4.3E-6	6.5E-5	4.3E-6	6.4E-12	3.2E-14	4.2E-12	4.3E-13
Xe-133	6.5E-3	4.3E-4	6.5E-3	4.3E-4	2.8E-10	1.3E-12	1.8E-10	1.8E-11
Xe-135m	6.2E-7	9.0E-9	6.2E-7	9.0E-9	9.3E-12	3.3E-14	6.2E-15	-
Xe-135	9.2E-5	4.0E-6	9.2E-5	4.0E-6	5.2E-11	1.7E-13	2.6E-14	-

TABLE 12.2–11 AIRBORNE CONCENTRATIONS INSIDE MAJOR BUILDINGS (μCI/CC) (HISTORICAL)

	Containment Building Prior to Recirculation		Containment Building After 16-Hour Recirculation (1)		Turbine Building		Fuel Building	
Isotope	**Design**	**Expected**	**Design**	**Expected**	**Design**	**Expected**	**Design**	**Expected**
Xe-137	2.1E-8	1.4E-9	2.1E-8	1.4E-9	3.1E-13	1.5E-15	-	-
Xe-138	2.8E-7	2.5E-8	2.8E-7	2.5E-8	2.8E-12	1.8E-14	-	-

NOTES:

(1) Using 99% filter efficiency

(2) 1.6E-4 = 1.6 x 10^{-4}

Historical, not subject to future updating. This table has been retained to preserve original design basis.

12.3 RADIATION PROTECTION DESIGN FEATURES

12.3.1 SHIELDING

The assessments performed to determine the original major shield designs were based on assumed source terms, occupancy times and acceptance criteria based on zone criteria. Although these criteria were used to establish the original shield design, they were never intended to establish requirements for the radiation protection program implementation during plant operation. As time evolves, source terms change. Acceptable doses have typically decreased with time as ambitious ALARA person-REM goals are established.

Current shielding requirements are non-specific and are established through the implementation of the Radiation Protection Program and ALARA Program. These programs evaluate the need for a combination of exposure saving principals such as reduced source term, decreasing occupancy time, or increased shielding. These programs use shielding as one method to help ensure compliance with 10 CFR 20.

This section provides the basis for the original plant shielding design. Although current dose rates may not be consistent with the zone maps in this chapter, these maps are not being changed to be current, as that would make them inconsistent with the original design basis criteria for the shielding. Recent Heath Physics surveys should be consulted for information on current station radiological conditions.

Radiation shielding is designed to ensure that radiation exposure to the general public and to personnel in-plant is kept to levels as low as is reasonably achievable (ALARA), consistent with the requirements set forth in 10 CFR 20 for normal operation and 10 CFR 50.67 for accident conditions and with the overall objectives set forth in USNRC Regulatory Guide 8.8. The original design of this radiation shielding was based upon radiation zone criteria which were established in support of the expected access requirements and durations of occupancy during normal operations, during refueling outages, and during accident situations. Descriptions of the zone criteria are presented in Table 12.3-1, and the detailed radiation zone criteria for normal and shutdown operations are illustrated on Figures 12.3-1 through 12.3-4 and 12.3-6 through 12.3-9. These figures do not represent operational requirements.

Radiation shielding is provided on the basis of maximum concentrations of radioactive materials within each shield region (e.g., 1 percent failed fuel at the original design basis core power level of 3636 MWt) rather than the annual average values. For batch processes, as an example, the point of highest radionuclide concentration in the batch process is assumed (e.g., just prior to draining of a tank). The shielding designs are, therefore, intentionally conservative in that the dose rates reflect maximum, rather than average, sources to be shielded. These maximum dose rates are based on anticipated occupancy requirements and are set such that the maximum exposure of plant personnel is within the limits set by 10 CFR 20. The average exposures are expected to be a small fraction of the limiting values because it is not expected that the plant would run at 1 percent failed fuel with all tanks full to capacity, all demineralizer beds at saturation, etc.

In computing the dose rates on which the confirmation of shielding thicknesses is based, a number of explicit and implicit conservative measures are included. Dose points are generally calculated along vertical shield surfaces opposite the most intense source in the vicinity. These calculations are based on the inherent assumption that plant personnel spend the required time in each zone in contact with the shield at this point. This is a demonstrably conservative approach, since the dose rate actually decreases dramatically as the dose points are moved along the surface of the shield due to the slant penetration involved. The additional reduction of intensity with distance is also ignored by this approach.

The shield wall thicknesses are derived from design basis fuel defect of 1 percent and dose rate limitation of adjacent zones and are expected to provide adequate protection for abnormal conditions which may occur during normal plant operations.

Zone designations are based on the annual occupational exposure limits, access requirements and occupancy time for the specific location in the plant as described in Table 12.3-1.

For the yard areas, the shield walls are designed to meet the Zone I criterion of 0.25 mRem per hour. The most significant structures which contribute to the yard dose rate are the containment, fuel building, waste disposal building, auxiliary building, refueling water storage tank, and the boron recovery tanks.

The calculated dose rate levels in the unrestricted areas are based upon full power normal plant operations assuming fuel defects producing expected quantities and concentrations of radionuclides consistent with NUREG-0017. At the site boundary, the calculated dose rate is approximately 0.43 mRem per year.

Dose rates are generally calculated at three and six foot levels above walking surfaces, particularly if significant sources are located on the next level above the zone within the building.

Dose rates for post-shutdown conditions are computed at the earliest reasonable time after shutdown. Subsequent decay is ignored for conservatism; i.e., the dose rate at that point in time is quoted despite the fact that the radiation levels continue to decay to lower values.

Transit times in coolant loops, etc, are computed as precisely as practicable with no intentional conservatism. Simplified models which are used to describe large components are intentionally devised not to overestimate component self-shielding. This provides an amount of conservatism which varies from component to component. In some instances no component self-shielding is included.

The modeling reflects the knowledge of specific components and limitations which exist when details on components are not available. The models allow for this uncertainty in a conservative manner, thus ensuring that the actual radiation leakage from the supplied component is less than or equal to predicted values.

Shielding in the Millstone 3 plant was designed using Stone & Webster Engineering Corporation's topical report, Radiation Shielding Design and Analysis Approach for Light Water

Reactor Power Plants (RP 8A) (SWEC 1975). This approach, recommended for guidance by the NRC in the Standard Review Plan, Section 12.3, incorporates the design features offered in Regulatory Guides 1.69 and 8.8. RP 8A defines the assumptions, codes, techniques, and parameters used in calculating shield thickness, material, and placement.

12.3.1.1 Primary Shielding

Primary shielding is provided to limit radiation emanating from the reactor vessel.

The primary shield is designed to:

1. attenuate neutron flux to minimize activation of plant components and structures;

2. reduce residual radiation from the core to a level that allows access to the region between the primary and secondary shields at a reasonable time after shutdown; and

3. optimize the combination of primary and secondary shielding by reducing the radiation level from the reactor so that it is commensurate with radiation levels from other sources.

The primary shield consists of a water filled neutron shield tank and a 4.5 foot thick reinforced concrete shield wall. The neutron shield tank has an annular thickness of 3 feet and is located between the reactor vessel and the concrete shield wall. To maintain the integrity of the primary shield, a streaming shield fabricated from borated silicon rubber (Dow Corning Sylgard 170 silicon elastomer or equivalent) is installed in the upper annular gap between the vessel flange and the neutron shield tank and around the nozzles. (Refer to Figure 12.3-12.)

This shield is designed to minimize the leakage of neutrons to the annular region and streaming to the upper levels of the containment, thus reducing the neutron dose rate on the operating floor, during normal operations, to acceptable levels.

It was estimated that the neutron dose rate in the annulus area between the containment wall and the crane wall at the operating floor level would not exceed 5 mRem per hour with the shield in place. Radiation protection surveys should be consulted to determine actual neutron dose rates.

12.3.1.2 Secondary Shielding

Secondary shielding consists of reactor coolant loop shielding, the crane wall, containment structure shielding, fuel handling shielding, auxiliary equipment shielding, waste storage shielding, control room shielding, and yard shielding.

Secondary shielding thicknesses within the containment structure are based on nitrogen 16 being the major source of radioactivity in the reactor coolant during normal operation. This source establishes a required shielding thickness of the reactor coolant loop shielding, crane wall, and containment structure wall. The shutdown radiation levels in the reactor coolant loop cubicles are established by the activities of the activated corrosion and fission products in the reactor coolant system.

The crane wall provides shielding for limited access to the annulus between the crane wall and the containment structure wall and provides additional exterior shielding during power operation.

The containment structure shielding consists of a steel lined reinforced concrete cylinder and hemispherical dome. This shielding, together with the crane wall, attenuates radiation during full power operation and during an accident. This shielding keeps radiation levels within acceptable levels at the outside surface of the containment structure and at the exclusion area boundary (EAB).

The fuel handling shielding, including both water and concrete, attenuates radiation from spent fuel assemblies, control rods, and reactor vessel internals to acceptable levels and permits the removal and transfer of spent fuel and control rods to the fuel pool in the fuel building.

The refueling cavity above the reactor is formed by a stainless steel-lined, reinforced concrete structure. This refueling cavity becomes a pool when filled with borated water to provide shielding during the refueling operation.

The depth of the shielding water in the cavity is such that the radiation dose rate at the surface of the water from a spent fuel assembly should not exceed approximately 2.5 mRem per hour during the short time intervals when the fuel handling operation brings the spent fuel assembly to its closest approach to the pool surface.

The cavity is large enough to provide storage space for the upper and lower internals and miscellaneous refueling tools.

The fuel pool in the fuel building is filled with water for shielding as discussed in Section 9.1.4.3.4. The fuel pool walls are a minimum of 6 foot thick concrete to ensure a dose rate less than 0.75 mRem per hour outside the fuel building and less than 2.5 mRem per hour inside both the fuel building and the adjacent auxiliary building from the fuel stored in the pool.

In order to preclude unacceptable radiation dose rates during fuel transfer, a special radiation shield, fabricated from carbon steel has been provided inside containment, where the fuel transfer tube traverses the gap between the containment wall and the refueling cavity wall. (Refer to Figure 12.3-11). The design basis for the shielding concept is a dose rate at the surface of the shield of approximately 50 mRem per hour and a dose rate at the personnel access hatch of approximately 5 mRem per hour.

Outside containment, the fuel transfer tube is inaccessible to personnel by means of backfill covering the transfer tube and a security fence between the containment and the fuel building, assuring limited access to this area.

Three radiation monitors with local audible and visible alarms as well as remote alarms in the control room, are used to monitor fuel transfer operations. Two radiation monitors are located in the passageway and access area adjacent to the fuel transfer tube in the containment building. The other monitor is in the fuel building.

Auxiliary building components may exhibit varying degrees of radioactive contamination due to the handling of various fluids. The function of shielding in this building is to protect operating and maintenance personnel working near the various auxiliary system components, such as those in the makeup and purification system, the boron recovery system, the radioactive liquid and gaseous waste systems, and the sampling system.

Typically, major components of systems are individually shielded so that compartments may be entered without having to shut down and possibly decontaminate the entire system. Potentially highly contaminated ion exchangers and filters are located in individual shielded cells in the auxiliary building. The concrete thicknesses provided around the shielded compartments was based on reducing the surrounding area dose rate to less than approximately 2.5 mRem per hour and the dose rate to any adjacent cubicle to less than approximately 100 mRem per hour.

In some areas, tornado missile protection in the form of concrete affords more shielding than that required for radiation protection.

The waste storage and processing facilities in the auxiliary building and the waste disposal building are shielded to provide protection for operating personnel in accordance with radiation protection design criteria.

Boron recovery tanks, which may be used to store “letdown” prior to its recycling to the plant or processing as waste, are shielded to reduce dose rates to accessible levels within the yard area.

12.3.1.3 Accident Shielding

12.3.1.3.1 Containment and Control Room Design

Accident shielding is provided by the containment structure, which is a reinforced concrete structure lined with steel. For structural reasons, the thickness of the cylindrical wall and the dome are 54 inches and 30 inches, respectively. These thicknesses are more than adequate to meet the shielding requirement during accident conditions.

The radiation design objective for the control room shielding helps limit the dose from external sources to personnel inside the control room to less than 5 Rem during any design basis accident. This dose includes: (1) the external radiation contribution from the postulated radioactive plume leaking from the containment for a period of 30 days; (2) the 30 day radiation dose from radioactivity inside the containment; (3) the 30 day radiation dose due to post-LOCA leakage from the ECCS located outside of the Millstone 3 containment; and (4) the radiation dose due to radioactive components within the control room boundary (e.g., buildup of halogens in filters). The Millstone 3 Control Room has also been evaluated for the 30 day dose due to a postulated LOCA at Millstone 2 and those results are within the limits of GDC-19. Shielding calculations show that the 2 foot thick concrete walls which enclose the control room are sufficient to ensure that the radiation dose inside the control room remains below the radiation design basis during any postulated accident. The control room design includes special treatment of shield wall penetrations and structural details which ensures that this facility remains acceptably leak-resistant.

12.3.1.3.2 Post-Accident Access to Vital Areas

A radiation and shielding design review was performed in accordance with NUREG-0737, Action Item II.B.2 (USNRC, 1980), in order to ensure personnel accessibility after a design-basis accident (DBA). The DBA considered for this evaluation was the loss-of-coolant accident (LOCA). The projected dose to complete each activity necessary to mitigate a DBA LOCA, en route to and in vital areas, is less than the 5 rem design limit of NUREG-0737. At Millstone 3, this requirement is met by providing sufficient shielding of components containing post-accident radioactive inventories, consistent with anticipated access routes and stay times.

Areas requiring accessibility (vital areas) are those areas where post-LOCA actions can be taken over the short-term to ensure the capability of operators to control and mitigate the consequences of an accident. A description of the post-accident activities is summarized below and in Table 12.3-3.

1. Locally trip the reactor trip breakers and bypass breakers

   This action is performed at the 43 foot 6 inches elevation in the auxiliary building MCC rod control area. This is done in the event that the reactor failed to trip. This action must take place as soon as possible. Thus, the 0 to 30 minute time frame is assumed. While this step is done only in the event of an ATWS (beyond the design basis scenario), it is conservatively included as a required operator action.

2. Local actions needed to realign Spent Fuel Pool Cooling, RBCCW and Service Water for spent fuel pool cooling

   FSAR 9.1.3.3 states that spent fuel pool cooling will be initiated approximately 4 hours after the LOCA. This requires operator action in the spent fuel pool building. The 2 to 8 hour time frame is assumed.

3. Powering the Plant Process Computer

   The Plant Process computer is normally not powered from an Emergency Bus. It is powered from an uninterruptible power supply that may last for only 30 minutes. Thus, the 0 to 30 minute category is assumed. The plant process computer is used for SPDS and OFIS. In order to restore power to the plant process computer, MCC 32-3T is energized on the 38 foot level in the turbine building.

4. Powering the SI accumulator valves

   For post-LOCA cooldown and depressurization, the SI accumulator isolation valves are closed to prevent injection of nitrogen that might interrupt natural circulation. It is necessary to repower the valves from the 24 foot 6 inch level in the auxiliary building. Since this would be done only after the plant is stabilized in preparation for a cooldown, the 30 minute to 2 hour time frame is assumed.

5. <u>Initiate hydrogen monitor</u>

FSAR Section 6.2.5.2 states that this system will be available to provide continuous monitoring within 1 hour and 30 minutes of an accident. For dose consequence evaluation, availability within 30 minutes was assumed for conservatism. Thus, the 0 to 30 minute category is assumed. Access to the hydrogen recombiner building is needed in order to initiate hydrogen monitoring.

6. <u>Deleted</u>

7. <u>Deleted</u>

8. <u>Repower Monitor and Maintain the porous concrete groundwater removal system</u>

A non-safety related pump (3SRW-P5) is credited with groundwater removal that circumvents the waterproof membrane that surrounds the containment structure and the containment structure contiguous buildings.

- 3SRW-P5 is normally powered from 32-4T. If "A" Train Emergency Bus is not able to supply power to 32-4T, then 3SRW-P5 can be repowered from 32-3U, "B" Train Emergency Bus. It is estimated that repower may take 1.5 hours. It is expected that for a design basis LOCA, this step is reached before 4 hours. The performance of this action is based on the radiological conditions near the RWST, which requires work outside the ESF building to be completed between 2 and 6 hours.

- The status of the groundwater removal system will be monitored and operated several times a day. These activities take place in the yard on the north side of the Refuel Water Storage Tank (RWST) at local panel 3SRW-CSP5. The 24 hour time frame and beyond is assumed for monitoring. Due to dose considerations near the RWST as the accident progresses, the activities at panel 3SRW-CSP5 may need to be completed in as little as 2 minutes.

- Should the single non-safety related groundwater sump pump become nonfunctional, it must be replaced or repaired. Due to dose considerations, the 1 day time frame and beyond is assumed for maintenance and repair activities for the sump pump. The sump pump is accessible from the ESFB roof. Access to the ESFB roof is achieved via the Hydrogen Recombiner Building stairway.

9. <u>Open the breakers for the non-safety grade sump pumps</u>

The operation of the non-safety grade sump pumps may mask the presence of a leak. Thus, the need to secure sump pumps in ECCS pump cubicles and common areas in the Auxiliary and ESF buildings. The 1 day to 4 day time frame is assumed. This action requires access to the 21 foot elevation of the ESF building and the 24 foot 6 inch elevation of the auxiliary building. However, if radiological conditions preclude entry into the ESF or Auxiliary Building, then the associated MCC's may be de-energized at its Load Centers in the 4 foot elevation of the Service Building. Therefore, local operator actions in the ESF and Auxiliary Building are not required.

10. Align Alternate AFW Pump Suction Source or Replenish Demineralized Water Storage Tank (DWST) Inventory

For a small break LOCA, steam generator inventory makeup beyond that provided by the DWST may be required for long term heat removal. Technical Specifications 3.7.1.3, "Demineralized Water Storage Tank" ensures that at least a 13 hour inventory is available. In the longer term, the AFW pumps can be aligned to the condensate storage tank (CST).Travel Route 8 reflects the travel route to manual valve 3FWA*HCV37 which is used to realign pump 3FWA*P2 to the CST.

Thus, the 8 hour to 24 hour category is assumed. This action is performed on the 21 foot elevation of the ESF building.

11. Reset MCC breakers for Diesel Generator keep warm systems

This action is taken when off site power is available and the running diesel generator is stopped. The keep warm system assures that the diesel generator would be maintained in the optimum condition for a subsequent start if a loss of off site power occurs later in the transient. This action is performed in the emergency diesel generator building.

In addition to the areas and activities defined above, the Control Room and Technical Support Center (TSC) require post-accident access and continued occupancy as discussed in NUREG-0737.

Post-accident control room habitability is discussed in Section 6.4. The post-accident dose consequences for the Control Room are presented in Table 15.0-8.

The potential radiation doses to a person occupying the TSC have been evaluated for the Unit 3 LOCA. The TSC is designed for continuous operation for the duration of the accident (i.e., 30 days). The building roof and walls provide adequate shielding to protect the occupants against direct radiation from the external radioactive cloud and from the containment during the postulated LOCA. Double vestibule doors are provided at the building entrance to minimize inleakage due to personnel ingress/egress. The TSC ventilation system is described in Section 9.4.13. The evaluated 30 day integrated dose for an individual occupying the TSC following the DBA is within the NUREG-0737 criteria of 5 rem whole body dose or equivalent.

It has been determined that post-accident access to areas addressed in NUREG-0737, which have not been identified above, is not required for Millstone 3.

Table 12.3-4 provides an estimate of the anticipated times after a LOCA that vital area access is required, with consideration given to the typical 30 minute minimum time frame assumed for operator action outside control room and the X/Q intervals assumed in the FSAR Chapter 15 accident analysis. Outside travel routes are shown on Figure 12.3-10 and are listed on Table 12.3-3. A general description of the ingress travel routes, primary and alternates, are described below (the egress path is the same as the ingress path except for alternate routes to the backup Chemistry Laboratory in travel route 4).

Travel route 1: The primary route is from the control building through the service building to the auxiliary building (no outdoor travel). The alternate route is from the control building to the service building to the exit between the service and auxiliary building to the north entrance to the auxiliary building.

Travel route 2: The primary route is from the control building through the service building to the exit between the service building and the auxiliary building, along the north side of the waste disposal building then south to outside of the ESF Building, (inside the Radioactive Materials Area fence). The alternate route is from the control building to the service building to the turbine building to the RR loading area, east along the roadway past the RWST to the outside of the ESF building.

Travel route 3: The primary route is from the control building through the service building corridor leading to the roadway beside the MSV building to the RCA gate south of the hydrogen recombiner building (HRB) and into the HRB. The alternate route is from the control building to the service building to the turbine building to the RR loading area to the RCA gate adjacent to the HRB.

Travel route 4: The primary route is from the control building through the service building corridor leading to the roadway beside the MSV building to the RCA gate south of the hydrogen recombiner building (HRB) and into the HRB. The alternate route is from the control building to the service building to the turbine building to the RR loading area to the RCA gate adjacent to the HRB. The sample analysis is performed in the MP3 chemistry lab which is on the egress path. Figure 12.3-10, sheet 4, provides two additional routes for sample analysis in the MP1/MP2 service building.

Travel route 5: The primary route is from the control building through the service building to the exit between the service building and the auxiliary building, following the roadway north and east of the waste disposal building, then entering the fuel building. The alternate route is from the control building through the service building corridor to the turbine building to the RR loading area, then east along the roadway past the RWST, north to the fuel building.

Travel route 6: The primary route is from the control building to turbine building 38 foot level. No alternate is given since doses would only increase.

Travel route 7: The primary route is from the control building to the emergency diesel generator building. No alternate is given since doses would only increase with any other route.

Travel route 8: The primary route is from the control building through the service building to the exit between the service and auxiliary buildings, along the north side of the waste disposal building, south to the ESF building or the turbine driven Auxiliary Feedwater Pump Room. The alternate route is from the control building through the service building corridor to the turbine building to the RR loading area, then east along the roadway past the RWST and into the ESF building or the turbine driven Auxiliary Feedwater Pump Room.

Travel route 9: The primary path is from the control building through the service building corridor to the turbine building auxiliary bay, lower level then across the road to the auxiliary building. The alternate path is from the control building to the service building, past the Chemistry Laboratory, exit the service building to the auxiliary building.

The following general assumptions and criteria are used as a basis for review of all vital areas and access routes as applicable:

1. The starting point for all activities is the Unit 3 Control Building.

2. In order for an access/egress pathway to be considered acceptable, the total dose for activities required for mitigation of the design basis accident (which includes the dose to perform the activity and the associated transit dose) must be no greater than the 10 CFR Part 50 Appendix A GDC-19 or 10 CFR 50.67 dose criteria. The determination of total dose is based on the earliest time post-LOCA when access to the designated vital area is required as identified in Table 12.3-4.

3. All calculated outside pathway doses are assumed to be comprised of contributions from (a) containment radiation (both direct shine and skyshine contributions) and (b) direct radiation from the overhead plume.

Gaseous and liquid LOCA source terms used in the review are not less than that stated in NUREG-0737, Section II.B.2, which provides the minimum source terms to be used for evaluation of the adequacy of radiation protection to the operators.

To determine post-accident doses to personnel for performance of and transit to identified activities, the following sources of radiation are considered.

1. Auxiliary Building

- Radiation from containment atmosphere shining through electrical penetrations.

- Radio iodine buildup in the SLCRS filter.

- Sump water in the safety injection system piping located below the elevation 24 foot 6 inches floor.

- Containment atmosphere shine through the personnel hatch and surrounding walls and floors.

- Sump water in safety injection and charging system piping and associated shine through walls and floors.

2. Fuel Building

   - Direct shine from containment

   - Plume shine

   - Shine from the RHR heat exchanger in the EFS building.

   - Shine from the fuel pool cooling pumps.

3. ESF Building

   - Shine from RSS and SIH piping

   - Shine through the wall from the Recirculation Coolers

   - Shine from RWST piping

   - Shine from Auxiliary Steam piping

4. Along routes from control building to the vital areas.

   - Skyshine from containment

   - Direct shine from containment.

   - Plume shine.

   - Direct shine from the RWST.

Systems containing sources of radiation which are identified in NUREG-0737 but which have not been identified in buildings discussed above are considered to be either irrelevant following an accident or negligible contributors to personnel exposure following an accident. For example, the GWS system is a negligible contributor of radiation following an accident because when an accident occurs, the only use of this system is for post-LOCA hydrogen purge as the result of a beyond design basis event.

The results of the dose calculations indicate that the plant shielding and design provide adequate protection to operators following a design basis LOCA to ensure compliance with the NUREG-0737 design dose requirements.

12.3.2 FACILITY DESIGN FEATURES

The Millstone 3 design is consistent with the guidance presented in Regulatory Guide 8.8, Revision 4, C2, which discusses specific features in the facility and equipment design that limit radiation exposure to levels that are ALARA. The following features have been incorporated.

12.3.2.1 Location and Design of Equipment to Minimize Service Time

In the auxiliary building, nonradioactive equipment, such as the reactor plant component cooling system and components used to process the waste evaporator distillate, are located outside high radiation cubicles in areas designated as Radiation Zones II or III (defined in Table 12.3-1). In the containment structure, nonradioactive equipment requiring servicing is typically located in Radiation Zone IV areas. Exceptions include those components attached to the reactor coolant system, such as the reactor coolant pump motor cooling equipment and the equipment support snubbers.

Major radioactive components which may require servicing are typically located in individually shielded cubicles. These cubicles are designed such that radiation contributions from adjacent cubicles is small compared to sources within the cubicle. The resultant dose rate in any cubicle in which equipment is being serviced is due to sources within the cubicle to radiation penetrating through shield walls from adjacent cubicles, and to radiation streaming through shield wall penetrations. The design basis for shield walls enclosing cubicles containing process equipment is discussed in Section 12.3.1. Shield wall arrangement and dimensions are shown for the Containment Building, Figure 3.8-60; Auxiliary Building, Figure 3.8-62; Fuel Building, Figure 3.8-63; and Waste Disposal Building, Figure 3.8-74.

Cubicle access openings generally incorporate a labyrinth design which precludes direct radiation shine. The openings are sized to allow for removal and replacement of minor fluid system components such as pumps and valves, as well as to provide access for maintenance equipment. For example, pump cubicle openings for all horizontal pumps except the charging pumps are of sufficient size to skid such pumps through the entranceway. The openings of other cubicles containing equipment requiring servicing are sized to allow the passage of components while still maintaining radiation safety conditions. Cubicles are sized to allow sufficient clearance around equipment for laydown of equipment and installation of temporary shielding as needed. The equipment service requirements for pull space and laydown space are provided within each cubicle, thus eliminating the need for dismantling of piping other than that directly connected to the equipment.

The corridor system is sized to allow hand cart and dolly access. Motor terminal boxes and other terminal boxes are located so as not to block access, and are separated from radioactive piping if possible. Platforms for servicing specific components are provided where necessary.

Certain components have design features which minimize service time. For example, the reactor coolant pump design includes an assembled cartridge seal which results in reduced time required for replacement. The cartridge seal is also expected to have a useful life which is double that of the older designs. The reactor coolant pump design also includes a spool piece to facilitate separation and replacement of the motor from the pump.

The reactor vessel nozzle welds insulation is fabricated in sections with a thin reflective metallic sheet covering and quick disconnect clasps to facilitate removal for inspection of the welds.

Typically, filters are designed to be removable from the top with lifting bails in the middle of the head. The filter assemblies usually have bolt lead-ins for tool entry, and the filters are contained in disposable cartridge assemblies. These features facilitate remote removal, disposal, and assembly.

The head closure system provided for Millstone 3 includes quick disconnect/connect stud tensioners which have quick-acting, hydraulically-operated stud gripper devices, as opposed to conventional tensioners which must be threaded onto the tops of the studs. Also provided are air-motor driven stud removal tools which can rapidly remove (or insert) the studs, in contrast to the much slower manual stud removal and insertion tools used in older designs. The stud tensioners are designed to operate simultaneously, as are the stud removal tools.

The primary system heat exchangers are designed such that the shell-to-tube sheet joint need not be broken for inspection. The shell and tube assembly can be lifted intact above the channel head to expose the tube ends for inspection and leak testing.

Pumps are typically designed with flanged connections to facilitate removal for maintenance. Depending on expected conditions, either canned pumps or pumps with high quality mechanical seals are used to reduce leakage and maintenance requirements.

12.3.2.2 Location of Instruments Requiring In Situ Calibration

Instruments which require in situ calibration are located, wherever possible, on exterior walls of shielded cubicles to minimize exposure of instrumentation and personnel. Instruments which cannot be located in this manner are located in the lowest practicable radiation area in the cubicles and are provided with convenient access. Where practical, instruments are designed for removal to low radiation areas for calibration and maintenance.

12.3.2.3 Location of Equipment Requiring Servicing in Lowest Practicable Radiation Field (or Movable to Lowest Practicable Radiation Field)

As indicated above, radioactive equipment requiring servicing is typically located in shielded cubicles with access openings sized for ease of equipment removal. As an example, pump cubicles are designed to allow removal of the pump to the lowest practicable radiation field.

Westinghouse has designed the Model F steam generators to reduce the radiation exposure during both normal operation and maintenance. The tube ends are designed to be flush with the tube sheet in the steam generator channel head to eliminate a potential crud trap. The steam generator manways (entrance to channel head) are sized to facilitate entrance and exit with protective clothing. Handholes to the secondary side are positioned to facilitate maintenance operations. Changes to increase steam generator reliability also reduces occupational radiation exposures. Such changes include improved steam generator tube support plates (stainless steel and quatrefoil flow holes) and the use of all-volatile treatment chemistry on the secondary side.

12.3.2.4 Valve Location and Selection

Valves are located in separate shielded valve cubicles or areas outside equipment cubicles to the greatest extent practicable to minimize maintenance exposure. Valve selections are usually based on “best product” available and maintenance time required. Westinghouse has supplied valves of the bolted body-to-bonnet forging type. This permits the use of ultrasonic testing in place of radiography for inspection and facilitates assembly and disassembly, resulting in reduced inspection and maintenance time. Additionally, manual valves under 2 inches in diameter are designed for zero stem leakage.

12.3.2.5 Penetrations of Shielding and Containment Walls by Ducts and Other Openings

There are numerous piping penetrations through shield walls in the auxiliary building which are directed into adjacent cubicles, into the pipe chases for radioactive piping, and into the corridors for nonradioactive piping. To the greatest extent practicable, penetrations through walls separating higher radiation zone areas from lower radiation zone areas are located above head level, in corners, and in positions which are offset from radiation sources in the higher radiation zone cubicles. This prevents line-of-sight radiation streaming from significant radiation sources to personnel working in adjacent cubicles. Noteworthy examples of this practice are provided as follows.

1. Electrical penetrations through shield walls are made to prevent direct line-of-sight to any significant radiation sources.

2. Instrument tubing penetrations through shield walls are made so as to prevent direct line-of-sight to any significant radiation sources.

3. Ventilation duct penetrations through shield walls are made at the highest possible elevation and at locations which minimize direct line-of-sight to significant radiation sources. Where direct line-of-sight penetrations through cubicle walls are unavoidable, penetration shields are often employed either inside or outside such cubicles.

12.3.2.6 Radiation Sources and Occupied Areas

Radiation sources (Section 12.2) are separated, as far as is practicable, from normally occupied areas by shield walls and cubicles. Piping runs are also located as far as practicable from equipment cubicles. Radioactive piping (e.g., process piping carrying radioactive materials) is typically located behind shielding and also routed around, rather than through, normally occupied areas wherever practicable. Valves are located in shielded valve areas where practical and are separated from equipment cubicles, pipeways, and areas of general access.

Physically locked barriers (i.e., locked doors) are provided for areas having radiation levels in excess of criteria as specified in the Technical Specifications.

12.3.2.7 Minimizing Spread of Contamination and Facilitation of Decontamination Following Spills

Typically sources of contamination from leaks or spills from components located in cubicles are prevented from spreading by cubicle entrance dikes and/or low point drains to enclosed collection sumps. Floor surfaces and walls are sealed or painted as required with a protective chemically-resistant coating to provide a surface which is easily decontaminated. Demineralized water hose stations are provided throughout the auxiliary building to allow flush water to be available to each cubicle in the auxiliary building. Systems containing radioactive fluids are usually fabricated of corrosion-resistant materials.

Airborne contamination is kept from spreading by ventilation systems which are described in Section 12.3.3. A personnel decontamination area is located in the radiation protection area. An equipment decontamination area is located in the waste disposal building.

12.3.2.8 Piping to Minimize Buildup of Contamination

Interior surfaces of systems in radioactive liquid service typically are made of stainless steel or other corrosion-resistant material.

The piping associated with these systems is normally routed to avoid sharp bends by carefully selecting the elevation between points and by attempting to run this piping at no more than two elevations between these points. Pockets and low points are also avoided. Pipe runs for spent resin sluicing are provided with large radius bends rather than welded elbows to prevent accumulation of resin fines and crud particles. Resin piping is also butt-welded where possible to minimize the potential for crud particles. Valve stations are designed to minimize the buildup of crud by minimizing the number of pockets and stagnant vertical legs.

Ventilation design features to minimize radioactive contamination buildup are discussed in Section 12.3.3.

12.3.2.9 Flushing or Remote Chemical Cleaning of Contaminated Systems

Means for flushing and draining of potentially highly radioactive tanks, lines, and other components are considered in fluid system design. Waste collection tank design includes provisions for internal flushing with spray nozzles to remove potential collections of particulate material. All heat exchangers are provided with chemical cleaning connections which are connected prior to servicing.

Flushing and vent connections are provided to allow flushing of piping systems for maintenance.

12.3.2.10 Ventilation Design

The ventilation systems are designed with sufficient capacity to control airborne radioactivity releases and concentrations during normal and maintenance conditions. The ventilation flow through equipment cubicles is based upon unrestricted air flow from general access areas into these cubicles. The design of ventilation systems typically ensures a positive flow from non-contaminated areas to potentially contaminated areas to prevent the spread of airborne radioactivity and to exhaust from the potentially contaminated areas. A more explicit description of ventilation systems is given in Section 12.3.3.

12.3.2.11 Radiation and Airborne Contamination Monitoring

Area and airborne radiation monitoring ensures that any substantial abnormal radioactivity release is promptly detected. The area and airborne radiation monitoring system is described in more detail in Section 12.3.4 and Section 11.5, respectively.

12.3.2.12 Temporary Shielding

The use of temporary shielding to facilitate maintenance tasks is considered on a case-by-case basis. Convenient means for transport and placement of such shielding are provided by access corridors and elevators in the auxiliary building and an elevator in containment.

12.3.2.13 Solid Waste Shielding

As shown on Figures 12.3-1 through 12.3-4, radioactive wastes in tanks, evaporators, process gas charcoal bed adsorbers and associated equipment are located in shielded cubicles. Solid waste is shielded both by the storage area walls in the waste disposal building and by individual transportation shields.

12.3.2.14 Remote Handling Equipment

As noted in Section 12.3.2.1, filters are designed for remote removal, disposal, and assembly. Equipment is provided for filter handling as well as for remote removal and replacement of ion exchange resins.

Stations for potentially radioactive system valves are, in general, arranged either in segregated shielded cubicles away from the equipment served and/or are provided with reach rods. The demineralizer and filter valves are in cubicles beneath the vessels and are also provided with reach rods.

12.3.2.15 Maximum Expected Failures of Fuel Element Cladding and Steam Generator

Design features such as shielding and radiation zones accommodate 1 percent fuel defects and primary to secondary steam generator tube leaks of 1,370 lb per day.

12.3.2.16 Sampling Stations

Sample points are provided with sample sinks and ventilation hoods, splash screens, and valves located outside each splash screen. Samples are provided with recirculation paths behind shield walls at sample sinks, with reach rods for operators.

12.3.2.17 Cobalt Impurity Specifications

Cobalt weight percentages for materials in contact with reactor coolant are considered in purchase specifications.

12.3.2.18 Reactor Cavity Filtration System

During refueling, the reactor cavity water may become turbid, making it difficult to observe the removal and replacement of fuel assemblies. The portable reactor cavity filtration system, consisting basically of a pump and four filters, provides capability for cleanup of this water, thus minimizing the time required for, and dose due to, fuel and equipment handling operations.

12.3.3 VENTILATION

This section provides the basis for the original plant ventilation design. Although current airborne levels may not be consistent with the tables in this chapter, these tables are not being changed to be current as that would make them inconsistent with the design basis criteria for the ventilation systems. Recent radiation protection surveys should be consulted for information on current radiological conditions.

12.3.3.1 Design Objectives

The function and design bases of the ventilation systems are given in Section 9.4. Consistent with these, the following specific objectives pertain to radiation protection and the commitment that occupational radiation exposures are ALARA, in accordance with Regulatory Guide 8.8.

1. The airborne radioactivity concentrations from radioactive sources released into the fuel building and turbine building, as shown in Table 12.2-11, are small fractions of values in Column 1, Table 1 of 10 CFR 20, Appendix B. Radwaste piping system and process components in the auxiliary building and the waste disposal building are separated from normally accessed areas by walls, and are provided with ventilation systems which supply air from clean, occupied areas and exhaust from duct openings located within the process system cubicles. Ultimately, routine plant surveys by plant radiation protection personnel provide appropriate controls and protective measures described in Section 12.5.3 when access is needed to areas which are not normally occupied.

2. Concentrations in areas accessible to administrative personnel are less than 25 percent of the concentrations given in Column 1, Table II of Appendix B to 10 CFR 20.

3. The airborne concentrations in all plant areas are ALARA.

4. The containment atmosphere filtration system, with only one of its two 12,000 cfm fan units in operation, is capable of reducing the airborne iodine concentration in the containment atmosphere to below 1 EC of I 131 in less than 16 hours of filter operation under the conditions of expected reactor coolant radioactivity concentration and leakage described in NUREG-0017.

5. The containment purge air system is capable of reducing airborne radiation levels in the containment to acceptable levels prior to and during extended personnel occupancy of the containment.

6. The fuel building ventilation system operates in the once-through mode without recirculation with the provision to exhaust through charcoal filters.

7. Typically, air flow within the auxiliary, waste disposal, and fuel buildings during normal operation is from areas of lower to higher potential airborne contamination and then to monitored vents with provisions for terminating or filtering the ventilation flow upon a high radioactivity alarm.

8. Systems are designed so that filters containing radioactivity can easily be maintained to minimize the radiation dose to personnel.

12.3.3.2 Design Description

Detailed descriptions of the ventilation systems for the plant buildings which contain radioactivity or potentially radioactive systems are given in the following sections:

Section	Title
9.4.1	Control room area ventilation system
9.4.2	Spent fuel pool area ventilation system
9.4.3	Auxiliary building area ventilation system
9.4.5	Engineered safety feature ventilation system
9.4.4	Turbine building area ventilation system
9.4.7	Containment ventilation
9.4.9	Waste disposal building ventilation system
9.4.10	Main steam valve building ventilation system
9.4.11	Hydrogen recombiner building ventilation
9.4.13	Technical Support Center building ventilation system

12.3.3.3 Personnel Protection Features

The recommendations of Regulatory Guide 1.52, as described in Section 1.8, are implemented in the design of the safety-related ventilation filter trains to help assure that occupation radiation exposures from service of these trains are ALARA.

This is accomplished by utilizing the following criteria.

1. Each filter train is housed in a shielded compartment, room, or cubicle except for the control building filters which occupy a common cubicle with the air conditioning unit.

2. Adequate aisle space is provided for both personnel and equipment adjacent to the service side of the filter trains, and above those sections which require top access (i.e., charcoal adsorber).

3. Convenient and accessible passageways and corridors from the filter trains to the elevators and equipment hatches are provided for transport of replaceable filter train components and the equipment used in accomplishing their replacement.

4. Replaceable elements, except for most downstream HEPA filters, are designed for ready removal from the clean filter side, and minimal radiation exposure of personnel. A portable cart-mounted vacuum conveying system is provided for draining and recharging gasketless-type charcoal adsorbers. Contaminated filters can be transported in shielded containers if necessary.

5. Rigid, hinged access doors are provided in accordance with ANSI N 509 for man-entry filter trains.

6. HEPA and prefilter arrangements are no more than 3 elements high to facilitate easy replacement without the use of ladders, temporary scaffolds, or platforms.

7. A minimum of 2.5 linear feet is maintained from mounting frame to mounting frame between banks of components for removal of filter elements.

8. Adequate vapor-tight lighting is provided on each side of the filter banks for man-entry filter trains.

9. (Deleted)

10. Drains are provided to convey water from moisture separators, maintenance or fire protection discharge out of the filter train.

11. Permanent test fittings are provided for initial and periodic field testing.

Filter train arrangement is discussed in Section 6.4.

12.3.3.4 Radiological Evaluation

Concentrations of airborne activity for the expected and design conditions in the containment structure, turbine building, and fuel building are tabulated in Table 12.2-11. The concentrations based upon design conditions are expected to envelope anticipated operational occurrences. The airborne concentrations are averages based on assumed total leak rates described in NUREG-0017 and the ventilation rates for the respective buildings. Corridors and areas normally occupied by operating personnel are expected to have negligible airborne activity concentrations since clean air ventilation flow is typically directed from areas with less potential for contamination (manned areas) to areas with greater potential for contamination.

Equipment cubicles are the most likely areas for airborne concentrations but are not normally occupied or accessible without prior survey and control. For purpose of quantification, the worst airborne concentration could conceivably exist in cubicles for which the combination of relatively high system volatile radionuclide concentrations and low cubicle ventilation rate would simultaneously exist for a given leak rate. A cubicle such as the letdown heat exchanger cubicle in the auxiliary building could develop airborne concentrations of approximately 7×10^{-5} Ci/cc assuming all the design basis leak rate takes place in that cubicle for expected coolant radioactivity concentrations.

Based on the above assumption, it is expected that other cubicles would have airborne concentrations of less than 7×10^{-5} Ci/cc.

Section 12.2 includes the models and parameters used as a basis for calculated radioactivity values.

12.3.4 AREA RADIATION AND AIRBORNE RADIOACTIVITY MONITORING

12.3.4.1 Purpose

The area radiation monitoring system (RMS) works in conjunction with the process, effluent, and airborne radiation monitoring group (Section 11.5). Its purpose is to protect plant personnel by measuring levels of radiation in various areas of the plant. It also provides a warning to operations of abnormal radiological conditions. If high radiation levels are monitored, the system sounds an alarm. It also produces a record of radiation levels.

12.3.4.2 System Design

The basis of the RMS area radiation monitoring group is the single channel, GM tube or ion chamber detector equipped with a dedicated microprocessor except the containment high range monitors which are analog and in designated cases, a rate meter. The microprocessor provides local display and control functions for the detector, computes and stores time-averaged detector outputs, stores all necessary operating parameters (e.g., alarm trip values), and also handles all communication between the detector and the RMS computer system. The rate meter, where provided, is located adjacent to the detector and provides local analog display. A high activity level is indicated by both audible and visible alarms which may be acknowledged at the rate meter. Area radiation monitors are located in normally accessible areas where changes in plant conditions could cause significant increases in personnel exposure rates in accordance with design criteria established in ANSI/ANS HPSSC-6.8.1-1981.

The RMS computer system provides centralized display and control, at the control room RMS workstations. A dual server based computer system, located in the control building, polls each monitors microprocessor every several seconds to obtain the latest readings, and to register any alarms present. All alarms are displayed on the control room RMS workstation and can be printed. Radiation alarms are annunciated both in the control room and if equipped with local indication locally at the microprocessor, and can be acknowledged at either location. The system operator also uses the RMS workstations to either output data from individual monitors or input commands to these non-Class 1E monitors. All commands sent are recorded in the message summary log. The last (30) 1 minute, 10 minute and hourly averages are stored and available for review at the RMS workstation for all radiation monitors except the containment high range. An RMS workstation is located in the radiation protection Office.

In addition, those monitors designated Class 1E except that containment high range monitors are connected directly to one of two control room 1E cabinets. The output of each monitor is digitally displayed and also recorded. A remote indication and control module (RIC) is furnished in the cabinets for each 1E monitor. The RIC handles all remote control functions for 1E monitors. The containment high range monitors are displayed on the 1E control room cabinets and recorded on the plant process computer. The 1E cabinets are connected by electronic isolators to the RMS computers to allow data from the 1E monitors to be displayed on the control room RMS workstations and to be written into the RMS computer.

The area RMS is calibrated both by a standard factory calibration and by onsite calibration. Factory calibration included checks for linearity and energy response. Sources traceable to national standards are used. Onsite calibration includes detector response using sources of known energy and strength. The frequency of onsite calibration of safety-related monitors is provided in the Technical Specifications.

Table 12.3-2 gives the mark numbers, names, locations, and ranges of the area monitors in the RMS. The following paragraphs provide a brief description of the different types of area monitors.

12.3.4.3 Class 1E Area Monitors

Four of the area monitors are designated Class 1E. (Section 8.3.1.1.2 discusses Class 1E power and its backup supply.) These are the two redundant fuel drop monitors and the two containment internal high range monitors. They differ from most other area monitors in that they use ion chamber detectors instead of GM detectors. The high range monitors are capable of withstanding a design basis accident inside containment. The fuel drop monitors are designed to operate in the normal containment environment and are discussed in Section 11.5.

The containment internal high-range monitors are located on the inside face of the annulus wall approximately 180° apart. Range is 1 to 10^8 R/hr.

These monitors are qualified based on the requirement of IEEE 323-1974, 279-1971, and 344-1975.

Due to the normally high operational dose rates inside containment, the dedicated microprocessors for all detectors located inside the containment structure are located within the auxiliary building.

As with the Class 1E process monitors (Section 11.5), the outputs of these devices are displayed and recorded on the control room Class 1E panels. The display and recording of the containment high range monitor is as required by Regulatory Guide 1.97.

12.3.4.4 Non-Class 1E Area Monitors

Non-Class 1E area monitors measure and transmit local radiation levels, and annunciate an alarm upon a high radiation level. All monitors except the hydrogen recombiner control room have analog display rate meters located adjacent to the detectors and are powered from normal AC power (Section 8.3.1.1.1). The hydrogen recombiner control room monitor is powered by inverted normal DC power. Their purpose is to protect plant personnel from excessive exposure and to provide a display radiation levels within the plant.

12.3.4.5 Airborne Radioactivity Monitoring

The process and effluent radiation monitoring system described in Section 11.5 includes normal range particulate and gas monitors. Their purpose is to monitor airborne radioactivity in areas that may be occupied by plant personnel and to facilitate finding radioactive leaks. These monitors sample air from the reactor containment, the ESF building, the control room, and from locations in the reactor plant heating and ventilation system upstream of the ventilation vent monitor. They are capable of detecting airborne activated corrosion products and fission products at levels below the derived air concentration of 10 CFR 20.

Both the particulate and gas detector channels of these monitors are provided with an “alert level” alarm, in addition to the “high” alarm, with setpoints established by operating personnel to allow observation of increases in airborne radioactivity. These monitors are polled every several seconds by the radiation monitoring computer system. Once elevated readings are noticed, personnel with portable air samplers can determine which area, associated with the ventilation stream with high radioactivity, contains the source of the problem.

12.3.5 REFERENCES FOR SECTION 12.3

12.3-1 NUREG-75/087, USNRC. Standard Review Plan, Revision 1.

12.3-2 Regulatory Guide 1.52, USNRC. Design, Testing, and Maintenance Criteria for Post Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants, Revision 2.

12.3-3 Regulatory Guide 1.69, USNRC. Concrete Radiation Shields for Nuclear Power Plants.

12.3-4 Regulatory Guide 1.70, USNRC. Standard Format and Contents of Safety Analysis Reports for Nuclear Power Plants, Revision 3.

12.3-5 Regulatory Guide 1.97, USNRC. Instrumentation for Light-Water-Cooled Nuclear Power Plant to Assess Plant and Environs Conditions During and Following Accident, Revision 2. (Compliance provided in a separate report, Section 1.7.4.)

12.3-6 Regulatory Guide 8.8, USNRC. Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low as is Reasonably Achievable, Revision 4.

12.3-7 Stone & Webster Engineering Corporation (SWEC) 1975. Radiation Shielding Design and Analysis Approach for Light Water Reactor Power Plants, RP-8A, May 1975.

TABLE 12.3–1 RADIATION ZONES

Zone Designation	Zone Description	Maximum Allowable Dose Rate* (mRem/hr)
I	Unrestricted area - Continuous access	≤ 0.25
II	Unrestricted area - Periodic access - 40 hrs/wk	≤ 2.5
III	Restricted area - Controlled periodic access - 6 hrs/wk	≤ 15
IV	Restricted area - Controlled infrequent access - 1 hr/wk	≤ 100
V	High radiation area - Not normally accessible	> 100

NOTE:

* Based upon the 5 rem per year criterion given in 10 CFR 20 and the maximum personnel occupancy time corresponding to each radiation zone.

TABLE 12.3–2 RADIATION MONITORING SYSTEM - AREA RADIATION DETECTOR LOCATION

Mark Number	Name	Location	Range mr/hr
3RMS-RE01	Refueling Machine	Containment – 51 feet 4 inches	$1\text{-}10^5$
3RMS-RE02	Fuel Transfer Tube	Containment – 3 feet 8 inches	$1\text{-}10^5$
3RMS-RE03	Incore Inst. Transfer Area	Containment – 24 feet 6 inches	$1\text{-}10^5$
3RMS*RE04A	Containment High Range Internal	Containment – 52 feet 4 inches	$10^3\text{-}10^{11}$
3RMS*RE05A	Containment High Range Internal	Containment – 51 feet 4 inches	$10^3\text{-}10^{11}$
3RMS-RE06	Decontamination Area	Fuel – 24 feet 6 inches	$0.1\text{-}10^4$
3RMS-RE07	HVAC Area	Auxiliary – 66 feet 6 inches	$0.1\text{-}10^4$
3RMS-RE08	Spent Fuel Pit Bridge/Hoist	Fuel – 52 feet 4 inches	$0.1\text{-}10^4$
3RMS-RE09	Auxiliary Bldg General (A)	Auxiliary – 18 feet 6 inches	$0.1\text{-}10^4$
3RMS-RE10	Auxiliary Bldg General (B)	Auxiliary – 4 feet 6 inches	$0.1\text{-}10^4$
3RMS-RE11	Auxiliary Bldg General (C)	Auxiliary – 4 feet 6 inches	$0.1\text{-}10^4$
3RMS-RE12	Auxiliary Bldg General (D)	Auxiliary – 24 feet 6 inches	$0.1\text{-}10^4$
3RMS-RE13	Auxiliary Bldg General (E)	Auxiliary – 24 feet 6 inches	$0.1\text{-}10^4$
3RMS-RE14	Auxiliary Bldg General (F)	Auxiliary – 24 feet 6 inches	$0.1\text{-}10^4$
3RMS-RE15	Auxiliary Bldg General (G)	Auxiliary – 43 feet 6 inches	$0.1\text{-}10^4$
3RMS-RE16	Volume Control Tank Cubicle	Auxiliary – 43 feet 6 inches	$0.1\text{-}10^4$
3RMS-RE17	Waste Disposal Bldg (A)	Waste Disposal – 4 feet 6 inches	$0.1\text{-}10^4$
3RMS-RE18	Waste Disposal Bldg (B)	Waste Disposal – 4 feet 6 inches	$0.1\text{-}10^4$
3RMS-RE19	Solid Waste Storage Area	Waste Disposal –24 feet 6 inches	$0.1\text{-}10^4$
3RMS-RE20	Sample Room	Auxiliary – 43 feet 6 inches	$0.1\text{-}10^4$
3RMS-RE21	Laboratory	Service – 24 feet 6 inches	$0.01\text{-}10^3$
3RMS-RE22	Control Room	Control – 47 feet 6 inches	$0.01\text{-}10^3$

TABLE 12.3–2 RADIATION MONITORING SYSTEM - AREA RADIATION DETECTOR LOCATION

Mark Number	Name	Location	Range mr/hr
3RMS-RE24	Waste Disposal Bldg (C)	Waste Disposal – 4 feet 6 inches	$0.1\text{-}10^4$
3RMS-RE25	Waste Disposal Bldg (D)	Waste Disposal – 4 feet 6 inches	$0.1\text{-}10^4$
3RMS-RE28	Fuel Building Pipe Rack	Fuel – 11 feet 0 inches	$0.1\text{-}10^4$
3RMS-RE29	Spent Fuel Cask Area	Fuel – 52 feet 4 inches	$0.1\text{-}10^4$
3RMS-RE31	Fuel Transfer Tube	Containment – 24 feet 6 inches	$0.1\text{-}10^4$
3RMS-RE32	Containment Sump Area	Containment – (-24 feet 6 inches)	$0.1\text{-}10^4$
3RMS-RE33	RHR Cubicle “A” Normal Range	ESF – 4 feet 6 inches	$0.1\text{-}10^4$
3RMS-RE34	RHR Cubicle “B” Normal Range	ESF – 4 feet 6 inches	$0.1\text{-}10^4$
3RMS-RE35	Incore Inst. Thimble Area	Containment – 3 feet 8 inches	$0.1\text{-}10^4$
3RMS-RE36	Fuel Pool Monitor	Fuel – 52 feet 4 inches	$0.1\text{-}10^4$
3RMS-RE37	Condensate Demineralizer	Cond. Polishing – 14 feet 6 inches	$0.01\text{-}10^3$
3RMS-RE38	Regeneration Area	Cond. Polishing – 38 feet 6 inches	$0.01\text{-}10^3$
3RMS*RE41	Fuel Drop Monitor	Containment – 51 feet 4 inches	$10^1\text{-}10^8$
3RMS*RE42	Fuel Drop Monitor	Containment– 51 feet 4 inches	$10^1\text{-}10^8$
3RMS-RE52	Recombiner Control Room	HRB– 24 feet 6 inches	$10^1\text{-}10^8$

TABLE 12.3–3 OPERATOR ACTIVITY LOCATIONS AND TIME DURATIONS

	Activity	Location	Approx. Duration (minutes)	Travel Route*
1	Locally trip the reactor trip breakers and bypass breakers	43 feet 6 inches Auxilary Building (MCC Rod Control)	< 5	9
2	Deleted			
3	Local actions needed to realign Spent Fuel Pool Cooling, RBCCW and Service Water for spent fuel pool cooling	Spent Fuel Building	< 15	5
4	Powering the Plant Process Computer	38 feet Turbine Building	< 10	6
5	Powering the SI accumulator valves	24 feet 6 inches Auxilary Building	< 5	9
6	Initiate hydrogen monitor	HRB	< 30	3
7	Deleted			
8	Deleted			
9	Deleted			
10	Monitor and maintain the porous concrete groundwater removal system			
	a. Repower sump pump	Outside of ESF Building	< 90	2
		ESF Building 38 feet 6 inches	< 10	2
	b. Monitor sump pump and operate system	Outside of ESF Building.	< 2	2
	c. Maintain/repair sump pump	ESF Building roof	< 240	3

TABLE 12.3–3 OPERATOR ACTIVITY LOCATIONS AND TIME DURATIONS

	Activity	Location	Approx. Duration (minutes)	Travel Route*
11	Open the breakers for the non-safety grade sump pumps in the ESF and Auxiliary buildings	21 foot ESF Building	< 15	2
		24 feet 6 inches Auxilary Building	< 15	1
		4 foot Service Building	< 15	
12	Deleted			
13	Align Alternate AFW Pump Suction Source or Replenish Demineralized Water Storage Tank (DWST) Inventory.	21 foot ESF Building	< 60	8
14	Reset MCC breakers for Diesel Generator keep warm systems	Emergency Diesel Generator Building	N/A **	7

* Figure 12.3-10 graphically depicts each route by route number.

** There are no appreciable dose rates in the Emergency Diesel Generator Building.

TABLE 12.3–4 ACTIVITY INITIATION TIME

Time Frame *	Activity	
0 to 30 minutes	1	Locally trip the reactor trip breakers and bypass breakers
	2	Deleted
	4	Powering the process computer
	6	Initiate hydrogen monitor
	14	Reset MCC breakers for Diesel Generator keep warm systems
30 minutes to 2 hours	5	Powering the SI accumulator valves
	7	Deleted
	9	Deleted
	12	Deleted
2 hours to 8 hours	3	Local actions needed to realign Spent Fuel Pool Cooling, RBCCW and Service Water for spent fuel pool cooling
	10	Repower porous concrete groundwater pump **
8 hours to 24 hours	10	Monitor porous concrete groundwater system
	13	Align Alternate AFW Pump Suction Source or Replenish Demineralized Water Storage Tank (DWST) Inventory
1 day to 4 days	10	Monitor and maintain the porous concrete groundwater system
	11	Open the breakers for the non-safety grade sump pumps in the ESF and Auxiliary Building
4 days to 30 days	8	Deleted
	10	Monitor and maintain the porous concrete groundwater system

* The starting time of the time frame listed is used for source term decay correction.

** Work must be completed between 2 to 6 hours.

12.4 DOSE ASSESSMENT

This section was applicable during the prestart-up period as it provided estimates of the occupational radiological consequences of future operation and the dose to workers during unit construction. Now that Millstone Unit 3 is operational, Annual Reports submitted to the NRC per Regulatory Guide 1.16 should be consulted for data on the station occupational person-rem requirements. Information on design dose rates in unrestricted areas has been moved to Section 12.3-1.

12.5 HEALTH PHYSICS PROGRAM

The regulatory guides and other references cited in this section were used as basis documentation for the development of the radiation protection program. Documented methods and solutions different from those set out in the guidance have also been incorporated in the radiation protection program.

12.5.1 ORGANIZATION

The radiation protection program is established to provide an effective means of radiation protection for permanent and temporary employees and for visitors at the station. To provide an effective means of radiation protection, the radiation protection program incorporates a philosophy from management (Section 12.1.1); employs qualified personnel to supervise and implement the program; provides appropriate equipment and facilities; and utilizes written procedures designed to provide protection of station personnel against exposure to radiation and radioactive materials in a manner consistent with Federal and State regulations (Section 13.5). The radiation protection program is developed and implemented through the applicable guidance of INPO 05-008, Regulatory Guides 8.2, Revision 1; 8.8, Revision 3; and 8.10, Revision 1-R.

The radiation protection department and line function management implement and enforce the radiation protection program. Dominion Corporate commitment to the radiation protection program is provided in DNAP-0100, "Dominion Nuclear Operations Standard."

The Radiation Protection Manager shall meet or exceed the qualifications specified in Regulatory Guide 1.8, Revision 1. The Site Vice President will designate the individual or position that will serve in the position of Radiation Protection Manager (RPM) that is required in the Administrative Section of the Technical Specifications for each Unit. Radiation protection technicians meet or exceed the qualifications specified in ANSI N18.1-1971. The radiation protection organization includes radiation protection operations, support and waste services.

The radiation protection department coordinates with all station, corporate, and contractor organizations to provide radiation protection coverage for all activities that involve radiation or radioactive material. The radiation protection department is organized to provide the following services:

1. preparation and implementation of radiation protection procedures for routine and nonroutine activities associated with the operation, maintenance, inspection, and testing at the station;

2. compliance with regulatory requirements for maximum permissible dose limits and contamination control;

3. maintenance of a personnel radiation dosimetry program and dosimetry records;

4. the surveying of station areas, maintenance of survey records, and the posting of survey results for daily activities within the station;

5. assistance in the station training program by providing specialized radiation protection training;

6. procurement, maintenance, and calibration of radiation detection instruments and equipment for assessment of the radiation areas;

7. procurement, maintenance, and issuance of protective clothing and equipment;

8. shipping, storage, and receiving of all radioactive material to assure compliance with regulatory requirements;

9. assistance in the decontamination of personnel, equipment, and facilities;

10. preparation, maintenance, and issuance of the required regulatory, station, and personnel reports that are associated with radiation or radiation exposure; and

11. preparation, maintenance, and implementation of the radiological respiratory protection program.

12. Ensure stop work authority when required by actual or potential radiological conditions.

The chemistry department is responsible for measuring the radioactive content of all gaseous and liquid effluents from the site in accordance with the requirements of the Technical Specifications, Radiological Effluent Monitoring and Offsite Dose Calculation Manual and 10 CFR 20.

It is a policy of the Millstone Power Station to keep personnel radiation exposure within the applicable regulations, and beyond that, to keep it as low as reasonably achievable.

12.5.2 EQUIPMENT, INSTRUMENTATION, FACILITIES

The criteria for purchasing the various types of portable and laboratory equipment used in the radiation protection and chemistry department is based on several factors. Portable survey and laboratory radiation detection equipment is selected to provide the appropriate detection capabilities, ranges, accuracy and durability required for the expected types and levels of radiation anticipated during normal operating or emergency conditions. Selection of respiratory protection equipment such as full-face masks, self-contained breathing apparatus, and respirator filters is made following the guidance of applicable approval regulations.

Radiation protection equipment, such as portable survey meters, is maintained by radiation protection. Survey equipment for use in emergency situations is stored in emergency kits which are located in such areas as the control room and the emergency operation facility. Special portable equipment, such as personnel air samplers, is available from radiation protection, and is utilized at the discretion of radiation protection supervision. Respiratory protection equipment is primarily stored at the respiratory storage and issue facilities.

Portable instruments for measuring radiation or radioactivity are used as required by 10 CFR 20, and by the provisions set forth in Regulatory Guide 1.97, Revision 2. Millstone 2 and 3 maintain a common inventory of hand-held radiation meters, electronic dosimeters, and National Voluntary Laboratory Accreditation Program (NVLAP) accredited individual monitoring devices. The Millstone Station radiation protection group maintains adequate supplies of hand-held radiation meters, secondary dosimeters, and NVLAP accredited dosimeters for normal station activities, multiple unit shutdowns, and/or potential accident conditions. The station will maintain an adequate supply of portable radiation protection instrumentation strategically located at the facility to ensure the radiation protection staff is properly equipped to perform their required functions. These instruments will be calibrated as specified by the manufacturers instructions and procedural requirements or as deemed necessary by radiation protection supervision. Calibration, operation, and maintenance procedures are followed for each specific type of instrument. Detailed records of calibration and maintenance of each instrument are maintained at the station. Calibrations are performed using radiation sources of known activity. These sources are calibrated or certified accurate by the National Institute of Standards and Technology (NIST). Calibration sources are stored by radiation protection. Actual calibration of equipment is performed in the calibration laboratories or other appropriate facilities.

The radiation protection group and chemistry group maintain appropriate laboratory instruments to perform the required radiological evaluations to support the station needs. Radiation protection or chemistry personnel check each counting system at regular intervals with standard radioactive sources to determine counting efficiencies, proper voltage settings, and background count rates. Records are maintained for each instrument or counting system. Repair and maintenance of laboratory equipment is performed by station personnel or through vendor repair contracts.

The Millstone site contains the following areas:

Unrestricted Area - access to which is neither limited nor controlled by the licensee.

Controlled Area - an area, outside a restricted area but inside the site boundary, access to which can be limited by the licensee for any reason.

Restricted Area - an area, access to which is limited by the licensee for the purpose of protecting individuals against undue risks from exposure to radiation and radioactive materials.

Radiologically Controlled Area/Radiological Control Area (RCA) - an area, posted with a sign by the licensee for the purpose of protecting individuals from exposure to radiation and/or radioactive materials. Dosimetry is always required within the RCA.

The Millstone Restricted Area generally corresponds to the area inside the protected area fence. Millstone, with three units, has a number of RCAs within the Restricted Area.

Within the RCA, Radiation Areas, High Radiation Areas, Technical Specification Locked High Radiation Areas, Very High Radiation Areas, Contaminated Areas, Airborne Radioactivity Areas, Radioactive Materials Areas, and Hot Particle Areas can be found.

A portal monitor and/or frisker and/or personnel contamination monitor is located at selected control points to detect the spread of radioactive contamination to the areas outside the RCA. At the discretion of radiation protection supervision, a personnel monitor or frisker is placed in specific areas at the station where contamination or the potential for contamination may be present.

Any areas where radioactive materials and radiation may result in doses in excess of the dose limits in 10 CFR 20, Section 20.1301 is surveyed, classified, and conspicuously posted with the appropriate radiation caution signs, labels, and signals in accordance with 10 CFR 20, Sections 20.1902 and 20.1903, except as described below.

The station employs administrative and physical security measures to prevent unauthorized entry of personnel into any high or very high radiation areas. The NRC granted the station approval in accordance with 10 CFR 20.1601(c), to use alternative methods for controlling access to high radiation areas in place of the controls required by 10 CFR 20.1601 (a) and (b). These alternative methods are described in the administrative section of each unit's Technical Specifications.

Very High Radiation Areas are those areas where an individual could receive in excess of 500 rads in 1 hour at 1 meter (3 feet) from a radiation source or from any surface that the radiation penetrates. These areas, in addition to the controls specified in the Technical Specifications, have a unique key and a specific procedure for entry into the area.

Traffic patterns normally discourage or prevent access to radiation or potential radiation areas. Warning signs, audible and visible indicators, barricades, or locked doors are employed to protect personnel from access to high radiation areas that may exist temporarily or semi-permanently as a result of unit operations and maintenance.

Radiation protection services and facilities around the site provide all workers the necessary protection and controls for work in radioactive environments.

Administrative radiation protection activities are centered around the radiation protection office. Standard office equipment, equipment storage areas, records storage, and some personnel dosimetry equipment are among the items to be included in the radiation protection office. Personnel decontamination supplies and equipment are stored in the radiation protection decontamination facility. This room contains stainless steel showers and sinks, with drains directed to the wastewater treatment system (Section 9.2.3). A low-background count laboratory is used for counting and/or identifying radioactivity in airborne and liquid samples in conformance with 10 CFR 20, and to 10 CFR 50 App. A General Design Criterion 64. The chemistry laboratory is used to perform chemical and elemental analyses of environmental effluents. All sink and floor drains in this room are directed to the wastewater treatment system; fume hood exhaust is directed to the ventilation system. Equipment used to perform routine counting and analyses on all plant radioactivity samples, as required by 10 CFR 20, are acquired, maintained and calibrated as appropriate.

All personnel entering contaminated areas are required to wear protective clothing. The nature of the work to be done, the contamination level in the area, and the total industrial risks, are the governing factors in the selection of protective clothing to be worn by individuals. Additional protective clothing stations are established at temporary dressing rooms or strategic locations, as required, to ensure efficient operations and to preclude the spread of contamination. Protective clothing available at the station includes the following:

- modesty garments
- shoe covers
- overshoe rubbers
- head covers
- gloves
- coveralls and lab coats

Additional items of specialized apparel are available for operations involving high-level contamination, such as:

- plastic or rubber suits
- surgeon's masks
- face shields
- bubble hoods

All protective clothing is cleaned and decontaminated at a vendor laundry, on-site laundry facility, or disposed of as radioactive waste.

Appropriate training and written guidance govern the proper use of protective clothing, where and how it is to be worn and removed, the decontamination facilities for personnel and equipment, and the areas to be used.

Respiratory protective equipment is available to qualified station personnel and issued to individuals, as required by actual or potential occupational risk of the work assignment. The respiratory protection program follows the guidance of Regulatory Guide 8.15, Revision 1, and complies with 10 CFR 20, Subpart H. Respiratory protection equipment is stored at the respiratory storage and issue facilities. Respiratory equipment may include:

- pressure demand full-face-piece air line respirators;
- continuous air flow hoods or suits;
- pressure demand full-face-piece self-contained breathing apparatus; and

- full-face mechanical filter respirators.

Respiratory protective equipment is cleaned, sanitized, repaired and decontaminated at respiratory storage and issue facilities, or at vendor facilities.

All radiation workers are issued NVLAP accredited dosimeters and are required to wear such dosimeters at all times while within any RCA. All other individuals who enter an RCA are required to wear an individual radiation monitoring device.

Electronic dosimeters or direct-reading pocket dosimeters are issued as an additional method for determining gamma exposure. All individuals are required to examine their dosimeters at frequent intervals while in radiation areas. The use, care, and testing of these direct reading dosimeters will follow applicable guidance of Regulatory Guides 8.4, Revision 1, and 8.28, Revision 0.

Special or additional dosimetry, such as finger ring dosimeters and teledosimetry, are issued under special conditions at the discretion of radiation protection supervision.

The NVLAP accredited individual monitoring devices are processed periodically at the discretion of radiation protection personnel. In addition, they can be processed promptly whenever it appears that an overexposure may have occurred.

Dosimeter records furnish the exposure data for the administrative control of radiation exposure. Exposure records for each individual are maintained in accordance with the guidance of Regulatory Guide 8.7, Revision 2.

12.5.3 PROCEDURES

All radiation protection procedures and methods of operation for ensuring that occupational radiation exposure is as low as reasonably achievable (ALARA) follow the provisions and suggestions of Regulatory Guides 8.8, Revision 3; 8.10, Revision 1-R; and 1.33, Revision 2, as applicable. Such procedures are implemented by qualified personnel whose qualifications meet the requirements of Regulatory Guide 1.8, Revision 3. In addition, all administrative and procedural practices associated with the monitoring of occupational radiation exposure follow the guidance of Regulatory Guides 8.2, Revision 1; 8.4, Revision 1; 8.7, Revision 2; 8.9, Revision 1; and 8.34, Revision 0.

Many radiation protection procedures at Millstone Nuclear Power Station are common to Units 2 and 3. Radiation protection procedures are an integral part of the ALARA program at the station.

Access to restricted/radiologically controlled areas is controlled by administrative and physical security measures as required by 10 CFR 20, Subparts G and J.

Station management assures entry control to high radiation areas through the administration of radiation work permits (RWPs) that stipulate purpose of entry, work location, radiological conditions, surveillance and dosimetry requirements, stay time, protective clothing, respiratory protective equipment, special tools, engineering controls, special personnel monitoring devices, and other procedural requirements.

The following are some objectives for issuing RWPs.

1. Provide a detailed assessment of the actual and potential radiation hazards that are associated with the job function and area.

2. Ensure that proper protective measures are taken to safely perform the required tasks in the area and to maintain the Total Effective Dose Equivalent as low as reasonably achievable (ALARA).

3. Provide a mechanism for individuals to acknowledge their understanding of the radiological conditions, the protective and safety equipment and measures required, and the willingness to follow the requirements designated on the RWP.

4. Provide a system for recording the sources (station systems and components), job types and functions, and personnel categories where exposures occur.

RWPs are issued for general and specific activities performed in radiation areas, contaminated areas, airborne radioactivity areas, and for all activities that require entrance into high radiation areas, and very high radiation areas as defined in 10 CFR 20, Section 20.1003. RWPs are also issued prior to maintenance or inspection of contaminated or radioactive equipment with removable contamination in excess of 1,000 dpm/100 cm^2 beta-gamma and/or 100 dpm/100 cm^2 alpha. RWPs are also required prior to entrance into the reactor containment of any unit.

Under limited situations and at the discretion of radiation protection supervision, continuous radiation protection personnel coverage may be substituted for an RWP, such as an emergency which threatens personnel or plant safety.

Radiation protection personnel routinely survey selected areas of the station to assess and control exposure to radiation and radioactive materials in accordance with 10 CFR 20, Section 20.1501. Depending on the type of survey required and anticipated types and levels of radioactivity, various portable instruments and techniques are used to perform these surveys. Results of all surveys are recorded and kept on file at the radiation protection office on a short term basis.If necessary, survey sheets may be posted. Permanent storage is provided by forwarding records to the nuclear records facility. Reporting practices for all normal and accident conditions comply with the regulations set forth in 10 CFR 20, Subpart M.

Area surveys are performed at scheduled frequencies, based on location, radiation levels, station status, and occupancy. All area survey readings are recorded and filed as required by 10 CFR 20, Section 20.2103 and Regulatory Guide 8.2, Revision 0. Caution placards, describing the radiological conditions, are posted to comply with the requirements of 10 CFR 20, Section 20.1902.

Surveys for contamination are used to assess containment of radioactive materials and the need for decontamination of an area.Contamination is measured at selected locations throughout the station, where the potential for the spread of contamination exists. Contamination surveys are made using the “smear” or “swipe” technique, or by using an appropriate portable instrument. Scheduled frequencies are based on location, radiation levels, station status and occupancy, or as required by actual operating conditions, and as directed by radiation protection supervision. Contamination surveys are performed on personnel, equipment, and in uncontrolled areas to ensure that radiological control methods are adequate. Personnel, equipment, and material leaving contaminated areas are monitored to prevent the spread of contamination into clean areas. Areas, equipment, and personnel that may be contaminated with radioactive material are decontaminated using applicable methods and techniques, such as those suggested in NCRP65 and IE Circular 81-07.

Levels of contamination are also used to judge the potential for airborne radioactive material and the need for monitoring air, and the use of engineering controls or respiratory protection.

It is management’s intent to control airborne radioactivity levels as effectively as practicable by proper preventive measures, engineering controls, and good housekeeping techniques. In the event of a radioactive airborne problem, every effort is made to promptly assess the situation. Section 12.3.4 provides information on the installed airborne radioactivity monitoring instrumentation.

Control of airborne radioactivity levels is assured through the use of the station’s heating, ventilation, and air-conditioning (HVAC) systems and portable air movers and filters. The HVAC systems provide controlled air movement and filtration capability for those areas with a high potential for airborne radioactivity problems. As required, special control techniques are used to minimize airborne exposure arising from special work projects. Respiratory protection equipment is available for use in those situations where airborne radioactivity hazards exist and where other control measures are inadequate at the location and time. Respiratory protection equipment use is assessed based upon the principle of keeping the Total Effective Dose Equivalent ALARA, consistent with minimizing total occupational risk.

The special control techniques used to minimize airborne exposure include decontamination of the component or area prior to performing work, keeping work surfaces damp while work is in progress, and using tents or glove bags in conjunction with appropriate, filtered ventilation systems.

Techniques for obtaining breathing zone air samples are grab samples taken in areas representative of the worker’s breathing zone and/or lapel air samplers.

Some conditions which require special air sampling include lifting the reactor vessel head, venting a contaminated system, and working on an open contaminated system.

In regard to reporting practices for airborne contamination surveys, radiation protection supervision is notified when airborne concentrations read 30 percent of DAC and the area requires posting if this condition persists for a sustained period.

All airborne contamination survey sheets are reviewed by radiation protection supervision and filed.

The air sampling program provides information on the potential inhalation of radioactive material by workers. The information is used to determine what remedial action or protective measures such as respirators, glove boxes, or engineering controls are necessary to protect the worker. Air samples are taken for all work on systems which have the potential for release of airborne radioactivity. Surveys are performed on a routine basis, depending on location, station status, and occupancy. In addition, surveys are performed whenever work is required on a known or potentially contaminated system that must be opened to the working environment or whenever welding, burning, or grinding is performed on a known or potentially contaminated system.

Surveys are also performed whenever the continuous air monitor indicates an airborne problem and prior to containment entry. Additional surveys are performed as deemed necessary by radiation protection supervision.

Prior to issuance and use of required respiratory protection equipment, each individual must have satisfactorily completed the following:

- a satisfactory medical evaluation to ensure that the individual is medically fit to use respiratory protection devices;

- training for the device to be used;

- a fit test (face sealing devices only); and

The air sampling and respiratory protection programs meet the recommendations and provisions of 10 CFR 20, Subpart H, Regulatory Guide 8.15, Revision 1, and NUREG-0041.

Special procedures control the handling or movement of material within and from restricted and radiologically controlled areas, such as the shipment and receipt of radioactive materials. These procedures comply with the regulations stipulated in 49 CFR 170-178, 10 CFR 70, 10 CFR 71, and 10 CFR 20.1906.

As previously discussed in this section, all radiation workers receive a NVLAP accredited individual monitoring device and direct-reading pocket ion chamber and/or electronic dosimetry to monitor personnel exposure. Exposure records are filed and retained for each individual in accordance with the recommendations of Regulatory Guides 8.2, Revision 1 and 8.7, Revision 2, and as required by 10 CFR 20, Subpart L. Any reports of overexposures and excessive levels and concentrations comply with the regulations of 10 CFR 20, Subpart M. Reports of personnel monitoring, and reports of theft or loss of licensed material are issued in accordance with the regulations required by 10 CFR 20, Subpart M.

The bioassay program at the Millstone Point Nuclear Power Station follows the guidance of Regulatory Guides 8.9, Revision 1 and meets the requirements of 10 CFR 20, Section 20.1204. The bioassay program includes:

- determination of the conditions under which bioassays should be required;
- selection of measurement techniques, measurement frequency, and program participants;
- action points and actions to be taken based on measurement results; and
- interpretation of measurement results in terms of location of radioactive material in the body, the quantity present, the rate of elimination, and the resulting dose commitment and
- use of personnel contamination monitors, located at RCA exits and the Protected Areas exits, which may serve as “passive monitors” for detection of internal contamination in lieu of periodic whole body counts for all workers.

A whole-body counter is located at the station as needed for in vivo measurement of station personnel, visitors, or support personnel. The whole-body counter provides preliminary background information, periodic evaluation, and emergency capability for detecting internal exposure conditions. Assessment of internal radiation exposure for station personnel may be performed, for example when:

- individuals have a known or suspected intake of four or more DAC hours within a calendar week,
- incidents involve contamination around the nose or mouth; and
- accidents involve a potential intake. Excreta samples from suspected individuals may be sent to a qualified laboratory for analysis.

Training in radiation protection principles and procedures is performed by the Nuclear Training Department or by qualified station personnel. New employees, contractors, and other supporting personnel receive validation of prior training and orientation training, as appropriate, before the beginning of their work assignments.

All permanent station personnel who are required to work in the RCA are required to successfully complete basic training courses and practical exercises to demonstrate their proficiency and competence.

All radiological workers participate in the radiological worker training program. The radiological worker training program maintains the proficiency of employees through training and annual retraining on selected material. Additional training is given for selected tasks which require increased radiation protection.

The content of the radiation protection related training program meets the intent of Regulatory Guide 8.27, Revision 0; Regulatory Guide 8.13, Revision 1; Regulatory Guide 8.29, Revision 0; and NUREG-0731. The program content is detailed in Section 13.2, Training Program. Details of the Emergency Plan which meet the intent of NUREG-0731, dated 1980, are given in Section 13.3, Emergency Planning.

Assessments are performed on all radiation protection procedures including those such as emergency procedures and instrument storage, calibration, and maintenance procedures, in addition to the procedures specifically required by Regulatory Guide 1.33, Revision 2.

12.5.4 REFERENCE FOR SECTION 12.5

12.5-1 Nuclear Regulatory Commission, Code of Federal Regulations 10 CFR Part 20, "Standards for Protection Against Radiation"

12.5-2 Nuclear Regulatory Commission, Code of Federal Regulations 10 CFR Part 50, App. A, "General Design Criteria for Nuclear Power Plants," "Domestic Licensing of Production and Utilization Facilities"

12.5-3 Nuclear Regulatory Commission, Code of Federal Regulations 10 CFR Part 70, "Domestic Licensing of Special Nuclear Material"

12.5-4 Nuclear Regulatory Commission, Code of Federal Regulations 10 CFR Part 71, "Packaging and Transportation of Radioactive Material"

12.5-5 Department of Transportation, Code of Federal Regulations 49 CFR 170-178, "Subchapter C-Hazardous Materials Regulations"

12.5-6 NUREG-0731, "Guidelines for Utility Management Structure and Technical Resources," 1980

12.5-7 NUREG-0800, USNRC. "Standard Review Plan", Revision 1.

12.5-8 Regulatory Guide 1.8, Rev. 3 "Qualification and Training of Personnel for Nuclear Power Plants"

12.5-9 Regulatory Guide 1.33, Rev. 2, "Quality Assurance Program Requirements"

12.5-10 Regulatory Guide 8.2, Rev. 1, "Guide for Administrative Practices in Radiation Monitoring"

12.5-11 Regulatory Guide 8.4, Rev. 1, "Personnel Monitoring Device - Direct-Reading Pocket Dosimeters"

12.5-12 Regulatory Guide 8.7, Rev. 2, "Instructions for Recording and Reporting Occupational Radiation Exposure Data"

12.5-13 Regulatory Guide 8.8, Rev. 3, "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations will be as Low as is Reasonable Achievable"

12.5-14 Regulatory Guide 8.9, Rev. 1, "Acceptable Concepts, Models, Equations and Assumptions for a Bioassay Program"

12.5-15 Regulatory Guide 8.10, Rev. 1-R, "Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As is Reasonably Achievable"

12.5-16 Regulatory Guide 8.13, Rev. 3. "Instruction Concerning Prenatal Radiation Exposure"

12.5-17 Regulatory Guide 8.15, Rev. 1, "Acceptable Programs for Respiratory Protection"

12.5-18 Regulatory Guide 8.27. "Radiation Protection Training for Personnel at Light-Water-Cooled Nuclear Power Plants", Revision 0.

12.5-19 Regulatory Guide 8.28, Rev. 0, "Audible-Alarm Dosimeters"

12.5-20 Regulatory Guide 8.29. "Instructions Concerning Risks from Occupational Radiation Exposure", Revision 1.

12.5-21 Regulatory Guide 8.34, Rev. 0, "Monitoring Criteria and Methods To Calculate Occupational Radiation Doses"

12.5-22 Regulatory Guide 1.33. "Quality Assurance Program Requirements", Revision 2.

12.5-23 INPO 05-008, "Guidelines for Radiological Protection at Nuclear Power Stations"

12.5-24 IE Circular 81-007, "Control of Radioactively Contaminated Material"

12.5-25 NCRP Number 65, "Management of Persons Accidently Contaminated with Radionuclides"

12.5-26 NUREG-0041, Rev. 1, "Manual of Respiratory Protection Against Airborne Radioactive Material"

TABLE 12.5–1 DELETED BY FSARCR 04-MP3-040

TABLE 12.5–2 DELETED BY FSARCR 04-MP3-040

FIGURE 12.2 - 1 ARRANGEMENT - OPERATING PERSONNEL ACCESS AND EGRESS (PLAN ELEVATION 3 FEET 8 INCHES AND ABOVE)

[redacted]

Withheld under 10 CFR 2.390 (d)(1)

FIGURE 12.2–2 ARRANGEMENT - OPERATING PERSONNEL ACCESS AND EGRESS (PLAN ELEVATION 24 FEET 6 INCHES AND ABOVE)

[redacted]

Withheld under 10 CFR 2.390 (d)(1)

FIGURE 12.2–3 ARRANGEMENT - OPERATING PERSONNEL ACCESS AND EGRESS (PLAN ELEVATION 38 FEET 6 INCHES AND ABOVE)

[redacted]

Withheld under 10 CFR 2.390 (d)(1)

FIGURE 12.2–4 I-131 CONCENTRATION CONTAINMENT (HISTORICAL)

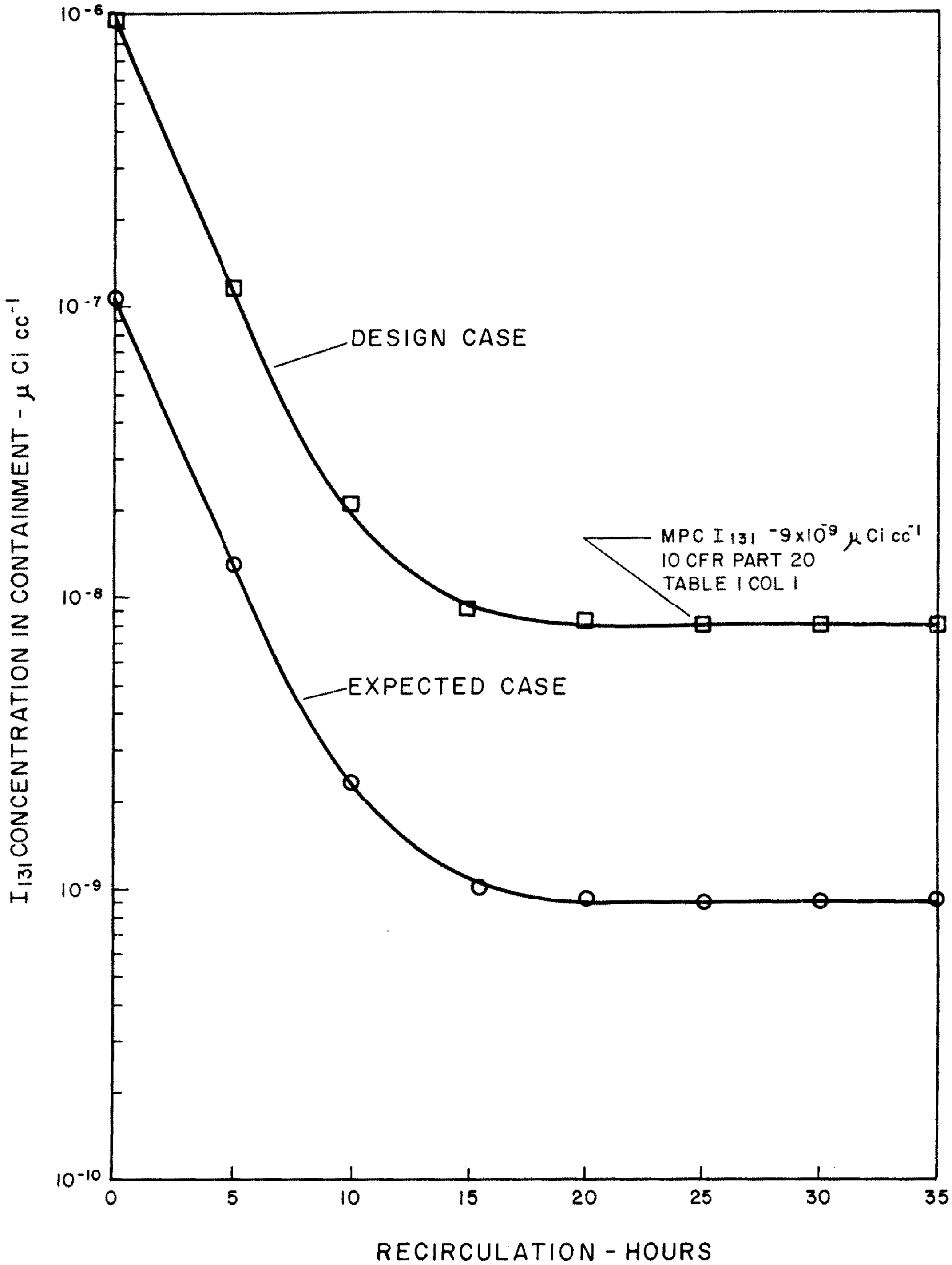


FIGURE 12.3–1 DESIGN BASIS RADIATION ZONES FOR SHIELDING (NORMAL OPERATIONS)

[redacted]

Withheld under 10 CFR 2.390 (d)(1)

FIGURE 12.3–2 DESIGN BAS[redacted]TION ZONES FOR SHIELD[redacted]RMAL OPERATIONS)

[redacted]

Withheld under 10 CFR 2.390 (d)(1)

[redacted] **FIGURE 12.3–3 DESIGN BASI[redacted]IATION ZONES FOR SHIELDIN[redacted]RMAL OPERATIONS)**

[redacted]

Withheld under 10 CFR 2.390 (d)(1)

FIGURE 12.3–4 DESIGN BASIS RADIATION ZONES FOR SHIELDING (NORMAL OPERATIONS)

[redacted]

Withheld under 10 CFR 2.390 (d)(1)

FIGURE 12.3-5 CONTAINMENT MONITORING SYSTEM

THIS FIGURE NOW IN SECTION 11.5 (FIGURE 11.5-2)

FIGURE 12.3–6 DESIGN BASIS RADIATION ZONES FOR SHIELDING (SHUTDOWN/REFUELING)

[redacted]

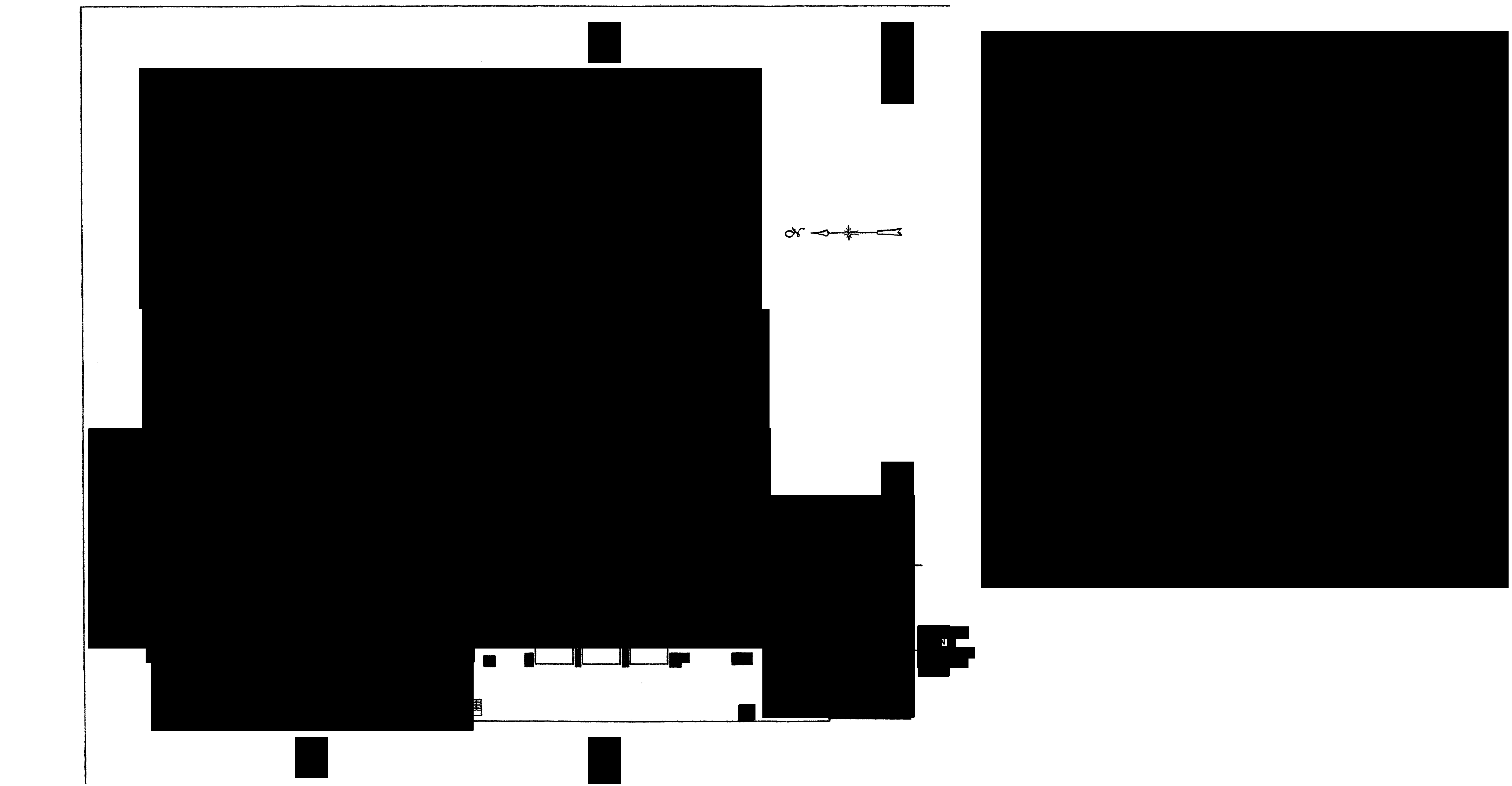
Withheld under 10 CFR 2.390 (d)(1)

FIGURE 12.3–7 DESIGN BASIS RADIATION ZONES FOR SHIELDING (SHUTDOWN/REFUELING)

[redacted]

Withheld under 10 CFR 2.390 (d)(1)

FIGURE 12.3–8 DESIGN BASIS [redacted]DIATION ZONES FOR SHIELDIN[redacted]HUTDOWN/REFUELING)



[redacted]

Withheld under 10 CFR 2.390 (d)(1)

FIGURE 12.3–9 DESIGN BAS[redacted] RADIATION ZONES FOR SHIELDIN[redacted] (SHUTDOWN/REFUELING)

[redacted]

Withheld under 10 CFR 2.390 (d)(1)

FIGURE 12.3–10 SHEET 1 ROUTES TO POST-ACCIDENT VITAL AREAS - ROUTE 1 TO AUXILIARY BUILDING

[redacted]

Withheld under 10 CFR 2.390 (d)(1)

FIGURE 12.3–10 SHEET 2 ROUTES TO POST-ACCIDENT VITAL AREAS - ROUTE 2 TO THE ESF BUILDING

[redacted]

Withheld under 10 CFR 2.390 (d)(1)

FIGURE 12.3–10 SHEET 3 ROUTES TO POST-ACCIDENT VITAL AREAS - ROUTE 3 TO THE HYDROGEN RECOMBINER BUILDING

[redacted]

Withheld under 10 CFR 2.390 (d)(1)

FIGURE 12.3–10 SHEET 4 ROUTES TO POST-ACCIDENT VITAL AREAS - ROUTE 4 TO THE HYDROGEN RECOMBINER BUILDING AND TO UNIT 2 CHEMISTRY LABS

[redacted]

Withheld under 10 CFR 2.390 (d)(1)

FIGURE 12.3–10 SHEET 5 ROUTES TO POST-ACCIDENT VITAL AREAS - ROUTE 5 TO FUEL BUILDING

[redacted]

Withheld under 10 CFR 2.390 (d)(1)

FIGURE 12.3–10 SHEET 6 ROUTES TO POST-ACCIDENT VITAL AREAS - ROUTE 6 TO TURBINE BUILDING

[redacted]

Withheld under 10 CFR 2.390 (d)(1)

FIGURE 12.3–10 SHEET 7 ROUTES TO POST-ACCIDENT VITAL AREAS - ROUTE 7 TO DIESEL GENERATOR

[redacted]

Withheld under 10 CFR 2.390 (d)(1)

FIGURE 12.3–10 SHEET 8 ROUTES TO POST-ACCIDENT VITAL AREAS - ROUTE 8 TO ESF BUILDING

[redacted]

Withheld under 10 CFR 2.390 (d)(1)

FIGURE 12.3–10 SHEET 9 ROUTES TO POST-ACCIDENT VITAL AREAS - ROUTE 9 TO AUXILIARY BUILDING

[redacted]

Withheld under 10 CFR 2.390 (d)(1)

FIGURE 12.3 - 11 FUEL TRANSFER TUBE SHIELDING

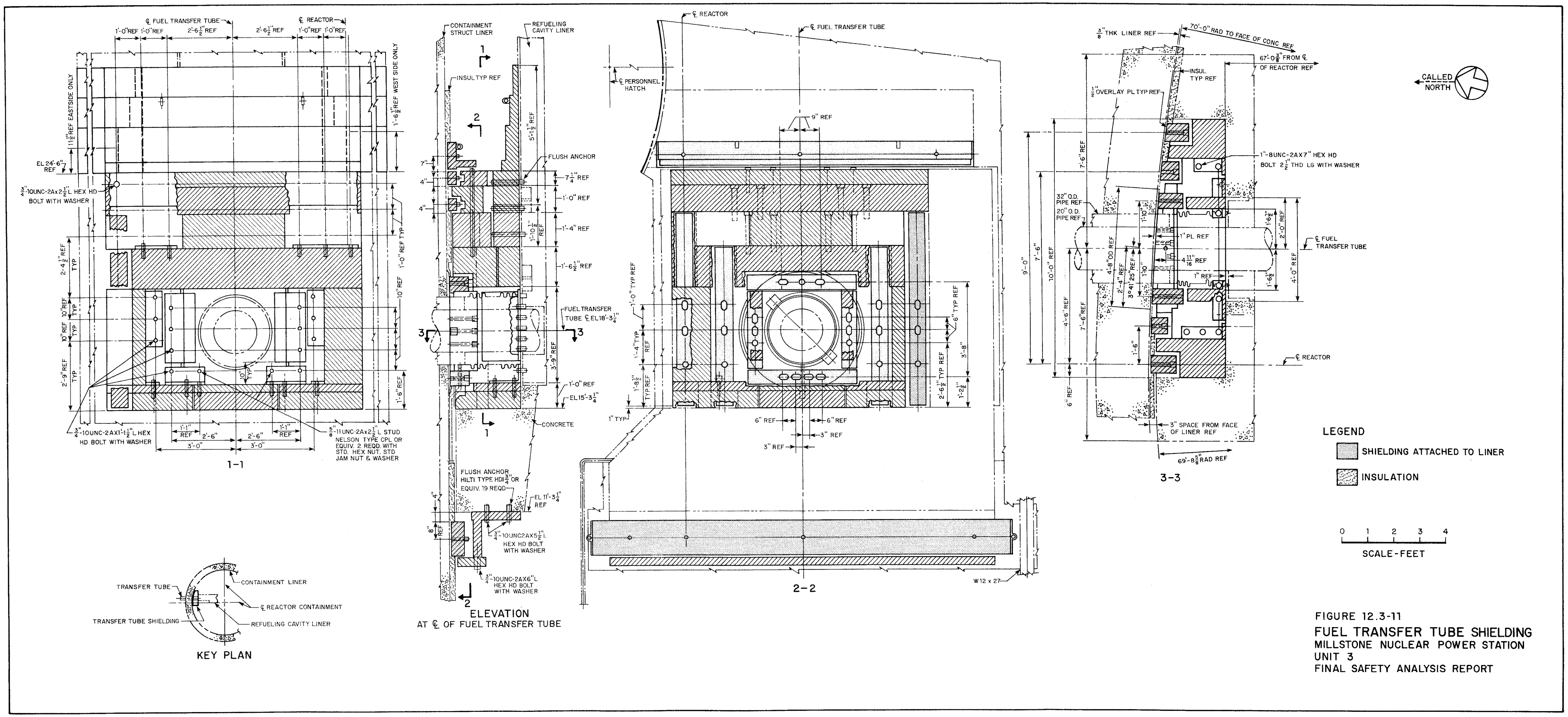


FIGURE 12.3 - 12 UPPER REACTOR CAVITY NEUTRON SHIELD

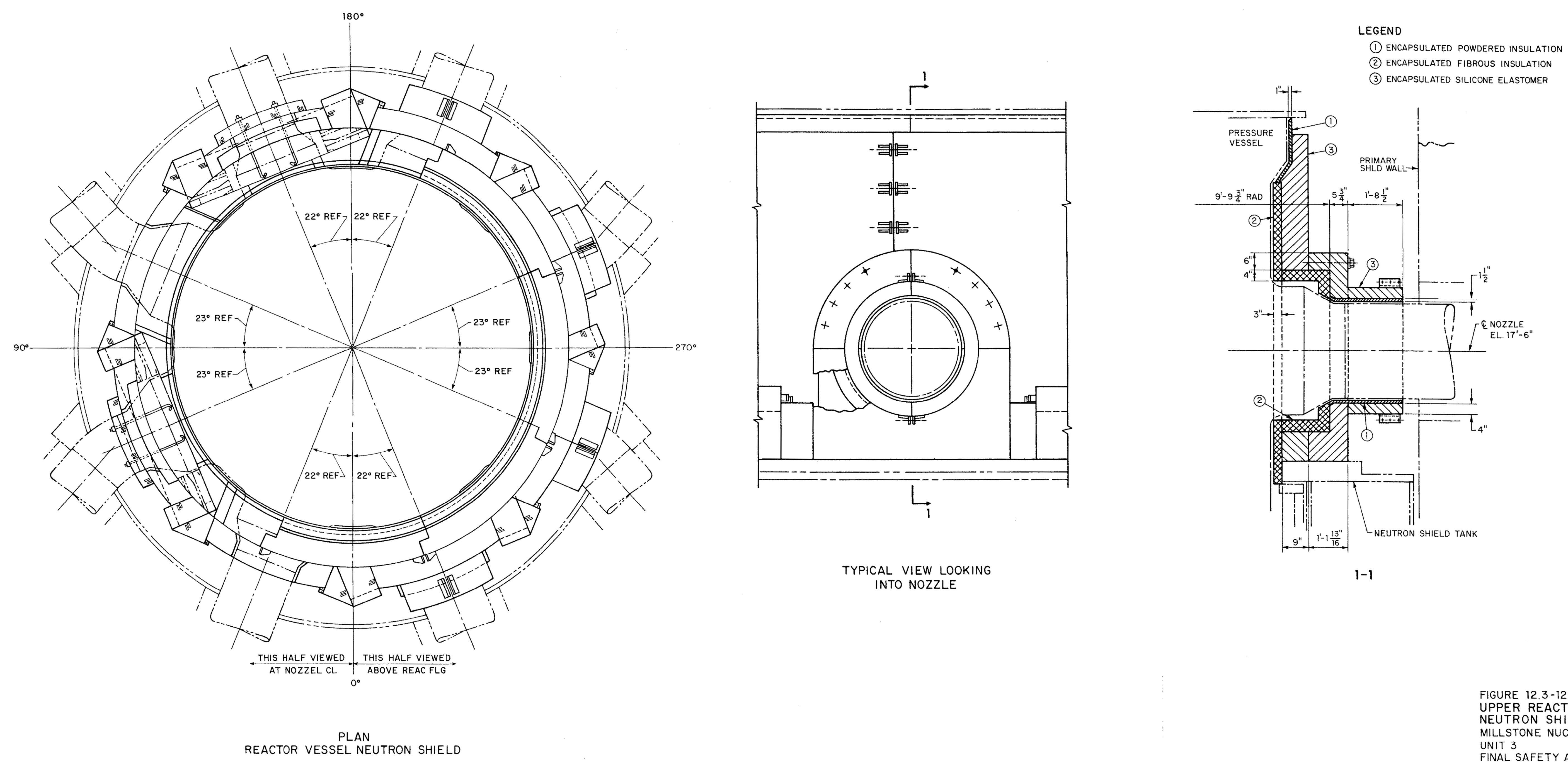


FIGURE 12.3-12
UPPER REACTOR CAVITY
NEUTRON SHIELD
MILLSTONE NUCLEAR POWER PLANT
UNIT 3
FINAL SAFETY ANALYSIS REPORT