GPU Nuclear

GPU Nuclear Corporation
Post Office Box 480
Route 441 South
Middletown, Pennsylvania 17057-0191
717 944-7621
TELEX 84-2386
Writer's Direct Dial Number:

(717) 948-8400

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

Three Mile Island Nuclear Station, Unit 2 (TMI-2)
Operating License No. DPR-73
Docket No. 50-320
SNM Accountability

Dear Sir:

By NRC letter dated October 17, 1985, GPU Nuclear was granted exemption from certain requirements for periodic inventory and reporting of the special nuclear materials (SNM) balance for Three Mile Island Unit 2 (TMI-2). As a condition of the exemption, GPU Nuclear is required to conduct an assessment of the SNM remaining at TMI-2 following the completion of the defueling effort. This assessment is referred to in the exemption as the "post-defueling survey." GPU Nuclear letter 4410-88-L-0162 dated September 30, 1988, submitted the initial Post-Defueling Survey Reports (PDSRs).

As stated in that submittal, the PDSR documents the GPU Nuclear assessment of the amount of residual SNM in the various facilities, systems, and components of the plant and describes the methodology utilized to determine the quantity of SNM in each case. The attached PDSR transmits the post-defueling survey results for the Reactor Building Miscellaneous Components.

The remaining PDSRs will be submitted as they are completed. A compilation of the individual PDSRs will form the basis for a final assessment of the quantity of residual SNM at TMI-2 for accountability purposes.

Sincerely,

R. L. Long
Director, Corporate Services/TMI-2

EDS/dlb

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cc: T. T. Martin - Regional Administrator, Region I
M. T. Masnik - Project Manager, PDNP Directorate
L. H. Thonus - Project Manager, TMI
F. I. Young - Senior Resident Inspector, TMI

TMI-2 POST-DEFUELING SURVEY REPORT

FOR

REACTOR BUILDING MISCELLANEOUS COMPONENTS

Approved: _____
Manager, TMI-2 Licensing

Approved: _____
Manager, TMI-2 Engineering

TMI–2 INTERNAL REVIEWS

AUTHORS: F. W. DEININGER

RESPONSIBLE TECHNICAL REVIEWER: M. H. HAGHIGHI

MANAGER, FUEL MEASUREMENTS & ANALYSIS: C. H. DISTENFELD 6/12/91

PDSR COMMITTEE CHAIRMAN: R. E. ROGAN 6/13/91

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SUMMARY

The estimate of record of the amount of uranium dioxide (UO_2) remaining in the RB Miscellaneous Components included in this Post Defueling Survey Report (PDSR) was 64.0 kg with a range from 31.6 to 85.4 kg, distributed approximately as follows:

Incore Guide Tubes (in "A" D-Ring)	21.0 kg UO_2
Fuel Transfer Canal/Transfer Tubes	18.9 kg UO_2
Upper Endfittings	5.9 kg UO_2
Core Flood System	4.9 kg UO_2
Drain Line from Tool Decon Facility	4.4 kg UO_2
Temporary RV Filter System	4.4 kg UO_2
Defueling Water Cleanup System	3.7 kg UO_2
Other Components	0.8 kg UO_2
Total	64.0 kg UO_2

The above summary table shows that 62% of the UO_2 remaining in the RB Miscellaneous Components was located in two (2) components, the Incore Guide Tubes and the Fuel Transfer Canal/Transfer Tubes. The 21.0 kg UO_2 in the Incore Guide Tubes (IGT) stored in the 'A' D-Ring was determined to be tightly compacted inside the 1.6 centimeter ID holes in the upper sections of the IGTs which was not accessible to the brushing tools. A small portion of the 18.9 kg UO_2 remaining in the FTC/Transfer Tubes was fuel debris that was added during drain down of the Reactor Vessel.

The miscellaneous components were measured for fuel content using gross gamma exposure rates, gamma spectroscopy, neutron interrogation and direct sampling and analysis. These methods are described in Section 3.0 and highlighted for each component in Table 1. Several miscellaneous components not addressed in this PDSR are discussed in other PDSRs as listed in Section 1.0, Introduction.

The term "uncertainty" in this Post Defueling Survey Report was used to represent the estimated error of each "estimate of record" and was taken as one sigma. The overall uncertainty for the total residual UO_2 remaining in the RB Miscellaneous Components is expressed as a range from 31.6 kg to 85.4 kg UO_2 because of the asymmetrical value for the Fuel Transfer Canal/Tranfer Tubes.

TMI-2 Post-Defueling Survey Report

for

The Reactor Building Miscellaneous Components

1.0 INTRODUCTION

This report presents the analysis of the amount of fuel (UO_2) remaining in the Reactor Building for miscellaneous components only. The content of this analysis addresses the fuel remaining in the following components/systems.

a. Reactor Coolant Drain Tank
b. Fuel Transfer Canal/Transfer Tubes
c. Core Flood System
d. Upper Endfittings
e. Tool Decon Facility (TDF)
f. Drain Line from TDF
g. Defueling Water Cleanup System
h. Temporary RV Filter System
i. Incore Guide Tubes
j. Defueling Tools

The remaining miscellaneous components in the Reactor Building are addressed in other Post-Defueling Survey Reports as listed below:

Miscellaneous Component	Addressed in PDSR
Rv Head	RV Head Assembly
Plenum	Plenum Assembly
RB Basement	RB Basement
Letdown Coolers	Letdown Cooler Room

This report is one in a series of reports prepared to fulfill the requirements of the TMI-2 SNM Accountability Program (Reference 1). All statistical uncertainties are expressed as ± one sigma limits (defined as one standard deviation) unless otherwise stated. Other segments of the RB are treated in separate Post-Defueling Survey Reports.

Section 2, "Background," describes the physical attributes of the individual components/systems and their relationship to the accident and subsequent cleanup activities. The boundaries of these components/systems with respect to this PDSR are also presented in this section.

Section 3, "Measurement Methods," describes the measurement methods utilized to assay the residual fuel (UO_2) in each of the components/systems. The majority of components/systems were assayed using passive nondestructive assay methods (i.e., gamma spectroscopy, gamma dose rate, or neutron interrogation).

Section 4, "Analysis," explains the methodology for arriving at the estimate of record of fuel (UO_2) in the RB using multiple measurements and sample analysis data and discusses supporting data, assumptions made, and calculations used.

Section 5, "Conclusion," presents the estimate of record and uncertainty for the amount of SNM remaining in the RB and supporting rationale leading to the conclusion that the estimate of record is reasonable based on the available data and analyses performed.

2.0 BACKGROUND

The March 1979 TMI-2 accident resulted in significant damage to the reactor core and subsequent release of fuel particles and fission products into the Reactor Coolant System (RCS) and other closely interconnected systems/components. The damaged core consisted of loose fuel pellets, resolidified fuel, structural metal components, loose rubble, and partial fuel assemblies. Therefore, fuel accountability by the normal method of accounting for individual fuel assemblies was not possible.

During the accident, core debris was transported to the Reactor Building (RB) as a result of core degradation and coolant flow from the reactor vessel (RV) through the pilot operated relief valve (PORV) and the RCS Makeup and Purification (MU&P) System. Approximately 10 kg of fuel (UO_2) was relocated to the RB sump and other RB locations during the accident sequence (Reference 2). 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' core flood tank (CFT) and 'A' D-ring; storage of upper endfittings; flushing of defueling tools; and transfer of the defueling canisters into the fuel transfer canal (FTC). Even though fuel was relocated to the RB during cleanup operations, RB residual fuel conditions were maintained significantly below the safe fuel mass limit (SFML), Reference 3. Further, a significant cleanup effort was undertaken with the primary purpose of reducing exposure rates but which also resulted in the removal of additional core debris.

2.1 Reactor Coolant Drain Tank

The reactor coolant drain tank (RCDT) was designed to absorb blowdown from the pressurizer relief valves and receive leakage from the RC pump seals and RC valves. The discharge from the RCDT was

designed to flow through a cooler to the reactor coolant bleed tanks (RCBT). The RCDT normally had a nitrogen cover gas and was vented to the RCBTs. The tank was protected by a 150 psig relief valve and a 200 psig rupture disk.

During the accident, reactor coolant was discharged from the RCS through the PORV to the RCDT which was located in the basement of the Reactor Building. The RCDT contained two (2) safety components: a relief valve which discharged to the RB sump and a rupture disk which discharged 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. During the accident, the operators realized that the PORV was not closed and they manually closed the pressurizer block valve. 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, PORV and block valve into the RCDT and subsequently into the reactor building through the rupture disk discharge.

2.2 Fuel Transfer Canal/Transfer Tubes

A small amount of fuel resides at the bottom of the fuel transfer canal (FTC), having been transported from the RV to the FTC as debris adherent to the outside of the fuel bearing canisters. Additional fuel was deposited in the FTC as a result of the placement and storage of the upper plenum assembly in the deep end of the FTC. 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 the form of sediment and surface films on the plenum surfaces. Prior to its

removal from the RV, the plenum was flushed to remove loose surface debris. The fuel content of the plenum was reported in Reference 4 and will not be included here. Additional fuel deposits are contained with the endfittings stored in a temporary storage container (TSC #3) which is stored in the FTC.

During drain down of the RV and connecting systems/components, water will be transferred to the deep end of the FTC. The FTC will act as a settling volume to collect any fuel fines removed during drain down. The RV will be pumped down in multiple phases. The first phase will be to drain the water from the RV using a jet pump. The jet pump's suction will be located in the area where the H-8 boat sample was removed during the RV sampling program. Then the RV will be refilled using water from the RCBTs. During drain down and refilling, the RV will be measured for remaining fuel (UO_2). After the fuel measurements have been completed, draining of the connected systems/components will begin. The RV will be drained to approximately half-full and then the drain pumps in the steam generators will be turned on so that they are draining to the RV. During the draindown of the connected systems/components, the RV water level will be maintained well below the nozzle level (elevation 311') to prevent water from flowing back into the cold legs via the nozzles.

The pumping rate in and out of the RV will be approximately 10 gpm. Once both steam generators have been drained, the RV will be drained. All of the water removed from the steam generators and RV will be pumped directly to the FTC. During the drain down, it will be necessary to continuously remove water from the FTC to prevent the FTC from overflowing. The water removed from the FTC will be pumped to a RCBT. The drain line from the RV to the FTC will be located at least fifteen (15) feet away from the suction line to the

RCBTs to prevent carry over of fuel fines.

2.3 Core Flood System

Two (2) core flood tanks (CFT) were installed as part of the original plant design as a passive low pressure injection system in the event of a loss of coolant accident. They were designed to inject borated water directly into the RV when the RCS pressure dropped below a predetermined level. Approximately eight (8) hours into the accident, the core flood tanks injected some water into the RV. However, twelve (12) minutes later, a high level alarm indicated that the core flood tanks were receiving water from the RCS even though a check valve was installed to prevent such an occurrence (Reference 2). Fuel fines could have been transported into the core flood system by this backflow. A more likely means of fuel transport took place after the initial core damage. The water level in the RV is believed to have dropped below the core flood nozzles on two (2) separate occasions, thereby draining the horizontal lines. It is likely that fuel was transported into these lines by the fluid expansions resulting from the high energy releases inside the RV or by the refilling operations.

During the lower core support assembly (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 prevented the possibility of further fuel transport. Storage of the LCSA components in the 'A' CFT permitted continuous progress in the RV defueling activities. Prior to removal from the RV, the LCSA segments were 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 adherent residual fuel.

2.4 Upper Endfittings

In September 1986, the core bore machine was installed in the RV to break up the large resolidified mass so that it could be removed and placed in fuel canisters. Loose upper endfittings that would interfere with drilling operations were removed from the surface of the debris bed. In some cases, several endfittings had been fused together and were too large to fit into the fuel canisters; they were placed in shielded drums filled with borated water and stored at elevation 347' in the reactor building near the RV head stand.

Currently, there are approximately seventeen (17) upper endfittings stored in six (6) containers. The maximum number of endfittings in a single container is four (4) (See Section 4.4).

2.5 Tool Decontamination Facility

The tool decontamination facility (TDF) was located on the 347' elevation of the RB and consisted of two (2) 12 foot square enclosures connected by an 8 foot square anteroom. One enclosure was used for high pressure/high temperature flushing of defueling tools; the other enclosure was used for cutting up the defueling tools. Because of the extensive use of this facility, residual fuel deposits remain in the form of surface contamination.

2.6 RB Drain Line From TDF

The RB basement boundary was taken to include all space below the 305' elevation with one exception. The exception is the RB drain line that was used to transfer defueling tool decontamination wash water to the basement. A separate Post Defueling Survey Report has

been submitted for the RB basement and determined the fuel Estimate of Record for the RB basement at 1.3 kg UO_2. The discharge path from the tool decontamination facility located on the 347' elevation of the RB is from the decon sink to the floor drain located within the decon facility. The discharge piping from the floor drain passes through the 347' elevation floor, turns nearly horizontal for approximately ten (10) feet and then is essentially vertical for approximately fifty-five (55) feet to a long horizontal flooded drain line under the 282' elevation basement floor (Reference 5). More than a dozen basement floor drains empty into the line. The residual fuel exists in the line as small pebbles as opposed to the colloidal particles that were transported to the RB sump (Reference 2). It is noteworthy that the defueling tools were flushed in the RV prior to being removed to minimize fuel transport.

2.7 Defueling Water Cleanup System

The defueling water cleanup system (DWCS) located in the RB was composed of three (3) parts: the interconnecting hoses, the manifold assembly, and the RV cleanup pumps located in the south end of the FTC canal. All of the filters have been removed and shipped off-site. The interconnecting hoses consisted of two (2) similar assemblies labeled train 'A' and 'B'. In aggregate the trains of reinforced rubber hose consisted of approximately twenty-eight hundred (2800) feet of nominal two (2)-inch diameter hose; fourteen hundred (1400) feet of nominal three (3)-inch diameter hose; and fourteen hundred (1400) feet of nominal four (4)-inch diameter hose (Reference 6). The fuel content was determined by direct gamma spectrometry of representative hose samples and extensive directional detector gross gamma scans for a larger sample.

The manifold assembly was about twenty-one (21) feet long and was

located along the north end of the FTC. It was composed of a complex of hard piping, valves and other components that interfaced between the RV and filter canisters to provide the cleaning action. Since the filter canisters have been removed, there is no allowance for fuel residual to the filters. Therefore, the entire manifold assembly was in the field of view of a directional detector and shielded only by individual pipes and hoses (Reference 6).

The third section of the DWCS was two (2) RV cleanup pumps. The two pumps contained suction lines from the RV and discharge lines to the manifold assembly. The pumps were located under water in the former fuel storage pit near the southeastern corner of the FTC. The pumps were emersion type, designed to fit inside vertical 10 inch steel casings. The suction side of the casings were terminated by 10 inch caps that were continued by schedule 40 pipe sections welded to the cap centerline. The short downgoing 4 inch schedule 40 pipe section became horizontal for 18 inches and then extended vertically up and out of the fuel storage pit. The entire DWC system was utilized over several years while defueling was in progress (Reference 5).

2.8 Temporary RV Filtration System

The function of the Temporary RV Filtration System (TRVFS) was to restore and maintain the visibility in the RV to allow continuation of the early defueling program. Investigation had lead to the discovery of micro-organism growth in the reactor coolant. Operating experience revealed that these micro-organisms were capable of plugging the filters prior to the collection of any significant quantity of core debris. The TRVFS system was operated as a temporary filter system while GPU developed a permanent program to control this phenomenon.

The TRVF System consisted of a pump, 1 1/2-inch diameter suction and discharge hoses, isolation valves, fittings, and filter assemblies. The filter assemblies were commercial type diatomaceous earth-filled swimming pool filters. The suction depth was controlled by fixed lengths of solid piping. The TRVFS was operated only when operations personnel were on the defueling platform. The TRVFS took suction from the IIF and/or the RV and returned the filtered water to the IIF. The initial flow rate through the filter was approximately 100 gpm. When the pressure drop across the filter reached a predetermined level (10 psi), the pump could no longer provide significant flow (Reference 7). Consequently, the filter was "backbumped" or cleaned. Backbumping was accomplished by stopping the flow and flexing the filter media which caused the filtered material and diatomaceous earth to fall to the bottom of the filter housing. The TRVFS was eventually replaced by the Defueling Water Cleanup System which is described in Section 2.7. The two (2) highly radioactive swimming pool filters are currently stored in the RB on elevation 305'.

2.9 Incore Nozzles

Sections of the flow distributor removed from the RV, which contained incore guide tubes, were too large to be placed in the 'A' CFT. Thirteen (13) such sections, containing 33 incore guide tubes, were stored in the 'A' D-ring. These sections were bagged and suspended in the 'A' D-ring in order to prevent interference with the RV defueling efforts. Each section was brushed and flushed prior to removal from the RV.

The thirteen (13) sections of the flow distributor stored in the 'A' D-ring, with incore guide tubes attached, generally consist of a one (1) to two (2) foot rectangularly shaped piece of the flow

distributor with up to four attached incore guide tubes of up to forty-seven (47) inches in length. The majority of the residual fuel associated with these sections was determined to be tightly compacted inside the 5/8-inch inside diameter holes in the upper section of the incore guide tubes and not accessible to the brush tools.

2.10 Defueling Tools

Typical defueling tools consisted of stainless steel pipe sections two (2) to four (4) inches in diameter and thirty (30) to forty (40) feet long with an end effector attached. For the purpose of this assessment, the defueling tools were modeled using Microshield (Reference 8). The inside surfaces of the pipe were assumed to be open to the internal environment of the reactor vessel and contain fuel debris. Routine practice required the working end of each tool removed from the RV be isolated in a plastic bag. This contamination control technique also prevented the spread of fuel bearing material to the tool racks or to surfaces below the tool storage areas. The radiation environment in proximity to the rack containing the most contaminated tools was not distinguishable from normal values above the 'B' D-ring.

3.0 MEASUREMENT METHODS

Post-defueling SNM measurement in the TMI-2 Reactor Building was a complex task. Several different methods were used to locate and quantify residual fuel. These methods included direct measurement by instrumentation, visual inspection, and sample collection and analysis. The methods selected were influenced by many factors including accessibility, configuration of component/area to be assayed, area radiation dose rates, measurement uncertainties, and equipment sensitivity.

The following includes a discussion of the various methods and the factors that influenced their selection. Five (5) general methods were used for fuel detection: detection of gamma rays, neutrons, alpha particles, sample and analysis, and visual inspection. Each detection method included a number of specific techniques that are described below.

3.1 Gamma Dose Rate

Gamma detection for fuel measurement included the use of the gross gamma exposure rates and gamma spectroscopy techniques. Gross gamma directional surveys were performed on areas, piping and components when exposure rates and accessibility permitted. The measurement system utilized a portable directional gamma survey instrument consisting of a directional probe and readout device. TMI-2 core debris contained fission and activation products which produced gamma-ray exposures. Exposure rate values were calculated based on the isotopic distribution of sample data. Using the reactor core debris sampling data, a correlation was developed between the gross gamma-ray output and the quantity of fuel present (Reference 9). Computer modeling was used to predict exposure rates for a given geometry of fuel for a particular location. Field measurements were performed for specific locations and compared with the computer

modeled output. This comparison was done on a point by point basis. Scaling the models with the measured exposure rates yielded an estimate of residual fuel. Three (3) areas/components (decon facility, DWCS-manifold and fuel transfer canal/tubes) were analyzed using gross gamma directional surveys.

3.2 Gamma-Ray Spectroscopy

Gamma-ray 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 a high-purity germanium (HPGe) detector. 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-ray radiation. Calibrated gamma-ray spectroscopy systems were used at TMI-2 to measure the quantity of Ce-144(Pr-144) and/or Eu-154 present in discrete locations or on components. The quantity of cerium or europium present was converted to the quantity of fuel present based upon the calculated ratios (Reference 10) and the actual measurements of the cerium/fuel and europium/fuel ratios.

The HPGe detector measurements were performed on the defueling tools, DWCS hoses, LCSA sections stored in the 'A' Core Flood Tank and incore guide tubes stored in the 'A' D-ring. The HPGe detectors are approximately 1 3/4" diameter x 1 3/4" long while the NaI detectors are approximately 1/2" diameter x 3/4" long. HPGe detectors have the advantage of much better energy resolution capability due to their relative size, than sodium iodide (NaI) detectors. In addition, they are much more sensitive to ambient gamma radiation levels. However, HPGe detectors require liquid nitrogen cooling to operate. HPGe detector measurements were

performed to identify both Ce-144/Pr-144 (2.19 MeV gamma radiation) and Eu-154 (1.27 MeV gamma radiation). The Ce-144/Pr-144-and Eu-154-to-fuel (UO_2) ratios were 152.5 and 42.6 μCi g^{-1}, respectively as of 8/1/87 (Reference 10).

The HPGe detector was housed in a 2-inch thick cylindrical shell lead shield. The detector was connected to a preamplifier, amplifier, and multi-channel analyzer (MCA) as described in Reference 11. The MCA and input-output device were at a remote location; as much as 100 feet of signal cable was connected between the preamplifier and spectroscopy system . The detector system was source checked periodically to ensure proper operation using standard Ce-144 and Eu-154 sources. After the calibration data had been collected, the standard sources were removed and the acquisition of data was performed.

3.3 Neutron Interrogation

Neutron emission from the induced fission reactions is directly proportional to fuel quantity. This active neutron assay method interrogates fuel with lower energy neutrons and detects induced higher energy fission neutrons.

Active neutron interrogation is more sensitive than some passive methods for quantifying small deposits of fuel. At TMI-2, an antimony-beryllium (Sb-Be) photoneutron interrogation method used an antimony photon source of 1.7 MeV gamma-rays to produce low-energy (approximately 0.024 MeV) interrogating neutrons via the Be (γ,n) reaction. These interrogating neutrons which are moderated by the beryllium ring (and polyethylene surrounding the Be-ring) impinged upon the fuel and induced fission reactions in the fissile material contained in the fuel. Some of the fission neutrons emitted from

the surrounding fuel were detected by a He-4 fast neutron proportional counter. The He-4 neutron counter system differentiated the higher-energy induced fission neutrons from the lower-energy photoneutron source and gamma-rays on the basis of the pulse height signal. With proper shielding, this system can operate effectively and efficiently in a substantial radiation field.

3.4 Direct Sampling and Analysis

Two types of samples were used at TMI-2 for residual fuel determinations: core debris and RCS components. Samples of core debris from fixed locations were analyzed to determine fuel and radionuclide content. Samples were analyzed by gamma spectroscopy, alpha counting, and chemical/physical techniques. Estimates of the total debris volume of concern were developed using visual aids and radiological models were incorporated into the analytical results to derive specific fuel quantities. Samples of RCS components were used to estimate the density of fuel fixed in surface films. Representative samples of various core debris deposits were extrapolated to represent deposits from the reactor vessel. The distant material has greater potential gamma activity per unit fuel weight. Because of the uncertainty in the sample representation, it was preferable to use sampling techniques in conjunction with other methods that measure fuel directly.

Two (2) areas/components (temporary reactor vessel filter system and the RC drain tank) were assayed using direct sampling and analysis.

4.0 ANALYSIS

The Reactor Building (RB) miscellaneous areas were measured for fuel (UO_2) by performing a series of non-destructive measurements described in Section 3.0, Measurement Methods. The estimate of record for each component/system was determined by performing gamma detection measurements or neutron interrogation measurements and/or engineering analyses. Engineering analyses were performed in areas where the total fuel quantity was believed to be insignificant. An insignificant quantity was defined as about 7 kg of UO_2 (Reference 1) or approximately 8% of the maximum safe fuel mass (Reference 3) for fuel removed from or remaining in the reactor vessel.

All assessment measurements were performed using a direct gamma reading or counting technique and are supported by formal engineering calculations containing data sheets, equipment calibration information and essential supporting information as well as data reduction. It is assumed, unless otherwise stated, that fuel estimates based on existing references are correct within the uncertainties stated. The result of these fuel assessments was to provide an "estimate of record" of residual fuel (UO_2) for the miscellaneous areas in the Reactor Building.

Two (2) major components contained 62% of the residual fuel in the RB Miscellaneous Components. They are the Fuel Transfer Canal/Transfer Tubes and the Incore Guide Tubes stored in the 'A' D-Ring. Other miscellaneous components contained quantities of fuel as listed in Table 2. The analysis for each component is discussed individually below.

The term "uncertainty" in this Post-Defueling Survey Report is used to represent the uncertainty of each "estimate of record" and is taken to be one sigma. The uncertainty was determined by the project engineer for each individual measurement and appropriate engineering calculations. The uncertainty value was derived by taking into consideration uncertainties in

mathematical modeling, measurement equipment, counting, standard values (fissionable material), randomness, and uncertainty in background levels. The range of uncertainty for the total residual UO_2 remaining in the RB Miscellaneous Components was determined using the asymmetrical term for the fuel transfer canal/transfer tubes and combining its range algebraically with the fuel quantity (45.1 kg UO_2 ±32%) estimated for the symmetrical terms. The parts leading to the overall uncertainty were considered for possible covarient uncertainty. Each ingredient is considered to be independent.

This results in an estimate of record of 64.0 kg UO_2 for all the RB Miscellaneous Components as shown in Table 2. After evaporation of the accident generated water stored in the components, the potential for fuel mobility will be greatly reduced because only small batches of water will accumulate and be processed during the post defueling storage period.

4.1 Reactor Coolant Drain Tank

The west end of the Reactor Coolant Drain Tank (RCDT) was visually inspected by lowering a camera from the 305' elevation in the RB. A four (4)-inch diameter hole was drilled in the concrete floor (elevation 305') and in the elbow of the eighteen (18)-inch vent line directly below. A small cylindrical underwater black and white camera was lowered into the RCDT. Because of limitations inherent in the equipment used to perform the video inspection, only the west end of the tank was examined. A sediment sampler was also used to collect samples of liquid and particulate matter from the inside bottom surface of the RCDT, directly beneath the rupture disk and vertical section of the vent line. Additional details are discussed in Reference 12.

Video surveys of the inside of the RCDT showed that the layer of

debris was quite thin in the area beneath the rupture disk. For the purpose of this PDSR, the debris on the bottom surface of the tank was assumed to be 0.16 cm thick and to uniformly cover one-eighth of the inner surface of the tank from end to end. The area of the deposited debris was about 52,000 cm^2, and its total volume was approximately 8320 cm^3. Based on the elemental analysis results for the RCDT solids sample presented in Reference 13 (the elements listed collectively account for about 94% of the sample weight), the density of the debris was about 6.2 g/cm^3. If we assume that one-half of the total debris volume was water for a packing fraction of 0.5, then the total mass of debris in the RCDT was about 26 kg (8.3×10^3 cm^3 x 6.2 $g/cm^3 \div 2$). See Reference 13 for details of results.

The microprobe analyses performed on the RCDT samples did not definitely identify uranium, although cerium, which is generally associated with uranium, was detected in each sample using gamma-ray spectroscopy measurement techniques. Neutron activation analyses subsequently performed on the RCDT solids quantified the uranium in the samples. The concentration of uranium measured through neutron activation analysis was 3.7 mg/g. The total fuel content of the RCDT was estimated as (26,000 gm) (3.7 mg/g) (1.1345 UO_2/U ratio) = 0.1 kg UO_2.

The uncertainty is dominated by two components. The lessor uncertainty is due to the radiochemical fuel analysis of the RCDT sample. EG&G Idaho and the Westinghouse Hanford Engineering Development Laboratory independently analyzed the sample. A reasonable uncertainty of 20% (Reference 14) was adopted for the fuel assay since specific error values were not published. The larger uncertainty is assigned to the determination of sediment volume lying on the bottom of the RCDT. Previously, a video estimate was performed on the upper tube sheet of the "B" Once Through Steam Generator,

Reference 15. Subsequently the work was reassessed by gross gamma methods, Reference 16. Generally the video method produced a result that was less accurate than the gross gamma method. Since the radiochemical and video uncertainties are independent, the overall uncertainty was propagated as the product of volume and concentration as the square root of the sum of the squares of the fractional errors. The overall uncertainty is 54%.

4.2 Fuel Transfer Canal/Transfer Tubes

During the TMI-2 Cleanup Program, fuel canisters were transferred from the RV through the FTC/transfer tubes for temporary storage in the Fuel Storage Pool A in the Auxiliary Building. The residual fuel debris in the FTC came from six (6) different sources. A major source of fuel debris was the fuel canister transfer when they were lowered into the FTC and rotated from vertical to horizontal for transfer to the Spent Fuel Pool A (SFPA). Three hundred and one (301) fuel canisters were transferred through the FTC prior to measurement of the FTC (Reference 17). The term fuel canister is used in the all inclusive sense; i.e., "fuel", "knockout" and "filter" canisters. Underwater gross gamma measurements were performed in the FTC during August and September 1989. Also a video inspection was made before and during these measurements to determine the distribution patterns for the residual fuel debris. The video inspection disclosed that there were not any areas where a significant depth of debris could be seen and in fact many areas appeared relatively clean. The measured fuel (UO_2) quantity of 12.2 kg used a Cs-137 activity to fuel ratio of 1800 μCi/g UO_2. Since the FTC measurement was performed in 1990, the new Cs-137 activity to fuel ratio value of 1355 μCi/g UO_2 (Reference 9) was determined which changes the FTC fuel uncertainty to +24%, -100%. The result of this improved Cs-137 fuel activity is that the FTC initially is estimated to contain 12.2 kg UO_2 +24%, -100%. The

basic measurement data and details of the data analyses are contained in Reference 18. Since the measured fuel quantity was the result of transferring three hundred and one (301) canisters, it was reasonable to assume that the increase in residual fuel can be prorated for the last forty-one (41) canisters (Reference 19) transferred after the measurements were taken. This additional fuel deposit was estimated by multiplying the last forty-one (41) canisters times the average fuel per canister transferred with the initial three hundred and one (301) canisters.

Average fuel/canisters = (12.2 kg) ÷ (301) = 0.041 kg UO_2/canister
Additional fuel deposited = (0.041 kg) (41 canisters) = 1.7 kg UO_2

Therefore, the total estimated fuel (UO_2) in the FTC as a result of transferring canisters to SFPA was 12.2 kg + 1.7 kg = 13.9 kg, +24%, -100%.

Another contributor to the fuel (UO_2) remaining in the FTC was the fuel debris remaining in the fuel transfer tubes which connect the FTC to the SFPA. The fuel transfer tubes were assumed to have fuel debris deposited in the lower sixty (60) degrees of arc of each tube because the tubes are basically horizontal. Reference 18 determined the residual fuel (UO_2) deposited in the transfer tubes as 0.5 kg, +34%, -92%.

A third contributor to the fuel (UO_2) remaining in the FTC was a storage container (designated TSC) containing three (3) endfittings from the original fuel assemblies. The fuel debris contents of this storage container is included in the total fuel estimate for the Upper Endfittings discussed in Section 4.4. Even though the container itself is stored in the deep end of the FTC, it seems appropriate to include its fuel estimate along with the other endfittings stored on

the 347' elevation in the Reactor Building.

A fourth contributor to the fuel (UO_2) remaining in the FTC was the upper reactor Plenum Assembly. Based on the video inspections (which saw no debris) and the fact that the plenum was flushed prior to placing it in the FTC, the plenum was assumed to contain less than 0.1 kg of fuel (UO_2) in the form of loose debris or rubble. Reference 20 describes the analysis utilized and assumptions made to estimate the quantity of fuel (UO_2) remaining on the plenum assembly. The fuel remaining on the plenum assembly is not included in the total fuel (UO_2) estimated for the FTC because it is addressed in a separate Post Defueling Survey Report (Reference 4).

A fifth contributor to the fuel (UO_2) remaining in the FTC was attributed to approximately seven-hundred (700) feet of three (3) different sizes of rubber hose (Reference 21 and 22) that were installed between the DWCS manifold and the filter canisters installed in the deep end of the FTC. This seven-hundred (700) feet of hose was removed from the work platform and placed in the FTC. The DWCS Engineering Calculation, Reference 6, summarized the fuel concentration that was measured on sample lengths of hose surfaces exposed to unfiltered RV water. The average fuel mass (UO_2) for several lengths of sample hoses was 1.5E-5 kg ±100% for approximately 486 square centimeters of inner surface area. Applying this average value to the seven-hundred (700) feet of hose (376999 cm^2 of inner surface) results in a total estimated fuel deposit of 0.01 kg ±100%.

The sixth and final contributor to the fuel (UO_2) remaining in the FTC is the fuel debris (UO_2) being transferred to the FTC during drain down of the RV and connecting systems/components.

Following the completion of pick and place activities in the lower

head region of the RV, the lower head was vacuumed to minimize the relocation of core debris to other surfaces in the vessel and to improve the visibility in the RV. The in-vessel vacuum system, utilizing a knockout canister and filter canister in series, was used for this evolution. Following the NRC-sponsored Lower Head Sampling Program, a final cleanup of the lower head region was performed during March 1990, and a revised estimate was determined. There appeared to be a fine dusting of material distributed over the entire bottom head surface except for two patches where granular material was somewhat deeper (an average of 0.3 cm deep). Based on this information, it was estimated that the remaining fuel debris in the lower head region was 8.1 kg UO_2 (Reference 23).

In order to measure the RV for remaining fuel (UO_2) it will be necessary to completely drain and refill the RV while making active and passive neutron measurements. It is estimated that most of the fuel debris transferred from the RV to the FTC during drain down will occur while the jet pump is draining the RV from level 8 (forging location) to the bottom of the lower head. Each time the RV is drained a quantity of core debris will be transferred to the FTC. It is necessary to perform this drain sequence at least once (1) during the RV fuel measurement program and again during the final drain down sequence.

Once the jet pump is installed in the lower head, the pump suction will be close enough to the bottom surface and the water velocity will be high enough to remove a quantity of the fine debris which is located near the pump intake port.

During the first drain down cycle, it is estimated that approximately 1.1 kg of UO_2 will be transferred from the RV to the FTC. This value, approximately 13% of the loose fuel on the lower head, was determined

using the volume of core debris existing within the range of flow velocities greater than 0.3 centimeters per second. This assumes that the pump picks up all particles uniformly. See Appendix A for definition of the area from which pump suction removes fuel debris. This is an area surrounding the jet pump approximately ten (10) inches distant from the intake port. It also assumes that the debris is initially 0.2 centimeters deep. Using an average debris density of 4.4 g/cm^3 (Reference 24), and an average UO_2 debris ratio of 0.6 (Reference 25), the total mass of UO_2 transferred from the RV to the FTC is estimated as 1.1 kg UO_2 (25.4 cm x 25.4 cm x π x 0.2 cm x 4.4 gm/cm^3 x 0.6).

After the vessel has been completely drained the first time, it will be refilled using water from the RCBTs via the 'B' makeup line (MU-V-16B). During refilling, it will be necessary to stop at discrete locations to perform a series of fuel measurements. This filling operation will tend to wash most of the residual fuel out of the '1B' cold leg and nozzle. The '1B' cold leg and nozzle were estimated (see discussion below) to contain approximately 26 kg UO_2. In addition, the filling action will tend to slosh water through a section of the downcomer region which will tend to move small quantities of available loose, fine debris from the horizontal surfaces down to the lower head. The velocity of the water inside the Core Support Assembly during refilling is estimated to be less than 0.04 cm/sec (100 GPM) and therefore, very little of the remaining fuel debris will be relocated inside the Core Support Assembly. Prior to performing the neutron measurements in the RV, the estimated quantity of loose fine debris in the lower Downcomer/Core Support Assembly region was approximately 18.1 kg UO_2 emerging from the thermal shield support blocks (15.2 kg) and core catchers (2.9 kg) - Reference 23. It is not unreasonable to assume due to localized higher water velocities that approximately 25% of the 18.1 kg or 4.5 kg UO_2 of loose debris will be

redistributed during the first complete filling of the RV. This is the first time since the accident that the RV will be completely drained and refilled, and a small amount of sloshing in a limited area will take place during filling.

The '1B' cold leg and nozzle contain a significant quantity of fuel debris which could be relocated during refilling of the RV. The '1B' cold leg was estimated to contain 2 kg UO_2 (Reference 26). The nozzle connected to the '1B' cold leg contains approximately 24 kg UO_2 which was estimated based on its volume of fuel debris (13,000 cm^3 per Reference 27), the density of the fuel debris (3.2 g/cm^3 per Reference 26) and the UO_2-to-fuel debris radio (58% per Reference 25). The product of the three (3) factors above is 24.1 kg UO_2. The fuel estimate for the '1B' cold leg and nozzle quantities is 26 kg UO_2. Combining the 26 kg in the '1B' cold leg and nozzle with the 4.5 kg from the thermal shield support blocks and core catchers (downcomer region), and the 7.0 kg remaining on the lower head, implies that approximately 38 kg UO_2 of loose debris could be relocated into the lower head region during the RV refilling sequence.

After the neutron measurements have been completed, it may be necessary to drain the RV a second time. During this second draining, it is estimated that a small amount of the loose fine debris relocated during refilling of the RV could be pumped to the FTC. Since the RV drain pump sucks from an area equivalent to approximately 3% of the lower head area and the lower head is sloped towards the drain pump, it would be reasonable to assume that 5% (1.9 kg UO_2) of the relocated fuel could be transferred to the FTC during the second drain down of the RV.

The next sequence will be to drain the OTSGs to the RV. It is assumed that a portion of the residual fuel in the OTSGs will be pumped to the

RV and consequently a portion of that fuel material will be transferred to the FTC. It will be necessary to partially drain the RV periodically while the OTSGs are being drained to the RV. During this sequential draining, it is assumed that loose fuel debris will be relocated within the RV. In addition, fuel fines coming from the OTSGs will settle out on horizontal surfaces in the RV and the lower head region.

The data collected during the Once Through Steam Generator (OTSG) measurement program was evaluated and fuel quantities in the 'A and B' OTSG lower heads and J-legs were estimated. This was accomplished by comparing the actual in situ exposure rates to calculated rates for known activity concentrations modeled with the Microshield Computer Program (Reference 8). The debris observed in both OTSGs was very low density material and was easily suspended while moving the video probe. This low density material was uniformly distributed except for a few areas which contained gravel like debris which corresponded to the highest exposure rates. The resulting fuel estimates prior to drain down for the 'A' OTSG lower head and J-legs were 0.29 kg and 0.67 kg respectively, and for the 'B' OTSG lower head and J-legs were 0.46 kg and 5.79 kg respectively (Reference 28). Even though the original fuel estimate for the 'A' OTSG lower head may increase significantly due to the new fluff sample data, the fuel transferred to the RV would be relatively small, significantly less than seven (7) kilograms.

The OTSGs will be drained using a TMI designed low flow (<1 gpm) pump to reduce the amount of core debris transferred during drain down. Three (3) pumps will be installed in each OTSG, one (1) in each J-leg and one (1) in the lower head of each OTSG. During a recent (February 1991) J-leg sampling program which used the actual drain pumps, it was observed that the drain pumps quickly picked up the fine debris

surrounding the intake port and then the effluent became relatively clear. It is estimated that approximately 1.0 kg of fuel (UO_2) will be transferred from the OTSGs to the RV. This estimate assumes there is core debris within the range of the suction flow, approximately six (6) centimeters away. It also assumes that the debris is approximately one (1) centimeter deep (estimated volume = $\pi(6)^2(1)=113$ cm^3). Using an approximate debris density of 6 g/cm^3, a packing fraction of 0.5 and an average UO_2 to debris ratio of 0.5 recently estimated for the J-leg fluff samples (Reference 29), the total mass of UO_2 transferred from the OTSGs to the RV is estimated as 1 kg UO_2 (6 pumps x 113 cm^3 volume/pump x 6 g/cm^3 x 0.5 x 0.5)

The next task is to estimate the quantity of fuel debris that will be transferred from the RV to the FTC during the sequential drain downs of the RV. While the OTSGs are being drained to the RV, it will be necessary to periodically drain the RV to the FTC to prevent the RV water level from rising too high. The water level in the RV needs to be kept well below the nozzles to prevent any chance of recirculation to the OTSGs. During the water transfer to the FTC, some (estimated 10%) of the loose fine core debris transferred from the OTSGs will be transferred to the FTC. It is assumed that 5% of the debris transferred from the OTSGs will be deposited in the pump suction area and another 5% of debris will slide into the pump suction area due to the slope of the lower head and the velocity of the water near the lower head surface. In addition to the fuel debris from the OTSGs, additional fuel debris will settle out on the lower head due to the water level sloshing in the RV during drain down of the OTSGs. It is estimated that 0.1 kg of fuel (UO_2) would be transferred to the FTC during this sequence.

The next phase will be to completely drain the RV for the final time. While the RV is being drained it will be necessary to drain the FTC at

the same time to prevent it from overflowing. During this final drain down, it is estimated that another 1.4 kg (5% of 28 kg on lower head) could be transferred to the FTC. This would mean that an estimated total of 4.5 kg of fuel (UO_2) will be transferred from the RV to the FTC during drain down of the OTSGs and RV. The uncertainty assigned to the total fuel debris (4.5 kg UO_2) tranferrred from the RV to the FTC during drain down is ±78%. This uncertainty is based on an average dimensional uncertainty of 20%, a fuel density uncertainty of 40% and a fuel debris ratio uncertainty of 53%. The value was calculated using standard propogated error formulas for a product of terms (Reference 30).

After the OTSGs and RV are completely drained, the FTC will be drained to the RCBTs using the same pumps used to drain the RV. Since the floor of the deep end of the FTC was relatively clean prior to starting drain down and the plenum covers approximately 36% of the floor area of the deep end of the FTC, only a fraction (64%) of the core debris pumped into the FTC during drain down (estimate 2.9 kg) could reach the floor of the FTC. There will be plenty of time for the fine debris to settle out in the FTC prior to draining. Based on the total FTC horizontal area (420 sq ft - Reference 31) that fuel fines could settle over and the area that the drain pump would normally suck clean (2.3 sq ft - Appendix A), it was estimated that approximately 1% of the fuel debris added to the FTC during drain down of the RV would be removed during drain down of the FTC. In order to simplify SNM Accountability, the amount of fuel removed from the FTC during drain down will continue to be included in the total quantity shown for the FTC. This will eliminate the need to revise the AFHB PDSR to account for any fuel relocated in the RCBTs during FTC drain down.

The total estimated fuel (UO_2) remaining in the FTC after drain down is as follows:

	Fuel (UO_2)	Uncertainty
Fuel washed off canisters	13.9 kg	+ 24%, -100%
Fuel on transfer tubes	0.5 kg	+ 34%, - 92%
Fuel in DWCS hoses	0.0 kg*	
Fuel added during drain down	4.5 kg	+ 78%, - 78%
Total Estimated Fuel	18.9 kg	+ 37%, - 95%

* Fuel content in DWCS hoses was estimated at 0.012 kg UO_2 which rounds off to 0.0 kg.

The estimated percentage uncertainties for the measurements, calculations and estimations are shown above. Total uncertainty was determined using an asymmetrical approach where the negative and positive uncertainties were added separately and a range was determined for the final uncertainty. The range was from 2.4 kg to 26.2 kg or 18.9 kg +37%, -95%.

It is acknowledged that the method of estimating fuel transfer during drain down was relatively uncertain but since the fuel quantity transferred to the FTC was relatively small, it was decided that additional fuel measurements would not be warranted.

4.3 Core Flood System

During the 1979 TMI-2 accident, fuel was deposited in the core flood system due to backflow from the reactor through one or more malfunctioning check valves. Additional fuel was deposited in the core flood tank '1A' during the defueling and removal of the lower core support assembly (LCSA). These additional fuel deposits were attributed to the following LCSA segments which were removed from the RV to permit continuous progress in the RV defueling activities.

13 Severed lower grid rib segments
4 Pie-shaped lower grid distributor segments
4 Pie-shaped lower grid forging segments
4 Incore guide support plate sections
12 Flow distributor plate sections
48 Lower grid rib support posts

Prior to removal from the RV, the LCSA segments were flushed and brushed to minimize the transfer of adherent core debris. The segments were also video inspected to ensure 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. Exposure rate measurements were performed on pipe sections to establish the fuel deposited in the Core Flood System.

Exposure rate measurements were taken along the core flood system pipes using a standard HP-220A probe in 1985 and a modified HP-220A probe in 1989. Both instruments utilized a hemispherical tungsten collimator which had an approximate front to back response ratio of 50:1. The standard HP-220A had a field of view of 140 degrees, and the modified HP-220A had a field of view of 90 degrees.

Measurements were conducted by placing the probe in contact with the pipes either on the side (vertical pipes) or on the bottom (horizontal pipes). A background measurement was conducted at each measurement location by facing the probe away from the pipes. This type of background measurement estimated the general area exposure rate from other sources in the reactor building. There were two (2) other sources of background, or non-fuel related, exposure rates: Cs-137 surface contamination on the pipe surfaces and Cs-137 in the aqueous phase of the pipe contents. The average exposure rate of the

contaminated surface was 11.3 mR/hr which was used as the Cs-137 surface contamination background. The contribution to background from dissolved Cs-137 was considered to be negligible. Thus, the background subtracted from each gross exposure rate consisted of the general area background (calculated for each measurement location) and the Cs-137 surface contamination exposure (11.3 mR/hr).

Two general source geometries were used to estimate the fuel quantity in the core flood system: the fuel film in the pipes and tanks, and the fuel deposits in the pipes. The average RCS fuel film mass per unit area was multiplied by the internal surface area of the core flood system to estimate the total fuel mass in films. This portion of the estimate is conservative because the core flood system was not exposed to circulating fuel debris for as long a time as the main portion of the RCS. Thus the average fuel film concentration in the core flood system was expected to be lower than that of the RCS.

The 1985 and 1989 gamma exposure rate measurements and radionuclide concentration data from core samples were used to estimate fuel deposited in the core flood pipe segments. The observed fuel per segment distribution was then applied to the remaining, unmeasured, portion of the piping. This approach assumed that no fuel debris was deposited in the tanks or vertical pipe sections.

Because most of the "A" side piping was inaccessible, the results from the "B" side were used to estimate the fuel mass in the "A" side. The assumption is that the linear concentration of fuel in the "B" pipe is equal to that in the "A" pipe.

In addition, pieces of the lower core support assembly were removed from the reactor vessel and stored in the "A" core flood tank which had its top cut off for this purpose.

The results of the calculations for each area are summarized below and calculation details are described in Engineering Calculation 4200-3232-90-067.

	A-Side	B-Side
Fuel (UO_2) total in films	0.008 kg	0.010 kg
Fuel (UO_2) total in deposits	0.890 kg	1.670 kg
LCSA pieces	2.300 kg	----
SUB TOTAL	3.198 kg	1.680 kg

Total Fuel Mass In Core Flood System: 4.9 kg of UO_2

The uncertainty of this estimate is ± 77%.

4.4 Upper Endfittings

A total of seventeen (17) upper endfittings are stored in six (6) separate storage containers. Five (5) of the endfitting storage containers are stored on elevation 347' in the Reactor Building. The sixth (6th) storage container, designated as TSC, is stored in the deep end of the fuel transfer canal. A neutron interrogation system was utilized to determine the amount of fuel remaining on the upper endfittings. The interrogation system utilized an antimony-beryllium (Sb-Be) photoneutron source and He-4 detectors. The Sb-Be source primarily produces neutrons with energy of 24 ± 2 keV. The half-life of a Sb gamma source is sixty (60) days. The measurements were conducted in the Reactor Building where the presence of other equipment caused electronic noise to interfere with the counting system. The noise problem was partially overcome by performing repetitive counts and then applying Chavenets' criterion (References 30 and 32) to the resultant data.

While each endfitting storage container was placed in the counting location, the background was measured using He-4 detectors without any Sb-Be source. Then known UO_3 calibration spikes were added to the storage container, and the Sb-Be source was introduced to produce interrogating neutrons. These neutrons produced fission in the U-235 in the residual fuel. Measurements were taken for all three (3) conditions. First, without the Sb-Be source; second, with the Sb-Be source; and third, with the Sb-Be and UO_3 sources. The fast neutron flux from fission of U-235 was measured using the three (3) He-4 neutron detectors and associated electronic equipment. Most of the fission neutrons had energies between one (1) to two (2) MeV which was much higher than the interrogating source neutrons.

The neutron interrogation system was calibrated for each geometry encountered by placing two (2) uranium calibration spikes into the containers with the endfittings. The spikes were cylindrical containers of natural uranium with known masses. The system efficiency factors were calculated from the increase in neutron count rate per U-235 mass in the spikes for counting geometry.

The data was collected for each storage container using three (3) He-4 detectors. Counting of the signals was performed with a computer using LABTECH NOTEBOOK software. This software collected the data for each preset interval and summed the results over a preset time.

Five (5) storage containers were measured using this technique while the sixth (6th) storage container, designated as 'TSC', was analyzed using the average fuel value obtained while counting the five (5) storage containers described above. The results of the measured and estimated residual fuel quantities were as follows:

Storage Container	Number of Endfittings	Fuel Quantity
#7	2	0.35 kg
#9	3	0.69 kg
#5	4	3.12 kg
#4	3	0.57 kg
#6	2	0.12 kg
TSC	3	1.04 kg
Totals	17	5.89 kg

The fractional uncertainties for the measurement and calculations were estimated as follows:

1. Calibration of Spike Masses: negligible
2. System Efficiency: ±57%
3. Measurement Error: ±63%

Total uncertainty was determined by quadrature error propagation which was the square root of the sum of the squares of both the efficiency and measurement errors.

$$\text{Sqrt}\ ((57\%)^2 + (63\%)^2) = \pm 85\%$$

The total fuel estimate for all of the endfittings stored in the Reactor Building was 5.9 kg of UO_2. The uncertainty associated with this estimate was ±85%. The results of the detail calculations for the endfitting storage containers are described in Engineering Calculation 4249-3232-90-079.

4.5 Tool Decon Facility (TDF)

The tool decon facility (TDF) consisted of three (3) rooms which were

located on the 347' level of the reactor building. The TDF had been used until January 31, 1990, for decon work; thus, only general area surveys dated after January 31, 1990, were used to determine residual fuel. The background gamma radiation level in the TDF due to general area Cs-137 contamination on the floor, walls and components was estimated as 15 mR/hr. It was assumed that negligible quantities of fuel were attached to the walls and ceilings of the TDF.

The Cs-137 to UO_2 ratio for core debris was taken to be 1400 ± 500 μCi Cs-137/g as of March 28, 1988 (Reference 9). Decay correction to the date of the area surveys (March 1, 1989) was neglected. Data for Co-60 and other gamma-ray emitters were not analyzed; however, the contribution of these radionuclides to the gamma exposure rate was estimated to be less than 20% of that due to Cs-137 alone.

The fuel in the TDF was assumed to be uniformly distributed over the floors and grating in the three (3) rectangular rooms. Readings in the left room (Area -1) were fairly uniform with the center reading 15 mR/hr. Thus, a model using a 12' x 12' rectangular area source with the dose point in the center was entered into Microshield (Reference 8) to calculate an exposure rate for the Area -1 fuel concentration factor mR/hr/(mg/cm^2). The actual Area -1 fuel concentration (mg/cm^2) was calculated by dividing the actual exposure rate (after background subtraction) by the Microshield factor. Multiplication of the Area -1 concentration by the floor area resulted in an estimate of less than 0.1 grams of fuel (UO_2) on the floor.

The fuel estimate for the middle room (Area -2) was calculated using the same approach. The resulting estimate for Area -2 was 5 grams of fuel (UO_2).

The right room (Area -3) had more variable exposure rates. Thus, Area

-3 concentration estimates based on each measurement point were calculated using Microshield. The four (4) corner points were assumed to be two (2) feet from one wall and three (3) feet from the other wall. The two (2) center points were assumed to be six (6) feet from the side walls and four (4) feet from the end walls. The average of the six (6) area concentration estimates multiplied by the floor area then resulted in a fuel estimate of 0.11 kg of UO_2 for the room.

The total fuel estimate for the entire TDF was 0.1 kg of UO_2. The uncertainty associated with these estimates was ± 100%. The results of the detail calculations for each area are described in Engineering Calculation 4249-3232-90-068.

4.6 Drain Line From TDF

There are numerous drain lines throughout the reactor building. Only one line was expected to have significant quantities of fuel; the line that services the defueling tool decon facility on the 347' elevation. This line begins at a floor drain in the tool decon facility, passes through the floor and extends ten (10) feet nearly horizontally to a riser. The riser drops down through the 305' level via vertical and forty-five (45) degree runs and empties into a horizontal run within the basement floor slab. The horizontal run in the basement slab extends approximately one-hundred and twenty (120) feet via a gradual slope into the RB sump. The basement had been under at least eight (8) inches of water since the accident, thus this horizontal run had been filled with water.

Other lines in the reactor building have not serviced fuel-related activities and thus are assumed to contain negligible quantities of fuel.

The fuel estimates in the drain line were made using gamma exposure rate measurements within the pipe. A Cs-137 to fuel mass ratio (Reference 9) was utilized to determine the exposure rate per unit mass of fuel within a pipe segment. The quantity of fuel within each pipe segment was then calculated using the exposure rate data and the exposure rate per unit mass fraction.

The exposure rate measurements were made using a miniature or "peanut" GM detector positioned at regular intervals within the drain pipe. Six detectors were included in the detector string, however, only detectors #1, #3 and #5 were used due to the availability of readout electronics. Detector #1 was surrounded with a cadmium energy compensation shield and was connected to a Ludlum 2200 single channel analyzer. Detectors #3 and #5 had no shield and were connected to Eberline SAM-II single channel analyzers. Detectors #3 and #5 experienced an unusually large amount of dead time during the measurements thereby nullifying the results. Only the data from detector #1 was used to calculate fuel.

The GM detectors were housed within a one-hundred (100) foot long by one-half (1/2) inch OD polyethylene tube which was water tight. The tube was attached to a plumber's snake (1/8" x 1" steel band) to cause it to lay flat in the bottom of the pipe. The detector tube insertion was incremented by one (1) and four (4) foot groups so that the detectors at one (1), three (3), and five (5) feet would provide count rate readings at each foot. However, because detectors #3 and #5 were later found to be malfunctioning, only the readings from detector #1 were used to establish the fuel mass.

The fuel estimates were obtained by subtracting the background exposure rate (5.2 R/hr) from each section and dividing by the Microshield exposure rate/mass factor (0.21 R/hr-g UO_2). The total

mass identified in the eight (8)-inch drain line under the basement floor was 2.5 kg of UO_2. The total mass in the six (6)-inch drain line under the 347' floor was 1.9 kg of UO_2. This results in a total fuel mass of 4.4 kg of UO_2 in the TDF drain line. The uncertainty of this total mass was ± 87% and includes the uncertainties due to calibration, geometry and shielding models used, and the variability in the Cs-137 to fuel ratio. See Engineering Calculation 4249-3232-90-073 for details of the data reduction.

4.7 Defueling Water Cleanup System

The Defueling Water Cleanup System (DWCS) was installed during the TMI-2 Cleanup Program to remove fuel fines from the RCS water and improve visibility in the RV during defueling. The DWC System consisted of circulating water pumps, filter canisters, a valve manifold and interconnecting hoses. All of the filter canisters have been shipped offsite for permanent storage.

The circulating water pumps, DWC-P-2A and DWC-P-2B were located in the new fuel pit in the southeast corner of the fuel transfer canal. A video inspection of the DWC-P-2A pump intake area was conducted on September 22, 1989. A depth of approximately 0.3 centimeters of fuel debris was estimated on the horizontal, annular surface of the transition area. It was assumed that DWC-P-2B contained an equal quantity of fuel. To estimate the fuel in the two (2) pump transitions, the depth (0.3 cm) was multiplied by the horizontal area (852 cm^2) of both transitions times the excore fuel density (4.7 g/cc) times the UO_2 to debris ratio (0.72). The results of this product was the total quantity (0.86 kg UO_2) of fuel residing on the horizontal surfaces of both pump transition areas.

Two other locations in the pumps were identified in the video inspection as containing fuel debris. The quantity was estimated as 98 and 131 cubic centimeters totaling (229 cm^3 x 4.7 x 0.72) = 0.78 kg UO_2 in each pump assembly. The total quantity of fuel estimated for both DWCS pumps was (0.86 kg + (2 x 0.78 kg) or 2.4 kg of UO_2 ± 100%.

The manifold consisted of piping, valves and other components that interfaced between the RV and filter canisters. It was located at the north end of the FTC. A series of gross gamma measurements were taken from each D-Ring using a hexadirectional probe. This probe utilized a hemispherical tungsten collimator which had an approximate front to back response of 50:1 and a field of view of 90 degrees. The background measurements consisted of pointing the detector away from the component, up and/or towards a wall of the RB. Thus the exposure rate detected was due to non-fuel related Cs-137 contamination on the RB surfaces. The resulting fuel related exposure rate for the manifold was the gross reading minus the average background readings collected on the D-Rings. The average background exposure rates for the "A" and "B" D-Rings were 5.05 and 2.37 mR/hr, respectively. The net exposure rates from the "A" and "B" D-Rings were 5.48 and 6.59 mR/hr, respectively.

The average net exposure rate approximately twenty-four (24) feet from the center of the manifold was ((5.48 + 6.59)/2) = 6.04 mR/hr. The exposure rate per unit mass factors for line and point source models with varying shielding thicknesses were calculated using the computer program Microshield (Reference 8). The results using the above data and the most probable shielding and geometry model (the best case) estimated the fuel quantity in the manifold as 1.1 kg UO_2 ± 50%.

The third and final component that made up the DWC System was the interconnecting hoses between the pumps, filters and manifold. Hose

sizes varied from 0.5 to 4 inches inside diameter and a total length of approximately 6550 feet. A series of gamma spectrometry and gamma exposure rate measurements were taken on one 1 foot samples and other hoses in place. The average fuel mass for the one 1 foot long hose samples was determined as 0.015 g ± 90% for the two (2) inch inside diameter hose which had 486 square centimeters of inside area per foot. The total inner surface area for all the DWCS hoses was determined as 614,986 square inches. Taking the product of the average fuel mass per area times the total area resulted in a fuel estimate of 0.122 kg ± 90% for all the hoses in the DWC System. A second method using gross gamma measurements was utilized to measure the hoses which resulted in a fuel estimate of 0.295 kg. Then the two (2) estimates were averaged to arrive at the final fuel estimate of 0.21 kg UO_2 ± 100%.

Details of the data analysis and measurements are contained in the Engineering Calculation No. 4249-3232-090-078.

After the DWC System had been measured, sections of the two (2)-and three (3)-inch diameter hoses were removed from the work platform and placed in the FTC. The quantity of fuel remaining on these hoses was estimated (see Section 4.2) as 0.012 kg UO_2 ± 100%. This amount was deducted from the total residual fuel deposits in all the hoses discussed above. The final estimate for the remaining hoses was (0.21 kg - 0.01 kg) or 0.20 kg UO_2 ± 100%. The final estimate for the DWC System in the RB was:

Component	Qty of Fuel(UO_2)	Uncertainty
DWC Pumps	2.4 kg	± 100%
Manifold Assembly	1.1 kg	± 50%
Interconnecting Hoses	0.2 kg	± 100%
Total	3.7 kg	± 67%

4.8 Temporary RV Filter System

Curie estimates were performed on the diatomaceous earth (DE) collection drums and filters that were used in the Temporary Reactor Vessel Filtration System (TRVFS). The isotopic analysis used to determine the residual fuel came from one of the collection drums and was considered typical of the DE in the filters. The Radiological Engineering Calculation, Reference 33, was assumed to be a valid estimate of the curie loading for the two filters. The calculation was rechecked by Fuel Measurements and Analysis (FM&A) and was found to be acceptable.

Using the data provided in Reference 28, the fuel loading for both filters was estimated as 3.80 kg and 0.59 kg of UO_2. This resulted in a total of 4.4 kg of UO_2 for the TRVFS. See Engineering Calculation 4800-3520-89-076 for details of the data reduction. The uncertainty associated with these values is ± 90% to a one-sigma confidence level.

4.9 Incore Guide Tubes

The flow distributor assembly (FDA) was cut up into twenty-six (26) pieces. It was determined that thirteen (13) plates each contained one to four incore instrument guide tubes (IIGTs). The plates were removed from the reactor vessel (RV) one at a time and moved to a measurement location on top of the 'A' D-ring.

The gamma-ray spectrometry measurements were performed using a lead shielded HPGe detector. The detector assembly was connected to a preamplifier, preamplifier power supply, amplifier, and a multi-channel analyzer (MCA). The detector system was calibrated using Ce-144 and Eu-154 sources for a full energy peak calibration. The

detector system was also source checked before each measurement to ensure proper operation using a standard Ce-144 source. The quantity of reactor fuel was based upon using Ce-144 as an analog for fuel (UO_2). The Ce-144-to-fuel ratio of 152.5 μCi/g (± 53%) was available on 8/1/87 (Reference 10). This activity was decayed to the date that the measurements were performed.

The calculation of the amount of fuel on each plate was performed using either the computer code QAD (Reference 34) or MICROSHIELD (Reference 8). The source region of the IIGTs was modeled as a volume inside the IIGTs to simplify the calculation. The source was assumed to be uniformly distributed in the source region. The maximum quantity of fuel in one IIGT was (assuming a density of 5 g/cm^3 (UO_2)) 1.42 kg. Two of the IIGTs, K-5 and M-10, were cut out of the flow distributor plates, stored in the 'A' D-ring and shipped offsite to a laboratory for analysis. This work was performed for the Nuclear Regulatory Commission. Both IIGTs were assayed as containing 1.42 kg of UO_2 each and thus, 2.8 kg was subtracted from the 23.8 kg of fuel measured on the flow distributor plates stored in the 'A' D-ring. The adjusted fuel quantity remaining on the flow distributor plates was 21.0 kg of UO_2. The estimated uncertainty in this fuel value was ± 54%. Details of the data analysis and measurements are contained in Reference 35.

4.10 Defueling Tools

Five (5) defueling tools were measured utilizing gamma-ray spectrometry and an HPGe shielded detector. The quantity of fuel was based upon the activity of Ce-144 and the known Ce-144-to-fuel ratio of 152.5 μCi/g decayed to the date the measurements were performed. The detector system was source checked before each measurement and after the final sequence of measurements to ensure proper operation

using a standard Ce-144 source. The calculation of the quantity of fuel on each tool was performed using either the computer code QAD (Reference 34) or MICROSHIELD (Reference 8). Computer modeling of source geometry was used to calculate the gamma flux per unit of fuel mass on each tool. The final mass estimate for each tool was then determined by dividing the measured gamma flux by the gamma flux per unit mass factor. The total residual fuel for the five (5) measured tools was 61 grams ± 60% of UO_2.

The five (5) measured tools were also measured using the gamma exposure rate method to arrive at an approximate fuel estimate for the remaining one-hundred and twelve (112) tools to be assayed. This data resulted in a gamma exposure rate-to-fuel ratio of 0.094 (R/hr)/gram. Sixty-nine (69) tools were measured using the gamma exposure rate radiation levels and were found to contain a total fuel mass of 318 grams ± 70% of UO_2.

Combining this data with the data above, results in an average fuel deposit of 5.1 grams/tool of UO_2. There were forty-three (43) other tools that were not measured due to radiological restrictions. The exact number of tools or tool pieces is uncertain due to multiple component tools, tools that have been cut-up, and the lack of data on some tool storage areas. Thus, without additional data, the conservative estimate of 117 total tools was used for this report. The number of tools, for which no actual or specific survey data is available, was taken as 117-74 or 43. The best estimate for the quantity of fuel on these tools is 43 x 5.1 g/tool or 220 grams ± 85% of UO_2. Adding this to the 74 measured tools (379 g), results in approximately 0.6 kg ± 75% of UO_2 for all one-hundred and seventeen (117) defueling tools. Details of the uncertainty analysis are discussed in Engineering Calculation 4240-3232-90-066.

5.0 CONCLUSION

The estimate of record of the quantity of fuel (UO_2) remaining in the Reactor Building Miscellaneous Components is 64.0 kilograms, (at one sigma). This uncertainty is based on the individual component uncertainties associated with each component addressed in this Post Defueling Survey Report. The range of the estimate of record extends from 31.6 to 85.4 kg UO_2.

This estimate of record is derived from the measured and analyzed data summarized in Table 2. The data shows that 62% of the remaining fuel is contained in two components, the Fuel Transfer Canal/Transfer Tubes and the Incore Guide Tubes stored in the 'A' D-Ring. These components are expected to remain static once drain down is completed. No future water flows are planned except possibly to remove condensation from the FTC. Additional measurements of the RB Miscellaneous Components are not considered to be justifiable based on ALARA considerations due to the small quantity of fuel (UO_2) measured to date. After final draindown of the RCS, the RB will be normally isolated from all systems outside the RB, and any operation that could result in a potential fuel movement will be closely controlled.

The goal of the TMI-2 defueling program was to remove more than 99% of the original core inventory of approximately 94,000 kg. In that context, the 64.0 kg quantity of UO_2 remaining in the RB Miscellaneous components is less than 10% of the total fuel (UO_2) remaining at TMI-2.

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TABLE 1

ASSAY METHODS UTILIZED TO MEASURE MISCELLANEOUS AREAS IN THE REACTOR BUILDING

AREA DESCRIPTION	ASSAY METHOD	CALCULATION NUMBER
Reactor Coolant Drain Tank	Sampling & Analysis	Reference 13
Fuel Transfer Canal/Transfer Tubes	Gamma Dose Rate	4800-4420-89-095
Core Flood System	Gamma Dose Rate	4200-3232-90-067
Upper Endfittings	Neutron Interrogation	4249-3232-90-079
Tool Decon Facility (TDF)	Gamma Dose Rate	4249-3232-90-068
Drain Line From TDF	Gamma Dose Rate	4249-3232-90-073
Defueling Water Cleanup System	Gamma Spec./Gamma Dose Rate	4249-3232-90-078
Temporary RV Filter System	Sampling & Analysis	4800-3520-89-076
Incore Guide Tubes	Gamma Spectroscopy	4200-3211-90-062
Defueling Tools	Gamma Spec./Gamma Dose Rate	4240-3232-90-066

TABLE 2

SUMMARY OF SNM INVENTORY FOR RB MISCELLANEOUS AREAS

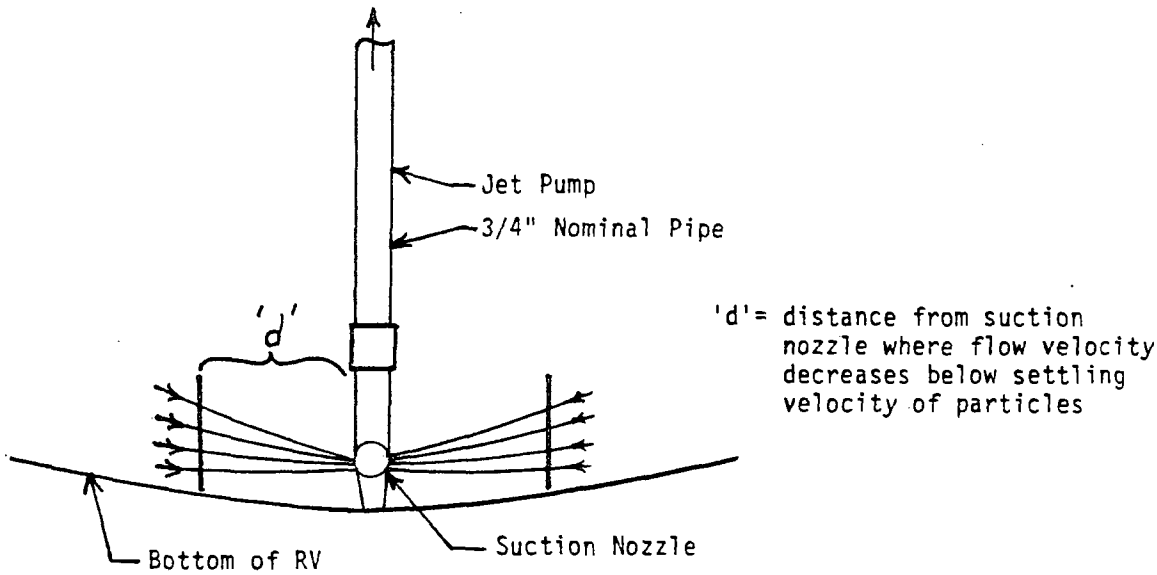
AREA DESCRIPTION	FUEL QUANTITY	UNCERTAINTY	REFERENCE
Symmetrical Data:			
Reactor Coolant Drain Tank	0.1 kg	± 54%	PDSR Paragraph 4.1
Core Flood System	4.9 kg	± 77%	PDSR Paragraph 4.3
Upper Endfittings	5.9 kg	± 85%	PDSR Paragraph 4.4
Tool Decon Facility (TDF)	0.1 kg	±100%	PDSR Paragraph 4.5
Drain Line From TDF	4.4 kg	± 87%	PDSR Paragraph 4.6
Defueling Water Cleanup System	3.7 kg	± 67%	PDSR Paragraph 4.7
Temporary RV Filter System	4.4 kg	± 90%	PDSR Paragraph 4.8
Incore Guide Tubes	21.0 kg	± 54%	PDSR Paragraph 4.9
Defueling Tools	0.6 kg	± 75%	PDSR Paragraph 4.10
Sub-Total	45.1 kg	± 32%	
Asymmetrical Data:			
Fuel Transfer Canal/Transfer Tubes	18.9 kg	± 37%, -95%	PDSR Paragraph 4.2
ESTIMATE OF RECORD (UO_2)	64.0 kg		

TOTAL UO_2 RANGE: Low Value: 31.6 kg, High Value: 85.4 kg

NOTE: The range was determined using the asymmetrical term for the Fuel Transfer Canal/Transfer Tubes and combining its range algebraically with the fuel quantity (45.1 kg UO_2 ± 32%) estimated for the symmetrical terms.

APPENDIX A

Pumping Velocity at Known Distance from Pump Suction Nozzle



Assumed Pumping Rate = 10 GPM

Water Velocity in 3/4 " pipe = 6.02 ft/sec or 72.2 in/sec.

Cross section area of 3/4 " pipe = 0.533 sq. in.

Assumed surface area of cylinder at distance 'd' from jet pump which surrounds pump suction nozzle for a uniform flow velocity = πd^2

Effective settling Area for lower vessel head = 112 sq. ft.

Assumed water velocity 'V_s' at distance 'd' from suction nozzle:

$$v_s = \left(\frac{area\ of\ pipe}{area\ of\ surface}\right)(velocity\ in\ pipe)$$

$$v_s = \frac{0.533\ in^2}{\pi d^2\ in^2}(72.2 in/\text{sec}) = \frac{12.25}{d^2} in/\text{sec}$$

(I)

$$at\ d = 6cm;\ V_s = 2.2\ in/\text{sec}\ or\ 5.6\ cm/\text{sec}$$

5.6 cm/sec is the approximate settling velocity for particle sizes less than 500 microns.

(2)

$$at\ d = 25.4\ cm\ or\ 10\ inches;\ V_s = 0.12in/\text{sec}\ or\ 0.30cm/\text{sec}$$

0.30 cm/sec is the approximate settling velocity for particle sizes less than 70 microns

Area enclosed by 'd' radius = 2.3 sq. ft.

Approximate percentage of lower head settling area = 3%

* *

Assumed fuel debris particle size distribution based on Lower Vessel SSC–1–2 Sample data with no particles larger than one millimeter (Reference 36).

Particle Size	Cummulative Percent Weight	Maximum Settling Velocity*
< 38 micron	10%	0.07 cm/sec
< 90 micron	30%	0.5 cm/sec
< 150 micron	46%	1.3 cm/sec
< 300 micron	67%	3.0 cm/sec
< 710 micron	91%	7.0 cm/sec
<1000 micron	100%	12.0 cm/sec

* Settling velocities determined from CRC Handbook of Chemistry and Physics, 68th Edition (Page F–233).