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US Nuclear Regulatory Commission
Washington, DC 20555

ATTN: Document Control Desk

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

Dear Sirs:

GPU Nuclear letter 4410-89-L-0070, dated July 5, 1989, transmitted the initial submittal of the Defueling Completion Report (DCR) which included the basis for determining that defueling activities in the Auxiliary and Fuel Handling Buildings (AFHB) have been completed. The letter also stated that subsequent submittals, forwarded as fuel removal activities are completed, would 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).

Attached, in part, is that portion of the DCR which provides the basis for determining that defueling activities in the RB have been completed (i.e., Section 5.2). The amount of residual fuel in the RB has been demonstrated to be less than 88 kg which is significantly below the Safe Fuel Mass Limit and a very small fraction of the defueling endpoint goal.

This submittal also includes minor corrections in previously submitted sections of the DCR which are annotated by change bars in the margin. Changed pages and the new material submitted herein are indicated as Revision 1 to the initial submittal of the DCR.

Sincerely,

M B Roche

M. B. Roche
Director, TMI-2

A001
1/1

RDW/emf

Attachment

cc: W. T. Russell - Regional Administrator, Region I
J. F. Stolz - Director, Plant Directorate I-4
L. H. Thonus - Project Manager, TMI Site
F. I. Young - Senior Resident Inspector, TMI

TABLE OF CONTENTS

TABLE OF CONTENTS

TABLES

FIGURES

TABLE OF CONTENTS

FIGURES (Cont'd)

APPENDICES

1.0 INTRODUCTION

1.1 Background

On May 27, 1988, the NRC issued License 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 Reactor Building. 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 Buildings (AFHB), Reactor Building (RB), Reactor Coolant System (RCS), and Reactor Vessel (RV)].

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 ensure 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 than 200°F.
2	Mode 2 shall exist when the following conditions are met: a. The Reactor Vessel and Reactor Coolant System are 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 Bore Machine
CFT	Core Flood Tank
CRA	Contral Rod Assembly
CSA	Core Support Assembly
CWST	Concentrated Waste Storage Tank
DCR	Defueling Completion Report
DF	Decontamination Factor
DHR	Decay Heat Removal
DOE	Department of Energy
DWCS	Defueling Water Cleanup System
ECCS	Emergency Core Cooling System
FHB	Fuel Handling Building
FTC	Fuel Transfer Canal
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
LS	Lead Screw
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

TABLE 1-2 (Cont'd)

ACRONYMS

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
RCDT	Reactor Coolant Drain Tank
RCP	Reactor Coolant Pump
RCS	Reactor Coolant System
RV	Reactor Vessel
SDS	Submerged Demineralizer System
SER	Safety Evaluation Report
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

2.1.2 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 approximately 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 core 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

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 particulates 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 debris and 7,000 kg of aglomerated core 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 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 rubbled.

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 into the "B" loop where a portion of the fuel 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, 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 transported outside the RV. However, a small quantity of fuel 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 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 phenomenon 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 stationary 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 through the pressurizer and PORV into the RCDT, and subsequently into the RB through the 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 was transported into the letdown coolers and associated piping.

In summary, a relatively small quantity of fuel (see Table 2-1) was released to the RB as a result of the accident due to the operation of the PORV and the MU&P System (References 2.13 through 2.17).

2.2.4 Auxiliary and Fuel Handling Buildings

A small quantity of fuel 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 was transported into the system as indicated by the fact that very little fuel 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 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 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 may have been transported to the Seal Injection System.

RCBTs A, B, and C also contained fuel 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 was transported into the AFHB (see Table 2-1), principally through the RCBTs and the MU&P System. Some of this fuel 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 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 of fuel. These components, which are currently stored in various RB locations, include the RV head, upper plenum assembly, internal RV structures (e.g., endfittings, LCSA grid plates, distributor plates, grid forging), 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.

TABLE 2-1

POST-ACCIDENT ESTIMATED EX-VESSEL
FUEL MATERIAL DISTRIBUTION
(References 2.12 through 2.17)

	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	0.1
Letdown Coolers	4
Core Flood System	1
Auxiliary/Fuel Handling Buildings	
Makeup and Purification System	6
Seal Injection System	1
Reactor Coolant Bleed Tanks A, B, and C	15
Waste Disposal Liquid System	1

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 primarily 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. Additionally, a directional gamma probe and a cadmium telluride gamma spectrometer were used to measure the "B" Core Flood Tank Line.

3.2 Neutron Detectors and Activation/Interrogation Techniques

Neutrons from spontaneous fission and (Y,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.

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

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 4.4). 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 Buildings 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:

4.1.2.3 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 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 SRSTs, 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 Buildings 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 MU&P 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 made 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 filters 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 following sections discuss those areas of the RB where decontamination activities were performed which resulted in the removal of residual fuel. Other locations in the RB which contain residual fuel (e.g., plenum) are described further in Section 5.2.

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.

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. Residual fuel in the "B" hot leg was scraped, flushed, and vacuumed into defueling canisters as part of RV defueling (Section 4.4).

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 was removed, as described in the above section, as part of the RV defueling.

4.3.4 RCS Fuel Removal Assessment

(To Be Published Later)

4.4.3.2 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 Bore Machine (Reference 4.27)

In September 1986, the CBM was re-installed to break the large resolidified mass into rubble 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.

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 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.

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 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

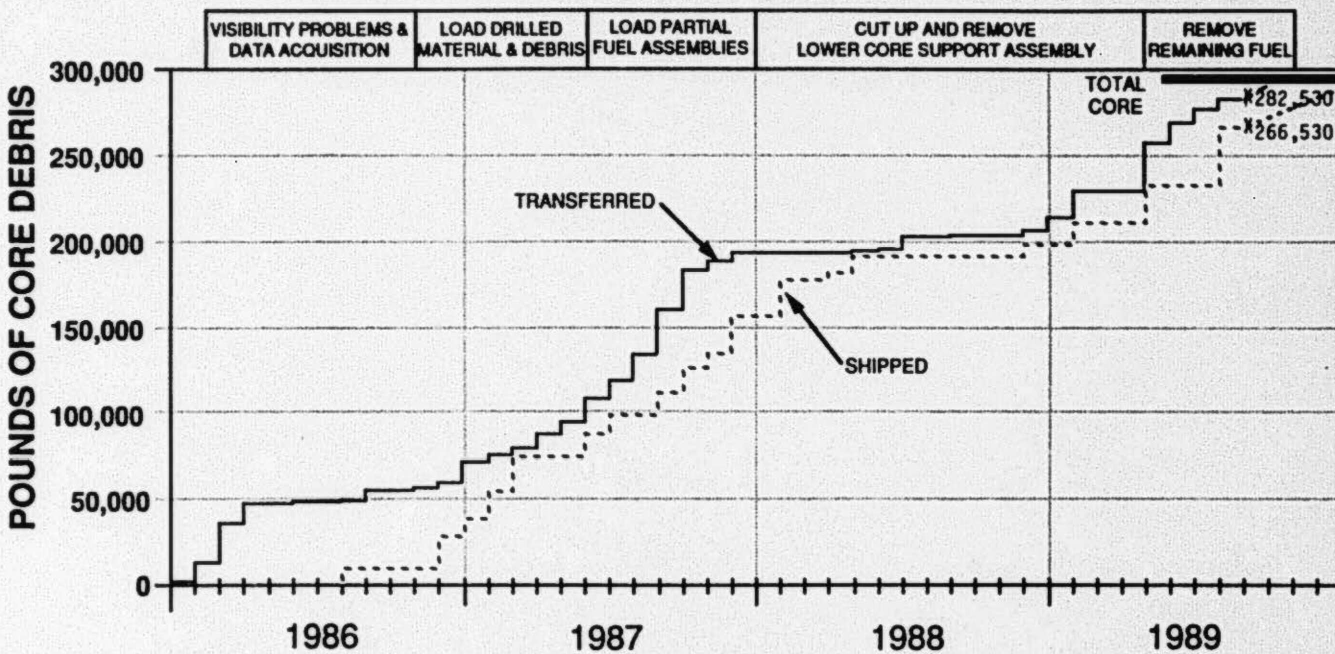
(To Be Published Later)

4.4.4 Reactor Vessel Fuel Removal Assessment

(To Be Published Later)

FIGURE

TMI-2 DEFUELING PROGRESS



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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 the 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 effcrt was urdertaken (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.

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 Liquid Waste Pumps (AX124), the Concentrated Waste Storage Tank (CWST) (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 20,000 gallons) to the AB sump (approximately 7600 gallons). The MWHT holds approximately three (3) times the volume of the AB sump. The maximum measured fuel content of the AB sump (less than 300 grams; Reference 5.4) was then multiplyed by a factor of 3 and rounded up to 1 kg for conservative purposes.

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) 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 MWHT, CHST, 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 an upward rounding of the summation of the measured fuel content of the referenced 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 SDS monitor 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 4350-6000 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.

5.2 Reactor Building

During the accident, fuel was transported to the RB as a result of the core degradation event and coolant flow from the RV through the PORV and RCS MU&P System. Table 2-1 reported that approximately 11 kg of fuel was transported to the RB during the accident sequence. Subsequent to the accident, fuel was relocated to the RB as a result of several cleanup operations including: transfer to and storage of structural RV components in the "A" CFT and "A" D-ring; storage of upper end fittings; flushing of defueling tools; and transfer of the defueling canisters into the FTC. Even though fuel was relocated to the RB during cleanup operations, RB residual fuel conditions were maintained significantly below the SFML. Further, a significant cleanup effort was undertaken (as described in Section 4.2) with the primary purpose of reducing exposure rates but which also resulted in the removal of additional core debris.

The following sections provide the current estimates of residual fuel remaining within the RB, not including the RCS and RV. These estimates are based on fuel measurements, visual inspections, and extensive evaluations of RB structures, systems, and components. The basis for each estimate is provided. As noted in Section 3.6, some of the reported residual fuel quantities are referred to as MDL indicating that the actual quantity of residual fuel is less than or equal to the reported value.

5.2.1 Reactor Vessel Head Assembly (Reference 5.7)

The RV head assembly was removed from the RV and placed on its storage stand on the 347' elevation in July 1984. Portions of the head structure that were exposed to reactor coolant include the dome, flange, leadscrews, leadscrew support tubes, and leadscrew motor housing. Only these components were considered when calculating fuel content in the head assembly. During and after the core degradation portion of the accident, the control rod assemblies were fully inserted into the core region. The leadscrews were, therefore, extended into the plenum area inside their support tubes. Because of the close proximity of the leadscrews to the head surfaces, leadscrew fuel deposition data is taken as an analog for fuel deposition on head surfaces.

In November 1982, three (3) leadscrews were removed for analysis. Fuel analyses were performed on two (2) of the samples by Battelle Columbus Laboratories, Science Applications, International Corporation, and Babcock and Wilcox. Also, a sample of a leadscrew support tube was analyzed for radionuclide activity on both internal and external surfaces.

The fuel content of the leadscrews was extrapolated from direct fuel assay of the leadscrew samples. The fuel content of the other RV head assembly components was calculated by:

- determining the Ce-144 activity on leadscrew surfaces by gamma spectroscopy and the fuel activity on the leadscrew surfaces by direct assay;

- adjusting the activity distribution as evidenced by the internal/external contamination ratio on the leadscrew support tube sample;

- dividing by the average Ce-144/fuel ratio determined for the leadscrews to get a fuel to surface area value;

- multiplying the fuel/area ratio by the corresponding surface area for the RV head assembly component in question.

Visual inspection was done of the RV head assembly and no desposits were observed in the structure. Considering the force of gravity and the RV head assembly geometry, gravel-like material is not expected to be on the RV head.

Summing the component fuel values produced the total fuel estimate for the RV head assembly. The preliminary estimate of fuel in the RV head assembly is 1.4 kg, primarily in the form of surface films.

5.2.2 Reactor Vessel Upper Plenum Assembly (Reference 5.8)

During reactor operation, the plenum is located directly above the reactor core and below the RV head assembly. It consists of a cover, CRA guide tube assemblies (guide tubes), upper grid (at the bottom of the plenum), and the flanged plenum cylinder with openings for reactor coolant flow (see Figures 5-5 and 5-6). CRA guide tube assemblies provide CRA alignment, protect CRAs from coolant cross-flow, and provide structural attachment of the grid assembly to the plenum cover. The leadscrews, which move the CRAs in and out of the core, were inside the guide tubes during the accident. The 69 guide tubes are vertical cylinders that constitute the majority of the surface area in the plenum assembly.

During the accident, fuel particles were transported to the plenum when large amounts of reactor coolant flow, steam, and hydrogen passed through it. Fuel was deposited in sediment and surface films on the plenum surfaces. In May 1985, the plenum was lifted from the RV and placed on a storage stand in the deep end of the FTC. The plenum was flushed to remove loose surface debris, prior to its removal from the RV.

The calculation of fuel loading in the plenum is based on analysis of samples from two (2) leadscrews and one (1) leadscrew support tube which are composed of similar material to the plenum and whose fuel deposition is believed to be representative of the plenum. The two (2) leadscrews were in the plenum during the accident and were removed before plenum lift. The fuel activity found on the leadscrews was extrapolated to the total surface area of the plenum components exposed to coolant flow. Data from the leadscrew support tube was used to correct for high and low flow areas in the plenum assembly.

A small fraction of the total surface area of the plenum consists of upward-facing horizontal surfaces. To account for the settling of fine sediment on these surfaces, the difference between threaded and non-threaded leadscrew surface activity was applied. Higher activity levels on threaded surfaces were assumed to be the result of settling of fine debris in the threads. A high and low flow correction was also applied to this portion of the calculation.

A conservative estimate of the residual fuel quantity in the plenum is:

Surface Films	1.5 kg
Silt/Sediment	0.6 kg
TOTAL	2.1 kg

5.2.3 Fuel Transfer Canal

The vast majority of the fuel in the FTC is contained inside the fuel, filter, and knockout canisters located in the fuel racks. The exact number of filled canisters will vary until all fuel bearing canisters have been transferred to Spent Fuel Pool "A" for shipment to INEL. The canisters are stored in an inherently subcritical array within the fuel storage racks. Further, during Mode 1 the TMI-2 Technical Specifications require that the water in the FTC will be borated to a concentration of 4350-6000 ppm. Therefore, subcriticality is currently ensured under all credible conditions notwithstanding that a very small amount of uncontained fuel may be accumulating at the bottom of the FTC, having been transported from the RV to the FTC as debris adherent to the outside of the fuel bearing canisters.

Since each canister is flushed prior to transfer from the RV, the quantity of uncontained residual fuel potentially accumulating in the FTC, 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 residual fuel in FTC will not pose a potential for communicating with other fuel locations in the RB. GPU Nuclear is currently performing fuel measurements of the FTC. Results of these measurements will be provided in a subsequent DCR submittal.

5.2.4 Core Flood System (References 5.9 through 5.13)

The core flood system consists of two (2) tanks and piping into the RV (see Figure 2.4). During LCSA defueling, the top of the "A" CFT was removed and the tank was used for storage of LCSA components. Additionally, the piping from the "A" CFT to the RV was cut and flanged which will prevent the possibility of fuel transport. Storage of the LCSA components outside but in proximity to the RV (e.g., in the "A" CFT) was deemed necessary to permit continuous progress in the RV defueling activities. Prior to removal from the RV, the LCSA segements were flushed and brushed to remove fuel. The segments were then video inspected to ensure that no visible fuel was present. Sample sections of each plate were measured by gamma spectroscopy and/or alpha measurements to determine the quantity of residual fuel. Extrapolation of fuel content in other sections was determined based on the fuel quantity of the measured sections. For example, two (2) of the four (4) quadrants of the lower grid distributor plate were measured for fuel content and determined to contain a total residual fuel quantity of 163 grams. These measurements were extrapolated for the other two (2) quadrants and a total residual fuel quantity of 320 grams of residual fuel was assigned to the lower grid distributor grid (Reference 5.10). Likewise, one (1) of the 11 pieces of the flow distributor plate was measured for fuel content (Reference 5.13). Its residual fuel value (i.e., 10 grams) was deemed to be representative of the remaining segments and a total residual fuel quantity of 110 grams was assigned for the flow distributor plate.

Based on the above approach, the "A" CFT, which contains the LCSA components, has been assigned a total of approximately 2.4 kg (References 5.9 through 5.13) of residual fuel, distributed as follows:

Components	Fuel (kg)
Lower Grid Rib Section	<0.1
Lower Grid Distributor Plate	<0.3*
Lower Grid Forging	1.7
Incore Guide Support Plate	<0.2*
Flow Distributor Plate	0.1
TOTAL	2.4

* = MDL value

The portion of the "B" core flood line between the CFT and the check valve was measured for fuel debris using both a directional gamma probe and a cadmium telluride gamma spectrometer. This measurement determined a maximum residual fuel quantity of 130 grams (Reference 2.12).

Measurement of the residual fuel in the "B" CFT and the "A" core flood line are planned and will be provided in a subsequent DCR submittal. Based on the residual fuel content in the "B" core flood line, the residual fuel quantity in these areas is not expected to substantially increase the current core flood system estimate. There are no post-defueling plans to remove the LCSA components stored in the "A" CFT due to the relatively small quantity of residual fuel involved.

5.2.5 D-Rings (Reference 5.13)

The only residual fuel in the D-rings above the basement level (basement is discussed in Section 5.2.9) is located in the flow distributor plate sections stored therein. Sections of the flow distributor plate removed from the RV which contained IIGTs were too large to be placed in the "A" CFT. These sections were bagged and suspended in the "A" D-ring in order to prevent interference with continued progress in the RV defueling efforts. These sections were brushed and flushed prior to removal from the RV.

Gamma spectrometry performed on 13 of the 14 segments placed in the D-rings, containing a total of 30 IIGTs determined that these segments contain 21 kg of residual fuel. The remaining segment which was not measured for residual fuel, contains three (3) IIGTs. Based on a simple arithmetic average of the amount of fuel per IIGT of the unmeasured segments (i.e., 21 kg per 30 IIGTs), it is reasonably estimated that the unmeasured segment contains approximately 2 kg of residual fuel (i.e., 0.7 kg per IIGT multiplied by 3 IIGTs). This estimate is believed to be conservative because the unmeasured segment was in the northwest quadrant of the flow distributor plate whereas the measured segments which contained the largest quantities of residual fuel were generally located in the southeast quadrant of the flow distributor plate. Thus, the total estimated amount of residual fuel in the "A" D-ring is 23 kg. Further assessment of the LCSA components in the "A" D-rings is provided in Section 6.0.

5.2.6 Upper Endfitting Storage Area

As described in Section 4.4.3.2.1, during RV defueling, loose upper endfittings were removed from the surface of the RV debris bed to allow access for defueling. These endfittings were too large to be inserted into fuel canisters; thus, they were placed in shielded drums filled with borated water (i.e., 4350-6000 ppm) and stored at elevation 347' in the RB. Storage of these upper endfittings is described in an NRC-approved SER (References 5.14 and 5.15).

Currently, there is a total of 18 upper endfittings stored in a total of five (5) containers in the endfitting storage area. The maximum number of endfittings in a single container is six (6). Reference 5.14 conservatively estimated that each endfitting could contain up to 3 kg of fuel if fuel were packed solidly within the flow spaces in the endfitting casings. Based on the maximum of six (6) endfittings per container, the maximum estimated fuel in any container would be 18 kg. This amount is significantly less than the SFML. Additionally, if all of the 18 endfittings were loaded with fuel to the maximum theoretical value (i.e., 3 kg of fuel), the total maximum amount of fuel is conservatively estimated to 54 kg. This quantity is also significantly less than the SFML. Furthermore, the upper endfitting storage area is neutronically decoupled from any other fuel bearing location; thus, subcriticality is assured.

GPU Nuclear is planning to measure the upper endfitting storage containers in order to quantify the amount of fuel in each container (Reference 16). This section will be updated in a subsequent DCR submittal to reflect the results of the survey program.

5.2.7 Reactor Coolant Drain Tank (Reference 4.20)

As described in Section 2.2.3, fuel was deposited in the RCDT as a result of the accident. This tank provided a settling point for particles escaping from the PORV before release to the RB basement. The RCDT has been inaccessible for defueling operations due to the high dose rates in the RB basement.

In 1983, sludge samples were collected and video inspections were performed. Analysis of the samples yielded a uranium concentration of 3.7 mg/g in the sludge. This, combined with an estimate of the quantity of sludge in the tank (2.6×10^4 g), adjusted to UO_2, produced an estimate of fuel in the RCDT of approximately 0.1 kg. This residual fuel quantity is deemed to be valid since there have been no defueling or decontamination activities performed in the RCDT.

5.2.8 Letdown Coolers (Reference 5.17)

The letdown cooler cubicle, located in the RB basement, contains the letdown coolers (MU-C-1A and 1B) and associated piping. This system was designed to cool the reactor coolant before it entered the rest of the MU&P System for processing. Portions of the MU&P System ran continuously before and during the accident, and have run since the accident, potentially transporting small amounts of core debris throughout the system. Residual fuel in most MU&P components is discussed in Section 5.1.

Fuel in the letdown cooler system was measured with a collimated, shielded sodium iodide gamma spectrometer. Calculations were made using computer codes to model the associated piping, coolers, and detector configurations. The calculated residual fuel content of the letdown coolers system is less than or equal to an MDL value of 4 kg.

5.2.9 RB Basement and Sump (Reference 5.18)

The RB basement consists of the space between the floors of elevations 282'6" and 305' of the RB, the RB sump, and the floor drains. Excluded from this section and treated elsewhere in this report is equipment (e. g., the letdown coolers and RCDT) located in the basement.

During the accident, reactor coolant was discharged from the RCS into the RCDT and then into the RB basement. Table 2-1 indicates that the RB basement/sump contained approximately 5 kg of fuel as a result of the accident. The reactor coolant that was discharged into the RB became mixed with sediment-bearing river water, RB spray water, decontamination water, condensation, and additional leakage from the RCS. The basement remained flooded for approximately two (2) years. During this period, sediment and fuel fines settled into a sludge on the basement floor. As discussed in Section 4.2, a significant portion of this sludge was removed during cleanup operations in the RB basement.

The sludge remaining after desludging operations was analyzed by sampling and gamma spectroscopy methods. Uranium concentrations measured in three (3) samples were combined with estimates of residual sediment volume to calculate the total residual fuel on the basement floor excluding the RCDT discharge area. A gamma scan was performed in the RCDT area since the maximum amount of fuel was initially expected to be located in the RCDT. The total fuel contained in the remaining basement sludge following cleanup operations is estimated to be approximately 1.1 kg.

Additionally, fuel particles from washdown of defueling tools was transported through the RB drain system to the RB sump. Reference 5.18 provides an initial estimate that 0.2 kg of fuel could have been added to the basement inventory from this activity. Thus, the total fuel in the RB basement is currently estimated to be 1.3 kg.

5.2.10 Miscellaneous Systems and Equipment

In addition to the residual fuel quantities reported in Sections 5.2.1 through 5.2.9, residual fuel is expected to be contained in various systems/equipment located in the RB which were utilized during the defueling effort. Included are the DWCS and the Defueling Tool Rack which contains the various long-handled tools used to defuel the RV. Residual fuel contained in these operating cleanup systems/equipment is expected to amount to a very small fraction of the SFML and will pose no criticality concern. For example, the NRC approved DWCS Technical Evaluation Report (TER) (Reference 5.19) states that the DWCS has been designed to prevent a possible critical configuration of fuel. Further, the DWCS will be internally flushed and partially disassembled prior to being decommissioned. This action will remove a portion of the internal deposits of residual fuel contained in the DWCS. Additionally, as discussed in Section 5.2.9, defueling tools are generally flushed prior to removal from the RV in order to remove any loose residual fuel. The estimate of residual fuel in these cleanup systems will be provided in a subsequent DCR submittal.

5.2.11 Criticality Assessment

Table 5-3 lists the total quantity of residual fuel in the RB exclusive of the RCS and RV. This table will be updated following the completion of remaining fuel measurements. As indicated, the total fuel mass remaining in the RB is well below the SFML of 140 kg presented in Appendix B. Subcriticality is further enhanced since most of the residual fuel is tightly adhered to RV components or in isolated areas within the RB. Fuel in this configuration is significantly less reactive than in the optimum conditions assumed in Appendix B (i.e., fuel pellets, optimum moderation with unborated water, and spherical geometry). Additionally, the current configuration prevents any significant debris transport, thus minimizing any interactive effects of the various fuel accumulations. The majority of residual fuel in the RB (i.e., "A" D-ring, letdown coolers, and upper endfitting storage containers) is located in areas which are neutronically decoupled from other fuel bearing locations and, consequently, there is no potential for a criticality event due to fuel transport. Thus, subcriticality within the RB is assured. The potential for fuel transport and interaction with the RCS and RV will be described in Sections 5.3 and 5.4.

5.2.12 Summary

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

The total estimated quantity of fuel in the RB, listed in Table 5-3, is significantly less than the SFML which assumes optimum moderation and infinite water reflector (worst case) conditions. Additionally, it is expected that the total quantity of residual fuel in the RB following the completion of remaining fuel measurements in the RB will continue to be significantly less than the SFML. Thus, subcriticality is assured.

The current estimate of residual fuel content in the RB is primarily concentrated in:

- o segments of the flow distributor plate containing IIGTs which are stored in the "A" D-ring;
- o upper endfitting storage containers; and
- o letdown coolers.

Each of these areas are addressed below.

<u>"A" D-Ring</u>

Residual fuel is attached to the sections of the flow distributor plate located in the "A" D-ring. GPU Nuclear decided to place these components in the "A" D-ring since there were too large to placed in the "A" CFT. The residual fuel quantity in the "A" D-ring does not pose a criticality concern. Additionally, it is anticipated that some of the LCSA components in the "A" D-ring may be shipped off-site for analysis; thus, the total residual fuel quantity in this area may be reduced. Section 6 of the DCR provides a further assessment of the LCSA components in the "A" D-ring.

Upper Endfitting Storage Containers

As stated in Sections 4.4.3.2.1 and 5.2.6, during the early defueling efforts loose upper endfittings were removed from the surface of the core debris bed to permit the continuation of RV defueling. These endfittings were too large to be inserted into fuel canisters; thus, they were placed in shielded drums filled with borated water and stored at elevation 347' in the RB. As stated in Section 5.2.6, it is conservatively estimated that the maximum amount of residual fuel contained in the upper endfitting storage containers is 54 kg which is much less than the SFML. GPU Nuclear currently plans to perform fuel measurements of each upper endfitting storage container as described in Reference 5.16. It is GPU Nuclear's belief that the residual fuel quantity determined as a result of these measurements will be significantly less than 54 kg. The disposition of the upper endfitting storage containers will be determined following their measurements.

Letdown Coolers

Section 2.2.3 states that fuel was transported to the letdown coolers, which are located in the RB basement, as a result of the TMI-2 accident. Due to the high dose rates in the RB basement, the letdown coolers are inaccessible and performance of defueling or water processing activities has been precluded. Fuel measurements of the letdown coolers have determined that their residual fuel quantity is less than or equal to an MDL value of 4 kg which is significantly below the SFML. Thus, the residual fuel in the letdown coolers does not pose a criticality concern.

The residual fuel in the remaining areas of the RB consists of finely divided small particle size sediment material with minor amounts of fuel found as adherent films on metal oxide surfaces. Decontamination activities in the RB served to remove residual fuel, especially in the RB basement where the residual fuel quantity was reduced by approximately 75% (see Tables 2-1 and 5-3). Post-defueling activities (e.g., flushing tanks/pipes, system draindowns) may result in the removal of additional small quantities of fuel. Thus, the quantity of residual fuel in the RB may be further reduced.

Based on the above analysis of the total estimated quantity of residual fuel, there is no potential for transport of fuel within the RB which could result in a critical mass. Thus, subcriticality is assured. The potential for fuel transport and interaction with the RCS and RV will be described in Sections 5.3 and 5.4. GPU Nuclear has determined that no further efforts for the specific purpose of removing fuel from the RB are appropriate or necessary to preclude criticality or otherwise demonstrate that defueling has been completed to the extent reasonably achievable.

However, a few components will be flushed/drained or removed as part of post-defueling activities which may result in the removal of additional small quantities of fuel. Thus, the quantity of residual fuel in the RB may be further reduced.

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
AX401	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 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
0.005**	AX019	WDL 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.

TABLE 5-3

RESIDUAL FUEL QUANTIFICATION IN THE REACTOR BUILDING[a]

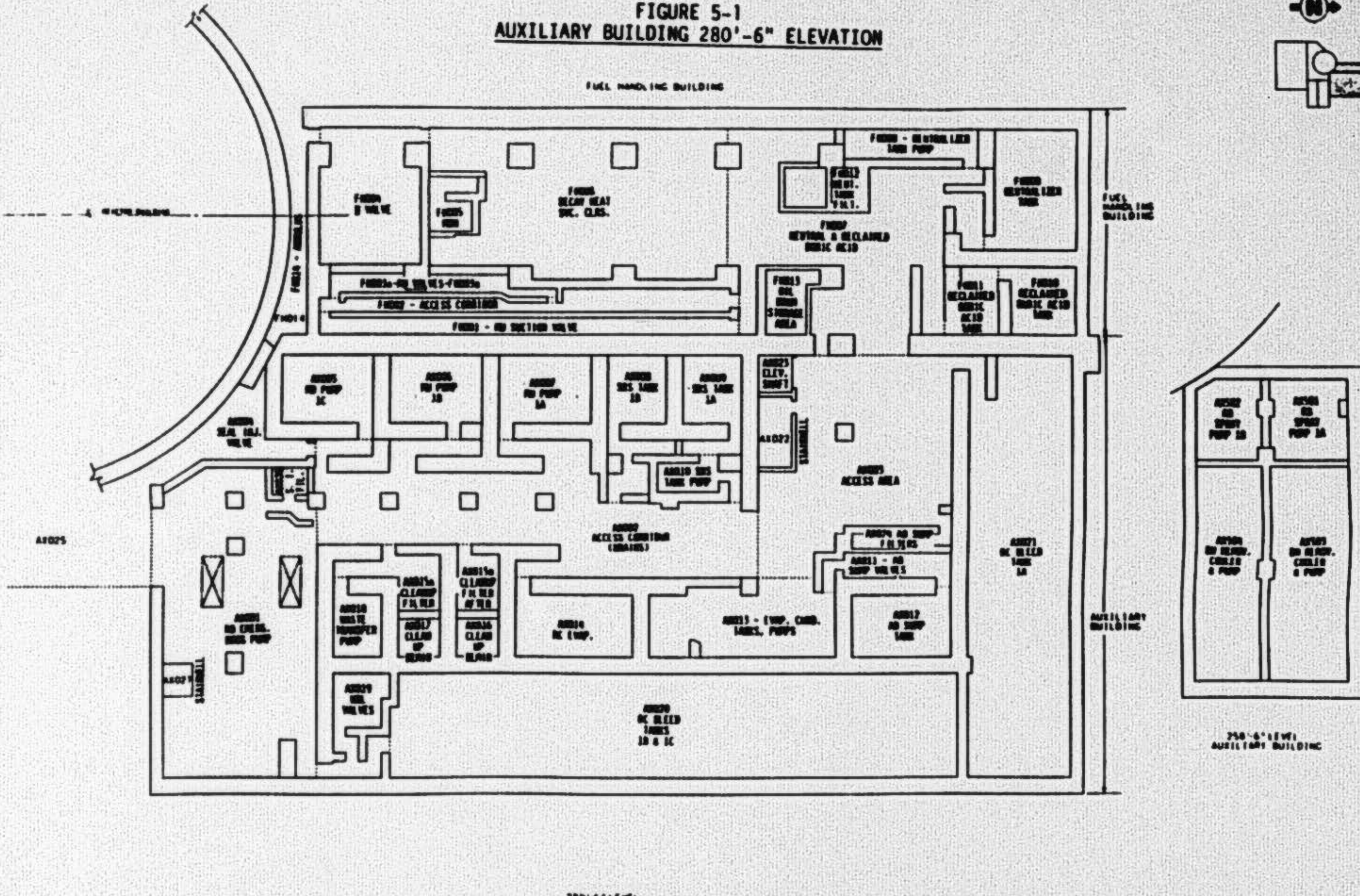
COMPONENT	RESIDUAL FUEL QUANTITY (KG)
RV Head	1.4[b]
RV Plenum	2.1
Fuel Transfer Canal	(a)
Core Flood System	2.4[b]
D-Rings	23
Upper Endfittings	< 54[b]
Reactor Coolant Drain Tank	0.1
Letdown Coolers	< 4 [c]
RB Basement/Sump	1.3
Cleanup Systems/Equipment (e.g. DWCS)	(b)
TOTAL	< 88 kg[b]

(a) - Excluding the RV and RCS.

(b) - To Be Updated in a Subsequent DCR Submittal.

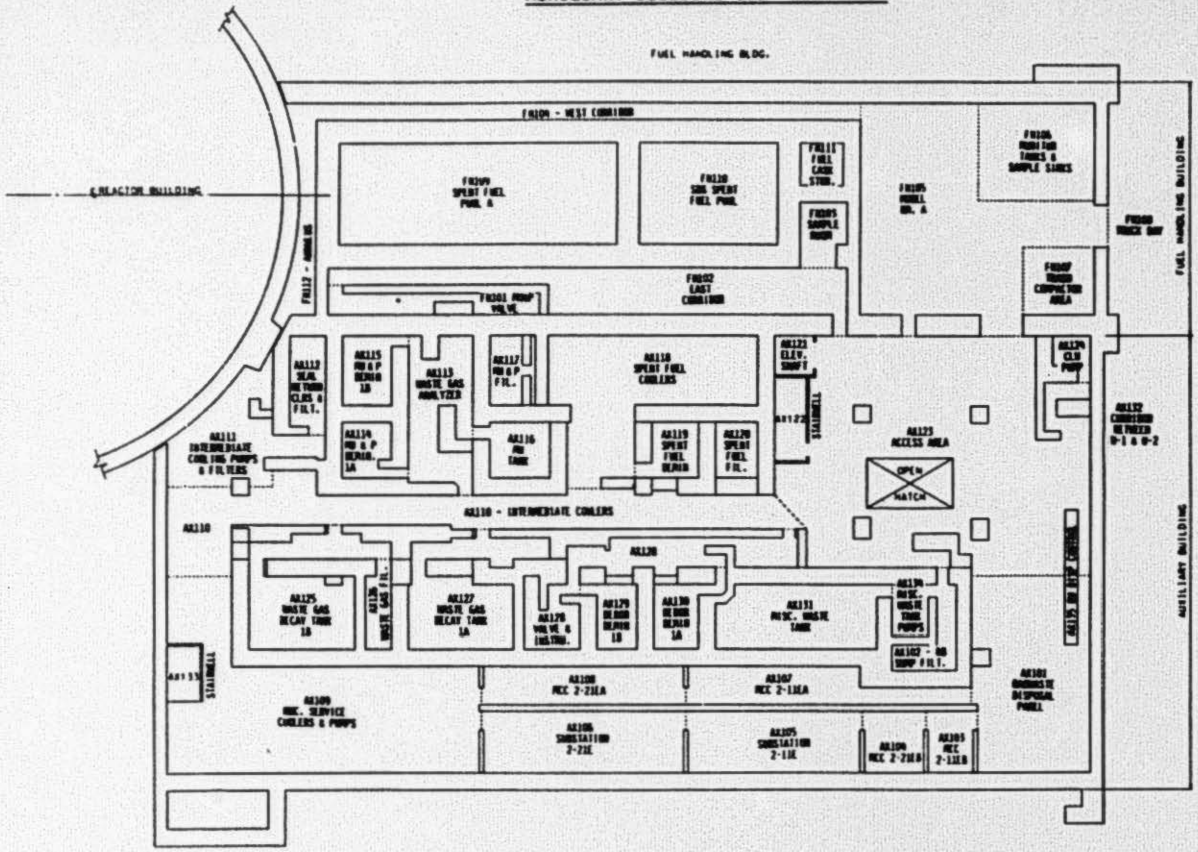
(c) - MDL

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



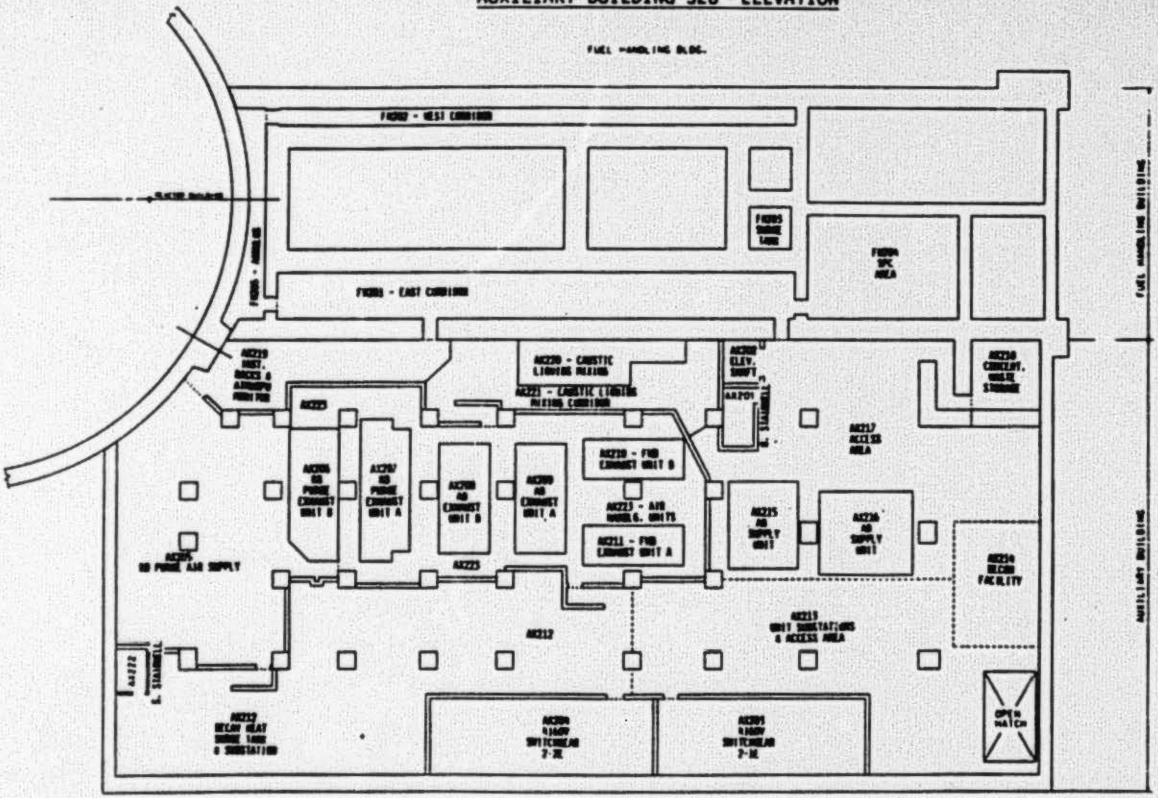
Rev. 1/0461P

FIGURE 5-2
AUXILIARY BUILDING 305' ELEVATION



305' LEVEL
AUXILIARY BLDG.

FIGU -3
AUXILIARY BUILDING 328' ELEVATION



328' LEVEL AUXILIARY BLDG.

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

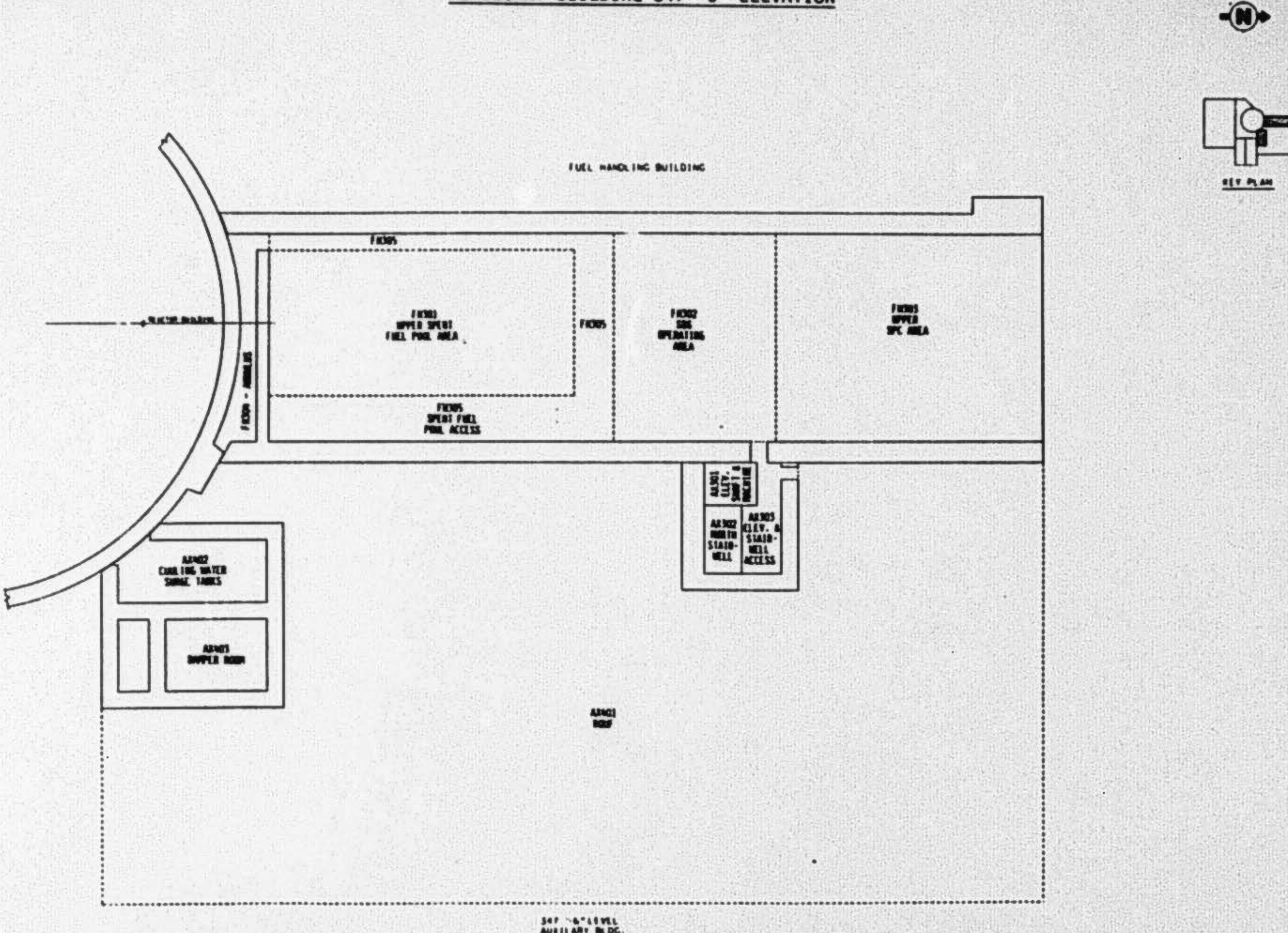


FIGURE 5-5

TMI-2 REACTOR - UPPER HALF
(Simplified)

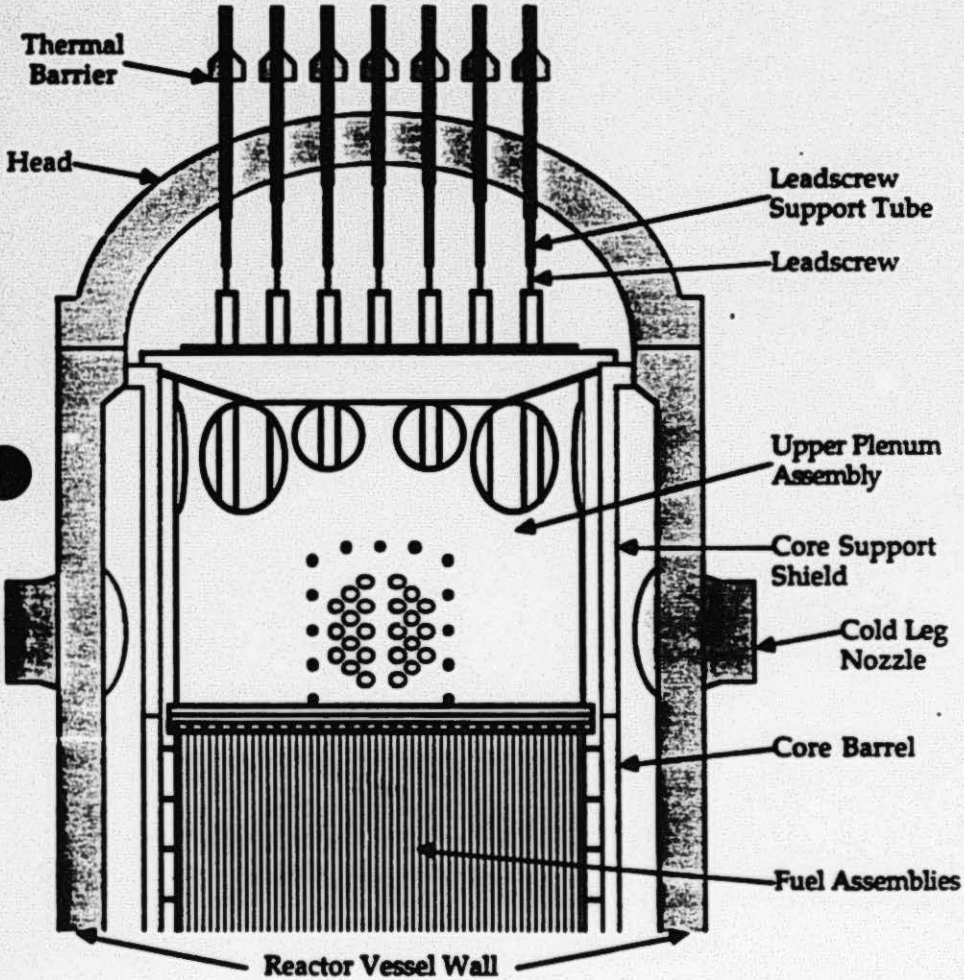
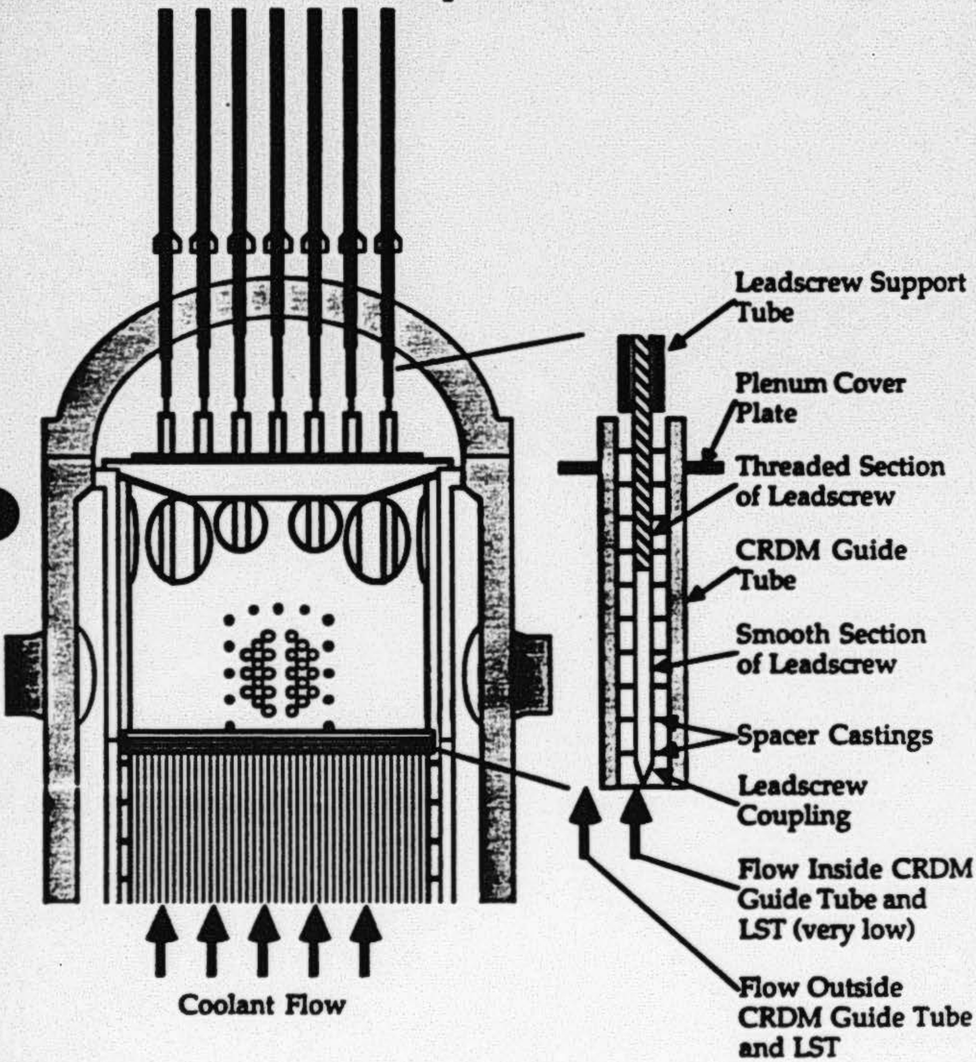


FIGURE 5-6

LEADSCREW AND LS SUPPORT TUBE

(Simplified)



APPENDIX A

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