GPU NUCLEAR

THREE MILE ISLAND

NUCLEAR STATION

UNIT 2

DEFUELING COMPLETION REPORT

EXECUTIVE SUMMARY

The Defueling Completion Report (DCR) provides the basis for concluding that the Three Mile Island, Unit 2 (TMI-2) facility has been defueled to the extent reasonably achievable and that the possibility of an inadvertent criticality is precluded. As a result of the extensive defueling efforts and the recently completed residual fuel characterization, the following assessments have been made:

- The total quantity of residual fuel is estimated to be less than 1125 kg (approximately 1% of the original core inventory). This fuel is primarily in the form of finely divided, small particle-size sediment material, resolidified material either tightly adherent to components or in areas inaccessible to defueling, and adherent films on surfaces contained within piping, tanks, and other components.

- Evaluation of the ex-vessel residual fuel has demonstrated that insufficient fuel resides in any discrete location to exceed the Safe Fuel Mass Limit (SFML) of 140 kg. Further, assuming the residual fuel could accumulate in one ex-vessel area, an unlikely event, the total quantity would not exceed the SFML. In the case of the Reactor Vessel (RV), a specific analysis was performed to demonstrate that a criticality event could not occur in any configuration of residual fuel.

- With the exception of a final cleanup following the completion of the NRC-sponsored Reactor Vessel Lower Head Sampling Program, removal of any significant quantity of fuel would require a tedious, labor-intensive effort with an attendant significant occupational exposure. Further, unique defueling techniques such as abrasive cleaning, high pressure water erosion, chemical cleaning, and component removal and/or disassembly of the primary system would be required. These unique techniques and material requirements would create radioactive waste forms and packages which are not amenable to accepted disposal options and, therefore, could require extended on-site storage or further processing.

Considering the extensive cleanup activity accomplished over the past ten years involving an average work force in excess of 1000 persons/year, the more than 3.6 million person-hours of cleanup activity, the major effort completed to quantify and characterize the residual fuel, the analyses performed which demonstrate that criticality has been precluded, and the evaluation that continued defueling activities are of no significant benefit to the health and safety of the public, GPU Nuclear concludes that TMI-2 has been defueled to the extent reasonably achievable and that transition to Facility Mode 2, as defined by the TMI-2 Technical Specifications, is appropriate.

BACKGROUND

As a result of the accident at TMI-2, GPU Nuclear has completed an extensive program to remove fuel from the facility. On May 27, 1988, the NRC issued License Amendment No. 30 which defined three facility modes for the TMI-2 facility. This amendment established that 60 days prior to transition to each successive facility mode, a report shall be submitted to the NRC providing the necessary basis and justification for the transition.

For transition from Mode 1 to Mode 2, the licensee was required to demonstrate that the RV and Reactor Coolant System (RCS) have been defueled to the extent reasonably achievable, the possibility of criticality in the Reactor Building (RB) is precluded, and there are no canisters containing core material remaining in the RB.

Additionally, GPU Nuclear was to provide a criticality analysis that addressed each separate quantity of residual fuel in each defined location within the RB. The criticality analysis was to estimate the quantity of fuel remaining, its location, its dispersion within the location, its physical form (i.e., film, finely fragmented, intact fuel pellets), its mobility, the presence of any mechanism that would contribute to the mobility of the material, the presence of any moderating or reflecting material, and its potential for a criticality event.

This DCR provides the required criticality analyses. The report also provides the basis for concluding that the TMI-2 facility has been defueled to the extent reasonably achievable and demonstrates that inadvertent criticality has been precluded. Its purpose is to document compliance with the NRC reporting requirements defined in License Amendment No. 30, as identified above, and provide the basis for the TMI-2 facility transition to Mode 2.

To provide a better understanding of the end-state condition, the report also describes the accident sequence as it relates to core debris transport, the defueling program, and the fuel survey program.

THE ACCIDENT

Substantial core damage within the RV and subsequent attempts to cool the core provided the primary pathway by which core debris was transported into the RCS, RB, and the Auxiliary and Fuel Handling Buildings (AFHB). Because the plant systems required cooldown, isolation, and water processing at various times during the plant stabilization and recovery periods, additional pathways existed for insoluble core debris transport. However, the majority of these pathways within the RB and the AFHB are contained by specific boundaries, filters, and/or flow restrictions, which significantly reduced any potential core debris transport. Of the total fuel (i.e., UO_2) available to be transported from the RV, an early estimate was that approximately 25 kg reached the AFHB locations, approximately 10 kg relocated to the RB sump and various other RB locations, and approximately 230 kg relocated throughout the RCS. Based on defueling experience, the amount of fuel relocated to the RCS is now judged to have been approximately 400 kg. The remaining fuel inventory was retained in the RV.

The first major debris relocation occurred within three hours of the start of the accident. During the initial accident sequence, the amount of water in the RCS decreased because RCS makeup was insufficient to compensate for coolant loss through the pilot operated relief valve (PORV) located on top of the pressurizer. When the last two reactor coolant pumps (RCP) were turned off, at approximately 100 minutes, the top of the core was uncovered and coolant water separated into steam and liquid phases. Rapid oxidation of the zircaloy cladding at the top of the core began at approximately 150 minutes. The heat generated from oxidation elevated fuel rod temperatures above the cladding melting point (2100°K) developing a molten mixture of fuel, cladding, and some structural steel. This mixture flowed downward and solidified around intact fuel rods near the steam/liquid level interface. The operation of RC-P-2B at 174 minutes, for approximately 6 minutes, resulted in the first major core relocation event when coolant was circulated into the RV following core degradation. Thermal-mechanical interaction of the coolant with the oxidized and embrittled fuel rod remnants in the upper core regions is believed to have fragmented and collapsed these standing remnants and formed the upper core cavity and debris bed.

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

The second major core relocation event occurred between 224 and 226 minutes, within about 100 seconds. It is believed that failure of the supporting crust occurred in the upper and/or center region of the consolidated mass of molten core material, probably near the core periphery [1.5 meters (5 feet) from the bottom of the core] on the east side. Molten core material from the core region flowed through a large hole in the baffle plates into the Upper Core Support Assembly (UCSA), circumferentially throughout the UCSA, and downward through the flow holes in the core former plates into the Lower Core Support Assembly (LCSA) at nearly all locations around the core. The majority of the molten material appeared to have flowed into the LCSA on the southeast side through the hole in the baffle plate and through the southeast core former plate flow holes. Some molten core material flowed through the LCSA structures and came to rest on the bottom head. Approximately 6000 kg of resolidified material was dispersed at various locations on the circumference of these structures. In several places, resolidified material completely filled the flow holes and columns of once-molten material were observed between the plates.

During the accident, small quantities of core debris and fission products were transported throughout the RCS. There were two methods of transport of core debris to ex-vessel locations. The primary transport method was a sequential operation of the RCPs: RC-P-2B, RC-P-1A, and RC-P-2A. The secondary transport method was attributed to the "burping" phenomenon during natural circulation. Reactor coolant was also discharged from the RCS through the PORV. The PORV discharges to the Reactor Coolant Drain Tank (RCDT) which is located in the basement of the RB.

A relatively small quantity of core debris was released to the RB as a result of coolant flow through the PORV and the Makeup and Purification (MU&P) System during the accident. In addition, a small quantity of core debris was transported to the AFHB via the MU&P System during the accident. The MU&P System is fed from the suction side of RC-P-1A; flow then proceeds through the letdown coolers and enters the AFHB where it proceeds through various components and eventually discharges to the Reactor Coolant Bleed Tanks (RCBT). Some of this core debris may have further relocated into other systems as part of the post-accident water processing and cleanup activities.

DEFUELING PROGRAM

Requirements and Objectives

The two major TMI-2 defueling requirements were to defuel the facility to the extent that "inadvertent criticality was precluded" and to defuel the RCS and RV to the "extent reasonably achievable." The first requirement, "assurance of subcriticality," was to be demonstrated by measurements and analysis. The second requirement, "extent reasonably achievable," was to be based on actual defueling performance. In order to satisfy these TMI-2 defueling requirements, the following performance objectives were integrated into the defueling program planning:

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

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

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

Achievement of these objectives formed the basis for concluding that defueling has been accomplished to the "extent reasonably achievable."

Defueling Activities By Area

Auxiliary and Fuel Handling Buildings

Cleanup activities in the AFHB were focused on facilitating personnel access to those areas and components required to maintain the RCS in a stable condition, prepare for and conduct filtration and ion exchange removal of soluble and insoluble radionuclides in reactor coolant, and reduce the overall AFHB curie content. The cleanup activities included water removal, surface decontamination, system flushing, tank sludge and demineralizer resin removal, and removal of various filters and the letdown block orifice. The initial cleanup phase of the AFHB took place during the early plant stabilization period. This effort consisted of removing the water that flooded the lower level of the AFHB during the accident and performing surface decontamination of the floors, walls, and equipment. The goal of this initial cleanup was to reduce the overall loose contamination throughout the AFHB and to reduce the requirement for respirators due to airborne radioactivity.

The largest single quantity of fuel material located in the AFHB was less than 10 kg and the overall AFHB residual fuel inventory including material transferred as a result of cleanup operations did not exceed 40 kg at any given time. The use of borated processed water for system flushes resolved any criticality safety concerns associated with AFHB cleanup. Because of the demonstrated lack of a critical fuel mass, there was no specific effort to "defuel" any AFHB component or area. Instead, fuel removal occurred as a by-product of dose reduction decontamination, water processing, sludge transfer, sludge processing, and/or resin removal.

Significant dose rate reductions were achieved in nearly all of the cubicles (factors of 10 to 1000). The most significant defueling accomplishments in the AFHB were the removal of approximately 3 kg of fuel from the MU&P demineralizers and 370 g of fuel by removal of the block orifice assembly.

Reactor Building

The major RB cleanup activity was directed to dose reduction and structural surface decontamination. The entire accessible part of the RB surface area above the 305' elevation was hydraulically flushed with processed water. All major access ways and floor surfaces on the 305' and 347' elevations were scabbled to remove embedded contaminants in the paint and concrete. An extensive effort was also undertaken to maintain surfaces clean and preclude recontamination by use of protective coatings, special sealant, and epoxy paints. Additional flushing was performed inside the upper elevations of both D-rings to allow entry for once-through steam generator (OTSG) and pressurizer defueling activities.

A second major cleanup effort in the RB was directed at basement and block wall sediment removal and dose reduction. The fission product activity had been absorbed into the concrete while the basement was flooded. To reduce the dose rates, it was necessary to remove the outer surface of the concrete walls. Scarification of walls in the RB basement was accomplished using a robotic system equipped with a high pressure hydraulic water lance. Leaching was accomplished by drilling holes in several sections of the block wall and recirculating low-pressure water from the RB basement through the block wall. As radioactive concentrations increased in the water, it was pumped from the RB to be processed by the Submerged Demineralizer System (SDS) and EPICOR II.

It was estimated that the RB basement scarification and desludging activities removed approximately 4900 kg of sediment which contained approximately 4 kg of fuel. A robotic desludging system desludged approximately 40% of the basement floor area (i.e., the area accessible to robotics); the removal efficiency of desludging was greater than 90%.

Reactor Coolant System

Defueling operations in the RCS were primarily focused on the major debris deposit locations (i.e., pressurizer, OTSG, hot legs, and decay heat drop line). Initially, the pressurizer was defueled using a submersible pump, a knockout canister, a filter canister, and an agitation nozzle. The second phase of pressurizer defueling was directed at the larger objects located on the bottom head. A remotely-operated submersible vehicle equipped with an articulating claw and a scoop was used to remove these larger pieces of debris.

Pick-and-place and vacuuming techniques were principally used to defuel the "A" and "B" OTSG upper tubesheets. Long-handled gripping tools were used to lift large pieces of debris into canisters and a vacuum system removed the smaller debris. The hot legs were initially defueled using a combination scraper and vacuuming tool. Additional residual core debris in the "B" hot leg was scraped, flushed, and vacuumed into defueling canisters as part of RV defueling. The in-vessel vacuum system was used to defuel the decay heat drop line. A deployment tool was developed to guide the vacuum hose into the decay heat drop line from the RCS "B" hot leg. All loose debris in the vertical portion of the decay heat drop line was vacuumed. Below the vacuumable loose debris, a hard compacted region of debris was encountered. A drain cleaning machine was used to penetrate this hard debris and size it so vacuuming could continue. The material was airlifted into the "B" hot leg and was removed as part of the hot leg defueling.

In summary, the RCS defueling operations removed greater than 90% of the debris in the pressurizer, decay heat drop line, and hot legs and approximately 70% of the debris in the OTSG upper tubesheets.

Reactor Vessel

The RV defueling operations spanned a five year period from October 1985 through January 1990 with an estimate of over 2 million person-hours involved. Defueling began with the removal of the upper fuel element endfittings and other loose debris, including vacuumable "fines," from the rubble bed. Loose debris was placed into fuel canisters using pick-and-place techniques. Additional core debris was broken into smaller pieces for canister loading. The first canisters of core debris were transferred from the RB to the Fuel Handling Building (FHB) in January 1986.

A major defueling effort involved the use of a core bore machine (CBM), which was initially used to obtain special Department of Energy (DOE) research samples of the core. Workers drilled a total of 409 closely spaced holes in the resolidified material to break up the hard mass and facilitate its removal. Defueling workers also used air-operated chisels to break up the large pieces to fit inside the defueling canisters. In some cases, defuelers were able to remove large sections of intact stub assemblies. Using a fuel assembly puller and a variety of cutting, snaring, and clamping tools, the peripheral assemblies were removed. In March 1987, assembly A-6 was removed essentially intact, creating the first opening to the LCSA. Essentially, all of the stub end assemblies were removed from the RV by the end of 1987.

Lower Core Support Assembly

The LCSA consisted of five massive plates: the lower grid rib section (LGRS), the lower grid distributor plate, the lower grid forging, the incore guide support plate, and the flow distributor. These plates were not designed to be removed; however, access was required to remove core debris located below them. In January 1988, the first phase of the LCSA removal operations began. This involved using the CBM to drill through all 52 incore instrument guide tube spider castings in order to free them from the LCSA. In March 1988, using the CBM, the LGRS plate was severed into 13 pieces. The pieces were removed using long-handled tools and stored underwater inside core flood tank (CFT) "A." In May 1988, workers installed the plasma arc torch and associated equipment. The torch used a high-velocity stream of high-temperature ionized nitrogen gas (i.e., plasma) to cut the LCSA plates into sections. A total of 72 remnant LGRS pieces were successfully severed and removed.

Substantial plasma arc cutting was performed on the remaining four plates. The plasma arc torch made over 1000 cuts, with considerable recutting needed to ensure severance. By March 1989, cutting was completed; the plates were cut into over 60 pieces. The sections of the plates that did not contain incore guide tubes were removed from the RV and placed inside CFT "A." The sections that did contain incore guide tubes were bagged and stored inside the "A" D-ring.

Bottom Head

With the LCSA removed, defuelers were able to begin removing approximately 30 tons of core debris from the RV bottom head. Workers, using airlifting and pick-and-place equipment (i.e., 30- to 40-foot long-handled tools), removed the loose debris from this area. Removal of the debris uncovered a monolith of resolidified debris in the bottom head. Defuelers removed the monolith in much the same way as one would demolish a concrete slab. Starting from the outside edges and working inward, workers used a cavitating water jet (cavijet) and an impact hammer with a chisel point to break up the resolidified debris. Pick-and-place and airlifting equipment were used to remove the debris from the vessel. In addition, the bottom head was vacuumed to remove the fine debris.

Upper Core Support Assembly

UCSA defueling involved removal of fuel debris from between the baffle plates and the core barrel (i.e., the core former region). Resolidified debris formed in this region during the accident. Loose debris also accumulated there from the accident and subsequent defueling efforts.

Before workers could access the core debris, they had to remove the baffle plates. In the spring of 1989, workers cut the plates into eight sections using the plasma arc torch. In August 1989, using an untorquing tool and a drill tool, workers removed a total of 864 bolts that held the baffle plates to the core barrel. After the baffle plates were removed, workers began defueling the UCSA. In the fall of 1989, the loose debris was vacuumed and the resolidified debris was sized using the cavijet and mechanical methods. The final bulk defueling removed debris from the bottom head that fell there during UCSA work. This was accomplished in December 1989. Following completion of bulk defueling, a significant quantity of residual fuel, mostly in the form of small particle-size sediment material, was removed from the RV in an effort to leave the RV as clean as possible. This activity was accomplished in January 1990.

Table ES-1 provides a list of major defueling program milestones and accomplishments. Figure ES-1 provides a time sequence of defueling progress and fuel shipment performance.

RESIDUAL FUEL

Fuel Survey Program

An extensive fuel measurements program was developed which included direct measurement by instrumentation, visual inspection, and sample collection and analysis. The methods selected were influenced by many factors including accessibility, measurement uncertainties, equipment sensitivity, geometry, source strength, and physical form of core debris and were complicated by high radiation backgrounds, complex shielding, and limited access to core debris locations. Five general methods were used for fuel detection (detection of gamma rays, neutrons, alpha particles; sample and analysis; visual evidence). Additional measurements will be conducted at selected plant locations as part of the Special Nuclear Material (SNM) accountability program. A matrix of residual core debris locations and measurement methods was developed for the fuel survey program.

Quantity of Fuel by Area

The total quantity of residual fuel (UO_2) is estimated to be less than 1125 kg distributed in four major plant locations as follows:

• Auxiliary and Fuel Handling Buildings	< 17 kg
• Reactor Building (excluding the RCS)	< 75 kg
• Reactor Coolant System (excluding the RV)	<133 kg
• Reactor Vessel	<900 kg

Auxiliary and Fuel Handling Buildings

Conservative estimates indicate that up to 25 kg of fuel material was transported to the AFHB during the accident sequence. Additionally, it was indicated that up to 15 kg of fuel material may have been relocated into the AFHB as part of water processing and defueling operations. Based on these estimates (up to 40 kg), it was concluded that AFHB residual fuel conditions were at all times substantially below any criticality concern.

The total quantity of fuel material remaining in the AFHB is estimated to be less than 17 kg; not including fuel material that remains in canisters within the "A" spent fuel pool which are awaiting shipment to INEL. The majority of the remaining AFHB fuel material resides in piping and components as finely divided, small particle-size sediment with small amounts fixed as adherent films on surfaces. There remains no potential for transport of a significant quantity of core debris to other locations outside the AFHB.

Reactor Building

Approximately 10 kg of fuel was transported to the RB during the accident sequence. Subsequent to the accident, approximately 70 kg of fuel was relocated to the RB as a result of several cleanup operations including: transfer to and storage of structural RV components in CFT "A" and the "A" D-ring; storage of upper endfittings; flushing of defueling tools; and transfer of the defueling canisters into the fuel transfer canal. Even though fuel was relocated to the RB during cleanup operations, RB residual fuel conditions were maintained significantly below the SFML of 140 kg. Further, a significant cleanup effort was undertaken with the primary purpose of reducing exposure rates but which also resulted in the removal of some fuel material.

The current estimate of residual fuel content in the RB is less than 75 kg with the largest single quantity located in the "A" D-ring (approximately 23 kg). The residual fuel in the remaining areas of the RB consists of finely divided, small particle-size sediment material with minor amounts of fuel found as adherent films on surfaces. The condition of this residual fuel prevents transport of a significant quantity of material, thus minimizing any interaction and accumulation potential. Decontamination activities in the RB served to remove residual fuel, especially in the basement where the residual fuel quantity was reduced by approximately 75%. Post-defueling activities (e.g., special sample collection) will result in the removal of additional small quantities of fuel. Thus, the quantity of residual fuel material may be further reduced.

Reactor Coolant System

Early estimates indicated that approximately 230 kg of fuel was transported within the RCS during the accident. Additionally, it was determined that some debris relocation from the RV occurred during defueling operations. The amount of fuel relocated to the RCS is now judged to have been approximately 400 kg. Following completion of the RCS defueling operations, the total quantity of residual fuel in the RCS was estimated to be less than 133 kg. Of the residual fuel in the RCS outside the RV, the largest discrete location of fuel is in the upper tubesheet of the "B" OTSG. This 36 kg of residual fuel exists as tightly adherent material not readily removable by available dynamic defueling techniques and, therefore, not readily transportable to other locations for accumulation. The remaining residual fuel is dispersed throughout the RCS in the form of finely divided, small particle-size sediment material and adherent films on surfaces. The condition of this residual fuel prevents transport of a significant quantity of material, thus minimizing any interaction and accumulation potential. Even if all this residual fuel is accumulated, the SFML would not be exceeded and, hence, criticality is precluded.

Reactor Vessel

Prior to defueling operations, the RV was estimated to have retained over 99% of the original fuel inventory. The original core fuel loading was 94 metric tonnes or 94,000 kg of fuel material. Extensive visual examination during and following RV defueling has quantified the amount, form, and location of residual fuel in the RV. The remaining quantity of fuel in the RV is <900 kg. The residual fuel consists primarily of finely divided, small particle-size sediment in inaccessible holes, crevices, and corners; surface films; and resolidified material either tightly adherent to the RV components or inaccessible for defueling. As a result of the extensive defueling effort, the remaining residual fuel is not easily removable and, therefore, not readily transportable between locations. While the total remaining fuel exceeds the SFML, it has been demonstrated in the criticality analyses that criticality is precluded. The following is a summary of residual fuel in the RV by region.

LOCATION	RESIDUAL FUEL (kg of UO2)
Work Platform and Suspended Equipment	< 31
Downcomer Region	<179
Internals Indexing Fixture Region	< 5
Core Support Shield Region	< 11
UCSA Region	< 86
LCSA Region	<429
Bottom Head Region	<152
Surface Coatings	< 3
TOTAL	<900

CRITICALITY ANALYSES

A criticality assessment was conducted to evaluate the quantities of residual fuel in plant systems and components. The criticality analyses demonstrated that as a result of the extensive TMI-2 defueling effort, criticality has been precluded. This conclusion is based on three separate evaluations: the SFML determination, the limiting RV criticality calculation, and the potential for criticality under accident conditions.

Safe Fuel Mass Limit

A revised SFML has been established for evaluating TMI-2 long-term storage conditions. This safe limit of 140 kg of UO_2 was based on the extensive data base collected from debris sampling, video inspection, and other defueling data gathered to characterize the residual fuel composition. A conservative spherical geometric model which consisted of a center region mixture of unborated water and fuel surrounded by 30 cm (1 foot) of unborated water reflector (effectively an infinite reflector) was used. The fuel composition was assumed to be TMI-2 average fuel including burnup effects, optimally moderated with unborated water, and with no credit taken for the presence of impurities in the fuel. In fact, impurities observed during sample analysis included structural and control material (e.g., zirconium, iron, boron, cadmium, silver, and indium). If credit is taken for impurities, a more reasonable representation of the residual fuel demonstrates that criticality is precluded for all quantities of fuel accumulation in any optimally moderated condition with unborated water (i.e., k_∞ less than 1). Nonetheless, a SFML of 140 kg was conservatively adopted.

Reactor Vessel Criticality Calculation

Since the total residual fuel quantity within the RV was estimated to be greater than 140 kg of UO_2, a special analysis of worst-case conditions within the RV was performed. This analysis used in-vessel inspections and assessments of debris locations and quantities to develop a specific three-dimensional analytical model of the end-state RV configuration. Conservative assumptions were made regarding the quantity and location of fuel remaining in the RV following the completion of in-vessel defueling activities. The regions modelled in detail were the bottom head, the LCSA, and the core former region. Significant conservative allowances included: no credit was allowed for the presence of structural or control impurities in the fuel; the entire bottom head was assumed to be covered with a 1.3-cm (0.5-inch) layer of fuel; a 0.6-cm (0.25-inch) layer of fuel, with a height of 3 meters (10 feet) was assumed to be attached to the core barrel in the core former region; each of the LCSA plates was modelled with a radial thickness that conservatively bounded the presence of fuel on the plate; the fuel was assumed to extend the entire 360° around the periphery of the RV; and the flow holes in each of the modelled LCSA plates were assumed to be filled with fuel and unborated water in an optimal mixture.

Considerably more fuel was included in the analytical model than was estimated to remain as residual fuel. The approximate fuel quantities modelled (kg UO_2) were 670 kg distributed on the bottom head, 5500 kg on the LCSA, and 600 kg on the UCSA for a total of greater than 6700 kg UO_2. By comparison, a final quantity of approximately 900 kg of residual fuel was estimated to remain in the entire RV based on the final visual surveys. The results of the criticality analysis using this extremely conservative model of the RV resulted in a neutron multiplication factor (k_{eff}) of 0.983 including a 0.025 Δk computer code bias. Because the k_{eff} was less than 0.99, it is concluded that the much smaller quantity of fuel actually remaining in the RV does not pose a criticality safety concern and criticality is precluded.

Criticality Under Accident Conditions

With approximately 900 kg of fuel residual in the RV, it can be postulated that a seismic event, aging and corrosion, or other non-mechanistic events could cause the residual fuel to accumulate in one area (e.g., the RV lower head) resulting in a potential for criticality. However, as evidenced by the dynamic defueling effort, the residual fuel has consistently resisted strong displacement attempts by aggressive mechanical methods.

Nonetheless, a realistic criticality analysis of the residual fuel, accounting for reasonably expected impurity levels from observed control and structural materials, has been performed assuming an optimal configuration. Even with an unlimited quantity of unborated water in the RV, which in itself is extremely unlikely, the calculated infinite neutron multiplication factor k_∞ is less than 0.99, including the 0.025 Δk computer code bias. Therefore, no quantity of residual core debris can result in a critical fuel configuration. Regardless, a stable and insoluble neutron poison material will be added to the bottom head of the RV to provide added margin and absolute assurance that no circumstance will result in a condition causing the residual fuel in the RV to become critical. Hence criticality is precluded for all credible potential conditions.

OCCUPATIONAL EXPOSURE

The actual occupational exposure to complete the defueling activities was well within the estimate of the Programmatic Environmental Impact Statement (PEIS) Supplement No. 1. The cumulative occupational dose for RV defueling is below 2000 person-rem. For the entire cleanup activity to date, the total occupational dose is below 6500 person-rem. Completion of defueling with less than the estimated collective dose was made possible by extensive design and engineering preparation of tools and equipment; use of dose reduction considerations incorporated as part of the defueling operations and support activities; ex-vessel dose reduction activities; airborne contamination control; and extensive worker briefings and training on mock-ups. No individual worker has received over 3.7 rem whole body dose in any one year since the initial cleanup activities began in 1979.

TABLE ES-1

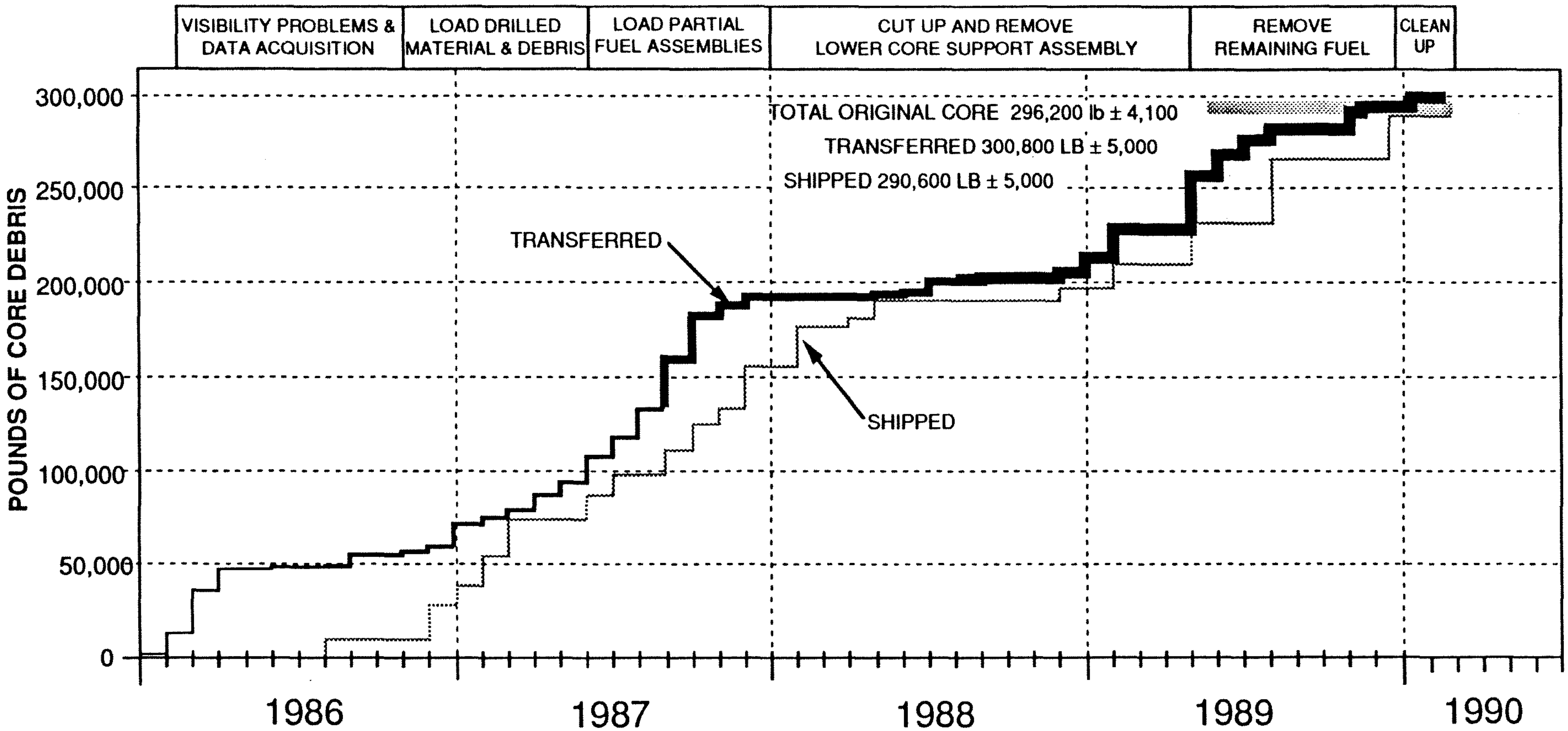
TMI-2 REACTOR VESSEL DEFUELING
MILESTONES AND ACCOMPLISHMENTS

DATE	MILESTONE ACTIVITY
October 1985	Defueling begins - removal of loose debris from top of core.
November 1985	First fuel canister loaded.
January 1986	Water clarity problems delay defueling.
May 1986	Defueling resumes with marginal water clarity.
July 1986	First core debris material shipped to Idaho National Engineering Laboratory (INEL) on special rail cask.
August 1986	CBM installed for Research and Development samples.
October/November 1986	"Swiss Cheese" of hard crust with 409 holes drilled for sizing and dislodging material.
January 1987	Startup of Modified Defueling Water Cleanup System (hydrogen peroxide and filters); startup airlifting (vacuuming) equipment.
February 1987	Water clarity is significantly improved via modified cleanup system (see January 1987).
March 1987	Removal of stub-end assembly begins.
September 1987	50% of core debris removed; stub-end assembly removal completed.
December 1987	First triple-cask shipment to INEL.
January 1988	LCSA disassembly begins using CBM.
April 1988	CBM cut the top LCSA plate into four quadrants (88 cuts between flow holes), removed and stored them in the "A" CFT.
November 1988	Completed cutting, removal, and storage of the largest LCSA plate (forging).
April 1989	Completed cutting, removal, and storage of the fifth LCSA plate (flow distributor).
August/September 1989	Baffle plates removed to gain access to USCA.

TABLE ES-1 (Cont'd)

TMI-2 REACTOR VESSEL DEFUELING
MILESTONES AND ACCOMPLISHMENTS

DATE	MILESTONE ACTIVITY
October 1989	UCSA defueling completed.
November 1989	LCSA defueling completed.
December 1989	RV bulk defueling operation completed.
January 1990	Completed final RV cleanup and video inspection.



TMI-2 Defueling Progress

FIGURE ES-1

DEFUELING COMPLETION REPORT

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1.0 INTRODUCTION

1.1 Background

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

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

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

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

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

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

1.2 Purpose

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

1.3 Report Organization

The DCR is structured to address four 4 separate areas of the TMI-2 facility [i.e., Auxiliary and Fuel Handling Buildings (AFHB), Reactor Building (RB), Reactor Coolant System (RCS), and Reactor Vessel (RV)]. Fuel removal and associated decontamination activities are discussed in detail for each area.

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

Section 6.0 of the DCR focuses on residual fuel and discuss possible alternatives and impacts associated with attempts to remove the remaining fuel. It should be recognized that it is not feasible, nor required, to remove all residual fuel from the facility prior to transition to Mode 2. However, the facility must have been defueled to the extent reasonably achievable, and inadvertent criticality must have been precluded.

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

1.4 Defueling Objectives and Guidelines

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

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

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

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

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

1.5 Residual Fuel Characterization

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

For clarifiation, the term core debris is defined as material which consists of a mixture of the original fuel (UO_2) plus structural material (i.e., stainless steel, zircaloy, inconel) and control rod material (i.e, silver, indium, and cadmium). Residual fuel and/or fuel material differ from core debris in that they represent the actual fuel content by weight (i.e, kg of UO_2). Where noted, the transition or conversion from core debris to fuel material is directly determined by the weight percent (wt%) of fuel within the core material.

TABLE 1-1

FACILITY MODES

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

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

TABLE 1-2

ACRONYMS

AB	Auxiliary Building
ACES	Automated Cutting Equipment System
AFHB	Auxiliary and Fuel Handling Buildings
ALARA	As Low As Is Reasonably Achievable
B&W	Babcock & Wilcox
cavijet	cavitating water jet
CBM	Core Bore Machine
CFT	Core Flood Tank
CPS	Canister Positioning System
CRA	Control Rod Assembly
CSS	Core Support Shield
CSA	Core Support Assembly
CWST	Concentrated Waste Storage Tank
DCR	Defueling Completion Report
DF	Decontamination Factor
DHR	Decay Heat Removal
DOE	Department of Energy
DWCS	Defueling Water Cleanup System
ECCS	Emergency Core Cooling System
EPRI	Electric Power Research Institute
FHB	Fuel Handling Building
FM&A	Fuel Measurement and Analysis
FTC	Fuel Transfer Canal
GM	Geiger-Müller
HEPA	High-Efficiency Particulate Absolute
HPGe	High-Purity Germanium
HPI	High Pressure Injection
ID	Inner Diameter
IGSP	Incore Guide Support Plate
IGTSP	Incore Guide Tube Support Plate
IIF	Internals Indexing Fixture

TABLE 1-2 (Cont'd)

ACRONYMS

IIGT	Incore Instrument Guide Tube
INEL	Idaho National Engineering Laboratory
IVFS	In-Vessel Filtration System
LCSA	Lower Core Support Assembly
LGRS	Lower Grid Rib Section
LGDP	Lower Grid Distributor Plate
LLD	Lower Limit of Detection
LOCA	Loss of Coolant Accident
LS	Leadscrew
LST	Leadscrew Support Tube
MDL	Minimum Detectable Level
MeV	Million Electron Volts
MU	Makeup
MU&P	Makeup and Purification
MWHT	Miscellaneous Waste Holdup Tank
NaI(Tl)	Thallium Drifted Sodium Iodide
NRC	Nuclear Regulatory Commission
OECD	Organization of Economic Cooperation and Development
OPG	Oxalic - Peroxide Gluconic
ORNL	Oak Ridge National Laboratory
OTSG	Once-Through Steam Generator
PBC	Peroxide Bicarbonate
PDMS	Post-Defueling Monitored Storage
PEIS	Programmatic Environmental Impact Statement
PORV	Pilot Operated Relief Valve
RB	Reactor Building
RCBT	Reactor Coolant Bleed Tank
RCDT	Reactor Coolant Drain Tank
RCP or RC-P	Reactor Coolant Pump

TABLE 1-2 (Cont'd)

ACRONYMS

RCS	Reactor Coolant System
RV	Reactor Vessel
RWP	Radiation Work Permit
SAR	Safety Analysis Report
SDS	Submerged Demineralizer System
SER	Safety Evaluation Report
SFML	Safe Fuel Mass Limit
SFP	Spent Fuel Pool
SIVR	Seal Injection Valve Room
SNM	Special Nuclear Material
SPC	Standby Pressure Control
SRD	Self-Reading Dosimeter
SRST	Spent Resin Storage Tank
SSCH	Surveillance Specimen Capsule Holder
SSTR	Solid-State Track Recorder
Si(Li)	Lithium Drifted Silicon
TB	Technical Bulletin
TER	Technical Evaluation Report
TLD	Thermoluminescent Dosimeter
TMI-2	Three Mile Island, Unit 2
TRVFS	Temporary Reactor Vessel Filtration System
UCSA	Upper Core Support Assembly
WDL	Waste Disposal Liquid

2.0 POST-ACCIDENT FUEL DISPERSION

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

Substantial core damage within the RV and subsequent attempts to cool the core provided the source material and primary pathway by which fuel was transported into the RCS, RB, and AFHB. Because the plant systems required cooldown, isolation, and water processing at various times during the plant stabilization and recovery periods, additional potential pathways existed for insoluble fuel material transport. However, the majority of these pathways within the RB and the AFHB are defined by specific boundaries, filters, and/or flow restrictions, which significantly reduced any potential fuel transport. Of the total fuel available to be transported from the RV, an early estimate was that no more than 25 kg reached the AFHB, no more than 10 kg relocated to the RB sump and various other RB locations, and no more than 230 kg relocated throughout the RCS (see Table 2-1). Based on defueling experience, the amount of fuel relocated to the RCS is now judged to have been approximately 400 kg. The remaining fuel inventory was retained in the RV. The following discussion represents the basis for fuel transport dispersion at TMI-2.

2.1 The Accident Scenario

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

The accident can be divided into the following five phases:

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

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

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

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

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

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

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

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

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

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

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

The second major core relocation event occurred between 224 and 226 minutes, within about 100 seconds. This event was indicated by the RCS pressure monitor, self-powered neutron detectors, and the source range neutron monitors. It is believed that failure of the supporting crust occurred in the upper and/or center region of the consolidated mass of molten core material, probably near the core periphery [1.5 meters (5 feet) from the bottom of the core] on the east side, as shown in Figure 2-1c. Visual inspections conducted during defueling indicated that the flow of molten core entered the core former on the east side and flowed around the core former and then down into the LCSA internals. Analysis of potential flow of molten core materials through fuel assembly locations indicated that all of the molten core material could have relocated into the LCSA internals and lower head in less than one minute through only one or two fuel rod assemblies.

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

Approximately 16 hours after the start of the accident, RC-P-1A was restarted and operated for approximately one week. Subsequently, RC-P-2A was placed in operation until April 27, 1979.

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

2.2 Post-Accident Condition of the Plant

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

The original core inventory included approximately 94,000 kg of UO_2 and 35,000 kg of cladding, structural, and control materials. Accounting for oxidation of core materials during the accident and for portions of the upper plenum structure that melted, the total amount of post-accident core debris was estimated to be 133,000 kg.
(293,212 lbs)

2.2.1 Reactor Vessel Internals

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

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

2.2.1.1 Upper Plenum Assembly

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

2.2.1.2 Core Region

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

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

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

The standing, undamaged fuel assembly stubs extended upward from the lower grid plate to the bottom surface of the resolidified region of the once-molten materials. These stubs varied in length from approximately 0.2 to 1.5 meters (8 inches to 5 feet). The longer partial fuel assemblies were located at the periphery of the resolidified mass. On the east side of the core, one fuel assembly was almost completely replaced with once-molten core material; this indicated a possible relocation path into the LCSA and core bypass region for molten material. The standing fuel assembly stubs and peripheral assemblies constituted about 45,000 kg of core debris.

2.2.1.3 Upper Core Support Assembly

This region consists of vertical baffle plates that form the peripheral boundary of the core; horizontal core former plates to which the baffle plates are bolted; the core barrel; and the thermal shield. There are a number of flow holes in the baffle and core former plates through which coolant flowed during normal operations. On the east side of the core, a large hole approximately 0.6 meter (2 feet) wide and 1.5 meters (5 feet) high, and extending across three baffle plates and three core former plates was discovered. Adjacent baffle plates on the east and southeast were warped possibly as a result of the high temperatures and the flow of molten material in the bypass region.

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

The circumference of the core region (i.e., the area behind the baffle plates) contained loose debris throughout. The depth of debris varied from approximately 1.5 meters (5 feet) on the east side to a few millimeters on the southwest side. There appeared to be a resolidified crust on the upper horizontal surfaces of the three bottom core former plates; this crust varied in thickness from approximately 0.5 to 4.0 cm (0.2 to 1.6 inches). It is estimated that approximately 4000 kg of core debris was retained in the UCSA region. In the small annulus between the core barrel and the thermal shield, fine particulates were observed but no major damage to these components was seen.

2.2.1.4 Lower Core Support Assembly

The LCSA region consists of five stainless steel structures. The structures vary in thickness from 0.025 to 0.33 meter (1 inch to 1 foot) with 0.080 to 0.15 meter (3 to 5 inches) diameter flow holes. Some molten core material flowed through these structures and came to rest on the lower head. There were approximately 6000 kg of resolidified material dispersed at various locations on the circumference of these structures. In several places, resolidified material completely filled the flow holes, and columns of once-molten material were observed between the plates. The largest accumulation of resolidified material appeared to have flowed into the LCSA from the east side of the core. Although most of the material was seen on the east to southeast side, many columns of resolidified material were also seen in the LCSA around the periphery of the core beneath the core bypass region.

2.2.1.5 Lower Head Region

The debris in the lower head region accumulated to a depth of 0.75 to 1 meter (2.5 to 3.3 feet) and to a diameter of 4 meters (13.2 feet). The spatial distribution of the material was neither uniform nor symmetric. The surface debris had particle sizes which varied from large rocks [up to 0.20 meter (8 inches)] to granular particles [less than 0.001 meter (0.04 inch)]. The larger rocks, especially in the northeast and southwest regions, were located near the periphery. The debris pile was lower at the vessel center than at the periphery, with granular or gravel-like material observed in the central region of the vessel. A large resolidified mass was identified between the loose debris bed and the lower head of the RV. This mass was approximately 0.5 meter (1.7 feet) in the center and 1.7 meters (5.7 feet) in diameter. A large cliff-like structure formed in the northern region from once-molten core material. The cliff face was approximately 0.38 meter (1.3 feet) high and 1.25 meters (4.2 feet) wide. It was estimated that approximately 12,000 kg of loose core debris and 7,000 kg of agglomerated core debris relocated into the lower head.

2.2.2 Reactor Coolant System

During the accident, small quantities of fuel debris and fission products were transported throughout the RCS (see Figure 2-4). The largest RCS components operated during the accident were the RCPs. RC-P-2B was the only pump which would respond to a "start" command 174 minutes into the accident. This pump was started and operated for approximately 6 minutes. The operation of this pump was the major driving force for the relocation of fuel from the RV. Coolant circulated through the RV by this pump caused a rapid quenching of the highly oxidized, high temperature fuel, which resulted in the fuel rods being shattered and rubbled.

As the RCP operated, the flow of the "B" loop was in a "forward" (i.e., normal) direction. The flow rate through the RV was sufficient to transport small amounts of fuel into the "B" loop where a portion of the fuel relocated into the "B" hot leg and settled out into the decay heat drop line. The decay heat drop line connects to the bottom of the horizontal section of the "B" hot leg and was found to contain some fuel, presumably as a result of the RC-P-2B operation. The coolant continued to flow up the "candy cane" and deposited fuel material on the "B" OTSG upper tubesheet. The tubesheet acted as a "strainer" for the collection of fuel transported outside the RV. However, a small quantity of fuel flowed down through the steam generator tubes and was deposited on the lower head of the "B" OTSG and J-legs. As the coolant continued to flow, relatively smaller quantities of fuel were then deposited in the "B" RCP and cold legs.

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

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

In summary, there were two methods of transport of fuel to ex-vessel locations. The primary transport method was a sequential operation of the RCPs: RC-P-2B, RC-P-1A, and RC-P-2A. The secondary transport method was attributed to the "burping" phenomenon during natural circulation. Table 2-1 provides an early estimate of the quantity of fuel relocated into the RCS during the accident sequence and resulting thermal hydraulic phenomena (References 2.12 through 2.14). Based on defueling experience, the amount of fuel relocated to the RCS is now judged to have been approximately 400 kg.

2.2.3 Reactor Building

Reactor coolant was discharged from the RCS through the PORV located on top of the pressurizer. The PORV discharges to the RCDT which is located in the basement of the RB (see Figure 2-5). The RCDT contains two safety components: a relief valve which discharges to the RB sump and a rupture disk which discharges to the RB floor adjacent to the RCDT cubicle. Both safety devices were believed to have performed their respective safety functions. The rupture disk was subsequently found in an open or ruptured condition, as expected. If the relief valve had initially operated during the pressure buildup in the RCDT, it would be expected to reseat after the rupture disk opened, thereby minimizing any continuous release to the RB sump via that pathway.

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

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

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

2.2.4 Auxiliary and Fuel Handling Buildings

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

The block orifice is the normal pressure reduction device for flow rates up to 45 gpm through the MU&P system. The block orifice and its isolation valve became blocked during the accident; subsequently, they were bypassed. As a result, very little fuel was measured in the block orifice and its associated piping. The letdown flow was directed to the letdown filters and purification demineralizers at very low rates during the accident and was then routed to RCBT "A" and the makeup tank. Letdown flow was lost several times during the accident due to flow blockage. More than 24 hours after the initiation of the accident, the purification demineralizers also were bypassed and letdown was directed to RCBT "B". Due to the flow blockage of the letdown coolers and restrictions in the block orifice, fuel transport to the filters, demineralizers, and RCBTs was limited.

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

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

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

2.3 Fuel Transport and Relocation Due To Cleanup Activities

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

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

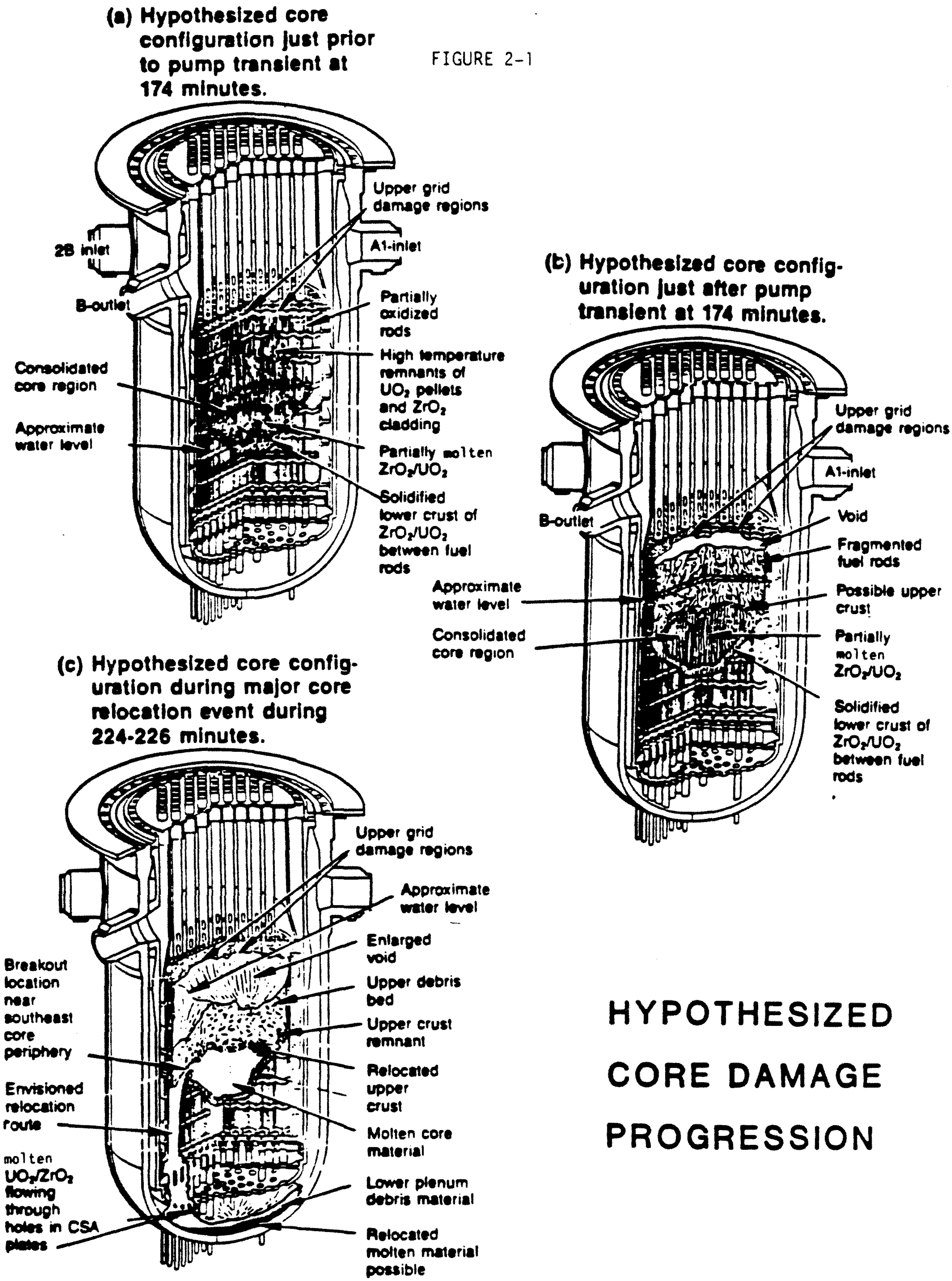
In the AFHB, the primary cause of fuel relocation from cleanup operations was water processing through the RCBTs, MWHT, SRSTs, and SDS monitoring tanks. Additionally, fuel material may have relocated into SFP "A" as part of fuel canister transfers from the RV. While every effort was made to flush residual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

TABLE 2-1

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

Reactor Coolant System	**Kilograms**
"A" Side	
Hot Leg	1
OTSG Upper Tube Sheet	1
Tube Bundle	3
Lower Head	1
J-Legs	1
Reactor Coolant Pumps	2
Cold Legs	1
"B" Side	
Hot Leg	8
Decay Heat Drop Line	30
OTSG Upper Tube Sheet	125
Tube Bundle	9
Lower Head	1
J-Legs	6
Reactor Coolant Pumps	20
Cold Legs	7
Pressurizer	12
Reactor Building	
RB Basement/Sump	5
Reactor Coolant Drain Tank	0.1
Letdown Coolers	4
Core Flood System	1
Auxiliary/Fuel Handling Buildings	
Makeup and Purification System	6
Seal Injection System	1
Reactor Coolant Bleed Tanks A, B, and C	15
Waste Disposal Liquid System	1

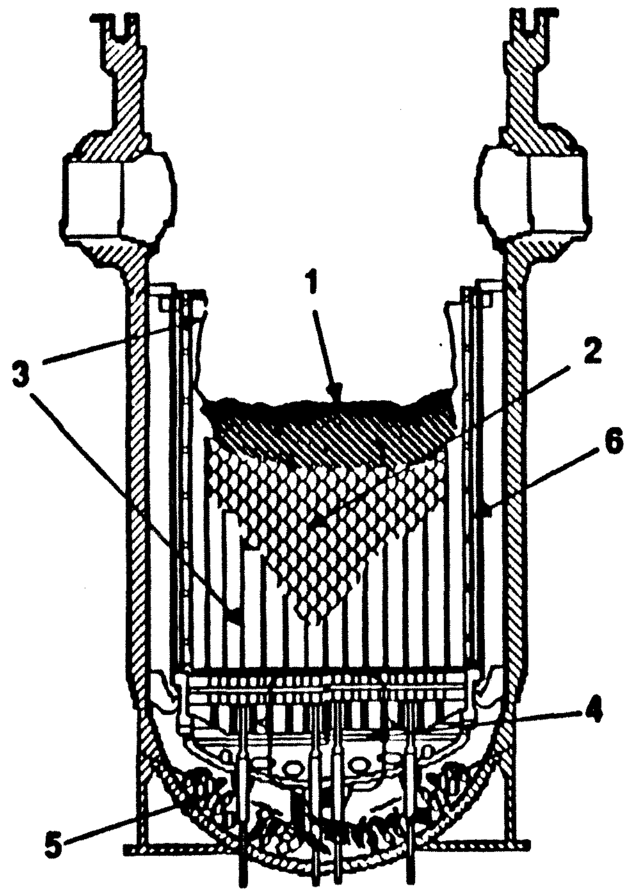
FIGURE 2-1



HYPOTHESIZED CORE DAMAGE PROGRESSION

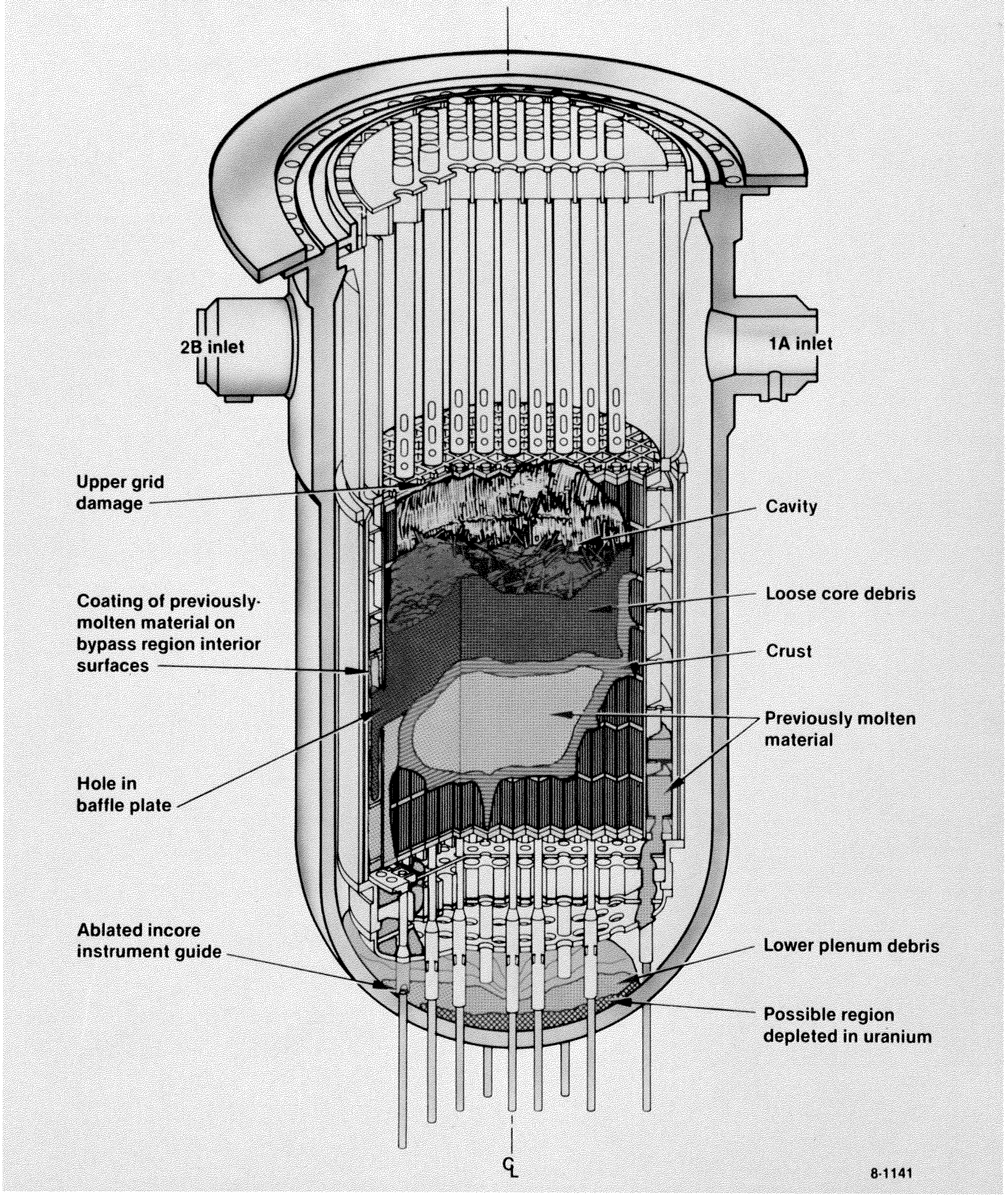
FIGURE 2-2

POST-ACCIDENT ESTIMATED CORE MATERIAL DISTRIBUTION



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

TMI-2 Core End-State Configuration



REACTOR COOLANT SYSTEM COMPONENTS

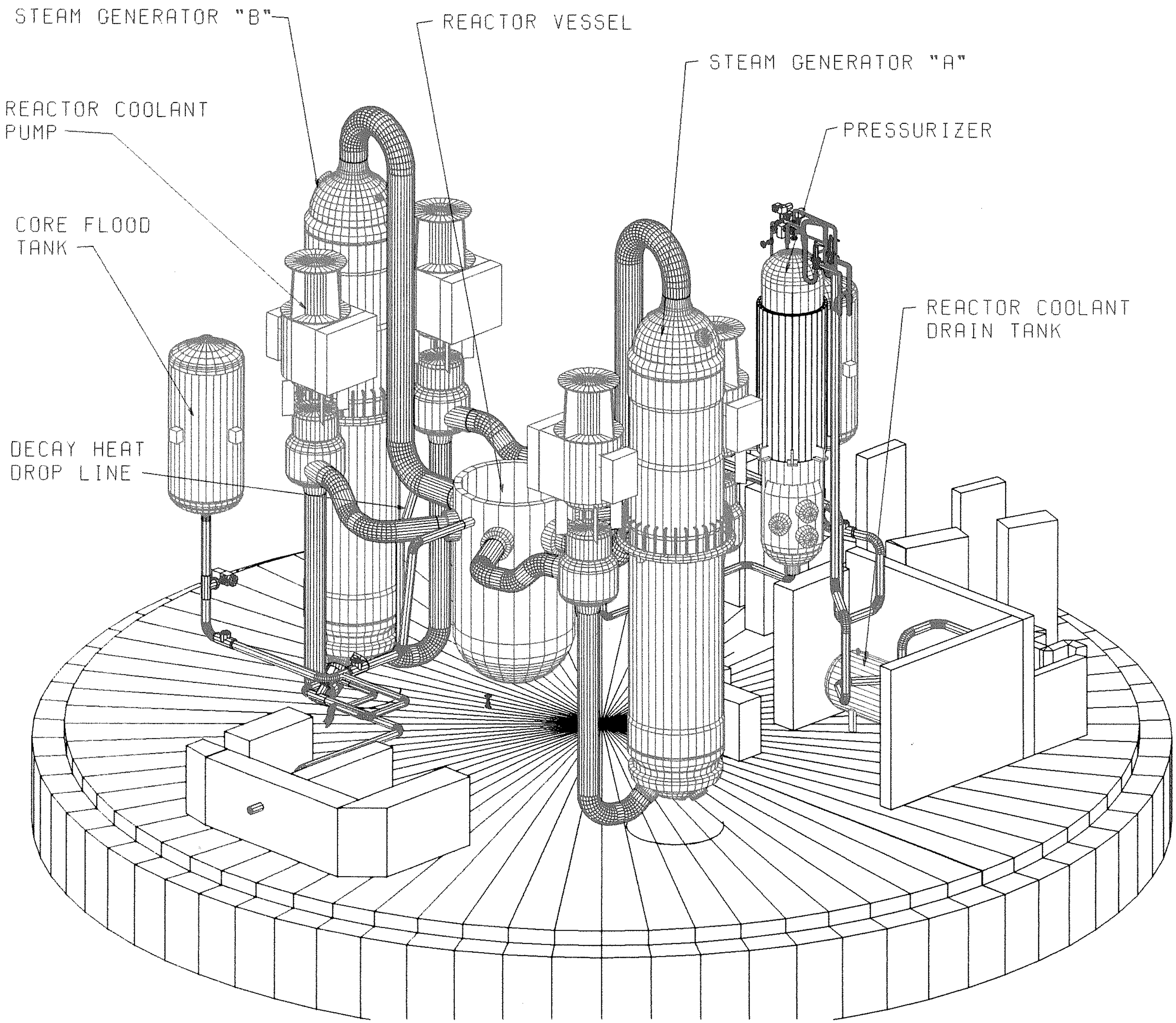


FIGURE 2-4

TMI-2 REACTOR BUILDING BASEMENT

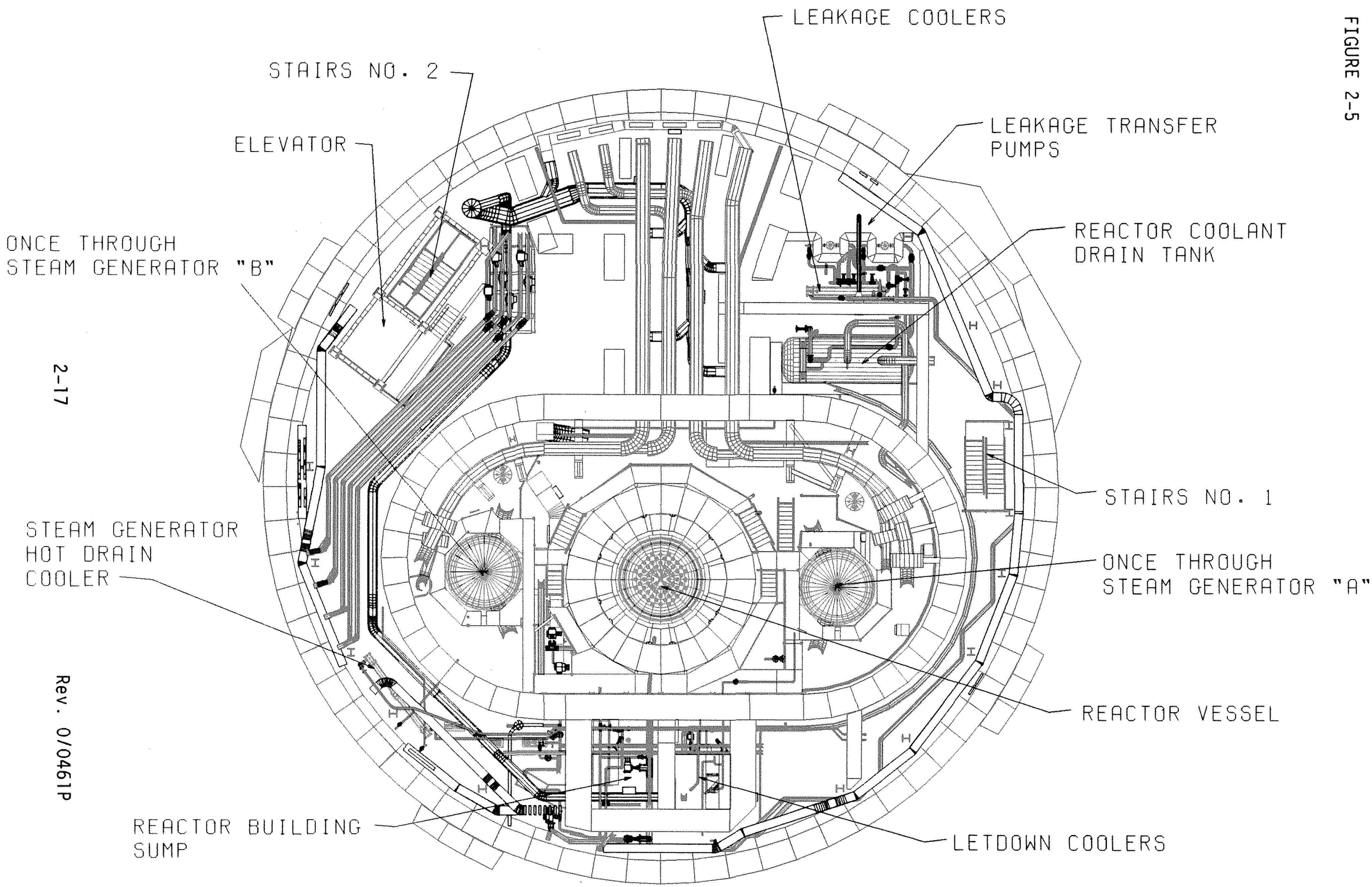


FIGURE 2-5

Rev. 0/0461P

3.0 FUEL SURVEY TECHNIQUES

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

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

3.1 Gamma Dose Rate and Spectroscopy Techniques

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

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

Two (2) gamma spectroscopy detector systems were primarily utilized for residual fuel measurements at TMI-2. NaI(Tl) detector measurements were performed in many AFHB cubicles from 1983 through 1987. The NaI(Tl) detector has a good efficiency and adequate sensitivity to detect the characteristic Ce-144 2.185 MeV gamma radiation. Limitations on the use of the NaI(Tl) detector in the AFHB were due to the relatively high ambient dose rates in several of the cubicles during early fuel characterization measurements and the relatively short half-life of Ce-144 (284 days).

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

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

3.2 Neutron Detectors and Activation/Interrogation Techniques

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

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

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

- Copper activation coupons become irradiated in the presence of a neutron flux. The Cu-64 then decays by positron emission resulting in two 0.511 MeV gamma rays scattered at 180°, with a 0.66% yield. By using a coincidence counting system consisting of two NaI(Tl) detectors, discrimination of this dual emission from background is possible. Copper activation is insensitive to gamma radiation, making this method particularly useful in areas of high gamma fields.

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

Active neutron interrogation is more sensitive than passive counting for quantifying small deposits of fuel. At TMI-2, a Sb-Be photoneutron interrogation method was used. This photoneutron interrogation method uses an Sb-Be photoneutron source to produce low-energy (approximately 0.024 MeV) interrogating neutrons via the Be (γ,η) reaction by irradiating beryllium with the 1.692 MeV gamma ray emitted from a Sb-124 isotope. These interrogating neutrons impinge upon the fuel and induce fission reactions in the fissile material contained in the fuel. Some of the fission neutrons returning from the surrounding fuel are detected by a He-4 fast neutron recoil proportional counter. The He-4 neutron counter can differentiate the higher-energy induced fission neutrons from the lower-energy photoneutron source and gamma rays on the basis of the pulse height signal; with directional shielding, it can also operate effectively and efficiently in a substantial radiation field.

3.3 Alpha Fuel Detectors

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

3.4 Direct Sampling and Analysis Techniques

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

3.5 Visual Inspection

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

3.6 Fuel Measurement Uncertainties

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

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

The technique of using video images to determine the quantity of core debris remaining produces an estimated quantity. There is an uncertainty associated with the estimate for each of the discrete quantities so estimated. Unlike a direct measurement, the uncertainty in an observed or video estimate is difficult to quantify. The magnitude of the uncertainty can itself only be an estimate and will not be the same in every location. This uncertainty is affected by several factors, including:

- Error in Estimated Dimensions - Because of camera angles, and the fact that an attempt is being made to ascertain extent in three dimensions from a two-dimensional image, there is an uncertainty in each of the three dimensions estimated for the volume of the mass. The vertical and lateral extent can be estimated quite accurately, perhaps to within 10%, provided there are nearby landmarks of known dimensions to serve as points of comparison. The depth of the object along the axis of view, however, is less well known because the material in the foreground may obscure the view of what is behind it. This is usually resolved by conservatively assuming the object fills the available space between its front surface and the structure behind it.

- Poor Visibility - In some instances, the view showed water that was murky from suspended material or biological growth, or the lighting was poor because of interfering structures and unwanted shadows. In these cases, uncertainties in extent of the material are resolved by conservative assumptions.

- Lack of Access - In a limited number of places, it was impossible to maneuver a camera into position to view a location. Reasonably conservative assumptions about the amount of fuel debris remaining in those places were made and are described in this report.

- Assumed Density and Composition of Debris - Once the volume was determined, an assumed composition was applied to the material. The density and composition of the samples varied within a range of 10 to 20%, with a few non-homogeneous particles outside this range. Whenever a deposit of material could not be positively identified, it was conservatively assumed to be fuel bearing.

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

3.7 Fuel Measurement Selection

Table 3-1 presents a matrix of fuel locations versus measurement methods for the various TMI-2 residual fuel locations. All areas containing residual fuel are listed and cross-referenced with the techniques selected for the area. Estimates were based on a review of accident flow data, radiological controls measurements, and existing fuel measurement data from similar locations or components.

TABLE 3-1

FUEL MEASUREMENT SELECTION

FUEL MEASUREMENT METHODS		Gamma	NaI	HPGe	Neutron	Alpha	Sample	Estimate*
AX004	Seal Injection Valve Room							X
AX005	MU Pump Room 1C	X						
AX006	MU Pump Room 1B		X					
AX007	MU Pump Room 1A		X					
AX008	Spent Resin Stor. Tank 1B							X
AX009	Spent Resin Stor. Tank 1A							X
AX010	Spent Resin Stor. Tank Pump							X
AX011	Aux. Sump Pump Valve Room		X					
AX012	Aux. Bldg. Sump Tank Room		X					
AX019	WDL Valves		X					
AX020	RC Bleed Tanks 1B, 1C		X					
AX021	RC Bleed Tank 1A			X				
AX024	Aux. Bldg. Sump Filters	X						
AX026	Seal Inj. Filters MU-F-4A,B		X					
AX102	RB Sump Filters							X
AX112	Seal Return Coolers & Filters		X					
AX114	MU&P Demin. 1A						X	
AX115	MU&P Demin. 1B						X	
AX116	Makeup Tank Room		X					
AX117	MU&P Filters 2A,B 5A,B		X					
AX124	Conc. Liquid Waste Pump							X
AX129	Deborating Demin. 1B		X					
AX130	Deborating Demin. 1A		X					
AX131	Misc. Waste Tank							X
AX134	Misc. Waste Tank Pumps							X
AX218	Conc. Waste Storage Tank Room							X

* Derived only by calculation and/or video examination (i.e., no actual measurements).

TABLE 3-1 (Cont'd)

FUEL MEASUREMENT SELECTION

FUEL MEASUREMENT METHODS		Gamma	NaI	HPGe	Neutron	Alpha	Sample	Estimate*
FH001	Makeup Suction Valve Room		X					
FH003a	Makeup Discharge Valve Room		X					
FH003b	Makeup Discharge Valve Room		X					
FH014	Annulus							X
FH101	MU&P Valve Room		X					
FH106	Monitor Tanks & Sample Sink							X
FH109	Spent Fuel Pool A	X						
FH110	Spent Fuel Pool B							X
FH111	Fuel Cask Storage							X
FH112	Annulus							X
FH302	SDS Operating Area							X

TABLE 3-1 (Cont'd)

FUEL MEASUREMENT SELECTION

FUEL MEASUREMENT METHODS		Gamma	NaI	HPGe	Neutron	Alpha	Sample	Estimate*
RB01	Letdown Coolers Cubicle		X					
RB02	RB Sump						X	X
RB03	RC Drain Tank Cubicle						X	X
RB04	RB Basement Floor		X				X	
RB11	Decay Heat Drop Line	X					X	
RB21	Reactor Coolant Pumps							X
RB22	RCS Cold Legs	X					X	
	RCS J-Legs	X			X			
RB23	Reactor Vessel						X	X
RB31	Pressurizer		X				X	
RB32	OTSGs	X	X		X	X	X	
RB33	Core Flood Tanks & Drain Lines			X		X	X	
RB34	Incore Instr Guide Tubes (LCSA)			X				
RB35	Upper Plenum Assembly						X	X
RB36	Reactor Vessel Head						X	X
RB37	RCS Hot Legs	X						
RB38	Pressurizer Surge Line		X					
RB40	Fuel Transfer Canal	X						
	Endfittings				X			
	DWCS	X						
	TRVFS	X					X	
	Defueling Tool Rack	X						
	RB Drains	X						
	Tool Decontamination Facility	X						

4.0 FUEL REMOVAL ACTIVITIES

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

4.1 Auxiliary and Fuel Handling Buildings

As discussed in Section 2.0, the TMI-2 AFHB was contaminated as a result of the March 1979 accident and subsequent plant stabilization and water processing activities. A small, but measurable, quantity of fuel was transported into the piping and tanks of the MU&P System and the WDL System components, which are located in the AFHB.

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

4.1.1 Cleanup Approach

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

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

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

The initial cleanup of the AFHB took place during the early plant stabilization phase of the TMI-2 cleanup program (Reference 4.4). This effort consisted of removing the water that flooded the lower level of the AFHB during the accident and performing surface decontamination of the floors, walls, and equipment. The goal of this initial cleanup was to reduce the overall loose contamination throughout the AFHB and to reduce the requirement for respirators due to airborne radioactivity. In addition, there was a need to reduce the radionuclide content of water that was stored in tanks in the AFHB. This latter task was accomplished by the EPICOR II System.

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

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

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

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

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

Significant dose rate reductions were achieved in nearly all of the cubicles; most cubicles were acceptably decontaminated (Reference 2.16). An example of the success achieved in removing residual fuel from the AFHB is the MU&P demineralizers. It is noteworthy that the block orifice removal resulted in the elimination of approximately 370 grams of the initially estimated 400 grams of fuel. Post-defueling activities such as final draindown, water processing, and fuel pool decontamination are expected to further reduce the AFHB residual fuel inventory.

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

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

4.1.2.1 Mechanical Decontamination

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

4.1.2.1.1 Hands-On Decontamination

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

4.1.2.1.2 High-Pressure Water Spray and Flushing

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

4.1.2.1.3 Kelly Vacuumac

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

4.1.2.1.4 Concrete Scabbling

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

4.1.2.2 Robotic Equipment

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

4.1.2.3 System Decontamination

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

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

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

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

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

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

4.1.2.5 Sludge and Resin Removal (References 4.10 and 4.11)

Resin removal was primarily performed in the MU demineralizers, the cleanup demineralizers, the spent fuel demineralizer, the deborating demineralizers and the evaporator condensate demineralizers. The AB Sump was desludged. The sludge and resin were deposited in the SRSTs or directly into a solidification liner, dewatered, prepared for shipment, and shipped offsite for disposal. Further details of the sludge and resin removal are provided in Section 4.1.3.

4.1.3 Auxiliary and Fuel Handling Buildings Cleanup Activities

4.1.3.1 Seal Injection Valve Room

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

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

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

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

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

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

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

The RCBT cubicles have been extensively decontaminated since the accident. Manual and robotic decontamination efforts have significantly reduced the airborne radionuclide concentrations. The RCBT 1A was flushed internally to remove sedimentation and residual fuel debris but was placed in service subsequently for water processing activities. The B and C RCBTs were not decontaminated internally.

4.1.3.3 Makeup and Purification Demineralizers (References 4.12 and 4.13)

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

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

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

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

4.1.3.4 Auxiliary Building Sump (Reference 4.14)

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

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

4.1.4 Auxiliary and Fuel Handling Buildings Fuel Removal Assessment

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

4.2 Reactor Building Fuel Removal and Decontamination Activities

As discussed in Section 2.0, the RB was contaminated as a result of the TMI-2 accident. A small, but measurable amount of fuel was transported to the RB as a result of: the accident, subsequent plant stabilization, and water processing activities (see Table 2-1). The following sections discuss those areas of the RB where decontamination activities were performed which resulted in the removal of residual fuel. Other locations in the RB which contain residual fuel (e.g., plenum) are described further in Section 5.2.

4.2.1 Cleanup Approach

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

4.2.2 Reactor Building Cleanup Equipment and Techniques

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

4.2.3 Major Reactor Building Cleanup Activities

4.2.3.1 General Area

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

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

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

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

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

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

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

4.2.3.4 Reactor Coolant Drain Tank (Reference 4.20)

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

4.2.4 Reactor Building Fuel Removal Assessment

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

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

4.3 Reactor Coolant System Defueling Operations

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

4.3.1 Reactor Coolant System Defueling Approach

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

4.3.2 Reactor Coolant System Defueling Equipment and Techniques

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

4.3.3 Reactor Coolant System Defueling Activities

4.3.3.1 Pressurizer Defueling Operations (Reference 4.21)

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

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

4.3.3.2 Pressurizer Spray Line

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

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

Pick-and-place and vacuuming techniques were used to defuel the "A" and "B" OTSG upper tubesheets. Long-handled gripping tools were used to lift large pieces of debris into canisters and a vacuum system removed the smaller debris. While this process essentially succeeded in defueling the "A" OTSG tubesheet, a crust of tightly adherent debris remained on the surface of the "B" OTSG tubesheet. Despite extensive efforts to remove this crust or to collect a sample for analysis by scraping, no further progress was achieved. It has been concluded that no further defueling of the "B" OTSG tubesheet is necessary or appropriate because the small amount of remaining fuel is tightly adherent and unlikely to be transported elsewhere in the system in the future due to a lack of a motive force and our demonstrated inability to remove it with dynamic defueling techniques.

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

The hot legs were defueled using a combination scraper/ vacuuming tool and the in-vessel vacuum system. Residual fuel in the hot legs was scraped, flushed, and vacuumed into defueling canisters as part of RV defueling (Section 4.4).

Further assessment of the dynamic defueling techniques applied in attempting to remove the tightly adherent residual fuel in the "B" OTSG upper tubesheets is provided in Section 6.0.

4.3.3.4 Decay Heat Drop Line (Reference 4.23)

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

4.3.4 Reactor Coolant System Fuel Removal Assessment

Extensive defueling operations were performed in the RCS with the goal of removing the majority of fuel transported to the RCS as a result of the accident. These activities were successful. For example, defueling operations removed greater than 90% of the debris in the pressurizer, decay hot drop line, and hot legs and approximately 70% of the fuel in the OTSG upper tubesheets. The residual fuel quantity in the RCS components is discussed in Section 5.3.

4.4 Reactor Vessel

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

4.4.1 Reactor Vessel Defueling Approach (References 4.24 and 4.25)

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

4.4.2 Reactor Vessel Defueling Equipment and Techniques

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

4.4.3 Reactor Vessel Defueling Activities

4.4.3.1 Initial Defueling Activities

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

4.4.3.2 Core Region Defueling

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

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

4.4.3.2.1 Use of the Core Bore Machine (Reference 4.27)

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

4.4.3.2.2 Core Topography and Drill String Removal

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

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

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

4.4.3.2.3 Stub Assembly Removal

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

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

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

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

4.4.3.3 Lower Core Support Assembly Disassembly and Defueling

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

4.4.3.3.1 Introduction

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

4.4.3.3.2 Initial Lower Core Support Assembly Drilling Operations

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

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

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

4.4.3.3.3 Lower Grid Rib Section Removal

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

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

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

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

4.4.3.3.4 Lower Grid Distributor Plate Removal

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

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

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

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

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

4.4.3.3.5 Lower Grid Forging Removal

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

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

4.4.3.3.6 Incore Guide Support Plate Removal

By the end of November 1988, preparations had begun to disassemble the incore guide support plate. A few loose fuel rod segments located on the plate were removed. Additionally, loose debris up to 33 cm (13 inches) below the incore guide support plate was cleared.

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

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

4.4.3.3.7 Flow Distributor Plate Removal

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

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

4.4.3.3.8 Lower Core Support Assembly Remnant Defueling

Following completion of LCSA plate removal, LCSA remnant defueling began. This consisted of removing the loose and resolidified debris that remained on the plate remnants. The primary defueling approach utilized high volume, low pressure water flush and low volume, high pressure cavitating water jet flush. Much of this work was done under conditions of poor to zero visibility due to the suspension of loose debris and was accomplished by indexing positioning tools to LCSA remnants to access specific target areas. High volume, low pressure water flush tools were used first to flush the loose debris off the remnants and into the lower head. The newly exposed resolidified debris was then dislodged with the cavitating water jet. This displaced material was then removed from the lower head as part of lower head defueling using airlifting and vacuuming as well as pick and place activities.

4.4.3.4 Lower Head Defueling

Lower head defueling commenced following the removal of the flow distributor plate which provided a large access hole to the lower head. Lower head defueling included the removal of the accident generated monolith and loose core debris on the lower head as well as post-accident generated debris that relocated to the lower head during the defueling of the other areas within the vessel.

This evolution involved sizing and conditioning of the resolidified material in the monolith with the impact hammer and the cavitating water jet; pick and place of the rods and large debris; and airlifting and vacuuming of loose core debris.

4.4.3.4.1 Loose Debris Defueling in the Lower Head

Prior to removal of the flow distributor plate, a large quantity of material was airlifted from the lower head to facilitate cutting and removal of that plate. When the final LCSA plate was removed, airlifting of the lower head was again performed to remove additional debris. The airlifting activity removed the bulk of the loose debris; pick and place activities removed the remaining loose debris. These activities uncovered a monolith of resolidified debris in the lower head. Following the conditioning and sizing of the monolith, airlifting was repeated in order to remove the remainder of the core debris.

4.4.3.4.2 Monolith Defueling in the Lower Head

The accident resulted in formation of a resolidified mass in the lower head which was irregular in shape varying in depth to less than two (2) feet in the center. This resolidified debris was sized and conditioned successfully using two (2) tools. The first, an impact hammer, was used to break up the central region where there was ready access from above. The monolith was broken up in much the same way as one would approach the demolition of a concrete slab, starting from the outside edges and working inward. The cavitating water jet was used to break up the remaining resolidified debris on the lower head which was located under the LCSA remnants and was inaccessible to the impact hammer. Pick and place and airlifting then removed the conditioned debris.

4.4.3.4.3 Vacuuming in the Lower Head

Following the completion of pick and place activities and airlifting in the lower head, the lower head was vacuumed to minimize the relocation of core debris to other surfaces in the vessel during use of the airlift and to improve visibility. The in-vessel vacuum system, a modified application of the in-vessel filtration system utilizing a knockout canister and filter canister in series, was used for this evolution.

4.4.3.5 Upper Core Support Assembly Defueling

UCSA defueling encompassed removing the fuel debris located between the baffle plates and the core barrel (i.e., the core former region). Resolidified debris formed in this region during the accident. Loose debris also was deposited during the accident and subsequent defueling operations elsewhere in the RV. The scope of this defueling effort includes gaining access to the core former region through the removal of the baffle plates and removal of the resolidified and loose debris.

4.4.3.5.1 Gaining Access to the Upper Core Support Assembly

To gain access to the UCSA required removal of the baffle plates. This was accomplished by cutting the baffle plates into eight sections using the plasma arc torch. Then, the bolts and screws that fastened the baffle plates to the former plates were removed. Bolt removal required use of a hydraulic untorquing tool and a drill tool. The drill tool was used when the untorquing tool either failed to remove the bolt or the untorquing tool could not be used. A total of 864 bolts and screws were removed. A third operation involved clearing the kerf and recutting the baffle plate cuts previously made by the plasma torch using an abrasive saw or drill.

4.4.3.5.2 Baffle Plate Handling

Baffle plate handling exposed the UCSA for defueling of the core former area. Two of the eight baffle plate sections were removed and hung from vent valve seats. The exposed area was defueled before removal of the next plate section. Handling of the plates essentially rotated each plate 90° from its original location to its final location.

4.4.3.5.3 Defueling of Upper Core Support Assembly

Defueling the UCSA included brushing, vacuuming, conditioning resolidified debris, and a second vacuuming.

The inboard and outboard surfaces of the baffle plates, the top and bottom surfaces of the former plates, and the inboard surface of the core barrel, which contained visible fuel, were brushed. The task was accomplished using hydraulically-powered counter-rotating brushes mounted on a pivoting deployment end effector.

Loose debris was vacuumed from the core former plates after removal of the baffle plates and again after conditioning the resolidified debris and brushing the plate surfaces. The in-vessel vacuum system was used for this task. Conditioning the resolidified debris in the UCSA was accomplished using mechanical methods and the cavijet system. While the baffle plates were removed, mechanical devices were used to clear the flow holes in the periphery of the grid rib section outboard of the baffle plates which contained core debris.

4.4.4 Reactor Vessel Fuel Removal Assessment

Figure 2-2 indicates that there was approximately 133,000 kg of core debris in the RV following the accident. The extensive defueling efforts described in the above sections have been very successful in removing this core debris. A summary of the efforts in defueling the various components of the RV is provided below.

4.4.4.1 Core Region

Figure 2-2 indicates that there was a total amount of 104,000 kg of core debris in the upper debris bed, resolidified mass, and intact fuel assemblies. Of this total, 99.9% was removed as a result of defueling activities. The small percentage of core debris remaining in this region is essentially in the R-6 incore location. Following extensive defueling in this area, some of this resolidified mass remains.

4.4.4.2 Lower Core Support Assembly

Figure 2-2 indicates that there was approximately 6000 kg of core debris contained in LCSA components following the accident. More than 90% of this core debris was removed during the extensive LCSA removal phase and LCSA remnant defueling. The majority of the residual core debris is in an inaccessible region between the forging and the ISGP.

4.4.4.3 Lower Head

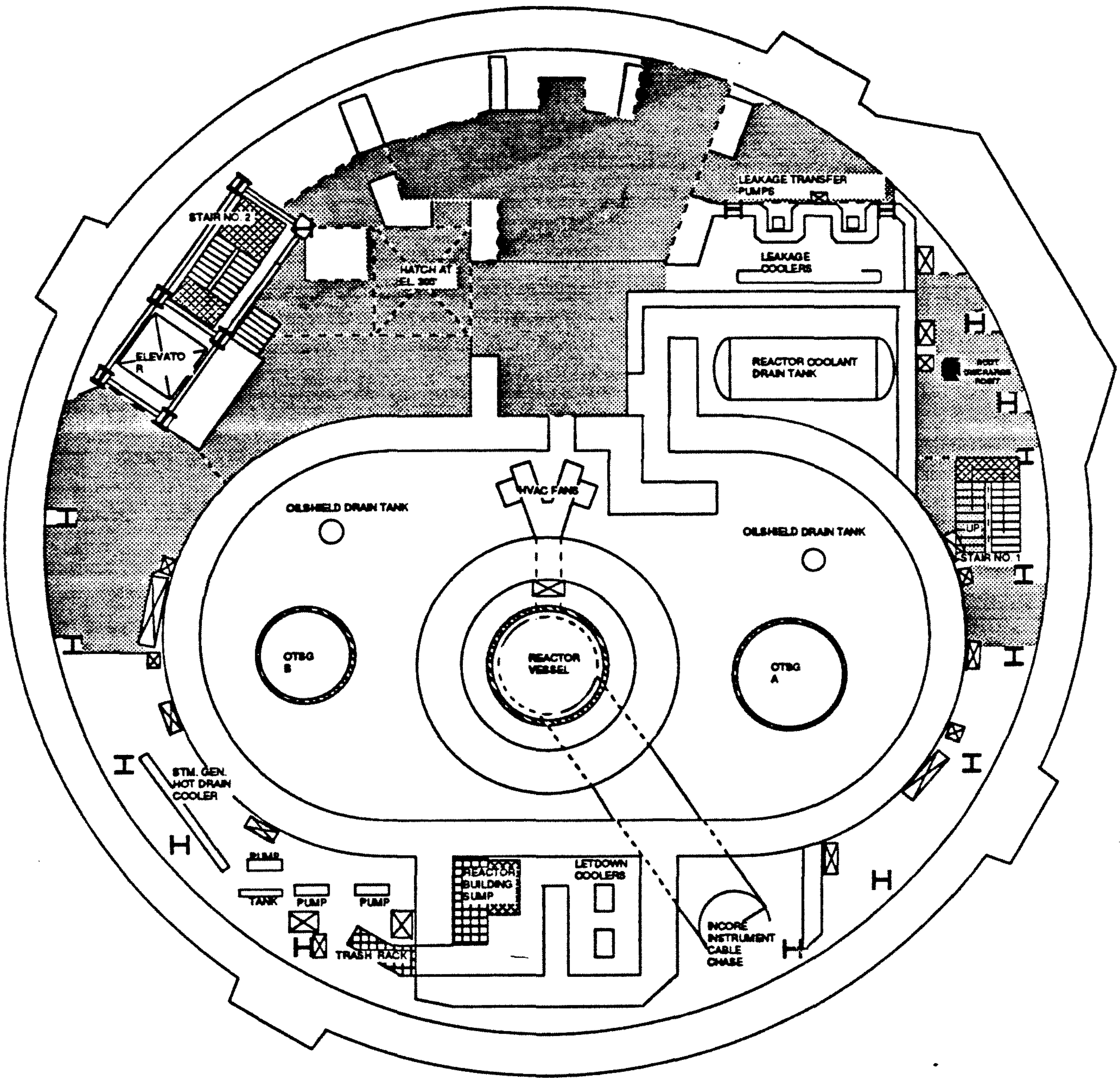
Figure 2-2 indicates that approximately 19,000 kg of core debris (i.e., 12,000 kg of loose debris and 7,000 kg of resolidified mass) existed in the RV lower head due to the accident. The defueling efforts in the lower head region, described in Section 4.4.3.4, have removed approximately 99% of this debris. Prior to the start of the NRC-sponsored Reactor Vessel Lower Head Sampling Program, the remaining material was primarily a layer of fine dust over the entire lower head surface that could not be effectively vacuumed. The sampling program has generated some additional debris which we will attempt to remove following the completion of the program.. A small amount of fuel may be removed as a result of this cleanup effort.

4.4.4.4 Upper Core Support Assembly

Video inspections of the UCSA indicated that there was approximately 4000 kg of core debris in this region, primarily behind the baffle plates. Following removal of the baffle plates, the core debris in this area was accessible for defueling activities (e.g., flushing, vacuuming, cavijet). Approximately 95% of this core debris was removed. The majority of the remaining material resides in an inaccessible 1-inch annular gap between the thermal shield and the core barrel.

FIGURE 4-1

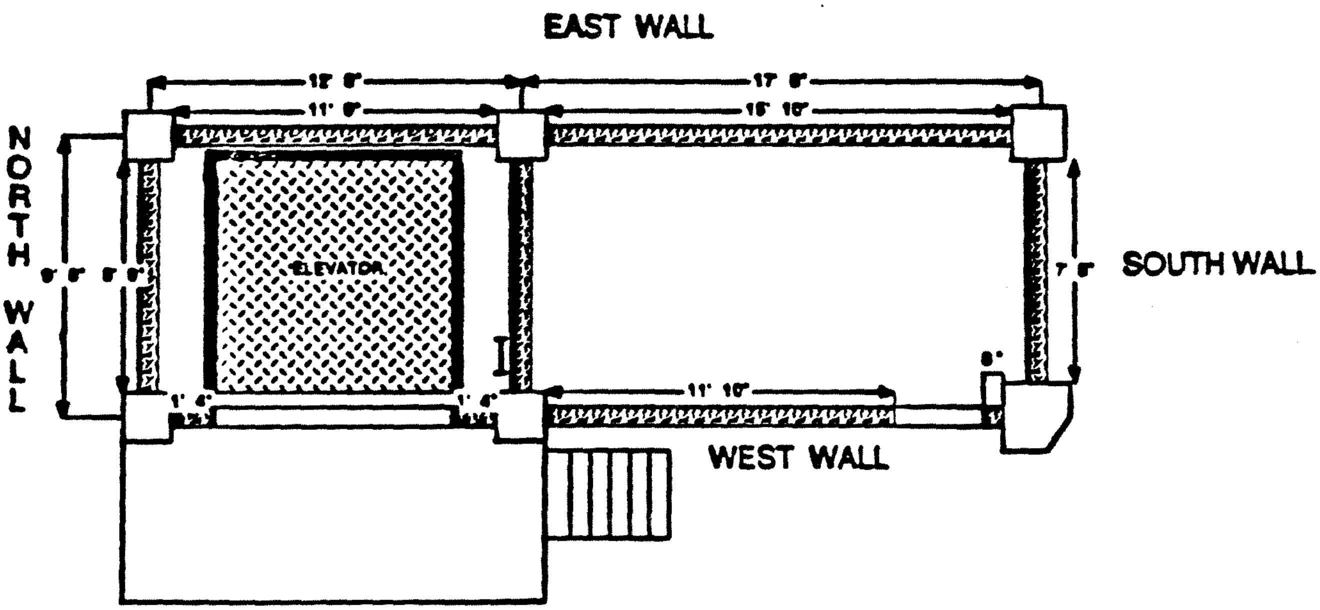
REACTOR BUILDING BASEMENT FLOOR PLAN (DESLUDGED)



NOTE: Shaded area represents desludged portion of basement.

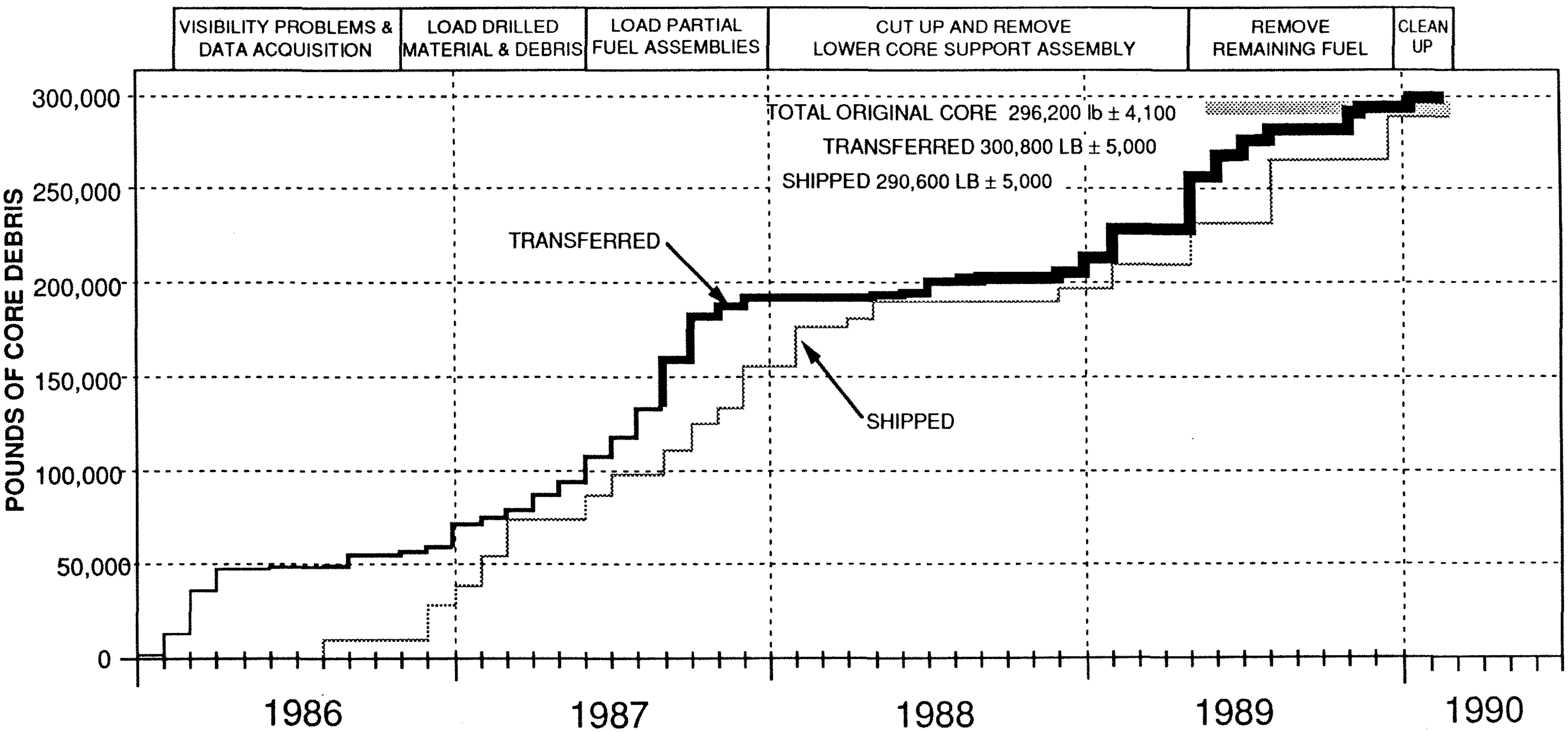
FIGURE 4-2

FACE IDENTIFICATION



SUMMARY:

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

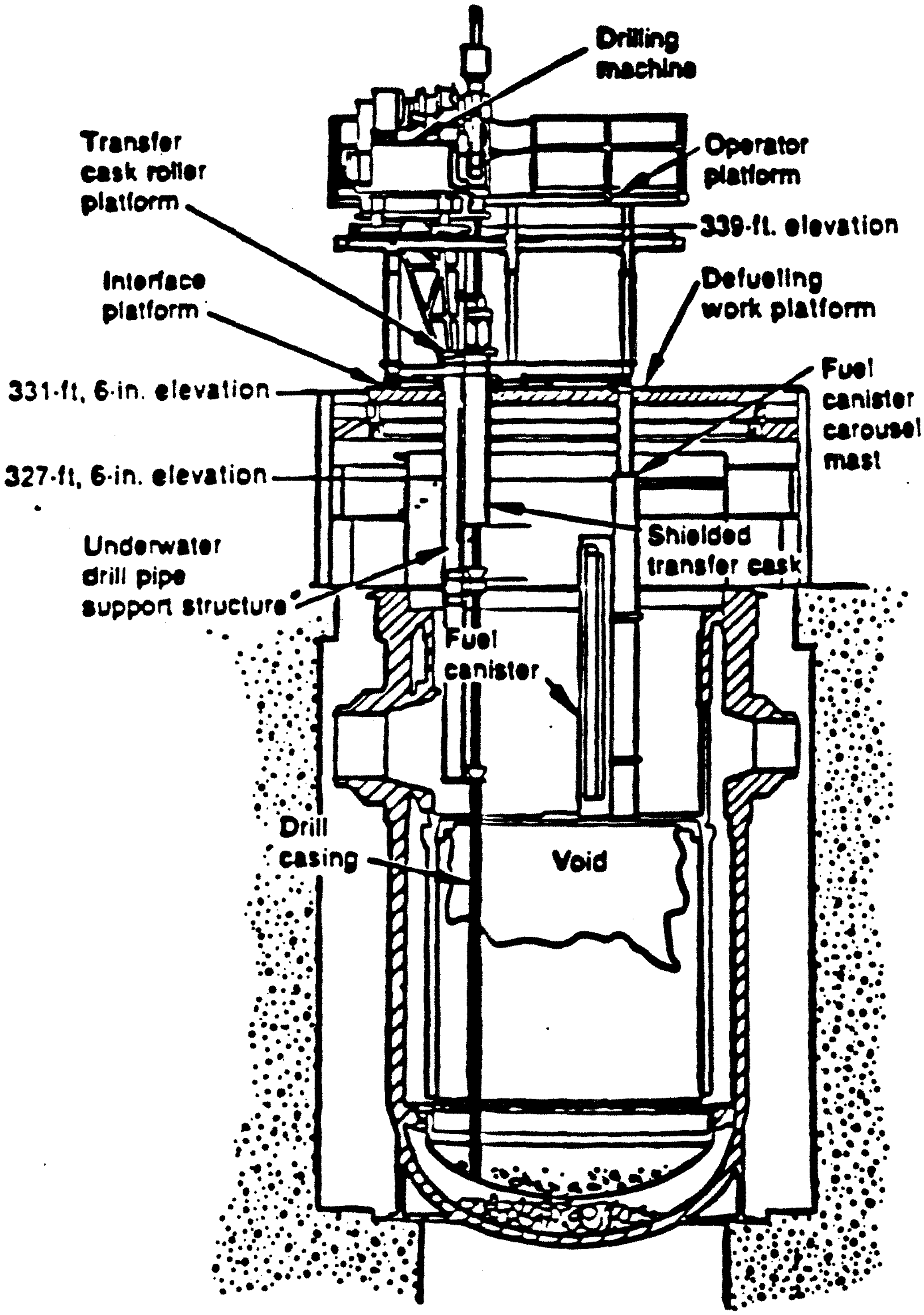


TMI-2 Defueling Progress

FIGURE 4-3

FIGURE 4-4

CORE BORE MACHINE



LOWER CORE SUPPORT ASSEMBLY

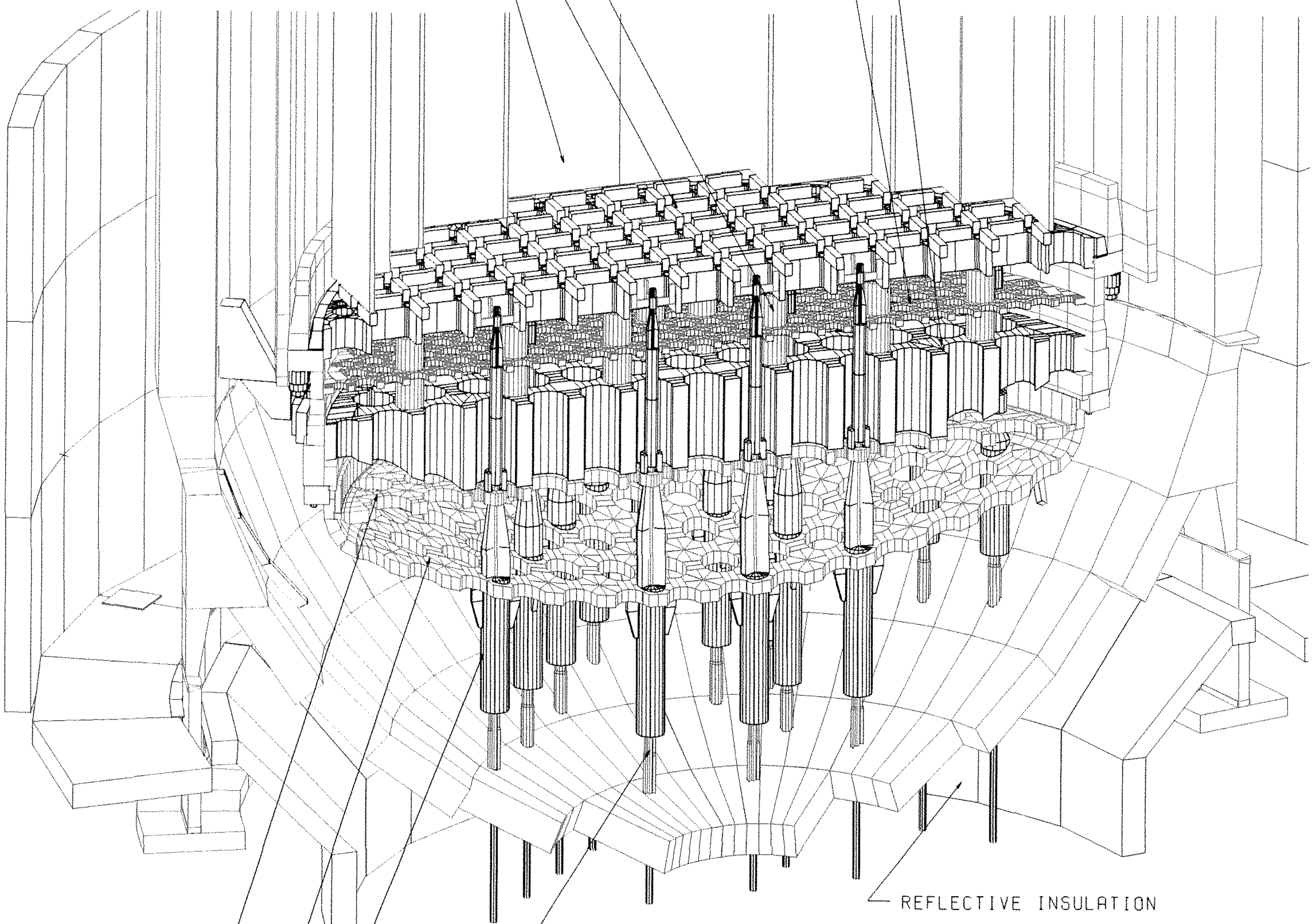


FIGURE 4-5
LOWER CORE SUPPORT ASSEMBLY

5.0 RESIDUAL FUEL QUANTIFICATION AND CRITICALITY ASSESSMENT

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

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

5.1 Auxiliary and Fuel Handling Buildings

During the accident, core debris was transported to the AFHB as a result of the core degradation event and the concurrent RCS MU&P System operation. Section 2.0 reported that approximately 25 kg of fuel material (i.e., UO_2) was transported to the AFHB during the accident sequence. Section 4.1.1 indicated that up to 15 kg of fuel may have been relocated into the AFHB as part of water processing and defueling operations (i.e., potentially a total of 40 kg). Based on these estimates, it could be concluded that AFHB residual fuel conditions were maintained significantly below the SFML during the accident and subsequent cleanup period. Nonetheless, a significant cleanup and decontamination effort was undertaken (as described in Section 4.1) to reduce dose rates and remove fuel where practical. These efforts have further reduced the remaining residual fuel content in the AFHB.

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

5.1.1 Auxiliary and Fuel Handling Buildings Cubicles

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

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

5.1.2 Areas Containing Fuel in the Auxiliary and Fuel Handling Buildings

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

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

These cubicles contain piping and/or tanks that are part of the resin transfer system. This system has been and will continue to be used to remove the highly radioactive resin from the MU demineralizer and the cleanup demineralizers ion exchangers. Those resins, in place during the accident, became contaminated with fuel debris.

The resin transfer system has been used to remove most of the resins from the cleanup demineralizers, essentially all of the resins from the "A" MU demineralizer, and most of the resins from the "B" MU demineralizer. Additionally, the resins have been removed from the spent fuel demineralizers and the deborating demineralizers.

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

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

Seven cubicles, listed below, had fuel deposited in the piping and/or tanks as a result of the accident. The fuel content of these cubicles was measured prior to the system flush and resin transfer activities (References 5.1 and 5.2).

Cubicle	Equipment/Name	Fuel Measurement
AX015a	Cleanup Filters	5 grams
AX015b	Cleanup Filters	5 grams
AX016	Cleanup Demineralizer	160 grams
AX017	Cleanup Demineralizer	160 grams
AX114	MU Demineralizer	580 grams
AX115	MU Demineralizer	420 grams
FH001	MU Valve Room	270 grams
	TOTAL =	1600 grams

This estimate is a total of the fuel measured in the seven cubicles before decontamination and resin transfer. Following these measurements, the cleanup filters, cleanup demineralizers, and 85% of the combined total of the A and B MU demineralizers resin were removed. Therefore, a reasonable estimate of the residual fuel content is 420 grams. For bounding purposes, 800 grams is used in Table 5-2.

5.1.2.2 Cubicle AX102 - Reactor Building Sump Pump Filter Room

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

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

The residual fuel content of AX102 has not been measured because the system is still in use. The residual fuel content will be measured as part of the SNM accountability program. However, a bounding estimate of the residual fuel content of AX102 is 200 grams. This estimate is conservative since any fuel deposited in the RB sump filters and piping as a result of the accident is believed to have been flushed into the filters and removed as part of the multiple (over 30) filter changeouts or by being flushed to the MWHT. The major use of the RB sump filters during the post-accident period has been to filter water transferred from the AB sump to the MWHT. Thus, a small quantity of fuel could have been transferred from the AB sump to the sump filters or associated piping. Therefore, the total measured content of the AB sump, less than 200 grams, was selected as the bounding estimate for the current residual fuel inventory for AX102.

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

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

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

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

Cubicles AX124 and AX218 are located in the AB and they contain the concentrated liquid waste pumps (AX124), the CWST (AX218), and associated piping. The CWST has been used since the accident as a holding tank for decontamination waste water.

Cubicles FH008, FH009, and FH012 contain the neutralizer tank pumps (FH008), the neutralizer tanks (FH009), and the neutralizer tank filters (FH012). The neutralizer system has been used as a batch tank to receive the effluent from the MWHT and feed it into the EPICOR II System for filtration and purification via ion exchangers. Although originally intended to be used to chemically treat waste liquid, the neutralizer system has not been used in that manner since the TMI-2 accident.

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

The bounding estimate of the residual fuel content in the two MWHT cubicles is 1 kg of fuel. This estimate was developed by comparing the volume of the MWHT (approximately 20,000 gallons) to the AB sump (approximately 7600 gallons). The MWHT holds approximately three times the volume of the AB sump. The maximum measured fuel content of the AB sump (less than 200 grams; Reference 5.3) was then multiplied by a factor of three and rounded up to 1 kg for conservative purposes.

The bounding estimate of the residual fuel content of the CWST cubicles is 0.5 kg. This estimate was developed by comparing the holding volume of the CWST (approximately 9600 gallons) to that of the AB sump (approximately 7600 gallons). The CWST holds about 1.3 times the volume of the AB sump. The maximum measured fuel content of the AB sump (less than 200 grams) was then multiplied by 1.3 and then rounded to 0.5 kg for conservative purposes.

The bounding estimate of the residual fuel content of the neutralizer tank cubicles is 1 kg of fuel. This estimate was developed by comparing the total volume of the two neutralizer tanks (approximately 19,300 gallons) in FH009 to the volume of the AB sump (approximately 7600 gallons). The neutralizer tanks hold about three times as much as the AB sump. The maximum measured fuel content of the AB sump (i.e., less than 200 grams) was then multiplied by three and rounded to 1 kg for conservative purposes.

All of the three bounding estimates are believed to be highly conservative because the fuel content of the AB sump is below the MDL of the measurement. Also, the estimates are conservative because of rounding up of all values.

Another benchmark for comparison of the bounding nature of the estimates of residual fuel in the MWHT, CWST, and neutralizer cubicles is the measured residual fuel content in the MU tank cubicles. The MU tank was used to receive and hold unfiltered reactor coolant for a considerable portion of the post-accident period. Measurement of the residual fuel content in the MU tank cubicle (Reference 5.4) found approximately 300 grams deposited in the cubicle, virtually all of it in the tank (volume approximately 4500 gallons). Although the tank volume is smaller than the volume of the AB sump, the water held was unfiltered reactor coolant, which is known to have a significantly greater fuel content than the surface decontamination and system flush waste water which was in the AB sump.

5.1.2.4 Cubicle FH014 - Annulus
Cubicle FH112 - Annulus
Cubicle FH205 - Annulus

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

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

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

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

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

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

5.1.2.6 Cubicle FH109 - Spent Fuel Pool "A"

The vast majority of the fuel in SFP "A" is contained inside the fuel, filter, and knockout canisters stored in the fuel racks. The exact number of filled canisters will vary until all fuel bearing canisters have been shipped from TMI-2 to INEL for long-term storage. The canisters are inherently subcritical by design (Reference 5.6) and are stored in a subcritical array within the fuel storage racks. Further, the TMI-2 Technical Specifications require that during Modes 1 and 2, the water in SFP "A" will be borated between 4350-6000 ppm. Therefore, subcriticality is ensured under all credible conditions. A very small amount of fuel is accumulating at the bottom of SFP "A". This material has been transported from the RV to SFP "A" as debris adherent to the outside of the fuel bearing canisters.

Calculations based on gross gamma measurements performed in SFP "A" estimate approximately 4.9 kg of residual fuel external to the defueling canisters (References 5.5 and 5.6). The remaining canister transfer activities are not expected to significantly increase this fuel quantity.

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

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

5.1.3 Summary

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

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

1. The total quantity of fuel in the AFHB (i.e., less than 17 kg of finely divided, small particle-size sediment material with minor amounts of fuel found as adherent films on surfaces, see Table 5-2), exclusive of the fuel in the canisters in SFP "A," is essentially a small fraction of the SFML which assumes optimum moderation and infinite water reflection (worst-case) conditions. The fuel in the canisters has been demonstrated to be critically safe under all conditions (Reference 5.7). Thus, subcriticality is ensured.

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

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

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

5.2 Reactor Building

During the accident, core debris was transported to the RB as a result of the core degradation event and coolant flow from the RV through the PORV and RCS MU&P System. Table 2-1 reported that approximately 10 kg of fuel (i.e., UO_2) were transported to the RB during the accident sequence. Subsequent to the accident, fuel was relocated to the RB as a result of several cleanup operations including: transfer to and storage of structural RV components in the "A" CFT and "A" D-ring; storage of upper endfittings; flushing of defueling tools; and transfer of the defueling canisters into the FTC. Even though fuel was relocated to the RB during cleanup operations, RB residual fuel conditions were maintained significantly below the SFML. Further, a significant cleanup effort was undertaken (as described in Section 4.2) with the primary purpose of reducing exposure rates but which also resulted in the removal of additional core debris.

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

5.2.1 Reactor Vessel Head Assembly (Reference 5.8)

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

In November 1982, three LSs were removed for analysis. Fuel analyses were performed on two of the samples by Battelle Columbus Laboratories; Science Applications, Inc.; International Corporation; and B&W. Also, a sample of an LST was analyzed for radionuclide activity on both internal and external surfaces.

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

- Determining the Ce-144 activity on LS surfaces by gamma spectroscopy and the fuel activity on the LS surfaces by direct assay;

- Adjusting the activity distribution as evidenced by the internal/external contamination ratio on the LST sample;

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

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

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

Summing the component fuel values produced the total fuel estimate for the RV head assembly. The estimate of fuel in the RV head assembly is approximately 1.3 kg, primarily in the form of surface films.

5.2.2 Reactor Vessel Upper Plenum Assembly (Reference 5.9)

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

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

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

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

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

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

5.2.3 Fuel Transfer Canal

A small amount of fuel may reside at the bottom of the FTC, having been transported from the RV to the FTC as debris adherent to the outside of the fuel bearing canisters. Based on gross gamma measurements performed in the FTC, it is estimated that there are approximately 12.7 kg of residual fuel external to the defueling canisters (Reference 5.5). The few remaining canister transfer activities are not expected to significantly increase this fuel quantity.

5.2.4 Core Flood System (References 5.10 through 5.14)

The core flood system consists of two tanks and piping into the RV (see Figure 2.4). Directional gamma measurements were made in the accessible parts of the core flood system. These measurements were related to fuel deposits, within the field of view of the instrument, using typical radiochemical results (Reference 5.15). The inaccessible parts of the "A" and "B" trains were quantified by assuming the deposition for components and per unit length of pipe were the same as for the measured parts (Reference 5.16).

The "B" CFT contains no measurable residual fuel deposits and a negligible amount (<10 grams) of surface films. The portion of the "A" core flood line between the CFT and the check valve contains less than 0.5 kg of residual fuel. The similar segment of the "B" core flood line contains less than 0.6 kg of residual fuel. The residual fuel estimate in the "A" and "B" core flood lines from the check valve to the RV is reported in Section 5.3.6.

During LCSA defueling, the top of the "A" CFT was removed and the tank was used for storage of LCSA components. Additionally, the piping from the "A" CFT to the RV was cut and flanged which will prevent the possibility of fuel transport. Storage of the LCSA components outside but in proximity to the RV (e.g., in the "A" CFT) was deemed necessary to permit continuous progress in the RV defueling activities. Prior to removal from the RV, the LCSA segments were flushed and brushed to remove fuel. The segments were then video inspected to ensure that no visible fuel was present. Sample sections of each plate were measured by gamma spectroscopy and/or alpha measurements to determine the quantity of residual fuel. Extrapolation of fuel content in other sections was determined based on the fuel quantity of the measured sections. For example, two of the four quadrants of the lower grid distributor plate were measured for fuel content and determined to contain a total residual fuel quantity of 163 grams. These measurements were extrapolated for the other two quadrants and an MDL value of less than or equal to 320 grams of residual fuel was assigned to the lower grid distributor plate (Reference 5.11).

Based on the above approach, the "A" CFT, which contains the LCSA components, has been assigned a total of approximately 3.3 kg (References 5.10 through 5.14, and 5.17) of residual fuel, distributed as follows:

Components	Fuel (kg)
Lower Grid Rib Support Posts	<1.1*
Lower Grid Rib Section	<0.1
Lower Grid Distributor Plate	<0.3*
Lower Grid Forging	1.7
Incore Guide Support Plate	<0.1*
TOTAL	3.3

* = MDL value

Flow distributor sections that do not contain IIGTs are also stored in the "A" CFT; however, the total amount of residual fuel is insignificant. There are no post-defueling plans to remove any of the LCSA components stored in the "A" CFT due to the relatively small quantity of residual fuel involved.

Therefore, the total residual fuel in the core flood system is 4.4 kg.

5.2.5 "A" D-ring (Reference 5.17)

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

Gamma spectrometry performed on 13 of the 14 segments placed in the D-rings, containing a total of 30 IIGTs, determined that these segments contain 22.2 kg of residual fuel. The remaining segment which was not measured for residual fuel, contains three IIGTs.

Based on an average of the amount of fuel per IIGT of the measured segments (i.e., 22.2 kg per 30 IIGTs), it is reasonably estimated that the unmeasured segment contains approximately 2.1 kg of residual fuel (i.e., 0.7 kg per IIGT multiplied by 3 IIGTs). This estimate is conservative because the unmeasured segment was in the northwest quadrant of the flow distributor whereas the measured segments which contained the largest quantities of residual fuel were generally located in the southeast quadrant of the flow distributor. Thus, the total estimated amount of residual fuel in the "A" D-ring is 24.3 kg. Further assessment of the LCSA components in the "A" D-ring is provided in Section 6.0.

5.2.6 Upper Endfitting Storage Area

As described in Section 4.4.3.2.1, during RV defueling, loose upper endfittings were removed from the surface of the RV debris bed to allow access for defueling. These endfittings were too large to be inserted into fuel canisters; thus, they were placed in shielded drums filled with borated water (i.e., 4350-6000 ppm) and stored at the 347' elevation in the RB. Storage of these upper endfittings is described in an NRC-approved SER (References 5.18 and 5.19).

Currently, there is a total of approximately 21 upper endfittings stored in a total of six containers in the endfitting storage area. The maximum number of endfittings in a single container is six . A neutron interrogation system was used to measure the amount of residual fuel in the upper endfittings. The analysis resulted in a total estimated amount of residual fuel in all six containers of 7.7 kg (References 5.20 and 5.21).

5.2.7 Reactor Coolant Drain Tank (Reference 4.20)

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

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

5.2.8 Letdown Coolers (Reference 5.22)

The letdown cooler cubicle, located in the RB basement, contains the letdown coolers (MU-C-1A and 1B) and associated piping. This system was designed to cool the reactor coolant before it entered the rest of the MU&P System for processing. Portions of the MU&P

System ran continuously before and during the accident, and have run since the accident, potentially transporting small amounts of core debris throughout the system. Residual fuel in most MU&P components is discussed in Section 5.1.

Fuel in the letdown cooler system was measured with a collimated, shielded sodium iodide gamma spectrometer. Calculations were made using computer codes to model the associated piping, coolers, and detector configurations. The calculated residual fuel content of the letdown coolers system is less than or equal to an MDL value of 3.7 kg.

5.2.9 RB Basement and Sump (Reference 5.22)

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

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

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

Additionally, fuel particles from washdown of defueling tools from the tool decontamination facility (Section 5.2.10) were transported to the RB sump. Reference 5.22 estimates that 0.2 kg of fuel could have been added to the basement inventory from this activity. Thus, the total fuel in the RB basement is estimated to be 1.3 kg.

5.2.10 Tool Decontamination Facility

The tool decontamination facility on the 347' elevation of the RB consists of two 3.6 meter by 3.6 meter (12 foot by 12 foot) enclosures connected by a 2.4 meter by 2.4 meter (8 foot by 8 foot) anteroom. One enclosure is used for high pressure/high temperature flushing of decontamination tools; the other enclosure is used for cutting up these decontaminated tools. Because of the extensive use of this facility, residual fuel deposits remain in the form of surface contamination. Radiation surveys were used to infer the surface activity on the floors of the decontamination facility. In addition, a "hot spot" existed under the grating in the decontamination flush trough. Based on a conservative evaluation of these radiation surveys (Reference 5.23), a total of 0.2 kg of residual fuel remains in the tool decontamination facility.

5.2.11 Miscellaneous Systems and Equipment

In addition to the residual fuel quantities reported in Sections 5.2.1 through 5.2.10, residual fuel is expected to be contained in various systems/equipment located in the RB which were utilized during the defueling effort. Included are the DWCS, the defueling tool racks which contain the various long-handled tools used to defuel the RV, the TRVFS, and the RB drain system.

The DWCS is composed of three parts: the interconnecting hoses, the manifold assembly, and the RV cleanup pumps located in the south end of the RB canal. The total fuel in the DWCS is estimated to be 2.3 kg (Reference 5.24). The aggregate amount of residual fuel on the various long-handled tools and the tool racks on which they are stored is estimated to be 4.8 kg (Reference 5.23). As discussed in Section 5.2.9, defueling tools are generally flushed prior to removal from the RV in order to remove any loose residual fuel. The TRVFS units were commercial diatomaceous earth-filled swimming pool filters that were used to fight organic growth. The total amount of residual fuel in the TRVFS units is estimated to be 4.4 kg (Reference 5.25).

The RB basement boundary was taken to include all space below the 305' elevation with one partial exception. The exception is the RB drain line that was used to transfer defueling tool decontamination wash water to the basement. The discharge path from the tool decontamination facility (Section 5.2.10) located on the 347' elevation of the RV is from the decon sink to the floor drain located within the decon facility. The discharge piping, from the floor drain, passes through the 347' elevation floor, turns nearly horizontal for approximately 3 meters (10 feet) and then is essentially vertical for approximately 16.5 meters (55 feet) to a long horizontal run under the 282' elevation basement floor. More than a dozen basement floor drains empty into the line. Small gamma detectors, strapped to a drain snake, were used to determine the amount of fuel remaining in this inaccessible region. This fuel exists as small pebbles as opposed to the colloidal particles that were transported to the RB sump (Section 5.2.9). The total amount of residual fuel in the RB drain line is estimated to be 5.1 kg (Reference 5.24).

5.2.12 Criticality Assessment

Table 5-3 lists the total quantity of residual fuel in the RB exclusive of the RCS and RV. As indicated, the total fuel mass remaining in the RB is well below the SFML of 140 kg presented in Appendix B. Subcriticality is further enhanced since most of the residual fuel is tightly adherent to RV components or is in isolated areas within the RB. Fuel in this configuration is significantly less reactive than in the optimum conditions assumed in Appendix B (i.e., fuel pellets, optimum moderation with unborated water, and spherical geometry). Additionally, the current configuration prevents any significant debris transport within the RB or to the RCS or RV, thus minimizing any interaction and accumulation potential. The majority of residual fuel in the RB is located in areas which are neutronically decoupled from other fuel bearing locations and, consequently, there is no potential for a criticality event. Thus, subcriticality within the RB is ensured.

5.2.13 Summary

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

The total estimated quantity of fuel in the RB, listed in Table 5-3, is significantly less than the SFML which assumes optimum moderation and infinite water reflector (worst-case) conditions. The residual fuel in the RB is primarily contained in segments of the flow distributor containing IIGTs which are stored in the "A" D-ring. The residual fuel in the remaining areas of the RB consists of finely divided small particle size sediment material with minor amounts of fuel found as adherent films on metal oxide surfaces. Decontamination activities in the RB served to remove residual fuel, especially in the RB basement where the residual fuel quantity was reduced by approximately 75% (see Tables 2-1 and 5-3). Post-defueling activities (e.g., flushing tanks/pipes, system draindowns, and the special sampling program) may result in the removal of additional small quantities of fuel. Thus, the quantity of residual fuel in the RB may be further reduced.

Based on the above analysis of the total estimated quantity of residual fuel, there is no potential for transport of fuel within the RB which could result in a critical mass. Thus, subcriticality is ensured. In addition, the condition of this residual fuel prevents transport of a significant quantity of material to either the RCS or the RV, thus minimizing any interaction and accumulation potential. Thus, GPU Nuclear has determined that no further efforts for the specific purpose of removing fuel from the RB are appropriate or necessary to preclude criticality or otherwise demonstrate that defueling has been completed to the extent reasonably achievable.

5.3 Reactor Coolant System

As described in Section 2.0, during the accident, fuel was transported to the RCS as a result of the core degradation event and operation of the RCPs. Section 2.2 reported that approximately 400 kg of fuel was transported throughout the RCS during the accident. Section 4.3 describes the defueling operations performed on these RCS components.

The following sections provide the current estimate of residual fuel in the RCS excluding the RB, which is provided in Section 5.2, and the RV, which is provided in Section 5.4. These estimates are based on fuel measurements and/or analyses of RCS components. The basis for each approach is provided within each section.

5.3.1 Pressurizer

Following the completion of pressurizer defueling operations in June 1988 (see Section 4.3.3.1), a small amount of core debris, consisting of small particles, remained in the pressurizer. A video examination of the debris at the bottom of the pressurizer was used to determine the volume of core debris.

A 100 gram sample was removed from the pressurizer in March 1988. Neutron interrogation and gamma spectrometry were used to analyze the sample. The neutron counts were compared to a natural uranium standard and the gamma counts were compared to standard Ce-144 and Eu-154 sources. From these comparisons, the uranium content of the sample was calculated. Scaling from the sample to the total quantity of residual debris in the pressurizer yielded the total fuel in the pressurizer. From this analysis, it has been calculated that 0.3 kg of fuel remains in the pressurizer (Reference 5.26).

Spectrometry measurements were made under the lower head of the pressurizer adjacent to the surge line elbow. The effect of fuel from the pressurizer was reduced by use of a 5-cm (2-inch) thick lead block shield. Results of the analysis indicate that no more than 0.2 kg of fuel remains in the pressurizer surge line (Reference 5.27).

As described in Section 4.3.3.2, debris in the pressurizer spray line was flushed back into the pressurizer and was subsequently removed during defueling operations. Therefore, it was assumed that no measurable quantity of residual fuel remained in the pressurizer spray line.

5.3.2 Decay Heat Drop Line (Reference 5.28)

A video inspection and gross gamma measurement of the decay heat drop line was performed after defueling of the decay heat drop line was completed in January 1989. This video inspection and gamma probing data indicated that the radiation levels measured in the horizontal portion of the decay heat drop line corresponded to small amounts of debris on the bottom internal surface of the line.

A sample of the decay heat drop line debris was analyzed by gamma spectrometry to determine the radionuclide distribution. The line was then modeled with a shielding code using the sample information as the source. By matching the model to the measured gamma exposures, it is calculated that 1.5 kg of fuel remains in the decay heat drop line.

5.3.3 Once-Through Steam Generators

5.3.3.1 Tubesheets/Upper Heads (References 4.22 and 5.29)

The estimate of fuel remaining on the "B" upper tubesheet was generated based on copper foil activation measurements performed in January 1989 (Reference 4.22). Four copper foils were placed inside the "B" OTSG above the tubesheet. They were activated by exposure to the fuel in this environment and were measured with a coincidence counting system. In addition, foils were positioned to measure background at the counting station and inside the "A" OTSG upper head. The "A" OTSG and background foils were activated to the same level, indicating an undetectable quantity of fuel on the "A" tubesheet using this method. Using the background and "A"tubesheet measurements as calibration data, the "B" tubesheet was calculated to contain 36.3 kg of residual fuel.

The "A" OTSG upper tubesheet had less than one liter of debris on it following the accident. Following defueling, the quantity of fuel on the tubesheet was so low as to be undetectable via copper foil activation coupons. The estimate of the residual fuel quantity in the "A" OTSG upper tubesheet is based on gamma radiation measurement data. The "A" OTSG upper tubesheet was calculated to contain 1.4 kg of residual fuel (Reference 5.29).

5.3.3.2 Tube Bundles (Reference 5.30)

Fuel in the OTSG tube bundles was measured using a gross gamma probing technique. Preliminary shielding code work showed that the gamma detectors proposed for these measurements could detect a tube plugged with fuel to a radius of 20 cm (8 inches). By probing a grid of 52 tubes, the whole OTSG tube bundle could be measured. The data were collected at 30-cm (1-foot) increments down the length of the 52 chosen tubes in each OTSG.

Analysis of the probing data indicates that there are no significant radiation sources within the tube region that are attributable to large fuel blockages. High radiation fields within the upper 1.8 meters (6 feet) of the "B" tube bundle are attributed to the upper tubesheet debris. Additionally, high dose rates were also associated with the water/air interface approximately halfway down the tube, possibly corresponding to a "bathtub ring" of boron and crud. Dose rates for all other areas were relatively uniform within the calculated deviation.

Based on the modeled steam generator core debris and corresponding dose rates, the residual fuel in the "A" and "B" OTSG tube bundles were calculated to be 1.7 and 9.1 kg, respectively.

5.3.3.3 Lower Heads/J-Legs (Reference 5.31)

The "A" and "B" OTSG lower heads and J-legs fuel measurements were performed using GM probe fuel measurement strings containing copper coupons which were inserted through the generator tube bundle to the lower head and associated J-legs. Miniature lights and videoprobes were also inserted through surrounding tubes and used to verify placement of the strings and location of debris. Fuel estimates based on in-situ exposure rates for the "A" and "B" OTSG lower heads and J-legs are 1.0 and 6.3 kg, respectively.

The copper coupons were removed from the strings and transferred to the DOE for subsequent independent fuel measurement assessment. The DOE estimates of residual fuel quantities in the "A" and "B" OTSG lower head and J-legs are 5.2 and 5.4 kg, respectively (Reference 5.32). For purposes of the DCR, the GPU Nuclear fuel estimates are reported. The GPU Nuclear fuel estimates are believed to be more representatives of the residual fuel in the "A" and "B" lower heads and J-legs based on the location of the GM counters and the sensitivity of the fuel measurements (i.e., Reference 5.32 states that the DOE fuel measurements have an uncertainty factor of two).

5.3.4 Hot Legs

The boundaries of the "A" and "B" hot legs extend from the RV to the respective steam generators. The residual fuel estimates are based on an integration of video observations, sample collection and analysis, and direct gamma measurements of deposits. The video observations provide a basis for establishing the sizes of the surfaces of deposition. Gamma measurements determined the depth and mass of debris when normalized to radiochemical analysis of typical debris, and of the specific material taken from the places of interest.

The amount of residual fuel in the "A" and "B" hot legs is estimated to be 0.8 and 1.7 kg, respectively (Reference 5.33).

5.3.5 Cold Legs/Reactor Coolant Pumps

The cold legs extend from the OTSGs through the RCPs to the RV. The residual fuel estimates for that portion of the cold legs from the OTSGs to the RCPs (i.e., the J-legs) is reported in Section

5.3.3.3. The residual fuel estimates for the cold legs and RCPs are based on an integration of video observations, sample collection and analysis, and direct gamma measurements of deposits.

The total amount of residual fuel in the four RCPs is 14.7 kg. The amount of residual fuel in the 1A, 2A, 1B, and 2B cold legs is estimated to be 5.5, 28.6, 12.5, and 8.8 kg, respectively (References 5.23 and 5.33).

5.3.6 Core Flood Lines

The boundary of the "A" and "B" core flood lines reported here is from the RV to the first check valve. The residual fuel estimate for the core flood system external to the first check valve is reported in Section 5.2.4. The residual fuel estimates are based on an integration of video observations, sample collection and analysis, and direct gamma measurements of deposits. The amount of residual fuel in the "A" and "B" core flood lines is estimated to be 0.7 and 1.2 kg, respectively (Reference 5.33).

5.3.7 Criticality Assessment

Table 5-4 lists the total quantity of residual fuel in the RCS exclusive of the RB and RV. The total quantity of fuel in the RCS is below the SFML of 140 kg. Subcriticality is further enhanced since most of the residual fuel is tightly adherent to RCS components. Fuel in this configuration is significantly less reactive than in the optimum conditions assumed in Appendix B (i.e., fuel pellets, optimum moderation with unborated water, and spherical geometry). Additionally, the current configuration prevents any significant debris transport within the RCS or to the RV or RB, thus minimizing any interaction and accumulation potential. Thus, subcriticality within the RCS is ensured.

5.3.8 Summary

Section 2.2 indicates that approximately 400 kg of fuel was deposited in RCS components. Subsequently, a significant amount was removed during extensive defueling operations performed in the RCS as described in Section 4.3. RCS defueling operations were performed in the pressurizer, the pressurizer spray line, the "A" and "B" OTSG upper tubesheet, the RCS hot legs, and the decay heat drop line. As a result of these defueling operations, the residual fuel in RCS components has been reduced significantly and does not pose a criticality concern. The largest measured quantity of residual fuel in the RCS is in the "B" OTSG upper tubesheet (36 kg). A variety of defueling techniques have been used on the tubesheets (e.g., pick-and-place, vacuuming, scraping). It has been determined that the remaining fuel in the "B" OTSG upper tubesheet exists as tightly adherent material not readily removable by available dynamic defueling techniques. Further assessment of the residual fuel in the "B" OTSG upper tubesheets is provided in Section 6.0.

The total estimated quantity of fuel (133 kg) in those portions of the RCS, listed in Table 5-4, is less than the SFML. There is no potential for transport of fuel within the RCS which could result in a critical mass. Thus, subcriticality is ensured. GPU Nuclear has concluded that no further efforts to remove fuel from these portions of the RCS are appropriate or necessary to preclude criticality or otherwise demonstrate that defueling has been completed to the extent reasonably achievable.

5.4 Reactor Vessel

Approximately 133,000 kg of core debris, including approximately 94,000 kg UO_2, remained in the RV after the accident; Figure 2-3 depicts the end-state configuration of the TMI-2 core prior to the start of defueling. Section 4.4 describes the extensive defueling operations performed in the RV to remove core debris resulting from the accident. The determination of residual quantities of core debris and its location in the RV has been an ongoing process since the start of defueling. This section will describe end-state conditions of the RV after defueling.

During the course of the cleanup program, the status of the RV and core debris inventory was continuously updated. Technicques included video inspections and sample acquisition analyses.

As a final confirmation of residual fuel quantities, the entire RV was re-examined using visual techniques in conjunction with final vessel cleanup. The final video examination confirmed the physical configurations of the remaining core debris. The potential for fuel transport and interaction with the RCS is described in Section 5.5.

The following sections provide the estimate of residual fuel in the RV. Although a variety of measurement techniques have been used extensively throughout the facility to determine residual fuel quantities, the radiation-based measurements have their own inherent uncertainties which are adversely impacted by the presence of borated water. Because the RV was filled with heavily borated water, video examination, in conjunction with fuel sampling and analyses, was selected as the preferred technique to determine residual fuel quantities in the RV to minimize overall uncertainties.

5.4.1 Composition of Residual Core Debris Deposits

There are three types of residual core debris deposits remaining in the RV:

- Loose, fine, granular debris
- Surface films
- Resolidified material

Samples of each type of material have been analyzed. The following sections summarize the available data utilized to characterize the residual material in the RV.

5.4.1.1 Sample Data

The types and locations of samples taken to date are summarized in the following table.

Type of Material Samples	Location	No. of Samples
Fine and granular debris	RV debris bed	11
Fine and granular debris	OTSG tubesheet	12 particles

Type of Material Samples	Location	No. of Samples
Surface deposit	LS pieces	3
Surface deposit	LCSA structure	10
Porous and non-porous rocks	Bottom head	16 particles
Resolidified mass	Core region R6 area	1

5.4.1.2 Residual Core Debris Deposits

5.4.1.2.1 Loose Debris (Type 1 Debris)

There are several areas in the RV internal structure that contain loose core debris that is inaccessible to defueling. There are two possible origins of this material: material generated as a result of the dynamics of the accident and material generated as a result of the defueling operations. The fact that these areas are inaccessible to defueling also makes them inaccessible for sampling; hence, no samples have been acquired. Samples of loose core debris have been retrieved from various areas in the RCS and the RV. Analytical results of the samples were used to determine the characteristics and properties of the material. The following are the data generated from the various loose core debris sample analyses.

Reference Sample	Fuel (%)	Density (g/cc)
OTSG-B core debris (Ref. 5.15)	53.5	4.2
RV debris bed samples (Ref. 5.34)	83.4	5.0

As a result of the accident, some fuel fragmentation occurred which generated loose debris. During the defueling program, additional loose debris was generated through the use of the CBM and the crust impact tool. It is believed that the loose debris generated during the defueling program exceeded the amount generated during the accident. Based on this belief, a mixture of 60/40 was picked to determine the characteristics of the loose debris. Therefore, it is assumed that the density and percent fuel of 40% of the loose core debris in the RV is represented by the "B" OTSG core debris sample and that the remaining 60% is represented by the debris bed samples. Based on this assumption, the average density of the loose debris in the RV is calculated to be 4.7 g/cc, and it contains an average of 72% UO_2.

5.4.1.2.2 Surface Films (Type 2 Debris)

The surface film deposits vary from area to area. Therefore, actual surface film density sample analysis results have been used to estimate fuel quantities in regions that are similar to the areas in which the samples were taken. The following table summarizes the application of the sample data to the respective regions.

Surfaces	Surface Film Fuel Deposit ($\mu g/cm^2$ UO_2)	Basis
• Horizontal		
Areas above the core region (CSS vertical surfaces)	427	Assumed 30% threaded, 70% unthreaded LS. This is conservative because a large fraction of the threaded portions are horizontal surfaces; the average surface deposit was used (Reference 5.8).
LCSA, bottom head region, and core former plates	1102	Average of the analysis results for the top surface of the LGDP samples (References 5.35 and 5.36).
• Vertical		
Baffle plates and lower grid remnants, vessel wall, core barrel, and thermal shield	27.3	Average of LGRS sample analysis results (Reference 5.10).

5.4.1.2.3 Resolidified Material (Type 3 Debris)

Two types of resolidified material have been observed in the rock samples taken from the bottom head region: material with a small amount of open surface porosity and material with little or no open surface porosity. Samples have been analyzed and the results indicate that the surface porosity is not indicative of the actual density of the material. Based on analysis of the samples (Reference 5.37), the average density of the resolidified material is 7.1 g/cc and contains an average of 73% UO_2.

5.4.2 Residual Core Debris

This section describes how video inspection techniques were used in conjunction with the data described in Section 5.4.1 to determine the quantity of residual core debris in the RV. A computer model of the RV and its internal components was developed and continuously revised to track defueling progress and to serve as an engineering tool to support defueling operations. This "ground state" image defined the dimensional characteristics of the remaining RV internals and, more importantly, the spaces where core debris could reside. Then, based on observation, the actual core debris was spatially represented in the augmented computer model.

Computer software was developed which used information contained in the geometry model to calculate the volume of modelled debris. Other visual factors such as surface texture, shape, apparent hardness or porosity, friability, color, and location were used to categorize the residual core debris into one of the three types discussed in Section 5.4.1.2. The average density for the type of material was used to convert from volume to mass and from core debris to fuel (UO_2) mass. A discussion of the uncertainties associated with this measurement technique is included in Section 3.6. Reference 5.38 provides summary information of the residual core debris calculations. It also provided a reference for the final video inspection of the RV.

In the following discussion, residual debris quantities are related to one of the three types of debris; a value for the amount of fuel (i.e., UO_2) is given in parentheses. The data for the residual fuel in the RV are summarized on Table 5-5. The various components in the RV are illustrated on Figure 5-7.

5.4.2.1 Work Platform Region and Suspended Equipment

The components that will remain in the vessel for long-term storage include the following (Figures 5-8 and 5-9):

- Westinghouse vacuum system pump module and support structure
- In-vessel Filtration/Vacuum System
- Canister Positioning System
- DWCS suction and discharge piping

These systems have pumps, piping, hoses, and structural support beams associated with them. As part of the final cleanup operation, the outer surfaces of the components were vacuumed or flushed. During the final video inspection, only the outside surfaces of these components could be inspected. The vertical surfaces were clean, and deposits of fine granular debris (Type 1) were observed to be on horizontal surfaces in several locations.

Three of the five canister positions in the CPS were occupied during the final cleaning and video inspection. Although the canisters will eventually be removed, it is assumed that loose core debris (Type 1) is still inside the sleeves. Based on the estimated volume, the total estimated residual core debris in the work platform region and suspended equipment is 43.0 kg (30.9 kg UO_2).

5.4.2.2 Downcomer Region (see Figures 5-7, 5-10, 5-11, and 5-12)

The access to the downcomer region is much more limited than to other regions, both for television cameras and for flushing and vacuuming. As a result, some core debris remains in this region. Many of the identified quantities of residual core debris were extrapolated based on the portion of the region accessible for viewing.

5.4.2.2.1 Cold Leg Flow Deflectors

Each of the four cold leg flow deflectors has a horizontal top surface area of approximately 1100 cm^2 (1.2 ft^2). Visual examinations indicate that there is a 0.8-cm (0.3-inch) layer (approximately) of fine granular core debris (Type 1) on each of the horizontal surfaces. Based on the estimated volume, a total of 16.5 kg of residual core debris (11.9 kg UO_2) remains on the four flow deflectors.

5.4.2.2.2 Hot Leg Bosses in the CSS

Each of the two hot leg bosses has a horizontal surface on the top of the outside diameter and a small pocket about 15 cm (6 inches) deep between the boss and the hot leg nozzle on the vessel. The surface area is approximately 2600 cm^2 (2.8 ft^2) for each boss. Visual examinations indicate that there is a fine dusting of material on the "A" hot leg boss top surface and a few pebbles in the pocket. For the "B" hot leg, there is a 0.6-cm (1/4-inch) layer (approximately) of fine granular core debris (Type 1) on the top surface and the pocket is filled to a depth of about 8 cm (3 inches). Based on the estimated volume, a total of 37.0 kg of residual core debris (26.6 kg UO_2) remains on the hot leg bosses.

5.4.2.2.3 Outer Surface of CSS; Surface Deposits on RV Cylindrical Shell; Thermal Shield Outer Surface

A visual inspection of this area was performed and no residual core debris was observed. Only the surface coatings accounted for in Section 5.4.3 are present on these vertical surfaces.

5.4.2.2.4 Surveillance Specimen Capsule Holders

Based on visual examination, there is a 1-cm (0.4-inch) layer (approximately) of loose debris (Type 1) remaining on the horizontal surfaces of the SSCHs. Based on the estimated volume, a total of 4.9 kg of residual core debris (3.5 kg UO_2) remains on these surfaces.

5.4.2.2.5 Thermal Shield Support Blocks (Top Surface)

Each of these support blocks has an upper horizontal surface area of approximately 350 cm^2 (0.4 ft^2). Visual examination indicates that a 0.6-cm (1/4-inch) layer (approximately) of loose debris remains on these surfaces. Based on the estimated volume, a total of 21.2 kg of residual core debris (15.2 kg UO_2) remains on these surfaces.

5.4.2.2.6 Thermal Shield Inner Surface and Annular Gap (see Figure 5-13)

An annular gap is formed between the core barrel and the thermal shield. The spacing between the thermal shield support blocks is approximately the same as the space covered by the blocks. However, the radial width of the annular gap is smaller than the thickness of the blocks. Visual examination performed at three widely separated locations indicate an accumulation of loose debris (Type 1) approximately 0.2 meter (9 inches) deep at the bottom of the gap. This depth of material is assumed to extend the entire circumference. Based on the estimated volume, a total of 164.9 kg of residual core debris (118.6 kg UO_2) remains in this gap.

5.4.2.2.7 Drain Holes at Bottom of Thermal Shield

The drain holes at the bottom of the thermal shield are inaccessible for inspection; therefore, it was conservatively assumed that they are filled with resolidified (Type 3) core debris, for a total of 0.3 kg (0.2 kg UO_2).

5.4.2.2.8 Core Catchers/Seismic Restraint Blocks

Five of the 12 core catchers were seen in the final video inspection. A layer of loose, fine debris (Type 1) of an average of 1-cm (0.4-inch) deep was seen on the top horizontal surface, and a 1.3-cm (0.5-inch) layer (approximately) was seen where the core catcher and the CSA meet. It is assumed that the remainder of the core catchers had similar deposits. Because the seismic restraint blocks are beneath an overhang in the lower grid shell forging, it is assumed that there is no loose debris on the top surface. Based on the estimate volume, a total of 4.0 kg of core debris (2.9 kg UO_2) remains on these surfaces.

5.4.2.3 Internals Indexing Fixture Region (see Figure 5-7)

5.4.2.3.1 RV Flange, IIF Flange, and CSS Flange

This region was flushed and vacuumed as part of the final vessel cleanup. The debris visible during the final video inspection included several small deposits of loose sand-like debris in crevices. The crevices included the space between the alignment keys on the IIF and the RV, and the point of attachment of the IIF bottom flange. There was also a small deposit of loose debris (Type 1) on top of two of the three CSA lifting lugs. In addition, there were small pockets of material on the CSS upper flange in hard-to-vacuum locations, such as behind the lifting lugs, and behind hoses which pass through holes in the flange. Based on the estimated volume, a total of 6.8 kg of residual core debris (4.9 kg UO_2) remains on these surfaces.

5.4.2.3.2 IIF Inside Surface

The IIF was installed after the accident to allow the water level to be maintained at an elevation higher than the RV flange. It was not exposed to any of the accident conditions. It was flushed and vacuumed as part of the final vessel cleanup. Visual examinations confirmed that core debris which might have been deposited on this vertical surface as a result of defueling operations has been removed. It was concluded that no measurable core debris remains on these surfaces.

5.4.2.4 Core Support Shield Region (see Figures 5-11 and 5-12)

5.4.2.4.1 Vent Valve Seats (Inner Surfaces)

The vent valves were removed in 1987 to gain access to the cold leg piping. The valve bosses which are built into the CSS remain. They have a limited amount of horizontal surface. Loose debris deposited on these surfaces was flushed and vacuumed during final vessel cleanup. However, there is a groove around the inside surface which is not accessible and in which loose debris (Type 1) was observed during the final video inspection. Based on the estimated volume, a total of 12.2 kg of residual core debris (8.7 kg UO_2) remains in this area.

5.4.2.4.2 Hot Leg Openings

A layer of (Type 1) silt was observed in each hot leg opening during the final video inspection. The "B" hot leg was revacuumed and re-inspected at the end of the final vessel video examination. The "B" hot leg had no debris remaining. For the inside surface of the "A" hot leg boss in the CSS, it is estimated that a total of 0.3 kg of loose core debris (0.2 kg UO_2) remains in this area.

5.4.2.4.3 LOCA Bosses

There are 13 LOCA bosses around each hot leg opening. Each is a cylinder 7.5 cm (3 inches) in diameter by 10 cm (4 inches) long. During the final video inspection, five bosses were observed to have a 0.5-cm (0.2-inch) layer (approximately) of fine debris (Type 1) on the top surface and three had a fine dusting of material on the top surface. The other 18 LOCA bosses were clear of material. Based on the estimated volume, a total of 1.1 kg of core debris (0.8 kg UO_2) remains on these surfaces.

5.4.2.4.4 Inner Surface of CSS

This is a vertical surface and the only residual core debris on this surface is a surface coating. Surface coatings are discussed in Section 5.4.3.

5.4.2.4.5 Top of Lower CSS Flange

This is a location where material dispersed by the airlift settled out. This material was flushed and vacuumed as part of the final vessel cleanup, leaving only a few pieces of rods and small rocks of resolidified material (Type 3) which could not be removed because of the tight clearance between the bolt circle and the CSS wall. Based on the estimated volume , a total of 1.4 kg of residual core debris (1.0 kg UO_2) remains in this area.

5.4.2.5 Upper Core Support Assembly Region (see Figures 5-11 and 5-14)

5.4.2.5.1 Baffle Plate Inside Surface

In the area of the R-6 fuel element on the southeast side of the core, there is a large hole melted through several baffle plates (see Figure 5-15). There are several small melt holes further south, near the P-4 fuel element area. Immediately adjacent to the melt holes, the baffle plate inside surface (the surface facing into the core region) has a visible crust of fused-on core debris which is estimated to be up to 2.5-cm (1-inch) thick in selected locations. Some of the material was dislodged by the rotary brush tool and removed in sheets. In the places where the cavijet was used on this material, it made visible score lines but did not dislodge it. It is assumed that a 0.2-meter (9-inch) wide band of core debris (Type 3) ranging from 0.3- to 0.6-cm (1/8- to 1/4-inch) thick remains around the perimeter of the hole. Based on the estimated volume, a total of 23.3 kg of resolidified core debris (17.0 kg UO_2) remains in this area.

5.4.2.5.2 Baffle Plate Outside Surface

As discussed above, material was fused to the edge of the melt hole on the inside surface of the baffle plate. Thus, it was assumed that such non-removable material also would be present around the hole on the outside surface, and perhaps at other locations where a blockage of water flow had occurred. Therefore, it is estimated that an amount of material equivalent to that observed to be on the inside surface [i.e., 23.3 kg of core debris (17.0 kg UO_2)], remains in this area.

5.4.2.5.3 Baffle Plate Flow Holes and Bolt Holes

In previous video inspections, resolidified material has been observed in dozens of the baffle plate flow holes. Brushing of both sides of the baffle plates with the rotary brush tool, as well as handling during plate removal and cavijet operations has removed most of this material. All of the baffle plate bolt holes were occupied by bolts at the time of the accident. In order to unbolt these bolts, residual core debris material in the area was removed by brushing and flushing. Any bolt which could not be removed in this way was drilled out. Accordingly, there is no residual core debris adherent in these holes.

After the baffle plates were hung in position, additional defueling using the airlift was performed. Fine debris was resuspended in the water and settled into the bottoms of the flow and bolt holes. The material in any one hole is negligible, but there are hundreds of holes. During the final video inspection, each hole was examined and the total amount of loose core debris in the flow holes, jackscrew holes, and bolt holes amounts to 14.6 kg (10.5 kg UO_2).

5.4.2.5.4 Former Plates Top and Bottom Surfaces

Most of the material in this area was identified as Types 1 or 3. The majority of the granular material (Type 1) was easily vacuumed. The remaining material is estimated to be 10.1 kg of core debris (7.4 kg UO_2). In addition, there was a thin layer of resolidified debris (Type 3) on half of the horizontal top surface of the former plates that did not come off when the material resting on the plate was removed. This debris coating varied in thickness from a few mils to as much as 0.6-cm (1/4-inch). It is estimated that this very thin layer of material along with a few discrete deposits of resolidified fuel, amounts to 44.7 kg (32.5 kg UO_2). Therefore, the total amount of material in this area is 54.8 kg of core debris (39.9 kg UO_2).

5.4.2.5.5 Former Plates Edge Holes

All of the former plate edge holes were occupied by bolts at the time of the accident. These bolts were removed during defueling operations. Accordingly, it was expected that there would be no residual core debris remaining in these holes. This has been confirmed by results of the final video inspection.

5.4.2.5.6 Core Barrel Inner Surface

Results of video inspections indicate that there is no residual core debris on the inner surface of the core barrel.

5.4.2.5.7 Orifice Holes to Thermal Shield Gap (see Figure 5-13)

During the final video inspection of this region, it was observed that ten of the thirty 1.9-cm (3/4-inch) orifice holes were filled with resolidified debris (Type 3). Each hole is approximately 6.4 cm (2.5 inches) deep. Therefore, based on the estimated volume, it is estimated that 1.3 kg of residual core debris (0.9 kg UO_2) remains.

5.4.2.6 Lower Core Support Assembly Region (see Figures 5-11 and 5-16)

5.4.2.6.1 LGRS Top Surface and Peripheral Flow Holes (see Figures 5-17 and 5-18)

The major portion of the LGRS was removed to provide access to lower elevations within the RV. Consequently, the remaining upper surfaces consist only of the bypass region peripheral section, the outer row of grid pads, and one partial grid cell at the R-6 fuel element location.

The final video inspection of the LGRS was performed while the baffle plates were removed for defueling. Most of the resolidified material was removed. Only material for which tools were not effective or which was not accessible with available tools was not removed. This amounted to 54.1 kg of resolidified debris (Type 3) and 2.5 kg of loose material in the form of a dust layer (Type 1) for a total of 56.6 kg of residual core debris (41.3 kg UO_2).

5.4.2.6.2 Between LGRS and LGDP

This area was cleaned using the cavijet. The final video inspection revealed that only two small masses, amounting to 1.7 kg of resolidified core debris (Type 3) remained in this area. The operation of the airlift near the end of the defueling sequence and the discharge from the vacuum system deposited a thin layer of fine debris (Type 1) uniformly on the surface of the LGDP. Also seen were several pieces of fuel rods that were assumed to be filled with fuel pellets. These remaining fuel rods were either wedged in place and could not be removed or in an inaccessible location. The total additional material, primarily fine debris, is 15.9 kg for a total of 17.6 kg of core debris (12.8 kg UO_2) remaining in this area.

5.4.2.6.3 LGDP Peripheral Flow Holes (see Figures 5-19 and 5-20)

In the final video inspection of this area, all but two holes were inspected and seen to be clear. The two which were not inspected were inaccessible due to interference by the hanging baffle plates where the camera would have had to be placed. Those two holes are conservatively assumed to be filled with resolidified debris (Type 3) for a total of 1.0 kg of residual core debris (0.7 kg UO_2).

5.4.2.6.4 Between LGDP and Forging

This area was cleaned using the cavijet. During the final video inspection, it was seen that three masses of core debris (Type 3) remain in the southeast quadrant. Based on the estimated volume, a total of 32.4 kg of residual core debris remains. The operation of the airlift near the end of the defueling sequence and the discharge from the vacuum system uniformly deposited a thin layer of fine debris (Type 1) on the surface of the forging. Also seen were several pieces of fuel rods that were assumed to be filled with fuel pellets. These remaining fuel rods were either wedged in place and could not be removed or in an inaccessible location. The total additional material, primarily fine debris, is 33.6 kg for a total of 66.0 kg of core debris (48.2 kg UO_2) remaining in this area.

5.4.2.6.5 Forging Peripheral Flow Holes (see Figures 5-21 and 5-22)

A flushing tool was inserted through a number of the holes in the forging to flush both the holes and the space below the holes. Additional cavijet and flush operations were performed. During the final video inspection, an attempt was made to insert a camera into each of the remaining holes in the forging. This effort was not entirely successful in that six of the 16 small diameter holes at the periphery were blocked from view. These holes were conservatively assumed to be filled with resolidified debris (Type 3). Of the remaining large diameter flow holes, 21 were observed to contain varying amounts of resolidified debris (Type 3). Based on the estimated volume, there is 153.1 kg of residual core debris (110.1 kg UO_2) remaining in these holes.

5.4.2.6.6 Inside Support Post Stubs (see Figure 5-23)

The inside support posts were cut off 5 to 10 cm (2 to 4 inches) above the top of the forging. It was relatively easy to vacuum loose debris (Type 1) out of the stubs. During the final video inspection, a number of small rocks (Type 3) were seen to be inside some of the stubs. Based on the estimated volume, a total of 1.9 kg of core debris (1.4 kg UO_2) remains in this area.

5.4.2.6.7 Between Forging and IGSP

The space between the forging and the IGSP is 1.2-cm (1/2-inch) wide where the forging is the thickest. Because of the taper on the forging, the space increases to 20 cm (8 inches) at the periphery. This location was cleaned with a flushing tool inserted through a number of the holes in the forging and operated to flush both the hole and the space below the hole. Based on the final video inspection, it is estimated that there is 200.6 kg of core debris (Type 3) in this region. The bulk of the remaining core debris is a single solidified mass in the southeast quadrant of this region. This mass is inaccessible to defueling because of the pattern of cutting of the forging. In any event, the configuration and condition of this mass is such that criticality is not a concern.

The operation of the airlift near the end of the defueling sequence and the discharge from the vacuum system deposited a thin layer of fine debris (Type 1) uniformly on the surface of the IGSP. Also seen were several pieces of fuel rods that were assumed to be filled with fuel pellets. These remaining fuel rods were either wedged in place and could not be removed or in an inaccessible location. The total additional material, primarily fine debris, is 41.8 kg for a total of 242.4 kg of core debris (174.3 kg UO_2) remaining in this area. This estimate also includes the flow holes in the IGSP.

5.4.2.6.8 Incore Guide Support Plate Flow Holes (see Figures 5-24 and 5-25)

The amount of core debris in the IGSP flow holes is included in the estimate for the space between the forging and the IGSP (Section 5.4.2.6.7).

5.4.2.6.9 Between IGSP and Flow Distributor

This area was inaccessible. The incore instrument guide tubes blocked the free rotation of the cavijet tool in most places. Based on the final video inspection, it is estimated that there is 12.0 kg of resolidified core debris (Type 3) remaining in this area. The operation of the airlift near the end of the defueling sequence and the discharge from the vacuum system deposited a thin layer of fine debris (Type 1) uniformly on the surface of the flow distributor. Also seen were several pieces of fuel rods that were assumed to be filled with fuel pellets. These remaining fuel rods were either wedged in place and could not be removed or in an inaccessible location. The total additional material, primarily fine debris, is 43.3 kg for a total of 55.3 kg of core debris (39.7 kg UO_2) remaining in this area.

5.4.2.6.10 Flow Distributor Flow Holes (see Figures 5-26 and 5-27)

The final video inspection confirmed that no measurable core debris remains in this area.

5.4.2.7 Bottom Head Region (see Figures 5-11, 5-28, and 5-29)

5.4.2.7.1 Head Surface

During the course of the defueling operations, the large masses of resolidified core material in the bottom head were broken up successfully with a variety of tools. For the most part, use of these tools was limited to the accessible region formed by the large hole that was cut into the lower internals. Portions of the bottom head beyond that region (under the overhanging remnant of the flow distributor) were not reached by the crust impact tools.

In addition, as the CSS, UCSA, and LCSA regions were defueled, much debris fell into the bottom head. In general, the resolidified debris was easy to break up and the remaining loose material was then airlifted or vacuumed into canisters. Four defueling attempts were made to flush and vacuum the remaining loose material from the bottom head. The residual material was micron-size particles, which circulated through the vacuum system and redistributed back to the bottom head as fine dust material.

The bottom head region was examined in the final video inspection. There appears to be a fine dusting of material (Type 1) over the entire bottom head surface except for patches where the granular material was somewhat deeper [an average of 0.2-cm (0.1-inch)]. Based on the estimated volume, a total of 145.6 kg of core debris (104.6 kg UO_2) remains in this area.

5.4.2.7.2 Incore Instrument Nozzles

Nineteen incore instrument nozzles have partial guide tubes standing above them which could have prevented material from falling down into the nozzles if the instrument string was still intact (including the spiral seal). Guide tubes around the remaining 33 nozzles have been removed and the instrument strings severed. A number of the nozzles were melted off, some to within 2.5 cm (1 inch) of the head surface. Because of limited access, it is not practical to perform a detailed inspection of this area. As a conservative upper limit, it is assumed that the 33 incore nozzles have the annular space between the instrument string and the inside of the nozzle/guide pipe filled with loose debris (Type 1) to a length of 2.4 meters (8 feet). Based on the estimated volume, a total of 40.8 kg of loose core debris (29.4 kg UO_2) remains in this area.

5.4.2.7.3 Standing Incore Guide Tubes

The standing incore guide tubes were examined in the final video inspection. A miniature video probe was inserted from below into the bottom end of each guide tube. Only the lower skirt could be inspected in this manner. For the guide tubes which were clear of debris at the bottom, it is assumed that there is no debris inside the tube for its entire length. This was the case for 17 of 19 tubes which remain in the vessel.

For incore guide tube #42 (O-5 core grid location), there was a small lump of resolidified debris (Type 3) bridging from the tip of the nozzle to the inside of the guide tube, approximately 4-cm (1.5-inches) in diameter. The remainder of the skirt area looked clear, so it is assumed that this material is a remnant of the debris which was on the bottom head.

For incore guide tube #45 (R-7 core grid location), a large amount of resolidified debris was seen inside the skirt. About half of the 0.5-cm (0.2-inch) thick skirt had been melted away and the debris filled the visible cross-section above the melted skirt. Therefore, it is assumed that the entire length of the central hole in the incore guide tube is filed with resolidified debris (Type 3). Based on the estimated volume, a total of 24.4 kg of core debris (17.6 kg UO_2) remains in this area.

5.4.3 Surface Film Deposits

In addition to the concentrations of residual core debris, there is some fuel bound to the surfaces of the components in the RV. These deposits were not distinguishable by video inspection because they are thin and relatively uniform over a wide area. This film accounts for a very small fraction of the total fuel left in the vessel, but is included in this report for completeness.

The likely fuel concentration in surface film coatings was estimated based on the available surface sample information, as discussed in Section 5.4.1.2.2 of this report. The surface area was calculated for each of the components remaining in the RV (Reference 5.39). The appropriate surface concentration of fuel was then applied to the calculated surface area, and the results are tabulated on Table 5-6.

5.5 Reactor Vessel Residual Fuel Criticality Assessment

5.5.1 Criticality Safety Analysis

This section provides a criticality safety analysis and assessment of the core debris remaining in the RV. Because the amount of fuel remaining in the RV is larger than the SFML, a separate criticality safety analysis was performed to demonstrate subcriticality. This analysis used in-vessel inspections of debris locations and quantities, as well as some conservative estimates of remaining fuel, to develop a specific three-dimensional analytical model of the RV end-state configuration. The criticality calculations presented in this section were performed by ORNL.

The analysis discussed in this section was developed for actual plant conditions and does not address postulated accident configurations (e.g., resulting from a seismic event). The criticality safety assessment for accident configurations is provided in Section 5.5.2 of this document.

Section 5.5.1.1 describes the assumptions and bases used to develop the geometrical model, including postulated core debris locations, used in this criticality safety assessment. Section 5.5.1.2 provides a detailed discussion of the analytical treatment of the core debris. Section 5.5.1.3 provides the results of the specific criticality safety analyses performed for this assessment and Section 5.5.1.4 presents the conclusions.

5.5.1.1 Geometrical Modelling

The analytical model used for this criticality assessment was developed prior to the completion of in-vessel defueling efforts. Consequently, conservative estimates had to be made regarding the quantity and locations of core debris that would remain in the RV following the completion of in-vessel defueling activities. These estimates were made based on in-vessel inspection data, debris removal experience, and the proposed defueling plans at the time of model development.

The three major areas where core debris was assumed to be left are the RV bottom head, the LCSA, and in the core former area (i.e., between the core former baffle plates and the core barrel) in the UCSA. Fuel accumulations in other locations within the vessel were considered to be too small (much less than the SFML) and/or separated from these three areas by a far enough distance [i.e., the equivalent of approximately 30 cm (12 inches) of water, Reference 5.40] so as not to cause a reactivity increase due to neutronic interaction between these areas. Additionally, the conservative debris quantities that were used in the model will more than compensate for the limited quantity of debris in those areas not specifically modelled. Details of the modelling of each of the regions are provided in the following sections.

Prior to developing the final analytical model, a number of preliminary analyses were performed to determine the most reactive regions of the RV model. In these analyses, various core debris configurations and RV modelling approaches were evaluated. The conclusion that was reached from these analyses was that the major influence on the neutron multiplication was the debris left in the LCSA region of the RV. Though the other regions did add reactivity, their effects were small in comparison to the LCSA region. Within the LCSA, the most reactive section was the open gap region below the grid forging and above the IGSP. Due to the limited inspection access of this region, and because of its overall reactivity importance, this region was assumed to contain more core debris than will actually remain there following in-vessel defueling activities. This assumption is discussed further in Section 5.5.1.1.3.

5.5.1.1.1 General Model Descriptions

At the time of the initial development of the geometrical model used in this criticality assessment, the main areas of the RV where core debris remained were the RV bottom head, the peripheral regions of the LCSA, the core former area, and the CSS. Additionally, a relatively small quantity of debris was located within the core region at fuel assembly location R-6. This debris configuration is represented in Figure 5-30. Further details regarding final quantities and locations of residual core debris are provided in Section 5.4 of this document.

The geometrical model, including the postulated core debris locations and quantities, that was used in the performance of this criticality safety assessment is shown in Figure 5-31. The figure shows an annular ring representing the RV internals and postulated debris accumulations located along the outer periphery of the RV. This annular ring was conservatively assumed to go 360° around the vessel. Due to the initial success at debris removal in R-6, it was assumed that essentially no debris would remain at the R-6 location. Additionally, due to the large separation distance [i.e., greater than 30 cm (12 inches)] between the CSS and the most reactive region of this model (the LCSA), the core debris that will remain in the CSS was not explicitly modelled. Thus, the only regions to be modelled in detail in the analytical model of the RV were the lower head, the LCSA, and the core former area. All additional regions of the RV, excluding the modelled debris accumulations and vessel internals, were assumed to be filled with unborated water.

5.5.1.1.2 Reactor Vessel Bottom Head

As shown in Figure 5-30, the bottom head contained a hard layer of debris in the central region and large quantities of loose debris along the peripheral areas. Using a conservative approach, a 1.2-cm (1/2-inch) layer of core debris covering the entire inside surface of the bottom head was assumed to exist in the model for this criticality assessment; this amount is greater than what was reported in Section 5.4.2.7. The presence of interferences in the bottom head region (e.g., the incore instrument nozzles) were conservatively neglected.

5.5.1.1.3 Lower Core Support Assembly

A large portion of the core debris resides in the peripheral regions of the LCSA. Additionally, preliminary analyses showed that the LCSA was the most reactive region of the analytical model developed for this criticality safety assessment. Consequently, the major focus and detail of the geometrical model occurred in the modelling of this region.

As a starting point and to simplify the model, each of the LCSA plates, except the LGRS, was assumed to be of the same radial thickness [i.e., 50 cm (20 inches), ΔR of Figure 5-31]. This radial thickness essentially corresponded to the distance at which the underside of the grid forging levels out (see Figures 5-16 and 5-32). For those plates which may have a radial thickness in excess of 50 cm (20 inches) at some locations along the plate inner circumference (e.g., the forging or the IGSP), it was assumed and subsequently verified that no significant debris accumulations exist in these locations. Figures 5-33 through 5-37 provide a view of the fuel remaining on each of the LCSA plates based on video inspections in September 1989. Subsequent defueling activities have removed some of the debris shown in these figures. These figures show the conservative approach associated with assuming a 50-cm (20-inch) radial thickness for the LCSA plates, as most areas of the plates are relatively clear and have little, if any, fuel that extends out near the 50-cm (20-inch) location. As it was assumed that no core debris would remain on those portions of the plates in excess of 50 cm (20 inches), these portions were not explicitly modelled. Rather, these portions of the plates were assumed to be incorporated into the effectively infinite water reflector applied at the inner radius of the annular ring of the model. For those locations where the LCSA plates have been cut such that the radial thicknesses are less than 50 cm (20 inches), the approach utilized conservatively overestimates the debris quantities and locations. This approach is conservative as analyses demonstrated that the neutron multiplication increased with increasing plate radial thickness due to the increase in the quantity of modelled core debris (Reference 5.41).

The LGRS was modelled as an annular ring with a 30-cm (12-inch) radial thickness because the outer grid pad cells form a confined region in which the debris has collected; that area is significantly less than 50 cm (20 inches) (see Figure 5-33).

An additional conservative approach included in the model was that all of the remaining holes in each of the LCSA plates were assumed to be filled with fuel and unborated water in an optimal mixture, even though, as a result of defueling operations, many of the holes were free of debris, as shown in Figure 5-33 through 5-37. Based on design drawings and available defueling records, the cross-sectional area of the remaining holes in each of the plates was determined. The percentages of hole area (filled with fuel/water) and steel for each of the plates are shown in Figure 5-31. These percentages were assumed to be representative of the entire plate and no attempt was made to account for localized areas of a plate in which the hole area might exceed the percentage determined for that particular plate. Using the percentages determined for each of the plates, the fuel/water mixture within the holes and the steel quantity were combined to get homogenized cross-section sets to represent each plate. The geometrical dimensions of the LCSA plates, including separation distances, were taken from applicable drawings and defueling records.

Though significant portions of the LCSA plates were free of debris, as seen in Figures 5-33 through 5-37, a debris layer was applied to each of the modelled LCSA plates to account for any debris that could not be removed from the plate surfaces. The actual layer dimensions for each plate were determined based on available defueling data. These debris layers were conservatively assumed to extend the entire 360° around the RV.

The thickness of the debris layer on top of the LGRS was assumed to be 5 cm (2 inches). This debris layer was modelled accounting for the presence of a 5-cm (2-inch) high "lip" on this plate (see Figure 5-32). It is assumed that the "lip" functions as a barrier to retain core debris in a region where it would not be readily accessible for removal.

Between the LGRS and the LGDP, two debris accumulations were modelled. The first was a 7.5-cm (3-inch) radial thickness accumulation located on the outer periphery of this region which extended axially the entire distance between the two plates. This layer of debris conservatively represented the potential accumulation of core debris in an area that was not readily accessible to defueling equipment. The second debris layer modelled in this region was a 0.6-cm (1/4-inch) thickness placed on top of the LGDP and the lower grid forging was similarly modelled with these two debris layers. Both of these regions are shown in Figure 5-31.

Due to the small separation distance between the lower grid forging and the IGSP [i.e., 1.2-cm (1/2-inch)], there has been only limited access of the area underneath the lower grid forging. However, to support the conservative approach, the amount of core debris modelled in this region was assumed to be much larger than that assumed for other areas of the LCSA. Based on the inspections that were completed, and considering the locations of debris accumulations on top of the lower grid forging as well as those underneath the IGSP, the core debris was seen to be concentrated in the outermost 25 cm (10 inches) of the region. Consequently, core debris was assumed to completely fill these 25 cm (10 inches) over the entire distance separating the lower grid forging from the IGSP. The remaining radial 20 cm (8 inches) of this region were assumed to be relatively free of significant debris accumulations and were modelled as unborated water only.

Underneath the IGSP, a 11.5-cm (4.5-inch) high layer of debris with a 20-cm (8-inch) radial thickness was modelled. This region represented the "knuckle" located at the outer edge of the flow distributor (see Figure 5-32) that was essentially inaccessible to defueling equipment and, thus, could potentially contain some accumulation of fuel debris.

The actual vertical offsetting of the plates (see Figure 5-30) was conservatively neglected. Instead, the LCSA plates were assumed to have a constant outer radius corresponding to the lower grid forging. Similarly, the core former and the bottom head regions were conservatively assumed to connect directly to the LCSA. No consideration was given for the offsetting of these regions. This approach placed the various regions closer to each other than they actually are, which would imply better neutron interaction between the modelled debris accumulations.

A 20-cm (8-inch) carbon steel region was placed on the outside of the annular ring to represent the RV wall. This approach essentially moved the water region between the core barrel and the vessel wall to outside the vessel. This was considered appropriate and conservative as only small accumulations of debris fines were expected to be found in this region and because analyses (Reference 5.42) have shown steel to be a better neutron reflector than unborated water. Finally, an unborated water reflector of effectively infinite thickness was placed outside the carbon steel region.

5.5.1.1.4 Core Former Area

To conservatively model remaining debris in the core former area, it was assumed that a 0.6-cm (1/4-inch) thick, 3-meter (10-foot) high layer remained attached to the core barrel. This conservative representation of the core former region also bounds the limited amount of core debris that remains in the regions above/outside the UCSA (e.g., RV downcomer, hot and cold leg penetrations, RV flange, CSS).

5.5.1.1.5 Neutronic Coupling of In- and Ex-Vessel Debris

The potential for neutronic coupling of the core debris that will remain in the RV and that debris remaining in the ex-vessel locations was considered in the development of the geometrical model for this criticality safety assessment. However, large separation distances exist between these other fuel locations (e.g., pressurizer, RB basement, D-rings) and the RV. Additionally, as shown in Figure 5-38, significant quantities of structural materials (e.g., equipment, concrete walls) present between the various core debris locations, which tend to decrease further any reactivity worth associated with coupling. Consequently, it was concluded that the effect of neutronic coupling of in- and ex-vessel debris accumulations would be negligible and, thus, no changes to the geometric model would be required to account for this effect. Further justification for this conclusion is presented in Section 5.5.1.3.3.

5.5.1.1.6 Conservative Approach Summary

As discussed in the above section, significant conservative estimates were built into the geometrical model for this analysis. This conservative approach is summarized below:

- Conservative values for layers of core debris were applied to the LCSA plates to represent core debris accumulations that may not be able to be removed from those surfaces.

- The entire inside surface of the RV bottom head was assumed to be covered with a 1.2-cm (1/2-inch) thick layer of core debris.

- A 0.6-cm (1/4-inch) thick layer of core debris, with a height of 3 meters (10 feet), was assumed to be attached to the core barrel in the core former region of the model.

- No credit was taken for the vertical offsetting of the LCSA plates or the other regions of the model, and the regions between the plates were conservatively modelled.

- Each of the modelled LCSA plates was modelled with a radial thickness that bounded the presence of core debris on the plate.

- The debris and vessel internals were assumed to extend 360° around the periphery of the RV.

- The holes in each of the modelled LCSA plates were assumed to be filled with core debris and unborated water in an optimal mixture.

- Unborated water was assumed to fill all portions of the RV, excluding the modelled debris accumulations and vessel internals.

- No credit was taken for the plans to eventually drain the RCS, essentially leaving the RV without a moderating medium.

- Considerably more core debris was included in the analytical model than will remain in the RV following defueling activities (see Table 5-7).

Considering the above approach, along with the conditions of the RV following defueling, it was concluded that the geometrical model described above, including the postulated core debris locations, was conservative and appropriately bound the RV configuration that exists.

5.5.1.2 Fuel Modelling

5.5.1.2.1 Enrichment

The original loading of the core included 56 assemblies of 1.98 wt% (batch 1), 61 assemblies of 2.64 wt% (batch 2), and 60 assemblies of 2.96 wt% (batch 3) U-235 enrichment. The loading pattern is shown in Figure 5-39. The enrichment of the fuel used in this evaluation was that corresponding to the homogeneous mixture of the three fuel batches. All 177 fuel assemblies were utilized to develop this homogeneous mixture of fuel and the resulting unburned enrichment of the homogeneous fuel was 2.54 wt%.

Defueling records indicated that approximately 65% of the batch 3 fuel was removed from the vessel as intact full- or partial-length fuel assemblies without any significant mixing with other fuel batches. Additionally, it is recognized that a significant portion of the batch 3 fuel that did mix with the other core debris was located in the upper core region and was removed during early defueling activities. Furthermore, the accident and defueling activities that have resulted in the relocation of the fuel debris to the areas of concern in this evaluation (i.e., LCSA, bottom head, and core former) have enhanced mixing of the debris within the vessel. The foregoing conditions indicate that the assumed homogeneity of the core debris was appropriate.

5.5.1.2.2 Fuel Burnup Worth

The TMI-2 fuel had experienced the equivalent of approximately 94 effective full-power days of burnup at the time of the accident (Reference 5.43). The net effect of burnup is to make the UO_2 fuel less reactive the longer it is burned in the reactor. Fuel burnup results in the depletion of U-235, a buildup of fission product poisons in the fuel material, and also a small buildup of fissile plutonium. The combination of these effects resulted in a net decrease in reactivity for the TMI-2 fuel. Because the degree of fuel burnup was well established at the time of the accident from plant data, some credit could be taken for its resultant negative effect on fuel reactivity. The conservative methods used to establish the amount of burnup credit taken in this analysis are described below.

Poisoning effects of the fission products were accounted for only if the fission products were identified as remaining with the fuel. The gaseous fission products were assumed to be released at the time of the accident and the soluble ones were assumed to have leached out of the fuel matrix.

Of the remaining non-soluble fission products, some become volatile under extremely high fuel temperatures and the formation of a zircaloy-fuel eutectic and, thus, were assumed not to remain with the fuel matrix. Of the non-soluble fission products, only the rare earths were considered to be stable under the TMI-2 accident conditions. Several of the rare earth elements act as significant neutron poisons in the fuel. To verify the presence of the rare earths in the fuel matrix, a literature search of fuel melt experiments was performed to support the RCS criticality analysis reported in Reference 5.43. The discussions presented in Appendix B of the Reference 5.43 evaluation regarding the presence of rare earth elements in the TMI-2 fuel were considered applicable for the current evaluation.

In the calculation of the fuel burnup effects, the average exposures for each of the three fuel batches were derived from existing plant data. These exposures and the core operating history were applied in the ORIGEN-S model in the SCALE system (Reference 5.44) to calculate the isotopic inventory at the time of the accident. Further details of the burnup analysis performed for the batch 3 fuel are presented in Reference 5.45. A similar burnup analysis was performed for batches 1 and 2 fuel. The incorporation of the burnup effects for the three batches produced a net U-235 enrichment of 2.24 wt%, plus associated plutonium buildup, for the homogeneous fuel mixture (Reference 5.46).

A review of the available enrichment sample data for TMI-2 demonstrated the appropriateness of the 2.24 wt% enrichment as well as indicating that mixing of the various fuel batches has indeed occurred within the RV. For example, the most comprehensive enrichment sample data evaluation performed at TMI-2, where 34 samples were taken from various locations within the bottom head (Reference 5.39), showed a weighted average U-235 enrichment of 2.23 wt%. Appendix B provides further details regarding these data. Additionally, the eight samples recently collected from fuel assesmbly location R-6 showed a weighted average U-235 enrichment of 2.5 wt% (Reference 5.47). Recognizing that R-6 was a batch 3 (2.96 wt%) fuel assembly, it was anticipated that the average enrichment of these samples would be larger that 2.24 wt%. Further consideration of these data indicated that mixing of the lower enrichment fuel batches 1 and 2 with the batch 3 fuel at R-6 did occur. This was evidenced by the comparison of the 2.5 wt% measured enrichment to the calculated enrichment of 2.67 wt% for batch 3 fuel after consideration of burnup effects. The lower measured value indicated that mixing occurred during the accident or during some of the early defueling operations. As these two sets of enrichment sample data were taken from within the vessel at or near locations in which core debris will remain, these data were considered to be representative of the core debris that was modelled in this evaluation.

The resultant fuel composition used in the analyses for this evaluation is given in Table 5-8. This table shows a U-235 enrichment of 2.24 wt% which, as discussed above, assumed a homogeneous mixture of the three fuel batches and includes burnup credit for each of the fuel batches. The table also indicates the buildup of plutonium and fission products which are a result of burnup. This was the same fuel composition that was used to develop the SFML of 140 kg defined in Appendix B of this document.

5.5.1.2.3 Lattice Structure

As with previous criticality safety analyses performed by ORNL (References 5.43, 5.48, and 5.49), the fuel was represented as a homogeneous medium for which the neutronic data corresponded to a dodecahedral lattice of spherically shaped fuel pellets as depicted in Figure 5-40.

Conservatively, it was assumed that there was nothing present in the fissile media but fuel pellets and unborated water. Thus, the negative reactivity effects due to the presence of cladding, fixed absorbers and structural materials were ignored. Another maximum reactivity assumption was the preservation of the pellet surface to mass ratio in the fuel pellet volume. This assumption enhanced the resonance shielding effect on the U-238 cross-sections.

Unborated water was used for the moderating medium. This assumption conservatively neglected that the RCS was required to be borated to approximately 5000 ppm during defueling activities and that the vessel will be drained following the completion of defueling operations. Additionally, the presence of residual boron which would remain in the RV following the draining process has been ignored. Furthermore, an optimum fuel volume fraction (i.e, resulting in a maximum k_∞) for fuel and unborated water (VF = 0.28) was determined and used in this evaluation.

5.5.1.2.4 Fuel Particle Size

The optimal fuel particle size for UO_2 particles moderated with unborated water was shown in previous analyses (Reference 5.49) to be greater than a standard-size fuel pellet. However, any core debris particles larger than standard-size pellets found in the core debris were considered to contain impurities, as melting and subsequent resolidification was the only credible means in which the larger particles could have been formed. In such a process the other materials within the vessel (e.g., cladding, structural materials, and poisons) would have intermixed with the fuel, thus reducing the reactivity of the debris. Sample results (References 5.15, 5.34, 5.39, 5.50, 5.51, and 5.52) support the conclusion that the debris is unlikely to be UO_2 without impurities, as all the debris samples evaluated to date have contained some amount of impurities. Furthermore, particle sizes less than a standard-size pellet have been shown to be less reactive than full pellets. Defueling experience has indicated that particle sizes much smaller than standard pellets are representative of the remaining core debris. Consequently, in accordance with the conservative method used, the spherical equivalent of standard full-size pellets was used for this evaluation.

5.5.1.2.5 Conservative Approach to the Fuel Model

As discussed in the above sections, a significant conservative approach was included in the development of the analytical fuel model. This conservative approach is summarized below:

- No credit was taken for the structural or solid materials existing in the debris, though sample data has shown the presence of impurities in all samples evaluated to date.

- Unborated water, optimally mixed with the core debris, was assumed for the moderating material in all fuel bearing regions of the model.

- The fuel particle size was assumed to be equivalent of standard full-size pellets.

- Actual fission product retention was considerably greater than that which was assumed in the analysis. For example, cesium was shown to be retained within previously molten corium which experienced temperatures in excess of 2900°K. Sample data also indicate that approximately 30% of the highly volatile fission products remained with the fuel within the RV following the accident. Additionally, a large percentage of the medium- and low-volatile fission products (e.g., cerium and ruthenium) have been shown to have remained within the RV (Reference 2.1).

- The fuel enrichment was represented as TMI-2 average burned fuel (i.e., homogenous mixture of all three fuel batches) which sample data have supported as being appropriate for this evaluation.

Quantification of the reactivity worth of some of these conservative values is presented in Section 5.5.1.3.2.

It was recognized that isolated regions within the vessel may have core debris accumulations for which some of the assumptions may not be bounding (e.g., the debris/stainless steel ratio in the slotted regions of the lower grid forging near fuel assembly location R-6 may exceed the 50%-50% ratio assumed in the analysis). However, in consideration of the significant conservative approach that has been applied, including those related to geometry, fuel, and impurities, it was concluded that the overall model used for this analysis was a conservative representation of the end-state RV configuration. Consequently, the analytical model was considered appropriate for use in this analysis.

5.5.1.3 Results

To evaluate the criticality safety consequences of leaving core debris in the RV following the comparison of in-vessel defueling activities, ORNL performed an analysis using the Monte Carlo computer program KENO V.a (Reference 5.53) and the model described in Sections 5.5.1.1 and 5.5.1.2. The results of this analysis, provided in Reference 5.54, are discussed in Section 5.5.1.3.1. Additional analyses were also performed by ORNL in order to quantify some of the conservative values associated with the more significant assumptions in the reference case analysis. These analyses are discussed in Section 5.5.1.3.2. Absence of neutronic coupling of the in-vessel debris with debris in other locations throughout the plant is addressed in Section 5.5.1.3.3.

5.5.1.3.1 Reference Case Results

The maximum resultant k_{eff} for the model described in Sections 5.5.1.1 and 5.5.1.2, hereafter referred as to the reference case, was 0.983 (Reference 5.54). This value includes a 2.5% Δk uncertainty bias (see Section 5.5.1.3.4). The criterion used to establish the acceptability of this value was that the calculated neutron multiplication, k_{eff}, including computer code uncertainty, did not exceed 0.99. This acceptance criterion was consistent with the previous licensing basis for the RCS during defueling (References 5.43, 5.48, and 5.49). As the calculated k_{eff} was within the acceptance criterion, it was concluded that the core debris remaining in the RV would be subcritical.

5.5.1.3.2 Quantification of the Conservative Method

As has been indicated previously, numerous conservatisms have been included in the reference case model used for this criticality safety assessment. Additional sensitivity analyses were performed to evaluate the reactivity worth of four assumptions. The four assumptions investigated were: 1) the fuel particle size; 2) the presence of impurities within the debris; 3) the fuel volume fraction of the debris/water mixture; and 4) the filling of all the remaining holes in the LCSA plates with an optimum mixture of debris and unborated water.

5.5.1.3.2.1 Effects of Fuel Particle Size

One significant conservative representation that was included in the analysis was that all the core debris was assumed to be the equivalent of standard full-size fuel pellets. As discussed in Section 5.5.1.2.4, particle sizes less than the size of full pellets were considered to be representative of the debris that will remain in the RV. To evaluate the sensitivity of the calculated neutron multiplication to the size of the fuel particle,a series of infinite cell calculations was performed with various postulated particle sizes. For each particle size evaluated, the fuel volume fraction was varied until the point of optimum moderation (i.e., maximum k_∞) was determined. The maximum k_∞ for each particle size is provided as Cases 2 through 5 of Table 5-9. An important note to these results is that the infinite cell calculations were performed with a U-235 enrichment of 2.67 wt%, corresponding to burned batch 3 fuel, and not the 2.24 wt% enrichment used in the analyses for the RV criticality safety assessment. Though the actual magnitudes of neutron multiplication presented in Cases 2 through 5 were higher than that used in the RV criticality assessment (i.e., Case 1 of Table 5-9), the trends seen in the table were considered applicable to the lower enriched fuel. Thus, it was clearly demonstrated that when the core debris particle size decreases, reductions occur in k_∞. Similar trends in the neutron multiplication, k_{eff}, would be experienced if the smaller particle debris were to be incorporated into the infinite system model representing the end-state condition of the RV.

5.5.1.3.2.2 Effects of Impurities

Another conservatism that was included in this analysis was that the core debris was assumed to be UO_2 without any impurities. This assumption was conservative in that all samples of TMI-2 debris accumulations collected to date have contained some impurities (see Table 5-10). Available sample data from within the RV (References 5.34, 5.39, and 5.50), the "B" OSTG tubesheet (Reference 5.15), the purification/makeup filters (Reference 5.51), and the pressurizer (Reference 5.52) all show the presence of impurities (e.g., boron, iron, zirconium, and cadmium). These samples have shown that the impurities, in particular the boron, were an integral part of the debris material and were not just simply surface deposition (Reference 5.47). Thus, these impurities were considered to be long-term constituents of the debris.

To access the reduction in reactivity due to the presence of impurities in the core debris, a series of infinite lattice cell calculations was performed. In these calculations, the average impurities identified in Table 5-11 were assumed to be mixed with optimally sized batch 3 fuel. Mixes 1, 2, and 3 considered burned batch 3 fuel (2.67 wt%), while Mix 4 utilized unburned batch 3 fuel (2.96 wt%). Unborated water was used as a moderating material for all four mixtures. For Mixes 1, 2, and 3, the mixture particle size and fuel volume fraction were varied until a maximum k_∞ value was determined, while for Mix 4, an optimum volume fraction was determined for a particle size that corresponded to standard full-size pellets.

Mixes 1 and 2 in Table 5-11 were developed based on the "B" OTSG sample results. These mixtures only considered the effects of boron, cadmium, iron, and zirconium. Any additional impurities were neglected and their mass was considered to be UO_2. For additional conservative representation and to account for measurement uncertainty, the average calculated impurity concentrations were further reduced by approximately 10%. The actual measured average impurity concentrations are shown in Table 5-10. As seen in that table, the "B" OTSG samples, in general, contained the minimum impurity concentrations of all of the sample data evaluated.

Sample results for core debris taken from the RV bottom head region (Reference 5.39), which are more representative of the debris that will remain in the RV (Reference 5.39), were used to develop Mix 3. The Mix 3 concentrations represent a numerical average of the impurities levels provided in Reference 5.39.

Mix 4 was also based on Reference 5.39 data. However, as boron was the main neutron poison of the impurities present, only it was considered. All other impurities were treated as UO_2. The boron concentration used in Mix 4 was developed based on a mass weighted average of the Reference 5.39 data.

The effects of impurities on the reactivity of the core debris can be seen by a comparison of the optimum k_∞ value for UO_2 without impurities (Case 2 of Table 5-9) and the optimum value considering the impurities (Cases 6 through 12). These differences between the values with and without impurities indicate the significant neutron poisoning effect due to the presence of only a very small quantity of impurities. Comparison of Cases 6 through 12 with Case 1 shows a substantial reduction in k_∞ even when considering the increased enrichments of Cases 6 through 12.

To further evaluate the effect of the presence of impurities, the fuel models developed for Mixes 1 (Case 6) and 2 (Case 7) were used in the finite system geometry developed for this analysis. For both cases, all the core debris shown in Figure 5-31 was assumed to be replaced with the appropriate fuel/impurity mixture. For the Mix 1 debris, the resultant k_{eff} was 0.836, while for Mix 2, k_{eff} was 0.972. Both of these results included the 2.5% Δk computer code uncertainty. If the larger impurity concentrations seen in the in-vessel samples (Mixes 3 and 4) would have been used in these analyses, even greater reductions in k_{eff} would have occurred. Consequently, based on the above results, it was concluded that neglecting the presence of impurities in the modelling of the core debris for the reference case model resulted in a significant safety margin in the analysis. As was noted above, the impurities are considered long-term constituents of the debris and, thus, the results presented above would be considered appropriate for the entire long-term storage conditions at TMI-2.

5.5.1.3.2.3 Effects of Fuel Volume Fraction

The next conservative representation included in the RV criticality safety assessment model was the effect of the fuel volume fraction of the core debris/water mixture. As was noted in Section 5.5.1.2.4, the fuel volume fraction was varied until the point of optimum moderation was found. This approach has no physical basis. It was merely a means to maximize the fuel/unborated water interaction. To quantify the effect of this assumption, a more physical basis was used in the determination of more realistic fuel volume fractions.

The first fuel volume fraction considered was 0.624, which was a measured value for randomly packed whole fuel pellets (Reference 5.55, Section 3, Page 35). The next value considered (0.74) corresponded to the maximum fuel volume fraction for spheres (as the fuel particles were actually modelled) in contact. Other values (0.5, 0.66, and 0.72) were also considered. These values were more realistic representations of the closely packed conditions of the debris accumulations at TMI-2 rather than the optimum fraction of 0.28.

The increased fuel volume fractions were also considered to be more representative of RV configurations in which the debris is not under water (i.e., less moderator present). As such, these cases would provide an indication of the reactivity reduction that would be expected when the RV is drained, as is currently planned, following defueling activities.

A series of infinite cell calculations was performed to assess the effect that an optimum fuel volume fraction had on k_∞. The enrichment of the fuel used for the volume fraction variations was the TMI-2 average, including burnup (i.e., 2.24 wt%). Standard whole fuel pellets were used for the particle size and unborated water was the moderating medium.

The results of the variations in fuel volume fraction are shown as Cases 13 through 17 in Table 5-9. As can be seen from this table, when fuel volume fractions more representative of the debris configurations at TMI-2 were used rather that optimum moderation conditions (Case 1), k_∞ decreases dramatically. Similarly, a dramatic decrease in neutron multiplication, k_{eff}, would occur when these fuel volume fractions were used in the finite system RV model.

Related to this, Reference 5.40 states that criticality is not possible for unmoderated uranium containing less than approximately 5 wt% U-235 further indicating that when the RV is completely drained subcriticality is ensured.

5.5.1.3.2.4 Effects of Filling Lower Core Support Assembly Plate Holes With Fuel

The final conservative representation to be evaluated for this document was the effect of filling all of the remaining holes in the LCSA plates with an optimal mixture of core and unborated water. Though it was anticipated that some debris would remain in these holes, it was not considered credible to assume that all the holes contained core debris. Therefore, to estimate the reactivity worth of the assumption utilized, an additional analysis was performed assuming that only about half of the plate holes contained debris. To accomplish this, the holes in the inner 25 cm (10 inches) of each plate remnant were assumed to be filled with only unborated water. In the remaining holes, along the outer periphery of each plate, an optimum fuel/unborated water mixture was assumed to exist. The k_{eff} for this case was 0.954, including the 2.5% Δk uncertainty bias, corresponding to a 2.9% Δk reduction from the reference case analysis. This decrease in neutron multiplication further demonstrates the significant conservative representation included in the reference case model. Table 5-12 provides a summary of all finite geometry cases performed for this criticality safety assessment.

5.5.1.3.3 Coupling of In- and Ex-Vessel Debris

As indicated previously, no model changes were incorporated into the reference case model to account for the potential neutronic coupling of the core debris within the vessel with the debris located in other plant areas. The basis for this assumption was that the debris within the vessel was well separated from the debris in the other locations. Figure 5-38 shows the location of the RV in relationship to the other plant components, many of which are where the ex-vessel debris will remain. For stable plant conditions, it was concluded that no fuel transport would occur between the in- and ex-vessel locations. Conditions caused by external events (e.g., seismic) were not considered part of this evaluation and are addressed in Section 5.5.2.

To demonstrate that the in- and ex-vessel debris would not be neutronically coupled, the result of a previous analysis is described. The model for this previous conservative analysis was developed to assess the degree of coupling between fuel masses located in the bottom head of an OTSG and the two adjoining "J" legs. The model utilized in the analysis conservatively bounds the separations seen between the end-state configuration of in- and ex-vessel core debris and, thus, the previous analysis provides a bounding assessment of the coupling of the in- and ex-vessel debris.

The model used in the previous analysis is shown in Figure 5-41. The upper sphere contained 120 kg of optimally moderated (with unborated water), 2.24 wt% enriched, standard sized UO_2 fuel pellets. The bottom half of this sphere was surrounded by 20 cm (8 inches) of carbon steel and the top by 20 cm (8 inches) of unborated water. Each of the bottom spheres also contained 120 kg of the same composition of core debris. However, these spheres were totally surrounded by only 7 cm (2.75 inches) of carbon steel. The spheres were placed in a triangular arrangement such that each sphere was in contact with the other two. This arrangement was evaluated by ORNL using KENO V.a and the resultant k_{eff} was 0.948 (Reference 5.42). The bottom two spheres were then removed from the model and the k_{eff} of only the upper sphere was determined to be 0.951. These two results were considered identical within the KENO V.a statistics. Consequently, these results showed that even with additional debris accumulations in relatively close proximity, the presence of the carbon steel effectively decoupled the neutronic interaction. Thus, based on the results, along with the large separation distances that exist between the in- and ex-vessel debris accumulations, and the presence of significant separation materials, it was concluded that the in- and ex-vessel debris would be effectively neutronically decoupled. Consequently, the conclusion that no model changes were required to account for the presence of ex-vessel debris accumulations was considered appropriate for this evaluation.

5.5.1.3.4 Computer Code Benchmarking

In Reference 5.43, an analytical bias of 2.5% Δk, including the KENO V.a statistical uncertainty, was established as an appropriate value for the highly borated systems being investigated in that report to define a safe boron concentration for the TMI-2 defueling program. Uncertainty values reported in the literature for unborated systems have been shown to be smaller than this value (Reference 5.45). Consequently, the 2.5% Δk value was considered conservative for the criticality safety analysis provided in this document. This uncertainty is considered appropriate for debris which contains small quantities of boron or other neutron poisons.

5.5.1.4 Conclusions

Based on the resultant k_{eff} of the reference case analysis performed in support of this criticality safety assessment (i.e., k_{eff} <0.99), it was concluded that the core debris that remains in the RV following the completion of in-vessel defueling operations will be subcritical. Additionally, the sensitivity analysis shows that a significant safety margin was inherently built into the reference case model developed for this evaluation. Furthermore, because the core debris in the vessel is well separated from ex-vessel debris locations, the effect of neutronic coupling between in- and ex-vessel debris accumulations will be negligible during post-defueling plant conditions.

The analyses were performed assuming that the debris was optimally moderated with unborated water; therefore, the above conclusions would be applicable whether or not the RV is drained. The results of the fuel volume fraction sensitivity study indicated that a significant reactivity reduction would be expected with reduced moderation, which would be representative of the draining of the vessel.

Unanticipated, post-defueling accident conditions were not considered in the development of the model for this criticality safety assessment; thus, the above conclusions may not apply for all accident conditions which could reconfigure the core debris. Discussions of core debris configuration resulting from post-defueling accidents and the corresponding safety assessments are presented in Section 5.5.2.

5.5.2 Criticality Event Analysis

Section 5.5.1 analyzed the existing residual core debris quantities, their configuration, and distribution in the RV, and demonstrated that there is no potential for a criticality event. This section evaluates potential relocation of residual core debris and moderator addition events to determine whether a criticality event could actually occur in the RV.

In order to attain criticality, sufficient quantities of core debris and moderator must accumulate in a favorable geometry in the absence of adequate quantities of neutron poisons. If insufficient quantities of either core debris or moderator are present, if they are in a non-critical configuration, or if sufficient quantities of neutron poisons are present, a criticality event cannot occur. The purpose of this section is to qualitatively review these three factors (i.e., core debris quantity and geometry, moderator, and neutron poisons) to show that a criticality event cannot occur in the RV.

The calculations in Appendix B establish 140 kg as the SFML for the purpose of determining the amount of fuel (i.e., UO_2) which could collect in a discrete volume and remain subcritical regardless of the consideration of other parameters. The objective of this section is to qualitatively review those other parameters and to demonstrate that a criticality event could not occur at TMI-2 for any credible conditions of fuel relocation and moderator addition. The approach used in this section is to examine: 1) the occurrence of conditions necessary to establish the minimum quantity and configuration of core debris required to support a criticality event (Section 5.5.2.1); 2) the potential for moderator addition into the RV (Section 5.5.2.2); and 3) the current presence of neutron poisons (Section 5.5.2.3).

5.5.2.1 Core Debris Considerations

A set of minimum conditions is required to support criticality. An adequate quantity of moderated TMI-2 fuel must be present for the potential of criticality to exist. The purpose of this section is to examine the quantity and configuration of the residual core debris after a postulated relocation event and evaluate the possibility of accumulating core debris to support criticality.

5.5.2.1.1 Safe Fuel Mass Limit

An SFML of 140 kg has been established; it is provided herein as Appendix B. As shown in Sections 5.1 through 5.3, there is no individual ex-vessel location that contains more than 140 kg of core debris. In fact, the RV is the only location where core debris could possibly accumulate in a quantity equal to or greater than the SFML of 140 kg. The calculation of the SFML is based on the following conservative assumptions:

- A spherical geometry was assumed to minimize the ratio of surface area to volume, thus maximizing k_{eff}
- The fuel (i.e., UO_2) was represented as TMI-2 average fuel (homogeneous mixture of all three fuel batches)
- The equivalent of full standard-size fuel pellets was used
- A moderating medium, unborated water, was in an optimal mixture with the fuel
- There was an effectively infinite water reflector
- No credit was taken for the large amount of structural and solid poison materials existing in the core debris (besides those created by burnup)
- A computer code bias of 2.5% Δk was assumed

If one or more of these assumptions is made less conservative, then the fuel mass required for criticality increases.

5.5.2.1.2 Residual Fuel Mass Greater than the Safe Fuel Mass Limit

With approximately 900 kg of residual fuel (UO_2) in the RV, it can be postulated that the drying and spalling of surface films, a seismic event, aging and corrosion, or other unidentified events could cause the residual core debris to accumulate in one area resulting in a potentially critical mass. However, as evidenced by the extensive defueling effort, the residual core debris and contained fuel has consistently resisted multiple removal attempts by aggressive mechanical means. Nonetheless, because the total amount of residual fuel in the RV exceeds the SFML, it is necessary to evaluate the significance of a relocation and accumulation of a larger quantity of residual fuel. It should be noted that because of the RV physical configuration, no significant quantity of fuel material can be transported into the RCS. Therefore, the following evaluation bounds any potential fuel relocation.

Relocation of a significant quantity of the residual core debris requires a transport mechanism. Drying and subsequent spalling of the surface film deposits could occur during PDMS;

however, the total quantity of fuel extant as surface films is less than 2.1 kg (Section 5.4.3). Therefore, this mechanism for relocating fuel can be discounted immediately as posing no threat of criticality. The other mechanistic (e.g., seismic, aging and corrosion) and non-mechanistic events that could cause relocation of the residual fuel will be considered together.

The worst-case accumulation of core debris which exceeds the SFML would occur if some of the debris remaining in the RV would relocate to the bottom head. If this occurred, it would be incredible for the debris to collect in a geometry that would resemble a sphere. Nothing is present in the vessel to allow collection in such a small and confined region. Most likely, debris would collect in the bottom head in a pile or layer. The relatively large surface area of such a geometry would significantly enchance neutron leakage and, thus, reduce k_{eff}. This effect was seen in the results of an analysis performed by ORNL assuming more than half of the remaining core debris in the RV (i.e., 500 kg) were to collect in the bottom head. The assumed configuration of the debris is shown in Figure 5-42. Region L_1 contains an optimal mixture of 500 kg of core debris and unborated water. Region L_2 contains approximately 500 gallons of unborated water. The height of L_2 is large enough to consider it an effectively infinite water reflector. This 500 gallons is significantly more water than is expected to accumulate in the RV throughout PDMS (see Section 5.5.2.2). The core debris was assumed to be 2.24 wt% U-235, standard fuel pellets, and contained no impurities. The resultant neutron multiplication, k_{eff}, for this configuration was 0.946 (Reference 5.56), including a 2.5% Δk computer code bias. This value is significantly below the k_{eff} criterion of 0.99. Thus, there is not a criticality safety concern for this configuration.

5.5.2.1.3 Summary

The first barrier to inadvertent criticality is prevention of the accumulation of fuel in a critical configuration. It is extremely unlikely that a significant amount of fuel will relocate and accumulate in one area of the RV. If the fuel does relocate, its geometry will probably be piles or layers of debris, significantly less reactive than the sphereical geometry assumed in the SFML analysis. Therefore, considering the type and degree of conservative values used in the SFML analysis, it is concluded that the geometrical conditions required for criticality are incredible and cannot occur, notwithstanding moderator considerations.

5.5.2.2 Moderator Considerations

The preceding section addressed core debris considerations for a postulated criticality event and assumed optimum moderation with unborated water. The second barrier to reducing the potential for criticality is to minimize the potential for the presence of large quantities of unborated moderator. As discussed above, a more realistic assessment is that residual fuel collecting on the bottom of the RV is unlikely to form a suspended sphere and is more likely to assume a less reactive configuration such as a layer or a pile. Notwithstanding the previous conclusion that the fuel conditions for criticality cannot occur regardless of moderator considerations, this section discusses the potential for moderator addition, including its possible sources, chemical content, and likely impact.

5.5.2.2.1 Water Addition

5.5.2.2.1.1 Expected Sources of Water

The amount of water remaining in the RV after draindown is expected to be less than ten gallons. This water will be borated; the criticality analyses assumed non-borated water.

There are currently no planned activities that use water in the RB during PDMS. Additionally, the potential for water ingress to the RV is further reduced by:

- Single-barrier protection to the RB will exist
- Piping systems within the RB will be drained to below the 313' elevation (i.e., below the elevation of the hot and cold leg entrances to the RV)
- The RV will be covered such that water intrusion is inhibited
- All operations using water in the RB will be administratively controlled

It is concluded, therefore, that introduction of water to the RV is unlikely.

In addition, it is estimated that no more than four gallons of water will accumulate during PDMS from condensation of the RB atmosphere under the passive breather mode conditions (Reference 5.57). The conservative approach applied in this calculation accounts for currently planned active ventilation periods during PDMS. Therefore, less than 14 gallons of water are expected to accumulate in the RV from planned or identified sources during PDMS.

5.5.2.2.1.2 Water Addition From Accident Conditions

The post-defueling accident scenarios examined were floods, fires, and seismic events. In all of the postulated scenarios, the expected value of water addition was determined to be negligible (Reference 5.58).

5.5.2.2.2 Addition of Other Moderators

Other likely moderator materials include oil, grease, plastics, and chemicals. The RV opening will be such that water intrusion will be inhibited. The RV will breathe through a filtered hole in the top of the cover. During normal conditions, this cover will remain in place. It can be postulated that during accident conditions, the cover could fall into the RV. However, the collapse of the cover would add structural material consisting of neutron reflectors, moderators, and poisons. Due to its physical characteristics, the cover material cannot become interstitially dispersed with the fuel. The existence of oil, grease, plastics, and chemicals in other than very small quantities is not likely, especially in close proximity to the RV. The potential storage of these items on the RV cover or in the RV is also considered highly unlikely due to the administrative controls on flammable material and the relative inactivity in the RB during Facility Modes 2, 3, and 4 (i.e., PDMS).

In any event, the resulting configuration (i.e., cover material, oil, grease, plastics, or chemicals in the RV) would be less reactive than the case of unborated water addition.

5.5.2.2.3 Summary

The second barrier to inadvertent criticality is to minimize the potential for unborated moderator addition. Due to isolation barriers and precautionary measures, it is expected that less than 14 gallons of water will accumulate in the RV from system draindown residual and potential condensation. This residual water will be borated to some degree. Further, the accident scenarios that could introduce a significant quantity of water into the RV are highly unlikely. Finally, the addition of moderator material other than water is extremely unlikely. Therefore, considering the realistic fuel configuration following relocation (i.e., a pile on the RV bottom head), there will not be sufficient moderator material present to support a criticality event.

5.5.2.3 Neutron Poison Considerations

The third barrier to inadvertent criticality is the existence of neutron poisons. In the RV during PDMS, neutron poisons will exist in four forms: as impurities in the residual fuel, as boron in any remaining water, as structural material, and as an insoluble poison that will be added to the RV. The Appendix B SFML analysis presents several fuel conditions; one fuel condition uses the earlier defined base case assumptions with two variations:

- Rather than a homogeneous mixture of TMI-2 fuel, a U-235 enrichment of 2.96 wt% is used
- The effects of a debris impurity concentration of 0.072 wt% boron is considered

The first variation from the base case model is actually more conservative (i.e., a U-235 enrichment of 2.96 wt% versus 2.24 wt%) than the 140 kg SFML analysis. This fuel enrichment assumption represents 100% Batch 3 fuel without taking credit for burnup. The second variation considers the presence of boron poisons in a concentration more representative of the samples collected from the TMI-2 RV (see Appendix B, Table 2) than the 140 kg SFML analysis. The base case model calculation for the SFML theoretically required for criticality (i.e., 140 kg) assumes no impurities.

Using the above assumptions, the calculated k_{∞} is less than 0.99, including the 2.5% ΔK computer code uncertainty (see Appendix B). Therefore, the theoretical mass of fuel required for criticality is infinite. That is, based on these assumptions, no quantity of residual fuel can attain criticality. Thus, given ideally conservative assumptions, except for the realistic consideration of interstitially mixed boron at 0.072 wt%, the quantity of core debris required for criticality increases beyond that theoretically possible to accumulate, even on a non-mechanistic basis.

The other impurities (e.g., cadmium, indium, iron, silver, and zirconium) found in the core debris have been conservatively neglected in the SFML calculation, even though their presence has been established and their contribution to maintaining a subcritical configuration is significant. In addition, all criticality analyses assume unborated water as the moderating medium. In fact, the water remaining in the RV will be highly borated which will also contribute to maintaining a subcritical configuration.

The boron impurity concentration used in the above analysis is representative of the residual fuel in the RV. Therefore, without consideration of the known existence of other poisons in the fuel, there is no quantity of fuel that can sustain criticality. Regardless, a stable and insoluble neutron poison material will be added to the bottom head of the RV to provide added margin and absolute assurance that no circumstance will result in a condition causing the residual fuel in the RV to become critical.

5.5.2.4 Conclusion

The theoretical conditions required for a criticality event include sufficient amounts of core debris and moderator in a critical configuration in the absence of sufficient quantities of neutron poisons. Therefore, the three barriers to inadvertent criticality are: 1) the prevention of the accumulation of fuel in a critical configuration; 2) minimizing the presence of unborated moderator; and 3) the continued existence of neutron poisons. Fuel movement in the RV under any circumstances is highly unlikely considering the inability of the extensive dynamic defueling efforts to displace the remaining debris. Assuming fuel relocation, its reconfiguration into the optimum condition (i.e., spherical) is even more unlikely. A more realistic configuration would be a pile or a layer which is significantly less reactive than the optimum conditions assumed in the Appendix B SFML analysis. Finally, the existence of interstitial neutron poisons in the core debris makes it essentially impossible for any postulated reconfiguration of the residual fuel in the RV to attain criticality.

The water remaining in the RV after system draindown is expected to be ten gallons with an additional four gallons potentially accumulating during PDMS via condensation. This residual water will be borated to some degree. The possibility of introducing a significant quantity of unborated water into the RV is considered incredible. There are no known credible accident scenarios that result in a significant water addition to the RV before or during PDMS. However, considering the realistic fuel configuration following relocation (i.e., a pile on the RV bottom head), nuclear criticality is not a concern.

The existence of neutron poisons, interstitially mixed in the residual fuel and inherent in the residual water, forms the final barrier to criticality. In fact, by assuming a small amount of impurities in the fuel debris, criticality is absolutely precluded (i.e., $k_{\infty} < 1.0$). Although not needed to assure reactivity control over the long-term, as an additional conservative measure, a stable and insoluble neutron poison will be added to the bottom head of the RV.

Therefore, it is concluded that a criticality event cannot occur in the RV.

SECTION 5.0

TABLES

TABLE 5-1

AFHB CUBICLES WHICH CONTAIN NO RESIDUAL FUEL

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

TABLE 5-1 (Cont'd)

AFHB CUBICLES WHICH CONTAIN NO RESIDUAL FUEL

DESIGNATION	NAME	EXPLANATION
AX135	Radwaste Disposal Control Panel	No water piping in area
AX201	North Stairwell	No water piping in area
AX202	Elevator Shaft	No water piping in area
AX203	4160 Switchgear 2-1E	No water piping in area
AX204	4160 Switchgear 2-2E	No water piping in area
AX205	RB Purge Air Supply	System design prevents fuel transport
AX206	RB Purge Exhaust - B	System design prevents fuel transport
AX207	RB Purge Exhaust - A	System design prevents fuel transport
AX208	AB Exhaust Unit B	System design prevents fuel transport
AX209	AB Exhaust Unit A	System design prevents fuel transport
AX210	FHB Exhaust Unit B	System design prevents fuel transport
AX211	FHB Exhaust Unit A	System design prevents fuel transport
AX212	Decay Heat Surge Tank	No fuel transport pathway
AX213	Unit Substation	No water piping in area
AX214	Decon Facility	No fuel transport pathway
AX215	FHB Supply Unit	System design prevents fuel transport
AX216	AB Supply Unit	System design prevents fuel transport
AX217	Access Area	No water piping in area
AX219	Instrument Racks	System design prevents fuel transport
AX220	Caustic Mixing Area	All pathways isolated since accident
AX221	Caustic Mixing Area	All pathways isolated since accident
AX222	South Stairwell	No water piping in area
AX223	Air Handling Units	System design prevents fuel transport
AX301	Elevator Shaft	No water piping in area
AX302	North Stairwell	No water piping in area
AX303	Elevator and Stairwell Access	No water piping in area
AX401	Roof	No water piping in area
AX402	Cooling Water Storage Tanks	No fuel transport pathway
AX403	Damper Room	System design prevents fuel transport
FH002	Access Corridor	No water piping in area

TABLE 5-1 (Cont'd)

AFHB CUBICLES WHICH CONTAIN NO RESIDUAL FUEL

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

TABLE 5-2

AFHB CUBICLES WHICH POTENTIALLY CONTAIN RESIDUAL FUEL[1]

FUEL QUANTITY (kg)	DESIGNATION	NAME	REFERENCE
0.03	AX004	Seal Injection Valve Room	Ref. 5.59
<0.003*	AX005	Makeup Pump - 1C	Ref. 5.60
0.07	AX006	Makeup Pump - 1B	Ref. 5.61
0.25	AX007	Makeup Pump - 1A	Ref. 5.62
	AX008	Spent Resin Storage TK-1B	Section 5.1.2.1
	AX009	Spent Resin Storage TK-1A	Section 5.1.2.1
	AX010	Spent Resin Storage Tank Pump	Section 5.1.2.1
	AX014	Reactor Coolant Evaporator	Section 5.1.2.1
	AX015a	Cleanup Filters	Section 5.1.2.1
	AX015b	Cleanup Filters	Section 5.1.2.1
0.8	AX016	Cleanup Demineralizer - 2A	Section 5.1.2.1
	AX017	Cleanup Demineralizer - 2B	Section 5.1.2.1
	AX114	MU&P Demin - 1A	Section 5.1.2.1
	AX115	MU&P Demin - 1B	Section 5.1.2.1
	AX119	Spent Fuel Demineralizer	Section 5.1.2.1
	AX129	Deborating Demineralizer - 1B	Section 5.1.2.1
	AX130	Deborating Demineralizer - 1A	Section 5.1.2.1
	FH001	MU Suction Valves	Section 5.1.2.1
<0.002*	AX011	AB Sump Pump and Valve	Ref. 5.3
<0.20*	AX012	AB Sump Pumps and Tank	Ref. 5.3
<0.01*	AX018	Waste Transfer Pump	Ref. 2.12
<0.005*	AX019	WDL Valves	Ref. 2.12
4	AX020	RCBTs 1B and 1C	Ref. 5.63
0.31	AX021	RCBT 1A	Ref. 5.64

* - Denotes Minimum Detectable Level

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

TABLE 5-2 (Cont'd)

AFHB CUBICLES WHICH POTENTIALLY CONTAIN RESIDUAL FUEL

FUEL QUANTITY (kg)	DESIGNATION	NAME	REFERENCE
0.005	AX024	AB Sump Filters	Ref. 5.65
<0.002*	AX026	Seal Injection Filters	Ref. 5.66
0.20	AX102	RB Sump Pump Filters	Section 5.1.2.2
0.29	AX112	Seal Return Coolers	Ref. 5.67
0.31	AX116	Makeup Tank	Ref. 5.4
0.04	AX117	MU&P Filters	Ref. 2.12
1	AX131	MWHT	Section 5.1.2.3
	AX134	Miscellaneous Waste Tank Pumps	Section 5.1.2.3
0.5	AX124	Concentrated Liquid Waste Pump	Section 5.1.2.3
	AX218	CWSTs	Section 5.1.2.3
0.002	AX501	RB Spray Pump - 1A	Ref. 5.68
0.002	AX502	RB Spray Pump - 1B	Ref. 5.68
0.002	AX503	DHR Cooler & Pump - 1A	Ref. 5.68
0.002	AX504	DHR Cooler & Pump - 1B	Ref. 5.68
<0.01*	FH003a	MU Discharge Valves	Ref. 2.12
<0.06*	FH003b	MU Discharge Valves	Ref. 2.12
1	FH008	Neutralizer Tank Pump	Section 5.1.2.3
	FH009	Neutralizer Tank	Section 5.1.2.3
	FH012	Neutralizer Tank Filters	Section 5.1.2.3
<1	FH014	Annulus	Section 5.1.2.4
	FH112	Annulus	Section 5.1.2.4
	FH205	Annulus	Section 5.1.2.4
0.71	FH101	MU&P Valve Room	Refs. 2.12 and 5.69
1	FH106	SDS Monitor Tanks	Section 5.1.2.5
	FH110	Spent Fuel Pool "B"	
4.9	FH109	Spent Fuel Pool "A"	Refs. 5.5 and 5.6
<17 kg = TOTAL			

TABLE 5-3

RESIDUAL FUEL QUANTIFICATION IN THE REACTOR BUILDING[a]

COMPONENT	RESIDUAL FUEL QUANTITY (kg)
RV Head Assembly	1.3
RV Upper Plenum Assembly	2.1
FTC	12.7
Core Flood System	4.4
"A" D-ring	24.3
Upper Endfitting Storage Area	7.7
RCDT	0.1
Letdown Coolers	<3.7[b]
RB Basement and Sump	1.3
Tool Decontamination Facility	0.2
Miscellaneous Cleanup Systems/Equipment	
DWCS	2.3
Defueling Tool Rack	4.8
TRVFS	4.4
RB Drains	5.1
TOTAL =	<75 kg

(a) - Excluding the RV and RCS

(b) - MDL

TABLE 5-4

RESIDUAL FUEL QUANTIFICATION IN THE RCS[a]

COMPONENT	RESIDUAL FUEL QUANTITY (kg)
Pressurizer (including surge line)	0.5
Decay Heat Drop Line	1.5
"A" Side	
OTSG Upper Tubesheet	1.4
Tube Bundle	1.7
Lower Head and J-Legs	1.0
Hot Leg	0.8
Cold Legs	34.1
Core Flood Line[b]	0.7
"B" Side	
OTSG Upper Tubesheet	36.3
Tube Bundle	9.1
Lower Head and J-Legs	6.3
Hot Leg	1.7
Cold Legs	21.3
Core Flood Line[b]	1.2
RCPs	14.7
TOTAL =	<133 kg

(a) - Excluding the RV

(b) - Between the RV and first check valve

TABLE 5-5

RESIDUAL FUEL QUANTIFICATION IN THE REACTOR VESSEL

LOCATION	DEBRIS TYPE *	CORE DEBRIS (kg)	RESIDUAL FUEL (kg UO2)
Work Platform Region and Suspended Equipment			
Westinghouse Vacuum Pump Module	1	1.3	0.9
In-Vessel Filtration System	1	22.7	16.3
Canister Positioning System	1	19.0	13.7
DWCS Inlet/Outlet Piping	N/A	0	0
Subtotal		43.0	30.9
Downcomer Region			
Cold Leg Flow Deflectors	1	16.5	11.9
Hot Leg Bosses in CSS	1	37.0	26.6
Outer Surface of CSS	N/A	0	0
Surface Deposits on RV Cylindrical Shell	N/A	0	0
Thermal Shield Outer Surface	N/A	0	0
Surveillance Specimen Capsule Holders	1	4.9	3.5
Thermal Shield Support Blocks (Top Surface)	1	21.2	15.2
Thermal Shield Inner Surface and Annular Gap	1	164.9	118.6
Drain Holes at Bottom of Thermal Shield	3	0.3	0.2
Core Catchers/Seismic Restraint Blocks	1	4.0	2.9
Subtotal		248.8	178.9
Internals Indexing Fixture Region			
RV Flange, IIF Flange, and CSS Flange	1	6.8	4.9
Internals Indexing Fixture Inside Surface	N/A	0	0
Subtotal		6.8	4.9

* Debris Type: 1 = Loose/Fine Debris
2 = Surface Film Material
3 = Resolidified Material

**Includes fuel rod pieces assumed full of fuel pellets; the weight per length of rod segment is 1 kg/m and contains 80.3 % UO_2 (Reference 5.70).

TABLE 5-5 (Cont'd)

RESIDUAL FUEL QUANTIFICATION IN THE REACTOR VESSEL

LOCATION	DEBRIS TYPE	CORE DEBRIS (kg)	RESIDUAL FUEL (kg UO2)
CSS Region			
Vent Valve Seats (Inner Surfaces)	1	12.2	8.7
Hot Leg Openings	1	0.3	0.2
LOCA Bosses	1	1.1	0.8
Inner Surface of CSS	N/A	0	0
Top of Lower CSS Flange	1	1.4	1.0
Subtotal		15.0	10.7
UCSA Region			
Baffle Plate Inside Surface	3	23.3	17.0
Baffle Plate Outside Surface	3	23.3	17.0
Baffle Plate Flow Holes and Bolt Holes	1	14.6	10.5
Former Plates Top and Bottom Surfaces	1,3	54.8	39.9
Former Plates Edge Holes	N/A	0	0
Core Barrel Inner Surface	N/A	0	0
Orifice Holes to Thermal Shield Gap	3	1.3	0.9
Subtotal		117.3	85.3
LCSA Region			
LGRS Top Surface and Peripheral Flow Holes	1,3	56.6	41.3
Between LGRS and LGDP	1,3	17.6**	12.8**
LGDP Peripheral Flow Holes	3	1.0	0.7
Between LGDP and Forging	1,3	66.0**	48.2**
Forging Peripheral Flow Holes	3	153.1	110.1
Inside Support Post Stubs	3	1.9	1.4
Between Forging and IGSP (includes IGSP Flow Holes)	1,3	242.4**	174.3**
Between IGSP and Flow Distributor	1,3	55.3**	39.7**
Flow Distributor Flow Holes	N/A	0	0
Subtotal		593.9	428.5
Bottom Head Region			
Head Surface	1	145.6	104.6
Incore Instrument Nozzles	1	40.8	29.4
Standing Incore Guide Tubes	3	24.4	17.6
Subtotal		210.8	151.6
Surface Film Deposits (See Table 5-6)	2	N/A	2.1
			TOTAL = 892.9 kg

TABLE 5-6

SURFACE FILM DEPOSITS

COMPONENT	SURFACE AREA (sq. in.)	RESIDUAL FUEL DEPOSIT (g UO2)
Work Platform and Suspended Equipment	92,475	254.8
Vessel Cylindrical Shell	175,873	31.0
Thermal Shield, Support Blocks, SSCH	174,583	30.7
Internals Indexing Fixture	32,405	89.3
Core Support Shield	114,805	316.3
Baffle Plates	165,115	29.1
Former Plates	25,790	183.4
Core Barrel	152,659	26.9
Lower Grid Rib Section	10,102	71.8
Lower Grid Distributor Plate	6,704	47.7
Lower Grid Forging and Support Post Stubs	25,224	179.3
Incore Guide Support Plate	11,992	85.3
Flow Distributor	21,188	150.6
Lower Grid Shell Forging	21,321	151.6
Standing Incore Instrument Guide Tubes	15,369	109.3
Bottom Head	47,807	339.9
TOTAL	1,093,412 ($705 m^2$)	2,097.0 (2.1 kg)

TABLE 5-7

COMPARISON OF MODEL TO ESTIMATED REMAINING FUEL MASSES

	MASS OF UO_2 (kg)	
LOCATION	ESTIMATED	MODEL
Bottom Head	151.6	670
UCSA	85.3	600
LCSA	428.5	5,500

NOTES:

1. Estimated quantities were taken from Section 5.4 of this document.
2. The other regions of the RV that contain debris were considered to be separated from the areas of interest by large distances [>30 cm (12 inches)] and/or to have smaller-than-SFML quantities.
3. The neutron multiplication of a fuel mass is not only influenced by mass; the configuration of the mass is also an important consideration.

TABLE 5-8

FUEL MODEL COMPOSITION

ISOTOPE	NUMBER DENSITY (atoms/barn-cm)
U-235	5.21 E-04
U-238	2.25 E-02
O-16	4.60 E-02
Pu-239	4.01 E-05
Pu-240	2.00 E-06
Pu-241	2.49 E-07
Sm-149	1.01 E-07
Sm-151	1.79 E-07
Eu-151	8.20 E-09
Eu-153	1.32 E-07
Eu-154	4.51 E-09
Eu-155	6.12 E-09

NOTES: 1. Only isotopes significant to the model are listed above.

2. Values are taken from Reference 5.71 and are for a particle size of 1.0724 cm (2.72 inches), with a fuel volume fraction of 0.28 and a U-235 enrichment of 2.24 wt%.

TABLE 5-9

QUANTIFICATION OF CONSERVATIVE VALUES

CASE	U-235 ENRICHMENT (wt%)	IMPURITIES	VF	PARTICLE DIAMETER (CM)	K∞
1	2.24	No	0.28	1.07	1.325 b
2	2.67	No	0.27	1.07	1.369 b
3	2.67	No	0.25	0.5	1.350 b
4	2.67	No	0.23	0.1	1.322 b
5	2.67	No	0.27	0.0	1.294 b
6	2.67	Mix 1	0.29	1.7	1.103 a
7	2.67	Mix 2	0.31	1.9	1.293 a
8	2.67	Mix 1	0.26	1.07	1.100 b
9	2.67	Mix 2	0.28	1.07	1.287 b
10	2.67	Mix 2	0.26	0.00	1.217 b
11	2.67	Mix 3	0.40	3.00	0.760 a
12	2.96	Mix 4	0.265	1.07	0.931 b
13	2.24	No	0.50	1.07	1.227
14	2.24	No	0.624	1.07	1.120
15	2.24	No	0.66	1.07	1.085
16	2.24	No	0.72	1.07	1.023
17	2.24	No	0.74	1.07	1.001

a. Optimized volume fraction and particle size

b. Optimized volume fraction for given particle size

NOTES: 1. For mixtures, see Table 5-11.

2. Results presented in this table are from References 5.54, 5.72 through 5.76, and were performed using XSDRNPM (Reference 5.77).

TABLE 5-10

AVERAGE IMPURITY CONCENTRATIONS

SAMPLE DESCRIPTION	ELEMENTAL COMPOSITION OF DEBRIS (wt%)					
	Cd	U	Fe	B	Zr	Reference
OTSG "B"	0.06	82.9	0.29	0.01	1.44	5.26
Core Debris	<LLD	73.6	0.74	0.48	11.2	5.27
Bottom Head	<LLD	64.7	2.2	0.072	12.8	5.31
Pressurizer	0.77	2.7	8.68	1.07	2.43	5.45
MUF-5B (B&W)	11	6	7	2	>25	5.44
MUF-5B (0104)	11.4	~5	5.7	0.62	5.4	5.44
MUF-5B (0105)	11.2	~5	5.22	0.64	5.7	5.44
MUF-5B (0111)	----	7.27	3.9	~0.1	12.6	5.44

NOTES: 1. Only significant elements included

TABLE 5-11

IMPURITY CONTENT OF CORE DEBRIS

(WEIGHT PERCENT)

ELEMENT	MIX 1	MIX 2	MIX 3	MIX 4
UO_2	98.416	98.470	83.79	99.928
Zr	1.260	1.260	12.70	0.000
Fe	0.261	0.261	2.44	0.000
B	0.009	0.009	0.11	0.072
Cd	0.054	0.000	0.00	0.000
Cr	0.000	0.000	0.75	0.000
Mo	0.000	0.000	0.15	0.000
Mn	0.000	0.000	0.06	0.000
TOTAL	100	100	100	100

NOTES: 1. These impurity concentrations were developed from review of References 5.15, 5.34, 5.39, 5.50, 5.51 and 5.52. Mixes 1 and 2 were based on data from the "B" OTSG and Mixes 3 and 4 were from the bottom head data (see Table 5-10).

TABLE 5-12

SUMMARY OF FINITE GEOMETRY KENO V.a ANALYSES

CASE DESCRIPTION	k_{eff}
Reference Case	0.983
Reference Case Geometry Modelling With Impurity Mix 1 Fuel	0.836
Reference Case Geometry Modelling With Impurity Mix 2 Fuel	0.972
Reference Case Fuel Modelling With Debris Removed From Selected LCSA Holes	0.954

NOTES: 1. All results include a 2.5% Δk computer code uncertainty bias and are taken from Reference 5.54.

2. See Section 5.5.1.3.2 for discussion of modelling of cases other than reference case.

SECTION 5.0

FIGURES

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

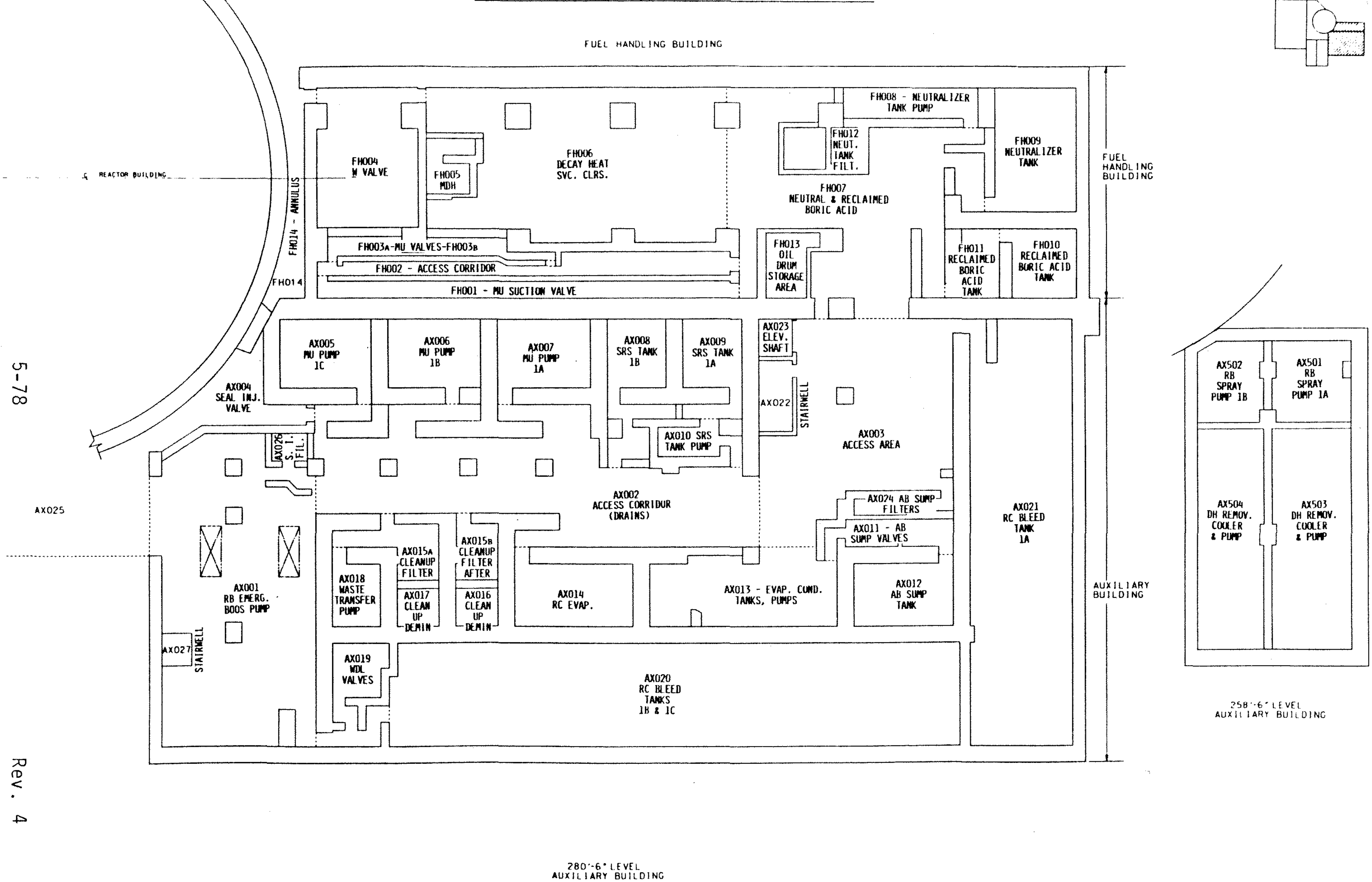
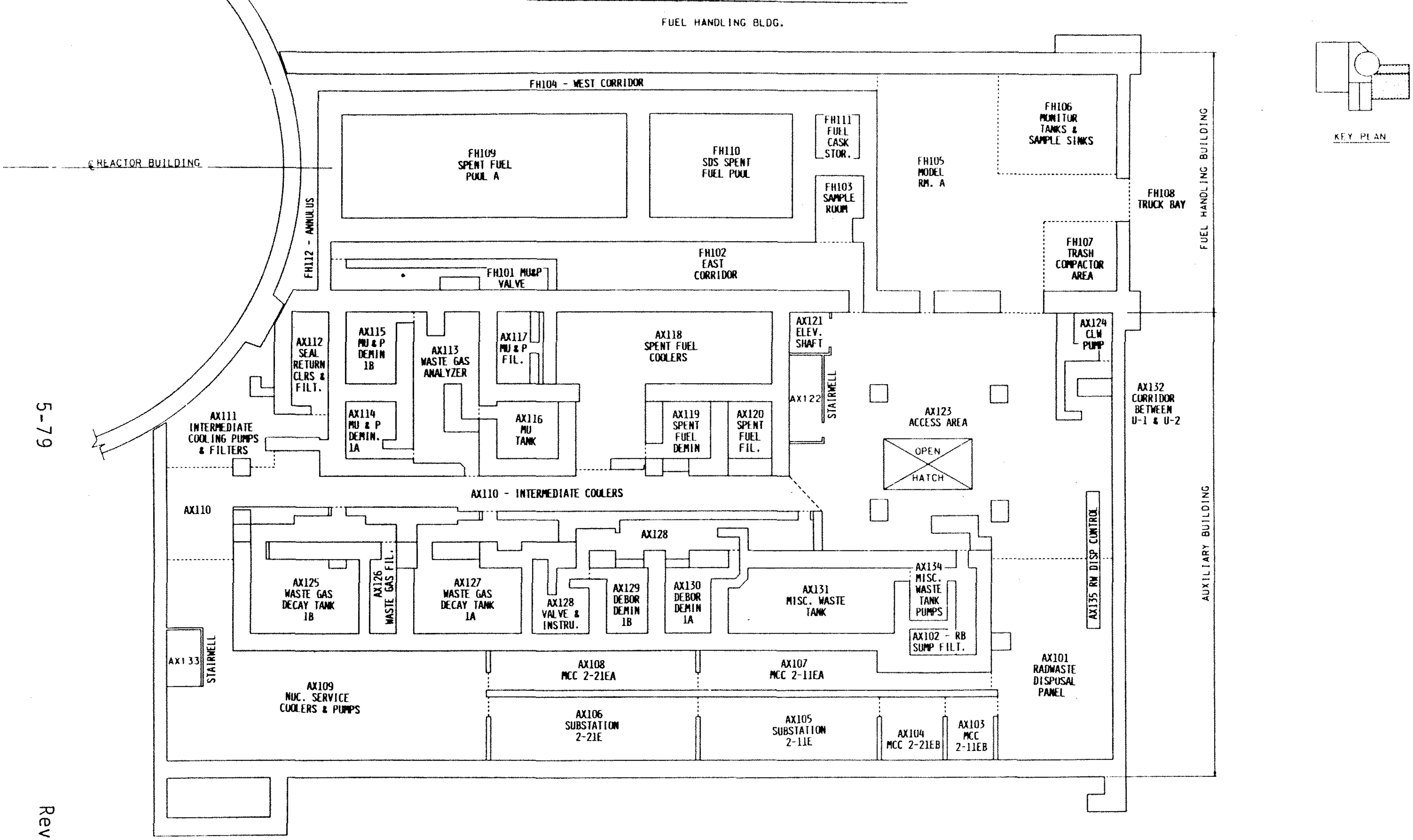


FIGURE 5-2
AUXILIARY BUILDING 305' ELEVATION



305' LEVEL
AUXILIARY BLDG.

FIGU 3
AUXILIARY BUILDING 328' ELEVATION

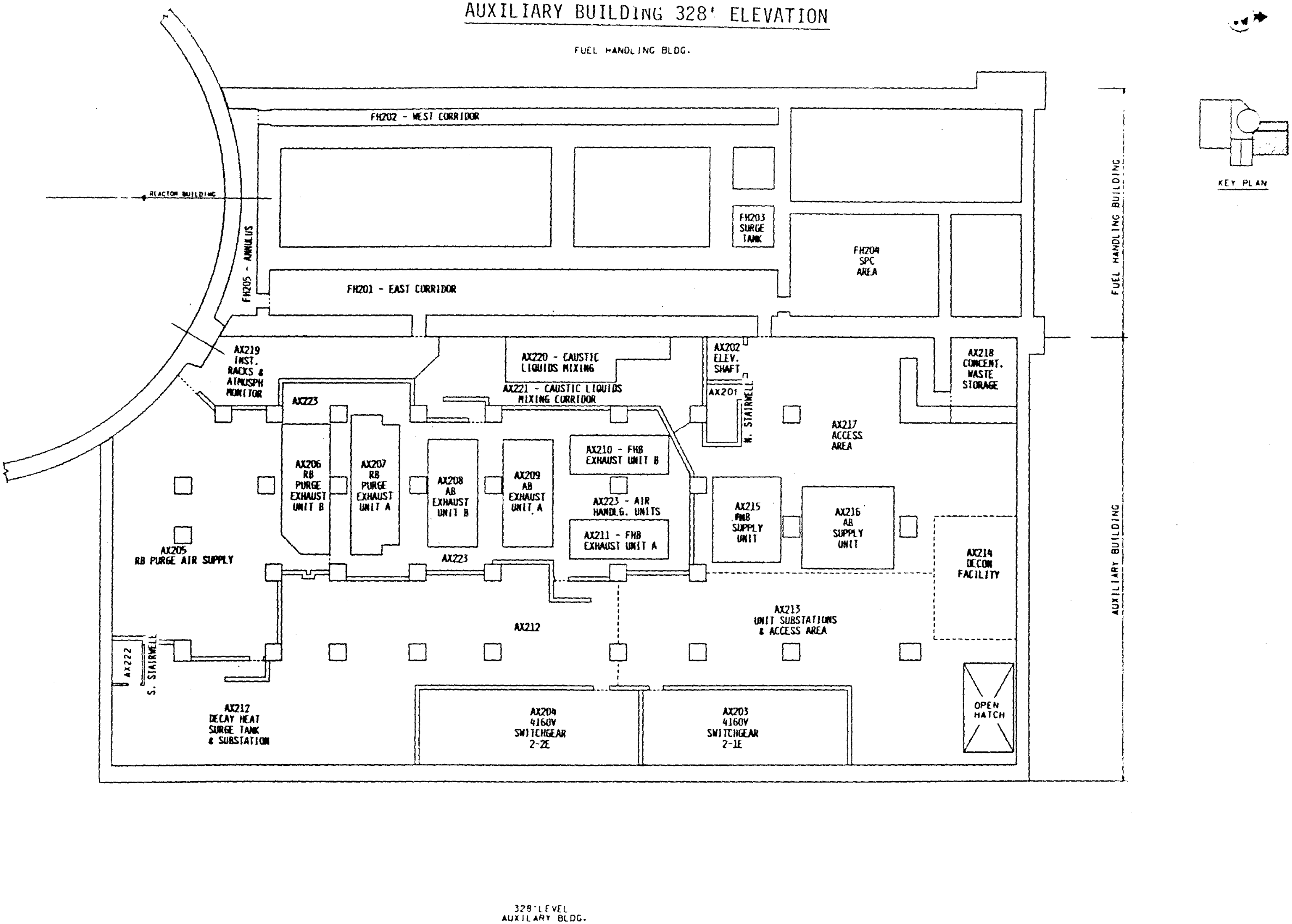
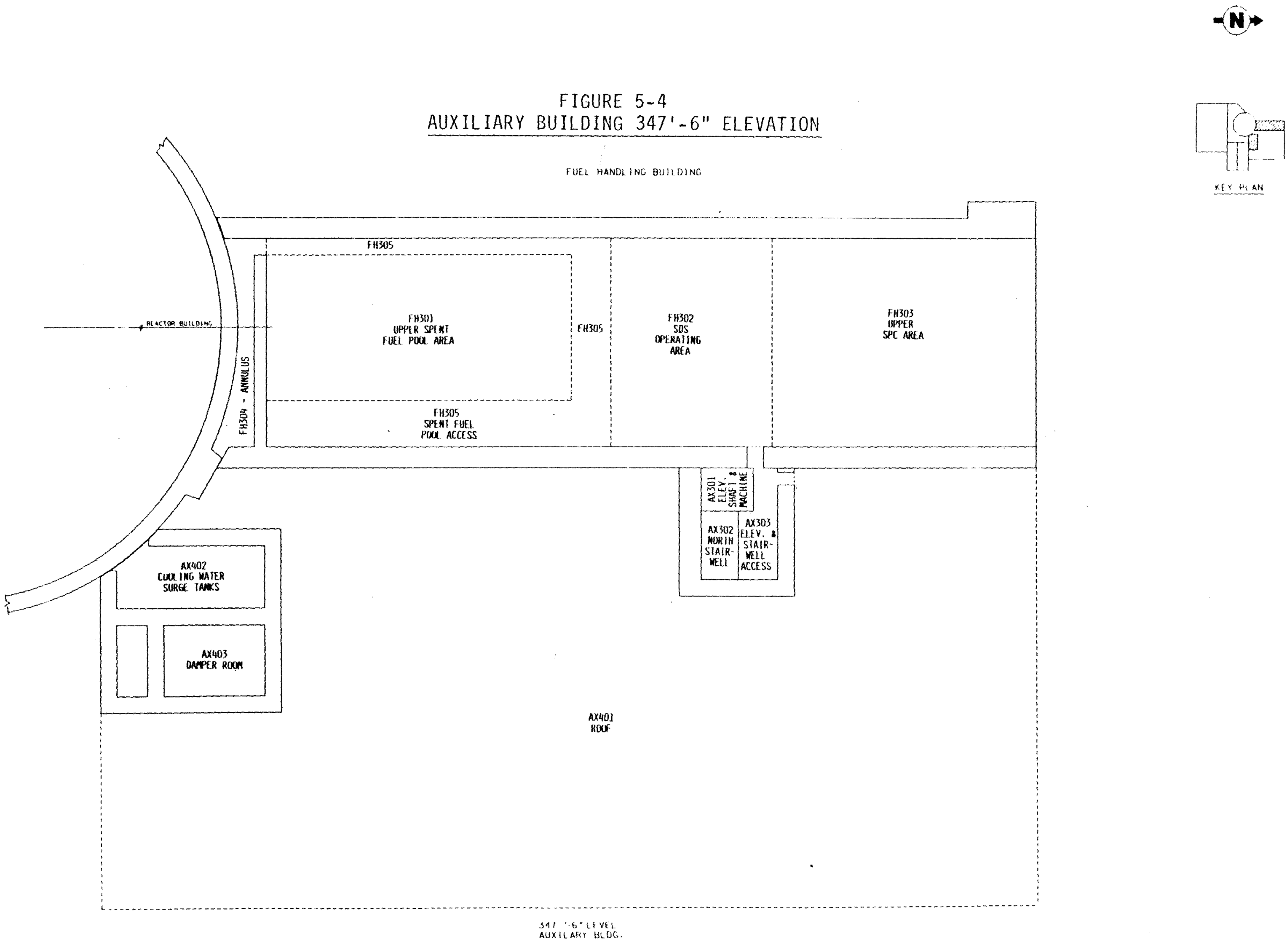


FIGURE 5-4
AUXILIARY BUILDING 347'-6" ELEVATION



TMI-2 REACTOR - UPPER HALF
(Simplified)

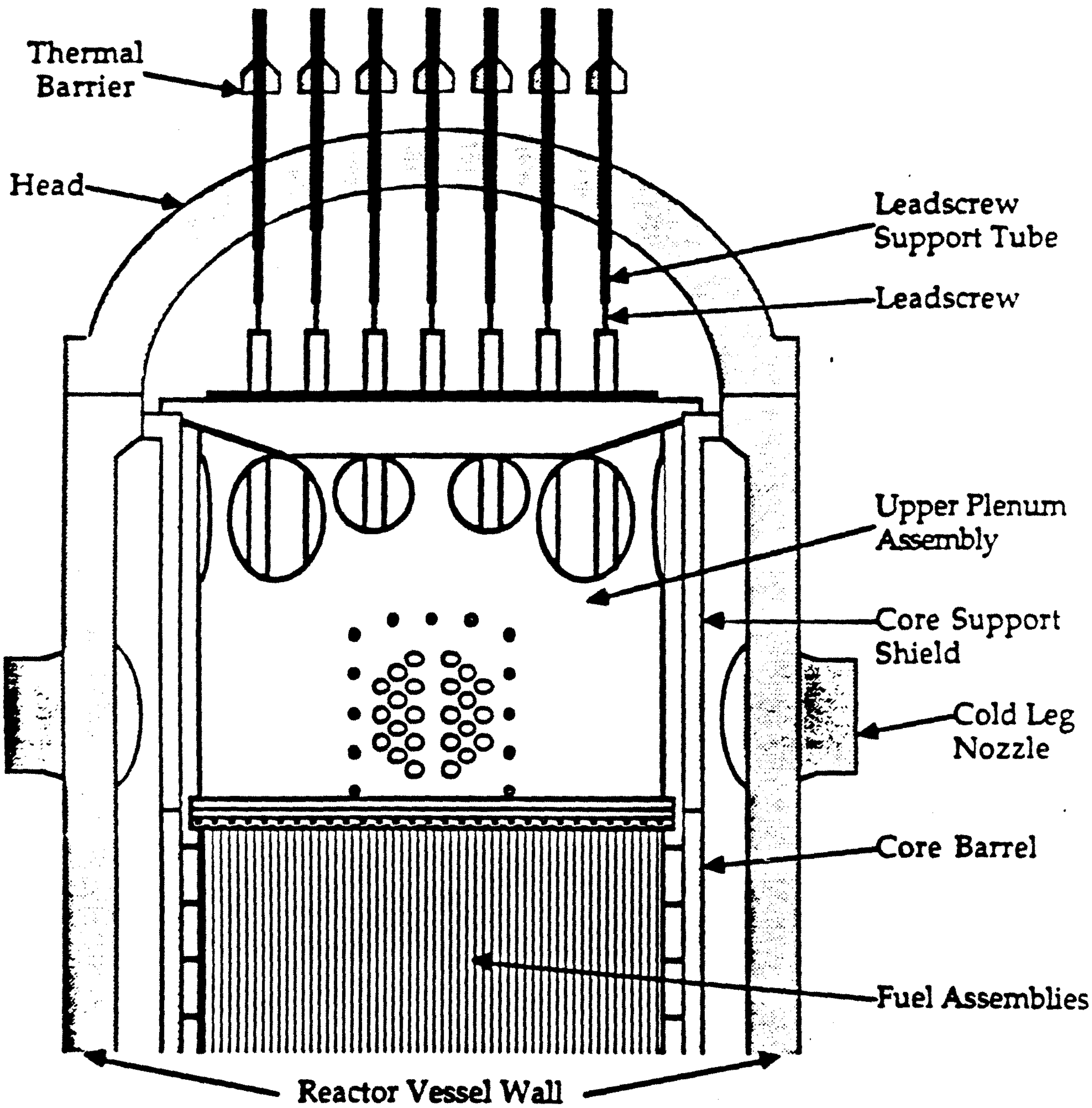
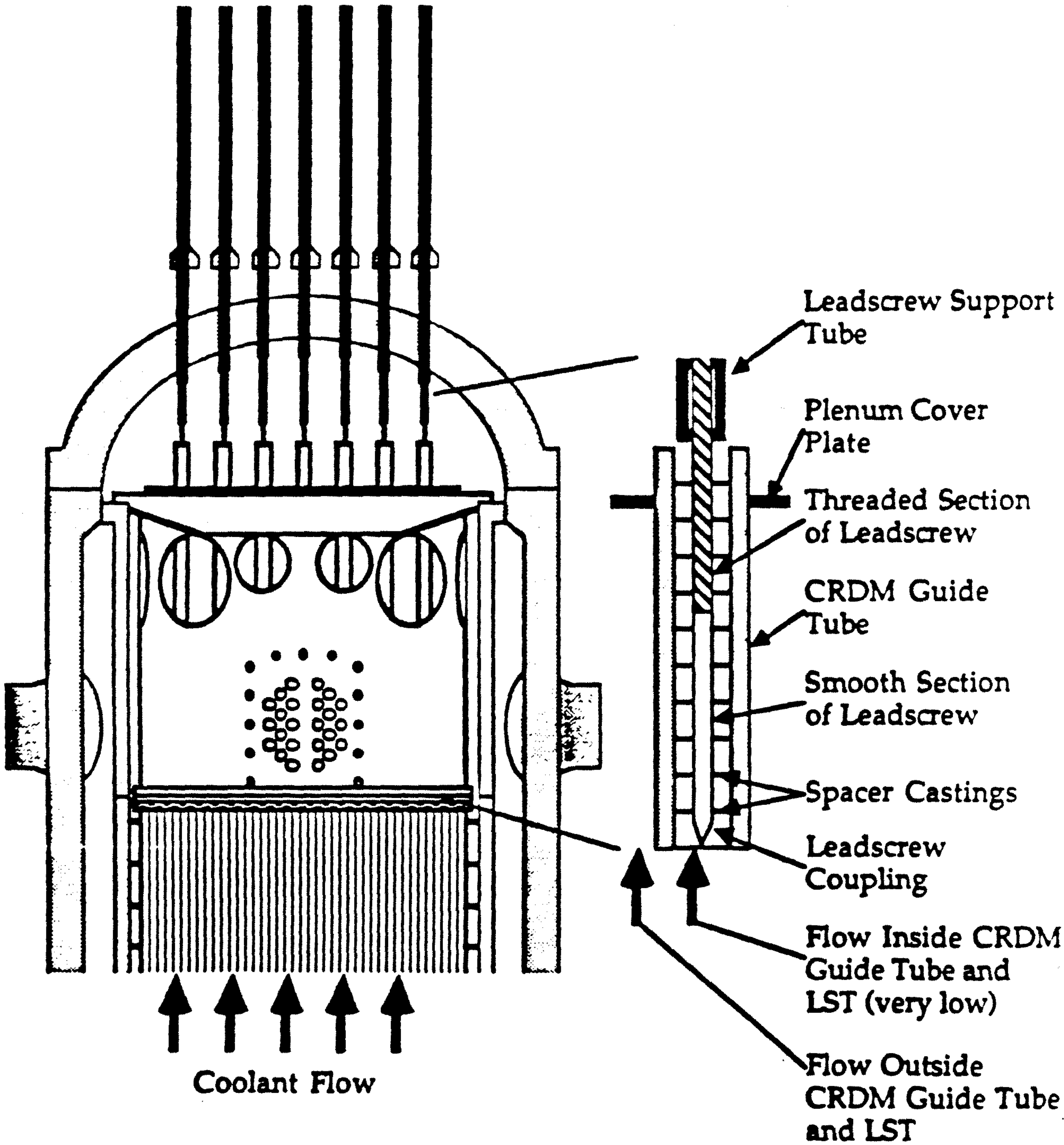
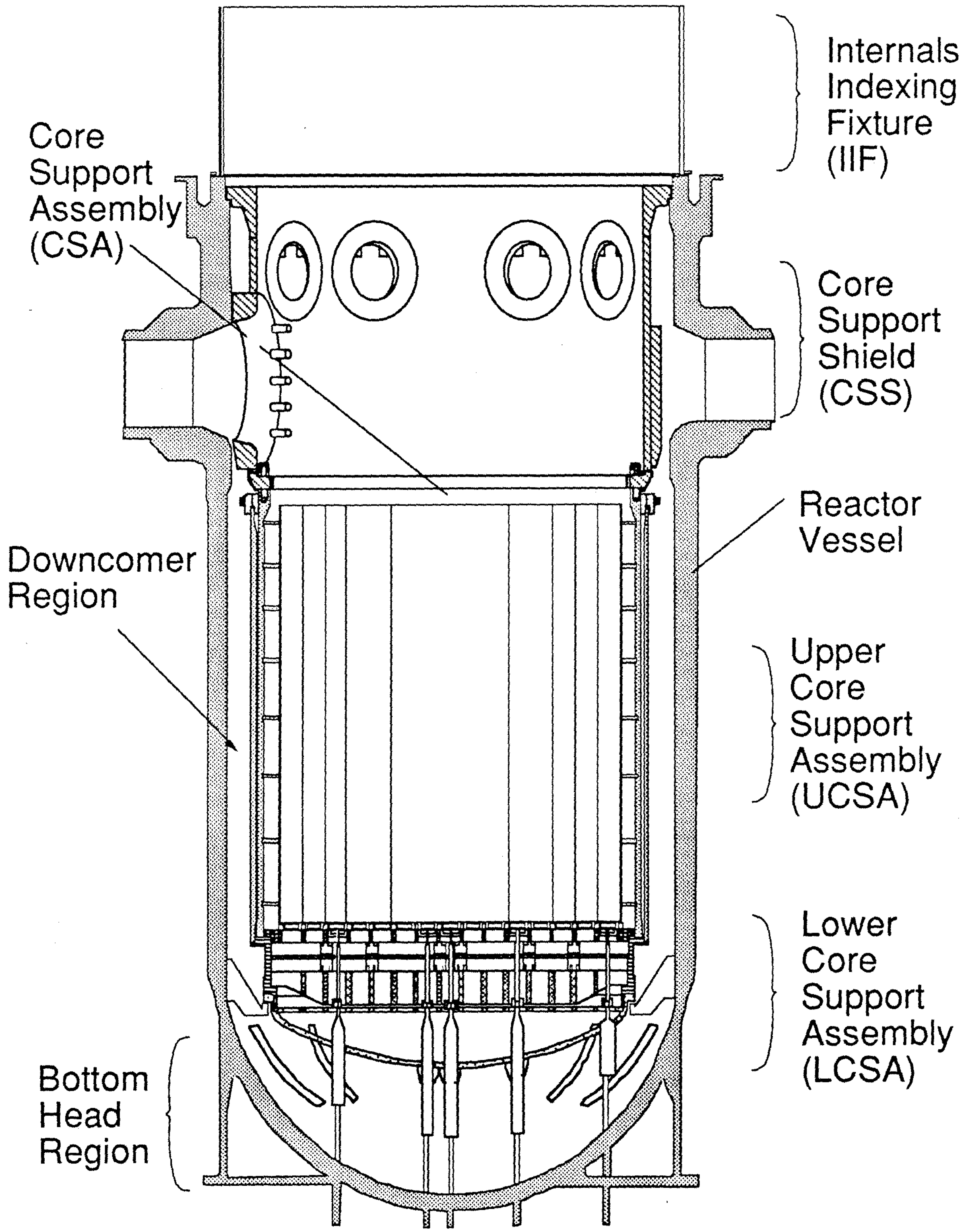


FIGURE 5-6

LEADSCREW AND LS SUPPORT TUBE

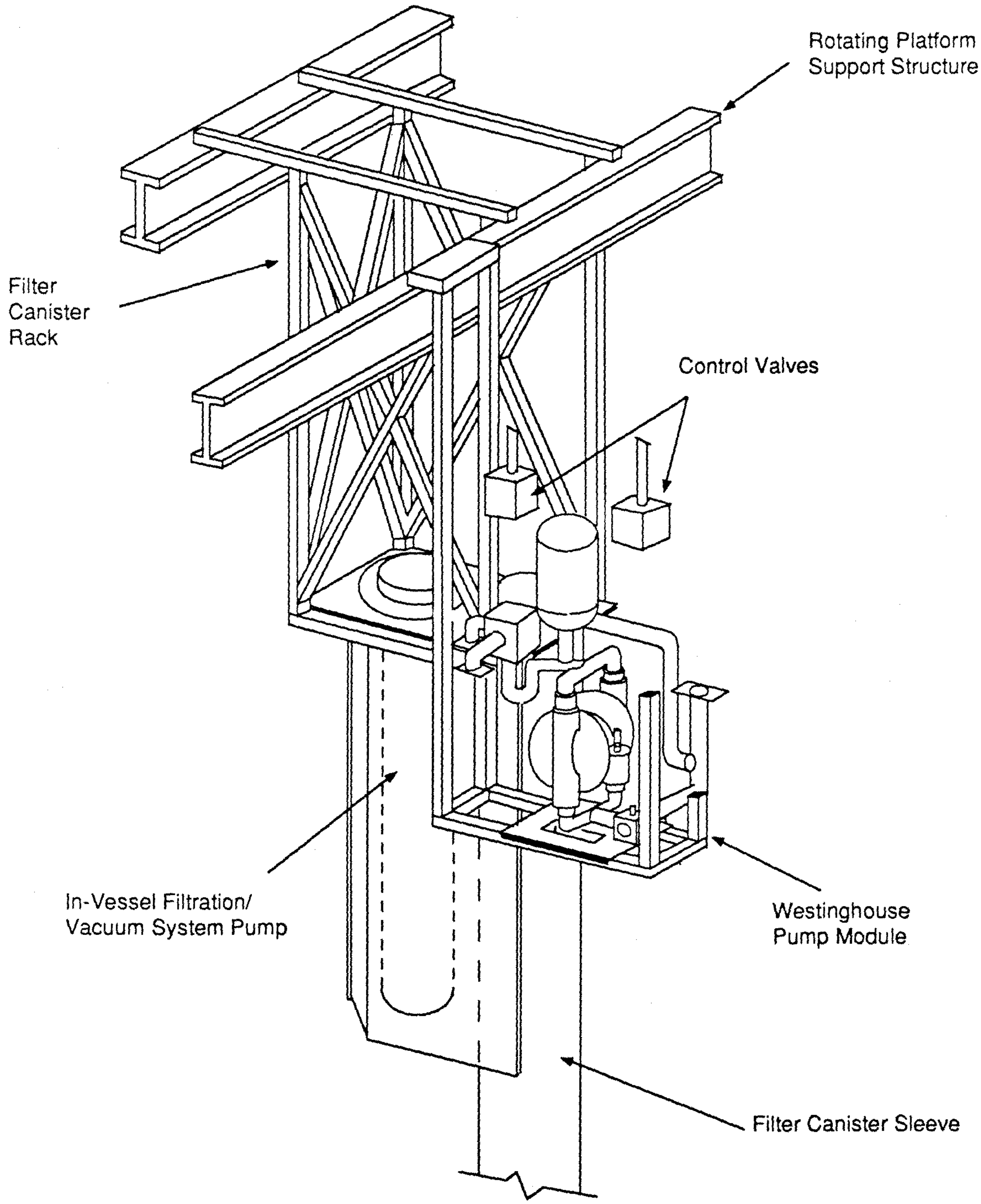
(Simplified)





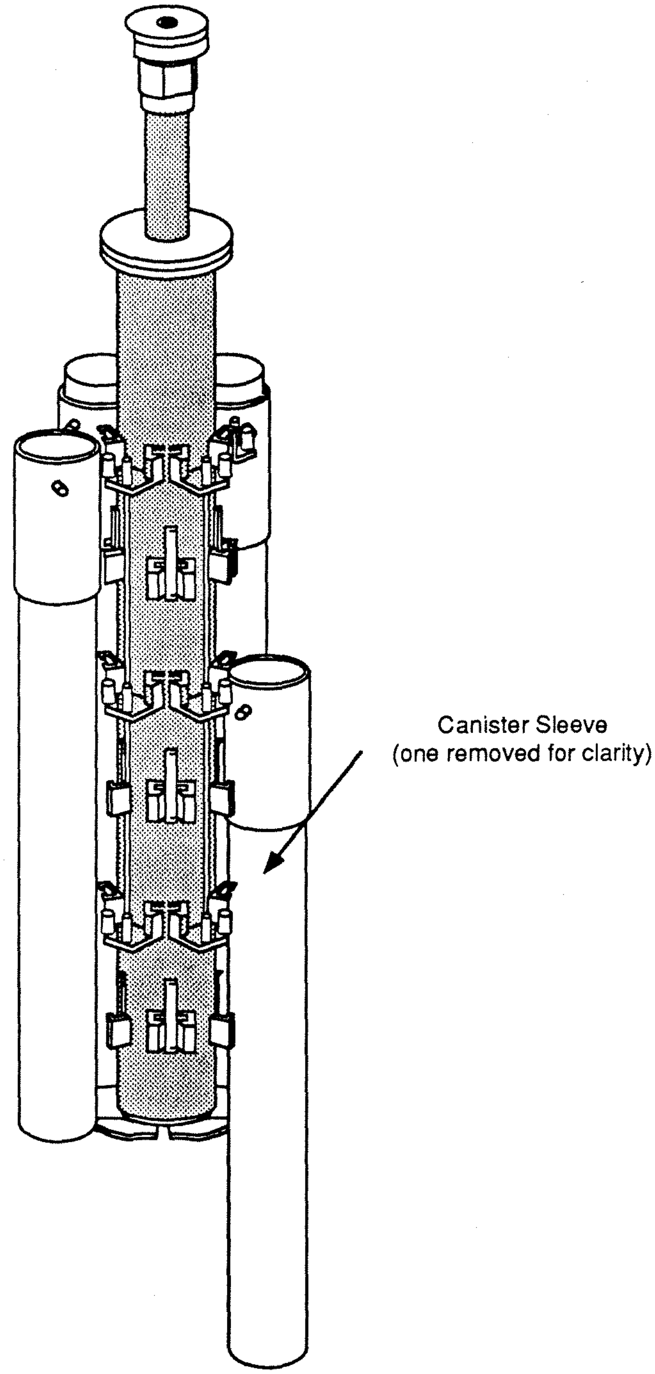
Reactor Vessel Cutaway View

FIGURE 5-7



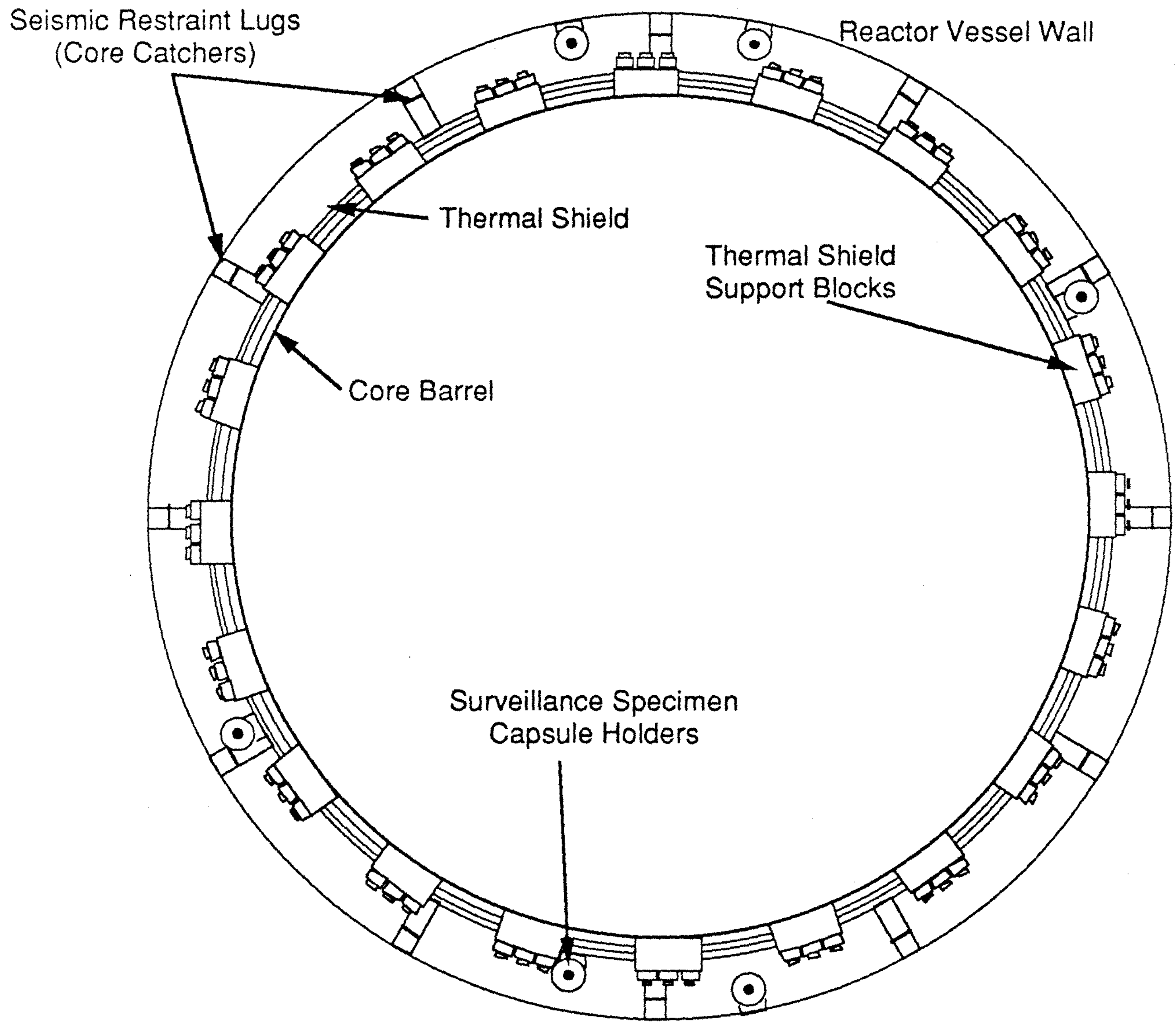
Defueling Equipment Remaining in the Vessel

FIGURE 5-8



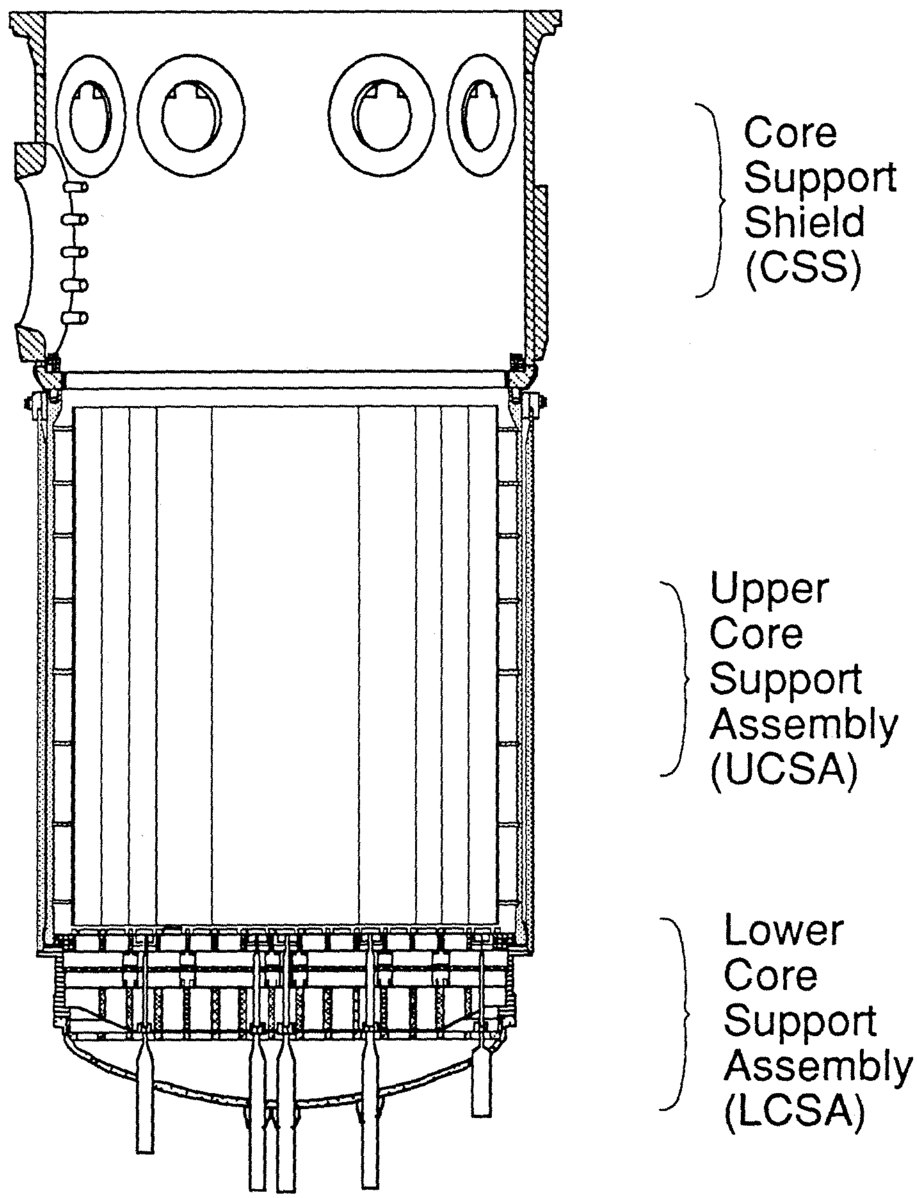
Canister Positioning System

FIGURE 5-9



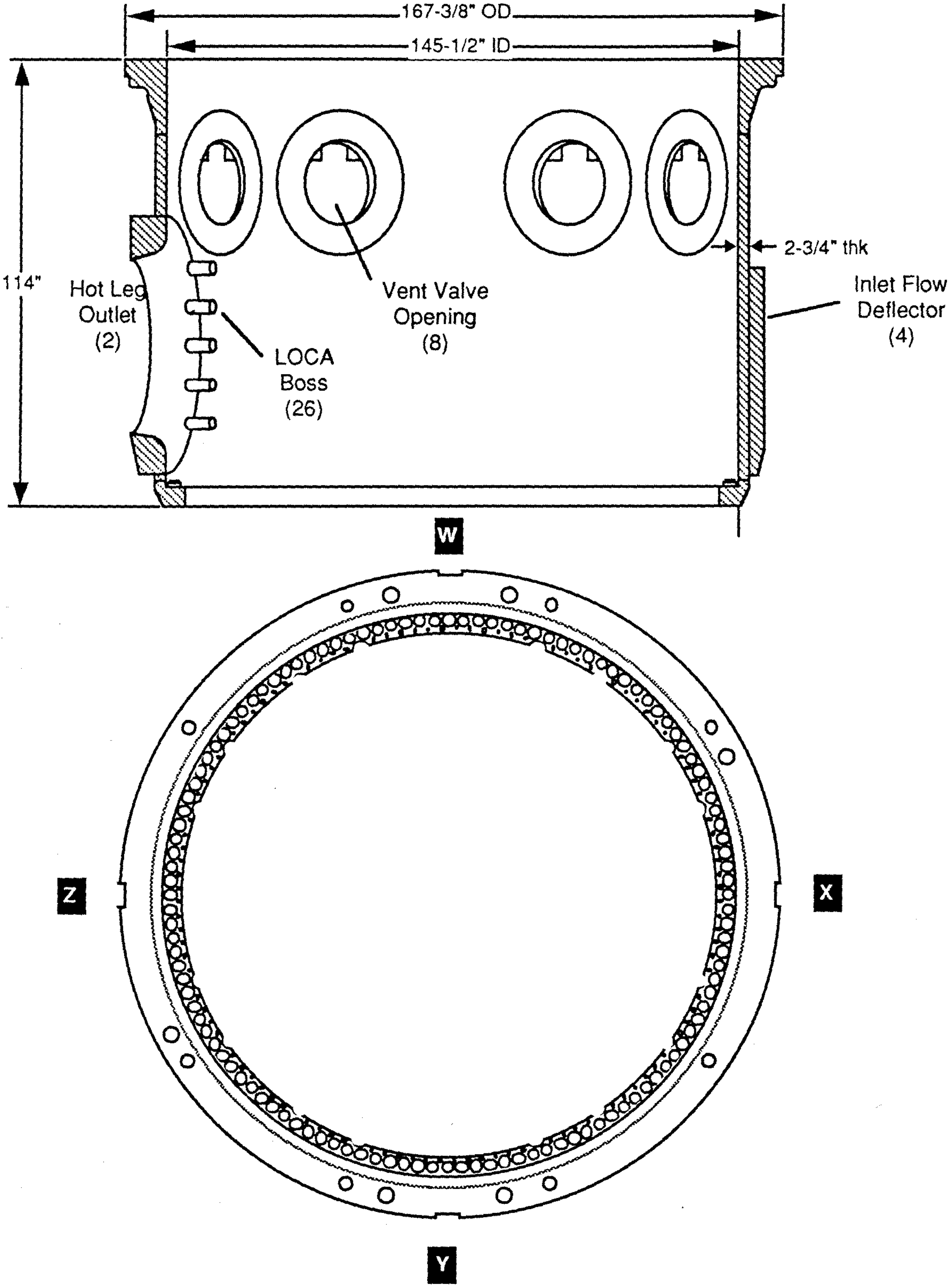
Downcomer Region

FIGURE 5-10



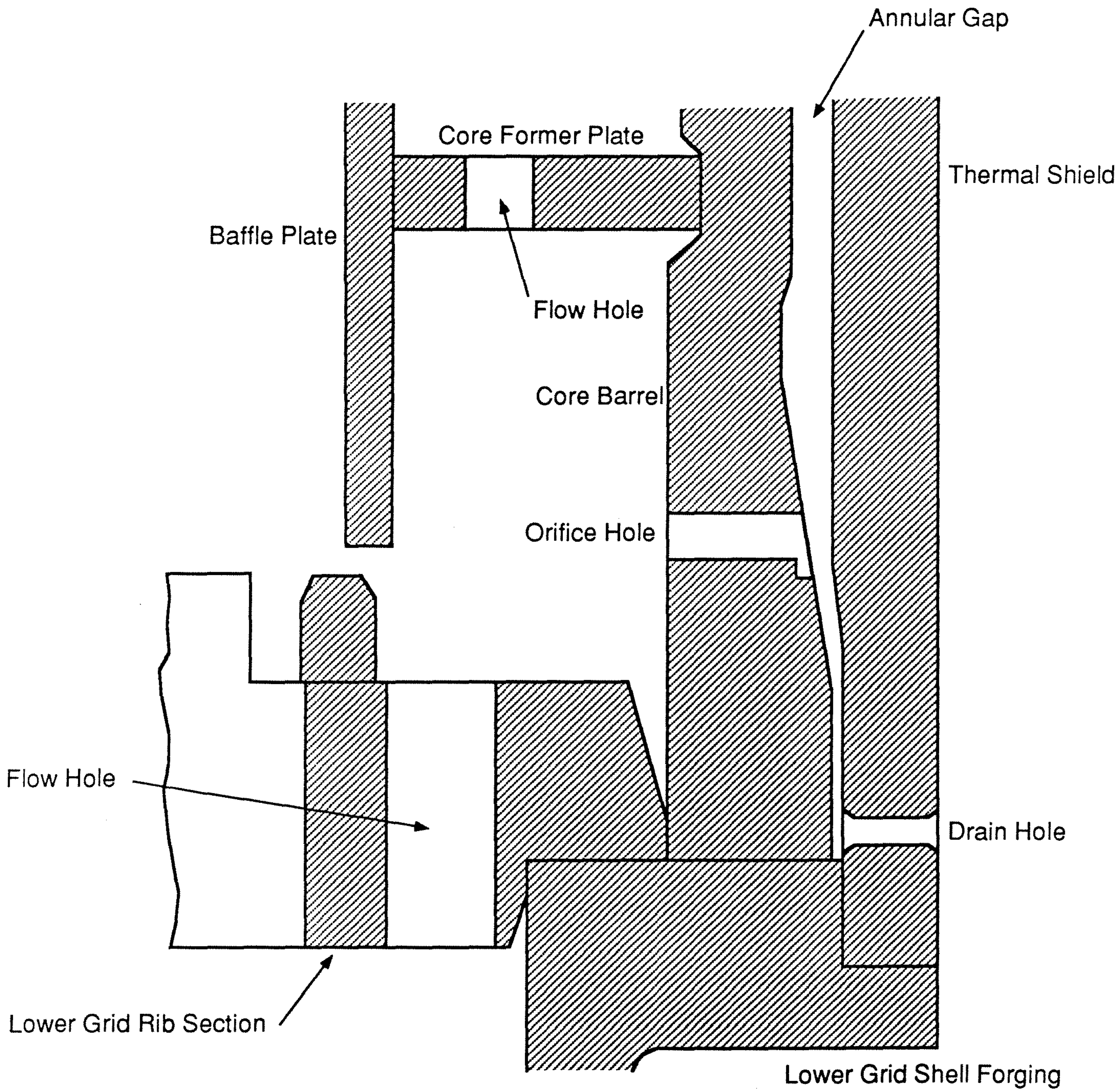
Core Support Assembly

FIGURE 5-11



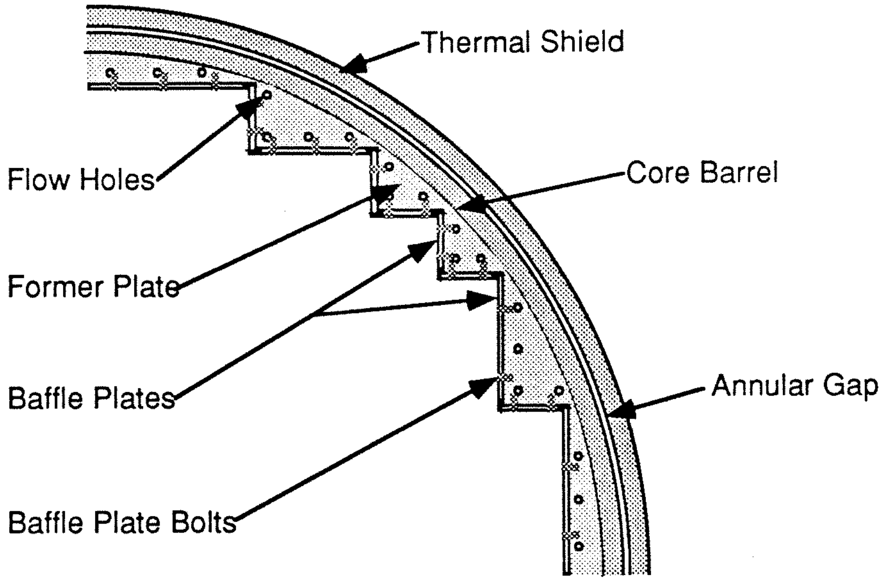
Core Support Shield

FIGURE 5-12

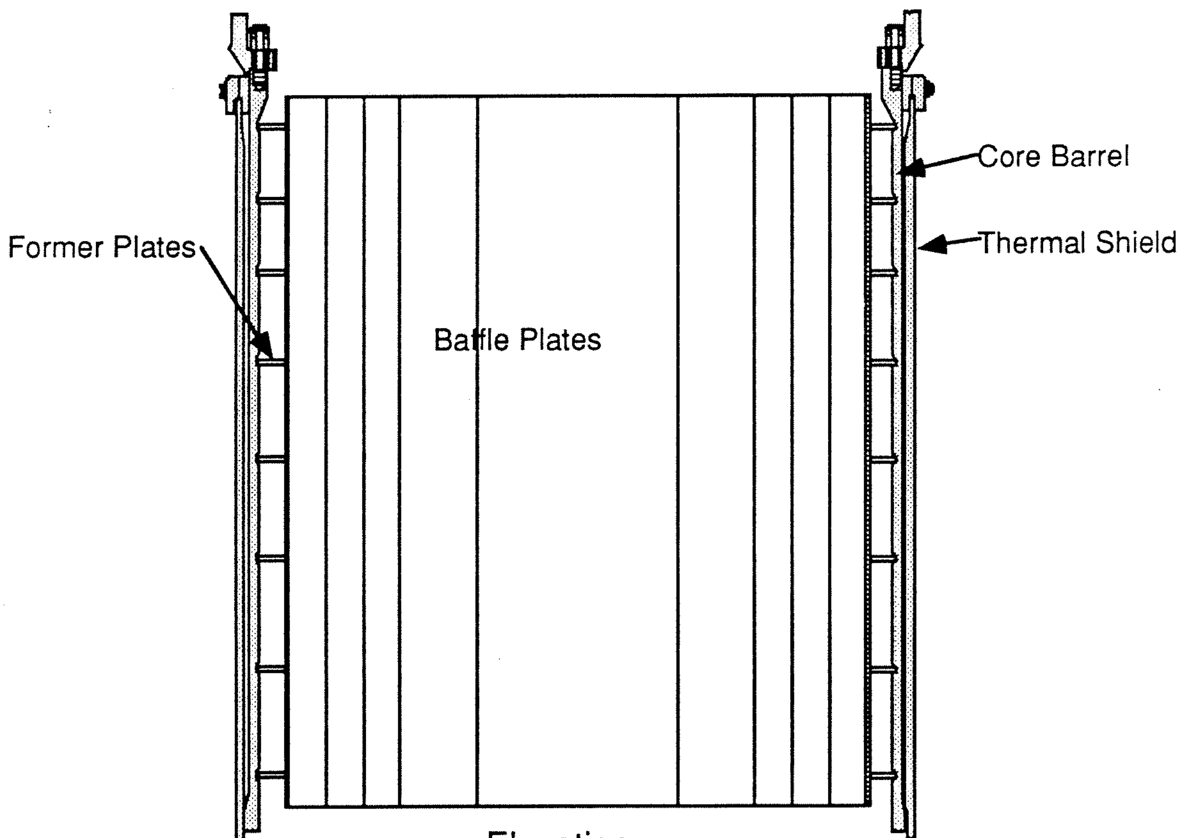


Thermal Shield Inner Surface and Annular Gap

FIGURE 5-13



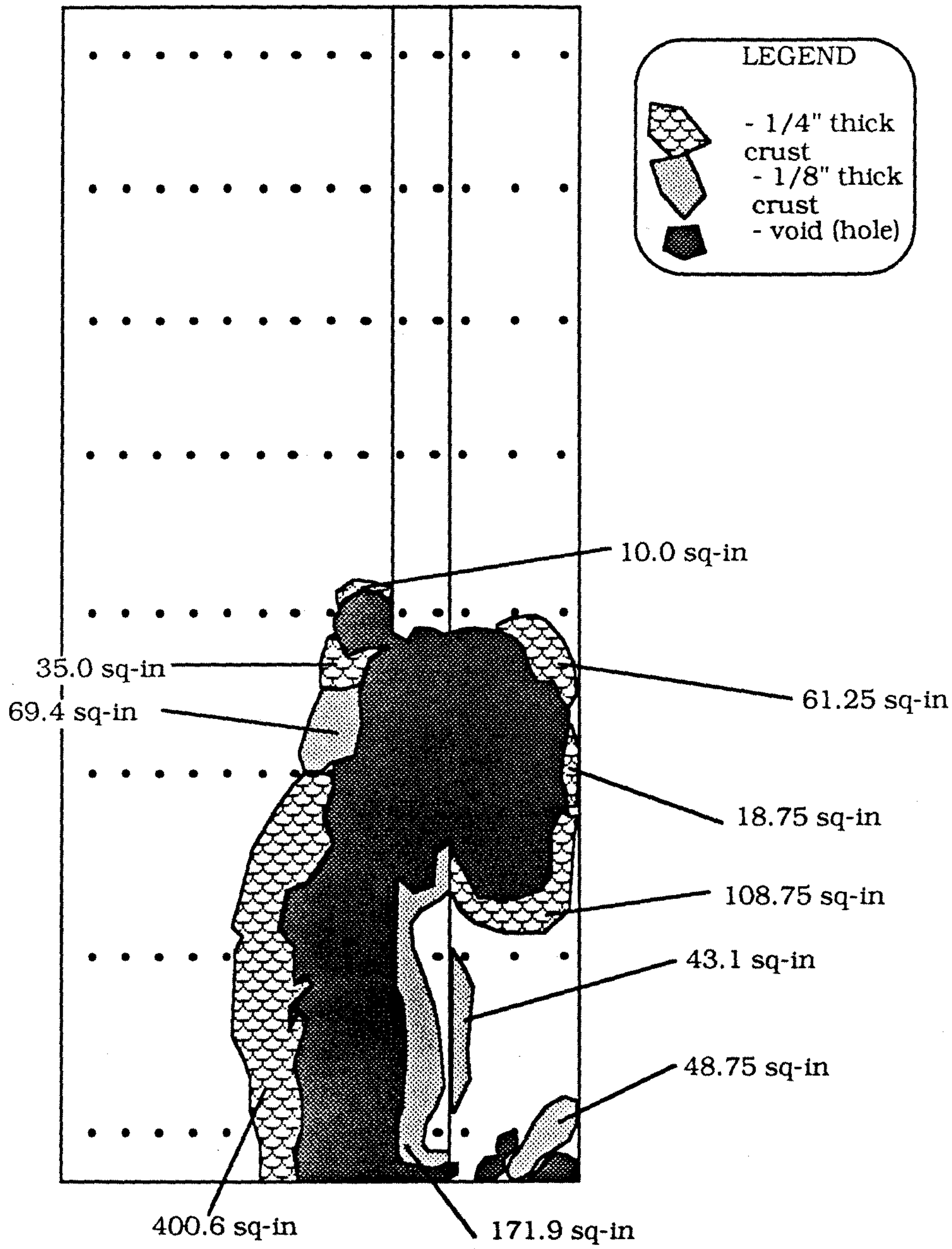
Plan View



Elevation

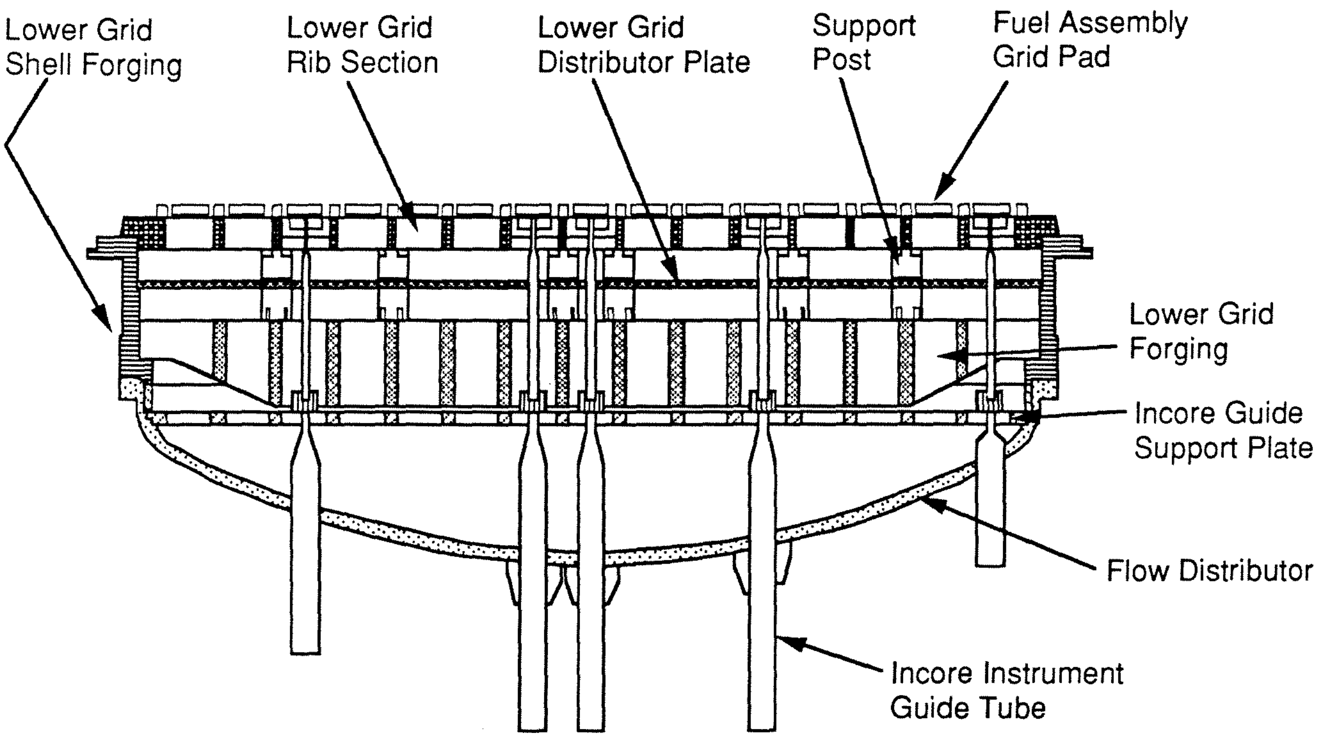
Upper Core Support Assembly

FIGURE 5-14



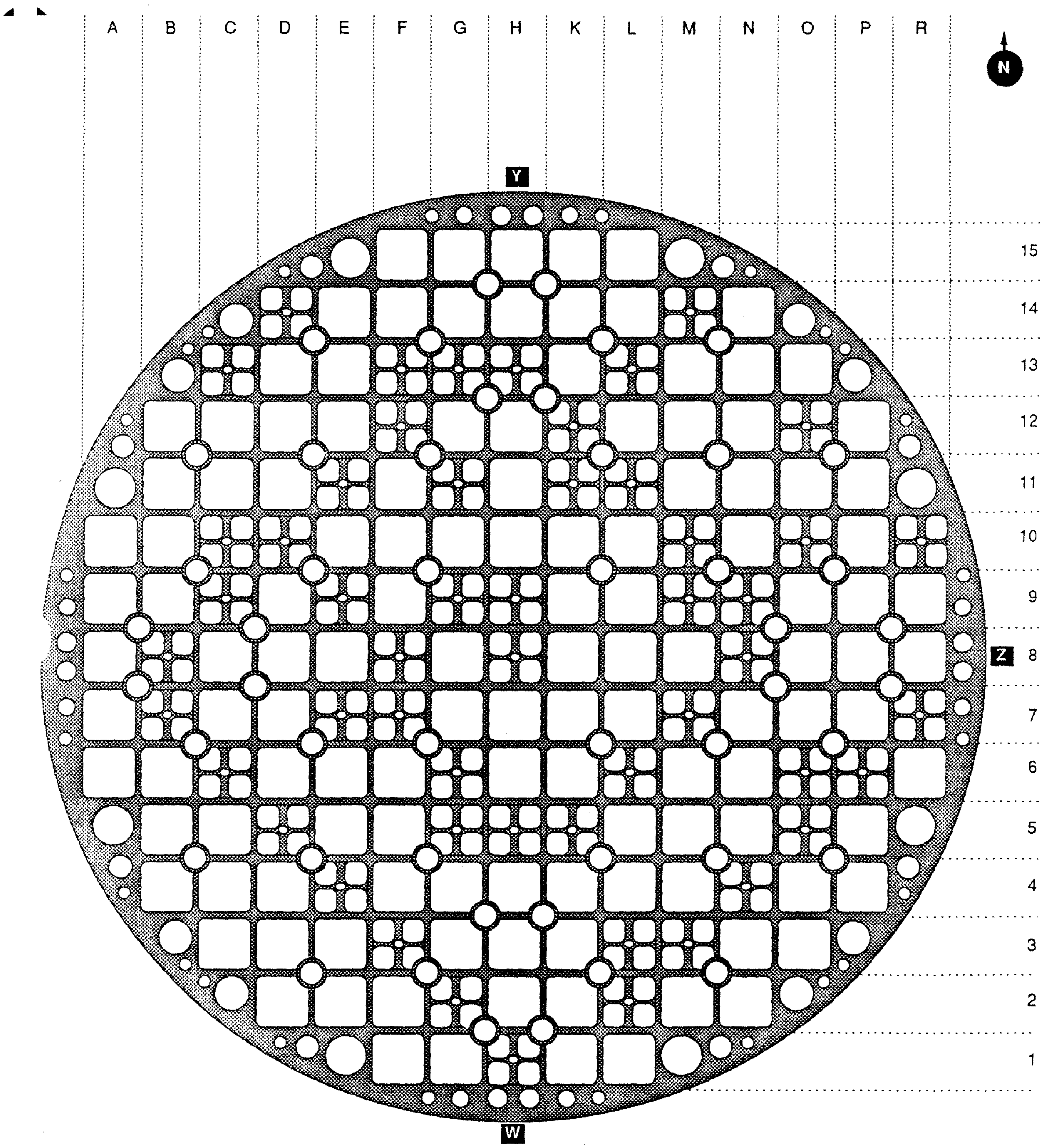
Melt Damage Near R-6 Grid Location

FIGURE 5-15



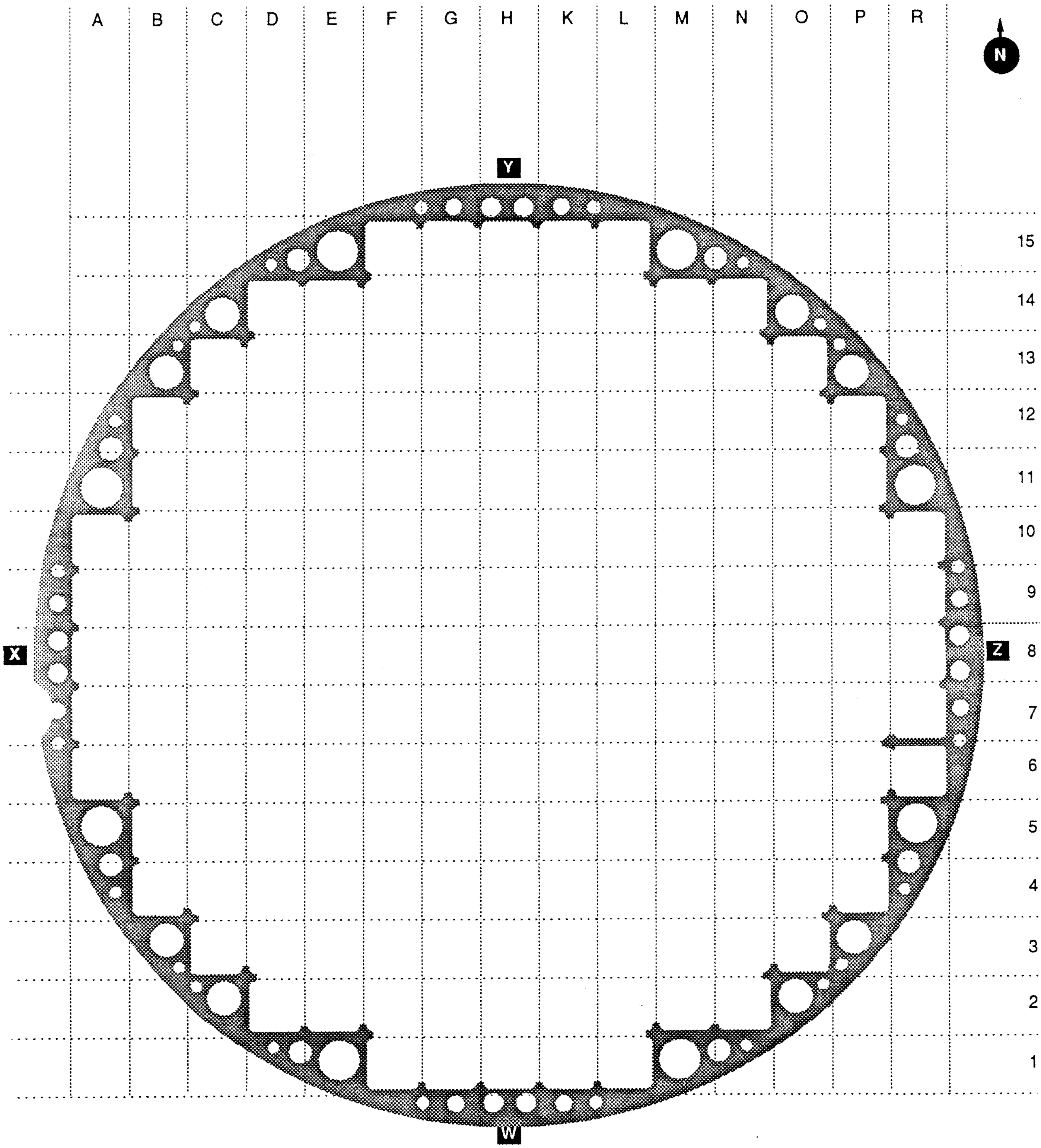
Lower Core Support Assembly (LCSA)

FIGURE 5-16



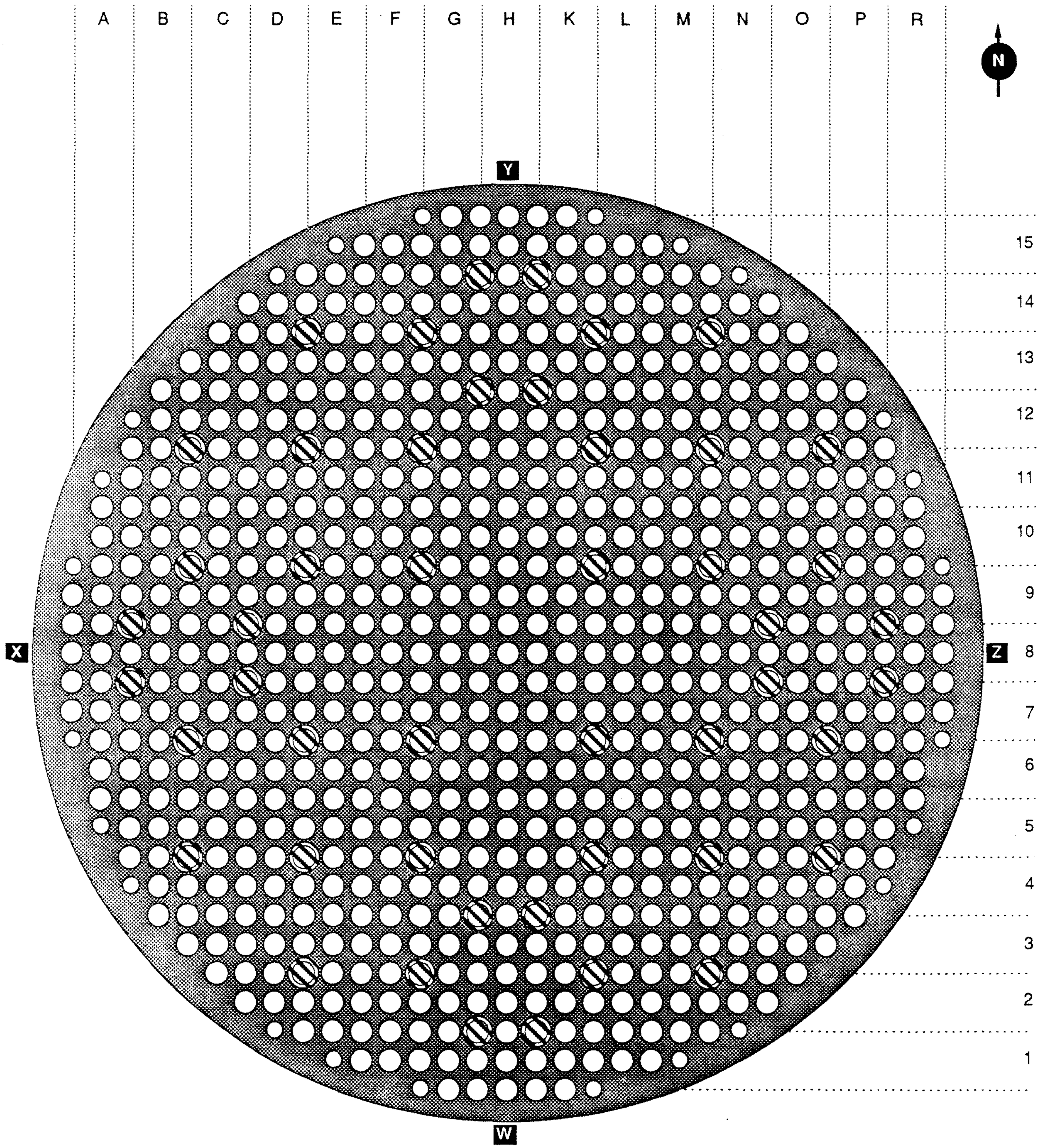
Lower Grid Rib Section

FIGURE 5-17



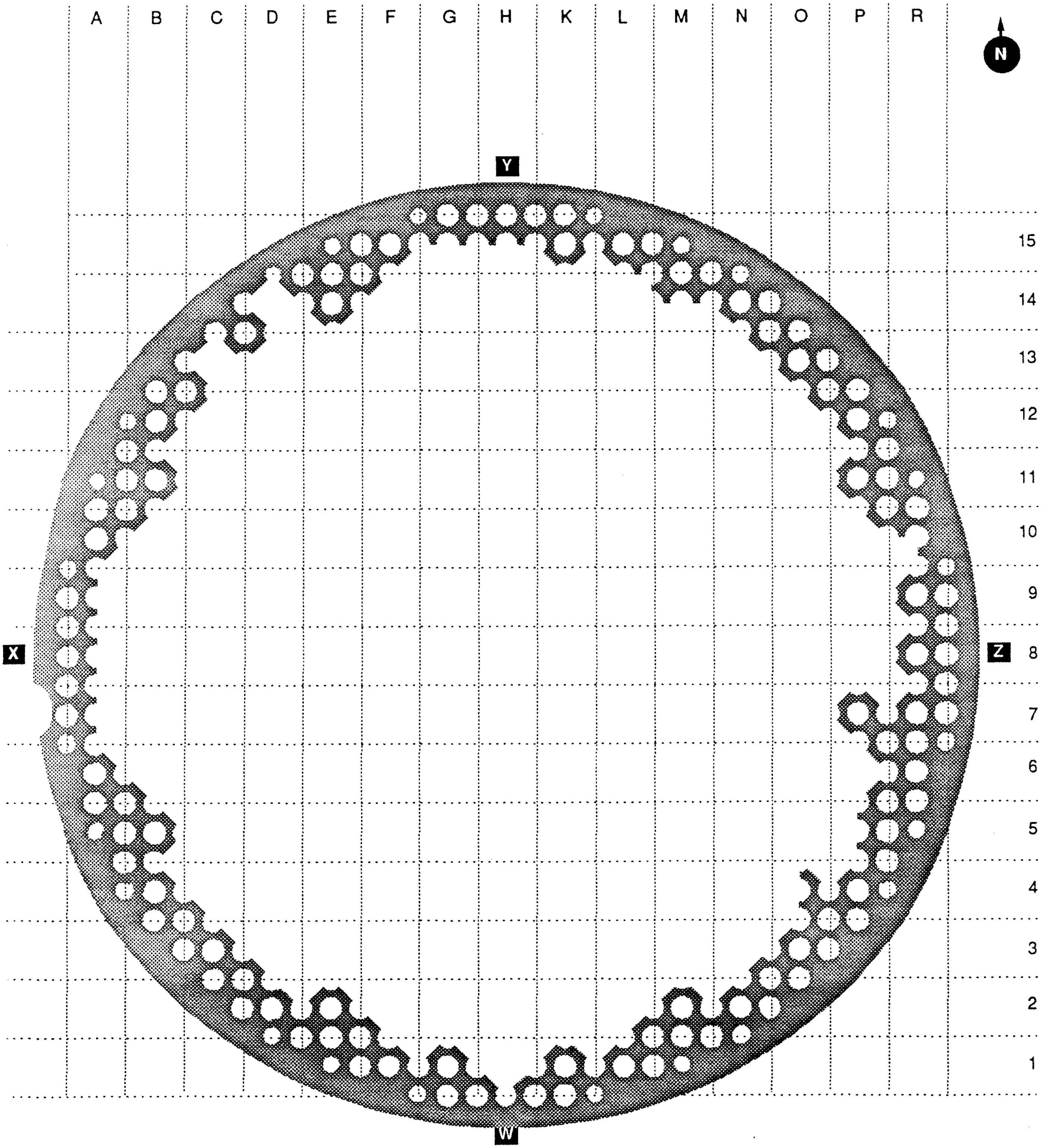
Lower Grid Rib Section After Cuts

FIGURE 5-18



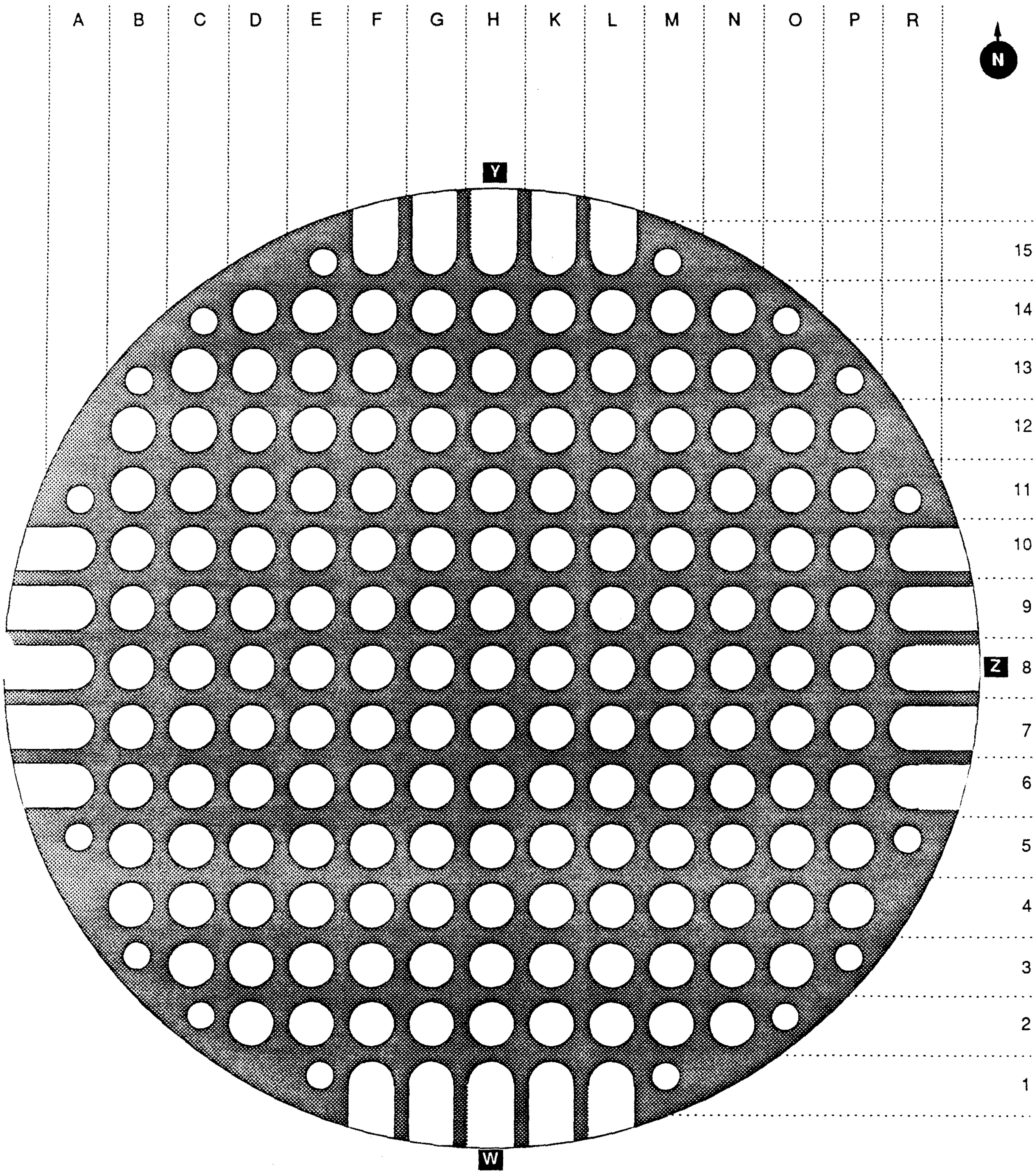
Lower Grid Distributor Plate

FIGURE 5-19



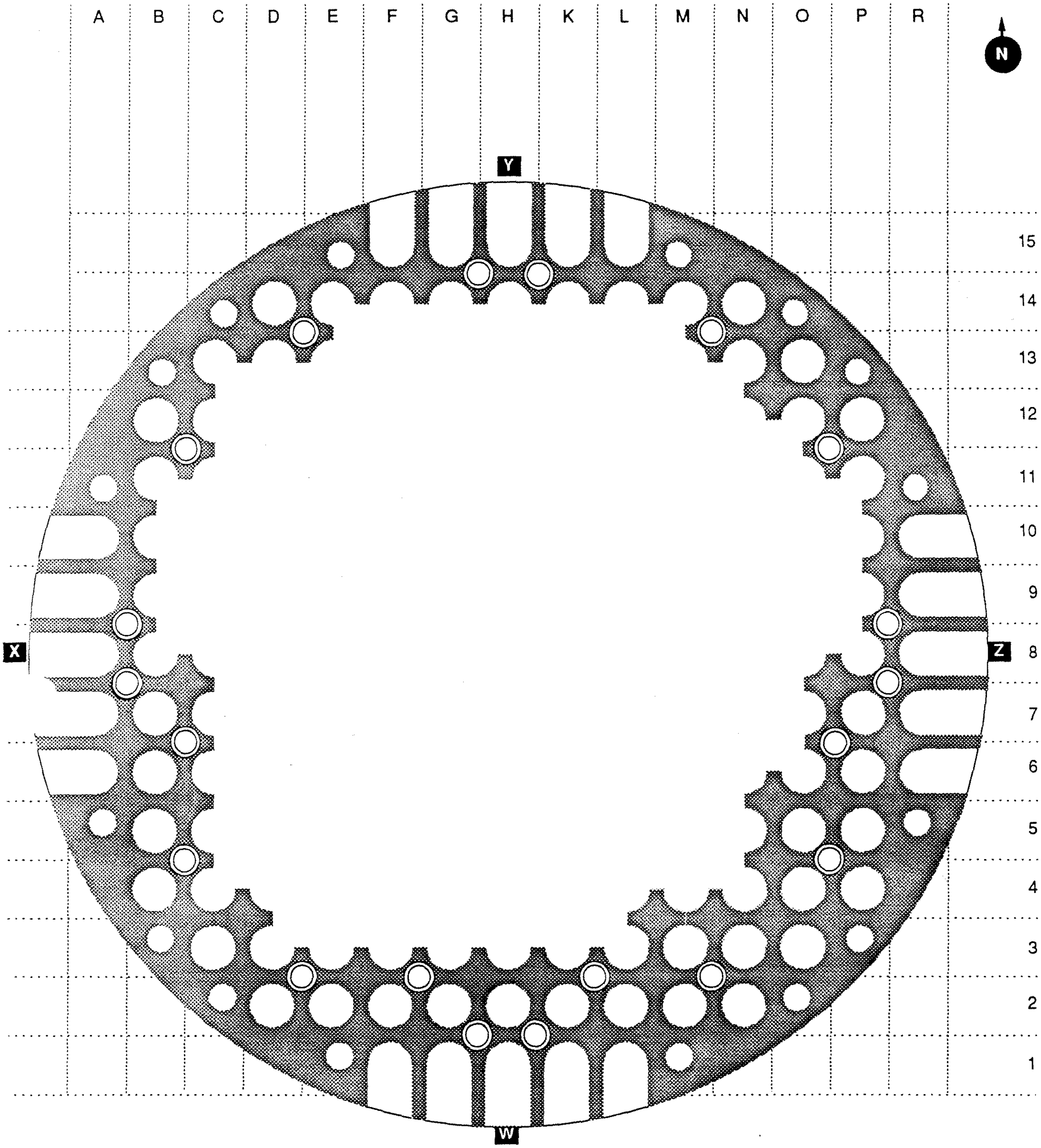
Lower Grid Distributor Plate After Cuts

FIGURE 5-20



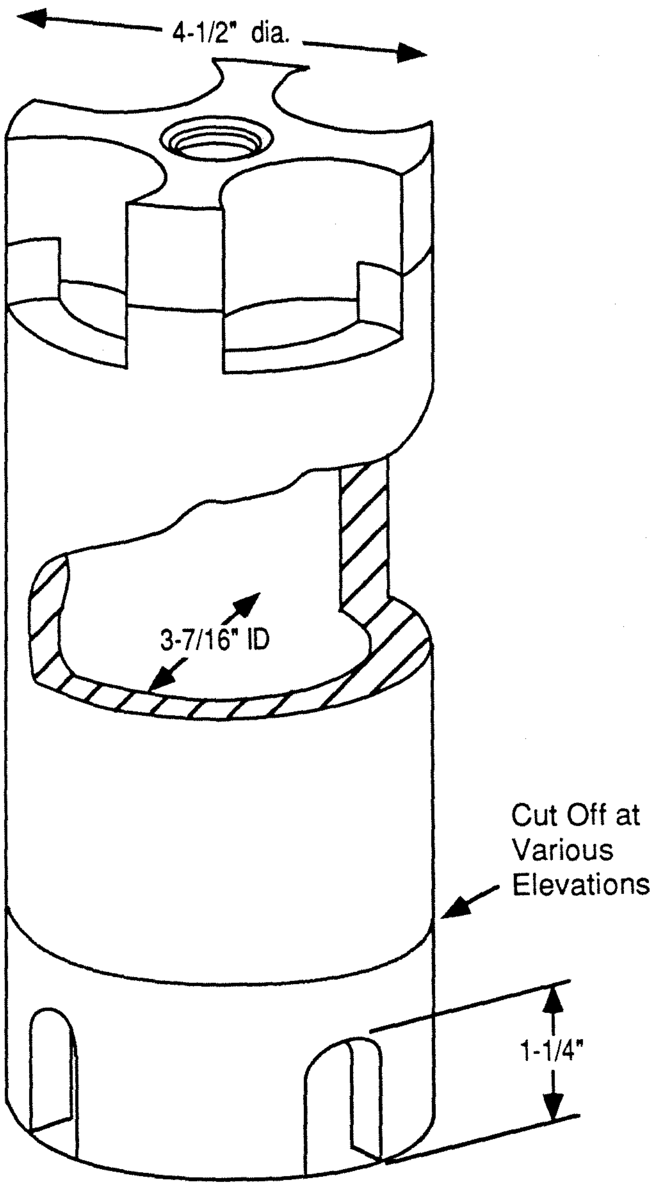
Lower Grid Forging

FIGURE 5-21



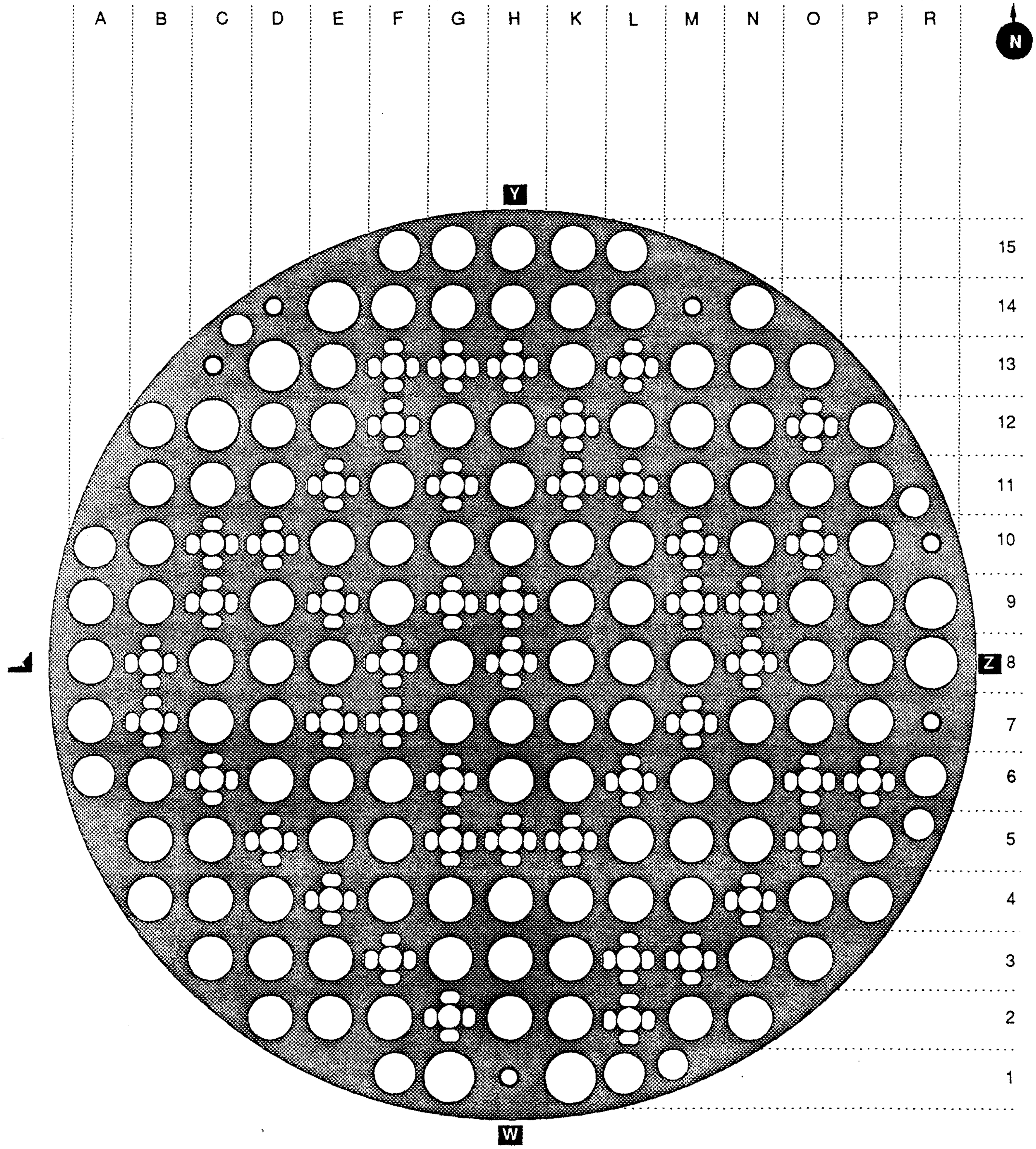
Lower Grid Forging After Cuts

FIGURE 5-22



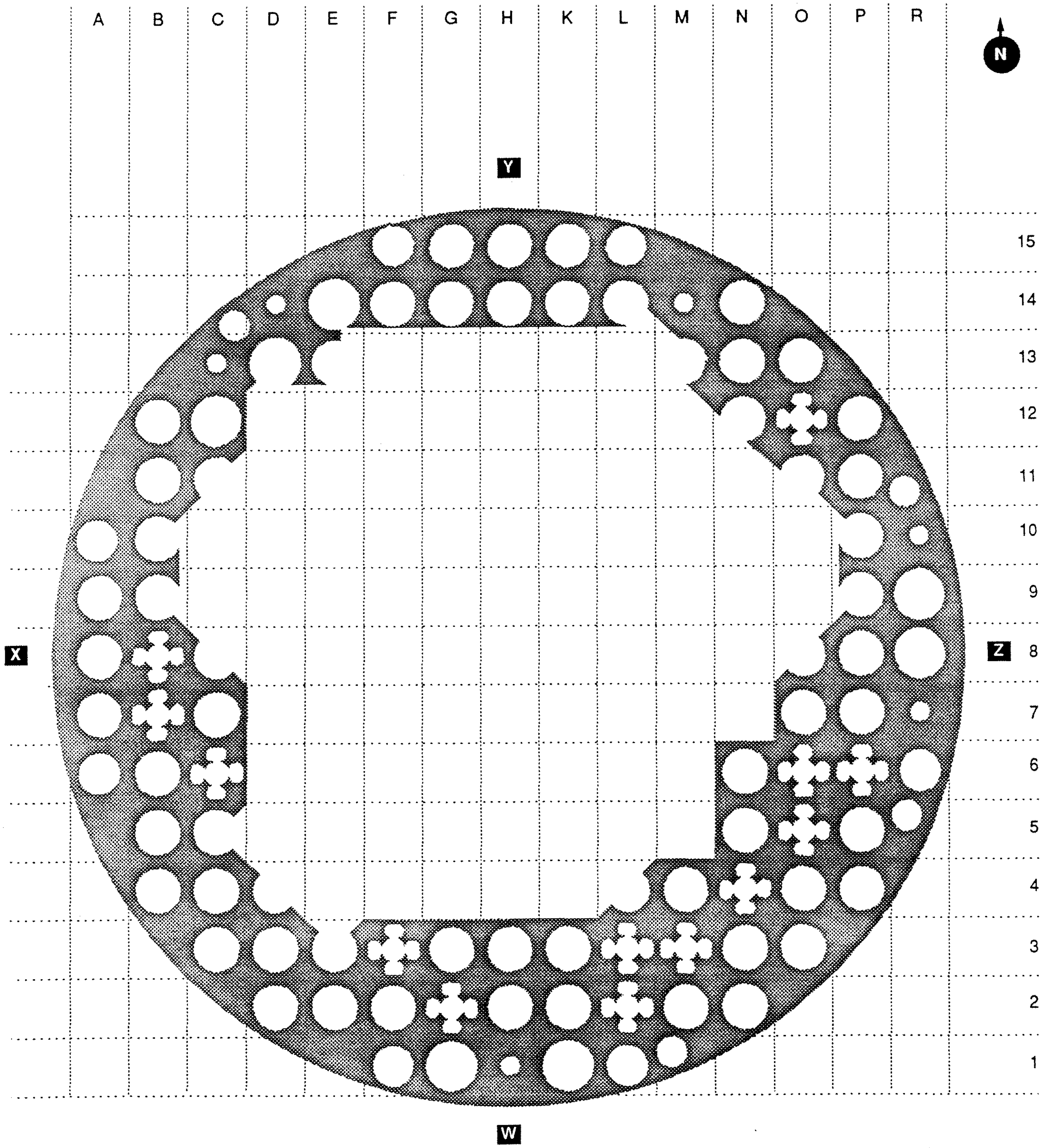
Support Post

FIGURE 5-23



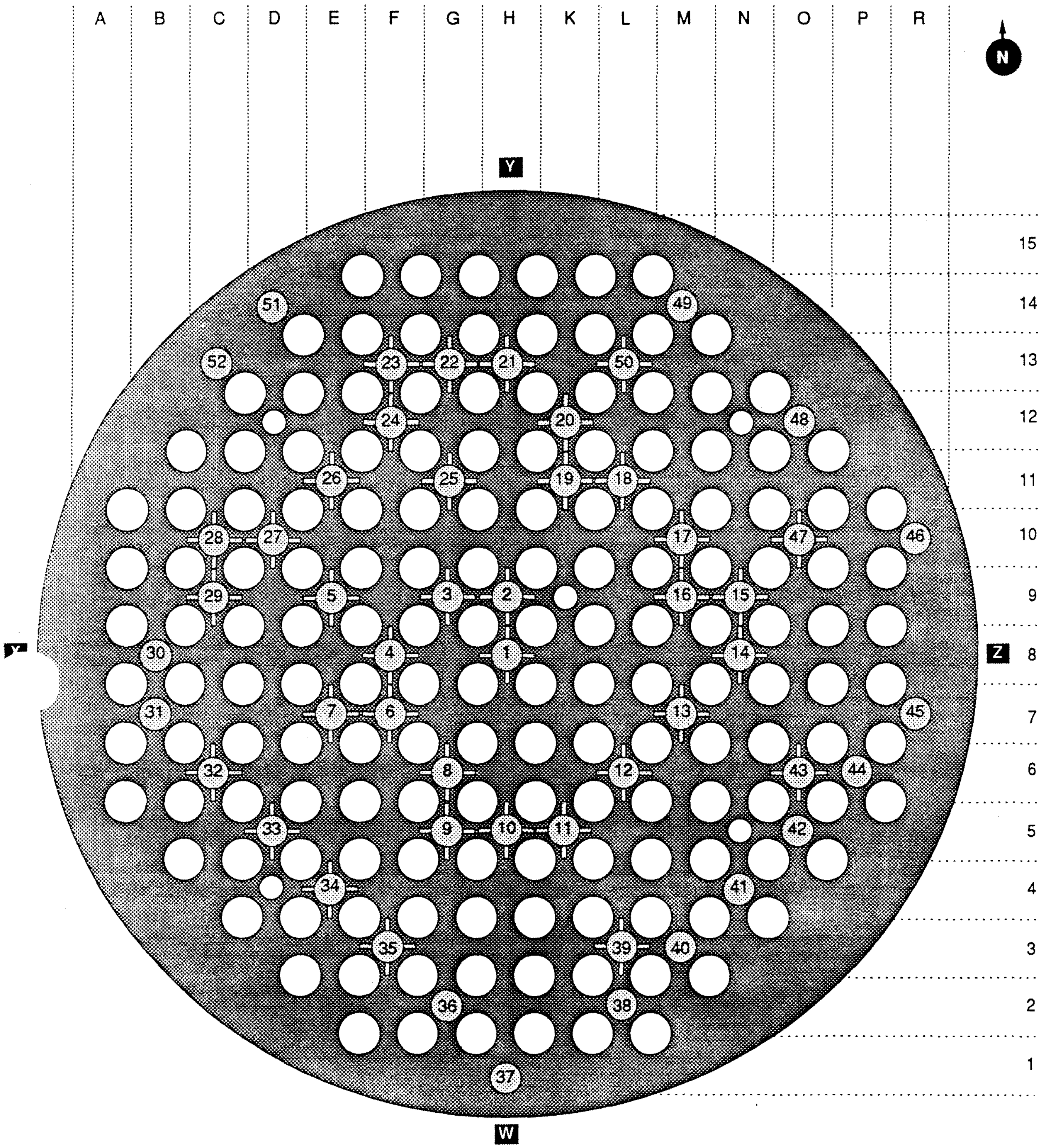
Incore Guide Support Plate

FIGURE 5-24



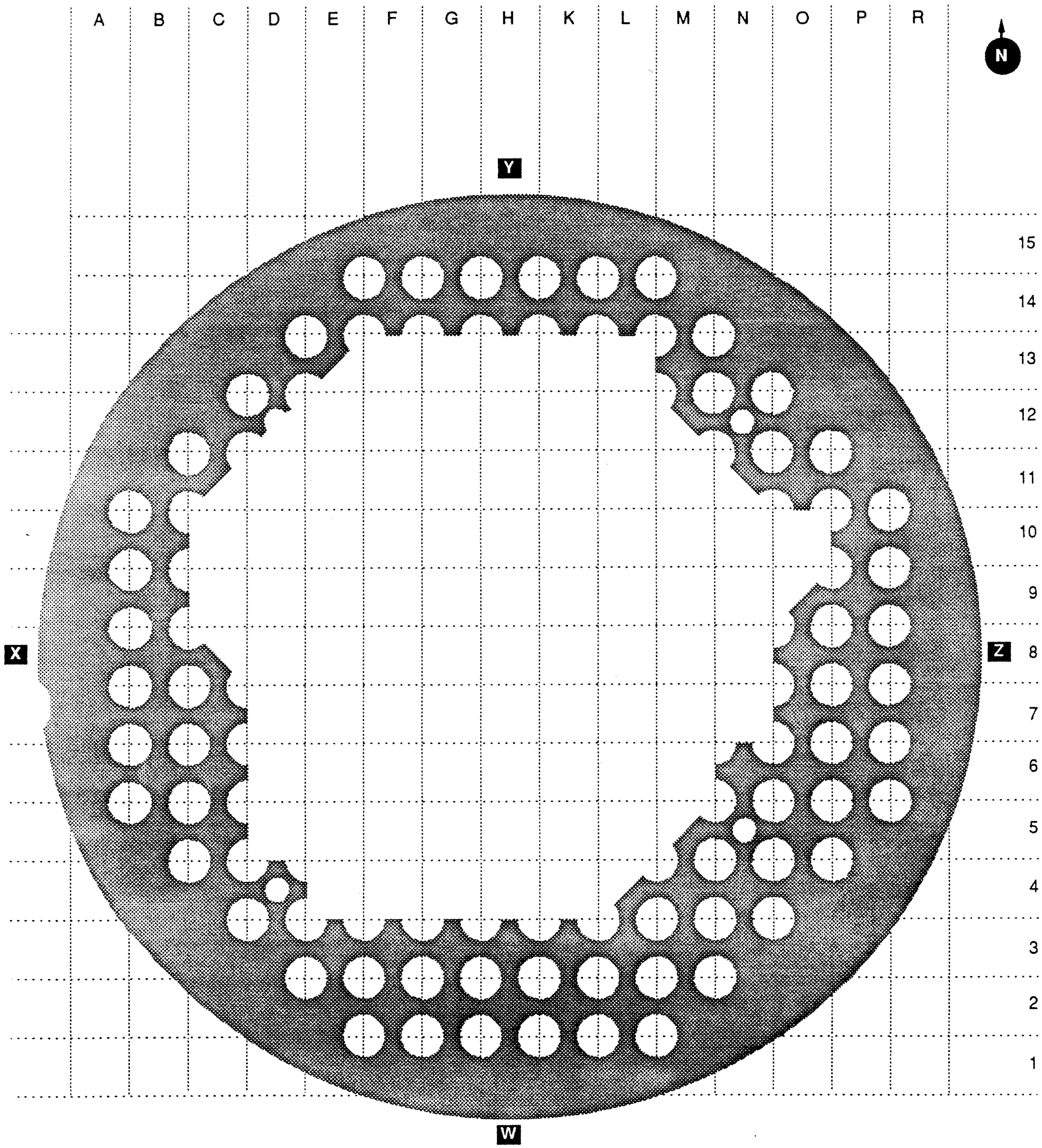
Incore Guide Support Plate After Cuts

FIGURE 5-25



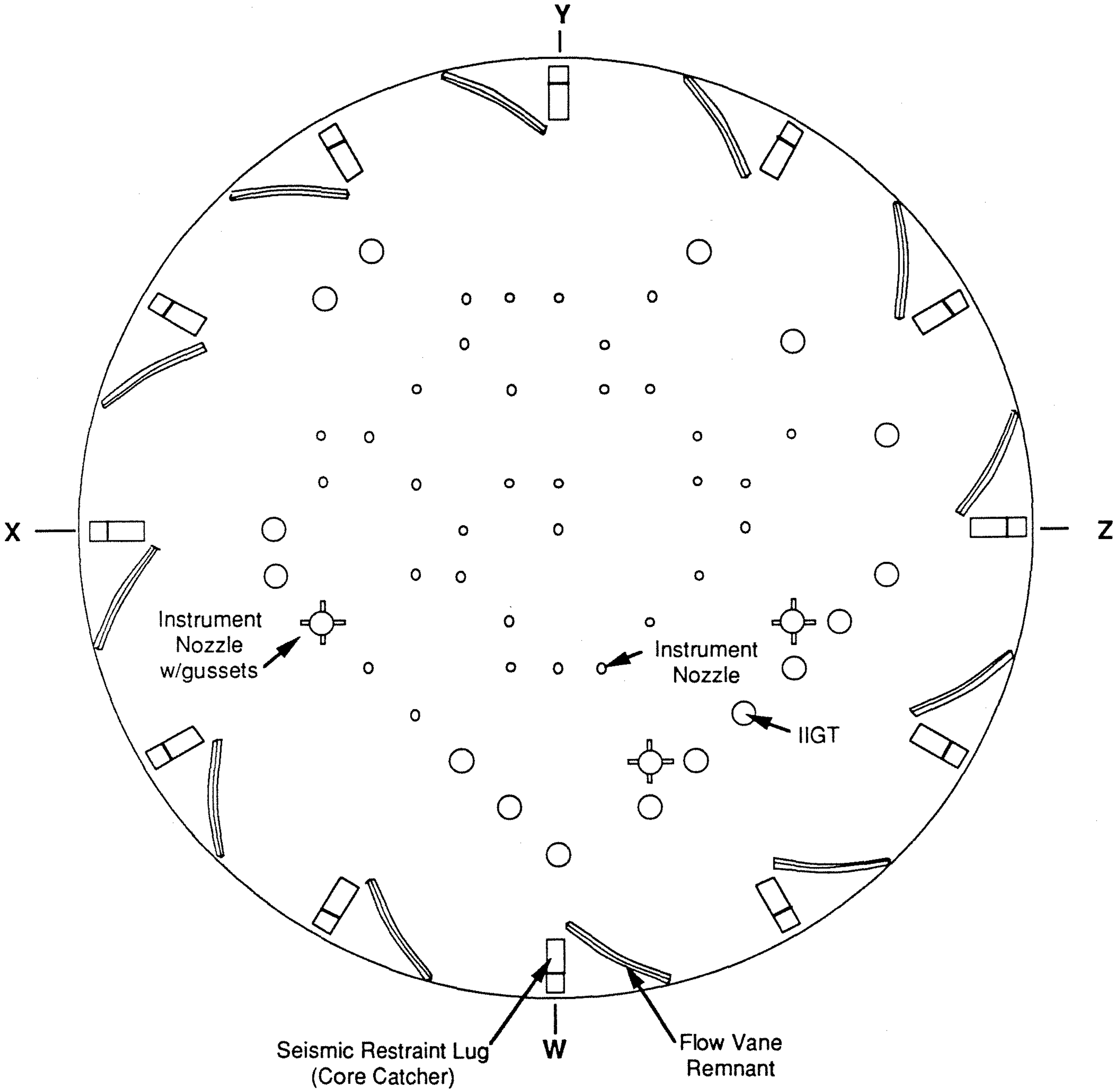
Flow Distributor

FIGURE 5-26



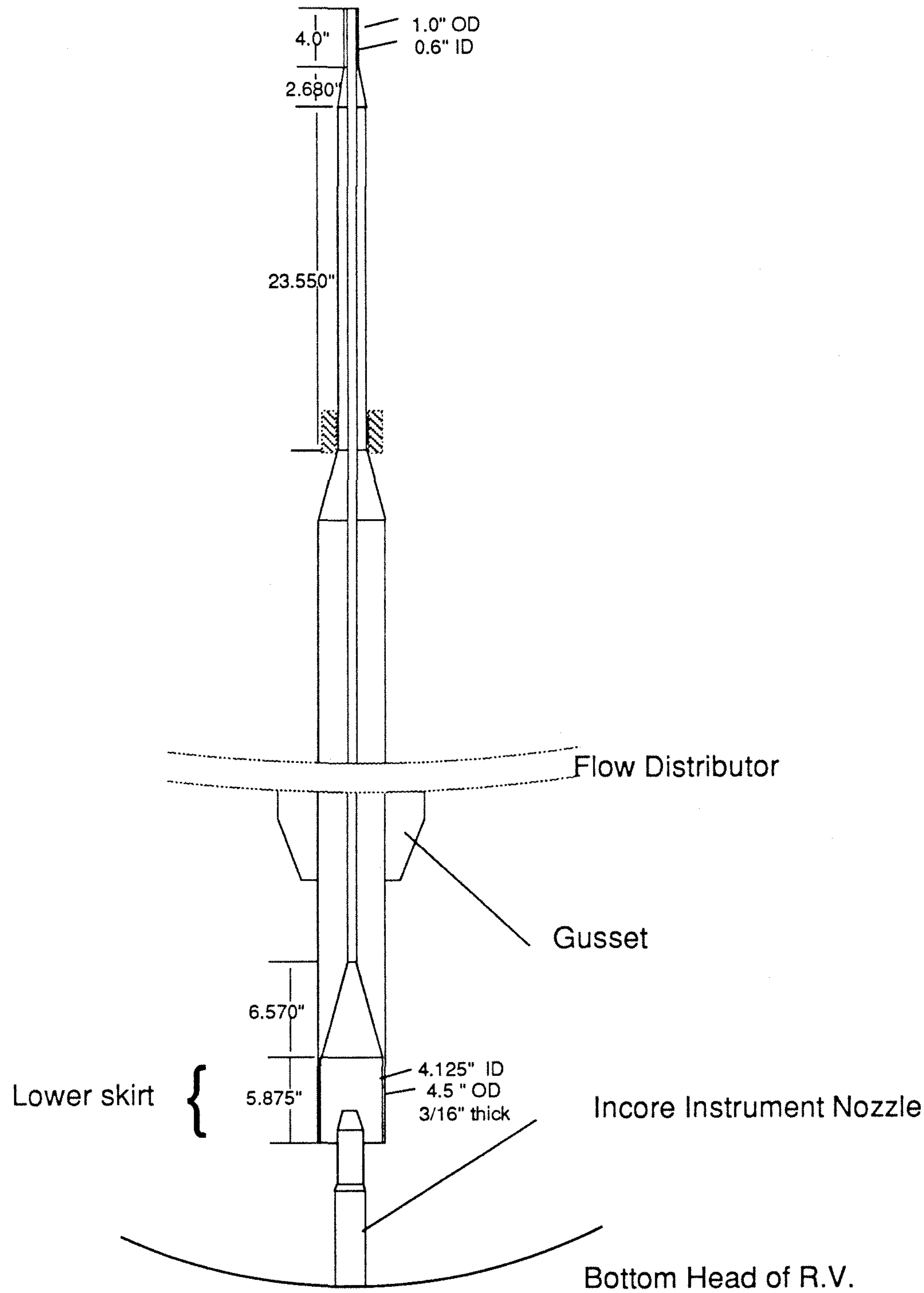
Flow Distributor After Cuts

FIGURE 5-27



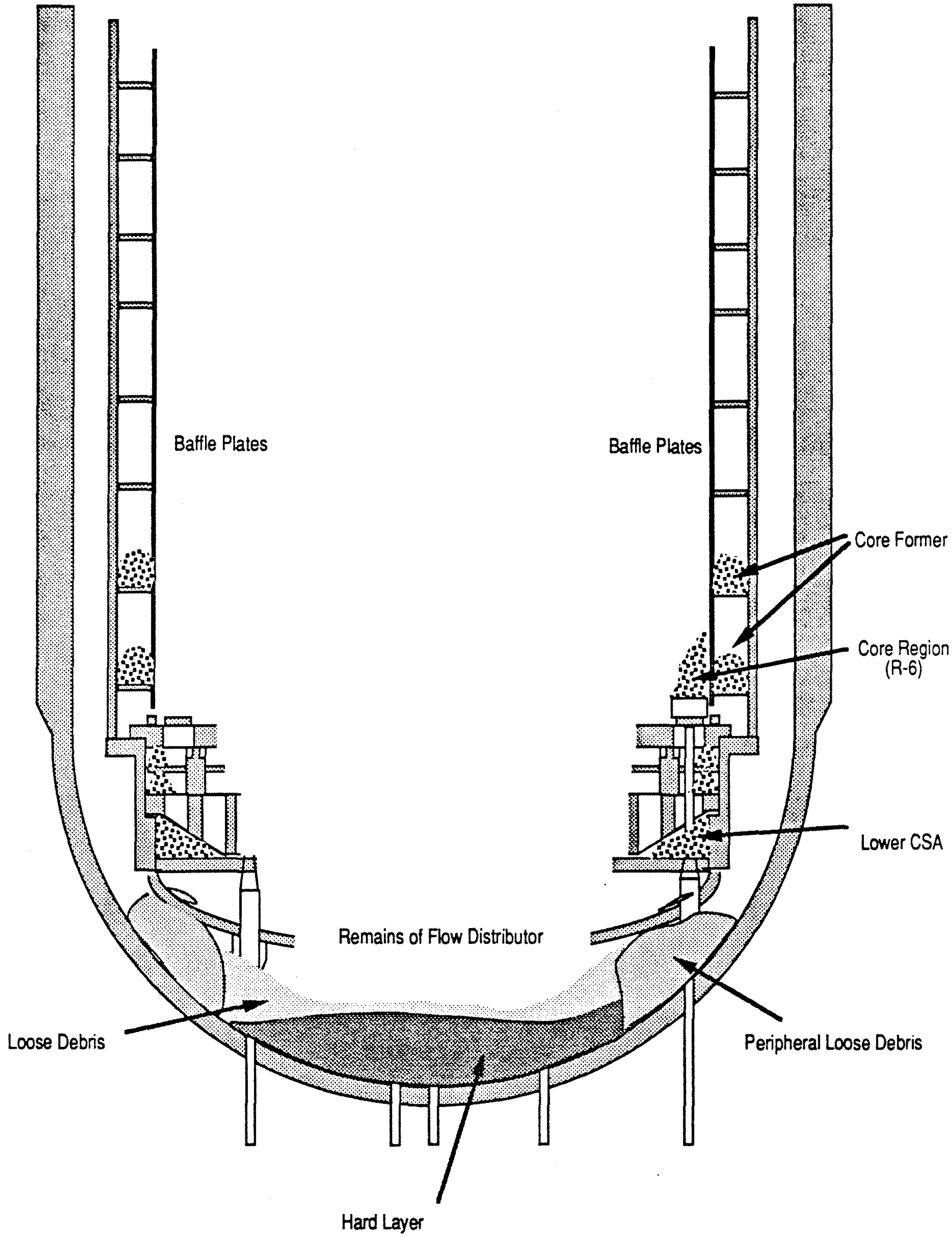
Bottom Head Arrangement

FIGURE 5-28



Incore Nozzle and Guide Tube Arrangement

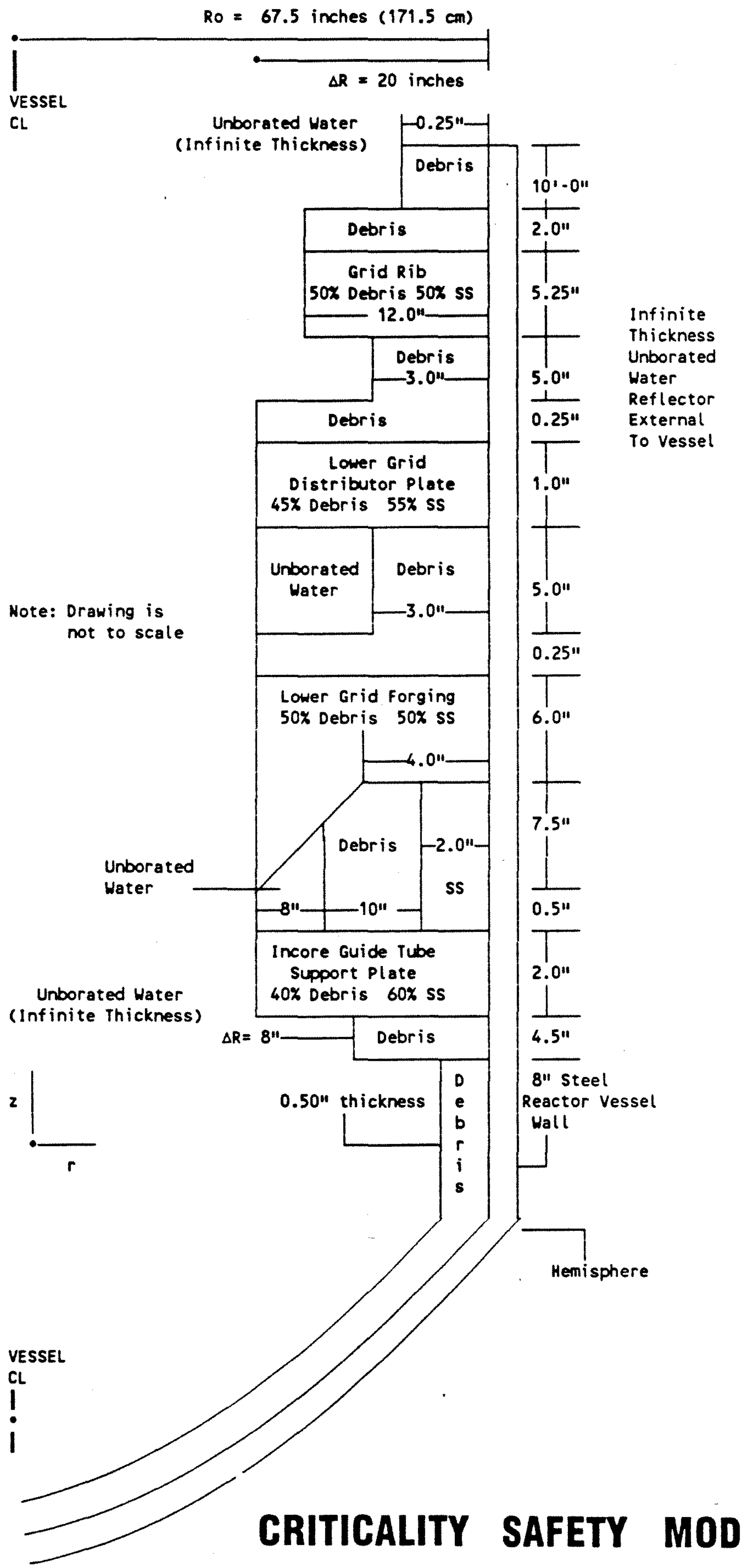
FIGURE 5-29



TMI-2 MATERIAL AT THE BOTTOM OF THE REACTOR VESSEL

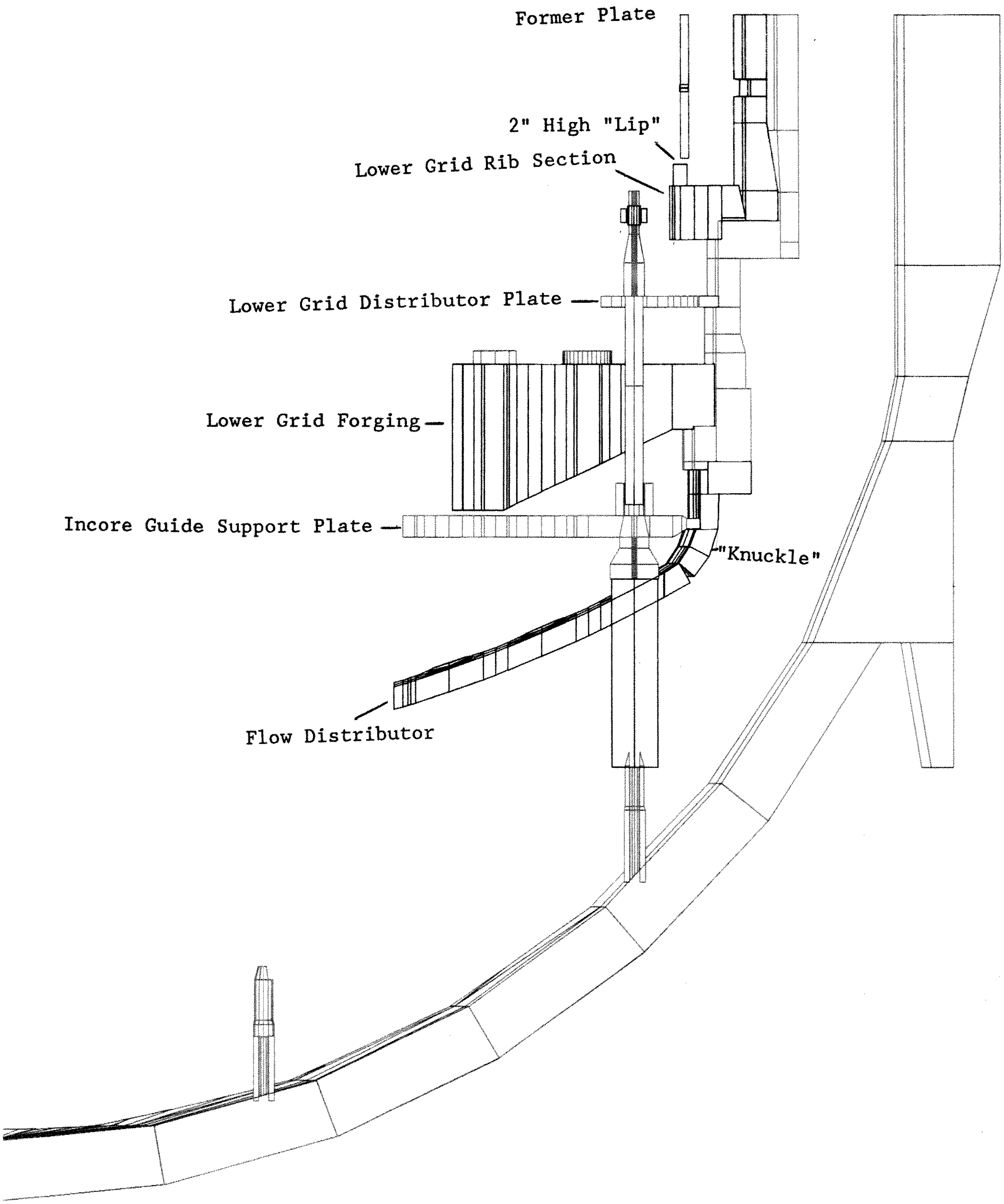
4/4/89

Figure 5-30



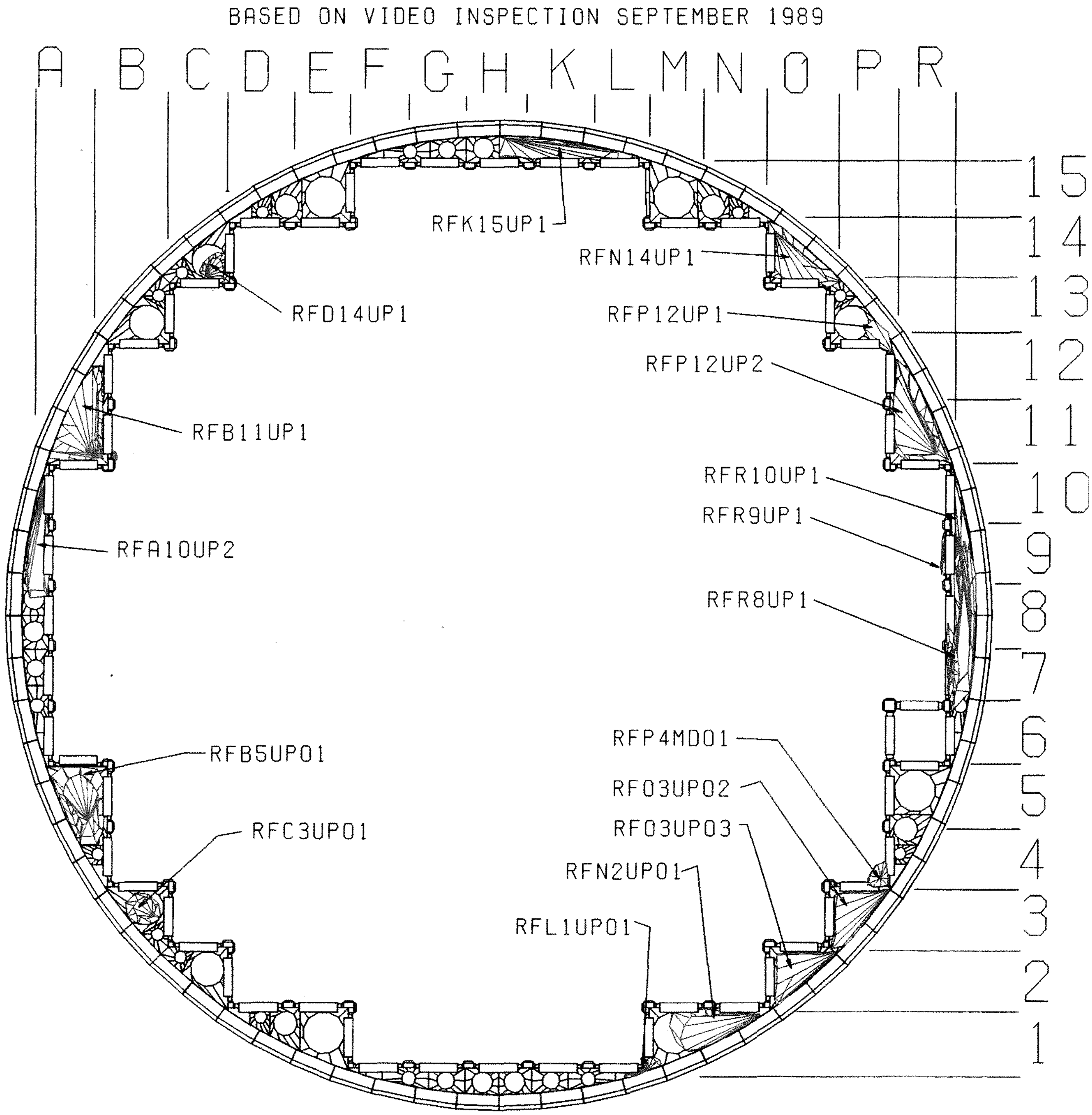
CRITICALITY SAFETY MODEL

Figure 5-31



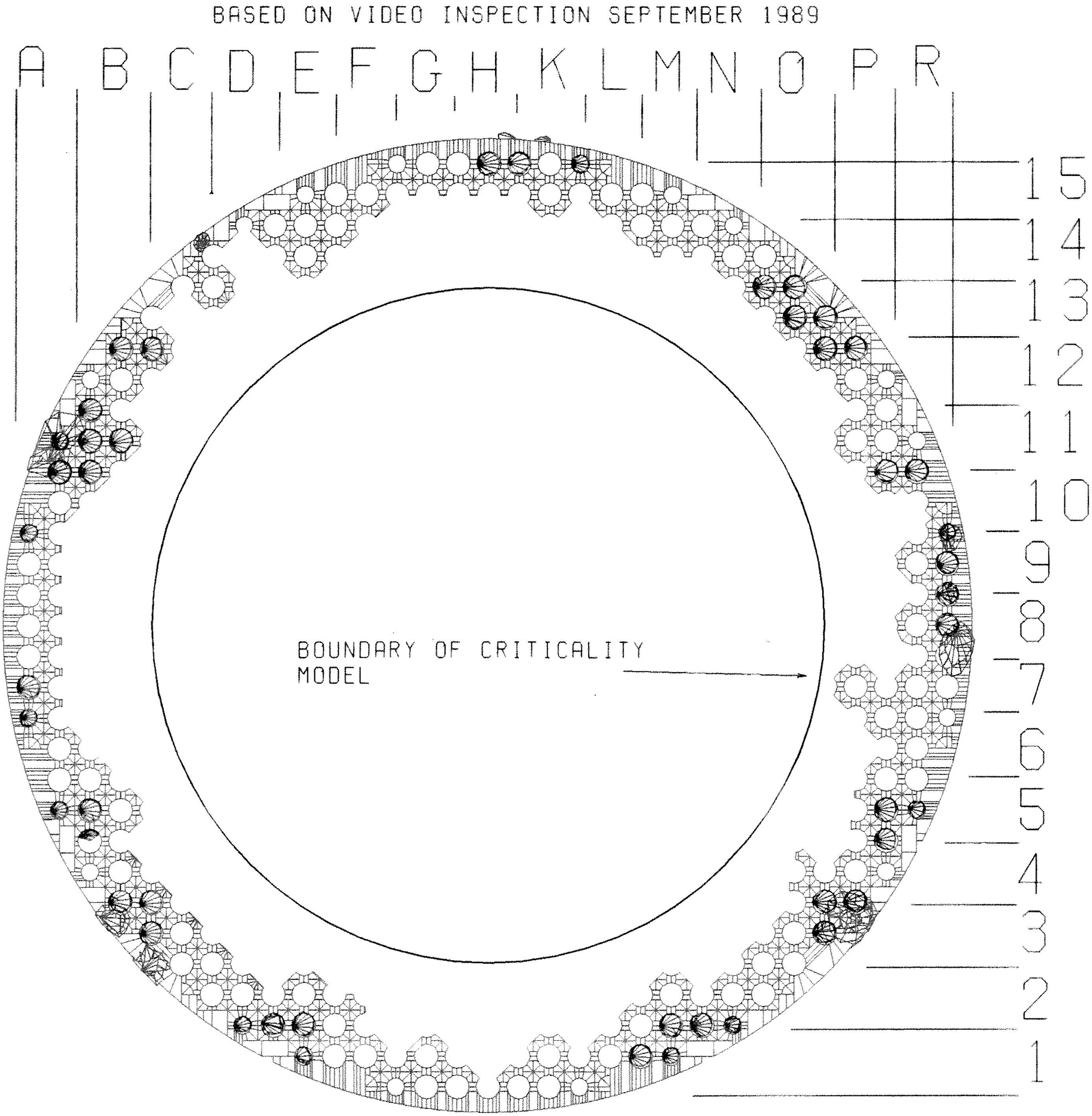
LOWER CORE SUPPORT ASSEMBLY

Figure 5-32



REMAINING LOWER GRID RIB SECTION

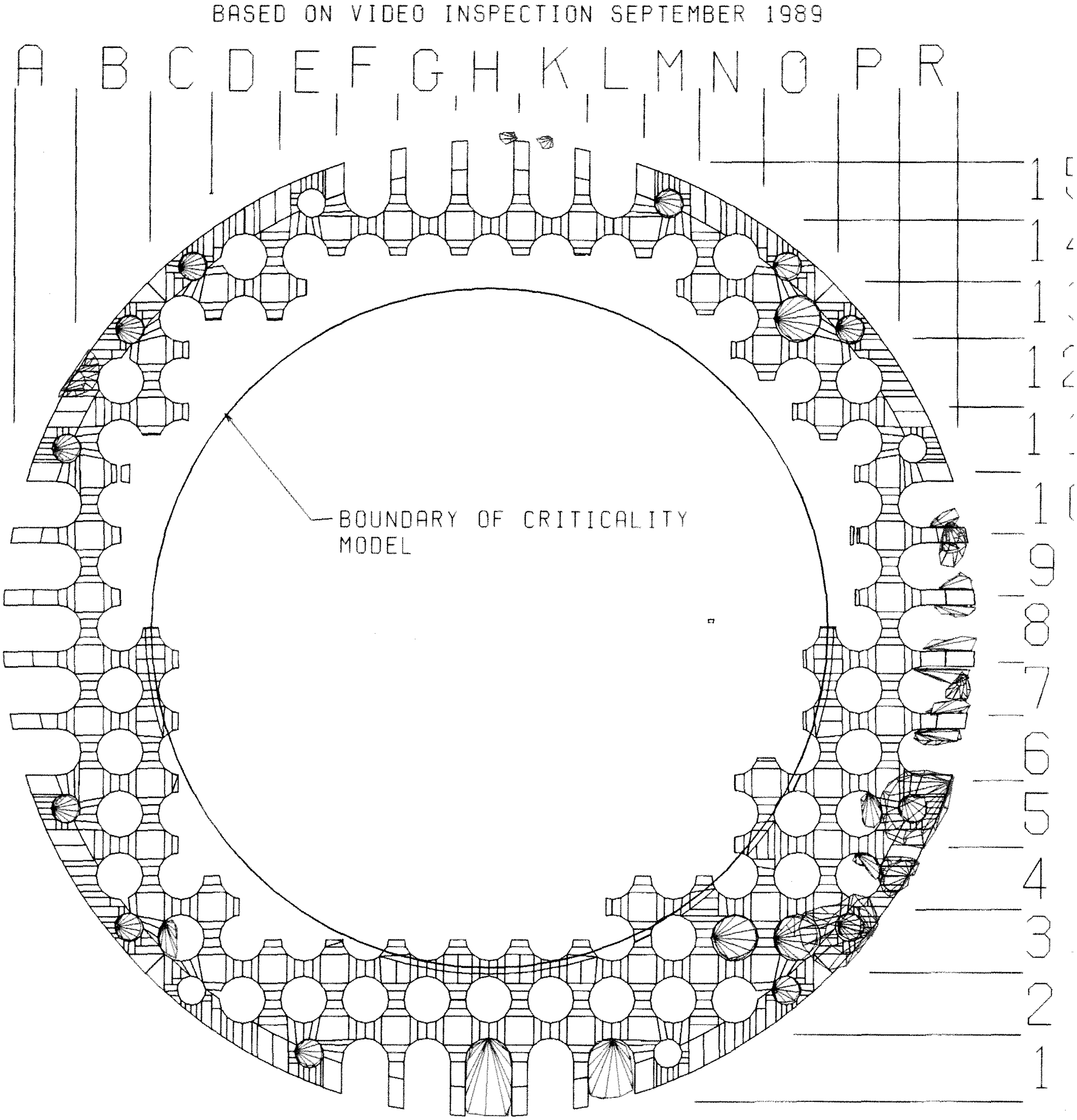
Figure 5-33



REMAINING FLOW DISTRIBUTOR PLATE

NOTE: FUEL OBSERVED FROM VIDEO INSPECTION DISPLAYED IN RED.
SUSPECTED FUEL LOCATION DISPLAYED IN GREEN.

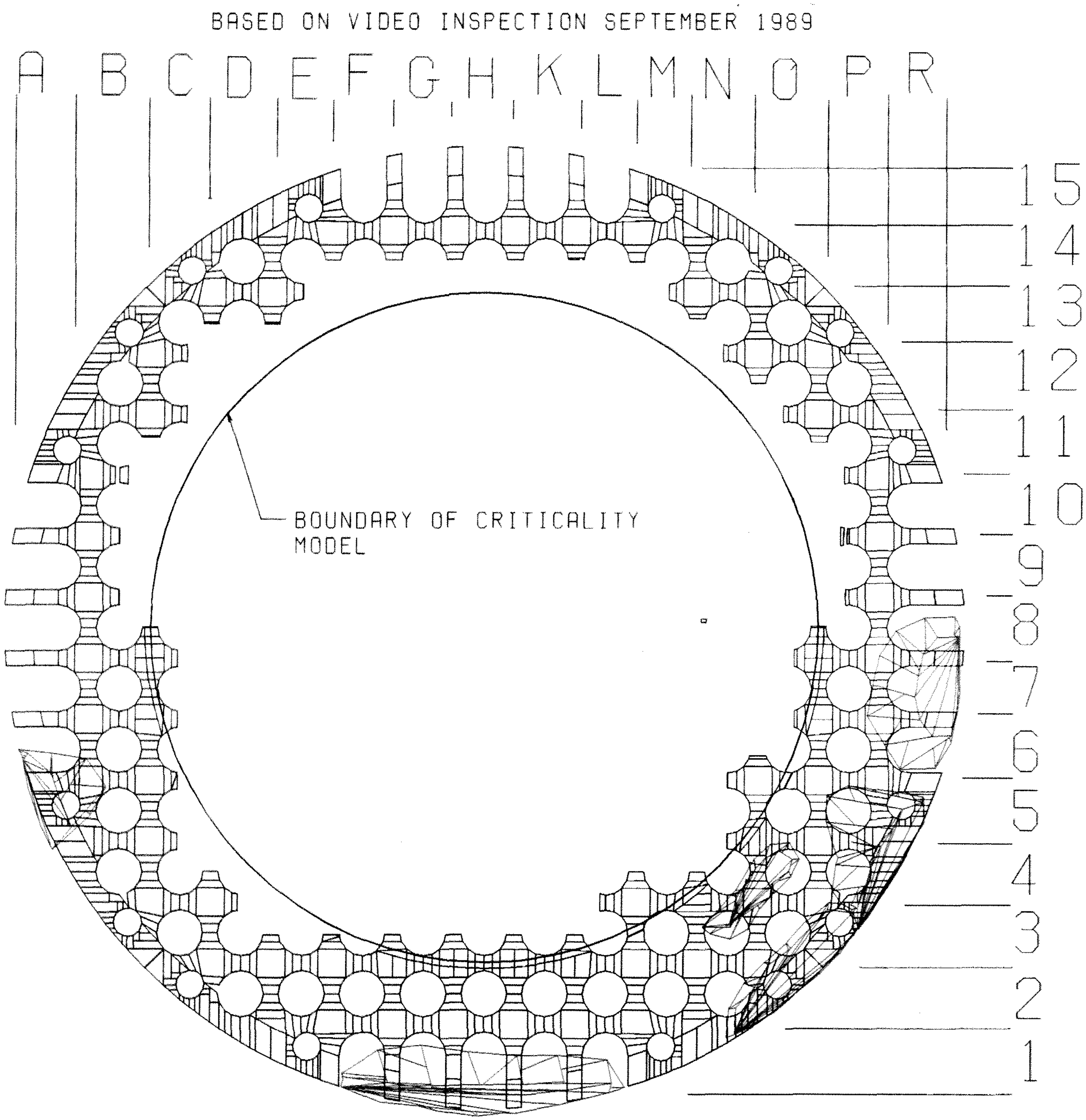
Figure 5-34



REMAINING LOWER GRID FORGING

NOTE: FUEL OBSERVED FROM VIDEO INSPECTION DISPLAYED IN RED ARE LOCATED ABOVE PLATE. SUSPECTED FUEL LOCATIONS ARE DISPLAYED IN GREEN AND ARE LOCATED IN FLOW HOLES OF PLATE.

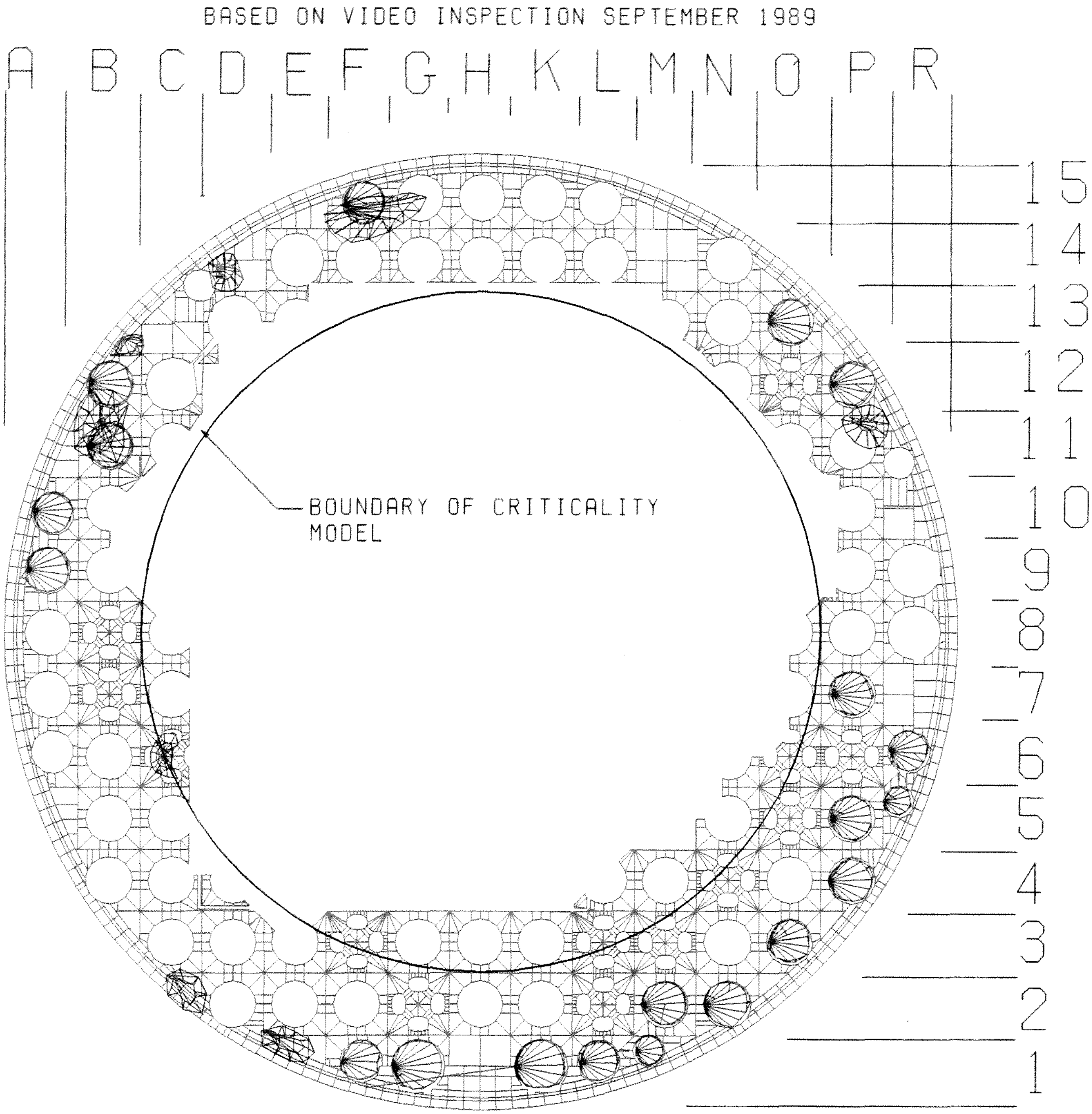
Figure 5-35



REMAINING LOWER GRID FORGING

NOTE: SUSPECTED FUEL LOCATIONS DISPLAYED IN GREEN.
SUSPECTED FUEL IS LOCATED IN "V" OF FORGING.

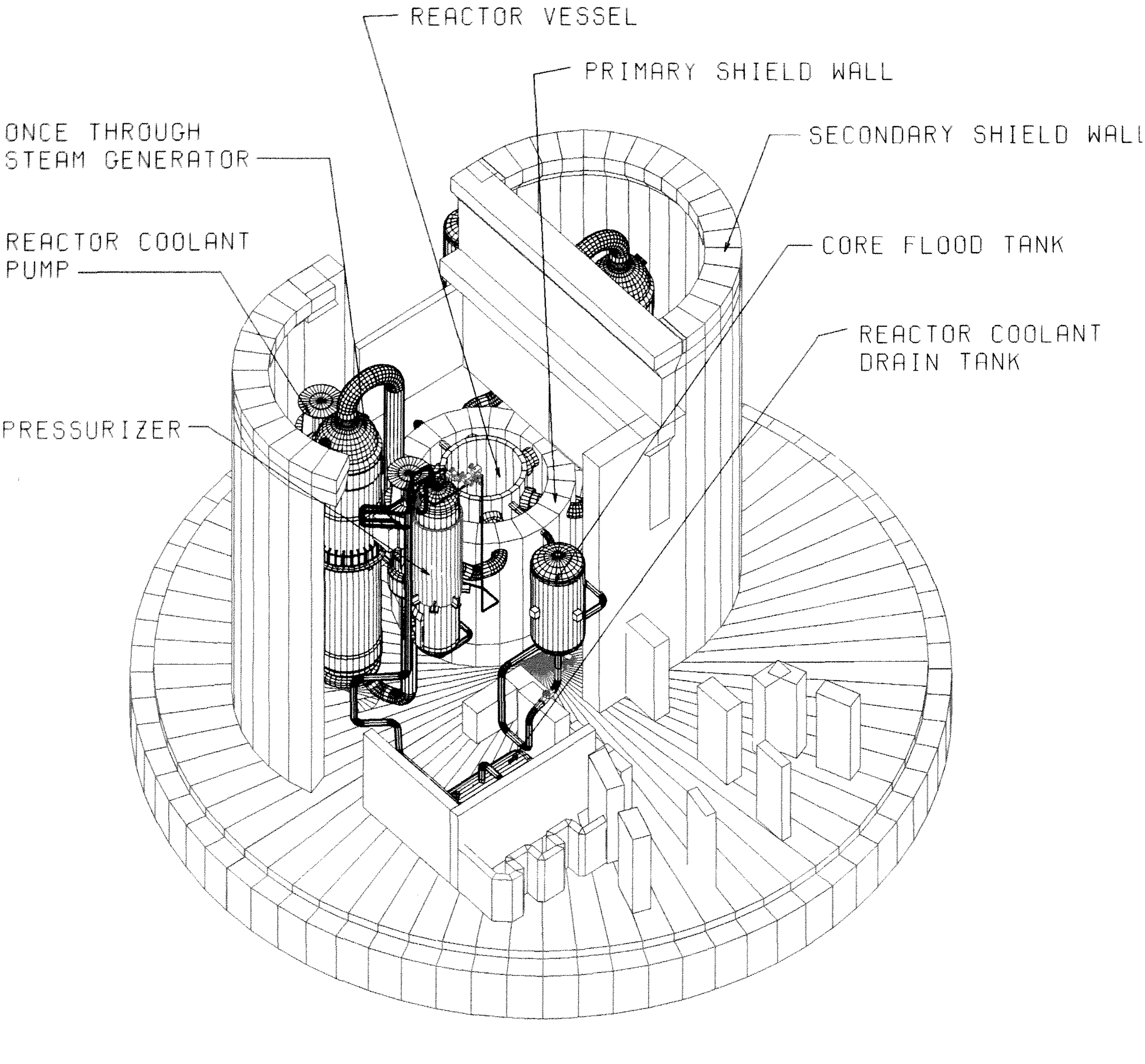
Figure 5-36



REMAINING INCORE GUIDE SUPPORT PLATE

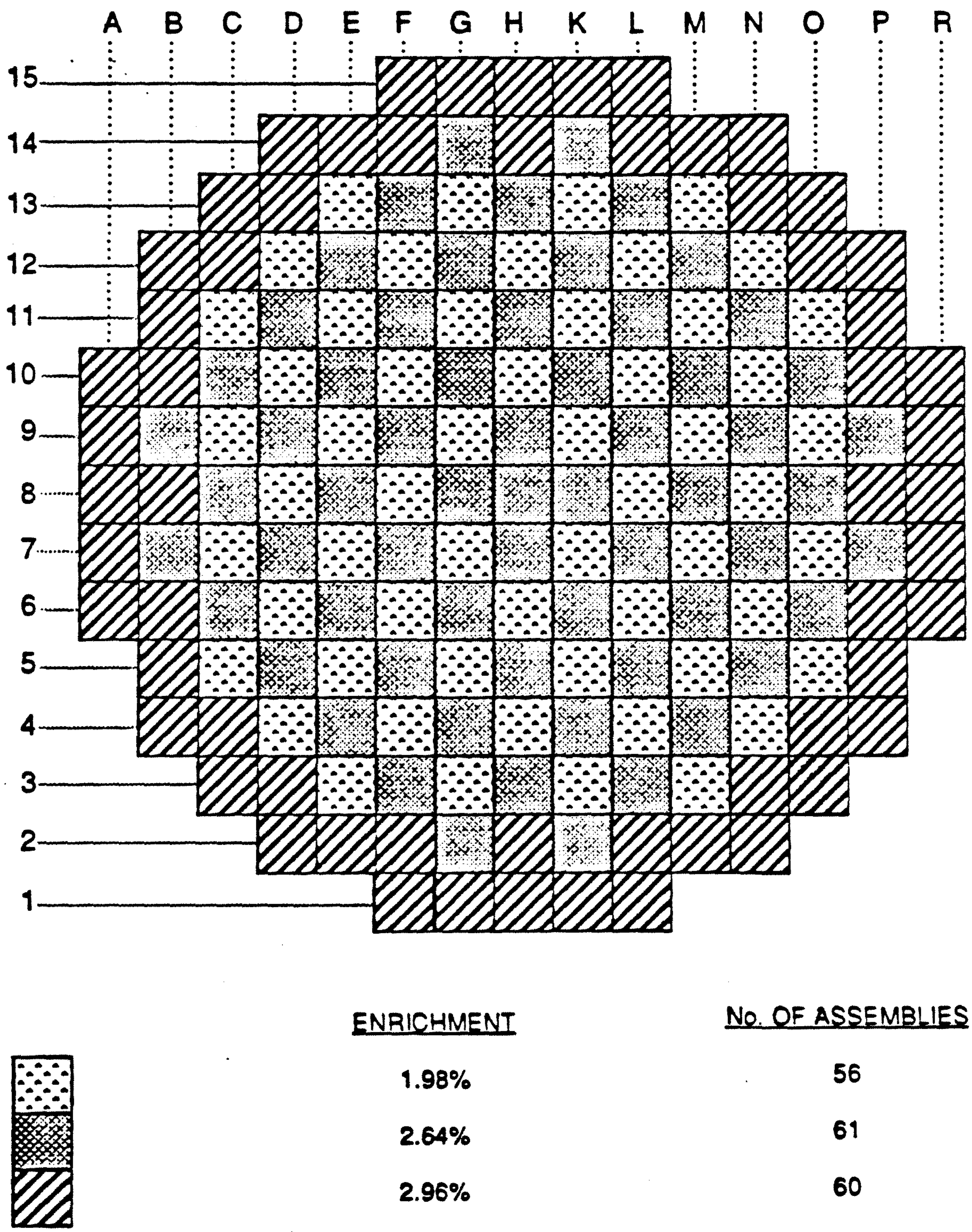
NOTE: FUEL OBSERVED FROM VIDEO INSPECTION DISPLAYED IN RED.
SUSPECTED FUEL LOCATIONS DISPLAYED IN GREEN.

Figure 5-37



THREE MILE ISLAND UNIT-2
REACTOR BUILDING

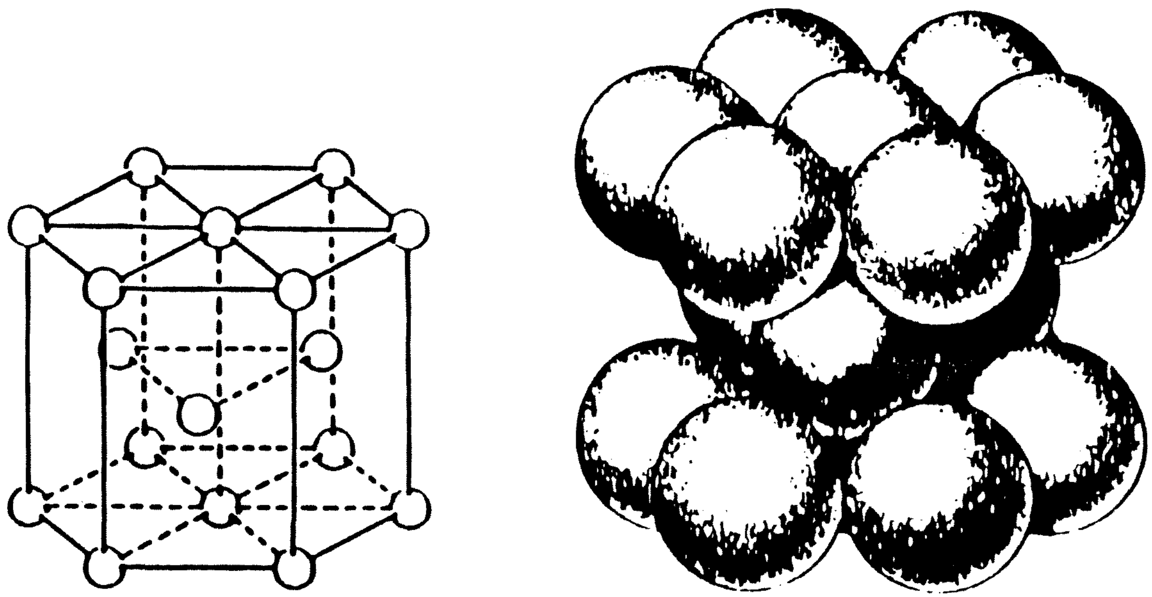
Figure 5-38



ENRICHMENT	No. OF ASSEMBLIES
1.98%	56
2.64%	61
2.96%	60

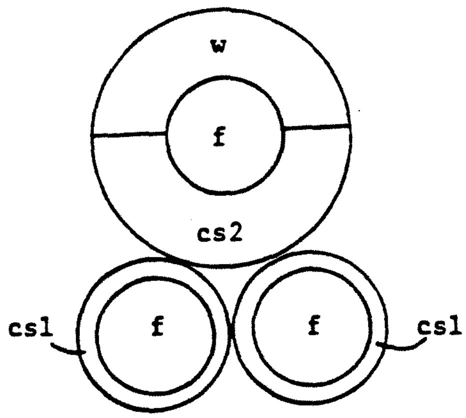
TMI-2 CORE ENRICHMENT PATTERN

Figure 5-39



FUEL DEBRIS LATTICE STRUCTURE

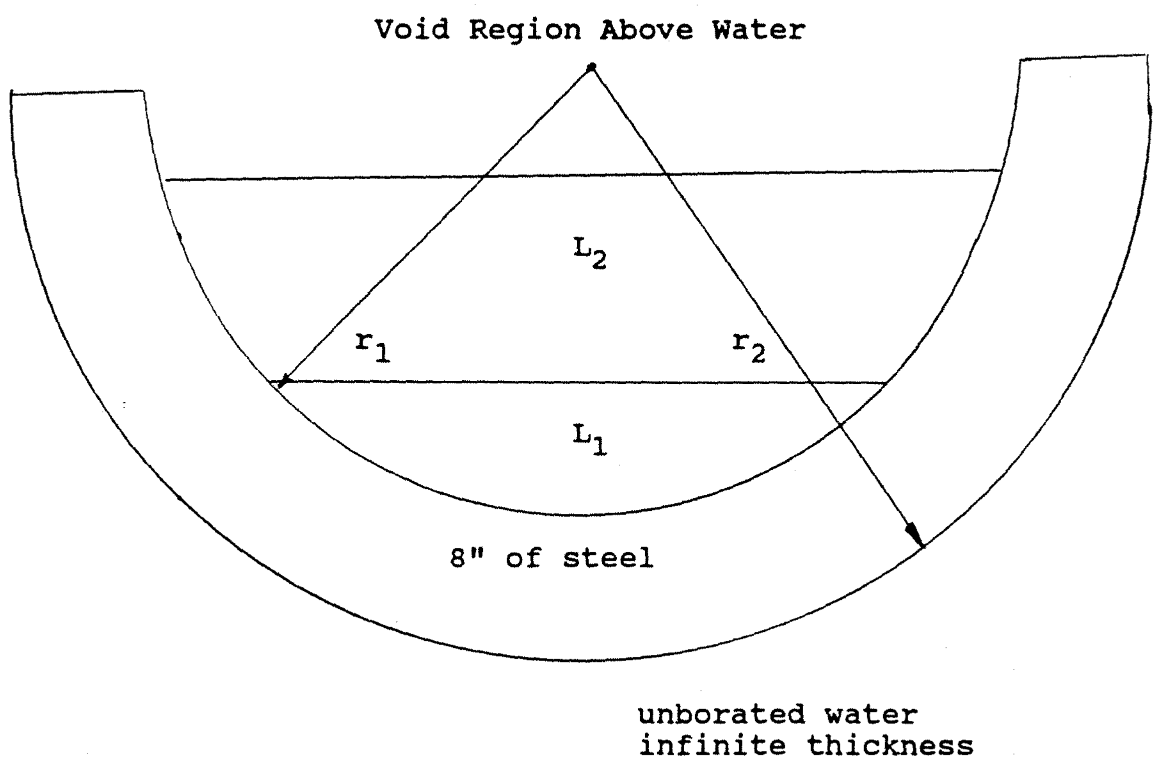
Figure 5-40



f = 120 kg fuel optimally moderated
cs1 = 2.75" carbon steel
cs2 = 8" carbon steel
w = 8" water

NEUTRON COUPLING MODEL

Figure 5-41



r_1 = 217.7 cm r_2 = 238.0 cm

L_1 = region containing 500 kg of core debris and unborated water

L_2 = region containing ~500 gallons of unborated water

BOTTOM HEAD FUEL MODEL

Figure 5-42

6.0 ASSESSMENT OF MAJOR RESIDUAL FUEL DEPOSITS

6.1 Introduction

Previous sections of this report describe defueling objectives and guidelines, core debris transport and the condition of the TMI-2 plant as a result of the accident, methods used to locate and quantify residual fuel, fuel removal activities undertaken during TMI-2 cleanup, and the characterization of residual fuel by quantity and location within TMI-2. Also included was a criticality assessment which demonstrates that subcriticality is ensured for all systems, structures, and components within the TMI-2 facility. Collectively, these sections demonstrate that a safe fuel condition has been achieved at TMI-2. Inadvertent criticality has been precluded.

This section provides a discussion of the major residual fuel deposits (i.e., UO_2) located within the TMI-2 facility with the intent of evaluating and demonstrating that reasonable and practical methods were considered and used to access and remove this fuel. The defueling operational objectives, defined in Section 1.0, established "reasonably achievable" as the removal of accessible fuel material utilizing technically practical methods. Included within these operational objectives are factors which include ALARA occupational dose considerations, accepted margins of health and safety to the general public, use of available technology, and overall feasibility based on schedule and resource impacts. In order to demonstrate attainment of the "reasonably achievable" objective, other defueling alternatives were evaluated for probability of success. In general, the alternative defueling methods considered were based on present-day technology; their applications were developed from defueling experience.

The TMI-2 cleanup effort has spanned ten years with a collective manpower effort of over 3.6 million person-hrs to complete. Total fuel removal represents approximately 99% of the original core loading. For comparative purposes, the unit cost and occupational exposure impacts associated with fuel removal through 1989 are approximately \$1900/kg and 0.02 person-rem/kg, respectively. All additional fuel removal activities identified have been determined to exceed these costs and person-rem impacts by orders of magnitude.

As presented in Section 4.0, a variety of fuel removal methods have been evaluated and used where appropriate. The major quantity of residual fuel remaining at TMI-2 is relatively small and physically configured such that it is fixed, isolated, and not susceptible to relocation. Residual quantities of fuel exist largely because inaccessibility or material characteristics (e.g., resolidified mass, thin films, fine dust material) make removal extremely difficult. The initial planning for defueling considered a broad range of defueling alternatives, such as grit and sandblasting, explosive charges, chemical decontamination, uranium oxide dissolution, metallic shredding, arc sawing, metal disintegration, mechanical shearing, lasers, and ultrasonic disintegration. For the most part, these alternatives were deemed impractical and are viewed as less applicable now because the fuel remaining is deposited in a few small, hard to access spaces and poses no safety concern. In general, removal of the remaining fuel would require more tedious, labor intensive efforts with an attendant occupational exposure and unique techniques beyond those considered, including more abrasive cleaning, higher pressure water erosion, chemical cleaning, and component removal and/or disassembly of the primary system. Additionally, these unique techniques and material requirements would create radioactive waste forms and packages which are not amenable to accepted disposal options and, therefore, could require extended onsite storage or further processing. In summary, the apparent impact of additional defueling is unwarranted when assessed in the context of occupational doses (ALARA), resource commitment, and the lack of a significant benefit to the health and safety of the public. The following sections present the basis for consideration of further defueling and the conclusion that no additional defueling is warranted.

6.2 Assessment Criteria

The areas considered in the assessment initially included those locations which contain residual fuel quantities exceeding 25% of the SFML (i.e., >35 kg); however, based on that threshold, all residual fuel locations, except for the RV and "B" OTSG tubesheet would be eliminated from the assessment. Therefore, a more encompassing and descriptive criterion of 10% of the SFML was selected. This definition of a significant residual fuel deposit clearly provides a conservative threshold above which interest may be expressed, and below which clearly no additional defueling activity can be justified based on the extensive defueling actions completed.

As presented in Section 5.0, approximately 99% of the fuel was removed, which supports a conclusion that the TMI-2 facility was defueled to the extent reasonably achievable. It is also clear that the collective quantity of fuel remaining at TMI-2 exceeds the SFML of 140 kg. However, considering the specific locations and conditions of residual fuel, all isolated volumes contain much less than the SFML except the RV. In addition, only a few locations contain residual fuel quantities which exceed 10% of the SFML. The following areas were considered in this assessment: AFHB [none identified (no area contains >10% of SFML]; RB ("A" D-ring); RCS ("B" OTSG tubesheet and cold leg 2A); and the RV.

For each of these locations, the following elements are discussed, where applicable, to demonstrate the acceptability of the current end-state condition:

- Quantity of residual fuel
- Potential options for additional fuel removal
- Schedule impacts for fuel removal
- Occupational exposure impacts
- Waste generation and resultant disposal impacts

6.3 Auxiliary and Fuel Handling Buildings

As discussed in Section 5.1, the total quantity of fuel remaining in the AFHB is less than 17 kg with the largest single discrete volume containing less than 5 kg (RCBTs). Therefore, no additional assessment of the AFHB is considered appropriate.

6.4 Reactor Building

One location of residual fuel in the RB, but not within the RV or RCS, that contains more than 14 kg is the "A" D-ring. As discussed in Section 5.2.5, 14 sections of the flow distributor containing 33 IIGTs are bagged and stored in the "A" D-ring. The total amount of residual fuel contained therein was estimated to be 24 kg based on measurements performed on 13 sections containing 30 IIGTs.

The 14 sections of the flow distributor stored in the "A" D-ring are those sections which have IIGTs attached. These sections are physically configured such that they would not fit into defueling canisters or into the "A" CFT without additional cutting and sizing. Each section is comprised of a 30- to 60-cm (1- to 2-foot) rectangularly shaped piece of the flow distributor with up to four attached IIGTs of lengths up to 100 cm (40 inches) long. These sections were mechanically brushed, water-lanced, and inspected prior to removal from the RV. The majority of the residual fuel associated with these sections was determined to be tightly compacted inside the 1.6-cm (0.6-inch) ID holes in the upper section of the IIGT which was not accessible to the brush tools.

The total amount of fuel in this area will be significantly reduced as part of an NRC/OECD research program designed to better understand the interaction of molten fuel materials with structural components. Special containers and shipping packages will be used to transport samples of the IIGT remnants to selected laboratories for further analysis. Thus, the estimated 24 kg of fuel will be substantially reduced as part of this program and may be less than 10% of the SFML. Final fuel measurements will be provided as part of the separate SNM measurement program. With the completion of the research program, no additional defueling activity is deemed necessary for the "A" D-ring storage location and its contents.

6.5 Reactor Coolant System

Two locations within the RCS contain quantities of residual fuel in excess of 14 kg and warrant discussion here. They are the "B" OTSG upper tubesheet and the 2A cold leg.

6.5.1 "B" Once-Through Steam Generator Upper Tubesheet

Although visual inspections and video records show no significant quantities of residual material, results of radiation measurements indicate that the "B" OTSG upper tubesheet contains 36 kg of fuel (Section 5.3.3.1). This remaining material bridges some of the gaps between tube stubs on the tubesheet. Several attempts to dry vacuum this area after completion of mechanical defueling were unsuccessful. Attempts to obtain a scrape sample also were unsuccessful. The residual fuel and associated fission products produce a radiation field inside the OTSG upper head of approximately 100 to 200 R/hr. This very high radiation field is the chief impediment to further defueling of this area because it prohibits hands-on access. The results of dynamic defueling techniques and visual examination indicate that the material is tightly adherent; increasingly dynamic means of defueling would be required to remove additional fuel material. The likelihood of material transport or its contribution to future in-containment airborne radioactivity are judged to be quite low because this residual fuel deposit has resisted extensive defueling attempts. Criticality safety has been demonstrated because the quantity of concern is a discrete volume which is much less than the SFML. Therefore, it would be impractical and inconsistent with the principles of ALARA to expose workers to these high radiation fields for extended periods when no significant risk exists and no proportionate increase in the margin of facility safety will be achieved. However, three potential methods of removing additional fuel were evaluated for purposes of this discussion.

a. High Pressure Flush With Tubes Plugged

Equipment at TMI-2 could be modified to supply 10,000 psi flush water. However, plugging of the more than 15,000 tubes would be required to prevent fuel transport to the OTSG lower head; a labor-intensive undertaking. It is estimated that it would require as much as 6 months and robotic techniques to complete this activity due to the high radiation levels inside the upper head and the high (250 to 500 mr/hr) radiation levels external to the OTSG head where workers would be located. Further, it is estimated that it could require between 25 and 50 person-rem of radiation exposure to install and operate the remote-tube-plugging system before defueling operations could begin. A water cleanup recirculation system would be used to remove accumulated water and loose debris and filter it. Based on engineering judgment, this operation was estimated to be approximately 80% effective (i.e., 28 kg of fuel removed). The removal costs could be as high as $125,000/kg with an exposure rate of 2 person-rem/kg.

b. Abrasive Cleaning

Abrasive cleaning of the tubesheet appears feasible. Fuel would be removed by an abrasive-erosion process similar to sandblasting. A high velocity vacuum system could pick up a majority of the grit and dislodged fuel. Some fuel fines and abrasive grit would become airborne. A second, low velocity filtration system could collect the grit and airborne radioactive material. Some fuel fines and abrasive material would be expected to fall through the tubes to the bottom head of the OTSG if the tubes were not plugged. Nevertheless, this process would be expected to capture 60 to 70% (approximately 20 kg) of the fuel on the tubesheet. Since the tubes would not need to be plugged, the required time and expense would be significantly reduced. Estimates of the cost are approximately $12,000/kg. However, operating the grit blasting equipment and recovery processes could produce exposures as high as 3 person-rem/kg.

c. Wire Brushing

A wire brush attached to a rotary drill could be substituted for a grit blasting system to remove the fuel. This technique has the advantage of not using airborne grit material which must be recaptured. The operating time would be significantly greater because the mechanical process of brushing around each of more than 15,000 tube stubs would be a slow and tedious activity. It is expected that a wire brushing system coupled with a high flow vacuum system could recover 75% (approximately 25 kg) of the fuel and could cost approximately $12,500/kg of recovered fuel with an estimated exposure of approximately 4 person-rem/kg.

Because of the extensive effort already undertaken to defuel the "B" OTSG tubesheet, the remaining small amount of residual fuel is not considered readily transportable or a potential significant contributor to airborne contamination levels. Hence, none of the above methods of additional fuel removal is considered practical or beneficial because the residual fuel poses essentially no risk. No further action is considered practical or warranted.

6.5.2 Reactor Coolant System 2A Cold Leg

As discussed in Section 5.3.5, a collective amount of approximately 56 kg of residual fuel is contained in the RCS cold legs. The largest single quantity in any one cold leg was estimated to be approximately 29 kg in the 2A cold leg. Each of the other three cold legs contains less than 10% of the SFML. The material is comprised of a finely dispersed sediment distributed uniformly on the lower section of the RCS piping and in the cold leg nozzles at the RV interface.

Cleanup of the 2A cold leg piping between RC-P-2A and the RV could be accomplished by cutting a hole in the RV core barrel with the plasma arc torch to gain access to the 2A cold leg. The cut would be approximately 1 meter (40 inches) in diameter. Once an access hole is cut, three alternatives exist for defueling the cold leg: 1) deploy a manually manipulated vacuum system, similar to that used in the RV, to achieve the greatest coverage possible; 2) deploy a vacuum via a mini-submarine; and 3) deploy a vacuum/flushing tool on a pipe crawler. If the material cannot be vacuumed, it may be possible to relocate the material to the RV where the debris may be allowed to fall into the RV for subsequent removal.

Another option for removing fuel from the 2A cold leg involves cutting an access hole through the D-ring structures and hot tapping a penetration through the stainless steel pipe wall between the cold leg and the underside of RC-P-2A. A conceptual plan was developed but not pursued because of the labor intensive effort required within the RB and the need to acquire new and different types of equipment not previously used at TMI-2. Based on the relatively small quantity of fuel (<15% of SFML) at issue, this defueling option was determined not to be warranted.

Each of the alternatives considered for defueling the 2A cold leg would require an extensive level of effort with commensurate occupational exposure equivalent to several person-months of defueling in the RB. The residual fuel quantity in the cold legs of the RCS is not readily transportable even when the loops are drained. The fuel is isolated such that it would not contribute to unplanned airborne radioactivity releases to the RB nor the outside environment. Therefore, it is concluded that further efforts to defuel the 2A cold leg are not reasonable or warranted.

6.6 Reactor Vessel Fuel Deposits

Defueling operations have resulted in the cumulative removal of approximately 99% of the original TMI-2 fuel inventory. The remaining RV residual fuel principally resides as a tightly adherent or granular material in difficult to access locations or as a finely dispersed film of material deposited on surfaces of structural components within the RV. The quantity and location of the remaining fuel has been demonstrated to be subcritical with no active safety mechanism required to ensure long-term subcriticality control (see Section 5.5). Due to the apparent resistance of the residual fuel to extensive, dynamic defueling activities previously described, including the cutting and removal of the major RV internals, it appears reasonable to assume that the potential for significant core debris relocation or transport within or out of the RV is minimal.

This section discusses the major residual fuel quantities and locations within the RV and evaluates alternative defueling activities. For purposes of this evaluation, the post-defueling configuration of the RV is subdivided into seven major areas: 1) work platform region and suspended equipment; 2) downcomer region; 3) IIF region; 4) CSS region; 5) UCSA region; 6) LCSA region; and 7) bottom head region. Included in this discussion of alternative defueling concepts is a summary view of the final quantities of fuel, the extent of previous defueling efforts, and the basis for concluding that RV defueling has been completed to the extent reasonably achievable. Consistent with the previous approach in defining ex-vessel fuel deposits as significant, a threshold of <10% of the SFML (i.e., <14 kg) for an isolated or discrete volume is established as the criterion for consideration of further defueling alternatives.

6.6.1 Work Platform Region and Suspended Equipment

As described in Section 5.4.2.1, the work platform and associated equipment were determined to contain approximately 30 kg of residual fuel. These locations included the IVFS, CPS, pump module, and platform framework. Most of the fuel resides in the IVFS and CPS. These areas have restricted access and, therefore, make further removal of fuel impractical. The work platform and associated equipment will be left in place as a convenient storage location for these large components and also to provide ready access to the RV for further surveys and inspections as needed.

6.6.2 Downcomer Region

As described in Section 5.4.2.2, the downcomer region was estimated to contain approximately 120 kg of residual fuel in the annular gap consisting of loose, granular debris and approximately 60 kg of residual fuel of finely dispersed material residing on vertical and horizontal surfaces within the downcomer region.

The fuel material accumulated in the annular gap (see Figure 5-13) is a result of airlifting activities. The material is confined within very limited access. There exist only four top access locations where holes in the core barrel flange line up with gaps between the thermal shield blocks to permit access devices [2.5-cm (1-inch) maximum width]. Because of this restricted access, fuel removal from this gap is impractical.

A second entry location to this region is via thirty 1.9-cm (3/4-inch) diameter horizontal orifice holes near the bottom of the core barrel where air (or water) could be injected to fluff the debris out of the gap. This requires removing the suspended baffle plates to gain access. The amount that could be washed out is speculative but believed to be less than 10% [holes are approximately 50 cm (15 inches) apart at the bottom of the gap]. The debris washed out will redeposit throughout the RV which would require extensive recleaning of the RV. The effort to raise the baffle plates, develop and deploy an air sparging system, plus RV recleaning will take an estimated 2 to 3 month effort at a cost of approximately $180,000.

The last examined option is to cut access holes in the core barrel and then vacuum the exposed debris. As in the case discussed in Section 6.5.2, this plan is not pursued because of the labor intensive effort required within the RV and the need to acquire new and different types of equipment not previously used at TMI-2.

In summary, while this area represents a large accumulation of debris, it is the area where the debris is well confined. Therefore, no hazard exists and no potential for relocation is present. Accordingly, recognizing the difficulty involved to achieve removal, there is no basis to expand the efforts required.

Besides the material contained in the annular gap, a relatively uniform distribution of loose debris exists in locations within the hot leg bosses, flow deflector, vent valve seat, thermal shield and CSS inner and outer surfaces, and the inner surface of the RV wall. The majority of this material accumulated as a result of cavijet and airlifting operations within the core region and the resulting resuspension of fine particles. This material settled onto the horizontal surfaces and crevices.

Two methods were considered for removal of additional loose debris.

a. Air Sparging

Air sparging utilizes an air flow system to cause general turbulence and local eddies to wash off material on horizontal surfaces. Tests indicated that the debris movement can only be accomplished at close range between debris and the air source. Due to limited accessibility into the downcomer region (14 access holes), it is improbable that air sparging would be effective. In addition, a substantial fraction of any dislodged material is expected to settle onto other surfaces, defeating the objective of air sparging.

b. Brush and Vacuuming

The brush and vacuuming methods utilize a brush, possibly rotational, to swab debris off of surfaces. The debris is then removed via the filtration system and/or vacuumed from the bottom of the vessel. As with the air sparging system, accessibility is so limited as to make the concept unrealistic since no more than 10% of the areas can be reached.

Because of the relatively small, uniformly distributed quantities of material dispersed throughout the large surface areas of the annulus and downcomer region, it was concluded that no effective (or measurable) method could be deployed. It was also determined that any attempts to remove this finely dispersed material would likely relocate the material to previously cleaned surfaces and, therefore, would provide no direct benefit. It was concluded that this region of the RV has been defueled to the extent reasonably achievable given the conditions and fuel quantities that exist.

6.6.3 Internals Indexing Fixture Region

As described in Section 5.4.2.3, the estimate of residual fuel located in the IIF region is approximately 5 kg. The majority of any core debris residing in the IIF region is finely dispersed, resuspended material which has been demonstrated to be of a low density and fuel content. Due to the relative small quantity of fuel material (<10% SFML) and the low probability of material relocation, no additional defueling activities were deemed appropriate or necessary for this area.

6.6.4 Core Support Shield Region

As described in Section 5.4.2.4, the estimate of residual fuel in this area which includes the LOCA vent valves, hot leg openings, LOCA bosses, and the CSS inside surface and top of the lower CSS flange is approximately 11 kg. The major quantity of fuel material is located on top of the vent valve seats. The majority of material was transported to this area by airlift operations from the lower portions of the RV.

Defueling operations included both flushing and vacuuming. The remaining material was in a groove around the inside surface which is not accessible. Due to the small quantities of fuel remaining in any one area (<10% SFML), no additional defueling actions were identified as necessary or appropriate.

6.6.5 Upper Core Support Assembly Region

As identified in Section 5.4.2.5, the estimate of residual fuel remaining in the UCSA region is approximately 85 kg. An extensive effort was undertaken to remove all accessible fuel in this region of the RV. The baffle plates were removed one at a time, both sides were cleaned with a power brush to remove visible debris deposits, and the former plates were vacuumed for loose debris. An impact tool was used to break up debris masses formed on the former plate, and a "poker" tool was used to clear debris lodged in any of the former plate flow holes; all but two of a total of nearly 200 flow holes were completely cleared of debris. For debris masses not dislodged and broken up by the mechanical tools, the cavijet hydraulic tool was used to remove fuel deposits on the former plates and the core barrel. Following break up of all visible fuel debris, a high volume flush was performed in each region to wash fuel debris into the bottom head region for subsequent removal.

The preponderance of the remaining residual fuel is tightly adherent resolidified material for which no additional degree of mechanical fuel removal is practical. This material is not transportable, is widely distributed, and well below a quantity which would present a criticality concern. The remaining potential means for removing this fuel is chemical cleaning. This alternative is discussed in detail in Section 6.7. Because of the small amount of widely distributed, tightly adherent residual fuel, it is not practical to implement an effort as extensive as chemical dissolution to remove this fuel.

In summary, all accessible quantities of core debris have been removed and all practical means for fuel removal have been exhausted; therefore, no further effort is deemed warranted or appropriate.

6.6.6 Lower Core Support Assembly Region

As described in Section 5.4.2.6, an estimate of approximately 430 kg of residual fuel material was identified in this region. The majority of this remaining LCSA material, which is resolidified fuel, has accumulated within the gap between the forging and the IGSP (approximately 170 kg) and the forging peripheral flow holes (approximately 110 kg).

Over 95% of the total LCSA was cut into sections for removal from the RV. This represented over 30 tons of structural material and 1350 m^2 (15,000 ft^2) of surface area. Each of the removed sections was wire brushed and flushed extensively to minimize the amount of fuel material transferred to storage in the RB. Additionally, the remaining LCSA structure was mechanically defueled. An extensive effort was undertaken using airlift, high volume flush and vacuuming in all accessible areas of the LCSA. Three separate defueling attempts utilized the cavijet in remaining sections of the LCSA.

Notwithstanding the extensive effort directed to defueling the LCSA, the following three major methods of removing fuel were considered.

a. Localized Explosive Techniques

Shaped charge explosives were previously considered for use on large agglomerated masses of fuel material. This earlier application was considered specifically for the hard crust areas of the core which contained a hardened mixture of both metallic (cladding and structural metals) and ceramic (UO_2) materials. This application was not developed nor pursued because of the eventual success achieved by the CBM drilling operations and defueling impact tools.

The application of explosives for defueling of the peripheral flow holes of the forging requires extensive operations planning and proof-of-principal and charge loading tests. Complete access to the forging and precise placement of the charges would be required. Because of operator safety considerations, a detailed explosive handling and training program would be required as well as procedures and quality assurance requirements. An extensive licensing effort would also be required. Licensing activity and testing could require from 6 months to 1 year or more to complete. Additionally, there is a potential safety risk involved in the application of explosive charges.

Based on the level of effort required to qualify this program, the uncertainty in gaining access and precisely positioning these explosive charges and the uncertainty of eventual success in dislodging the fuel, no additional effort was deemed appropriate in developing explosive removal techniques.

b. Removal and Disassembly of the LCSA

Another potential method of removing the remaining fuel located within the LCSA is to remove the LCSA from the RV and selectively cut and disassemble the five plates to access the residual fuel. The LCSA was originally installed within the RV as one monolith structure containing the CSS, the UCSA/core barrel, and the LCSA. In order to access and remove the remaining LCSA plates, including the downward-facing bolts which connect the LCSA to the core barrel, the entire monolith structure would have to be removed as a complete unit. Due to the excessively high dose fields (approximately 3000 R/hr) associated with the neutron-activated structural steel, this entire operation would require water shielding and flooding of the FTC with over 300,000 gallons of borated water. As a minimum, the following supporting operations would be required: removal of the existing defueling work platform and associated equipment; relocation of the upper plenum assembly from the deep end of the FTC to a new location; establishment of a CSS, UCSA, and LCSA disassembly station with all support and handling equipment and storage fixtures; design, fabrication, and installation of a LCSA disassembly facility; and provision for fuel loading and storage.

Because the water elevation in the FTC would be raised approximately 7.5 meters (25 feet), the resulting water level in SFP "A" would also need to be raised an equivalent amount because the pools are common via the two fuel transfer tubes. The impact of higher water elevations in both the RB and FHB would require some major modifications in fuel canister handling equipment and operational procedures including relocation of the fuel canister dewatering station in SFP "A". Additionally, a new positioning or elevator system would be required to raise fuel canisters to the dewatering and transfer station. Within the RB, the existing service crane would no longer be useful as an overhead crane for transporting components or defueling equipment. The crane would have to be raised to a higher elevation or replaced by a new crane. The temporary dam installed between the deep and shallow ends of the FTC would also have to be removed to provide sufficient clearance to transport the RV internals and monolith structure (i.e., CSS, UCSA, and LCSA).

In summary, an extensive and detailed planning effort would be required to implement the removal and disassembly of the LCSA. The disassembly activity is technically feasible, not withstanding the first-of-a-kind effort required within the RB. An extensive design and fabrication effort would be required for both the monolith (i.e., CSS, UCSA, and LCSA) disassembly station and the LCSA (five separate plates) disassembly facility. The total cost and schedule burden for design, fabrication, and installation of the disassembly station and facility was estimated to be comparable to that of the original defueling platform, equipment, and canister loading hardware (i.e., approximately $20 million and 12 to 16 months). Assuming all 430 kg of residual fuel could be removed by this method, the cost would approximate $50,000/kg. While no major dose intensive activities were identified, extensive person-hrs, comparable to RV defueling, would be required. An estimate of approximately 350 person-rem would be accumulated over a 12 to 16 month expanded defueling activity.

The added defueling resource burdens were estimated at a cost of approximately $50,000/kg, at a schedule impact of 12 to 16 months, and an estimated additional accumulated dose of approximately 1 person-rem/kg. Accordingly, it was concluded that additional defueling activities were not deemed necessary or prudent.

c. Chemical Dissolution of the Lower Core Support Assembly

See Section 6.7 which discusses chemical dissolution techniques.

6.6.7 Bottom Head Region

As described in Section 5.4.2.7, an estimate of approximately 150 kg of residual fuel material remains in this region. The majority of this material is a fine dust (approximately 100 kg) widely distributed over the bottom head.

Initial defueling operations in this region included extensive mechanical defueling and airlift operations. Large quantities of debris were removed exposing the bottom head cladding surfaces and permitting inspection of the nozzles and welds. Four defueling attempts were made to flush and vacuum the remaining loose material from the bottom. The quantity of residual material was micron-size particles, which circulated through the vacuum system and redistributed back to the bottom head as fine dust material.

Some additional resolidified material remained attached to the inner surfaces of the peripheral standing guide tubes and some loose debris resides in the penetration holes of the incore nozzles. The quantity of material adherent to the guide tubes (approximately 20 kg) is inaccessible and, therefore, defueling is impractical. Approximately 30 kg of fuel remains in the incore instrument nozzles. A small quantity of this remaining fuel could possibly be removed by abrasive saw cutting of the remaining incore nozzles. However, this activity would have no significant effect on the total fuel quantity remaining and, therefore, is judged not to be effective.

Based on the extensive defueling effort undertaken on the bottom head region, no additional mechanical removal options were identified as appropriate. The only method identified for possible further removal of the fuel films and adherent material was chemical dissolution. Section 6.7 describes this technique and its application to the RV.

6.7 Chemical Dissolution

Chemical dissolution is the process of using dilute chemical solutions to dissolve fuel in bulk form, in powder form, or as a film on metallic surfaces. Chemical solutions used for this purpose were tested, under the sponsorship of EPRI, with direct applications to TMI-2 (Reference 6.1). Although some success has been achieved at test facilities, none of these testing facilities replicated the conditions of fuel failure and melting or the system volumes found at TMI-2. To date, use of chemical dissolution as a possible means of removing failed fuel material at TMI-2 has been restricted to laboratory scale experimentation and demonstration.

The key to ensuring the dissolution of UO_2 is to use a strong oxidizing agent, such as hydrogen peroxide, to convert the tetravalent uranium in UO_2 to the U^{+6} valence state where the highest solubility occurs. Oxalic acid, gluconic acid, and bicarbonate provide the solute capacity. Laboratory tests have shown that two processes will dissolve finely-divided UO_2 in a few hours. Unfortunately, these two candidates (OPG, a buffered oxalic-peroxide gluconic solution, and PBC, a peroxide bicarbonate solution) were not as successful when tested with TMI-2 core debris solids. In these experiments, only about 10 to 30% of the solidified uranium or uranium-zirc oxide eutectic were dissolved in 20 hours of solution exposure. The major dissolution of fuel bearing material occurred within the first 2 to 3 hours of testing.

Two variations of the OPG-AP-CITROX decontamination processes were tested for their ability to remove radioactivity from TMI-2 primary system artifact samples. Both the concentrated OPG-AP-CITROX and the more modified process using concentrated OPG followed by a dilute solution of AP-CITROX were found to be approximately equal in effectiveness for dissolving radioactive oxide films. Removal rates ranged from 44 to 82%. Both the dilute and concentrated OPG solvents proved to be unstable in solution due to peroxide decomposition. This decomposition was found to be accelerated by contact with large surface areas of stainless steel as would be found in the RV and RCS. The dilute OPG solution had an effective life ranging from 12 to 24 hours in stainless steel test systems and would be expected to be considerably shorter in the RV or RCS. (This instability of hydrogen peroxide has been demonstrated often at TMI-2 following addition of hydrogen peroxide to the RV to control the microorganisms appearing in the water.) Additional laboratory testing was performed to investigate the effect of various peroxide stabilizers in an OPG solution. Although several alternatives were tested, results were inconclusive.

To utilize the OPG-AP-CITROX dissolution methods at TMI-2 would require, as a minimum, the following:

a. Further laboratory testing on TMI-2 artifacts to determine the minimum required concentrations of chemicals in each solvent solution and to find a suitable peroxide stabilizer.

b. The design of various fluid systems and components to fill and circulate the dissolution solutions within the RV, process them as required, and remove them when the solution is to be changed. (To fill the RV and access all contaminated surfaces require at least 30,000 gallons of solvent solution.)

c. The procurement and installation of system components (pumps, valves, a condenser, one or two ion exchange columns, and a large heat exchanger). Flow rates of approximately 500 gpm may be required.

d. If the RV is to be flooded above the elevation of the nozzles of the reactor coolant piping (315' elevation), large [70 cm (28 inch) ID] plugs will have to be inserted to prevent chemical solutions from exiting the vessel. These plugs would need seals which are not affected by the corrosive uranium dissolving chemical solutions and portions of the upper core support cylinder would have to be removed to provide access.

e. Laboratory tests would be required to determine the proper methods of processing and disposing of the solvent and rinse wastes. At least 150,000 gallons of liquid would have to be processed. Dilute solutions could be processed through an ion exchanger with the residual liquid further processed by evaporation. The concentrated solvent wastes could be solidified in portland cement after some pH adjustment. Both ion exchange resins and concrete cylinders would eventually require shipment to proper waste storage facilities under DOE authority, as will the contaminated equipment installed to circulate and process the solvents. At this time, no high level waste storage facility in the United States will accept and store these solidified wastes.

Estimates were made of the potential cost, schedule and radiation exposures which would be encountered if a satisfactory chemical solvent system were added to the TMI-2 defueling program. This chemical decontamination process was estimated to cost approximately $6 million. While no major dose-intensive activities were identified, extensive person-hours, comparable to RV defueling, would be required. The costs and person-rem do not include the corresponding plant operating costs and personnel exposures to keep the TMI-2 site active for an additional three years.

It is assumed that these chemicals are capable of dissolving 60% of the fuel fines and fuel adhering to the vessel surfaces in the form of scale, and as much as 40% of the solidified fuel-bearing material principally in the RV bottom head. Based on this assumed performance potential, approximately 500 kg of fuel material would be removed from the RV at a removal cost of $11,000/kg and 3.0 person-rem/kg (i.e., roughly six times the unit cost and 15 times the person-rem impact of earlier bulk fuel removal activities discussed in Section 6.1). In addition, no existing waste storage facility would accept the high level wastes generated by this activity. Since the potential for acceptance of these wastes in the near future is very low, this fact alone would preclude any further consideration of attempts to remove residual fuel by the chemical dissolution processes.

6.8 Summary Assessment

For each of the identified facility areas containing residual fuel in excess of 10% of the SFML, potential method(s) for further fuel removal were identified and assessed. These methods, for the most part, are believed to be technically feasible. Each method, however, is attended by some incremental impact relative to the degree of complexity, uncertainty of success, significant occupational exposures, major additional cost and schedule impacts, and waste forms not amenable to disposal without additional onsite storage or processing requirements. This section provided a discussion of the major residual fuel deposits (i.e., UO_2) located within the TMI-2 facility with the intent of evaluating and demonstrating that reasonable and practical methods were considered and used to access and remove this fuel. Based on the following summation of plant conditions and alternative defueling considerations, it is concluded that the TMI-2 facility has been defueled to the extent reasonably achievable and all identified defueling objectives have been completed.

1. Collectively, the TMI-2 facility contains approximately 1% of the original inventory of fuel.

2. A significant effort has been expended to clean up the TMI-2 facility. Over ten years and more than 3.6 million person-hrs were required to complete the task.

3. Subcriticality has been ensured for all locations within the facility including systems, structures, and components.

4. Based on previous studies (i.e., PEIS) and the included criticality analyses (Section 5.5), the credible events and postulated accidents analyzed have been demonstrated to have no significant impact on the health and safety of the public.

5. Additional fuel removal alternatives have been evaluated and demonstrated to add incremental impacts which substantially exceed one or both of the defueling parameters (i.e., >$1900/kg and/or >.02 person-rem/kg of fuel removed) without a significant return in fuel removal or safety enhancement.

6. These additional defueling alternatives would be better suited to the eventual dismantlement and decommissioning of the TMI-2 facility. In the interim, substantial decay of existing fission and activation products will occur, thereby substantially reducing the potential occupational exposures.

7. The physical condition and long-term stability of the TMI-2 facility, including the existing residual fuel, has been demonstrated to be safe. Risk to the health and safety of the public has been essentially eliminated; the impact to the environment is non-existent.

Based on these considerations, no additional defueling actions are deemed necessary or beneficial.

7.0 OCCUPATIONAL EXPOSURE

7.1 Defueling Dose Estimates

The initial estimate of cumulative dose to complete the TMI-2 defueling effort was contained in the PEIS published in March 1981 (Reference 7.1). At that time, RB radiological conditions were still being characterized and the extent of reactor core damage was unknown. Occupational dose totals for fuel removal and primary system decontamination were estimated at 900 to 4100 person-rem. In October 1984, PEIS Supplement 1 was issued (Reference 7.2) which outlined several alternative plans for the TMI-2 cleanup and associated cumulative doses. The primary goal pursued at the time of PEIS Supplement 1 was dose reduction, followed by defueling and decontamination. The cumulative occupational dose estimate for reactor disassembly and defueling as presented in the PEIS was 2600 to 15,000 person-rem; the total TMI-2 cleanup estimate was 13,000 to 46,000 person-rem.

The actual cumulative occupational dose for defueling and defueling support activities is below 2000 person-rem. The total TMI-2 cleanup occupational dose to date is less than 6500 person-rem.

GPU Nuclear has proposed the near-term implementation of PDMS without further preparation for decommissioning. By postponing preparation for decommissioning, the final cumulative dose total will be reduced through natural decay of the radioactive material in the plant and the expected advances in cleanup technology. An evaluation included in Supplement No. 3 of the PEIS (Reference 7.3) estimates the dose savings to range from 3600 to 9100 person-rem. The PDMS SAR (Reference 7.4) estimates the dose savings to range from 4500 to 9800 person-rem. The large dose savings due to PDMS will significantly reduce the overall occupational dose total for TMI-2.

7.2 Defueling Dose Reduction and Radiological Protection

Completion of defueling with less than the estimated cumulative dose was made possible by a number of factors, some of which are discussed below.

7.2.1 Design Engineering of Defueling Tools and Equipment

Prior to installation of defueling equipment, extensive engineering was applied to ensure that the defueling activities could be performed in compliance with the ALARA principle in dose reduction. The defueling equipment was specifically designed to minimize dose to the workers. Designs included the shielded work platform with its 15-cm (6-inch) thick steel plates, shielding for the service and auxiliary work platforms, the canister transfer shield and shield collar, and the internal vertical shielding under the shielded work platform. Detailed descriptions of the engineered defueling equipment designs are contained in the Defueling SER (Reference 4.26).

As new defueling tools and equipment were developed, the need for Radiological Controls Department considerations became apparent. Tooling designs that minimized fuel traps, provided for proper flushing and drainage when removed from the RV, and allowed maintenance activities without high worker doses were a direct benefit of the Radiological Controls Department review of tool design prior to fabrication and installation.

7.2.2 Defueling Platform Dose Reduction

In addition to the upfront dose reduction considerations incorporated in the defueling design engineering, numerous actions were taken at the defueling platform to ensure that dose to defueling personnel was ALARA. These steps included:

1. Installation of lead shielding on the work slot handrails reduced dose rates to personnel involved in work activities at the slot.

2. Installation of a storage container (dose reduction box) in a corner of the defueling platform allowed interim storage of high radiation items near the work site. Other highly radioactive tools and equipment were removed from the defueling platform for both interim and long-term storage.

3. Frequent decontamination of the defueling platform contributed to reduced usage of respiratory protection equipment and minimized the spread of hot particles and the potential for skin contaminations. Periodic use of a hot water flush system also reduced dose rates in and around the work slot area by flushing high radiation debris back into the RV.

4. Implementation of the technique of "flush and wipe" for defueling tools during removal from the RV was successful in reducing the radiation and contamination levels on the defueling platform as well as on equipment prior to storage or maintenance. This technique also served to reduce the spread of hot particles.

5. Sealing and plugging of crevices and small holes on the defueling platform reduced the number of high radiation hot spots generated during defueling activities.

6. Frequent processing of the RCS reduced the concentrations of Cs-137 and Sr-90. This not only reduced dose rates at the work slot but also reduced the amount of contamination on items removed from the RV.

7.2.3 Dose Reduction for Defueling Support Activities

Actions undertaken to reduce the occupational dose to defueling personnel in other areas include:

1. Occupational exposures to workers in transit between the defueling platform and the RB airlock access point were reduced by use of the enclosed stairwell. With extensive shielding in and around the enclosed stairwell, the average round trip dose for the workers was reduced by over 50%.

2. Repair and modification to defueling tools were accomplished in a specifically designated area on the 347' elevation of the RB (i.e., the "tool repair area"). The area was chosen for its low dose rates and accessibility to the polar crane. During the course of the defueling effort, segregated work areas were set up in the tool repair area to allow maintenance of high radiation/contamination tooling (e.g. plasma arc cutting equipment, underwater lights, core bore drill bits). These special areas minimized the spread of high level contamination to other areas of the RB.

3. A facility was built on the 347' elevation of the RB to permit decontamination and cutting of defueling equipment in a contained, ventilated environment. The facility had direct access to the service and polar cranes. Use of the facility proved valuable for disposal of unneeded defueling equipment and also for gross decontamination of highly contaminated equipment.

4. A shielded, low-dose-rate polar crane operator station was established on top of the "A" D-ring to minimize dose to the crane operator when polar crane movements were necessary. The shielding was upgraded, as necessary, when tasks performed in adjacent areas could increase radiation fields at the polar crane operator station.

7.2.4 Ex-Vessel Defueling Dose Reduction

The numerous tasks required to complete ex-vessel defueling and fuel characterization received intensive Radiological Controls Department input. Whereas in-vessel defueling involved thousands of person-hours in comparatively low radiation fields, ex-vessel activities often involved short-term entries into very high radiation fields. Some of the ex-vessel activities (e.g., pressurizer and OTSG) were the focus of ALARA decision analyses performed, as required by the TMI-2 ALARA Program, to evaluate different work options in order to optimize radiation protection for the task.

7.2.5 Airborne Contamination Controls

Airborne Radioactivity levels were minimized during the defueling effort by use of a 4,000 scfm filtration unit used to off-gas the RV. The ventilation unit created a negative airflow into the defueling work slot and prevented airborne radioactivity generated under the platform from affecting personnel working on the platform.

Airborne radioactivity levels were generally controlled such that much of the early defueling activity was performed without respiratory protection. Later in the defueling effort, due to core debris drilling with the CBM, core debris airlift operations, and the consequent hot particle contamination in and around defueling areas, respiratory protection equipment became standard.

7.2.6 Defueling Worker Training

Worker involvement in maintaining low occupational exposures was achieved, in part, by extensive use of mock-up training, including a full-scale replica of the defueling platform and RV in a non-radiologically controlled area. Defueling equipment and techniques were practiced at the mock-up prior to actual task performance in the RB . Other full-scale mock-ups included the OTSG, pressurizer, and the RCS hot and cold legs.

In addition to required pre-job briefings, videotapes were often utilized to provide additional instruction in activities such as contamination control at the defueling work slot, protective clothing undressing techniques, and proper handling of radioactive material in the RB. Seminars were used to discuss topics such as hot particle and contamination controls. All of the above resulted in a well-trained work force participating in the defueling effort. Informed workers constituted a major aspect of our radiation protection program.

7.2.7 Defueling Radiological Considerations

Radiological considerations included:

1. Dose rate limits on defueling tools stored in accessible equipment storage areas and racks

2. Portable area radiation monitors for use, as necessary, in the defueling work slot, on the defueling platform, and at the crane operator station

3. Dose rate hold points in work documents which required specific approval for removal and handling of highly radioactive material out of the RV

4. Maximum contamination limits for each defueling work area (When these limits were exceeded, decontamination was performed.)

5. Survey and monitoring standards for Radiological Controls Technicians to ensure radiological conditions were properly assessed

6. An aggressive hot particle control and fuel spill cleanup program

7. Radiological goals for defueling activities to maintain doses ALARA and to reduce skin contaminations.

7.3 Defueling Radiological Dose Statistics

The dose information contained in this section was obtained from one of two sources: SRDs and TLDs. SRD values are those dose records obtained immediately as personnel exit the work area. SRDs, due to their design, tend to overestimate actual dose received. TLD values are inherently more accurate than SRD values and are normally used as "doses-of-record." Because TLDs may be worn by personnel for multiple tasks, using TLD dose information on a task basis is difficult.

7.3.1 Reactor Vessel Defueling and Defueling Support

A chronology of RV defueling activities is presented in Table 7-1, along with the dose expended for each activity. The data was obtained from defueling radiological statistics compiled on a daily basis. The statistics were used throughout the defueling effort to trend various radiological parameters, such as average dose per working hour, so that variations might be addressed rapidly. Tables 7-2 and 7-3 summarize both RV defueling operations and defueling support activities for the entire defueling effort. All dose totals were derived from SRD values.

7.3.2 Ex-Vessel Defueling and Fuel Characterization

Major ex-vessel defueling and characterization activities are summarized in Table 7-4 for each activity. The dose data for each activity was obtained from dose information contained in applicable ALARA reviews, RWPs, and exposure tracking numbers. All dose totals were derived from SRD values.

7.3.3 Cumulative Dose Summaries

A comparison of dose expended per category of work activity is provided in Table 7-5. The categories are broader than those in the previous tables, with the Defueling Support category encompassing many more tasks than were included on Table 7-3. The data illustrates that defueling activities since 1986 have accounted for over half the total dose expended for the TMI-2 cleanup.

A summary of the TMI-2 annual worker dose, as determined by TLDs, is included in Table 7-6. The total clearly shows the impact of the round-the-clock defueling effort since the beginning of 1986. No individual worker has received over 3.7 rem whole body dose in any one year since the initial cleanup activities began in 1979.

TABLE 7-1

REACTOR VESSEL DEFUELING ACTIVITIES

DATES	ACTIVITIES	DOSE (person-rem)
10/30/85 - 04/14/86	Initial Defueling Activities: Pick-and-Place Spade Bucket, Canister Transfers	96
04/15/86 - 06/20/86	Video Inspections, Vacuum System, Pick-and-Place	16
06/21/86 - 08/10/86	Core Sample Acquisition Program (CBM)	24
08/11/86 - 08/31/86	Core Region Pick-and-Place, Spade Bucket, and Chisel	15
09/01/86 - 11/16/86	Endfitting Removal and CBM Rubblize Core Debris	44
11/17/86 - 05/24/87	Core Region Pick-and-Place, Airlift, Chisel	81
05/25/87 - 12/13/87	Stud Fuel Assembly Removal, Pick-and-Place, and Airlift	114
12/28/87 - 04/09/88	CBM Drilling of LGRS	41
04/22/88 - 04/29/88	LGRS Removal into CFT-1A	10
04/30/88 - 06/05/88	Plasma Arc Cutting of Lower Grid Rib Periphery	8
06/06/88 - 07/02/88	Plasma Arc Cutting of LGDP	9
07/07/88 - 07/11/88	LGDP Removal into CFT-1A	6
07/12/88 - 08/03/88	Forging Cleaning via Airlift and Pick-and-Place	14
08/03/88 - 11/14/88	Plasma Arc Cutting of Forging, Incore Guide Tubes, and Support Posts	42
11/16/88 - 11/23/88	Forging Removal into CFT-1A	3
11/26/88 - 12/18/88	IGTSP Cleaning and Abrasive Saw of Support Posts/Incore Guide Tubes	19
12/22/88 - 01/06/89	Plasma Arc Cutting of IGTSP	3

TABLE 7-1 (Cont'd)

REACTOR VESSEL DEFUELING ACTIVITIES

DATES	ACTIVITIES	DOSE (person-rem)
01/07/89 - 01/12/89	IGTSP Removal into CFT-1A	4
01/13/89 - 02/23/89	Flow Distributor Cleaning	29
02/28/89 - 03/31/89	Plasma Arc Cutting of Flow Distributor	17
04/01/89 - 04/12/89	Plasma Arc Cutting of Baffle Plates	5
04/21/89 - 04/28/89	Flow Distributor Removal into CFT-1A and "A" D-ring	8
04/29/89 - 07/09/89	Bottom Head Defueling via Pick-and-Place and Airlift	50
07/12/89 - 08/08/89	Baffle Plate Bolt Removal	21
08/15/89 - 08/26/89	Airlift, Pick-and-Place, Cavijet LCSA	6
08/27/89 - 09/25/89	Baffle Plate Kerf Cleaning and Bolt Removal	18
09/26/89 - 10/29/86	Baffle Plate Removal and Core Former Region Drilling	21
10/30/89 - 12/16/89	Final Flush of LCSA and Final Defueling of Bottom Head	21
12/27/89 - 01/30/90	Final RV Cleanup and Inspection	16
10/30/85 - 01/30/90	Total Defueling Operations	761

NOTE: This table does not include defueling support activities, or ex-vessel defueling and characterization activities. All dose totals are from SRDs.

TABLE 7-2

REACTOR VESSEL DEFUELING OPERATIONS

TIME PERIOD	DOSE (person-rem)	RWP PERSON-HOURS	PERSON ENTRIES	DOSE RATE (mrem/hr)	DOSE/ENTRY (mrem)
1985	12	1,463	400	8.2	30
1986	200	19,101	5,398	10.5	37
1987	179	19,103	5,627	9.4	32
1988*	154	16,154	4,778	9.5	32
1989**	202	18,267	5,677	11.1	36
TOTAL	747	74,088	21,880	10.1	34

* Includes LGRS, LGDP, and Forging Removal.

** Includes IGSP and Flow Distributor Removal.

NOTE: All dose values are from SRDs.

TABLE 7-3

REACTOR VESSEL DEFUELING SUPPORT

TIME PERIOD	DOSE (person-rem)	RWP PERSON-HOURS	PERSON ENTRIES	DOSE RATE (mrem/hr)	DOSE/ENTRY (mrem)
1985	3	281	90	10.1	32
1986	125	7,019	3,479	17.8	36
1987	186	12,002	5,372	15.5	35
1988	242	17,137	6,738	14.1	36
1989	230	13,756	5,555	16.7	41
TOTAL	786	50,195	21,234	15.7	37

TABLE 7-4

EX-VESSEL DEFUELING AND FUEL CHARACTERIZATION ACTIVITIES

	DATE	ACTIVITIES	DOSE (person-rem)
Once-Through Steam Generators	1984/1985	FM&A	9
	1986	Remove "A" & "B" manways, upper and bottom head FM&A	27
	1987	"A" lower head sampling "A & B" tubesheet defueling	38
	1988/1989	Bottom head, J-Leg, tubebundle, tubesheet FM&A	47
		TOTAL =	121
Pressurizer	1985	Remove manway, inspect and sample FM&A	28
	1986	Sludge sample and spray line defueling preps	4
	1987	Spray line hot tap/flush, pressurizer vacuuming	19
	1988	Pressurizer defueling (mini-sub) and surge line FM&A	26
		TOTAL =	77

TABLE 7-4 (Cont'd)

EX-VESSEL DEFUELING AND FUEL CHARACTERIZATION ACTIVITIES

	DATE	ACTIVITIES	DOSE (person-rem)
Incore Instrumentation Probe	1985	Reactor Vessel and Cavity FM&A	6
Cold Leg/Reactor Coolant Pump/ Decay Heat Drop Line	1987	FM&A	11
Hot Legs (incl. Decay Heat Drop Line)	1987&1989	Defueling	14
Decay Heat Drop Line	1988/1989	Defueling	28
Endfittings	1989	FM&A	11
Hot Leg/Cold Leg/Core Flood Line	1989	FM&A	5
		TOTAL =	75

NOTE: All dose values are from SRDs.

TABLE 7-5

WORKER DOSE FOR MAJOR ACTIVITIES

1986 - 1989

	DOSE (person-rem)	% OF TOTAL
Defueling Operations (Reactor Vessel Only)	698	20%
Defueling Support (Tool Repairs, Water Cleanup)	1,058	31%
Reactor Building Miscellaneous (Robotics, Crane Operations, Radwaste, etc.)	765	22%
Decontamination (Outside the Reactor Building)	424	13%
Routine Operations (Operations, Chemistry, Radiological Controls, Outside Reactor Building)	277	8%
Ex-Vessel Defueling (Pressurizer, OTSG, etc.)	216	6%
	TOTAL = 3,438	100%

NOTE: All person-rem totals are corrected TLD values.

TABLE 7-6

ANNUAL WORKER DOSE

YEAR	DOSE (person-rem)	MAXIMUM WORKER WHOLE BODY DOSE (rem)
1979*	418	4.5
1980	193	2.1
1981	138	2.0
1982	384	3.0
1983	373	2.7
1984	514	3.7
1985	722	3.5
1986	907	3.4
1987	975	3.5
1988	917	3.6
1989	639	3.5
	TOTAL = 6,180	

* From March 28, 1979, through December 31, 1979.

NOTE: All person-rem totals are corrected TLD values.

8.0 CONCLUSIONS

The purpose of this DCR is to provide the basis for concluding that the TMI-2 facility has been defueled to the extent reasonably achievable and to demonstrate that inadvertent criticality has been precluded. This report provides the basis for the TMI-2 facility transition to Mode 2.

This basis includes criticality analyses that addressed the quantity of residual fuel in each defined location and the potential for fuel relocation. The analyses have estimated the quantity of fuel remaining, its location, its dispersion within the location, its physical form (i.e., film, finely fragmented, intact fuel pellets), its mobility, the presence of any mechanism that would contribute to the mobility of the material, the presence of any moderating or reflecting material, and its potential for a critical event. Each issue was addressed to the extent appropriate for a given quantity of fuel.

In summary, GPU Nuclear has concluded that the TMI-2 cleanup has progressed to the extent that an inadvertent criticality is precluded, the RV and RCS are defueled to the extent reasonably achievable, and that the prerequisites for the transition to Facility Mode 2, as defined in Technical Specifications Table 1.1, have been satisfied.

8.1 Residual Fuel Quantification

The total quantity of residual fuel is estimated to be less than 1125 kg distributed in four major plant locations as follows:

- Auxiliary and Fuel Handling Buildings < 17 kg
- Reactor Building (excluding the RCS) < 75 kg
- Reactor Coolant System (excluding the RV) <133 kg
- Reactor Vessel <900 kg

Assessment of the known residual fuel quantities demonstrates that there is insufficient (i.e., <140 kg SFML) residual fuel present in any discrete location, except the RV, to exceed the SFML even if it were to accumulate in one area. In the case of the RV, a specific analysis was performed to demonstrate that a criticality event could not occur.

The estimated residual fuel quantity represents approximately 1% of the original 94,000 kg of UO_2 core inventory and demonstrates a substantial defueling effort. The residual fuel estimate was developed based on a variety of methods including direct measurement by instrumentation, visual inspection, and sample collection and analysis. The methods selected were influenced by many factors including accessibility, background radiation levels, measurement uncertainties, and equipment sensitivity. GPU Nuclear plans to conduct an extensive SNM measurement program as part of the overall facility fuel accountability program. The post-defueling SNM survey will provide the final residual fuel estimates.

8.2 Residual Fuel Location and Forms

As a result of the TMI-2 accident and subsequent cleanup activities, less than 1% of the fuel material was dispersed to system tanks and components external to the RV. The following is a summary of fuel locations and forms in the TMI-2 facility.

8.2.1 Auxiliary and Fuel Handling Buildings

The residual fuel in the AFHB is located throughout the two buildings in numerous pipes and tanks. This fuel is in the form of finely divided particulate and sediment material with minor amounts of fuel found as adherent films. This quantity of fuel is substantially below the SFML of 140 kg; thus, within the AFHB, there is no potential for fuel accumulation which could result in a critical mass.

8.2.2 Reactor Building, Excluding the Reactor Coolant System

The residual fuel in the RB outside the RCS consists of finely divided particulate and sediment material located within piping systems, in the RB basement, in the FTC, affixed to incore guide tubes stored in the "A" D-ring, and as adherent films on surfaces of the RV head assembly, the RV plenum, LCSA pieces stored in the CFT, and upper endfittings stored in containers located on the RB 347' elevation. Because the total amount of these residual fuel quantities is less than 140 kg, there is no potential for fuel accumulation within the RB to result in a critical mass in the RB.

8.2.3 Reactor Coolant System, Excluding the Reactor Vessel

Of the residual fuel in the RCS, the largest discrete location of fuel is in the upper tubesheet of the "B" OTSG. Approximately 36 kg of residual fuel exists as tightly adherent material not readily removable by available dynamic defueling techniques and, therefore, not readily transportable to other locations for accumulation. The remaining quantity of residual fuel is dispersed through the RCS in the form of finely divided particulate and sediment material and adherent films. The condition of this residual fuel prevents any significant fuel transport, thus minimizing any potential for fuel accumulation or interaction. Even if all of this residual fuel is accumulated in one location, the SFML would not be exceeded. Hence, criticality is precluded.

8.2.4 Reactor Vessel

Extensive visual examination and sample analyses during and following RV defueling has quantified the amount, form, and location of residual fuel in the RV. The majority of residual fuel remains as resolidified material, either tightly adherent to the RV components or inaccessible for defueling, as granular material largely located within and around inaccessible areas, and as a fine dust not amenable to further vacuuming. A small quantity of fuel material exists as films isolated on internal surfaces of piping, tanks, and RV internals. A separate analysis was performed to demonstrate that criticality is precluded in the RV in its present condition. An additional evaluation was performed to consider the potential effects of a substantial relocation of residual fuel as a result of an external event such as a seismic event and other external or non-mechanistic events to demonstrate that criticality under those conditions is precluded. Based on these analyses, it is concluded that the residual fuel in the TMI-2 RV is suitable for long-term storage with criticality precluded.

8.3 Criticality Analyses

As discussed above, analyses have demonstrated that criticality has been precluded as a result of the extensive TMI-2 defueling effort. This conclusion is based on three evaluations: the SFML determination, a bounding RV criticality calculation, and the potential for criticality under accident conditions.

8.3.1 Safe Fuel Mass Limit

A revised SFML has been defined for assessment of the long-term storage conditions at TMI-2. This limit is based on the extensive data base developed from debris sampling, video inspection, and other defueling programs to characterize residual fuel composition. The SFML establishes that the calculated neutron multiplication, k_{eff}, does not exceed 0.99, including a computer code uncertainty bias of 2.5% Δk.

The conservative spherical geometric model which consisted of a center region containing an optimal mixture of unborated water and fuel surrounded by 12 inches of unborated water reflector (i.e., effectively an infinite reflector) was used. The fuel composition was assumed to be TMI-2 average fuel including burnup effects, optimally moderated with unborated water and excluding any credit for the presence of impurities in the fuel (e.g., observed structural and control material, including zirconium, iron, boron, cadmium, silver, and indium).

This highly conservative calculation resulted in a SFML of 140 kg of UO_2. Introducing a conservative assumption regarding the effect of interstitially mixed boron (0.072 wt%), as observed in some sample analyses, an infinite neutron multiplication factor (k_∞) of less than unity is calculated. Thus, no amount of core debris accumulation could result in a criticality event.

In summary, a more realistic representation of the residual fuel demonstrates that criticality is precluded for all quantities of fuel accumulation even when optimally moderated with unborated water (i.e., $k_\infty < 1$). Nonetheless, a SFML of 140 kg was conservatively adopted.

8.3.2 Reactor Vessel Criticality Calculation

Because the total residual fuel quantity within the RV was estimated to be greater than 140 kg of UO_2, a special analysis of worst-case conditions within the RV was performed. This analysis used in-vessel inspections of debris locations and quantities, as well as conservative debris removal estimates, to develop a specific three-dimensional analytical model of the end-state RV configuration. Conservative assumptions were made regarding the quantity and location of fuel remaining in the RV following the completion of in-vessel defueling activities. The regions modelled in detail were the bottom head, the LCSA, and the core former region. Significant conservative allowances included in the development of the analytical model are:

- Unborated water optimally mixed with the fuel with no credit allowed for the presence of impurities in the fuel
- Conservative amounts of fuel layers placed on the LCSA plates
- The entire bottom head covered with a 1.2-cm (0.5-inch) layer of fuel
- A 0.6-cm (1/4-inch) layer of fuel, with a height of 3 meters (10 feet), assumed to be attached to the core barrel in the core former region of the model
- Each of the LCSA plates having a radial thickness that conservatively bounded the presence of fuel on the plate
- The fuel was assumed to extend the entire 360° of the periphery of the RV
- The holes in each of the modelled LCSA plates were assumed to be filled with fuel and unborated water in an optimal mixture
- Considerably more fuel was included in the analytical model than estimated measurements of residual fuel. In the model, over 670 kg of UO_2 were distributed on the bottom head, 5500 kg on the LCSA, and 600 kg on the UCSA, for a total of greater than 6700 kg UO_2, as compared to less than 900 kg total estimated residual fuel based on visual surveys of the entire RV.

The results of analysis of the RV utilizing this extremely conservative model for criticality resulted in a neutron multiplication factor of 0.983 including a 2.5% Δk uncertainty bias. Because the k_{eff} was less than 0.99, it is concluded that the much smaller quantity of fuel actually remaining in the RV does not pose a criticality safety concern (i.e., criticality is precluded).

8.3.3 Potential for Criticality During Accident Conditions

With approximately 900 kg of residual fuel in the RV, it can be postulated that a seismic event, aging and corrosion, or other unknown event could cause the residual fuel to accumulate in one area resulting in a potential for criticality. However, as evidenced by the extensive defueling effort, the residual fuel has consistently resisted strong displacement atttempts by aggressive methods.

Nonetheless, a realistic criticality analysis of the residual fuel, accounting for reasonably expected impurity levels (i.e., observed control and structural materials) has been performed for the maximum potential fuel accumulation in an optimal configuration. Even with an unlimited quantity of unborated water, the calculated infinite neutron multiplication factor k_{∞} is less than 0.99, including the 2.5% Δk computer code bias. Therefore, no physically achievable quantity of residual core debris can result in a critical fuel configuration. Regardless, a stable and insoluble neutron poison material will be added to the bottom head of the RV to provide added margin and absolute assurance that no circumstance will result in a condition causing the residual fuel in the RV to become critical. Hence, criticality is precluded for all credible or incredible conditions.

8.4 Defueling Objectives and Guidelines

In addition to precluding criticality, the defueling program had as an objective the removal of fuel to the extent "reasonably achievable" within technically practical methods. Implicit in this operational objective were an aggressive ALARA program to limit occupational doses to the defueling staff, ensurance of health and safety to the general public, use of reasonably available technology, and overall feasibility based on schedule and resource impacts. In order to demonstrate attainment of the "reasonably achievable" objective, additional defueling alternatives were evaluated for the remaining major residual fuel deposits.

In general, removal of the remaining fuel would require a more tedious, labor-intensive effort with attendant significant occupational exposure. Further, unique defueling techniques (i.e., abrasive cleaning, high pressure water erosion, chemical cleaning, and component removal and/or disassembly of the primary system) would be required. These unique techniques and material requirements would create radioactive waste forms and packages which are not amenable to accepted disposal options and, therefore, could require extended on-site storage or further processing.

8.5 Summary

In summary, additional defueling is unwarranted when assessed in the context of occupational doses (ALARA), resource commitments, and no significant benefit to the health and safety of the public. Considering the extensive cleanup activity accomplished over the past ten years, the major effort completed to quantify and characterize the residual fuel, the analyses performed which demonstrate that criticality has been precluded, and that continued defueling activities are of limited benefit, GPU Nuclear concludes that TMI-2 has been defueled to the extent reasonably achievable and that transition to Facility Mode 2, as defined by TMI-2 Technical Specifications, is appropriate.

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APPENDIX B

CRITICALITY SAFETY EVALUATION FOR THE TMI-2 SAFE FUEL MASS LIMIT

GPU Nuclear

GPU Nuclear	**No.** 15737-2-N09-005
Title Criticality Safety Evaluation for Increasing the TMI-2 Safe Fuel Mass Limit	**Page** i **of** 26

Rev.	SUMMARY OF CHANGE	Approval	Date
0	Original Issue		
1	Changed to include discussion of effects of impurities and to remove small fuel particle discussion.		
2	Incorporate change in Boron Weight percent for impurities case.		

FORM NG-7310.06-2 (11/83)

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1.0 INTRODUCTION

1.1 Background

In the early stages of the TMI-2 cleanup activities, analyses were performed to establish limits on the amount of fuel debris that could collect in any plant component without posing a criticality safety concern (Reference 1). The significant assumptions used in that analysis included a fuel enrichment of 3 wt% U-235, unborated water reflection and moderation, and a maximum fuel rod diameter of 0.4 inches. The 3 wt% enrichment approximately corresponds to the unburned condition of the highest enriched batch 3 fuel - 2.96 wt%. The unburned enrichments for the other fuel batches at TMI-2 were batch 1 - 1.98 wt% and batch 2 - 2.64 wt%.

Based on the compilation of data presented in Reference 2, Reference 1 reported that the minimum critical mass for unborated water reflected and moderated 3 wt% UO_2 fuel rods of a maximum diameter of 0.4 inches was 93 kg of UO_2. A factor of safety of approximately 75% was then applied, thus establishing the critically safe fuel mass limit for the TMI-2 defueling operations at 70 kg. This limit provided the criterion for the maximum amount of fuel which could collect in an isolated unit and be assured to remain subcritical regardless of other parameter values. This limit has been applied to the various defueling activities at TMI-2 unless it was demonstrated by a specific evaluation that a larger mass would be maintained subcritical.

1.2 Purpose

The purpose of this document is to develop a refined safe fuel mass limit for use at TMI-2 during the remaining defueling activities and in evaluating long term storage conditions (i.e., Post Defueling Monitored

Storage, Mode 4). This limit is to be developed based on more realistic and consequently less conservative assumptions than those used in the Reference 1 analyses. Justification for using these less conservative assumptions is provided by the significant data that have been collected from debris samplings, video inspections, and other defueling data which were unavailable at the time of the Reference 1 analyses. These data provide a better understanding of the accident scenario and the actual debris configuration and composition, thus permitting the creation of a refined and more realistic model of the fuel debris.

1.3 Criterion for Allowable Fuel Mass

The criterion used to establish the acceptability of a quantity of fuel is that the calculated neutron multiplication, k_{eff}, does not exceed 0.99, including a computer code uncertainty bias. This acceptance criterion is consistent with the previous licensing basis for the reactor coolant system during defueling (References 3,4,5).

2.0 MODELLING

2.1 Geometrical Considerations

The safe fuel mass limit model developed for the present evaluation is shown in Figure 1. As noted from this figure, the innermost region of the model consisted of a mixture of unborated water and fuel debris. A spherical geometry was chosen in an effort to minimize the surface area to volume ratio, thus maximizing the neutron multiplication. Surrounding the fuel region was an effectively infinite thickness (~12 inches) of unborated water reflector. The radius of the innermost region was varied until the calculated k_{eff} (including a 2.5% Δk uncertainty bias, see Section 2.5) reached the licensing limit of 0.99.

2.2 Base Case Fuel Model

Consistent with previous criticality safety analyses for TMI-2 (References 3,4,5), the fuel was represented as a homogeneous medium for which the neutronic data corresponded to a dodecahedral lattice structure of spherically shaped fuel pellets (Figure 2). The composition of the fuel was assumed to be TMI-2 average fuel (i.e., the homogeneous mixture of the three fuel batches). As with the Reference 5 analyses, burnup effects were considered in all three fuel batches. In each fuel batch the effects of uranium depletion, fissionable plutonium generation and rare earth fission product generation were considered. The procedure used for the quantification of batches 1 and 2 burnup was similar to that previously used for the batch 3 fuel (see Reference 3), using the actual exposure histories for batches 1 and 2. The result of incorporating the burnup effects produced a net U-235 enrichment of 2.24 wt%, plus associated plutonium, for the homogeneous mixture. The reasonableness of this enrichment is discussed in Section 2.3. The composition of the fuel used in the analysis is provided in Table 1.

Other major assumptions considered in this evaluation were that the equivalent of standard, full sized fuel pellets were used for the fuel particle size, the fuel was assumed to be optimally moderated with unborated water (VF=0.28), and no credit was taken for the presence of impurities in the fuel debris. All sample data collected to date have shown that it is not credible to assume that the debris is UO_2 without impurities (References 6,7,10,14-16). Debris samples have been collected in many areas of the plant, including within the vessel, the pressurizer, the "B" steam generator and the makeup filters. These samples are considered representative of any debris that is still remaining at TMI-2. Each sample collected has shown the presence of impurities, (e.g., zirconium, iron, boron, cadmium). Table 2 provides a summary of the impurity content of the various debris samples. These samples have shown that the impurities, in particular the boron, are an integral part of the debris material and are not just surface deposition (Reference 17). Thus these impurities are considered to be a long term constituent of the fuel debris. Additional studies were performed (see Section 2.4) to evaluate the effects that these impurities have on the reactivity of the fuel debris and, thus, the corresponding effect on the calculated safe fuel mass limit.

2.3 Justification of Fuel Model

As discussed above, the fuel used in this evaluation was the TMI-2 average fuel, including the effects of burnup in all three fuel batches. This approach resulted in a net U-235 enrichment of 2.24 wt%, plus associated plutonium buildup. The enrichment was an analytically developed number which is considered to be a best estimate value, though it was developed based on conservative assumptions. The use of the fuel model with this enrichment is considered appropriate for use in this evaluation based on the following :

- The analytical approach used to determine the burnup credit for the fuel was conservative. Only limited credit is taken for the presence of fission product poisons, thus increasing the neutron multiplication of the modelled fuel. See References 3 and 5 for more details regarding the burnup evaluation.

- Defueling data indicates that the majority of the highest enrichment batch 3 fuel was removed from the reactor vessel 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 was collected from the lower head, where 34 samples were collected. Table 3 provides details of these samples. The weighted average U-235 enrichment for these samples was 2.23 wt%.

- The more recent enrichment data collected at fuel assembly location R-6 indicates a weighted average enrichment of 2.5 wt%, as shown in Table 4. Recognizing that R-6 is a batch 3 fuel assembly, it was anticipated that these samples would show an average enrichment which was greater than 2.24 wt%. Using the limited credit analytical approach for burnup, as discussed previously, of the batch 3 fuel, the resultant average enrichment is 2.67 wt%. The measured lower enrichment of 2.5 wt% indicates some mixing of the lower enriched fuel batches with the batch 3 fuel located at R-6.

- The enrichment data available for TMI-2 samples is provided in Table 5. This table includes the samples collected from lower head, R-6, and other ex-vessel locations. Typically uncertainties associated with the enrichment analyses were approximately 10%. It should be recognized that when interpreting the ex-vessel location data there was only a limited number of samples collected at these locations. Furthermore, a review of Tables 3 and 4 shows that a large variation in enrichment can occur even with samples that were collected in close proximity to one another.

- The accident scenario and subsequent defueling operations have enhanced mixing of the fuel debris, both within and external to the vessel. 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 predominantly batch 3 fuel. The debris is well mixed, and thus contains a substantial percentage of batches 1 and 2 fuel.

As discussed above, the analytically determined enrichment of 2.24 wt%, plus the associated plutonium buildup, is considered to be a best estimate value though it was developed using conservative assumptions.

The most comprehensive enrichment sample data collected to date at TMI-2, as well as other defueling data, supports this enrichment as being reasonable. Consequently, it is concluded that the use of the 2.24 wt% U-235 enrichment is appropriate for use in this evaluation. However, it is important to recognize that the use of this enrichment does not imply that the enrichment of all individual small samples of fuel debris throughout the plant will be less than 2.24 wt%. As shown in Tables 3 and 4, there is a wide variation in the enrichment data even among samples collected in close proximity to one another. Therefore, based on the available enrichment data with the recognition of the potential for sample variations and measurement uncertainties, as well as considering the potential fuel relocation pathways, it is concluded that the 2.24 wt% enrichment is a appropriate average for any significant (i.e.,>70 kg) accumulation of debris.

2.4 Quantifying the Effects of Impurities

The base case fuel model for this evaluation, as described Section 2.2, neglected the presence of any impurities in the fuel debris. This assumption is conservative in that all samples of TMI-2 debris accumulations collected to date have shown that the debris contains impurities (see Table 2). Available sample data from within the reactor vessel (References 6,7,10), the "B" steam generator tube sheet (Reference 14), the purification/makeup filters (Reference 15), and the pressurizer (Reference 16) all show the presence of impurities. Due to the numerous locations sampled and considering the potential fuel relocation pathways, these samples are considered to be representative of the fuel debris remaining at TMI-2. Of the above samples, those collected from the "B" steam generator tube sheet contained the largest percentage of fuel (i.e., on the order of 80 wt% uranium). On the other hand the sample results from the purification/makeup system filters showed approximately 5 wt% uranium, with a significant percentage of impurities.

Previous analyses showed that the impurities having the most effect on the neutron multiplication were boron and cadmium, mainly due to the large absorption cross sections of these elements. Review of the sample data results indicated that the samples taken from the "B" steam generator tube sheet contained relatively smaller amounts of these impurities when compared to the other samples, while, as noted above, containing a larger percentage of fuel. Consequently the "B" steam generator tube sheet sample data were conservatively used to develop a fuel debris model to assess the reactivity worth of the impurities.

From the data provided in Reference 14, an average impurity content was developed for use in an impurity effect evaluation. This average only considered the effects of boron, iron and zirconium. The poisoning effect of the cadmium was conservatively neglected. Any additional impurities were neglected and their mass was considered to be UO_2. For additional conservatism, and to account for measurement uncertainty, the impurity concentrations derived from Reference 14 were reduced by approximately 10% before being used in the impurity effects evaluation. The actual impurity concentration analyzed is shown in Table 6. Also reported in this table for comparison purposes is the initial TMI-2 core composition by elemental weight percent. Thus it can be seen that the impurity concentrations used in this analysis represent a significant conservatism over what would be derived assuming a homogeneous mixture of the initial core composition.

The enrichment of the fuel used in this analysis was conservatively chosen to be that corresponding to the highest enriched batch 3 fuel, including the effects of burnup. This resulted in a net U-235 enrichment of 2.67 wt%. An optimal amount (VF=0.28) of unborated water was used as the moderating material. The debris particle size considered was standard whole pellets.

An additional evaluation was performed to assess the sensitivity of the safe fuel mass limit calculations to the concentration of boron in the debris. In this case unburned batch 3 fuel (2.96 wt%) was assumed to contain 0.072 wt% natural boron. This boron concentration is almost a factor of ten larger than that considered for the above described impurity evaluation, though it is more representative of the samples collected from the lower head and conservatively represents the sample data for other plant locations outside of the reactor vessel (see Table 2). No other impurities were considered in this case. Additionally, standard fuel pellets size particles and optimum moderation with unborated water (VF=0.265) were assumed.

As with the base case model, in both impurity effects evaluations, the radius of the inner region of Figure 1 was varied until the resultant neutron multiplication was 0.99, including the 2.5% Δk computer code uncertainty bias.

The results of the impurity effects evaluations are presented in Section 3.2 of this document.

2.5 Computer Code Benchmarking

In Reference 3 an analytical uncertainty bias of 2.5% Δk, including the KENO V.a (Reference 11) statistical uncertainty, was established as an appropriate value for the highly borated systems being investigated in that report to define a safe boron concentration for the TMI-2 defueling program. Uncertainty values reported in the literature for unborated systems have been shown to be somewhat lower than this value (Reference 12). Consequently, the 2.5% Δk value is considered conservative for the criticality safety analyses provided in this evaluation. This bias is also considered acceptable and applicable for the XSDRNPM analyses performed in this evaluation since previous analyses (References 3, 4) demonstrate the good agreement between the KENO V.a and the XSDRNPM generated results.

2.6 Conservatisms

In the development of the base case criticality safety model for this evaluation, conservative assumptions were utilized. A summary of these conservatisms follows:

- Unborated water, in an optimal mixture with fuel debris, was assumed for the moderating medium.

- No credit was taken for the large amount of structural and solid poison materials existing in the debris.

- The equivalent of full standard sized fuel pellets was utilized.

- A spherical geometry, which minimizes the ratio of surface area to volume, thus maximizing k_{eff}, was utilized.

- The fuel was represented as TMI-2 average fuel (homogeneous mixture of all three fuel batches).

- An effectively infinite water reflector was utilized.

- The computer code uncertainty used in this analysis was conservatively assumed to be 2.5% Δk.

It is recognized that isolated regions may have fuel debris accumulations in which some of the assumptions may not be bounding (e.g., enrichment of isolated debris chunks may be greater than 2.24 wt%). However, considering all of the model assumptions, including those for geometry and fuel modelling, it is concluded that the overall model used for this evaluation is conservative and appropriate for determining a critically safe fuel mass limit for TMI-2.

3.0 RESULTS

3.1 Base Case Allowable Mass

As previously described, the significant assumptions used in the base case model of this evaluation were:

- **Optimum moderation using unborated water;**
- **No credit for structural or solid poison materials;**
- **Full size fuel pellets; and**
- A 2.24 wt% U-235 enrichment.

The results of this analysis were that the inner radius of Figure 1 resulting in a k_{eff} of 0.99 was 22.7 cm. This corresponds to a fuel mass of 141 kg. This analysis was performed by Oak Ridge National Laboratory (ORNL) using the computer program XSDRNPM (Reference 8) and was reported in Reference 9. Based on this result, a 140 kg safe fuel mass limit was adopted.

The corresponding volume of unborated water associated with this fuel mass was determined by:

$$V = 4/3 * \pi * r^3 * (1-VF)/3785.43$$

where:

V = volume of unborated water (gallons)
r = inner radius of Figure 1 (22.7 cm, Reference 8)
VF = fuel volume fraction (0.28, Reference 8)
1/3785.43 = conversion factor (cm^3 to gallons)

The resulting water volume was 9.3 gallons. This volume did not include the infinite thickness water reflector assumed in the analysis.

One of the most obvious results of this analysis is that the maximum allowable fuel limit (140 kg) essentially doubles the 70 kg limit that was established in Reference 1. Although a different analytical technique was used to develop the initial 70 kg value, the main reason for the increase in the amount of allowable fuel in the current evaluation is that the net U-235 enrichment in the current evaluation (2.24 wt%) was lower than the Reference 1 enrichment (3 wt%). Two factors contributing to this lower enrichment were the assumed mixing of the three fuel batches along with the incorporation of fuel burnup effects in all three fuel batches. As stated previously, the burnup effects incorporate selected fission products, some of which are strong neutron absorbers.

As mentioned above, the analytical technique used in the present evaluation did not significantly contribute to the difference in the allowable mass when compared to the Reference 1 value (140 vs. 70 kg). The k_{eff} criterion established for the calculations for the present evaluation (0.99, including a 2.5% Δk uncertainty bias) yielded an allowable mass that was approximately 75% of the minimum calculated critical mass (i.e., k_{eff}=1.00). The approach utilized in Reference 1 was that the allowable mass was set at a value corresponding to approximately 75% of the minimum critical mass. Consequently, the two approaches should provide similar allowable masses given similar baseline assumptions.

3.2 Effects of Impurities

The major assumptions considered in the first impurity effects evaluation were:

- **Optimum moderation using unborated water;**
- **Full size fuel pellets;**
- **A 2.67 wt% U-235 enrichment; and**
- **Minimal credit for impurities (0.009 wt% B, no Cd, Table 6).**

The result of this impurity effects evaluation was that the inner radius resulting in a k_{eff} of 0.99 was 23.8 cm (Reference 18). This resulted in a UO_2 mass of 169 kg. This case demonstrates the significant effect that a very small amount of impurities present in the fuel debris can have on the calculated safe fuel mass limit, even when considering an increased fuel enrichment. This result is considered applicable for the entire Post Defueling Monitored Storage period, as the sample results indicate that the boron was incorporated into the debris during the accident, and was not simply surface deposited boron. Therefore, the boron content is considered to be an integral part of the debris and, thus, expected to remain with the debris for the long term (Reference 17).

The major assumptions considered for the second impurity effects evaluation were:

- Optimum moderation using unborated water;
- Full size fuel pellets;
- A 2.96 wt% U-235 enrichment; and
- A debris impurity concentration of 0.072 wt% B.

The calculated infinite neutron multiplication factor (k_∞) for this case was 0.931 (Reference 22). This value is substantially lower than the 1.287 calculated for the impurity effects evaluation with the 0.009 wt% natural boron. With k_∞ less than the allowable k_{eff} (i.e., 0.99, including the addition of 2.5% Δk for computer code uncertainty), the allowable fuel mass becomes infinite. Thus it can be seen that if the additional effects of the larger measured boron concentrations or the cadmium were considered in the safe fuel mass limit analysis, the increase to the allowable fuel mass would be dramatic, even with the fuel enrichment being increased to 2.96 wt%.

Even though the safe fuel mass limit as calculated when considering the minimum presence of impurities (169 kg) is larger than the 140 kg adopted from the base case model, the 140 kg is still considered the bounding value and is thus considered the appropriate safe fuel mass limit for fuel debris accumulations within the TMI-2 plant.

4.0 CONCLUSIONS AND LIMITATIONS

This report shows that when more realistic assumptions are made regarding the composition of the fuel debris remaining at TMI-2, the critically safe fuel mass limit can be increased to 140 kg. This increase occurs even though of the major modelling assumptions used in the base case model, only the fuel enrichment was adjusted to be made more realistic. No attempt was made to adjust the other three major assumptions (i.e., impurity concentration, moderation, and particle size).

The above limit is considered applicable for isolated accumulations of fuel debris (i.e, those accumulations of fuel that will remain physically and neutronically decoupled from other fuel accumulations) at TMI-2. Fuel accumulations are considered neutronically decoupled if the equivalent of 12 inches of water separates the accumulations (Reference 13).

Based on the available sample data as well as considering the various fuel relocation pathways, the three cases presented in this evaluation are considered to bound any accumulations of fuel debris remaining at TMI-2 which are in excess of the 70 kg limit established in Reference 1. Also the degree of conservatism for a particular assumption can be modified and still demonstrate the appropriateness of the 140 kg limit. As examples of this, additional cases were provided where batch 3 fuel enrichments were used, with minimal credit for impurities and all other significant assumptions unchanged. These cases showed allowable masses in excess of 140 kg.

The above limit of 140 kg is not considered applicable in cases where the fuel debris is surrounded by a thick lead reflector (e.g., the shipping casks), as under certain conditions lead can be a better neutron reflector than unborated water. In such cases, separate evaluations will be performed.

5.0 REFERENCES

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Table 1: Fuel Model Composition for Base Case Model

Isotope	Number Density (atoms/barn-cm)
U-235	5.21 E-04
U-238	2.25 E-02
O-16	4.60 E-02
Pu-239	4.01 E-05
Pu-240	2.00 E-06
Pu-241	2.49 E-07
Sm-149	1.01 E-07
Sm-151	1.79 E-07
Eu-151	8.20 E-09
Eu-153	1.32 E-07
Eu-154	4.51 E-09
Eu-155	6.12 E-09

Notes: 1. Only the more significant isotopes are listed above.

2. All values are taken from Reference 21

Table 2 : Average Impurity Concentrations

Sample Description	Elemental Composition of Debris (wt %)					
	Cd	U	Fe	B	Zr	Reference #
OTSG "B"	0.06	82.9	0.29	0.01	1.44	14
Core Debris	<LLD	73.6	0.74	0.48	11.2	10
Lower Head	<LLD	64.7	2.2	0.072	12.8	6
Pressurizer	0.77	2.7	8.68	1.07	2.43	16
MUF-5B (B&W)	11	6	7	2	>25	15
MUF-5B (0104)	11.4	~5	5.7	0.62	5.4	15
MUF-5B (0105)	11.2	~5	5.2	0.64	5.7	15
MUF-5B (0111)	--	7.27	3.9	~0.1	12.6	15

Table 3 : Lower Head Core Debris Sample Enrichment Data (Ref. 6)

Sample #	U-235 (mg)	U-238 (mg)	Total U (mg)	Enrichment (wt%)
11-1-C-400	0.38	16.8	17.18	2.21
11-1-C-401	0.62	21.3	21.92	2.83
11-4-D-403	0.34	13.25	13.59	2.50
11-4-D-402	0.49	21.5	21.99	2.23
11-4-D-404	0.60	27.52	28.12	2.13
11-6-B-405	0.41	16.30	16.71	2.45
11-6-B-406	0.46	19.86	20.32	2.26
11-6-B-407	1.12	47.08	48.20	2.32
7-1-B-410	1.15	52.11	53.26	2.16
7-1-B-409	0.45	19.24	19.69	2.29
7-1-B-412	0.64	27.37	28.01	2.28
7-1-B-411	1.09	44.97	46.06	2.37
7-1-B-408	0.29	10.96	11.25	2.58
11-4-B-413	0.69	30.75	31.44	2.19
11-4-B-414	0.57	23.92	24.49	2.33
11-4-B-415	1.12	49.27	50.39	2.22
11-4-B-416	1.02	39.89	40.91	2.49
11-7-C-419	0.33	12.65	12.98	2.54
11-7-C-418	0.44	17.87	18.31	2.40
11-7-C-417	1.60	77.41	79.01	2.03
11-2-C-422	0.51	21.31	21.82	2.34
11-2-C-424	0.34	12.93	13.27	2.56
11-2-C-421	1.59	73.2	74.79	2.13
11-2-C-420	0.79	31.95	32.74	2.41
11-1-A-427	0.48	23.64	24.12	1.99
11-1-A-428	0.68	37.66	38.34	1.77
11-5-C-433	0.63	28.25	28.88	2.18
11-5-C-435	1.69	80.74	82.43	2.05
11-5-C-434	1.46	63.13	64.59	2.26
11-5-C-431	1.57	70.64	72.21	2.17
11-5-C-432	1.41	64.77	66.18	2.13
11-5-C-430	0.71	23.01	23.72	2.99
11-5-C-436	1.30	60.25	61.55	2.11
11-5-C-437	0.74	33.82	34.56	2.14
Totals	27.71	1215.32	1243.03	
Weighted Average				2.23

Note: Average Enrichment = Total (U-235)/ Total (Total U)

Table 4 : R-6 Sample Enrichment Data

Sample	Sample Weight (mg)	U Content (%)	Total U (mg)	U-235 (mg)	Enrichment (wt%)
R6-1A	468.2	74	346.5	8.3	2.4
R6-1B	204.9	77	157.8	3.8	2.4
R6-2A	399.8	76	303.8	7.3	2.4
R6-2B	265	77	204.1	5.3	2.6
R6-3A	347.5	76	264.1	6.3	2.4
R6-3B	502	77	386.5	8.9	2.3
R6-4A	495.2	74	366.4	9.5	2.6
R6-4B	477.8	75	358.4	9.3	2.6
Totals			2387.6	58.7	
Weighted Average					2.5

Notes: 1. Data taken from Reference 17

2. For each sample: U-235 = weight * (% U) * enrichment

3. Average Enrichment = Total (U-235)/ Total (Total U)

4. Uncertainty for uranium content and enrichment data are 10%-15%

Table 5 : TMI-2 Sample Enrichment Data

Sample Description	# of Samples	Average Enrichment (wt%)	Reference
Lower Head	34	2.23 (*)	6
Make Up Filter	1	2.30	19
RCBT	1	2.29	19
MUF 5B (0104)	1	2.56	15
MUF 5B (0105)	1	2.43	15
MUF 5B (0111)	1	2.81	15
R-6	8	2.5 (**)	17

Note: Uncertainties for enrichments are typically on order of 10-15%

* - See Table 3 ** - see Table 4

Table 6: Impurity Content of Fuel Debris (wt%)

Component	First Impurity Evaluation	Second Impurity Evaluation	TMI-2 Core Average*
UO_2	98.470	99.928	74.3 **
Zr	1.260	0.000	18.0
Fe	0.261	0.000	3.0
B	0.009	0.072	0.1
Cd	0.000	0.000	0.1
Others	0.000	0.000	4.5

* Taken from Reference 6

** U plus O percentages from Reference 6 were combined to get UO_2 percentage

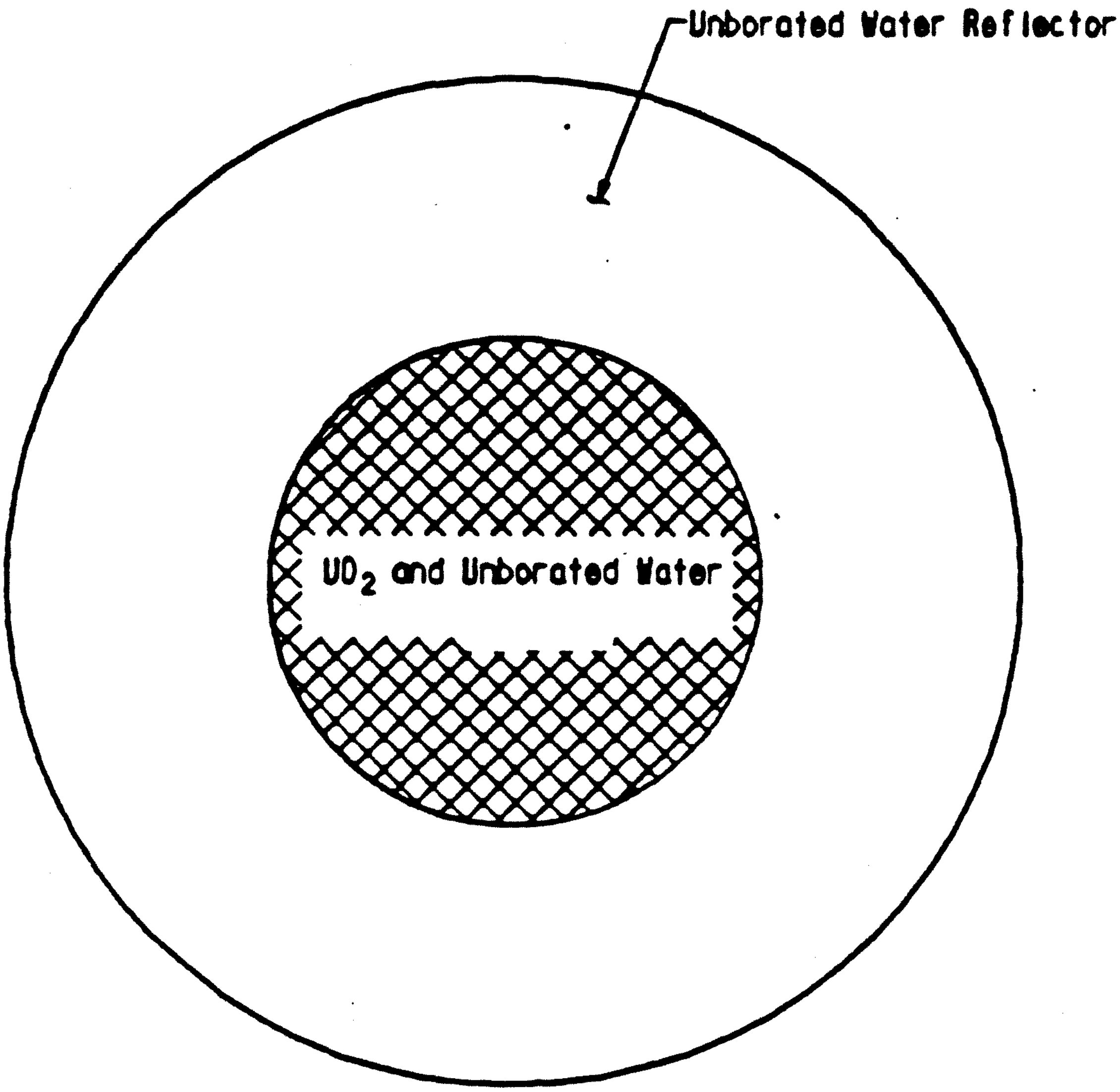

FIGURE 1: GENERIC SPHERE MODEL

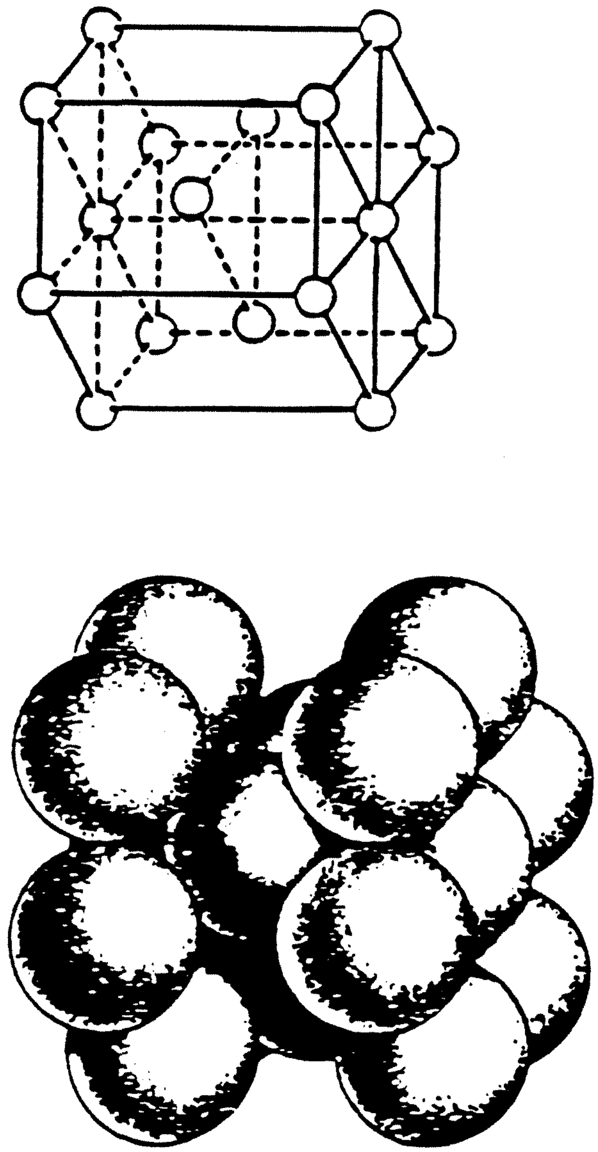

Figure 2: Fuel Model Lattice Structure