CHAPTER 11 – RADIOACTIVE WASTE AND RADIATION PROTECTION

TABLE OF CONTENTS

SECTION	TITLE
11.0	RADIOACTIVE WASTE AND RADIATION PROTECTION
11.1	DESIGN PERFORMANCE OBJECTIVES
11.1.1	DESIGN BASIS
11.2	RADIOACTIVE WASTE DISPOSAL SYSTEMS SUMMARY
11.2.1	RADIOACTIVE LIQUID WASTE DISPOSAL SYSTEM
11.2.1.1	SYSTEM FUNCTIONS
11.2.1.2	SYSTEM DESCRIPTION
11.2.1.3	METHODS OF OPERATION
11.2.1.4	SYSTEM DESIGN EVALUATION
11.2.2	RADIOACTIVE GAS WASTE DISPOSAL SYSTEM
11.2.2.1	SYSTEM FUNCTION
11.2.2.2	SYSTEM DESCRIPTION
11.2.2.3	METHODS OF OPERATION
11.2.2.4	SYSTEM DESIGN EVALUATION
11.2.3	RADIOACTIVE SOLID WASTE DISPOSAL SYSTEM
11.2.3.1	SYSTEM FUNCTION
11.2.3.2	SYSTEM DESCRIPTION
11.2.3.3	SYSTEM STORAGE CAPACITY
11.2.3.4	SYSTEM DESIGN EVALUATION
11.2.4	OPERATING POLICY ON RADIOACTIVE DISCHARGES TO THE ENVIRONMENT
11.2.5	DILUTION FACTORS IN SUSQUEHANNA RIVER
11.2.6	MAJOR CHANGES TO RADIOACTIVE WASTE TREATMENT SYSTEMS
11.3	RADIATION SHIELDING
11.3.1	DESIGN CRITERIA
11.3.2	GENERAL DESCRIPTION
11.3.2.1	PRIMARY SHIELD
11.3.2.2	SECONDARY SHIELD
11.3.2.3	REACTOR BUILDING
11.3.2.4	CONTROL ROOM
11.3.2.5	AUXILIARY BUILDING
11.3.2.6	SPENT FUEL SHIELDING
11.3.2.7	ORIGINAL STEAM GENERATOR STORAGE FACILITY (OSGSF) SHIELDING
11.3.3	MATERIALS
11.4	RADIATION MONITORING SYSTEM
11.4.1	DESIGN BASIS
11.4.1.1	SYSTEM DESCRIPTION
11.4.2	AREA MONITORING SUBSYSTEM DESCRIPTION
11.4.3	ATMOSPHERIC MONITORING SUBSYSTEM DESCRIPTION
11.4.4	LIQUID MONITORING SUBSYSTEM DESCRIPTION
11.4.5	POSTACCIDENT RADIATION MONITORING

TABLE OF CONTENTS (cont'd)

SECTION	TITLE
11.4.6	POST ACCIDENT RADIOLOGICAL SAMPLING
11.4.7	NORMAL OPERATION IODINE SAMPLING SYSTEM
11.4.8	SUPPLEMENTAL AIR SAMPLERS AND MONITORS
11.5	RADIOLOGICAL HEALTH AND SAFETY
11.5.1	RADIATION WORK PERMITS
11.5.2	PERSONNEL MONITORING SYSTEM
11.5.3	PERSONNEL PROTECTIVE EQUIPMENT
11.5.4	ACCESS CONTROL
11.5.5	RADIOLOGICAL CONTROL LABORATORY FACILITIES
11.5.6	RADIOLOGICAL CONTROL INSTRUMENTATION
11.5.7	MEDICAL PROGRAM
11.5.8	RADIATION PROTECTION PROGRAM
11.5.8.1	FOUNDATION FOR THE TMI-1 RADIOLOGICAL CONTROLS PROGRAM
11.5.8.2	DELETED
11.5.8.3	AUDITS, REVIEWS, AND REPORTS ON THE TMI-1 RADIOLOGICAL CONTROLS PROGRAM
11.5.8.4	RADIOLOGICAL CONTROLS TRAINING
11.5.8.5	CONTROL OF EXTERNAL RADIATION EXPOSURE
11.5.8.6	CONTROL OF INTERNAL RADIATION EXPOSURE
11.5.8.7	CONTROL OF RADIOACTIVE CONTAMINATION
11.5.8.8	ORGANIZATION FOR RADIOLOGICAL CONTROLS
11.6	RADIOACTIVE MATERIALS SAFETY
11.6.1	MATERIALS SAFETY PROGRAM
11.6.1.1	CONTROL OF RADIOACTIVE MATERIALS
11.6.1.2	RECEIPT OF RADIOACTIVE SOURCES
11.6.1.3	STORAGE OF RADIOACTIVE SOURCES
11.6.1.4	SOURCE PREPARATION AND HANDLING
11.6.1.5	DISPOSAL OF RADIOACTIVE WASTE RESULTING FROM SOURCE PREPARATION
11.6.1.6	RADIOLOGICAL SURVEYS
11.6.1.7	SOURCE INVENTORY
11.6.2	FACILITIES AND EQUIPMENT
11.6.3	PERSONNEL AND PROCEDURES
11.6.4	REQUIRED RADIOACTIVE SOURCES
11.7	REFERENCES

LIST OF TABLES

TABLE	TITLE
11.1-1	DESIGN BASIS ESCAPE RATE AND COEFFICIENTS FOR FISSION PRODUCT RELEASE
11.1-2	DESIGN BASIS REACTOR COOLANT ACTIVITIES FROM ONE PERCENT DEFECTIVE FUEL
11.1-3	DESIGN BASIS RADIOACTIVE WASTE QUANTITIES
11.2-1	DISPOSAL SYSTEM COMPONENT DATA
11.2-2	EQUIPMENT IN PRIMARY COOLANT AND MISCELLANEOUS WASTE TRAINS
11.2-3	ASSUMPTIONS FOR ESTIMATING DESIGN BASIS ANNUAL AVERAGE LIQUID WASTE RELEASE TO THE RIVER
11.2-4	DESIGN BASIS ESTIMATE OF ANNUAL RELEASE OF RADIOACTIVE LIQUIDS TO RIVER
11.2-5	WASTE GAS SYSTEM COMPONENT DATA
11.2-6	ASSUMPTIONS FOR 90-DAY HOLDUP BASIS ESTIMATE OF ANNUAL AVERAGE WASTE GAS RELEASE
11.2-7	DESIGN MAXIMUM 90-DAY HOLDUP BASIS WASTE GAS RELEASE ANALYSIS SUMMARY
11.2-8	ACTIVITY LEVELS AND FRACTIONS OF MPC IN EVAPORATOR CONDENSATE STORAGE TANKS AS A FUNCTION OF THE QUANTITY OF WATER IN WHICH THE DESIGN BASIS ACTIVITY APPEARS
11.2-9	ACTIVITY LEVEL AND FRACTION OF MPC IN COOLING TOWER EFFLUENT, UPSTREAM AND DOWNSTREAM RIVER WATER FOR THE DESIGN BASIS QUANTITY OF ACTIVITY AS A FUNCTION OF: WASTE WATER QUANTITY AND FLOW RATE, COOLING TOWER EFFLUENT FLOW RATE AND RIVER WATER FLOW RATE
11.3-1	PRINCIPAL SHIELDING
11.4-1	RADIATION MONITORING SYSTEM POWER SUPPLY SOURCES
11.4-2	RADIATION MONITORS
11.4-3	DELETED
11.5-1	DELETED

LIST OF FIGURES

FIGURE	TITLE
11.2-1	DELETED
11.2-2	DELETED
11.2-3	DELETED
11.2-4	DELETED
11.2-5	DELETED
11.2-6	DELETED
11.2-7	WASTE GAS SYSTEM COMPRESSOR SCHEMATIC
11.3-1	DELETED
11.4-1	RADIATION MONITORING SYSTEM BLOCK DIAGRAM
11.4-2	RADIATION MONITORING SYSTEM INTERLOCKS
11.5-1	DELETED
11A	Appendix A
	POST ACCIDENT SHIELDING EVALUATION

11.0 RADIOACTIVE WASTE AND RADIATION PROTECTION

11.1 DESIGN PERFORMANCE OBJECTIVES

The radioactive waste disposal systems and the structures housing them are designed and constructed to meet or exceed the applicable federal regulations for the containment, control, and release or disposal of radioactive liquids, gases, and solids generated as a result of normal and emergency operation of the plant.

Plant structures are designed and constructed, and equipment is located within them, in order to attenuate the radiation received by plant personnel in the discharge of their normal duties to levels not exceeding applicable federal regulations for occupational exposure. The structures also provide shielding of the general public such that applicable federal regulations for radiation exposure are not exceeded during operation of the unit.

The radiation monitoring systems are designed to detect, measure, and alarm or control, as required, radiation and releases of radioactive solids, liquids, and gases produced as a result of normal and emergency operation of the plant. These systems aid in the protection of the general public and plant personnel from exposures to radiation or radioactive materials in excess of those allowed by the applicable federal regulations.

11.1.1 DESIGN BASIS

Radioactivity accumulation in the Reactor Coolant System and associated waste handling equipment was determined for radwaste on the basis of fission product leakage through clad defects in 1 percent of the fuel. The activity levels were computed for a design basis core assuming full power operation of 2535 MWt for a 460 day initial core cycle and a 310 day second core cycle with no defective fuel followed by operation over a third 310 day core cycle with 1 percent defective fuel. Continuous reactor coolant purification at a rate of one reactor coolant system volume per day was used with a zero removal efficiency for Kr, Xe, Cs, Mo, and Y and a 99 percent removal efficiency for all other nuclides.

The current core cycle is described in UFSAR Section 3.2. Whereas the current core power level and burnup are not as described above (i.e., power level and fuel cycle duration have been increased), the results on Table 11.1-2 provide a conservative estimate of the radioactive material inventory in the reactor coolant system to determine the expected radioactive materials processing and handling requirements.

The quantity of fission products released to the reactor coolant during steady-state operation is based on the use of escape rate coefficients (sec^{-1}) as determined from experiments involving purposely defected fuel elements (Reference 1, 2, 3, and 4). Values of the escape rate coefficients used in the calculations are shown in Table 11.1-1.

Calculations of the activity released from the fuel were performed with a digital computer code which solves the differential equations for a five member radioactive chain for buildup in the fuel, release to the coolant, removal from the coolant by purification and leakage, and collection on a resin or in a holdup tank. The fission product activity levels in the reactor coolant at the end of the third core cycle are shown in Table 11.1-2. Table 11.1-2 also includes the corrosion product activity levels normally anticipated in the reactor coolant as extrapolated from Reference 5.

The tritium activity level in the reactor coolant is based on a tritium production rate, by ternary fission within the fuel, of 6.88×10^{15} atoms/sec. One percent of the tritium produced by ternary fission is assumed to escape the fuel clad and appear in the reactor coolant in associated water molecules. The production rate of tritium, due to neutron irradiation of boron and lithium in the reactor coolant, is estimated to be 0.26×10^{15} atoms/sec, and the tritium produced by these mechanisms is also assumed to appear in the reactor coolant in associated water molecules. The total rate of appearance of tritium in the reactor coolant is, therefore, 3.3×10^{14} atoms/sec. Considering the rated power upgrade by 1.3 percent, from 2535 Mwt to 2568 Mwt, the total rate was conservatively assumed to be increased by a factor of 1.5 percent (Reference 26) resulting in the rate of 3.35×10^{14} atoms/sec. Assuming that all tritium produced in and released to the reactor coolant is ultimately released, the average annual discharge of tritium would be 509.5 curies.

The estimated volumes of radioactive waste generated during plant operation, for sizing and selecting equipment, are listed in Table 11.1-3.

TABLE 11.1-1
(Sheet 1 of 1)

DESIGN BASIS ESCAPE RATE COEFFICIENTS FOR FISSION PRODUCT RELEASE

Element	Escape Rate Coefficient (sec^{-1})
Xe	1.0×10^{-7}
Kr	1.0×10^{-7}
I	2.0×10^{-8}
Br	2.0×10^{-8}
Cs	2.0×10^{-8}
Rb	2.0×10^{-8}
Mo	4.0×10^{-9}
Te	4.0×10^{-9}
Sr	2.0×10^{-10}
Ba	2.0×10^{-10}
Zr	1.0×10^{-11}
Ce and other rare earths	1.0×10^{-11}

TABLE 11.1-2
(Sheet 1 of 1)

DESIGN BASIS REACTOR COOLANT ACTIVITIES
FROM ONE PERCENT DEFECTIVE FUEL

Isotope	Activity (ûCi/ml)	Isotope	Activity (ûCi/ml)
Fission Products			
Kr 85m	1.6	I 131	3.3
Kr 85	9.3	I 132	5.0
Kr 87	0.87	I 133	3.9
Kr 88	2.8	I 134	0.52
Rb 88	2.8	I 135	2.0
		Cs 134	4.9
Sr 89	0.042	Cs 136	0.83
Sr 90	0.0038	Cs 137	49.0
Sr 91	0.048	Cs 138	0.76
Sr 92	0.018	Mo 99	5.6
Xe 131m	2.5	Ba 139	0.084
Xe 133m	2.9	Ba 140	0.068
Xe 133	260	La 140	0.022
Xe 135m	0.97	Y 90	0.97
Xe 135	6.2	Y 91	0.23
Xe 138	0.53	Ce 144	0.0029
Corrosion Products			
Cr 51	0.0014		
Mn 54	0.00016		
Co 58	0.0084		
Fe 59	0.00016		
Co 60	0.0045		
Zr 95	0.011		

Total Activity = 353 ûCi/ml

E[(1)] for this isotopic mix = .37 MeV

(1) E is defined in Technical Specification 3.1.4.1

TABLE 11.1-3
(Sheet 1 of 3)

DESIGN BASIS RADIOACTIVE WASTE QUANTITIES

Waste Source	Quantity per Year (ft^3)	Assumptions and Comments
a. Liquid Waste		
1) Reactor Coolant System:		
For selecting tank and equipment capacities	60,600	2 cold startups at beginning of core life; cold startups thereafter at 77.5 day intervals, continuous shim bleed during operation; boron removal via deborating demineralizers during last 55 days of core life and shutdown for refueling. Total cycle time of 340 days. Therefore, full power operation for 335 days per year.
2) Sampling and laboratory drains	400	12 samples per week at 5 gals per sample
3) Purification demineralizers sluice	160	80 ft^3 resin/yr sluiced at 2 ft^3 sluice water/ft^3 resin
4) Cation demineralizers	288	144 ft^3 resin/yr sluiced at 2 ft^3 sluice water/ft^3 resin
5) Deborating demineralizers	1,080	2 resin regenerations/yr at 13.5 ft^3 resin at 40 ft^3 of regenerating solution and rinses/ft^3 resin
6) Condensate demineralizers	96	48 ft^3 resin/yr sluiced at 2 ft^3 sluice water/ft^3 resin

TABLE 11.1-3
(Sheet 2 of 3)

DESIGN BASIS RADIOACTIVE WASTE QUANTITIES

Waste Source		Quantity per Year (ft^3)	Assumptions and Comments
7)	Precoat filters	240	8 filter bed changes/yr at 30 ft^3 sluice water/ filter bed change.
8)	Miscellaneous system leakage, decontamination, etc.	70,000	1 gpm accumulation rate
9)	Laundry[1]	7,300	150 gpd accumulation rate
10)	Showers[1]	14,600	10 showers per day at 30 gal per shower
b.	Gaseous Waste[3]		
1)	Off-gas from reactor coolant letdown	1,500	Degas at 25 cm^3 H_2 per liter concentration
2)	Off-gas from reactor coolant sampling	10	Degas at 25 cm^3 H_2 per liter concentration
3)	Makeup tank gas inventory	1,000	Vent once per year
4)	Off-gas from pressurizer	60	Vent once per year
c.	Solid Waste		
1)	Purification resin	80	Resin replacement twice per year
2)	Cation demineralizers	144	Resin replacement four times per year
3)	Deborating resin	4	One-quarter of one resin bed replaced per year

TABLE 11.1-3
(Sheet 3 of 3)

DESIGN BASIS RADIOACTIVE WASTE QUANTITIES

Waste Source		Quantity per Year (ft^3)	Assumptions and Comments
4)	Condensate demineralizers resin	48	Resin replacement four times per year
5)	Filter precoat	16	Filter bed replacement eight times per year at 2 ft^3/replacement
6)	Evaporator bottoms[(2)]	811	

[1] Laundry waste comes from lavatory drains only and amounts to approximately 730 ft^3 per year. This is all processed in the liquid waste system through the miscellaneous waste evaporator. The Laundry Waste Tank was disconnected from the Sanitary Waste System in 1986. The contribution of this waste to evaporator bottoms remains the same as the original license basis. The original license basis was 7,300 ft^3 of laundry waste per year. It was assumed 10 percent of this would be processed in the liquid waste system and the remainder discharged to sanitary waste. (10 percent of 7,300 ft^3 is 730 ft^3, and 10 percent of 14,600 ft^3 is 1,460 ft^3).

[2] Based on the following assumptions:

a) Concentrate from 90 percent of item (a)(1) is reclaimed for reuse. Remainder is concentrated by a factor of 20 for packaging (303 ft^3).

b) Item (a)(2) is concentrated by a factor of 10 for packaging (40 ft^3).

c) Items (a)(3), (a)(4), (a)(6), (a)(7), and (a)(8) are concentrated by a factor of 500 for packaging (141 ft^3).

d) Item (a)(5) is concentrated by a factor of 10 for packaging (108 ft^3).

e) 10 percent of Items (a)(9) and (a)(10) are concentrated by a factor of 10 for packaging (219 ft^3).

[3] Gaseous waste quantities are in standard cubic feet per year.

11.2 RADIOACTIVE WASTE DISPOSAL SYSTEMS SUMMARY

The radioactive waste disposal systems provided in the plant are listed below with some brief information:

System	Comments
Spent fuel and control rod handling and packaging	Fuel transfer canal in Reactor Building, spent fuel pools and shipping cask pit in Fuel Handling Building, refueling bridges in both pool areas. See Sections 9.4 and 9.7 for details.
Incore detector removal and packaging	Special equipment at 346 elevation of Reactor Building
Out-of-core detector removal and packaging	
Purification filter removal and packaging	Shipping cask at 305 elevation of Auxiliary Building
Liquid waste disposal system	Floor and equipment drains, Reactor Building sump and reactor coolant drain tank, cooler and pump in Reactor Building, bulk of system equipment on Elevations 276, 281, and 305 of Auxiliary Building, including drains and sumps, the Chemical Cleaning Building and Elevation 305 of the TMI-2 Auxiliary Building.
Waste gas system	Reactor Building vent header system; bulk of vent headers and system equipment on Elevations 281 and 305 of Auxiliary Building.
Solid waste disposal and packaging	Disposal manifolds, decontamination, storage of waste packages on 305 Elevation of Auxiliary Building, the Waste Handling and Packaging Facility, Interim Solid Waste Staging Facility, Solid Waste Staging Facility, and the Respirator Cleaning and Laundry Maintenance Facility.

The systems treated in detail below are the Liquid Waste Disposal System, the Gas Waste Disposal System, and the Solid Waste Disposal and Packaging System. Additionally, in order to store onsite, until plant decommissioning, the Once-Thru Steam Generators (OTSGs) that were replaced with new Enhanced OTSGs (EOTSGs) as part of the TMI-1 Steam Generator Replacement Project, an Original Steam Generator Storage Facility (OSGSF) was constructed. The OSGSF meets all requirements for this OTSG storage, and in addition will be used to store RCS hot leg piping that was removed for replacement as part of the TMI-1 SGR. The OSGSF is designed to be used as a non-occupied mausoleum whose use is limited to the storage of these large contaminated components until plant decommissioning.

11.2.1 RADIOACTIVE LIQUID WASTE DISPOSAL SYSTEM

The radioactive liquid waste disposal equipment and piping systems are all housed within structures that are designed in accordance with seismic Category I and are designed to retain their integrity in the highly unlikely event of an aircraft incident with the exception of that equipment located in the Chemical Cleaning Building (CCB). The CCB has a seismic bath tub but is not aircraft hardened. Some limited piping is located underground or in non-seismic structures. The major equipment components of the liquid waste disposal system are tanks, pumps, precoat filters, demineralizers, evaporators, coolers, and floor and equipment drains with associated sumps. All such major components which would normally (per design basis) present a radiation hazard to plant personnel are contained within cubicles that are separated from normally occupied areas and adjacent equipment by thick concrete walls. Access to the cubicles is restricted by concrete walls forming labyrinth entry ways. The building ventilation system provides an air velocity barrier (from occupied area to the cubicle) across the entry way door to prevent airborne radioactive particulates or gases from contaminating the atmosphere of occupied areas. Area radiation monitors are strategically located to warn of excessive radiation levels within the area surveyed. Section 11.4 describes the radiation monitoring system in detail.

11.2.1.1 System Functions

The Liquid Waste Disposal System provides operating service functions to the reactor coolant system and spent fuel pools in addition to the collection, containment, and processing of miscellaneous wastes for reuse or disposal. The functions provided are listed below for each case.

a. Operating Service Functions to Reactor Coolant System and Spent Fuel Pools

    1) Chemical shim and volume control for the Reactor Coolant System.

    2) Pressurizer relief suppression, containment, and collection.

    3) Drain and fill the Reactor Coolant System.

    4) Clean up spent fuel pool water.

    5) Process reactor coolant and refueling water for reuse or disposal.

    6) Process spent fuel pool water for reuse or disposal.

b. Miscellaneous Functions

    1) Process miscellaneous wastes from the following:

        a. Radioactive laboratory drains
        b. Building and equipment drains and sumps, including applicable ones in TMI-2
        c. Regeneration of deborating resins
        d. Discharge of spent resins from demineralizers
        e. Discharge of used precoat from precoat filters

2) Process potentially radioactive shower drain waste for disposal.

3) Safely dispose of waste liquids from both functions 1) and 2) above to the river.

4) Provide a means for collection, containment and sampling of potentially contaminated oil for final dispositioning.

The Liquid Waste Disposal System provides for the recovery of concentrated boric acid and purified water from the reactor coolant, the refueling water, and the spent fuel pool water processed in it.

11.2.1.2 System Description

The liquid waste disposal system process flow diagrams are shown on Drawings 302690, 302691, 302692, 302693, 302695 Sh 1, 302695 Sh 2, and 302697. Liquid Waste Disposal System component data are given in Table 11.2-1. Liquid waste processing equipment is essentially divided into two separate trains.

One train provides operating service functions to the Reactor Coolant System and the spent fuel pools (as listed in Subsection 11.2.1.1). This equipment is hereafter referred to as the reactor coolant train. The second train provides storage, treatment, and/or concentration for reuse or disposal for all miscellaneous radioactive wastes. It is hereafter referred to as the miscellaneous waste train. The equipment associated with the reactor coolant train is shown principally on Drawings 302690 and 302691. The reactor coolant evaporator and reclaimed boric acid storage tanks are shown on Drawing 302692. The reactor coolant and waste evaporators are cross-connected so either evaporator can serve either function if required.

The equipment of the liquid waste disposal system normally associated with the reactor coolant train and that associated with the miscellaneous waste train are identified in Table 11.2-2.

Controls that permit the operator to select tanks and pumps or demineralizers and a process piping route, as required for the function to be performed, are provided for the equipment and process lines involved in chemical shim and volume control and the filling or draining of the reactor coolant system.

Once selected, the process equipment and process lines are reserved exclusively to the function set up. The selection process is accomplished manually with separation of the individual trains as provided by procedural control.

Controls similar to those above are provided for cleanup and/or evaporation of reactor coolant and cleanup of spent fuel pool or refueling water. These controls permit the operator to set up a process route for the function required until the process path is terminated by the operator. Thus, the integrity of feed solutions to the reactor coolant system, bleed solutions from the reactor coolant system, spent fuel pool or refueling water are preserved in all stages of processing.

The only portion of the equipment normally utilized for cleanup of reactor coolant or spent fuel pool water that is utilized for processing miscellaneous waste solutions is precoat filter B. This is not a normal process route for cleanup of miscellaneous wastes. Automatic interlocks are not used in the manual mode (normal operating mode) and procedural controls are provided to prevent the inadvertent routing of miscellaneous wastes through any other item of reactor coolant processing equipment downstream of precoat filter B. The normal miscellaneous waste process routes can be selected by the operator for the function required until the operator terminates the process. Other process routes are provided for functions of infrequent occurrence or emergency situations. The process route selection is accomplished manually with separation of the individual trains by procedural control.

Liquid Waste Disposal System piping is so arranged that all liquids collected in equipment of either the reactor coolant or miscellaneous waste trains must be routed through the reactor coolant or miscellaneous waste evaporators and the condensate demineralizers and be collected in the waste evaporator condensate storage tanks or CC-T-2 before they can be discharged into the effluent from the mechanical draft cooling tower basin, or reused. Portable demineralizers may be used instead of the permanent evaporator and/or condensate demineralizers. A minimum effluent flow rate of 5000 gpm is always maintained from the mechanical draft cooling tower basin for dilution of radioactive liquid wastes during their release.

During the design of Unit 2, certain radwaste interconnections were designed to provide means of transfer for liquid radwaste between the units. With the accident at TMI-2, this transfer could have resulted in excessive contamination of Unit 1 from Unit 2 liquid radwaste. Therefore, methods were established to isolate the interconnections.

The following liquid waste disposal interfaces between Units 1 and 2 are isolated:

a. The connection from the Unit-2 Reactor Coolant Bleed Holdup Tanks to the Unit-1 Reactor Coolant Waste Evaporator.

b. The ability to transfer condensate from the Unit-1 Miscellaneous Waste Evaporator to Unit-2 Evaporator Condensate Test Tanks.

c. An interconnection to permit the movement of evaporator concentrates between units. Reclaimed boric acid as well as bottoms from the miscellaneous waste evaporator could have used this path.

d. The ability to move spent ion exchange resin between units.

NRC approval was received to allow reconnection of the interfaces. The following interface has been reestablished:

a. The contents of the Unit-2 Neutralizer Tanks, Contaminated Drain Tanks, Reactor Coolant Bleed Holdup Tanks, Auxiliary Building Sump Tank, and Miscellaneous Waste Holdup Tank can be transferred to Unit-1 for storage and processing in the Unit-1 Liquid Waste Disposal System. Similarly, Unit-1 miscellaneous waste water can be transferred to the Unit-2 Miscellaneous Waste Holdup Tank. However, the Unit-2 Neutralizer Tanks, Contaminated Drain Tanks, and the Reactor Coolant Bleed Tanks are administratively isolated. The Unit-2 Auxiliary Building Sump Tank and the Miscellaneous Waste Holdup Tank are also part of the Unit-2 Post Defueling Monitored Storage (PDMS) Program.

The three (3) sumps (asterisked in Table 11.2-2) normally collect groundwater that may seep into the areas they drain. Since they are located in areas that might conceivably receive radioactively contaminated water in the event of a pipe leak or rupture, the effluent from the sumps is intermittently discharged into the miscellaneous waste storage tank or into the Auxiliary Building sump.

It is anticipated that the activity level of the contents of the laundry waste tank (collects water from the decon showers and lavatory drains) will normally be less than 10^{-7} ûCi/cm^3. (Starting in 1986, the hot laundry was not used so water from this source was not sent to the laundry waste tank. This change would not alter the activity of the water appreciably from the original license basis of 10^{-7} ûCi/cm^3.) The contents will be pumped to the miscellaneous waste evaporator or the neutralizer feed tank for processing in the liquid waste system.

Condensate collected in the waste evaporator condensate storage tanks may be transferred to the reactor coolant bleed tanks for feed to the primary system, rather than being discharged to the river. It is also possible to transfer the waste evaporator condensate tank contents to the reclaimed water tank (of the Chemical Addition System) for miscellaneous uses throughout the radioactive waste systems.

All concentrates are packaged in the solid waste disposal system for shipment offsite. As an alternative, concentrated liquid waste may be shipped offsite to a licensed processor for volume reduction prior to disposal. Concentrate collected in the reclaimed boric acid tanks may be reused in the reactor coolant system or spent fuel pools or packaged for shipment offsite.

Any wastes stored in the evaporator condensate storage tanks or the reclaimed boric acid tanks may be reprocessed through the waste evaporator or Reactor Coolant Evaporator for further decontamination or concentration.

Potentially contaminated oil from plant components such as pumps, motors, etc. can be placed in a collection tank where it will be contained and sampled. If contaminated, the tank can be gravity drained to the Solid Waste Disposal System for processing or it can be drained for disposal as appropriate. This separate system is located on the 331' 0" elevation in an area of the Auxiliary Building known as the Chemical Addition Room.

11.2.1.3 Methods Of Operation

While within the equipment of the liquid waste disposal system, the integrity of all solutions involved in chemical shim adjustment, volume control of reactor coolant inventory, and filling or draining of the reactor coolant system are protected by procedural controls.

Control of tritium concentration in the primary coolant is achieved by disposal, as required, of condensate produced from the evaporation of reactor coolant. The amount of condensate that must be disposed of will depend on the actual building rate of tritium in the reactor coolant.

All normally radioactive liquids collected within the liquid waste system have to pass through an evaporator, as condensate, and a mixed bed demineralizer prior to being collected for reuse or disposal to the river. A portable demineralizer may be used instead of the permanent evaporator and/or evaporator demineralizer. Such disposals to the river are on a batch basis with activity analyses (including an isotopic breakdown) of the samples from the batch being obtained prior to disposal. Based on the batch analysis and the diluent flow rate from the mechanical draft cooling tower basin, a maximum flow rate for the disposal of the batch is determined. The flow rate of each such batch disposal of radioactive liquids is controlled to ensure that the activity in the basin effluent being discharged to the river is within ODCM limits. Set points on the flow and activity monitors (providing direct surveillance over the discharge of a batch) are set accordingly before initiating the discharge.

After liquid waste enters the effluent from the mechanical draft cooling tower basin, the mixture travels approximately 275 ft before discharging into the west channel of the Susquehanna River, approximately 600 ft downstream of the river water intake structure.

Batches of liquid waste are not disposed of in the effluent from the mechanical draft cooling tower basin if the effluent's flow rate is less than 5000 gpm. Dilution credit is taken for cooling tower effluent flow rates up to 38,000 gpm. On an average annual basis for liquid waste disposal, the summation is made on the basis that the flow rate of cooling tower basin effluent has been the minimum 5000 gpm throughout the year. This method of operation, under design basis conditions of reactor coolant activity and quantity, assures that the activity in the cooling tower basin effluent is within ten times the concentration levels specified in 10CFR20.1001-20.2401, Appendix B, Table 2.

All concentrates produced from the evaporators are either reclaimed for reuse or packaged for shipment offsite. The contents of the miscellaneous waste evaporator feed tank will be sampled and analyzed before disposal. All slurries (spent resin and filter precoat), produced as a result of radioactive system operation, are packaged for shipment offsite. Potentially contaminated oil is sampled for radiological content. It is disposed of by gravity drain to the Solid Waste Disposal System or by other appropriate solid waste management methods.

The processing equipment will be decided sometime in the future.

11.2.1.4 System Design Evaluation

Since most of the liquid waste disposal system equipment is contained within shielded cubicles (drained below grade) located inside Class I structures designed to withstand the hypothetical aircraft incident, it is not considered credible that any single accident could violate the multiple barriers required to release significant quantities of radioactive liquids to the environment. Two tanks and their pumps are located in the Chemical Cleaning Building (CCB) along with the space for a rented demineralizer system. The CCB has a seismic bath tub but is not aircraft hardened. The CCB has been previously evaluated (NUREG 0591) and used to hold liquid from the TMI-2 accident and to process that liquid through demineralizers located in its lower level. The radiological content of the accident water bounds that of any water expected to be generated by TMI-1.

Discharges of liquid waste are initiated in accordance with strict administrative procedures. If an operator inadvertently initiates a release of excessive amounts of liquid waste, either of two radiation monitors would terminate the release.

An analysis relative to the release of liquid wastes to the river was made and is summarized in Table 11.2-4. The analysis simulates operation of the plant wherein primary coolant is continuously let down for chemical shim adjustment and immediately goes through one cycle of cleaning followed by concentration in the evaporator for boric acid recovery. The condensate from the evaporation is processed through the evaporator condensate demineralizers, as necessary, to the evaporator condensate storage tanks. Subsequently, the condensate is sampled, analyzed, chemically adjusted for pH if required, and disposed of to the river via the mechanical draft cooling tower basin effluent or reused. A summary of assumptions for the analysis is presented in Table 11.2-3.

Based on the above indicated systems and equipment, the design basis waste liquid quantities generated annually are 49,000 μCi of mixed fission products (excluding tritium) and 5.095×10^{8} μCi of tritium. Using the 5000 gpm design basis number for the annual average effluent flow rate, the Unit 1 annual discharge volume for which dilution credit may be taken is 1.02×10^{13} ml. This results in an annual average concentration of mixed fission products (excluding tritium) and tritium in the plant effluent of 4.8×10^{-9} μCi/ml and 4.99×10^{-5} μCi/ml, respectively. This annual average concentration of mixed fission products (including tritium) is within concentration levels of 10CFR20.

The detailed analysis of compliance with 10CFR50 Appendix I, in accordance with Reg. Guides 1.109, 1.110, 1.111, and 1.112, is presented in References 18 and 19.

The Waste Oil Storage System has the capacity for storage of up to 700 gallons of oil in WDL-T-17 and 25,000 gallons of oil in WDL-T-27. WDL-T-17 satisfies ASME codes, it is mounted to meet Seismic Class II criteria, and is located within a Seismic Class I curbing which represents a bathtub. The associated piping serving as the principal boundary of containment of the oil is within the curbing. WDL-T-27 was designed and fabricated to NFPA and 1968 ASME code, Sec VIII. It was built Seismic I and is located in a Seismic I structure. Collected oil is typically not contaminated, however, oil has been found with gross activity levels on the order of 1.0×10^{-5} μC/ml. Worst case pathway analyses show that offsite concentrations resulting from release of the tanks is well below regulatory limits. The designs therefore provide sufficient provisions for safe operation. Oil with activity levels orders of magnitude greater than analyzed can be contained without a safety concern.

11.2.2 RADIOACTIVE GAS WASTE DISPOSAL SYSTEM

The equipment and piping of the waste gas system are all housed within structures that are designed in accordance with seismic Category I and are designed to retain their integrity in the highly unlikely event of the hypothetical aircraft incident. The major equipment components of the system are gas decay tanks and gas compressors. The gas spaces of those tanks of the liquid waste system that are listed in Table 11.2-1 as being vented to the vent header system also serve as part of the waste gas system. All major components of the waste gas system which would normally (per design basis) present a radiation hazard to plant personnel are contained within cubicles that are separated from normally occupied areas and adjacent equipment by thick concrete walls. Access to the cubicles is restricted by concrete walls forming labyrinth entry ways, each of which is provided with a door. The building ventilation system provides an air velocity barrier between occupied areas and the interiors of the cubicles to prevent possible contamination of the atmosphere of the occupied areas. The area radiation monitors serving the auxiliary building provide surveillance over equipment of the liquid waste disposal and waste gas disposal systems. Section 11.4 describes the radiation monitoring system in detail.

11.2.2.1 System Function

The waste gas system provides for the safe collection and storage of gases evolved from reactor coolant in all tanks or items of equipment where this might occur.

The following items are indicated in Table 11.2-1 as being vented to the atmosphere of the cubicles in which they are located: spent resin storage tank, used filter precoat tank, neutralizer tank, neutralizer feed tank, neutralized waste storage tank, evaporator condensate storage tank, miscellaneous waste evaporator feed tank, laundry waste tank, and the precoat filters. Although some of these items may contain highly radioactive material, essentially none of it is in the form of radioactive gases. In all cases, the radioactive material contained in these items of equipment is either in the form of crystalline, dissolved ionic, or resin-fixed ionic solids.

Also, the water in these items is either clean water (not primary coolant that has been provided from the plant supply for regenerating resin, flushing resin or filter precoat from their process beds, and so forth) or previously degassed primary coolant (miscellaneous waste evaporator feed tank only) that contains only dissolved ionic solids. Therefore, the probability of radioactive gases evolving from the normal contents of these items of equipment is extremely low. However, in the unlikely event that significant amounts of radioactive gas did escape from any of these tanks, they would pass into the exhaust ventilation system for the Auxiliary and Fuel Handling Buildings, through the roughing, HEPA, and charcoal filters, and be sensed by the unit vent radiation monitor.

Volumetrically, hydrogen is the principal gas evolved in those tanks which normally might receive reactor coolant and is vented to the low pressure vent header system or to the waste gas compressors. The radioactive fission products, activated dissolved gases, and so forth, contribute an extremely small fraction of the total volume of gas liberated. The waste gas system has been designed to provide a blanket of inert nitrogen gas in which to collect the gases evolved from the reactor coolant. The mixture of gases collected (nitrogen, hydrogen, and radioactive gaseous isotopes) is compressed and stored for decay of the radioactive components prior to recycle (as blanket gas) or disposal to the atmosphere.

The low pressure portions of the vent header system are protected from overpressure by relief valves off the piping of the vent header proper, and by water filled loop seals or relief valves on the liquid waste storage tanks whose gas spaces form a portion of the system. Protection of tanks and piping against excessive vacuum in the event of a combination of highly unlikely equipment malfunctions has been provided. The waste gas decay tanks are protected from overpressure by individual relief valves, as are the waste gas compressors.

11.2.2.2 System Description

Drawing 302694 and Figure 11.2-7 are process flow diagrams of the Waste Gas Disposal System and the waste gas system compressors, respectively. Component data for the waste gas system are given in Table 11.2-5.

The waste gas vent header system is essentially split into two sections, one section within the Reactor Building and one section within the Auxiliary Building. Condensing water vapor or liquids, entering the section of the vent header system within containment, drain to the reactor coolant drain tank, while those entering the vent header system within the Auxiliary Building drain to the miscellaneous waste storage tank. The vent header from the reactor coolant drain tank discharges to the miscellaneous waste storage tank. The gas spaces of the miscellaneous waste storage tank and the three reactor coolant bleed tanks are joined as an intermediate gas storage area and discharge the gases they collect to the suction of the waste gas compressors via an intermediate waste gas delay tank. Make up tank gas sample return and waste gas release are routed directly downstream of the waste gas delay tank and upstream of the waste gas compressors to avoid hydrogen pockets in the low pressure radwaste liquid and gas tanks. Prior to makeup tank venting, the decay tank will be filled with diluting nitrogen to insure that the H_2 content complies with ODCM limitations.

The compressed gas portion of the waste gas system starts at the waste gas compressors and includes the three waste gas decay tanks. These tanks provide for a minimum of 30 days of storage for gases during normal operation prior to release to the atmosphere. Release is possible prior to the minimum decay time of 30 days if calculations done in accordance with Reference 27 indicate that the radioactive gas concentration is within 10CFR20 and 10CFR50, Appendix I limits. When the currently filled decay tank is pressurized to 80 psig, an automatic sequencing system preferentially selects a new waste gas decay tank for filling based on the pressure within the tank (i.e., it being less than 80 psig) and that waste gas is not being discharged from it at that time. Administrative approval is required to manually initiate either recycle or release to the atmosphere of waste gases stored in any waste gas decay tank.

11.2.2.3 Methods of Operation

Except for initiating the makeup tank sample and waste gas venting and the recycle or disposal of compressed waste gases stored in the waste gas decay tanks, the operation of the waste gas system is entirely automatic. One waste gas compressor comes on automatically, removing gases from the vent header system as required to maintain the pressure in the system at a maximum of about 16.4 psia.

The second waste gas compressor is on standby and automatically starts, as required, to backup the running compressor. Before receiving waste gas, the decay tanks will be filled with diluting nitrogen to insure that the H_2 content complies with ODCM limitations.

Compressed waste gases are sampled shortly after the completion of filling of a waste gas decay tank. The analysis of this sample is the basis for determining whether the gas in the tank should be reused (as makeup blanket gas to the vent header system) or disposed of to the environment. If stored gas is to be recycled, recycling may be initiated at any convenient time following analysis of the initial sample.

Any release of radioactive gases from the waste gas system will be made as follows:

a. After analysis of samples taken from batches prior to their release, and establishing flow and radiation level alarm point per Reference 27.

b. Only through paths that require positive manual operation in order to effect the release.

c. Through a path in which the gas is monitored twice, once as it leaves the decay tanks, and again after it mixes with other gases in the Auxiliary Building ventilation system. Either monitor will terminate the gas discharge automatically in the event its set point is exceeded.

11.2.2.4 System Design Evaluation

Waste Gas Disposal System equipment and piping (external to the Reactor Building) are designed for pressures considerably in excess of those capable of being applied. The maximum gas pressure capability in the low pressure vent header system is positively limited to approximately 8.0 psig by relief valves and the overflow loop seals on the tanks vented to it. A highly unlikely combination of equipment malfunctions and operator inattention is required to blow the water in these loop seals. The absolute minimum volume provided by the gas spaces of tanks associated with the low pressure vent header system is approximately 4000 ft^3. The maximum gas flow rate anticipated in the system (80 scfm) would have to occur for a minimum of about 27 minutes, with no removal of gases by either waste gas compressor, before the loop seals would blow. All gas flows in the waste gas system of this order of magnitude are initiated by operator action, by either pumping water into the system from a source not connected to the vent header system, or by venting the pressurized gas space of a tank to the system. Therefore, the operator has adequate time to terminate the gas displacement or discharge in the event that both waste gas compressors fail.

The design pressure of the high pressure waste gas system piping and equipment is 150 psig while the design discharge capability of the waste gas compressors is only 80 psig. Further, each waste gas decay tank is protected by its own relief valve, which is set to relieve the tank at 85 psig. The relief valve on the discharge of the compressors is also set to relieve at 85 psig. Consequently, it is not considered credible that a rupture or major failure resulting from overpressure could occur in the piping or other components of the high pressure portion of the waste gas system.

Accidental discharges resulting from the relieving of a waste gas decay tank or compressor relief valve are not considered credible as the operator will have approximately 8 minutes (between receiving the alarm that the automatic sequencer has not been able to transfer waste gas discharge to a fresh tank, and the popping of the relief valve on the overfilled tank) in which to take the action required to get an empty, or partially empty tank on the line, and/or to terminate the gas displacement or discharge into the vent header system.

The potential for adverse concentrations of hydrogen and oxygen in the Waste Gas (WG) System is very low. A nitrogen overpressure is maintained on the WG headers; therefore, the only source of hydrogen or oxygen in the system during operation is from the Reactor Coolant System (RCS). The Makeup tank is vented directly to a Waste Gas Decay Tank (WGDT) with nitrogen dilution to minimize the potential for a combustible mixture in the vent header. The WGDT can withstand the pressure peak of a detonation.

Continuous online gas analyzers at Unit 1 monitor for hydrogen and oxygen in the waste gas system. The analyzers are described in Section 9.2.2.

Since all the piping and equipment are housed in hypothetical aircraft incident proof Class I structures, within cubicles enclosed by concrete shield walls, it is not considered credible that any physical damage could occur to the waste gas system that would release radioactive gases to the environment from any of the components of the waste gas system in an uncontrolled manner.

Although such an event is not considered credible, an analysis of the rupture of a waste gas decay tank is presented in Chapter 14 to demonstrate that the results of such an accident are well below the limits of the 10 CFR 100 guidelines.

All normal releases of radioactive waste gases from the Waste Gas Disposal System will be made in a controlled manner, with double monitoring of the release, to assure that annual average activity levels at or beyond the site boundary will be within the limits of 10CFR20.

An analysis was made of waste gas release from the Waste Gas Disposal System. The analysis assumes a maximum 90 day holdup period for compressed radioactive gases prior to their release to the environment, that all gas compressed during the year is released (no gas recycled) and averages the release over the year. Assumptions of this analysis are presented in Table 11.2-6 and the results of the analysis are summarized in Table 11.2-7.

This analysis indicates that Kr85 is essentially the sole contributor (99.9 percent) to the activity in the waste gas system's discharge to the unit vent, and that the annual average concentration resulting at the site boundary is 0.00263 MPC for the mixture estimated to be discharged.

A detailed analysis which demonstrates compliance with 10CFR50, Appendix I gaseous releases in accordance with Regulatory Guides 1.109, 1.110, 1.111, and 1.112 is presented in References 18, 19, and 27.

11.2.3 RADIOACTIVE SOLID WASTE DISPOSAL SYSTEM

Interim Mobile Solidification System for packaging radioactive solid and liquid wastes is located outside of the Auxiliary Building. Filled packages will normally be loaded aboard a truck adjacent to the Solidification Building.

Packaging of wastes for offsite shipment is in Department Of Transportation approved containers supplied and transported by a subcontractor licensed for such activity. Shipping packages are shielded with overpacks, as required.

Five general types of waste are produced, processed, and shipped from the TMI site as solid radioactive waste. These wastes are:

a. Concentrated liquid waste (evaporator bottoms)

b. Used precoat (spent powdered resin)

c. Spent resin (bead type)

d. Dry compactible trash

e. Dry noncompactible trash

Dry trash is either shipped offsite directly following compaction (to reduce the volume) or shipped to an offsite processor for decontamination and/or volume reduction prior to recycle or disposal. Appropriate packaging of dry trash is performed in accordance with applicable shipping and disposal regulations.

Concentrated liquid waste*, and contaminated used precoat and spent resin will be solidified prior to being shipped offsite for disposal where required by applicable regulations. When solidification is not required for contaminated precoat and spent resin, they will be properly dewatered prior to being shipped offsite for disposal. Contaminated used precoat and spent resin may be shipped to a licensed offsite processor for volume reduction prior to disposal. Permanently installed plant equipment does not currently exist to solidify radwaste.

A two part program has been initiated to solidify these wastes. For the short term, until the permanent system is available, Unit 1 will use a contractor supplied mobile solidification system.

* Note: If not being shipped for disposal, liquid waste may be shipped in liquid form to a licensed processor for volume reduction prior to disposal. The shipment must comply with DOT regulations for shipment and license conditions of the recipient.

11.2.3.1 System Function

The function of the Solid Waste Disposal System is to package radioactive solid and concentrated liquid wastes in such a manner as to ensure minimum exposure of unit personnel during the packaging process, produce waste packages that provide protection for the public during their transportation from the unit to an offsite processor or to the ultimate disposal site, and meet ultimate disposal requirements for waste packages sent for direct disposal.

11.2.3.2 System Description

The Solid Waste Disposal System consists of the spent resin (WDL-T-4) and used precoat (WDL-T-5) tanks, and the slurry pump and associated piping system in the Auxiliary Building, as well as the following facilities outside of the Auxiliary Building: The Solidification Building, the Waste Handling and Packaging Facility (WHPF), the Interim Solid Waste Staging Facility (ISWSF), the Solid Waste Staging Facility (SWSF), and the Respirator Cleaning and Laundry Maintenance (RLM) Facility.

All waste materials are packaged for shipment, processing and disposal IAW the regulations found in 49 CFR 173, 10CFR 61 and 10 CFR 71 for the safe transport of radioactive material.

The Solidification Building (located immediately south of the auxiliary building) is the packaging area for spent resin, used precoat and concentrated waste. Spent resin and used precoat are packaged into disposable liners and placed inside of a shielded container inside the solidification building. The shielding is designed to be sufficient to minimize exposure of operating personnel for the worst case of spent resin or used precoat resins. A disposal liner without additional shielding is used for packaging concentrated waste. The spent resin and used precoat are both dewatered sufficiently to be shipped to an off-site vendor for additional volume reduction prior to the final disposal of the material. Concentrated waste is packaged and shipped as a liquid. The full container of approximately 1000 gallons of concentrate waste is transferred from the solidification building to a shielded container for physical protection during its transportation to an offsite vendor for additional processing/drying prior to final disposal.

The approximate radwaste production rates are as follows:

Evaporator Bottoms	10 ft^3/week
Used Precoat	<1 ft^3/week
Spent Resin	<2 ft^3/week

The available storage capacity for unsolidified radwaste is as follows:

Used Precoat	300 ft^3
Spent Resin	500 ft^3

The WHPF was designed for processing and packaging Dry Activated Waste (DAW) and contaminated tools and equipment. The following functions may be performed in the WHPF:

a. Sectioning and disassembly of large pieces of equipment to a size that will fit into packages such as a 55 gallon drum or a 4 ft x 4 ft x 6 ft container, or 20' cargo containers. This size reduction is accomplished by use of plasma arc and oxy-acetylene torches as well as hand-held tools.

b. Decontamination of tools and equipment by an abrasive blaster, as required.

c. Compaction of DAW.

d. Packaging non-compactable trash and equipment into Low Specific Activity (LSA) boxes, drums, or 20 foot cargo containers.

e. Temporary staging of radioactive material prior to, during, and after processing.

f. Transferring radioactive waste after processing, sorting, and/or packaging to an onsite staging facility.

The Interim Solid Waste Staging Facility (ISWSF) is an above ground storage facility used for temporary storage of radioactive material that is properly packaged for off-site shipment.

The RLM facility is designed for maintenance and cleaning of respirators and clothing used for prevention of contamination. The Operations in the RLM Facility include the temporary staging of these radioactive materials, and processing and compaction and packaging of DAW.

The Solid Waste Staging Facility (SWSF) is an underground facility used for storage of high dose rate resins. The SWSF design meets the requirements of Reg. Guide 1.143, July 1978.

11.2.3.3 System Storage Capacity

Prior to the accident at TMI-2, the solid radwaste system installed at TMI-1 was shared by both TMI-1 and TMI-2. The TMI-2 design utilized the TMI-1 solid radwaste system to provide for the solidification of TMI-2 waste by transferring all TMI-2 wastes to TMI-1 and by solidifying them at TMI-1. As a result of the isolation/separation requirement of the October 2, 1985 NRC letter to GPUN (See GPUN document reference 5211-85-3239) authorizing TMI-1 restart (and lifting the 1979 Commission imposed shutdown order) and containing this Condition of Operation (as contained in Commission Order CLI-85-9, dated May 29, 1985), the TMI-1 solid radwaste system has been completely separated from TMI-2. However, by letter dated August 30, 1989 (Letter No. C311-89-2073) GPUN advised the NRC of certain planned activities that would involve use of TMI-1 facilities to process material contaminated at TMI-2. These activities included use of the TMI-1 laboratory to analyze TMI-2 samples, use of tools and instruments having residual contamination from TMI-2 at TMI-1, use of common shipping containers and the TMI-1 trash compactor for radioactive waste from both units and consolidation of decontamination facilities. By letter dated January 10, 1990, the NRC staff agreed that the scope of the Condition of Operation provided in the October 2, 1985 NRC letter did not preclude the types of activities discussed in GPUN letter of August 30, 1989.

The Solid Waste Disposal System also provides capability for spent resin and used precoat to be shipped either solidified with cement or dewatered in licensed shipping casks. If the waste is dewatered, the water is returned to the floor drain system and subsequently to the auxiliary building sump.

Storage capacity is provided within the liquid waste system estimated to be sufficient to provide the periods indicated below between packaging of the various wastes:

Type of Waste	Period between packagings of waste
Reactor Coolant quality concentrated liquids	about 26 weeks (average)
Spent resins	about 1 year
Used precoat filter material	about 2 years
Potentially Contaminated oil	about 2 years

Miscellaneous waste concentrated liquids are transferred directly to a liner. Liquids are transferred to liner packaging for offsite processing.

Based on prior analyses of samples obtained from the wastes to be packaged and the PCP, the operator determines the approximate maximum quantities of the wastes that may be put into a container. The system also has the ability to solidify contaminated oil, if necessary. The maximum capacity of the contaminated waste oil tank (WDL-T-17) is 750 gallons, which approximates the total system capacity.

11.2.3.4 System Design Evaluation

LSA waste (solidified evaporator bottoms, dry trash and other LSA waste) can be stored in existing space in the TMI-1 Auxiliary Building. Other storage space is available outside the Auxiliary Building, and has the capacity of:

a. 100, 55 gallon drums unshielded (compacted trash)

b. 20, 100 ft^3 LSA boxes unshielded

This amount of storage would provide storage for up to six months. Solidified evaporator bottoms and packaged used precoat and spent resin could be stored in the SWSF. Fifty-five gallon drums and boxes of uncompacted waste will be stored in the ISWSF. That Facility has the capacity for a minimum of six months of solid waste storage of material other than evaporator bottoms, used precoat, and spent resin.

11.2.4 OPERATING POLICY ON RADIOACTIVE DISCHARGES TO THE ENVIRONMENT

The policy for the control of radioactive discharges to the environment from the unit is (1) to discharge the minimum quantity of radioactive materials (either liquids or gases) to the environment which is consistent with the capability of the equipment provided in the unit, and (2) to meet all the conditions of the operating license and all applicable federal, state, and local regulations.

With respect to disposal of liquid wastes, no specific holdup times are utilized to take advantage of radioactive decay since the degree of reduction in the activity discharged from the unit achieved by radioactive decay is insignificant when compared to that which can be achieved by the equipment installed in the unit for decontaminating the liquid wastes. This equipment consists of cation and mixed-bed demineralizers, precoat filters, and waste evaporators, and is so interconnected that liquid wastes in any stage of processing may be reprocessed as required to reduce activity levels and quantities in the effluent from the unit. Although no credit is taken for radioactive decay of liquid wastes, it is anticipated that there will be a minimum period of about two days between the time a batch of primary coolant is letdown and the time that demineralized distillate (produced from that batch) is discharged into the effluent from the mechanical draft cooling tower basin.

For the design basis quantities and activity levels of liquid wastes to be discharged to the environment (see Table 11.2-3, Item e.2), credit was taken for only one pass of reactor coolant through the cation demineralizer prior to evaporation and subsequent demineralization of the evaporator distillate which is to be discharged to the environment. Since the unit is designed to recover and reconcentrate boric acid from the reactor coolant letdown for the purpose of using the concentrated boric acid produced as makeup to the Reactor Coolant System, it is necessary to achieve an average decontamination factor of about 200 for the reactor coolant (letdown over a core lifetime) prior to evaporation to ensure that the activity level in the reclaimed boric acid does not contribute to buildup of activity in the Reactor Coolant System or offer any significant radiation hazard to operating personnel. The actual decontamination factor required varies from a minimum of about 60 for letdown produced during dilution of the refueling water boric acid concentration, to 700 for letdown produced at the point in core lifetime when resin deboration replaces bleed and feed as the means of adjusting chemical shim concentration. Therefore, the actual decontamination factor achieved in the cation demineralizers during normal operation is, conservatively, about four times greater than that assumed here.

With respect to the discharge of tritium to the environment, there is no commercial equipment available to remove tritium from the liquid wastes prior to dilution in the cooling tower effluent.

In implementing the above stated policy with respect to disposal of gaseous wastes from the radwaste system, a combination of holdup for radioactive decay and filtration through roughing, HEPA and charcoal filters is utilized prior to release. For the radioactive waste gases stored in the waste gas system, a design maximum storage capacity is provided to permit storage of radioactive gases for periods of up to a maximum of 90 days during normal operation prior to release to the environment. However, it is anticipated that a minimum holdup time, prior to release, of about two weeks might be anticipated during periods when equipment has failed, malfunctioned, or is unavoidably out of service. For waste gas releases during normal operation, the minimum holdup period prior to release of gas from the waste gas system is set at 30 days decay time. Releases prior to 30 days are allowed after calculations are performed (Reference 27) to verify that the radioactive gas concentrations and their associated dose or dose commitment to an individual, in an unrestricted area, are within 10CFR20 and 10CFR50, Appendix I limits. When purging of the Reactor Building atmosphere is required to obtain access, these releases will be made through roughing and HEPA filters for removal of particulates and charcoal filters for the removal of iodine. The Kidney Filter system is utilized to decrease particulate and iodine radioactivity prior to purging.

TMI-1 has equipment to maintain the radioactive waste releases to the environment to values as low as reasonably achievable (ALARA). The design basis analysis presented herein (in which activity levels associated with 1 percent failed fuel are postulated) indicates that the annual average mixed, identified fission product (other than tritium) concentration activity is at 1/12,500 of its 10CFR20 MPC level and is 0.015 times the 10CFR20 MPC level for tritium in the mechanical draft cooling tower effluent released to the river and that the waste gases released to the atmosphere through plant vent achieve 1/380 of their 10CFR20 MPC level at the site boundary on an average basis. Therefore, the equipment provided is capable of achieving environmental activity levels substantially below applicable 10CFR20 MPC values for all radioactive wastes even under conservative assumptions of activity release from the fuel and equipment decontamination capability. Further, redundant equipment is provided for all phases of liquid waste processing so that loss of cleanup capability due to equipment failure is unlikely.

References 18 and 19 demonstrate compliance with 10CFR50, Appendix I requirements for liquid and gaseous releases. Potential pathways for releases and subsequent doses to population are also analyzed in References 18 and 19.

Section 11.2.5 presents analysis of the activity concentration in the Susquehanna River downstream of the plant for the design basis (activity levels in reactor coolant associated with 1 percent failed fuel) case.

11.2.5 DILUTION FACTORS IN SUSQUEHANNA RIVER

Three Mile Island is located in a section of the Susquehanna River known as Lake Frederick (refer to Figure 2.1-3). Lake Frederick is ponded by the York Haven Dam and is divided into three major flow channels by Three Mile Island and Shelley Island. A section of the York Haven Dam blocks the east channel approximately one mile downstream of the TMINS. The remainder of the York Haven Dam extends in a south, southeasterly direction from the southern tip of Three Mile Island to the west bank of the Susquehanna River, where the York Haven Hydroelectric Station is located. At this point, the west bank of the river lies east of the unit. The river flow in this area is from north to south. TMINS is located near the north end of the Three Mile Island and discharges into the center channel of the pond about 1500 ft downstream of the unit.

The nominal flow capacity of the York Haven Hydroelectric Station is 16,600 cfs with all the turbines running. Therefore, river flows below approximately 17,000 cfs can pass entirely through the hydrostation. The hydrostation can induce spillage over the dam below these flows, however, by reducing the number of turbines operating. Flows in excess of about 17,000 cfs or in excess of the capacity of the operating turbines spill over the York Haven Dam and pass through the Conewago Falls. The hydrostation operates as a run-of-river unit when river flows drop below the operating capacity of the turbines. The operating schedule of the turbines is adjusted such that river flows entering Lake Frederick leave at the same flow rate.

Travel time and mixing characteristics of TMINS liquid effluents were studied by Sutron Corp. in the early 1980s. Sutron modeled characteristics at river flows ranging from 4700 cfs to 85,500 cfs (Reference 29). This accounts for about 92% of the river flow conditions near TMINS. Sutron found the distribution of river flows through the three channels varies with total river flow. The west channel remains the most constant, ranging from 55% of the total flow at extremely low river flows to 53% during river flows of 85,500 cfs. The west channel also contains the talweg of the river. The proportion of flow in the center channel varies inversely with flows in the east channel. The east channel flow varies from 0% when the river is not spilling over the east dam to 15% at flows of 85,500 cfs. Conversely, about 45% of the river passes through the center channel when the east channel has zero flow. The proportion of river flow in the center channel drops to about 32% when the east channel flows increase to 15%.

When the entire river flow passes through the hydrostation, liquid effluents from TMINS enter the center channel of the pond, travel along the west bank of Three Mile Island, finally reaching the York Haven Dam. The surface areas of the center channel impacted by TMINS effluents increases to about 625 feet at the dam. From there, this band travels along the face of the dam narrowing to approximately 200 feet at the head race section of the hydrostation. Throughout this reach, the amount of river flow impacted by discharges from TMINS remains fairly constant at about 15%. The York Haven Hydrostation discharges towards the eastern bank of the river downstream of the York Haven Dam. At extremely low flows, this results in a complete mixing of the water impacted by TMINS effluents with the rest of the river.

At river flows above the operating capacity of the hydrostation, the portion of the river impacted by TMINS effluents travels along and eventually over the dam and through the Conewago Falls. Flows over the dam are then compressed towards the east side of the river by the discharge from the hydrostation. This area of the river, the York Haven Dam, Conewago Falls and the discharge from the hydrostation, accounts for the most significant contribution to mixing of TMINS effluents with the rest of the river. At medium flows, approximately 35% of the area of the river just downstream of the hydrostation is impacted by TMINS effluents. This increases to about 68% as the total river flow increases to about 85,000 cfs. At medium and high flows, effluents from TMINS do not mix completely with the Susquehanna River as far downstream as Lake Clark.

Dilution of TMINS liquid effluents follow similar patterns to the mixing characteristics. The most substantial dilution occurs as the TMINS effluent enters the Susquehanna River at the main station outfall. Only a small amount of additional dilution occurs from the outfall to the York Haven Dam. However, total dilution exceeds 80% (dilution factor of 5) prior to spilling over the dam or reaching the York Haven Hydroelectric Station during all flow conditions. Additional dilution, second only to the initial discharge, occurs as the river passes over the dam, through the Conewago Falls or through the hydrostation. Effluents are diluted by essentially 100% by the time effluents reach the head waters to Lake Clark.

The criteria for radioactive releases from the unit that establish the maximum activity level in the downstream Susquehanna are: (1) that the mechanical draft cooling tower basin effluent cannot exceed ten times the concentration levels specified in 10CFR20.1001-20.2401, Appendix B, Table 2 while being discharged to the river, and (2) that dilution credit will be taken for as much as 38,000 gpm of mechanical draft cooling tower basin effluent. The criterion that establishes the maximum activity level in the downstream Susquehanna is that radioactive liquids will not be discharged to the river when the mechanical draft cooling tower basin effluent flow rate is below 5000 gpm.

The river flow rates, which will be considered in this analysis to establish maximum and minimum activity levels in the two applicable sections of the downstream Susquehanna, are: (1) 1700 cfs (765,000 gpm) which is the minimum daily river flow rate recorded, and (2) 34,000 cfs (15,300,000 gpm) which is the annual average river flow rate.

Since batch releases of radioactive liquid wastes from the unit will be made at flow rates between 5 and 30 gpm, the capacity of an evaporator condensate storage tank (7500 gallons) sets the maximum and minimum duration of batch radioactive liquid releases.

The maximum frequency of discharge of batches of radioactive liquid waste to the river is determined by the maximum quantity of radwaste liquids estimated to be produced in the plant per year. This is indicated on Table 11.1-3 to be 135,054 ft^3 or approximately 1,000,000 gal per year. This is equivalent to a design basis maximum of 135 seventy five hundred gallon batches per year or one batch disposed approximately every two and a half days. (Of the 135,054 ft^3 of liquid waste produced, processing by evaporation results in 811 ft^3 of evaporator bottoms; 6,060 ft^3 reclaimed for reuse; and, 128,183 ft^3 of discharge). However, the average activity level in the 1,000,000 gallons per year would be approximately one sixth of that indicated in Table 11.2-4 since the design basis quantity of activity is contained in the 135,054 ft^3 of liquid instead of in the 23,500 ft^3 of design basis letdown. This also represents the minimum design basis activity level in the batches of radioactive liquid waste disposed to the river and is more nearly representative of the activity level normally present in the evaporator condensate storage tanks since these collect condensate from the evaporation of both reactor coolant and miscellaneous liquid wastes.

Table 11.2-8 presents an isotopic breakdown of the activity levels and fractions of MPC to be anticipated in the evaporator condensate storage tanks when the annual design basis quantity of activity is mixed with 23,500 ft^3 of design basis letdown and with 135,000 ft^3 of waste water. The mixed isotope and tritium activity levels and fractions of MPC are utilized in Table 11.2-8 to calculate the activity levels and fractions of MPC for these two constituents in the mechanical draft cooling tower basin effluent, the river between the point at which mechanical draft cooling tower basin effluent discharges into it, and at the tail race of the York Haven Hydro Plant ("upstream river" in Table 11.2-9) and the river downstream of the York Haven Hydro Plant tail race ("downstream river" in Table 11.2-9) as functions of mechanical draft cooling tower basin effluent to liquid waste dilution ratios, section of the river, and river flow rates.

The mechanical draft cooling tower basin effluent (CTE) dilution of the liquid waste (LW) can vary from a maximum of 7600 (38,000 gpm CTE to 5 gpm LW) to a minimum of 167 (5000 gpm CTE to 30 gpm LW). However, Table 11.2-9 analyzes only those dilution ratios which would result in the cooling tower effluent entering the river at MPC level, or below, per the administrative limits for liquid waste disposal indicated above.

From Table 11.2-9, it is apparent that tritium may be the limiting isotope since its fraction of MPC is appreciably greater than that of the mixed isotopes in all cases. For all cases of mechanical draft cooling tower basin effluent to liquid waste dilution ratios studied, the mixed isotopes in the mechanical draft cooling tower basin effluent flowing into the river were considerably below their MPC level. Thus, on the design basis case studied, mixed isotopes are insignificant relative to tritium. This occurs because there are no practical methods of removing tritium from the reactor coolant letdown into the liquid radwaste system.

11.2.6 MAJOR CHANGES TO RADIOACTIVE WASTE TREATMENT SYSTEMS

Initiated safety related changes to the radioactive waste system (liquid, gaseous and solid):

1. Shall be reported to the Commission in the Annual Radioactive Effluent Release Report (TMI-1 Technical Specification 6.9.4) for the period in which the evaluation was reviewed. The discussion of each change shall contain:

    a. A summary of the evaluation that led to the determination that the change could be made in accordance with 10 CFR 50.59;

    b. Sufficient detailed information to totally support the reason for the change without benefit of additional or supplemental information;

    c. A detailed description of the equipment, components and processes involved and the interfaces with other plant systems;

    d. An evaluation of the change which shows the predicted releases of radioactive materials in liquid and gaseous effluents and/or quantity of solid waste that differ from those previously predicted in the license application and amendments thereto;

    e. An evaluation of the change which shows the expected maximum exposures to individuals in the unrestricted area and to the general population that differ from those previously estimated in the license application and amendments thereto;

    f. A comparison of the predicted releases of radioactive materials, in liquid and gaseous effluents and in solid waste, to the actual releases for the period prior to when the changes are to be made;

    g. An estimate of the exposure to plant operating personnel as a result of the change; and

    h. Documentation of the fact that the change was reviewed and approved.

TABLE 11.2-1
(Sheet 1 of 12)

DISPOSAL SYSTEM COMPONENT DATA

Equipment Tanks:

Item No.	Name/ Function	Type	Capacity, ft^3 (Each Tank) Total	Capacity, ft^3 (Each Tank) Liquid	Design Temp. (F)	Design Press. (psig)	Material Body	Material Lining	Vented To	Design Code	Seismic Design	Comments
WDL-T-1A WDL-T-1B WDL-T-1C	R.C. bleed tanks/contain reactor cool-ant letdown and feed to primary	H/S	12,050	11,050	200	25	S.S.	None	V.H.	ASME III-C	Class I	Maximum operating temp./pressure= 235 F/18 psig, -3 psig. Contains one primary system volume.
WDL-T-2	Misc. waste storage tank/ contain misc. radioactive wastes	H/S	3,124	2,718	200	25	S.S.	None	V.H.	ASME III-C	Class I	Maximum operating temp./pressure= 235 F/18 psig, -3 psig.
2-WDL-T-2	Misc. waste Hold-Up Tank/ Contain Misc. Radioactive Wastes	H/S	2,609	2,601	150	20	S.S	None	Cell Atm.	ASME III-C	Class I	In TMI-2 Aux. Bldg.
2-WDL-T-5	Aux. Bldg. Sump Tank/ Collect Drains	H/S	430	422	80	20	S.S.	None	Cell Atm.	ASME III-C	Class I	In TMI-2 Aux. Bldg.
WDL-T-3	R.C. coolant drain tank/ quench press. discharge	V/L	780	561 (Normal)	300	55	S.S.	None	V.H.	ASME III-C	Class I	Rupture disk provides overpressure relief
WDL-T-4	Spent resin storage tank/ contains spent resins	V/L	1,120	980	150	15	S.S.	None	Cell Atm.	ASME III-C	Class I	Nominal resin capacity 550 ft^3 or two years' retention at design
WDL-T-5	Used filter precoat tank/ contains used precoat from filters	V/L	590	510	150	15	S.S.	None	Cell Atm.	ASME III-C	Class I	Nominal precoat capacity 300 ft^3 or about four years' retention at design basis

Legend: 1) Type tank/supports; H/S - horizontal/saddle; V/L - vertical/legs

TABLE 11.2-1
(Sheet 2 of 12)

DISPOSAL SYSTEM COMPONENT DATA

Equipment Tanks:

Item No.	Name/ Function	Type	Capacity, ft^3 (Each Tank) Total	Liquid	Design Temp. (F)	Design Press. (psig)	Material Body	Material Lining	Vented To	Design Code	Seismic Design	Comments
WDL-T-7A WDL-T-7B	Reclaimed boric acid tanks/store reclaimed boric acid	V/L	920	728	150	15	S.S.	None	V.H.	ASME III-C	Class I	Nominal three months' storage per a.1) of Table 11.1-3
WDL-T-8	Neutralizer tank/neu- tralize resin regenerating solutions	V/L	194	166	150	15	C.S.	Phe- no- line 368 Sys- tem	Cell Atm.	ASME III-C	Class I	Four batches/ resin regeneration.
WDL-T-9	Neutralizer feed tank/ stores resin regenerating solutions	V/L	666	582	150	15	C.S.	Phe- no- line 368 Sys tem	Cell Atm.	ASME III-C	Class I	Stores one batch of resin re- generating solution and sub- sequent rinses. Reserve cap. to WDL-T-2.
WDL-T-10	Neutralized waste storage tank/stores neutralized resin regenerating solutions	V/L	779	695	150	15	C.S.	Phe- no- line 368	Cell Atm.	ASME III-C	Class I	Stores one batch of resin re- generating solution and sub- sequent rinses. Reserve capacity to WDL-T-2.

TABLE 11.2-1
(Sheet 3 of 12)

DISPOSAL SYSTEM COMPONENT DATA

Equipment Tanks:

Item No.	Name/ Function	Type	Capacity, ft^3 (Each Tank) Total	Liquid	Design Temp. (F)	Design Press. (psig)	Material Body	Material Lining	Vented To	Design Code	Seismic Design	Comments
WDL-T-11A WDL-T-11B	Evaporator condensate storage tank/ stores evap. condensate for reuse or disposal	V/L	1,234	1,024	150	15	C.S.	Phe-no-line 368 Sys-tem	Cell Atm.	ASME III-C	Class I	Nominal three day storage for condensate per design basis Table 11.1-3. N2 blanketed to preserve O2 free purity of condensate. The inventories in these two tanks principally govern releases of radioactive liquids to the environment.
WDL-T-12	Laundry waste tank/stores wastes from "hot" showers	H/S	1,352	1,114	200	15	C.S.	Phe-no-line 368 Sys-tem	Cell Atm.	ASME III-C	Class I	Nominal 1-year storage for "hot" shower wastes.
CC-T-1	Off-Spec. Water Tank	V/L	11,494	11,494	250	75	S.S.	None	Cell Atm.	ASME VIII-Div 1		Inside CCB Bathtub
CC-T-2	Clean Water Tank	V/L	17,871	16,681		ATM	S.S.	None	Cell Atm.	ASME VIII-Div 1		Inside CCB Bathtub

Total Liquid Capacity in Liquid Waste System Tanks = 76,042 ft^3

TABLE 11.2-1
(Sheet 4 of 12)

DISPOSAL SYSTEM COMPONENT DATA

Process Equipment:

Item No.	Name/ Function	Each Resin Volume (ft)	Each Design Flow Rate (gpm)	Type of Resin	Each Tank Volume (ft^3)	Design Temp. (F)	Design Press. (psig)	Mat'l. of Constr.	Vented To	Design Code	Seismic Design	Comments
	Demineralizers											
WDL-K-1A WDL-K-1B	Deborating demins./remove borate ion from primary coolant/remove lithium ion from primary coolant when using cation resin	40	70	Anion or Cation	80	120	150	S.S.	V.H.	ASME III-C	Class II	Resin is regenerated in place. Spent resin dumped to spent resin tank.
WDL-K-2A WDL-K-2B	Cation demins./ remove fission products from primary coolant and spent fuel pools water	36	70	Cation or Mixed Bed	80	120	150	S.S.	V.H.	ASME III-C	Class II	Spent resin dumped to spent resin tank.
WDL-K-3A WDL-K-3B	Condensate demins./polish condensate from reactor coolant and waste evaporators	12	15	Mixed Bed	16	120	150	S.S.	V.H.	ASME III-C	Class II	Spent resin dumped to spent resin tank.

TABLE 11.2-1
(Sheet 5 of 12)

DISPOSAL SYSTEM COMPONENT DATA

Process Equipment:

Item No.	Name/ Function	Type	Design Flow Rate (gpm)	Filter Medium	Tank Volume (ft^3)	Design Temp. (F)	Design Press. (psig)	Mat'l. of Constr.	Vented To	Design Code	Seismic Design	Comments
	Filters											
WDL-F-1A WDL-F-1B	Precoat filters/remove particulates from spent fuel pool water and primary coolant	Precoat	150	Filter Aid Media	25	200	150	S.S.	Cell Atm.	ASME III-C	Class II	Used precoat material dumped to the used precoat tank.
WDL-F-2	Resin trap/ deborating demins.	Cart-ridge	140	Micro-Wynd	2	120	150	S.S.	V.H.	ASME III-C	Class II	Trapped resin is backflushed to spent resin tank.
WDL-F-3A WDL-F-3B	Resin trap/ cation demins.	Cart-ridge	70	Micro-Wynd	1	120	150	S.S.	V.H.	ASME III-C	Class II	Trapped resin is backflushed to spent resin tank.
WDL-F-4	Resin trap/ condensated demins.	Cart-ridge	30	Micro-Wynd	<1	120	150	S.S.	V.H.	ASME III-C	Class II	Trapped resin is backflushed to spent resin tank.
WDL-F-5	Aux sump recirc filter/removes particulates from sump water.	Cart-ridge	360	Various	5	90	150	S.S.	Cell Atm.	None	Class II	

TABLE 11.2-1
(Sheet 6 of 12)

DISPOSAL SYSTEM COMPONENT DATA

Process Equipment:

Item No.	Name/ Function	Type	Design Flow Rate (gpm)	Type Tube-sheet to Tube Constr.	Type Head to Tube sheet Constr.	Design Temp. (F)	Design Press. (psig)	Mat'l Of Constr.	Vented To	Design Code	Seismic Design	Comments
	Coolers											
WDL-C-1	Reactor coolant drain tank cooler/cools contents of reactor coolant drain tank	Shell and Tube	30	Welded	Welded	250	150	S.S.	V.H.	ASME III-C	Class I	

TABLE 11.2-1
(Sheet 7 of 12)

DISPOSAL SYSTEM COMPONENT DATA

Process Equipment:

Item No.	Name/ Function	Type	Design Flow Rate (gpm)	De-con-tamina-tion Factor	Feed Tank Volume (ft^3)	Design Temp. (F)	Design Press. (psig)	Mat'l. Of Constr.	Vented To	Design Code	Seismic Design	Comments
	Evaporators											
WDL-Z-1A	Reactor coolant evaporator/ concentrates reactor coolant and spent fuel pools water	AMF Vacuum Dis-tilla-tion	12.5	10^4	133	250	22.5	S.S.	V.H.	ASME VIII	Class II	Design conditions given for evaporator section
WDL-Z-1B	Miscellaneous waste evaporator/ concentrates radioactive liquids	AMF Vacuum Dis-tilla-tion	12.5	10^4	133	250	22.5	S.S.	Cell Atm.	ASME VIII	Class II	Design conditions given for evaporator section
	Oil Skimmer											
WDL-Y-4	Aux sump oil skimmer/removes oil from sump water surface	-	240 (gpd)	n/a	n/a	200	-	Ceramic	-	None	Class II (anti-falldown)	Skimmer tube design temp 200ºF

Total Liquid Capacity in Liquid Waste System Process Equipment = 673 ft^3

TABLE 11.2-1
(Sheet 8 of 12)

DISPOSAL SYSTEM COMPONENT DATA

Sumps:

Name/Function	Type	Capacity (ft^3) Full (to 1' from top)	Sump Depth	Sump Pumps	Sump Pump Capacity (gpm)	Head (ft)	Comments
Reactor Building/collects water from floor and misc. equip. drains within reactor building	S.S. lined concrete pit	1,170	7'-6"	None Gravity drain via 6" line	-	-	Grating box keeps particulates out of outlet lines. Drains to Auxiliary Building Sump.
Auxiliary Building/collects water from floor and misc. equip. drains within Auxiliary and Fuel Handling Buildings	S.S. lined concrete pit	1,380	7'-10"	WDL-P-5A WDL-P-5B	150 150	75 75	Auxiliary Bldg. sump pumps are S.S. construction and Class I Seismic Design. FSAR section 6.4.5 describes sump pump use in protecting ECCS equipment in the Aux. Building.
Spent fuel pit room/ collects water from floor and misc. equipment drains in Fuel Handling Building	S.S. lined concrete pit	204	6'-8"	WDL-P-2A WDL-P-2B	20 20	35 35	
Heat exchanger vault/collects ground water seepage and water from misc. floor and equipment drains within HX vault	S.S. lined concrete pit	120	6'-0"	WDL-P-3A WDL-P-3B	20 20	35 35	Not considered part of liquid waste system. Water normally collected is that from ground water seepage, the river or the closed cooling system. Remote potential for contamination.
Borated water tank tunnel/ collects ground water seepage into borated water tank tunnel	Concrete pit lined with protective coating	42	3'-0"	WDL-P-4A WDL-P-4B	20 20	35 35	Not considered part of liquid waste system.

TABLE 11.2-1
(Sheet 9 of 12)

DISPOSAL SYSTEM COMPONENT DATA

Sumps:

Name/Function	Type	Capacity (ft^3) Full (to 1' from top)	Sump Depth	Sump Pumps	Sump Pump Capacity (gpm)	Sump Pump Capacity Head (ft)	Comments
Tendon access gallery/ collects ground water seepage into tendon access gallery	Concrete pit lined with protective coating	27	2'-6"	WDL-P-1A WDL-P-1B	20 20	60 60	Not considered part of liquid waste system. Remote potential for contamination.

Total Liquid Capacity in Liquid Waste System Sumps - 2,754 ft^3

TABLE 11.2-1
(Sheet 10 of 12)

DISPOSAL SYSTEM COMPONENT DATA

Equipment Pumps:

Item No.	Name/Function	Type	Design Flow (gpm)	Design Head (ft H_2O)	Design Temp. (F)	Design Press. (psig)	Mat'ls. of Constr.	Seismic Design	Comments
WDL-P-6A WDL-P-6B WDL-P-6C	Waste transfer pumps/ inject feed to primary syst. via Makeup and Purification System and process reactor	Vertical centrifugal with mechanical seals	140	227	150	150	S.S.	Class II	
WDL-P-7A WDL-P-7B	Miscellaneous waste transfer pumps/ transfer misc. wastes to processing	Horizontal centrifugal with mechanical seals	50	100	150	250	S.S.	Class II	
WDL-P-8	Reactor coolant drain tank pump/circulate drain tank contents for cooling or transfer to waste processing	Horizontal centrifugal with mechanical seals	30	97	250	150	S.S.	Class I	

Piping: All piping is 304 seamless S.S. fabricated and erected in accordance with USAS Draft Code B31.7 Class N.3.
Seismic design class as indicated on Drawings 302690, 302691, 302692, 302693, 302695 and 302697.

Valves: Most valves are 304 S.S. bodies, Seismic design class as indicated on Drawings 302690, 302691, 302692, 302693, 302695 and 302697
Air operated valves WDL-V-636/639 are of SA-351 construction.
The bulk of the remaining valves are diaphragm type except within containment and isolated valves.

TABLE 11.2-1
(Sheet 11 of 12)

DISPOSAL SYSTEM COMPONENT DATA

Equipment Pumps:

Item No.	Name/Function	Type	Design Flow (gpm)	Design Head (ft H_2O)	Design Temp. (F)	Design Press. (psig)	Mat'ls. of Constr.	Seismic Design	Comments
WDL-P-9A WDL-P-9B	Neutralizer pumps/transfer neutralized wastes	Horizontal centrifugal with mechanical seals	30	97	250	150	S.S.	Class II	
WDL-P-10	Decant pump/ transfer decant	Horizontal centrifugal with mechanical seals	30	97	150	150	S.S	Class II	
WDL-P-11	Slurry pump/ transfer slurry	Horizontal centrifugal with mechanical seals	80	160	150	150	S.S	Class II	

Piping: All piping is 304 seamless S.S fabricated and erected in accordance with USAS Draft Code B31.7 Class N.3.
Seismic design class as indicated on Drawings 302690, 302691, 302692, 302693, 302695 and 302697.

Valves: Most valves are 304 S.S. bodies, Seismic design class as indicated on Drawings 302690, 302691, 302692, 302693, 302695 and 302697.
Air operated valves WDL-V-636/639 are of SA-351 construction.
The bulk of the remaining valves are diaphragm type except within containment and isolated valves.

TABLE 11.2-1
(Sheet 12 of 12)

DISPOSAL SYSTEM COMPONENT DATA

Equipment Pumps:

Item No.	Name/Function	Type	Design Flow (gpm)	Design Head (ft H_2O)	Design Temp. (F)	Design Press. (psig)	Mat'ls. of Constr.	Seismic Design	Comments
WDL-P-13A* WDL-P-13B	Boric acid recycle pumps/ transfer reclaimed boric acid	Horizontal centrifugal with mechanical seals	70	160	150	150	S.S.	Class II	
WDL-P-14A WDL-P-14B	Evaporator condensate pumps/transfer evaporator condensate	Horizontal centrifugal with mechanical seals	60	230	200	150	S.S.	Class II	
WDL-P-15	Laundry waste pump/transfer laundry waste	Horizontal centrifugal with mechanical seals	30	97	150	150	S.S.	Class II	
WDL-P-16	Reactor drain pump/drain primary system	Horizontal centrifugal with mechanical seals	200	75	200	150	S.S	Class II	
ALC-P-5	Clean Water Tank Pump/ Transfer Water For Release	Horizontal Centrifugal With Mechanical Seals	200	100	230	200	S.S.	Class II	
ALC-P-12	Off-Spec Water Tank Pump/ Transfer Misc Wastes To Processing	Horizontal Centrifugal With Mechanical Seals	200	189	150	150	S.S		Pump is within CCB BathTub

Piping: All piping is 304 seamless S.S fabricated and erected in accordance with USAS Draft Code B31.7 Class N.3.
Seismic design class as indicated on Drawings 302690, 302691, 302692, 302693, and 302695 and 302697.

Valves: Most valves are 304 S.S. bodies, Seismic design class as indicated on Drawings 302690, 302691, 302692, 302693, and 302695 and 302697.
Air operated valves WDL-V-636/639 are of SA-351 construction.
The bulk of the remaining valves are diaphragm type except within containment and isolated valves.

Flow rates will vary for other modes of operation (recirculation, transfer to bleed tanks).
The most severe (design basis) case is 10 gpm (minimum) at 180 ft of head.

TABLE 11.2-2
(Sheet 1 of 3)

EQUIPMENT IN PRIMARY COOLANT AND MISCELLANEOUS WASTE TRAINS

Primary Coolant Train	Miscellaneous Waste Train
Raw Water Sources:	
WDL-T-1A, 1B and 1C Reactor coolant bleed tanks	Reactor and Auxiliary Building sumps
WDL-T-3 Reactor coolant drain tank	WDL-T-2 Misc. waste storage tank
	WDL-T-12 Laundry waste storage tank
	WDL-T-8 Neutralizer mixing tank
	WDL-T-9 Neutralizer feed tank
	WDL-T-10 Neutralized waste storage tank
	WDL-T-4 Spent resin tank
	WDL-T-5 Used precoat tank
	Borated water tank tunnel sump*
	Heat exchanger vault sump*
	Tendon access gallery sump*
	Spent fuel pit room sump
	2-WDL-T-2 Misc Waste Holdup Tank
	2-WDL-T-5 Aux. Bldg. Sump Tank
	CC-T-1 Off Spec Water Tank

* These three sumps normally collect ground water that may seep into the area they drain.

TABLE 11.2-2
(Sheet 2 of 3)

EQUIPMENT IN PRIMARY COOLANT AND MISCELLANEOUS WASTE TRAINS

Primary Coolant Train	Miscellaneous Waste Train
Liquid Processing Equipment:	
WDL-C-1 Reactor coolant drain tank heat exchanger	WDL-P-7A and 7B Misc. waste transfer pumps
WDL-P-8 Reactor coolant drain tank pump	WDL-P-15 Laundry waste pump
WDL-P-16 Reactor drain pump	WDL-P-9A, 9B Neutralizer pumps
WDL-P-6A, 6B and 6C Waste transfer pumps	WDL-Z-1B Misc. waste evaporator
WDL-K-1A and 1B Deborating demineralizers	WDL-T-10 Decant pump
WDL-K-2A and 2B Cation	ALC-P-12 Off Spec Water Tank Pump
demineralizers	WDL-F-5 Aux Sump Filter
WDL-F-1A, 1B Precoat filters	WDL-Y-4 Aux Sump Oil Skimmer
WDL-Z-1A Reactor coolant waste evaporator	
WDL-F-3A, 3B Cation demineralizer resin trap	
WDL-F-2, Deborating demineralizer resin trap	

TABLE 11.2-2
(Sheet 3 of 3)

EQUIPMENT IN PRIMARY COOLANT AND MISCELLANEOUS WASTE TRAINS

Primary Coolant Train	Miscellaneous Waste Train
Solidification Processing Equipment:	
WDL-T-7A and 7B Reclaimed boric acid tanks	WDL-P-11 Slurry Pump
WDL-P-13A and 13B Boric acid recycle pumps	
Liquid Effluent Release Equipment:	
	WDL-K-3A and 3B Evaporator condensate demineralizers
	WDL-T-11A, 11B Evaporator condensate storage tanks
	WDL-P-14A, 14B Evaporator condensate pumps
	WDL-F-4 Evaporator condensate demin resin trap
	CC-T-2 Clean Water Tank
	ALC-P-5 Clean Water Tank Pump

TABLE 11.2-3
(Sheet 1 of 1)

ASSUMPTIONS FOR ESTIMATING DESIGN BASIS ANNUAL AVERAGE LIQUID WASTE RELEASE TO THE RIVER

a. The reactor is the only source of radioactive materials and all activity escaping the primary system, by whatever route, is represented by the 23,500 ft^3 of annual letdown utilized in developing the values in Table 11.1-3.

b. Calculations are based on a equilibrium core 340 day cycle: 310 days full power operation followed by 30 day shutdown for refueling with letdown occurring during 255 of the full power days and deboration by the demineralizers during the remaining 55 days of full power operation.

c. Reactor thermal power level is 2535 Mwt from fission.

d. Reactor coolant activity per Table 11.1-2.

e. Decontamination factors in process equipment:

   1) Purification Demineralizer - 100 for all nuclides except Cesium, Yttrium, Molybdenum, Krypton, Xenon, and Tritium which are 1.

   2) Cation Demineralizer - 50 for all nuclides except Yttrium and Molybdenum for which it is 2, and Krypton, Xenon, and Tritium which is 1.

   3) Evaporator - 10^4 for all nuclides except Krypton, Xenon, and Tritium. The DF for Krypton and Xenon and Tritium is 1.

   4) Condensate Demineralizer - 20 for all nuclides except Yttrium and Molybdenum for which it is 5, and Krypton, Xenon, and Tritium which are 1.

f. Effluent from the mechanical draft cooling tower basin at the minimum flow rate of 5000 gpm.

g. No radioactive decay considered.

h. The primary coolant letdown is processed sequentially through Items e.1), e.2), and e.3), above, and the condensate from Item e.3) is processed through Item e.4).

TABLE 11.2-4
(Sheet 1 of 2)
DESIGN BASIS ESTIMATE OF ANNUAL RELEASE OF RADIOACTIVE LIQUIDS TO THE RIVER

Isotopic Breakdown of Annual Quantities of Activity Released to River

Isotope	Annual Release to River (Ûci)	Concentration in Cooling Tower Effluent (Ûci/cm^3)	MPC (Ûci/cm^3)	MPC Fraction
Fission Products				
Rb 88	1.86	1.91×10^{-13}	3×10^{-6}	0.637×10^{-7}
Sr 89	0.028	2.80×10^{-15}	3×10^{-6}	0.933×10^{-9}
Sr 90	0.0025	2.61×10^{-16}	3×10^{-7}	0.870×10^{-9}
Sr 91	0.032	3.20×10^{-15}	7×10^{-5}	0.457×10^{-10}
Sr 92	0.012	1.20×10^{-15}	7×10^{-5}	0.171×10^{-10}
Y 90	6440.0	6.44×10^{-10}	2×10^{-5}	0.322×10^{-4}
Y 91	1530.0	1.54×10^{-10}	3×10^{-5}	0.513×10^{-5}
Mo 99	37200.0	3.73×10^{-9}	2×10^{-4}	0.186×10^{-4}
I 131	2.19	2.20×10^{-13}	3×10^{-7}	0.733×10^{-6}
I 132	3.33	3.34×10^{-13}	8×10^{-6}	0.418×10^{-7}
I 133	2.59	2.60×10^{-13}	1×10^{-6}	0.260×10^{-7}
I 134	0.346	3.47×10^{-14}	2×10^{-5}	0.174×10^{-8}
I 135	1.33	1.33×10^{-13}	4×10^{-6}	0.333×10^{-7}
Cs 134	326.0	3.27×10^{-11}	9×10^{-6}	0.363×10^{-5}
Cs 136	55.3	5.54×10^{-12}	9×10^{-5}	0.620×10^{-7}
Cs 137	3260.0	3.27×10^{-10}	2×10^{-5}	0.164×10^{-4}
Cs 138	50.4	5.08×10^{-12}	3×10^{-6}	0.169×10^{-5}
Ba 139	0.057	5.74×10^{-15}	3×10^{-6}	0.191×10^{-8}
Ba 140	0.045	4.54×10^{-15}	3×10^{-5}	0.151×10^{-9}
La 140	0.015	1.47×10^{-15}	2×10^{-5}	0.735×10^{-10}
Ce 144	0.0019	1.94×10^{-16}	1×10^{-5}	0.194×10^{-10}
Corrosion Products				
Cr 51	0.00093	9.36×10^{-17}	2×10^{-3}	4.68×10^{-14}
Mn 54	0.00011	1.10×10^{-17}	1×10^{-4}	1.07×10^{-13}
Co 58	0.00559	5.62×10^{-16}	1×10^{-4}	5.62×10^{-12}
Fe 59	0.00011	1.10×10^{-17}	6×10^{-5}	1.77×10^{-13}
Co 60	0.00299	3.00×10^{-16}	5×10^{-5}	6.0×10^{-12}
Zr 95	0.00732	7.36×10^{-16}	6×10^{-5}	1.23×10^{-12}

[1] MPC value not listed in 10CFR20. Utilized MPC value given in 10CFR20 for unlisted isotopes that are not alpha emitters.

TABLE 11.2-4
(Sheet 2 of 2)

DESIGN BASIS ESTIMATE OF ANNUAL RELEASE OF RADIOACTIVE LIQUIDS TO THE RIVER

Tritium

H_3	5.095×10^8	4.92×10^5	3×10^5	0.015

Total Activity and Fraction of MPC of Annual Release to River

Type of Activity	Annual Quantity Released (Ûci)	Fraction of MPC in Cooling Tower Effluent
Mixed Fission Products	48,730	0.000080*
Tritium	5.095×10^8	.015

* For the mixture of isotopes, excluding tritium, considered above.

TABLE 11.2-5
(Sheet 1 of 3)

WASTE GAS SYSTEM COMPONENT DATA

Equipment Tanks:

Material Item No.	Name/Function	Type
WDG-T-1A WDG-T-1B WDG-T-1C	Waste gas decay tanks/store compressed waste gases at 80 psig for reuse or disposal	Vertical with leg supports
WDG-T-2	Waste gas delay tank/removal of entrained water vapor and delay of gases for decay of short-lived isotopes	Vertical with leg supports

	Capacity (ft^3 each)		Design	
Item No.	Volumetric	Gas (80 psig)	Temp (F)	Press (psig)
WDG-T-1A WDG-T-1B WDG-T-1C	1,125	6,750[1]	150	150
WDG-T-2	400	Not Applicable	150	25

[1] Equivalent to ft^3 of gas in the low pressure vent header system at 16.0 psia.

Total Storage Capacity, in terms of ft^3 of gas in Low Pressure Vent Header System - 20,250.

TABLE 11.2-5
(Sheet 2 of 3)

WASTE GAS SYSTEM COMPONENT DATA

Item No.	Body	Lining	Design Code	Seismic Design	Comments
WDG-T-1A WDG-T-1B WDG-T-1C	C.S.	Zinc Chromate Primer	ASME-III-C	Class II	Equipped with relief valve to discharge at 85 psig and drain connection for draining condensed water vapor
WDG-T-2	C.S. Chromate Primer	Zinc	ASME-III-C	Class II	Internally baffled for water de-entertainment and delay of gases. Drain connection provided

Piping: Piping is 304 S.S. and C.S. fabricated and erected in accordance with USAS Draft Code B31.7 Class N.3.

Valves: Valves are 304 S.S. and C.S. bodies, as applicable to the piping run. Bulk of valves are diaphragm type except within containment and isolation valves.

TABLE 11.2-5
(Sheet 3 of 3)

WASTE GAS SYSTEM COMPONENT DATA

Compressor Units:

	Design Performance (each)		Structural Design		
Item No.	Name/Function	Type	Flow (scfm)	Disch. Pr. (psig)	Inlet Pr. (psia)
WDG-P-1A WDG-P-1B	Waste gas compressors/ compress waste gases for storage.	Horizontal, rotary, water sealed	40	80	16.0

Item No.	(F)	Temp. (psig)	Press. Const.	Mat'ls of Design	Seismic Comments
WDG-P-1A WDG-P-1B	140	100	C.S.	Class II	Not positive displacement compressors. ASME-III-C and ASME-VIII applied, as required

Piping: Piping is 304 S.S. and C.S. fabricated and erected in accordance with USAS Draft Code B31.7 Class N.3.

Valves: Valves are 304 S.S. and C.S. bodies, as applicable to the piping run. Bulk of valves are diaphragm type except within containment and isolation valves.

TABLE 11.2-6
(Sheet 1 of 1)

ASSUMPTIONS FOR ESTIMATE OF ANNUAL AVERAGE WASTE GAS RELEASE

a. Annual volume of primary coolant degassed is 23,500 ft^3.

b. Radioactive gas concentrations are per Table 11.1-2. All tritium is assumed in the form of water which has been removed from the gas prior to its compression into storage. A reduction factor of 10,000 was applied to the iodine activity to account for its removal in the purification demineralizers, by partitioning in the letdown stream and by plate-out.

c. The compressed gas mixture is stored for a maximum 90 days to permit decay of the radioactive components.

d. The release is assumed to occur at a constant rate throughout the year.

e. The MPC of the mixture of gases estimated to be disposed is used.

f. The long term atmospheric diffusion is used to determine annual average concentrations at the site exclusion boundary, i.e., $X/Q = 4.5 \times 10^{-6}$ sec/m^3.

TABLE 11.2-7
(Sheet 1 of 1)

90 DAY HOLDUP DESIGN BASIS WASTE GAS RELEASE ANALYSIS SUMMARY

Isotope	Activity Released to Environment Ci/yr)	Fraction of MPC at Site Boundary
KR 85	550	2.63×10^{-3}
KR 85m, KR 87, KR 88	Negligible	Negligible
Xe 131m	8.99	3.02×10^{-6}
Xe 133	1.28	6.13×10^{-8}
Xe 133m, Xe 135m, Xe 135, Ce 138	Negligible	Negligible
I 131	0.988×10^{-4}	1.42×10^{-7}
I 132, I 133, I 134, I 135	Negligible	Negligible
Total	5560	0.00263

TABLE 11.2-8
(Sheet 1 of 3)
ACTIVITY LEVELS AND FRACTIONS OF MPC IN EVAPORATOR CONDENSATE STORAGE TANKS AS A FUNCTION OF THE QUANTITY OF WATER IN WHICH THE DESIGN BASIS ACTIVITY APPEARS

Activity Fission Products	Half Life Units as Indicated	Design Basis Activity Thru Evaporator Cond. Storage Tanks (ûCi)	_PC (ûCi/cm3i)
Rb 88	17.8 minutes	1.86	3×10^{-6}
Sr 89	52 days	0.028	3×10^{-6}
Sr 90	28 years	0.0025	3×10^{-7}
Sr 91	9.7 hours	0.032	7×10^{-5}
Sr 92	2.7 hours	0.012	7×10^{-5}
Y 90	64 hours	1,530.0	3×10^{-5}
Mo 99	67 hours	37,200.0	2×10^{-4}
I 131	8 days	2.19	3×10^{-7}
I 132	2.3 hours	3.33	8×10^{-6}
I 133	20.8 hours	2.59	1×10^{-6}
I 134	53 minutes	0.346	2×10^{-5}
I 135	6.7 hours	1.33	4×10^{-6}
Cs 134	2.1 years	326.0	9×10^{-6}
Cs 136	13.5 days	55.3	9×10^{-5}
Cs 137	30 years	3,260.0	2×10^{-5}
Cs 138	32 minutes	50.4	3×10^{-6}
Ba 139	85 minutes	0.057	3×10^{-6}
Ba 140	12.8 days	0.045	3×10^{-5}
La 140	40 hours	0.015	2×10^{-5}
Ce 144	285 days	0.0019	1×10^{-5}
Subtotals		48,873.5394	
Corrosion Products			
Cr 51	28 days	0.000931	2×10^{-3}
Mn 54	290 days	0.000106	1×10^{-4}
Co 58	71 days	0.00559	1×10^{-4}
Fe 59	45 days	0.00106	6×10^{-5}
Co 60	5.2 years	0.00299	5×10^{-5}
Zr 95	65 days	0.00732	6×10^{-5}
Subtotals		0.017043	
Grand Total Mixed Isotopes		48,873.55	
Tritium	12 years	5.095×10^{8}	3×10^{-3}

TABLE 11.2-8
(Sheet 2 of 3)

ACTIVITY LEVELS AND FRACTIONS OF MPC IN EVAPORATOR CONDENSATE STORAGE TANKS AS A FUNCTION OF THE QUANTITY OF WATER IN WHICH THE DESIGN BASIS ACTIVITY APPEARS

Activity Fission Products	Activity Level in Evaporator Condensate Storage Tanks (ûCi/cm^3(10^8) 23,500 ft^3 Letdown	135,000 ft^3 Letdown	Fraction of MPC in Evaporator Condensate Storage Tanks 23,500 ft^3 Letdown	135,000 ft^3 Letdown
Rb 88	0.28	0.0487	0.0009	0.00016
Sr 89	0.0042	0.00073	Negligible	Negligible
Sr 90	0.00038	0.000066	Negligible	Negligible
Sr 91	0.0048	0.00084	Negligible	Negligible
Sr 92	0.0018	0.00031	Negligible	Negligible
Y 90	970.0	168.85	0.485	0.084
Y 91	230.0	40.04	0.077	0.013
Mo 99	5600.0	974.81	0.28	0.049
I 131	0.33	0.057	0.011	0.0019
I 132	0.50	0.087	0.00063	0.00011
I 133	0.39	0.068	0.0039	0.00068
I 134	0.052	0.009	Negligible	Negligible
I 135	0.20	0.035	0.0005	Negligible
Cs 134	49.0	8.53	0.0054	0.00095
Cs 136	8.3	1.44	0.00092	0.00016
Cs 137	490.0	85.30	0.245	0.043
Cs 138	7.6	1.32	0.025 3	0.0044
Ba 139	0.0086	0.0015	Negligible	Negligible
Ba 140	0.0068	0.0012	Negligible	Negligible
La 140	0.0022	0.0038	Negligible	Negligible
Ce 144	0.0003	0.000052	Negligible	Negligible
Subtotal	7356.68108	1280.599778	1.135	0.1979

TABLE 11.2-8
(Sheet 3 of 3)

ACTIVITY LEVELS AND FRACTIONS OF MPC IN EVAPORATOR CONDENSATE STORAGE TANKS AS A FUNCTION OF THE QUANTITY OF WATER IN WHICH THE DESIGN BASIS ACTIVITY APPEARS

Activity Fission Products	Activity Level in Evaporator Condensate Storage Tanks (ûCi/cm^3(10^8) 23,500 ft^3 Letdown	135,000 ft^3 Letdown	Fraction of MPC in Evaporator Condensate Storage Tanks 23,500 ft^3 Letdown	135,000 ft^3 Letdown
Corrosion Products				
CR51	0.00014	0.000024	Negligible	Negligible
Mn54	0.000016	0.0000028	Negligible	Negligible
Co58	0.00086	0.00015	Negligible	Negligible
Fe59	0.000016	0.0000028	Negligible	Negligible
Co60	0.00046	0.00008	Negligible	Negligible
2r95	0.00113	0.00020	Negligible	Negligible
Subtotal	0.002622	0.0004596	0.0	0.0
Grand Totals Mixed				
Isotopes	7356.68	1280.60	1.135	0.1979
Tritium	7.72×10^7	1.31×10^7	257	44

TABLE 11.2-9
(Sheet 1 of 6)

ACTIVITY LEVELS AND FRACTION OF MPC IN COOLING TOWER EFFLUENT,UPSTREAM AND DOWNSTREAM RIVER WATER FOR THE DESIGN BASIS QUANTITYOF ACTIVITY AS A FUNCTION OF: WASTE WATER QUANTITY AND FLOW RATE,COOLING TOWER EFFLUENT FLOW RATE AND RIVER WATER FLOW RATE

Evap. Cond. Waste Water Quantity (ft^3/yr)	Activity ûCi/cc F/MPC	In Evap. Cond. (LW)	Dilution Ratio CTE/LW	In CTE to River
23,500	MI ûCi/cc	73.6 x 10^{-6}		2.84 x 10^{-7}
(Minimum)	T ûCi/cc	0.772		0.003
	MI F/MPC	1.135	5000 gpm	0.0044
	T F/MPC	257.	19.5 gpm	1.0
	MI ûCi/cc	73.6 x 10^{-6}		2.84 x 10^{-7}
	T ûCi/cc	0.772	(257)	0.003
	MI F/MPC	1.135		0.0044
	T F/MPC	257.		1.0
	MI ûCi/cc	73.6 x 10^{-6}		9.62 x 10^{-9}
	T uCi/cc	0.772		0.0001
	MI F/MPC	1.135	38,000 gpm	0.000148
	T F/MPC	257.	5 gpm	0.0338
	MI ûCi/cc	73.6 x 10^{-6}		9.62 x 10^{-9}
	T ûCi/cc	0.772	(7600)	0.0001
	MI F/MPC	1.135		0.000148
	T F/MPC	257.		0.0338

MI = Mixed Isotopes, T = Tritium

TABLE 11.2-9
(Sheet 2 of 6)

ACTIVITY LEVELS AND FRACTION OF MPC IN COOLING TOWER EFFLUENT, UPSTREAM AND DOWNSTREAM RIVER WATER FOR THE DESIGN BASIS QUANTITYOF ACTIVITY AS A FUNCTION OF: WASTE WATER QUANTITY AND FLOW RATE,COOLING TOWER EFFLUENT FLOW RATE AND RIVER WATER FLOW RATE

Evap. Cond. Waste Water Quantity (ft^3/yr)	Activity ûCi/cc F/MPC	Upstream Dilution Ratio River/CTE	In Upstream River	River Flow Rate CFS	Downstream Dilution Ratio River/CTE
23,500	MI ûCi/cc		$5.68E^{-8}$		
(Minimum)	T ûCi/cc		0.0006	1700	20
	MI F/MPC		0.00088		
	T F/MPC	5	0.2		
	MI ûCi/cc		$5.68E^{-8}$		
	T ûCi/cc		0.0006	34,000	403
	MI F/MPC		0.00088		
	T F/MPC		0.2		
	MI ûCi/cc		$1.92E^{-9}$		
	T ûCi/cc		0.00002	1700	20
	MI F/MPC		0.00003		
	T F/MPC	5	0.00676		
	MI ûCi/cc		$1.92E^{-9}$		
	T ûCi/cc		0.00002	34,000	403
	MI F/MPC		0.00003		
	T F/MPC		0.00676		

MI = Mixed Isotopes, T = Tritium

TABLE 11.2-9
(Sheet 3 of 6)

ACTIVITY LEVELS AND FRACTION OF MPC IN COOLING TOWER EFFLUENT, UPSTREAM AND DOWNSTREAM RIVER WATER FOR THE DESIGN BASIS QUANTITYOF ACTIVITY AS A FUNCTION OF: WASTE WATER QUANTITY AND FLOW RATE,COOLING TOWER EFFLUENT FLOW RATE AND RIVER WATER FLOW RATE

Evap. Cond. Waste Water Quantity ft^3/yr	Activity ûCi/cc F/MPC	In Downstream River/CTE	Time to Disposals Batch (Hours)	Time Elapsed Between Batch Disposals (Days)
23,500	MI ûCi/cc	1.42×10^{-8}		
(Minimum)	T ûCi/cc	0.0015		
	MI F/MPC	2.2×10^{-4}		
	T F/MPC	0.5	6.4	15.3
	MI ûCi/cc	7.05×10^{-10}		
	T ûCi/cc	7.46×10^{-6}		
	MI ûCi/cc	1.09×10^{-5}		
	T F/MPC	0.0025		
	MI ûCi/cc	4.81×10^{-10}		
	T ûCi/cc	5×10^{-6}		
	MI F/MPC	7.4×10^{-6}		
	T F/MPC	1.69×10^{-3}	25	14.4
	MI ûCi/cc	2.39×10^{-11}		
	T ûCi/cc	2.48×10^{-7}		
	MI F/MPC	3.67×10^{-7}		
	T F/MPC	8.39×10^{-5}		

MI = Mixed Isotopes, T = Tritium

TABLE 11.2-9
(Sheet 4 of 6)

ACTIVITY LEVELS AND FRACTION OF MPC IN COOLING TOWER EFFLUENT, UPSTREAM AND DOWNSTREAM RIVER WATER FOR THE DESIGN BASIS QUANTITYOF ACTIVITY AS A FUNCTION OF: WASTE WATER QUANTITY AND FLOW RATE,COOLING TOWER EFFLUENT FLOW RATE AND RIVER WATER FLOW RATE

Evap. Cond. Waste Water Quantity (ft^3/yr)	Activity ûCi/cc F/MPC	In Evap Cond. (LW)	Dilution Ratio CTE/LW	In CTE to River
135,000	MI ûCi/cc	12.8×10^{-6}		7.60×10^{-8}
(Maximum)	T ûCi/cc	0.131		7.84×10^{-4}
	MI F/MPC	0.1980	5000 gpm	1.18×10^{-3}
	T F/MPC	44.	30 gpm	0.263
	Mi ûCi/cc	12.8×10^{-6}		7.60×10^{-8}
	T ûCi/cc	0.131	(167)	7.84×10^{-4}
	Mi F/MPC	0.1980		1.18×10^{-3}
	T F/MPC	44.		0.263
	MI ûCi/cc	12.8×10^{-6}		1.00×10^{-8}
	T ûCi/cc	0.131		1.03×10^{-4}
	MI F/MPC	0.1980	38,000 gpm	1.55×10^{-4}
	T F/MPC	44.	30 gpm	0.0347
	Mi ûCi/cc	12.8×10^{-6}		1.00×10^{-8}
	T ûCi/cc	0.131	(1267)	1.03×10^{-4}
	MI F/MPC	0.1980		1.55×10^{-4}
	T F/MPC	44.		0.0347
	MI ûCi/cc	12.8×10^{-6}		1.67×10^{-9}
	T ûCi/cc	0.131		1.72×10^{-5}
	Mi F/MPC	0.1980	38,000 gpm	2.59×10^{-5}
	T F/MPC	44.	5 gpm	5.79×10^{-3}
	MI ûCi/cc	12.8×10^{-6}		1.67×10^{-9}
	T ûCi/cc	0.131	(7600)	1.72×10^{-5}
	MI F/MPC	0.1980		2.59×10^{-5}
	T F/MPC	44.		5.79×10^{-3}

MI = Mixed Isotopes, T = Tritium

TABLE 11.2-9
(Sheet 5 of 6)

ACTIVITY LEVELS AND FRACTION OF MPC IN COOLING TOWER EFFLUENT, UPSTREAM AND DOWNSTREAM RIVER WATER FOR THE DESIGN BASIS QUANTITY OF ACTIVITY AS A FUNCTION OF: WASTE WATER QUANTITY AND FLOW RATE, COOLING TOWER EFFLUENT FLOW RATE AND RIVER WATER FLOW RATE

Evap. Cond. Waste Water Quantity (ft^3/yr)	Activity ûCi/cc F/MPC	Upstream Dilution Ratio River/CTE	In Upstream River	River Flow CFS	Downstream Dilution Rate River/CTE	Ratio
135,000	MI ûCi/cc		$1.52E^{-8}$			
(Maximum)	T ûCi/cc		$1.57E^{-4}$	1700	153	
	MI F/MPC		$2.36E^{-4}$			
	T F/MPC	5	0.053			
	MI ûCi/cc		$1.52E^{-8}$			
	T ûCi/cc		$1.57E^{-4}$	34,000	3060	
	MI F/MPC		$2.36E^{-4}$			
	T F/MPC		0.053			
	Mi ûCi/cc		$2.00E^{-9}$			
	T ûCi/cc		$2.06E^{-5}$	1700	20	
	MI F/MPC		$3.10E^{-5}$			
	T F/MPC	5	0.0069			
	MI ûCi/cc		$2.00E^{-9}$			
	T ûCi/cc		$2.06E^{-5}$	34,000	403	
	MI F/MPC		$3.10E^{-5}$			
	T F/MPC		0.0069			
	MI ûCi/cc		$3.34E^{-10}$			
	T ûCi/cc		$3.44E^{-6}$	1700	20	
	MI F/MPC		$5.18E^{-6}$			
	T F/MPC	5	$1.16E^{-3}$			
	MI ûCi/cc		$3.34E^{-10}$			
	T ûCi/cc		$3.44E^{-6}$	34,000	403	
	MI F/MPC		$5.18E^{-6}$			
	T F/MPC		$1.16E^{-3}$			

MI = Mixed Isotopes, T = Tritium

TABLE 11.2-9
(Sheet 6 of 6)
ACTIVITY LEVELS AND FRACTION OF MPC IN COOLING TOWER EFFLUENT, UPSTREAM AND DOWNSTREAM RIVER WATER FOR THE DESIGN BASIS QUANTITYOF ACTIVITY AS A FUNCTION OF: WASTE WATER QUANTITY AND FLOW RATE,COOLING TOWER EFFLUENT FLOW RATE AND RIVER WATER FLOW RATE

Evap. Cond. Waste Water Quantity ft^3/yr	Activity ûCi/cc F/MPC	In Downstream River/CTE	Time to Disposals Batch (Hours)	Time Elapsed Between Batch Disposals (Days)
135,000	MI ûCi/cc	4.97×10^{-10}		
(Maximum	T ûCi/cc	5.12×10^{-6}		
	MI F/MPC	7.71×10^{-6}		
	T F/MPC	1.72×10^{-3}	4.2	2.52
	MI ûCi/cc	2.48×10^{-11}		
	T ûCi/cc	2.56×10^{-7}		
	MI F/MPC	3.86×10^{-7}		
	T F/MPC	8.5×10^{-5}		
	Mi ûCi/cc	5.01×10^{-10}		
	T ûCi/cc	5.15×10^{-6}		
	MI F/MPC	7.75×10^{-6}		
	T F/MPC	1.73×10^{-3}	4.2	2.52
	Mi ûCi/cc	2.49×10^{-11}		
	T ûCi/cc	2.56×10^{-7}		
	MI F/MPC	3.85×10^{-7}		
	T F/MPC	8.61×10^{-5}		
	Mi ûCi/cc	8.36×10^{-11}		
	T ûCi/cc	8.6×10^{-7}		
	MI F/MPC	1.3×10^{-7}		
	T F/MPC	2.9×10^{-4}	25	1.66
	Mi ûCi/cc	4.14×10^{-12}		
	T ûCi/cc	4.27×10^{-8}		
	MI F/MPC	6.43×10^{-8}		
	T F/MPC	1.44×10^{-5}		

MI = Mixed Isotopes, T = Tritium

11.3 RADIATION SHIELDING

Radiation shielding for Unit-1 is provided principally by concrete walls, floors, and ceilings or roofs that are also structural members of buildings, which are designed to meet the requirements of the controlling criterion (i.e., structural strength or radiation shielding). Therefore, where structural strength is the controlling criterion, the radiation shielding provided is in excess of requirements.

11.3.1 DESIGN CRITERIA

The design criteria for radiation shielding for Unit 1 are based on the following:

a. Ensuring that during normal plant operation the radiation dose to operating personnel and to the general public is within the limits set forth in 10CFR20.

b. Providing the necessary protection for operating personnel so that a postulated accident may be terminated without excessive radiation exposure to the operators or to the general public.

c. Protecting certain components from excessive radiation damage or activation.

To comply with radiation exposure limits specified in 10CFR20, the shielding is designed to attenuate radiation levels throughout the plant, from direct and scattered radiation, to the dose rate levels specified for each of the areas indicated below:

Full-Power Operation Conditions (1 percent failed fuel)

Location	Dose Rate (mr/hr)
Office, Control Room, and Turbine Building	0.5
Reactor Building: Accessible areas	15.0
Auxiliary Building: Non-Controlled areas	1.5
Fuel Handling Building: Non-Controlled areas	1.5

11.3.2 GENERAL DESCRIPTION

Radiation shielding is divided into the following categories, i.e., primary, secondary, Reactor Building, Control Room, spent fuel (including fuel transfer), and auxiliary shielding. Each of these categories is described below.

11.3.2.1 Primary Shield

The primary shield serves to protect plant personnel, surrounding structures, and primary system components from the neutron and gamma radiation escaping the core and the reactor vessel. It is an annular cylinder of reinforced concrete which surrounds the reactor vessel and extends upward from the Reactor Building floor to the bottom of the fuel transfer canal. The shield thickness is 5 ft up to the height of the reactor vessel flange, where the thickness is reduced to 4.5 ft. It is designed to meet the following specific radiation attenuation objectives:

a. To sufficiently reduce (in conjunction with the secondary shield) the radiation escaping the reactor vessel during normal full power operation to allow limited access to the Reactor Building.

b. To sufficiently reduce the radiation, escaping the reactor vessel after shutdown, to permit limited access to the reactor coolant system equipment.

c. To sufficiently attenuate the neutron flux, escaping the reactor vessel, to prevent excessive activation of plant components and structures over the life of the plant.

11.3.2.2 Secondary Shield

The secondary shield is a 4 ft thick reinforced concrete structure surrounding the primary shield and the bulk of the primary system equipment, including piping, pumps, and steam generators. It is designed to attenuate radiation sufficiently to permit limited access to certain areas within the Reactor Building during full power operation. In addition, the secondary shield supplements the primary shield by attenuating neutron and gamma radiation escaping from the primary shield.

11.3.2.3 Reactor Building

The Reactor Building is a reinforced, prestressed containment structure with 3.5 ft thick cylindrical wall and a 3.0 ft thick dome that completely surrounds the nuclear steam supply system. During full-power operation, this building will attenuate any radiation escaping from the primary-secondary shield complex such that radiation levels from the source outside the Reactor Building will be less than 0.5 mr/hr. In addition, the Reactor Building structure will shield personnel from radiation sources inside the building following a Maximum Hypothetical Accident (MHA) or other postulated events inside containment resulting in elevated radiation levels. The building walls are of sufficient thickness to allow personnel a reasonable time period in which to evacuate the immediate vicinity of the Reactor Building following the MHA without excessive radiation exposure.

Other significant radiation shielding inside the Reactor Building is listed in Table 11.3-1.

11.3.2.4 Control Room

The Control Room shield is 2 ft thick and is designed to permit continuous occupancy by essential Control Room personnel following a Maximum Hypothetical Accident. It enables full control and shutdown procedures to be carried out without hazard to the Control Room operators. This shielding assures that the integrated whole body dose for 90 days following an MHA will not exceed 5 rem.

11.3.2.5 Auxiliary Building

Auxiliary building includes all concrete walls, covers, doors, and removable blocks which provide radiation protection of personnel from the numerous sources of radiation occurring from equipment housed within the Auxiliary Building. The major components and their required shielding are given in Table 11.3-1.

11.3.2.6 Spent Fuel Shielding

Radiation shielding is provided for personnel protection during all phases of spent fuel removal and storage. Operations requiring shielding of personnel are: spent fuel removal from the reactor vessel, spent fuel transfer through the refueling canal and transfer tubes, spent fuel storage, and spent fuel shipping cask loading prior to transportation. Normally, the spent fuel assembly rests within the spent fuel storage rack where it is covered with a minimum water shield depth of 23 ft. During spent fuel removal and transfer operations in the Spent Fuel Pool, the minimum water depth above the top of active fuel is 6.5 ft. Under this condition, the dose rate received by fuel handling personnel from the element being handled is less than 15 mr/hr.

The 4 ft thick concrete walls of the fuel transfer canal and the 5 ft thick spent fuel pit walls supplement the water shielding and limit the maximum continuous radiation dose levels in working areas adjacent to the spent fuel area to less than 1.5 mr/hr.

The refueling water and concrete walls also provide shielding from activated control rod clusters and reactor internals which will be removed at times during refueling periods. Although dose rates will normally be less than 1.5 mr/hr in working areas, certain manipulations of fuel assemblies, rod clusters, or reactor internals may produce short term exposures in excess of 1.5 mr/hr. However, the radiation levels will be closely monitored during refueling operations to establish the allowable exposure times for plant personnel in order not to exceed the integrated doses specified in 10CFR20.

11.3.2.7 Original Steam Generator Storage Facility (OSGSF) Shielding

In order to store, until site decommissioning, the Once-Thru Steam Generators (OTSGs) that were replaced with new Enhanced OTSGs (EOTSGs) as part of the TMI-1 Steam Generator Replacement Project (SGR), an Original Steam Generator Storage Facility (OSGSF) was constructed. The OSGSF meets all requirements for this OTSG storage, and in addition will be used to store RCS hot leg piping that was removed for replacement as part of the TMI-1 SGR. The radiological design of the OSGSF meets the radiation shielding requirements of 40 CFR 190, 10 CFR 20, and the Plant License. The building is designed to have a maximum contact dose rate of 0.2 mR/hour on the exterior wall surface, and also provides a locking access control entrance door behind a concrete labyrinth designed to provide shielding.

11.3.3 MATERIALS

The material used for the primary, secondary, Reactor Building, Control Room, auxiliary, and spent fuel shields is concrete with a density of approximately 150 lb/ft^3.

TABLE 11.3-1
(Sheet 1 of 3)
PRINCIPAL SHIELDING

Reactor Building

Component	Concrete Thickness (ft)
Primary shield (below flange)	5
Primary shield (above flange)	4.5
Secondary shield	4
Reactor Building vertical walls	3.5
Reactor Building dome	3
Side walls of fuel transfer canal	4
End walls of fuel transfer canal	3 to 4
Floor of fuel transfer canal	4
Reactor coolant drain tank room walls	1.5
Minimum water over active fuel during transfer (Reference Sterns Rogers Dwg. No. 21804-1)	8.1 (water)

Auxiliary Building

Component	Adjacent Concrete Wall Thickness (ft)
Miscellaneous waste storage tank	3
Reactor coolant bleed tanks	3
Waste transfer pumps	2 to 4
Waste gas compressors	2 to 4
Waste gas delay tank	2 to 6
Reactor coolant waste evaporator	2 to 4
Miscellaneous waste evaporator	2 to 4

TABLE 11.3-1
(Sheet 2 of 3)
PRINCIPAL SHIELDING

Auxiliary Building

Component	Concrete Thickness (ft)
Boric acid recycle pumps	2 to 4
Reclaimed boric acid tanks	3
Used filter precoat tank	3
Spent resin storage tank	3
Makeup tank	3
Makeup and purification pumps	2 to 3
Waste gas decay tanks	3 to 4
Condensate demineralizer	2.5
Makeup and purification demineralizers	3.5
Deborating demineralizers	3 to 5
Cation demineralizers	3 to 5
Precoat filters	2 to 3
Waste drumming area	3

Fuel Handling Building

Component	Concrete Thickness (ft)
Side walls of spent fuel pools	5
End walls of spent fuel pools	5
Bottom of spent fuel pools	5

TABLE 11.3-1
(Sheet 3 of 3)

PRINCIPAL SHIELDING

Fuel Handling Building

Component	Concrete Thickness (ft)
Pools water-gate wall	5
Spent fuel shipping cask pit walls	1.5
Minimum water over active fuel during transfer	6.5 (water)
Minimum water over active fuel in storage racks	23 (water)

11.4 RADIATION MONITORING SYSTEM

11.4.1 DESIGN BASIS

The Radiation Monitoring System detects, indicates, annunciates, and records the radiation level at selected locations inside and outside the plant to verify compliance with 10CFR20 limits.

11.4.1.1 System Description

The Radiation Monitoring System, as shown on Figure 11.4-1 and Table 11.4-1, is divided into the following subsystems:

a. Area monitoring

b. Atmospheric monitoring

c. Liquid monitoring

The Radiation Monitoring System instrumentation receives power from the battery-backed, inverter-fed, vital instrument buses as per Table 11.4-1. Associated motors are fed from engineered safeguards supplies.

All of the monitors except RM-A12, RM-A13, RM-L12, WHP-RIT-1, RLM-RM-1, and the CCB monitor (ALC-RMI-18) display their measured variable and alarm condition on the radiation monitoring panel located in the Unit 1 Control Room. Each monitor is equipped with two adjustable alarm levels and a channel failure and/or loss of power indication.

Interlock functions performed by the radiation monitoring system are shown on Figure 11.4-2.

11.4.2 AREA MONITORING SUBSYSTEM DESCRIPTION

The Area Monitoring Subsystem performs the following three basic functions: monitors in-plant radiation levels, monitors radiation levels for containment isolation, and provides monitoring for containment dome airborne activity.

The Area Monitoring Subsystem is comprised of 24 channels equipped with gamma-sensitive ionization chambers and 2 channels equipped with scintillation detectors installed at selected locations throughout the unit. Each channel has a Light indication corresponding to the radiation alarm levels and a "Fail" light, which, when out, indicates a loss of power to the channel.

Channels RM-G1 through RM-G3, RM-G5 through RM-G7, and RM-G9 through RM-G15 provide in-plant monitoring and have a range of 0.1 mR/hr to 10^7mR/hr. The setpoints of RM-G6 and RM-G7 are lowered during refueling operations in accordance with plant procedures.

Radiation channel RM-G9 is located in the Fuel Handling Building to monitor radiation levels in the spent fuel storage area. The high alarm and alert setpoints are provided to call attention to an increase in radiation level and off-standard conditions. RM-G9 is utilized to ensure compliance with 10 CFR 50.68(b), which TMI committed to as part of TS Amendment No. 260.

If RM-G9 become inoperable, portable survey instrumentation, having the appropriate ranges and sensitivities to fully protect individuals involved in refueling operation, is used until the permanent instrumentation is returned to service.

During fuel handling and refueling operations the radiation levels in the Reactor Building are monitored by RM-G6 and RM-G7. RM-G6 and RM-G7 are utilized to ensure compliance with 10 CFR5 0.68(b), which TMI committed to as part of TS Amendment No. 260.

RM-G6, RM-G7, and RM-G9 are required to be functional when handling fuel. RM-G6, RM-G7, and RM-G9 are determined to be functional by completion of source checks, channels tests, and calibrations. Source checks are performed on a weekly basis for RM-G6, RM-G7, and RM-G9. Channel tests are performed on a monthly basis for RM-G6, RM-G7, and RM-G9. Calibrations are performed on a quarterly basis for RM-G6 and RM-G7 and on a yearly basis for RM-G9.

If RM-G6 and RM-G7, become non-functional, movement of fuel in the reactor core shall cease until permanent instrumentation is returned to service or portable survey instrumentation , having the appropriate ranges and sensitivities to fully protect individuals involved in refueling operation, have been installed.

Radiation channels RM-G16 through RM-G20 provide monitoring of activity for containment isolation. Channels RM-G16 and RM-G17 monitor Once Through Steam Generator (OTSG) A and OTSG B samples, and provide alarm and isolation functions. Channel RM-G18 monitors reactor coolant samples and provides alarm and isolation. Channel RM-G19 monitors the reactor coolant pump seal return and provides an alarm when activity limits are reached, with containment isolation of the system by operator action. Channel RM-G20 monitors the reactor coolant drain tank pump discharge, with alarm and isolation on detection of activity limits. Containment isolation is further described in Section 5.3. The channels are ionization chambers with a range of 0.1 mR/hr to 10^7 mR/hr.

Channels RM-G22 through RM-G27 perform post accident monitoring functions, and are described in Section 11.4.5.

The Area Monitoring Subsystem is summarized on Table 11.4-2.

11.4.3 ATMOSPHERIC MONITORING SUBSYSTEM DESCRIPTION

The Atmospheric Monitoring Subsystem performs the following basic functions: effluent monitoring, emergency release monitoring, and in-plant air monitoring, and is part of the containment leak detection system.

The atmospheric monitoring subsystem consists of 15 channels located inside and outside the plant. Their function, sensitivity, and locations are as follows:

a. Control Tower Air Intake: Channel RM-A1

   This channel monitors the air intake into the Control Room and is located in the control tower. It measures airborne activity with a particulate/iodine/gas (PIG) type monitoring system.

The RMA-1 channels have a range of 10 - 10^6 cpm. An interlock is provided to place the Control Building in the recirculation mode.

b. Reactor Building Air Sample Line: Channel RM-A2

This channel is used to monitor the activity level of the air inside the Reactor Building. The sampling unit is a PIG type system located in the Intermediate Building, with sample air drawn through a Reactor Building penetration. The readout of the monitor will be used to detect leaks in systems containing primary coolant.

The RM-A2 channels have a range as follows:

RM-A-2P	10-10^7 cpm
RM-A-2I	10-10^7 cpm
RM-A-2G	10-10^6 cpm

Postaccident Reactor Building atmospheric sampling is described in Section 11.4.6.

c. Fuel Handling Building Exhaust Ventilation Duct: Channel RM-A4 Auxiliary Building Exhaust Ventilation Duct: Channel RM-A6

These in-plant air monitors are a PIG type system with sampling lines located before the filters. They are used for Auxiliary and Fuel Handling Building air activity monitoring. Their function is to provide an early detection of activity released in their particular location.

The RM-A4 and RM-A6 channels have a range of 10-10^6 cpm.

Channels RM-A4 and RM-A6 will initiate shutdown of the related ventilation air supply system should their preset radiation setpoint be exceeded.

d. Condenser Vacuum Pump Exhaust: Channel RM-A5 Low Range and Channel RM-A15

The Condenser Vacuum Pump Exhaust Channel RM-A5 Low Range and Channel RM-A15 are gaseous effluent monitors that will detect leakage between the primary and secondary systems.

Radiation monitors, RM-A5 (Low Range) and RM-A15 are beta scintillation detectors.

RM-A5 Low Range and RM-A15 each have a range of 10 to 10^7 cpm. The monitors sample the common discharge of the condenser vacuum pumps, exhausting to the suction of the vacuum pumps. A high alarm initiates the MAP-5 iodine particulate sequential sampling. See Section 11.4.6.

The approximate monitoring range is 10^{-6} microcuries per cubic centimeter to 10^{-1} microcuries per cubic centimeter for Xe-133. Alarm setpoints are consistent with the EPRI Guidelines and the Technical Specification shutdown limit of 0.1 GPD (144 GPD) primary to secondary leakage.

The extended range for the postaccident monitoring channel RM-A5 High Range is described in Subsection 11.4.5.

e. Waste Gas Discharge: Channel RM-A7

This channel is an effluent gas monitor that detects and controls the activity released from the waste gas decay tanks. It will close the tanks' discharge valve if the activity is above its preset level. RM-A7 has a range of 10 - 10^6 cpm.

f. Auxiliary and Fuel Handling Building Exhaust: Channel RM-A8

This channel is part of the effluent monitoring system that monitors the activity released from the Auxiliary and Fuel Handling Buildings through the unit vent. This channel is a particulate/iodine/gas (PIG) type monitor. The sample point for this monitor is located downstream of the filter unit. Should the gas channel's preset level be exceeded, it will initiate the shutdown of the ventilation air supply system of the Auxiliary and Fuel Handling Buildings as well as close the waste gas decay tanks' discharge valve and MAP 5 iodine/particulate sampling will be initiated. The monitoring range is 10 - 10^6 cpm. Further information is provided in item (g) below.

The extended range for postaccident monitoring Channel RM-A8 High Range is described in Section 11.4.5.

g. Reactor Building Purge Exhaust: Channel RM-A9

The activity of the air discharged to the unit vent when purging the Reactor Building is measured by a PIG type effluent monitor. Should the channel's preset level be exceeded, the monitor will initiate closure of the purge valves and reactor building sump outlet isolation valves. MAP-5 iodine particulate sampling is initiated on RM-A9 high alarm on gas channel. The monitoring range is 10 to 10^6 cpm.

The radiation monitors RM-A8 and RM-A9 take a continuous sample of air from the Auxiliary and Fuel Handling Buildings' concrete ventilation exhaust duct located against the outside of the Auxiliary Building. This sample withdrawal location meets the requirements of ANSI 13.1-1969.

A building housing the monitors is located on top of the exhaust duct abutting the Auxiliary Building. Concrete walls and roof provide radiation shielding from external sources.

The ambient temperature is thermostatically controlled between 650F and 900F in the building housing the monitors and the background radiation is less than 100 mR/hr under normal conditions. The ambient temperature of the building is bounded by a temperature range of 70 to 110ºF.

The extended range for postaccident monitoring channel RM-A9 High is described in Subsection 11.4.5.

h. Radiochemical Laboratory Monitor: Channel RM-A12 and Spent Fuel Area Monitor: Channel RM-A13

These two channels are movable cart mounted PIG type monitors which will be used for in-plant air monitoring of particular plant areas during sampling and/or refueling. Their ranges are 10 - 10^6 cpm.

i. Fuel Handling Building ESF Ventilation System Exhaust: Channel RM-A14

The activity of the air discharged by the FHB ESF Ventilation system is measured by an offline radiation monitoring system comprising an isokinetic nozzle, a valve and manual sampling rack, a local control unit (LCU), a noble gas skid and a remote digital ratemeter. The isokinetic nozzle is mounted on the ventilation system common discharge duct in the ESF enclosure on the Auxiliary

Building roof (elevation 331'). Directly below it in the ESF enclosure is the valve and manual sampler rack, containing the tritium and fission gas grab sample points and the particulate and iodine filters. The noble gas skid is located near the local control unit in the Auxiliary Building HVAC equipment room on elevation 312 feet. The skid contains the detection units and sample pump. The remote digital ratemeter is located in the PRF panel of the Main Control Room and provides signal relay, alarm report and setpoint recall functions.

The effluent radiation monitor is powered from the Engineered Safeguard Valves Motor Control Center MCC-1C to provide reliable power in the event of ES activation of either of the diesel generators.

The environment of the remote digital ratemeter is controlled by the Control Room ventilation system; the environment of the noble gas skid and local control unit is controlled by the Fuel Handling and Auxiliary Building ventilation system. The valve and manual sampler and the isokinetic nozzle are located in the ESF enclosure, whose environment is controlled within the range of 700F to 1200F by fans and heaters in the enclosure. Sample lines are heat traced to prevent condensation in the gas sample.

j. Chemical Cleaning Building (CCB) Ventilation Exhaust: Channel ALC-RMI-18

This channel monitors the Chemical Cleaning Building (CCB) exhaust with a PIG type monitoring system.

ALC-RMI-18 ratemeters have a range from 10 to 10^6 cpm. There is no interlock associated with this monitor. Sampling for tritium activity is performed off of the monitor.

k. Waste Handling and Packaging Facility (WHPF) Exhaust: Channel WHP-RIT-1.

This channel monitors the WHPF exhaust with a particulate radiation monitor. The monitor has a range of 10-100,000 cpm. A high alarm will initiate shutdown of the WHPF ventilation air exhaust system. Sampling for particulate activity is performed off of the monitor.

l. Respirator Cleaning and Laundry Maintenance (RLM) Facility Exhaust: Channel RLM-RM-1

This channel monitors the RLM exhaust with a particulate radiation monitor. The monitor has a range of 10-100,000 cpm. There is no interlock associated with this monitor. Sampling for particulate activity is performed off of the monitor.

11.4.4 LIQUID MONITORING SUBSYSTEM DESCRIPTION

The Liquid Monitoring Subsystem performs the following basic functions: effluent monitoring, leak detection, and monitoring of the Reactor Coolant System activity.

The Liquid Monitoring Subsystem consists of nine liquid monitors. Four of these monitors (RM-L2, RM-L3, RM-L4, and RM-L9) are used for leak detection monitoring of closed cooling loops which act as barriers against release of activity to the river. These four monitors have identical scale, sensitivity and range data. The scales are $10\text{-}10^{6}$ cpm.

RM-L5 is part of leak detection monitoring and consists of a GM tube monitor designed to detect changes in the radioactivity concentrations of the spent fuel pool water. These changes may indicate the presence of a leaking fuel element. The scale is $10\text{-}10^{6}$ cpm.

The primary coolant letdown, after passing through a delay line, is monitored by RM-L1, which is equipped with one gamma radiation scintillation low range detector and one GM tube gamma high range detector. Both detectors have scales of $10\text{-}10^{6}$ cpm and their response is given in terms of an equilibrium mixture of fission products in the reactor coolant. Since the ranges of the two detectors overlap, they provide redundant measurement of the gross gamma activity resulting from failed fuel from a fractional percent to greater than 1 percent.

The liquid effluent monitor RM-L6 monitors the radioactive liquid waste water prior to dilution by the Mechanical Draft Cooling Tower basin effluent. The RM-L6 monitor has a $10\text{-}10^{6}$ scale. Should its preset level be reached, it will initiate closing of its related liquid waste discharge valve.

Channel RM-L7 monitors the plant effluent discharge to the river and is a backup for RM-L6 and RM-L12. Its scale is $10\text{-}10^{6}$ cpm.

RM-L10 has been physically removed from the plant. It was used to monitor water routed into the turbine building sump prior to it being pumped to the Industrial Waste Treatment System (IWTS) sump. The original design basis for RM-L10 was to ensure that the turbine building sump contribution to the IWTS did not exceed 10CFR20 MPC limits. Administrative controls have been established to secure the turbine building sump pumps when high secondary system activity is indicated during plant operation and to sample and isotopically analyze the turbine building sump during such a condition. These measures maintain adequate control of turbine building sump input to the IWTS. IWTS discharge is monitored by RM-L12 as described below.

RM-L12 is a gamma scintillator on the discharge of the Industrial Waste Treatment System (IWTS) and Industrial Waste Filter System (IWFS) before this effluent is diluted by the flow from the mechanical draft cooling tower basin. This monitor is especially designed and calibrated to insure that concentrations of isotopes in all liquid effluents, except water monitored by RM-L6, do not exceed ten times the concentration levels specified in 10CFR20.1001-20.2401, Appendix B, Table 2. Its range is $10\text{-}10^{6}$ cpm. When its high radiation setpoint is exceeded, it will terminate both the IWTS and IWFS discharge to the river. RM-L12 is surveilled in accordance with the requirements of the Offsite Dose Calculation Manual Section 3.1.1.

A composite proportional flow sampler is installed on the common discharge piping from the turbine building sump and the powdex sump to the Industrial Waste Treatment System. An additional composite proportional flow sampler is located near RM-L7. Piping and control valves are provided to allow flushing of radiation monitor RM-L6 following a discharge of the waste evaporator condensate storage tank.

The RM-L6 monitor has a removable sampler liner to facilitate decontamination.

11.4.5 POSTACCIDENT RADIATION MONITORING

The postaccident radiation instrumentation, in conformance with NUREG 0578 and NUREG 0737, consists of the following:

a. High-Range Effluent Monitors

The Auxiliary Building, the Fuel Handling Building, and the Reactor Building ventilation exhausts, the discharges from the condenser vacuum pumps, and OTSG safety valves and atmospheric dump are considered as potential significant effluent paths. Therefore, extended range monitors are provided to assure monitoring and readout capability for postulated accidents.

The purpose of the high range monitors is to provide extended ranges to radiation monitors RM-A8, RM-A9 and RM-A5. The high range monitors also provide additional monitoring capability and include radiation monitors RM-G24, RM-G25, RM-G26, and RM-G27. The monitors and their locations are listed in Table 11.4-1. Also provided are three additional sampling stations, one for the Auxiliary and Fuel Handling Building ventilation exhaust, one for the Reactor Building ventilation exhaust, and one for the condenser vacuum pump exhaust.

A GM detector for extended range gas monitoring of the Auxiliary and Fuel Handling Building ventilation exhaust is provided in the cabinet of radiation monitor RM-A8. The detector has a scale of 10-10^6 cpm and a corresponding upper range of about 10^3 ûCi/cm^3. The detector is mounted in the RM-A8G Channel Lead Pig to monitor the sample lines inside RM-A8's cabinet. A readout module tagged RM-A8G High is located in the Control Room with a local panel meter provided on the cabinet.

A GM detector for extended range gas monitoring of the Reactor Building Purge ventilation exhaust is provided in the cabinet of radiation monitor RM-A9. The detector has a scale of 10-10^6cpm, and a corresponding upper range on the order of 10^2 ûCi/cm^3. The detector is mounted in the RM-A9's cabinet. A readout module tagged RM-A9G High is located in the Control Room with a local panel meter provided on the cabinet.

An ionization chamber is provided for extended range gas monitoring of the Reactor Building exhaust and has a scale of 0.1 to 10^7 mR/hr and a corresponding upper range on the order of 10^6 ûCi/cm^3. The upper range covers the expected concentration for the postulated worst case accident concentrations. A readout module tagged RM-G24 is located in the Control Room.

A GM detector for extended range gas monitoring of the condenser vacuum pump discharge is provided in the shield of radiation monitor RM-A5. The detector has a scale of 10 to 10^6 cpm and a corresponding upper range on the order of 10^2 ûCi/cm^3.

Although RM-A5 (High Range) measures the noble gas from the same sample chamber as RM-A5 (Low Range) it has a thin lead partition between the RM-A5 Low gas chamber to attenuate it to the proper high range. A readout module tagged "RM-A5 High" is located in the Control Room.

Also, an ionization chamber RM-G25 is provided for further extended range monitoring the condenser vacuum pump discharge. The detector has a scale of 0.1 to 10^7 mR/hr with a corresponding upper range for Xe^{133} on the order of 10^5 Ci/cm^3. Its response to Kr-85 has a upper range on the order of 10^6 ûCi/cm^3. A readout module tagged RM-G25 is located in the Control Room.

Two NaI detectors are provided for gas monitoring of the "A" and "B" OTSG 12" steam lines (atmospheric dump valves, bypass valves, and EF-P1). The detector has a scale of 10 to 10^7 cpm. The monitors provide detection of significant releases and provide information for release assessment. Readout modules are provided in the Control Room, one tagged RM-G26 and the other tagged RM-G27.

Although the instrumentation of item a. does not actuate safety equipment, nor is it required by safety analyses, it is appropriate to provide surveillance requirements to assure reliable postaccident performance of the instrumentation.

The equipment of item a. is designed for postaccident environment usage; however, the equipment is not single failure proof nor redundant, nor protected from casualty events. The equipment of item a. is seismically supported but is not seismically qualified.

b. High-Range Containment Radiation Monitor

1) Design Description

Two high range radiation monitoring channels, RM-G22 and RM-G23, are provided to monitor containment radiation levels during and following a postulated accident. Radiation levels are continuously detected, indicated, and alarmed should the radiation level exceed a predetermined set point. RM-G22 and RM-G23 provide input to the Safety Parameter Display System via the Plant Process Computer.

2) Detailed System Description

Two ionization chamber radiation monitors are mounted in the containment area on the secondary shield wall above elevation 346 feet.

The detectors have a range of 1 R/hr to 10^7 R/hr. These detectors are constructed of hermetically sealed, stainless steel outer surfaces and are ion chamber detectors containing no active electronics. The detector walls are strong enough to withstand LOCA pressures, but thin enough to provide full response to Xe^{133} (80 keV energy level) and are able to measure down to 60 keV gamma photons. The detectors are hermetically sealed, parallel plate, three terminal, guarded ionization chambers, operated in the saturation mode. All external surfaces are schedule 316 stainless steel. The mounting brackets, of the same material, are integral to the housing. Electrical connections are brought out through compression fittings, stainless steel flex hoses, through a gasketed pull-box and piping and finally a potted pullbox to hermetically seal them from LOCA steam. Triaxial cable which is environmentally qualified for nuclear accident service is used between the detector and the electrical penetration.

The electrical installation is designated Nuclear Safety Related to include the detectors and their associated electrical cabling up to ratemeters on the Main Control Room vertical panel PRF (panel right front) and including signal isolators providing isolation from connections to the Plant Process Computer. The readout consists of two readout modules. These readouts present a seven-decade logarithmic range across a 4 inch meter. Each readout contains its own independent power supply and dual independent radiation alarms. The alarm set point is adjustable over the full scale range and is verified by periodic checks. The alarm circuits actuate local panel indicating lights and relays for external annunciators.

The system automatically initiates a self check to assure the integrity of the electrode configuration and electrical operation of the detector/cable/readout system. A success check (no equipment fault) turns on a front panel indicator light which remains on until a fault is detected, thereby providing fail safe indication of proper operation of the system. An equipment fault initiates an alarm.

For manual tests, a test button on the readout panel injects a signal level which approximates one third of full scale signal. This tests all alarm lights and relays and allows test-check of all phases of the system.

Depressing the radiation alarm indicator light causes the meter to indicate the alarm set point.

The containment high radiation monitoring system is designed for continuous operation in all plant operating modes and abnormal accident conditions.

The containment high radiation monitoring system is a completely independent system and interfaces with no other system, except for its source of power, the PRF vertical panel, the plant annunciator system, and the Plant Process Computer.

Calibration and frequency of calibration are provided in accordance with plant procedures.

The high radiation emergency evacuation alarm system is described in Section 11.5.4.

11.4.6 POSTACCIDENT RADIOLOGICAL SAMPLING

Note: Technical Specifications Amendment #253 eliminated the requirements to maintain a Post Accident Sampling System. The Post Accident Sampling System will be maintained for contingency actions and long-term post accident recovery operations (see Section 1.3.2.10 Postaccident Sampling).

A description of the sampling system is provided in Section 9.2.2. Post Accident Radiological sampling associated with the Radiation Monitoring System is described as follows:

Postaccident analysis of reactor coolant samples and the containment atmosphere is recognized as a means to better define long-term recovery operations.

The key parameters to be determined include containment atmosphere spectrum, containment hydrogen concentration, reactor coolant boron concentration, reactor coolant chloride, reactor coolant total gas, and reactor coolant gross beta-gamma activity. The online hydrogen monitoring capability is described in Chapter 6.

A Reactor Building atmosphere sampling station is provided at the Reactor Building atmosphere radiation monitor RM-A2 station. The sample line for radiation monitor RM-A2 includes a three way ball valve to divert samples to the postaccident sampling system. The sampling system provides samples for radiological and hydrogen analysis of the Reactor Building atmosphere. The postaccident sampling system is capable of providing a sample of Reactor Building atmosphere following an accident coincident with a loss of offsite power and with limited personnel exposure. The sampling system is located in a postaccident accessible area in the Intermediate Building. Only the amount of Reactor Building gaseous atmosphere required for the analysis will be transported away from the sampling station. The existing shielding will provide reduction of radiation levels such that any individual in this area will not be exposed to more than 5 rem to the whole body and 75 rem to the extremities. Samples of the Reactor Building atmosphere are drawn by an eductor.

Once flow has been established from the local control panel, a sample is drawn from the sample station either through a septum on a 25cc sample bulb (using a syringe) or removal of the entire sample bulb from the system. After sampling, it is transported to the chemistry lab for analysis. To minimize condensation in the sample line, heat tracing is applied. The unused portion of a sample will be disposed of at the sampling station by sending it back into the containment. The system design will limit containment air loss from a rupture of the sample lines by utilizing passive flow restrictions. Pressure protection is provided to prevent exceeding the rating of the sampling devices. Valves located in postaccident high radiation areas are provided with remote operators.

All lines are capable of being flushed with either instrument air or compressed air/gas from bottles. The RM-A2 sample line and containment isolation valves are capable of being operated with bottle air sources in the event the instrument air system is inoperable.

The high range effluent radioiodine and particulate sampling analysis is provided by the MAP-5 sampling system that includes silver zeolite cartridges. The system design and operation both decrease the activity on the cartridges so they can be handled and decrease the xenon to iodine ratio. Counting of the cartridges is accomplished by use of an NaI crystal connected to a single or dual channel analyzer with appropriate window and discrimination settings for the 364 keV gamma of I-131, or by use of a GeLi/MCA (Multi-Channel Analyzer) system. The postaccident portion of the sampling system would be placed in service following an accident and is located in an area exhibiting low background.

Three processor-controlled iodine and particulate sampling stations are provided and allow a sample to be obtained independent of radiation monitors RM-A5, RM-A8, and RM-A9. The stations are processor-controlled solenoid valves with a valve in one or more of the three parallel filter cartridges provided in each station. The sampling times for each filter cartridge are adjustable and provided on each local control panel as part of the sampling station. The filter cartridges must be manually removed and analyzed elsewhere. Each station provides iodine and particulate samples of high and low radiation level. Two of the stations are located adjacent to the Auxiliary Building on top of the concrete exhaust duct of the Auxiliary and Fuel Handling Building ventilation exhaust duct. Of these two stations, one samples the Auxiliary Building ventilation exhaust and the other the Reactor Building ventilation exhaust. The third station is associated with radiation monitor RM-A5 and provides samples of the condenser vacuum pump discharge.

The sampling stations are normally idle. Upon receipt of a high radiation signal from RM-A8G, the Auxiliary and Fuel Handling Building exhaust monitor MAP-5 sampling station is initiated. Upon receipt of a high radiation signal from RM-A9G, the Reactor Building vent monitor MAP-5 sampling station is initiated. Upon receipt of a high radiation signal from RM-A5G, the condenser vacuum pump discharge monitor MAP-5 sampling station is initiated. Once any of these MAP-5 stations are initiated, they are automatically sequenced. The sample station process controllers control station provides the capability of local manual control (start/stop) and remote control (Control Room). An annunciator alarm is provided in the Control Room when the sampling station vacuum pump for either the RM-A8 or RM-A9 MAP-5 sampler fails to start after a predetermined interval. The RM-A5 associated MAP-5 sampler does not require any sample vacuum pump since it draws its sample from the discharge side of the main condenser vacuum pump header and returns it to the suction side of the pumps.

For the sampling stations, three sampling cartridges are installed in parallel and, by using sequenced solenoid valves, the sample flow through each filter cartridge is established.

11.4.7 NORMAL OPERATION IODINE SAMPLING SYSTEM

A condenser vacuum pump exhaust continuous iodine sampling station is provided to allow a continuous sample of radioiodine offgas effluents. This system adds the capability to continuously sample particulate effluents as well as radioiodine effluents. The station samples from the common discharge header of the condenser vacuum pumps during normal plant operations (as required by the TM1-1 Operating License). The system is further described in Section 9.2.2.6.

11.4.8 SUPPLEMENTAL AIR SAMPLERS AND MONITORS

Numerous installed air sampling instruments are located at various locations in the TMI-1 Auxiliary and Fuel Handling Buildings. Numerous regulated air samplers, and beta particulate air monitors are in service. The subject air monitors supplement the Radiation Monitoring System.

TABLE 11.4-1
(Sheet 1 of 5)

RADIATION MONITORING SYSTEM
POWER SUPPLY SOURCES

Type of Monitor	Tag Number	Location	Vital Bus Power Supply IA	IB	IC	ID
Area Gamma	RM-G1	Control Room	X			
Area Gamma	RM-G2	Radiochemical Laboratory			X	
Area Gamma	RM-G3	Sampling room				X
Area Gamma	RM-G5	Reactor Building near personnel access hatch	X			
Area Gamma	RM-G6	Reactor Building			X	
	RM-G7	Fuel Handling area Floor Elev. 346' D-Ring Wall North/South 2 monitors				X
Area Gamma	RM-G8*	Reactor Building high range				
Area Gamma	RM-G9	Fuel Handling Building				X
Area Gamma	RM-G10	Auxiliary Building entrance elevation 305	X			
Area Gamma	RM-G11	Make-up Demin. Area		X		
Area Gamma	RM-G12	Solid Waste (Trash Compacting Area)		X		

*RM-G8 has been removed from service

TABLE 11.4-1
(Sheet 2 of 5)
RADIATION MONITORING SYSTEM
POWER SUPPLY SOURCES

Type of Monitor	Tag Number	Location	Vital Bus Power Supply IA	IB	IC	ID
Area Gamma	RM-G13	Auxiliary Building entrance elevation 281		X		
Area Gamma	RM-G14	Waste evaporator			X	
Area Gamma	RM-G15	Heat exchanger vault elevation 271				X
Area gamma	RM-G16	OTSG A sample lines		X		
Area gamma	RM-G17	OTSG B sample lines		X		
Area gamma	RM-G18	Pressurizer reactor coolant sample line		X		
Area gamma	RM-G19	Reactor coolant pump seal return		X		
Area gamma	RM-G20	Reactor coolant drain tank pump discharge		X		
Area gamma	RM-G22	RB High range				X
Area gamma	RM-G23	RB High range	X			
Area gamma	RM-G24	RB Purge duct	X			
Area gamma	RM-G25	Condenser vacuum pump exhaust	X			

TABLE 11.4-1
(Sheet 3 of 5)

RADIATION MONITORING SYSTEM
POWER SUPPLY SOURCES

Type of Monitor	Tag Number	Location	Vital Bus Power Supply IA	IB	IC	ID
Area gamma	RM-G26	OTSG A Main Steam Line				X
Area gamma	RM-G27	OTSG B Main Steam Line		X		
Atmospheric Radiation	RM-A1	Control Room vent duct				X
Atmospheric Radiation	RM-A2	Reactor Building air sample line			X	
Atmospheric Radiation	RM-A4	Fuel handling vent duct				X
Atmospheric Radiation	RM-A5 (Low)	Condenser vacuum pump exhaust		X		
Atmospheric Radiation	RM-A5 (High)	Condenser vacuum pump exhaust		X		
Atmospheric Radiation	RM-A6	Auxiliary Bldg. vent duct				X
Atmospheric Radiation	RM-A7	Gas decay tank discharge				X
Atmospheric Radiation	RM-A8(Low)	Auxiliary Building & Fuel Handling Bldg. Vent duct			X	
Atmospheric Radiation	RM-A8 (High)	Auxiliary Building and Fuel Handling Building Vent duct	X			

TABLE 11.4-1
(Sheet 4 of 5)
RADIATION MONITORING SYSTEM
POWER SUPPLY SOURCES

Type of Monitor	Tag Number	Location	Vital Bus Power Supply IA	IB	IC	ID
Atmospheric Radiation	RM-A9 (Low)	Reactor Building Purge Duct	X			
Atmospheric Radiation	RM-A9 (High)	Reactor Building Purge Duct	X			
Atmospheric Radiation	RM-A12	Radio/Chem Sampling Room	(Portable)			
Atmospheric Radiation	RM-A13	Spent Fuel Pool Area (Portable)	(Portable)			
Atmospheric Radiation	RM-A14	ESF Vent Duct Discharge			X	
Atmospheric Radiation	RM-A15	Backup to RMA-5 2nd Floor Turbine Bldg.			X	
Atmospheric Radiation	ALC-RMI-18	CCB/Epicore II 304' Elevation		N/A		
Atmospheric Radiation	WHP-RIT-1	WHPF Mechanical Equipment Room		N/A		
Atmospheric Radiation	RLM-RM-1	RLM Mechanical Equipment Room		N/A		
Liquid Radiation (Low)	RM-L1	Primary coolant Letdown (low)		X		
Liquid Radiation (High)	RM-L1	Primary coolant letdown (high)	X			
Liquid Monitor	RM-L2	Decay heat closed cooling water "A"	X			

TABLE 11.4-1
(Sheet 5 of 5)

RADIATION MONITORING SYSTEM
POWER SUPPLY SOURCES

Type of Monitor	Tag Number	Location	Vital Bus Power Supply IA	IB	IC	ID
Liquid Monitor	RM-L3	Decay heat closed cooling water "B"		X		
Liquid Radiation	RM-L4	Nuclear service closed cooling water		X		
Liquid Monitor	RM-L5	Spent fuel cooling water			X	
Liquid Monitor	RM-L6	Liquid waste discharge				X
Liquid Monitor	RM-L7	Plant effluent line			X	
Liquid Monitor	RM-L9	Intermediate closed cooling water	X			
Liquid Monitor	RM-L10*	Turbine Bldg. Sump				
Liquid Monitor	RM-L12	IWTS and IWFS Discharge	(IWFS-MCC)			

* Monitor retired as described in Section 11.4.4

TABLE 11.4-2
(Sheet 1 of 9)
RADIATION MONITORS
MONITORS FOR PROCESS/SPECIFIC SYSTEMS MONITORING

MONITOR	SYSTEM	LOCATION	DETECTOR	RANGE	INTERLOCKS
RM-G1	Control Room	Control Room Control Bldg. EL. 355'	Ion	0.1-10^7 mR/hr	None
RM-G2	Radiochem. Lab	Radiochem. Lab Control Bldg. EL. 306'	Ion	0.1-10^7 mR/hr	None
RM-G3	Sampling Room	Nuclear Sampling Room Control Bldg. EL. 306'	Ion	0.1-10^7 mR/hr	None
RM-G5	RB Personnel Access Door	Personnel Hatch Reactor Bldg. EL. 305'	Ion	0.1-10^7 mR/hr	None
RM-G6	Fuel Hdlg Area – RB North	D-Ring Wall - RB North EL. 346'	Ion	0.1-107 mR/hr	None
RM-G7	Fuel Hdlg Area – RB South	D-Ring Wall – RB South EL. 346'	Ion	0.1-10^7 mR/hr	None
RM-G8**	RB Hi Range	Top of Elevator Shaft RB	Ion	0.1-10^7 mR/hr*	None
RM-G9	Fuel Handling Building	Fuel Handling Building East Wall	Ion	0.1-10^7 mR/hr	Trips AH-E10 Closes AH-D120, 121, 122
RM-G10	Aux. Bldg. Entrance	Auxiliary Bldg. EL. 305'	Ion	0.1-10^7 mR/hr	None

Notes:

* To compensate for lead shielding, indicated dose rate must be multiplied by a factor of 100.

** RM-G8 has been removed from service.

TABLE 11.4-2
(Sheet 2 of 9)

RADIATION MONITORS
AREA GAMMA MONITORS

MONITOR	SYSTEM	LOCATION	DETECTOR	RANGE	INTERLOCKS
RM-G11	MU Demineralizer Area	Demineralizer Area Aux. Bldg. EL. 305'	Ion	$0.1\text{-}10^7$ mR/hr	None
RM-G12	Solid Waste (Trash Compacting Area)	Radwaste Compacting Area Aux. Bldg. EL. 305'	Ion	$0.1\text{-}10^7$ mR/hr	None
RM-G13	Aux. Bldg. Entrance EL. 281'	Aux Bldg. EL. 281'	Ion	$0.1\text{-}10^7$ mR/hr	None
RM-G14	Waste Evaporator Area	Near Waste Evaporators Aux. Bldg. EL. 281'	Ion	$0.1\text{-}10^7$ mR/hr	None
RM-G15	Heat Exchanger Vault	Heat Exchanger Vault Aux. Bldg. EL. 271'	Ion	$0.1\text{-}10^7$ mR/hr	None
RM-G16	"A" OTSG Sampling	Turbine Bldg. EL. 305'	Ion	$0.1\text{-}10^7$ mR/hr	Closes CA-V4A, 5A
RM-G17	"B" OTSG Sampling	Turbine Bldg. EL. 305'	Ion	$0.1\text{-}10^7$ mR/hr	Closes CA-V4B, 5B
RM-G18	RCS Sample	Aux. Bldg. EL. 305'	Ion	$0.1\text{-}10^7$ mR/hr	Closes CA-V1, 2,3,13

TABLE 11.4-2
(Sheet 3 of 9)

RADIATION MONITORS
AREA GAMMA MONITORS

MONITOR	SYSTEM	LOCATION	DETECTOR	RANGE	INTERLOCKS
RM-G19	RCP Seal Return	Aux. Bldg. EL. 305'	Ion	$0.1\text{-}10^7$ mR/hr	Alarm Only Operator Must Close MU-V-25 or 26
RM-G20	RCDT Discharge	Aux. Bldg. EL. 305'	Ion	$0.1\text{-}10^7$ mR/hr	Closes WDL-V303,304 WDG-V3, V4
RM-G22	RB Hi Range	Sec. Shield RB EL. 346'	Ion	$1\text{-}10^7$ R/hr	None
RM-G23	RB Hi Range	Sec. Shield RB EL. 346'	Ion	$1\text{-}10^7$ R/hr	None
RM-G24	RM-A9 Gas HI-HI	Concrete Exh. Duct RB Block Bldg. Roof	Ion	$0.1\text{-}10^7$ mR/hr	None
RM-G25	RM-A5 Gas HI-HI	Vacuum Exhaust Turbine Bldg. EL. 355'	Ion	$0.1\text{-}10^7$ mR/hr	None
RM-G26	"A" OTSG 12" Steam Line (Bypass)	Interm. Bldg. EL. 295'	NaI	$10\text{-}10^7$ cpm	None
RM-G27	"B" OTSG 12" Steam Line (Bypass)	Interm. Bldg. EL. 295	NaI	$10\text{-}10^7$ cpm	None

TABLE 11.4-2
(Sheet 4 of 9)

RADIATION MONITORS
LIQUID MONITORS

MONITOR	SYSTEM	LOCATION	DETECTOR	RANGE	INTERLOCKS
RM-L1 Lo	RCS Letdown	Aux. Bldg. (Near Seal Return Cooler) EL. 281'	NaI	$10\text{-}10^6$ cpm	None
RM-L1 Hi	RCS Letdown	Aux. Bldg. (Near Seal Return Cooler) EL. 281'	GM Tube	$10\text{-}10^6$ cpm	Closes MU-V2A/2B
RM-L2	"A" DHCCW	Aux. Bldg. (Near Top of "A" DHR Vault EL. 281'	NaI	$10\text{-}10^6$ cpm	None
RM-L3	"B" DHCCW	Aux. Bldg. (Near Top of "A" DHR Vault EL. 281'	NaI	$10\text{-}10^6$ cpm	None
RM-L4	NSCCW	Aux. Bldg. (Near NSCCW Pumps) EL. 305'	NaI	$10\text{-}10^6$ cpm	None
RM-L5	Spent Fuel	FH Bldg. (Near SF Coolers/Pumps) EL. 305'	GM Tube	$10\text{-}10^6$ cpm	None
RM-L6	Liquid Radioactive Waste Discharge	Aux. Bldg. (Near WECST) EL. 281'	NaI	$10\text{-}10^6$ cpm	Closes WDL-V257

TABLE 11.4-2
(Sheet 5 of 9)

RADIATION MONITORS
LIQUID MONITORS

MONITOR	SYSTEM	LOCATION	DETECTOR	RANGE	INTERLOCKS
RM-L7	Plant Water Discharge	RM-L7 Bldg. (Near MDCT)	NaI	10-10^6 cpm	Closes WDL-V257
RM-L9	ICCW	FH Bldg. (Near ICCW Surge Tank) EL. 348'	NaI	10-10^6 cpm	None
RM-L10*	Turbine Bldg. Sump	Turbine Bldg. (Near Powdex Panels) EL. 322'	NaI	10-10^5 cpm	Trips SD-P9A/9B
RM-L12	IWTS/IWFS Effluent	IWTS/IWFS Bldg.	NaI	10-10^6 cpm	Trips IW-P16,17,18 Trips IW-P29, 30 Closes IW-V73,279

* Monitor retired as described in Section 11.4.4

TABLE 11.4-2
(Sheet 6 of 9)

RADIATION MONITORS
ATMOSPHERIC MONITORS

MONITOR	SYSTEM	LOCATION	DETECTOR	RANGE	INTERLOCKS
RM-A1	Control Room	Control Bldg. EL. 380'	P) β_{Scint} G) β_{Scint}	10-10^6 cpm	Control Bldg. placed in Recirc. mode: Trips AH-E-17A/B, 20A/B,21,26, 93A/B, 94A/B, and 95A/B. Closes AH-D-28, and AH-D-617
RM-A2	RB Atmosphere	Interm. Bldg. EL. 295'	P) β_{Scint} I) NaI G) β_{Scint}	10-10^7 cpm 10-10^7 cpm 10-10^6 cpm	None
RM-A4	FH Bldg.	Aux. Bldg. (Near Radwaste Panel) EL. 306'	P) β_{Scint} G) β_{Scint}	10-10^6 cpm	Trips AH-E-10 Closes AH-D-120,121,122
RM-A5	Condenser Off-gas Monitor	Turbine Bldg. EL. 322'	G) β_{Scint}	10-10^7 cpm	Starts MAP-5 Iodine Sampler
RM-A5 Hi	Condenser Off-gas Monitor Hi Range	Turbine Bldg. EL. 322'	G) GM Tube	10-10^6 cpm	None

TABLE 11.4-2
(Sheet 7 of 9)

RADIATION MONITORS
ATMOSPHERIC MONITORS

MONITOR	SYSTEM	LOCATION	DETECTOR	RANGE	INTERLOCKS
RM-A6	Aux. Bldg.	Aux. Bldg. (Near Radwaste Panel) EL. 306'	P) β_{Scint} G) β_{Scint}	$10\text{-}10^6$ cpm	Trips AH-E-11
RM-A7	Waste Gas Disposal Effluent	Aux. Bldg. (Near Beckman Analyzer) EL. 306'	G) β_{Scint}	$10\text{-}10^6$ cpm	Closes WDG-V47

TABLE 11.4-2
(Sheet 8 of 9)

RADIATION MONITORS
ATMOSPHERIC MONITORS

MONITOR	SYSTEM	LOCATION	DETECTOR	RANGE	INTERLOCKS
RM-A8	Aux./FH Bldg.	RM-A8/9 Bldg. (Near BWST)	P) β_{Scint} I) NaI G) β_{Scint}	$10\text{-}10^6$ cpm	Trips AH-E-10,11 Closes WDG-V47 Starts MAP-5 Iodine Sampler
RM-A8 Hi	Aux./FH Bldg.	RM-A8/9 Bldg. (Near BWST)	G) GM Tube	$10\text{-}10^6$ cpm	None
RM-A9	RB Purge Exhaust	RM-A8/9 Bldg. (Near BWST)	P) β_{Scint} I) NaI G) β_{Scint}	$10\text{-}10^6$ cpm	Closes AH-V1A/B/C/D Closes WDG-V534,535 Starts MAP-5 Iodine Sampler
RM-A9 Hi	RB Purge Exhaust	RM-A8/9 Bldg. (Near BWST)	G) GM Tube	$10\text{-}10^6$ cpm	None
RM-A12	Radio/Chem Sampling Room	Movable	P) β_{Scint} I) NaI G) β_{Scint}	$10\text{-}10^6$	None
RM-A13	Spent Fuel Area	Movable	P) β_{Scint} I) NaI G) β_{Scint}	$10\text{-}10^6$ cpm	None

TABLE 11.4-2
(Sheet 9 of 9)

RADIATION MONITORS
ATMOSPHERIC MONITORS

MONITOR	SYSTEM	LOCATION	DETECTOR	RANGE	INTERLOCKS
RM-A14	FHB ESF Ventilation System	ESF Ventilation System Enclosure EL. 331' (Roof of Aux. Bldg.)[1]	G) β_{Scint}	10^{-6}-10^{3} ûCi/cc Xe^{133}	None
RM-A15	Condenser Off-gas Back-up	Turbine Bldg. EL. 322'	G) β_{Scint}	10-10^{7} cpm	Starts MAP-5 Iodine Sampler
ALC-RMI-18	CCB/ Epicore II Exhaust System	CCB/ Epicore II 304' Elevation	P) β_{Scint} G) β_{Scint}	75 µci 10^{-6} - 10^{-1} µCi/cc	None
WHP-RIT-1	WHPF Exhaust System	WHPF Mechanical Equipment Room	P)GM Tube	10-10^{5} cpm	Trips WHPF Ventilation
RLM-RM-1	RLM Exhaust System	RLM Mechanical Equipment Room	P)GM Tube	10-10^{5} cpm	None

(1) Locations of sample point Detectors are located in Auxiliary Building Main Ventilation Room EL. 312'

Table 11.4-3

DELETED

11.5 RADIOLOGICAL HEALTH AND SAFETY

The Director - Radiological Health and Safety is responsible for radiation protection and contamination control for the nuclear facility. This responsibility includes administration of the Radiation Protection Program. All personnel and visitors are required to follow procedures established for protection against radiation and contamination. The Training Department personnel train station personnel in radiation safety, including location and evaluation of radiological problems, and making recommendations for control or elimination of radiological hazards. The Radiological Controls personnel function in an advisory capacity to assist all personnel in carrying out their radiation safety responsibilities, and to audit all aspects of plant operation and maintenance to assure safe conditions and compliance with applicable regulations concerning radiation protection. Per the TMI-1 Operational Quality Assurance Plan, audits of the Radiological Controls Program are conducted by qualified persons outside the Radiological Health and Safety Department.

Administrative controls assure that all procedures and requirements relating to radiological protection are followed by all station personnel. The procedures that control radiation exposure are subject to the same review and approval as those that govern all other station procedures (see Section 12.3, Procedures). These procedures include a Radiation Work Permit system.

11.5.1 RADIATION WORK PERMITS

A Radiation Work Permit is required by all personnel prior to any work or entry into restricted areas that would involve any of the following:

a. High radiation area

b. Airborne radioactivity area

c. Contaminated area

d. Those radiation areas specified in applicable procedures.

This procedure provides an excellent means of controlling access to areas where personnel could receive excessive exposures.

Only in the event that the safety of the plant or its personnel is endangered may entry be made into a Radiation Work Permit area without a Radiation Work Permit. In such a case, the entry will be made by qualified personnel carrying monitors, and a Radiation Work Permit will be executed shortly thereafter.

11.5.2 PERSONNEL MONITORING SYSTEM

Dose is maintained within the limits established in 10CFR20. All personnel entering a radiation area will wear dosimetry devices as specified by procedure or the Radiation Work Permit. Types of dosimetry that are readily available are self-reading dosimeters, alarm capable dosimeters, thermoluminescent dosimeters, and extremity dosimetry. This personnel monitoring equipment will also be available on a day-to-day basis for those persons, employees, or visitors not assigned to the station who have occasion to enter Radiation Control Areas, or to perform work involving possible exposure to radiation. Records of radiation exposure history and current occupational exposure are maintained by the Radiological Controls section for each individual requiring personnel monitoring. The external radiation dose to personnel will be determined by means of a self reading dosimeter for each Radiation Work Permit entry. Thermoluminescent Dosimetry will be processed annually or more frequently as necessary.

11.5.3 PERSONNEL PROTECTIVE EQUIPMENT

Special protective or anticontamination clothing will be furnished and worn as necessary to protect personnel against contact with radioactive contamination. Change areas are conveniently located for proper utilization of this protective clothing.

Respiratory protective equipment is also available for the protection of personnel against airborne radioactive contamination. The equipment consists of full-face filter masks, self contained air breathing units, or air-supplied masks and hoods. The first line of defense against airborne contamination in the work area is the ventilation system. However, respiratory protective equipment is provided should its use become necessary.

Maintenance of the above equipment is in accordance with the manufacturer's recommendations and 10CFR20, "Standard for Protection Against Radiation." The use and maintenance of this equipment are under the direct control of the Radiological Controls Section. All personnel will be trained in the use of this equipment, utilizing it in the performance of work.

11.5.4 ACCESS CONTROL

Change area facilities are provided where personnel may obtain clean protective clothing required for station work. These facilities are divided into clean and contaminated sections. The contaminated section of the change area is used for the removal and handling of contaminated protective clothing after use. Showers, sinks, and necessary monitoring equipment are also provided to aid in the decontamination of personnel.

Equipment decontamination facilities are provided at the station for large and small items of plant equipment and components.

All facilities and equipment necessary to exclusively support the radiation protection and decontamination activities of Unit 1 are provided.

Provision is made for decontamination of work areas throughout the station. Appropriate written guidance governs the proper use of protective clothing, where and how it is to be worn and removed, and how the change area and decontamination facilities for personnel, equipment, and station areas are to be used.

Administrative and physical security measures are employed to prevent unauthorized entry of personnel to any high radiation areas. Warning signs, audible and visual indicators, barricades, and locked doors will be used as necessary. The Radiation Work Permit system is also utilized to control access to high radiation areas.

The emergency evacuation monitoring system speakers, alarm lights, and page phones are provided in work areas to include high-noise areas so as to advise working personnel in these areas that an emergency evacuation signal is initiated (References 20 and 21).

Should the radiation monitors detect an increase in the level of radiation or airborne activity in a work area, an alarm will be initiated in the Control Room. The installed emergency lights (red flashing) have a minimum of 2 foot-candles at a distance of 2 feet.

The installed sound alarms are 6 dB minimum above the ambient noise level in the work areas.

The mounting of equipment to be installed in the vicinity of safety related equipment is designed so that no hazardous missiles will be created due to a seismic event.

Routinely, an operational test is performed to assure the emergency alarm system is operational.

11.5.5 RADIOLOGICAL CONTROL LABORATORY FACILITIES

The unit includes a Radiological Control Laboratory with facilities and equipment for detecting, analyzing, and measuring alpha, beta, or gamma radiation and for evaluating any radiological problem which may be anticipated. Counting equipment (such as G-M, scintillation, and proportional counters) is provided in an appropriately shielded environment for detecting and measuring all types of radiation as well as a High Purity Germanium detector/MCA for the identification of specific radionuclides.

11.5.6 RADIOLOGICAL CONTROL INSTRUMENTATION

Portable radiation survey instruments are provided. A variety of instruments have been selected to cover the entire spectrum of radiation measurement problems anticipated at TMI-1. Sufficient quantities have been obtained to allow for use, calibration, maintenance, and repair. This includes instruments for detecting and measuring alpha, beta, gamma, and neutron radiation. In addition to portable radiation monitoring instruments, fixed monitoring instruments, i.e., count rate meters, are located at exits from radiation control areas. These instruments are intended to prevent any contamination on personnel, material, or equipment from being spread into unrestricted areas. Appropriate monitoring instruments are available at various locations within the radiation control areas for contamination control purposes. The station has permanently installed area and process radiation monitoring systems. These systems monitor airborne particulate and gaseous radioactivity, including iodine, as well as external radiation levels. The systems present audible alarm and radiation level indication in the areas of concern in addition to reading out in the Control Room.

11.5.7 MEDICAL PROGRAM

A physical examination and medical surveillance program is provided to establish and maintain records of the physical status of those employees who must meet specific OSHA and regulatory physical criteria in order to perform their duties at the Three Mile Island nuclear facility. Medical examinations shall be performed as required. The site Medical Department is responsible for the program.

11.5.8 RADIATION PROTECTION PROGRAM

11.5.8.1 Foundation For The TMI-1 Radiological Controls Program

The objective of the Radiological Controls Program is to control radiation hazards, to avoid accidental radiation exposures, to maintain exposures within the regulatory requirements, and also to maintain exposures to workers and the general population as low as is reasonably achievable. These philosophies, policies, and objectives are based on and implement the regulations of the NRC as contained in Title 10 of the Code of Federal Regulations, Parts 19, 20, 50, and 71, and appropriate Regulatory Guides, specifically 8.2 (1973), 8.8 Rev. 3 (1978), 8.9 Rev. 1 (1993), 8.10 Rev. 1-R (1975), 8.13 Rev. 3 (1999), and 8.14 (1977).

Specific details as to how the Radiation Protection Program is implemented is promulgated in site and company procedures and includes those applicable procedures addressed in Reg. Guide 1.33 Rev. 2 (1978), App. A, paragraph 7, and paragraph 8 (1) (aa), (bb). Requirements governing release of radioactive liquids and gases to the environment and the disposal of solid radioactive waste are addressed in the Technical Specifications and the Off-Site Dose Calculation Manual (ODCM). The site Radiological Health and Safety Department will concur in planned batch offsite radioactive releases prior to the release being made.

The Radiological Controls procedures are written to provide an effective Radiological Controls Program at TMI-1. Procedures provide adequate guidance and specify appropriate methods or techniques to insure that the performance of each activity is in accordance with sound radiological control principles and is in compliance with applicable regulatory provisions. The Radiological Controls procedures are prepared, reviewed, approved, and controlled as described in company administrative procedures.

The TMI-1 Radiological Controls Program is fully integrated into each and every phase of operations at TMI-1. The TMI-1 Radiological Controls Program, when carried out as specified, will assure that the operation of Unit 1 will be performed with personnel who work at the site incurring radiation exposure as low as can reasonably be achieved (ALARA).

In order to meet this objective, the program must be carried out by each person involved in the TMI-1 activities. There is no group or person involved in the TMI-1 operations who does not have some degree of responsibility for the Radiological Controls Program.

11.5.8.2 Deleted

11.5.8.3 Audits, Reviews, And Reports On The TMI-1 Radiological Controls Program

Each individual is responsible for maintaining his or her radiation exposure as low as reasonably achievable while completing the scope of work he or she is required to perform. Each is required to comply with the applicable procedures of the TMI-1 Radiological Controls Program and the specific radiological controls prescribed for work in which he or she is engaged.

In order to ensure that these requirements are being met and to assist all site personnel in understanding and complying with these requirements, a system of audit and review procedures has been established, including criteria for timely and appropriate corrective action. The following assessments and review procedures are used:

a. Radiological control technicians monitor and aid the performance of each individual's radiological work practices.

b. Self assessments of the Radiological Controls Program will be conducted.

c. Radiological assessments are conducted throughout the Radiological Controls Program on an ongoing basis. This assessment function reports through Nuclear Oversite management.

d. Quality Assurance audits of the TMI-1 Radiological Controls Program are conducted by technically qualified persons from outside the Radiological Controls Department. These audits are conducted in accordance with procedures as outlined in the Operational Quality Assurance Plan. The TMI Nuclear Oversite Department schedules these audits and provides personnel from their own department and/or outside contractors as appropriate to conduct the audits. These audits are performed in accordance with the TMI-1 Operational Quality Assurance Plan, and all procedures in the TMI-1 Radiological Controls procedures at least every 24 months.

e. In addition to these reviews and audits, various systems are employed to identify radiological control deficiencies. A radiological control deficiency is defined as a violation of an established procedure or practice. Such deficiencies are recorded in Internal Assessments, Surveillance Inspection Reports, Corrective Action Program Process, or other reporting mechanisms. The purpose of these systems is to identify deficiencies whose correction will result in an improved Radiological Controls Program. There shall be at least one reporting system for individuals to report deviations from good radiological practices.

f. The NRC also inspects and reviews the TMI-1 Radiological Controls Program.

g. In the event all the preceding measures fail to prevent a radiological incident, an investigation shall be conducted to determine the causes of the incident and to determine the corrective actions and improvements needed.

11.5.8.4 Radiological Controls Training

a. Periodic radiological controls training is given to ensure that each person understands the radiological conditions to which he is exposed, understands his responsibility to minimize his own exposure to radiation, and understands his own responsibilities for complying with radiological control procedures. Personnel occupationally exposed to radiation receive instruction on the effects of radiation and the risks associated with radiation exposure.

b. General radiological indoctrination is given to those not directly involved with radiation so that they understand not to enter areas requiring TLDs and not to cross radiation barriers. The indoctrination includes explanation of the radiological environment in which they work.

c. Those who require access to areas controlled by Radiation Work Permits receive extensive training and are required to pass a radiological examination on their practical abilities, including use of dosimetry, frisking, anticontamination clothing, and response to unusual situations. In addition, those who need to use a respirator receive respirator training. Retraining and both written and practical examinations are conducted at least annually. In addition, spot checks are made to ensure that they retain the required knowledge during the period between examinations. Special briefings and extra training, including use of mockups where applicable, may be required for work involving higher than usual exposures to radiation and radioactivity.

11.5.8.5 Control Of External Radiation Exposure

Control of radiation exposure is based on the assumption that any exposure, no matter how small, involves some risk; however, exposure within the accepted limits represents a risk small compared with normal hazards of life. Therefore, the policy of TMI-1 is to maintain exposures to individuals and total man-rems as low as is reasonably achievable (ALARA). Line management from all departments as well as each individual worker shall take an active role in radiation exposure reduction.

To aid in exposure reduction, administrative radiation exposure control levels shall be established. Radiation person-rem exposure goals shall be established for each major job and for each year. Work involving radiation exposure shall be preplanned. Major exposure jobs shall require that radiological controls be incorporated in the design, that written procedures be prepared, and that pre-job briefing and rehearsals be conducted prior to commencing work. A Radiation Work Permit will be required for any work or entry to restricted areas that would involve or create any of the following: (1) high radiation area, (2) airborne radioactivity area, and (3) contaminated area, or (4) those radiation areas specified in applicable procedures.

Restricted areas used to control personnel access to radiation and radioactive materials are defined, access controlled, and posted in accordance with 10CFR20 except as allowed by Technical Specification step 6.12.1.

Radiological Controls personnel are exempt from the RWP issuance requirements during the performance of their assigned radiation protection duties providing they are following radiological control procedures for entry into High Radiation Areas.

To evaluate radiological conditions, radiation surveys are conducted for air activity, removable surface contamination, and external radiation at regular intervals. Surveys are performed in order to (1) monitor the suitability of control measures, (2) evaluate the needs for additional controls, (3) evaluate trends for ALARA purposes, and (4) evaluate radiological conditions in areas routinely entered without radiation work permit coverage. Surveys in unrestricted areas are provided to insure the effective control of radioactive material. Unusual conditions detected in the performance of either a routine or special survey are immediately brought to the attention of Radiological Controls management. Portable radiation survey instruments are calibrated at least annually, except for direct reading dosimeters which will be calibrated at least semiannually, to assure a consistent, reliable, and predictable response to radiation levels. Records of surveys are maintained on file. An administrative program is used to verify the calibration of personnel and field monitoring instruments.

11.5.8.6 Control Of Internal Radiation Exposure

It is the policy of Exelon to prevent significant unplanned exposure to personnel from radioactivity associated with TMI-1. Intakes may be planned consistent with TEDE-ALARA concepts and other occupational health considerations. (TEDE – Total Effective Dose Equivalent).

11.5.8.7 Control Of Radioactive Contamination

Radioactive surface contamination is controlled in order to minimize possible inhalation or ingestion of radioactivity and to minimize buildup of radioactivity in the environment. Measures are taken to contain radioactivity and to minimize the number and extent of areas contaminated in order to minimize personnel radiation exposure, to simplify subsequent personnel and area or facility decontamination, and to minimize the need to rely on anticontamination clothing.

The surface contamination level release limit for uncontrolled use of items is 5000 dpm/100cm^2 for total beta-gamma contamination, 300 dpm/100cm^2 for total alpha contamination, 1000 dpm/100cm^2 for smearable beta-gamma contamination, and 20 dpm/100cm^2 for smearable alpha contamination.

The skin contamination action level is ≥100 counts per minute above background, as indicated by a pancake probe frisk. (This is equivalent to 5000 dpm/100cm^2 beta-gamma.)

Emphasis is placed on planning, training, and working in order to minimize the number of occurrences of radioactive surface contamination of a person's skin or of areas not controlled for radioactive surface contamination, and the amount of radioactivity involved. Each such occurrence is reviewed in detail to determine how to correct deficiencies and improve control of radioactivity.

11.5.8.8 Organization For Radiological Controls

The Director - Radiological Health and Safety is responsible for ensuring that a high quality Radiological Controls Program is established and maintained. It is the responsibility of the Radiological Health and Safety Department to evaluate radiological conditions and recommend precautionary measures. The Radiological Health and Safety Department is organized as shown on Figure 12.1-2.

At times when demands upon the Radiological Health and Safety Department are sufficiently heavy to require a temporary increase in staff, qualified contractor personnel will be used. These personnel will be fully integrated into the department under the direction of the Director - Radiological Health and Safety. Support services (instrument calibration, respirator protection, bioassay, TLD/dosimetry, and training) may be provided by Radiological Controls or other licensee organizations. These support services will be administered by procedures which define the organization interface required to insure that the quality of services provided meets the commitments of the Radiation Protection Plan.

Qualifications for the key radiological managers contained in NRC Regulatory Guide 1.8, Rev. 1-R (1975), will be met as far as practicable. Where the combination of a strong manager and experience in radiological controls cannot practicably be obtained in the same person, a deputy will be appointed who meets the NRC's guidance.

TABLE 11.5-1

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11.6 RADIOACTIVE MATERIALS SAFETY

11.6.1 MATERIALS SAFETY PROGRAM

11.6.1.1 Control Of Radioactive Materials

The following program will be implemented to assure the safe storage, handling, and use of sealed and unsealed special nuclear, source, and byproduct materials. In addition to the definition of 10CFR20, any material with surface contamination in excess of the limits specified in Section 11.5.8.7 shall be handled as radioactive. A radioactive material control system shall be established to ensure radioactive material is not lost or misplaced in a location where personnel could unknowingly be exposed to radiation and to prevent the uncontrolled spread of radioactivity to areas where the public might be affected. This system shall include the following requirements:

a. The number of areas in which radioactive materials are stored shall be minimized.

b. Any long term radioactive materials storage area shall be approved by the Director - Radiological Health and Safety.

c. The numbers of radioactive items and the amount of radioactivity in storage shall be minimized.

d. Radioactive items shall be identified as radioactive before removing them from a restricted area.

e. Radioactive materials removed from the Protected Security Area or removed from a restricted area outside the Protected Security Area shall be controlled in accordance with an accountability procedure which ensures the materials are not lost or improperly handled during transfer or subject to unauthorized removal. This accountability procedure shall require inventory of radioactive materials which remain outside such areas.

f. Each incoming or outgoing shipment of radioactive material shall be handled in strict compliance with detailed written procedures.

Each case in which radioactive material is lost or unaccounted for shall be reviewed in detail to determine the potential radiation exposure personnel might unknowingly receive, to correct deficiencies, and to improve control of radioactive materials.

11.6.1.2 Receipt Of Radioactive Sources

The method of receipt, date, and description of the contents will be recorded. Each radioactive sealed source will be surveyed for removable contamination prior to use. Records of the survey will be maintained. Each sealed source and/or source container will be smeared using cotton swabs or small filter papers. These smears will be counted in the counting laboratory for alpha and/or beta gamma activity, as appropriate. If leakage is confirmed from any source, the source will be repaired or replaced.

Each source will be numbered and recorded in the Source Accountability System.

11.6.1.3 Storage Of Radioactive Sources

Each source will have an assigned storage location and an assigned custodian responsible for the security of the source.

11.6.1.4 Source Preparation And Handling

Only personnel trained in Radiological Controls practices are permitted to handle radioactive sources. These personnel will wear dosimeters/TLDs while handling radioactive sources as required by a Radiation Work Permit . When radioactive material containers are opened, a radiation and contamination survey will be taken as appropriate.

Protective clothing, consisting of rubber gloves, lab coats, and shoe covers, will be worn, as required, during the preparation of the radioactive calibration sources.

All liquid or gaseous radioactive sources will be opened under a ventilation hood. The area where the opened radioactive material is handled will be appropriately marked and posted. Radiological Controls practices, appropriate to the material being handled, will be followed.

Standards prepared at the station will be carefully made so as to limit the spread of radioactivity. For liquid sources, a cover is provided for the source container to prevent spillage during transport from the source storage area to the source preparation area and vice versa.

A smear sample will be taken of each sealed source for detection of leaks or removable contamination. The detection limit of instrumentation used should be able to detect the presence of 0.005 uCi of removable contamination.

11.6.1.5 Disposal Of Radioactive Waste Resulting From Source Preparation

Liquid waste, which is expected to result from source preparation, will be stored in a radioactive waste storage area prior to disposal.

Solid waste consisting of absorbent paper and kimwipes, etc., will be handled in accordance with solid waste handling procedures.

11.6.1.6 Radiological Surveys

A radiation survey and a survey for contamination will be conducted in the laboratory source preparation area after the sources are prepared. In addition, a radiation and contamination survey will be conducted once per week for laboratory areas in use. Leak testing for radioactive surface contamination will be conducted in accordance with Technical Specifications 4.13 and 3.10.1. Sealed sources are exempt from such leak tests when the source contains 100 ûCi or less of beta and/or gamma emitting material or 5 ûCi: or less of alpha emitting material. The survey detection limit of contamination must be equal to or less than 0.005 ûCi.

Routinely scheduled radiation and contamination surveys will be conducted in and around the source storage areas.

11.6.1.7 Source Inventory

A source inventory will be maintained current at all times in accordance with Technical Specification 3.10.2. Records of the inventory will be kept by Radiological Controls.

11.6.2 FACILITIES AND EQUIPMENT

Three Mile Island Unit 1 includes a Radiological Control Laboratory with facilities and equipment for detecting, analyzing, and measuring all types of radiation and for evaluating radiological problems which may occur. Counting equipment (such as G-M, scintillation, and proportional counters) are provided in an appropriately shielded counting room for detecting and measuring all types of radiation.

The unit radiochemistry lab and adjacent area are equipped with the following equipment:

a. Hoods with HEPA and charcoal filters

b. Laboratory sinks which drain to the liquid radioactive waste disposal system

c. A continuous particulate radiation monitor

d. A personnel contamination monitoring device

11.6.3 PERSONNEL AND PROCEDURES

The Director - Radiological Health and Safety, Manager - Radiological Engineering, Manager - Radiological Field Operations, Group Radiological Controls Supervisor, and Manager – Radwaste and Chemistry are responsible for the handling and monitoring of the sealed and unsealed special nuclear, source and byproduct materials. Resumes for these key personnel are available at the TMI-1 site.

Site procedures are written to provide instructions and guidance on radiation protection, sampling, laboratory analyses used to determine concentration and species of radioactive liquids and gases, and calibration of laboratory equipment.

Surveillance procedures are written to ensure compliance with Technical Specification requirements.

11.6.4 REQUIRED RADIOACTIVE SOURCES

Various radioactive sources are required with various radionuclides, quantities and form.

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