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U. S. Department of Energy Three Mile Island Research and Development Program 1989 Annual Report

Idaho National Engineering Laboratory
U.S. Department of Energy • Idaho Operations Office

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U.S. DEPARTMENT OF ENERGY THREE MILE ISLAND RESEARCH AND DEVELOPMENT PROGRAM 1989 ANNUAL REPORT

July 1990

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ABSTRACT

Defueling of the Three Mile Island Unit 2 (TMI-2) reactor continued through 1989. This report summarizes that work and other TMI-2 related cleanup, research, and development activities. The major topics in this report include:

- Waste immobilization
- Core debris transportation, receipt, and storage
- Accident Evaluation Program
- Technical Integration Program.

Significant progress was made toward completing the U.S. Department of Energy (DOE) programmatic effort of support to cleanup the TMI-2 facility. Completion of this effort is expected during 1990.

SUMMARY

Fuel and Waste Handling and Disposition Program

During 1989, the fuel waste handling and disposition program continued. Throughout the year, fuel debris canister preparation, canister acceptance, cask loading, and core rail cask shipping receipt and storage was conducted in accordance with the established procedures of the program.

By the end of 1989, 46 shipments of core debris had been transported in NUPAC 125–B rail casks, which hold seven fuel canisters each. A total of 322 canisters have been transported to the Idaho National Engineering Laboratory (INEL). About 98% of the core debris is now in storage at the INEL. This amount accounts for approximately 131,836 kg of the original TMI–2 core debris, which was estimated to be 134,945 kg. TMI–2 core debris will be stored at the INEL for up to 30 years or until a national high level waste repository becomes available.

Activities related to researching alternate core debris storage methods continued during 1989. Research progress in TMI–2 waste disposition was recorded during the year.

Accident Evaluation Program

Defueling operations and examination of core debris samples during 1989 provided additional information to (a) update the final TMI–2 accident scenario and (b) characterize the distribution and relocation of fission products and core materials inside the reactor core during the accident. This includes a discussion of events that lead to the formation of the central molten material and the relocation of molten material to the bottom of the reactor vessel.

Several calculations were performed during the year to evaluate the effect of molten core movement on reactor vessel internal components and the lower head of the reactor vessel.

Two specific accomplishments were completed during 1989: (a) the TMI–2 computer data base was completed and (b) standard problem exercise calculations on TMI–2 were completed by the task group of the Organization for Economic Cooperation and Development (OECD). A final Standard Problem Report will be published during 1990.

During 1989, final sample examinations, addressed as part of the TMI–2 Accident Evaluation Program, were completed. The results were reported in papers published in the August through December, 1989 issues of *Nuclear Technology*. These papers were presented at the TMI–2 Imbedded Topical Meeting, of the American Nuclear Society/European Nuclear Society International Conference, held October 30 – November 4, 1988 in Washington, D.C. A listing of the published papers is included in Appendix B.

Technical Integration Program

Defueling of the reactor vessel core continued during 1989. Disassembly, removal, and defueling of the lower core support assembly (LCSA), initiated during 1988, was completed during the year. Disassembly of the LCSA was necessary to gain access to fuel debris located on the lower head area of the reactor vessel.

Defueling of the lower head region of the reactor vessel was accomplished during 1989. This effort involved the removal of (a) loose core debris and (b) required the use of an impact tool to break up a mass of rock–like, resolidified material located at the bottom of the lower head region of the reactor vessel.

Defueling of the lower head region was followed by disassembly and defueling of the upper core support assembly (UCSA). UCSA disassembly and defueling was accomplished by removing core baffle plates and the defueling of the core debris that had migrated behind the baffle plates into the core former region of the vessel.

During 1989 the major bulk defueling operations of the TMI–2 reactor vessel were completed. Final defueling cleanup and inspection of the reactor vessel started during December, 1989. This activity is expected to be completed early in 1990.

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1. INTRODUCTION

Throughout 1989, defueling of the TMI-2 reactor and associated cleanup and research activities continued. These activities included:

- Removal of the remaining two plates of the original five plates that comprised the lower core support assembly. This was necessary to access the lower head region of the reactor vessel for defueling.

- Defueling of the lower head region of the reactor vessel.

- Disassembly of the upper core support assembly which involved disassembly of the core baffle plates and removing fuel debris located behind the baffle plates in the core former region of the vessel.

- Completing the major bulk defueling operations inside the vessel and initiating final cleanup and inspection activity.

- Transporting 12 cask loads of core debris from Three Mile Island to the INEL, bringing the total to 46 cask loads that have been received and processed at the INEL for storage. This represents to date 98% of the original postaccident debris material.

- Completing analytical work that determined the final TMI-2 accident scenario and the relocation of the molten material inside the nuclear vessel.

- Completion of the TMI-2 computerized data base that may be used to evaluate potential severe reactor accidents; international use of the data base by the Organization for Economic Cooperation and Development, TMI-2 Task Force to create a standard problem exercise to assess potential accident scenarios.

- Completion of final core material sample examinations, conducted as part of the TMI-2 Accident Evaluation Program, with results reported in papers published in the August through December, 1989 issues of *Nuclear Technology*. A listing of the published papers is included in Appendix B.

- Progress toward closure of the overall DOE programmatic effort with completion expected during FY-1990. This progress included reduction of staff at the Technical Integration Office at TMI; further transfer of programmatic responsibility to the INEL; and planning for the final efforts in reporting and records management including budget closeout.

2. FUEL AND WASTE HANDLING AND DISPOSITION PROGRAM

2.1 Waste Disposition

Steps taken during 1989 in the disposition of TMI–2 materials temporarily stored at the INEL include the following:

- Continuing the preparations for the disposition of the last four EPICORE–II prefilters, which were developed as a water treatment system used in decontaminating the approximately 2,120,000 liters of contaminated water generated by the TMI–2 accident.[a] The filters are to be disposed in concrete high–integrity containers (HICs) at the Radioactive Waste Management Complex (RWMC) located at the INEL. The HICs were developed for the disposal of EPICOR II prefilters used at TMI–2 and for the 46 filters that were previously disposed at the U.S. Ecology commercial disposal facility located in the state of Washington. The last four filters have been in a Nuclear Regulatory Commission (NRC) research program at the INEL and at other DOE laboratories. This disposition will mark the end of the NRC program of sampling the contents of the filters. The last samples of the program were taken during 1989.

  Planning for disposition included the preparation of detailed operating procedures for (a) sealing the four EPICOR II prefilters, identified as PF–8, PF–9, PF–20, and PF–27, in HICs, (b) loading each of the HICs into a transport cask, and (c) transporting the HICs to the RWMC for burial.

  Disposal of the HICs is expected to be completed early in 1990.

- Inventorying TMI equipment and material stored at the Test Area North (TAN), Central Facilities Area (CFA), and Test Reactor Area(TRA) facilities located at the INEL. The inventory of the TMI equipment and material stored at TAN has been completed; inventory of TMI equipment and material located at CFA and TRA is in process.

  Once the inventory has been completed at both facilities, each item will be evaluated for proper final disposition.

2.2 Core Transportation

Throughout the year, canister preparation and cask loading cycles were conducted in accordance with the program started in 1986. The process of transporting TMI–2 core debris to the INEL continued during 1989.[b] Table 1 shows all transports of debris through the end of 1989. In 1986, 5 shipments arrived at the INEL, 17 shipments arrived in 1987, 12 shipments in 1988, and 12 shipments in 1989. A total of 46 shipments have arrived at the INEL from 1986 through the end of 1989 (see Table 1).

2.3 Core Receipt and Storage

Figure 1 shows the percentage of TMI–2 core debris that has been transferred from Three Mile Island for storage at the INEL. The figure shows that approximately 98% of the core debris was transferred to the INEL by the end of 1989. This amount accounts for approximately 131,836 kg of the original TMI–2 core debris, which was estimated to be 134,945 kg.

The core debris was transported in casks by railcar to the INEL. The casks were delivered to the TAN facility by tractor/trailer where the TMI–2 debris canisters inside the casks were removed and stored in a water pit at the TAN Hot Shop.

During 1989, an alternate storage method for TMI–2 core debris was studied. The study evaluated the use of dry casks and development of a canister drying system for storing debris. The conceptual design effort for the dry casks was completed.

Storage of TMI–2 core debris at the INEL is planned for up to 30 years or until a national high–level waste depository becomes available.

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a. A detailed description of the EPICOR II water treatment system and HICs can be found in GEND Report No. 064, *U.S. Department of Energy Three Mile Island Research and Development Program 1988 Annual Report*, dated April, 1989.

b. The detailed procedures for loading, transport, receipt, and storage of a rail cask from TMI to the INEL is discussed GEND Report No. 064, *U.S. Department of Energy Three Mile Island Research and Development Program 1988 Annual Report*, dated April, 1989.

Table 1. Summary of core debris shipping campaign

Cask Shipment Number	Rail Shipment Number	TMI Shipping Date	Arrival at INEL	Return to TMI	Accumulated Number of Canisters
001	001	07/20/86	07/24/86	08/12/86	7
002	002	08/31/86	09/04/86	10/09/86	14
003	002	08/31/86	09/04/86	09/26/86	21
004	003	12/14/86	12/17/86	01/23/87	28
005	003	12/14/86	12/17/86	12/30/86	35
006	004	01/11/87	01/14/87	02/04/87	42
007	005	02/01/87	02/04/87	02/26/87	49
008	006	02/15/87	02/18/87	03/07/87	56
009	007	03/22/87	03/26/87	04/20/87	63
010	007	03/22/87	03/26/87	04/16/87	70
011	008	06/21/87	06/25/87	07/10/87	77
012	008	06/21/87	06/25/87	07/15/87	84
013	009	07/26/87	07/30/87	08/19/87	91
014	009	07/26/87	07/30/87	08/26/87	98
015	010	09/13/87	09/17/87	10/03/87	105
016	010	09/13/87	09/17/87	10/10/87	112
017	011	10/25/87	10/29/87	11/15/87	119
018	011	10/25/87	10/29/87	11/24/87	126
019	012	11/15/87	11/19/87	12/08/87	133
020	013	12/20/87	12/24/87	01/09/88	140
021	013	12/20/87	12/24/87	01/18/88	147
022	013	12/20/87	12/24/87	01/27/88	154
023	014	02/07/88	02/11/88	02/27/88	161
024	014	02/07/88	02/11/88	02/24/88	168
025	014	02/07/88	02/11/88	03/07/88	175
026	015	04/10/88	04/14/88	05/09/88	182
027	015	04/10/88	04/14/88	04/27/88	189
028	015	04/10/88	04/14/88	05/03/88	196
029	016	05/22/88	05/25/88	06/25/88	203
030	016	05/22/88	05/25/88	07/03/88	210
031	016	05/22/88	05/25/88	07/11/88	217
032	017	12/18/88	12/22/88	01/25/89	224
033	017	12/18/88	12/22/88	01/27/89	231
034	017	12/18/88	12/22/88	02/06/89	238
035	018	02/19/89	02/23/89	03/18/89	245
036	018	02/19/89	02/23/89	03/28/89	252
037	018	02/19/89	02/23/89	03/12/89	259
038	019	06/18/89	06/22/89	07/08/89	266
039	019	06/18/89	06/22/89	07/15/89	273
040	019	06/18/89	06/22/89	07/23/89	280
041	020	08/13/89	08/16/89	08/31/89	287
042	020	08/13/89	08/16/89	09/07/89	294
043	020	08/13/89	08/16/89	09/18/89	301
044	021	12/17/89	12/21/89	02/24/90	308
045	021	12/17/89	12/21/89	02/10/90	315
046	021	12/17/89	12/21/89	03/05/90	322

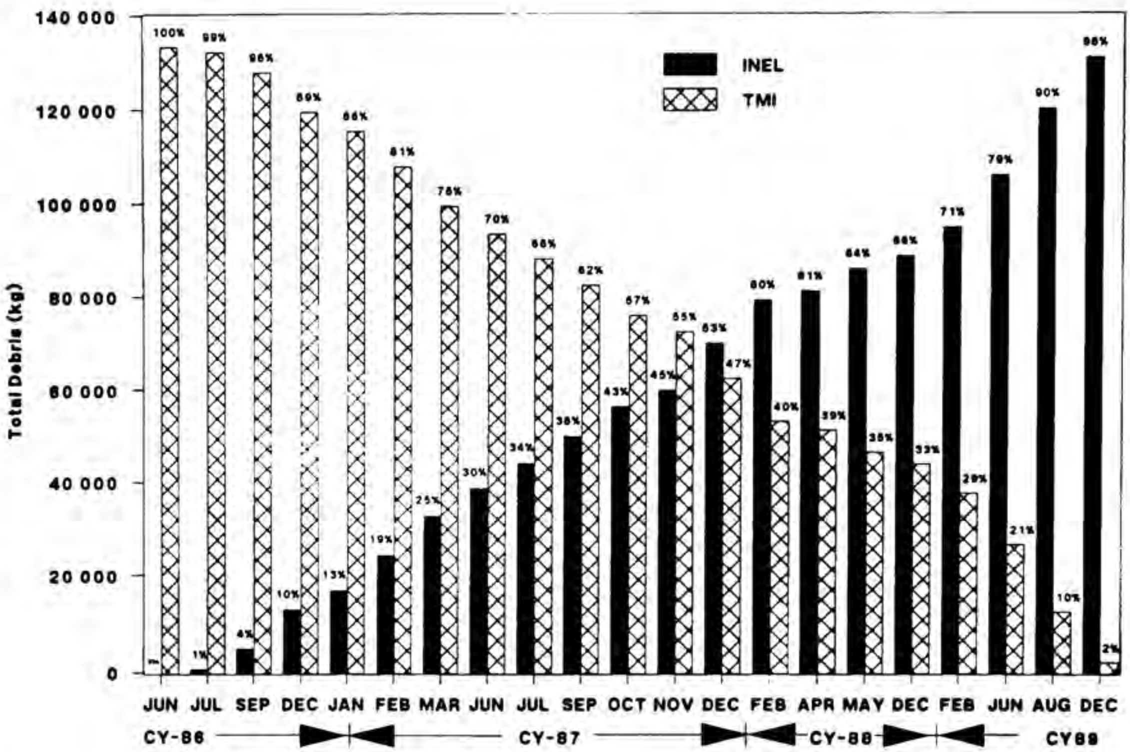

Figure 1. Location of TMI–2 core debris—Three Mile Island versus Idaho National Engineering Laboratory.

2.3.1 Canister Needs. Through the end of 1989, 423 canisters have been purchased for the defueling program (283 fuel, 65 knockout, and 75 filter). Of these, 331 canisters have been used (266 fuel, 55 filter, and 10 knockout).

2.4 Abnormal Waste

Three shipments of abnormal wastes have occurred: two in 1987 and one in 1988. None occurred in 1989. The wastes were transported to the INEL and are stored in concrete casks until disposal facilities exist for this class of waste or until the waste is otherwise processed for disposal. The casks are located on a concrete pad outside the TAN Hot Shop.

CUNO filters were used for all three waste shipments. CUNO filters were used in the submerged demineralizer system (SDS), water treatment system at TMI–2 to remove contaminated particles from contaminated water generated by the TMI–2 accident.

Methods for disposing the CUNO filters have been developed by EG&G Idaho, Inc. The process appears to have good promise for final disposal of the filters. First, the filter elements, core debris fines, and sludge would be dissolved with an acid. Second, the acid solution would be neutralized, mixed with cement, and solidified in 208–liter capacity drums. The drums would then be disposed. The CUNO filters and any residue would be disposed of separately. But, because of the expense of the process, GPU Nuclear requested that alternative disposal methods be studied during 1989. Accordingly, disposition of the CUNO filters had not been initiated by the end of 1989.

The contract period expired in December, 1989 between DOE and GPU Nuclear for acceptance of abnormal wastes. Therefore, it is presently not expected for DOE to accept any additional abnormal wastes. Also, at the INEL, responsibility for the continued monitoring of the wastes transferred from the TMI–2 Program to the TAN Hot Shop at the INEL.

3. ACCIDENT EVALUATION PROGRAM

The March, 1979 accident at the Three Mile Island Unit 2 (TMI-2) nuclear reactor was the most severe accident to occur at a commercial operating power reactor in the United States. At least 45% of the core was molten during the accident and about 20 metric tons of molten core material relocated from the central core region and came to rest on the lower head region of the reactor vessel.

The progression of the TMI-2 accident was mitigated by the presence of cooling water in the pressure vessel. Although the integrity of the lower head of the reactor pressure vessel was challenged by the molten core material, it did not fail. Therefore, the molten core material was confined within the reactor pressure vessel boundary. Very little fission product release occurred as a result of the accident.

TMI-2 provides a wealth of information that is being used to expand upon the present knowledge of severe nuclear reactor accidents, reactor core melt progression, and the associated behavior of fission products. This information can be used in the future to aid in accident management and recovery.

DOE has sponsored the TMI-2 Accident Evaluation Program to take full advantage of this important wealth of information. The objectives of the DOE Accident Evaluation Program are to:

- Understand what happened during the accident in terms of core melt progression; fission product retention and transport; and material temperatures, oxidation, and interactions
- Develop an accident scenario based on this understanding
- Produce a computerized data base containing TMI-2 research results, examination data, and supporting analyses; using the data base to produce a standard problem exercise to determine possible nuclear accident scenarios.

Sample acquisition and examination of materials from the reactor pressure vessel, the reactor coolant system, and the reactor containment and auxiliary buildings were conducted during the DOE Accident Evaluation Program. *U.S. Department of Energy Three Mile Island Research and Development Program 1988 Annual Report*, GEND Report No. 064, dated April, 1989 provides details and results of these examinations.

The following sections of this report describe the progress of the Accident Evaluation Program during 1989. The sections are:

- Final TMI-2 accident scenario
- TMI-2 data base and standard problem exercise
- Sample acquisition and examination program
- Information and industry coordination program.

3.1 Final TMI-2 Accident Scenario

The final TMI-2 accident scenario has been refined and formulated. It is based upon examinations performed during 1989. Specific examinations conducted during 1989 included (a) a study of the lower head of the reactor vessel after the removal of fuel and debris, and (b) a study of the core former region of the reactor vessel that was exposed when the core baffle plates were disassembled and removed during the defueling operations.

3.1.1 Accident Scenario. The accident at the TMI-2 pressurized-water reactor, including the relocation of core material to the lower head of the reactor vessel, started with a series of events progressing from problems in the secondary coolant water system.

Water from the secondary backup pumps failed to reach steam generators due to improperly closed valves in the water lines. With no water flowing through the steam generators, reactor coolant water heated and expanded. The water level in the reactor pressurizer started to rise and the system pressure increased. The increase in pressure caused the reactor to scram. The pressure increase also actuated a pressure relief valve causing the valve to open properly to relieve system pressure. Upon reaching specified pressure level, the relief valve was designed to close. This would stabilize the reactor system pressure. However, the pilot-operated relief valve stuck open and failed to close. With the relief valve stuck in the open position, pressure continued to fall and primary coolant water flowed out of the reactor core without being replaced.

During the first 100 minutes after the accident began, although there was coolant water loss, the primary coolant pumps provided two–phase coolant water to the core. This prevented overheating. However, shortly after 100 minutes two reactor coolant pumps were turned off. Steam and water separated, the water level fell, and the top of the core fuel assemblies started to uncover.

By 140 minutes, the core water level had dropped to about mid–core and the upper regions of the core heated sufficiently (1100 to 1200 K) to cause ballooning and rupturing of the metal cladding on fuel rods and control rods.

3.1.1.1 Formation of Molten Core Material. When the temperature reached about 1100 K, the silver, indium, and cadmium control material melted but was contained within the stainless steel cladding of the control rods. The first molten material to flow downward resulted from eutectic interaction (a) between the Inconel spacer grids and the zircaloy fuel rod cladding, and (b) between the zircaloy guide tubes and the stainless steel cladding of the control rods. This occurred at a temperature of about 1500 K. Temperatures of 1500 K were reached between 150 and 165 minutes.

Once the first liquid melt formed, additional eutectic interaction occurred more rapidly due to enhanced atomic mobilities in a liquid as opposed to a solid state. The eutectic interaction formed between Inconel and zircaloy reaction products, and the stainless steel control rod cladding which would allow the already molten silver, indium, and cadmium to join in the eutectic formation sequence. Together, the combination of liquid cladding, structural material, and control material was capable of flowing downward.

By 165 minutes, the water level had dropped to about 1 m above the bottom of the core. Any molten material that flowed down and reached the water would be expected to resolidify.

Sometime between 165 and 174 minutes, the temperatures in the upper regions of the core increased more rapidly when the melting temperature of cladding reached about 1700 K. Rapid oxidation of the zircaloy cladding begins at 1700 K. The heat associated with the oxidation of the zircaloy caused temperatures to elevate much more rapidly. Also, the stainless–steel control rod cladding melted at about 1700 K. This caused much more molten silver, indium, cadmium control material to be released at locations other than spacer grids. This material interacted with the zircaloy control rod guide tubes as did the molten stainless steel. After the cladding melting temperature was reached (2125 to 2245 K depending on oxygen content), the molten zirconium started to dissolve the uranium dioxide fuel pellets. This added uranium to the downward flowing melt. This downward flowing melt would also be expected to resolidify when it reached the water level.

Beginning at 174 minutes, sufficient coolant water to fill the reactor vessel was injected into the reactor using the 2B primary coolant pump. The water caused the oxidized fuel rod cladding in the upper region of the core to shatter. This caused the fuel rod assemblies to collapse into a bed of debris. This lead to the beginning of the formation of the upper core debris bed region. Although the 2B pump was on for 19 minutes, significant flow in the B–loop hot leg was measured for only 15 seconds.

At 200 minutes, the emergency core coolant system was activated and injected additional coolant water into the reactor vessel. This filled the reactor vessel in 7 to 10 minutes. The upper core debris bed probably quenched during this period.

The degraded core material continued to heat up after 180 minutes. By 224 minutes, calculations indicate that the central molten region, containing up to 42% of the core, was formed. A resolidified crust that had formed prevented the molten material from relocating and moving to other locations inside the core vessel.

3.1.1.2 Relocation of Molten Core Material. The primary relocation of molten material occurred between 224 and 226 minutes. It is believed that the supporting crust failed near the upper, peripheral region on the east side of the vessel core. The molten material relocated through peripheral fuel assemblies on the east side of the core and through the core former region where core baffle plates had melted exposing the core former region. The partitioning of the molten core material flow, through the peripheral fuel assemblies and the core former region, has not been resolved. As a result of the relocation of the molten material, an estimated 19.2 metric tons of molten core material flowed to the lower head region of the reactor vessel and challenged the integrity of the reactor vessel.

When the molten corium (i.e., molten core material) relocated to the lower head region of the reactor, a pressure pulse occurred. It started at 224 minutes and ended at 240 minutes, indicating that heat transfer and steam generation within the lower head debris was significant for at least 15 minutes. A source range monitor response indicated that some core material, probably molten corium from the consolidated molten region in the center of the core, may have continued to relocate through the core former region. This occurred between 230 and 930 minutes.

At 930 minutes after reactor scram, forced coolant water flow through the reactor pressure vessel was reestablished with one of the A-loop primary pumps. The accident had terminated.

3.1.2 Molten Core Movement Calculations. Several calculations were performed to evaluate the effect of molten core movement on reactor vessel internal components and on the lower head of the reactor vessel. These calculations included:

- The potential for an energetic molten fuel and coolant interaction (i.e., steam explosion which did not occur)
- The thermal ablation of the core baffle plates, core former plates, and reactor lower head by the impingement of molten core materials
- The thermal failure of the incore instrument guide tubes
- Creep rupture of the lower head under conditions of elevated temperature and pressure.

3.1.2.1 Molten Fuel and Coolant Interaction. A 20,000 to 40,000 kg mass of corium material (i.e., molten core material) is believed to be necessary to produce marginal failure through thermal degradation in the lower head region of a reactor vessel. However, a corium mass of only 500 kg was calculated to have been present, during the relocation of the TMI-2 core, to engage in a steam explosion. This indicates that the theoretical possibility for the failure of the TMI-2 reactor pressure vessel was not evident. In any event, the high pressure condition inside the TMI-2 vessel, when the molten core relocated, most likely prevented the occurrence of a classical steam explosion altogether.

3.1.2.2 Baffle Plate, Former Plate, and Lower Head Thermal Analysis. The time required for baffle plate melt-through was assessed, on the basis of conduction-controlled heat transfer, for an assumed geometry of semi-infinite molten debris coming in contact with a steel slab of finite thickness. Complete melt-through of the baffle plate wall thickness was calculated to occur in about 5 minutes. Calculations were also performed for a jet of corium material impinging on a baffle plate. Results of the corium jet impingement calculations indicate baffle plate melt-through will occur in 3 to 26 seconds. The length of time is dependent upon the diameter of the jet, the superheat in the jet of molten material, and the relocation duration.

An assessment was also made of (a) the migration behavior of corium material during relocation through the core former region and (b) the potential for melt ablation of the 0.03175-m thick, stainless steel core former plates.

A comparison of the area of the accident-induced hole in the baffle plates and the total area of the 80 holes in the former core plates at any elevation, indicated that the core former region would fill faster than it would drain. Hence the relocation of corium around the core periphery in the core former region. The time required for a completely corium-filled region between two core former plates to drain was calculated to be 15 seconds. Calculations indicated that the time required to start the melting of core former plates, at a distance of 15 cm from the baffle plate melt-through location on the core, is more than one minute. Thus, these calculations can only predict core former plate melting close to the baffle plate melt-through location.

The potential of thermal damage to the lower head of the reactor vessel was assessed for a jet impingement of corium material. Calculations indicated that it would take a jet impingement time of 15 to 20 minutes to ablate half way through the thickness of the lower head. For a jet impingement time of 1 to 2 minutes, which is believed to be the time that was required for the TMI-2 corium material to relocate to the lower head, little thermal damage would result to the lower head of the reactor vessel.

3.1.2.3 Incore Instrument Guide Tube Thermal Analyses. Several thermal analyses were performed to assess the potential for melting of the incore guide instrument tubes that penetrated the 13-cm thickness of the lower head. These analyses were conducted prior to observing that a few stainless steel guide tubes had melted during the relocation of corium material (see Section 4.1.2).

The analyses indicated that melting of the incore instrument guide tubes would be expected if they came in contact with (a) corium at temperatures in the range from 1600 to 1800 K, or (b) "metallic-like" debris at temperatures greater than 1620 K. Thermal attack by molten stainless steel approximately 200 K above its melting point was also assessed to lead to guide tube melting. However, corium solidification and plugging around instrument guide tubes were predicted. This would prevent core material from escaping the reactor vessel.

3.1.2.4 Lower Head Thermal Analysis. The thermal response of the lower head of the reactor vessel was calculated using the COUPLE/FLUID code for three assumed debris configurations covering the following range of conditions: (a) an upper bound case, (b) an intermediate bound case, and (c) a lower bound case. Thermal response was calculated for quenched and unquenched debris beds in each configuration. In all three cases for an unquenched debris bed, the temperatures increased with time, whereas, in all three cases for a quenched debris bed, the temperatures eventually decreased.

This study concluded that for the unquenched debris bed calculations, the ultimate strength of the 13-cm thick, lower head steel would be expected to be reached and the vessel would fail. Also, during the quenched upper bound case, although the calculations found that the temperatures decreased, the lower head would be expected to fail due to the high temperature conditions present in the reactor vessel. Thus, only the quenched lower bound and quenched intermediate bound represent realistic cases for the TMI-2 lower head.

Calculations were also performed to (a) determine the creep rupture potential of the vessel lower head and (b) estimate the margin-to-failure of the lower head. The calculations were conducted using the ABAQUS code for a quenched debris configuration for an intermediate bound case (porous debris bed resting on lower head). Results of the calculations indicated that the lower head wall would be expected to plastically deform, but both the plastic and creep strains would be in the 1% range, whereas creep rupture strains of about 35% are expected at 783 K. Thus, creep rupture of the vessel lower head is not expected and the margin-to-failure appears to be quite large.

3.2 TMI-2 Data Base and Standard Problem Exercise

During 1989, two specific accomplishments were completed: (a) the TMI-2 computer data base was completed and (b) standard problem exercise calculations on TMI-2 were completed by an international task force of the Organization for Economic Cooperation and Development (OECD).

The TMI-2 data base was completed and was distributed to various groups who use the data base to assess potential severe reactor accidents. The data base contains TMI-2 research results, examination data, and supporting analyses.

The TMI-2 standard problem exercise was an international standard problem for the OECD TMI-2 Task Force. A number of countries using a variety of codes, performed calculations using the TMI-2 data base to determine possible accident scenarios. The results of these calculations are being summarized in the final OECD Standard Problem Report that is scheduled for completion during 1990. The report will summarize differences in the various code-calculated results for the TMI-2 accident and will identify limitations in the codes used to perform the calculations.

3.3 Sample Acquisition and Examination Program

3.3.1 Core Examinations. During 1989, final sample examinations, addressed as part of the core examination program, were completed. Most of the results were reported in papers published in the August through December, 1989 special issues of *Nuclear Technology*. These issues summarize the results of the TMI-2 Accident Evaluation Program.

Final examinations of loose debris from the lower head of the reactor vessel indicate that the debris was U-Zr-O (uranium-zirconium-oxygen) that had reached temperatures greater than 2800 K. The examination of these samples may be continued, as part of the OECD-sponsored vessel investigation program, to obtain a better assessment of fuel and material interaction on the lower head of the reactor vessel.

Final reports to be published as part of the sample acquisition and examination program are the final Core Bore Examination Report and the Fission Product Inventory Report. These reports are scheduled for completion during 1990.

3.3.2 Distribution of Core Materials. The final distribution of core materials inside the TMI-2 reactor vessel, as determined during 1989, is shown in Table 2. The table lists the postaccident distribution of core materials at different core regions inside the TMI-2 reactor vessel and outside the vessel.

The largest percentage of the core materials (33%) consisted of intact fuel rods located in the core periphery and at the bottom of the reactor core. Examination of these damaged assemblies indicated that they had not lost any core materials inventory and that they were not subjected to high temperatures.

The remaining core material repositories ranged in composition from the prior molten fuel and structural materials in the central core consolidated region, to the

Table 2. Estimated postaccident core materials distribution

Core region	Estimated Quantity (kg)	Uncertainty[a] (%)	Percent of Total Core (%)
Intact fuel assemblies (Partially or fully intact)	44500	5	33.4
Central core region resolidified mass	32700	5	24.5
Upper core debris bed	26600	5	19.9
Prior molten material on the Lower reactor vessel head	19100	20	14.3
Lower core support assembly[b]	5800	40	4.3
Upper core support assembly[b]	4200	40	3.2
Outside the reactor vessel	100	—[c]	0.3

a. The uncertainty estimates are based on defueling operations. Those areas of the reactor vessel that have been defueled have relatively low uncertainties, whereas those which have not have relatively high uncertainties.

b. The lower core support assembly is that portion of the reactor vessel below the core which includes the lower grid assembly and five plates. The upper core support assembly is a coolant flow region outside the vertical core baffle plates, which make up the peripheral boundary of the core.

c. Estimates of the amount of fuel material outside the reactor vessel are based on nondestructive evaluations of reactor components in the reactor and auxiliary buildings. They range from 60 to about 430 kg.

mixture of intact and previously molten materials in the upper core debris bed.

The central consolidated mass region was composed of a center area of prior molten fuel materials surrounded by layers of crust material with differing compositions. Analysis of the data indicated that the upper crust was composed of 2450 kg of debris with an average density of 8.3 g/cm^3, and the lower crust was composed of 8760 kg of material with an average density of 7.3 g/cm^3. These nominal values have associated uncertainties of 30 to 40% due to the heterogeneity and distribution of the debris in the crust layers. This resulted in a total of 25,990 kg of prior molten debris between the upper and lower layers of crust.

The upper core debris bed (20% of core mass) was composed of a mixture of relatively intact fuel materials and prior molten fuel, structural material, and control material. The debris bed was made up of relatively friable material which had a density from 3 to 5 g/cm^3. Intact cladding shards and control material fragments were present in the debris.

The prior molten material that relocated to the lower head of the reactor vessel (19,100 kg) has been examined only at the surface of the bed. Examination of this material, which may not be representative of all material on the lower head, indicated that the debris was a homogeneous mixture of fuel materials (uranium, zirconium) with relatively small amounts of structural and control materials. Nondestructive examinations of the lower head suggest that structural materials may have relocated to this area of the reactor vessel. Examinations are in progress to better define the composition of this debris.

The remaining repositories in the reactor vessel (10,000 kg) have not been characterized, but are expected (a) to be similar in composition to the material found on the surface of the debris bed on the lower head of the reactor vessel or (b) to contain metallic materials similar to those found in the lower crust of the central consolidated mass region.

3.4 Information and Industry Coordination

During 1989, the information and industry coordination program principally focused upon (a) responding to special information requests and (b) completing the reporting of core examination results in the August through December, 1989 special issues of *Nuclear Technology*. These issues summarize the TMI–2 accident evaluation program. Funding was provided to allow authors to complete modifications to their papers as identified by the reviewers at *Nuclear Technology*.

The work of the information program, upon completion of the publication of the special issues of *Nuclear Technology*, is essentially completed. However, information requests and responses to public requests is expected to continue through 1990.

4. TECHNICAL INTEGRATION PROGRAM

4.1 Reactor Evaluation Program

During 1989 defueling of the TMI–2 reactor vessel continued. This effort resulted in the cumulative removal to date of 99.7% of the original post–accident debris material.

Defueling activities during 1989 included the following:

- Final disassembly and defueling of the lower core support assembly (LCSA)
- Defueling of the lower head of the reactor vessel
- Disassembly of the upper core support assembly (UCSA) baffle plates and defueling of debris trapped behind the plates
- Initiation of final reactor vessel defueling cleanup and inspection.

4.1.1 LCSA Disassembly and Defueling. Disassembly of the lower core support assembly (LCSA) was necessary to (a) gain access to the fuel debris located below the LCSA on the lower head of the reactor vessel and (b) gain access to debris located among the five plates that comprised the LCSA. Disassembly of the LCSA was initiated during 1988 and was completed during April, 1989.

Figure 2 shows the TMI–2 reactor vessel and the location of the LCSA inside the vessel. Figure 3 shows the construction of the LCSA and the plates that required disassembly.

The LCSA consists of a series of five interconnected, stainless steel plates (see Figure 3). The plates, from top to bottom, are: (a) lower grid rib section, (b) lower grid distributor plate,(c) lower grid forging plate, (d) incore guide tube support plate, and (e) the flow distributor head plate.

The first three plates were disassembled during 1988; the latter two plates during 1989.

Disassembly required (a) drilling to remove the vertical cylindrical members of the LCSA (support posts and incore guide tubes) and (b) cutting a large center section of each plate into pieces. The cut pieces were then removed from the reactor vessel and placed in storage inside the modified core flood tank 'A' which is located in the Reactor Building.

The following primary equipment was used in disassembling the LCSA:

- A core bore machine that used a coring–type drill bit for removing the LCSA vertical support posts and incore guide tubes. This drill rig was installed on the shielded work platform of the reactor vessel (see Figure 2).
- An automated cutting equipment system (ACES) that was used to cut the LCSA plates into sections. ACES was built with a plasma arc torch that used a high–velocity stream of high–temperature, ionized, nitrogen gas (i.e., plasma). To position the plasma arc torch cutter, ACES used a robotic arm that was attached to a computer–controlled bridge and trolley system that was suspended over the LCSA.

A detailed description of this equipment and the operation of the equipment is discussed in GEND Report No. 064, *U.S. Department of Energy Three Mile Island Research and Development Program 1988 Annual Report*, dated April, 1989. The report also details the disassembly of the first three of the five plates of the LCSA: lower grid rib section, lower grid distributor plate, and lower grid forging plate. Disassembly of the remaining two plates, the incore guide tube support plate, and the flow distributor head plate, is described in the following sections.

4.1.1.1 Disassembly of the Incore Guide Tube Plate. Most of the disassembly effort of the incore guide tube plate, the fourth in a series of five stainless steel plates in the LCSA, was completed at the end of 1988. Final disassembly and removal was accomplished early in 1989.

Prior to disassembly of the incore guide tube plate, loose debris and a few loose fuel rod segments located on the plate were removed. This was accomplished using the midi–airlift defueling method. It used a small nozzle and lift pipe to lift large–size and loose debris into buckets for loading into fuel canisters. A pump lift vacuum system was used to clear loose debris in areas from up to 30 cm below the incore guide tube plate.

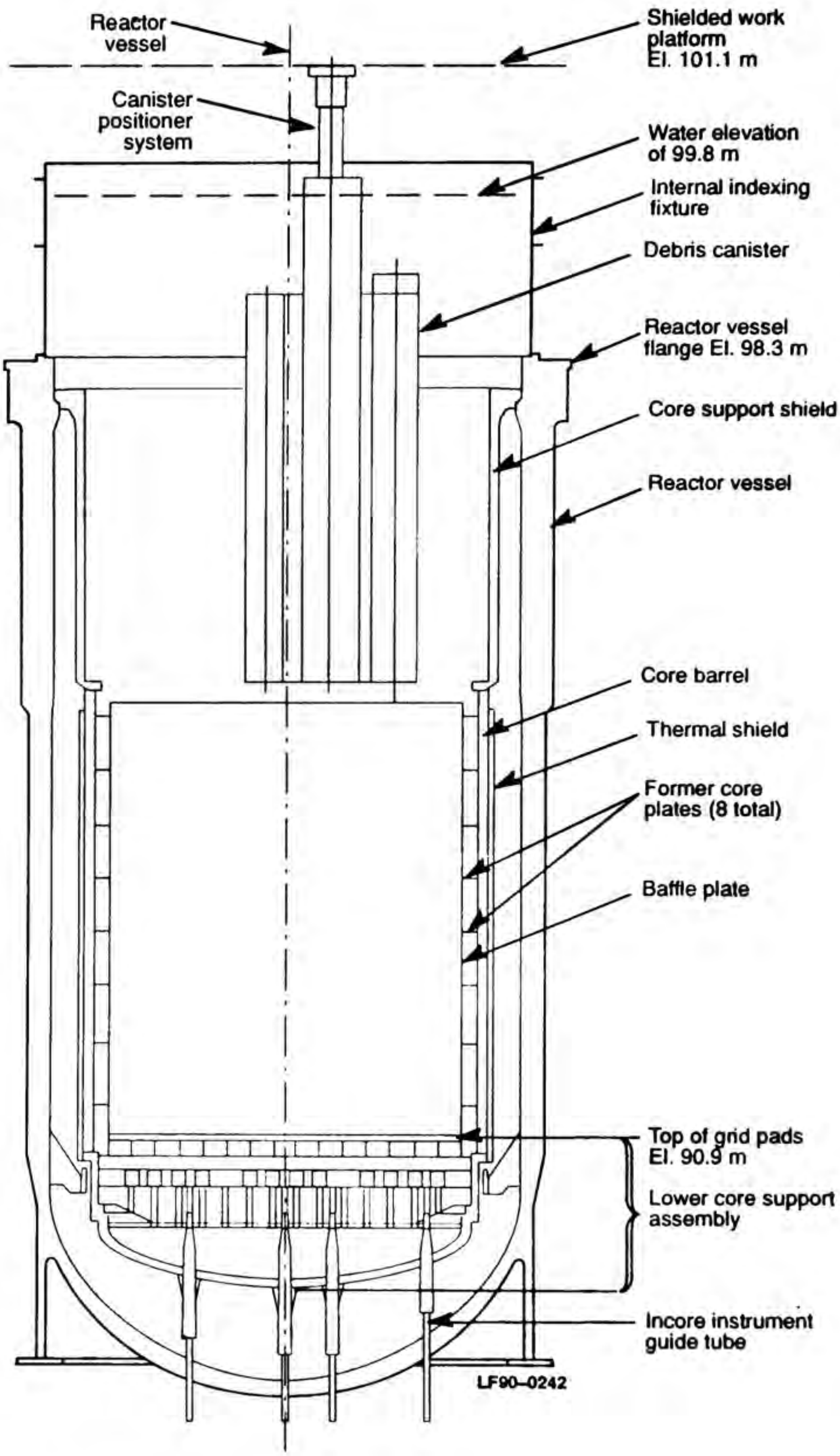

Figure 2. TMI–2 reactor vessel and defueling platform.

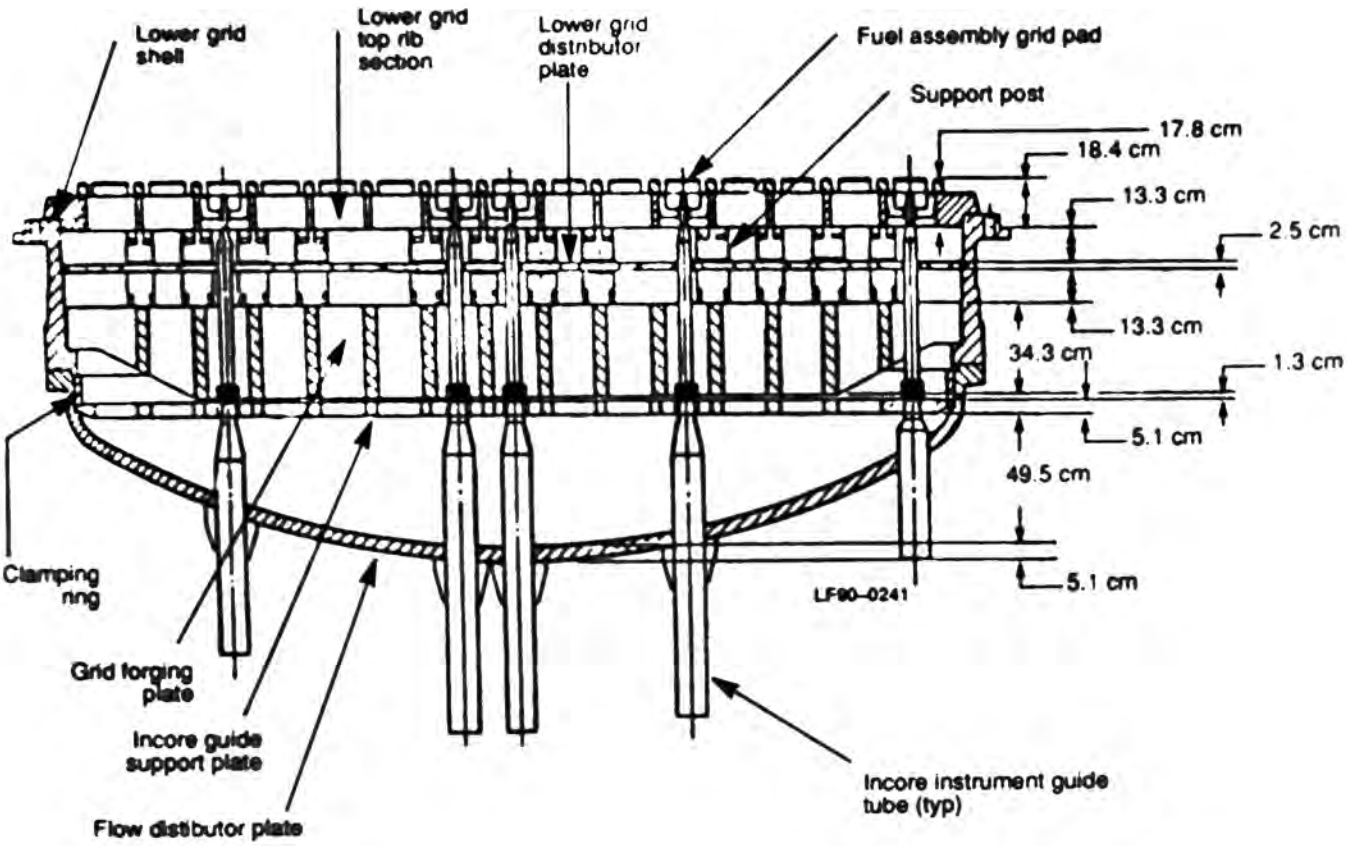

Figure 3. TMI-2 lower core support assembly.

Following removal of the pump lift vacuum from the reactor vessel, a hydraulically driven, abrasive saw was lowered into the vessel. This tool was used to cut the upper stubs of the incore instrument guide tubes (see Figure 3). These 36–cm high stubs penetrated the incore guide tube support plate and grid forging plate, the latter of which was removed from the reactor vessel during 1988. After cutting off the stubs, the saw was used to cut retaining locking weld tabs off the hold–down nuts. These nuts connected the incore instrument guide tubes to the incore guide tube support plate. A modified socket wrench was then used to remove the hold–down nut and a three–point gripper tool was used to remove the underlying washer of each incore instrument guide tube. This operation was repeated for 33 of the original 52 incore instrument guide tubes.

By disassembling the incore guide support plate in this manner, a substantial amount of plasma arc torch cutting time was saved. The severed stubs of the incore instrument guide tubes, hold–down nuts, and washers were then removed from the vessel.

In preparation for cutting the incore guide tube support plate, a specially designed, hydraulically driven, rotary brush was used to clean the plate to remove any remaining fuel debris. The plasma arc torch of the ACES was then reinstalled in the vessel to cut a large center section from the incore guide support plate. A total of 25 cutting operations were required to cut the center section into four quadrants.

After removing the plasma arc torch, the four quadrants were lifted from the LCSA by a polar crane, flushed under water, and transferred from the reactor vessel to the core flood tank 'A' for storage.

Three of the quadrants were removed without difficulty. The fourth quadrant could not be removed because one cut on the outer periphery of the LSCA was not made all the way through the plate. Instead of reinstalling the ACES with its bridge, a long pole ("pogo stick") with used to deploy the plasma arc torch to make the final cut. The quadrant was then removed.

4.1.1.2 Disassembly of the Flow Distributor Head Plate. Prior to disassembly of the 5–cm thick, flow distributor head plate, the fifth in a series of five stainless steel plates in the LCSA, defueling operations were conducted to remove debris from the top of the flow distributor plate in preparation of cutting the plate into sections.

Pick–and–place tooling was used to remove large fuel debris from the flow distributor head plate. The large debris and fuel rod segments were removed using a hydraulically operated, long–handled pick–and–place gripper. The gripper was used to remove debris that was too large for airlift vacuum removal. Defueling was also conducted using the midi–airlift vacuuming method to collect smaller, loose debris located on the flow distributor head plate and also for capturing loose debris up to six inches below the flow distributor plate in the lower head region of the reactor vessel. Access to the lower head region was gained through flow holes in the flow distributor plate. The debris was placed in buckets which were then emptied into fuel canisters.

Following removal of the defueling equipment, the plasma arc torch of the automated cutting equipment system was installed inside the reactor vessel to cut the flow distributor head plate into sections. A total of 104 cutting operations were required to cut the distributor head plate into 26 pieces. The large number of pieces were required because incore instrument guide tubes protruded through the plate. This increased the complexity of the cutting patterns. Thirteen of the 26 pieces contained incore instrument guide tubes that were cut free. To facilitate removal of 13 plate pieces containing guide tubes, nuts were installed on the majority of the 30 guide tubes in order to provide lifting points.

The pieces were removed from the reactor vessel and placed in storage. After the pieces were removed, access to the lower head region of the reactor vessel was possible through the irregularly shaped hole in the flow distributor plate. The hole measured roughly 254 cm across.

Figure 4 illustrates the remains of the flow distributor head plate, the peripheral remains of the LCSA, and the configuration of the lower region of the reactor vessel after the LCSA was disassembled and removed from the reactor vessel.

4.1.2 Reactor Vessel Lower Head Defueling. Disassembly of the LCSA provided direct overhead access to fuel debris located in the lower head region of the reactor vessel. Defueling operations began with pick–and–place defueling of loose debris. Large debris was removed using a hydraulically operated, long–handled core debris digger, and pick–and–place gripper. The debris was collected in debris buckets and was loaded into fuel canisters for storage in core flood tank 'A'. Midi–airlift defueling was performed to airlift smaller debris into buckets.

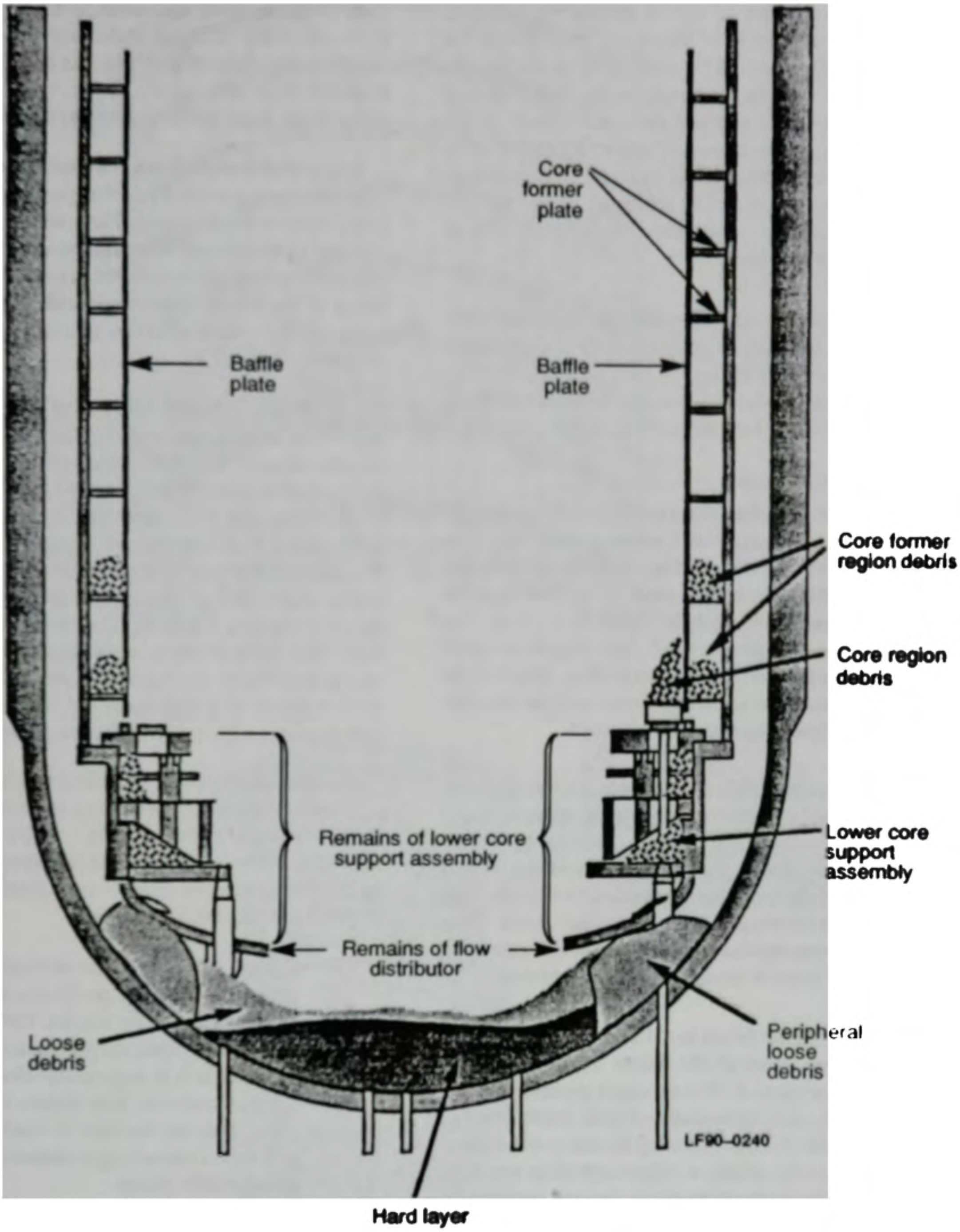

Figure 4. Illustration showing the configuration of the TMI-2 reactor vessel and the location of core debris after the lower core support assembly was removed.

The high volume of airlift defueling activity resulted in a high level of suspended particles in the reactor coolant water. The suspended particles obscured water visibility throughout the vessel during airlift defueling. To enhance the water clarity in the reactor vessel, a new in–vessel water filtration system was installed. Although the system was unable to maintain water clarity during airlifting, it increased the water visibility recovery time afterward by 50% to approximately two hours.

Pick–and–place and midi–airlifting defueling operations were alternately performed to collect loose debris prior to breaking up the hard–packed, resolidified mass of material, located in the center of the debris bed at the bottom of the lower head of the vessel (see Figure 4).

Most of the rock–like mass readily broke apart after methodical conditioning with a core impact tool. The tool consisted of a 46–cm long, star chisel that was driven into the resolidified mass by a 168–kg slide weight dropped from a height from 1.8 to 3.7 m. The impact tool was applied in a full, 360–degree sweep of the lower head region that was accessible. Much of the mass was reduced to small size debris compatible with midi–airlift defueling operations.

Approximately 80% of the mass was broken up. A central region of the mass measuring approximately 30 to 60–cm wide and 20 cm deep could not be broken up—possibly due to the congestion of the debris around it and the limitation on the height that the slide weight could be lifted to apply an impact force. Once clear access was obtained, defuelers were successful in reducing the mass to loadable sizes for removal.

After most of the debris in the lower vessel head had been removed, most of the incore instrument guide tubes were exposed. These tubes penetrated the 13–cm thick, bemispherically–shaped lower head of the reactor vessel. Prior to cutting the tubes and removing them from the vessel, a video inspection was conducted. Video inspection of the incore instrument guide tubes indicated damage to some of the tubes. Of the 52 tubes, 8 appeared to be severely damaged; approximately 20 showed some damage. At two tube locations, only 2.5 to 3.8 cm remained of the original tubes. Surface discontinuities were observed where the two tubes are welded into the lower head.

After cleaning the surface with a brush tool, another video inspection was conducted in the area where the surface discontinuities were seen. Several cracks were confirmed in the immediate area of the two tubes. The largest crack appeared to be approximately 0.16 cm deep, 0.24 cm wide, and 15 cm long. The cracks were apparently confined to the 0.48–cm thick, stainless steel liner of the 13–cm thick, carbon steel lower head. Some indication of rust was also observed.

It appeared that the cracks did not extend into the incore instrument guide tubes that penetrated the lower head of the reactor vessel. The cracks only radiated into the heat affected zone associated with the welds on the incore guide tubes. Nothing indicated any weakening of the lower head vessel itself, nor any hazard with respect to the final stages of defueling the reactor vessel.

4.1.3 UCSA Disassembly and Defueling. The upper core support assembly (UCSA) includes the vessel area located immediately above the lower core support assembly (see Figures 2 and 4). This area consists of an interfacing wall comprised of 36 vertical core baffle plates that surround and shape the core region. The plates are bolted to eight horizontal rows of core former plates that are attached to the circular core barrel (see Figures 2 and 4). The immediate location behind the baffle plates is referred to as the core former region (see Figure 4). Figure 5 shows a diagram of the reactor vessel that illustrates the eight rows of core former plates with the baffle plates removed.

Disassembly of the core baffle plates was necessary in order to remove fuel debris that migrated behind the baffle plates to the core former region during the TMI–2 accident (see Figure 4). Removal of the core baffle plates from the core former plates required the following disassembly operations:

- Cutting the baffle plates vertically at eight locations around the periphery of the reactor vessel baffle plate region. Cutting was accomplished using the plasma arc touch which was attached to a specially–designed manual manipulator. The plates were cut into eight sections because it would be difficult and time consuming to remove the 36 individual baffle plates.

- Removing 864 bolts that secured the baffle plates to the eight rows of core former plates. To remove the bolts, either a hydraulic wrench or drill was used. The hydraulic wrench was used to unscrew most bolts in the normal fashion. A drill was used to drill out galled, broken, or heat–distorted bolts.

Each baffle plate section was removed by installing two clamps connected by a cable to the top of the section. The section was then pried away from the

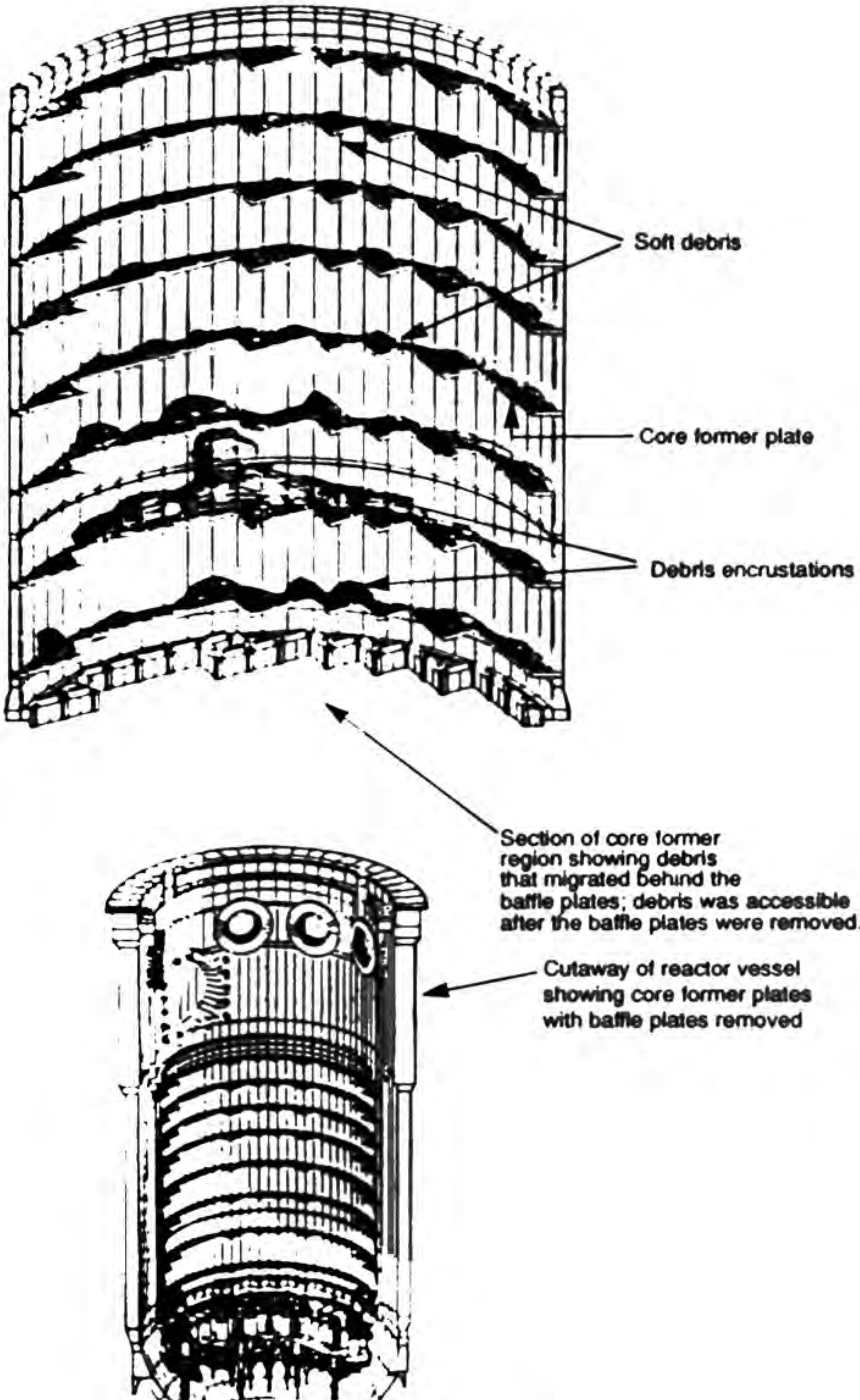

Figure 5. Cutaway of TMI-2 reactor vessel and section of core former region.

horizontal core former plates. The baffle plate section was then cleaned front and back and transferred to two clamp hooks on the opposite side of the reactor vessel where it was suspended by a cable.

After the section was removed, the exposed debris that had collected behind it on the core former region was removed by vacuuming. Where debris was too large for vacuuming or debris adhered to structures, long–handled tools were used to knock the debris loose or to relocated it into the lower regions of the reactor vessel for defueling. A high pressure, water cavijet, capable of discharging 0.76 l/s at 82,737 kPa, was employed for removing resolidified material.

The process of removing a section and defueling the exposed core former regions behind it continued until all eight areas were cleaned. As each baffle plate section was removed, and eventually hung in an alternate position corresponding in shape to its original position, the core former regions that had been defueled were recovered. In an effort to prevent agitated, fine debris from redepositing in the recently cleaned core former region, protective shrouds were installed on the repositioned baffle plate sections.

Disassembly of the upper core support assembly was the last major area inside the reactor vessel to be defueled. Upon completion of the core former region defueling effort, defueling of the lower regions of the reactor vessel was conducted to remove debris that had accumulated there during defueling of the core former region. This was the last major bulk defueling operation inside the vessel.

4.1.4 Final Reactor Vessel Defueling. Final cleanup and inspection activity inside the reactor vessel involves vacuuming loose debris and fuel fines. Some minor pick–and–place defueling may also be necessary. Final defueling started at the top of the vessel and will methodically progress downward to the bottom of the reactor vessel.

Final defueling activity inside the reactor vessel started in December, 1989. Final cleanup and final inspection of the vessel are expected to be completed early in 1990.

APPENDIX A

1989 TMI-2 TECHNICAL BULLETINS AND TPO/TMI DOCUMENTS

APPENDIX A

1989 TMI–2 TECHNICAL BULLETINS AND TPO/TMI DOCUMENTS

Index of 1989 TMI–2 Technical Bulletins

TB Number	Title	Dated
TB–86–12	Defueling Canisters Transfer Log	
	Rev. 7	01–11–89
	Rev. 8	07–21–89
	Rev. 9	10–19–89
TB–86–28	Auxiliary Building Sump, Sump Tank, and Valve Gallery Reactor Fuel Quantification	
	Rev. 1	09–21–89
TB–86–33	Rev. 16—Offsite Shipment of Defueling Canisters	04–11–89
	Rev. 17	07–18–89
	Rev. 18	08–15–89
	Rev. 19	12–21–89
TB–86–37	Deposition of Fuel on the Inside Surfaces of the RCS	
	Rev. 1	11–29–89
TB–86–38	Summary of Fuel Quantities External to the Reactor Vessel	
	Rev. 2	01–10–89
	Rev. 3	01–24–89
TB–87–21	Conditions in the Reactor Vessel (A Summary)	
	Rev. 2	01–10–89
TB–89–01	Validation of Microshield 3 Computer Code	02–10–89
TB–89–02	Lower Head Debris Topography	02–27–89
TB–89–03	Estimated Core Material Distribution	03–09–89
TB–89–04	Dose Rates from a Drained Reactor Vessel	03–28–89
TB–89–05	Incore Instrument Guide Tube Gamma Scan	03–21–89

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TB–89–06	Upper Plenum Assembly Post–Draindown Dose Rates	04–12–89
TB–89–07	Video Inspections of Incore Instrument Guide Tubes	05–23–89
TB–89–08	Final Core Material Estimates	10–19–89
TB–89–09	Radionuclide Corrections at TMI–2 for 10 CFR 61 Compliance—Summary of EPRI Report NP–6454, Project 2558–2	12–06–89
TB–89–11	Fuel Estimate of Upper Endfitting Storage Containers Stored in the 347' Elevation of the Reactor Building	11–22–89
TB–89–12	R–6 and Lower Vessel Debris Final Examination Results DWA–43–89	12–12–89

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SNM–88–05	Pressurizer Fuel Valve After June 1988 Defueling	
	Rev. 2	09–29–89
SNM–88–07	Letdown Cooler Room SNM Accountability Summary	
	Rev. 1	09–29–89
SNM–89–01	Lower Core Support Assembly (LCSA) Lower Grid Rib Forging (LGF) SNM Accountability Summary	01–25–89

Index of 1989 TPO/TMI Documents

Document No./Type	Title	Revision/ Date
TPO/TMI-125 Data Report	Reactor Building Radiological Characterization (Vol. 1)	Rev. 2 12-89
TPO/TMI-125 Data Report	Reactor Building Radiological Characterization (Vol. 2)	Rev. 2 12-89
TPO/TMI-191 Planning Study	A Study on Long-Term Disposition of the Reactor Vessel (An Evaluation of Options)	Rev. 0 10-89

APPENDIX B

NUCLEAR TECHNOLOGY
THREE MILE ISLAND UNIT 2 PAPERS
AUGUST 1989 – DECEMBER 1989

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Bottomley, Paul David Coquerelle, Michel	Metallurgical Examination of Bore Samples from the Three Mile Island Unit 2 Reactor Core
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Cronenberg, August W. Tolman, E. L.	Thermal Interaction of Core Melt Debris with Three Mile Island Unit 2 Vessel Components
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Hoffman, Peter Hagen, Siegfried J. L. Schanz, Gerhard Skokan, Alfred	Reactor Core Materials Interactions at Very High Temperatures
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