TECHNICAL EVALUATION
OF TMI-2 POST-DEFUELING MONITORED STORAGE FACILITY OPERATING LICENSE NO. DPR-73

GPU NUCLEAR CORPORATION

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ABSTRACT

Three Mile Island Unit 2 (TMI-2) is a pressurized-water reactor that experienced a loss-of-reactor-coolant accident in 1979, which resulted in serious damage to the reactor fuel and the spread of radioactive contamination throughout the facility. Following mitigation of the accident and stabilization of the facility, major efforts were made to remove the fuel and clean up the contamination. In 1988, the licensee proposed to place the facility in post-defueling monitored storage (PDMS) and to amend the facility license to a possession-only license (POL). The basic criterion for transition to PDMS is assurance that the health and safety of the public is protected by conformance to all applicable Commission regulations. The evaluation of the safety significance of PDMS is dependent on the conditions of the facility at the time the plant begins long-term storage and on the proposed actions of the licensee during the storage period.

Seven prerequisites for placing the TMI-2 facility into PDMS have been identified. Each of these prerequisites is defined and the actions taken to ensure its completion described. The Nuclear Regulatory Commission (NRC) staff has also identified six environmental protection systems that are important in providing reasonable assurance that the facility can be safely maintained during PDMS. The NRC staff concluded (based on their review of material received from the licensee and on independent evaluation and measurements by the NRC staff) that the prerequisites have been met, and the environmental protection systems are satisfactory. The staff further concluded that the proposal to place TMI-2 into PDMS conforms to all applicable Commission regulations and can be implemented without significant risk to the environment or the public.

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ABBREVIATIONS

ac	alternating current
AFHB	auxiliary and fuel-handling building
ALARA	as low as is reasonably achievable
CFR	Code of Federal Regulations
dc	direct current
DCR	Defueling Completion Report
EPA	U.S. Environmental Protection Agency
FPPE	Fire Protection Program Evaluation
GPU or GPUNC	General Public Utilities Nuclear Corporation
HEPA	high-efficiency particulate air
HVAC	heating, ventilation, and air conditioning
MDL	minimum detectable limit
NRC	U.S. Nuclear Regulatory Commission
ODCM	Offsite Dose Calculation Manual
PEIS	programmatic environmental impact statement
PDMS	post-defueling monitored storage
POL	possession-only license
REMP	Radiological Environmental Monitoring Plan
SAR	Safety Analysis Report
SCBA	self-contained breathing apparatus
SD	system descriptions
SFML	safe fuel mass limit
SNM	special nuclear material
TER	Technical Evaluation Report
TMI-1	Three Mile Island Unit 1
TMI-2	Three Mile Island Unit 2
VAC	voltage alternating current

EXECUTIVE SUMMARY

The purpose of this technical evaluation is to evaluate the safety significance of the licensee's proposal to place the Three Mile Island Unit 2 facility into post-defueling monitored storage (PDMS). This evaluation forms the basis for the requirements and controls to be maintained during storage to ensure public health and safety and protection of the environment. A brief history of the facility and its current status are provided. The prerequisites for PDMS are given and the environmental protection issues that accompany these prerequisites during PDMS are addressed in this report.

FACILITY HISTORY AND CURRENT STATUS

Three Mile Island Unit 2 (TMI-2), a 890-megawatt electric pressurized-water reactor, was issued an operating license on February 8, 1978. On March 28, 1979, an accident at the TMI-2 facility involved a loss of reactor coolant and resulted in serious damage to the reactor fuel. When coolant was restored, radioactive contamination in the form of core debris and fission products was distributed by the cooling water throughout the reactor coolant system. A portion of the water, carrying core debris and fission products as dissolved and particulate material, escaped from the reactor coolant system and flowed into the reactor building basement. Exposed surfaces in the reactor building and the auxiliary and fuel-handling building (AFHB) were contaminated with material in the reactor coolant and from radionuclides that became airborne as steam that had escaped from the reactor coolant system and then had condensed during and shortly after the accident.

On July 20, 1979, the U.S. Nuclear Regulatory Commission (NRC) issued an order* suspending the authority of the licensee, General Public Utilities Nuclear Corporation (GPU or GPUNC), to operate the facility and requiring that the licensee maintain the facility in a shutdown condition in accordance with approved operating and contingency procedures. Although its authority to operate the facility was suspended, the licensee retained an operating license, possessing a Class 103 (10 CFR Part 50.22) license. The staff initially made the decision for the facility to retain the operating license because the exact extent of the damage was unknown and some people still believed, in the absence of a true understanding of the extent of the damage, that the facility could be refurbished and could operate again. Later, when the true extent of the damage was understood, the NRC staff concluded that permitting the licensee to maintain an operating license during defueling and partial clean up allowed for the most effective regulatory oversight by the NRC. An order dated February 11, 1980, issued new proposed Technical

Specifications, which modified or replaced all Technical Specifications in Appendix A and sections of Appendix B. These proposed Technical Specifications became final on January 27, 1987. Since that time, the Technical Specifications have been amended 15 times.

The NRC is responsible for regulating the TMI-2 cleanup operations to ensure the protection of the health and safety of the public and the TMI-2 occupational work force and the protection of the environment. The cleanup activities at the TMI-2 facility were conducted in compliance with the appropriate Federal and State regulations. NRC responsibilities include reviewing and approving the licensee's proposals for cleanup actions, overseeing the licensee's implementation of approved activities, coordinating with other Federal and State governmental agencies regarding their activities in the cleanup, and informing local officials and the public about the status of the cleanup. The NRC has reviewed the licensee's cleanup activities and has acted upon license amendment requests when appropriate.

Following mitigation of the accident and stabilization of the facility, the licensee's major efforts have included partial decontamination of contaminated areas, removal of the fuel, and removal and treatment of the accident-generated water. Currently, the reactor vessel and the reactor coolant system have been defueled to the extent reasonably achievable (all fuel removed that is reasonably accessible within technically practical methods), and the possibility of criticality in the reactor building has been precluded. The facility has been extensively modified to facilitate defueling and decontamination activities. The facility is not operable. All canisters containing core material have been shipped to an offsite location. No canisters containing core material remain in the reactor building. Technical Specifications (including removing the requirements for criticality monitoring and for the presence of operators in the control room) have also been reduced.

The licensee has proposed placing the facility in post-defueling monitored storage. In the Safety Analysis Report (SAR) for Post-Defueling Monitored Storage (PDMS), the licensee states that "TMI-2 has been defueled and decontaminated to the extent that the plant is in a safe, inherently stable condition suitable for long-term management and any threat to the public health and safety has been eliminated. This long-term management condition is termed Post-Defueling Monitored Storage." PDMS was proposed by the licensee in a letter dated December 2, 1986. In a letter dated August 16, 1988, the licensee requested that the TMI-2 license (No. DPR-73) be amended to authorize implementation of the proposed PDMS plant configuration. Transmitted with the August 16, 1988, letter were the proposed amended facility license, proposed PDMS Technical Specifications, and the PDMS SAR. The requested action would allow the licensee to place the TMI-2 facility in monitored storage. The licensee has indicated that the proposed monitored storage of TMI-2 would not extend beyond the operation of Three Mile Island Unit 1 (TMI-1). In response to the licensee's request, the NRC evaluated the environmental impacts associated with the licensee's proposal to place the TMI-2 facility into storage at the conclusion of defueling. This analysis was published in August 1989 as Supplement 3 to the "Programmatic Environmental Impact Statement Related to Decontamination and Disposal of Radioactive Waste Resulting from March 28, 1979 Accident, Three Mile Island Nuclear Station, Unit 2." The Pacific Northwest Laboratory under the direction of the NRC staff has prepared this technical evaluation to assess the licensee's proposal to place the TMI-2 facility into PDMS and to ensure that the proposed action is within the scope of the Programmatic Environmental Impact Statement (PEIS) and its supplements. The staff adopts this evaluation and where the term "staff" appears in this document it refers to observations, analyses, or conclusions made by the Pacific Northwest Laboratory and adopted by the NRC staff. This report provides the technical basis for requirements imposed upon the licensee and documents commitments made by the licensee in support of their license amendment for PDMS. The actual requirements will be incorporated into the TMI-2 license with the issuance of the PDMS amendment.

PREREQUISITES FOR POST-DEFUELING MONITORED STORAGE

The basic criterion for transition to PDMS is compliance with all applicable Commission regulations and assurance that the health and safety of the public is protected. The evaluation of the safety significance of PDMS is dependent on the conditions of the facility at the time the plant begins long-term storage. The NRC staff has evaluated PDMS on the basis of the initial conditions identified by the licensee (in the PDMS SAR) and described below.

Prerequisites for PDMS are based on the NRC staff evaluation of information provided by the licensee in the PDMS SAR, 1.1.2.1. The prerequisites are as follows:

1. Defueling of the facility to the extent reasonably achievable and to such a degree that a nuclear criticality is precluded.

2. Shipment off site of all fuel and core debris that have been removed from the reactor and associated systems.

3. Removal of water, to the extent practicable, from the reactor coolant system; draining of the fuel transfer canal; and isolation of the fuel transfer tubes. To the extent that the spent fuel pools are needed to store the accident-generated water, water may remain in these pools after the start of PDMS.

4. A reduction of potential for release of radioactive material from the facility above the design objectives specified in Title 10 of the Code of Federal Regulations, Part 50 (10 CFR Part 50), Appendix I, for offsite dose consequences and a reduction of the potential for instantaneous concentrations of released material within the limits specified in 10 CFR Part 20.

5. Shipment off site or packaging and staging for shipment of remaining radioactive waste from the major TMI-2 decontamination activities.

6. Determination and reduction of radiation levels within the facility to allow plant monitoring, maintenance, and inspections.

Additionally, the NRC staff has identified a seventh prerequisite for PDMS:

7. Definition and establishment of an adequate surveillance program for PDMS environmental protection systems to ensure public health and safety.

Each of these prerequisites is discussed below.

Reduction of Potential for Accidental Criticality

The licensee's "Defueling Completion Report" (DCR) describes the methods and efforts used to remove fuel from the facility; specifies, as far as possible, locations and quantities of fuel remaining; and provides the technical basis for ensuring that an accidental criticality is precluded. For the purposes of this document, fuel is defined as UO_2 (uranium dioxide). Core debris is a mixture of fuel, structural and adsorber materials resulting from the accident at TMI-2 and the subsequent cleanup. The NRC has conducted an independent evaluation of the DCR to verify that an accidental criticality is precluded. This evaluation included reviewing calculations, conducting onsite inspections, and viewing videotapes of defueled areas. The NRC staff agrees with the licensee that, considering the location, form, and quantity of fuel (i.e., UO_2) and core debris remaining in the facility and the safeguards enacted by the licensee, an accidental criticality is precluded.

Removal of Fuel and Core Debris From the TMI Site

The NRC staff has verified that all fuel canisters containing core debris have been removed from the reactor facility and shipped off site.

Removal of Water

Before the beginning of PDMS, the reactor vessel, the reactor coolant system, the reactor building fuel canal, and the fuel transfer tubes will be drained of water. The reactor vessel will be covered to minimize the potential for water entry. The submerged demineralizer system and spent fuel pool B will be drained and shielded, as required. The fire mains within the reactor building will be closed with valves and drained. It is estimated that the reactor vessel will contain approximately 10 gallons (38 liters) of residual water. However, because of the distribution, amount, and physical state of fuel located in the reactor vessel, this quantity of water will not contribute to a criticality. In addition, the quantity of water that will remain in the reactor coolant system is not enough to transport radioactive material within the facility. There may be some water stored outside containment awaiting processing in the Accident Generated Water evaporator after PDMS begins. This water is considered to be in temporary storage prior to disposal and may remain in these locations until completion of the evaporation operations during the beginning of PDMS.

Reduction of the Potential for Release of Radioactive Material

The potential for release of any significant quantity of radioactive material from TMI-2 during PDMS has been minimized by the removal of as much of the fuel and core debris as is reasonably achievable and the decontamination of large sections of the reactor building and the surfaces, equipment, and piping in the auxiliary and fuel-handling building (AFHB). The major source of radioactive material remaining in the facility is inside the reactor building. This building constitutes a sealed and leaktight enclosure, except during entry for inspection and measurements, and except for the use of a breather system that is equipped with a high-efficiency particulate air (HEPA) filter. Releases of radioactive material during PDMS are governed by the Code of Federal Regulations (CFR) 10 CFR Part 20, Appendix B, Table II, and 10 CFR Part 50, Appendix I, as referenced in the TMI-2 proposed Technical Specifications for PDMS (Chapter 9 of the PDMS SAR). The radioactivity concentration in gaseous and liquid effluents from TMI-2 to the environment during PDMS shall not exceed the values specified in 10 CFR Part 20, Appendix B, Table II. In addition, the design objective annual exposure values specified in 10 CFR Part 50, Appendix I, apply during PDMS. Releases of radioactive material to the atmosphere must also meet the Environmental Protection Agency's (EPA's) environmental standards for the uranium fuel cycle specified in 40 CFR Part 190. Releases for radioactive material to water must meet EPA's National Interim Primary Drinking Water Standards specified in 40 CFR Part 141.

Appendix I to 10 CFR Part 50 states as a design objective, that the calculated annual total quantity of all radioactive iodine and radioactive material in particulate form above background to be released in effluent to the atmosphere cannot result in an estimated annual dose or dose commitment to any individual in an unrestricted area in excess of 15 millirem to any organ. The design objective also states that the calculated annual total quantity of all radioactive material above background to be released to the atmosphere cannot result in an estimated annual air dose from gaseous effluent to an individual in an unrestricted area in excess of 10 millirad for gamma radiation and 20 millirad for beta radiation, except if reasonable assurance is provided that the proposed higher quantity will not result in an estimated annual external dose from the effluent in excess of 5 millirem to the total body or 15 millirem to the skin. The dose permitted by these design objectives compares to an estimated 50-year dose commitment from a 1-year release of 0.22 millirem to the total body and 2.8 millirem to the bone of the maximally exposed offsite individual at TMI-2 during PDMS.

For routine (normal) activities, the design objective from 10 CFR Part 50, Appendix I, states that the annual total quantity of all radioactive material above background that is released in liquid effluent should be limited such that it would not result in an estimated annual dose or dose commitment for any individual in an unrestricted area (considering all pathways of exposure) in excess of 3 millirem to the total body or 10 millirem to any organ. This dose is comparable to the estimated 50-year dose commitment of less than 0.0009 millirem to the total body and 0.0009 millirem to the bone of the maximally exposed offsite individual at TMI-2 during PDMS.

10 CFR Part 100 applies to any accident at TMI-2 during PDMS. This regulation limits the dose to an individual located at any point on the site boundary for 2 hours immediately following onset of the postulated fission product release to a whole-body dose of 25 rem or a total radiation dose of 300 rem to the thyroid from iodine exposure. The licensee evaluated seven potential accident scenarios in the PDMS SAR. Four of these accidents were based on the

possibility that minor or limited decontamination activity* may be conducted to maintain the facility in reasonable condition. The seven accidents include (1) a vacuum canister failure during decontamination activities, (2) accidental spraying of concentrated contamination with high-pressure spray during decontamination activities, (3) accidental cutting of contaminated pipe during decontamination activities, (4) accidental break of a contaminated pipe during decontamination activities, (5) a fire in the containment building, (6) an open penetration during surveillance or maintenance activities, and (7) rupture and release of contaminated resins of a single makeup and purification demineralizer. The staff reviewed and performed independent evaluations of the seven potential accidents identified by the licensee in the PDMS SAR. The staff also evaluated an eighth potential accident, a fire in the D-rings inside containment. In all cases, the resulting doses would be significantly lower than those permitted by 10 CFR Part 100 for determination of exclusion areas. The accident that would result in the maximum offsite dose is the fire in the containment building. The estimated 50-year dose commitment from a fire in the reactor building without operation of the ventilation system is a 49-millirem dose to the total body and a 51-millirem dose to the bone of the maximally exposed offsite individual.

Removal of Radioactive Waste Resulting From Major Decontamination Activities

All of the radioactive waste resulting from major decontamination activities has been shipped off site or packaged and staged for shipment off site, except the waste from the accident-generated water disposal activities. (The removal of waste resulting from accident-generated water disposal activities is discussed in the NRC staff's Safety Evaluation for the Accident-Generated Water, dated September 11, 1989.) Some of these wastes may remain on site awaiting shipment after the commencement of PDMS. Some radioactive components and equipment will remain in the reactor building, including the reactor head assembly, the upper plenum assembly, the upper end fittings, sections of the flow distributor that contain incore instrument guide tubes, and fuel-removal tooling. These remaining components and equipment are not flammable and do not add significant quantities of radioactive material to the inventory in the reactor building. Consequently, they do not represent an increased risk to the health and safety of the public.

Reduction of Radiation Levels To Allow Plant Maintenance and Surveillance During Post-Defueling Monitored Storage

During PDMS, entry into the reactor building and into the AFHB will be made periodically to conduct inspections, surveillance, radiological surveys, radiological waste processing, remedial decontamination, and some maintenance to support these activities, as well as preventive maintenance on a limited number of operational systems. During the initial period of PDMS, monthly entries will be made. After facility stability has been verified and a database has been established, the frequency of entry may be reduced.

* Although few activities are expected to be conducted during PDMS, routine surveillance, preventive maintenance, and stabilization activities will be conducted if migration of radioactive material is detected.

Decontamination and shielding have been performed to reduce radiation dose rates in areas requiring access by personnel. Radiation exposures to personnel will be maintained within the limits established by 10 CFR Part 20.

Definition and Establishment of a Surveillance Program

During PDMS, the licensee will be required to or has committed to conduct surveillance programs to ensure the maintenance of environmental protection systems. These programs include surveillance of reactor containment building isolation, surveillance of the reactor containment building and the AFHB ventilation and filtration systems, surveillance of the fire protection system and the support air monitoring systems (including electrical, effluent monitoring, and environmental monitoring systems), and oversight of administrative systems. Administrative systems include organizational structure, staff qualifications, records, independent safety reviews, procedures; occupational radiation protection, a quality assurance plan, an emergency plan, and other administrative control activities. Operability and surveillance requirements for these systems are contained or referenced in this document, which forms the licensing basis for PDMS. Referenced documents include, but are not limited to, the PDMS SAR, the Fire Protection Program Evaluation, the Offsite Dose Calculation Manual, the Radiological Environmental Monitoring Plan, the PDMS Quality Assurance Plan, and the TMI-2 Technical Specifications. Operability, surveillance, and monitoring requirements are described in this document and the referenced documents to ensure that the facility is maintained in the configuration analyzed by this evaluation.

ENVIRONMENTAL PROTECTION ISSUES DURING POST-DEFUELING MONITORED STORAGE

Considering the post-accident, inoperable and essentially defueled condition of the reactor, TMI-2 has no safety-related structures, systems, or components. Safety-related structures, systems, and components are those necessary to ensure the capability of shutting down the reactor and maintaining it in a shutdown condition. Although there are no safety-related structures, systems, or components at TMI-2, the license for TMI-2 ensures that the facility is maintained in an acceptable condition and that the environment is protected during PDMS. In contrast to the concern of ensuring safe shutdown of an operating plant or maintaining safe shutdown in the pre-PDMS TMI-2 facility, the principal safety concern during PDMS is the inadvertent release of radioactive materials into the environment. For this reason, the staff has identified structures, systems, and components that provide reasonable assurance that the facility can be maintained in a defueled condition without undue risk to the health and safety of the public. These systems, called PDMS environmental protection systems, are (1) the reactor containment structure; (2) the reactor containment and AFHB purge, breather, ventilation, and filtration systems; (3) the fire protection system; (4) the reactor vessel; (5) the flood protection system; and (6) the support and monitoring systems. A review of these systems by the NRC staff has indicated that these systems can provide a reasonable assurance that the facility can be maintained in a defueled condition without undue risk to the health and safety of the public or the environment.

CONCLUSIONS

On the basis of the material received from the licensee and independent evaluation and measurements, the staff concludes that the proposal to place TMI-2 into PDMS is within applicable regulatory limits and can be implemented without significant risk to the environment or the public.

1 INTRODUCTION

Post-defueling monitored storage (PDMS) was initially proposed in a letter from the licensee [General Public Utilities (GPU) Nuclear Corporation (GPUNC)] to the U.S. Nuclear Regulatory Commission (NRC) dated December 2, 1986 (GPU 1986). However, the concept of PDMS was first introduced by the Three Mile Island Unit 2 (TMI-2) Advisory Panel on April 12, 1984 (NRC 1984b). The approach to PDMS was expanded when the licensee submitted its environmental evaluation of PDMS on March 11, 1987 (GPU 1987a).

In response to the licensee's proposal to place the TMI-2 facility into PDMS at the end of defueling, the NRC staff evaluated the environmental impacts associated with the licensee's proposal. A draft supplement to the original programmatic environmental impact statement (PEIS) related to the decontamination and disposal of radioactive wastes resulting from the accident was published as NUREG 0683, Supplement 3, in April 1988 (NRC 1988b). This draft supplement was circulated to Federal, State, and local government agencies and to interested members of the public for comment. The final supplement, published in August 1989 (NRC 1989a), contains the NRC staff's evaluation of the environmental impacts of the licensee's proposal for PDMS, as well as a number of alternatives, and established ranges for the expected plant conditions and the expected radiation exposure.

By letter dated August 16, 1988 (GPU 1988b), the licensee requested that TMI-2 License No. DPR-73 be amended to a possession-only license and that the license authorize implementation of the proposed PDMS plant configuration. The requested action would allow the licensee to place the TMI-2 facility in monitored storage. In a letter dated June 23, 1989 (GPU 1989d), the licensee indicated that the proposed monitored storage of TMI-2 would not extend beyond the operation of Three Mile Island Unit 1 (TMI-1).

The licensee transmitted the proposed amended facility license, the proposed PDMS Technical Specifications, and the PDMS Safety Analysis Report (SAR) with its letter of August 16, 1988. The licensee supplied additional supporting information (GPU 1989a, 1989b, 1989c, 1989e, 1989f, 1989g, 1990a, 1990b, 1990c, 1991a, 1991b, 1991c, 1991d, 1991e, and 1992) partly as a response to NRC staff requests (NRC 1989b, 1989c, 1989d, 1989e, 1990a, and 1990b) resulting from NRC's detailed review of the licensee's proposal for PDMS of the TMI-2 facility.

Pacific Northwest Laboratory under the direction of the NRC staff has evaluated the PDMS SAR through Amendment 15 and prepared this technical evaluation report. For the purposes of this document, the terms "NRC staff" or "staffs" refer to evaluations performed for this document by the Pacific Northwest Laboratory under the direction of the NRC staff and the results of which are adopted by the NRC staff. Additionally, the staff has evaluated the PDMS proposal against the applicable requirements of 10 CFR Part 20; 10 CFR Part 50; 10 CFR Part 50, Appendix A (general design criteria); 10 CFR Part 100; and 40 CFR Part 190.

Because the TMI-2 reactor has been defueled, there are no safety-related systems that pertain to safe shutdown that are required to maintain the unit during PDMS. Thus, the purpose of this technical evaluation report is to evaluate the licensee's proposal to place the TMI-2 facility into PDMS and to ensure that the proposed action is within the scope of the PEIS and its supplements. This document provides the basis for requirements necessary to ensure public health and safety and protection of the environment.

Section 2 of this technical evaluation provides a brief regulatory history of the TMI-2 facility. Section 3 provides a description of PDMS. Section 4 discusses the status of the facility before entry into PDMS, and Section 5 lists and describes the major prerequisites for facility configuration at the start of PDMS. Section 6 provides a detailed discussion of the structures, systems, and components that are used to ensure that the environmental protection systems are satisfactory and that the facility can be safely maintained during PDMS. The conclusion is provided in Section 7, and reference material is listed in Section 8.

2 REGULATORY HISTORY

Three Mile Island Unit 2 was issued an operating license on February 8, 1978. The accident on March 28, 1979, involved a loss of reactor coolant and resulted in serious damage to the reactor fuel. On July 20, 1979, the NRC issued an order (NRC 1979a) suspending the licensee's authority to operate the TMI-2 facility and requiring that the licensee maintain the facility in a shutdown condition in accordance with approved operating and contingency procedures. Initially, because the exact extent of the damage was unknown, it appeared (in the absence of a true understanding of the extent of the damage) that the facility could be refurbished and would operate again. A subsequent order, dated February 11, 1980 (NRC 1980), provided new proposed Technical Specifications, which modified or replaced all Technical Specifications in Appendix A. These Technical Specifications were contested by a member of the public and were not issued in final form until January 27, 1987. The licensee retains a 10 CFR Part 50 license since the Atomic Energy Act of 1954, as amended, requires a license for possession of a defueled reactor.

On July 20, 1981, the NRC issued an exemption to the requirements of 10 CFR Part 50.71 (e) for License No. DPR-73. The exemption deleted the requirement to periodically update the TMI-2 FSAR to reflect true plant conditions. The exemption required the licensee to use instead System Descriptions (SDs) and Technical Evaluation Reports (TERs) for documenting changes made to the facility during the cleanup. These documents are required to be updated annually. The licensee has proposed using the August 16, 1988, PDMS SAR as the licensing basis document for PDMS and will periodically update the PDMS SAR to reflect current plant conditions. (See proposed Technical Specification 6.8.1.3b and GPU 1991e, SAR 3.1.1.56.)

Following mitigation of the accident and stabilization of the facility, the licensee's efforts have been focused largely on the removal and treatment of the accident-generated water, decontamination, and removal of the reactor fuel.*

In 1988, Amendment 30 to the TMI-2 license (NRC 1988a) defined the plant conditions corresponding to three facility modes. Mode 1 was defined as the condition wherein the reactor was subcritical with an average reactor coolant temperature of less than 200°F. The facility was in Mode 1 shortly after the

* A more detailed description of the facility's history and the cleanup efforts can be found in the Programmatic Environmental Impact Statement, NUREG 0683 dated March, 1981 (NRC 1981) as supplemented.

accident and continued so to the end of active defueling. Mode 2 was defined as the facility condition in which (1) the reactor vessel and the reactor coolant system were defueled to the extent reasonably achievable, (2) the possibility of a criticality in the reactor building was precluded, and (3) all canisters containing core material were removed from the reactor building. Mode 3 was defined as meeting the conditions of Mode 2 plus the removal of any canisters containing core material from the TMI-2 site. The TMI-2 facility is currently in Mode 3. The number of Technical Specifications, including the need for criticality monitoring and for the presence of operators in the control room, was also reduced when the facility progressed into Mode 3.

All postaccident operations at TMI-2 were conducted under the following regulatory objectives: (1) to maintain reactor safety and control of radioactive releases; (2) to ensure that environmental impacts of cleanup are minimized and that radiation exposures to workers, to the public, and to the environment are within regulatory limits and as low as is reasonably achievable; and (3) to ensure interim safe storage and/or disposal of radioactive wastes from cleanup operations (NRC 1984a). During cleanup activities, the NRC staff has, where appropriate, issued license amendments to the TMI-2 Technical Specifications requested by the licensee that reflected changing plant conditions.

During the extended period of PDMS, the NRC staff will conduct periodic inspections of the licensee's compliance with the licensing basis documents (including but not limited to the licensee's Safety Analysis Report, the Fire Protection Program Evaluation (FPPE), the Offsite Dose Calculation Manual (ODCM), the Radiological Environmental Monitoring Program (REMP), the PDMS Quality Assurance Plan, and PDMS Technical Specifications) and the requirements of the Code of Federal Regulations (CFR).

3 DESCRIPTION OF POST-DEFUELING MONITORING STORAGE

The licensee proposes to place the TMI-2 facility in long-term monitored storage until TMI-1 is decommissioned. The licensee stated that TMI-2 will remain in monitored storage no later than the end of TMI-1 operation (GPU, 1990a), at which time both units would be decommissioned. The present TMI-1 license expires on April 19, 2014. The licensee, in a recent amendment, requested a license extension for TMI-2 to April 19, 2014 (this amendment request is being processed separately). Therefore, if PDMS begins in 1991, and the licensee is allowed to defer decommissioning until the expiration date of the TMI-1 license (2014), then the duration of PDMS would be 23 years (the length of time between 1991 and 2014). Thus, for the purposes of this technical evaluation, the staff used a storage period of 23 years. If the license extension request is not granted, then the duration of PDMS would be less than 23 years. A storage period of less than 23 years would be encompassed by this evaluation. A storage period of more than 23 years would not necessarily invalidate this technical evaluation, although additional analyses (such as analyses of surveillance programs) may be required for significantly longer storage periods. At the end of the storage period, the facility would be decommissioned. Decommissioning is not evaluated in this technical evaluation.

The licensee has stated (GPU 1987b) that a monitored storage period would be beneficial for several reasons: (1) occupational dose in the plant would be reduced during monitored storage because of natural decay of radioactive contamination; (2) a monitored storage period would allow time for continued development of decontamination technology so that the most effective and efficient techniques may be applied; and (3) further reduction in occupational exposure could be achieved through the use of advanced robotic technology, automatic cleaning and chemical cleaning techniques, and advanced waste treatment methods developed during PDMS.

The licensee has further indicated (GPU 1988a) that during the PDMS period, both the developing technology for radioactive waste packaging and volume reduction could result in a reduction in the total volume of radioactive waste generated following PDMS. In addition, the licensee has stated that placing the TMI-2 facility in storage until the decommissioning of TMI-1 would allow a more efficient use of the site decommissioning work force and eliminate any possible impact of TMI-2 decontamination and decommissioning efforts on the operating TMI-1 facility.

During PDMS, the TMI-2 facility would be in long-term monitored storage, similar to the decommissioning mode SAFSTOR, in which the facility is secured, monitored, and maintained in a manner that ensures the protection of the public health and safety for an extended period.

4 STATUS OF FACILITY BEFORE POST-DEFUELING MONITORED STORAGE

The March 28, 1979, accident at the TMI-2 facility involved a loss of reactor system coolant and resulted in serious damage to the reactor fuel. When coolant was restored, radioactive contamination in the form of core debris and fission products was distributed by the cooling water throughout the reactor coolant system. A portion of the water, carrying core debris and fission products as dissolved and particulate material, escaped from the reactor coolant system and flowed into the reactor building basement. Exposed surfaces in the reactor building and the auxiliary and fuel-handling building (AFHB) were contaminated with material in the reactor coolant and from radionuclides that became airborne as steam that had escaped from the reactor coolant system and then had condensed during and shortly after the accident.

TMI-2 cannot operate and has not operated since the accident. Following mitigation of the accident and stabilization of the facility, the licensee's efforts have focused largely on the removal and treatment of the accident-generated water, decontamination of the facility, and removal of the fuel.

4.1 Disposal of Accident-Generated Water

Approximately 2.3 million gallons (8.7 million liters) of the accident-generated water (as defined in the February 27, 1980, agreement between the City of Lancaster, Pennsylvania; Metropolitan Edison Company; and the NRC) have been processed through decontamination systems and placed in storage while awaiting final disposition. The NRC staff evaluated the environmental impact of the disposition of the accident-generated water in Final Supplement 2 to the PEIS (NRC 1987). A continuous effluent monitoring system with an operating ventilation system is being used to monitor releases during disposition of the accident-generated water. Disposal of the accident-generated water may not be completed before the facility is ready to enter PDMS. As of December 21, 1991, over a million gallons (more than 3.79 million liters) have been evaporated. However, the disposition of the accident-generated water will not have a significant effect on the PDMS configuration of the facility. The NRC staff evaluated the effect of disposing of the accident-generated water in the safety evaluation for the accident-generated water evaporator (NRC 1989f) and in PEIS Supplement 2 (NRC 1987).

4.2 Decontamination of Facilities

The licensee has conducted an extensive decontamination program since the accident. The recovery activities were primarily focused on removal of the reactor fuel. Extensive decontamination was undertaken to expand access to the facility and to ensure that occupational exposures were as low as is reasonably achievable (ALARA). The decontamination activities to date are discussed for the AFHB, the reactor building, and other contaminated facilities.

4.2.1 Auxiliary and Fuel-Handling Building

The interior of the AFHB and 26 piping systems in the AFHB were contaminated as a result of the accident, although less severely than the reactor building. Cleanup of the AFHB began shortly after the accident. Considerable amounts of debris and contaminated equipment have been removed, contaminated systems have been flushed, and the building and equipment have been decontaminated. Decontamination of the AFHB, which consists of 137 areas (cubicles), has resulted in the radiation and contamination levels shown in Table 4.1. Several of the cubicles contain air-handling units, in which the radiological condition changes with each regularly scheduled filter change. Likewise, the status of the truck bay cubicle varies because of the ongoing work activities.

As part of the decontamination program, the licensee has established contamination-level goals for entrance into PDMS for each area of the AFHB. The contamination-level goals to be reached for most areas before placement into PDMS have been established and are also given in Table 4.1. If the decontamination goals cannot be met because of the unique situation at TMI-2 or ALARA considerations, the licensee will provide an evaluation of the specific situation to the NRC.

The radiological assessment of the facility will be completed by the licensee and verified by the NRC staff before entry into PDMS (GPU 1991a, SAR 5.3.2). Radiological surveys will serve as a baseline for PDMS. When PDMS begins, most of the cubicles and the corridors will have been decontaminated to levels similar to those in an undamaged reactor facility nearing the end of its operating life; thus, these cubicles could be accessible. The licensee has indicated (GPU 1991a, SAR 7.1.2.2) that the auxiliary building will be locked but will be accessible for periodic surveillance entries and other limited activities.

The fuel-handling portion of the AFHB was decontaminated and used during cleanup for handling of the defueling canisters. Defueling operations recontaminated the facility. Now that fuel removal has been completed, the fuel pools will be drained and the facility d ntaminated. The fuel-handling portion of the AFHB that is above elevati 347 feet 6 inches is in direct contact with the fuel-handling building in the TMI-1 facility and will remain ventilated and filtered through the TMI-1 operations.

Levels of residual removable contamination for the AFHB are given in Table 4.2. The licensee has estimated (GPU 1991d, SAR Tables 5.3-5 and 5.3-6) that less than 0.5 curie of removable contamination is present in the AFHB and in other contaminated buildings at TMI-2. No data is available for 17 of the areas for various reasons, including changing radiological conditions as preparations are made for PDMS. However, the licensee has committed

Table 4.1 Decontamination Goals and Post-Defueling Monitored Storage Radiological Conditions in the Auxiliary and Fuel-Handling Building[a]

Cubicle Number	Area Description	Specific Decontamination Goals: General Area Dose Rate (mR/hr)	Specific Decontamination Goals: Surface Contamination <7 ft/Overheads (dpm/100 cm²)	PDMS Radiological Conditions[b]: General Area Dose Rate (mR/hr)	PDMS Radiological Conditions[b]: Surface Contamination <7 ft/Overheads (dpm/100 cm²)
AX001	Reactor Building Emergency Cooling Booster Pumps Area	<2.5	<1,000/<10,000	0.2 - 1.2	<1,000/<10,000
AX002	Access Corridor	<2.5	<1,000/<10,000	0.4 - 1.5	<1,000/<10,000
AX003	Access Area	<2.5	<1,000/<10,000	0.3 - 2.4	<1,000/<10,000
AX004	Seal Injection Valve Room	<1,000	<50,000/<50,000	60 - 150	150,000/<50,000
AX005	Makeup and Purification Pump 1C Room	<500	<50,000/<50,000	6 - 14	60,000/<50,000
AX006	Makeup and Purification Pump 1B Room	<500	<50,000/<50,000	10 - 100	240,000/<50,000
AX007	Makeup and Purification Pump 1A Room	<500	<50,000/<50,000	4 - 60	40,000/<50,000
AX008	Spent Resin Storage Tank 1B Room	<500	<150,000[c]/<500,000[c]	30 - 2,500	150,000/475,000
AX009	Spent Resin Storage Tank 1A Room	<500	<550,000[c]/<50,000	100 - 3,000	550,000/<50,000
AX010	Spent Resin Transfer Pump Room	<500	<10,400,000[c]/<50,000	75 - 300	10,400,000/<50,000
AX011	Auxiliary Building Sump Tank Pumps and Valve Room	<50	<5,000/<50,000	3 - 8	10,000/<50,000
AX012	Auxiliary Building Sump and Tank Room	<50	<5,000/<50,000	100	90,000/<50,000
AX013	Evaporator Condensation Tanks, Pumps, and Demineralizer Room	<500	<1,000/<10,000	0.2 - 14	2,000/<10,000

See footnotes at end of table.

Table 4.1 (cont'd)

Cubicle Number	Area Description	Specific Decontamination Goals: General Area Dose Rate (mR/hr)	Specific Decontamination Goals: Surface Contamination <7 ft/Overheads (dpm/100 cm^2)	PDMS Radiological Conditions[(b)]: General Area Dose Rate (mR/hr)	PDMS Radiological Conditions[(b)]: Surface Contamination <7 ft/Overheads (dpm/100 cm^2)
AX014	Reactor Coolant Evaporator Room	<500	<50,000/<50,000	3 - 20	100,000/<50,000
AX015a	Cleanup Filters Room	<500	<50,000/<50,000	10 - 25	35,000/<50,000
AX015b	Cleanup After Filters Room	<500	<50,000/<50,000	50 - 100	25,000/<50,000
AX016	Cleanup Demineralizer 2A Room	<500	<50,000/<50,000	10 - 14	500,000/<50,000
AX017	Cleanup Demineralizer 2B Room	<500	<50,000/<50,000	100 - 220	110,000/<50,000
AX018	Waste Transfer Pumps Room	<500	<50,000/<50,000	2 - 10	70,000/<50,000
AX019	Waste Disposal Liquid Valve Room	<500	<50,000/<50,000	0.2 - 5	7,000/<50,000
AX020	Reactor Coolant Bleed Holdup Tanks 1B and 1C Room	<500	<50,000/<50,000	12 - 450	290,000/<50,000
AX021	Reactor Coolant Bleed Holdup Tank 1A Room	<500	<50,000/<50,000	1 - 60	1,000/<50,000
AX022	North Stairwell	<2.5	<1,000/<10,000	0.5	<1,000/<10,000
AX023	Elevator Shaft (inside cab)	<10	<50,000/N/A	3.2	<1,000/N/A
AX024	Auxiliary Building Sump Filters Room	<500	<50,000/<50,000	4 - 25	60,000/<50,000
AX025	Area Between Service, Control, and Reactor Building	<500	<1,000/<10,000	1 - 10	1,000/<10,000
AX026	Seal Injection Filters Room	<500	<50,000/<50,000	5	60,000/<50,000

See footnotes at end of table.

Table 4.1 (cont'd)

Cubicle Number	Area Description	Specific Decontamination Goals: General Area Dose Rate (mR/hr)	Specific Decontamination Goals: Surface Contamination <7 ft/Overheads (dpm/100 cm^2)	PDMS Radiological Conditions(b): General Area Dose Rate (mR/hr)	PDMS Radiological Conditions(b): Surface Contamination <7 ft/Overheads (dpm/100 cm^2)
AX027	South Stairwell	<2.5	<1,000/<10,000	<0.2	<1,000/<10,000
AX101	Radwaste Disposal Control Panel Area	<2.5	<1,000/<10,000	<0.2	<1,000/<10,000
AX102	Reactor Building Sump Pumps Filters Room	<1,000	<50,000/<50,000	10	5,000,000/<50,000
AX103	Motor Control Center 2-11EB Room	<2.5	<1,000/<10,000	<0.2 - 0.4	<1,000/<10,000
AX104	Motor Control Center 2-21EB Room	<2.5	<1,000/<10,000	<0.2	<1,000/<10,000
AX105	Substation 2-11E Room	<2.5	<1,000/<10,000	0.4	<1,000/<10,000
AX106	Substation 2-21E Room	<2.5	<1,000/<10,000	<0.2	<1,000/<10,000
AX107	Motor Control Center 2-11EA Room	<2.5	<1,000/<10,000	<0.2	<1,000/<10,000
AX108	Motor Control Center 2-21EA Room	<2.5	<1,000/<10,000	<0.2	<1,000/<10,000
AX109	Nuclear Services Coolers and Pump Area	<2.5	<1,000/<10,000	<0.2	<1,000/<10,000
AX110	Intermediate Coolers Area	<2.5	<1,000/<10,000	<0.2	<1,000/<10,000
AX111	Intermediate Cooling Pumps and Filters Room	<50	<1,000/<10,000	<0.2 - 4	<1,000/<10,000
AX112	Seal Return Coolers and Filter Room	<1,000	<50,000/<50,000	10 - 45	40,000/<50,000
AX113	Waste Gas Analyzer Room	<50	<50,000/<50,000	1 - 15	20,000/<50,000

See footnotes at end of table.

Table 4.1 (cont'd)

Cubicle Number	Area Description	Specific Decontamination Goals: General Area Dose Rate (mR/hr)	Specific Decontamination Goals: Surface Contamination <7 ft/Overheads ($dpm/100\ cm^2$)	PDMS Radiological Conditions[b]: General Area Dose Rate (mR/hr)	PDMS Radiological Conditions[b]: Surface Contamination <7 ft/Overheads ($dpm/100\ cm^2$)
AX114	Makeup and Purification Demineralizer 1A Room	"AS IS"	"AS IS"	>100,000	(d)
AX115	Makeup and Purification Demineralizer 1B Room	"AS IS"	"AS IS"	>125,000	(d)
AX116	Makeup Tank Room	<500	<50,000/<50,000	30 - 1,000	10,000/<50,000
AX117	Makeup and Purification Filters Room	<1,000	<42,500,000[c]/<50,000	2,000	42,500,000/<50,000
AX118	Spent Fuel Coolers and Pumps Area	<2.5	<1,000/<10,000	0.3 - 3	1,000/<10,000
AX119	Spent Fuel Demineralizer Room	<2.5	<1,000/<10,000	0.2 - 2.4	4,000/<10,000
AX120	Spent Fuel Filters Room	<2.5	<1,000/<10,000	0.2 - 2	30,000/<10,000
AX121	Elevator Shaft (inside cab)	<2.5	<1,000/<10,000	0.2	<1,000/<10,000
AX122	North Stairwell	<2.5	<1,000/<10,000	<0.2	<1,000/<10,000
AX123	Access Area	<2.5	<1,000/<10,000	<0.2	<1,000/<10,000
AX124	Concentrated Liquid Waste Pump Room	<500	<50,000/<50,000	2 - 26	8,000/<50,000
AX125	Waste Gas Decay Tank 1B Room	<500	<50,000/<50,000	<0.2	<1,000/<50,000
AX126	Waste Gas Filter Room	<500	<50,000/<50,000	<0.2	<1,000/<50,000
AX127	Waste Gas Decay Tank 1A Room	<500	<50,000/<50,000	<0.2 - 0.5	<1,000/<50,000

See footnotes at end of table.

Table 4.1 (cont'd)

Cubicle Number	Area Description	Specific Decontamination Goals: General Area Dose Rate (mR/hr)	Specific Decontamination Goals: Surface Contamination <7 ft/Overheads (dpm/100 cm^2)	PDMS Radiological Conditions[b]: General Area Dose Rate (mR/hr)	PDMS Radiological Conditions[b]: Surface Contamination <7 ft/Overheads (dpm/100 cm^2)
AX128	Valve and Instrument Room	<500	<50,000/<50,000	<0.2 - 2	<1,000/<50,000
AX129	Deborating Demineralizer 1B Room	<500	<50,000/<50,000	0.2	<1,000/<50,000
AX130	Deborating Demineralizer 1A Room	<500	<50,000/<50,000	<0.2	<1,000/<50,000
AX131	Miscellaneous Waste Holdup Tank Room	<50	<5,000/<50,000	5 - 160	15,000/<50,000
AX132	Corridor Between Unit 1 and Unit 2	<2.5	<1,000/<10,000	<0.2	<1,000/<10,000
AX133	South Stairwell	<2.5	<1,000/<10,000	<0.2	<1,000/<10,000
AX134	Miscellaneous Waste Tank Pumps Room	<50	<50,000/<50,000	3 - 20	500,000/<50,000
AX135	Radwaste Disposal Control Panels	<2.5	<1,000/<10,000	<0.2	<1,000/<10,000
AX201	North Stairwell	<2.5	<1,000/<10,000	<0.2	<1,000/<10,000
AX202	Elevator Shaft	<2.5	<1,000/<10,000	<0.2	<1,000/<10,000
AX203	4160V Switchgear 2-1E Room	<2.5	<1,000/<10,000	<0.2	<1,000/<10,000
AX204	4160V Switchgear 2-2E Room	<2.5	<1,000/<10,000	<0.2	<1,000/<10,000
AX205	Reactor Building Purge Air Supply and Hydrogen Control Exchange Area	<2.5	<1,000/<10,000	<0.2 - 5	<1,000/<10,000
AX206	Reactor Building Purge Air Exhaust Unit B	<50	N/A	12	N/A

See footnotes at end of table.

Table 4.1 (cont'd)

Cubicle Number	Area Description	Specific Decontamination Goals: General Area Dose Rate (mR/hr)	Specific Decontamination Goals: Surface Contamination <7 ft/Overheads (dpm/100 cm^2)	PDMS Radiological Conditions[b]: General Area Dose Rate (mR/hr)	PDMS Radiological Conditions[b]: Surface Contamination <7 ft/Overheads (dpm/100 cm^2)
AX207	Reactor Building Purge Air Exhaust Unit A	<50	N/A	15	N/A
AX208	Auxiliary Building Exhaust Unit B	<50	N/A	0.5	N/A
AX209	Auxiliary Building Exhaust Unit A	<50	N/A	0.5	N/A
AX210	Fuel Handling Building Exhaust Unit B	<50	N/A	<0.2	N/A
AX211	Fuel Handling Building Exhaust Unit A	<50	N/A	<0.2	N/A
AX212	Decay Heat Surge Tank and Substation Area	<2.5	<1,000/<10,000	<0.2	<1,000/<10,000
AX213	Unit Substations and Access Area	<2.5	<1,000/<10,000	<0.2 - 1.5	<1,000/<10,000
AX214	Decontamination Facility	<2.5	<1,000/<10,000	<0.2 - 2	1,000/<10,000
AX215	Fuel Handling Building Supply Unit	<2.5	N/A	<0.2	N/A
AX216	Auxiliary Building Supply Unit	<2.5	N/A	<0.2	N/A
AX217	Access Area	<2.5	<1,000/<10,000	<0.2	<1,000/<10,000
AX218	Concentrated Waste Storage Tank Room	<500	<50,000/<50,000	7 - 32	2,000/<50,000
AX219	Instrument Racks and Atmospheric Monitor Area	<2.5	<1,000/<10,000	<0.2	<1,000/<10,000

See footnotes at end of table.

Table 4.1 (cont'd)

Cubicle Number	Area Description	Specific Decontamination Goals: General Area Dose Rate (mR/hr)	Specific Decontamination Goals: Surface Contamination <7 ft/Overheads (dpm/100 cm^2)	PDMS Radiological Conditions[b]: General Area Dose Rate (mR/hr)	PDMS Radiological Conditions[b]: Surface Contamination <7 ft/Overheads (dpm/100 cm^2)
AX220	Caustic Liquids Mixing Area	<500	<1,000/<10,000	1.5	1,000/<10,000
AX221	Caustic Liquids Mixing Area Corridor	<500	<1,000/<10,000	<0.2 - 1	<1,000/<10,000
AX222	South Stairwell	<2.5	<1,000/<10,000	<0.2	<1,000/<10,000
AX223	Air Handling Units General Area	<2.5	<5,000/<10,000	<0.2 - 8	<1,000/<10,000
AX301	Elevator Shaft and Elevator Machine Room	<2.5	<1,000/<10,000	<0.2	<1,000/<10,000
AX302	North Stairwell	<2.5	<1,000/<10,000	<0.2	<1,000/<10,000
AX303	Elevator and Stairwell Access	<2.5	<1,000/<10,000	<0.2	<1,000/<10,000
AX401	Roof	<2.5	<1,000/N/A	<0.2	<1,000/N/A
AX402	Cooling Waste Surge Tanks Room	<500	<50,000/<50,000	<0.2	<1,000/<50,000
AX403	Damper Room	<500	<50,000/<50,000	<0.2	<1,000/<50,000
AX501	Reactor Building Spray Pump 1A Room	<25	<5,000/<50,000	1 - 16	15,000/<50,000
AX502	Reactor Building Spray Pump 1B Room	<25	<950,000[c]/<50,000	78 - 140	950,000/<50,000
AX503	Decay Heat Removal Cooler and Pump 1A Room	<25	<50,000/<50,000	1 - 60	50,000/<50,000
AX504	Decay Heat Removal Cooler and Pump 1B Room	<25	<50,000/<50,000	2 - 5	60,000/<50,000

See footnotes at end of table.

Table 4.1 (cont'd)

Cubicle Number	Area Description	Specific Decontamination Goals: General Area Dose Rate (mR/hr)	Specific Decontamination Goals: Surface Contamination <7 ft/Overheads (dpm/100 cm^2)	PDMS Radiological Conditions[b]: General Area Dose Rate (mR/hr)	PDMS Radiological Conditions[b]: Surface Contamination <7 ft/Overheads (dpm/100 cm^2)
FH001	Makeup Suction Valve Room	<500	<200,000[c]/<50,000	5 - 180	180,000/<50,000
FH002	Access Corridor	<2.5	<1,000/<10,000	<0.2 - 5	<1,000/<10,000
FH003a	Makeup Discharge Valve Room	<1,000	<50,000/<100,000	40 - 200	100,000/<100,000
FH003b	Makeup Discharge Valve Room	<1,000	<50,000/<100,000	30 - 500	40,000/<100,000
FH004	Westinghouse Valve Room	<500	<50,000/<50,000	0.2 - 25	75,000/<50,000
FH005	Mini Decay Heat Vault	<500	<50,000/<50,000	1.5	3,000/<50,000
FH006	Decay Heat Service Coolers Area	<500	<1,000/<10,000	0.2 - 3	<1,000/<10,000
FH007	Neutralizer and Reclaimed Boric Acid Access Area	<500	<1,000/<10,000	0.2 - 8	1,500/<10,000
FH008	Neutralizer Tanks Pumps Room	<500	<50,000/<50,000	3 - 10	15,000/<50,000
FH009	Neutralizer Tanks Room	<500	<50,000/<50,000	20 - 250	225,000/<50,000
FH010	Reclaimed Boric Acid Tank Room	<500	<50,000/<50,000	2	3,000/<50,000
FH011	Reclaimed Boric Acid Pump Room	<500	<50,000/<50,000	5 - 30	150,000/<50,000
FH012	Neutralizer Tanks Filters Room	<500	<50,000/<50,000	<0.2	2,000/<50,000
FH013	Oil Drum Storage Area	<500	<1,000/<10,000	80	<1,000/<10,000

See footnotes at end of table.

Table 4.1 (cont'd)

Cubicle Number	Area Description	Specific Decontamination Goals: General Area Dose Rate (mR/hr)	Specific Decontamination Goals: Surface Contamination <7 ft/Overheads (dpm/100 cm^2)	PDMS Radiological Conditions[b]: General Area Dose Rate (mR/hr)	PDMS Radiological Conditions[b]: Surface Contamination <7 ft/Overheads (dpm/100 cm^2)
FH014	Annulus	<500	<250,000[c]/<50,000	16 - 120	<230,000/<50,000
FH101	Makeup and Purification Valve Room	<500	<50,000/<50,000	80 - 1,000	220,000/<50,000
FH102	East Corridor	<2.5	<1,000/<10,000	0.4 - 1.6	<1,000/<10,000
FH103	Sample Room	<50	<50,000/<50,000	0.3 - 4	3,000/<50,000
FH104	West Corridor	<2.5	<1,000/<10,000	<0.2	<1,000/<10,000
FH105	Model Room	<2.5	<1,000/<10,000	0.2 - 120	<1,000/<10,000
FH106	Monitor Tanks and Sample Sink Area	<2.5	<1,000/<10,000	0.6 - 1.2	2,000/<10,000
FH107	Trash Compactor Area	<2.5	<1,000/<10,000	0.5	1,000/<10,000
FH108	Truck Bay	<2.5	<1,000/<10,000	<0.2	<1,000/<10,000
FH109	Spent Fuel Pool A	<2.5	<1,000/<10,000	See FH301	See FH301
FH110	Submerged Demineralizer System Spent Fuel Pool	<2.5	<1,000/<10,000	(e)	(e)
FH111	Fuel Cask Storage	<1,000	<1,000/<10,000	(e)	(e)
FH112	Annulus	<100	<50,000/<50,000	1 - 25	4,000/<50,000
FH201	East Corridor	<2.5	<1,000/<10,000	0.4 - 8	3,000/<10,000
FH202	West Corridor	<2.5	<1,000/<10,000	0.2	3,000/<10,000

See footnotes at end of table.

Table 4.1 (cont'd)

Cubicle Number	Area Description	Specific Decontamination Goals		PDMS Radiological Conditions[b]	
		General Area Dose Rate (mR/hr)	Surface Contamination <7 ft/Overheads (dpm/100 cm²)	General Area Dose Rate (mR/hr)	Surface Contamination <7 ft/Overheads (dpm/100 cm²)
FH203	Surge Tank Area	<500	<50,000/<50,000	(d)	(d)
FH204	Standby Pressure Control Area	<500	<1,000/<10,000	<0.2 - 2	5,000/<10,000
FH205	Annulus	<100	<50,000/N/A	<0.2 - 100	5,000/N/A
FH301	Upper Spent Fuel Pool A Area				
	(Below waterline, from 308-ft to 328-ft elevation)	<2.5	2,500,000[c]/N/A	(e)	2,500,000[f]/N/A
	(Above waterline, from 328-ft to 348-ft elevation)	<2.5	<350,000[c]/N/A	0.2 - 6	350,000/N/A
	(Above fuel pool)	<2.5	<1,000/<10,000	0.6 - 12	<1,000/<10,000
FH302	Submerged Demineralizer System Operating Area	<2.5	<1,000/<10,000	0.2 - 3.2	<1,000/<10,000
FH303	Upper Standby Pressure Control Area	<2.5	<1,000/<10,000	<0.2 - 1.6	<1,000/<10,000
FH304	Annulus	<500	<50,000/<50,000	0.3 - 1.2	7,000/<50,000
FH305	Spent Fuel Pool Access Area	<2.5	<1,000/<10,000	0.6 - 12	<1,000/<10,000

(a) Data was obtained from Table 5.3-2 of GPU 1991a.
(b) The radiological conditions in this table reflect the most current survey data existing in January 1991. Surface contamination levels in the overheads are assumed to be at PDMS decontamination goals unless otherwise noted.
(c) These contamination values reflect the existing condition. Further cubicle decontamination and cleanup will be evaluated on a case-by-case basis; however, it is not possible to set a reasonable goal at this time.
(d) This area is currently inaccessible; no meaningful data exist at this time.
(e) This area is currently under water; no meaningful data exist at this time.
(f) The surface contamination level below the water line in spent fuel pool "A" (FH301) was estimated by surveying the "bathtub ring" immediately above the current water level.

Notes: N/A - not applicable. dpm - disintegrations per minute.

Table 4.2 Surface Contamination in the Auxiliary and Fuel-Handling Building[a]

Cubicle Number	Area Description	Principal Isotopes[b]	Curies[c]
AX001	Reactor Building Emergency Cooling Booster Pumps Area	A	2.61E-4
AX002	Access Corridor	A	2.90E-4
AX002a	Nitrogen Piping System	A	(d)
AX003	Access Area	A	1.35E-4
AX004	Seal Injection Valve Room	C	9.41E-4
AX005	Makeup and Purification Pump 1C Room	B	7.54E-4
AX006	Makeup and Purification Pump 1B Room	B	7.54E-4
AX007	Makeup and Purification Pump 1A Room	B	7.54E-4
AX008	Spent Resin Storage Tank 1B Room	A	6.82E-3
AX009	Spent Resin Storage Tank 1A Room	A	3.92E-3
AX010	Spent Resin Transfer Pump Room	A	7.33E-2
AX011	Auxiliary Building Sump Tank Pumps and Valve Room	B	8.39E-5
AX012	Auxiliary Building Sump and Tank Room	A	3.33E-4
AX013	Evaporator Condensate Tanks, Pumps, and Demineralizer Room	A	1.08E-4
AX014	Reactor Coolant Evaporator Room	A	3.73E-4
AX015a	Cleanup Filters Room	A	2.48E-4
AX015b	Cleanup After Filters Room	A	2.48E-4
AX016	Cleanup Demineralizer 2A Room	A	4.23E-4
AX017	Cleanup Demineralizer 2B Room	A	4.23E-4

See footnotes at end of table.

Table 4.2 (cont'd)

Cubicle Number	Area Description	Principal Isotopes[b]	Curies[c]
AX018	Waste Transfer Pumps Room	A	2.99E-4
AX019	Waste Disposal Liquid Valve Room	A	6.94E-4
AX020	Reactor Coolant Bleed Holdup Tanks 1B and 1C Room	B	7.28E-3
AX021	Reactor Coolant Bleed Holdup Tank 1A Room	A	1.20E-3
AX022	North Stairwell	A	3.85E-6
AX023	Elevator Shaft (inside cab)	A	1.08E-4
AX024	Auxiliary Building Sump Filters Room	B	2.27E-4
AX025	Area Between Service, Control, and Reactor Building	A	9.36E-5
AX026	Seal Injection Filters Room	C	4.85E-5
AX027	South Stairwell	A	3.89E-6
AX101	Radwaste Disposal Control Panel Area	A	8.38E-5
AX102	Reactor Building Sump Pumps Filters Room	A	1.82E-4
AX103	Motor Control Center 2-11EB Room	A	4.16E-5
AX104	Motor Control Center 2-21EB Room	A	4.16E-5
AX105	Substation 2-11E Room	A	9.58E-5
AX106	Substation 2-21E Room	A	1.08E-4
AX107	Motor Control Center 2-11EA Room	A	1.16E-4
AX108	Motor Control Center 2-21EA Room	A	8.86E-5

See footnotes at end of table.

Table 4.2 (cont'd)

Cubicle Number	Area Description	Principal Isotopes[b]	Curies[c]
AX109	Nuclear Services Coolers and Pumps Area	A	2.36E-4
AX110	Intermediate Coolers Area	A	1.32E-4
AX111	Intermediate Cooling Pumps and Filters Room	A	1.11E-4
AX112	Seal Return Coolers and Filter Room	B	3.60E-4
AX113	Waste Gas Analyzer Room	A	7.20E-4
AX114	Makeup and Purification Demineralizer 1A Room	A	4.23E-5
AX115	Makeup and Purification Demineralizer 1B Room	A	5.23E-5
AX116	Makeup Tank Room	B	3.80E-4
AX117	Makeup and Purification Filters Room	B	2.75E-1
AX118	Spent Fuel Coolers and Pumps Area	A	1.29E-4
AX119	Spent Fuel Demineralizer Room	A	1.20E-5
AX120	Spent Fuel Filters Room	A	4.51E-6
AX121	Elevator Shaft (inside cab)	A	6.61E-6
AX122	North Stairwell	A	1.77E-6
AX123	Access Area	A	3.33E-4
AX124	Concentrated Liquid Waste Pump Room	A	1.96E-4
AX125	Waste Gas Decay Tank 1B Room	A	5.79E-4
AX126	Waste Gas Filter Room	A	1.50E-4
AX127	Waste Gas Decay Tank 1A Room	A	5.79E-4

See footnotes at end of table.

Table 4.2 (cont'd)

Cubicle Number	Area Description	Principal Isotopes[b]	Curies[c]
AX128	Valve and Instrument Room	A	1.60E-4
AX129	Deborating Demineralizer 1B Room	A	1.81E-4
AX130	Deborating Demineralizer 1A Room	A	1.88E-4
AX131	Miscellaneous Waste Holdup Tank Room	A	4.62E-4
AX132	Corridor Between Unit 1 and Unit 2	A	(d)
AX133	South Stairwell	A	6.05E-5
AX134	Miscellaneous Waste Tank Pumps Room	A	3.56E-4
AX135	Radwaste Disposal Control Panels	A	(d)
AX201	North Stairwell	A	3.59E-5
AX202	Elevator Shaft	A	5.85E-6
AX203	4160V Switchgear 2-1E Room	A	1.47E-4
AX204	4160V Switchgear 2-2E Room	A	1.53E-4
AX205	Reactor Building Purge Air Supply and Hydrogen Control Exchange Area	A	2.69E-4
AX206	Reactor Building Purge Air Exhaust Unit B	A	(d)
AX207	Reactor Building Purge Air Exhaust Unit A	A	(d)
AX208	Auxiliary Building Exhaust Unit B	A	(d)
AX209	Auxiliary Building Exhaust Unit A	A	(d)
AX210	Fuel Handling Building Exhaust Unit B	A	(d)
AX211	Fuel Handling Building Exhaust Unit A	A	(d)

See footnotes at end of table.

Table 4.2 (cont'd)

Cubicle Number	Area Description	Principal Isotopes[b]	Curies[c]
AX212	Decay Heat Surge Tank and Substation Area	A	4.22E-4
AX213	Unit Substations and Access Area	A	3.33E-4
AX214	Decontamination Facility	A	(d)
AX215	Fuel Handling Building Supply Unit	A	(d)
AX216	Auxiliary Building Supply Unit	A	(d)
AX217	Access Area	A	3.69E-4
AX218	Concentrated Waste Storage Tank Room	A	1.87E-4
AX219	Instrument Racks and Atmospheric Monitor Area	A	1.11E-4
AX220	Caustic Liquids Mixing Area	A	6.29E-5
AX221	Caustic Liquids Mixing Area Corridor	A	1.68E-4
AX222	South Stairwell	A	3.14E-5
AX223	Air Handling Units General Area	A	5.28E-4
AX301	Elevator Shaft and Elevator Machine Room	A	(d)
AX302	North Stairwell	A	(d)
AX303	Elevator and Stairwell Access	A	(d)
AX401	Roof	A	(d)
AX402	Cooling Water Surge Tanks Room	A	(d)
AX403	Damper Room	A	(d)
AX501	Reactor Building Spray Pump 1A Room	B	3.93E-5
AX502	Reactor Building Spray Pump 1B Room	A	6.25E-3

See footnotes at end of table.

Table 4.2 (cont'd)

Cubicle Number	Area Description	Principal Isotopes[b]	Curies[c]
AX503	Decay Heat Removal Cooler and Pump 1A Room	A	1.45E-3
AX504	Decay Heat Removal Cooler and Pump 1B Room	A	6.73E-4
FH001	Makeup Suction Valve Room	B	3.11E-3
FH002	Access Corridor	B	1.10E-4
FH003a	Makeup Discharge Valve Room	B	3.84E-4
FH003b	Makeup Discharge Valve Room	B	6.27E-4
FH004	Westinghouse Valve Room	A	1.22E-3
FH005	Mini Decay Heat Vault	A	9.13E-5
FH006	Decay Heat Service Coolers Area	A	2.73E-4
FH007	Neutralizer and Reclaimed Boric Acid Access Area	A	3.03E-4
FH008	Neutralizer Tanks Pumps Room	B	6.03E-4
FH009	Neutralizer Tanks Room	B	4.96E-4
FH010	Reclaimed Boric Acid Tank Room	A	1.69E-5
FH011	Reclaimed Boric Acid Pump Room	A	6.45E-5
FH012	Neutralizer Tanks Filters Room	A	9.20E-5
FH013	Oil Drum Storage Area	A	2.73E-5
FH014	Annulus	A	2.90E-3
FH101	Makeup and Purification Valve Room	A	5.09E-4
FH102	East Corridor	A	1.90E-4
FH103	Sample Room	A	1.85E-4

See footnotes at end of table.

Table 4.2 (cont'd)

Cubicle Number	Area Description	Principal Isotopes[b]	Curies[c]
FH104	West Corridor	A	1.72E-4
FH105	Model Room	A	3.06E-4
FH106	Monitor Tanks and Sample Sink Area	A	1.00E-4
FH107	Trash Compactor Area	A	8.62E-5
FH108	Truck Bay	A	(d)
FH109	Spent Fuel Pool A	A	1.20E-1
FH110	Submerged Demineralizer System Spent Fuel Pool	A	4.49E-5
FH111	Fuel Cask Storage	A	8.36E-7
FH112	Annulus	B	1.72E-5
FH201	East Corridor	A	1.88E-4
FH202	West Corridor	A	1.36E-4
FH203	Surge Tank Area	A	1.08E-4
FH204	Standby Pressure Control Area	A	1.63E-4
FH205	Annulus	B	1.43E-4
FH301	Upper Spent Fuel Pool A Area	A	1.76E-4
FH302	Submerged Demineralizer System Operating Area	A	2.01E-4
FH303	Upper Standby Pressure Control Area	A	4.08E-4

See footnotes at end of table.

Table 4.2 (cont'd)

Cubicle Number	Area Description	Principal Isotopes[b]	Curies[c]
FH304	Annulus	B	1.03E-4
FH305	Spent Fuel Pool Access Area	A	4.99E-4

(a) Data were obtained from Table 5.3-5 of GPU 1991d.
(b) The principal isotopes and their relative distribution are defined below:

A		B		C	
Isotope	Percent	Isotope	Percent	Isotope	Percent
Sr-90	7.44	Sr-90	39.4	Sr-90	82.5
Cs-137	92.6	Cs-137	60.9	Cs-137	17.5
		Pu-238	0.0002	Pu-238	0.00006
		Pu-239	0.0023	Pu-239	0.0007
		Pu-240	0.0006	Pu-240	0.0002
		Pu-241	0.0288	Pu-241	0.0084
		Am-141	0.0004	Am-141	0.0001

(c) These are calculated values based on the specific decontamination goals given in Tables 5.2-2 and 5.2-3 of GPU 1991a.
(d) No estimate available.

(GPU 1990c, SAR 7.2.4.2) to performing a final radiological assessment before entry into PDMS. The NRC staff will perform confirmatory measurements.

4.2.2 Reactor Building

The primary decontamination effort has been focused on the reactor building and the reactor vessel itself. Extensive decontamination activities were undertaken in the containment building to reduce dose rates to facilitate fuel removal. The accident-generated water that had collected in the basement has been removed to the extent practicable and processed to remove most of the radioactivity, and is awaiting final disposition (see Section 4.1). Regions of the reactor building were decontaminated by high- and low-pressure flushing, hydroblasting, scabbling of concrete, and removal of material and equipment. In addition, highly contaminated areas were shielded to reduce dose rates in areas necessary for personnel access and work. The major sources of contamination remaining in the reactor building include core debris in crevices and plated on surfaces in the reactor vessel and reactor coolant system (discussed in Section 4.3) and sludge, core debris, and fission product material absorbed and plated onto surfaces in the containment building basement.

Estimates of the amount of radioactive material inside the reactor building are given in Section 2.2 of PEIS Supplement 3 (NRC 1989a). Large amounts of cobalt-60 may be present within the metal of the reactor vessel and not easily available for disposal. Strontium-90/yttrium-90 and cesium-137 are the major radionuclides that could potentially be released from the reactor building (see GPU 1990c, SAR 8.1.2). Table 4.3 provides the estimated quantities of these two isotopes in the various locations in the reactor building.

Table 4.3 Estimated Quantity of Cesium-137 and Strontium-90[a]

Location	Cesium-137 (curies)	Strontium-90 (curies)
Concrete block wall	19,000[b]	910[c]
Sludge on basement floor	350[d]	400[d]
D-rings	17,000[d]	830[d]
Floors, walls, overhead structures	7,000[d]	300[d]
Total	43,350	2,440

(a) Data were obtained from Table 4.3 of PEIS Supplement 3 (NRC 1989a).
(b) An estimated 20,000 curies of cesium-137 is present in the concrete block wall (GPU 1988b). However, since this estimate was made, approximately 7 percent of the activity in the concrete block wall has been leached from the structure, leaving an estimated 19,000 curies.
(c) A ratio of 21:1 (based on leach rate tests [ANS 1988]) was applied to the cesium-137 curie estimate before leaching (20,000 curies), and a conservative 43 curies of strontium-90 (GPU 1988d) was assumed to have been removed during leaching of the concrete block wall.
(d) GPU 1988b.

4.2.3 Other Contaminated Facilities

Ten areas (facilities) outside the AFHB and the reactor building were also contaminated during the accident:

- diesel generator "A", elevation 281 feet
- service building, elevation 281 feet
- tendon access gallery
- service building, elevation 305 feet
- turbine building M-20 area
- turbine building M-20 area sump

- turbine building, elevation 281 feet
- containment air control envelope building
- processed water storage tank sump
- borated water storage tank area.

The licensee considers four of these areas to be decontaminated to their decontamination endpoint goals: diesel generator "A", elevation 281 feet; the tendon access gallery; the turbine building M-20 area; and the borated water storage tank area. On the basis of measurements, the licensee has estimated that there is less than 0.00038 curie of loose radioactive material available for release from 7 of the 10 contaminated areas (Table 4.4). The licensee does not have final data for the service building, elevation 305 feet; the turbine building, elevation 281 feet; or the containment air control envelope building. However, the licensee has committed (GPU 1991a, SAR Section 5.3.2) to survey those facilities and provide the information in the PDMS SAR before entry into PDMS. The purpose of this survey is to establish a radiological baseline for the facility prior to entry into PDMS. Additional radioactive material is present in closed systems within the listed areas that is considered to be contained and not available for release. This material is not indicated in Tables 4.2, 4.3, or 4.4.

The licensee has committed (GPU 1991a, SAR 5.3.2) to complete the plant radiation and contamination surveys before entry into PDMS. Proposed PDMS license condition 2.F requires the licensee to submit the results of the completed radiological survey to the NRC prior to entry into PDMS. The NRC staff will perform confirmatory measurements.

4.3 Removal of the Fuel

After extensive evaluation and study, the reactor head and upper plenum assembly were removed to permit access to the fuel in the reactor core. The internals indexing fixture was placed on top of the reactor vessel, and a rotating, shielded work platform was placed on top of it. Working from this platform and using vacuum systems and long-handled tools, workers removed from the reactor most of the fuel (estimated by the licensee to be more than 99 percent [GPU 1990c, SAR 4.3.1]). For the purposes of this document, fuel is defined as UO_2 (uranium dioxide). Core debris is defined as a mixture of fuel, structural material, and absorber material that resulted from the accident at TMI-2 and the subsequent cleanup. This material was placed in canisters and shipped off site. The quantity of fuel remaining in the TMI-2 facility was measured using a variety of radiation measurement techniques, such as gamma dose rate and spectroscopy, neutron detection and activation/interrogation, alpha particle detection, direct sampling and analysis, and visual inspection to determine the amount and location of remaining core debris. These techniques are described in detail in the "Defueling Completion Report" (GPU 1989h, 1989i, 1989j, 1990d, 1990e, and 1990f). Estimates based on measurements, sample analyses, and visual observations indicate that no more than 159 pounds (72.4 kilograms) of residual fuel (i.e., UO_2) remains in the reactor building, excluding the reactor vessel and reactor coolant system; less than 199 pounds (90.2 kilograms) in the reactor coolant system; and less than 1339 pounds (609 kilograms) in the reactor vessel. Detailed estimates of the quantity and location of residual fuel (i.e., UO_2) in the reactor building and reactor coolant system (including the reactor vessel) are given in Tables 4.5 and 4.6, respectively.

Table 4.4 Surface Contamination - Other Buildings[a]

Cubicle Number	Area Description	Curies[b]
DG000	Diesel Generator "A", Elevation 281 feet	3.12E-4
SB000	Service Building, Elevation 281 feet	7.36E-7
SB500	Tendon Access Gallery	6.39E-5
-	Service Building, Elevation 305 feet	ND[c]
SB002	M-20 Area	3.34E-6
SB002	M-20 Area Sump	7.36E-7
-	Turbine Building, Elevation 281 feet	ND
-	Containment Air Control Building	ND
RA101	Processed Water Storage Tank Sump	3.01E-7
RA104	Borated Water Storage Tank Area	5.52E-7

(a) Data were obtained from Table 5.3-6 of GPU 1990c.
(b) The principal isotopes and their relative distribution are defined below:

Isotope	A Percent
Sr-90	7.4
Cs-137	92.6

(c) ND - no data.

Table 4.5 Final Residual Fuel Inventory Estimates by Location in the Reactor Building[a]

Location	Pounds	Fuel Quantity (i.e., UO_2) (kilograms)
Reactor Vessel Head Assembly	2.9	1.3
Reactor Head Plenum Assembly	4.6	2.1
Fuel Transfer Canal	41.6	18.9
Core Flood System	9.7	4.9[b]
Incore Instrument Guide Tubes in A D-Ring	46.2	21.0
Upper Endfitting Storage Area	13.0	5.9[c]
Reactor Coolant Drain Tank	0.2	0.1
Letdown Coolers	8.1	3.7[c]
Reactor Building Basement and Sump	2.9	1.3
Tool Decontamination Facility	0.2	0.1
Defueling Water Cleanup System	8.1	3.7
Defueling Tool Rack	1.3	0.6[d]
Temporary Reactor Vessel Filtration System	9.7	4.4
Reactor Building Drains	9.7	4.4
Total	<159.3	<72.4

(a) Data were obtained from Table 4.3-1 of GPU 1991e.
(b) One fuel container containing three upper endfittings will remain stored in the deep end of the fuel transfer canal and rather than in the endfitting storage container area as stated in GPU 1990e. There are 1.4 kilograms of residual fuel in this container. The fuel values in this table have been changed accordingly (GPU 1991a).
(c) Minimum detectable limit (MDL)
(d) This value was changed from that given in GPU 1990e by GPUNC Calculation 4240-3232-90-066, "Reactor Fuel Quantity on Defueling Tools," Revision 0, dated October 1990.

Table 4.6. Final Residual Fuel Inventory Estimates by Location in the Reactor Coolant System and the Reactor Vessel[a]

Location	Pounds	Fuel Quantity (i.e., UO_2) (kilograms)
Reactor Vessel	1339	608.8
Ex-vessel Reactor Coolant System		
Pressurizer (including surge line)	1.1	0.5
Once-Through Steam Generator A Side		
Upper Tube Sheet	3.1	1.4
Tube Bundle	3.7	1.7
Lower Head and J-Legs	9.7	4.0
Hot Leg	2.0	0.9
Cold Legs	15.8	7.2
Core Flood Line	1.3	0.6
Once-Through Steam Generator B Side		
Upper Tube Sheet	79.2	36.0
Tube Bundle	20.0	9.1
Lower Head and J-Legs	22.2	10.1
Hot Leg	4.0	1.8
Cold Legs	9.2	4.2
Core Flood Line	0.9	0.4
Reactor Coolant Pumps	13.6	6.2
RCS Surface Films	10.1	4.6
Decay Heat Drop Line	3.3	1.5
Subtotal	199.2	90.2
Total	1538	699

(a) Data were obtained from Table 4.3-1 of GPU 1991e.

Table 4.7 Final Residual Fuel Inventory Estimates by Location in the Auxiliary (SNM) and Fuel-Handling Building[a]

Cubicle Number[b]	Area Description	Pounds	Fuel Quantity (i.e., UO_2) (kilograms)
AX004	Seal Injection Valve Room	0.07	0.03
AX006	Makeup and Purification Pump 1B Room	0.2	0.07[c]
AX007	Makeup and Purification Pump 1A Room	0.5	0.23[c]
AX012	Auxiliary Building Sump Tank Room	0.2	0.10
AX015a/b	Cleanup Filters Room	0.2	0.10[c]
AX019	Waste Disposal Liquid Valve Room	0.02	0.01
AX020	Reactor Coolant Bleed Holdup Tanks 1B and 1C Room	7.7	3.5
AX021	Reactor Coolant Bleed Holdup Tank 1A Room	0.7	0.31
AX024	Auxiliary Building Sump Filters Room	0.04	0.02
AX102 AX131 AX134	Reactor Building Sump Pump Filters Room Miscellaneous Waste Holdup Tank Room Miscellaneous Waste Tank Pumps Room	0.22	0.10
AX112	Seal Return Coolers and Filter Room	0.66	0.30[c]
AX114	Makeup and Purification Demineralizer 1A Room	2.3	1.06
AX115	Makeup and Purification Demineralizer 1B Room	0.3	0.13
AX116	Makeup Tank Room	0.7	0.31
AX117	Makeup and Purification Filters Room	0.13	0.06

See footnotes at end of table.

Table 4.7 (cont'd)

Cubicle Number[b]	Area Description	Pounds	Fuel Quantity (i.e., UO_2) (kilograms)
AX128	Instrument and Valve Room	0.02	0.01
AX218	Concentrated Waste Storage Tank Room	0.02	0.01
AX501	Reactor Building Spray Pump 1A Room	0.02	0.01
AX502	Reactor Building Spray Pump 1B Room	0.02	0.01
AX503	Decay Heat Removal Cooler and Pump 1A Room	0.02	0.01
AX504	Decay Heat Removal Cooler and Pump 1B Room	0.02	0.01
FH001	Makeup Suction Valve Room	1.0	0.46
FH002 FH004 FH014	Access Corridor Westinghouse Value Room Annulus	0.35	0.16
FH003a	Makeup Discharge Valve Room	0.02	0.01
FH003b	Makeup Discharge Valve Room	0.2	0.10
FH101	Makeup and Purification Valve Room	0.7	0.32
FH109	Spent Fuel Pool A	8.3	3.8
FH112	Annulus	0.02	0.01
	Embedded Valves and Piping (Makeup System)	0.4	0.17
	Embedded Valves and Piping (Waste Disposal Liquid System)	0.9	0.04
Total SNM Inventory		25.3	11.46

(a) Data were obtained from Table 4.3-2 of GPU 1991e.
(b) All locations not listed contain less than 0.011 pounds (0.005 kilograms) UO_2 per area.
(c) Minimum detectable limit (MDL).

During the accident, core debris was transported by the cooling water through the reactor coolant system and into the AFHB, primarily through the make-up and purification system and the seal injection system. Some of this core debris may have further relocated into other systems as part of the post-accident water processing and cleanup activities. Core debris has been removed from these systems. The licensee estimates (GPU 1991e, PDMS SAR Table 4.3-2) that less than 96 pounds (12 kilograms) of fuel (i.e., UO_2) in the form of plated material on the interior surfaces of piping and as particulate material in dead legs, tanks, and so forth, remain in the AFHB. Of the 137 cubicles in the AFHB, 105 have been determined to contain less than 0.005 kilograms per area of residual fuel (i.e., UO_2) (PDMS SAR Table 4.3-2). The estimated quantity of residual fuel (i.e., UO_2) in the cubicles ranges from the minimum detectable limit to almost 9 pounds (3.8 kilograms), with less than 26 pounds (12 kilograms) total. The estimated quantity and location of residual fuel (i.e., UO_2) in the AFHB are given in Table 4.7.

The safe fuel mass limit (SFML) and the corresponding criticality analysis for the TMI-2 facility are discussed in Section 5.1 of this report.

5 PREREQUISITES FOR POST-DEFUELING MONITORED STORAGE

The basic criterion for reviewing the proposed PDMS is compliance with Commission regulations regarding radioactive releases and maintenance of PDMS environmental protection systems, components, and structures. The first six prerequisites for placing the TMI-2 facility into PDMS listed below are based on NRC staff evaluation of information provided by the licensee in the PDMS SAR 1.1.2.1 (GPU 1989e). The seventh prerequisite below was identified by the NRC staff.

1. Defueling of the facility to the extent reasonably achievable and to such a degree that a nuclear criticality is precluded.

2. Shipment off site of all fuel and core debris that have been removed from the reactor and associated systems.

3. Removal of water, to the extent practicable, from the reactor coolant system; draining of the fuel transfer canal; and isolation of the fuel transfer tubes. To the extent that the spent fuel pools are needed to store the accident-generated water before disposal, water may remain in these pools after the start of PDMS.

4. A reduction of the potential for release of radioactive material from the facility within the design objectives specified in 10 CFR Part 50, Appendix I, for offsite dose consequences; and a reduction of the potential for instantaneous concentrations of released material within the limits specified in 10 CFR Part 20.

5. Shipment off site or packaging and staging for shipment of remaining radioactive waste from the major TMI-2 decontamination activities.

6. Determination and reduction of radiation levels within the facility so that necessary and required plant monitoring, maintenance, and inspections can be performed.

7. Definition and establishment of a surveillance program for PDMS environmental protection systems to ensure public health and safety.

Each of these prerequisites is discussed below.

5.1 Reduction of Potential for Accidental Criticality

Defueling of the TMI-2 reactor and removal of fuel from the reactor coolant system and associated components have been completed to the extent reasonably achievable. The licensee defined defueling to the extent reasonably achievable (GPU 1990d) as implementation of the following objectives:

1. All fuel will be removed that is reasonably accessible within technically practical methods;

2. Sufficient fuel will be removed to ensure the absence of a potential criticality regardless of degree of accessibility and level of difficulty; and

3. Residual fuel that is not reasonably accessible by practical means and has been determined to have no significant impact on public health and safety may remain.

The NRC staff concurred with the licensee's definition of "defueled to the extent reasonably achievable." In assessing technical accessibility and practicality, the NRC staff evaluated whether new technology would have to be developed to remove additional core debris. The NRC staff also evaluated the additional costs both in dollars and in person-rem of radiation exposure to remove incremental quantities of core debris. The average unit cost per kilogram of core debris removed during defueling, based on information given in the licensee's DCR (GPU 1990e), was $1900 and 0.02 person-rem. The staff determined, based on information given in the licensee's DCR (GPU 1990e), that to remove significant quantities of additional core debris, it would be an order of magnitude more expensive monetarily and two orders of magnitude higher in personnel exposure.

Residual fuel is primarily distributed as plated material on the internal surfaces of the reactor vessel and components, reactor coolant pipes, pressurizer, steam generators, and reactor coolant pumps; as solid and particulate material in the lower portions of the reactor vessel; as particulate material in tanks, demineralizers, and dead legs in the piping systems; and as sludge in the reactor building basement and the auxiliary and fuel-handling building (AFHB) floor drains.

The licensee has provided the results of its analysis of the fuel quantities remaining in the TMI-2 facility. It submitted its initial report, "Defueling Completion Report" (GPU 1989h), to the NRC by letter dated July 5, 1989. It submitted three amendments and a final report (GPU 1989i, 1989j, 1990d, and 1990e) by February 22, 1990. The report was supplemented by a letter dated April 12, 1990 (GPU 1990f), documenting the results of the final cleanup following the lower head sampling program, and containing a revised criticality analysis that made use of the January 1990 video inspection results. The report as supplemented provides the licensee's estimate of the quantity of fuel remaining and its location, form, potential for mobility, and potential for criticality. The facility is divided into four major areas: (1) the AFHB, (2) the reactor building (outside the reactor coolant system and the reactor vessel), (3) the reactor coolant system, and (4) the reactor vessel. Individual locations within these larger areas were evaluated in detail. The fuel estimates (as shown in Tables 4.5 and 4.6) are based on accident flow models, radiation measurements, and visual observations.

The NRC staff reviewed the licensee's "Defueling Completion Report," including the quantification of the fuel and the analysis of the potential for a criticality. In addition to the independent analysis of the criticality potential at the TMI-2 facility, the NRC staff independently verified, on an audit basis, the licensee's estimates of fuel remaining at TMI-2 following the defueling effort. The staff examined the potential for the licensee to have overlooked significant quantities of fuel and concluded that the licensee had identified all locations containing significant amounts [more than 9 pounds (4 kilograms)] of residual fuel, i.e., UO_2 (NRC 1990c). The NRC staff conservatively chose 4 kilograms of fuel as significant because lesser quantities would not change the conclusions of the staff's and the licensee's criticality analyses. The licensee's measurement results and the NRC staff review were used (1) to provide information regarding the quantity of fuel remaining in the TMI-2 facility during the PDMS period proposed by the licensee and (2) to ensure that the possibility of an inadvertent criticality was precluded for both routine conditions and conditions involving the accidental shifting or movement of fuel.

The staff has also performed verification measurements of the fuel quantities remaining in selected areas of the TMI-2 facility (NRC 1991a). These measurements were performed in five cubicles of the AFHB and on four incore instrument guide tube bundles in the reactor building. The goal of the measurements was to provide an independent verification of the licensee's measurement of residual fuel.

On the basis of the results of the measurements, the staff concluded that the licensee's analysis methodology generally ensured a conservative fuel estimate. In all but one location, the licensee's estimates of the fuel quantities were higher than the NRC staff's central estimates. The single exception was incore instrument guide tube bundle 7; however, the licensee's estimate of fuel remaining in the guide tube bundle was within the range estimated by the NRC staff.

To evaluate the potential for a criticality in the remaining fuel, the NRC staff reviewed the licensee's calculation of the safe fuel mass limit (SFML) for the TMI-2 fuel, that is, the quantity of fuel below which there would be no possibility of an accidental criticality.

The licensee calculated the SFML by assuming an optimum fuel geometry (size and shape of the fuel), optimum moderation, and an infinite water reflector. These assumptions provide the optimum conditions for a criticality. The licensee assumed that the fuel was uniformly mixed. The licensee calculated the average enrichment of the three regions of the core after burnup to be 2.24 percent. Using these assumptions, the licensee calculated the SFML to be 309 pounds (140 kilograms). The staff has found the licensee's SFML of 309 pounds (140 kilograms) acceptable only for the fuel that was transported out of the reactor vessel during the March 28, 1979, accident and subsequent defueling activities. Fuel samples taken by the licensee in the reactor vessel during the defueling process have shown localized areas where the fuel enrichment exceeded 2.24 percent. On the basis of an enrichment of

2.67 weight percent (wt%) uranium-235 corresponding to the burnup of the more highly enriched fuel, the NRC staff determined the appropriate SFML to be 205 pounds (93 kilograms) of fuel (i.e., UO_2) in the reactor vessel (NRC 1990c) and 309 pounds (140 kilograms) outside the reactor vessel. The assumption is that fuel outside the reactor vessel is well mixed and the average enrichment value (2.24 percent) applies. Thus, two SFMLs apply: 205 pounds (93 kilograms) for fuel (i.e., UO_2) located in the reactor vessel and 309 pounds (140 kilograms) for fuel (i.e., UO_2) located outside the reactor vessel.

The staff compared the appropriate SFML with the quantity of fuel remaining in four areas in the TMI-2 facility (1) the AFHB, (2) the reactor building, (3) the reactor coolant system, and (4) the reactor vessel. It evaluated each area separately because each area is physically isolated from the others and there is no transport mechanism available to cause inadvertent transport of fuel and core debris from one area to the other. Each of the areas is discussed below.

5.1.1 Auxiliary and Fuel-Handling Building

The estimated quantity of fuel (i.e., UO_2) in the AFHB as given in the PDMS SAR (GPU 1991e) is less than 26 pounds (12 kilograms) (Table 4.7). The largest quantity of fuel in a single cubicle within the AFHB is 8.3 pounds (3.8 kilograms) in FH109, spent fuel pool "A". The total quantity [26 pounds (12 kilograms)] is 8.6 percent of the SFML for areas outside the reactor vessel. Verification measurements performed by the NRC staff indicate that the licensee's estimate of fuel quantities in the AFHB is conservatively high and the actual quantities of fuel (i.e., UO_2) remaining in the areas measured by the NRC staff were lower or within the range of the quantities reported by the licensee.

5.1.2 Reactor Building

The estimated quantity of fuel (i.e., UO_2) in the reactor building (not including the reactor coolant system or the reactor vessel) as given in the PDMS SAR (GPU 1991e) is less than 160 pounds (72.4 kilograms) (Table 4.5). The largest quantity of fuel in a single location is in the sections of the flow distributor removed from the reactor vessel that contained incore instrument guide tubes that were bagged and suspended in the "A" D-ring. The licensee's measurements and estimates indicate that there are 46 pounds (21 kilograms) of fuel (i.e., UO_2) in this location. Verification measurements performed by the NRC staff indicate that this is a conservatively high estimate of the remaining quantity of fuel and that the actual quantity of fuel remaining in this location is the same as, or less than, that reported by the licensee (NRC 1991a). This is 15 percent of the ex-vessel SFML of 309 pounds (140 kilograms), using 2.24 wt% uranium-235 enrichment, for areas outside the reactor vessel. If the fuel (i.e., UO_2) from all locations within the reactor building [less than 160 pounds (72.4 kilograms) not including the reactor coolant system or reactor vessel] were to be combined, the total would be only slightly over 50 percent of the ex-vessel SFML.

5.1.3 Reactor Coolant System

The estimated quantity of fuel (i.e., UO_2) in the reactor coolant system outside the reactor vessel (see Table 4.6) as given in the PDMS SAR (GPU 1991e) is 199 pounds (90.2 kilograms). This estimated is below the SFML of 309 pounds (140 kilograms) for areas outside the reactor vessel using an enrichment of 2.24 wt% uranium-235. The largest quantity of fuel (i.e., UO_2) in a single location is in the "B" once-through steam generator upper tube sheet, where an estimated maximum of approximately 80 pounds (36 kilograms) is located. This amount of residual fuel exists primarily as tightly adherent material and was not readily removable using available dynamic defueling techniques and is not readily transportable to other locations for accumulation. The remaining residual fuel is dispersed throughout the reactor coolant system in the form of finely divided, small-particle-size material and adherent film on surfaces. It should be noted that even if the remaining fuel became mobile and nonmechanistically accumulated in the lowest point of each half of the reactor coolant system, the greatest possible quantity that could accumulate would be less than the 309-pound (140-kilogram) SFML (for areas outside the reactor vessel). The separation of the two sides (A and B sides) of the reactor coolant system provides an additional margin of safety in isolating the remaining quantities of fuel.

The NRC staff reviewed the licensee's estimates, including videotapes of inspections of the inside of the reactor coolant system, and concluded that the licensee's estimates were reasonable and conservative (NRC 1990c).

5.1.4 Reactor Vessel

On the basis of the results of the licensee's post-lower head sampling program cleanup (GPU 1991e), the estimated quantity of fuel (i.e., UO_2) in the reactor vessel is 1339 pounds (608.8 kilograms) (see Table 4.6). This amount was significantly greater than the SFML for the reactor vessel of 205 pounds (93 kilograms). For this reason, the licensee performed a separate criticality safety analysis. For the analysis, the licensee used in-vessel inspections of core debris locations and quantities to develop a specific three-dimensional analytical model of the fuel in the reactor vessel rather than making worst-case assumptions regarding the geometry and reflectivity. For the purposes of the criticality analysis, fuel was specifically modeled in the reactor vessel bottom head, the lower core support assembly, and the core former area (i.e., the area between the core former baffle plates and the core barrel) in the upper core support assembly. Figure 5.1 shows the location of the reactor components. In other areas of the reactor vessel, the fuel accumulations within the vessel were considered to be too small or were separated from those areas where fuel was located by enough distance (the equivalent of approximately 12 inches [30 centimeters] of water) so as not to cause a reactivity increase as a result of neutronic interaction between the areas.

Fuel was modeled as if it extended radially in from the core barrel to the maximum distance where fuel was observed on each lower core support assembly plate. Also, the full periphery of each plate was assumed in the model to be

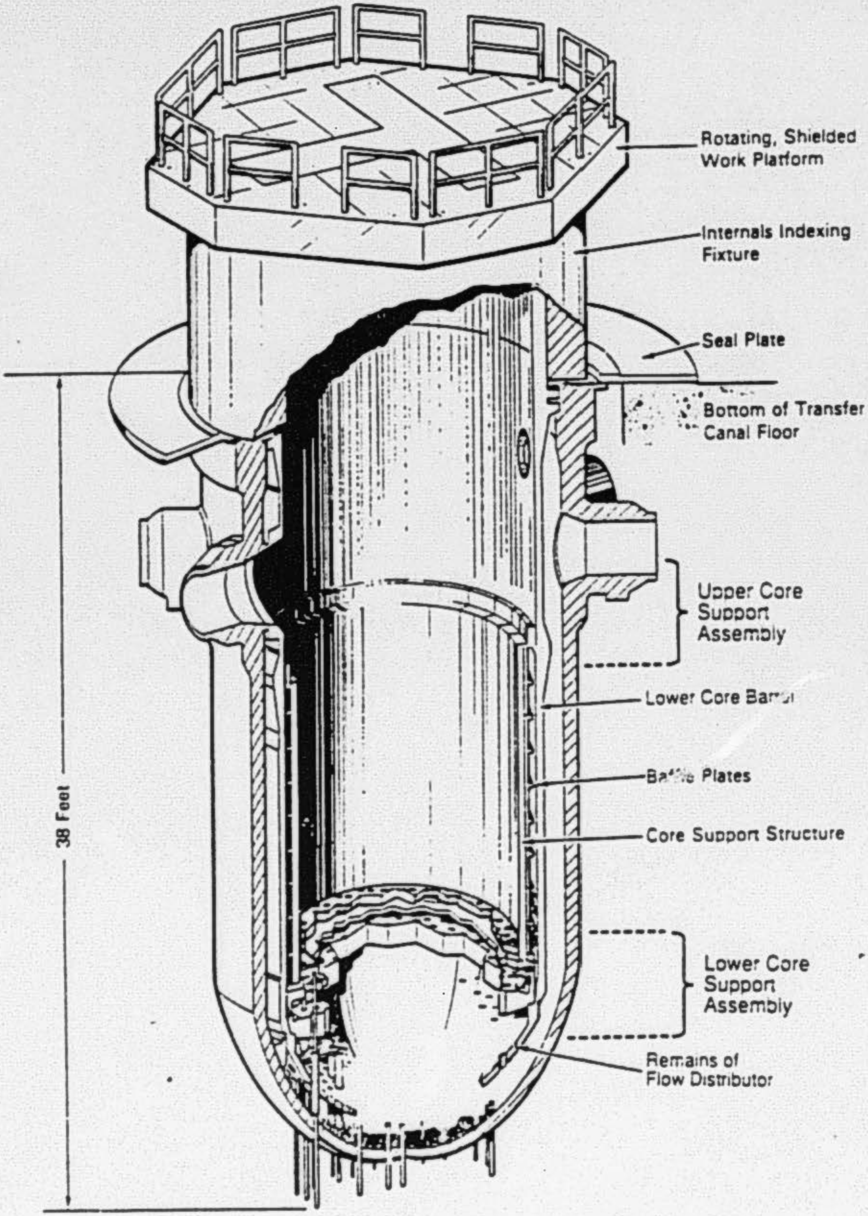

Figure 5.1 Cutaway View of TMI-2 Vessel Following Completion of Defueling Effort

loaded with fuel even though some areas did not contain fuel (this latter assumption probably had only a marginal effect on the reactivity of the system). The amount of fuel (i.e., UO_2) that was assumed in this model, 6400 pounds (2910 kilograms), was conservative when compared to that which is estimated to remain in the vessel (1339 pounds [608.8 kilograms]). Additional assumptions included a fuel enrichment of 2.96 wt% uranium-235 before burnup, with no credit allowed for the presence of structural and solid poison materials in the fuel or for moderation with unborated water. A k_{eff}* of 0.95 was used as the NRC's acceptance criterion, based on the limit allowed in Standard Technical Specifications (NRC 1991b) for spent fuel storage. The results of the licensee's analysis gave a k_{eff} of 0.945, indicating subcriticality. The NRC staff reviewed the licensee's calculations and verified that the assumptions used were highly conservative and that the model used and the calculations made were correct, thus verifying that subcriticality is ensured.

The staff reviewed the licensee's reactor vessel measurements. It selected several areas within the reactor vessel with the potential for uninventoried fuel and examined videotapes of the locations in detail. It determined that there was some additional fuel that the licensee had failed to include in the estimates; however, the amounts were insignificant (probably less than 2.2 pounds [1 kilogram]) relative to the reported quantity of fuel (i.e., UO_2) in the reactor vessel (1339 pounds [608.8 kilograms]). The staff decided to enlarge the audit sample size and examine additional areas. After additional evaluation and review, the staff concluded that the licensee had looked at and characterized the remaining fuel in all locations within the reactor vessel. The staff determined after reviewing the videotapes of the selected locations within the reactor vessel and the calculation of residual fuel based on the video inspections that the licensee's estimates of residual fuel (GPU 1990f) are reasonable and conservative.

Given the type and physical condition of the fissile material present, criticality would be possible only if this material were to accumulate in greater quantities at some location. In all areas outside the reactor vessel the SFML of 140 kilograms has not been reached, and intermixing of the fuel within these areas could only be accomplished by deliberate action. The only area where fuel might be able to accumulate is in the lower head of the reactor vessel. In its analysis of the result of a redistribution of the available material into the lower head, the licensee assumed that 772 pounds (350 kilograms) is available for redistribution (this is all loose, fine granular debris and surface films but not material that is fused to the reactor internals). The licensee's model calculations indicate that, using the most conservative assumptions, this redistribution would result in a maximum k_{eff} of 0.913. The NRC staff concluded that this is a conservative analysis, since much of this core debris is in the annular gap and thus would

* The k_{eff} can be defined as the effective multiplication factor. For a k_{eff} of 1, the reactor is just critical; neutrons are produced at the same rate as they are lost, and the system is balanced. If k_{eff} is less than 1, the fuel is subcritical, and more neutrons are consumed than are produced. In this case, the chain reaction is not self-sustaining. For a k_{eff} greater than 1, the system is said to be supercritical, and the number of neutrons and the power level increase with each generation.

not be available to be redistributed into the lower head. In addition, a criticality is precluded even under accident conditions by the absence of sufficient water and the presence of neutron poisons, including (1) impurities in the residual fuel, (2) boron in any remaining water, (3) structural material, and (4) a stable and insoluble-neutron poison, which has been added by the licensee to the bottom head of the reactor vessel following draining of the reactor coolant system (GPU 1991e, SAR 4.3.5).

In the criticality analysis, the licensee also considered the possibility of neutronic coupling of the core debris within the vessel and the core debris located in other areas of the facility. However, this possibility was discounted as the core debris in these areas is well separated from the core debris in other locations and no identifiable methods exist for transporting the fuel into or out of the vessel.

The results of the analyses indicate that there is no potential for a criticality in the fuel remaining in the TMI-2 facility during either normal or accident conditions. The conservatism built into the model and the safeguards contained in the licensee's commitment as a prerequisite to PDMS to remove water from the reactor vessel, the licensee's commitment (GPU 1991e, SAR 4.3.5) to add a neutron poison into the reactor vessel, as well as license restrictions on deliberate fuel movement (proposed PDMS Technical Specification 3.2.1.1 and 3.2.1.2), would further preclude the possibility of a criticality.

If any fuel is removed from the reactor vessel in the future, the 205-pound (93-kilogram) SFML will apply to that fuel. This 205-pound (93-kilogram) SFML would be applied based on the potential for an enrichment of 2.67 wt% uranium-235 of the residual fuel to exist as a result of little or no mixing of fuel having occurred. Also, if the fuel in the reactor vessel is rearranged outside the analyzed geometries used in the reactor vessel criticality analysis, the 205-pound (93-kilogram) SFML will apply to the rearranged fuel. To ensure that the criticality calculations remain valid and that the geometry of the remaining fuel remains as defined in the criticality calculations, the license conditions prohibit taking any action that would result in the movement of more than 45 percent of the SFML (93 pounds [42 kilograms]) from or within the reactor vessel without specific prior approval of the NRC (proposed PDMS Technical Specifications 3.2.1.1 and 3.2.1.2).

Based on review of the licensee's estimates of residual fuel and calculations of SFML and independent evaluation, calculations and verification, the staff concludes that the prerequisite that the facility has been defueled to the extent reasonably achievable and to such a degree that a nuclear criticality is precluded has been met.

5.2 Removal of Fuel and Core Debris From the Three Mile Island Site

All defueling canisters containing core debris from the reactor and associated systems have been removed from the reactor building and shipped off site. Residual fuel (i.e., UO_2) remains in the facility in the form of core debris lodged in structural gaps and crevices, or as plateout in the pipes and on surfaces in the reactor coolant system and associated structures.* The licensee estimates that the total quantity of fuel remaining in the TMI-2 facility is less than 1723 pounds (783 kilograms) (GPU 1991e). This is estimated by the licensee to be less than 1 percent of the original inventory of fuel. The NRC staff has verified that all remaining defueling canisters containing core debris have been removed from the reactor facility (NRC 1990d).

Based on review of the licensee's records and observation of activities, the staff concludes that the prerequisite that all fuel and core debris that have been removed from the reactor and associated systems has been shipped off site has been met.

5.3 Removal of Water

Evaporation of the accident-generated water was begun in January 1991, and removal and processing of the water will be completed either before or near the start of PDMS. The NRC staff evaluated the processing and disposal of the accident-generated water in a safety evaluation (NRC 1989f). In addition to removal of the accident-generated water, the fire mains within the reactor building will be closed with valves and drained before PDMS to minimize the potential for introduction of water into the reactor vessel.

As far as possible, water in the reactor vessel, the reactor coolant system, the reactor building fuel canal, and the fuel transfer tubes will be removed. The reactor vessel will then be covered to minimize the potential for water entry. The licensee has drilled holes in the canal seal plate to prevent the refueling canal from filling. The submerged demineralizer system and "B" spent fuel pool will be drained and shielded to permit personnel to enter to conduct surveillance activities. However, some residual water will remain in the facility; it is estimated (GPU 1991a, Section 6.2.27.2) that the reactor vessel will contain less than 10 gallons (38 liters) of water. The quantity of water that will remain throughout the reactor coolant system is not enough to transport radioactive material within the facility.

Based on observation of the facility, and review of the licensee's actions and commitments specified in the PDMS SAR, the staff concludes that the prerequisite to remove, to the extent practicable, water from the reactor coolant system; drain the fuel transfer canal; and isolate the fuel transfer tubes will be met.

5.4 Reduction of the Potential for the Release of Radioactive Material

The potential for release of any significant quantity of radioactive material from TMI-2 during PDMS has been minimized by the removal of as much of the fuel and core debris as reasonably achievable and the decontamination of large sections of the reactor building and AFHB surfaces, equipment, and piping. The major source of radioactive material remaining in the facility is inside the reactor building (see Section 4.2). Both routine and accidental releases of the remaining radioactive contamination by atmospheric and liquid pathways are considered in this section. Transfer of contamination by pests is also considered, although not quantified.

5.4.1 Routine Atmospheric Releases

Gaseous effluent released from TMI-2 is limited by 10 CFR Part 20 (Appendix B, Table II, Column I) and the design objectives of Appendix I of 10 CFR Part 50. For gaseous effluent from TMI-2, the dose from radionuclides (other than noble gaseous effluent) must not exceed 15.0 millirem to any organ for the calendar year. The concentration of radioactive material in air released to the atmosphere must not exceed the values specified in 10 CFR Part 20, Appendix B, Table II, Column I. The U.S. Environmental Protection Agency's environmental standards for the uranium fuel cycle given in 40 CFR Part 190 require that "the annual dose equivalent does not exceed 25 millirem to the whole body, 75 millirem to the thyroid, and 25 millirem to any other organ of any member of the public as the result of exposures to planned discharges of radioactive materials, radon and its daughters excepted, to the general environment from uranium fuel cycle operations and to radiation from these operations." The proposed PDMS Technical Specification limits are based on the design objective annual exposure values specified in 10 CFR Part 50, Appendix I. Appendix I states that the total quantity of all radioactive iodine and radioactive material in particulate form above background to be released in effluent to the atmosphere will not result in an estimated annual dose or dose commitment to any individual in an unrestricted area from all pathways of exposure in excess of 15 millirem to any organ. The calculated annual total quantity of all radioactive material above background to be released to the atmosphere also will not result in an estimated annual air dose from gaseous effluent at any location near ground level that could be occupied by individuals in unrestricted areas in excess of 10 millirad for gamma radiation and 20 millirad for beta radiation.

As indicated in Section 4.2, the major portion (approximately 46,000 curies) of the remaining residual contamination is in the reactor building. Thus, the calculated annual dose from routine releases from the TMI-2 facility is based on activity released from the reactor building. The licensee has stated in the PDMS SAR (GPU 1991a, SAR 7.2.4.3) that during PDMS, the reactor containment will not be actively ventilated except before and during entries. Periodic entries will be made during PDMS into the reactor containment and the AFHB for measurement and surveillance activities. Before entry into the containment, the containment will be ventilated (GPU 1991a, SAR 7.2.4.3) using the reactor building purge system. Effluent from the reactor building will be routed through the reactor building purge system to the station vent and will pass through two high-efficiency particulate air (HEPA) filter banks in series. Maintenance of the operability of the HEPA filters is contained in the PDMS SAR (GPU 1991d, SAR 7.2.1.3.2). The station vent will be continuously monitored during reactor containment ventilation by an effluent monitor in the vent; HP-R-219 or HP-R-219A (GPU 1991d, SAR 7.2.4.3). When the reactor containment is not actively ventilated, a containment atmospheric breather will be used to maintain pressure equilibrium between the atmosphere and the reactor building. The containment atmospheric breather is a 6-inch (15.2-centimeter) former hydrogen control system line in which a HEPA filter, 24 inches by 24 inches (61 centimeters by 61 centimeters), has been installed between the reactor building and the AFHB exhaust fan. The breather thus provides a HEPA-filtered pathway for effluent from the containment. Because the breather terminates inside the AFHB, rather than emptying directly to the atmosphere, some material carried by the air will be deposited by impaction and plateout as the air passes through the AFHB. The breather is the most probable pathway for passive ventilation from the containment building because the line is very large compared to other potential leak paths. There is an isolation valve between containment and the HEPA filter that will automatically close upon receipt of a containment pressure increase of 0.25 psi. The purpose of this isolation is to protect the breather HEPA filter in the event of a significant fire in the reactor building (GPU 1991b).

A total of 50 discharges of reactor building atmosphere particulate content per year was conservatively estimated by the licensee (GPU 1991, SAR 8.1.2.1) for routine release calculations. This included both active ventilation of the containment using the reactor building purge system before entry for measurement and surveillance activities, and passive air changes through the atmospheric breather as a result of atmospheric pressure changes (conservatively estimated by the licensee to be 10 per year).

Although the amount of radioactive material in the AFHB that is available for release is much smaller than the amount in the reactor building, there is some potential for a small offsite release. Unlike the reactor building, the AFHB is not designed to be leak-tight. Early in the PDMS period, the AFHB ventilation system will be continuously operated and the release monitored (License Condition 2D of License DPR-73). This circumstance will provide for real-time monitoring of the airflow from the reactor building and the AFHB to the environment by way of the effluent monitors in the station vent (HP-R-219 or HP-R-219A). The staff has determined that the normal ventilation system should be operated and continuous effluent monitoring conducted until an appropriate database has been established. The licensee has committed (GPU 1991e, SAR 7.2.4.3) to conduct a special monitoring program of AFHB airborne levels of radioactive material for at least a 1-year period before PDMS, and for at least 1 year after entry into PDMS (see Proposed PDMS License Condition 2.D). This special monitoring program will be temporarily suspended when activities in the AFHB are expected to generate significant airborne levels of radioactive material. The assumption is made (with the exception of activities generating significant airborne levels of radioactive material) that the operation of the ventilation system will result in greater resuspension of radioactive material as a result of the air movement than if the ventilation system were shut down. Thus, measurement of radioactive material being resuspended and removed from the building by the ventilation system during normal PDMS operations will provide a conservative indication of potential releases from the AFHB in the absence of ventilation.

The licensee has estimated the offsite dose from a 1-year routine atmospheric release from the TMI-2 facility (GPU 1990c, SAR Table 8.1-5)) assuming no operation of the AFHB ventilation system except before entry of personnel. The licensee's estimate was based on measurements of air samples made during the period of time prior to routine reactor building entries. A first order rate equation was used to determine the source and depletion parameters which approximate the observed behavior of the air concentrations following the initial krypton purge. The sink and source estimates were used to predict equilibr[illegible] of 1.0×10^{-8} $\mu Ci/cm^2$ of cesium-137 and 3.0×10^{-9} $\mu Ci/cm^2$ for strontium-90/yttrium-90. The licensee conservatively assumed a total of 50 discharges of the reactor building atmosphere particulate content per year (as indicated previously). This resulted in 2.8×10^{-4} Ci/year of cesium-137,

8.5×10^{-5} Ci/year of strontium-90/yttrium-90 and 3.7×10^{-8} Ci/year of transuranics. The licensee estimated a dose commitment of 0.01 millirem/year to the total body of the maximally exposed offsite individual and 0.02 millirem/year to the bone of the maximally exposed offsite individual. The licensee's estimates are also considerably smaller than the design objectives specified in 10 CFR Part 50, Appendix I; the limits given in 40 CFR Part 190; and in the licensee's proposed PDMS Technical Specifications.

The NRC staff has also independently estimated the offsite dose from a 1-year routine release from the TMI-2 facility. The staff identified four major sources of potentially suspendible contamination in PEIS Supplement 3 (NRC 1989a): (1) the enclosed stairwell/elevator structure; (2) the sludge residue on the reactor building basement floor; (3) the remaining surface contamination on the concrete walls, equipment, overhead structures, and so forth; and (4) the surface contamination on the walls and equipment located in the D-rings. A resuspension factor* of 0.000002/meter (Clayton 1970; Dunster 1962) was applied to the fraction of contamination deemed to be potentially suspendible. Credit was taken for the presence of double-stage HEPA filters in the reactor building purge system and of double-stage HEPA filters in the station vent for air that left the reactor building by way of the atmospheric breather system.

The NRC staff has estimated the amount of radioactive material to be released annually during the PDMS period as 0.0012 curie of strontium-90/yttrium-90 and 0.013 curie of cesium-137.** This estimate is based on the assumption that a fraction of the contamination present in the reactor building becomes resuspended and is either released to the atmosphere during active ventilation of the containment or by way of the breather pathway during ventilation of the AFHB.

The staff's estimated release from the reactor building during a 1-year period (assuming operation of the AFHB ventilation and filtration system) would result in a 50-year dose commitment*** of approximately 1.6 millirem to the bone of the maximally exposed offsite individual and approximately 0.16 millirem to the total body of the maximally exposed offsite individual. The estimated dose from a 1-year routine release from the TMI-2 facility is considerably smaller than the design objective specified in 10 CFR Part 50, Appendix I; the limits given in 40 CFR Part 190; and in the licensee's proposed PDMS Technical Specifications.

Although the licensee has committed to initially operate the AFHB ventilation system, the NRC staff has also evaluated the release of contamination for the situation in which the ventilation system is not operating. In this case, credit was taken for the HEPA filter in the atmospheric breather, but

* The resuspension factor is the ratio of air contamination ($\mu Ci/m^3$) to the surface contamination ($\mu Ci/m^2$).

** Additional isotopes may be released as given in PEIS Supplement 3 (NRC 1989a). However, strontium-90/yttrium-90 and cesium-137 account for more than 95 percent of the dose.

*** 50-year dose commitment, as calculated by the NRC, cannot be directly compared to an annual dose as calculated by the licensee.

no credit was taken for the additional deposition of material in the AFHB from impaction and plateout or for the HEPA filters in the station vent. Instead, the breather was modeled as if it emptied directly to the atmosphere to ensure a conservative evaluation. The staff calculated a possible release of 0.0021 curie of strontium-90/yttrium-90 and 0.023 curie of cesium-137. This gives a 50-year dose commitment* of approximately 0.28 millirem to the total body of the maximally exposed offsite individual and approximately 2.8 millirem to the bone of the maximally exposed offsite individual. Again, the estimated dose from a 1-year routine release from the TMI-2 facility is considerably smaller than the design objectives specified in 10 CFR Part 50, Appendix I; the limits given in 40 CFR Part 190; and in the licensee's proposed PDMS Technical Specifications.

5.4.2 Routine Liquid Releases

The limits for radioactivity in liquid effluent released from TMI-2 are specified in 10 CFR Part 20 (Appendix B, Table II, Column 2) and in the design objectives of 10 CFR Part 50, Appendix I. These regulations limit the estimated annual dose or dose commitment from the liquid effluent released from TMI-2 to the site boundary to a dose less than or equal to 3 millirem to the total body and less than or equal to 10.0 millirem to any organ for the calendar year. Also, the concentration of radioactive material released at any time from TMI-2 to unrestricted areas is limited to the concentrations specified in 10 CFR Part 20, Appendix B, Table II, Column 2. Releases of radioactivity to any body of water must also meet EPA's environmental standards for the uranium fuel cycle specified in 40 CFR Part 190. These standards require that the annual dose equivalent not exceed 25 millirem to the whole body, 75 millirem to the thyroid, and 25 millirem to any other organ of the body. In addition, the release of radioactivity to water must meet EPA's National Interim Primary Drinking Water Standards specified in 40 CFR Part 141 that limit beta particle and photon radioactivity from manmade radionuclides in community water systems to that level that "shall not produce an annual dose equivalent to the total body or any internal organ greater than 4 mrem/year." This standard applies to concentrations at community water intakes downstream of the discharge point. The licensee's current Technical Specification limits are based on the design objective annual exposure values specified in 10 CFR Part 50, Appendix I (see Section 5.4.1).

Routine liquid releases from TMI-2 during PDMS are expected to be no more than 5000 gallons (19,000 liters) each year (GPU 1987a). These releases would be from monitored sources and do not include the accident-generated water that may be processed during the first years of PDMS. The impact of processing the accident-generated water is discussed in the safety evaluation for accident-generated water (NRC 1989f). The major sources of liquids that could result in contaminated liquid releases from the facility (other than accident-generated water) are expected to be from groundwater inleakage primarily at the interface (the cork seal) between the AFHB and the reactor building, collected precipitation, and occasional small quantities of fluids used for minor decontamination jobs (during necessary maintenance or inspec-

* 50-year dose commitment, as calculated by the NRC, cannot be directly compared to an annual dose as calculated by the licensee.

tion activities or as a result of the spread of contamination). Collected liquids, other than accident-generated water, will be analyzed for contamination and processed through the EPICOR II system or the TMI-1 radwaste system, if necessary, to ensure that discharges are less than those specified in regulatory requirements. The capability to process potentially contaminated liquid will be maintained during PDMS.

The licensee estimated (GPU 1990c, SAR 8.1.2.2) the amount of radioactive material to be released by way of liquid pathways. Based on previous experience, the cesium-137 and strontium-90/yttrium-90 concentrations achieved by the EPICOR processing system are 4×10^{-6} mCi/ml and 1×10^{-5} mCi/ml, respectively (GPU 1992). Based on 5000 gallons (19,000 liters), this results in a projected offsite dose (GPU 1990c) of 0.005 millirem/year to the bone and 0.002 millirem/year to the total body of the maximally exposed offsite individual.

The NRC staff has also independently estimated (NRC 1989a) the amount of radioactive material to be released by way of liquid pathways as approximately 7.6×10^{-4} curie/year as shown in Table 5.1. The 50-year dose commitment to the bone of the maximally exposed offsite individual from a 1-year release is approximately 0.0009 millirem. A dose of 0.0009 millirem is also estimated for a 1-year release to the total body of the maximally exposed individual.* Both the NRC staff and licensee's calculated doses resulting from a 1-year release (as given here) are considerably smaller than the design objectives cited in 10 CFR Part 50, Appendix I; the limits given in 40 CFR Part 190; the limits given in 40 CFR Part 141; and in the licensee's proposed PDMS Technical Specifications.

5.4.3 Accidental Atmospheric Releases

For an accident situation, the guidance provided in 10 CFR Part 100 for determination of exclusion areas is used. This guidance states that an exclusion area is required of such size that an individual located at any point on its boundary for 2 hours immediately following onset of the postulated fission product release would not receive a total radiation dose in excess of 25 rem to the whole body or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure.

The licensee estimated the dose that would result from a number of accidents that could potentially release radioactive material to the environment. These accidents include (1) a vacuum canister failure, during decontamination activities, (2) the accidental spraying of concentrated contamination with a

* These doses are based on information given in PEIS Supplement 3 (NRC 1989a) and are shown in Table 3.6 of that document for releases lasting for 5, 23, and 33 years.

Table 5.1 Rates of Routine Liquid Release to the Susquehanna River During Post-Defueling Monitored Storage[a]

Radionuclide	Release Rate, Ci/yr[b]
Tritium	3.0×10^{-8}
Carbon-14	1.7×10^{-8}
Selenium-79	3.6×10^{-9}
Strontium-90/Yttrium-90	3.9×10^{-5}
Niobium-93m	7.4×10^{-9}
Technetium-99	1.2×10^{-7}
Ruthenium-106/Rhodium-106	1.3×10^{-6}
Cadmium-113m	2.1×10^{-9}
Antimony-125	9.0×10^{-7}
Tellurium-125m	2.5×10^{-7}
Tin-126/Antimony-126m	2.5×10^{-9}
Cesium-134	5.6×10^{-6}
Cesium-135	2.5×10^{-9}
Cesium-137/Barium-137m	7.0×10^{-4}
Samarium-151	1.8×10^{-5}

(a) Data were obtained from NRC 1989a.
(b) Release rate is for the first year of PDMS. Release rates for subsequent years are based on the first-year release rates and account for radioactive decay.

high pressure spray during decontamination activities, (3) accidental cutting of contaminated piping during decontamination activities, (4) an accidental break of contaminated piping during decontamination activities, (5) a fire inside the contaminant building, (6) an open penetration, and (7) a rupture and release of resins from the makeup and purification demineralizers.

The NRC staff performed independent evaluations of the seven potential accidents identified by the licensee in the PDMS SAR. The staff also evaluated an eighth potential accident, a fire in the D-ring inside containment. The staff analyzed each type of accident separately. The results of the accident analyses are shown in Table 5.2 and are discussed below and compared to the results obtained by the licensee.

Decontamination Activity Accidents

During PDMS, if movement of radioactive material is detected, some decontamination activities may be conducted to mitigate the impact and permit normal surveillance activities. During these decontamination activities, an accident could occur. Such an accident in the reactor containment during the PDMS could result in an unanticipated release of radioactive material to the environment. In its PDMS SAR, the licensee evaluated four potential decontamination accident scenarios that could occur during decontamination activities.

Table 5.2 Estimated Dose from Accidents During Post-Defueling Monitored Storage

	Maximum Individual Dose, mrem			
	with Operation of Ventilation System		with No Operation of Ventilation System	
Accident	Total Body	Bone	Total Body	Bone
1. Decontamination Activity Accidents				
A. Vacuum Canister Failure	1.2×10^{-4} (a)	1.1×10^{-3} (a)	-	-
B. High-Pressure Spray of Contamination	1.4×10^{-5} (a)	1.3×10^{-4} (a)	-	-
C. Cutting Contaminated Pipe	7.9×10^{-8} (a)	8.3×10^{-8} (a)	-	-
D. Break of Contaminated Pipe	4.8×10^{-8} (a)	5.1×10^{-8} (a)	-	-
2. Fire in Containment				
A. Elevator/Stairwell	0.02	0.13	1.6	13
B. D-rings	0.49	0.51	49	51
3. Containment Penetration Failure	Not Applicable[b]		2.6	27
4. Release of Resins from Makeup and Purification Demineralizer	0.20	0.25	20	25

(a) From Murphy and Holter 1982.
(b) Since the release is into the turbine building, the operation of the auxiliary building ventilation system is not expected to have any effect on the offsite dose.

The decontamination activity accidents include: (1) failure of a vacuum canister, (2) spraying of contamination with high-pressure spray, (3) cutting of a contaminated pipe, and (4) a break in a contaminated pipe. The licensee developed scenarios for these accidents based on activities evaluated for the decommissioning of a generic pressurized-water reactor following an accident (Murphy and Holter 1982) in a study performed for the NRC. For each scenario the licensee looked at three cases: (1) the reactor building purge system operating, (2) the reactor building isolated and at negative pressure, and (3) the reactor building under passive ventilation. In all cases, the licensee's estimate of the effects of these accidents is several orders of magnitude greater than the effects calculated for a generic facility by Murphy and Holter for purposes of comparison. The dose estimates given in Murphy and Holter are shown in Table 5.4. Although the activities analyzed by the licensee are not expected to occur, the licensee's analysis is conservative, and in all four cases radiological emissions are significantly lower than those permitted by 10 CFR Part 100 for determination of exclusion areas (a total radiation dose to the whole body in excess of 25 rem or a total radiation dose in excess of 300 rem to the thyroid from iodine exposure). For the vacuum canister failure, the licensee estimated 4.8 millirem to the maximally exposed individual (assuming the reactor building purge system is operating). A dose to the maximally exposed individual of 4.0 millirem was calculated by the licensee for the accidental spraying of contaminated liquid (assuming the reactor building is at slightly negative pressure and that the total source term release is exhausted to the environment in the first reactor building air change subsequent to the activation of the reactor building purge system). The dose from the accidental cutting of a contaminated pipe and the dose from the accidental breaking of a contaminated pipe were estimated by the licensee to be 0.005 millirem (assuming the reactor building is at slightly negative pressure and the total source term release is exhausted to the environment in the first reactor building air change subsequent to the activation of the reactor building purge system).

Fire in the Containment

The licensee's analysis (GPU 1990c, SAR 8.2.5) of a fire in containment was based on a fire occurring in the combined "A" and "B" D-rings (although the occurrence of such a fire is considered to be incredible). It was assumed that the D-rings contained 16,600 curies of cesium-137, 830 curies of strontium-90/yttrium-90 and 0.6 kilograms of residual fuel on the suspended defueling tools. One percent of the contamination and 100 percent of the fuel on the tools was assumed to be loose, surface activity that is available to become airborne in a fire. A suspension factor of 0.001 was used for both contamination and fuel. Plateout of the airborne source was not considered. The licensee calculated (GPU 1990c, SAR 8.2.5) a dose to the maximally exposed individual of 13.8 millirem as a result of a fire in the reactor containment building, assuming the reactor building ventilation system was operating. A dose of less than 13.8 millirem was determined by the licensee for the case where the ventilation system was not operating and the reactor building was vented passively through the breather.

In PEIS Supplement 3, the NRC staff analyzed an accidental atmospheric release that resulted from a fire in the stairwell/elevator structure of the reactor containment building. To evaluate the accident scenario involving a fire, the staff used the following conservative assumptions: the accident would occur early in the storage period, before appreciable decay of the radionuclides occurred; 20 percent of the stairwell/elevator structure below the 8-foot (2.4-meter) mark would be involved in the fire; 20 percent of the activity in the stairwell/elevator structure would be involved in the fire, although the contamination in the structure is not distributed uniformly; and the 7.1 pounds (3.2 kilograms) of core debris thought to remain on the floor of the basement after desludging would also be involved in the fire (even though desludging has occurred in the area of the stairwell/elevator structure and measurements taken before desludging indicate that core debris is not located near the stairwell/elevator structure). The fraction of radioactive material to be released into the reactor building atmosphere during the burning of the contaminated material was assumed to be 0.0005, based on studies by Mishima and Schwendiman (1973). The amount released from the building would be further reduced because the HEPA filters would remove at least 99 percent of the radioactive particulates.* The fraction of the radioactive particulate material that would penetrate the single-stage HEPA filter used when the reactor building was secured but not actively ventilated was conservatively assumed to be 0.01 (NRC 1978). The amount of radioactive material calculated for release during this accident is estimated to be 0.02 curie (Table 5.3). The resulting bone dose to the maximum offsite individual is 13 millirem, and the total body dose is estimated to be 1.6 millirem. These doses are significantly lower than those permitted by 10 CFR Part 100 for determination of exclusion areas. Operation of the AFHB ventilation system during this accident would further reduce the doses by a factor of 100.**

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* Each filter has an in-place tested efficiency of at least 99.95 percent for removal of particulates of 0.3-micron (0.0003-millimeter) diameter. Therefore, only a fraction (0.0005) of the particulates in the building atmosphere would pass through the first stage to the atmosphere. The staff, however, in evaluating the potential for release of radioactive material used a more conservative penetration factor of 0.01 (corresponding to 99-percent efficiency). Regulatory Guide 1.140 (NRC 1979c) gives guidelines for operating nuclear power plants, specifiying the conservative penetration factor of 0.01 (corresponding to 99-percent efficiency) for filtration systems that test, in place, to an efficiency of 99.95 percent or more.

** The AFHB will be ventilated through two HEPA filters in series. Each filter has an in-place tested efficiency of at least 99.95 percent for removal of particulates of 0.3-micron (0.0003-millimeter) diameter. Therefore, only a fraction (0.0005) of the particulates in the building atmosphere would pass through the first stage and a similar fraction (0.00000025 of the initial particulates) would pass through the second stage to the atmosphere. The staff, however, in evaluating the potential for release of radioactive material used a more conservative penetration factor of 0.01 (corresponding to 99-percent efficiency). Regulatory Guide 1.140 (NRC 1979c) gives guidelines for operating nuclear power plants, specifying the conservative penetration factor of 0.01 (corresponding to 99-percent efficiency) for filtration systems that test, in place, to an efficiency of 99.95 percent or more.

Table 5.3 Postulated Accidental Atmospheric Release From a Fire in the Stairwell/Elevator Structure During Post-Defueling Monitored Storage[a]

Radionuclide	Release Rate, Ci/yr[b]
Tritium	7.9×10^{-7}
Carbon-14	4.4×10^{-7}
Manganese-54	3.2×10^{-10}
Iron-55	8.5×10^{-7}
Cobalt-60	3.6×10^{-6}
Nickel-63	9.5×10^{-7}
Selenium-79	9.7×10^{-8}
Krypton-85	3.0×10^{-6}
Strontium-90/Yttrium-90	1.1×10^{-3}
Zirconium-93	2.7×10^{-9}
Niobium-93m	2.0×10^{-7}
Technetium-99	3.3×10^{-6}
Ruthenium-106/Rhodium-106	3.8×10^{-5}
Cadmium-113m	5.5×10^{-8}
Antimony-125	2.5×10^{-5}
Tellurium-125m	7.1×10^{-6}
Tin-126/Antimony-126m	6.6×10^{-8}
Cesium-134	1.5×10^{-4}
Cesium-135	6.6×10^{-8}
Cesium-137/Barium-137m	1.9×10^{-2}
Cerium-144/Praseodymium-144	1.1×10^{-6}
Praseodymium-144m	1.6×10^{-8}
Promethium-147	2.0×10^{-5}
Samarium-151	4.8×10^{-4}
Europium-152	4.0×10^{-10}
Europium-154	5.2×10^{-7}
Europium-155	1.5×10^{-6}
Uranium-234	2.0×10^{-8}
Uranium-235/Thorium-231	6.8×10^{-10}
Uranium-236	6.1×10^{-10}
Uranium-237	3.5×10^{-10}
Uranium-238/Thorium-234/Protactinium-234m	4.6×10^{-9}
Plutonium-238	1.2×10^{-7}
Plutonium-239	1.5×10^{-6}
Plutonium-240	4.1×10^{-7}
Plutonium-241	1.5×10^{-5}
Americium-241	4.0×10^{-7}

(a) Data were obtained from NRC 1989a.
(b) Assumes accident occurs during the first year of PDMS.

The NRC staff performed an additional evaluation of the fire for this technical evaluation involving a fire inside the D-rings in the containment. The major source of combustible material during PDMS in the reactor building is the oil in the reactor coolant pump oil reservoir. For the evaluation, the staff assumed that half of the remaining oil in two of the four coolant pumps (a total of 138 gallons [522 liters] of oil) spilled onto the floor in the D-rings and was subsequently nonmechanistically ignited. The computer code FIRIN (Chan et al. 1989) was used to model conditions in the reactor building during a fire. The FIRIN model simulated burning of the fuel, generation of smoke and combustion gases, plugging of filters, and transfer of heat to walls, ceiling, floor, and equipment in containment. Sufficient oxygen was assumed to exist to fuel the fire. Plugging and failure of the HEPA filter in the passive breather line were calculated to occur (the filter was assumed to fail at 3.2 psig overpressure (Burchsted et al. 1976). While the containment atmospheric breather system is designed to provide automatic isolation of the filter in the event of a 0.25 psi overpressure of the containment building, this protective feature was not considered in calculating the impact of a fire in the containment building. The activity release that resulted was based on (1) 17,000 curies of cesium-137 and 830 curies of strontium-90 in the D-rings, (2) an estimated 50 percent of the activity available for release, (3) an assumed release rate of 0.152 wt%, and (4) particles larger than 10 micrometers (95 percent of particles) settling out in the reactor building or in the AFHB downstream of the breather line. Assuming no ventilation of the AFHB during the accident, a release of 0.65 curie of cesium-137 and 0.03 curie of strontium-90 would occur. The resulting dose to the maximum offsite individual would be 49 millirem to the total body and 51 millirem to the bone. This dose is due only to external radiation from the plume and the ground and from inhalation; it does not include the food pathway or drinking water. This dose is higher than that discussed above for the fire in the stairwell, although it is significantly lower than that permitted by 10 CFR Part 100 for the determination of exclusion areas. Assuming the AFHB ventilation system is operating during the fire, the maximum doses would be decreased by a factor of 100 (considering the 99-percent efficiency of filtration of the air by the HEPA filters in the station vent). This would result in the release of 0.0065 curies of cesium-137 and 0.0003 curies of strontium-90/yttrium-90. The resulting dose to the maximum offsite individual would be 0.49 millirem to the total body and 0.51 millirem to the bone. Considering the relatively high flash point of the oil (450°F) (GPU Nuclear Calculation Sheet 4710-3220-87-037, "RB Fire HEPA Filter $\Delta P/\Delta t$," dated December 16, 1987), the removal of the majority of the combustible material from the building to minimize the potential for heating material, and the lack of potential ignition sources since systems will normally not be energized inside containment, the probability of this fire is extremely remote.

Although operation of the AFHB ventilation system would reduce the offsite dose to the public by a factor of approximately 100 for either fire scenario, the staff has concluded that operation of the AFHB ventilation system is not required in the event of a fire in containment. This conclusion is based on the calculated dose to the public without operation of the ventilation system which would be a fraction of the 10 CFR Part 100 limits.

Containment Penetration Failure

The licensee's analysis of an open penetration assumed that an unfiltered pathway was created to the environment. The licensee assumed that (1) the containment was under passive ventilation, (2) an unfiltered pathway was open for a quarter of a year, (3) there were 2.5 reactor building air changes with the environment, and (4) 100 percent of the release was through an unfiltered pathway.

The licensee (GPU 1991b, SAR 8.2.6) calculated a dose to the maximally exposed individual of 0.88 millirem. This resulted from 0.0014 curies of cesium-137, 0.0004 curies of strontium-90/yttrium-90, and lesser amounts of transuranics.

The NRC staff also analyzed the failure of a penetration between the containment building and the turbine building. It was assumed that a 14-inch diameter penetration between the two buildings nonmechanistically failed. This failure resulted in an unfiltered pathway to the turbine building and ultimately to the environment. The unfiltered pathway was assumed to be open for one quarter of a year. It was assumed that the reactor building ventilation system was not operating at this time and that 100 percent of the air was released (i.e., 2.5 reactor building air changes, based on the 10 reactor building air changes per year via the passive breather as discussed in Section 5.4.1). It was also assumed that there was no settling or plateout of material within the turbine building but that the particulates were released directly to the atmosphere. This scenario resulted in the release of 0.02 curie of strontium-90 and 0.22 curie of cesium-137, and a dose to the maximally exposed individual of 2.6 millirem to the total body and 27 millirem to the bone, assuming all pathways. The resulting dose is significantly lower than that permitted by 10 CFR Part 100 for the determination of exclusion areas.

Release of Resins From a Makeup and Purification Demineralizer

The licensee also calculated (GPU 1991b, SAR B.8) the dose resulting from the release of resins from a makeup and purification demineralizer vessel rupturing non-mechanistically and the contents spilling onto the floor of the cubicle. The inventory of the demineralizer included 530 curies of cesium-137, 100 curies of strontium-90/yttrium-90, and 1.1 pounds (0.5 kilograms) of fuel. The licensee used an airborne release factor of 0.0001. The licensee's calculation resulted in an estimated 0.45-millirem dose to the maximally exposed individual.

The NRC staff also analyzed the release of contaminated resins from a ruptured demineralizer unit in the auxiliary building. The inventory of the demineralizer includes approximately 100 curies of strontium-90, 530 curies of cesium-137, and 500 grams of fuel. The primary isotopes in the fuel are plutonium-239, plutonium-240, plutonium-241, samarium-151, europium-154, and europium-155. An airborne release fraction of 0.0005 was used. In addition, it was assumed that all airborne activity would be filtered by the HEPA filters in the AFHB ventilation system before it was released into the atmosphere. The calculated doses to the maximally exposed individual were 0.20 millirem for the total body and 0.25 millirem to the critical organ (the bone) from inhalation and external exposure. These doses are significantly lower than those permitted by 10 CFR Part 100.

For the case in which the AFHB ventilation system is not operating, airborne activity would be released directly to the atmosphere, rather than being filtered through the ventilation system. The calculated doses would increase by a factor of 100, resulting in estimated doses of 20 millirem to the total body to the maximally exposed individual and 25 millirem to the critical organ (the bone) from inhalation and external exposure. These doses are still significantly lower than those permitted by 10 CFR Part 100.

5.4.4 Accidental Liquid Releases

In the PDMS SAR (GPU 1991d), the licensee postulated no accidental liquid releases during PDMS. The NRC staff also concluded that no accidental liquid releases would occur during PDMS. During PDMS, water-processing capabilities will be available to dispose of the small amount of liquid produced by inleakage, condensation, and small amounts of decontamination. Liquids that are not directly releasable pursuant to 10 CFR Part 20, Appendix B, Table II, Column 2, will be collected in the miscellaneous waste holdup tank, transferred to the chemical cleaning building, and then processed through the EPICOR II system before final sampling and discharge. On the basis of the environmental assessment prepared by the NRC staff in 1979 on the use of the EPICOR II system at TMI-2 (NRC 1979b), there are no credible accidents that would result in a liquid release to the environment during the transfer or processing of the liquids produced during PDMS. The operating history of this system in the intervening time has not altered this conclusion. This processing is the only activity during PDMS that will involve contaminated or potentially contaminated liquids.

5.4.5 Transfer of Contamination by Pests

The staff has identified a potential for bird, rodent, bat, and insect intrusion into contaminated areas of the facility. If this intrusion occurs, some potential exists for periodic transfer of contamination to uncontrolled areas. However, the licensee maintains a contract with a local pest control service that provides insect, rodent, spider, and bird control and removal of live animals. This service will be maintained during PDMS. Additionally, the licensee has committed in the PDMS SAR (GPU 1991a, SAR Supplement 3, A-16) to a surveillance program under which a limited number of carcasses, when available, will be analyzed for gamma-emitting isotopes as part of the nonroutine Radiological Environmental Monitoring Program.

5.5 Removal of Radioactive Waste Resulting From Major Decontamination Activities

All the radioactive waste resulting from major decontamination activities has been shipped off site or packaged and staged for shipment off site, except the waste from the accident-generated water disposal activities. The accident-generated water located in the reactor vessel, fuel pools, fuel transfer canal, and building sumps will be pretreated to remove the majority of the particulate material before being processed as part of the accident-generated water disposal activities. The removal of the waste resulting from the accident-generated water disposal activities is discussed in the safety evaluation in support of TMI-2 license Amendment 35 dated September 11, 1989, for the accident-generated water (NRC 1989f). Radioactive components, such as the reactor head assembly, upper plenum assembly, upper end fittings, sections of the flow distributor that contain incore instrument guide tubes, and fuel-removal tooling, remain in the reactor building. These remaining components and equipment are not flammable and do not add significant quantities of radioactive material to the inventory in the reactor building. Consequently, they do not degrade the level of protection of the health and safety of the public.

Based upon observation of activities and review of records, the staff concludes that the prerequisite that the remaining radioactive waste from the major TMI-2 decontamination activities be shipped offsite or packaged and staged for shipment has been met.

5.6 Reduction of Radiation Levels To Allow Plant Maintenance and Surveillance During Post-Defueling Monitored Storage

During PDMS, personnel will enter the reactor building and the AFHB periodically to conduct inspections, surveillance, radiological surveys, radiological waste processing, remedial decontamination, and some maintenance to support these activities, as well as preventive maintenance on a limited number of operational systems. During the initial period of PDMS (a minimum of 6 months), the licensee has committed in Section 7.2.4 of the PDMS SAR to conduct monthly entries. After facility stability has been verified and a database has been established, the frequency of entry may be diminished. Decontamination has been performed and shielding has been applied to reduce radiation dose rates in areas requiring access by personnel. It is expected that occupational radiation exposure rates would increase following the removal of water from the reactor vessel as a result of the cobalt-60 activation of the baffle plates in the reactor vessel. However, shielding will be applied in critical locations to reduce the dose rates. Radiation exposures to personnel will be maintained within the limits established by 10 CFR Part 20.

Based upon reviews of the results of decontamination efforts and radiological surveys and activities to be conducted during PDMS, the staff concludes that the prerequisite that radiation levels within the facility be determined and reduced so that necessary and required plant monitoring, maintenance and inspections can be performed has been met to allow commencement of PDMS.

5.7 Definition and Establishment of a Surveillance Program

During PDMS, the licensee will be required to conduct surveillance programs to ensure the maintenance of environmental protection systems. These programs include surveillance of reactor containment building isolation, surveillance of the reactor containment building and the AFHB ventilation and filtration systems, surveillance of the fire protection system and the support air monitoring systems (including electrical, effluent monitoring, and environmental monitoring systems), and oversight of administrative systems. Administrative systems include organizational structure, staff qualifications, records, independent safety reviews, procedures, occupational radiation protection, a quality assurance plan, an emergency plan, and other administrative control activities. These surveillance functions are covered by the proposed PDMS Technical Specifications, various review plans, commitments given the licensee in the PDMS SAR, and requirements in this technical evaluation report.

Based upon review of the surveillance program specified by the PDMS SAR and the proposed PDMS Technical Specifications, the staff concludes that the prerequisite that a surveillance program for PDMS environmental protection systems to ensure public health and safety be defined and established has been met.

6 POST-DEFUELING MONITORED STORAGE ENVIRONMENTAL PROTECTION SYSTEMS

Considering the defueled status of the reactor and the maximum potential for offsite dose from credible accidents, TMI-2 has no safety-related structures, systems, or components. Safety-related structures, systems, and components are those that are necessary to ensure (1) the integrity of the reactor coolant pressure boundary, (2) the capability to shut down the reactor and to maintain it in a shutdown condition, and (3) the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to the limits given in 10 CFR Part 100.

Although there are no safety-related structures, systems, or components at TMI-2, the NRC staff has identified six structures, systems, and components that are important in providing reasonable assurance that the facility can be safely maintained during PDMS and that are used to provide environmental protection:

1. Reactor Vessel - to preclude the possibility of an inadvertent criticality.

2. Reactor Containment Structure - to ensure containment of the remaining radioactive material during the PDMS period.

3. Reactor Containment and Auxiliary and Fuel Handling Building (AFHB) Purge, Breather, Ventilation, and Filtration Systems - to control radioactive effluents.

4. Fire Protection System - to detect and mitigate any effects of a fire within the facility.

5. Flood Protection - to minimize the intrusion of water into the facility.

6. Support and Monitoring Systems - to support PDMS, includes such systems as the electrical system and the radiological monitoring system, to ensure fire detection and a radiological monitoring capability.

The following sections present a description, the current licensing basis, the PDMS licensing basis, and the conclusions of the NRC staff as to the acceptability of each of these structures, systems, and components to ensure protection of the environment during PDMS.

6.1 Reactor Vessel

6.1.1 Reactor Vessel - System Description

As a result of the accident at TMI-2, some reactor fuel and core debris were dispersed from the reactor vessel into the reactor coolant system, the reactor containment building, and the AFHB. As described in Section 5.1, the licensee has defueled the facility to the extent reasonably achievable, and this has been independently verified by the NRC staff.

The reactor vessel is the only component within TMI-2 that contains sufficient residual fuel to exceed the safe fuel mass limit (SFML). The licensee's Defueling Completion Report (GPU 1990d and e), the NRC staff's analysis of the Defueling Completion Report (NRC 1990c), and Section 5.1.4 of this technical evaluation report provide an analysis and evaluation of the SFML for the reactor vessel and the potential for an inadvertent criticality in its current location and form. All three analyses indicate that there is no potential for a criticality in the fuel remaining in the reactor vessel during either normal or accident conditions. Much of the remaining fuel (i.e., UO_2) (approximately 262 pounds [119 kilograms]) is in the annular gap and would not readily move. However, rearrangement of fuel, either purposefully or as a result of an impact such as a heavy load dropped from the polar crane, may result in relocation of the reactor internal components and movement of the fuel. In addition, fuel and core debris may move to the bottom of the reactor vessel as a result of long-term degradation of reactor internal components or flaking of the surface films, fines, and granular debris.

The PDMS SAR (GPU 1991e, SAR 4.3.6) states that "Control of SNM at TMI-2 during PDMS relies upon isolation boundaries and control of access to components which contain SNM. Isolation boundaries will be maintained, as necessary to prevent relocation of significant SNM quantities." The PDMS SAR (GPU 1991e, SAR 4.3.6) further states that the reactor coolant system, which contains the largest quantity of SNM outside the reactor vessel, will be drained to the extent practicable and isolated within the containment building.

The licensee has stated in the PDMS SAR (GPU 1991d, SAR 7.2.2.2) that fire mains within the reactor building will be closed with valves and drained before PDMS to minimize the potential for introduction of water into the reactor vessel. In addition, the reactor vessel will be covered. Holes have been drilled in the canal seal plate to prevent it from filling and flooding the reactor vessel (see Section 5.3 of this document).

Water has been removed from the reactor vessel and the reactor coolant system to the extent practical. To the extent that the spent fuel pools are needed to store the accident-generated water before disposal, the water may remain in these pools after the start of PDMS.

6.1.2 Reactor Vessel - Current Licensing Basis

Currently, no Technical Specifications relate to the fuel remaining in the TMI-2 facility.

6.1.3 Reactor Vessel - PDMS Licensing Basis

To maintain the fuel in the reactor vessel in the analyzed geometry during PDMS, the proposed PDMS Technical Specifications limit activities that could alter the geometry of the fuel in the reactor vessel. To ensure maintenance of the required conditions, the proposed PDMS Technical Specifications stipulate the following:

1. Loads in excess of 50,000 pounds (22,700 kilograms) are prohibited from travel over the reactor vessel unless the activity is approved by the NRC by a docketed safety evaluation (proposed PDMS Technical Specifications 3.3.1), and

2. No more than 93 pounds (42 kilograms) of fuel (i.e., UO_2) may be removed from the reactor vessel or rearranged outside the analyzed geometries in the "Defueling Completion Report" (GPU 1990d, 1990e) without prior NRC approval. When more than 93 pounds (42 kilograms) of fuel (i.e., UO_2) in the reactor vessel have been removed or rearranged, the licensee must suspend all further fuel removal or rearrangement activities and submit a safety analysis to the NRC for approval of this activity and any further fuel removal or rearrangement activities (proposed PDMS Technical Specifications 3.2.1.1 and 3.2.1.2).

3. The licensee will notify the NRC if there is a change in conditions that could affect the geometry of the fuel in the reactor vessel, that is, flood, seismic event, and so forth (proposed PDMS Technical Specifications 3.2.1.2).

Based on the results of the reactor vessel criticality analyses, no neutron poison is required to preclude an inadvertent criticality. However, the licensee has stated in the PDMS SAR 4.3.5 that an insoluble neutron poison will be placed in the bottom of the reactor vessel (GPU 1991e). The licensee also has the capability of inserting a video camera into the reactor vessel to verify fuel location if it is determined at a later time that such an examination is required.

6.1.4 Reactor Vessel - Conclusion

The NRC staff finds that with the proposed PDMS Technical Specifications limiting the mass of loads over the reactor vessel and limiting the quantity of fuel (i.e., UO_2) that can be removed or rearranged, and with the licensee's actions to increase the margin of safety by adding an insoluble neutron poison to the reactor vessel and minimizing the potential for water accumulation in the reactor vessel, the possibility of an inadvertent criticality is precluded.

6.2 Reactor Containment Structure

6.2.1 Reactor Containment Structure - System Description

Most of the remaining residual fuel and the remaining removable contamination (fission products and activation products) are located in the reactor building. During PDMS, the reactor building and associated systems will be used as the environmental barrier for the residual contamination (which could potentially result in offsite exposure) that remains inside the containment structure.

The reactor building is a reinforced concrete structure composed of cylindrical walls with a flat foundation and a dome roof. The cylinder has an inside diameter of 130 feet (39.62 meters), a wall thickness of 4 feet (1.22 meters), and a height of 157 feet (47.85 meters) from the top of the foundation slab to the spring line. The roof is a shallow dome with a large radius of 110 feet (33.53 meters), a transition radius of 20 feet 6 inches (6.25 meters), and a thickness of 3 feet 6 inches (1.07 meters). The foundation slab is bearing on rock and is 11 feet 6 inches (3.51 meters) thick reinforced with conventional mild steel reinforcing. The cylindrical wall is prestressed with a post-tensioning system in the vertical and hoop directions. The dome roof is prestressed using a three-way post-tensioning system. The inside surface of the containment structure is lined with carbon steel. The nominal liner plate thickness is 3/8 inch (0.95 centimeter) for the cylinder, 1/2 inch (1.27 centimeter) for the dome, and 1/4 inch (0.635 centimeter) for the base. A 2-foot- (0.61 meter-) thick concrete slab is above the base liner plate. The structure provides biological shielding during normal and unanticipated conditions. The steel liner encloses the equipment and systems that remain inside the containment and ensures that the upper limit of potential leakage of radioactive material will not be exceeded under the worst unanticipated conditions.

A complete listing of containment penetrations is provided in Table 6.1, which also gives the service originally provided by the penetration, line sizes, the isolation valve or other isolation mechanism, and its status during PDMS. Twelve penetrations that were modified after the accident to provide special functions for the cleanup period are listed in Table 6.2. The piping penetration modifications installed during the cleanup period were designed to withstand 5 psi of pressure and have been tested to hold 1.2 to 1.5 times this pressure for not less than 10 minutes (GPU 1991b, SAR 3.7.1.2 and 7.2.1.1). Seven penetrations that are considered operational during PDMS are shown in Table 6.3.

Containment isolation valves were designed to provide a barrier on the system lines that penetrate the containment so that no event can result in loss of isolation or leakage in excess of Technical Specification limits. Except when not required by activities conducted under approved procedures, containment isolation is maintained by ensuring that all penetrations required to be closed during accident conditions, except those listed in Table 6.4, have at least one deactivated automatic valve secured in the isolated position, or at

Table 6.1 Containment Isolation[a]

Penetration Number	Service	Operational System	Line Size (inch)	Isolation Valve	Status
R-401	Reactor Building Basement Level Indication	Yes	1/2	RBS-IV-1009	Operational
			1/2	RBS-IV-1011	Operational
			1/2	RBS-IV-1013	Operational
			1/2	RBS-IV-1014	Operational
R-524	Fuel Transfer Canal Fill Line	No	10	SF-V105	L.C.[b]
R-525	Decay Heat Coolant Letdown	No	12	DH-V3	Deenergized-L.C.
			1/2	DH-V225	Manual-L.C.
R-526	Steam Generator "A" Sample Line	No	1/2	CA-V8	Deenergized-L.C.
R-527	Core Flooding Tank Bleed and Sample	No	1	CF-V144	Deenergized-L.C.
R-528	Steam Generator "B" Sample Line	No	1/2	CA-V9	Deenergized-L.C.
R-529	Reactor Coolant Drain Pump Discharge	Yes	4	WDL-V1125	Deenergized-L.C. (deactivated portion)
R-530	Steam Generator Side Vent and Drain	No	2	SV-V55	Deenergized-L.C.
			1	SV-V66	Manual-L.C.
R-531	Decay Heat Closed Cooling Water for Reactor Coolant Leak Recovery System	No	8	DC-V115	Deenergized-L.C.

See footnotes at end of table.

Table 6.1 (cont'd)

Penetration Number	Service	Operational System	Line Size (inch)	Isolation Valve	Status
R-532	Fuel Transfer Tube	No	30	FH-V1B	L.C.
R-533	Fuel Transfer Tube	No	30	FH-V1A	L.C.
R-535	Demineralized Water	No	3	DW-V28	Manual-L.C.
R-536	Spare	N/A[c]	N/A	N/A	N/A
R-537	Nitrogen and Fill to Core Flooding Tank	No	1	CF-V145	Manual-L.C.
			1	CF-V114B	Manual-L.C.
			1/2	CF-V129B	Manual-L.C.
R-538	Pressurizer Steam and Water Space Sample Line	No	1/2	CA-V10	Deenergized-L.C.
R-539	Defueling Water Cleanup System Isolation	No	8	DC-V103	Deenergized-L.C.
			1/2	DC-V137	Manual-L.C.
R-541	Letdown Line to Purification Demineralizer	No	2 1/2	MU-V376	Deenergized-L.C.
R-542	Defueling Water Cleanup System Borated Water Flush	No	3	DH-V187	L.C.
			1	DH-V205	Manual-L.C.
R-543	Reactor Building Nitrogen Header	No	1	NM-V52	Air Disabled-L.C.
R-544	Nitrogen and Fill to Core Flooding Tank	No	1	CF-V146	Manual-L.C.
			1/2	CF-V129A	Manual-L.C.
			1	CF-V114A	Manual-L.C.

See footnotes at end of table.

Table 6.1 (cont'd)

Penetration Number	Service	Operational System	Line Size (inch)	Isolation Valve	Status
R-545-A	Building Spray Pressure Sensing	No	1	BS-V146	Manual-L.C.
R-545-B	Defueling Water Cleanup System Sample Isolation	No	3/4 3/4	DWC-V038 DWC-V040B	Manual-L.C. Manual-L.C.
R-545-C	Defueling Water Cleanup System Sample Isolation	No	3/4 3/4	DWC-V037 DWC-V040A	Manual-L.C. Manual-L.C.
R-545-D	Reactor Coolant Pump Seal Water Return	No	2	MU-V377	Deenergized-L.C.
R-546	Pressurizer, Reactor Coolant, Once-Through Steam Generator, and Core Flooding Tank Vents	No	4	WDG-V199	Deenergized-L.C.
R-547	Reactor Building Sump Pump Discharge	No (Deactivated portion)	4	WDL-V1126	L.C.
R-548	Fire Protection	Yes	4	FS-V639	Operational
R-549	Reactor Building Inlet Purge Line	Yes	36 4 1/2	AH-V1B AH-V90B AH-V149	Operational L.C. Manual-L.C.
R-550	Reactor Building Inlet Purge Line	Yes	36 4 1/2	AH-V1A AH-V90A AH-V151	Operational L.C. Manual-L.C.

See footnotes at end of table.

Table 6.1 (cont'd)

Penetration Number	Service	Operational System	Line Size (inch)	Isolation Valve	Status
R-551	Reactor Building Outlet Purge Line	Yes	36	AH-V4A	Operational
			10	AH-V52	Operational
			4	AH-V120A	L.C.
			1/2	AH-V153	Manual-L.C.
R-552	Reactor Building Outlet Purge Line	Yes	36	AH-V4B	Operational
			10	AH-V7	Air Disabled-L.C.
			10	AH-V81	Air Disabled-L.C.
			4	AH-V120B	L.C.
R-553	Defueling Water Cleanup System to Reactor Building Isolation	No	2	WDL-V1092	Deenergized-L.C.
R-554-A	Instrument Air, Purge	No	1/2	AH-V60	Deenergized-L.C.
R-554-B	Air Sample Supply (Radiation Detection)	No	1	AH-V101	Deenergized-L.C.
			1/2	AH-V169	Manual-L.C.
R-554-C	Building Spray Pressure Sensing	No	1	BS-V147	Manual-L.C.
R-554-D	Instrument Air to Defueling Water Cleanup System	No	3/4	DWC-V316	Manual-L.C.
			3/4	DWC-V318	Manual-L.C.
R-555-A	Air Sample Supply (Radiation Detection)	No	1	AH-V105	Deenergized-L.C.
			1/2	AH-V168	Manual-L.C.

See footnotes at end of table.

Table 6.1 (cont'd)

Penetration Number	Service	Operational System	Line Size (inch)	Isolation Valve	Status
R-555-B	Air Sample Return (Radiation Detection)	No	1/2 1/2	AH-V102 AH-V171	Deenergized-L.C. Manual-L.C.
R-555-D	Air Sample Return (Radiation Detection)	No	1/2 1/2	AH-V107 AH-V170	Deenergized-L.C. Manual-L.C.
R-557	To Reactor Coolant Pump Oil and Motor Coolers	No	8 1/2	NS-V72 NS-V210	Air Disabled-L.C. Manual-L.C.
R-558	From Reactor Coolant Pump Oil and Motor Coolers	No	8	NS-V81	Air Disabled-L.C.
R-559	Intermediate Cooling to Roller Nut Drive Cooling Coils	No	3	IC-V5	Air Disabled-L.C.
R-561	High-Pressure Water	No	1 1	TDW-V001 TDW-V003	Manual-L.C. Manual-L.C.
R-561	Decontamination Service Air	No	1 3/4	DSA-V004 DSA-V006	Manual-L.C. Manual-L.C.
R-562-A	Instrument Air Supply	No	1/2	AH-V62	Deenergized-L.C.
R-562-B	Pressure Transfer Fans	No	1 1/2	AH-V5 AH-V147	Deenergized-L.C. Manual-L.C.
R-562-C	Building Spray Pressure Sensing	No	1	BS-V148	Manual-L.C.

See footnotes at end of table.

Table 6.1 (cont'd)

Penetration Number	Service	Operational System	Line Size (inch)	Isolation Valve	Status
R-562-D	Reactor Building Sludge Transfer	No No	1 1	WDS-FV612 WDS-FV614	Manual-L.C. Manual-L.C.
R-563	Intermediate Cooling System	No	6 1/2	IC-V4 IC-V207	Air Disabled-L.C. Manual-L.C.
R-565	Processed Water Supply to Reactor Building	No	3 1	PW-V69 PW-V99	Manual-L.C. Manual-L.C.
R-566	Service Air	No	2 1/2	SA-V20	L.C.
R-567	Intermediate Cooling System	No	6	IC-V3	Air Disabled-L.C.
R-569	Secondary System Flush and Drain	No	3	SV-V18	L.C.
R-570	High-Pressure Injection Line	No	2 1/2 1/2	MU-V16A MU-V315	Deenergized-L.C. Manual-L.C.
R-571-A	Integrated Leak Rate Test	No	1	Blind Flange	
R-571-C	Building Spray Pressure Sensing	No	1	BS-V149	Manual-L.C.
R-571-D	Integrated Leak Rate Test	No	1	Blind Flange	
R-572	High Pressure Injection Line and Makeup	No	1/2 2 1/2 2 1/2	MU-V316 MU-V16B MU-V18	Manual-L.C. Deenergized-L.C. Air Disabled-L.C.

See footnotes at end of table.

Table 6.1 (cont'd)

Penetration Number	Service	Operational System	Line Size (inch)	Isolation Valve	Status
R-573	Reactor Coolant Pump Seal	No	2	MU-V378	Deenergized-L.C.
R-574	Water Supply		3/4	MU-V330	Manual-L.C.
R-575			1/2	MU-V364	Manual-L.C.
R-576			1	MU-V439	Deenergized-L.C.
R-577	Reactor Building Air	No	8	RR-V5A	Deenergized-L.C.
	Coolers		1	RR-V28A	Manual-L.C.
			1/2	RR-V86	Manual-L.C.
R-578	Reactor Building Air	No	6	RR-V11A	Air Disabled-L.C.
	Coolers		1/2	RR-V75A	Manual-L.C.
			6	RR-V25A	Air Disabled-L.C.
R-579	Reactor Building Air	No	8	RR-V5B	Deenergized-L.C.
	Coolers		1	RR-V28B	Manual-L.C.
			1/2	RR-V88	Manual-L.C.
R-580	Reactor Building Air	No	8	RR-V5C	Deenergized-L.C.
	Coolers		8	RR-V6C	Deenergized-L.C.
			1	RR-V28C	Manual-L.C.
			1/2	RR-V90	Manual-L.C.
			1/2	RR-V92	Manual-L.C.
R-581	Reactor Building Air	No	6	RR-V11C	Air Disabled-L.C.
	Coolers		1/2	RR-V75C	Manual-L.C.
			6	RR-V25C	Air Disabled-L.C.
R-582	Reactor Building Air	No	6	RR-V11B	Air Disabled-L.C.
	Coolers		1/2	RR-V75B	Manual-L.C.
			6	RR-V25B	Air Disabled-L.C.

See footnotes at end of table.

Table 6.1 (cont'd)

Penetration Number	Service	Operational System	Line Size (inch)	Isolation Valve	Status
R-583	Reactor Building Spray Inlet Line	No	8	BS-V1B	Deenergized-L.C.
			3	BS-V130B	Manual-L.C.
R-584	Reactor Building Air Coolers	No	8	RR-V6D	Deenergized-L.C.
			1	RR-V28D	Manual-L.C.
			1/2	RR-V94	Manual-L.C.
			1	RR-V98	Manual-L.C.
R-585	Reactor Building Air Coolers	No	6	RR-V11D	Air Disabled-L.C.
			1/2	RR-V75D	Manual-L.C.
			6	RR-V25D	Air Disabled-L.C.
R-586	Reactor Building Spray Inlet Line	No	8	BS-V1A	Deenergized-L.C.
			3	BS-V130A	Manual-L.C.
R-587	Reactor Building Air Coolers	No	8	RR-V6E	Deenergized-L.C.
			1	RR-V28E	Manual-L.C.
			1/2	RR-V96	Manual-L.C.
R-588	Reactor Building Air Coolers	No	6	RR-V11E	Air Disabled-L.C.
			1/2	RR-V75E	Manual-L.C.
			6	RR-V25E	Air Disabled-L.C.
R-589	Decay Heat Coolant Supply	No	10	DH-V4A	Deenergized-L.C.
R-590	Decay Heat Coolant Supply	No	10	DH-V4B	Deenergized-L.C.
R-591	High-Pressure Injection Line	No	2 1/2	MU-V16C	Deenergized-L.C.
		No	1/2	MU-V366	Manual-L.C.

See footnotes at end of table.

Table 6.1 (cont'd)

Penetration Number	Service	Operational System	Line Size (inch)	Isolation Valve	Status
R-592	High-Pressure Injection Line	No	2 1/2	MU-V16D	Deenergized-L.C.
			1/2	MU-V368	Manual-L.C.
R-593	Sump Penetration Sleeve and Drain Line	No	18	DH-V6A	Deenergized-L.C.
R-594	Sump Penetration Sleeve and Drain Line	No	18	DH-V6B	Deenergized-L.C.
R-616	Emergency Feedwater to Once-Through Steam Generator "B"	No	6	EF-V12B	Deenergized-L.C.
			4	EF-V33B	Deenergized-L.C.
			3/4	EF-V36	Manual-L.C.
			6	Blind Flange	
			6	Blind Flange	
R-617	Feedwater to Once-Through Steam Generator "B"	No	20	FW-V17B	Deenergized-L.C.
			6	FW-V19B	Deenergized-L.C.
			3/4	FW-V68B	Manual-L.C.
			3/4	FW-V35B	Manual-L.C.
			2	GR-V7B	Manual-L.C.
R-618	Feedwater to Once-Through Steam Generator "A"	No	20	FW-V17A	Deenergized-L.C.
			6	FW-V19A	Deenergized-L.C.
			3/4	FW-V68A	Manual-L.C.
			3/4	FW-V35A	Manual-L.C.
			2	GR-V7A	Manual-L.C.

See footnotes at end of table.

Table 6.1 (cont'd)

Penetration Number	Service	Operational System	Line Size (inch)	Isolation Valve	Status[a]
R-619	Main Steam to Turbine	No	24	MS-V7B	Deenergized-L.C.
			10	MS-V15B	Deenergized-L.C.
			3/4	MS-V224	Manual-L.C.
			1	MS-V51B	Manual-L.C.
R-620	Main Steam to Turbine	No	24	MS-V4B	Deenergized-L.C.
			6	MS-V1B	Manual-L.C.
			3/4	MS-V2B	Manual-L.C.
			4	MS-V11B	Deenergized-L.C.
			1	MS-V50B	Manual-L.C.
			3/4	MS-V225	Manual-L.C.
			2	GR-V1B	Manual-L.C.
R-621	Main Steam to Turbine	No	24	MS-V7A	Deenergized-L.C.
			10	MS-V15A	Deenergized-L.C.
			1	MS-V51A	Manual-L.C.
			4	MS-V11A	Deenergized-L.C.
			3/4	MS-V227	Manual-L.C.
R-622	Main Steam to Turbine	No	24	MS-V4A	Deenergized-L.C.
			6	MS-V1A	Manual-L.C.
			3/4	MS-V2A	Manual-L.C.
			1	MS-V50A	Manual-L.C.
			3/4	MS-V226	Manual-L.C.
			2	GR-V1A	Manual-L.C.

See footnotes at end of table.

Table 6.1 (cont'd)

Penetration Number	Service	Operational System	Line Size (inch)	Isolation Valve	Status
R-623	Emergency Feedwater to Once-Through Steam Generator "A"	No	6	EF-V12A	Deenergized-L.C.
			4	EF-V33A	Deenergized-L.C.
			6	Blind Flange	
			6	Blind Flange	
R-626	Sump Sucker Feedline (SWS-P-1 Retagged WDL-P-2C)	No	2	SWS-FV-001	Blind Flanged

(a) Data obtained from Table 7.2-2 of GPU 1990c.
(b) L.C. - locked closed.
(c) N/A - not applicable.

Table 6.2 Modified Containment Penetrations[a]

Penetration Number	Modification
R-401	The penetration was modified in late summer 1979 to allow access to obtain samples of the reactor building sump water. Following successful completion of the sampling program, further changes were made to use the penetration for reactor building water level measurement by adding a manometer system to the sampling tube. A further modification was made to provide a more permanent closure in consideration of future potential increase of reactor building water level to the extent of flooding the penetration. This final modification consisted of removing the 12-inch (30.5-centimeter) gate valve and the special cover assembly outboard of the valve and welding a closure assembly to the penetration.
R-537	The penetration was modified to provide a flow path into and out of the reactor building for the defueling water cleanup system. This modification consisted of adding a pipe for hose connections to the isolation test connections inside and outside the containment and adding a blind flange on the outboard side to isolate the system from the plant nitrogen.
R-539	The penetration was modified to provide a flow path into and out of the reactor building for the defueling water cleanup system. This modification consisted of tying-in with pipe[b] and adding a second containment isolation valve on the outboard side and providing piping for a hose connection on the inboard side.
R-542	The penetration was modified to provide a flow path into and out of the reactor building for the defueling water cleanup system. This modification consisted of tying-in with pipe[b] and adding a second containment isolation valve on the outboard side and providing piping for a hose connection on the inboard side.
R-544	The penetration was modified to provide a flow path into the reactor building for a high pressure decontamination water supply. This modification consisted of replacing an existing 1-inch (2.5-centimeter) line with a 1-inch (2.5-centimeter) XXS carbon steel pipe.
R-545	This spare penetration was modified to provide a flow path into and out of the reactor building for the defueling water cleanup system. This modification consisted of adding a double valve pressure boundary on the outboard side and providing piping for a hose connection on the inboard side.

See footnotes at end of table.

Table 6.2 (cont'd)

Penetration Number	Modification
R-546	The penetration was modified to provide a flow path into and out of the reactor building for the defueling water cleanup system. This modification consisted of providing piping for a hose connection on the inboard side and tying-in with pipe[b] on the outboard side.
R-553	The penetration was modified to provide a flow path into and out of the reactor building for the defueling water cleanup system. This modification consisted of providing piping for a hose connection on the inboard side and tying-in with pipe[b] on the outboard side.
R-554	The penetration was modified to provide a source of clean compressed air for use with pneumatic controls and operators. This modification consisted of replacing two outboard containment isolation valves with three valves and adding a flow limiter and quick disconnect on the inboard side.
R-562	The penetration was modified to provide a flow path for sludge transfer from the reactor building to the spent resin storage tank in the auxiliary building. This modification consisted of adding a double-valve pressure boundary on the outboard side and piping for a hose connection on the inboard side.
R-565	The penetration was modified to provide a means of transferring shield water to the containment sump. This modification consisted of adding a piping spool assembly to the outboard side of the penetration.
R-626	The penetration was initially modified to insert an antenna and camera arrangement into the containment. These were subsequently removed, and the penetration was modified to allow pumping out of the reactor building basement. This modification consisted of installing a new spool piece and piping. For PDMS, all existing piping will be removed and a cover bolted over the penetration with a single pipe centered in the cover. This pipe will be blind flanged for future use.

(a) Information was obtained from GPU 1991a, Supplement 1, Table 2-1.
(b) "Tying-in with pipe" can be effected in one of two ways: (1) cutting the existing pipe completely and adding a "T" fitting or (2) cutting a circular hole in the existing pipe and welding a new pipe to it. In either case, the result is two parallel flow paths where one previously existed.

Table 6.3 Penetrations Considered Operational During Post-Defueling Monitored Storage

Penetration Number	Service	Line Size (inch)	Isolation Valve
R-529	Reactor Coolant Drain Pump Discharge	4	WDL-V1125
R-548	Fire Protection	4	FS-V639
R-549	Reactor Building Inlet Purge Line	36	AH-V1B
R-550	Reactor Building Inlet Purge Line	36	AH-V1A
R-551	Reactor Building Outlet Purge Line	36 10	AH-V4A AH-V52
R-552	Reactor Building Outlet Purge Line	36	AH-V4B
R-401	Reactor Building Basement Level Indication	1/2 1/2 1/2 1/2	RBS-IV-1009 RBS-IV-1011 RBS-IV-1013 RBS-IV-1014

Data obtained from GPU 1991, Table 7.2.2

Table 6.4 Penetrations That Maintain Isolation Using a Check Valve Inside the Containment[a]

Penetration Number	Outside Containment Isolation Valve
R573	MU-V-379
R574	MU-V-380
R575	MU-V-381
R576	MU-V-382
R577	RR-V-5A
R579	RR-V-5B
R584	RR-V-6D
R587	RR-V-6E
R580	RR-V-5C and 6C
R583	BS-V-1B and 130B
R586	BS-V-1A and 130A
R589	DH-V-4A
R590	DH-V-4B
R591	MU-V-16C
R592	MU-V-16D
R537	CF-V-114B
R539	DC-V-103
R542	DH-V-187
R544	CF-V-114A
R557	NS-V-72
R559	IC-V-5
R563	IC-V-4
R566	SA-V-20
R570	MU-V-16A
R572	MU-V-16B

(a) Data were obtained from Table B-1 GPU 1989f.

least one closed manual valve or a blind flange. Table 6.4 lists 25 penetrations that maintain isolation using a check valve inside the containment as the isolation valve (although they also have associated outside containment isolation valves) and 14 penetrations that have single isolation valves. The remaining penetrations have double isolation valves or welded or bolted flanges. There will be one exception during PDMS: the containment breather isolation valve, which will normally remain open to the AFHB atmosphere through the breather filter and will generally be maintained at ambient or possibly slightly negative pressure with respect to the atmosphere. (See Section 5.4.1 of this report.)

The containment structure also has two air locks to allow entry and exit of personnel and equipment. Both air locks are manually operated and require no outside power to function.

6.2.2 Reactor Containment Structure - Current Licensing Basis

Under the current Technical Specification requirements, primary containment isolation is maintained. Containment isolation exists when each penetration is closed by an accessible manual valve, a welded or bolted flange, or a deactivated automatic valve secured in the closed position. A containment penetration may be opened following an approved procedure, provided it can be closed as specified above. In addition, the equipment hatch must be closed and sealed and each containment air lock must be operable. Containment isolation is verified quarterly. However, penetrations that are closed by bolted or welded blind flanges are not required to be surveyed. Containment air locks are demonstrated operable after each opening (if not demonstrated operable within the last 72 hours) by verifying the seal leak rate and at least once every 3 months by performing a mechanical operability check of each air lock, including a visual inspection and lubrication if necessary.

In addition, the current Technical Specifications require that the primary containment pressure be maintained between 0 psig to not less than 12.2 psia through the use of the containment purge system.

6.2.3 Reactor Containment Structure - PDMS Licensing Basis

The proposed PDMS Technical Specifications specify that primary isolation be maintained. The proposed PDMS Technical Specifications require quarterly verification of containment isolation with the following exceptions:

1. Isolation valves that are locked closed shall be verified annually on a quarterly staggered test basis. If a valve is found to be out of position, a verification check of all locked-closed isolation valves shall be performed.

2. An independent verification of all changes in isolation valve position shall be performed.

3. Bolted or welded blind flanges that form a containment isolation boundary will be visually inspected for signs of degradation and/or leakage every 5 years on an annually staggered test basis. If a problem is discovered with a flange, a complete verification shall be performed.

The proposed PDMS Technical Specifications 3.1.1.3 also require that each containment air lock will be operable with at least one door closed. In addition, each containment air lock will be demonstrated operable once every 3 months by a mechanical operability check, including a visual inspection and lubrication, as necessary, and visual inspection of the door seals for degradation (proposed PDMS Technical Specification 4.1.1.3). An exception to this requirement on the operability of the doors is made when the air lock is being used for transit entry and exit in accordance with site-approved procedures. When both air lock doors are opened simultaneously, the following conditions should be verified (proposed PDMS Technical Specification 4.1.1.3):

1. The capability exists to expeditiously close at least one air lock door.

2. The air lock doors and containment purge are configured to restrict the outflow of air in accordance with site-approved procedures.

3. The air lock doors are cycled to ensure mechanical operability within seven days before the opening of both doors.

The proposed PDMS Technical Specifications 3.1.1.2 require that the unfiltered leak rate from containment with the reactor building breather closed shall be less than 1/100 of the rate through the reactor building breather. If the unfiltered leak rate from containment with the reactor building breather closed is greater than 1/100 of the rate through the reactor building breather, or if the trend indicates that the 1/100 value will be exceeded within 1 year, then the licensee is required to

1. Identify the excessive leak path

2. Make necessary repairs or adjustments

3. Perform an additional unfiltered leak rate test

4. Prepare and submit a special report to the NRC within the next 30 days.

To ensure that this leak rate is not exceeded, an unfiltered leak rate test will be conducted periodically (proposed PDMS Technical Specification 3.1.1.2 and license condition 2E) to verify the leak-tightness of the reactor containment structure and that the breather is the primary leak pathway.

6.2.4 Reactor Containment Structure - Conclusion

The integrity of the reactor containment structure must be maintained during PDMS to ensure that releases of radioactive material are maintained as low as is reasonably achievable and within the limits established in the proposed

PDMS Technical Specifications. The proposed maintenance of containment isolation and routine surveillance for verification provide a level of surveillance equivalent to that currently implemented, except for an increased level of surveillance of bolted or welded blind flanges. The NRC staff agrees that an increase in surveillance for these items is appropriate, considering the reduced probability for chance discovery of an open flange by personnel as a result of fewer entries into the facility and the longer time periods between entries.

Requirements for surveillance of the atmospheric breather filter are contained in the PDMS SAR 7.2.1.2 (GPU 1991b) and discussed in Section 6.3.2 of this technical evaluation report. These requirements specify testing to ensure proper seating and filtration efficiency, and verification that the integrity of the filter is maintained.

Thus, the NRC staff concludes that maintenance of the building in the current configuration, as described above and in Section 7.2 of the PDMS SAR, will provide control of radioactive material to ensure that potential releases are maintained within the limits required by the regulations and prescribed in the proposed PDMS Technical Specifications. The required surveillance will further ensure that containment isolation capability is verified and maintained.

6.3 Reactor Containment and Auxiliary and Fuel Handling Building Purge, Breather, Ventilation and Filtration Systems

The ventilation and filtration systems maintain a negative pressure on the buildings to ensure that any leakage is into the buildings, that potentially contaminated air is filtered before it is discharged to the atmosphere, and that this air is directed to the station vent, where it can be monitored and the concentration of radioactive material in the air measured.

The capability for active ventilation of the reactor building, the auxiliary building, and the fuel-handling building will be maintained during PDMS (GPU 1991d; SAR 7.2.1.3.2, 7.2.6.1, and 7.2.6.2). A schematic of the ventilation system is given in Figure 6.1. Three systems, the reactor building purge system, the auxiliary building ventilation system, and the fuel-handling building ventilation system, will be operational. The reactor building purge systems will be discussed first, followed by a discussion of the ventilation systems in both portions of the auxiliary and fuel-handling building.

6.3.1 Reactor Building Purge System

6.3.1.1 Reactor Building Purge System - System Description

Currently, the reactor containment building is normally ventilated continuously using the reactor building purge exhaust system. This system consists of two containment purge exhaust units and associated duct work, dampers, and filters. During active ventilation, the purge units (25,000 cubic feet per minute [708 cubic meters per minute] each) draw air from the D-rings through HEPA filters and discharge either to the station vent or back into the containment. As effluent from the reactor building is routed through the

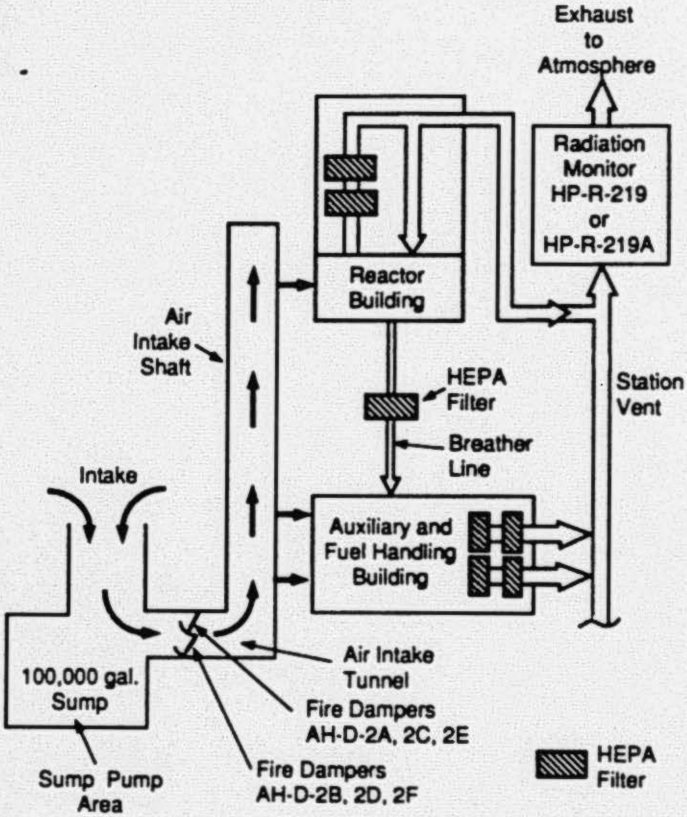


FIGURE 6.1. Ventilation System During Post-Defueling Monitored Storage

reactor building purge system to the station vent, it passes through two HEPA filter banks in series. The station vent is continuously monitored during reactor containment ventilation by an effluent monitor in the vent (HP-R-219 or HP-R-219A). A local differential pressure indicator is installed across each HEPA filter. These differential pressure indicators are checked monthly while the ventilation system is in service.

During PDMS, the reactor building purge exhaust system will not be operated continuously. Periodic entries will be made into the reactor building for measurement and surveillance activities. Before reactor containment entry, the containment will be ventilated as necessary, using the reactor building purge system (GPU 1992, SAR 7.2.4.3). At other times, the ventilation system will not be operated, although it will be kept operational as specified in the SAR 7.2.1.3 (GPU 1991d). Thus, during PDMS, it is anticipated that the ventilation systems could be out of service for extended periods, and, therefore, the HEPA filters would not be checked during this time. However, because exhaust lines from the reactor building are closed, there would be no release through the reactor building vent. The reactor building purge system will not be returned to service to ventilate the containment for entry unless the applicable surveillance requirements for the filter systems have been met. The HEPA filter systems will be leak-tested in place after they are installed and whenever filters are replaced to ensure that they meet the required performance specifications.

A containment atmospheric breather (GPU 1991b, SAR 7.2.1.2) will be used to maintain pressure equilibrium between the AFHB and the reactor building when the containment is not actively ventilated. The system is designed to provide a specific pathway through which the containment atmosphere can be aspirated to maintain pressure equilibrium with the environment external to the containment. The containment atmospheric breather also provides a HEPA-filtered pathway for effluent from the containment. The breather is a 6-inch (15-centimeter) former hydrogen control system line in which a HEPA filter (24 inches by 24 inches [61 centimeters by 61 centimeters]) has been installed between the reactor building and the exhaust fan (AH-E-34). The containment atmospheric breather exhausts into and draws air from the interior of the AFHB. The breather is the most probable pathway for passive ventilation because the line is large compared to other potential leak paths. There will be an isolation valve between containment and the breather HEPA filters that will automatically close upon receipt of a containment pressure increase of 0.25 psi (GPU 1991b, SAR 7.2.1.2.2). The purpose of this isolation is to protect the breather HEPA filters in the event of a significant fire in the containment.

During PDMS, when the auxiliary building ventilation system is operating but the reactor containment is not being actively ventilated, a slight negative pressure will be maintained on the reactor containment building through the breather line from the AFHB. Two sample filter papers will be placed downstream of the HEPA filter and before the location where it exhausts into the auxiliary building. Airflow in the containment atmospheric breather may be continuously reversing direction, depending on the mode of operation of the ventilation systems and atmospheric pressure. The second filter paper downstream from the HEPA filter will be used to protect the first filter paper from any extraneous radioactivity entering from the auxiliary building. The two filter papers will be removed semiannually, and the first paper downstream of the HEPA will be analyzed for radionuclide content. The licensee will report the results of the assay of the first filter paper (assuming a filtration efficiency of 50 percent, although the efficiency of the sample filter paper will most likely be greater) as though a like quantity of radionuclides was released. This will provide a qualitative estimate of the amount of radioactive material exhausted from the reactor containment building (GPU 1991d, SAR 7.2.1.2.3). Since the deposition on the filter paper will be cumulative, it will provide determinative (but not real-time) monitoring during PDMS as a result of changes in ambient atmospheric pressure.

6.3.1.2 Reactor Building Ventilation - Current Licensing Basis

The current Technical Specification 3.6.3.1 requirement ensures that at least one train of the reactor building purge exhaust system is operable at all times and capable of performing its functions, including all necessary instrumentation, controls, normal and emergency electrical power sources, cooling or seal water, lubrication or other auxiliary equipment that is required for the system, subsystem, train, component, or device. Requirements also include flow tests and filter tests at specified frequencies and periodic verification of differential pressure. The current Technical Specifications also establish requirements for monitoring of the gaseous effluent and verification of the operability and calibration of the monitoring equipment. The HEPA filter systems were leak tested in place when they were installed and whenever the filters are replaced to ensure that they meet the required performance specifications. The internal pressure of the reactor building is kept negative in relation to the atmosphere by active ventilation of the building. The atmospheric breather is not currently in use. The atmospheric breather system is a modification planned specifically for PDMS.

6.3.1.3 Reactor Building Ventilation - PDMS Licensing Basis

The licensee has committed in Section 7.2.1.3.2 of the PDMS SAR (GPU 1991d) to maintain the reactor building purge exhaust system in an operational condition to support PDMS activities (surveillance and maintenance entries) in the reactor containment building. Maintenance of an operational condition includes leak testing of filters and monitoring of the effluent during operation of the ventilation system. However, this system is not a safety related system necessary to mitigate the consequences of an accident and limit offsite dose to within 10 CFR Part 100 limits considering the post-accident, inoperable essentially defueled condition of the facility.

The licensee has also committed to ensure that the atmospheric breather system is the predominant pathway for effluent and influent to the building (during those times that the reactor building ventilation system is not being operated) and the effluent is filtered and monitored. As described in Section 6.2.3 of this report and in the proposed PDMS Technical Specifications 3.1.1.2, periodic testing and surveillance will be conducted to verify that the atmospheric breather system remains the predominant pathway. The HEPA filter installation in the containment atmospheric breather provides in-place leak testing of the filter and housing. The HEPA filter will be leak tested before installation (SAR 7.2.1.2.2 and B-3 of Supplement 3; GPU 1990c and 1991b). In addition, the HEPA filter installation will be tested in place before it is used and whenever the filter is replaced. The frequency of reinstallation or replacement of the filter will depend on the results of the leak testing and other surveillance to ensure integrity of the installation.

6.3.1.4 Reactor Building Ventilation - Conclusion

The NRC staff has concluded that the licensee's program will ensure that the effluent from the reactor building will be filtered and monitored during both active and passive ventilation, that the atmospheric breather system will be the most probable pathway during passive ventilation, that tests and surveillance activities will ensure that the atmospheric breather system remains the predominant pathway, and that the operability of the reactor containment purge system will be maintained to provide ventilation before entry of personnel into the enclosed reactor containment building.

6.3.2 Auxiliary and Fuel Handing Building Ventilation System

6.3.2.1 Auxiliary and Fuel Handling Building Ventilation - System Description

The auxiliary building's side of the AFHB's ventilation system exhausts through dual HEPA filters into the station vent. The station vent is required to be continuously monitored during ventilation system operation using an effluent monitor in the vent stack, HP-R-219 or HP-R-219A. Operation of the auxiliary building ventilation system provides a negative pressure on the AFHB and on the reactor containment building through the breather. The AFHB is not a leak tight structure. Currently, the potential movement of radioactive material outside the building is minimized by maintaining a negative air pressure between the building and the outside atmosphere.

The fuel-handling building side of the AFHB is essentially a four-floor building that shares a common wall with the auxiliary building side of the AFHB. On all levels, there are doors that allow the passage of personnel through the common wall. The effluent from the fuel-handling building ventilation system exhausts to the station vent, which is continuously monitored during ventilation system operation. The space in the fuel handling building above the 347-foot elevation is common to TMI-1 and TMI-2, as are the truck bay (elevation 305 feet) and the standby pressure control pit (elevation 328 feet). The operating floor (elevation 347.5 feet) is ventilated by the TMI-1 ventilation system, which maintains a constant negative pressure in the area. Air withdrawn through the TMI-1 ventilation system is filtered by the Unit 1 station vent, which is continuously monitored.

The licensee has stated in the PDMS SAR 7.2.6.1 (GPU 1990c) that during PDMS, the auxiliary building ventilation system and filters will be kept in an operational condition to support PDMS activities. However, they will not be required to operate continuously. The licensee has further stated (GPU 1991e, SAR 7.2.4.3) that the ventilation system will be operated during PDMS for a minimum of 1 year. License Condition 2D states that prior to terminating continuous operation of the auxiliary and fuel handling building ventilation systems, the special monitoring program on particulate releases will be completed. The program shall include at least 1 year of data prior to entry into PDMS and at least 1 year of data after entry into PDMS. A report containing the results of the program and containing sufficient data and analyses to demonstrate that the release rate of particulates with half-lives greater than 8 days from the AFHB will be less than 0.00024 μCi/sec when averaged over any calendar quarter shall be submitted to the NRC staff at least 60 days prior to terminating continuous operation of the AFHB ventilation systems.

The licensee has also stated in the PDMS SAR 7.2.6.2 (GPU 1990c) that the fuel handling building ventilation system and filters will be kept in an operational condition and operated as required to support PDMS activities. The fuel handling building ventilation system provides fresh air to and filters effluent air from the portion of fuel handling building below the spent fuel pool operating deck. The effluent from the fuel-handling building ventilation system exhausts to the station vent where it is continuously monitored during system operation. The lower three floors of the auxiliary building and fuel handling building (281-foot, 305-foot, and 328-foot elevations) are open to each other allowing for the free passage of air. When the ventilation system is not operating, the AFHB ventilation will equalize to atmospheric pressure via the HEPA filtered exhaust line which will remain open.

The TMI-2 spent fuel pool operating deck is open to the truck bay and TMI-1 spent fuel pool operating deck. This common volume will be actively ventilated from the TMI-1 fuel handling building ventilation system. The lower elevations of the TMI-2 fuel handling building will be isolated from this common volume by maintaining doors which communicate between the two areas closed, sealing wall and floor penetrations, and closing leak tight isolation dampers on the communicating portions of the ventilation system.

6.3.2.2 Auxiliary and Fuel Handling Building Ventilation - Current Licensing Basis

The current TMI-2 Technical Specifications for the AFHB ventilation system provide assurance that gaseous effluent from the facility is controlled, filtered, and monitored before release to the environment. The operation of the ventilation system provides a negative pressure within the facility to ensure that contamination is retained within the building. Thus, atmospheric control is maintained by active ventilation of the facility. The current Technical Specifications specify that the auxiliary building air cleanup system and the fuel handling building air cleanup exhaust system shall be kept operable, with one of the four system air cleanup exhaust fans operable in each system. The operability requirement for these systems requires operability surveillance and testing of supporting components such as HEPA filters and differential pressure control. Both the auxiliary building and the fuel handling building exhaust systems discharge into the station vent, thus providing continuous monitoring of radioactive material concentrations in the effluent. The current Technical Specifications require operation of the effluent monitor and appropriate maintenance and calibration.

6.3.2.3 Auxiliary and Fuel Handling Building Ventilation - PDMS Licensing Basis

Gaseous effluent released from the AFHB must be kept below the limits specified by the licensee in the ODCM and the PDMS SAR. License Condition 2D states that prior to terminating continuous operation of the auxiliary and fuel handling building ventilation systems, the special monitoring program on particulate releases will be completed. The program shall include at least 1 year of data prior to entry into PDMS and at least 1 year of data after entry into PDMS. A report containing the results of the program and containing sufficient data and analyses to demonstrate that the release rate of particulates with half-lives greater than 8 days from the AFHB will be less than 0.00024 $\mu Ci/sec$ when averaged over any calendar quarter shall be submitted to the NRC staff at least 60 days prior to terminating continuous operation of the AFHB ventilation systems. In addition, the licensee shall operate the AFHB ventilation system until the accident-generated water is no longer being processed or transferred within the AFHB. Operability and surveillance requirements for the AFHB ventilation systems are provided in the PDMS SAR (GPU 1990c, SAR 7.2.6).

6.3.2.4 Auxiliary and Fuel Handling Building - Conclusion

The operation of the auxiliary building air cleanup system during PDMS will provide assurance that potential movement of radioactive material outside the AFHB is minimized and that gaseous effluent discharged from the facility will be monitored. Demonstration by the licensee that the release rate of particulates with half-lives greater than 8 days from the AFHB will be less than 0.00024 μCi/sec when averaged over any calendar quarter, will ensure that releases of radioactive material to the environment are less than the limits specified in 10 CFR Part 20.

6.4 Fire Detection and Protection System

Fire protection for a reactor facility is of regulatory concern to ensure safe shutdown of the reactor and to provide control of releases of radioactive material. Because of the post-accident, inoperable, essentially defueled condition of TMI-2, safe shutdown of the reactor is no longer of concern. The objective of the fire protection system during PDMS is to limit offsite doses if a fire were to occur. The quantity of radioactive material that could potentially be released to the public is dependent on the magnitude of the fire. The NRC staff analyzed the offsite release from fires that could occur during PDMS (Section 5.4.3 of this report). Offsite releases from any credible fire would result in only a small percentage of the dose specified in 10 CFR Part 100 for determination of exclusion areas. Nevertheless, because of the unique nature of TMI-2, some fire detection and manual suppression capability is advisable.

6.4.1 Fire Detection and Protection System - System Description

The objective of a fire protection system is to limit offsite doses if a fire were to occur. This is accomplished by (1) providing zone detection systems, (2) providing automatic fire suppression to areas of the facility and systems which contain significant amounts of combustibles and possible ignition sources, (3) providing hose reel and hose cabinet stations in areas with combustibles, and (4) providing a trained fire brigade to respond to fires.

The staff concluded, based on the post-accident, inoperable, essentially defueled condition of the facility, and the results of analyses that demonstrate that the maximum credible fire would not result in offsite doses in excess of 10 CFR Part 100 limits, that a limited fire protection program was appropriate for TMI-2 during PDMS.

Fire protection is provided during PDMS by a modification of the original TMI-2 fire protection system, as well as by fire potential reduction. During PDMS, the zone detection systems originally provided at TMI-2 will be operational in specific areas of the plant. A total of 28 fire protection zone detection systems will be operational in the TMI-2 facility during PDMS; 22 have been deactivated. Deactivation of the 22 fire protection zones reflect the completion of the current cleanup activities and the reduction in five hazard or in some cases the actual elimination of the structure that was monitored. A list of these systems and their status for PDMS is given in Table 6.5. The zone detectors will be present on all elevations of the reactor building and fuel-handling building. Only operational area ionization detectors are listed in the table; all duct smoke detectors have been deactivated. In addition, equipment-related detectors installed on various components within the plant to monitor a specific hazard and automatically trip the associated fire suppression system have been deactivated for areas in which the hazard has been removed and the related water suppression system has been deactivated.

Remote monitoring capabilities will be provided in the TMI-1 control room, a continuously staffed location, by the PDMS alarm and monitoring system which will identify the specific zone in which the fire is located.

The halon systems protecting the air intake tunnel and relay room have been deactivated. Portable fire extinguishers and self-contained breathing apparatus (SCBA) for firefighting response are located throughout the facility, as shown on Figures 6.2 and 6.3. Portable fire extinguishers are located in the control building south corridor (305-foot elevation), along the west wall of the turbine building (305-foot elevation), in the control building north corridor (305-foot elevation), in the auxiliary building (305-foot elevation just outside the entry to the reactor building), and just outside the service building. Each of these locations has from two to four 20-pound (9-kilogram) ABC dry chemical extinguishers. In addition, the fire brigade lockers in the Unit 1/Unit 2 corridor, the laundry and respirator facility, and the waste-handling and packaging facility are available for fire brigade use.

The deluge systems for the auxiliary transformers and east turbine building wall are maintained in the turbine building. Fire service systems in miscellaneous facilities will be maintained operable as required to support operations (e.g., the waste-handling and packaging facility, the respirator cleaning facility, and the administration building). The deluge systems in the auxiliary building and the control building will be deactivated. There are no deluge systems in the reactor containment building.

All portions of the fire protection system located inside buildings in areas where the fire hazard risk is small have been deactivated (GPU 1991a, SAR 7.2.2.2 (1)). The system has been configured so that the deactivated portions can be reactivated by valves, if necessary.

Table 6.5 Fire Protection Zone Detection Systems[a]

Building	Elevation	Zone	PDMS Status OS[b]	PDMS Status DS[c]
Auxiliary Building	258' 6" and 280' 6"	2	X	
	305' 0"	3	X	
	328' 0"	4A		X
		4B		X
		4C		X
	328' 0" and 347' 6"	4D	X	
Chlorinator House	---	5		X[d]
Circulating Water Pump House	---	6		X[d]
Coagulator Building	---	7	X	
Control Building	280' 6"	8A	X	
		8B	X	
	Mezzanine	8C	X	
	280' 6"	8D	X	
		8E	X	
	351' 6"	8F		X
	305' 0" and 351' 6"	9	X	
	305' 0"	9A	X	
		9B		X
	331' 6"	10	X	
	351' 6"	11A		X
		11B	X	
Control Building Area	280' 6"	12A	X	
		12B	X	
Fire Pump House	312' 0"	15	X	
Fuel-Handling Building	280' 6"	16	X	
	305' 0"	17	X	
	328' 0"	18A		X
		18B		X
	347' 6"	19	X	
Mechanical Draft Cooling Tower	-----	20		X

See footnotes at end of table.

Table 6.5 (cont'd)

Building	Elevation	Zone	PDMS Status OS(b)	DS(c)
Reactor Building	328' 0"	22A		X
		22B		X
		22C		X
	305' 0" and 347' 6"	22D	X	
	347' 6"	22E	X	
	282' 6"	22F	X	
River Water Pump House	280' 9" and 312' 0"	23		X
Service Building	280' 6"	24	X	
	305' 0"	25	X	
	322' 0" and 331' 6"	26A		X
		26B	X	
		26C		X
		26D		X
		26E		X
Turbine Building	281' 6"	27A		X
		27B	X	
	305' 0"	28A		X
		28B		X
		30	X	
	331' 6"	29	X	

(a) Data obtained from Table 7.2.4 of GPU 1991e.
(b) OS - operational status.
(c) DS - deactivated status.
(d) Except detection circuits from miscellaneous yard structures and outbuildings to remain operational as required.

The yard fire main will be kept pressurized using the station fire pumps in TMI-1 and the altitude tanks as a backup pressure source. The TMI-2 fire protection system draws its supply water from the tie-in to the yard fire main. Freeze protection has been added to applicable portions of the fire main because of the lack of normal heating and ventilation in parts of the facility. The principal means of freeze protection outside of the reactor building during PDMS will be electrical heat tracing, which will be maintained using approved maintenance procedures. Large piping manifolds for the fire protection system may be enclosed in a room outside the reactor building in non-combustible areas and provided with a small space heater.

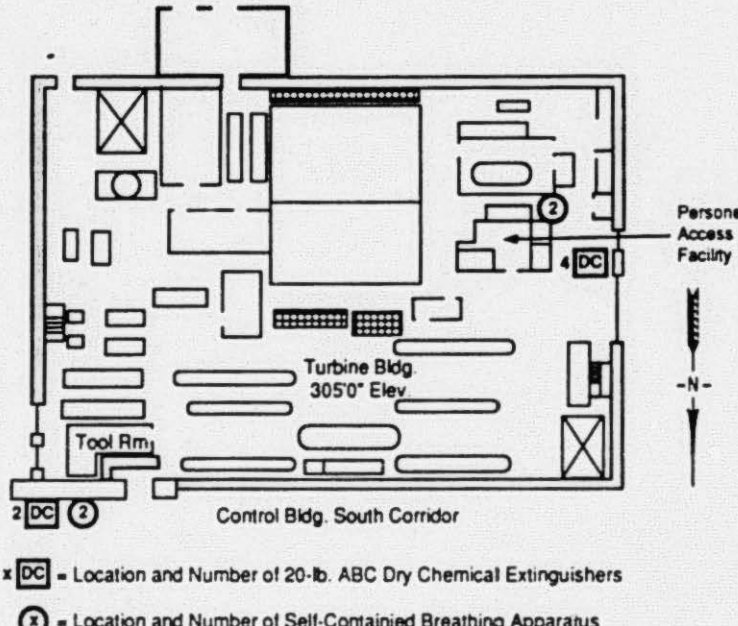
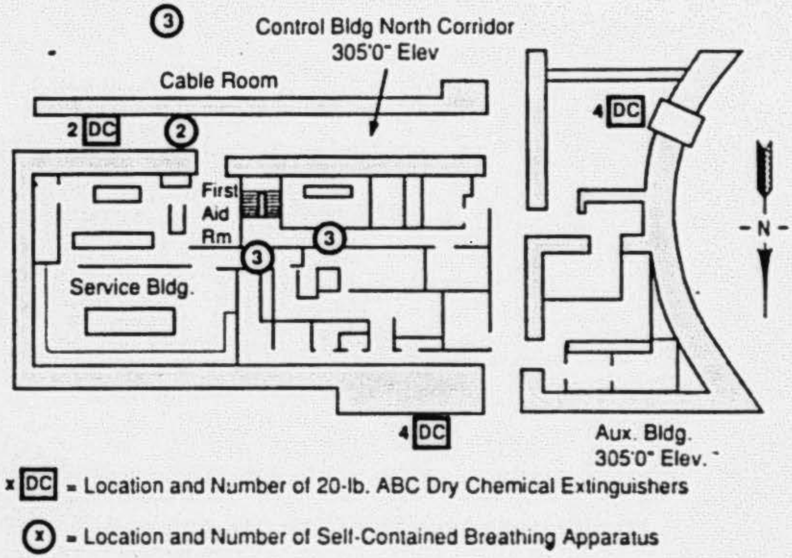


FIGURE 6.2. Portable Extinguisher and Self-Contained Breathing Apparatus Locations in the Turbine Building and Control Building South Corridor

The 12-inch (30.5-centimeter) fire service loop, which runs through the diesel generator building, the AFHB, the control building area, and the turbine building (east and west), has been isolated. The diesel generator building has been redesignated for use with TMI-1. As part of the modifications to support the licensee's use of the diesel generator building for TMI-1, the fire system line will be cut and blanked off at the fuel-handling building, where the fire system line runs from the diesel generator building. This modification eliminates the need for freeze protection of the fire system in the AFHB from this end of the fire system loop. To prevent the pipe from repressurizing as a result of seat leakage through the isolation valve supplies on the east and west turbine building headers, 1-inch (2.5-centimeter) drain valves were installed and are piped to turbine building sumps. These drain valves, which are normally open, will be inspected monthly during freezing weather [October through April (GPU 1991a, SAR 7.2.2.2k)].

Sufficient redundancy exists with other multiple fire service loops and sectional controls so that only multiple serious impairments would require placing this fire loop in service to provide a normal flow path. The loop can



<u>FIGURE 6.3</u>. Portable Extinguisher and Self-Contained Breathing Apparatus Locations in the Control Service and Auxiliary Buildings

be placed in service by opening valve FS-V652 and closing the drain valves. There is sufficient flow in the 12-inch (30.5-centimeter) fire service loop so that the drain valves do not have to be closed for the fire suppression system to be functional.

River water and fire pump houses are deactivated for PDMS but will provide a passive pathway for intake water to diesel fire pump FS-P-1, which will be used only as an emergency backup water-supply source. This reserve pump will be placed in service, if necessary, to maintain system operability requirements (there are three other fire pumps; only two are needed for system operability). During freezing weather, the pump will be in layup or heat will be available by heat trace or by the building's heating, ventilation, and air conditioning (HVAC) system.

The TMI-2 Fire Protection Program establishes administrative controls to minimize the presence of flammable or combustible liquids and materials in the TMI-2 facility. In addition, the licensee has taken the following specific actions to reduce the fire potential during PDMS:

1. Transient combustibles inside the containment and the AFHB will be removed to the extent practicable (GPU 1991a, SAR 7.2.2.2g). This includes most plant items installed after the accident. Fire loading of transient combustibles will be maintained at less than a

1-hour loading of 80,000 BTU/square foot (2.2 million BTU/square meter).

2. The oil has been drained from the main turbine, main feedwater pumps, emergency feedwater pumps, steam generator feedwater pumps, condensate booster pump reservoirs, and hydrogen seal oil unit (GPU 1991a, SAR 7.2.2.2h).

   Each of the four reactor coolant pumps has two sumps containing 120 gallons (454 liters) each of oil (flash point 450°F) in the upper reservoir and 18 gallons (68 liters) each in the lower reservoir, for a total of 138 gallons (522 liters) per pump. Approximately 50 percent of this oil has been removed. Thus, approximately 276 gallons (1048 liters) of oil will remain. The oil in the reactor coolant pumps will be at ambient temperature and pressure. Any leak would be a slow leak. The pumps are equipped with drip pans with flash screens to prevent the collected oil from becoming involved in a fire. The pumps are separated so that only two are adjacent thus decreasing the probability that the whole inventory of oil would be involved in one fire. Because of ALARA considerations, the licensee currently has no plans to remove the remainder of the oil (GPU 1990c, SAR 6.3.51).

3. The charcoal filters have been removed from all HVAC systems in TMI-2 (GPU 1991a, SAR 7.2.2.2i).

4. Most electrical systems have been deenergized, thus removing the major ignition source (GPU 1990c, SAR Supplement 1, Number 17).

6.4.2 Fire Detection and Protection System - Current Licensing Basis

The current Technical Specifications provide requirements for surveillance activities and for demonstrating operability of the fire suppression water systems and for the fire hose stations. The licensee prepared a Fire Protection Program Evaluation for TMI-2, which was reviewed and approved by the NRC staff in August 1987.

6.4.3 Fire Detection and Protection System - PDMS Licensing Basis

The licensee's requirements for maintenance and surveillance of the fire detection and protection system are specified in the PDMS SAR 7.2.2 (GPU 1991a) and will be specified in the PDMS in the Fire Protection Program Evaluation (FPPE). The FPPE specifies fire detection, followed by manual suppression of the fire. With most of the electrical circuits deenergized and less frequent occupancy during PDMS, there will be a significant decrease in possible sources of fire. In addition, there will be fewer combustibles in the facility.

The licensee has stated in the PDMS SAR (GPU 1991d, SAR 7.2.2.2b) that the operable portion of the fire detection and alarm systems will be tested every 6 months by channel functional tests and tests of supervised circuits. Nonsupervised circuits between the local panels and the remote monitoring station in the TMI-1 control room will be tested every 31 days to demonstrate operability (GPU 1991d, SAR 7.2.2.2b).

The licensee has stated in the FPPE that responsibility for the TMI-2 fire protection system and response to fires has been assumed by TMI-1 staff and will be maintained and controlled under the FPPE requirements for both TMI-1 and TMI-2. The station fire brigade, maintained at TMI-1, will be fully trained and familiar with system configurations, plant layout, and procedures for TMI-2. The station fire brigade will be under the supervisory control of TMI-1 staff. Upon detection of a fire, the station fire brigade will respond to the specific location in TMI-2.

6.4.4 Fire Detection and Protection System - Conclusion

On the basis of the low probability of fire during PDMS and on the very low consequence if a fire were to occur (see PDMS TER Section 5.4.3), the NRC staff concludes that the fire detection and prevention system is acceptable and provides reasonable assurance of early detection and suppression of a fire. However, there is still some risk that a fire could occur. Therefore, the fire detection capability described in the PDMS SAR and the FPPE will be maintained during PDMS, and the licensee has committed to manual suppression of fire by the fire brigade as indicated in the FPPE. The TMI-2 Fire Protection Program will be updated before entry into PDMS.

6.5 Flood Protection

The TMI site was designed to be protected against the probable maximum flood established by the U.S. Army Corps of Engineers. The island on which both the TMI-1 and TMI-2 reactors are located is within the 500-year flood plain (0.2 percent chance of flooding in any given year) but not within the 100-year flood plain as determined by the U.S. Army Corps of Engineers (NRC 1987).

6.5.1 Flood Protection - System Description

Although the station grade is above the water surface profile for the probable maximum flood, dikes are provided around the site to protect the station from wave action associated with the design basis flood. In addition, structures are completely protected at the exterior faces rather than the individual equipment or systems within. The water stops between adjacent building walls and mats were designed to withstand a maximum water head in excess of that associated with the probable flood level. The exterior sliding doors and flood panels are provided with water-tight seals (GPU 1990c, SAR 7.1.4.2).

The following items are specific design features for flood control (GPU 1990c, SAR 7.1.4.2):

1. There are no external openings in the reactor containment building below the probable maximum flood level.

2. There are no external openings in the TMI-2 fuel-handling building that require flood protection. The railroad door in the Unit 1 portion of the fuel-handling building is designed to be water-tight.

3. Flood panels are provided for all entrances to the control building.

4. A flood panel is provided for the entrance to the auxiliary building.

5. The openings in the air intake tunnel are located higher than the probable maximum flood level.

6. Doors and entrances (not flood-protected) to the control building area are either water-tight or are provided with flood panels. All openings that are potential leak paths (e.g., ducts, pipes, conduits, and cable trays) are sealed.

In addition to the specific flood protection design provisions, such as the building flood panels, site diking, and portable, gas-driven flood pumps, the site is included in an early warning system provided by the Federal-State River Forecast Center, National Weather Service, Harrisburg, Pennsylvania (GPU 1990c, SAR 7.1.4.3). Upon notification by the center of potential high river flows, TMI-1 personnel implement site emergency flood protection procedures that include the entire TMI site.

6.5.2 Flood Protection - Current Licensing Basis

The current Technical Specifications define measurement requirements at various river water levels and specify the level at which flood protection measures are to be implemented to ensure protection of the facility and prevent an inadvertent release of radioactive material from the facility. The current Technical Specifications also require that the dike be inspected for potential degradation.

6.5.3 Flood Protection - PDMS Licensing Basis Specifications

The license has committed to provide flood protection for the TMI-2 facility in Section 7.1.4 of the PDMS SAR. In addition, the licensee will prepare a revised site flood protection plan that will be completed in late 1992. Many of the flood protection requirements are applicable to the entire TMI site and are included in the TMI-1 Technical Specifications (e.g., TMI-1 Technical Specification 3.1.4.1 requires inspection of the dikes around the TMI site every six months). In addition, requirements have been established in the administrative control section 6.7.1 of the proposed PDMS Technical Specifications to provide procedures and programs for maintenance of the PDMS condition including implementation of the flood protection program.

6.5.4 Flood Protection - Conclusion

The NRC staff has concluded that because the TMI-2 facility has been defueled and the facility has been significantly decontaminated, the quantity of radioactive material that could be released as the result of a flood is significantly reduced from that of an operating reactor. However, because there is still a potential for a release, the requirements for flood protection have been kept in place. The staff finds that the flood protection requirements are adequate to minimize the probability of an inadvertent release of radioactive material from the facility.

6.6 Support and Monitoring Systems

The staff has identified five support and monitoring systems that are related to personnel and environmental protection and surveillance during PDMS: (1) electrical systems, (2) effluent monitoring systems, (3) environmental monitoring systems, (4) administrative systems, and (5) surveillance program.

6.6.1 Electrical Systems

6.6.1.1 Electrical Systems - System Description

The following electrical systems will be partly or fully operational during PDMS:

1. Most of the existing lighting systems will be operational.

2. In the reactor containment, reactor building circuits will be deenergized except for those necessary for PDMS monitoring, inspection, and surveillance equipment. During entry to the reactor building, circuits for lighting and power for required equipment will be energized.

3. In the auxiliary building, the 480/277-voltage alternating current (Vac) power to lighting, fire detectors, and sump level indication circuits will be energized. Selected loads to welding receptacles, heaters, pump motors, and fan motors will be energized.

4. In the fuel-handling building, low voltage circuits (120/208 Vac) to lighting and fire detection will be energized.

5. Portions of the TMI-2 auxiliary electrical distribution system will be operational and energized to provide power for the PDMS support systems and their associated controls and instrumentation. Power will be available for area lighting, receptacles, heating, and ventilation to support PDMS activities (GPU 1991d, SAR 7.2.5.1.2).

Installed emergency lighting will not be maintained during PDMS. Normal lighting will be available throughout the TMI-2 facility (GPU 1991c, SAR 7.2.5.2.1). Personnel entering the buildings will carry flashlights for use during loss of normal lighting. Emergency response personnel will carry 8-hour portable emergency lights when entering the buildings. These emergency lights will be staged with emergency response crew equipment.

During PDMS, certain electrical systems are important for the appropriate functioning of the ventilation and fire detection systems, as well as for lighting for routine surveillance and emergency firefighting response. The ventilation systems require electricity to operate and will be used as specified in Section 6.3 of this report. Electrical systems are also required to operate the monitors for measuring the radioactive material concentrations in both liquid and gaseous effluents. Also, the normal operation of the fire detection systems requires the support of the electrical power system. Although the loss of power to the fire protection system would not increase the potential for a fire, it could increase the potential consequences of a fire, if one were to occur, by delaying its detection.

During PDMS, most of the electrical feeds, particularly to the reactor building, will be deenergized to minimize the potential for providing an ignition source for fires. Because the remaining energized systems could cause a fire, the staff has analyzed the maximum credible fire and has found that the offsite dose consequences are acceptable and are within 10 CFR Part 100 limits (see PDMS TER section 5.4.3). Systems used for PDMS surveillance activities may require energizing from local control stations before they are used. The TMI-2 auxiliary electrical distribution system consists of two full-sized auxiliary transformers (2A and 2B) connected to two separate 230-kV buses (GPU 1991d, SAR 7.2.5.1.2). The station direct current (dc) batteries are deactivated. However, dc power during PDMS will be supplied through a group of four static rectifiers to the 2-1dc and 2-2dc buses. Direct current backup power supplies are provided to support radiation monitoring and fire protection systems during a temporary loss of power. Loads have been consolidated where practicable, using bus tie-breakers to reduce the number of energized circuits.

6.6.1.2 Electrical Systems - Current Licensing Basis

Electrical power is required to provide fire detection capability, monitoring of radioactivity, operation of ventilation systems, lighting for entry and layup activities, and for mitigation of accidents. The current Technical Specifications require that the determination of operability of specified electrical buses be determined at least once every 7 days by verifying correct breaker alignment and power availability.

6.6.1.3 Electrical Systems - PDMS Licensing Basis

During PDMS, electrical power will not be required to mitigate the consequences of an accident. However, electrical power will be necessary for fire detection capability, monitoring of radioactivity, and lighting for maintenance and surveillance activities. The licensee has committed in the PDMS SAR (GPU 1991d, SAR 7.2.5) to maintain portions of the TMI-2 auxiliary electrical distribution system operational and energized to provide reliable power for the PDMS support systems and their associated controls and instrumentation. These systems are not considered safety related systems necessary to mitigate the consequences of an accident and limit offsite dose to within 10 CFR Part 100 limits considering the post-accident, inoperable, essentially defueled condition of the facility. Because of the deactivation of the reactor and its associated support systems, Class 1E emergency diesel-backed power systems are no longer required.

6.6.1.4 Electrical Systems - Conclusion

Considering the post-accident, inoperable, and essentially defueled condition of the facility, electrical power is not required to maintain the safety of the facility. The need for electrical power to maintain non-safety related systems during PDMS will be essentially the same as it is currently, and the licensee has committed to maintain that electrical power capability.

6.6.2 Effluent Monitoring Systems

6.6.2.1 Effluent Monitoring Systems - System Description

The reactor building purge system and the AFHB ventilation systems exhaust through HEPA filters into the station vent. The station vent is continuously monitored during operation of the ventilation system using an effluent monitor in the vent stack (HP-R-219 or HP-R-219A). The operation of the ventilation systems is described in Section 6.3 of this report.

6.6.2.2 Effluent Monitoring System - Current Licensing Basis

Measurement of radioactive material concentrations in effluent from the facility is required to quantify releases to the environment and to demonstrate that releases from the facility are within the current Technical Specifications and Federal regulations. The current Technical Specifications specify requirements for effluent monitoring, both gaseous and liquid, including the type of sampling, frequency, and analyses as specified in the Recovery Operations Plan.

6.6.2.3 Effluent Monitoring System - PDMS Licensing Basis

During PDMS, radioactive material released in liquid and gaseous effluents must be measured to ensure that the limits specified in 10 CFR Part 20, and the design objectives of 10 CFR Part 50, Appendix I, are not exceeded and that the licensee complies with the requirements of the Radiological Environmental Monitoring Plan (REMP) and the Offsite Dose Calculation Manual (ODCM) (see License Condition 2.F). Monitoring equipment, calculational methodology, and sampling and monitoring frequency are specified in the REMP and the ODCM.

6.6.2.4 Effluent Monitoring System - Conclusion

The NRC staff concludes that the proposed effluent monitoring and analysis specifications for PDMS as provided in the REMP and the ODCM will ensure that radioactive releases from TMI-2 will be adequately measured and quantified.

6.6.3 Environmental Monitoring Systems

6.6.3.1 Environmental Monitoring Systems - System Description

Appendix I, 10 CFR Part 50, specifies that releases of radioactive material to unrestricted areas must be kept as low as is reasonably achievable (ALARA) and provides numerical guides for complying with the ALARA requirements in 10 CFR Part 50.34a and 10 CFR Part 50.36a. The guides are defined in terms of an estimated annual dose or dose commitment for any individual in an unrestricted area from all pathways of exposure. The licensee must verify that the impact on the environment from the radioactive materials released is within the guidelines established. This verification is accomplished by measuring quantities of radioactive materials released to the environment and the concentration of radioactive materials in the actual environment and calculating the potential doses to members of the public from the materials released.

6.6.3.2 Environmental Monitoring Systems - Current Licensing Basis

Appendix B of the current Technical Specifications contains the specifications for the licensee's radiological monitoring program for the TMI site. The program consists of collecting samples from the environment, analyzing the samples for radioactivity, and interpreting the results. Samples of air, soil, water, fin fish, milk, fruits, vegetables, groundwater, and precipitation are collected and analyzed to assess the critical pathways to man and to estimate potential doses. Thermoluminescent dosimeters and a real-time gamma monitoring system are placed in the environment to measure ambient gamma radiation levels.

Sampling locations have been established that take into consideration meteorology, population distribution, hydrology, and land-use characteristics of the area. Both indicator and control sample locations have been established to ensure the validity of the data collected. The TMI staff routinely reviews and evaluates the results of sample analyses and conducts investigations if levels requiring administrative action or anomalous values are discovered.

Radiological environment operating reports are submitted annually to the NRC for review (proposed PDMS Technical Specification 6.8.1.1).

6.6.3.3 Environmental Monitoring Systems - PDMS Licensing Basis

The Radiological Environmental Monitoring Program (REMP) for the TMI site will remain fully operational and will undergo continuous review and revision as necessary to ensure adequate evaluation of the environmental impact. Because rodent activity could result in the movement of radioactive material out of the facility buildings, the licensee has committed (GPU 1991a, SAR S3-7) to analyze a limited number of carcasses for gamma-emitting isotopes as part of the non-routine radiological environmental monitoring program. The carcasses will be obtained from catch-all traps near the TMI cafeteria. The licensee also maintains a contract with a local pest control service, on an as-needed basis, that controls insects, rodents, spiders, and birds and removes live animals.

6.6.3.4 Environmental Monitoring Systems - Conclusion

The staff finds that the licensee's program of environmental surveillance is adequate to verify TMI's compliance with environmental release requirements.

6.6.4 Administrative Systems

6.6.4.1 Administrative Systems - System Description

Administrative systems are required to ensure implementation of the requirements for organizational structure, staff qualifications, records, independent safety reviews, procedures, occupational radiation protection, a quality assurance plan, an emergency plan, and other administrative control activities.

6.6.4.2 Administrative Systems - Current Licensing Basis

The current Technical Specifications provide requirements for organizational structure and responsibilities, staff qualifications for radiation protection, management and review personnel, records, independent safety reviews, procedures, a radiation protection plan, a quality assurance plan, an emergency plan, and other administrative control activities.

6.6.4.3 Administrative Systems - PDMS Licensing Basis

During PDMS, administrative controls will be required to ensure that organizational structure and responsibilities, staff qualifications for radiation protection, management and review personnel, records, independent safety reviews, procedures, a radiation protection plan, a quality assurance plan, an emergency plan, and other administrative control activities are maintained as appropriate for the defueled and nonoperating monitored storage status of the facility. Occupational radiation protection during PDMS is defined in the radiation protection plan described in the PDMS SAR and required by the proposed PDMS Technical Specifications. The limited scope quality assurance program for PDMS is documented in the TMI-2 PDMS Quality Assurance Plan (GPU 1988c). Specific security provisions for TMI-2 are documented in the TMI Modified Amended Physical Security Plan. Because of the post-accident, inoperable and essentially defueled condition of TMI-2 during PDMS, there is no potential for any significant offsite radioactive releases. Because of the existence of TMI-1 on the same site, emergency planning requirements for the site are dominated by TMI-1. Emergency planning necessary for TMI-2 has been incorporated in the integrated corporate emergency plan, which has been reviewed and approved by the NRC.

6.6.4.4 Administrative Systems - Conclusion

The NRC staff finds that the administrative controls specified in the licensing basis documents are adequate to ensure acceptable administrative control during PDMS.

6.6.5 Surveillance Program

6.6.5.1 Surveillance Program - System Description

The licensee will conduct surveillance programs during PDMS to ensure the maintenance of environmental protection systems including surveillance of reactor containment building isolation (proposed PDMS Technical Specifications 4.1.1.1, 4.1.1.2, and 4.1.1.3), surveillance of reactor containment building and AFHB ventilation and filtration systems (GPU 1992, SAR 7.2.1.3, 7.2.4.3, 7.2.6.1, and License Condition 2D), surveillance of the fire protection system and support and monitoring systems (including electrical, effluent monitoring, and environmental monitoring systems), ODCM (proposed PDMS Technical Specification 6.7.4), and oversight of administrative systems (proposed PDMS Technical Specification Section 6). Administrative systems include organizational structure, staff qualifications, records, independent safety reviews, procedures, occupational radiation protection, a quality assurance plan, an emergency plan, and other administrative control activities.

The licensee has stated (GPU 1991a, SAR 7.2.4) that routine radiological surveillance of the AFHB and the reactor containment buildings will be conducted to verify the stability of the conditions. Radiological surveillance activities include air sampling to determine levels of airborne contamination, wipe surveys to determine levels of loose surface contamination, and radiation dose rate surveys to determine potential changes in radiological status.

The licensee has also stated (GPU 1990c, SAR 5.3) that radiological surveys performed in support of work activities during the cleanup were used to establish the pre-PDMS radiological status. Section 5.3 states that the summary of the radiological conditions as contained in the PDMS SAR will be updated when final radiological conditions have been determined and final decontamination results become available. The licensee has also stated (GPU 1991a, SAR 7.2.4) that radiological surveys will be conducted periodically in the AFHB and the reactor containment building to monitor radiological conditions. Preselected locations for containment surveys are shown in Figures 6.4 and 6.5. Fixed dosimeters may also be placed in various locations and replaced periodically to measure dose rates over a longer period. The licensee will review the results of the radiological surveys and evaluate them for trends in changes in contamination levels, movement of contamination, and changes in dose rates. The radiological surveys will also detect changes in the radiological status of the facilities that may require corrective action.

6.6.5.2 Surveillance Program - Current Licensing Basis

The current facility surveillance requirements included in the REMP, current Technical Specifications, and Recovery Operations Plans provide for the environmental protection systems necessary to preclude criticality; ensure reactor containment building isolation; ensure ventilation, filtration, and measurement of gaseous effluent being released to the environment; ensure collection and monitoring of liquid effluent being released from the facility; ensure prevention or detection and mitigation of fires; ensure the oversight of necessary administrative systems; and ensure monitoring of the facility to determine radiological conditions.

6.6.5.3 Surveillance Program - PDMS Licensing Basis

During PDMS, TMI-2 will conduct surveillance programs to ensure that environmental protection is maintained. These surveillance programs will ensure isolation of the reactor containment building (proposed PDMS Technical Specifications 4.1.1.1, 4.1.1.2, and 4.1.1.3), operability of reactor containment building and AFHB ventilation and filtration systems (GPU 1992, SAR 7.2.1 and 7.2.6), operability of the fire protection system (GPU 1991a, SAR 7.2.2, License Condition 2F, and the PDMS FPPE), and functioning of support and monitoring systems (proposed PDMS Technical Specification 5.2.1.2, Section 6, License Condition 2F, and GPU 1992, SAR 7.2.4.2) (including electrical, effluent monitoring, and environmental monitoring systems), oversight of administrative systems, and periodic measurement of radiation

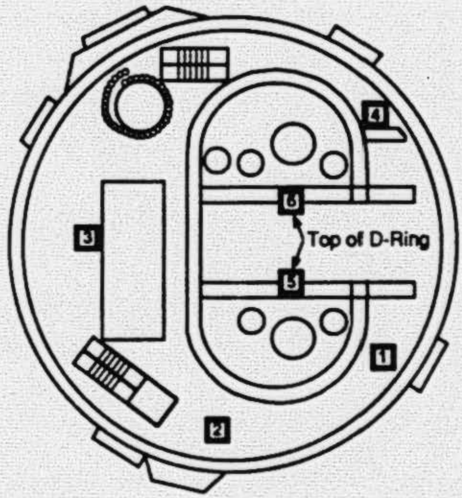


Figure 6.4 Radiation Survey Locations in the Reactor Building Elevation 347 Feet

and contamination levels to verify radiological conditions. Administrative systems include organizational structure, staff qualifications, records, independent safety reviews, procedures, occupational radiation protection, a quality assurance plan, an emergency plan, and other administrative control activities.

The PDMS surveillance programs described in the preceding sections are listed below:

- Maintenance of reactor vessel geometry, Section 6.1.3
- Reactor containment isolation, Section 6.2.3
- Reactor building breather and ventilation system, Section 6.3.1.3
- Auxiliary and fuel handling building ventilation system, Section 6.3.2.3
- Fire protection system, Section 6.4.3
- Flood protection, Section 6.5.3
- Support and monitoring systems, Section 6.6.1.3, 6.6.2.3, 6.6.3.3, and 6.6.4.3

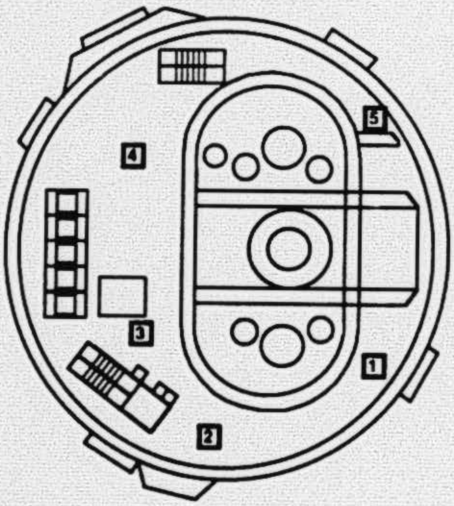


Figure 6.5 Radiation Survey Locations in the Reactor Building Elevation 305 Feet

6.6.5.4 Surveillance Program - Conclusion

The NRC staff finds that these surveillance programs will ensure maintenance of the environmental protection systems during PDMS. In addition, the NRC staff finds that the licensee's radiological surveillance activities during PDMS will be adequate to verify continued stability of radioactive material within the facility and to identify conditions that may require corrective action.

7.0 CONCLUSION

On the basis of the material received from the licensee and independent evaluation and measurements by the NRC staff, the NRC staff concludes that the entry of TMI-2 into PDMS will not decrease the margin of safety for workers and the public.

The staff makes the following findings:

1. Defueling of the reactor has been accomplished to the extent reasonably achievable.

2. All fuel (i.e., UO_2) and core debris removed from the reactor and associated systems have been shipped off site.

3. The results of analyses indicate that there is no potential for a criticality in the fuel remaining in the TMI-2 facility during either normal or accident conditions. The conservatism built into the model and the additional measures being taken by the licensee including removal of water, addition of a neutron poison into the vessel, and restrictions on deliberate fuel movement, would further preclude the possibility of a criticality.

4. Remaining radioactive waste from the major TMI-2 decontamination activities has been shipped off site or packaged and staged for shipment off site.

5. Radiation levels within the facility have been reduced to such levels that necessary and required plant monitoring, maintenance, and inspections can be performed.

6. Radiological control of activities during PDMS will be conducted in accordance with the approved Radiation Protection Plan and in compliance with the regulatory requirements of 10 CFR Part 20. This process will ensure adequate control of occupational exposure and protection of workers.

7. The licensee's proposed surveillance program is adequate to monitor the PDMS environmental protection systems.

8. The environmental monitoring for TMI-2 during PDMS will be included in the TMI Site Radiological Environmental Monitoring Plan and will ensure adequate environmental surveillance and control.

9. Fire protection at the TMI-2 facility during PDMS will be accomplished according to the approved TMI-2 Fire Protection Program Evaluation (FPPE) and will ensure that the risk of fire is within the bounds analyzed by this evaluation.

10. The requirements delineated in the proposed PDMS Technical Specifications provide assurance that the facility will be maintained in an environmentally safe condition.

11. The TMI-2 facility can safely be placed in long-term monitored storage, and the facility configuration during storage under both routine and accident conditions will not result in impacts that exceed those identified in the staff's PEIS Supplement 3 (NRC 1989a).

8.0 REFERENCES

American Nuclear Society (ANS). 1988. Technical Conference on Nuclear Fission. Fifty Years of Progress in Energy. Security and the Topical Meeting TMI-2 Accident: Materials Behavior and Plan Recovery Technology, October 30 to November 4, 1988. American Nuclear Society, Washington, D.C.

Burchsted, C. A., A. B. Fuller, and J. E. Kahn. 1976. Nuclear Air Cleaning Handbook. ERDA 76-21, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

Chan, M. K., M. Y. Ballinger, and P. C. Owczarski. 1989. "User's Manual for FIRIN." NUREG/CR-3037, U.S. Nuclear Regulatory Commission, Washington, D.C.

Clayton, R. F. 1970. Monitoring of Radioactive Contamination on Surfaces. Technical Report Series No. 120, International Atomic Energy Agency, Vienna, Austria.

Dunster, J. J. 1962. "Surface Contamination Measurements as an Index of Control of Radioactive Materials." Health Physics 8:353-356.

General Public Utilities Nuclear Corporation (GPU). 1986. "Three Mile Island Cleanup Program - Post-Defueling Monitored Storage (PDMS)." Letter from F. R. Standerfer to W. D. Travers, NRC, dated December 2, 1986. GPU Document 4410-86-L-0201.

General Public Utilities Nuclear Corporation (GPU). 1987a. "Environmental Evaluation for TMI-2 Post-Defueling Monitored Storage." Letter from F. R. Standerfer, GPU Nuclear Corporation, to NRC, dated March 11, 1987. GPU Document 4410-87-L-0025.

General Public Utilities Nuclear Corporation (GPU). 1987b. Technical Plan, TMI-2, Cleanup Program Post-Defueling Monitored Storage. TPO/TMI-188, Rev 0, GPU Nuclear Corporation, Middletown, Pennsylvania.

General Public Utilities Nuclear Corporation (GPU). 1988a. "Draft Programmatic Environmental Impact Statement, Supplement No. 3." Letter from F. R. Standerfer, GPU Nuclear Corporation, to M. T. Masnik, U.S. NRC, dated July 12, 1988.

General Public Utilities Nuclear Corporation (GPU). 1988b. "Post-Defueling Monitored Storage Proposed License Amendment and Safety Analysis Report." Letter from F. R. Standerfer, GPU Nuclear Corporation, to NRC, dated August 16, 1988. GPU Document 4410-88-L-0068.

General Public Utilities Nuclear Corporation (GPU). 1988c. "Post-Defueling Monitored Storage Quality Assurance Plan." Letter from F. R. Standerfer, GPU Nuclear Corporation, to NRC, dated August 25, 1988. GPU Document 4410-88-L-0133.

General Public Utilities Nuclear Corporation (GPU). 1988d. Evaluation of Block Wall Leaching, 13 June-17 August 1988. Technical Bulletin TB-88-11, Rev. 0, GPU Nuclear Corporation, Middletown, Pennsylvania.

General Public Utilities Nuclear Corporation (GPU). 1989a. "Technical Specifications for Post-Defueling Monitored Storage." Letter from M. B. Roche to NRC, dated January 9, 1989. GPU Document 4410-88-L-0188.

General Public Utilities Nuclear Corporation (GPU). 1989b. "Additional Information on the Post-Defueling Monitored Storage Safety Analysis Report." Letter from M. B. Roche, GPU Nuclear Corporation to J. F. Stolz, NRC, dated February 9, 1989. GPU Document 4410-89-L-0009.

General Public Utilities Nuclear Corporation (GPU). 1989c. "Additional Information on the Post-Defueling Monitored Storage Safety Analysis Report." Letter from M. B. Roche, GPU Nuclear Corporation, to NRC, dated March 31, 1989. GPU Document 4410-89-L-0019/0449P.

General Public Utilities Nuclear Corporation (GPU). 1989d. Letter from P. R. Clark, GPU Nuclear Corporation, to V. Stello, U.S. NRC. dated June 23, 1989.

General Public Utilities Nuclear Corporation (GPU). 1989e. "Post-Defueling Monitored Storage Safety Analysis Report, Amendment 4." Letter from M. B. Roche, GPU Nuclear Corporation, to NRC dated June 26, 1989. GPU Document 4410-89-L-0055/0449P.

General Public Utilities Nuclear Corporation (GPU). 1989f. "Post-Defueling Monitored Storage Safety Analysis Report, Amendment No. 5." Letter from M. B. Roche, GPU Nuclear Corporation, to NRC, dated October 10, 1989. GPU Document 4410-89-L-0095/0449P.

General Public Utilities Nuclear Corporation (GPU). 1989g. "Post-Defueling Monitored Storage Safety Analysis Report, Amendment 6." Letter from M. B. Roche, GPU Nuclear Corporation, to NRC, dated November 22, 1989. GPU Document 4410-89-L-0116/0449P.

General Public Utilities Nuclear Corporation (GPU). 1989h. "Defueling Completion Report, Initial Submittal." Letter from M. B. Roche, GPU Nuclear Corporation, to NRC, dated July 5, 1989. GPU Document 4410-89-L-0070.

General Public Utilities Nuclear Corporation (GPU). 1989i. "Defueling Completion Report, Second Submittal." Letter from M. B. Roche, GPU Nuclear Corporation, to NRC, dated August 18, 1989. GPU Document 4410-89-L-0078/0477P.

General Public Utilities Nuclear Corporation (GPU). 1989j. "Defueling Completion Report, Third Submittal." Letter from M. B. Roche, GPU Nuclear Corporation, to NRC, dated October 9, 1989. GPU Document 4410-89-L-0102/0477P.

General Public Utilities Nuclear Corporation (GPU). 1990a. "Post-Defueling Monitored Storage Safety Analysis Report, Amendment 7." Letter from M. B. Roche, GPU Nuclear Corporation, to NRC, dated June 21, 1990. GPU Document 4410-90-L-0039/0449P.

General Public Utilities Nuclear Corporation (GPU). 1990b. "Post-Defueling Monitored Storage Safety Analysis Report, Amendment 8." Letter from R. L. Long, GPU Nuclear Corporation, to NRC, dated October 15, 1990. GPU Document 4410-90-L-0075.

General Public Utilities Nuclear Corporation (GPU). 1990c. "Post-Defueling Monitored Storage Safety Analysis Report. Amendment 9." Letter from R. L. Long, GPU Nuclear Corporation, to NRC, dated November 7, 1990. GPU Document 4410-90-L-0078.

General Public Utilities Nuclear Corporation (GPU). 1990d. "Defueling Completion Report, Fourth Submittal." Letter from M. B. Roche, GPU Nuclear Corporation, to NRC, dated January 18, 1990. GPU Document 4410-89-L-0128.

General Public Utilities Nuclear Corporation (GPU). 1990e. "Defueling Completion Report, Final Submittal." Letter from M. B. Roche, GPU Nuclear Corporation, to NRC, dated February 22, 1990. GPU Document 4410-90-L-0012/0477P.

General Public Utilities Nuclear Corporation (GPU). 1990f. "Results of Post-Lower Head Sampling Program Cleanup." Letter from M. B. Roche, GPU Nuclear Corporation, to NRC, dated April 12, 1990. GPU Document 4410-90-L-0026/0531P.

General Public Utilities Nuclear Corporation (GPU). 1990g. "Decommissioning Financial Assurance Report for Oyster Creek, TMI-1, TMI-2, and Saxton Nuclear Experimental Station." From M. P. Morrell, GPU Service Corp. to NRC, dated July 26, 1990.

General Public Utilities Nuclear Corporation (GPU). 1991a. "Post-Defueling Monitored Storage Safety Analysis Report. Amendment 10." Letter from R. L. Long, GPU Nuclear Corporation, to NRC, dated February 19 1991. GPU Document C312-91-2005.

General Public Utilities Nuclear Corporation (GPU). 1991b. "Post-Defueling Monitored Storage Safety Analysis Report. Amendment 11." Letter from R. L. Long, GPU Nuclear Corporation, to NRC, dated April 19, 1991. GPU Document C312-91-2023.

General Public Utilities Nuclear Corporation (GPU). 1991c. "Post-Defueling Monitored Storage Safety Analysis Report. Amendment 12." Letter from R. L. Long, GPU Nuclear Corporation, to NRC, dated June 21, 1991. GPU Document C312-91-2047.

General Public Utilities Nuclear Corporation (GPU). 1991d. "Post-Defueling Monitored Storage Safety Analysis Report. Amendment 13." Letter from R. L. Long, GPU Nuclear Corporation, to NRC, dated August 28, 1991. GPU Document C312-91-2068.

General Public Utilities Nuclear Corporation (GPU). 1991e. "Post-Defueling Monitored Storage Safety Analysis Report. Amendment 14." Letter from R. L. Long, GPU Nuclear Corporation, to NRC, dated October 9, 1991. GPU Document C312-91-2080.

General Public Utilities Nuclear Corporation (GPU). 1992. "Post-Defueling Monitored Storage Safety Analysis Report. Amendment 15." Letter from R. L. Long, GPU Nuclear Corporation, to NRC, dated January 13, 1992. GPU Document C312-92-2004.

Mishima, J., and L. C. Schwendiman. 1973. "Fractional Airborne Release of Uranium (Representing Plutonium) During the Burning of Contaminated Wastes)." BNWL-1730, Pacific Northwest Laboratory, Richland, Washington.

Murphy, E. S., and G. M. Holter. 1982. "Technology, Safety and Costs of Decommissioning Reference Light Water Reactors Following Postulated Accidents." NUREG/CR-2601, U.S. Nuclear Regulatory Commission, Washington, D.C.

U.S. Code of Federal Regulations (40 CFR Part 190). 1988. "Environmental Radiation Protection Standards for Nuclear Power Operations." Title 40, Part 190.

U.S. Code of Federal Regulations (10 CFR Part 20). 1989. "Standards for Protection Against Radiation." Title 10, Part 20.

U.S. Code of Federal Regulations (10 CFR Part 50). 1989. "Domestic Licensing of Protection and Utilization Facilities." Title 10, Part 50.

U.S. Code of Federal Regulations (10 CFR Part 100). 1989. "Reactor Site Criteria." Title 10, Part 100.

U.S. Code of Federal Regulations (40 CFR Part 141). 1990. "National Primary Drinking Water Regulations." Title 40, Part 141.

U.S. Nuclear Regulatory Commission (NRC). 1978. "Design, Testing, and Maintenance Criteria for Post-Accident Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration on Adsorption Units of Light-Water-Cooled Nuclear Power Plants." Regulatory Guide 1.52, Rev. 2, Office of Standards Development, U.S. Nuclear Regulatory Commission, Washington, D.C.

U.S. Nuclear Regulatory Commission (NRC). 1979a. Order for Modification of License. July 20, 1979.

U.S. Nuclear Regulatory Commission (NRC). 1979b. "Environmental Assessment: Use of EPICOR-II at Three Mile Island, Unit 2." NUREG-0591, U.S. Nuclear Regulatory Commission, Washington, D.C.

U.S. Nuclear Regulatory Commission (NRC). 1979c. "Design, Testing, and Maintenance Criteria for Normal Ventilation Exhaust System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants." Regulatory Guide 1.140, Rev. 1, Office of Standards Development, U.S. Nuclear Regulatory Commission, Washington, D.C.

U.S. Nuclear Regulatory Commission (NRC). 1979d. "Investigation into the March 28, 1979 Three Mile Island Accident by the Office of Inspection and Enforcement." Investigative Report No. 50-320/79-10. NUREG-0609, Office of Inspection and Enforcement, U.S. Nuclear Regulatory Commission, Washington, D.C.

U.S. Nuclear Regulatory Commission (NRC). 1980. Modifications to Technical Specifications, February 11, 1980.

U.S. Nuclear Regulatory Commission (NRC). 1981. "Final Programmatic Environmental Impact Statement related to decontamination and disposal of radioactive wastes resulting from March 28, 1979, accident Three Mile Island Nuclear Station, Unit 2." NUREG-0683, Vol. 1, U.S. Nuclear Regulatory Commission, Washington, D.C.

U.S. Nuclear Regulatory Commission (NRC). 1984a. "NRC Plan for Cleanup Operations at Three Mile Island, Unit 2." NUREG-0698, Rev. 2, U.S. Nuclear Regulatory Commission, Washington, D.C.

U.S. Nuclear Regulatory Commission (NRC). 1984b. Meeting of the TMI-2 Advisory Panel for the Decontamination of TMI-2, April 12, 1984, Harrisburg, Pennsylvania.

U.S. Nuclear Regulatory Commission (NRC). 1987. "Programmatic Environmental Impact Statement Related to Decontamination and Disposal of Radioactive Waste Resulting from March 28, 1979 Accident, Three Mile Island Nuclear Station, Unit 2, Final Supplement Dealing with Disposal of Accident-Generated Water." NUREG-0683, Final Supplement 2, U.S. Nuclear Regulatory Commission, Washington, D.C.

U.S. Nuclear Regulatory Commission (NRC). 1988a. Amendment 30, License No. DPR-73. GPU Nuclear Corporation, Docket No. 50-320, Three Mile Island Nuclear Station, Unit No. 2, Amendment to Facility Operating License. Date of Issuance: May 27, 1988.

U.S. Nuclear Regulatory Commission (NRC). 1988b. "Programmatic Environmental Impact Statement Related to Decontamination and Disposal of Radioactive Waste Resulting from March 28, 1979 Accident, Three Mile Island Nuclear Station, Unit 2. Draft Supplement Dealing with Post-Defueling Monitored Storage and Subsequent Cleanup." NUREG-0683, Draft Supplement 3, U.S. Nuclear Regulatory Commission, Washington, D.C., April 1988.

U.S. Nuclear Regulatory Commission (NRC). 1989a. "Programmatic Environmental Impact Statement Related to Decontamination and Disposal of Radioactive Waste Resulting from March 28, 1979 Accident, Three Mile Island Nuclear Station Unit 2, Final Supplement Dealing with Post-Defueling Monitored Storage and Subsequent Cleanup." NUREG-0683, Final Supplement 3, U.S. Nuclear Regulatory Commission, Washington, D.C., August 1989

U.S. Nuclear Regulatory Commission (NRC). 1989b. "Request for Additional Information on the Post-Defueling Monitored Storage (PDMS) Safety Analysis Report (SAR)." Letter from J. F. Stolz, NRC, to M. B. Roche, GPU Nuclear Corporation, dated January 3, 1989.

U.S. Nuclear Regulatory Commission (NRC). 1989c. "Request for Additional Information for Evaluation of PDMS Safety Analysis Report." M. T. Masnik, NRC to M. B. Roche, GPU Nuclear Corporation, dated July 14, 1989.

U.S. Nuclear Regulatory Commission (NRC). 1989d. "Request for Additional Information for Evaluation of PDMS Safety Analysis Report (TAC 69115)." Letter from M. T. Masnik, NRC, to M. B. Roche, GPU Nuclear Corporation, dated August 22, 1989.

U.S. Nuclear Regulatory Commission (NRC). 1989e. "Requests for Additional Information to Evaluation of the Defueling Completion Report and the PDMS Safety Analysis Report." Letter from M. T. Masnik, NRC, to M. B. Roche, GPU Nuclear Corporation, dated December 12, 1989.

U.S. Nuclear Regulatory Commission (NRC). 1989f. "Safety Evaluation by the Office of Nuclear Reactor Regulation Related to Amendment No. 35 to Facility Operating License No. DPR-73, GPU Nuclear Corporation, Three Mile Island Nuclear Station Unit No. 2." Docket No. 50-320, dated September 11, 1989.

U.S. Nuclear Regulatory Commission (NRC). 1990a. "Request for Additional Information for Evaluation of PDMS Safety Analysis Report." Letter from M. T. Masnik, NRC, to M. B. Roche, GPU Nuclear Corporation, dated March 2, 1990.

U.S. Nuclear Regulatory Commission (NRC). 1990b. "Request for Additional Information Regarding Evaluation of Post-Defueling Monitored Storage Proposed License Amendment and SAR." Letter from M. T. Masnik, NRC, to R. L. Long, GPU Nuclear Corporation, dated August 6, 1990.

U.S. Nuclear Regulatory Commission (NRC). 1990c. "Subject: Inspection 50-320/90-03." Letter from E. C. Wenzinger, NRC, to GPU Nuclear Corporation, dated June 14, 1990.

U.S. Nuclear Regulatory Commission (NRC). 1990d. "Fuel Canister Verification at TMI-2." Letter from L. H. Thonus, F. I. Young, and R. R. Brady to J. F. Stolz, dated April 26, 1990.

U.S. Nuclear Regulatory Commission (NRC). 1991a. "Fuel Verification Measurements in TMI Unit 2." Letter from S. H. Weiss, NRC, to R. L. Long, GPU Nuclear Corporation, dated March 22, 1991.

U.S. Nuclear Regulatory Commission (NRC). 1991b. "Standard Technical Specifications - Babcock and Wilcox Plants." NUREG-1430, Vols. 1, 2, and 3, U.S. Nuclear Regulatory Commission, Washington, D.C.