11.A POSTACCIDENT SHIELDING EVALUATION

11.A.1 GENERAL

Following the accident at TMI-2, a shielding evaluation identified the necessary action to be taken for providing additional shielding in vital areas in which equipment may be unduly degraded by the large radiation fields in a postaccident environment and in which personnel may receive excessive exposures (References 1, 2, 3, and 4).

Systems, components, and areas considered subject to postaccident large radiation fields include: decay heat removal, reactor building spray recirculation, makeup and letdown, waste gas, Rad Waste Control Center, and Liquid and Gas Waste Panel.

11.A.2 SOURCE TERMS FOR POSTACCIDENT SHIELDING EVALUATION

For the evaluation of a similar plant, Midland was used to represent TMI. The Midland core was modelled as an equilibrium cycle of 310 EFPD (effective full power days) at a power level of 2552 MWth, and used to develop the source terms in Table 11A-1. A comparison of dose rates using the above power history with dose rates from a 2568 MWth, 661 EFPD core showed that calculated dose rates increased by less than 5%. This TMI Cycle 10 core as modelled is typical of expected future TMI cycles. Therefore, there are no significant changes to the postaccident exposure rates expected due to this revised inventory, since this modest increase is accommodated by known conservatisms in the evaluation. [Reference 11]

The activity assumed for liquid source term calculations is based on the following:

a. Liquid Containing Systems

   One hundred percent of the core equilibrium noble gas inventory, 50 percent of the core equilibrium halogen inventory, and 1 percent of all others are assumed to be mixed in the reactor coolant and liquids recirculated by the high pressure injection (HPI) and low pressure injection (LPI) systems. In determining the source term for recirculated, depressurized cooling water, it is assumed that the water contains no noble gases.

b. Gas Containing Systems

   One hundred percent of the core equilibrium noble gas inventory and 25 percent of the core equilibrium halogen activity are assumed to be mixed in the containment atmosphere.

Two liquid source terms are used in the evaluation. For systems or portions of systems which contain postaccident recirculation fluid, the source term is based on diluting the liquid inventory discussed previously with the minimum expected volume of fluid in the bottom of the Reactor Building postaccident, including the core flood tanks (CFTs), borated water storage tank (BWST), and reactor pressure vessel (PV). This is designated as the Recirculation source.

For systems which will contain postaccident fluid from the reactor coolant system which will not be diluted as noted above, the source term is based on diluting the liquid inventory discussed above with the volume of fluid in the reactor coolant system. This is designated as the Reactor Coolant source.

The activity assumed for the containment gaseous source term calculations was based on 100 percent of the noble gas core inventory and 25 percent of the halogen core inventory. The containment airborne source term was based on diluting the isotopic inventory of Subsection 11.A.2b with the air contained in the containment free volume. This is designated as the "Containment Gas" source.

The inventories and source terms discussed above are calculated for the time period immediately after the postulated accident. For other time periods, the decay parameters given in References 5 and 6 are used to adjust the source terms for radioactive decay. The sources are converted to standard shielding source term format as a function of time after the postulated accident and are used as the basic input data to the shielding codes.

11.A.3 CALCULATION OF DOSE RATES

Both the Shield Design Calculation Code (SDC), Reference 7, and the Gilbert/Commonwealth-developed SPOT1 code have been used in performing the dose rate calculations. The SDC code uses the methodology of Reference 8, which represents the cylindrical source by a equivalent line source. The SPOT1 code uses the methodology originally presented in Reference 9, which represents the cylindrical source by an equivalent cylindrical annular segment. These codes give comparable calculational results; therefore, the use of one code or the other is based on convenience.

These codes are used to calculate dose rates at the midplane laterally from cylindrical sources. Dose rate data are calculated for explicit pipe segments containing various sources. Dose rate data are calculated for shielding and unshielding conditions and as a function of time after the postulated accident. Dose rate data from tankage are also calculated utilizing these shielding codes.

The dose rate was determined at a representative location within a given area, and that dose rate was used as the general area dose rate. The criteria used for selection of a representative location was two-fold: first, that the dose rate should be reasonably representative of the area and, secondly, that the dose rate should be conservative for the area. Usually, the main contribution of the dose rate comes from unshielded piping in the immediate area. Other sources farther away or behind shield walls usually contribute significantly less to the general area dose rate.

As indicated above, midplane dose rate data versus lateral distance are calculated for explicit pipe segments containing various sources. Then, the distance from a given pipe to the representative location within the area is determined from the physical piping layout drawings. This distance is then used to determine the dose rate from the dose rate versus distance data. This being done for all pipes in the area, the dose rate contributions are added to obtain the total dose rate at the representative location within the area. Where significant, dose rate contributions from tankage or shielded pipes are also to obtain the total dose rate for an area.

Denoting time at the inception of an accident as time T=O, and noting that some operations may be performed at times significantly after T=O dose rate may generally be reduced by approximately a factor of 5 for T=8 hours, and by a factor of about 30 for T=5 days.

11.A.4 CALCULATION OF DOSES TO PERSONNEL DURING POSTACCIDENT ACCESS TO VITAL AREAS

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

The doses received during travel are determined by calculating dose rates at selected locations (or at a single location if the dose rate along the travel route is relatively uniform) along the travel route and multiplying the dose rates by the appropriate travel time from each selected location along the travel route. It is assumed that the individual travels at a rate of 50 feet per minute along the travel paths.

Doses received while performing a given operation are determined by multiplying the dose rate for the given area by the time assumed to perform the operation. Dose rates for the given vital area are determined using the methodology discussed above.

11.A.5 ACCEPTANCE CRITERIA

The acceptance criteria for personnel access will be based on the following guidelines:

a. The postaccident dose rate in areas requiring continuous occupancy should not exceed 15 mr/hour.

b. The postaccident dose rate in areas which do not require continuous occupancy should be such that the dose to an individual during a required access period is less than 5 rem whole body or its equivalent.

c. The minimum radioactive source terms used in the evaluation will be equivalent to the source terms recommended in Regulatory Guide 1.4 as clarified by documents of Reference 1, 2, 3, and 4.

These guidelines are design objectives and will not be a basis for limiting access in the event of an accident.

The acceptability of equipment was determined based on the results of the review of electrical safety related (Class 1E) equipment conducted in response to I&E Bulletin 79-01B and reported separately. Further information on equipment qualification is contained in Appendix 6B of this FSAR.

11.A.6 REVIEW OF PLANT AREAS FOR POSTACCIDENT ACCESS REQUIREMENTS

Areas which will or may require occupancy to permit an operator to aid in the mitigation of or recovery from an accident have been designated as vital areas. The Control Room, Technical Support Center (TSC), Operations Support Center (OSC), sampling station and sample analysis area, post-LOCA hydrogen control system, motor control centers, instrument panels, emergency power supplies, security center, and radwaste control panels are included within this designation.

Specific designation and discussion of these and other areas where postaccident access might prove useful follow. All areas evaluated are identified by Roman numerals and are shown on Figures 11.A-1 through 11.A-10. A summary of the dose rate by area is given in Table 11.A-2.

This review is predicted on the fact that letdown of reactor coolant outside containment will not be employed when coolant activity is at unsatisfactory levels. Letdown will be automatically terminated via the containment isolation system and will not be reestablished if activity levels are unacceptably high.

Plant procedures address postaccident letdown as it relates to postaccident shielding.

High pressure injection and low pressure injection piping and components located outside the Reactor Building will contain only the initial injection source following an accident. Also, there will be no accumulation of radioactive gas resulting from the accident in the waste gas system and components located outside containment.

The radiation dose rates identified herein are based on the conservative assumption that any systems and components may contain the sources listed in Section 11.3.1 will contain those sources at T=0. Actual recirculation of spilled reactor coolant may not occur until the clean water in the borated water storage tank is expended.

a. Areas Requiring Access

1) Area I

Area I is located in the Fuel Handling Building at the 305-ft elevation, as shown on Figure 11.A-10. This area is part of the travel route to other areas in the Auxiliary Building.

A radiation dose rate of 30 mrem/hour in this area results from the reactor coolant sample inlet line routed through Area V as shown on the figures.

The time to travel through the area will be less than a minute with resultant exposure of less than 0.5 mrem.

2) Area II

Area II is located on the 305-ft elevation of the Auxiliary Building as shown on Figure 11.A-3. It contains the liquid and waste gas control panels and the decay heat removal pumps remote oilers. Also, Area II is in the travel route to Areas III and IV on this elevation.

The source of radiation in this area is the Auxiliary Building HVAC exhaust duct which runs the length of the area about 15 feet above the floor. The dose rate from this source is less than 0.1 rem/hour at T=0 if all of the 0.01 percent per day containment leakage goes through this duct.

A dose rate of less than 1.0 rem/hour at T=0 results from an assumed 1-gpm makeup system liquid leakage inside the Auxiliary Building. This dose rate assumes a partition factor of 1 for the noble gases and 0.1 for the halogens. The 1-gpm leak rate greatly exceeds the anticipated leakage in the Auxiliary Building through modifications for the leakage reduction program.

Postaccident access to Area II may be desired to perform liquid and gas waste transfers and read waste gas decay tank pressure. Occupancy time to perform these activities are 10 minutes and 1 minute, respectively, during which radiation exposures of less than 183 and 18 mrem, respectively, would be experienced by Area II occupants.

Access to Area II may also be required to add oil to the decay heat removal pump bearings via the remote oilers. Occupancy time to perform this operation is 5 minutes with a resultant exposure of less than 92 mrem.

3) Areas III and IV

Areas III and IV are located on the 305' elevation of the Auxiliary Building as shown on Figure 11.A-3. Access to Area IV will be required immediately after an accident, but before starting recirculation from the

Reactor Building sump, to open Makeup and Purification valve MUV-198 to bypass the reactor coolant pump seal injection filters, unlock and open DHV-64 for long term boron precipitation control, unlock and close CAV-371 to isolate potential high activity leak path from RCS to Auxiliary Building atmosphere via relief CAV-446. Access to Area III may be required to reset any thrown circuit breakers at engineered safeguards motor control centers (MCCs) 1A and 1B located in the area. An occupancy time of five minutes is estimated for any of these operations.

A radiation dose rate in Area IV at T=0 (start of recirculation) of $1.2x10^4$ rem/hour results from the following Makeup and Purification piping and components: reactor coolant pump seal injection piping, high pressure injection piping to cold legs 'C' and 'D', and the seal injection filters and valves. This would result in a dose of 1000 rem to an operator occupying the area for the 5 minutes required to open MUV-198. A procedural control for the operation of MUV-198, DHV-64, and CAV-371 has been instituted to operate the valves prior to starting recirculation.

The radiation dose rate in Area III at T=0 (start of recirculation) would be 6,300 rem/hour, generated from the piping and components in Area IV. Assuming access is not required to Area III until 8 hours after an accident (i.e. T=8 hours) the general area dose rate would be 1,300 rem/hour resulting in a dose to the operator of 110 rem for a five minute staytime to reset breakers. A shield wall consisting of iron and concrete has been erected on the 305' elevation Auxiliary

Building between Area IV and the MCCs (reference 15). The resulting dose rates to Area III are 191 rem/hour at T=0 (start of recirculation) and 38 rem/hour at T=8 hours. The dose to the operator at T=8 hours to access the 1A and/or 1B MCC breakers would be 3.2 rem for the five minute staytime.

A remote valve operator has been installed for DHV-64 and situated so the valve handwheel receives the benefit of the shield wall, with access to the handwheel resulting in doses identical to Area III.

The estimated radiation dose for travel to and from Areas III and IV is negligible in comparison to the dose received during area occupancy.

4) Areas XI and XII

Areas XI and XII are located on the 281’ elevation of the Auxiliary Building (reference Figure 11.A-2). Two operations in these areas that may be required after an accident are: (1) boron precipitation control and (2) continued operation of the decay heat removal system if the postulated accident were the result of a break in one of the low pressure injection lines and the decay heat pump in the unfaulted line failed. These operations require access into Areas XI and XII to operate valves DHV-38A and B to achieve proper system alignment. These valves are located in the decay heat pump pits with reach rod extensions for operation from elevation 281’.

A dose rate in Area XI of 7,000 rem/hour at T=0 emanates from Decay Heat Removal System piping associated with the crossover lines from the decay heat coolers to the makeup pumps via valves DHV-7A and B, the piping legs back to MUV-14A and B, and the radiation sources located in Areas XIII and XIV.

A dose rate in Area XII of $2.0x10^4$ rem/hour at T=0 results from decay heat piping associated with injection from the decay heat pumps via DHV-4A and B, reactor building sump crossover via DHV-12A, the piping leg back to DHV-14 A and B, and the decay heat drop line to DHV-12A and B via DHV-3. The occupancy time to realign DHV-38A and B is five minutes, and if performed at T=0, the dose to the operator would be 580 rem for Area XI and 1700 rem for Area XII.

For post-accident conditions, low pressure injection will be initiated using the Borated Water Storage Tank as the suction source. Since the BWST water has very low levels of radioactivity, radiation levels at the remote manual operation point for DHV-38A and B would not increase to unacceptable levels until the suction of the decay heat and building spray pumps are switched to the reactor building sump. During operation via the BWST, DHV-38A and B would be opened, if required, with minimal dose, in accordance with emergency operating procedures.

The estimated radiation dose to travel to and from Areas XI and Area XII is negligible when compared to the dose received during area occupancy.

5) Areas XIII and XIV

Areas XIII and XIV, as identified on Figure 11.A-2, are the general locations of decay heat valves DHV-12A and B and DHV-64 on the 281’ elevation of the Auxiliary Building. DHV-12A and B are manual valves of which DHV-12A is locked closed and DHV-12B is locked open during power operation. Access to unlock and open DHV-64 is required for boron precipitation control. DHV-64 has had a remote valve operator installed and valve operation is accomplished on the 305’ elevation of the Auxiliary Building. Access to DHV-12A and B is not required for boron precipitation control, since DHV-12B is locked open.

A radiation dose rate of 1.7×10^4 rem/hour at T=0 in Area XIV emanates from decay heat removal piping associated with injection from the decay heat removal pumps via valves DHV-4A and B, reactor building sump crossover via DHV-12A, and the decay heat drop line to DHV-12A and B via DHV-3.

A radiation dose rate of 2.1×10^4 rem/hour at T=0 in Area XIII emanates from the same piping sources as for Area XIV.

The estimated occupancy time to unlock and open/close DHV-12A or B is five minutes. The resulting doses to the operator at T=0 is 1,750 rem for DHV-12A and 1,400 rem for DHV-12B. The estimated radiation dose for travel to and from Areas XIII and XIV via Areas VI, VII, and VIII is 165 rem.

For long-term boron precipitation control, neither DHV-12A or DHV-12B has to be operated post-accident. Leaving DHV-12B locked open during power operation, in conjunction with opening DHV-64 prior to low pressure injection from the reactor building sump, will ensure an active means of boron precipitation control is available. Thus, the need for personnel access into these areas is eliminated.

6) Area XV

Area XV, located on the 281 feet elevation of the Auxiliary Building as shown on Figure 11.A-2, contains the engineered safeguards motor control center (MCC) 1C.

A radiation dose rate of 25 rem/hour at T=0 in this area comes from the unshielded reactor coolant sample recirculation line routed above MCC 1C. The rate conservatively assumes a pure, undiluted reactor coolant sample is drawn at T=0.

Access to Area XV may be required after an accident to reset thrown circuit breakers in MCC 1C. Occupancy time for resetting circuit breakers is estimated to be 2 minutes resulting in a radiation dose of 0.83 rem to the operator.

7) Area XVI

Area XVI comprises two associated plant functions: the Nuclear Sampling Room (Area XVIA) and the Radiochemistry Laboratory (Area XVIB) on the 306' elevation of the Control Building as shown on Figure 11.A-4.

The radiation sources in these areas are due to the reactor coolant sampling and recirculation lines running into the Nuclear Sampling Room, which in an unshielded configuration, would result in a general area dose rate of 276 rem/hour if sampling pure, undiluted reactor coolant at T=0 (reference 12). Direct radiation from the reactor building contributes only 0.5 mrem/hour to the area dose rates.

Lead shielding has been installed around the reactor coolant sample lines in the Nuclear Sampling Room, effectively reducing the dose rate in expected occupied areas to 104 rem/hour (reference 14). Post Accident Sampling methodology is detailed in Section 9.2.2 of this FSAR.

8) Area XVIII

Areas XVIIIA and XVIIIB are located in the Intermediate Building as shown on Figures 11.A-7 and 11.A-8. These areas are the locations of the emergency feedwater pumps and associated piping and components. Postaccident access to these areas may be desirable for inspection and maintenance of equipment.

A dose rate at T=0 of 0.5 mrem/hour in Area XVIII results from direct radiation from the Containment Building.

9) Deleted

10) Areas XVII, XXI, and XXII

Area XVII is the Operations Support Center, shown on Figure 11.A-4. Area XXII contains the Control Room and Technical Support Center, both shown on Figure 11.A-6. Area XXI is the Backup Technical Support Center, as shown on Figure 11.A-5.

The Control Room and the Technical Support Center are on the 355 feet elevation of the Control Building. The Backup Technical Support Center is located on the 322 feet elevation of the Control Building. The Operations Support Center is located in the Health Physics Laboratory on the 306 feet elevation of the Control Building.

The radiation dose rates for these areas come primarily from the reactor coolant sampling and recirculation lines running into the Nuclear Sampling Room (Area XVIA on Figure 11.A-4) and secondarily from direct radiation from the Containment Building. The reactor coolant sampling and recirculation source is conservatively estimated to contain the reactor coolant source at T=O. The Containment Building direct radiation source is less than 0.5 mrem/hour.

At T=0, the radiation dose rates in the Backup Technical Support Center (Area XXI), the Control Room/Technical Support Center (Area XXII), and the Operations Support Center (Area XVIIB) and monitor area (Area XVIIA) are 480, 1.7, 18, and 340 mrem/hour, respectively. The radiation dose rate in these areas would be negligible if reactor coolant sampling was not assumed at T=O.

11) Area XXIV

Area XXIV is the Diesel Generator Building, as shown on Figure 11.A-9. Access to Area XXIV is required postaccident for operation/maintenance of the diesel generators.

The radiation dose rate in Area XXIV comes from the Containment Building. At T=0 it is less than 10 mrem/hour. The estimated dose accumulated during travel from the Control Building through the Turbine Building to Area XXIV is negligible.

12) Security Access Center

The security access center is on a direct line of sight with the containment at a distance at about 285 feet. The postaccident dose rate at this location is 750 mr/hr at T=0. It will take approximately 10 days for the dose rate to decay to 15 mr/hr. The dose rate average over 30 days is less than 15 mr/hr.

b. Areas for Which Postaccident Access is Not Required

The remaining areas of Figures 11.A-1 through 11.A-10 are those for which postaccident access is not necessary. Access to these areas is not needed because:

1) These areas contain equipment and piping necessary for accident mitigation but do not require personnel access, or
2) These areas do not contain any components for which postaccident operator access is necessary or desirable.

These areas are IX - Makeup pump components, XIX - Leak rate air dryer area and piping components, and XXIII - Decay Heat pits.

11.A.7 REVIEW OF RADIATION EFFECTS ON ELECTRICAL EQUIPMENT

The review of safety related (Class 1E) electrical equipment has been performed and is detailed in Appendix 6B of this FSAR.

11.A.8 CONCLUSIONS OF POSTACCIDENT SHIELDING EVALUATION

a. The TMI-1 plant shielding has areas where postaccident radiation will preclude operator access. A summary of modifications and administrative controls for postaccident access is given in Table 11.A-3.

Table 11.A-2 lists the plant areas evaluated and identifies the corresponding dose rates.

Areas where calculated dose rates and operator doses preclude postaccident access without appropriate modifications and/or administrative controls are:

Areas III and IV - Makeup and Purification System seal injection filter bypass valve MUV-198 and engineered safeguards MCCs 1A and 1B.

Areas XI and XII - Decay Heat Removal System valves DHV-38A and B for boron precipitation control.

Areas XIII and XIV - Decay Heat Removal System valves DHV-12A and B and DHV-64 for boron precipitation control and system operation.

b. Dose rates in the following areas requiring continuous occupancy exceed 15 mrem/hour at T=0 but the dose rates averaged over 30 days are less than 15 mrem/hour:

Area XVIIB - Operations Support Center

Area XXI - Backup Technical Support Center

\- Security Access Center

c. The calculated dose rates for the remaining areas of the plant where postaccident access might be desirable are low enough to allow access for the expected period of occupancy.

11.A.9 REFERENCES

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2. Letter from H. R. Denton (NRC) to All Operating Nuclear Power Plants, Subject: Discussion of Lessons Short-Term Requirements, dated October 30, 1979.

3. Letter from D.G. Eisenhut (NRC) to All Licensees of Operating Plants and Applicants for Operating Licenses and Holders of Construction Permits, Subject: Preliminary Clarification of TMI Action Plan Requirements, dated September 5, 1980.

4. Clarification of TMI Action Plan Requirements, Office of Nuclear Reactor Regulations, Division of Licensing, U.S. Nuclear Regulatory Commission, NUREG-0737, Section II.B.2, October 1980.

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

6. A. Tobias, "Data for the Calculation of Gamma Radiation Spectra and Beta Heating from Fission Products (Rev. 3)." RD/B/M2669, CNDC (73) P4, Central Electricity Generation Board, Research Dept., Berkeley Nuclear Laboratories, United Kingdom, June (1973).

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

8. Reactor Shielding Design Manual, ed. by T. Rockwell, Technical Information Service, Dept. of Commerce, Washington, D.C. (1956).

9. H. Oho and A. Tsuruo, "An Approximate Calculation Method of Flux for Spherical and Cylindrical Sources with a Slab Shield".

10. K. Shure and O. J. Wallace, "Compact Tables of Functions for Use in Shielding Calculations". Nuclear Science and Engineering, 56 pp. 89-94, January 1975.

11. GPUN Technical Data Report 121, “Design Review of Plant Shielding and Radiation Qualification for Post Accident Operations Outside Containment.”

12. GPUN Technical Data Report 183, “Shielding and Exposure Study for Post Accident Sampling.”

13. GPUN Technical Data Report 494, “TMI-1 Post-Accident Sampling Radiological Analysis.”

14. GPUN Calculation 6612-92-024, “Primary System Upgrade MM-9227.”

15. GPUN Engineering Change Memo #242, “Post-Accident Shielding.”

16. Letter from J. F. Stoltz (NRC) to TMI-1: Safety Evaluation by the Office of Nuclear Regulation Supporting NUREG-0737 Item II-B.2, “Plant Shielding,” dated December 26, 1984.

17. GPUN Calculation C-1101-211-5300-030, “MUF-4 Radiological Effect.”

18. GPUN Calculation C-1101-211-5300-031, “MUF-4 Modification, Radiological Effect-Area 15.”

Table 11.A-1
(Sheet 1 of 1)

ISOTOPIC CORE INVENTORY

Isotope	Total Core Inventory (Ci)	Isotope	Total Core Inventory (Ci)
Br 84	1.57×10^{7}	Xe 133	1.27×10^{8}
Br 85*	2.19×10^{7}	Xe 135m	3.26×10^{7}
Kr 83m	9.25×10^{6}	Xe 135	2.09×10^{7}
Kr 85m	2.19×10^{7}	Xe 138	1.17×10^{8}
Kr 85	5.30×10^{5}	I 129*	1.80×10^{0}
Kr 87	4.00×10^{7}	I 131	7.35×10^{7}
Kr 88	5.60×10^{7}	I 132	8.62×10^{7}
Sr 89	7.42×10^{7}	I 134	1.60×10^{8}
Sr 90	3.99×10^{6}	I 135	1.27×10^{8}
Sr 91	9.72×10^{7}	Cs 134	1.27×10^{6}
Sr 92	9.50×10^{7}	Cs 136	8.02×10^{5}
Y 90	3.96×10^{6}	Cs 137	4.99×10^{6}
Y 91	9.85×10^{7}	Cs 138	1.23×10^{8}
Mo 99	1.28×10^{8}	Ba 137m	4.67×10^{6}
Ru 106	2.29×10^{7}	Ba 140	1.25×10^{8}
Xe 131m	4.38×10^{5}	La 140	1.27×10^{8}
Xe 133m	3.07×10^{6}	Ce 144	7.50×10^{7}

* Deleted as insignificant for subsequent calculations.

Table 11.A-2
(Sheet 1 of 1)

RADIATION DOSE RATE BY AREA

Area	Figure Reference	Dose Rate at T=0 (mrem/hr)
I	11.A.10	3.0×10^{1}
II	11.A-3	1.1×10^{3}
III	11.A-3	1.9×10^{5}
IV	11.A-3	1.2×10^{7}
V	11.A-10	1.3×10^{4}
VI	11.A-2	$1.5 \text{ X } 10^{4}$
VII	11.A-2	8.8×10^{4}
VIII	11.A-2	2.1×10^{6}
IX	11.A-2	1.9×10^{6}
X	11.A-2	1.7×10^{1}
XI	11.A-2	7.0×10^{6}
XII	11.A-2	2.0×10^{7}
XIII	11.A-2	2.1×10^{7}
XIV	11.A-2	1.7×10^{7}
XV	11.A-2	2.5×10^{4}
XVIA/B	11.A-4	1.0×10^{5}
XVIIA	11.A-4	3.4×10^{2}
XVIIB	11.A-4	1.8×10^{1}
XVIII	11.A-8	0.5×10^{0}
XIX	11.A-8	6.1×10^{3}
XX	11.A-8	1.4×10^{5}
XXI	11.A-5	4.8×10^{2}
XXII	11.A-6	1.7×10^{0}
XXIII	11.A-1	4.0×10^{7}
XXIV	11.A-9	1.0×10^{1}

TABLE 11.A-3
(Sheet 1 of 4)

POST-TMI-2 MODIFICATIONS FOR POSTACCIDENT ACCESS

Required Action	Location	Dose
Manually open valve MUV-198 to bypass seal injection filters	305-ft elevation of the Auxiliary Building, Area IV, Figure 11.A-3	1000 rem for valve operation, negligible for travel
Manually close valve CAV-371 to isolate flow path thru CAV-446	305-ft elevation of the Auxiliary Building, Area IV, Figure 11.A-3	1000 rem for valve operation, negligible for travel
Reset any thrown circuit breakers in motor control centers 1A, 1B	305-ft elevation of the Auxiliary Building, Area III, Figure 11.A-3	110 rem for circuit breaker reset operation, negligible for travel
Manually operate DH-V38A and B for boron precipitation control	281-ft elevation of the Auxiliary Building, Areas XI and XII, Figure 11.A-2	580 rem in Area XI, 1700 rem in Area XII, negligible for travel

TABLE 11.A-3
(Sheet 2 of 4)

POST-TMI-2 MODIFICATIONS FOR POSTACCIDENT ACCESS

Major Source of Radiation Dose	Modification and/or Administrative Control
Reactor coolant pump seal injection piping, filters and valves; high pressure injection piping to cold legs "C" and "D"	Emergency operating procedures have been changed to require manual opening of MUV-198 prior to initiating recirculation from reactor building sump (i.e., before the BWST is depleted).
Reactor coolant pump seal injection piping, filters and valves; high pressure injection piping to cold legs "C" and "D"	Emergency operating procedures have been changed to require manual closing of CAV-371 prior to initiating recirculation from reactor building sump (i.e., before the BWST is depleted).
Reactor coolant pump seal injection piping, filters and valves; high-pressure injection piping to cold legs "C" and "D"	A shield wall has been constructed in Area IV, to shield the MCCs from the radiation sources in Area IV.
Makeup and purification system and decay heat removal system piping	Emergency operating procedures have been changed to open DHV-38A or B as necessary to accommodate single failure criteria.

TABLE 11.A-3
(Sheet 3 of 4)

POST-TMI-2 MODIFICATIONS FOR POSTACCIDENT ACCESS

Required Action	Location	Dose
Unlock and open valves DHV-12A & B and DHV-64 for boron precipi-tation control.	281-ft elevation of the Auxiliary Building, Areas XIII and XIV, Figure 11.A-2	1750 rem for Area XIII, 1400 rem for Area XIV, 165 rem for travel
Sampling and analyzing reactor coolant in nuclear sampling room and radiochemistry laboratory	306-ft elevation of the Control Building, Area XVI, Figure 11.A-4	Unshielded dose rates in the Nuclear Sampling Room are 276 rem/hour.

TABLE 11.A-3
(Sheet 4 of 4)

POST-TMI-2 MODIFICATIONS FOR POSTACCIDENT ACCESS

Major Source of Radiation Dose	Modification and/or Administrative Control
Decay heat system piping	Operating procedures require DHV-12B to be locked open during power operations. A valve operator reach rod extension has been installed on DHV-64, with the handwheel positioned to benefit from the shield wall installed in Area IV on the 305' Auxiliary Building. Emergency operating procedures have been changed to require manual opening of DHV-64 prior to initiating recirculation from the reactor building sump (i.e. before the BWST is depleted).
Reactor coolant sample and sample lines	Shielding has been installed in the Nuclear Sampling Room to reduce dose to Chemists performing post-accident reactor coolant sampling. Refer to Section 9.2.2 of this FSAR.