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

Dear Sirs:

Three Mile Island Nuclear Station, Unit 2 (TMI-2)
Operating License No. DPR-73
Docket No. 50-320
Defueling Completion Report, Initial Submittal

On May 27, 1988, the NRC issued License Amendment No. 30 which revised the TMI-2 Technical Specifications. The amendment established license conditions and associated applicability statements which reflect current and anticipated future plant conditions through the remainder of the TMI-2 Cleanup Program. The amendment provides for three (3) distinct facility modes and defines the prerequisites for transition from the current plant condition (Facility Mode 1) through the completion of defueling (Facility Mode 2) to the completion of fuel shipment (Facility Mode 3). At least 60 days prior to transition to each successive facility mode, a report must be submitted which provides the basis for the proposed transition.

In accordance with the provisions of License Amendment No. 30, GPU Nuclear is currently preparing a report entitled, "Defueling Completion Report (DCR)," which provides the basis for facility transition from Mode 1 to Mode 2. Attached is the first of four (4) submittals which will constitute the complete report. This initial submittal describes the accident sequence and resultant fuel redistribution within the Reactor Vessel and ex-vessel locations, provides a detailed description of ongoing and completed defueling activities, discusses the general defueling approach, and provides the basis for determining that defueling activities in the Auxiliary and Fuel Handling Buildings (AFHB) have been completed. Subsequent submittals will provide the basis for concluding that defueling has been completed in the Reactor Building (RB), the Reactor Coolant System (RCS), and the Reactor Vessel (RV) and will be forwarded as fuel removal activities are brought to completion in Unit 2.

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License Amendment No. 30 established residual fuel criteria for facility transition from Mode 1 to Mode 2 that were specific to the RB, RCS, and RV. GPU Nuclear is submitting the attached information to demonstrate that other locations of the TMI-2 facility (i.e., AFHB), which contained small quantities of residual fuel, also underwent cleanup to the degree necessary to ensure they meet the defueling completion criteria. The amount of residual fuel in the AFHB has been demonstrated to be significantly below the Safe Fuel Mass Limit of 140 kg and is estimated to be a very small fraction of the defueling endpoint goal.

The purpose of this initial submittal is to provide early input to the NRC Staff on defueled areas of the plant. Additional submittals will follow closely the completion of fuel removal activities in the remaining areas of the plant. GPU Nuclear plans the final submittal of the DCR shortly after defueling completion and understands that the prescribed 60-day review period does not begin until receipt of that submittal. However, GPU Nuclear hopes that early submittal of significant portions of the DCR will facilitate an expeditious review and a full 60-day period will not be required to complete a favorable review.

Additional selective decontamination and system draindown operations will occur as part of post-defueling activities. These activities may, in some instances, result in reducing the residual fuel quantities reported in the DCR. As you are aware, GPU Nuclear will conduct an extensive Special Nuclear Material (SNM) measurement program (post-defueling survey) as part of the overall facility SNM accountability program. The post-defueling survey will result in the "estimate of record" of residual SNM and will account for any variation in residual fuel quantities resulting from post-defueling activities. However, final SNM accountability is not a prerequisite for transition to facility Mode 2.

Sincerely,

M B Roche

M. B. Roche
Director, TMI-2

RDW/emf

Attachment

cc: F. I. Young - Senior Resident Inspector, TMI
W. T. Russell - Regional Administrator, Region I
J. F. Stolz - Director, Plant Directorate I-4
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THREE MILE ISLAND

NUCLEAR STATION

UNIT 2

DEFUELING COMPLETION REPORT

Rev. 0/0461P

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1.0 INTRODUCTION

1.1 Background

On May 27, 1988, the NRC issued Licensed Amendment No. 30 which provides three (3) facility modes for the TMI-2 facility (see Table 1-1). The plant conditions defined for each successive mode reflects continued progress in removing core material from the TMI-2 facility. At least 60 days prior to transition to each successive facility mode, a report shall be submitted to the NRC providing the necessary basis and justification for the transition. Specifically, the Technical Specifications require a detailed report prior to transition from Mode 1 to Mode 2 affirming that:

1. The Reactor Vessel and Reactor Coolant System are defueled to the extent reasonably achievable,

2. The possibility of criticality in the Reactor Building is precluded, and

3. There are no canisters containing core material in the Reactor Building.

In conjunction with issuance of License Amendment No. 30, the NRC granted GPU Nuclear an exemption from 10 CFR 70.24 for the criticality monitoring requirements in the TMI-2 RB. This action imposed the following mode transition provision:

> "Prior to transition to Mode 2, the licensee will provide a criticality analysis that will address each separate quantity of residual fuel in each defined location. The criticality analysis will estimate the quantity of fuel remaining, its location, its dispersion within the location, its physical form (i.e., film, finely fragmented, intact fuel pellets), its mobility, the presence of any mechanism that would contribute to the mobility of the material, the presence of any moderating or reflecting material, and its potential for a critical event. In this submittal the licensee must demonstrate that the cleanup has progressed far enough such that an inadvertent criticality is precluded..."

1.2 Purpose

This report entitled, "Defueling Completion Report (DCR)," provides the basis for concluding that the TMI-2 facility has been defueled to the extent reasonably achievable and demonstrates that inadvertent criticality has been precluded. Its purpose is to document compliance with the NRC reporting requirements identified above and provide the basis for the TMI-2 facility transition to Mode 2.

1.3 Report Organization

The DCR is structured to address four (4) separate areas of the TMI-2 facility [i.e., Auxiliary and Fuel Handling Building (AFHB), Reactor Building (RB), Reactor Coolant System (RCS), and Reactor Vessel (RV)].

Fuel removal and associated decontamination activities are discussed in detail for each area. Using this structured approach, the report will be submitted as four (4) separate, partial reports, the sum of which will constitute the DCR.

The report is organized to include detailed discussions of the post-accident fuel transport and dispersion conditions (Section 2.0); the survey techniques utilized for residual fuel measurements (Section 3.0); the major fuel removal accomplishments and methods (Section 4.0); and the resultant residual fuel quantification, by location, and criticality analyses for each fuel location, as appropriate (Section 5.0). The completed DCR also contains an assessment of major residual fuel deposits (Section 6.0); cumulative occupational exposures and projected occupational doses attendant to unplanned further attempts to defuel beyond the level deemed ALARA (Section 7.0); and the overall findings and conclusions (Section 8.0).

The concluding sections of the DCR (i.e., Sections 6, 7, and 8), provided as part of the final submittal, focus on residual fuel and discuss possible alternatives and impacts associated with attempts to remove some or all of the remaining fuel. It should be recognized that it is not feasible, nor required, to remove all residual fuel from the facility prior to transition to Mode 2. However, the facility must have been defueled to the extent reasonably achievable and inadvertent criticality must have been precluded.

Table 1-2 identifies acronymns used in this report. Appendix A provides a list of references as they appear in this report. Appendix B is the GPU Nuclear docketed safety evaluation for the Safe Fuel Mass Limit.

1.4 Defueling Objectives and Guidelines

In order to meet the defueling completion goals and satisfy the NRC requirements for mode transition, the following guidelines and fuel removal objectives were integrated into the defueling operations planning:

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

2. Sufficient fuel will be removed to assure 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 not be removed.

Implementation of these objectives forms the basis for concluding whether defueling has been achieved to the extent reasonably achievable.

1.5 Residual Fuel Characterization

The DCR represents a characterization of residual fuel for those TMI-2 facility locations which may have been exposed to fuel relocation as a result of the accident. As such, the DCR provides a bounding case analysis with added conservatism in fuel estimates where final system measurements may not have been practical, or possible, because of continuous use and/or the need for plant systems for further water processing and final draindown operations. These fuel estimate conservatisms were added to insure that bounding condition values (i.e., maximum expected fuel quantities) were considered for several facility locations identified within the report. GPU Nuclear plans to conduct an extensive SNM measurement program as part of the overall facility fuel accountability program. This post-defueling survey will account for any variation in residual fuel estimates and conservatism added as part of the DCR chararcterization effort.

TABLE 1-1

FACILITY MODES

MODE	PLANT CONDITION
1	The reactor shall be subcritical with an average reactor coolant temperature of less 200°F.
2	Mode 2 shall exist when the following conditions are met: a. The Reactor Vessel and Reactor Coolant System is defueled to the extent reasonably achievable. b. The possibility of criticality in the Reactor Building is precluded. c. There are no canisters containing core material in the Reactor Building.
3	Mode 3 shall exist when the conditions for Mode 2 are met and no canisters containing core material are stored on the TMI-2 site.

NOTE: Mode 2, criterion c, has been interpreted by GPU Nuclear to refer to defueling canisters that are used for defueling operations in the RB. Though not specified as part of Table 1-1, DWCS filter canisters in use for water cleanup during and after the NRC-sponsored RV Lower Head Sampling Program are expected to contain small amounts of fuel fines and may remain in the RB during Mode 2.

TABLE 1-2

ACRONYMS

AB	Auxiliary Building
ACES	Automated Cutting Equipment System
AFHB	Auxiliary and Fuel Handling Buildings
ALARA	As Low As Is Reasonably Achievable
CBM	Core Boring Machine
CSA	Core Support Assembly
CWST	Concentrated Waste Storage
DCR	Defueling Completion Report
DF	Decontamination Factor
DHR	Decay Heat Removal
DOE	Department of Energy
DWCS	Defueling Water Cleanup System
ECCS	Emergency Core Support System
FHB	Fuel Handling Building
GEM	Geiger-Mueller Counter
HEPA	High-Efficiency Particulate Absolute
HPGe	High-Purity Germanium
HPI	High Pressure Injection
IIGT	Incore Instrument Guide Tube
INEL	Idaho National Engineering Laboratory
LCSA	Lower Core Support Assembly
MDL	Minimum Detectable Level
MeV	Million Electron Volts
MU	Makeup
MU&P	Makeup and Purification
MWHT	Miscellaneous Waste Holdup Tank
NaI(Tl)	Thalluim Drifted Sodium Iodide
NRC	Nuclear Regulatory Commission
OTSG	Once-Through Steam Generator
PORV	Pilot Operated Relief Valve
RB	Reactor Building
RCBT	Reactor Coolant Bleed Tank

TABLE 1-2 (Cont'd)

ACRONYMS

RCDT	Reactor Coolant Drain Tank
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RV	Reactor Vessel
SDS	Submerged Demineralizer System
SFML	Safe Fuel Mass Limit
SIVR	Seal Injection Valve Room
SNM	Special Nuclear Material
SRST	Spent Resin Storage Tank
SSTRs	Solid-State Track Recorders
Si(Li)	Lithium Drifted Silicon
TMI-2	Three Mile Island, Unit 2
UCSA	Upper Core Support Assembly
WDL	Waste Disposal Liquid

2.0 POST-ACCIDENT FUEL DISPERSION

This section provides a summary discussion of the accident sequence as it relates to fuel material transport within the RV and from the RV to ex-vessel locations. Included are sections which describe the most likely supposition of the core accident scenario, the post-accident condition of the plant, and the fuel transport mechanisms within the RCS, RB, and AFHB. The bases for the following findings and conclusions are derived primarily from the results of visual examinations, analytical evaluations, and the experience and data derived from defueling operations.

Substantial core damage within the RV and subsequent attempts to cool the core provided the source material and initial pathway by which fuel debris was transported into the RCS, RB, and AFHB. Because the plant systems required cooldown, isolation, and water processing at various times during the plant stabilization and recovery periods, additional potential pathways existed for insoluble fuel material transport. However, the majority of these pathways within the RB and the AFHB are defined by specific boundaries, filters, and/or flow restrictions, which significantly reduced any potential fuel transport. Of the total fuel debris available to be transported from the RV, it was conservatively estimated that no more than 25 kg reached the AFHB locations, no more than 15 kg was relocated to the RB sump and various other RB locations, and no more than 230 kg was relocated throughout the RCS (see Table 2-1). The remaining core inventory was retained in the RV. The following discussion represents the basis for fuel transport dispersion at TMI-2.

2.1 The Accident Scenario

A postulated scenario of the accident was developed using currently available data from in-vessel and ex-vessel defueling operations and the accident transient sequence information (References 2.1 and 2.2). This data base included measurements from on-line instrumentation, visual observations, and supporting analytical studies as well as other experimental data from independent research facilities (Reference 2.3).

The accident can be divided into the following five (5) phases:

Phase I, Time 0-100 Minutes: Loss-of-Coolant with the RCS Pumps Operating.
Phase II, Time 100-174 Minutes: Initial Core Heatup and Degradation.
Phase III, Time 174-224 Minutes: Degraded Core Heatup and Relocation.
Phase IV, Time 224-230 Minutes: Core Relocation to LCSA.
Phase V, Time After 230 Minutes: Long-Term Cooling of Degraded Core.

2.1.1 Phase I - Loss-of-Coolant (0-100 Minutes)

The first phase of the accident is the time interval from the turbine trip until the A-loop RCPs were turned off at 100 minutes. The RCPs provided 2-phase cooling to the core during this period, preventing core overheating and damage. During the first phase of the accident, the amount of water in the RCS decreased because the RCS makeup was insufficient to compensate for coolant loss through the PORV.

2.1.2 Phase II - Initial Core Heatup and Degradation (100-174 Minutes)

When the last two RCPs were turned off, at approximately 100 minutes, the top of the core was uncovered and coolant water separated into steam and liquid phases. Temperatures in the upper regions of the core then increased more rapidly. The core liquid level dropped to approximately the mid-core elevation at approximately 140 minutes and fuel rod temperatures at the top of the core increased sufficiently (1100°K) to cause cladding rupture. During this period, the operators realized that the PORV was open. They manually closed the pressurizer block valve, thus limiting further loss of-coolant and gaseous fission product release from the RCS to the RB. However, the block valve had to be cycled (i.e., opened and closed) frequently to maintain RCS pressure during this period.

Rapid oxidation of the zircaloy cladding at the top of the core began at approximately 150 minutes. The heat generated from oxidation elevated fuel rod temperatures above the cladding melting point (2100°K) developing a molten mixture of fuel, cladding, and some structural steel. This mixture flowed downward and solidified around intact fuel rods near the coolant liquid level interface. The responses of incore instrumentation and source range monitors indicated that a large region of partially molten core materials formed by 174 minutes, as shown in Figure 2-1a. It is conjectured that the first molten material to flow was a mixture consisting primarily of UO_2, steel, zircaloy, and silver, with some indium and cadmium. As this molten flow stopped at the coolant level interface, it formed a thin layer, or crust, which later supported additional molten material in the core region.

2.1.3 Phase III - Degraded Core Heatup and Relocation (174-224 Minutes)

Operation of the RC-P-2B at 174 minutes for approximatly 6 minutes, resulted in the first major core relocation event when coolant was circulated into the RV following core degradation. Thermal-mechanical interaction of the coolant with the oxidized and embrittled fuel rod remnants in the upper core regions is believed to have fragmented and collapsed these standing remnants and formed the upper core cavity and debris bed. The configuration is shown in Figure 2-1b.

After approximately 25 minutes of further coolant heating and steam formation in the core, the ECCS was initiated at 200 minutes and subsequently filled the RV in 7 to 10 minutes. Studies of debris bed cooling indicate that final quenching of the upper core debris bed probably occurred during the last several minutes of this time period (Reference 2.4). It is postulated that effective cooling of the molten core material was limited to the surrounding crust material. Thus, the amount of molten material in the central region likely continued to increase in size and temperature because of decay heat from retained fission products and lack of coolant flow through the damaged core. Calculations simulating the accident suggest that a molten pool of approximately 50% of the original core materials was formed within the consolidated region by 224 minutes into the accident (Reference 2.5). This is consistent with the observed molten material found in the resolidified core mass, the CSA, and the lower head regions (Figure 2-2).

The interaction of the injected water with the upper debris bed during this period and the flow pattern of steam and gas exiting the core through the upper plenum have been assessed. The observed damage pattern to the upper fuel assembly grid was consistent with expected flow patterns, considering the location of the exit flow orifices. Rapid oxidation within the debris bed and the subsequent interaction between the upper grid structure and the high temperature gases exiting the core at high velocity probably caused the observed limited damage.

2.1.4 Phase IV - Core Relocation to Lower Core Support Assembly (224-230 Minutes)

The second major core relocation event occurred between 224 and 226 minutes, within about 100 seconds. This event was indicated by the RCS pressure monitor, self-powered neutron detectors, and the source range neutron monitors. It is believed that failure of the supporting crust occurred in the upper and/or center region of the consolidated mass of molten core material, probably near the core periphery (1.5 meters from the bottom of the core) on the east side, as shown in Figure 2-1c. Visual inspections conducted during defueling indicated that the flow of molten core entered the core former on the east side and flowed around the former and then down into the LCSA internals. Analysis of potential flow of molten core materials through fuel assembly location indicated that all of the molten core material could have relocated into the LCSA internals and lower head in less than 1 minute through only one or two fuel rod assemblies.

2.1.5 Phase V - Long-term Cooling of Degraded Core (after 230 Minutes)

Approximately 16 hours after the start of the accident, RC-P-1A was restarted and operated for approximately one (1) week. This pump was replaced by RC-P-2A which operated until April 27, 1979.

There was no evidence of any additional major relocation of molten core materials into the LCSA and lower head after the second core relocation. Thus, the post-accident configuration of the core presented in Figure 2-1c represents the final, stable, and coolable configuration for the materials in the core, LCSA, and lower head regions. Detailed thermal analyses have evaluated the long-term cooling of the consolidated molten mass within the core region. Results of these studies suggest that cooling of this mass occurred over many days to weeks. It was also concluded, based on analyses and observations, that the RV maintained full integrity during all phases of the accident sequence and the subsequent defueling period. Therefore, only a small fraction of the original fuel inventory was relocated outside the RV and was contained within selected RCS pathways.

2.2 Post-Accident Condition of the Plant

An accurate determination of the post-accident state of the plant was required to understand the accident progression and fuel transport mechanisms. Additionally, a thorough knowledge of the properties of the post-accident core debris was necessary to anticipate the conditions to be encountered in defueling the RV and removing fuel from the RCS, RB, and support systems in the AFHB. Detailed analysis of fuel including dispersion and general properties was also essential to completion of the final criticality assessment. This information was developed from several sources (References 2.6 through 2.11): visual inspections of RV internals, metallurgical/radiochemical examinations of samples acquired during the course of defueling, and readings from on-line instrumentation and experimental data developed from smaller-scale tests conducted at various independent facilities.

The original core inventory included approximately 94,000 kg of UO_2 and 35,000 kg of cladding, structural, and control materials. Accounting for oxidation of core materials during the accident and for portions of the upper plenum structure that melted, the total amount of post-accident core debris was estimated to be 133,000 kg.

2.2.1 Reactor Vessel Internals

During the accident sequence discussed in Section 2.1, peak temperatures ranged from approximately 3100°K at the center of the core (molten UO_2), to 1255°K immediately above the core and 723°K at hot leg nozzle elevations. Approximately 50% of the original core became molten. Lower portions of three (3) baffle plates on the east side of the core melted and some of the molten core material flowed into the core bypass region. Approximately 30,000 kg of molten materials flowed from the core to the core bypass region and through the lower internals. Approximately 19,000 kg came to rest on the RV lower head. Figure 2-3 illustrates the major RV components and the post-accident configuration of the core.

The post-accident condition of the upper plenum assembly, original core region, core bypass region, the UCSA, the LCSA, and lower head region are described in the following sections.

2.2.1.1 Upper Plenum Assembly

The upper plenum assembly, which was removed intact, had two (2) damaged zones. Localized variations of damage were evident in each zone. For example, in the limited area above one fuel assembly, ablation of the stainless steel structure was observed; however, grid structures adjacent to the ablated zone appeared to be undamaged. In some regions, the once-molten grid material had a foamy texture, which occurs when stainless steel oxidizes near its melting point. A once-molten mass close to this grid material appeared to be unoxidized, suggesting that some of the hot gases exiting the core were oxygen deficient. The damage to the upper plenum assembly indicated that the composition and temperature of gases exiting the core varied significantly within the flow stream. Only a small quantity of fuel debris was measured within the plenum assembly.

2.2.1.2 Core Region

Figures 2-2 and 2-3 illustrate the end-state configuration of the original core region. A core void or cavity existed at the top of the original core region. Below that, a bed of loose debris rested on a resolidified mass of material that was supported by standing fuel rod stubs. The stubs were surrounded by intact portions of fuel assemblies. A previously molten, resolidified mass was encapsulated by a distinct crust of material in which other fragments and shards of cladding could be identified.

The core void was approximately 1.5 meters deep with an overall volume of 9.3 cubic meters. Of the original 177 fuel assemblies, 42 partially intact assemblies were standing at the periphery of the core void. Only two (2) of these fuel assemblies contained more than 90% of their full-length cross-sections with the majority of fuel rods intact. The other assemblies suffered varying degrees of damage ranging from ruptured fuel rods to partially dissolved fuel pellets surrounded by once-molten material.

The loose debris bed at the base of the core cavity ranged in depth from 0.6 to 1 meters and consisted of whole and fractured fuel pellets, control rod spiders, endfittings, and resolidified debris totaling approximately 26,000 kg. Beneath the loose debris bed was a large resolidified mass approximately 3 meters in diameter. This mass varied in depth from 1.5 meters at its center to 0.25 meters at its periphery and contained approximately 33,000 kg of core debris. The center of this solid metallic and ceramic mass consisted of a mixture of structural, control, and fuel material that reached temperatures of at least 2800°K and possibly as high as 3100°K during the accident. The upper crust of this mass, which consisted of the same material and also reached 2800°K, contained intact fuel pellets near the periphery. The lower crust consisted of once-molten stainless steel, zircaloy cladding, and control rod materials resolidified in flow channels surrounding intact and partially dissolved fuel pellets. The thickness of this lower crust, based on initial video examinations, was estimated to be approximately .01 meters on the average. The resolidified mass was shaped like a funnel extending down toward the fuel assembly lower endfittings near the center of the core.

The standing, undamaged fuel assembly stubs extended upward from the lower grid plate to the bottom surface of the resolidified region of the once-molten materials. These stubs varied in length from approximately 0.2 to 1.5 meters. The longer partial fuel assemblies were located at the periphery of the resolidified mass. On the east side of the core, one (1) fuel assembly was almost completely replaced with once-molten core material; this indicated a possible relocation path into the LCSA and core bypass region for molten material. The standing fuel assembly stubs and peripheral assemblies constituted about 45,000 kg of core debris.

2.2.1.3 Upper Core Support Assembly

This region consists of vertical baffle plates that form the peripheral boundary of the core; horizontal core former plates to which the baffle plates are bolted; the core barrel; and the thermal shield (Figure 2-3). There are a number of flow holes in the baffle and core former plates through which coolant flowed during normal operations. On the east side of the core, a large hole approximately 0.6 meters wide and 1.5 meters high, and extending across three (3) baffle plates and three (3) core former plates was discovered. Adjacent baffle plates on the east and southeast were warped possibly as a result of the high temperatures and the flow of molten material in the bypass region.

It was concluded that molten core material from the core region flowed through the large hole in the baffle plates into the UCSA, circumferentially throughout the UCSA, and downward through the flow holes in the core former plates into the LCSA at nearly all locations around the core. The majority of the molten material appeared to have flowed into the LCSA on the southeast side through the hole in the baffle plate and through the southeast core former plate flow holes.

The circumference of the core region (i.e., the area behind the baffle plates) contained loose debris throughout. The depth of debris varied from approximately 1.5 meters on the east side to a few millimeters on the southwest side. There appeared to be a resolidified crust on the upper horizontal surfaces of the three (3) bottom core former plates; this crust varied in thickness from approximately 0.5 to 4.0 cm. It is estimated that approximately 4000 kg of core debris was retained in the UCSA region. In the small annulus between the core barrel and the thermal shield, fine particulate were observed but no major damage to these components was seen.

2.2.1.4 Lower Core Support Assembly

The LCSA region consists of five (5) stainless steel structures. The structures vary in thickness from 0.025 to 0.33 meters with 0.080 to 0.15 meter diameter flow holes.

Some molten core material flowed through these structures and came to rest on the lower head. There was approximately 6000 kg of resolidified material dispersed at various locations on the circumference of these structures. In several places, resolidified material completely filled the flow holes and columns of once-molten material were observed between the plates. The largest accumulation of resolidified material appeared to have flowed into the LCSA from the east side of the core. Although most of the material was seen on the east to southeast side, many columns of resolidified material were also seen in the LCSA around the periphery of the core beneath the core bypass region.

2.2.1.5 Lower Head Region

The debris in the lower head region accumulated to a depth of 0.75 to 1 meter and to a diameter of 4 meters. The spatial distribution of the material was neither uniform nor symmetric. The surface debris had particle sizes which varied from large rocks (up to 0.20 meters) to granular particles (less than 0.001 meters). The larger rocks, especially in the northeast and southwest regions, were located near the periphery. The debris pile was lower at the vessel center than at the periphery, with granular or gravel-like material observed in the central region of the vessel. A large resolidified mass was identified between the loose debris bed and the lower head of the RV. This mass was approximately 0.5 meters thick in the center and 1.7 meters in diameter. A large cliff-like structure formed in the northern region from once-molten core material. The cliff face was approximately 0.38 meters high and 1.25 meters wide. It was estimated that approximately 12,000 kg of loose core material and 7,000 kg of aglomerated fuel debris relocated into the lower head.

2.2.2 Reactor Coolant System

During the accident, small quantities of fuel debris (Table 2-1) and fission products were transported throughout the RCS (see Figure 2.4). The largest RCS components operated during the accident were the RCPs. The RC-P-2B was the only pump which would respond to a "start" command 174 minutes into the accident. This pump was started and operated for approximately 6 minutes. The operation of this pump was the major driving force for the relocation of fuel debris from the RV. Coolant circulated through the RV by this pump caused a rapid quenching of the highly oxidized, high temperature fuel which resulted in the fuel rods being physically shattered and rubblized.

As the RCP operated, the flow of the "B" loop was in a "forward" (i.e., normal) direction. The flow rate through the RV was sufficient to transport small amounts of fuel debris into the "B" loop where a portion of the fuel debris relocated into the "B" hot leg and settled out into the decay heat drop line. The decay heat drop line connects to the bottom of the horizontal section of the

"B" hot leg and was found to contain some fuel debris, presumably as a result of the RC-P-2B operation (see Table 2-1). The coolant continued to flow up the "candy cane" and deposited fuel material on the "B" OTSG upper tube sheet. The tube sheet acted as a "strainer" for the collection of fuel debris transported outside the RV. However, a small quantity of fuel debris flowed down through the steam generator tubes and was deposited on the lower head of the "B" OTSG and J-legs. As the coolant continued to flow, relatively smaller quantities of fuel debris were then deposited in the "B" reactor coolant pump and cold legs.

At approximately 16 hours, the RC-P-1A pump was started. The operation of this pump deposited finely divided silt-like debris in the top of the "A" OTSG and the bottom of the "B" OTSG due to reverse flow in the "B" OTSG loop. RC-P-1A, which experienced excessive pump vibration, operated for approximately one (1) week and was replaced by RC-P-2A, which operated until April 27, 1979. This pump was shutdown because all pressurizer level indicators failed.

Cold shutdown conditions (i.e., RCS temperature below 100°C) were established on the evening of April 27, 1979. After all RCP operations were terminated, the system circulation and cooldown was achieved by natural convection/circulation heat transfer. This natural circulation continued into approximately October 1979. Eventually, there was insufficient thermal driving head to maintain continuous natural circulation and a flow transient in the RCS, referred to as the "B" loop "burp," began to occur frequently over a period of several months. This phenomena occurred because the coolant in the "B OTSG and "B" loop cold legs gradually cooled until the density of this coolant increased sufficiently to initiate natural circulation flow in the "B" loop. The flow was sustained until the warmer fluid from the RV displaced the cold fluid in the "B" OTSG and cold leg. Repositioning of the coolant of different densities continued until hydraulic balance was achieved. The coolant was then stationery for several days until another "burp" occurred. This repeated flow rate phenomenon was believed to have transported small quantities of finely divided fuel debris from the RV to the steam generators and other RCS locations in both RCS loops.

In summary, there were two (2) methods of transport of fuel to ex-vessel locations. The primary transport method was a sequential operation of the RCPs: RC-P-2B, RC-P-1A, and RC-P-2A. The secondary transport method was attributed to the "burping" phenomenon during natural circulation. Table 2.1 provides an estimate of the quantity of fuel relocated into the RCS during the accident sequence and resulting thermal hydraulic phenomenon (References 2.12 through 2.14).

2.2.3 Reactor Building

Reactor coolant was discharged from the RCS through the PORV located on top of the pressurizer. The PORV discharges to the RCDT which is located in the basement of the RB (see Figure 2-5).

The RCDT contains two (2) safety components: a relief valve which discharges to the RB sump and a rupture disk which discharges to the RB floor adjacent to the RCDT cubicle. Both safety devices were believed to have performed their respective safety functions. The rupture disk was subsequently found in an open or ruptured condition, as expected. If the relief valve had initially operated during the pressure buildup in the RCDT, it would be expected to reseat after the rupture disk opened, thereby minimizing any continuous release to the RB sump via that pathway.

At approximately 138 minutes into the accident, the operators realized that the PORV was not closed and they manually closed the pressurizer block valve. Further loss of coolant and gaseous fission product release from the primary coolant system to the RB was essentially terminated. However, the block valve had to be cycled repeatedly to maintain system pressure. This cycling of the block valve permitted the transport of fission products, noble gases, and small quantities of fuel debris through the pressurizer and PORV into the RCDT, and subsequently into the RB through the rupture disk discharge. The largest concentration of fuel debris in the RB was located on the basement floor in the vicinity of the RCDT rupture disk discharge.

The MU&P System was operated during the accident and recovery period. The MU&P System inlet piping is fed from the RCS on the suction side of the RC-P-1A. The first major components in this system are the letdown coolers which are located in the basement of the RB (see Figure 2-5). Thus, some fuel debris was transported into the letdown coolers and associated piping.

In summary, a relatively small quantity of fuel debris (see Table 2-1) was released to the RB principally through the RCDT onto the basement floor. The remaining quantities of fuel debris in the RB external to the RCS were identified with tank and piping systems. Table 2.1 provides an estimate of the quantity of fuel relocated into the RB during the accident sequence (References 2.13 through 2.16).

2.2.4 Auxiliary and Fuel Handling Buildings

A small quantity of fuel debris was transported to the AFHB during the accident. The majority of this material was transported through the MU&P System and into the RCBTs. This system is fed from the RCS cold leg side of the "A" loop through the letdown coolers and discharges into the AFHB via the RCBTs. Although this system communicates through a large number of the cubicles in the AFHB, only a small amount of fuel debris was transported into the system as indicated by the fact that very little fuel debris was measured in upstream components such as the block orifice, MU&P demineralizer filters, MU&P demineralizers, and the makeup filters.

The block orifice is the normal pressure reduction device for flow rates up to 45 gpm through the MU&P system. The block orifice and its isolation valve became blocked during the accident;

subsequently, they were bypassed. As a result, very little fuel debris was measured in the block orifice and its associated piping. The letdown flow was directed to the letdown filters and purification demineralizers at very low rates during the accident and was then routed to RCBT "A" and the makeup tank. Letdown flow was lost several times during the accident due to flow blockage. More than 24 hours after the initiation of the accident, the purification demineralizers also were bypassed and letdown was directed to RCBT "B". Due to the flow blockage of the letdown coolers and restrictions in the block orifice, fuel transport to the filters, demineralizers, and RCBTs was limited.

Another potential pathway for transport of fuel debris to the AB was through the Seal Injection System. The Seal Injection System return line, which is downstream of the reactor coolant pump seals, receives reactor coolant pump seal return water. As a result of this, potential trace amounts of fuel debris may have been transported to the Seal Injection System.

RCBTs A, B, and C also contained fuel debris as a result of their use during the accident, interconnection with the MU&P System, and as a result of RCS water processing and removal of water from the RB sump and the AB sump.

In summary, a relatively small quantity of fuel debris was transported into the AFHB (see Table 2-1), principally through the RCBTs and the MU&P System. Some of this fuel debris may have further relocated into other systems as part of the post-accident water processing and cleanup activities (References 2.13 and 2.14).

2.3 Fuel Transport and Relocation Due To Cleanup Activities

As a result of the accident sequence and resultant cleanup activities, a small, but measurable quantity of fuel debris was transported into the various plant systems, tanks, and components. These cleanup activities were a necessary part of restoring conditions in the plant and significantly assisted recovery operations in meeting defueling completion objectives.

In the RB, the majority of the post-accident fuel material relocation from cleanup and defueling operations was attributed directly to the transfer of RV components. Major components have been removed from the RV which contained relatively small quantities (<10 kg) of fuel debris. These components, which are currently stored in various RB locations, include the RV head, upper plenum assembly, internal RV structures (i.e., endfittings, LCSA grid plates, distributor plates, grid forging, etc.), and contaminated equipment/tools. In all cases, these components and equipment were physically cleaned and decontaminated to the extent practical and surveyed for fuel content before storage. Some additional small amount of fuel material was relocated to the RB basement as part of tool flushing and building decontamination activities. In each case, the effect of this fuel material relocation is quantified as part of the fuel measurement activities reported herein.

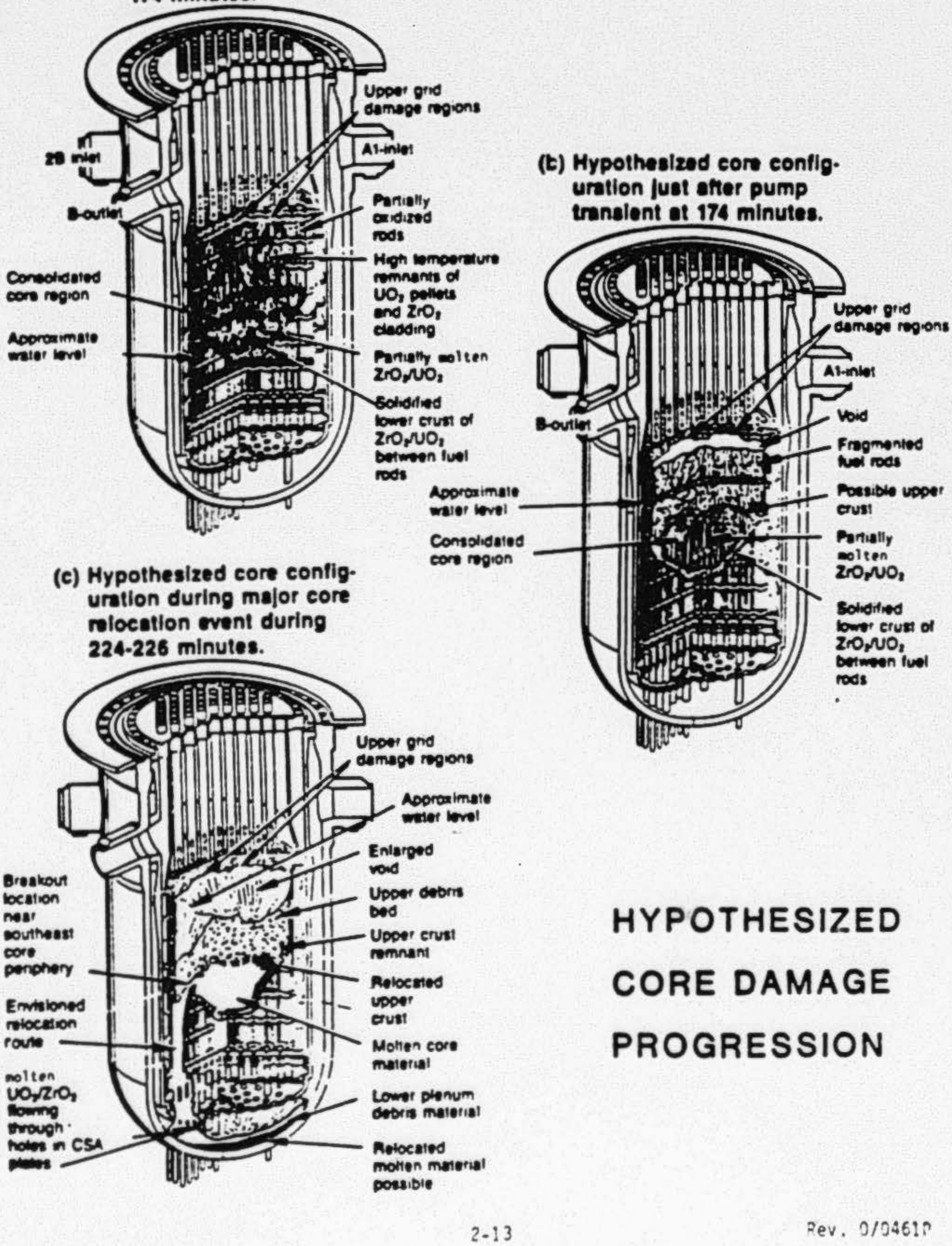
In the AFHB, the primary cause of fuel debris relocation from cleanup operations was water processing through the RCBTs, MWHT, SRSTs, and SDS monitoring tanks. Additionally, fuel debris material may have relocated into the FHB Spent Fuel Pool "A" as part of fuel canister transfers from the RV. While every effort was made to flush residual fuel material from the external surfaces of the defueling canisters, a small quantity of uncontained fuel material may have been transferred into the "A" fuel pool as part of handling and movement of over 300 defueling canisters. Post-defueling cleanup activities are expected to reduce the amount of residual fuel and ensure subcriticality.

TABLE 2-1

POST-ACCIDENT ESTIMATED EX-VESSEL
CORE MATERIAL DISTRIBUTION
(References 2.12 through 2.16)

	Kilograms
Reactor Coolant System	
"A" Side	
Hot Leg	1
OTSG Upper Tube Sheet	1
Tube Bundle	3
Lower Head	1
J-Legs	1
Reactor Coolant Pumps	2
Cold Legs	1
"B" Side	
Hot Leg	8
Decay Heat Drop Line	30
OTSG Upper Tube Sheet	125
Tube Bundle	9
Lower Head	1
J-Legs	6
Reactor Coolant Pumps	20
Cold Legs	7
Pressurizer	12
Reactor Building	
RB Basement/Sump	5
Reactor Coolant Drain Tank	1
Letdown Coolers	4
Core Flood System	1
Auxiliary/Fuel Handling Building	
Makeup and Purification System	6
Seal Injection System	1
Reactor Coolant Bleed Tanks A, B, and C	15
Waste Disposal Liquid System	1

FIGURE 2-1



(a) Hypothesized core configuration just prior to pump transient at 174 minutes.

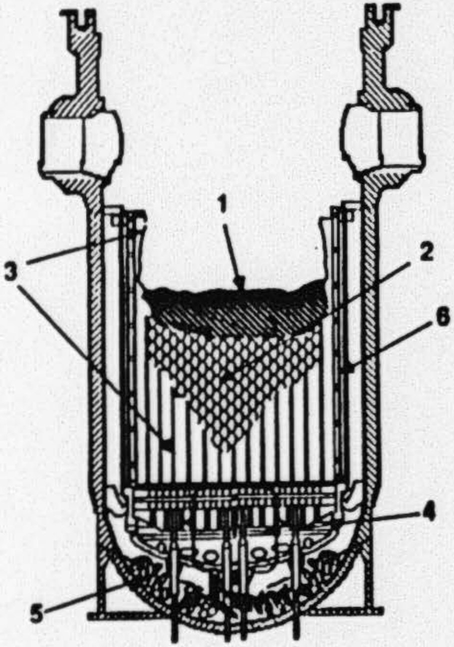
(b) Hypothesized core configuration just after pump transient at 174 minutes.

(c) Hypothesized core configuration during major core relocation event during 224-226 minutes.

HYPOTHESIZED CORE DAMAGE PROGRESSION

FIGURE 2-2

POST-ACCIDENT ESTIMATED CORE MATERIAL DISTRIBUTION



ZONE	DESCRIPTION	ESTIMATED QUANTITY (KG)
1	Upper Debris Bed	26,000
2	Resolidified Mass	33,000
3	Intact Assemblies	45,000
4	LCSA (loose debris and resolidified mass)	6,000
5	Lower Head (loose debris and resolidified mass)	12,000 7,000
6	UCSA (loose debris and resolidified mass)	4,000
	TOTAL -	133,000

TMI-2 Core End-State Configuration

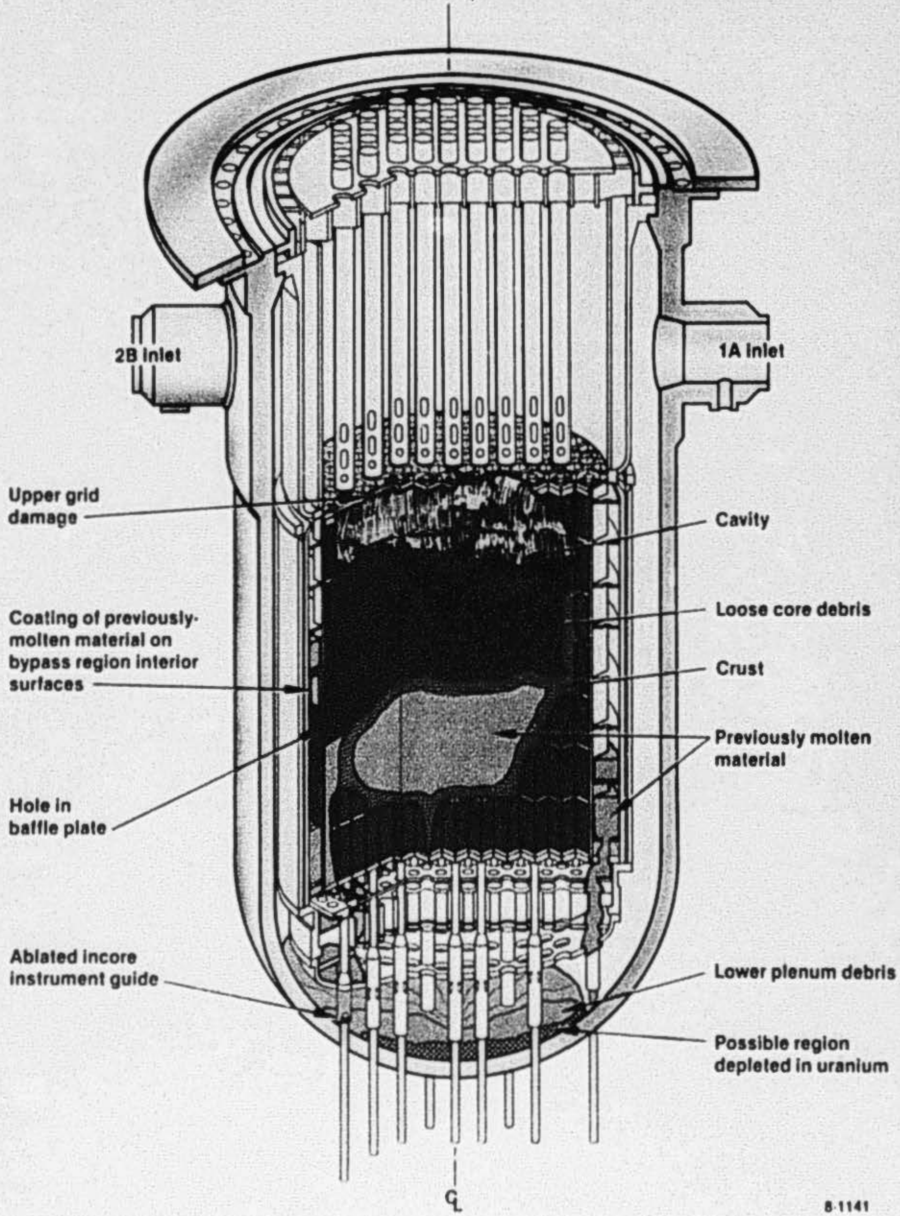
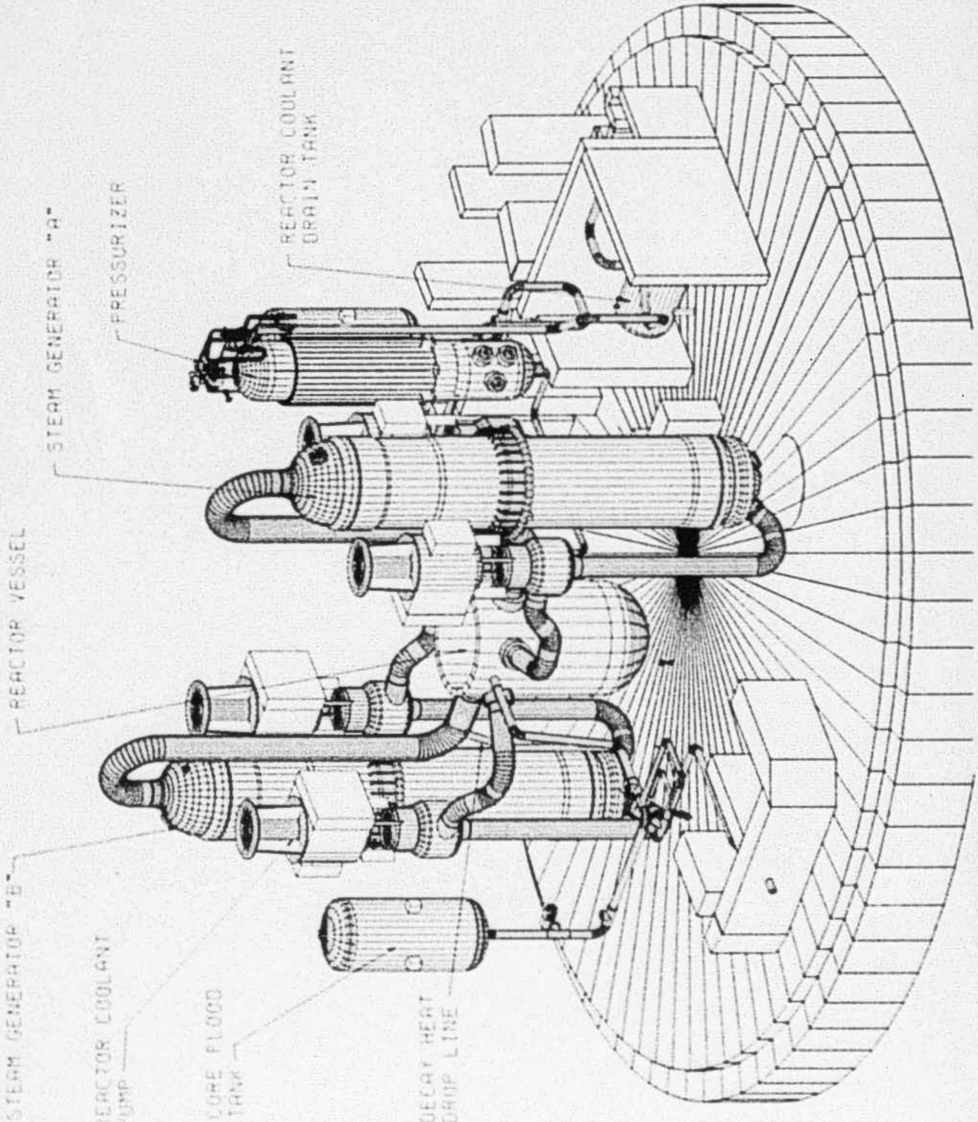


FIGURE 2-4

REACTOR COOLANT SYSTEM COMPONENTS



TMI-2 REACTOR BUILDING BASEMENT

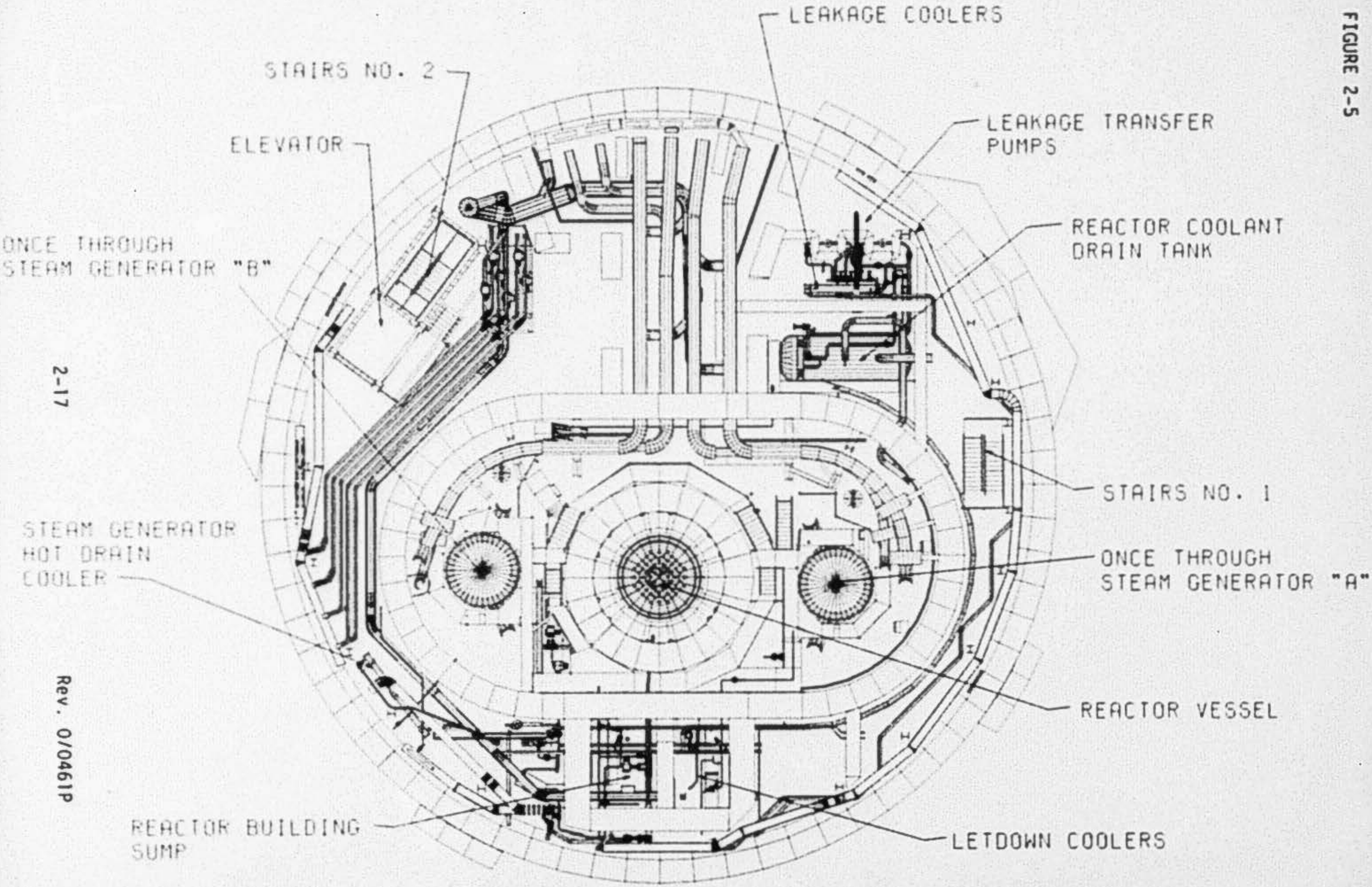


FIGURE 2-5

3.0 FUEL SURVEY TECHNIQUES

This section describes the various methods used at the TMI-2 facility to locate and quantify residual fuel (Reference 3.1 and 3.2). These methods included direct measurement by instrumentation, visual inspection, and sample collection and analysis. The methods selected were influenced by many factors including accessibility, measurement uncertainties, and equipment sensitivity. The actual measurement techniques employed for the various locations are identified. Additional measurements will be conducted at selected plant locations as part of the SNM accountability program. While these measurements will serve to confirm the data contained in the DCR, SNM measurements are not prerequisites for transition from Mode 1 to Mode 2. The following includes a discussion of the various methods and the factors that influenced their selection.

Because of the diverse locations and quantities of fuel dispersed throughout the TMI-2 facility, measurement of residual fuel required a variety of methods. Measurement methods were matched to geometry, source strength, and physical form of fuel debris. Complicating factors included high radiation backgrounds, complex shielding, and limited access to fuel locations. Five (5) general methods were used for fuel detection (detection of gamma rays, neutrons, alpha particles; sample and analysis; visual evidence). Each detection method included a number of specific techniques that are described below.

3.1 Gamma Dose Rate and Spectroscopy Techniques

Gamma detection for fuel measurement included the use of gross gamma dose rate and gamma spectroscopy techniques. Gross gamma fuel estimates were performed in the AFHB to generate fuel estimates for some cubicles. The technique used was gamma measurement with a shielded directional detector. Measurements were taken at numerous locations on pipes and components in a cubicle. Possible fuel distributions were modeled based on the cubicle geometry, accident history, and analysis of gamma flux from debris samples. Matching the models with the measured dose rates yielded an estimate of residual fuel in the cubicle.

Gamma spectroscopy was used to quantify the amount of a particular radioactive isotope present by measuring the characteristic gamma radiation emitted. Typically, the emitted gamma radiation was detected by sodium iodide or pure germanium material. The detected radiation impulses were converted to an electrical signal which, when processed by an analyzer, identified the relative energy of the originally emitted gamma radiation. Gamma spectroscopy was used at TMI-2 to measure the quantity of Ce-144 and/or Eu-154 present in discrete locations. The quantity of Cerium or Europium present was converted to the quantity of residual fuel present based upon the calculated ratios and actual measurements of Cerium/fuel and Europium/fuel ratios.

Two (2) gamma spectroscopy detector systems were utilized for residual fuel measurements at TMI-2. NaI(Tl) detector measurements were performed in many AFHB cubicles from 1983 through 1987. The NaI(Tl) detector has a good efficiency and adequate sensitivity to detect the characteristic

Ce-144 2.185 MeV gamma radiation. Limitations on the use of the NaI(Tl) detector in the AFHB were due to the relatively high ambient dose rates in several of the cubicles during early fuel characterization measurements and the relatively short half-life of Ce-144 (284 days).

HPGe detector measurements also have been performed. HPGe detectors have the advantage of a much better energy resolution capability, compared to NaI(Tl) detectors, but a lower relative detection efficiency. In addition, they are much more sensitive to ambient gamma radiation levels. HPGe detectors also require liquid nitrogen cooling to operate. HPGe detector measurements were performed to identify both Ce-144 (2.185 MeV gamma radiation) and Eu-154 (0.723 MeV and 1.274 MeV gamma radiation).

A Si(Li) Compton recoil gamma ray spectrometer was used to quantify the 2.185 MeV Ce-144 gamma radiation in the A and B MU&P demineralizer cubicles. This detector obtains a continuous spectra which is then used to determine the intensity of the 2.185 MeV gamma radiation. The technique utilizes a shielded directional gamma probe to isolate and quantify fuel deposits inside piping and/or components in each cubicle.

3.2 Neutron Detectors and Activation/Interrogation Techniques

Neutrons from spontaneous fission and (γ,n) reactions are directly proportional to fuel quantity. However, the neutron emission of TMI-2 fuel is quite small, approximately 0.2 neutrons/grams-seconds. Passive neutron detection methods may be used to detect this small flux but are likely to result in a high minimum detectable limit. Active neutron assay methods interrogate fuel with a neutron source and detect induced fission neutrons. Active methods, where practical, are more accurate for small amounts of residual fuel as well as for direct measurement of the U-235 content.

Passive neutron detection methods used at TMI-2 included solid state track recorders, copper activation foils/coupons, and BF_3 detectors (Reference 3.3).

- SSTRs were used to estimate the quantity of residual fuel in the MU&P A and B demineralizer cubicles. The SSTRs used 93% enriched U-235 foils, which are attached to a metal support plate and layered between two (2) lucite sheets. The enriched U-235 foil emits induced fast fission neutrons that create visible tracks in the lucite sheets. The fission neutrons are induced by thermalized neutrons emitted via spontaneous fission of the fuel being measured. The number of visible tracks is proportional to the thermal neutron flux, which is proportional to the quantity of fuel present.

- Copper activation coupons become irradiated in the presence of a neutron flux. The Cu-64 then decays by positron emission resulting in two (2) 511 MeV gamma rays scattered at 180°, with a 0.66% yield. By using a coincidence counting system consisting of two (2) NaI(Tl) detectors, discrimination of this dual emission from background is possible. Copper activation is insensitive to gamma radiation, making this method particularly useful in areas of high gamma fields.

- A BF_3 neutron detection system consists of BF_3 thermal neutron detection tube moderated by polyethylene material. The polyethylene thermalizes fast neutrons from the fuel debris so the BF_3 system can count them more efficiently. This technique is also useful for areas with moderately high gamma background radiation levels.

Active neutron interrogation is more sensitive than passive counting for quantifying small deposits of fuel debris. At TMI-2, an Antimony (Sb) - Beryllium (Be) photoneutron interrogation method was used. This photoneutron interrogation method uses an Sb-Be photoneutron source to produce low-energy (approximately .024 MeV) interrogating neutrons via the Be (γ,n) reaction by irradiating Beryllium with the 1.692 MeV gamma ray emitted from a Sb-124 isotope. These interrogating neutrons impinge upon the fuel and induce fission reactions in the fissile material contained in the fuel. Some of the fission neutrons returning from the surrounding fuel debris are detected by a He-4 fast neutron recoil proportional counter. The He-4 neutron counter can differentiate the higher-energy induced fission neutrons from the lower-energy photoneutron source and gamma rays on the basis of the pulse height signal; with directional shielding, it can also operate effectively and efficiently in a substantial radiation field.

3.3 Alpha Fuel Detectors

Alpha particle detection was used to quantify fuel debris on both steam generator tube surfaces and on RCS component sample surfaces. Because of their short range and high potential for absorption, alpha detection is only used for fuel distributed in thin films. To measure fuel on the RCS surface area, a thin-walled alpha detector was deployed into a number of OTSG tubes. The tubes were first swabbed to remove dirt, loose films, and water so that alpha particles contained in the adherent films could reach the detector. Alpha scanning was also performed on samples of stainless steel RCS components.

3.4 Direct Sampling and Analysis Techniques

Two (2) types of samples were used at TMI-2 for residual fuel determinations: core debris and RCS components. Samples of core debris from fixed locations were analyzed to determine fuel and radionuclide content. Samples were analyzed by gamma spectroscopy, alpha counting, and chemical/physical techniques. Estimates of debris volume or radiological models then incorporated the analytical results to derive fuel quantities. Samples of RCS components were used to estimate the density of fuel fixed on surface films. Representative samples of various core debris deposits were extrapolated to represent the total surface area of similar components. A difficulty with the sampling program is ensuring that samples are representative of the fuel content of the area being assayed. Because of the inherent uncertainty, it is preferable to use sampling techniques in conjunction with other methods that measure fuel directly.

3.5 Visual Inspection

As an aid to defueling operations, miniature radiation resistant video cameras and underwater lights have been used extensively to locate fuel debris concentrations. These tools can also be used to estimate fuel quantity. Using video cameras, the physical extent of debris deposits is mapped in three (3) dimensions, using known reference points or landmarks as dimensional indicators. Given good lighting conditions, the vertical and lateral extent can be estimated fairly accurately, but depth (dimension along line-of-sight) is much less easily determined. The physical distribution is then used to estimate volume. Surface texture and other subtle factors are used by experienced inspectors to identify and compare the subject debris deposit to other similar material for which sample analysis data exists. The sample data for fuel material is then used to estimate the density and composition of the deposit. Visual inspection techniques are not suitable for estimating surface films since it is impossible to determine film thickness from a two-dimensional video image.

3.6 Fuel Measurement Uncertainties

Some of the" estimate of record" quantities reported for residual fuel are referred to as the MDL. By definition, this means that the measurement technique did not detect a statistically significant number of events (counts) related to fuel material. Therefore, the true quantity of fuel debris believed to be in the target area, system, or component is equal to or less than the reported MDL with 95 % confidence. For example, if the residual fuel quantity is reported to be a MDL value of 3 kg, the true quantity of residual fuel could be any value from 0 to 3 kg.

Physical measurement of fuel quantities is subject to imprecisions. The accuracy of fuel debris measurements is significantly impacted by the inaccessibility of the fuel locations, high background gamma radiation dose rates, unknown distribution characteristics and low neutron emission rates for TMI-2 fuel. The accuracy of a measurement also is impacted by the fuel tracer characteristics. The preferred high energy fuel tracer isotope Ce-144 has a very short half-life (i.e., 284 days) as compared to Eu-154, which has a longer half-life (i.e., 8.8 years), but is a much lower energy tracer isotope. Errors are also due in part to the combined effects of counting statistics, representativeness of samples to the whole, high radiation interference to background which elevates minimum detection levels, complex and undefined fuel distribution geometries, lack of personnel access requiring use of remote measurement techniques, and varying signal absorption rates due to the presence of structural members and blanketing layers.

Because of the inability to control some of the variables described above, fuel measurements are inherently attended by some level of uncertainty. These uncertainties are minimized to the extent possible by the judicious selection of measurement techniques and a graded application of resources. In any event, the variables which impact the precision and accuracy of fuel measurements will result in some uncertainties, but these uncertainties are accounted for in the bounding values reported herein.

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3.7 Fuel Measurement Selection

Table 3-1 presents a matrix of fuel locations versus measurement methods for the various TMI-2 residual fuel locations. All areas containing residual fuel are listed and cross-referenced with the techniques selected for the area. Locations marked with an "X" have been measured as of this writing. Locations marked with an "E" indicate estimates of residual fuel have been performed based on a review of accident flow data, radiological controls measurements, and existing fuel measurement data from similar locations or components. Locations marked with a "P" indicate areas where measurements are proposed to be performed.

TABLE 3-1

FUEL MEASUREMENT SELECTION

Fuel	Measurement Methods	Gamma	NaI	HPGe	Cu Foil	Alpha	Sample	Analysis
AX004	Seal Injection Valve Room							E
AX005	MU Pump Room 1C	X						
AX006	MU Pump Room 1B		X					
AX007	MU Pump Room 1A		X					
AX008	Spent Resin Stor. Tank 1B							E
AX009	Spent Resin Stor. Tank 1A							E
AX010	Spent Resin Stor. Tank Pump							E
AX011	Aux. Sump Pump Valve Room		X					
AX012	Aux. Bldg. Sump Tank Room		X					
AX019	WDL Valves		X					
AX020	RC Bleed Tanks 1B, 1C		X					
AX021	RC Bleed Tank 1A			X				
AX024	Aux. Bldg. Sump Filters	X						
AX026	Seal Inj. Filters MU-F4A,B		X					
AX102	RB Sump Filters							E
AX112	Seal Return Coolers & Filters		X					
AX114	MU&P Demin. 1A						X	
AX115	MU&P Demin. 1B						X	
AX116	Makeup Tank Room		X					
AX117	MU&P Filters 2A,B 5A,B		X					
AX124	Conc. Liquid Waste Pump							E
AX129	Deborating Demin. 1B		X					
AX130	Deborating Demin. 1A		X					
AX131	Misc. Waste Tank							E
AX134	Misc. Waste Tank Pumps							E
AX218	Conc. Waste Storage Tank Rm							E

X=COMPLETED, P=PROPOSED, E=ESTIMATED

TABLE 3-1 (Cont'd)

FUEL MEASUREMENT SELECTION

Fuel Measurement Methods		Gamma	NaI	HPGe	Cu Foil	Alpha	Sample	Analysis
FH001	Makeup Suction Valve Room		X					
FH003a	Makeup Discharge Valve Room		X					
FH003b	Makeup Discharge Valve Room		X					
FH014	Annulus							E
FH101	MU&P Valve Room		X					
FH106	Monitor Tanks & Sample Sink							E
* FH109	Spent Fuel Pool A	P					P	
* FH110	SDS Spent Fuel Pool							E
FH111	Fuel Cask Storage							E
FH112	Annulus							E
FH302	SDS Operating Area							E

* - See Section 5.1.2.8 for a discussion of the planned fuel measurement in Spent Fuel Pool "A".

X=COMPLETED, P=PROPOSED, E=ESTIMATED

TABLE 3-1 (Cont'd)

FUEL MEASUREMENT SELECTION

Fuel Measurement Methods		Gamma	NaI	HPGe	Cu Foil	Alpha	Sample	Analysis
RB01	Letdown Coolers Cubicle		X					
RB02	RB Sump						X	
RB03	RC Drain Tank Cubicle						X	
RB04	RB Basement Floor		X				X	
RB05	Under Reactor Vessel							E
RB06	Letdown Line							E
RB11	Decay Heat Drop Line	X						
RB21	Reactor Coolant Pumps							P
RB22	RCS Cold Legs							P
	RCS J-Legs	X			X			
RB23	Reactor Vessel							P
RB31	Pressurizer		X			X	X	
RB32	OTSG's	X	X		X	X	X	
RB33	Core Flood Tanks & Drain Lines			X		X	X	
RB34	Incore Guide Tubes			X				
RB35	Upper Plenum Assembly						X	
RB36	Reactor Vessel Head						X	
RB37	RCS Hot Legs							P
RB38	Pressurizer Surge Line		X					
RB39	Pressurizer Spray Line							E
RB40	Fuel Transfer Canal	P					P	
	Endfittings							E

X=COMPLETED, P=PROPOSED, E=ESTIMATED

4.0 FUEL REMOVAL ACTIVITIES

This section provides a detailed discussion of the major fuel removal activities undertaken during the TMI-2 cleanup. As part of fuel removal activities, an extensive post-accident plant characterization was conducted. The resultant fuel location data base and building dose rate maps were incorporated into the defueling planning. For those areas of the TMI-2 facility which had relatively small, but measurable quantities of residual fuel, the focus was on dose reduction to support personnel access and/or assure proper plant system operations, maintenance, and surveillance. Defueling to the extent reasonably achievable was expected to be achieved as a byproduct of decontamination and dose reduction activities. The locations not requiring extensive defueling were predominantly identified with the AFHB cubicle areas and the general RB areas. The major fuel removal activities were focused on the RCS and the RV locations. The following provides a summary discussion of the major fuel removal activities, including details of the defueling approach, the equipment and techniques utilized, and an assessment of the relative effectiveness of fuel removal activities for the major structures, systems, and components within the TMI-2 facility.

4.1 Auxiliary and Fuel Handling Buildings

As discussed in Section 2, the TMI-2 AFHB was contaminated as a result of the March 1979 accident and subsequent plant stabilization and water processing activities. A small, but measurable, quantity of fuel was transported into the piping and tanks of the MU&P System and the WDL System components, which are located in the AFHB.

The AB contains the support systems which were originally designed to purify the reactor coolant, remove soluble radionuclides, and provide for the addition and/or removal of water treatment chemicals. In addition, the FHB provides the storage location for the TMI-2 defueling canisters prior to shipment.

4.1.1 Cleanup Approach

Cleanup activities in the AFHB were focused on facilitating personnel access to those areas and components required to maintain the RCS in a stable condition, prepare for and conduct filtration and ion exchange removal of soluble and insoluble radionuclides in reactor coolant, and reduce the overall AFHB curie content. The cleanup activities included water removal, surface decontamination, system flushing, tank sludge and demineralizer resin removal, and removal of various filters and letdown block orifice.

The amount of fuel relocated to the AFHB as a result of the TMI-2 accident and subsequent water processing and decontamination activities has been estimated to be significantly less than the SFML of 140 kg (Appendix B). However, early in the AFHB cleanup activities, there was a concern that some tanks and/or piping runs might contain significant quantities of fuel sediment. Therefore, system and tank flushes were performed using borated water.

Subsequent measurements (References 4.1, 4.2, and 4.3) of suspected fuel deposit locations have determined that the largest single quantities in discrete volumes were less than 10 kg and that the overall AFHB residual fuel inventory probably did not exceed 40 kg at any given time. The use of borated processed water for system flushes resolved any criticality safety concerns associated with AFHB recovery. Because of the demonstrated lack of a critical fuel mass, there was no dedicated effort to "defuel" any AFHB component or area. Instead, fuel removal occurred as a byproduct of dose reduction decontamination, water processing, sludge transfer, sludge processing, and/or resin removal.

The initial cleanup of the AFHB took place during the early plant stabilization phase of the TMI-2 cleanup program (Reference 4). This effort consisted of removing the water that flooded the lower level of the AFHB during the accident and performing surface decontamination of the floors, walls, and equipment. The goal of this initial cleanup was to reduce the overall loose contamination throughout the AFHB and to reduce the requirement for respirators due to airborne radioactivity. In addition, there was a need to reduce the radionuclide content of water that was stored in tanks in the AFHB. This latter task was accomplished by the EPICOR II system.

The corridors of the AFHB were successfully decontaminated. Most of the accident-generated water contained in the AFHB was processed. General area access to the AFHB no longer required respirators. Nonetheless, after the initial cleanup, a significant decontamination task remained. Several cubicles remained highly contaminated and had high general area dose rates. In addition, many of the surfaces that had been decontaminated were becoming recontaminated as radionuclides initially absorbed into the concrete surfaces began to leach out. As a result , a significant decontamination program (Reference 4.5) was undertaken and a system for removal of tank sludge and demineralizer resins was designed, fabricated, and installed. The overall objective of these efforts was to ensure that the AFHB would not pose a threat to public health and safety as a result of a long-term radionuclide inventory which could contribute to unacceptable airborne radioactivity levels.

In the second phase of the AFHB recovery program, the conditions of the highly contaminated cubicles in the AFHB were determined. In some cases, this was possible by routine radiological survey techniques. However, in several instances it was necessary to use remotely deployed radiation monitoring devices or specially designed robotic equipment. To implement this program, the assistance of DOE national laboratories and major universities was obtained. State of the art robotic deployment and radiation monitoring equipment was utilized. Unique solutions to the problems of decontaminating highly contaminated equipment, components, piping, and tanks were developed.

The next step in the AFHB cleanup program was the implementation of the specific decontamination techniques that had been developed. Surface decontamination techniques are described in detail in Section 4.1.2. In general, they included water flushing, mechanical abrading (commonly referred to as scabbling) and/or removal of surface coatings and subsurface layers, and actual removal of concrete surfaces followed by recoating and painting in some areas.

System decontamination methods also had been developed. In general, all MU&P and WDL System piping and components were flushed with processed water. In addition, the letdown block orifice and the makeup filters were removed. Finally, several portions of the MU&P and WDL piping and tanks were physically isolated and drained.

Removal of the sludge deposited in some of the piping and tanks in the AB was performed using a specially designed sludge removal and processing system. This system also was used to remove resin from the cleanup and MU&P demineralizers.

Significant dose rate reductions were achieved in nearly all of the cubicles; most cubicles were acceptably decontaminated (Reference 4.6). An example of the success achieved in removing residual fuel from the AFHB is the MU&P demineralizers. It is noteworthy that the block orifice removal resulted in the elimination of approximately 370 grams of the initially estimated 400 grams of fuel. Post-defueling activities such as final draindown, water processing, and fuel pool decontamination are expected to further reduce the AFHB residual fuel inventory.

4.1.2 Auxiliary and Fuel Handling Building Cleanup Equipment and Techniques (Reference 4.7)

As discussed above, the decontamination and cleanup of the AFHB required the use of mechanical decontamination methods, state of the art robotic equipment, surface treatment with strippable coatings, and, in the case of some porous concrete surfaces, removal of concrete that had absorbed radionuclides. In addition, water processing system components such as piping, tanks, and pumps required internal system flushes, sludge removal, and resin sluicing. The detailed description of the major equipment and techniques used to accomplish these tasks is described below.

4.1.2.1 Mechanical Decontamination

Mechanical decontamination is defined as the removal of radioactive contamination by rubbing, washing, brushing, or mechanical abrading. The equipment and techniques used to perform mechanical decontamination in the AFHB included:

4.1.2.1.1 Hands-On Decontamination

Hands-on decontamination of contaminated surfaces consisted of cleaning unwanted material from dirty surfaces by wiping, washing, and/or brushing surfaces, usually with water, detergent, or an abrasive grit in order to remove the contamination. Additional decontamination was accomplished by use of mechanically-powered hand brushes or floor brushes.

4.1.2.1.2 High-Pressure Water Spray and Flushing

Many cubicles and surfaces in the AFHB were sprayed with high-pressure water to remove tightly adherent contamination. Water temperature and pressure varied depending on the target object and surrounding equipment. Temperatures varied from ambient to over 65°C. Similarly, pressures varied from 60 psi to nearly 6000 psi. Chemical agents and abrasive grit were not utilized with high-pressure water spraying.

4.1.2.1.3 Kelly Vacuumac

The Kelly Vacuumac is a steam cleaning machine, somewhat similar to a conventional carpet steam cleaning machine. The vacuumac utilizes a steam wand to direct steam and hot water at a target surface. The loosened contamination and condensate water are then vacuumed into a 55 gallon drum. The Kelly Vacuumac was often used in conjunction with other mechanical decontamination methods, such as high-pressure water spray and scabbling.

4.1.2.1.4 Concrete Scabbling

Scabbling (i.e., removal of a portion of a concrete surface) was required to overcome the leaching phenomena observed after the first phase of AFHB decontamination. Scabbling of the concrete floors in many of the contaminated lower level corridors and cubicles was performed by breaking the surface, vacuuming the residue, and packaging it for disposal as radioactive waste. Scabbling removed the surface coating and as much as 1/8 inch of material for each pass; most scabbling involved 1 or 2 passes. Scabbling was generally followed by surface recoating and painting. For surfaces scabbled to a depth greater than that achieved with 2 passes, additional surface repair was required prior to coating and painting.

4.1.2.2 Robotic Equipment

Characterization of the radiological environment and cleanup of several areas of the AFHB was performed with the assistance of robotics. These devices were used to deploy cameras for visual inspection, radiation monitoring instrumentation, and decontamination equipment.

4.1.2.3 System Decontamination

The internal surfaces of some piping and components were contaminated with both fission products and residual fuel as a result of the accident and subsequent water processing activities. In the AFHB, this contamination was a significant contributor to the overall general dose rate in several cubicles. A program of piping, tank, and pump system flushes was implemented to remove as much of the internal system contamination as practical. In two (2) cases (i.e., the letdown block orifice and the makeup filters), the removal of internal components resulted in fuel removal. System flushing of internal piping, tanks and other components was performed utilizing processed water. Systems suspected to contain fuel were flushed using borated water. All piping and components which had high radiation dose rates and/or were suspected of containing residual fuel, with the exception of selected in-service components, were flushed. The following systems underwent internal system flushing: MU&P System; WDL System; OTSG Recirculation System; Spent Fuel System; DHR System; and Nitrogen System.

In addition to flushing, resins and filters which were highly contaminated as a result of fission product deposition were also removed. Wherever possible, system piping, tanks, pumps, filter housings, and resin tanks were left in a drained condition and were physically isolated by closed, tagged valves.

4.1.2.4 Block Orifice and Makeup Filters (References 4.8 and 4.9)

The TMI-2 block orifice was originally designed to reduce the reactor coolant pressure from the operating system pressure to the pressure of the MU&P System. As discussed in Section 2.2.4, during the accident, the block orifice was clogged and flow through the block orifice was lost. Letdown flow was restored during the accident by bypassing the block orifice. Subsequent radiation surveys of the block orifice, performed in 1982, revealed significant fission product content and a small amount of residual fuel. The block orifice was removed from the letdown flowpath of the MU&P System in 1986. The block orifice was surveyed for residual fuel content prior to shipment offsite. Gamma spectroscopy measurement of the determined that approximately 400 grams of fuel were originally deposited in the block orifice of which approximately 370 grams were removed with the block orifice.

The TMI-2 MU filters were originally installed downstream of the block orifice and upstream of the makeup demineralizers. The filters were designed to remove insoluble contaminants from reactor coolant prior to purification by the demineralizers. During the TMI-2 accident, the MU filters became clogged after the block orifice was bypassed and reactor coolant was routed directly to them. Letdown flow was restored by bypassing the MU filters after the MU filters became blocked.

The TMI-2 MU filters that were in use during the accident have been removed and shipped offsite. A small amount of fuel (estimated to be less than 100 grams) was deposited in them during the accident.

4.1.2.5 Sludge and Resin Removal (Reference 4.10 and 4.11)

Resin removal was primarily performed in the MU Demineralizers, the Cleanup Demineralizers, the Spent Fuel Demineralizer, the Deborating Demineralizers and the Evaporator Condensate Demineralizers. The AB Sump was desludged. The sludge and resin were deposited in the SRST's, dewatered, prepared for shipment, and shipped offsite for disposal. Further, details of the sludge and resin removal are provided in Section 4.1.3.

4.1.3 Auxiliary and Fuel Handling Building Cleanup Activities

4.1.3.1 Seal Injection Valve Room

The SIVR was highly contaminated as a result of the accident. An apparent leak in the seal injection flow instrumentation resulted in the deposition of a very significant amount of crystalline boric acid on the floor of the cubicle. The resulting ambient radiation dose rates and airborne concentration of radioactive materials were very high. A long, complex, and difficult decontamination effort was required to cleanup the SIVR and stabilize it for monitored storage.

The cleanup and decontamination of the SIVR required careful preparations. The presence of a large amount of highly contaminated boric acid posed a potential for the creation of hazardous levels of airborne concentration of radioactive material. Fission products in the water that contained the boron crystals were deposited on and absorbed into the unsealed concrete floor and wall surface as the water evaporated. This required scabbling of the concrete surfaces, recoating, and sealing of the scabbled areas. In preparation for the large scale decontamination activities (e.g., scabbling), accessible penetrations between the SIVR and the remainder of the AFHB were sealed. In addition, a HEPA filtration ventilation unit was installed along with a gasket seal plexiglass access door.

Most of the large-scale decontamination of the SIVR was performed using remotely operated robotic equipment. The boron crystal deposits on the floor were removed and the floor was scabbled. Following scabbling, a layer of concrete was added and the floor was repainted and flushed.

Although the SIVR did have a very significant fission product deposition bound in the crystalline boric acid deposits, it did not contain a significant amount of residual fuel.

4.1.3.2 Reactor Coolant Bleed Tanks 1A, 1B, and 1C

The RCBTs 1A, 1B, and 1C are the tanks to which the reactor coolant is letdown. These three (3) tanks are similar in configuration and size. Each tank holds approximately 80,000 gallons.

During the TMI-2 accident, reactor coolant was letdown directly to the RCBTs. Much of this letdown was unfiltered because of the need to bypass the makeup letdown filters and demineralizers. The letdown of unfiltered reactor coolant resulted in the deposition of a small amount of fuel in the RCBTs. Subsequent to the accident, the RCBTs have been used to receive reactor coolant letdown or other waste water during the cleanup program.

The RCBT cubicles have been extensively decontaminated since the accident. Manual and robotic decontamination efforts have significantly reduced the airborne radionuclide concentrations. The RCBT 1A was flushed internally to remove sedimentation and residual fuel debris but was placed in service subsequently for water processing activities. The B and C RCBTs have not been decontaminated internally because they have been in-service continuously. Additional measurements of those tanks are planned.

4.1.3.3 Makeup and Purification Demineralizers (References 4.12 and 4.13)

The TMI-2 MU&P demineralizers were designed to maintain water purity in the reactor coolant. During the TMI-2 accident the demineralizer resins became heavily loaded with fission products and a small, but measurable, amount of fuel as a result of receiving both filtered and unfiltered reactor coolant. The demineralizers were removed from service on the second day of the accident and were never returned to service.

Post-accident gamma surveys of the demineralizer cubicles detected dose rates in excess of 1000 R/hr. Subsequent radiation measurements and resin sampling were performed utilizing remotely operated and robotic equipment.

During 1984 and 1985, the Cs-137 content of the MU demineralizer resins was reduced when the cesium was eluted from the resins by a sodium borate solution. Following the elution process, preparations were maded to sluice the MU demineralizer resins to the SRSTs. A total of 51 separate resin transfer operations were performed employing a variety of techniques. As a result, the "A" makeup demineralizer resins were almost completely transferred to the spent resin storage tanks. Only 0.006 m^3 of the initial 0.7 m^3 of resin remains. In addition, approximately 75% of the resin was removed from the "B" demineralizer; 0.2 m^3 of resin remains.

The makeup demineralizer resin removal process has resulted in the transfer, solidification, and shipment for waste burial of over 1 kg of residual fuel and nearly 1300 curies of radioactivity.

4.1.3.4 Auxiliary Building Sump (Reference 4.14)

The TMI-2 AB sump was contaminated as a result of the flooding during the accident. In addition, subsequent decontamination of several cubicles resulted in the draining of decontamination water to the sump via the building drains. Analysis of the sludge in the AB sump indicated a small amount of fuel was present. Although direct gamma measurement of the sump did not detect fuel-related radiation, it is likely that a very small quantity of fuel (i.e., approximately 300 grams) was deposited in the sump.

The AB sump was extensively decontaminated, flushed, and desludged. Debris was removed from the sump and the remaining sediment was processed and shipped offsite for disposal.

4.1.4 Auxiliary and Fuel Handling Buildings Fuel Removal Assessment

The decontamination and dose reduction activities in the AFHB were primarily intended to reduce personnel exposure. A secondary objective of the cleanup activities was to place the AFHB in a long-term stable condition. Some fuel was removed from the AFHB as a result of the cleanup activities. The majority of the fuel removed was obtained as a result of the makeup demineralizer resin removal, water processing, system flushing and draining activities, and removal of the various failters and the block orifice.

4.2 Reactor Building Fuel Removal and Decontamination Activities

As discussed in Section 2.0, the RB was contaminated as a result of the TMI-2 accident. A small, but measurable amount of fuel was transported to the RB as a result of: the accident, subsequent plant stabilization, and water processing activities (see Table 2-1). The principal pathway was through the RCDT rupture disk into the RB basement.

4.2.1 Cleanup Approach

Because of the relatively small quantity of fuel, the major RB cleanup activity was directed to dose reduction and structural surface decontamination. A systematic RB cleanup plan was developed to reduce dose rates to the extent that access could be achieved to defuel the RV (References 4.15 and 4.16). The implementation of the RB cleanup plan required extensive resources over eight (8) years to reduce surface and embedded radionuclide contamination and to preclude further recontamination. Since the primary location of residual fuel was in the basement, an extensive effort was made to scarify and desludge the basement. Approximately 40% of the RB basement area was desludged (see Figure 4-1). Additional activities were conducted to remove and displace the solid, particulate contamination from all surfaces above the RB basement (elevations 305' and above). The following presents a summary discussion of the specific cleanup techniques used and locations involved. Also included is an assessment of the effectiveness of these activities in removing fuel from the RB.

4.2.2 Reactor Building Cleanup Equipment and Techniques

The methods utilized for the RB cleanup involved techniques to remove building surface contamination which was predominately radiocesium and strontium with only trace quantities of fuel. These methods included high pressure flushing using lances and robotics, scabbling of floor surfaces by mechanical means, scarification of walls using high-pressure water, sludge and debris removal by sludge pump and robotic equipment, and leaching of the basement block wall using a pump system for recirculation and periodic processing of waste water for contamination removal.

4.2.3 Major Reactor Building Cleanup Activities

4.2.3.1 General Area

During 1981 and 1982 the entire accessible part of the RB surface area above the 305' elevation was hydraulically flushed with processed water. This surface flushing included those areas up to the top of the building dome at the 478' elevation and all major vertical walls and horizontal surfaces. A substantial amount of surface contamination and debris was flushed to the RB basement areas for further processing and removal.

During 1983 and 1984, all major access ways and floor surfaces on the 305' elevation and 347' elevation were scabbled to remove embedded contaminants in the paint and concrete (see Section 4.1.2.1.4 for a description of scabbling). An extensive effort was also made to maintain surfaces clean and preclude recontamination by use of protective coatings and special sealant, epoxy paints. Additional flushing was performed inside of both D-rings in the upper elevations to allow entry for OTSG and pressurizer defueling activities.

4.2.3.2 Reactor Building Basement Scarification and Desludging (Reference 4.17 and 4.18)

During 1986, 1987, and part of 1988, activities in the RB were directed to basement fuel removal and dose reduction. The fission product activity had been absorbed into the concrete while the basement was flooded. In order to reduce the dose rates, it was necessary to remove the concrete surface layer. Scarification of walls in the RB basement was accomplished using a robotic system equipped with a high pressure hydraulic water lance. Accessible basement concrete walls and pillars were scarified using this method. Debris created from this process was allowed to collect on the basement floor to be removed during desludging operations.

After scarification, the robotic unit was retooled with an air-operated sludge pump to remove debris. The sludge and debris were transferred from the floor to a specially designed tank for subsequent transfer to the AB for final processing and disposal. Over 40% of the basement floor surface area was desludged. Some desludging had been conducted prior to scarification to remove a large amount of river water sediment which co-mingled with fuel. The total sludge debris removed from the basement was estimated at approximately 4900 kg of which only a small fraction (<5 kg) was fuel.

4.2.3.3 Reactor Building Basement Block Wall Cleanup Activities (Reference 4.19)

During 1988, cleanup activities in the RB basement were directed at dose reduction of a highly contaminated concrete block wall (see Figure 4-2). The block wall, which surrounded the RB elevator and adjacent stairway, acted as a collection reservoir for radionuclide particulates when the basement was flooded early after the accident. Soluble contaminants and the resulting radiological doses had to be significantly reduced by use of a water leaching process. Leaching was accomplished by drilling holes in several sections of the block wall and recirculating low-pressure water from the RB basement through the block wall. As radioactive concentrations increased in the water, it was pumped from the RB to be processed by the SDS. The cleaned (processed) water was returned to the RB for reuse. The entire activity was conducted remotely by using robotically-mounted drills and handling equipment.

4.?.3.4 Reactor Coolant Drain Tank (Reference 4.20)

In 1983, characterization of the inside surface of the RCDT for fuel removal was undertaken. Access was gained by cutting through the 305' elevation floor and the wall of the RCDT discharge line. Samples of liquid and particulate material were collected from locations directly beneath the rupture disk and vertical section of the rupture line. Visual inspections also were conducted in these regions. Based on both visual and sample analysis, it was concluded that less than 1 kg of residual fuel remained in the RCDT and no further fuel removal activity was deemed necessary.

4.2.4 Reactor Building Fuel Removal Assessment

The overall decontamination and defueling activities in the RB were extensive and resulted in substantial occupational dose reduction for personnel who performed defueling operations in the RCS and RV.

It was estimated that the RB basement scarification and desludging activities removed approximately 4900 kg of sediment. The robotic desludging system desludged approximately 40% of the basement floor area and the removal efficiency of desludging was estimated to be greater than 90%. The major area of residual fuel in the RB basement was determined to be near and adjacent to the RCDT rupture disk discharge line. This area was fully accessible and the majority of the residual material in this area was removed as part of the desludging operations. Post-defueling activities including water removal, decontamination, and system draindown may further reduce the current estimate of residual fuel in the RB.

4.3 Reactor Coolant System Defueling Operations

As a result of the accident, fuel was transported throughout the RCS. Estimates of the fuel quantities in equipment and piping outside the boundary of the RV were determined based on component-specific methods (e.g., sampling, remote visual inspections, and gamma spectroscopy). These methods also were used to identify the potential need to remove fuel from ex-vessel locations. Figure 2-4 shows the configuration of the RCS components.

4.3.1 Reactor Coolant System Defueling Approach

Defueling operations in the RCS were primarily concentrated on the major fuel deposit locations (i.e., Pressurizer, OTSG, and Decay Heat Drop Line). The defueling activities in the RCS resulted in significant removal of fuel and reduction of dose rates. The defueling process of the RCS is described below.

4.3.2 Reactor Coolant System Defueling Equipment and Techniques

Defueling the RCS required the use of water cleanup systems, defueling tools similar to those used in the RV, and robotic equipment. The detailed description of the major equipment and tasks used to defuel the RCS is presented in the following sections.

4.3.3 Reactor Coolant System Defueling Activities

4.3.3.1 Pressurizer Defueling Operations (Reference 4.21)

The Pressurizer was defueled using a submersible pump, a knockout canister, a filter canister, and an agitation nozzle. The fuel fines and debris were first suspended in the pressurizer water by pumping processed water from the DWCS through an agitation nozzle. The Pressurizer water was then pumped through a knockout canister and a filter canister to remove most of the suspended fuel fines and debris. The water was then returned to the RV through the existing DWCS.

After the initial effort was completed, visual inspections indicated that large pieces of debris (up to 5 cm wide by 10 cm long by 2.5 cm thick) remained on the bottom of the Pressurizer. These pieces had been buried by the loose debris and were not previously visible. A remotely-operated submersible vehicle equipped with an articulating claw and a scoop was used to remove these larger pieces of debris.

4.3.3.2 Pressurizer Spray Line

Debris in the Pressurizer Spray Line was flushed back into the Pressurizer and RCS cold leg 2A using water processed through the DWCS. Although the effort did not result in removing fuel from the primary system, it did relocate the debris for removal in subsequent defueling operations.

4.3.3.3 Once-Through Steam Generators and Hot Legs (References 4.22 and 4.23)

Pick-and-place and vacuuming techniques were used to defuel the "A" and "B" OTSG upper tube sheets. Long-handled gripping tools were used to lift large pieces of debris into canisters and a vacuum system removed the smaller debris. While this process essentially succeeded in defueling the "A" OTSG tubesheet, a crust of tightly adherent debris remained on the surface of the "B" OTSG tubesheet. It has been concluded that no further defueling of the "B" OTSG tubesheet is necessary or appropriate because of the small amount of remaining fuel.

The OTSG tubes were surveyed to detect blockages and adherent fuel-bearing films. GM counters and alpha detectors were used. The lower head of the OTSGs and the J-Legs were surveyed using GM counters and activation foils. No further defueling efforts are planned.

The hot legs were defueled using a combination scraper/vacuuming tool and the Westinghouse vacuum system. Additional defueling activities are planned in the "B" hot leg as part of RV defueling (Section 4.4). The debris will be transferred to the RV for removal.

4.3.3.4 Decay Heat Drop Line (Reference 4.23)

The in-vessel vacuum system was used to defuel the Decay Heat Drop Line. A deployment tool was developed to guide the vacuum hose into the Decay Heat Drop Line from the RCS "B" hot leg. All loose debris in the vertical portion of the Decay Heat Drop Line was vacuumed. Below the vacuumable loose debris, a hard compacted region of debris was encountered. A drain cleaning machine was used to penetrate this hard debris and size it so vacuuming could continue. The material was airlifted into the "B" hot leg and will be removed, as described in the above section, as part of the RV defueling.

4.3.4 RCS Fuel Removal Assessment

(To Be Published Later)

4.4 Reactor Vessel

The TMI-2 RV core region, LCSA, lower head region, and UCSA presented a unique defueling challenge. As a result of the accident, the fuel forms, locations, and accessibility for removal in each region varied greatly. The core region consisted of an upper debris bed, a resolidified mass, and partially intact assemblies. The LCSA consisted of the original series of five (5) plates, with core debris scattered throughout. The lower head region consisted of hard and loose debris beds. The UCSA consisted of essentially intact baffle plates with core debris trapped between them and the core barrel.

4.4.1 Reactor Vessel Defueling Approach (References 4.24 and 4.25)

The activities associated with the defueling of the TMI-2 RV were primarily the removal of core material from the RV, encapsulation of these materials within specially-designed canisters, and placement of the canisters into the storage racks located in Spent Fuel Pool "A". These canister were subsequently shipped to INEL for analysis and storage. The defueling process has been divided into five (5) major activities as described in Section 4.4.3

4.4.2 Reactor Vessel Defueling Equipment and Techniques

Defueling the RV presented a unique and challenging environment; special tools and equipment were developed specifically for defueling the RV. These included long-handled pick-and-place tools, the core boring machine, the plasma arc torch, and many other specially-designed tools which were vital to the completion of the defueling activities. Further details are presented in the following sections and are also provided in the NRC-approved Defueling Safety Evaluation Report (Reference 4.26).

4.4.3 Reactor Vessel Defueling Activities

4.4.3.1 Initial Defueling Activities

Initial in-vessel defueling operations began in October 1985 and consisted of removal of fuel element endfittings and other loose debris, including vacuumable "fines", from the rubble bed. The first step was to use manual, long-handled tools to rearrange core debris that interfered with completing the installation of the defueling equipment (e.g., fuel canister positioning system). Loose debris pieces were then picked-and-placed into fuel canisters. Additional core debris was broken into smaller pieces (i.e., sized) for future canister loading. Final preparations to operate the fines/debris vacuum system were completed. The first canisters of core debris were transferred from the RB to the "A" spent fuel pool in January 1986. Figure 4-3 shows the TMI-2 defueling progress from January 1986.

4.4.3.2 Core Region Defueling

Core region defueling consisted of removal of debris from the core region of the RV which remained after the completion of initial defueling. This phase differed from initial defueling in that significant sizing operations were performed (e.g., separating and cutting of fused fuel assemblies and other large pieces of core debris). Removal of the "hard crust" was also accomplished during this phase. Some activities performed during core region defueling were similar to those performed during initial defueling (e.g., pick and place).

This phase was initiated in the summer of 1986 when defueling shifted from pick-and-place operations to preparing the debris bed for a core sample acquisition program using the CBM. This operation went smoothly after resolution of initial difficulties with indexing the drill to target location. After sample drilling was completed and the CBM was removed, defueling resumed. The core region proved to be much harder to defuel than anticipated. Efforts to break up and remove the debris with long-handled tools were unsuccessful.

4.4.3.2.1 Use of the Core Boring Machine (Reference 4.27)

In September 1986, the CBM was re-installed to rubblize the large resolidified mass using a solid-faced drill bit (see Figure 4-4). Loose upper endfittings that would interfere with drilling operations were removed from the surface of the debris bed. Because several endfittings had been fused together and were too large to be inserted into fuel canisters, they were placed in shielded drums filled with borated water and stored at elevation 347' in the RB. In late October and early November 1986, the CBM was used to drill a total of 409 closely spaced holes in the resolidified material at the center of the core debris bed to break up the hard mass and facilitate its removal.

4.4.3.2.2 Core Topography and Drill String Removal

In late November 1986, core topography and video surveys were performed. The results indicated that the core drilling operations performed in October and November were not completely successful in breaking the resolidified material into easily removable pieces. In addition, a number of rocks exceeding 0.3 meters in diameter were identified which were believed to have fallen from the peripheral region surrounding the drilled area. This peripheral region consisted of undrilled, resolidified material and standing fuel assembly elements. Finally, several broken drill strings were located on or embedded in the drilled surface of the debris bed and required removal.

The majority of the drill strings were removed from the core debris bed and loaded into canisters. Additional attempts were made to resize the larger rocks of agglomerated material and to load the loose debris that was created. Given limited visibility, the crust impact (manual, long-handled) tool could not be used efficiently to break up the rocks. Although much of the smaller,loose debris proved difficult to remove, some areas of accessible rubble were defueled.

Defueling operations in 1987 began with removal of loose debris from the RV. A funnel, designed to withstand the impact of an air-operated chisel, was used to overcome the problem of sizing and loading debris that was too large to fit into canisters. Debris pieces were lifted into the funnel, which was suspended above the canister, and the chisel was used to break the rocks into pieces small enough to fall into the canister.

4.4.3.2.3 Stub Assembly Removal

After this operation, the focus of defueling shifted to removal of stub assemblies. The upper 40 to 70 inches of the peripheral assemblies were removed using a variety of cutting, snaring, and clamping tools. Portions of fuel assemblies removed in this manner were loaded into fuel canisters. In March 1987, assembly A-6 was removed essentially intact. That provided the first opening to the lower grid. Subsequently, fuel assemblies A-7 and B-6 were successfully removed marking the start of several months of stub assembly removal.

A fuel assembly puller was designed and fabricated to engage stub assemblies below the lower endfitting. Once engaged, the tool loosened and raised the assembly and allowed another tool to grasp it. The grasping tool was then used to load the assembly into the fuel canister. Although this technique worked, it was cumbersome and several fuel assemblies were dropped onto the debris bed during transfer.

A modified fuel assembly puller was introduced in August 1987. This tool had a long spike which engaged the stub assembly and reduced the number of assemblies dropped once extracted from the grid. Another fuel assembly handling tool was designed to grasp the raised assembly from the side and deposit it into a fuel canister. As a result of these tool improvements, productivity increased significantly.

The resolidified mass of core debris outboard of the central core region was broken up as stub assemblies were raised from beneath it. Portions of this debris were loaded along with the stubs. Some debris and loose rods fell onto the lower grid and into the lower internals. Stub assembly R-6 was not removed because it was severely damaged and was fused to the surrounding structure when resolidified.

4.4.3.3 Lower Core Support Assembly Disassembly and Defueling

The LCSA (see Figure 4-5) consists of a series of five plates: the lower grid rib section, the lower grid distributor plate, the lower grid forging, the incore guide support plate, and the flow distributor plate. Removal of these plates was necessary to access the core debris in the reactor lower head region. The disassembly of the LCSA began in January of 1988.

4.4.3.3.1 Introduction

Early observations reinforced the viability of a plan to use the plasma arc torch to cut RV structural material. However, inspections of the LCSA during 1987 revealed additional core debris and a significant number of broken fuel rods trapped between the LCSA plates. Cutting of a much larger hole in the LCSA than planned would be necessary to gain access to the additional debris. The larger hole would require a minimum of 2000 cuts with the plasma arc torch; such an undertaking would stress the reliabilty of the equipment. Therefore, in 1988, a new concept was developed which used both the plasma arc torch (linear cutting) and the core bore machine (circular cutting).

4.4.3.3.2 Initial Lower Core Support Assembly Drilling Operations

RV defueling operations were suspended to prepare for the LCSA defueling and disassembly operations. Following the installation of three drill guide plates, which provided the drill string lateral stability and alignment into the RV lower grid, workers installed the CBM on the shielded work platform. Drilling operations began in January 1988.

The first phase of the LCSA drilling operations involved drilling through all 52 incore instrument guide tube (IIGT) spider castings, which anchor the top of the IIGT to the center of the guidecell. This was the first step in freeing the IIGTs from the RV lower grid, thus permitting their removal from the LCSA.

The first pass drilling was successfully completed in January 1988 with no significant problems. Second-pass drilling of the 15 peripheral, non-gusseted incore IIGT positions was begun in February 1988. Defuelers completed drilling 14 of these positions down through the lower grid distributor plate to the top of the grid forging. Interference from the remains of the R-6 fuel element prevented access to the incore drilling target at R-7, the only remaining ungusseted IIGT. Following the installation of a specially-fabricated drilling guide and a flat-faced junkmill drill bit, enough of the mass was removed to provide access for drilling (at least through the distributor plate).

4.4.3.3.3 Lower Grid Rib Section Removal

The lower grid support post removal phase began in early March 1988. Forty-eight support posts were drilled through the grid rib section, the grid pad, and the lower grid flow distributor plate. Given the successful demonstration of the ability of the drill rig to cut, it was decided to use the CBM to finish severing the lower grid rib section.

A total of 16 ligament cuts were completed. These cuts, in conjunction with drilling the support posts, produced 13 severed pieces of lower grid which were removed and stored underwater inside core flood tank "A". (It was necessary to cut off the top of the tank to receive the grid pieces.) Before removal, the pieces were flushed and inspected for visible fuel and gamma-scanned to determine the quantity of adherent fuel.

In May 1988, workers completed the installation and checkout of the plasma arc torch and associated support equipment. The plasma arc torch used a high-velocity stream of high-temperature ionized nitrogen gas (i.e., plasma) to cut the LCSA plates into sections. To position the plasma arc torch, the ACES employed a robotic manipulator arm attached to a computer-controlled bridge and trolley system suspended over the LCSA.

The plasma arc torch equipment was used first to make trim cuts as a follow-on to the CBM defueling on the lower grid rib section periphery. Remnant trim cutting was completed in June 1988. A total of 72 remnant pieces were severed and removed.

4.4.3.3.4 Lower Grid Distributor Plate Removal

The lower grid distributor plate was cleared of loose debris using pick-and place tooling. Loose debris was loaded into fuel canisters using long-handled tools.

In preparation for cutting the lower grid distributor plate, the cutting equipment was removed and the trimmed pieces that could potentially interfere with lower grid distributor plate cutting were cleared. The remaining incore instrument strings also were trimmed down to the instrument guide tube nozzle.

Following re-installation of the plasma arc torch and support equipment, cutting of the one inch thick lower grid distributor plate began. In sectioning the lower grid distributor plate, a cutting pattern was used that took advantage of previous cuts made by the CBM; the result was four (4) roughly pie-shaped pieces.

Two of the planned severance cuts could not be completed on the first quadrant due to the presence of previously molten debris near the bottom of the lower grid distributor plate.

Consequently, an irregular cut was made around the interfering area. Additional supports were installed in the LCSA to control sagging as the plate was cut. Approximately 85 cuts of various lengths were required to sever the lower grid distributor plate at its periphery. The pieces were flushed, brushed (in an effort to minimize the transfer of adherent core debris), removed from the RV, and transferred to Core Flood Tank "A" for storage.

4.4.3.3.5 Lower Grid Forging Removal

The lower grid forging was the third LCSA components to be disassembled. The grid forging holes required for plasma arc equipment access had to be cleaned of debris. Fifteen holes were found to contain potentially interfering loose debris and fuel rod stubs. Consequently, using long-handled, hydraulically-operated vise grip pliers, the fuel rod segments were removed and placed temporarily inside a debris dumpster for eventual loading into fuel canisters. Other loose debris from the top of the grid forging and the flow holes as well as a region several centimeters below the bottom of the grid forging was removed using the airlift equipment. Airlifted material was loaded into top-loading, bottom-dumping debris buckets for subsequent transfer to fuel canisters. Completion of this pre-conditioning activity helped maintain plasma arc stability and minimized undesirable fusion of core debris during plasma arc cutting.

By the end of October 1988, all 34 of the IIGTs identified for removal had been cut with the plasma arc torch, removed, and loaded into fuel canisters. All 28 of the support posts identified for removal were also cut and loaded into fuel canisters. Plasma arc cutting of the lower grid forging was completed in November 1988. A total of 71 required forging severance cuts were made. A large center section of this plate was severed into four (4) roughly pie-shaped pieces. A hydraulically-operated brushing tool was applied to the top surfaces of the lower grid forging plate sections and each section was cycled through a special spray system designed to reduce contamination levels. These sectioned pieces were then removed from the RV and placed inside Core Flood Tank "A".

4.4.3.3.6 Incore Guide Support Plate Removal

By the end of November 1988, preparations had begun to disassemble the incore guide support plate. A few loose fuel rod segments located on the plate were removed. Additionally, loose debris up to 13 inches below the incore guide support plate was cleared.

In preparation for cutting the incore guide support plate, a hydraulically-driven rotary brush was used to clean the plate. The plasma arc torch and supporting equipment was re-installed and cutting of a large center section from the incore guide support plate was begun. By the end of December, 1988, the plate was sectioned into four, roughly pie-shaped pieces. All 25 cuts, including recuts required to section this plate, were cleaned and verified.

In early January 1989, the cut quadrants of the incore guide tube support plate were lifted from the LCSA, flushed, and transferred to Core Flood Tank "A" for storage.

4.4.3.3.7 Flow Distributor Plate Removal

Following completion of the incore guide support plate removal, loose debris and small pieces of fuel rods were vacuumed from above and below the flow distributor plate. Long-handled tools were used to pick-and-place larger pieces of debris, much of which had originated in the core region and had accumulated on the flow distributor plate as the result of defueling operations.

In late February 1989, the cutting of the flow distributor began. The plasma arc torch made 104 cuts, with numerous recuttings needed to ensure severance. The flow distributor was cut into 26 pieces. By the end of March, the cutting was complete. The sections of the flow distributor plate that did not contain incore guide tubes attached were removed from the RV and placed inside Core Flood Tank "A". The sections of the plate that contained incore guide tubes were bagged and stored inside the "A" D-Ring.

4.4.3.4 Lower Head

(To Be Published Later)

4.4.3.5 Upper Core Support Assembly Defueling

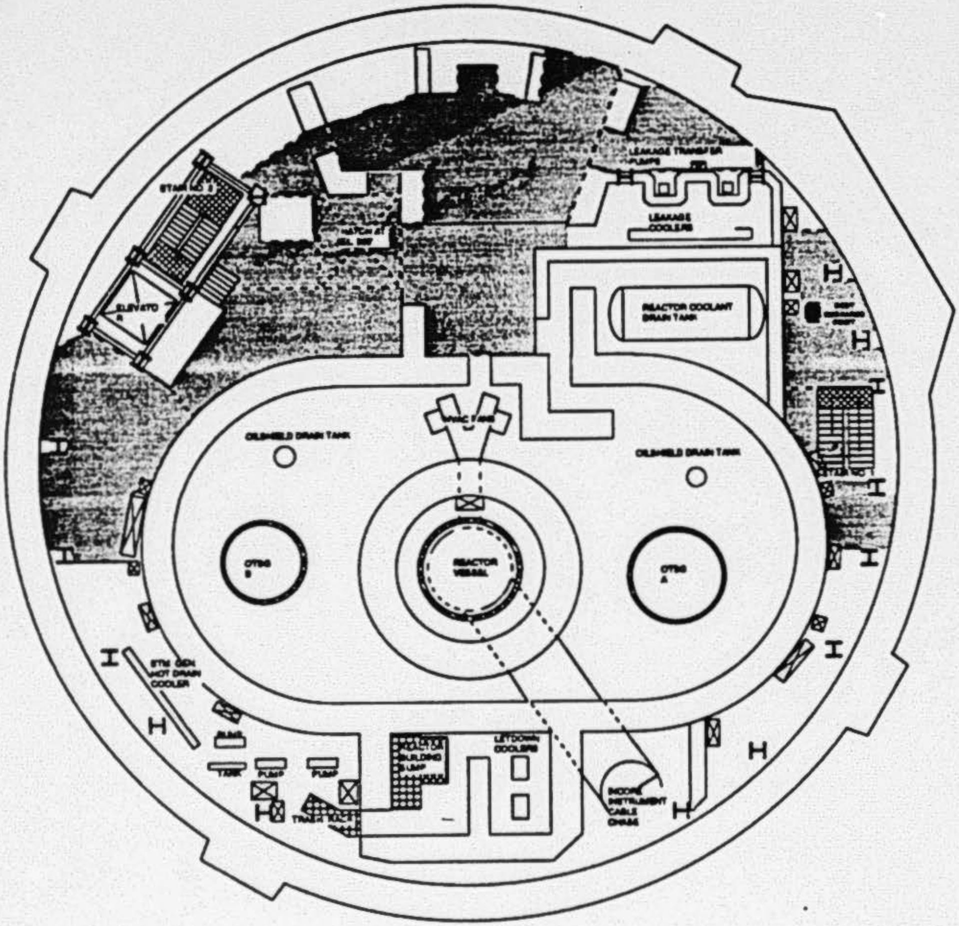
(To Be Published Later)

4.4.4 Reactor Vessel Fuel Removal Assessment

(To Be Published Later)

FIGURE 4-1

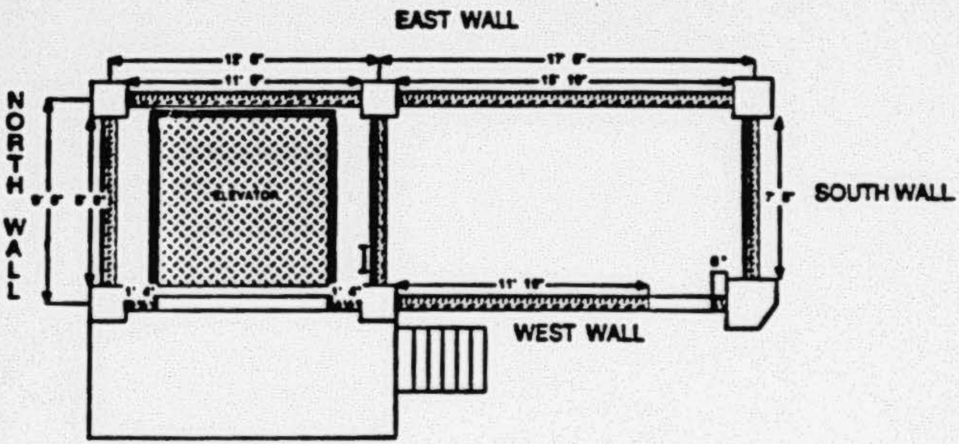
REACTOR BUILDING BASEMENT FLOOR PLAN (DESLUDGED)



NOTE: Shaded area represents desludged portion of basement.

FIGURE 4-2

FACE IDENTIFICATION

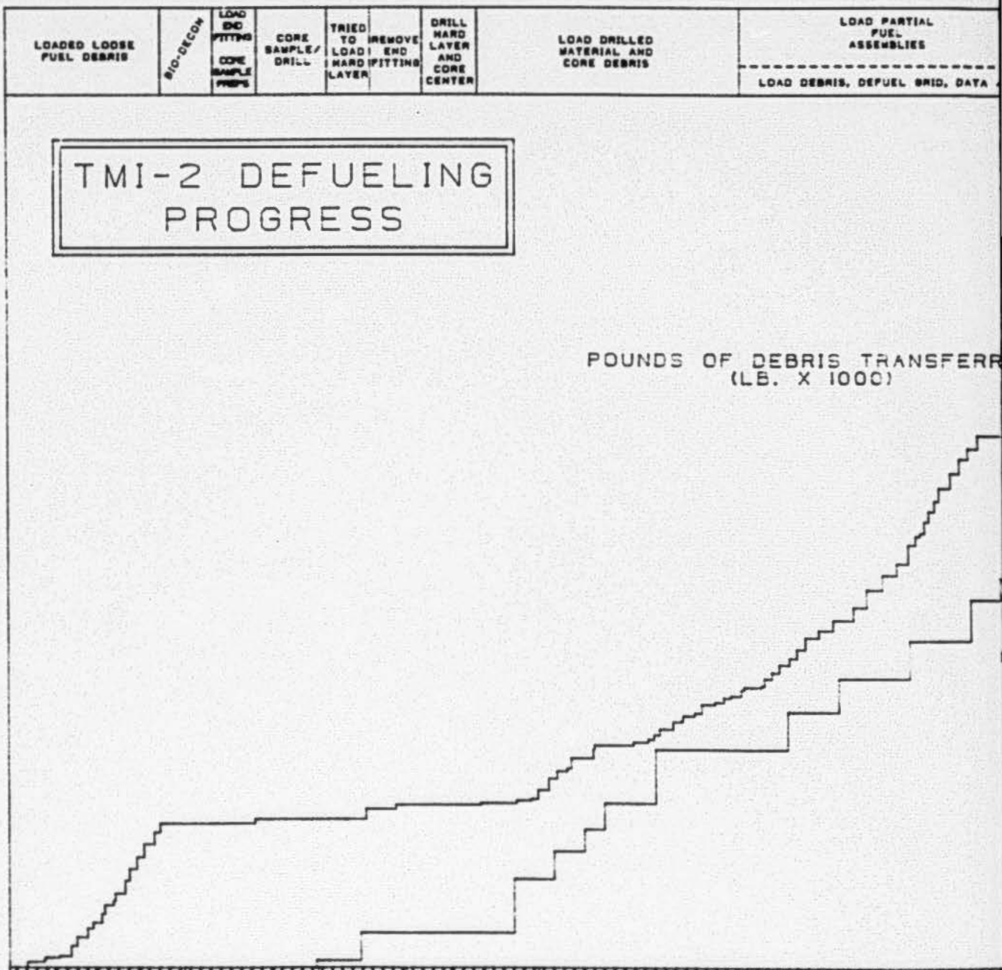


SUMMARY:

Leaching of the 282'-6" elevation block wall that encloses the stairwell and elevator shaft reduced the radionuclide content by 33% in the areas treated. The leaching resulted in a total removal of 1200 curies of Cesium-137, which represents a removal of 7.1% of the total block wall radioactivity. This estimate is supported by both the exposure rate data taken on the block wall and by water sample analysis.

FIGURE 4-3

TMI-2 DEFUELING PRO



GRESS

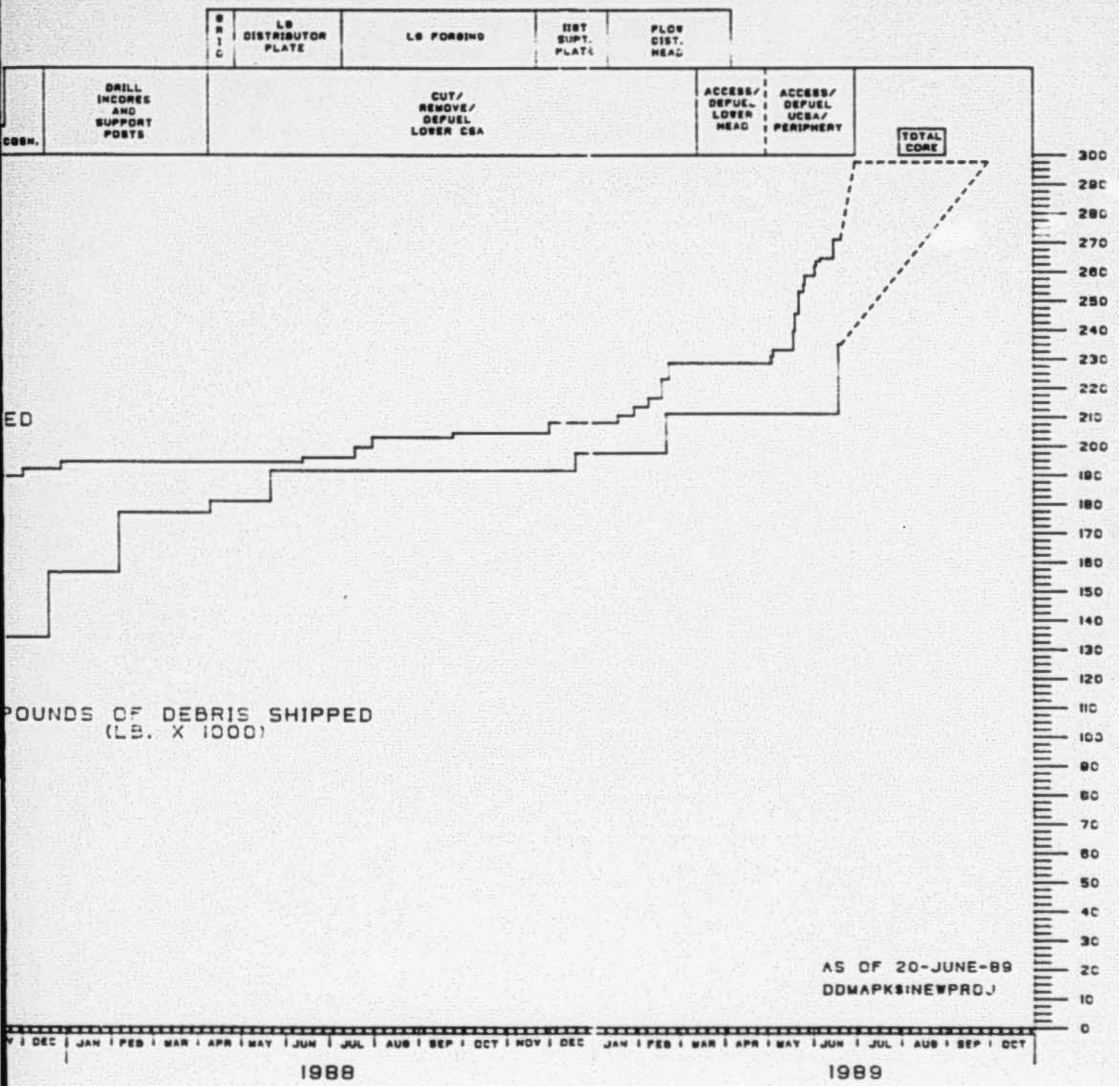


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FIGURE 4-4

CORE BORE MACHINE

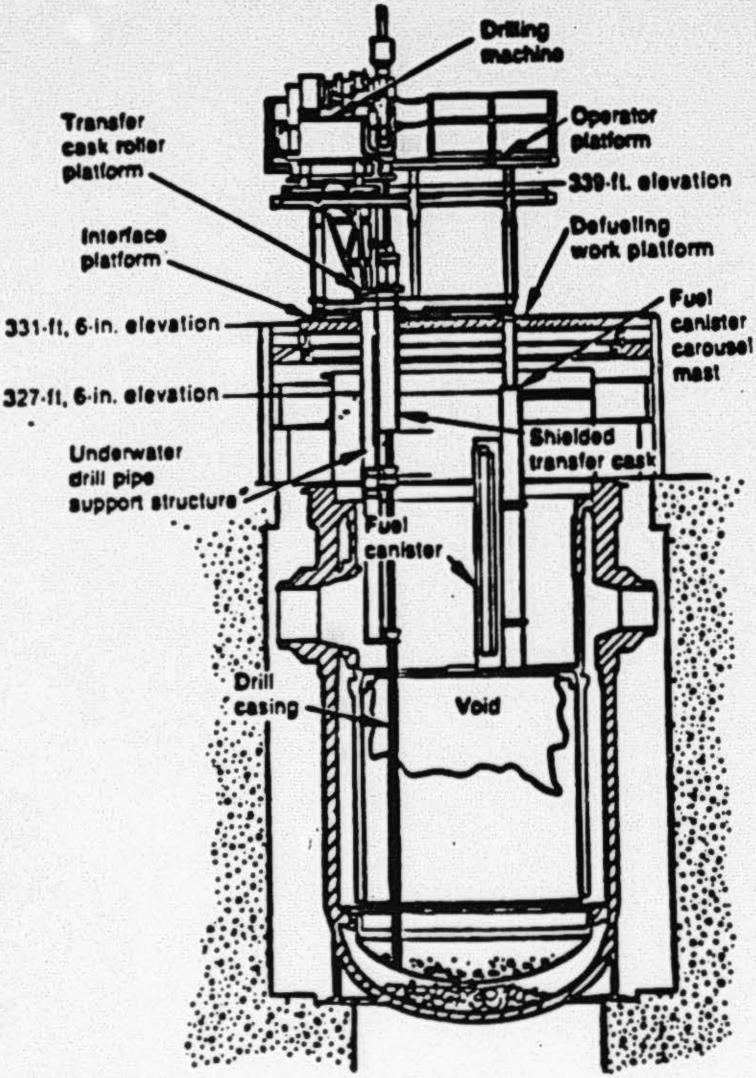
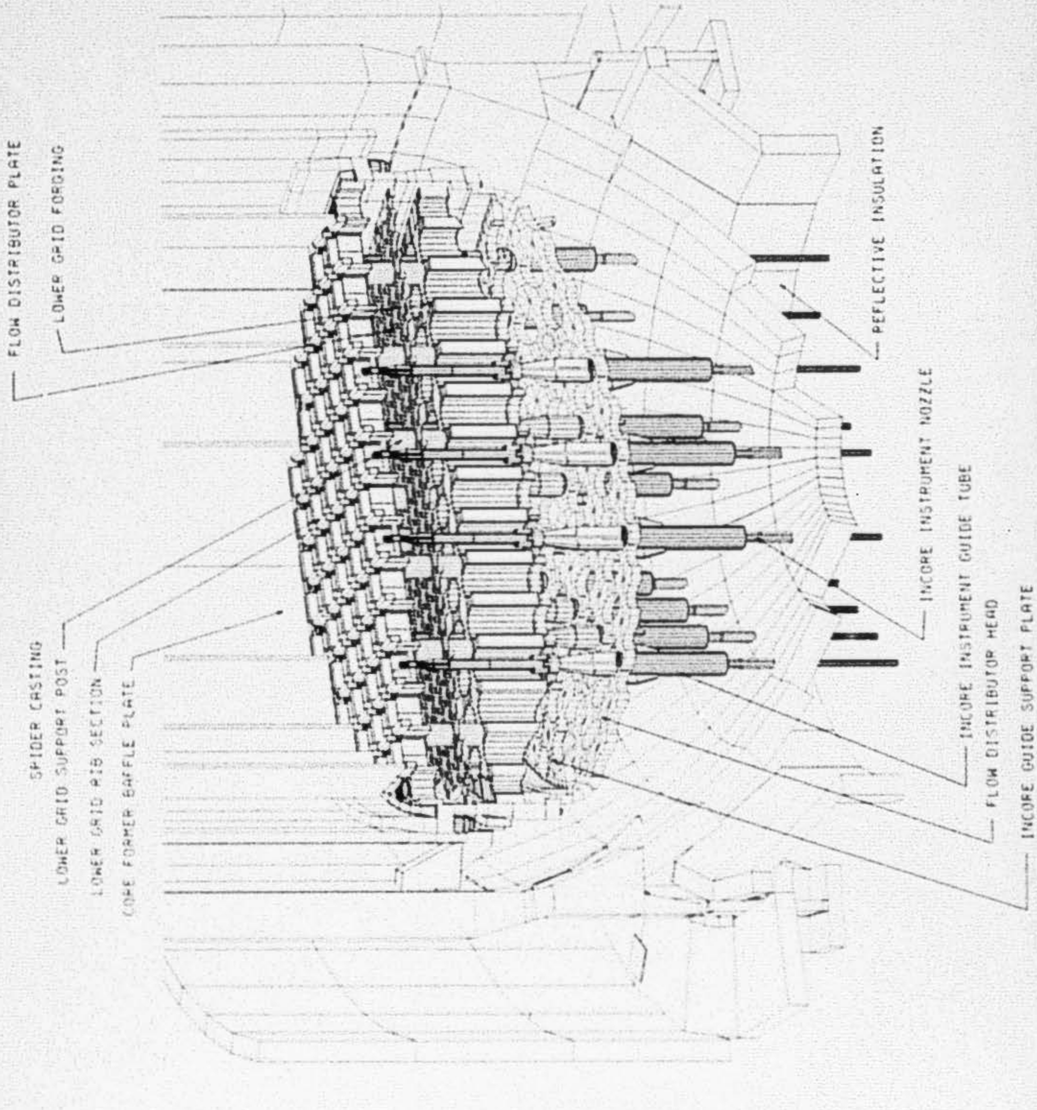


FIGURE 4-5

LOWER CORE SUPPORT ASSEMBLY

LOWER CORE SUPPORT ASSEMBLY



5.0 RESIDUAL FUEL QUANTIFICATION AND CRITICALITY ASSESSMENT

This section provides a characterization of residual fuel by quantity and location within TMI-2. To facilitate discussion, this section is subdivided to address the AFHB, RB, RCS, and RV.

The criticality assessment for those ex-vessel locations within Unit 2 that are demonstrated to have residual fuel quantities significantly below the SFML are not re-evaluated in this document, except to demonstrate the lack of a credible means for fuel material to be relocated. The SFML was developed with consideration for optimum moderation and infinite water reflector (worst case) conditions. These moderator and reflector considerations bound expected conditions within the AFHB, RB and RCS. For those locations and components (i.e., essentially in-vessel) which contain residual fuel quantities greater than the SFML, a more detailed criticality assessment and analysis will be provided.

5.1 Auxiliary and Fuel Handling Buildings

During the accident, residual fuel was transported to AFHB as a result of the core degradation event and the concurrent RCS MU&P System operation. Section 2.0 reported that approximately 25 kg of fuel material was transported to the AFHB during the accident sequence. Section 4.1.1 indicated that up to 15 kg of fuel may have been relocated into the AFHB as part of water processing and defueling operations (i.e., potentially a total of 40 kg). Based on these estimates, it could be concluded that AFHB residual fuel conditions were maintained significantly below the SFML during the accident and subsequent cleanup period. Nonetheless, a significant cleanup and decontamination effort was undertaken (as described in Section 4.1) to reduce dose rates and remove fuel where practical. These efforts have further reduced the remaining residual fuel content in the AFHB.

The following sections provide the current estimates of residual fuel within the AFHB. These estimates are based on extensive evaluations of the plant systems and building configurations, fuel measurement within various system pathway sources and tank locations, and a systems analysis approach for bounding fuel quantities in groupings of cubicles and/or system boundaries. The basis for each approach is provided within each Section.

5.1.1 Auxiliary and Fuel Handling Buildings Cubicles

All of the cubicles in the AFHB (see Figures 5-1 through 5-4) were reviewed to determine if fuel could have been transported into the cubicle and/or the piping and tanks located in the cubicle as a result of the TMI-2 accident and subsequent defueling or decontamination activities. It was concluded that the AFHB areas/cubicles listed in Table 5-1 contain no residual fuel.

It was also concluded that the AFHB cubicles listed in Table 5-2 potentially contain residual fuel. The results of actual fuel measurements are listed in Table 5-2. The rationale for inferring the fuel content in those areas/cubicles where fuel measurements were not performed (i.e., respective bounding fuel estimate) is presented in the following discussion.

5.1.2 Discussion of Areas Containing Fuel Debris in the Auxiliary and Fuel Handling Buildings

The following sections address those cubicles where fuel measurements were not performed and provides the basis for establishing boundary estimates for residual fuel.

5.1.2.1 Cubicle AX004 - Seal Injection Valve Room

The SIVR contains piping which is part of the MU&P and RCP Seal Injection Systems. In addition, The Seal Injection system piping that runs through the cubicle is the piping pathway from the RCP seals to and from the Seal Injection Filters.

Small amounts of residual fuel may have been relocated to the SIVR as a result of the normal RCS seal injection and makeup flow during the TMI-2 accident. In addition, small amounts of residual fuel may have been relocated into the MU&P HPI piping in the cubicle as a result of RCS makeup after the accident while the RCS was in the pressurized and level control mode.

The SIVR was highly contaminated as a result of a seal injection piping leak during the accident. However, it is unlikely that any significant amounts of residual fuel currently reside in the SIVR. Measurement of the MU&P System piping immediately upstream of the cubicle detected approximately 130 grams total residual fuel in the three Makeup Pump cubicles. Measurement of the Seal Injection Filters in 1987 detected less than 50 grams in the filters.

A bounding estimate of the residual fuel in the SIVR has been made by summing the measured residual fuel quantities in the three makeup pump cubicles. The bounding estimate of the residual fuel quantity in the SIVR is 0.13 kg. This estimate is believed to be conservative because portions of the Seal Injection piping have been flushed and drained, and because the Seal Injection filters had very little fuel loading (Reference 5.1).

5.1.2.2 Cubicle AX008 - Spent Resin Storage Tank 1B
Cubicle AX009 - Spent Resin Storage Tank 1A
Cubicle AX010 - Spent Resin Storage Tank Pump
Cubicle AX014 - Reactor Coolant Evaporator
Cubicle AX015a - Cleanup Filters
Cubicle AX015b - Cleanup Filters
Cubicle AX016 - Cleanup Demineralizer 2A
Cubicle AX017 - Cleanup Demineralizer 2B
Cubicle AX114 - Makeup and Purification Demineralizer 1A
Cubicle AX115 - Makeup and Purification Demineralizer 1B
Cubicle AX119 - Spent Fuel Demineralizer
Cubicle AX129 - Deborating Demineralizer 1B
Cubicle AX130 - Deborating Demineralizer 1A
Cubicle FH001 - MU Suction Valves

The cubicles listed above contain piping and/or tanks that are part of the resin transfer system. This system has been and will continue to be used to remove the highly radioactive resin from the Makeup Demineralizer and the Cleanup Demineralizer ion exchangers. Those resins, in place during the accident, became contaminated with fuel debris.

The resin transfer system has been used to remove all of the resins from the cleanup demineralizers, essentially all of the resins from the "A" makeup demineralizer, and most of the resins from the "B" makeup demineralizer. Additionally, some resins have been removed from the Spent Fuel Demineralizers and the Deborating Demineralizers.

Final residual fuel measurements will not be performed until resin transfer operations are concluded. Nonetheless, a bounding estimate of the maximum residual fuel content of the cubicles can be made based upon the measured fuel content of the cubicles prior to the initiation of resin transfer activities.

Seven (7) of 14 cubicles (i.e., AX008, AX009, AX010, AX014, AX119, AX129, and AX130) were not contaminated with fuel as a result of the TMI-2 accident. The systems' piping and tanks in these cubicles were not in the makeup, letdown, or waste disposal liquid flowpaths at that time. Therefore, these seven (7) cubicles do not contribute to the bounding estimate of total residual fuel in the 14 cubicles; however, as a result of the resin transfer operations they may contain residual fuel.

Seven (7) cubicles, listed below, had fuel deposited in the piping and/or tanks as a result of the accident. The fuel content of these cubicles was measured prior to the system flush and resin transfer activities (Reference 5.2 and 5.3).

Cubicle	Equipment/Name	Fuel Measurement	Fuel Content
AX015a	Cleanup Filters	Yes	5 grams
AX015b	Cleanup Filters	Yes	5 grams
AX016	Cleanup Demineralizer	Yes	160 grams
AX017	Cleanup Demineralizer	Yes	160 grams
AX114	MU Demineralizer	Yes	1500 grams
AX115	MU Demineralizer	Yes	1500 grams
FH001	MU Valve Room	Yes	260 grams
		TOTAL=	3590 grams

This estimate is a total of the fuel measured in the seven (7) cubicles before decontamination and resin transfer. Following these measurements, the cleanup filters, cleanup demineralizers, and 85% of the combined total of the A and B MU demineralizers resin were removed. Therefore, a reasonable estimate of the residual fuel content is 710 grams. For bounding purposes, 800 grams is used in Table 5-2.

5.1.2.3 Cubicle AX021 - Reactor Coolant Bleed Tank 1A

The RCBT 1A cubicle contains one of the three (3) 80,000 gallon tanks that are used as a reservoir for reactor coolant. RCBT 1A was drained and decontaminated after the TMI-2 accident but has subsequently been returned to service as a drain tank for reactor coolant. Preliminary results of a survey of RCBT 1A indicate that the residual fuel in RCBT 1A is approximately 1 kg. This estimate is currently being verified and will be updated, if necessary, as part of the final DCR submittal.

5.1.2.4 Cubicle AX102 - Reactor Building Sump Pump Filter Room

The RB sump pump filters (WDL-F-8A, 8B), filter housings, and associated piping are located in the AX102 cubicle. The RB sump filters were used during the TMI-2 accident to filter the water from the flooded RB basement as it was pumped to the Auxiliary Building. Post-accident sampling of the sludge in the RB basement found it contained a small quantity of fuel. Therefore, some fuel may have been transferred from the RB basement and deposited in AX102 during the accident as a result of the water transfer.

Since the TMI-2 accident, there has been no transfer of water from the RB to the Auxiliary Building sump via the RB sump filters. The RB sump filters that were installed during the accident were removed during 1980 and disposed as radioactive waste. Subsequent to the accident, the RB sump filters have been used routinely to filter water transferred from the Auxiliary Building sump to the MWHT. During the time from 1980 to the present, there have been over 30 filter changeouts of the RB sump filters.

The residual fuel content of AX102 has not been measured because the system is still in use. The residual fuel content will be measured after the defueling program is completed.

A bounding estimate of the residual fuel content of AX102 is 300 grams. This estimate is conservative since any fuel deposited in the RB sump filters and piping as a result of the accident is believed to have been flushed into the filters and removed as part of the multiple (over 30) filter changeouts or by being flushed to the MWHT. The major use of the RB sump filters during the post-accident period has been to filter water transferred from the AB sump to the MWHT. Thus, a small quantity of fuel could have been transferred from the AB sump to the sump filters or associated piping. Therefore, the total measured content of the AB sump, less than 300 grams, was selected as the bounding estimate for the current residual fuel inventory for AX102.

5.1.2.5 Cubicle AX131 - Miscellaneous Waste Holdup Tank
Cubicle AX134 - Miscellaneous Waste Tank Pumps
Cubicle AX124 - Concentrated Liquid Waste Pumps
Cubicle AX218 - Concentrated Waste Storage Tank
Cubicle FH008 - Neutralizer Tank Pumps
Cubicle FH009 - Neutralizer Tanks
Cubicle FH012 - Neutralizer Tank Filters

All of the cubicles listed above have been identified as potential locations of small quantities of residual fuel because either filtered reactor coolant and/or surface decontamination waste water has been stored in or pumped through each cubicle. The residual fuel content has not yet been measured in these cubicles because the tanks, piping, and/or filters in each cubicle are still in service.

These cubicles have been grouped together as a single section in the DCR because, for the most part, they have been primarily used to hold and transfer surface decontamination waste water and the bounding estimate for the residual fuel content in each cubicle has been developed based upon a single logical approach.

Cubicles AX131 and AX134 are located in the AB and they contain the MWHT (AX131), the Miscellaneous Waste Tank pump (AX134), and associated piping. The MWHT System has been used since the TMI-2 accident as a holding tank for water effluent from the SDS off-gas separator tank, water generated during the dewatering of SDS filters and ion exchangers, and waste water from system flush and surface decontamination activities.

Cubicles AX124 and AX218 are located in the AB and they contain the Concentrated Waste Storage Tank (CWST) (AX124), the Concentrated Liquid Waste Pumps (AX218), and associated piping. The CWST has been used since the accident as a holding tank for decontamination waste water.

Cubicles FH008, FH009, and FH012 contain the Neutralizer Tank Pumps (FH008), the Neutralizer Tanks (FH009), and the Neutralizer Tank Filters (FH012). The Neutralizer System has been used as a batch tank which received the effluent from the MWHT and fed it into the EPICOR II System for filtration and purification via ion exchangers. Although originally intended to be used to chemically treat waste liquid, the Neutralizer System has not been used in that manner since the TMI-2 accident.

A bounding estimate of the residual fuel content for each of the cubicles associated with the MWHT, the CWST, and the Neutralizer System has been developed based upon a comparison of each system tank volume with the AB sump and the extrapolation of the fuel characterization measurement of the AB sump to each system. This approach for developing the bounding estimate is believed to be conservative because the MWHT, the CWST, and the Neutralizer Tank all received and held surface and system flush decontamination liquids for a substantial portion of the cleanup period. These same liquids were also held and stored in the AB sump for a substantial portion of the cleanup period. Basing the estimate on a comparison of tank volumes is believed to be adequate because fuel characterization measurements of the residual fuel in the makeup system found the preponderance of the fuel deposited in tanks as compared to piping. This is due to the conditions in the larger tanks which are much more conducive to settling of suspended fuel as compared to those conditions in the system piping. Tanks have relatively low effluent flow rates and considerably more residence time for liquid contents than piping.

The bounding estimate of the residual fuel content in the two (2) MWHT cubicles is 1 kg of fuel. This estimate was developed by comparing the volume of the MWHT (approximately 19,000 gallons) to the AB sump (approximately 7600 gallons) (the MWHT holds approximately three (3) times the volume of the AB sump), multiplying the maximum measured fuel content in the AB sump (less than 300 grams) by a factor of 3 and then rounding up to 1 kg.

The bounding estimate of the residual fuel content of the CWST cubicles is 0.5 kg of fuel. This estimate was developed by comparing the holding volume of the CWST (approximately 9600 gallons) to that of the AB sump (approximately 7600 gallons). The CWST holds about 1.3 times the volume of the AB sump. The maximum measured fuel content of the AB sump (less than 300 grams) was then multiplied by 1.3 and then rounded to 0.5 kg for conservative purposes.

The bounding estimate of the residual fuel content of the Neutralizer Tank cubicles is 1 kg of fuel. This estimate was developed by comparing the total volume of the two (2) Neutralizer Tanks (approximately 19,300 gallons) in FH009 to the volume of the AB sump (approximately 7600 gallons). The Neutralizer Tanks hold about three (3) times as much as the AB sump. The maximum measured fuel content of the AB sump (i.e., less than 300 grams, Reference 5.4) was then multiplied by three (3) and rounded to 1 kg for conservative purposes.

All of the three (3) bounding estimates are believed to be highly conservative because the fuel content of the AB sump is below the minimum detectable level of the measurement. Also, the estimates are conservative because of rounding up of all values.

Another benchmark for comparison of the bounding nature of the estimates of residual fuel in the MWST, CWST, and Neutralizer cubicles is the measured residual fuel content in the MU tank cubicles. The makeup tank was used to receive and hold unfiltered reactor coolant for a considerable portion of the post-accident period. Measurement of the residual fuel content in the makeup tank cubicle (Reference 5.5) found approximately 300 grams deposited in the cubicle, virtually all of it in the tank (volume approximately 4500 gallons). Although the tank volume is smaller than the volume of the AB sump, the water held was unfiltered reactor coolant, which is known to have a significantly greater fuel content than the surface decontamination and system flush waste water which was in the AB sump.

5.1.2.6 Cubicle FH014 - Annulus
Cubicle FH112 - Annulus
Cubicle FH205 - Annulus

These cubicles represent the annulus area between the RB and the FHB. The area contains piping that runs between the RCS and MU&P System. The piping in the annulus could contain residual fuel because it is in the letdown and makeup pathway.

The annulus has not been measured for residual fuel content because the piping is still in use. Measurements will be performed after RCS draindown.

A bounding estimate of the residual fuel content of the annulus has been developed based upon fuel characterization measurements of the Makeup Valve Room, FH101, the Makeup Suction Valve Room, FH001, and the Makeup Discharge Valve Cubicles FH003a and FH003b. The piping in the annulus connects the RCS letdown path to the AFHB and the makeup pump discharge back to the RCS. By extrapolation of the results of fuel characterization measurements performed in those cubicles which are in the letdown flowpath immediately downstream of the annulus (FH003a, FH003b) and in the cubicles which contain the piping from the Makeup Pump discharge to the annulus, a bounding estimate of less than 1 kg for the residual fuel content in the annulus was obtained. This estimate is conservative since it is based on a upward rounding of the summation of the measured fuel content of the refereneced cubicles. These cubicles contain piping flowpaths for reactor coolant immediately upstream and downstream of the annulus and contain many more locations favorable for fuel deposit than the piping in the annulus.

5.1.2.7 Cubicle FH106 - Submerged Demineralizer System Monitor Tanks
Cubicle FH110 - Spent Fuel Pool "B"

The Spent Fuel Pool "B" contained the SDS piping and tanks. The SDSmonitor tanks were used to collect water processed by the SDS. Because the SDS was specifically designed to remove all insoluble particles and includes prefilters, post-filters and ion exchangers, the effluent water sent to the monitor tanks contained little or no residual fuel. Due to the extensive filtration, it is conservatively estimated that the residual fuel in the Spent Fuel Pool "B" and Monitor Tanks is expected to be much less than 1 kg.

The Spent Fuel Pool "B" and Monitor Tanks have not been measured to date for their residual fuel content because the SDS was in service until August 1988. The residual fuel inventory will be measured as part of the SDS isolation and cleanup activities.

5.1.2.8 Cubicle FH109 - Spent Fuel Pool "A"

The vast majority of the fuel in Spent Fuel Pool "A" is contained inside the fuel, filter, and knockout canisters stored in the fuel racks. The exact number of filled canisters will vary until all fuel bearing canisters have been shipped from TMI-2 to INEL for long-term storage. The canisters are inherently subcritical by design (Reference 5.6) and are stored in a subcritical array within the fuel storage racks. Further, the TMI-2 Technical Specifications require that during Modes 1 and 2, the water in Spent Fuel Pool "A" will be borated between 4500-6000 ppm. Therefore, subcriticality is ensured under all credible conditions. A very small amount of fuel is accumulating at the bottom of the Spent Fuel Pool "A". This material has been transported from the RV to the Spent Fuel Pool "A" as debris adherent to the outside of the fuel bearing canisters.

Due to the ongoing canister transfer activities in Spent Fuel Pool "A", it is not possible to ascertain the exact amount of the uncontained residual fuel. Thus, for purposes of the DCR, a residual fuel quantity has not been assigned to the Spent Fuel Pool "A". However, since each canister is flushed prior to transfer, the quantity of uncontained residual fuel in Spent Fuel Pool "A", as a result of canister transfer operations, is expected to be a very small fraction of the SFML and will pose no criticality concern. Additionally, the fuel in Spent Fuel Pool "A" does not pose a potential for communicating with other fuel locations in the AFHB. Prior to draindown of the Spent Fuel Pool "A", the residual fuel quantity in this cubicle will be determined and subcriticality will be assured.

5.1.2.9 Cubicle FH111 - Fuel Cask Storage
Cubicle FH302 - SDS Operating Areas

These are the locations where the supporting equipment for SDS processing and the fuel transfer cask are located. Basically composed of access walkways and equipment operating locations, these two (2) areas are routinely kept clean and, in most cases, are not contaminated. There is currently no known residual fuel in these two areas.

5.1.3 Summary

The collective evaluation of the material presented in this report demonstrates that an acceptable end to fuel removal activities has been achieved in the AFHB.

Quantification of the cumulative amount of residual fuel remaining in the AFHB demonstrates assured subcriticality. It has been concluded that any further efforts for the specific purpose of removing fuel from the AFHB will result in unnecessary additional occupational exposure with no attendant proportional benefit realized in terms of removal of substantial quantities of fuel or increasing the margin of safety.

1. The total quantity of fuel in the AFHB (i.e., less than 23 kg of finely divided small particle size sediment material with minor amounts of fuel found as adherent films on metal oxide surfaces, see Table 5-2), exclusive of the fuel in the canisters in the Spent Fuel Pool "A," is far less than the SFML which assumes optimum moderation and infinite water reflector (worst case) conditions. The fuel in the canisters has been demonstrated to be critically safe under all conditions (Reference 5.6). Thus, subcriticality is assured.

2. The residual fuel in the AFHB, not contained in defueling canisters, is located throughout the two buildings in numerous pipes and tanks. Most of these components have been flushed and decontaminated. A few components will be flushed and drained as part of post-defueling activities and this may result in the removal of additional small quantities of fuel. Thus, the quantity of residual fuel in the AFHB may be further reduced.

3. There is no potential for fuel transport within the AFHB to result in a critical mass. Thus, subcriticality is assured in the AFHB.

GPU Nuclear had determined that no additional fuel removal activities are appropriate or necessary within the AFHB to preclude criticality or otherwise demonstrate that defueling has been completed to the extent reasonably achievable.

TABLE 5-1

AFHB CUBICLES WHICH CONTAIN NO RESIDUAL FUEL

DESIGNATION	NAME	EXPLANATION
AX001	RB Emergency Pumps	No fuel transport pathway
AX002	Access Corridor	No waste piping in area
AX003	Access Area	No waste piping in area
AX013	Evaporator Condensate Tank Pumps	All pathways isolated prior to and following the accident
AX022	North Stairwell	No waste piping in area
AX023	Elevator Shaft	No waste piping in area
AX027	South Stairwell	No waste piping in area
AX101	Radwaste Disposal Panel	No waste piping in area
AX103	MCC 2-11 EB	No waste piping in area
AX104	MCC 2-21 EB	No waste piping in area
AX105	Substation 2-11E	No waste piping in area
AX106	Substation 2-21E	No waste piping in area
AX107	MCC 2-11 EA	No waste piping in area
AX108	MCC 2-21 EA	No waste piping in area
AX109	Nuclear Service Coolers and Pump	All pathways isolated since accident
AX110	Intermediate Coolers	All pathways isolated since accident
AX111	Intermediate Cooling Pump	All pathways isolated since accident
AX113	Waste Gas Analyzer	System design prevents fuel transport
AX118	Spent Fuel Coolers	All pathways isolated since accident
AX120	Spent Fuel Filters	All pathways isolated since accident
AX121	Elevator Shaft	No waste piping in area
AX122	North Stairwell	No waste piping in area
AX123	Access Area	No waste piping in area
AX125	Waste Gas Decay TK-1B	System design prevents fuel transport
AX126	Waste Gas Filter Room	System design prevents fuel transport
AX127	Waste Gas Decay TK-1A	System design prevents fuel transport
AX128	Valve and Instrument Room	System design prevents fuel transport
AX132	Corridor Between U1 & U2	All pathways isolated since accident
AX133	South Stairwell	No waste piping in area

TABLE 5-1 (Cont'd)

AFHB CUBICLES WHICH CONTAIN NO RESIDUAL FUEL

DESIGNATION	NAME	EXPLANATION
AX135	Radwaste Disposal Control Panel	No waste piping in area
AX201	North Stairwell	No waste piping in area
AX202	Elevator Shaft	No waste piping in area
AX203	4160 Switchgear 2-1E	No waste piping in area
AX204	4160 Switchgear 2-2E	No waste piping in area
AX205	RB Purge Air Supply	System design prevents fuel transport
AX206	RB Purge Exhaust - B	System design prevents fuel transport
AX207	RB Purge Exhaust - A	System design prevents fuel transport
AX208	AB Exhaust Unit B	System design prevents fuel transport
AX209	AB Exhaust Unit A	System design prevents fuel transport
AX210	FHB Exhaust Unit B	System design prevents fuel transport
AX211	FHB Exhaust Unit A	System design prevents fuel transport
AX212	Decay Heat Surge Tank	No fuel transport pathway
AX213	Unit Substation	No waste piping in area
AX214	Decon Facility	No fuel transport pathway
AX215	FHB Supply Unit	System design prevents fuel transport
AX216	AB Supply Unit	System design prevents fuel transport
AX217	Access Area	No waste piping in area
AX219	Instrument Racks	System design prevents fuel transport
AX220	Caustic Mixing Area	All pathways isolated since accident
AX221	Caustic Mixing Area	All pathways isolated since accident
AX222	South Stairwell	No waste piping in area
AX223	Air Handling Units	System design prevents fuel transport
AX301	Elevator Shaft	No waste piping in area
AX302	North Stairwell	No waste piping in area
AX303	Elevator and Stairwell Access	No waste piping in area
'01	Roof	No waste piping in area
AX402	Cooling Water Storage Tanks	No fuel transport pathway
AX403	Damper Room	System design prevents fuel transport
FH002	Access Corridor	No waste piping in area

TABLE 5-1 (Cont'd)

AFHB CUBICLES WHICH CONTAIN NO RESIDUAL FUEL

DESIGNATION	NAME	EXPLANATION
FH004	West Valve Room	All pathways isolated since accident
FH005	Mini Decay Heat Sevice Coolers	All pathways isolated since accident
FH006	Decay Heat Service Coolers	All pathways isolated since accident
FH007	Neutralizer and Reclaimed Boric Acid	All pathways isolated since accident
FH010	Reclaimed Boric Acid Tank	All pathways isolated since accident
FH011	Reclaimed Boric Acid Pump	All pathways isolated since accident
FH013	Oil Drum Storage	No waste piping in area
FH102	East Corridor	No waste piping in area
FH103	Sample Room	System flushed periodically no deposits
FH104	West Corridor	No waste piping in area
FH105	Model Room A	No waste piping in area
FH107	Trash Compactor	No waste piping in area
FH108	Truck Bay	No waste piping in area
FH111	Fuel Cask Storage	See Section 5.1.2.9
FH201	East Corridor	No waste piping in area
FH202	West Corridor	No waste piping in area
FH203	Surge Tank Area	All pathways isolated since accident
FH204	Standby Pressure Control Area	System design prevents fuel transport
FH302	SDS Operating Area	See Section 5.1.2.9
FH303	Upper SPC Area	System design prevents fuel transport
FH305	Spent Fuel Pool Access	System design prevents fuel transport

TABLE 5-2

AFHB CUBICLES WHICH POTENTIALLY CONTAIN RESIDUAL FUEL(1) (2)

FUEL QUANTITY (kg)	DESIGNATION	NAME	REFERENCE
0.13	AX004	Seal Injection Valve Room	Section 5.1.2.1
0.003**	AX005	Makeup Pump - 1C	TB SNM 89-03*
0.066	AX006	Makeup Pump - 1B	TB SNM 87-02
0.062**	AX007	Makeup Pump - 1A	Eng. Calculation 4550-3211-87-027
	AX008	Spent Resin Storage TK-1B	Section 5.1.2.2
	AX009	Spent Resin Storage TK-1A	Section 5.1.2.2
	AX010	Spent Resin Storage Tank Pump	Section 5.1.2.2
	AX014	Reactor Coolant Evaporator	Section 5.1.2.2
	AX015a	Cleanup Filters	Section 5.1.2.2
	AX015b	Cleanup Filters	Section 5.1.2.2
0.8	AX016	Cleanup Demineralizer - 2A	Section 5.1.2.2
	AX017	Cleanup Demineralizer - 2B	Section 5.1.2.2
	AX114	MU&P Demin - 1A	Section 5.1.2.2
	AX115	MU&P Demin - 1B	Section 5.1.2.2
	AX119	Spent Fuel Demineralizer	Section 5.1.2.2
	AX129	Deborating Demineralizer - 1B	Section 5.1.2.2
	AX130	Deborating Demineralizer - 1A	Section 5.1.2.2
	FH001	MU Suction Valves	Section 5.1.2.2
0.002**	AX011	AB Sump Pump and Valve	TB 86-28
0.300**	AX012	AB Sump Pumps and Tank	TB 86-28
0.01**	AX018	Waste Transfer Pump	TB 86-38

* - TB refers to a TMI-2 Technical Bulletin
** - Denotes Minimum Detectable Level

(1) - Based on current available data, this table will be updated as necessary as further data become available. Wherever uncertainties exist as to the quantity of fuel, the upper bound estimate is used.

(2) - The predominant form of residual fuel identified in the AFHB is finely divided, small particle size, sediment material with minor amounts of fuel found as asherent films on metal oxide surfaces.

TABLE 5-2 (Cont'd)

AFHB CUBICLES WHICH POTENTIALLY CONTAIN RESIDUAL FUEL(1) (2)

FUEL QUANTITY (kg)	DESIGNATION	NAME	REFERENCE
0.005**	AX019	HDL Valves	TB 86-38
14	AX020	RCBTs 1B and 1C	TB 87-12
1	AX021	RCBT 1A	Section 5.1.2.3
0.005	AX024	AB Sump Filters	TB SNM 89-02
0.002**	AX026	Seal Injection Filters	TB SNM 87-04
0.300	AX102	RB Sump Pump Filters	Section 5.1.2.4
0.292	AX112	Seal Return Coolers	TB SNM 88-03
0.309	AX116	Makeup Tank	Eng. Calculation 4550-3211-87-038
0.04	AX117	MU&P Filters	TB 86-38
1	AX131	Miscellaneous Waste Holdup Tank	Section 5.1.2.5
	AX134	Miscellaneous Waste Tank Pumps	Section 5.1.2.5
0.5	AX124	Concentrated Liquid Waste Pump	Section 5.1.2.5
	AX218	CWSTs	Section 5.1.2.5
0.002	AX501	RB Spray Pump - 1A	TB 86-47
0.002	AX502	RB Spray Pump - 1B	TB 86-47
0.002	AX503	DHR Cooler & Pump - 1A	TB 86-47
0.002	AX504	DHR Cooler & Pump - 1B	TB 86-47
0.008**	FH003a	MU Discharge Valves	TB 86-38
0.060**	FH003b	MU Discharge Valves	TB 86-38

* - TB refers to a TMI-2 Technical Bulletin
** - Denotes Minimum Detectable Level (MDL)

(1) - Based on current available data, this table will be updated as necessary as further data become available. Wherever uncertainties exist as to the quantity of fuel, the upper bound estimate is used.

(2) - The predominant form of residual fuel identified in the AFHB is finely divided, small particle size, sediment material with minor amounts of fuel found as adherent films on metal oxide surfaces.

TABLE 5-2 (Cont'd)

AFHB CUBICLES WHICH POTENTIALLY CONTAIN RESIDUAL FUEL[(1) (2)]

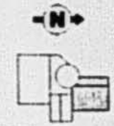
FUEL QUANTITY (kg)	DESIGNATION	NAME	REFERENCE
1	FH008	Neutralizer Tank Pump	Section 5.1.2.5
	FH009	Neutralizer Tank	Section 5.1.2.5
	FH012	Neutralizer Tank Filers	Section 5.1.2.5
<1	FH014	Annulus	Section 5.1.2.6
	FH112	Annulus	Section 5.1.2.6
	FH205	Annulus	Section 5.1.2.6
0.71	FH101	MU&P Valve Room	TB 86-38/86-21
1	FH106	SDS Monitor Tanks	Section 5.1.2.7
	FH110	Spent Fuel Pool "B"	
------ (3)	FH109	Spent Fuel Pool "A"	Section 5.1.2.8
<23 kg = TOTAL			

(1) - Based on current available data, this table will be updated as necessary as further data become available. Wherever uncertainties exist as to the quantity of fuel, the upper bound estimate is used.

(2) - The predominant form of residual fuel identified in the AFHB is finely divided, small particle size, sediment material with minor amounts of fuel found as adherent films on metal oxide surfaces.

(3) - No Value Assigned; See Section 5.1.2.8 for details.

FIGURE 5-1
AUXILIARY BUILDING 280'-6" ELEVATION



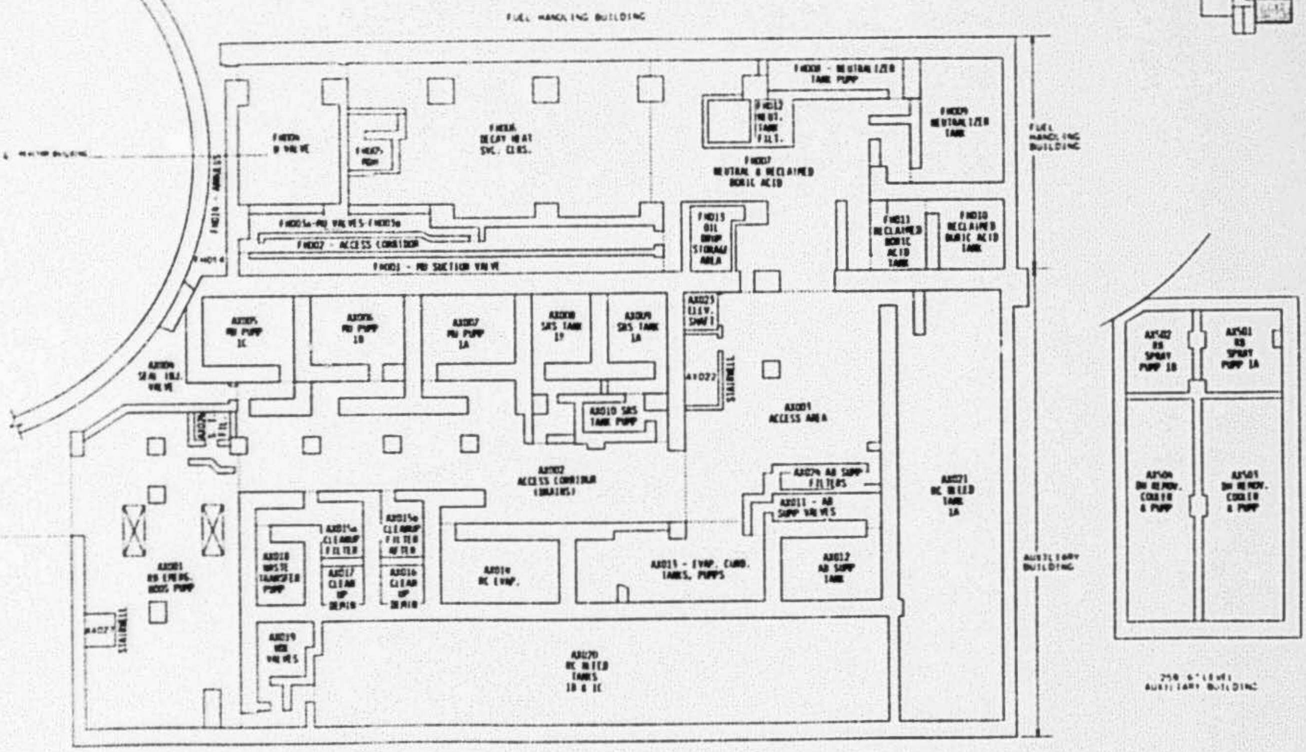


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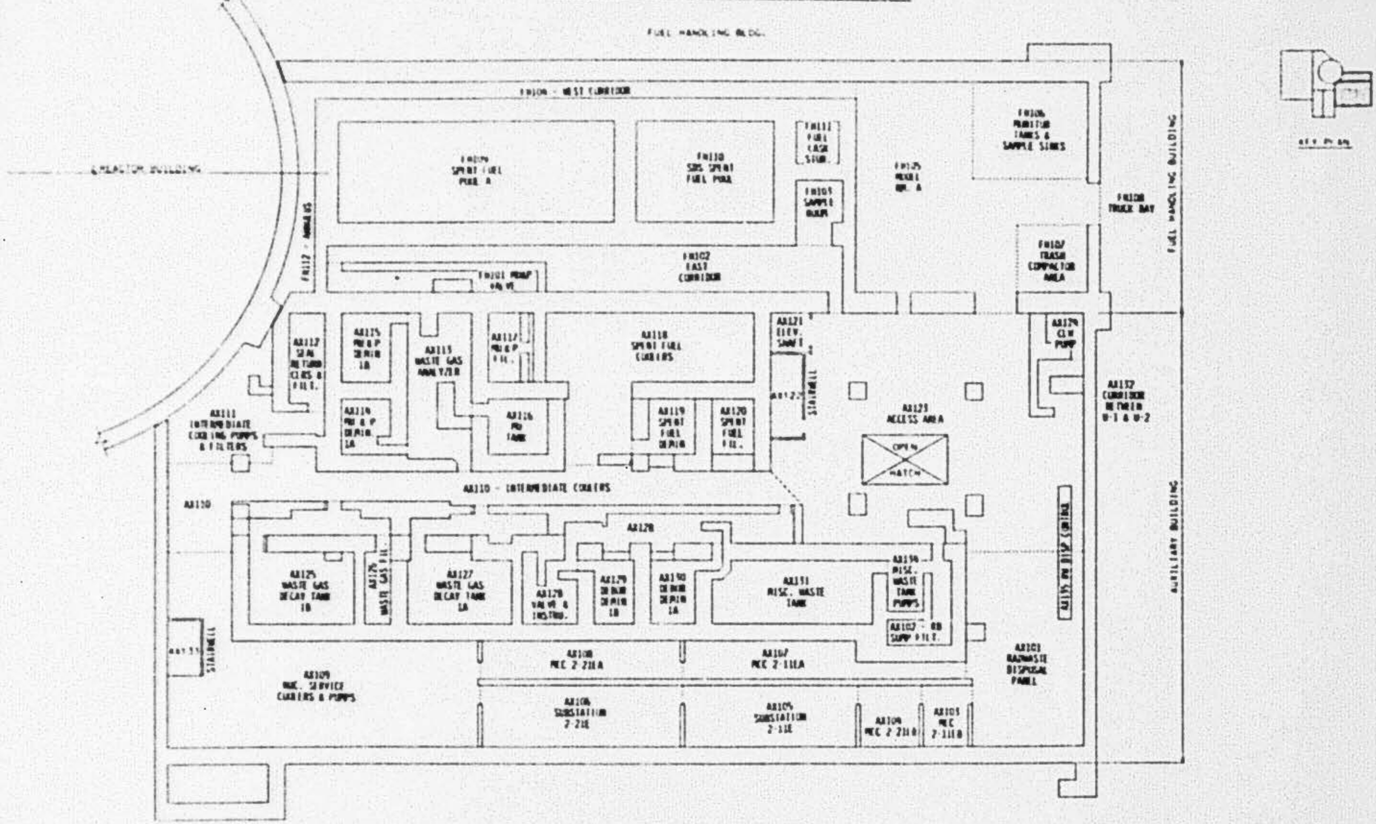
FIGURE 5-2
AUXILIARY BUILDING 305' ELEVATION



305' LEVEL AUXILIARY BLDG.

FIGURE 5-3
AUXILIARY BUILDING 328' ELEVATION

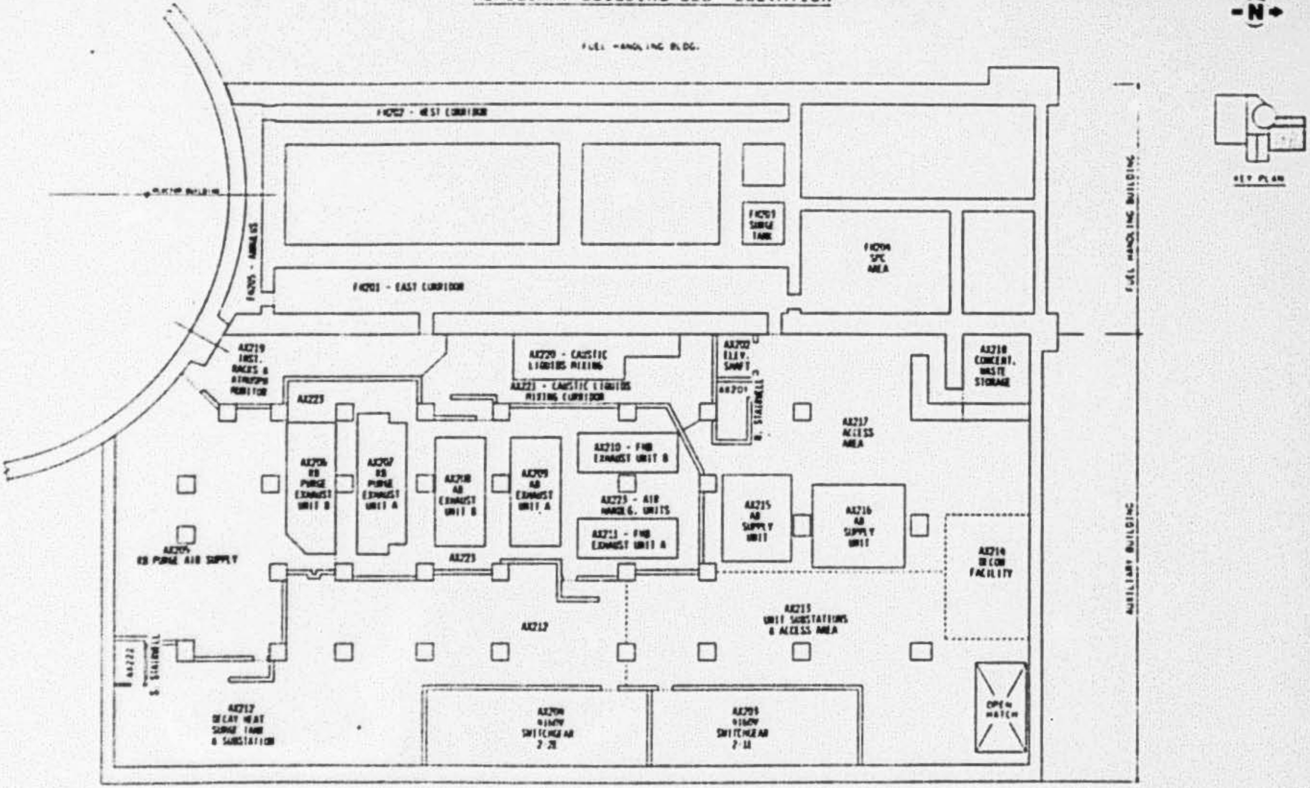


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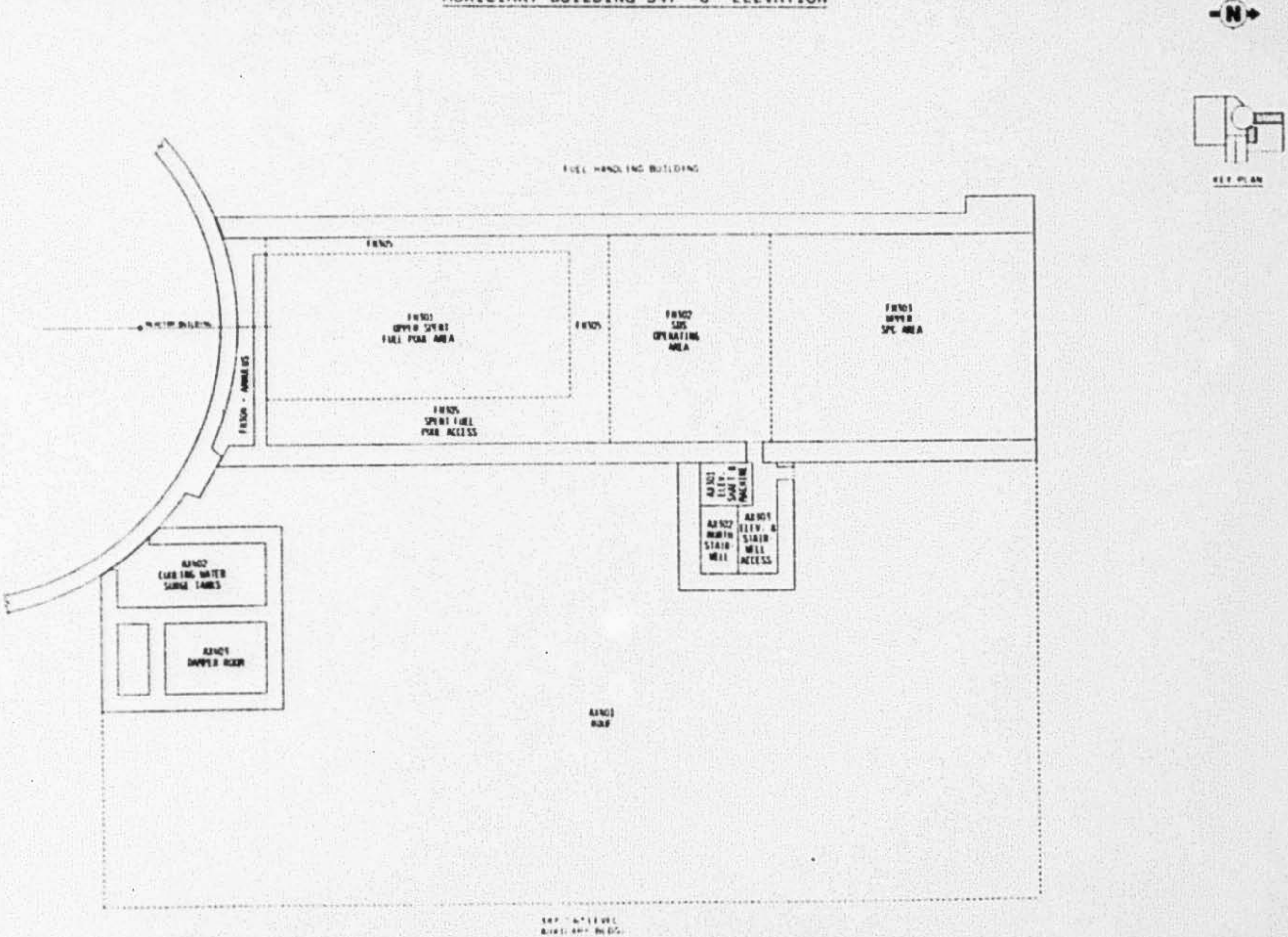
FIGURE 5-4
AUXILIARY BUILDING 347'-6" ELEVATION



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APPENDIX A

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