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US Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, DC 20555

Dear Sirs:

Three Mile Island Nuclear Station, Unit 2 (TMI-2)
Operating License No. DPR-73
Docket No. 50-320
Environmental Evaluation for TMI-2 Post-Defueling Monitored Storage

Attached for your use, in developing a supplement to the Programmatic Environmental Impact Statement Related to Decontamination and Disposal of Radioactive Wastes From March 28, 1979 Accident, Three Mile Island Nuclear Station Unit 2 (PEIS), is the GPU Nuclear Environmental Evaluation of TMI-2 Post-Defueling Monitored Storage (PDMS).

This document evaluates the potential impact of PDMS on the public health and safety and estimates occupational doses during PDMS.

Based on this evaluation, GPU Nuclear concludes that:

1. The analysis of potential off-site radiological consequences resulting from routine and accident conditions during PDMS demonstrates that cleanup operations will have progressed to a point where any threat to public health and safety has been eliminated.

2. Off-site consequence estimates for PDMS confirm the PEIS conclusion that the potential environmental consequences are dominated by occupational exposures.

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3. Occupational exposures associated with the recovery operation are expected to be substantially lower than the revised estimates of the PEIS, and are likely to fall within the range of the original estimates.

4. PDMS maintenance and monitoring activities will result in no more than insignificant increases in total recovery occupational exposure and future occupational exposure associated with activities related to recommissioning or decommissioning can be expected to be reduced as a result of natural decay and potential improvements in decontamination technology.

Sincerely,

F. R. Standerfer
Director, TMI-2

FRS/JJB/eml

Attachment

cc: Regional Administrator - Region 1, Dr. T. E. Murley
Director - TMI-2 Cleanup Project Directorate, Dr. W. D. Travers

ENVIRONMENTAL EVALUATION

OF

TMI-2 POST-DEFUELING MONITORED STORAGE

GPU NUCLEAR CORPORATION

THREE MILE ISLAND - UNIT 2

MARCH 1987

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1.0 INTRODUCTION

The TMI-2 Cleanup Program has as its objective the elimination of the radiological hazards to the public resulting from the March 29, 1979, accident and minimization of on-site worker exposure. The program has progressed from the initial efforts to stabilize the plant conditions through major cleanup steps, including the processing of the accident water and decontamination of major portions of the Auxiliary and Fuel Handling Buildings (AFHB) and Reactor Building (RB). Currently, efforts to remove the fuel from the reactor vessel and decontaminate the lower levels of the RB are well underway. Following completion of these steps but prior to the final activities necessary to either recommission or decommission the plant, GPU Nuclear intends to maintain the plant in the Post-Defueling Monitored Storage (PDMS) mode. This concept is described in Reference 1.

The environmental impacts of the cleanup program have been examined in detail in the Programmatic Environmental Impact Statement (PEIS) (Reference 2). The PEIS projected the most significant environmental impact of the recovery operations to arise from the radiation dose received by the work force and concluded that effluents from the plant would not significantly contribute to the environmental impact. Several alternative sequences of recovery operations are addressed in the PEIS and its supplements. Although a number of similarities exist between one of the alternatives considered in Supplement 1 (Reference 3), as described in the next section, none of the alternatives provide an exact match to the conditions and work sequence proposed in the PDMS concept.

PDMS represents an intermediate period following completion of the cleanup and establishment of safe, stable, and secure plant conditions but prior to effecting the final disposition of the plant. This additional time period could influence the overall environmental impact of the recovery operations. A positive influence of this monitored storage period arises from the natural decay of any remaining radionuclides, which will lower occupational exposures during the final phase of work. In addition, projected but presently intangible benefits (e.g., improvements in robotics or decommissioning technology) could have a beneficial impact on the cost, as measured in dollars and person-rem, of the overall recovery program.

On the other hand, the potential for routine releases over an extended period of time could result in an increase in total plant effluents. Also, the need to monitor and maintain the plant during this time period will result in some occupational exposure.

The purpose of the discussion presented herein is to examine the effect of the implementation of the PDMS concept on the overall environmental impact of the recovery. The potential benefits mentioned above are considered in a qualitative manner only. However, potential negative impacts are quantified in order to demonstrate that the total environmental impact of implementing PDMS is well within the bounds estimated for the recovery program in the PEIS. Some of the specific details discussed in this evaluation concerning expected plant conditions during PDMS may vary based on further recovery activities; however, any potential off-site environmental impact will be bounded by the evaluation provided herein.

2.0 COMPARISON WITH PEIS ALTERNATIVES

Supplement 1 to the PEIS (Reference 3) describes three (3) alternatives to the recovery program plan examined in the PEIS. The base case recovery program described in the PEIS consists of AFHB and RB dose reduction, followed by defueling and subsequent final decontamination efforts, including primary system decontamination.

The alternatives considered include: 1) postponement of defueling until a more extensive cleanup of the RB is achieved; 2) "phased defueling," including the removal of fuel fines through the reactor vessel head prior to head lift; and 3) defueling followed by a delay of RB cleanup pending development of robotic decontamination technology.

The third alternative is labeled "monitored, interim storage" in the PEIS supplement. It bears considered resemblance to the PDMS in concept as well as in name. Alternative 3 envisions completion of most of the AFHB decontamination and sufficient RB dose reduction to permit efficient defueling of the reactor. These activities would be followed by placing "the reactor and containment building in interim, monitored storage, and (performing) final building cleanup using robotics sometime in the future, when appropriate technology and devices become available" (Reference 3, page 2-29). The future robotic cleanup of the RB is assumed to involve "equipment removal, building and equipment decontamination, shielding removal and decontamination, and buildings surveys... The tasks would be performed without workers routinely being in the reactor building" (Reference 3, page 30-31).

The major differences between Alternative 3 and the proposed PDMS concept lies in the degree of cleanup accomplished prior to placing the facility into monitored storage. While Alternative 3 implies little or no cleanup of the RB prior to storage (other than that required for defueling), the PDMS conditions include RB cleanup sufficient to ensure elimination of any threat to the health and safety of the public. This includes the achievement of radiological conditions comparable to those typical for operating nuclear power plants in nearly all regions of the Turbine Building and the AFHB. Those regions of the RB which require routine access will be accessible with routine radiological precautions (e.g., protective clothing). The lower interior of the D-rings and certain areas on the 282' level will be the major areas in the RB to which access cannot be gained utilizing standard radiological protective measures. Nevertheless, sufficient cleanup of the 282' level is planned to eliminate sources of loose contamination which contain sufficient radionuclide inventory to present a threat to public health and safety. In addition, reduction of radiation fields will be adequate to permit limited personnel access to selected areas of the basement if the need should arise.

PDMS conditions in the AFHB will allow routine access to all areas except a few sealed-off cubicles (e.g., the seal injection valve room and the makeup and purification demineralizer cubicles) to which access is not required.

To date, considerable progress has been made towards achieving the radiological conditions anticipated for PDMS. Section 4.1 describes the current experience with respect to occupational exposure under conditions approaching the conditions expected after completion of cleanup activities. This experience indicates that exposures resulting from PDMS activities will be a small fraction of the total occupational exposure resulting from TMI-2 Cleanup activities. It is likely that the reductions in occupational exposure experienced during final plant disposition, arising from natural decay and the exploitation of improved technology for future in-plant work, would far exceed the minor increase resulting from PDMS activities.

Assuming postponement of the cleanup of major portions of the AFHB and the RB, Reference 3 concluded that the safety of Alternative 3 requires additional evaluation when the radionuclide inventories remaining during the storage phase become known. Analyses presented herein demonstrate that the radionuclide inventories remaining during PDMS are sufficiently reduced to eliminate any threat to the health and safety of the public.

The PDMS conditions are sufficiently similar to Alternative 3 to benefit from the potential occupational dose reductions described in the PEIS Supplement while differing from the Alternative 3 in the degree of RB cleanup.

Sufficient cleanup will be completed prior to PDMS to ensure the protection of the health and safety of the public.

3.0 IMPACT ON PUBLIC HEALTH AND SAFETY

The potential for release of significant quantities of radionuclides during PDMS is substantially reduced from that during normal operation, or any of the post-accident recovery phases, as a result of the reduced radionuclide inventory and the absence of inherent driving forces for transport processes. The assessment of any impact PDMS on public safety, therefore, hinges on identification of processes or events that could either alter the potential for transport of the remaining radionuclide inventory or provide unanticipated transport mechanisms to the environment. The approach taken herein is to review a range of potential unanticipated events to establish the bounding conditions of potential off-site releases. The radiological consequences associated with routine releases, as well as the bounding conditions, are estimated in the subsequent sections.

3.1 Routine Releases

Atmospheric releases to the environment during routine PDMS operations will be limited to any airborne contamination releases as a result of intermittent operation of the AFHB and the RB ventilation systems or other filtered monitored pathways. Both passive and forced ventilation discharges will involve controlled, HEPA-filtered, and monitored paths.

Liquid systems, except for systems needed to occasionally process batches of contaminated liquids, will have been drained and deactivated prior to PDMS. The major sources of such liquids are expected to be groundwater inleakage, collected precipitation, and occasional small quantities of fluids used for local groundwater. Rainwater and groundwater inleakage is anticipated primarily at the interface (i.e., the cork seal) between the AFHB and the RB. Such inleakage will be collected and analyzed for any contamination. The capability to process this liquid will be maintained to ensure that discharges are well within regulatory requirements.

3.2 Accidental Releases

Unanticipated events that could result in radionuclide releases exceeding those during normal conditions can be grouped into internally and externally initiated events.

Internally generated events include those unanticipated occurrences arising from the conditions or activities postulated during PDMS. Since there are no major activities planned for PDMS, it is difficult to envision an accidental event involving a major fraction of the remaining inventory of radionuclides. Although no specific mechanisms could be identified, it was assumed that a fire in the AFHB or the RB could involve a major fraction of any loose surface activity in the building. Thus, the non-mechanistic fire scenario was used to establish bounding conditions for the evaluation of potential off-site consequences of internally generated events. A number of other events that could be postulated, based on the types of activities considered within the scope of PDMS, were reviewed to verify that such events would not exceed the bounding conditions. Minor "industrial accidents" could be postulated, including exposures during radiation surveys, leaks or sprays during wet vacuum operation, mishaps during local decontamination operation, accidental cutting of contaminated piping, and fires or leaks during occasional dry and liquid waste processing. Events of this type were analyzed with respect to the anticipated off-site radiological impact in a generic study of a pressurized water reactor (PWR) decommissioning following an accident (Reference 4). The releases estimated as a result of this analysis are shown in Table 1. Each of these events involves only a minor, local source of radionuclides. Hence the total releases are quite small.

TABLE 1

RELEASES FROM INDUSTRIAL ACCIDENTS DURING POST-ACCIDENT PREPARATIONS FOR SAFE STORAGE

(From Reference 4)

Accident Event	Release to Atmosphere (uCi)
Gross Leak During In-Situ	
Chemical Contamination:	
Spray Leak	1.1×10^{2}
Liquid Leak	2.5×10^{-1}
Vacuum Bag Rupture	5.0×10^{0}
Accidental Spraying of Concentrated Contamination With High Pressure Spray	6.0×10^{-1}
Accidental Cutting of Contaminated Pipe	1.8×10^{-1}
Accidental Break of Contaminated Piping During Inspection	1.1×10^{-1}
Fire Involving Contaminated Clothing or Combustible Waste	3.0×10^{-2}
Fire in Contaminated Sweeping Compound	3.8×10^{-4}

The largest release listed in Table 1 is estimated to occur from a leak of chemical decontamination solution which results in an aerosol consisting of concentrated contaminants. This event is not directly applicable to TMI-2, since chemical decontamination methods are not anticipated during PDMS. An equivalent event for TMI-2 would be water spills associated with local decontamination efforts. Such events would not result in significant liquid or airborne source terms.

The external events for the TMI-2 site have been tabulated and analyzed in detail in the Final Safety Analysis Report (FSAR), and include seismic events, floods, windstorms, and aircraft crashes. The FSAR (Reference 5) analyses for these events were reviewed to assess a potential to create conditions resulting in releases to the environment which would exceed the bounding values postulated for PDMS. Since the original plant design included provisions to protect against these external events, it can be assumed that this protection remains, except where specifically negated by modifications resulting from the accident and subsequent recovery operations.

The protective measures that continue to have the potential for direct impact on off-site releases during PDMS are structures and system boundaries that serve as barriers for 1) residual fuel or 2) significant quantities of radionuclides.

With regard to preventing in-plant mobility of residual fuel, systems originally classified as seismic category I will provide isolation and contain residual fuel with a single exception (i.e., the non-seismic portions of the waste disposal liquid (WDL) system). However, this system does not contain a sufficient quantity of fuel to constitute a criticality hazard. Further, the addition of the loose contamination from the WDL system to the building's loose contamination source in case of system failure represents an insignificant increment.

Flood protection is included in the original design bases. Although there are no plant systems remaining that required flood protection to avoid potential airborne releases, a liquid release could be postulated as a result of flooding. Such a postulated release is prevented by the dike, which remains intact and provides flood protection throughout the PDMS phase.

The barriers against airborne releases assumed in this assessment (i.e., AFHB and RB structures) are designed to withstand external events up to and including the safe shutdown earthquake (SSE), windstorms, and the impact from an aircraft crash (Reference 5). As there have been no structural modifications to these buildings, it can be assumed that they will continue to provide protection against these events. (Minor modifications (e.g., to containment penetrations) do not affect the structural integrity of the RB.)

The review of potential safety hazards summarized in the previous paragraphs indicates that the conditions for potential accident off-site releases can be bounded by the assumption of a major fire in the AFHB or the RB. Such an event, although no considered likely in view of the limited presence of combustibles and ignition mechanisms (e.g., many electrical circuits will be de-energized) in both buildings, would provide the largest driving force for the transport of radionuclides and was selected as the bounding condition for accidental releases presented herein.

3.3 Source Terms

The inventory of radionuclides remaining on-site during PDMS will be greatly reduced from that existing prior to the accident or during any of the phases of the recovery operations. This results from 1) removal of the fuel which represents the largest concentration of radionuclides, 2) processing and shipping radioactive waste, and 3) natural decay. The remaining radioactivity can be characterized as residual contamination located primarily in either closed piping systems that were drained but not aggressively decontaminated, or surface films closely adherent to equipment or structural surfaces. An exception is the RB basement (282' level).

The radioactivity in the RB basement is dominated by the block wall enclosing the stairwell and elevator. Radionuclides (primarily cesium and strontium) were absorbed into this concrete block structure during the period when the wall was (partially) submerged in the highly contaminated water collected in the RB basement during and following the accident. Since the radioactive material is embedded in the concrete, it is not readily available as a near term source for airborne release. Over longer periods of time, however, mechanisms related to diffusion and leaching by cyclic changes in moisture content may transport a fraction of the radionuclides in the block wall to the surface where it can become available for suspension. Even though this fraction is expected to be small, the large inventory of the block wall (i.e., an estimated 17,000 Ci of Cs and 8000 Ci of Sr) could make any suspension of radionuclides reaching the surface a significant airborne source term. For the purpose of this analysis, a fraction (i.e., conservatively estimated to be 3%) of the radionuclide inventory in the block wall, and to a lesser degree from other surfaces containing some residual contamination, is assumed to reach the surface and contribute to the inventory of "suspendable" contamination. This results in a total of about 1000 Ci of "suspendable" contamination in the vicinity of the block wall which, for the purpose of this analysis, is considered the largest single potential source for airborne contamination.

Since this potential source, as well as other potential sources (e.g., inside the D-rings and other difficult to access regions of the RB basement) for airborne radioactivity, has existed for a number of years, its effect on RB atmospheric particulate concentrations can be deduced from existing measurements. The derivation of a maximum airborne particulate level for use as a source term for routine releases during PDMS is given in Appendix A.

This source term is based on the radionuclide mix observed in the RB atmosphere to date, which consists primarily of long-lived cesium and strontium isotopes.

Samples of sump liquid and sediment in the RB basement available to date, however, indicate a higher fraction of strontium than contaminants in other areas of the RB. Analysis of core samples of the block wall and concrete in the RB basement indicate cesium to strontium ratios of near 1:1, or lower, from comparisons of β and γ emissions, and a ratio of approximately 2:1 for a sample analyzed by destructive assay. The conservative 1:1 Cs to Sr ratio was used for the assumed postulated accident airborne concentrations during PDMS.

Another important factor in the consideration of residual contamination is the content of transuranic elements. Although the quantity and configuration of fuel remaining after completion of defueling is insufficient to be of concern with respect to criticality, it is necessary to examine the potential contribution it could make to radiological source terms.

The core inventory of significant transuranic element (e.g., greater than 0.1% of the core transuranic inventory, on a curie basis) remaining after eight (8) years of decay as shown in Table 2. On the basis of the samples analyzed to date, as well as the analyses of the course of the accident, the transuranic elements can be assumed to be associated with residual fuel. For an eight (8) year decay time, Table 2 indicates a plutonium/americium mix of 1.23 E+5 Ci for the total of 98,000 kg of fuel, or 1.25 Ci per kg of fuel.

Most of the residual fuel remaining during PDMS will be fixed in the form of very thin surface films on reactor components or deposits in inaccessible locations of the reactor coolant system. Therefore, they will not contribute significantly to the airborne source term. During the accident, a small quantity of fuel fines were carried to the RB basement by reactor coolant escaping through the struck-open relief valve. About 1.7 to 3.2 kg of these fuel fines are estimated to be mixed with other solid materials in the sediment in the RB basement based on information available in mid-1986. Much of this sediment will be removed prior to PDMS. However, because of the difficult operating conditions in the RB basement, it is assumed that a fraction of the fuel fines in the sediment will remain after removal of the bulk of the sediment. As a reference point for the calculation of potential off-site consequences, it is assumed that no more than about 2 kg of fuel will remain as a suspendable source during PDMS.

Additionally, various other radionuclides will be present in the residual contamination; however, they are not specifically evaluated here as the most significant radionuclide contribution to the off-site dose impact results from the transuranics listed on Table 2 and Sr-90 and will bound consideration of these other radionuclides.

3.3.1 Airborne Releases

The method of suspension and fraction of the inventory contributing to the source term is a function of the scenario postulated. For routine releases, the airborne concentration was estimated from the most applicable observations of RB airborne concentrations to date (see Appendix A). It was assumed that the RB airborne concentrations will reach equilibrium between intermittent operation of the RB purge system. Each purge operation is assumed to result in the discharge of the entire contents of the RB atmosphere. A total of 50 discharges of RB atmosphere particulate content per year was conservatively assumed for the routine release contributions.

For the postulated bounding conditions (i.e., a fire affecting essentially all of the suspendable inventory in the vicinity of the block wall (e.g., 1000 Ci) and the 2 kg of "loose" fuel) an experimentally determined suspension fraction was used (Reference 8).

3.3.2 Liquid Releases

Inleakage of groundwater and precipitation are anticipated to be the major sources of liquids during PDMS. Such inleakage, which has occurred periodically (primarily at the juncture of the AFHB and the RB) is kept under control by periodic maintenance. Based on experience to date, and the anticipated lower frequency of maintenance during PDMS, an annual inleakage of 5000 gallons is estimated. To the extent that such inleakage becomes contaminated by any residual contamination on floors and in sumps, it will be processed before discharge. Data based on actual operational experience to date has shown that cesium and strontium concentrations of 4E-6 uCi/ml and 1E-5 uCi/ml can be achieved by the EPICOR processing system. This system or an equivalent processing system will be kept operational during PDMS to ensure water processing capability. The liquid effluent source term, therefore, is based on the assumption of a discharge of 5000 gallons annually at the above noted radionuclide concentrations. The source terms resulting from these considerations, as well as several of the controlling parameters, are summarized in Table 3.

3.4 Off-Site Dose Estimates

Environmental doses for normal releases of cesium and strontium were quantified with the MIDAS code. This is the model used to estimate doses for the TMI-2 quarterly report to the NRC.

This model calculates the dispersion of radionuclides in three (3) ways: normal airborne dispersion (X/Q), depleted dispersion (which accounts for various removal processed like settling-out), and deposition (D/Q).

The MIDAS calculations use the "delta T", or the difference in temperature between sensors at 33 feet and 150 feet, of the on-site meteorological tower to determine the atmospheric (Pasquill) stability class. The model then uses the stability class, other meteorological parameters (e.g., wind speed and direction) and plant parameters including stack height, stack diameter, and stack flow rate to determine the atmospheric dispersion.

The meteorological data used by MIDAS is automatically collected from the on-site tower and stored electronically for future use. Each meteorological tower sensor (there are about 20) is polled by a computer every ten (10) seconds. The ten (10) seconds results are averaged into a 15 minute average. The 15 minute averages centered on the hour are used as hourly values. The hourly values are used for routine dose calculations. The calculations reported herein were based on one (1) hour average meteorological data for the period January 1, 1986, to December 31, 1986. The calculation, therefore, is based on about 8600 different sets of meteorological conditions.

The accident dose calculations for the report were generated by selecting meteorological conditions which correspond to the worst case short-term dispersion characteristics. The TMI-2 FSAR X/Q value of 6.1E-4 sec/m^3 was used for this purpose.

TABLE 3

SUMMARY OF SOURCE TERMS [1]

ROUTINE RELEASES:	
Airborne Concentration	Cs-137 1×10^{-8} uCi/ml Sr-90 3×10^{-9} uCi/ml
Air changes in Reactor Building & AFHB	50 per year
Filter efficiency	99%
Particulate source term	Cs-137 2.8×10^{-4} Ci/year Sr-90 8.5×10^{-5} Ci/year
Transuranic source term [2]	2×10^{-8} Ci/year
ACCIDENTAL (PUFF) RELEASE:	
Suspension fraction	5×10^{-4}
Filter efficiency	99%
Fission product source term [3]	5×10^{-3} Ci
Transuranic source term [2]	1.2×10^{-5} Ci
LIQUID RELEASES:	
Quantity	5000 gal/year
Cs source term	7.6×10^{-5} Ci/year
Sr source term	1.9×10^{-4} Ci/year

1 - Other radionuclides will be present in the residual contamination, however, they are not specifically evaluated here as, based on previous evaluations, the most significant radionuclides for off-site dose impact are the transuranics and Sr-90.

2 - Isotopic composition is as described in Table 2.

3 - Isotopic composition assumes a 1:1 ratio of Cesium-137 to Strontium-90.

To calculate the dose to the public, MIDAS employs numerous data files which describe the area around TMI in terms of population distribution and foodstuffs production. The area around TMI is subdivided into 16 equal meteorological sectors (N, NNE, NE, ENE, E, ESE, SE, etc.). Each sector is then represented in each set of data files and dispersion is calculated for each sector separately. Data files include such information as the distance from the plant vent to the site boundary in each sector, population groupings, milk cows, milk goats, gardens of more than 500 square feet and meat animals.

MIDAS also contains dose conversion factors for 75 radionuclides for each of four (4) age groups (adults, teenagers, children, and infants), seven (7) pathways (inhalation, ground dispersion, plume direct dose, and ingestion of cow milk, goat milk, vegetables, and meats), and eight (8) organs (total body, thyroid, liver, skin, kidney, lung, bone, and gastrointestinal tract (GI)). The age groups, pathways, and organs are those specified in Regulatory Guide 1.109.

The atmospheric dispersion is combined with the dose conversion factors and applied to each age group, pathway, and organ to estimate the dose to an individual by integrating by sector, distance, and time (meteorology changes) to determine the dose, distance and direction to the maximum exposed individual.

The fission product source term described in the previous section were used to estimate maximum hypothetical individual and population doses with this methodology. The routine releases were averaged over one (1) year, and were assumed to be released at a constant rate. The releases were treated as ground level releases for purposes of estimating dispersion. This assumption results in conservative estimates of the degree of dispersion, compared to elevated release points. The assumption is appropriate for the plant stack because there will be essentially no exit velocity for the plume.

The results of this calculation, which includes the fission product and transuranic source terms, are summarized in Table 4. The total annual airborne pathway population dose is calculated to be 0.27 person-rem to the bone. The highest individual dose is estimated to occur at a

TABLE 4

ESTIMATED DOSE COMMITMENTS RESULTING FROM ROUTINE AND ABNORMAL CONDITIONS DURING PDMS

I. Population Doses	Person-rem/year
Population Doses From Routine Releases Via Airborne Pathway:	
Bone Dose	0.27
Total Body Dose	0.07
Population Doses From Routine Releases Via Liquid Pathway:	
Bone Dose	0.09
Total Body Dose	0.02
Total Population Dose:	
Bone Dose	0.36
Total Body Dose	0.09
II. Doses To Maximally Exposed Individual	mrem/yr
Dose From Routine Releases Via Airborne Pathway:	
Bone Dose	0.06
Total Body Dose	0.02
Dose From Routine Releases Via Liquid Pathway:	
Bone Dose	0.005
Total Body Dose	0.002
III. Maximum Accidental Release	
Dose to Maximally Exposed Individual From Accidental Release (bone dose limiting)	4 mrem

distance of about 500 meters. A child occupying this position for one (1) year would receive a maximum bone dose of 0.06 mrem.

The off-site doses from accidental releases of fission products and transuranics were also evaluated using the emergency dose calculation routine from MIDAS, which is used by the TMI Environmental Controls staff to calculate off-site doses for emergency planning purposes. The meteorological conditions for this case were chosen to produce a X/Q of 6.1E-4 which is normally used as the worst case accident X/Q for this type of evaluation. The accident calculations include only the inhalation dose pathway since it is expected that the other pathways (e.g., the ingestion pathways) can be restricted by protective actions under accident conditions. For this case, the maximum dose to an individual from fission products and transuranics is calculated to be 3.9 mrem to the bone.

Liquid pathway doses were also evaluated using MIDAS, based on the 94-year average Susquehanna River flow. MIDAS liquid dose calculations consider three (3) pathways: fish ingestion, water ingestion, and shoreline exposure. The four (4) age groups and eight (8) organs of Regulatory Guide 1.109 were also considered. Based on this evaluation, the maximum hypothetical dose to an individual would be 0.005 mrem to the bone of an adult, primarily through the freshwater fish ingestion pathway. Liquid population dose, which considers all users of Susquehanna River water downstream of TMI-2, is calculated to be 0.092 man-rem to the bone.

For comparison, the largest potential doses which could be postulated on the basis of the actual measurements of the Radiological Environmental Monitoring Program (REMP) for TMI (Units 1 and 2) are shown in Table 5 for several recent years. It should be noted that the dose estimates of this table are hypothetical doses which could be accrued to a maximally exposed individual on the basis of the measured effluents from TMI-2. The comparison of these dose estimates with the calculated values of Table 4 shows that actual airborne releases from TMI-2 during recent years of cleanup activities were significantly smaller than the projections made for PDMS. It is concluded, therefore, that the calculated values of Table 4 are conservative estimates of the likely impact of

TABLE 5

MAXIMUM POTENTIAL DOSES ESTIMATED FROM EFFLUENT MEASUREMENTS*

	1982	1983	1984	1985	NRC Limit**
Doses to Hypothetical Maximally Exposed Individual from Airborne Releases:					
Limiting Organ	0.005	0.011	0.00002	0.006	15.0
Total Body	6.0E-9	0.00008	2.0E-7	0.002	5.0
Doses to Hypothetical Maximally Exposed Individual from Liquid Release:					
Limiting Organ	0.82	0.02	0.56	0.004	10.0
Total Body	0.55	0.001	0.38	0.001	3.0

*Source: (Annual Environmental Monitoring Report for Three Mile Island Nuclear Station, Prepared by TMI Environmental Controls, GPU Nuclear Corporation, (Submitted to NRC on Docket 50-320).

**Source: 10 CFR 50 Appendix I, "Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion 'As Low As Reasonably Achievable' for Radioactive Material in Light-Water-Cooled Nuclear Power Reactor Effluents."

PDMS on the environment. Nevertheless, the estimated doses for PDMS are such small fractions of the normal background doses that they can be considered insignificant.

3.5 Site Environmental Monitoring Programs

GPU Nuclear maintains an extensive REMP for TMI. This comprehensive program is operated at a level which satisfies the Technical Specifications. The current TMI REMP, which used state-of-the-art equipment, will be able to monitor any radiation and radioactive materials in critical exposure pathways.

The program consists of collecting samples from the environment, analyzing them for radiation and radioactivity content, and interpreting the results. With emphasis on the critical pathways to man, samples from the aquatic, atmospheric, and terrestrial environments are collected. These samples include air, soil, water, finfish, milk, fruits, vegetables, groundwater, and precipitation. Thermoluminescent Dosimeters (TLDs) and a real-time gamma monitoring system are placed in the environment to measure ambient gamma radiation levels.

Sampling locations have been established by considering meteorology, population distribution, hydrology, and land use characteristics of the local TMI area. The sampling locations are divided into two (2) classes: indicator and control. Indicator locations are those which are expected to show plant effects, if any exist. These locations were primarily selected on the basis of where the highest predicted environmental concentrations would occur. While the indicator locations are typically within a few miles of the plant, the control stations are generally at least ten (10) miles from the plant. Therefore, control samples are collected at locations which should be unaffected by plant operations. They provided a basis on which to evaluate fluctuations at indicator locations relative to natural background radiation and radioactivity and fallout from prior nuclear weapons tests.

The analytical results are routinely reviewed and evaluated by a qualified staff of scientists. Investigations are conducted when administrative action levels have been reached or when anomalous values are discovered. Action levels have been set low enough to provide early detection of potential environmental impacts. Corrective action is initiated, if the investigation concludes that the radiation or radioactivity is related to TMI.

A strong quality assurance program is conducted in accordance with guidelines provided by the NRC and as required by the Technical Specifica- tions. It is documented by written policies, procedures, and records. This program is designed to identify possible deficiencies so that immediate corrective action can be taken. It also provides a measure of the quality of the results. The quality assurance program is implemented by:

- Auditing the analytical laboratories
- Requiring the analytical laboratories to participate in a NRC-approved quality assurance program (Interlaboratory Comparison Program)
- Requiring analytical laboratories to split and perform duplicate analyses on every tenth sample (recounts are performed when samples cannot be split)
- Splitting samples, having the samples analyzed by independent laboratories, and then comparing the results for agreement

Radiological Environment Operating Reports are submitted annually to the NRC for review. These reports include summaries, interpretations, and analyses of the results and discuss observed impacts, if any, to the environment from plant operations. The results of land use censuses and the Interlaboratory Comparison Program are also included. Special reports are prepared and submitted to the NRC in the event that a reporting level, as defined by the Technical Specifications, is exceeded.

The radionuclide inventories will be substantially reduced during PDMS and the mechanisms for transport and dispersal of residual activity will be essentially non-existent. Nonetheless, the REMP will remain fully operational, undergo continuous review, and, if necessary, will be modified in response to changing site or plant conditions which could impact the environment.

4.0 OCCUPATIONAL PERSONNEL DOSES DURING PDMS

The PEIS concluded that the most significant environmental impact associated with the recovery operations will result from the radiation doses received by the work force. The above analysis of the off-site consequences associated with normal and accidental releases supports that conclusion. The following discussion explores the potential impact of PDMS on occupational doses.

4.1 Current Status

Significant progress has been made in the decontamination of the AFHB and the RB. Although substantial additional effort is necessary in several areas to achieve the radiological conditions anticipated for PDMS, a brief review of the experience to date and current radiological conditions provides a perspective of the likely contribution of PDMS to the total occupational exposure associated with the recovery.

The annual accumulations of occupational doses since the accident are shown in Table 6. The sum of all individual doses for all workers involved in the recovery of the TMI-2 plant was 3517 rem through October 1986. This total, accumulated over seven (7) years, is less than the collective dose reported by some operating nuclear plants for a single year (Reference 11). In 1985, a total dose of 722 rem was received by all persons working at the TMI-2 plant notwithstanding significantly increases in-containment activity and commencement of core defueling. This dose also is less than the total doses reported by many operating plants. Plants which reported lower total doses tend to be small plants which did not have outage work during the year. Among plants undergoing refueling or repair outages, the TMI-2 plant occuptional does was among the lowest reported.

As shown in Table 6, total personnel exposures declined for the years 1979 through 1981. In 1979, considerable work was expended to stabilize the TMI-2 reactor and perform initial cleanup operations. There was no work in the RB from April 1979 through June 1980. Most of the

TABLE 6

THREE MILE ISLAND NUCLEAR STATION UNIT 2
SUMMARY OF ANNUAL WHOLE BODY EXPOSURE BY INCREMENT

		NUMBER OF INDIVIDUALS EXPOSURE INCREMENT - REM									
YEAR	TOTAL MONITORED	NOT MEASURABLE	0-1	1-2	2-3	3-4	4-5	>5	TOTAL[1] PERSON-REM	AVERAGE[2] MEASURABLE DOSE (REM)	HIGHEST INDIVIDUAL DOSE (REM)
1979	5695	3613	1963	94	17	5	3	0	488	0.23	4.5
1980	5909	4631	1249	28	1	0	0	0	193	0.15	2.1
1981	3115	2272	817	25	1	0	0	0	138	0.16	2.0
1982	1161	274	761	118	33	2	0	0	384	0.42	3.0
1983	1303	535	614	128	26	0	0	0	373	0.49	2.7
1984	1946	1274	472	138	44	18	0	0	514	0.77	3.7
1985	1793	546	961	184	72	30	0	0	722	0.58 (3)	3.5

NOTES:

1. Data shown are for the entire calendar year 1979. For the cleanup period, the 1979 total dose is 418 person-rem.

2. The average is calculated by dividing the total person-rem by the number of persons in the exposure increments 0-1, 1-2, 2-3, 3-4, and greater than 5.

3. In 1985, improvements in the GPU Nuclear radiation dosimetry system decreased the minimum reportable dose from 10 millirem (0.01 rem) per month to 1 millirem (0.001 rem) per month. A portion of reduction in the average measurable dose in 1985 compared to 1984 is attributable to this change.

radiological work performed in 1980 and 1981 was related to cleanup of contaminated water in tanks and collection sumps. The work activities in radiological areas increased steadily in the period 1982 through 1984. Major work to accomplish shielding and decontamination took place in 1982 and 1983. In 1984, the bulk of radiation exposures was related to defueling preparations including removal of the top of the reactor vessel in preparation to commence defueling.

In 1979 and 1980, almost all persons working at TMI-2 were monitored for exposure whether or not they entered radiologically controlled areas. After 1980, the number of persons monitored was substantially reduced since more than half were not entering radiologically controlled areas. In 1983 and 1984, the number of persons monitored increased. However, the number of persons with measurable doses remained stable. The increases corresponded to an increase in the work activities associated with cleanup activities.

The decontamination work completed to date has significantly reduced the radiation fields in most areas of the plant. Emphasis has been placed on decontamination and shielding in high traffic areas and those areas in which the largest fraction of job-hours are expended. For example, hallways and generally accessible areas of the Auxiliary Building are now accessible in street clothing. Figure 1 shows the radiation fields on the 347'-6" level of the RB. Average fields on this level have decreased from hundreds of mrem/hr at the beginning of recovery to less than 25 mrem/hr for most traveled areas in 1985.

Of particular interest is the dose rate on the defueling platform directly above the reactor vessel, since a high level of work activity will have to be sustained in this location to complete defueling of the reactor. Figure 2 shows the average defueling platform doses for the first six (6) months of defueling operations. Approximately 50,000 lbs. of core materials have been removed during this time period, which represents one-sixth of the total core to be removed

Not all areas of the plant have been cleaned up to such desirable low levels. The RB basement, in particular, continues to be inaccessible because of high radiation fields. Although these high radiation areas

FIGURE 1

ISO-EXPOSURE MAP OF ELEVATION 347'-6" FLOOR
(As of August 1985)

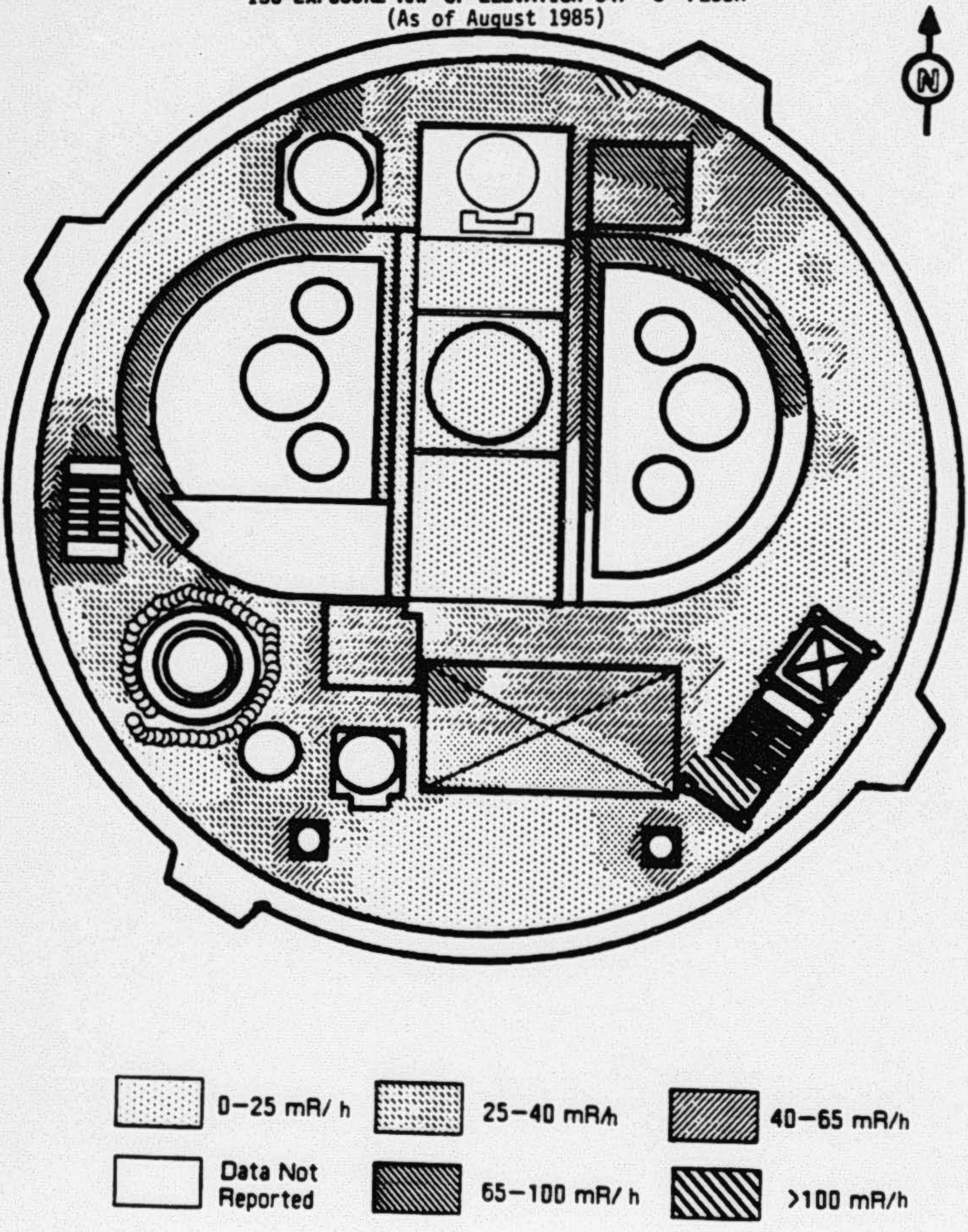
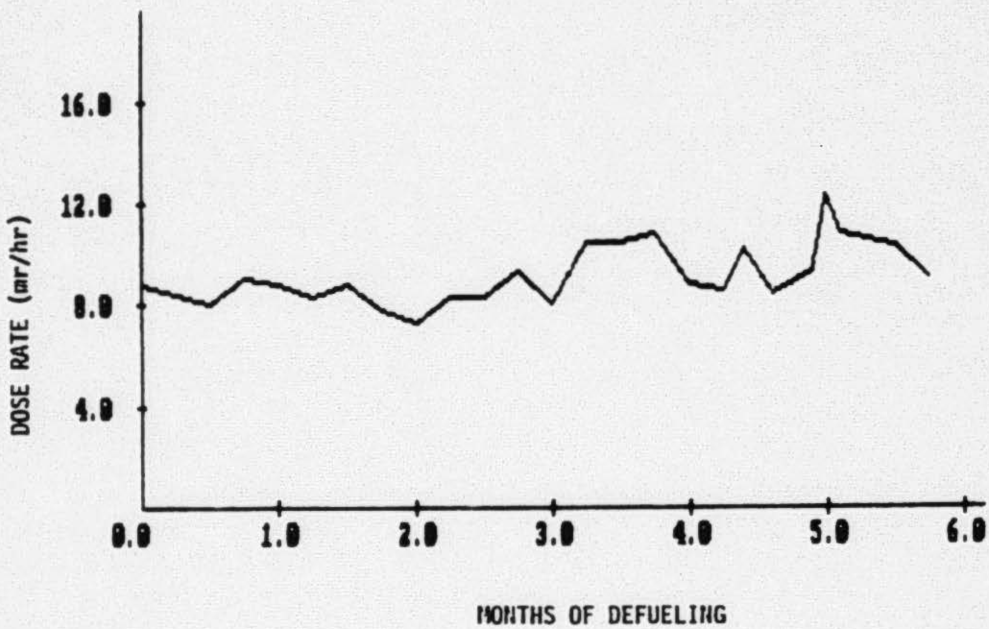


FIGURE 2

DEFUELING PLATFORM DOSE RATES
FOR FIRST SIX MONTHS OF DEFUELING



will not be cleaned up to levels comparable to routine access areas, a decontamination and dose mitigation program will be conducted prior to PDMS. This cleanup work will be performed remotely by specially designed robots; therefore, it is not expected to result in major increases in person-rem expenditure. Although no need for access to the 282' level of the RB has been identified, either prior to or during PDMS, the projected 282' level dose mitigation measures will enable limited access to selected areas of the basement, should the need arise as a result of unforeseen circumstances.

4.2 PDMS Occupational Dose Estimates

Personnel entries into the AFHB and the RB during PDMS will be limited. Under normal conditions entries will be required for inspections, technical specification surveillances, radiological surveys, radiological waste processing, remedial decontamination and some maintenance to support these activities as well as preventive maintenance for a limited number of operational systems. In addition to normal operations, entries may be required in order to respond to unforeseen circumstances.

Doses resulting from internal uptakes are expected to be minimal. It is planned that the ventilation system will be operated prior to and during the entries in order to purge the building air volume.

Airborne concentrations will be sampled remotely and upon initial entry in order to verify that the protection factor for the prescribed respiratory protective equipment will be adequate to maintain exposures less than one (1) MPC-hour/hour. Based on previous experience during the worst case conditions in the RB, it is expected that a protection factor of 1000 will be more than adequate for the anticipated PDMS conditions. Personnel will be monitored for internal uptakes in accordance with GPU Nuclear procedures to verify the adequacy of the respiratory protective equipment.

Reactor Building (RB)

Activities in the RB during PDMS will consist primarily of surveillance to assess the radiological and physical conditions of the building. These activities will be carried out from the floors at elevation 305' and 347'; access to the basement or inside of the D-rings will not be required. Under the scenario evaluated, entries will be made quarterly. (Note: During the initial period of PDMS, it is expected that entries will be made monthly.) Entry team stay times will be limited to the time necessary to carry out certain designated visual and radiological surveys in order to minimize doses.

The current experience with worker doses in the RB is summarized in Figure 3. On the basis of this experience, the exposure rate during PDMS can be expected to be less than 0.1 man-rem/man-hour. Based on this experience, the external personnel doses resulting from surveillance and monitoring activities in the RB are expected to be less than two (2) person-rem annually (less than six (6) person-rem annual during the initial period of PDMS). The sources of exposure will be due largely to residual contamination remaining in RB surfaces and equipment.

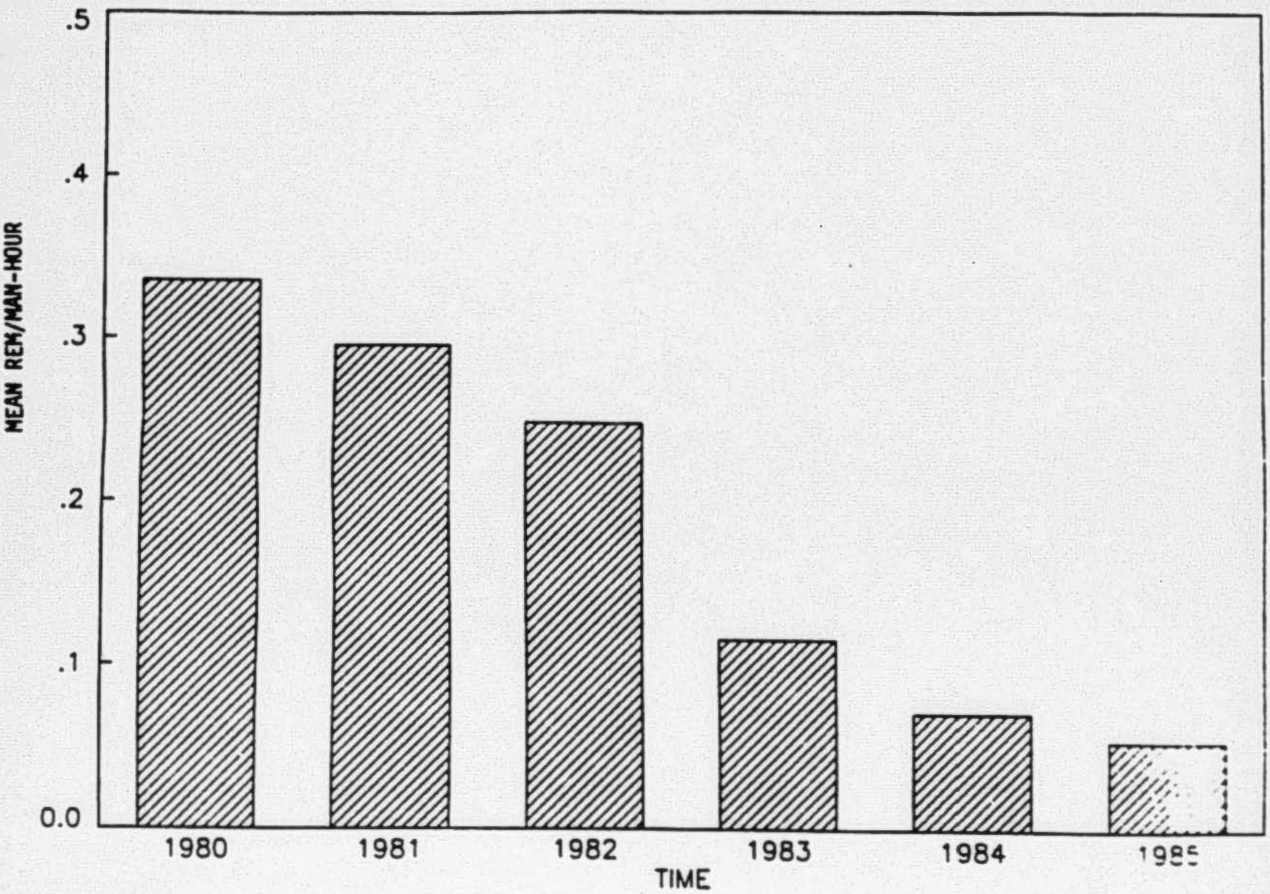
Auxiliary and Fuel Handling Buildings (AFHB)

Activities in the AFHB will consist primarily of surveillance, radiation monitoring, and occasional water processing. Access will also be required for the maintenance and operation of systems required to support PDMS (e.g., the ventilation, domestic water, fire suppression, waste disposal liquid, radiation monitors). These activities will be limited to a few cubicles and the AFHB corridors. Some continuing maintenance activities may include remedial decontamination, housekeeping, and response to natural events (e.g., installing flood doors). Most of these activities will take place in very low dose rate areas.

For the scenario described above, the annual collective dose is expected to be less than one (1) person-rem. Sources contributing to the external dose rate will be largely due to residual contamination in system piping, tanks, and components. In general, dose rates will be less than 2.5 mrem/hour.

FIGURE 3

TMI-2 REACTOR BUILDING
MEAN DOSE RATES



Doses resulting from internal uptakes are expected to be minimal. Most accessible areas will have loose surface contamination levels less than the limits established for unrestricted use. In the event that airborne radioactivity concentrations are anticipated, air sampling will be performed to determine the need for respiratory protection equipment. If required, respiratory protection equipment will be prescribed to keep exposures below an average of one (1) MPC-hour/hour. Personnel will be monitored in accordance with GPU Nuclear procedures to verify the adequacy of the respiratory protection equipment.

5.0 CONCLUSIONS

On the basis of the above analysis, the following has been concluded:

1. The analysis of potential off-site radiological consequences resulting from routine and accident conditions during PDMS demonstrates that cleanup operations will have progressed to a point where any threat to public health and safety has been eliminated.

2. Off-site consequence estimates for PDMS confirm the PEIS conclusion that the potential environmental consequences are dominated by occupational exposure.

3. Occupational exposures associated with the recovery operation are expected to be substantially lower than the revised estimates of the PEIS, and are likely to fall within the range of the original estimates.

4. PDMS maintenance and monitoring activities will result in no more than insignificant increases in total recovery occupational exposure while the future occupational exposure associated with recommissioning or decommissioning can be expected to be reduced as a result of natural decay and potential improvements in decontamination technology.

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APPENDIX A

AIRBORNE SOURCE TERMS DURING PDMS

At the conclusion of the ongoing Recovery Program, the primary source of radioactive material which could cause a significant off-site release (e.g., damaged fuel) will have been essentially removed. The remaining radioactivity can be characterized as residual loose contamination or fixed contamination which is either deeply embedded in solid materials (e.g., activation products in the reactor vessel and structural materials), or distributed in thin films adherent to surfaces which have been flushed but not aggressively decontaminated (e.g., by chemical solutions or mechanical surface removal). This remaining inventory of radionuclides, represents a potential airborne source.

Although the fixed contamination is not an issue in the near term, over the longer term a small fraction of this inventory may become available for suspension as a result of the aging of surface conditions (e.g., rusting of steel surfaces, chalking of paints, or flaking of concrete surfaces). The mechanisms of suspension of surface contamination resulting from these aging process are diffusion, diffusiophoresis, thermophoresis, air motion, and evaporation. While it is not practical to attempt quantification of each of these processes, the upper bound of their combined effect can be determined empirically from the observation of airborne contamination levels in the Reactor Building (RB).

The RB atmosphere has experienced conditions spanning a broad range of the variables affecting the formation of airborne contamination (i.e., temperature, humidity and concentration gradients, and changes in air flow) which readily encompass the conditions envisioned for PDMS. The most important factor affecting airborne contamination levels today is human activities associated with decontamination, plant modification, and defueling activities (e.g., foot traffic, vibrations from machinery, cutting, grinding, and welding). Since such activities will not take place during PDMS, RB air samples taken during the period prior to commencement of the now routine RB entry activities is likely to be more representative of PDMS conditions than the current RB atmospheric concentrations. The period immediately prior to, during and following the 1980 krypton purge, in particular, would approximate the effects of intermittent operation of the RB purge system separated by quiescent periods of no entries. Figure A-1 is a plot of the airborne particulate concentration during period.

During the krypton purge, the concentrations reached a low point of 1.2 E-10 uCu/cc for Cs-137 and 6.4 E-12 uCi/cc for Sr-90. Mixing the RB atmosphere during the last days of the purge (at an increased flowrate) increased this low point by nearly an order of magnitude between July 8 and 11. In the days following the purge, the Cs and Sr concentrations increased gradually to peak of 3.8 E-10 and 1.3 E-10 uCi/cc, respectively, on July 26, 1980. Subsequent airborne concentrations were affected by the beginning of RB entries, associated "mini-purges", and other human activities.

The rate of build-up of activity following purging can be estimated by the first-order rate equation.

$$\frac{dCi}{dt} = \frac{\Sigma Si}{V} - \Sigma\gamma i Ci - \frac{Q}{V} Ci$$

Where S_i is the net sum of all sources (uCi/day) of radionuclide i, $\Sigma\gamma i$ is the sum of all concentration-dependent removal rate constants and Q is the net exhaust flow of the ventilation (purge) system from the RB free volume (V). For the initial conditions of interest (i.e., negligibly small concentrations at the beginning and no purge flow during this period), this equation has the simple solution:

$$Ci(t) = \frac{\Sigma si}{\Sigma\gamma i} (1-e^{-\Sigma\gamma i t})$$

Where $si = \frac{Si}{V}$. The source and depletion parameters in this equation which approximate the observed behavior following the krypton purge are:

Σs = 5.0 E-11 (uCi/cc/day) for Cs-137
Σs = 1.5 E-11 (uCi/cc/day) for Sr-90, and
$\Sigma\gamma$ = .005 (day-1) for both isotopes

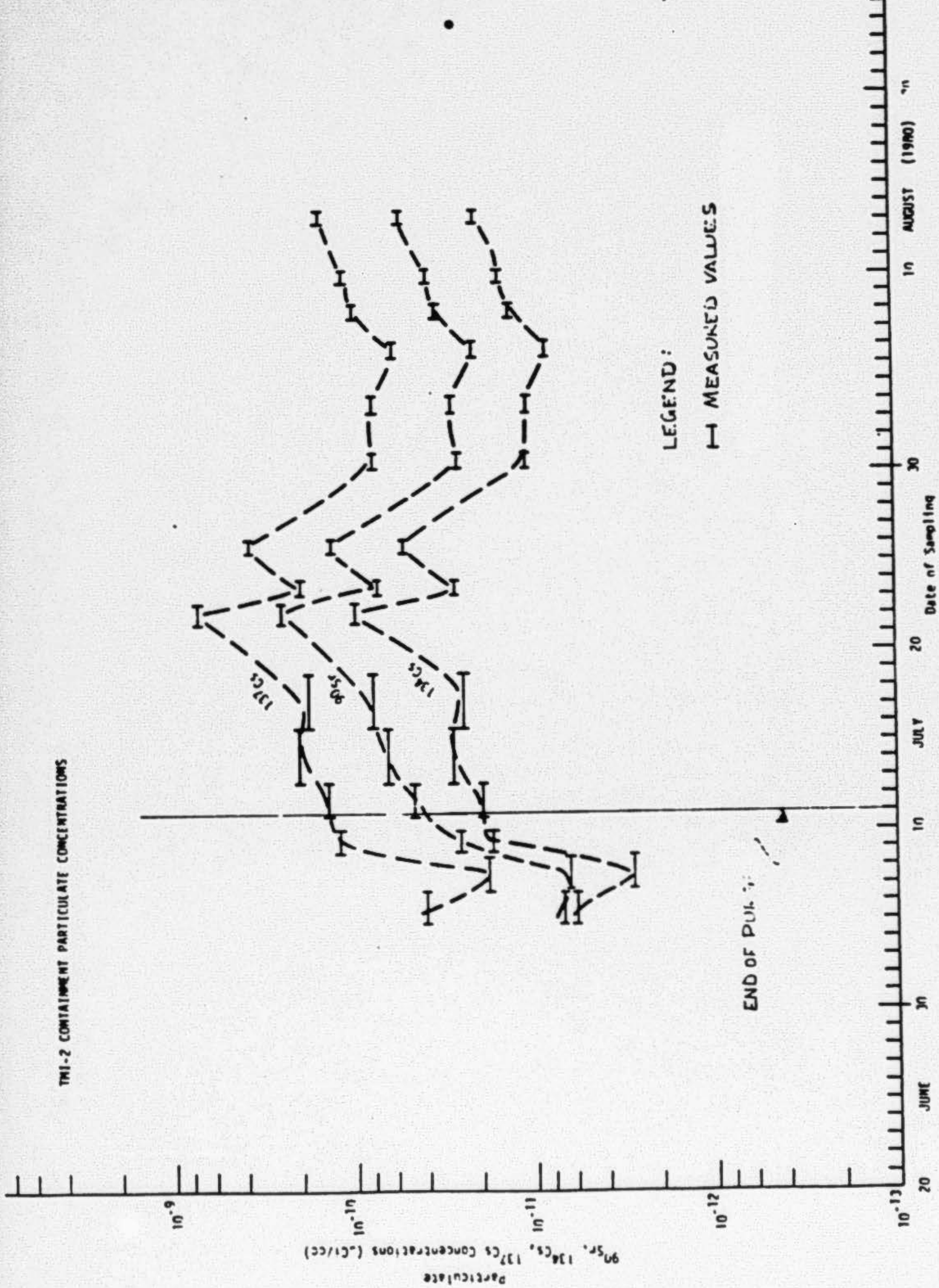
If the RB atmosphere were allowed to reach equilibrium (which has been prevented more recently by personnel entries, associated "mini-purges," and other activities) the equilibrium concentration would be:

$$Ci = \frac{\Sigma si}{\Sigma\gamma i}$$

The above source and sink estimates would predict equilibria at 1.0 E-8 and 3.0 E-9 uCi/cc, respectively, for Cs-137 and Sr-90.

The equilibrium value for Cs-137 is used as an estimator of the maximum RB particulate concentration during PDMS. This is considered conservative as it exceeds the actually observed level at any time after initial accident conditions. The removal of many of the potential sources of Cs and Sr in the RB resulting from the decontamination of large surfaces areas, removal of the basement water and sediment, and flushing the sealing of contaminated systems prior to PDMS, is expected to further reduce airborne concentrations during PDMS.

FIGURE A-1



TMI-2 CONTAINMENT PARTICULATE CONCENTRATIONS
LEGEND:
MEASURED VALUES
^{137}Cs
^{90}Sr
^{134}Cs
10^{-9}
10^{-10}
10^{-11}
10^{-12}
10^{-13}
Particulate
^{90}Sr, ^{134}Cs, ^{137}Cs Concentrations (μCi/cc)
20
JUNE
30
10
JULY
20
30
10
AUGUST (1980)
Date of Sampling