TMI-2 SOLUTIONS

May 5, 2021
TMI2-RA-COR-2021-0010

10 CFR 50.90

U.S. Nuclear Regulatory Commission
ATTN; Document Control Desk
Washington, DC 20555-0001

Subject: Supplemental Information to License Amendment Request – Three Mile Island, Unit 2, Decommissioning Technical Specifications

Three Mile Island, Unit 2
NRC Possession Only License No. DPR-73
NRC Docket No. 50-320

Reference:

1) Letter TMI2-RA-COR-2021-0002 from van Noordennen (TMI-2 Solutions LLC) "License Amendment Request – Three Mile Island, Unit 2, Decommissioning Technical Specifications," (ML21057A047) dated February 19, 2021.

On February 19, 2021, TMI-2 Solutions submitted an application for amendment of license for Three Mile Island Nuclear Station, Unit 2 ("TMI-2") to revise the Possession Only License and the associated Technical Specifications to support the transition of TMI-2 from a Post-Defueling Monitored Storage (PDMS) condition to that of a facility undergoing radiological decommissioning (DECON). This letter transmits a supplement to the application submitted on February 19, 2021 (Reference 1). Specifically, this letter transmits additional information related to a calculation that was submitted in Reference 1 that provides the basis for establishing a new Safe Fuel Mass Limit (SFML). Attachment 1 contains a report that contains a consensus on the inputs to the SFML calculation previously submitted in Reference 1. Attachment 2 contains a calculation of a bias determination for low enrichment Uranium.

In accordance with 10 CFR 50.91(b)(1), a copy of this submittal has been sent to the Commonwealth of Pennsylvania.

A001
NRR

In the event that the NRC has any questions with respect to the content of this document or wishes to obtain any additional information, please contact me at 860-462-9707.

Sincerely,

Gerard van Noordennen

Gerard van Noordennen
Senior Vice President Regulatory Affairs

Attachments:

Attachment 1 – TMI-2 Safe Fuel Mass Limit Computational Input Consensus

Attachment 2 – MCNP Versus 6.2 Bias Determination for Low Enrichment Uranium Using the ENDF/B-VIII.0 Cross Section Library

cc w/Enclosures

Ted Smith, NRC Project Manager
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STATE OF CONNECTICUT)
) SS.
COUNTY OF NEW LONDON)

Gerard van Noordennen, being duly sworn according to law deposes and says:

I am the Senior Vice President Regulatory Affairs of TMI-2 Solutions, LLC, and as such I am familiar with the contents of this correspondence and the attachments thereto. The matters set forth therein regarding TMI-2 Solutions, LLC, their affiliates, and Three Mile Island Nuclear Station, Unit 2 are true and correct to the best of my knowledge, information and belief.

Gerard van Noordennen

Gerard van Noordennen

Subscribed and Sworn to before me this 5th day of May 2021

Notary Public of Connecticut

Bonnie Stanford
Notary Public
My Commission Expires 12/31/2021

Attachment 1

TMI-2 Safe Fuel Mass Limit Computational Input Consensus

TMI-2 Safe Fuel Mass Limit Computational Input Consensus

Revision 0

Authored By: Digitally signed by Megan Pritchard Date: 2021.01.28 09:25:07 -06'00'
Megan Pritchard, Nuclear Criticality Safety Engineer — Date

Reviewed By:
Ray Klein, Project Manager — Date: 1/28/2021

Approved By: Digitally signed by Guy Rhoden DN: cn=Guy Rhoden, c=US, o=NS-TS, email=guy.rhoden@ns-ts.com Date: 2021.01.28 11:40:34 -08'00'
Guy Rhoden, Responsible Engineer — Date

- [ ] Non-Proprietary
- [ ] Proprietary
- [ ] Restricted Information

- [X] New
- [ ] Title Change
- [ ] Revision
- [ ] Rewrite
- [ ] Cancellation

Effective Date: 01/28/2021

Record of Revisions

Revision	Revision Title Description and Reason for Change
0	Initial Issue

Table of Contents

1.0 Introduction

The purpose of this report is to develop a consensus on the inputs that will be utilized in the computational development of the safe fuel mass limit (SFML) for Three Mile Island (TMI)-2 decommissioning. It is intended that a single SFML can be utilized for all work in TMI-2 decommissioning and bound all credible operational conditions. The input parameters for consensus include core debris composition (i.e., impurities concentration), fuel enrichment, fuel density, fuel burn-up and resulting composition, fuel pellet geometry, and modeling geometry, including moderation and reflection conditions. Several of these parameters were first developed in historic analyses and reports and are selected for use. Review of previous analyses, calculations and data allow a graded and informed application of conservatisms such that the SFML can be derived using conservative yet realistic inputs.

2.0 Remaining Core Inventory

About 225 minutes after the March 1979 accident, instruments indicated, and later analysis revealed a major core relocation. One effect of the accident was to shatter the ceramic UO_2 pellets and to mix the UO_2 with other metal oxides creating a gravel-like debris bed covering a mass of resolidified composite material. The late 1980s de-fueling operations shattered the resolidified material into gravel-like material visually indistinguishable from the other debris. As an example of what was accomplished during defueling, when fuel and support structures were cut out of the RV, Figure 2-1 provides the end-state graphic of where the incore guide support plates were cut.

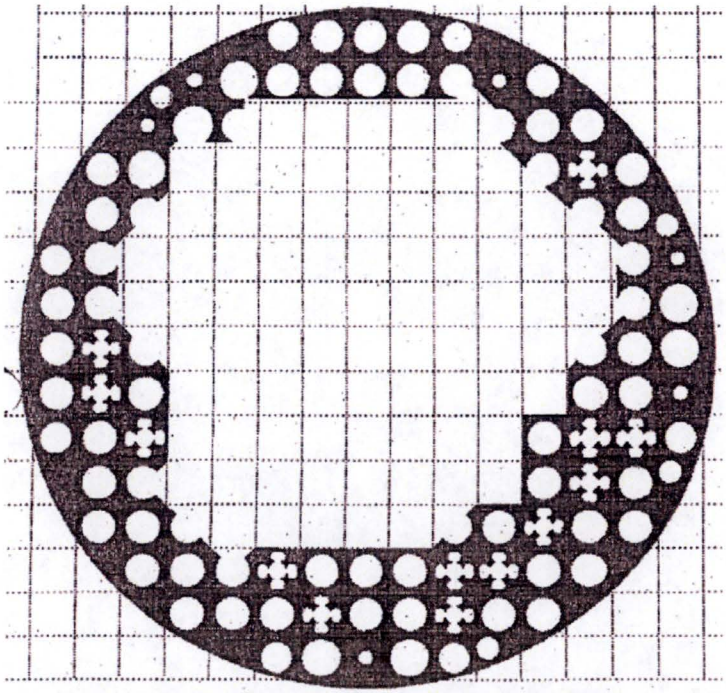

Figure 2-1. Incore Guide Support Plates After Cuts (Ref. [1], Figure 5-25)

After defueling, the residual fuel was observed to be composed of finely divided material, thin crusts, and solid masses. Small amounts of debris were left in hard to access locations, and a dusting of residual fines was widely deposited on horizontal surfaces. Peripheral lower core support assembly (LCSA) coolant flow holes and other similar channels that could not be defueled house solid masses of resolidified prior molten fuel.

Estimations of the remaining fuel debris inventory have been on-going since defueling began in 1985 with multiple updates and refinements to the values. Following defueling, the three major areas where core debris resides are the reactor vessel (RV) bottom head, the LCSA, and the core former area (i.e., between the core former baffle plates and the core barrel) in the upper core support assembly (UCSA) (Ref. [2]). Therefore, in the Post Defueling Survey Program, the primary focus of accurate estimates was on the RV. A Survey Report was written for each primary area/system of the TMI-2 facility, with the main focus being on the fuel remaining in the RV. Those areas and the estimated remaining fuel mass are listed in Table 2-1.

Each *Post Defueling Survey Report* provides an *Estimate of Record* and supplies all fissile mass data presented in the body of this document. The fuel material composition does not come from these documents but instead from the Appendix B to the *Defueling Completion Report* (Ref. [3]). Supporting documentation for the fuel debris impurity composition is from independent analytical reports listed in Section 3.0.

Table 2-1. Remaining Fuel Mass Estimates of Record (Ref. [4], Enclosure 2, Table 1)

Area/System	Estimated Mass (kg UO_2)	Uncertainty (kg UO_2)	Reference Section	Reference Document
Plenum	2.1	± 1.9	2.5	[5]
Letdown Coolers	3.7	± 2.0	2.6	[6]
Pressurizer	0.3	± 0.2	2.6	[6]
Reactor Vessel Head	1.3	± 0.9	2.7	[7]
Reactor Building Basement	1.3	± 0.7	2.6	[6]
'A' and 'B' OTSG	62.3	± 9.7	2.4	[8]
Auxiliary and Fuel Handling Buildings	11.5	± 5.8	2.2	[9]
Reactor Building Miscellaneous Components	64.0	± 26.9	2.8	[10]
Reactor Coolant System	25.8	± 11.1	2.3	[11]
Reactor Vessel	925	± 370	2.1	[4]
Total	**1097**	**± 371**		

Each area is broken down in the following sections to summarize that area's Survey Report and estimated remaining inventory. The degree of robustness in each estimate and whence its uncertainty (all reported at 1σ) is derived can be deduced. Although not all statements have a reference provided, it can be assumed that the general information is derived from the respective Survey Report unless otherwise referenced.

2.1 Reactor Vessel

2.1.1 Initial Estimates

The determination of residual quantities of core debris and its location within the RV was an ongoing process from the start of defueling. Initial techniques included video inspection and sample acquisition and analysis. Extensive visual examination of nearly all accessible parts of the RV internals were made during and following RV defueling. The video tapes were analyzed to infer the amount, form, and location of the residual fuel in the RV (Ref. [2]). The physical extent of debris deposits was mapped in three dimensions. Given good lighting conditions, the vertical and lateral extent could be estimated fairly accurately, but depth was much less easily determined. The physical distribution was then used to estimate volume. Surface texture and other factors were used to identify and compare the subject debris deposit to other similar material for which sample analysis data existed.

The archived video tapes showed the residual fuel in the RV to consist primarily of finely divided, small particle-size sediment in inaccessible holes, crevices, and corners (Type I); surface films (Type II); and resolidified material either tightly adherent to the RV components or inaccessible for defueling (Type III). A drawing depicting the fuel debris at the base of the RV is provided in Figure 2-2.

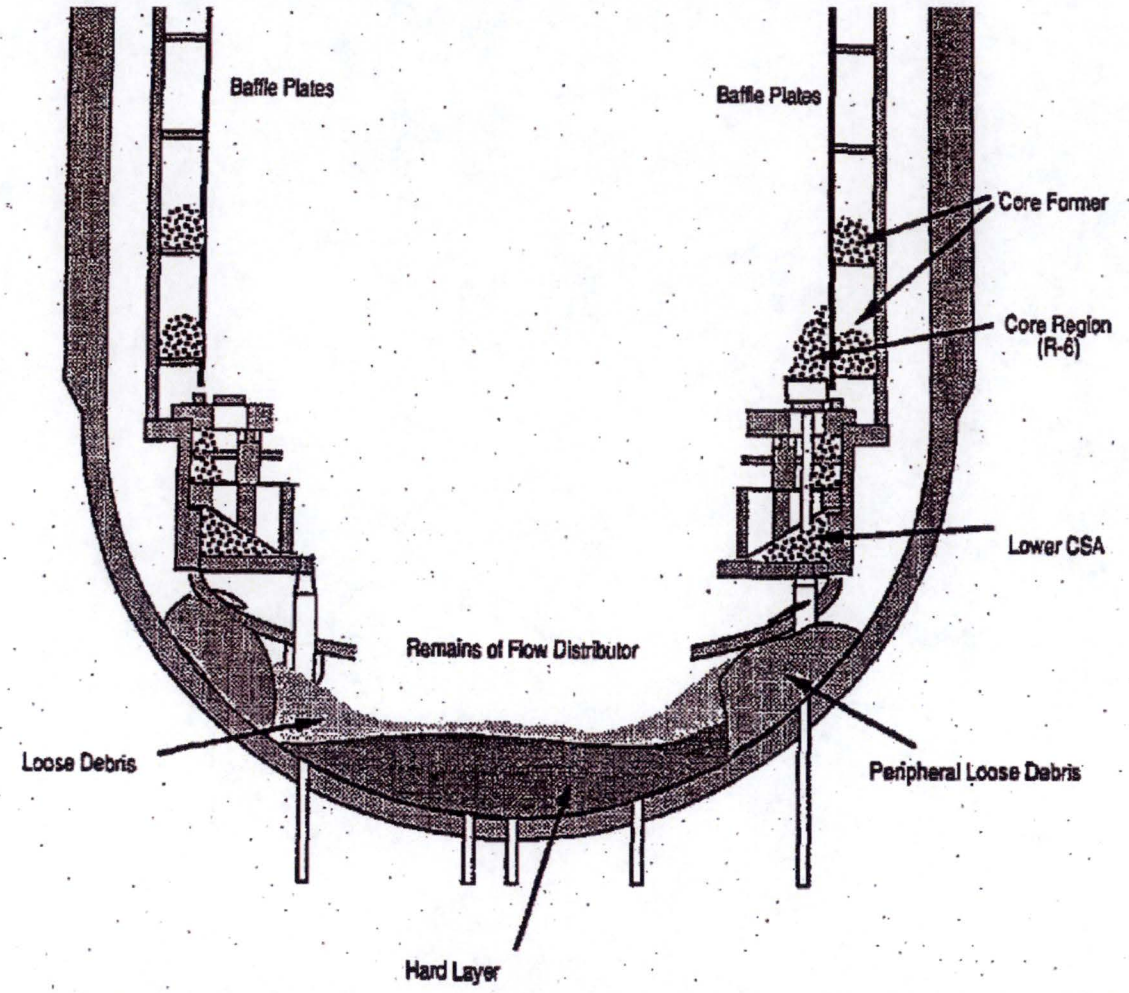

Figure 2-2. Remaining Fuel Debris at the Bottom of the Reactor Vessel (Ref. [1], Figure 5-30)

Approximately 50 samples of residual core debris were analyzed for density and fuel (UO_2) content. An average density and representative fuel content were determined for each type of material. A mass estimate was then obtained by multiplying the estimated core debris volume by the average density and UO_2 percentage corresponding to the particular type of material (i.e., Type I, II, or III).

The initial estimate using visual inspection and sample analysis was assumed to represent a conservative upper bound of the quantity of residual fuel in the RV. However, the range of uncertainty was extensive, and the video estimate was not sufficient for the Special Nuclear Material (SNM) Accountability Program. For this reason, a secondary estimate was performed.

2.1.2 Secondary Estimate

The final estimate of record is based on underwater video inspections and passive neutron measurements of the residual fuel, adjusted to account for measurement bias. The passive

neutron measurement technique utilized the results of the video inspections performed previously (Ref. [2]) to divide the RV into nine zones which separated the major fuel deposits by elevation (see Figure 2-3). The passive neutron measurement data was reviewed by a group of scientists (referred to as the Rasmussen Committee) and measurement bias adjustments were made to establish the final estimate of record for the RV.

The visual estimate of the residual fuel in the RV was 630 kg UO_2 (Ref. [1], [2]). The data presented showed a residual fuel inventory of 609 kg; however, for comparison with the passive neutron measurement estimate and the estimate of record, the 21 kg extant in the hot and cold leg nozzles (Ref. [12]) was added to achieve the stated 630 kg. The passive neutron measurements produced raw neutron counts for a series of locations within the RV (each zone) under known conditions. A minimum and maximum fuel value for each zone was determined and the average, using passive neutron measurement, was 1322 kg UO_2 (Ref. [4]).

Since the passive neutron estimate was more than twice the visual estimate, which was considered conservative, the Rasmussen Committee was commissioned to evaluate the measurement data, measurement technique, and residual fuel estimate. The Committee concluded that there were five systematic measurement biases in the passive neutron measurement techniques and analysis. These biases, related to boron variations, UO_2 particle size, calibration error, data analysis, and in-scattering of neutrons, were difficult to quantify with any reasonable accuracy. Biases were assigned to each zone and the estimate of record was adjusted accordingly to 925 kg UO_2. Figure 2-3 provides a comparison of the three fuel estimates (i.e., visual estimate, passive neutron estimate, and estimate of record).

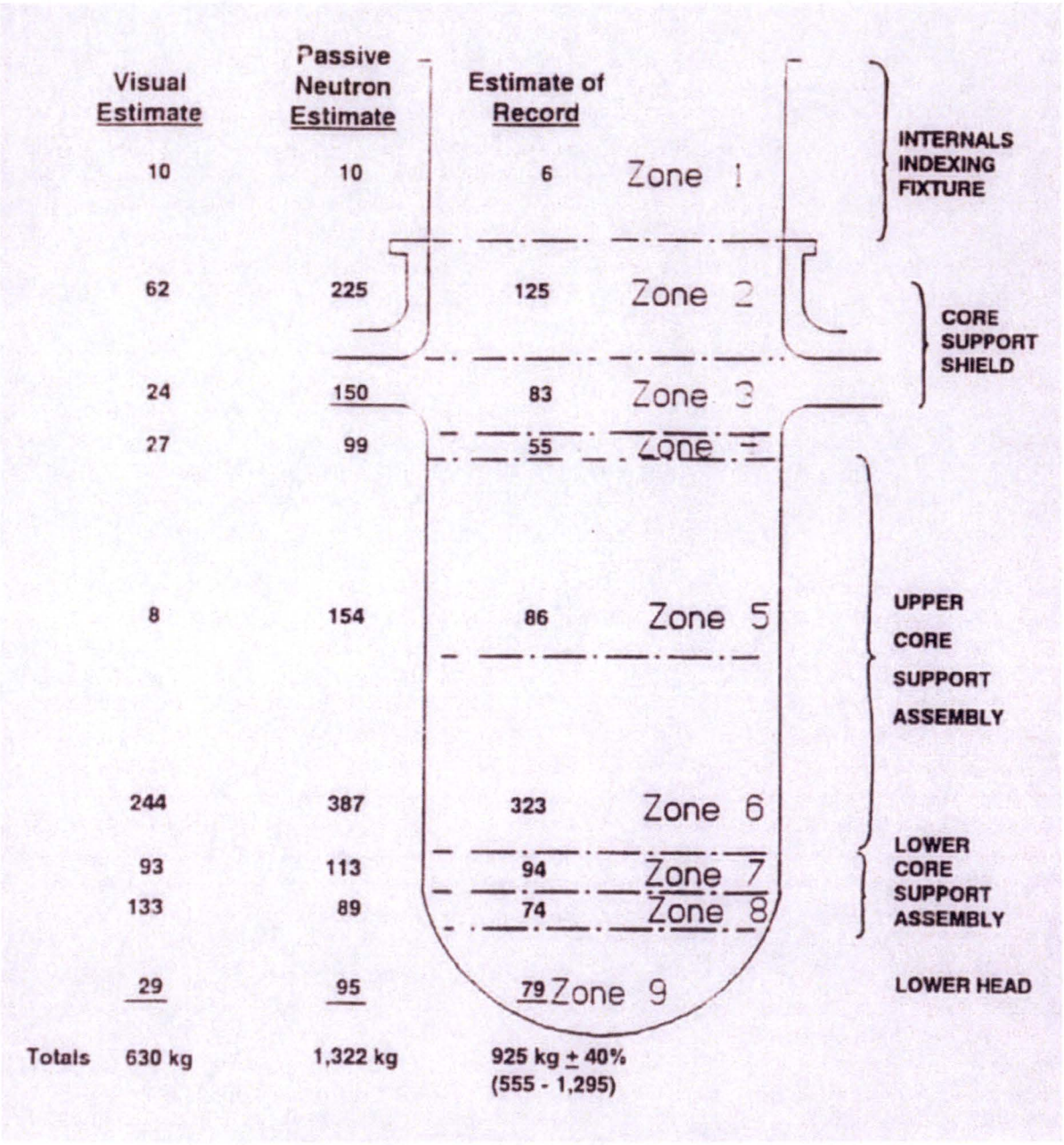

Figure 2-3. Comparison of RV Fuel Estimates (Ref. [4], Figure 2)

The passive neutron measurement report (Ref. [13]) includes an uncertainty estimate of approximately ± 15% for the estimated non-bias adjusted RV content of 1322 kg. The Rasmussen Committee judged the uncertainty to be ± 40% (at 1σ) when bias adjustments are made to the passive neutron estimate.

The estimate of record for residual fuel in the RV is 925 kg UO_2 with the Committee's assigned uncertainty of ± 40%. This is about 50% larger than the video estimate and 29% smaller than the

passive neutron measurement. If 2σ uncertainties (i.e., 95% confidence level) are considered, the estimate of record with its estimated 2σ uncertainty of 80% bounds both the visual estimate (630 kg) plus any reasonable uncertainty, and essentially bounds the unadjusted passive neutron estimate of 1322 kg plus its 2σ uncertainty (30%). Therefore, it has been considered reasonable to adopt 925 kg UO_2 as the estimate of record for the RV. This estimate is used in the justification of the SFML.

2.2 Auxiliary and Fuel Handling Buildings

The Auxiliary and Fuel Handling Buildings (AFHB) are divided into two parts that are separated by a common wall. One part, the Auxiliary Building, contains tanks, pumps, piping, and other equipment used for processing and storage of water for the reactor and primary cooling system, and for the treatment of radioactive wastes. The other part, the Fuel Handling Building, contains fuel handling and storage equipment and a limited number of tanks, filters, pumps and coolers, and other facilities. The interior of the AFHB was contaminated by radioactive material as a consequence of the accident and cleanup activities.

2.2.1 Initial Estimates

Early estimates (Ref. [1]) reported that approximately 25 kg of UO_2 was transported to the AFHB during the accident sequence. Additionally, certain defueling water processing activities were considered to have transported as much as an additional 15 kg of fuel from the reactor building into the AFHB for a total of 40 kg. Subsequent decontamination and water processing have reduced the AFHB residual fuel inventory to its current level of approximately 12 kg UO_2.

The majority of the 26 systems in the AFHB were either never contaminated or have been flushed and drained in preparation for post-defueling monitored storage. Most of the remaining fuel is contained within two systems: the Makeup and Purification System (MU&P), and the Liquid Radwaste Disposal System (WDL), with the remainder in various components, tanks, and pipes. The residual fuel is believed to be in the form of finely divided particles and sediment material with some minor amounts of fuel remaining as adherent film.

Post-defueling measurement methods for the AFHB were limited to gamma scintillation (NaI detector) for all except 10 areas. In those 10 areas, gross gamma exposure rates were measured in six and individual samples of fuel/resin were taken from the remaining four. Extrapolations were made of residual fuel in 29 areas that were based on measurements taken in other areas subjected to similar accident conditions. Details of the NaI detector measurements can be found in References [14] and [15], and details of the HPGe detector measurements in Reference [16]. The core debris sample data was correlated with the gross gamma measurements to estimate the amount of fuel present. Details of this analysis can be found in Reference [17].

2.2.2 Secondary Estimate

The final estimate of record for the AFHB was performed by employing gamma detection measurements and engineering analyses. Engineering analyses were performed in areas where the total quantity was believed to be insignificant and where accident flowpaths were established.

Two major plant systems contained 67% of the residual fuel in the AFHB. The two systems, as stated previously, are the MU&P and the WDL. The other supporting systems contained the remaining quantities of fuel as listed in Table 2-2. The analysis for each plant system can be found in the appendices of Reference [9]. Some individual areas within a system were judged to not require a fissile material assessment as it was not considered credible for them to contain fuel material.

Table 2-2. Summary of UO_2 Inventory for AFHB Systems (Ref. [9], Table 2)

System	Fuel Quantity (kg UO_2)	Uncertainty
Makeup & Purification	2.81	± 27%
Liquid Radwaste Disposal	4.13	± 71%
Solids Radwaste Disposal	0.01	± 57%
Balance of Systems	3.80	+34%, -92%
Other Areas	< 0.005	-
Total	10.75	

The entire MU&P System was determined to contain 2.81 kg UO_2 ± 27% plus 0.60 kg UO_2 (minimum detectable level [MDL] value). Conservative fuel mass assignments were made in a few areas of the MU&P that were unmeasured. These were based on their operational history during the accident and their service during the recovery phase. The WDL System was determined to contain 4.13 kg UO_2 ± 71% plus 0.11 kg UO_2 (MDL value). Similar conservative estimates were made in a few areas of the WDL that were unmeasured. The largest single deposit in the AFHB is in Spent Fuel Pool A, at 3.8 kg UO_2, which was surveyed while flooded.

To finalize the estimate of record for the AFHB, a total of 88 areas were measured or analyzed for fuel content, and 43 areas were evaluated or judged not to contain any fuel. Approximately 10% of the process piping was assumed to be embedded in concrete walls and floors. Each system has a total contribution estimation of 0.21 kg of UO_2 for embedded piping.

The overall estimate of record for the AFHB is 11.5 kg UO_2 with an overall range of values from 4.22 kg to 15.78 kg UO_2. The estimate of record includes the total in Table 2-2 and the total MDL of 0.71 kg. The range includes a 1σ uncertainty to represent the error of each estimate and as a total represents approximately 1.5% of the remaining TMI-2 fuel inventory.

2.3 Reactor Coolant System

The Reactor Coolant System (RCS) was designed to transfer thermal energy from the reactor core to the once through steam generators (OTSGs). The RCS consists of the RV, two vertical OTSGs, four shaft sealed reactor coolant pumps, an electrically heated pressurizer, and interconnecting piping. During the TMI-2 accident, core debris was transported to the reactor

building as a result of the core degradation event and coolant flow from the RV through the pilot operated relief valve (PORV) and the RCS MU&P System.
The Post-Defueling Survey Report for the RCS (Ref. [11]) analyzes multiple sections of the RCS as follows. Any remaining parts of the RCS are considered in separate areas.

- Reactor coolant pumps: Core debris was circulated through the four reactor coolant pumps during their short operation following the accident.
- Core flood lines "A" and "B": During repressurization following the accident, back pressure may have forced fuel debris into the core floor piping on the "A" side (Ref. [18]). Fuel debris has been detected beyond the appropriate valve and observed in the piping between the RV and valves.
- Cold legs 1A/B and 2 A/B: The RCS coolant flow deposited fuel debris in these pipe sections.
- Hot legs A and B: Coolant flow exiting the RV deposited fuel debris in the 12-ft horizontal section of these two pipe segments.
- Decay heat drop line: During the accident, RCS coolant exiting the RV moved past the entrance of the decay heat line and deposited fuel debris into the vertical piping segment.
- RCS nozzles: The hot leg nozzles are assumed to contain no fuel debris because of the fuel removal efforts, which effectively removed the debris deposits. The cold leg nozzles are considered separately as part of the RV.
- Pressurizer lines: The PORV stuck open early in the accident resulting in coolant flow through the pressurizer surge line into the pressurizer and out through the relief valve. The valve was cycled opened and closed subsequent to core damage. Fuel debris in the pressurizer spray line was flushed back into the pressurizer and removed during defueling operations. It is assumed that no measurable quantity of fuel remains in the pressurizer spray line.

Estimates of record for fuel debris in each of the above sections of the RCS were made using engineering analysis, video inspection, and gross gamma measurements, in some cases combined with the use of computer modeling. Additionally, reactor fuel was present in the form of surface films on RCS components. Several RCS components were analyzed to determine the quantity of UO_2 present within the adherent and loosely adherent surface film layers through sampling. The total fuel estimate for RCS surface film is 4.6 kg UO_2 with an uncertainty of $\pm$ 60% (at 1σ). A summary of the fuel inventory for each of the RCS areas, in addition to the total surface films, is in Table 2-3.

Table 2-3. Summary of UO_2 Inventory for RCS Areas (Ref. [11], Table 2)

Area/Component	Fuel Quantity (kg UO_2)	Uncertainty
Reactor Coolant Pumps	6.2	± 55%
Core Floor Lines 'A' and 'B'	1.0	
Cold Legs 1 A/B and 2 A/B	9.6	
Hot Legs 'A' and 'B'	2.7	
Decay Heat Drop Line	1.5	
Pressurizer Lines	0.2	± 60%
Surface Films	4.6	± 60%
Total	**25.8**	**± 43%**

The estimate of record of the quantity of UO_2 in the RCS is 25.8 kg ± 43%. The data shows that 61% of the remaining fuel is contained in two component groupings: the reactor coolant pumps and the cold legs (Ref. [11]). The UO_2 remaining in the RCS components represents less than 4% of the total remaining TMI-2 inventory.

2.4 Steam Generators

The OTSGs are vertical, straight tube and shell boilers in which reactor coolant is on the tube side and the secondary coolant is on the shell side. The TMI-2 'A' and 'B' OTSGs were used for transferring heat from the RCS to the secondary system.

As a result of the accident, fuel was transported through the hot legs into the OTSGs by a variety of pathway mechanisms which include drain and refill of the steam generator and inlet piping, forced circulation from the coolant pumps, natural circulation, and fluid movement resulting from energy releases into the reactor vessel (Ref. [19]). Fuel fines and sediment material were transported to the OTSGs by reactor coolant through the hot legs to the steam generator upper head and tube sheet. Much of the larger particulate material settled out on the upper tube sheet with the finer material being transported through the tube bundles to the lower head and outlet piping. During late 1987, the debris material on the upper tube sheets was removed.

As part of the effort to develop an estimate of record, gamma measurement surveys and neutron activation measurements, in some cases coupled with computer models, were performed for fuel characterization of the 'A' and 'B' OTSG upper tube sheets, tube bundles, lower heads, and associated cold legs during September 1988 to mid-January 1989. The composition of the debris used in modeling was based on analysis of samples taken from the J-legs of both steam generators (Ref. [20]). Fuel measurement strings and copper foils for neutron activation were used to deploy gross gamma probes into difficult to access areas of the OTSGs and assess the possibility of fuel debris blocking the OTSG tubes.

The estimates of record for each area of the OTSGs are provided in Table 2-4. The total UO_2 estimate of record for the 'A' OTSG is 7.1 kg ± 33% and for the 'B' OTSG is 55.2 kg ± 17%.

The uncertainty associated with these fuel estimates (except for the 'B' OTSG upper tube sheet) is based on modeling geometry errors, source calibration errors, Geiger-Mueller probe response errors, and cesium-fuel ratio errors. The uncertainty associated with the 'B' OTSG upper tube sheet is a result of neutron activation measurement errors.

Table 2-4. Summary of UO_2 Inventory for Once-Through Steam Generator (Ref. [8], Table 1)

Area	'A' OSTG (kg UO_2)	'B' OTSG (kg UO_2)
Upper Tube Sheet	1.4 kg ± 21%	36.0 kg ± 18%
Tube Bundle	1.7 kg ± 48%	9.1 kg ± 48%
Lower Head	1.4 kg ± 55%	2.2 kg ± 55%
RCP-1 J-Leg	1.5 kg ± 55%	1.9 kg ± 55%
RCP-2 J-Leg	1.1 kg ± 55%	6.0 kg ± 55%
Total	**7.1 kg ± 33 %**	**55.2 kg ± 17%**

2.5 Upper Plenum Assembly

The plenum is normally located directly above the reactor core. It consists of a cover, control rod assembly (CRA) guide tube assemblies, upper grid (the bottom of the plenum), and a flanged plenum cylinder with openings for reactor coolant outlet flow. The leadscrews, which move the CRAs in and out of the core, were inside the guide tubes during the March 1979 accident. A simple schematic showing the plenum and leadscrews is provided in Figure 2-4.

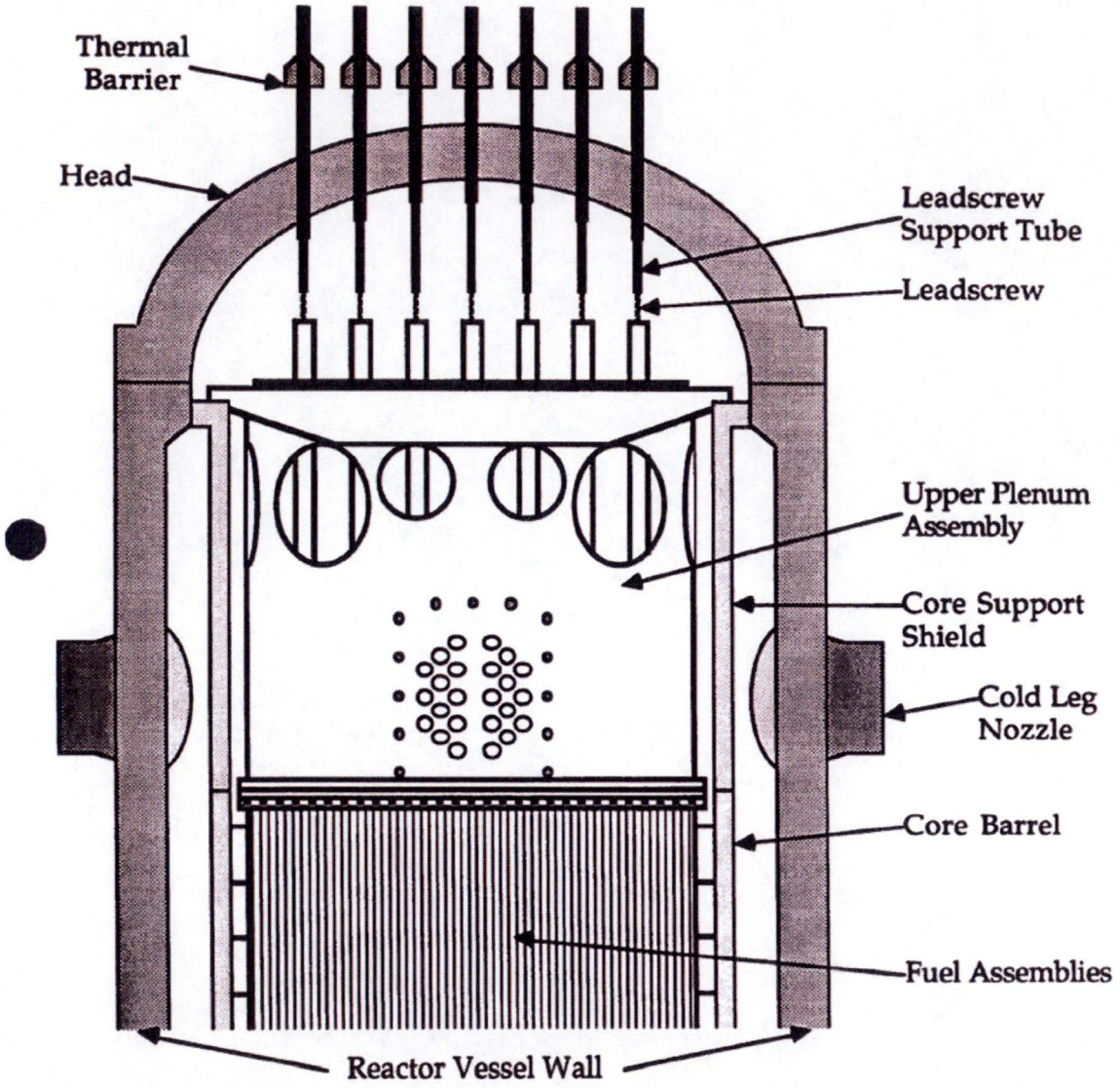

Figure 2-4. Upper Half of TMI-2 Reactor

During the March 1979 accident, fuel was transported to the plenum when large amounts of reactor coolant flow, steam, and hydrogen passed through it, depositing sediment and surface films on the plenum surfaces.

The 69 guide tubes are vertical cylinders that constitute the majority of the surface area (~69%) in the plenum assembly. Each guide tube has 10 spacers and a base plate that guides the CRAs. These structures are the primary horizontal settling area for sediment in the plenum; however, they only represent a relatively small fraction (~3.5%) of the total plenum surface area. The total surface area of the plenum is calculated to be 4.3×10^6 cm^2. The plenum's upward facing

horizontal surfaces, where debris could collect, include the cover plate, the top of the upper grid, and the spacer castings in the control rod guide tubes.

The estimate of fuel loading in the plenum is based on analysis of samples from two leadscrews and one control rod guide tube section. Fuel contamination in the plenum, as on the leadscrews, is assumed to be contained in the adherent film layers and fine sediment on metal surfaces. The surface deposits and films from samples were analyzed by delayed neutron emission at Idaho National Engineering Laboratory (INEL). Video inspections of a representative area (50% of plenum surface area) did not reveal any lodged debris or granular particles (Ref. [21]), indicating that granular fuel debris was not a significant contributor to the plenum UO_2 inventory.

The amount of fuel in the plenum was calculated using the leadscrew fuel concentration and distribution data, the leadscrew support tube contamination data, and the video inspection observations (Ref. [22]). The estimate of record of the amount of residual UO_2 in the upper plenum assembly is based upon the extrapolation of UO_2 quantities remaining on two sample leadscrews and a CRA support tube. The estimate of record is considered to be adequate because the leadscrews and CRA support tubes are composed of similar material to the plenum and were exposed to the same environment during and after the TMI-2 accident. Therefore, fuel deposition should be representative of the plenum.

The estimate of record for UO_2 remaining in the plenum structure is 2.1 kg UO_2, made up of 1.5 kg surface films and 0.6 kg silt/sediment. This estimate has an associated uncertainty of -45% to + 140%.

2.6 Pressurizer, Reactor Building Basement, Letdown Cooler Room

The TMI-2 pressurizer was intended to establish and maintain reactor coolant system pressure within the prescribed limits, and to provide a steam surge chamber and water reserve to accommodate RCS density changes during operation.

During the accident, the pressurizer water level was raised and lowered to thermally control the reactor pressure. Core material was relocated to the pressurizer, most likely as a result of this process. This is supported by the existence of a trace of core material in the reactor coolant drain tank (RCDT) and in the reactor building basement.

Quantification of residual UO_2 in the pressurizer was accomplished by estimating the volume of residual core debris after defueling and applying results from sample analysis. The estimate of record for fuel debris in the pressurizer is 0.3 kg $\pm$ 52% (at 1σ).

The reactor building basement consists of the space between two floors of the reactor building. Although the basement space houses several major components, the estimate of record for the basement does not include the fuel retained in those components. The estimate does include the entire floor area, including sediment, water, and walls. A cutout of the reactor building showing the basement is provided in Figure 2-5.

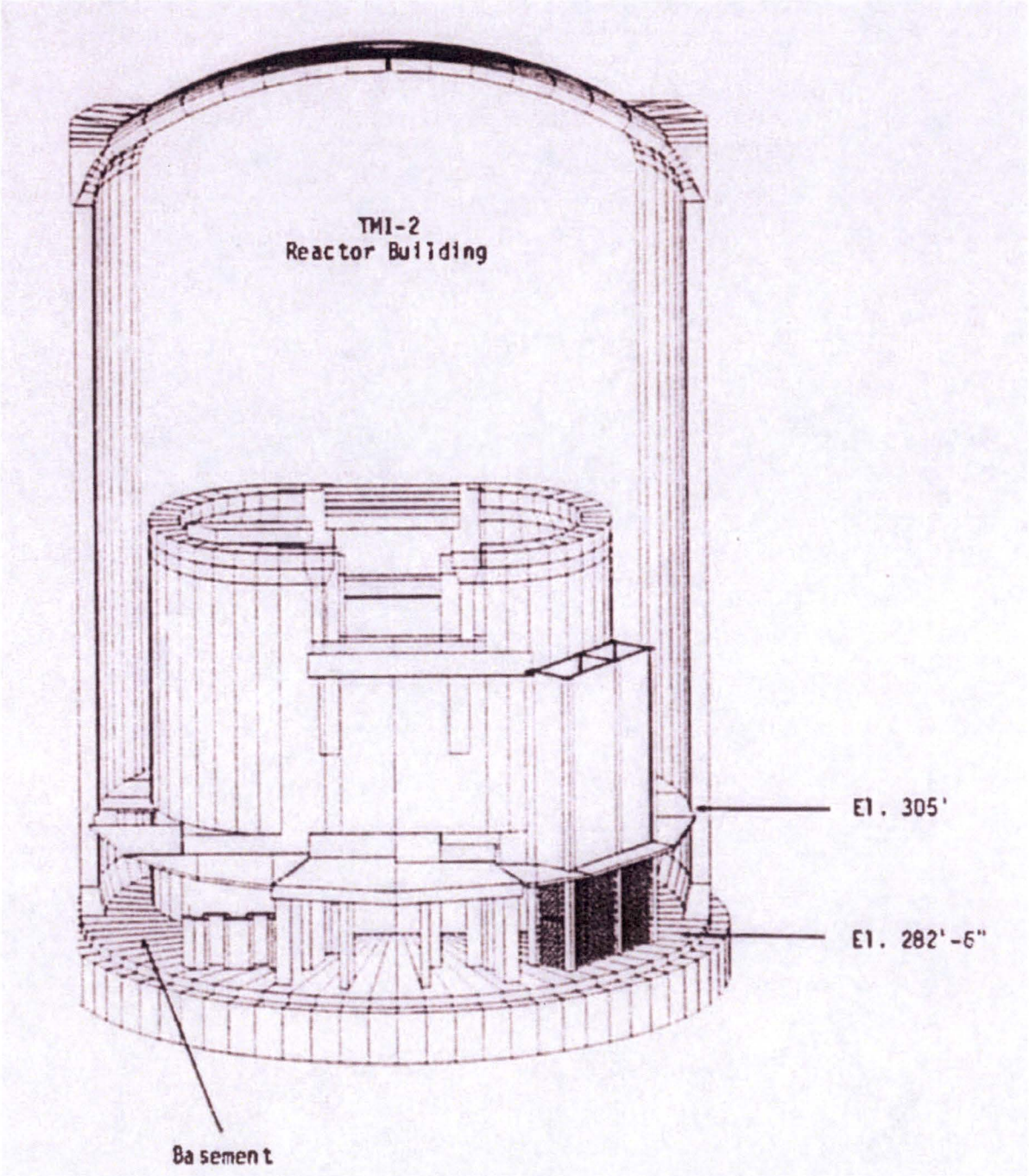

Figure 2-5. Reactor Building Basement Floor

During the accident, reactor coolant was discharged from the RCS to the reactor building basement via the PORV and RCDT. The discharged reactor coolant increased pressure in the RCDT and caused its rupture, which vented the RCDT to the reactor building basement. The reactor coolant that discharged into the RCDT and basement contained reactor fuel fines from the damaged core.

Quantification of residual UO_2 in the reactor building basement sediment was accomplished through remote sampling and gamma spectroscopy. Measurements on the basement concrete core did not indicate the presence of reactor fuel. The quantity of dissolved UO_2 in the water was low and therefore, it is assumed that the walls also contain minimal reactor fuel. The estimate of record for the UO_2 remaining in the basement is 1.3 kg ± 54% (at 1σ).

The letdown coolers are in the reactor building basement with the purpose to reduce the RCS water temperature from 293°C to 49 °C to prevent damage to the MU&P demineralizer resins. The RCS flow through the letdown coolers continued following the accident, at reduced flow rates. They have continued to be used as part of the letdown path from the RCS to maintain the RV water level, at a reduced flow rate. Because they are one of the low points of the MU&P system, they were examined explicitly for residual fuel.

Quantification of residual UO_2 in the letdown cooler room was accomplished through measurement via insertion of a collimated lead shielded sodium iodide detector through a penetration. Neutron interrogation and gamma spectroscopy were determined not feasible. Computer modeling was then used to apply the measurements to the geometry of the room; however, the tracer isotope was not positively identified so the estimate of record is assigned the MDL of ≤ 3.7 kg UO_2 ± 53% (at 1σ). The maximum flow through the associated piping is considered sufficient to have long ago removed fine fuel debris from the letdown coolers and to have prevented buildup of fuel debris (Ref. [6]).

2.7 Reactor Head

The RV head assembly covers the RV and serves as a containment barrier for the reactor coolant (see Figure 2-4). It has a hemispherical top portion attached to a cylindrical flange bottom portion. The hemispherical segment accommodates a protrusion of leadscrew motor housings (LSMHs), leadscrew support tubes (LSTs) and leadscrews. These components within the RV head are shown in Figure 2-6.

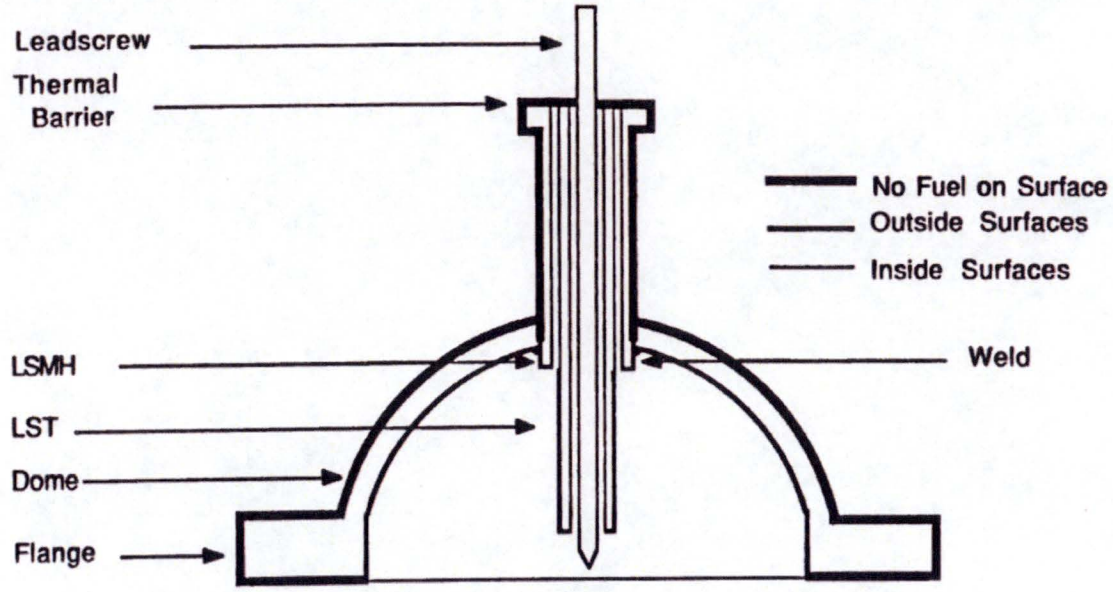

Figure 2-6. RV Head Assembly Component Surfaces (Ref. [7], Figure 4)

During the 1979 accident, fuel relocated to the RV head assembly when the water boiled, and steam rose within the vessel. The possibility of gravel-like deposits being trapped in the RV head assembly component is highly unlikely due to gravity and the RV head assembly orientation. Therefore, contributions to gravel-like material are insignificant when compared to surface films and loose contamination contributions.

The total fuel quantity was derived as the sum of three general increments: the fuel content on the leadscrews, the fuel on the exposed underhead surfaces, and the fuel on the hidden or inside underhead surfaces. The leadscrew increment was based on direct assay of the fuel content of various sections of two leadscrews. Radiochemical assays of a piece of leadscrew support tube and of the previous sections of leadscrew were used to infer fuel content for the remainder. Fuel analyses were performed by Battelle Columbus Laboratories, Scientific Applications Inc., and Babcock & Wilcox.

The estimate of record of UO_2 remaining in the RV head components is summarized in Table 2-5 for a total of 1.3 kg, primarily in the form of surface films and loose contamination. The associated uncertainty is -65% to +80%. It was calculated by extrapolation of data reported on residual UO_2 measured on the E9, B8, and H8 leadscrews. The locations of these are marked in Figure 2-7.

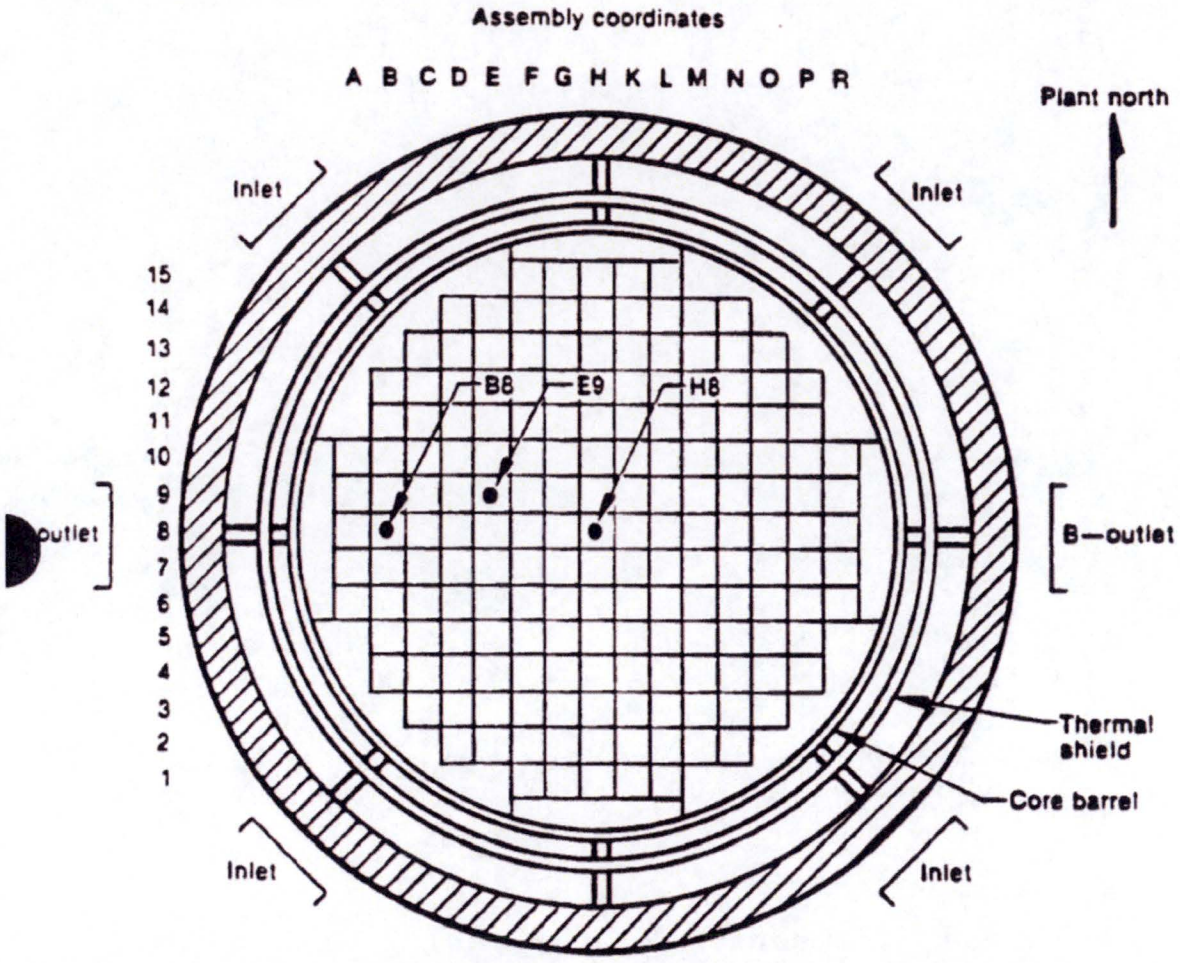

Figure 2-7. Sample Leadscrew Locations Shown in the TMI-2 Core (Ref. [7], Figure 3)

Table 2-5. Summary of UO_2 Inventory for RV Head Components (Ref. [7], Table 1)

Component	Number of Components	Total Fuel Quantity (g UO_2)
Flange	1	110.1
Dome (minus LSML holes)	1	174.6
LSMH	69	120.1
LST	69	457.9
Leadscrew	66	386.1
Total	-	**1248.8**

2.8 Miscellaneous Components

During the accident, approximately 10 kg of UO_2 was relocated to the reactor building sump and other reactor building locations. After the accident, fuel was relocated to the reactor building as a result of several cleanup operations including: transfer to and storage of structural RV components in the 'A' core flood tank and 'A' D-ring; storage of upper endfittings; flushing of defueling tools; and transfer of the defueling canisters into the fuel transfer canal.

The majority (62%) of the fuel debris in these components is in the fuel transfer canal (FTC)/transfer tubes and incore guide tubes stored in the 'A' D-ring. During the TMI-2 cleanup, 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 different sources, a major one being the vertical to horizontal rotation for transfer to Spent Fuel Pool A. The other sources are detailed in Reference [10]. Details of the data analysis performed on the incore guide tubes can be found in Reference [23].

Visual inspection was used to locate the fuel deposits, and measurements were performed using gross gamma exposure rates, gamma spectroscopy, neutron interrogation, and direct sampling and analysis. Engineering analyses were performed in areas where the total fuel quantity was believed to be insignificant (< 7 kg UO_2 or < 8% of the existing SFML).

The total estimate of record for the reactor building miscellaneous components is 64.0 kg UO_2. The breakdown on a component/system basis is provided in Table 2-6. Including 1σ uncertainty, the total residual UO_2 is expressed as a range from 31.6 kg to 85.4 kg UO_2. The UO_2 remaining in the reactor building miscellaneous components represents less than 10% of the total fuel remaining at TMI-2.

Table 2-6. Summary of UO_2 Inventory for RB Miscellaneous Components (Ref. [10], Table 2)

Area/Component	Fuel Quantity (kg UO_2)	Uncertainty
Reactor Coolant Drain Tank	0.1	± 54%
Core Floor System	4.9	± 77%
Upper Endfittings	5.9	± 85%
Tool Decon Facility (TDF)	0.1	± 100%
Drain Line From TDF	4.4	± 87%
Defueling Water Cleanup System	3.7	± 67%
Temporary RV Filter System	4.4	± 90%
Incore Guide Tubes	21.0	± 54%
Defueling Tools	0.6	± 75%
Fuel Transfer Canal/Transfer Tubes	18.9	-95%, +37%
Total	**64.0**	± 32%[(a)]

(a) The symmetrical uncertainty is associated with the subtotal that does not include the Fuel Transfer Canal/Transfer Tubes, which has an asymmetrical uncertainty.

The estimates of record for each system or area were a significant effort undertaken by experts in order to establish the most appropriate bounding yet realistic quantification of fissile material remaining. In describing the location of remaining fuel debris, it is apparent that a significant portion of material is scattered and held up within components with little expectation of relocation without intentional effort. Because the lower head of the RV is the lowest general point in the largest area, it would act as a settling location for disturbed material during decommissioning. Even so, there is still a measurable amount of material in other components that is unlikely to have any flowpath to the lower head.

3.0 Debris Characterization

Immediately after the TMI-2 accident, four organizations (both state and federal government and private industry) with interests in both plant recovery and accident data acquisition formally agreed to cooperate in these areas. These organizations, commonly referred to as the GEND Group – GPU Nuclear Corporation, Electric Power Research Institute, Nuclear Regulatory Commission, and Department of Energy – were all actively involved in reactor recovery and accident research (Ref. [24]).

The TMI-2 Accident Evaluation Program report (Ref. [24]) defines the program required to implement the DOE assignments and contains the guidelines and requirements for TMI-2 sample acquisition and examinations. A major task of the TMI-2 Accident Evaluation Program was to examine fuel samples taken from the core and lower vessel. The TMI-2 Accident Evaluation Program Sample Acquisition and Examination Plan (Ref. [25]) developed a test and inspection plan to characterize 1) the TMI-2 equipment that may have experienced core damage, and 2) the

fission product inventory. The equipment characterization program included both sample acquisition and examination, and in-situ measurements. Fission product characterization included locating the fission products as well as determining their chemical form and material association.

Fuel removal was initiated on November 12, 1985. During 1986, fuel removal was limited to the core cavity walls and floor and consisted of upper end fittings from fuel, control rod and burnable poison rod assemblies, partial fuel assemblies, and unsegregated loose debris. This represents a cumulative total weight of 51,000 lbs of 300,000 lbs of material that was present in the reactor core. A total of 49 fuel canisters had been loaded and transferred to the TMI Fuel Handling Building by October 1986 and 21 of them were shipped to INEL. The core findings of this effort were[1] (Ref. [25]):

- Large regions of the core exceeded cladding melting (~2200K), and significant fuel liquefaction by molten zircaloy and some fuel melting occurred with temperatures up to at least 3100K.
- Core materials relocated into the RV lower plenum region from the core, leaving a void in the upper core region equivalent to approximately 26% of the original core volume. Between two and twenty metric tons of core and structural material now reside in the space between the bottom head and elliptical flow distributor.
- Fission product retention in core materials is significant, and the retention of fission products outside the core was primarily in RCS water, water in the basement, and in basement sediment.

The highest priority of the sample and examination plan was to provide data that directly characterized core damage progression and fission product release from the fuel. Next in relative importance are data that characterized retained fission products in the containment basement, fission product chemical form, and structural damage within the lower plenum. The lowest priority data are those related to fission product retention in the primary cooling system and structural peak temperatures. Some of the more important data needs are identified below along with subsequent references that reported piecemeal results. This was merely the beginning of planned sampling efforts and is not meant to be an exhaustive list of all samples taken or all reports produced.

- Core bore samples – 30 core bore samples (10 high, 10 medium, 10 low) allow interpretation of the core damage (i.e., cladding melting, fuel liquefaction and relocation, freezer/melting/slumping of core materials). See Reference [26] for results.
- Core debris grab samples – 11 samples (small localized and large volume) provided physiochemical data to evaluate material interactions and fission product behavior. See References [27], [28], and [29] for results.

1. These findings were dictated prior to the defueling efforts of the late 1980s. Statements to the amount of fuel debris and its location are not necessarily valid statements as to the state of material as of the time of this writing.

- Intact rod segments – segments from fuel rods, control rods, and poison rods provided information on peak temperature, material interactions, retained fission products, and fission product chemical form. See Reference [30] for results.
- Core support assembly samples – provided information on the interaction between core materials and stainless-steel structures. See Reference [31] for position H8 leadscrew support tube results.
- Reactor Coolant System samples – provided information on the fission product chemical form and fission product interaction with structural materials. See Reference [32] for B OTSG tube sheet results and Reference [33] for liquid RCS sample results.

Note that for each sample retrieved during the sampling campaigns, it was processed into specimens which were then analyzed individually. One sample can have multiple specimens of varying sizes and elemental makeup and may have been processed at different facilities.

During defueling, the majority of the material in the debris bed of the core was removed. For this reason, more detail is given on the sample results for the lower head region than those for the core debris bed since this is the bulk of the material that will be handled during decommissioning.

3.1 Reference Data for Lower Vessel Head

In 1985, 16 pieces of prior-molten debris from the lower reactor vessel (i.e., lower head) were obtained through the annulus, between the thermal shield and the reactor vessel. The pieces were intended to be both representative of the region "typical material" and a material that "looked different." They were divided into subsamples and analyzed in two laboratories. Eight[2] of the subsamples, which broke into more pieces during shipping, were examined at INEL to obtain data on the physical properties, material structure constituents, and radiochemical composition of the debris. Portions of the same subsamples were sent to Argonne National Laboratory for metallurgical analysis, and Westinghouse Hanford for elemental analysis. The analysis and results are documented in GEND-INF-0084, *Examination of Debris from the Lower Head of the TMI-2 Reactor* (Ref. [27]) and summarized in this section. The remaining eight pieces were reserved for examination at the TMI on-site laboratory. Note that each piece received by a laboratory is broken down further for the various analyses performed. "Piece" is used to refer to the debris chunk as a whole and "sample" or "subsample" is used to refer to the slices/sections broken down for analysis.

The density of the pieces ranged from 6.57 g/cm^3 to 8.25 g/cm^3, with an average value of 7.07 g/cm^3 and an average porosity of about 25%. Two small areas of ceramic material containing only U and Zr were found in the largest piece and it appeared the UO_2 was subjected to temperatures near the melting point; therefore, the minimum peak temperature of this piece was near 3100K.

2. Not all examinations were performed on each sample piece at INEL. Elemental analysis was only performed on seven pieces.

Elemental analyses were performed on the dissolved lower vessel debris samples for 17 elements that constitute the principal components of the TMI-2 core. This work was performed at Hanford Engineering Development Laboratory. As explained below, the data indicate that the composition of the debris is similar to what would be expected if the principal components of the original core were mixed, melted, and a fraction subsequently relocated to the lower plenum of the reactor. The general appearance of the lower plenum debris and the microstructure of the subsamples indicate that they were once molten. The particles of pure UO_2 have changes in porosity that indicate it may have been near the melting point.

Although these analyses were performed prior to the defueling, most of the remaining material is in the same region as these analyzed pieces. Regardless, the data continues to aid in understanding the composition of the fuel debris remaining in the core and serves to aid in creating a homogeneous impurity composition for development of a revised SFML.

For comparison to referenced analytical data, Table 3-1 lists the average composition (over all components) of each of the core elemental constituents. Table 3-2 lists the highest, lowest, and average weighted distribution for the pieces and compares them to the core composition (initially provided in Table 3-1). These results come from 37 subsamples taken from the seven[3] pieces, with results binned per piece (an average of subsamples from the same piece). The suite of analytical results for these pieces and subsamples can be found in Appendix E of Reference [27]. Elements of particular interest to the SFML determination are U and B, which are seen in relatively similar concentrations in the analytical average to the initial core average.

3. There are nine Debris Piece IDs used in a following table. Two of the ID numbers are associated with the same initial piece.

Table 3-1 – TMI-2 Core Composition by Element Weight Percent (Ref. [27], Table 8)

Element	Composition Including Oxygen (wt.%)	Composition Including Oxygen (wt.%)[(a)]
U	71.94	65.8
Zr	19.70	18.0
O	--	8.5
Fe	3.28	3.0
Ag	1.91	1.8
Cr	1.09	1.0
Ni	0.99	0.9
In	0.35	0.3
Sn	0.35	0.3
Al	0.19	0.2
B	0.15	0.1
Cd	0.12	0.1
Mn	0.10	0.08
Nb	0.05	0.04
Si	0.04	0.04
C	0.04	0.04
Mo	0.03	0.03
Gd	0.01	0.01
Ti	0.01	0.01
N	0.01	0.01
Co	0.01	0.01

(a) Oxygen content of 11.8 wt.% for UO_2 is assumed.
Additional oxygen may range from 4 to 7%, depending on degree of oxidation of zircaloy and structural materials.

Table 3-2 – Elemental Composition Among Lower Vessel Debris Pieces (deduced from Ref. [27], Table 9)

Element	TMI-2 Core Average (wt.%)	Highest Distribution (wt.%)	Lowest Distribution (wt.%)	Average Distribution[a] (wt.%)	Average Distribution[c] (wt.%)
U	65.8	69.5	62.3	65.1	N/A
Zr	18	15.0	11.7	12.6	N/A
Ag	1.8	0.22	Below detect	0.22	0.024
Al	0.2	Not in analysis	Not in analysis	Not in analysis	Not in analysis
Cu	0.001	0.46	Below detect	0.335	0.074
Fe	3.0	3.70	1.83	2.44	N/A
B	0.1	0.36	Below detect	0.125	0.11
Cr	1.0	0.99	0.58	0.75	N/A
Si	0.04	Not in analysis	Not in analysis	Not in analysis	Not in analysis
Sn	0.3	Below detect	Below detect	Below detect	0
Mg	Negligible	Not in analysis	Not in analysis	Not in analysis	Not in analysis
In	0.3	Below detect	Below detect	Below detect	0
Cd	0.1	0.065	Below detect	0.045	0.01
Gd	0.01	Below detect	Below detect	Below detect	0
Mn	0.08	0.089	Below detect	0.075	0.058
Mo	0.03	0.21	0.093	0.15	N/A
Nb	0.04	Below detect	Below detect	Below detect	0
Ni	0.9	0.26	Below detect	0.235	0.104

(a) Since detection limits are not all known, the average only includes reported values. Below detect results are omitted from the average.
(b) "Not in analysis" indicates results are omitted because the sample was contaminated with the element during handling.
(c) This average includes below detect results, averaged in as zeros.

A summarized discussion of the analytical results is divided into the four principal groups of materials present in the core:

- Uranium fuel and zircaloy cladding
- Ag-In-Cd control rod materials
- Neutron poisons (B_4C-Al_2O_3 and Gd_2O_3-UO_2)
- Structural materials (stainless steel and Inconel)

3.1.1 Uranium Fuel and Zircaloy Cladding

The first principal group of materials considered was uranium fuel and zircaloy cladding. The analytical results for concentration indicated a narrow range of U and Zr composition for all pieces. The lowest U concentration distribution was 62.3 wt.% (Ref. [27], Table 9); the highest was 69.5 wt.%, with an average for all pieces of 65.1 wt.%. For Zr, the lowest concentration distribution was 11.7 wt.% (Ref. [27], Table 9); the highest was 15.0 wt.%, with an average of 12.6 wt.%. Tin, the other principal component of the cladding, was not measurable. These data indicate that the lower vessel debris is relatively homogeneous and that U and Zr are the two principal components of the debris.

The core average concentrations of U and Zr were expected to be about 65 and 18 wt.%, respectively, excluding oxidation of the zircaloy and structural materials. The variation from expected Zr concentration to that measured is likely due to a) oxidation of the Zr, which would reduce the total amount of Zr present relative to the U content, and b) analysis uncertainty. The total oxygen content of the debris, assuming 65 wt.% U and 18 wt.% Zr and stoichiometric oxidation, would be 14 to 16 wt.%. Therefore, if the total measured material contents listed in Table 3-2 have oxygen added based on the actual composition of the subsamples and in the appropriate amounts to oxidize U, Zr, and other structural material, the data indicate that within the uncertainty, all material in the subsamples has been accounted for through inductively coupled plasma analysis.

The average Zr/U ratio expected in the core is 0.28 and the average measured ratio in the pieces is 0.19. However, the U and Zr concentrations throughout the debris pieces are quite consistent, with results in the range of 60 – 70% and 10 – 15%, respectively. This is the range of concentrations expected if the fuel, cladding, and core structural materials were melted, oxidized, and mixed. These data suggest that the lower vessel debris pieces were relatively evenly mixed and transported to the lower reactor vessel without significant ablation of the structural materials in the flow pathway. This is substantiated by the structural material results (i.e., Fe, Cr, Ni, and Mn), which are nearly the same as the expected core average concentrations of these elements.

3.1.2 Control Rod Materials

Small amounts of Ag and Cd (from control rods) were measured in the lower vessel debris, although substantially less than the expected average concentration of those elements in the core. The measured concentrations were also near the detection limits of the equipment (Ag: 0.2 – 0.4 wt.%; In: 2-4 wt.%). Indium was not measurable in the lower vessel debris, but the detection limit was higher than for Ag.

The control rod materials (Ag, In, Cd) are significant fractions of the original core mass [i.e., Ag (1.8 wt.%), In (0.3 wt.%), and Cd (0.1 wt.%)]. These elements were essentially not measurable in the lower vessel pieces. There were three positive detects (one Ag and two Cd) among the three elements in the nine analyzed pieces (Ref. [27], Table 9). This data suggests that the lower vessel debris (upper layer of lower head material) was substantially depleted of control rod materials. It is possible that the control rod materials relocated to the lower reactor vessel head (lower layer of material) (Ref. [27]).

3.1.3 Poison Materials

The neutron poisons present in the TMI-2 core consist of material from the two types of burnable poison rods and the substantial amounts of boron in the reactor coolant. The two types of burnable poison rods are boron carbide/alumina rods, and gadolinia rods. Of the two possible poisons (B and Gd) in the TMI-2 reactor, only boron was found in measurable concentrations of these samples (other than one ' below detect', 0.066 to 0.36 wt.% on average). Identifying the source of the boron, either from coolant or poison rods, was not possible. Gd was shown to be highly concentrated in the core debris bed (see Section 3.3).

Boron concentration results from each subsample (i.e., slice, chunk) of the mother debris piece are provided in Table 3-3. Some subsamples are obtained from the edge of the debris piece, and some from the interior. It is not clear whence each originated; however, the original analysts state that the data does indicate that the boron poison is deposited at a relatively constant concentration evenly through the interior of the subsamples. The concentrations found in the samples are consistent with the amount of boron in the reactor coolant (Ref. [27]).

Table 3-3. Boron Concentration Results for Each Lower Vessel Debris Subsample (deduced from Appendix E, Ref. [27])

Debris Piece ID	Sample ID	Boron Concentration (wt.%)
11-1-C	400	< 0.068
	401	< 0.046
11-4-D	402	< 0.053
	403	0.12
	404	< 0.044
11-6-B	405	0.64
	406	0.071
	407	< 0.049
7-1-B	409	0.14
	411	0.063
	412	0.079
11-4-B	414	0.088
	415	0.050
	416	0.059
11-7-C	417	0.039
	418	0.16
	419[a]	0.089
11-2-C	421	0.059
	422	0.068
	423	0.083
	424	< 0.082
	425	0.075
11-1-A	427	0.18
	428	0.069
11-5-C	429	0.081
	430	0.092
	431	0.11
	432	0.098
	434	0.033
	435	0.038
	436	0.051
	437	0.11

(a) Analysis results indicate possible loss of sample material.

3.1.4 Structural Material

The structural material concentrations listed in Table 3-2 indicate that substantial amounts of structural material have been retained in the prior-molten material. The element with the highest concentration is Fe. This element has relatively constant concentrations, ranging from 1.8 to 3.7 wt.%. There is also no evidence of significant gradients within, or concentrations of Fe on the surface of the particles, which suggests that the Fe was distributed in the particles when the debris was molten (Ref. [27]).

The average concentration of Cr in the core would be expected to be 1.0 wt.% if all core materials were evenly distributed in the mass. The lower vessel debris data gives concentrations ranging from 0.6 to 1.0 wt.%, which indicates that Cr is present in the lower vessel debris in the expected concentrations. For most particles, there is no evidence of concentration gradients (Ref. [27]).

The Ni concentrations measured in the lower vessel debris range from 0.2 to 0.26 wt.% where this element was measurable (about 50% of the samples). The detection limit (around 0.2 wt.%) is near the expected average concentration.

The concentration of Mn measurable at most sample locations ranged from 0.05 to 0.09 wt.%. This compares with an expected average concentration of 0.08 wt.%, suggesting similarity to those observed for Cr and Fe.

Molybdenum was measurable at most sample locations at concentrations higher than the expected core average concentration of 0.03 wt.%. The measured concentrations range from 0.04 to 0.27 wt.% (Ref. [27], Appendix E), which suggests that some of the structural material present in the lower vessel debris is Inconel 718 from the spacer grids, as there is no Mo present in 304 stainless steel. In the full-length fuel assembly, there is 15.1 kg of 304 stainless steel, mostly in the cladding of the control rods and 6.8 kg of Inconel. The measured concentration indicates that although there is more than expected amounts of Mo from the Inconel 718, there is probably a substantial fraction of 304 stainless steel also present.

Copper was measurable in two pieces at concentrations ranging from 0.2 to 0.69 wt.% (Ref. [27], Table E-4). Copper is a minor constituent of Inconel (~0.3 wt.%) but makes up only 0.001 wt.% of the expected core mass. The presence of this element suggests that it is present as accumulations in the lower vessel.

3.2 Reference Data for the Core Bores

A substantial amount of the core was removed during defueling in the late 1980s. A summary of the core bore results, like the core debris bed results, is provided to gain more understanding of how core constituents mixed. The results of the sample analysis performed on the core bores is documented in GEND-INF-092, *TMI-2 Core Bore Examinations* (Ref. [26]) in two volumes. The following information is extracted from this document. The central core and intact core component results are not discussed since it is assumed that all this material was removed during defueling.

In 1986, 10 assembly locations were identified for retrieving a 6.1 cm diameter core from the lower core region. These 10 core bores provided two pieces from the center of the core, two pieces from the near mid-radius, and six pieces from around the core periphery (note: only nine of 10 cores contained any sample material). These pieces were subdivided into 66 samples and further divided into subsamples for various analyses at multiple facilities.

The primary focus for the SFML determination is poison materials and structural materials intermixed with the core debris. In the upper crust, Al and B poisons were present at concentrations like those averages for the core, while Gd was elevated over the core average, indicating accumulation in the upper crust. Of note, Mo was also found in the upper crust of these samples in a Ni/Mo ratio like those found in Inconel grid spacers.

In the lower crust, the Al concentration was higher than the core average, but the average is biased by one sample outlier, suggesting it came from a poison assembly. B was present in the lower crust samples at concentrations between 0.05 and 0.26 wt.%. Gd was present in a range of concentrations from 0.02 to 2.9 wt.% (the core average is 0.01 wt.%), which suggest significant accumulation of Gd in the lower crust. Mo was also found in the lower crust at ratios approximately 0.75 times those for Inconel. This suggests that the crust was formed to a significant extent from the grid spacers as well as some stainless-steel components.

In the peripheral crust, Al was present at concentrations close to the core average. B from controls rods and reactor coolant was found in most samples at concentrations between 0.03 and 0.19 wt.%. Gd was present in a range of concentrations from 0.02 to 0.09 wt.%, which is higher than the core average and suggests accumulation in the peripheral crust. The Fe/Cr/Ni ratios of the peripheral crust are closer to that of stainless steel than Inconel, but the presence of Mo suggests again that grid spacers were contributors to the formation of the crust. The Ni/Mo ratios in these samples indicated that some stainless-steel did contribute to the crust formation, however.

In summary, the average U concentrations in the core borings indicate relatively similar concentrations in all regions of the lower core with the exception of the lower crust. The lower crust contained only about half the expected amount of U, indicating the presence of significantly more structural components in that region. The average concentration for Zr indicated similar concentrations in all core regions. The concentration of Ag in the lower core was almost twice as high in all parts than the core average. The concentration of In are three to four times the core average in all parts. The Cd concentrations are similar to or less than the core average. The average Fe concentrations were similar in the upper and lower crusts, and there are concentrations at two to five times the core average found in the central region. The Ni is concentrated in the metallic phases, with only a small amount in the ceramic samples. The Mo data indicate that Mo is significantly concentrated in all metallic regions of the lower core at concentrations 10 to 20 times the core average. The Ni/Mo ratios in the metallic phase are near those for Inconel, suggesting that the crust layers had a substantial contribution from the grid spacers.

3.3 Reference Data for Core Debris Bed

A substantial amount of core debris bed material was removed during defueling in the late 1980s. A summary of the sample results is provided here to gain more understanding of the elemental composition of the fuel debris and degree of mixing.

A sampling of core debris from positions H8 (center) and E9 (radius) was performed in 1983-1984, removing 11 pieces of material from three depths: surface of the rubble bed, 3 inches deep into the bed, and 22 inches deep into the bed. The pieces were divided and dispersed between two laboratories. The results of the sample analysis performed at the Lynchburg Research Center is documented in GEND-INF-060 Volume 2, *TMI-2 H8A Core Debris Sample Examination Final Report* (Ref. [28]). The results of the remaining sample analyses performed at INEL, Rockwell Hanford Operations, and Argonne National Laboratories (East and West) are documented in GEND-INF-075, *TMI-2 Core Debris Grab Samples – Examination and Analysis Part 1* (Ref. [29]).

The schematic in Figure 3-1 depicts the location of the extracted pieces for the debris bed and marks the lower head region. Although not depicted in the figure, the lower head region likely has two layers: a surface/top layer, and a prior-molten bottom layer.

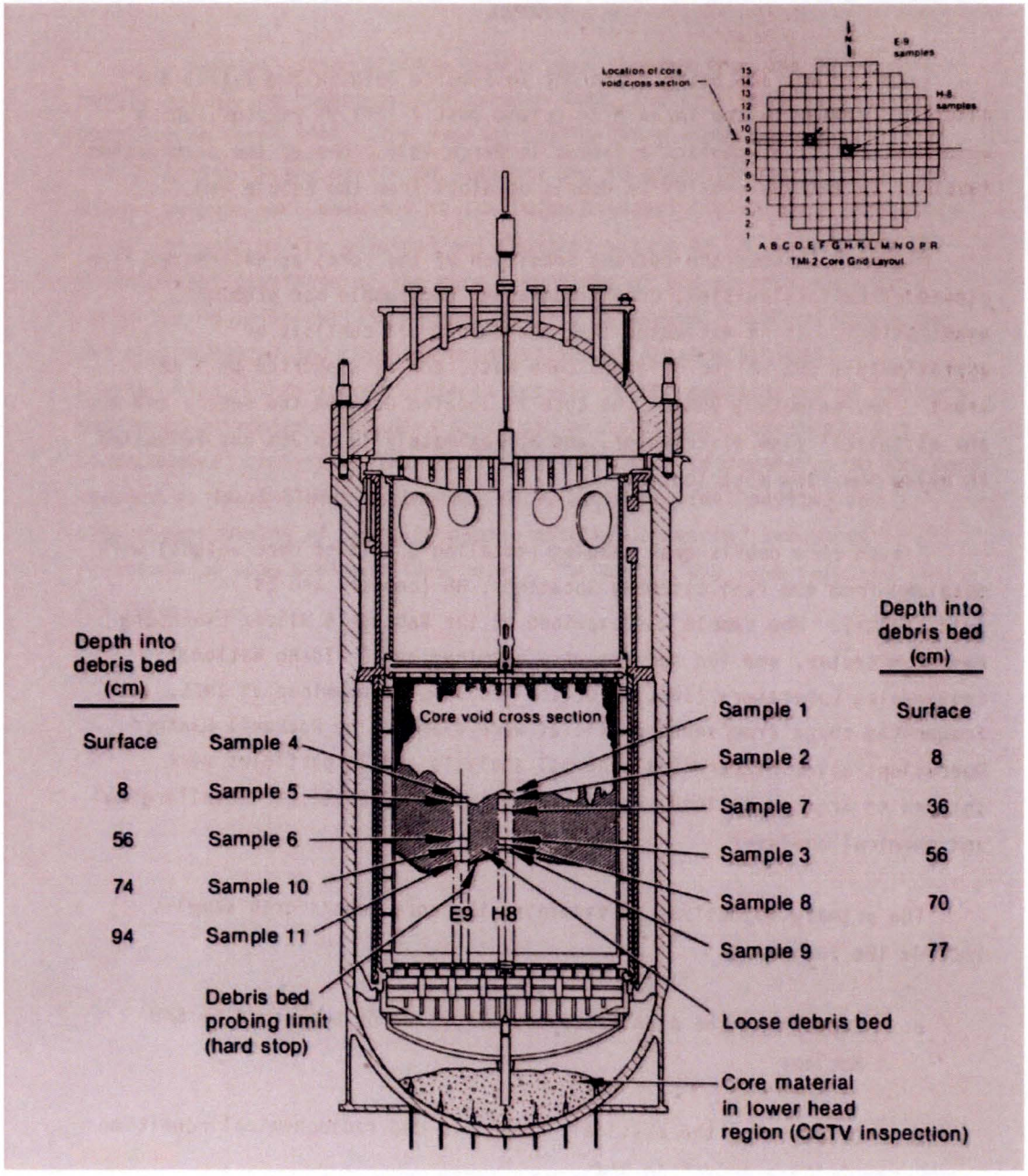

Figure 3-1 – Sampling Locations within the TMI-2 Core (Ref. [29], Figure S-1)

3.3.1 Lynchburg Elemental Results

Lynchburg received and processed a piece from the H8 location, 3 inches below the rubble bed surface. The subsamples in the piece appeared to be a high temperature reaction product from portions of two or more components. Small pieces could be identified as fuel pellet fragments, oxidized cladding, or combinations with reaction product material. The radionuclides found in

the Lynchburg piece were typical of irradiated Pressurized Water Reactor (PWR) fuel assemblies. The largest particle size observed in this sample was 1 cm which is approximately equivalent to the size of a typical PWR fuel pellet.

The Lynchburg analysis on the dissolved subsample resulted in an average, normalized, uranium content of 82.6 wt.%, zirconium content of 13.9 wt.%, iron content of 2.7 wt.% and nickel content of 0.9 wt.%. The as-built TMI-2 active core region contained about 94.5 wt.% UO_2 + Zr-4 and approximately 2.3 wt.% stainless steel + Inconel. The U/Zr ratio for the active core regions is 3.6. The average, normalized content was close to the average as-built active core elemental distribution, but with a higher ratio of U to Zr (approximately 5.9). The total of Ni, Cr, and Fe is about 1.3 wt.% higher than the as-built core. The magnetic fraction of the sample (7%) also contains about 2 wt.% Ag, which is associated with control rods. The remainder of the magnetic fraction is Ni, Fe, and Cr.

3.3.2 INEL Elemental Results

INEL received and processed ten of the eleven pieces, with additional analysis provided by Rockwell Hanford Operations and Argonne National Laboratory. Metallurgical examination indicated regions of prior molten U-Zr-O with traces of non-fuel rod materials (Al, Cr, Fe, and Ni). For some subsamples, these materials are a significant portion of the total subsample weight. Control rod materials (Ag, In, and Cd) were not as commonly found as structural materials.

The elemental analysis performed here used five groups of core components similar to the lower head analysis report, but also included a fifth group for elemental Te. Analysis for control rod materials with emphasis on Ag indicates that as much as 90% of the Ag relocated from the debris bed. There is also significant depletion ($\leq$ 50%) of Zr from the debris bed. The burnable poison rod materials, Al and Gd, were measured in most of the samples. There is a wide distribution of Gd, which is of particular interest, because only four fuel assemblies in the core contained Gd control material.

Structural materials are well mixed in the debris bed in concentrations similar to their original concentration in the fueled portion of a fuel assembly. Elemental Te is concentrated near the surface of the debris bed, similar to Al.

The degree of mixing and relocation was evaluated and suggests that molten core materials mixed vigorously to produce the degree of homogeneity observed, although it may not be representative of the entire debris bed. The original uranium enrichments for the H8 and E9 fuel assemblies were compared to the measured enrichment of samples and there was no correlation, indicating significant physical relocation of fuel at these locations.

3.4 Reference Data for Steam Generator Tube Sheet

An examination of loose core debris from the B-loop steam generator upper tube sheet was part of a fission product inventory evaluation, and also provided insight into the deposition of core material outside of the RV. The upper tube sheet is at the top of the steam generator, just below the coolant inlet. The sample was collected through a vacuum with a sock-type filter. All

information below is derived from GEND-INF-090, *TMI-2 B-Loop Steam Generator Tube Sheet Loose Debris Examination and Analysis* (Ref. [32]). The destructive and non-destructive analysis took place at Babcock & Wilcox in Lynchburg, Virginia via INEL.

The debris in the upper tube sheet of the B-loop steam generator is believed to have been carried to that location from the core region when the reactor coolant pump 2B was restarted (174 to 192 minutes after accident initiation). The debris appears similar to a) the sample from 8 cm below the surface of the core debris bed and b) some of the particles in the rock-size samples taken from the RV lower plenum.

The 12 subsamples were reaction products consisting primarily of U and Zr oxides. Fe, Ni, and Cr were trace elements within the sample matrix. Two of the 12 subsamples examined were Ag, In, and Cd, indicating control rod origin.

Of the 12 subsamples examined, the highest, lowest, and average distribution of each element is listed in Table 3-4. The first average is calculated among those subsamples that had a detectable concentration of the element and the second includes a concentration of zero for those below detect. Listed next to those elements that had a low value that was below the detectable limit, the number of subsamples that did not detect the element is in parentheses. Approximately 29 g of material remained on the filter has an elemental distribution independent from Table 3-4.

Table 3-4 – Elemental Composition Among Steam Generator Tube Sheet Samples (deduced from Ref. [32], Table 12)

Element	TMI-2 Core Average (wt.%)	Highest Distribution (wt.%)	Lowest Distribution (wt.%)	Average Distribution[(b)] (wt.%)	Average Distribution[(c)] (wt.%)
U	65.8	84.4	Below detect (2)	48.0	38.4
Zr	18	73.8	0.006	18.6	N/A
Ag	1.8	73.6	Below detect (2)	9.27	7.42
Al	0.2	0.352	Below detect (1)	0.198	0.179
Cu	0.001	0.059	Below detect (1)	0.021	0.019
Fe	3.0	1.19	0.128	0.623	N/A
B	0.1	0.672	Below detect (1)	0.146	0.131
Cr	1.0	0.371	0.011	0.166	N/A
Si	0.04	6.49[(a)]	Below detect (2)	1.97	1.58
Sn	0.3	1.74	Below detect (8)	1.40	0.28
Mg	Negligible	0.126	Below detect (2)	0.043	0.035
In	0.3	14.7	Below detect (8)	9.77	1.95
Cd	0.1	13.4	Below detect (1)	1.98	1.79
Gd	0.01	0.014	Below detect (9)	0.014	0.0014
Mn	0.08	0.042	Below detect (1)	0.020	0.018
Mo	0.03	0.025	Below detect (7)	0.017	0.0052
Nb	0.04	0.041	Below detect (7)	0.035	0.011
Ni	0.9	0.187	0.010	0.088	N/A

(a) It is likely that Si is a contaminant from the furnace that was used to ash the filter papers during dissolution.
(b) Since detection limits are not all known, the average only includes reported values. Below detect results are omitted from the average.
(c) This average includes below detect results, averaged in as zeros.

The primary constituent in all the samples was U, although the highest fraction was found in the larger particle sizes. The filter material was less than 5 wt.% metallic core elements overall. All the particle size fractions contained all core materials, with the largest contribution coming from UO_2 and Zircaloy cladding. Smaller fractions are from control rod material and Fe-Cr-Ni alloys such as stainless steel and Inconel. The large presence of Si is mostly likely due to contamination from the furnace ash.

Of particular interest to the SFML determination is the consistent presence of B in almost all the subsamples, even those identified with control rod origin. The only subsample with a B concentration below the detectable level was 'Particle 2' (see Ref. [32], Table 12) which is identified as a partially oxidized piece of zircaloy cladding. 'Particle 3' and 'Particle 4', identified as a fuel pellet fragment and resolidified portion of a control rod, respectively, have measurable B concentration found in both subsamples.

3.5 Summary

Over the years since samples were retrieved at TMI-2 and analyzed at various laboratories, substantial effort has been put into using those results to inform operational decontamination and decommissioning (D&D) decisions, validate accident computer codes, estimate fission product

release, and understand the nature of the TMI-2 accident. For the purposes of calculating a SFML, general conclusions can be drawn from the abundance of analytical sample information, particularly that available for the lower head of the RV since this is most representative of the remaining fuel debris. The last three bullets are not discussed in the previous sections but are part of the summary of the analytical results and discussed later in the document.

- Impurities, whether from cladding, structural material, or control rods, exist throughout the TMI-2 core. Control rod material was not as commonly found in remaining core debris as structural material.
- The U/Zr compositions in the core are indicative of mixing of all core constituents.
- Boron is evenly distributed throughout sample particles at the same concentration as in the reactor coolant, although it is not known if the elemental boron originated in the coolant or poison rods.
- Gadolinium is distributed among debris bed material at similar concentrations to the original burnable poison rod loading, indicating vast mixing. Gadolinium was not readily identified in the lower head or steam generator tube sheet as it was heavily concentrated in the core crusts.
- Most of the control rod material accumulated in the core debris bed or elsewhere, not in the lower head region.

As mentioned above, these last three bullets are not discussed in the above sections but are included here.

- Temperatures during the accident reached levels to initiate and propagate core melting, peaking at greater than 2800 K. There are few examples of pure UO_2, indicating isolated temperatures greater than 3100 K.
- Mixing of different assemblies' enrichment has occurred, and little or none of the 2.96 wt.% $^{235}U/U$ is contained in the lower head region.
- Core debris density measurements ranged from about 6 g/cm^3 to less than 9 g/cm^3, and other debris had density as low as around 4 g/cm^3. A presumed intact fuel component had a density of 9.97 g/cm^3.

4.0 Parameters

4.1 Fuel Enrichment

The unburned fuel enrichments for the three batches were 1.98 wt.% (batch 1, 56 assemblies), 2.96 wt.% (batch 3, 60 assemblies) and 2.64 wt.% (batch 2, 61 assemblies) $^{235}U/U$. As a homogeneous mixture, the unburned overall enrichment was 2.54 wt.% (Ref. [1]). The TMI-2 fuel experienced the equivalent of approximately 94 effective full-power days of burnup at the time of the accident (Ref. [34]). Applying this burnup to the highest enriched batch reduces the

enrichment to 2.67 wt.% ^{235}U/U. Batch 3 fuel was located at the core periphery and sustained less damage than the batch 1 and 2 fuel located at the core center, thus was easier to remove in bulk during defueling and likely sustained less mixing than the other two. The sum of the accident results and the 1980s defueling techniques results in an unquantifiable bias toward preferential removal of batch 3 fuel. The fuel that remains is largely in the form of either once molten, resolidified masses located in the LCSA or widely dispersed fines.

The actual exposure history for each fuel batch, using existing plant data, was applied to calculate burnup effects. The exposures and core operating history were applied to the ORIGEN model to calculate isotopic inventory at the time of the accident. The incorporation of the burnup effects for each fuel batch was used to produce a net ^{235}U enrichment of 2.24% for a homogeneous mixture of the three batches (Ref. [35]). The remaining fuel in the core is biased to an enrichment below this core averaged "burned" enrichment, with localized areas of resolidified mass that may exceed it. Use of 2.24 wt.% ^{235}U/U is considered conservative for use in the SFML calculation for multiple reasons, as summarized from Reference [3]:

- Defueling data indicates that the majority of the highest enrichment batch 3 fuel was removed from the RV as partial or full length fuel assemblies. Thus, most of the remaining fuel debris at TMI-2 is expected to consist mainly of the lower enrichment batches 1 and 2 fuel, in relatively equal amounts.
- The most comprehensive enrichment data available from the TMI-2 samples collected is that from the lower head, where 34 samples were collected. The weighted average of the ^{235}U for these samples was 2.23 wt.% ^{235}U/U (Ref. [3], Table 3).
- Enrichment data collected at location R-6 indicates some mixing of the lower enriched fuel batches with the batch 3 fuel located at R-6.
- Review of sample enrichment data shows a large variation in enrichment can occur in samples collected in close proximity to one another, indicating mixing of fuel batches.
- Observations, data, and assessments using postulated accident scenarios indicate that it would be incredible to expect that any significant debris accumulation (i.e., > 70 kg), except at isolated locations within the vessel (e.g., R-6) would be predominately batch 3 fuel. The debris is well mixed and thus contains a substantial percentage of batches 1 and 2 fuel.

Enrichment sample data and prior defueling data support an enrichment of 2.24 wt.% ^{235}U/U as a reasonable homogeneous value for the remaining TMI-2 core. It is important to recognize that this enrichment value does not imply that the enrichment of all individual small samples of fuel debris throughout the plant are less than 2.24 wt.% ^{235}U/U. Rather, application of this enrichment value is appropriate for any significant (i.e., > 70 kg) accumulation of debris. For this reason, this enrichment is applied in calculation of the revised SFML.

4.2 Fuel Debris Composition

References [27], [28], and [29] analytically estimate the peak and sustained temperatures in the core in order to better estimate fission product release and fuel composition. There are other important temperature dependent phenomena in the reactor core. One of the most important

phenomena considered in a TMI-2 scenario is the interaction of UO_2 and zircaloy. Melting of cladding and dissolution of the UO_2 fuel (liquefaction) can begin at temperatures as low as 2030K. Other low-temperature eutectics may form between the fuel and rod materials (Zr and UO_2) and other structural materials (Inconel, stainless steel, and control rod materials). Other reactor components have lower melting temperatures than the UO_2 fuel and many of these components can form eutectics with the fuel rod materials, thus, lowering fuel melting temperatures and changing physical properties. It is estimated that about 20% of the core material in the debris bed experienced temperatures up to fuel melting. The lower plenum material could have attained temperatures of at least 2810 K.

Poisoning effects of the depleted ^{235}U can be accounted for because the ^{235}U was not chemically affected by the transients during the accident and is assumed to have remained in the RV. However, poisoning effects of some of the fission products can be accounted for only if they are identified as remaining in the fuel. The gaseous fission products were assumed to be released at the time of the accident and the soluble ones are assumed to have leached out of the fuel matrix at the time of the accident. This was confirmed at the time of issue of Reference [34]. Of the remaining non-soluble fission products, some become volatile under extremely high fuel temperatures and the formation of the zircaloy-UO_2 eutectic. Of the non-soluble fission products, rare earths can be considered stable under the TMI-2 accident conditions and can act as a significant neutron poison (Ref. [34], Appendix B).

The only historical report that provides the explicit burned composition of the fuel (UO_2 component of the debris) is the 1989 *Criticality Safety Evaluation for Increasing the SFML* (Ref. [3]). For this reason, the composition from that report is utilized as the starting point for the calculation of the new SFML. This composition accounts for burnup effects in all three fuel batches but no melted and comingled cladding or structural material. It is assumed that the burned composition was also decayed to 1988 when the calculations were performed. Comparisons of the calculated fission product composition with the analytical data indicated, at that time, that the calculational results were appropriate (Ref. [27]). Table 4-1 lists all isotopes included in that burned composition and their associated number densities.

In each batch, the effects of uranium depletion, fissionable plutonium generation, and rare earth fission product production are considered. The referenced composition includes some nuclides that may have decayed at least one additional half-life in the 30-plus years since this burn-up analysis was performed. An ORIGEN decay calculation performed on the composition in Table 4-1 out to year 2022 is used in the SFML determination.

Table 4-1 – Burned Fuel Composition (Ref. [1], Table 5-8)

Isotope	Number Density (atoms/barn-cm)
^{235}U	5.21E-4
^{238}U	2.25E-2
^{16}O	4.60E-2
^{239}Pu	4.01E-5
^{240}Pu	2.00E-6
^{241}Pu	2.49E-7
^{149}Sm	1.01E-7
^{151}Sm	1.79E-7
^{151}Eu	8.20E-9
^{153}Eu	1.32E-7
^{154}Eu	4.51E-9
^{155}Eu	6.12E-9

4.3 UO_2 Density

Reference [28] indicates a 92.5% theoretical density for UO_2 (10.14 g/cm^3) from the as-built design and material characteristics for the TMI-2 core. Fuel debris matrix densities reported for the samples analyzed range from 4.59 g/cm^3 to 9.97 g/cm^3. A visual and gamma scan of one particle of the Lynchburg sample indicated unreacted fuel with a matrix density of 10.0 g/cm^3 (Ref. [28]). This is very close to the as-fabricated density of 10.14 g/cm^3. The ^{137}Cs and ^{144}Pr content of the particle revealed irradiated UO_2 that saw lower local peak time/temperature than other particles in the sample. This helps to validate the effect of lowered UO_2 density following irradiation. The relative density as a function of burnup is expressed as (Ref. [36]):

$$\frac{\rho}{\rho_{th}} = 0.963 - 1 \cdot 10^{-4} \cdot (burnup)^{3/2}$$

Full theoretical density (10.97 g/cm^3) as well as a reduced density (10.55 g/cm^3) based on burnup history, and the as-fabricated density (10.14 g/cm^3) are used in the initial SFML scoping calculations; however, the effect on reactivity is minimal. All calculations are performed at the most limiting density for that case.

4.4 Fuel Pellet and Arrangement

The standard cylindrical fuel pellet as loaded into the initial TMI-2 core had the dimensions of 1.778 cm long by 0.9398 cm diameter (Ref. [37], §3.1.1), producing a radius of around 0.5 cm. Applying the as-built UO_2 density of 10.14 g/cm^3, the surface area to mass ratio for the pellet is 0.5307, and applying the theoretical UO_2 density, the ratio is slightly smaller. Preservation of the design pellet surface-to-mass ratio in the specification of a spherical pellet volume has been practiced in multiple criticality safety analyses performed to date (Ref. [38]). Preserving this ratio enhances the resonance shielding effect of the ^{238}U cross sections. For the SFML calculation, the upper bound on the fuel pellet radius is that dimension associated with

preserving this ratio, which is approximately 0.6 cm. Further, defueling experience indicated that particles sizes much smaller than standard pellets were representative of the remaining core debris (Ref. [2]), thereby supporting the use of an upper radius limit of 0.6 and no larger.

The fuel pellet radius was varied from 0.2 cm to 0.6 cm, and the pellet separation from 0.1 cm to 0.4 cm in scoping calculations in an effort to identify the optimum conditions for maximum reactivity. The fuel volume fraction for randomly packed whole fuel pellets is around 0.63, and a volume fraction of 0.74 corresponds to the maximum that can be achieved for spheres in contact in a hexagonal lattice. A pellet radius of 0.6 cm with a separation of 0.1 cm results in a fuel volume fraction of approximately 0.634, slightly higher than that for randomly packed spherical pellets. A pellet radius of 0.6 cm with a separation of 0.4 cm results in a fuel volume fraction of 0.34. As the pellet size and separation distance (i.e., unit cell pitch) are varied, the optimum moderation can be found. Since the fuel is low enriched, moderation is necessary to achieve criticality. A higher fuel volume fraction (i.e., less moderator present) in each modeled unit cell results in a reduced reactivity. For this reason, the pellet radius and pitch are not varied beyond these specified values.

For each calculation to determine the SFML, the optimum fuel pellet size and separation distance is used. It is recognized that fuel pellet sizes larger than 1.2 cm may exist due to fuel melting. However, sample data have shown, and temperature-dependent phenomena indicate that particles of pure UO_2 are highly unlikely.

4.5 Core Geometry

The exact geometry of the remaining fuel in the TMI-2 core is unknown, other than it is dispersed within the RV and outside the RV to a much lesser extent. In order to bound all possible geometric arrangements during D&D, a spherical and annular fissile material geometry is analyzed. The first model includes a homogenized fuel sphere placed into a moderated lattice, which fills a spherical geometry meant to represent all remaining fuel debris. The second model also includes the homogenized fuel sphere placed into a moderated lattice but is dispersed into an annular configuration with a small amount of fuel debris also at the bottom of the annulus. The annular model is intended to simulate a more realistic material configuration within the RV although it is not bounding. Each geometric arrangement is surrounded by an effectively infinite water or concrete reflector.

It was determined in the scoping calculations that the spherical geometry is bounding of the annular, or any other, arrangement. For this reason, the spherical geometry is the limiting case used to develop the SFML. The annular configuration may be used to qualify margin.

4.6 Moderator and Reflection

The degree of moderation and reflection in the TMI-2 core is unknown. In order to bound all possible scenarios during D&D, full water reflection (i.e., 30 cm) and optimum moderation is applied to each calculation. The sensitivity of concrete reflection is also examined. For the TMI-2 fuel enrichments, neutron moderation is required to produce a critical configuration, and the

amount of neutron moderation determines the reactivity of the system. The SFML calculations assume there is sufficient water to provide the necessary moderation to achieve a maximum k_{eff}.

Moderating and reflecting water is not borated and does not contain any impurities. Previous defueling work in the TMI-2 facility has used borated water (Ref. [1]), which was subsequently drained. Residual boron would remain in the RV following the draining; however, this is ignored for the SFML determination as part of the moderating water, which is conservative.

4.7 Impurities

As summarized in Section 2.0, all samples of TMI-2 debris accumulations collected have shown that the debris contains impurities, whether those be from structural material, control rods, burnable poison rods, coolant poison, or cladding. In many cases, impurities from each category were present in the sample. These are considered long-term impurities of the debris, not surface contaminants. The borosilicate glass added to the RV in the 1990s (Ref. [39]) is not considered as an impurity as it was added after most sampling campaigns, but its presence would serve to neutronically isolate any fuel debris that is located or could relocate to the bottom of the RV. This would further reduce reactivity of the system.

Sampling occurred in the reactor vessel (Ref. [27], [28], [29]), "B" steam generator tube sheet (Ref. [32]), purification/makeup filters, and the pressurizer, of which the latter two were not detailed in this document but also indicated the presence of impurities (Ref. [40], [41]). Further, it was demonstrated through sampling that most of the fuel debris in the RV mixed to create some degree of homogeneity. A clear example of this mixing is the presence of Gd at a location at least 30 cm from its position in the TMI-2 core, and present in many samples taken from the core debris bed. Although Gd was not found in the lower head or tube sheet. Another example is the consistent and relatively stable presence of B in all the lower head samples (see Table 3-3).

Due to the numerous locations sampled and considering the potential fuel relocation pathways, these samples are considered representative of the fuel debris remaining in TMI-2. Core debris in the lower head region of the RV is most representative of what remains in the RV at the present time. For this reason, the impurity concentration used in the SFML calculation is derived from the sampling results for the RV. A composition representing the RV lower head was initially derived as 'Mix 3' in the Defueling Completion Report (Ref. [1], Table 5-11). In this mix, the fuel composition (UO_2+fission products) makes up 83.79 wt.% of the core debris while the remainder (16.21 wt.%) is impurities. The debris composition with impurities is listed in Table 4-2. Because this mix was already derived and reported in previous documents, it is used in the SFML determination.

Table 4-2 – Impurity Content of Core Debris ('Mix 3')

Component	Weight Percent (%)
Fuel Composition	83.79
Zr	12.70
Fe	2.44
B	0.11
Cd	0.00
Cr	0.75
Mo	0.15
Mn	0.06

Homogenization of the impurity content of the RV lower head in the modeled fuel mixture represents a significant conservatism over what would be derived assuming a homogeneous mixture of the initial core composition, where the UO_2 component is 65.8 wt.% of the TMI-2 core mass (Ref. [27]).

Review of the sample data results indicate that samples taken from the "B" steam generator tube sheet contained relatively smaller amounts of these impurities when compared to other samples, with a larger percentage of fuel. On the other hand, the sample results from the purification/makeup system filters showed approximately 5 wt.% U, with a significant percentage of impurities. The elemental results for the lower head are closer to the original core elemental percentages and likely are a result of total mixing of all constituents and a good representation of a core average for a homogenized model intended to represent the maximum credible configuration.

Varying concentrations of impurities were added to previous criticality calculation models to assess the reactivity worth. Reference [3] performed an impurity effect evaluation using data in Reference [32], reduced by 10%, and only including boron, iron, and zirconium isotopes. Natural boron was added homogeneously to the fuel mixture at 0.072 wt.%, which is bounding of sample data for plant locations outside of the RV. This boron concentration is not representative of the average concentration for the fuel debris remaining in the RV, which accounts for ~84% of remaining material.

The 'Mix 3' impurity composition is in alignment with the lower head region elemental sample results summarized in Section 3.1 and debris bed results summarized in Section 3.3. Nevertheless, this percentage serves as the upper bound of the analyzed impurities for the SFML and the full percentage of impurities is not credited.

The impurity composition within the core debris is varied in the analysis from 0% to 16.21%. In addition, the boron and iron content are varied independently since it is assumed that the greatest reactivity worth comes from those constituents. Cadmium would also be expected to have a measurable reactivity worth but is neglected in the 'Mix 3' composition.

The weight fraction of the 'Mix 3' impurity (more specifically, the boron fraction) applied to each model has a significant effect on the neutron multiplication factor. Although the impurity mix has multiple constituents, the boron has the highest neutron absorption cross section for the energy region of interest and therefore the greatest impact on k_{eff}.

In the SFML calculation, the overall weight fraction of the 'Mix 3' composition in the core debris and the percent of total boron are adjusted. Credit for 100% of the impurities identified in 'Mix 3' and the entirety of the boron are not assumed.

5.0 Conclusion

Development of a SFML for the TMI-2 decommissioning relies on a graded application of input parameters based on conditions both known and unknown within the TMI-2 reactor to develop a maximum credible configuration of fissile debris. For those parameters that are unknown (e.g., moderation, fissile material geometry), a theoretically bounding condition (e.g., optimal moderation, spherical geometry) can be applied. For parameters such as core debris composition, known conditions can be applied cautiously and conservatively, using credibly bounding ranges of values, when specifics are not known (e.g., impurities) or more exactly when specifics are known (e.g., enrichment, burn-up, decay).

6.0 References

[1] GPU Nuclear Corporation, "TMI Nuclear Station Unit 2 Defueling Completion Report Rev. 4," 1989.

[2] GPU Nuclear Corporation, "Letter 4410-90-L-0026/0531P, Results of Post-Lower Head Sampling Program Cleanup," April 12, 1990.

[3] GPU Nuclear Corporation, "4710-3220-89-1 Rev. 2, Criticality Safety Evaluation for Increasing the TMI-2 Safe Fuel Mass Limit," Submitted to Nuclear Regulatory Commission as Appendix B to the Defueling Completion Report Rev. 4, 1989.

[4] E. Schrult, "Letter C312-93-2004, TMI-2 Post-Defueling Survey Report for the Reactor Vessel," GPU Nuclear Corporation, Middleton, PA, February 1, 1993.

[5] GPU Nuclear Corporation, "4410-88-L-0162/0356P, TMI-2 Post-Defueling Survey Report for The Upper Plenum Assembly," Middletown, PA, September 30, 1988.

[6] GPU Nuclear Corporation, "4410-89-L-0097/0356P, TMI-2 Post-Defueling Survey Report for the Pressurizer, Reactor Building Basement, and Letdown Cooler Room," three reports packaged together, Middletown, PA, September 22, 1989.

[7] GPU Nuclear Corporation, "4410-90-L-0019/0356P, Post-Defueling Survey Report for the Reactor Vessel Head Assembly," Middletown, PA, March 14, 1990.

[8] C. Distenfeld, "C312-91-2064 Rev. 1, TMI-2 Post-Defueling Survey Report for the 'A' and 'B' Once-Through Steam Generators," GPU Nuclear Corporation, Middletown, PA, August 20, 1991.

[9] F. Deininger, "Letter C312-91-2045, TMI-2 Post-Defueling Survey Report for the Auxiliary & Fuel Handling Buildings," GPU Nuclear Corporation, Middletown, PA, June 7, 1991.

[10] F. Deininger, "C312-91-2052, TMI-2 Post-Defueling Survey Report for Reactor Building Miscellaneous Components," GPU Nuclear Corporation, Middletown, PA, June 18, 1991.

[11] P. Deininger, "Letter C312-91-2055, TMI-2 Post-Defueling Survey Report for the Reactor Coolant System," GPU Nuclear Corporation, Middletown, PA, July 3, 1991.

[12] GPU Nuclear Corporation, "Memorandum 4850-90-008, Fuel Assessment of Hot, Cold, and Core Flooding Nozzles, Tool, Tool Rack, and Tool Decon Facility," February 12, 1990.

[13] GPU Nuclear Corporation, "Memorandum, 4240-92-114, Results of Passive Neutron Measurement Program," August 21, 1992.

[14] GPU Nuclear Corporation, "TPO/TMI-051 Rev. 0, Location and Characterization of Fuel Debris in TMI-2," Project Planning and Analysis Department, Middletown, PA, April 1984.

[15] GPU Nuclear Corporation, "TPO/TMI-187 Rev. 0, Instrument Selection for Residual Fuel Measurements," MIddletown, PA, January 1987.

[16] P. Babel, "TB86-41 Rev. 2, Ce-144, Eu-154, and Eu-155 as Tracers for Fuel Debris," GPU Nuclear Corporation, Middletown, PA, February 1988.

[17] T. Rahon, "4200-3232-90-063 Rev. 0, Redetermination of the Cs-137 to UO2 Ratio," GPU Nuclear Corporation, Middletown, PA, October 1990.

[18] Nuclear Safety Analysis Center, "NSAC-80-1, Analysis of Three Mile Islant - Unit 2 Accident," March 1980.

[19] GPU Nuclear, "TPO/TMI-051 Rev. 0, Location and Characterization of Fuel Debris in TMI-2," Middletown, PA, April 1984.

[20] GPU Nuclear Corporation, "Calculation No. 4249-4400-91-004, Evaluation of Fluff Samples - OTSG 'A' & 'B'," Middletown, PA, April 8, 1991.

[21] P. P. a. A. Department, Director, *Library Videotapes E1.O through E55.O.* [Film]. Middletown, PA: GPU Nuclear Corporation.

[22] A. Kelsey, "4550-3211-87-031 Rev. 4, Calculation of Reactor Plenum Assembly SNM Accountability," GPU Nuclear Corporation, Middletown, PA, September 23, 1988.

[23] M. Haghighi, "4200-3211-90-062 Rev. 0, Incore Instrument Guide Tube Reactor Fuel," GPU Nuclear Corporation, Middletown, PA, February 21, 1990.

[24] E. Tolman, J. Broughton, R. McCardell, S. Langer, R. Hobbins, W. Domenico and P. Davis, "EGG-TMI-7048, TMI-2 Accident Evaluation Program," EG&G Idaho, prepared for the U.S. Department of Energy Idaho Operations Office, Idaho Falls, ID, February 1986.

[25] M. Russell, R. McCardell, M. Peters, M. Martin, J. Carlson and J. Broughton, "EGG-TMI-7132, TMI-2 Accident Evaluation Program Sample Acquisition and Examination Plan," EG&G Idaho, prepared for the U.S. Department of Energy Idaho Operations Office, Idaho Falls, ID, January 1986.

[26] D. Akers, C. Olsen, M. Russell, R. McCardell and B. Pregger, "GEND-INF-092, TMI-2 Core Bore Examinations," EG&G Idaho, Inc, prepared for U.S. Department of Energy Idaho Operations Office, Idaho Falls, ID, January 1990.

[27] C. S. Olsen, D. W. Akers and R. K. McCardell, "GEND-INF-084, Examination of the Debris from the Lower Head of the TMI-2 Reactor," EG&G Idaho, Prepared for the U.S. Department of Energy Idaho Operations Office, Idaho Falls, Idaho, January 1988.

[28] G. O. Hayner, "GEND-INF-060 Vol. 2, TMI-2 H8A Core Debris Sample Examination Final Report," Prepared for EG&G Idaho, Inc and the U.S. DOE Three Mile Island Operations Office, Lynchburg, Virginia, 1985.

[29] D. W. Akers, E. R. Carlson, B. A. Cook, S. A. Ploger and J. O. Carlson, "GEND-INF-075, TMI-2 Core Debris Grab Samples - Examination and Analysis Part 1 & 2," EG&G Idaho, Inc., Idaho Falls, Idaho, 1986.

[30] S. Jensen, D. Akers, R. Garner and G. Roybal, "GEND-INF-082, Examination of the TMI-2 Core Distinct Components," Prepared for the U.S. Department of Energy Three Mile Island Operations Office, September 1987.

[31] M. Failey, V. Pasupathi, M. Landow, M. Stenhouse, J. Ogden and R. Denning, "GEND-INF-067, Examination of the Leadscrew Support Tube From Three Mile Island Reactor Unit 2," Battelle Columbus Laboratories, prepared for EG&G Idaho, Inc. and the U.S. Department of Energy Three Mile Island Operations Office, Columbus, OH, March 1986.

[32] G. Hayner and T. Hardt, "GEND-INF-090, TMI-2 B-Loop Steam Generator Tube Sheet Loose Debris Examination and Analysis," Prepared for EG&G Idaho, Inc and the U.S. DOE, Lynchburg, VA, 1989.

[33] T. Cox, J. T. Horan and R. Nitschke, "GEND-INF-039, Final Analysis Data on TMI-2 Reactor Coolant System and Reactor Coolant Bleed-Tank Samples," EG&G Idaho, prepared for U.S. Department of Energy Three Mile Island Operations Office, Idaho Falls, ID, June 1983.

[34] GPU Nuclear Corporation, "Letter 4410-84-L-0199, Reactor Coolant System Criticality Report," November 8, 1984.

[35] *Letter, C.V. Parks, Oak Ridge National Laboratory, to D.S. Williams, GPU Nuclear Corporation,* November 6, 1987.

[36] M. Marchetti, D. Laux and L. Fongaro, "Physical and Mechanical Characterization of Irradiated Uranium Dioxide with a Broad Burnup Range and Different Dopants Using Acoustic Microscopy," *Journal of Nuclear Materials,* vol. 494, pp. 322-329, 2017.

[37] G. Neeley, D. Napolitano, M. Erighin, H. Hassan, D. Raine and B. Bingham, "R003-03-002106 Rev. 0, Core Nuclear Design Codes and Methods Qualification," Babcock & Wilcox Nuclear Energy, August 2010.

[38] R. Westfall, J. Knight, P. Fox, O. Hermann and J. Turner, "GEND-071, TMI Criticality Studies: Lower Vessel Rubble and Analytical Benchmarking," Martin Marietta Energy Systems, prepared for the U.S. Department of Energy Three Mile Island Operations Office, Oak Ridge, TN, May 1986.

[39] S. Weiss, "Letter to GPU Nuclear Corporation, Three Mile Island Unit 2 Reactor Vessel Criticality Safety Analysis," Nuclear Regulatory Commission, Washington, DC, July 6, 1993.

[40] R. E. Mason, R. R. Hobbins, B. A. Cook and P. E. MacDonald, "EGG-TMI-6181, Interim Report on the TMI-2 Purification Filter Examination," EG&G Idaho, Inc, Idaho Falls, Idaho, 1983.

[41] "Analytical Report 12705, TMI-2 Pressurizer Debris, Phase 2 Metallic and Radionuclide Distribution," Westinghouse Electric Corporation Advanced Energy Systems Division, 1986.

Attachment 2

MCNP Version 6.2 Bias Determination for Low Enrichment Uranium Using the ENDF/B-VIII.0 Cross Section Library