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TECHNICAL REPORT

TMI-2 REACTOR VESSEL LOWER HEAD INTEGRITY PROGRAM PLAN (DRAFT)

Richard K. McCardell

Prepared for the
U.S. NUCLEAR REGULATORY COMMISSION

EG&G Idaho

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(Draft Preliminary Report for Comment)

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ABSTRACT

A plan for the accomplishment of the TMI-2 Vessel Integrity Program, which is sponsored by the United States Nuclear Regulatory Commission in partnership with the Organization for Economic Co-operation and Development, is presented. The end state core conditions and the presently understood scenario of the TMI-2 accident are described and the analysis that has been performed to understand the attack of the reactor pressure vessel internals and the vessel lower head are reviewed. A method for determining the margin-to-failure for the TMI-2 lower head is incorporated in the program plan which should supply analytical tools based on the understanding of the accident that can be used for accident mitigation and management. The integration of the NRC/OECD Vessel Integrity Program and the DOE Accident Evaluation Program is described. The results of these two programs should provide a complete and accurate understanding of the TMI-2 accident.

SUMMARY

The recovery, cleanup, and debris inspection and examination of the Three Mile Island Unit 2 (TMI-2) pressurized water reactor plant, which underwent a small break loss-of-coolant accident on March 28, 1979, has provided a wealth of information that is being used to expand upon the present knowledge of severe accident and core melt progression and the associated behavior of fission products, and that can be used in the future to aid in accident management. The United States Department of Energy (DOE) has sponsored the TMI-2 Accident Evaluation Program to take full advantage of this important information. The DOE Accident Evaluation Program (AEP) included sample acquisition and examination for materials within the reactor pressure vessel, within the reactor coolant system, and within the containment and auxiliary buildings. Although the DOE AEP has or will cover all of these areas with respect to the program objectives, when the DOE program was established it was not known that almost 20 metric tons of previously molten material had relocated to the lower head of the TMI-2 reactor pressure vessel and therefore there was no provision in the DOE program for acquisition and examination of samples of the reactor vessel lower head nor of the material directly adjacent to the vessel lower head.

Although almost 20 metric tons of molten core materials relocated from the TMI-2 core region and came to rest on the reactor pressure vessel lower head, the lower head did not fail. Since the margin-to-failure of the reactor vessel lower head is vitally important to future accident management strategies, the United States Nuclear Regulatory Commission (NRC) formed an agreement with the Organization for Economic Co-operation and Development (OECD) to investigate the TMI-2 reactor pressure vessel lower head and companion samples of debris adjacent to the lower head. This program is denoted as the NRC/OECD TMI-2 Vessel Integrity Program and this document is a plan for the accomplishment of the NRC/OECD program.

In order to understand the TMI-2 Vessel Integrity Program Plan, it is necessary to have an understanding of the post-accident core conditions.

the way that it is believed that the accident progressed (the accident scenario), and the analysis that has been performed to date on the challenges to the reactor pressure vessel internals and lower head resulting from the relocation of the molten core debris.

The post-accident core conditions were determined from: visual inspections of reactor vessel internals; metallurgical/radiochemical examinations of samples acquired during the course of defueling the reactor; readings from on-line instrumentation during the accident; and experimental data developed from tests sponsored by the NRC in various facilities including ACRR, LOFT, NRU, and PBF.

The major physical damage to the plant was limited to reactor pressure vessel internals. Small quantities of fission products were transported to the containment building through the stuck open pilot-operated relief valve (PORV, which was the "small break" causing the loss-of-coolant) and to the auxiliary building through the letdown, makeup and purification systems. Only a very small amount of core materials were transported from the core region to the reactor coolant system by coolant flow and this consisted of fine silt and small particles.

During the accident, peak temperatures ranged from approximately 3100 K at the center of the core (molten UO_2) to 1255 K immediately above the core to 723 K at the hot leg nozzle elevations. The upper grid structure (stainless steel) was ablated in two zones. At least 42% of the original core melted during the accident. Lower portions of three baffle plates on the east side of the core melted and some of the molten material flowed into the core bypass region.

The end-state configuration of the original core region included a core void or cavity at the top of the core region that was surrounded by portions of 42 of the original 177 fuel assemblies. Below that, a loose debris bed rested on a resolidified mass of material that was supported by standing fuel rod stubs that, in turn, were surrounded by portions of fuel assemblies. Most of the loose debris bed consisted of shattered and resolidified material that contained regions of previously-molten U-Zr-O, indicating peak temperatures greater than 2200 K. There were also many debris bed samples of previously molten $(U,Zr)O_2$, indicating peak temperatures greater than 2800 K, and a few samples of previously molten UO_2, indicating temperatures of 3100 K.

The large, crust-enclosed, resolidified mass below the debris bed was approximately 3 m in diameter, 1.5 m thick at its center, and 0.25 m thick at its periphery. The center of this resolidified mass consisted of a mixture of structural, control, and fuel material that reached temperatures of 2700 K and possibly as high as 3100 K during the accident. The bulk composition of the metallic phases in the central region consisted primarily of Fe-Ni and Ag-In phases and comprised about 15% of the central region. Thus, the central portion of the resolidified mass contained much more metal than the upper debris bed which was essentially devoid of metallics. The upper and lower crust regions of the resolidified mass also contained metallics. The upper consisted of 25% metallics which were in three phases, Fe-Ni, Ag-In-U, and Ni-Sn. Ceramics, which comprised the other 75% of the upper crust, reached temperatures of at least 2700 K. The lower crust consisted of previously molten metallic material surrounding vertical fuel pellet stacks. There was a high percentage of zirconium in the lower crust metallics which consisted of three phases, Zr-Fe-Ni-Cr, Ag-In, and Zr-Ni-In. The maximum temperature of the lower crust was between 1400 and 2200 K.

The core bypass region consists of vertical core 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. Inspection of the baffle plates indicated that they were in good condition with one notable exception: on the east side of the core there was a large hole approximately 0.6 m wide and 1.5 m high, extending across the lower portion of three baffle plates. Molten core material from the core region flowed through this hole into the core bypass region.

The core support assembly region consists of five stainless steel structures: four flow plates and a lower grid assembly. Liquiefied material flowed through the periphery of these structures and came to rest on the lower head. There was resolidified material at various locations on the circumferences of these structures. The largest accumulation of resolidified material appeared to have flowed into the core support assembly from the east side of the core.

The debris resting on the lower head accumulated to a depth of 0.75 to 1 m above the lowest head elevation and to a diameter of 4 m. Lower head particle sizes ranged from large "rock" (up to 0.2 m) to granular sizes (less than 1 mm). The peak temperature of the lower head debris was in the range of 2700 to 3100 K.

The TMI-2 accident was initiated by a series of events progressing from problems in the secondary system that eventually led to opening of the PORV which stuck open. For the first 100 min after the accident was initiated, although there was coolant loss, the primary coolant pumps provided two-phase coolant to the core preventing overheating. However, shortly after 100 min the last two reactor coolant pumps were turned off and the top of the core started to uncover and water separated into steam and liquid phases. By 140 min the core liquid level had dropped to about mid core and the upper regions of the core would have heated sufficiently (1100 to 1200 K) to result in ballooning and rupture of the cladding. When the temperature reached about 1100 K the silver, indium and cadmium control material melt but is contained within the stainless steel cladding. The first melt to form that is free to flow downward results from the eutectic interaction between the Inconel spacer grids and the Zircaloy fuel rod cladding; and, between the Zircaloy guide tubes and the stainless steel cladding of the control rods at about 1500 K. Temperatures of 1500 K were reached between 150 and 165 min. Once the first liquid is formed additional eutectic (such as 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) forms more rapidly due to enhanced atomic mobilities in the liquid as opposed to the solid state. The combination of liquid cladding, structural and control materials is then capable of flowing downward. By 165 min the water level had dropped to about 1 m above the bottom of the core and any molten material that flowed down to this elevation would be expected to freeze.

The temperatures in the upper regions of the core increased more rapidly when the cladding temperatures reached about 1700 K (sometime between 165 and 174 min) because rapid oxidation of the Zircaloy cladding begins at this temperature and the heat of oxidation from the zircaloy elevates temperatures much more rapidly. Also, the stainless steel control rod cladding melts at about 1700 K releasing much more molten silver, indium, cadmium control material at locations other than spacer grids which interacts with the Zircaloy control rod guide tubes as does the molten stainless steel. After the cladding melting temperature was reached (2125 to 2245 K depending on oxygen content) the molten zirconium began dissolution of the UO_2 fuel pellets and added uranium to the downward flowing melt. This downward flowing melt would also freeze when it reached the water level.

Beginning at 174 min, sufficient coolant was injected into the reactor vessel by operation of the 2B primary coolant pump to fill the reactor vessel. Oxidized fuel rod cladding in the upper region of the core would be expected to have shattered and formed the upper debris bed when the coolant injection occurred leading to the beginning of the formation of the upper core void region. Although the 2B pump was on for 19 min, significant flow in the B-loop hot leg was measured for only 15 s.

At 200 min the ECCS coolant injection was initiated and subsequently filled the reactor vessel in 7 to 10 min. The upper debris bed probably quenched during this time period. The degraded core continued to heat up after 180 min and by 224 min calculations indicate that the central molten region, containing up to 42% of the core, was formed. The primary core relocation event occurred between 224 and 226 min. It is believed that the supporting crust failed near the upper, peripheral region on the east side of the core. The molten material relocated through peripheral fuel assemblies on the east side of the core and through the core bypass region where the baffle plates had melted. The partitioning of the molten core material flow through the assemblies and the core bypass has not been resolved. As a result of the relocation event, 19.2 metric tons of molten core material flowed to the lower head region and challenged the integrity of the reactor pressure vessel.

When the molten corium relocated to the lower head a pressure pulse occurred and lasted from 224 to 240 min indicating that heat transfer and steam generation within the lower head debris was significant for at least 15 min. The source range monitor response indicates that some core material, probably molten corium from the consolidated region in the center of the core, may have continued to relocate to the core bypass region between 230 min and 15.5 hours. At 15.5 hours after reactor scram, forced coolant flow through the reactor pressure vessel was reestablished with one of the A-loop primary pumps.

Several calculations were performed to evaluate the response of the reactor pressure vessel internals and lower head to the attack resulting from molten core movement. These include: (a) the potential for an energetic molten fuel-coolant interaction (which did not occur); (b) the thermal ablation of the core baffle plates, former plates, and reactor lower head by the impingement of molten core materials; (c) the thermal failure of the instrument penetration tube nozzles; and, (d) creep rupture of the lower head under conditions of elevated temperature and pressure.

A 20,000 to 40,000 kg corium mass is believed to be necessary to produce marginal reactor vessel lower head failure. However, a corium mass of only 500 kg was calculated to have been available during the relocation of the TMI-2 core for participation in a steam explosion event indicating that the theoretical possibility for the failure of the TMI-2 reactor pressure vessel is not evident. In any event, the high pressure condition in the TMI-2 vessel at the time of the core relocation most likely prevented the occurrence of a classical steam explosion altogether.

The time for baffle plate melt-through was assessed on the basis of conduction controlled heat transfer, for the assumed geometry of semi-infinite molten debris in contact with a steel slab of finite thickness. Complete melt-through of the baffle wall thickness was calculated to occur in about 5 min. Calculations were also performed for a molten jet impinging on the baffle plate. Results of the molten jet impingement calculations indicate a 3-to-26 s baffle plate melt-through time depending upon the diameter of the jet, the superheat in the jet corium, and the relocation duration.

An assessment was also made of the migration behavior of corium during relocation through the core bypass and of the potential for melt ablation of the 0.03175-m-thick stainless steel former plates. Comparison of the area of the hole in the baffle plates and the area of the 80 holes in the former plates at any elevation, indicates that the core bypass region could fill faster than it could drain. Hence the relocation of corium around the core periphery in the core bypass region. The time required for a completely corium-filled region between two former plates to drain was calculated to be 15 s. Calculations indicated that the time to melt initiation of the former plates at a distance of 15 cm from from the baffle plate melt-through location is more than a minute. Thus, the calculations would only predict former plate melting close to the baffle plate melt-through location.

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

Thermal analyses were performed to assess the potential for failure of the instrument penetration tube nozzles prior to the observation that the

R6 stainless steel nozzle guide tube had melted. The analyses indicated that melting of the nozzles could be expected if they were in contact with corium at temperatures in the range of 1600 to 1800 K, or with "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 nozzle melt failure. However, corium freezing and plugging in the instrument tubes were predicted, which would prevent core material from escaping the reactor vessel.

The thermal response of the reactor vessel lower head was calculated using the COUPLE/FLUIO code for three assumed debris configurations (an upper and lower bound and an intermediate case) and for quenched and unquenched debris beds in each configuration. For all three cases using an unquenched debris bed the temperatures trend ever upward with time, whereas for a quenched debris bed all three cases eventually decrease. For all the unquenched debris bed calculations and for the quenched upper bound case the ultimate strength of the lower vessel head steel would be expected to be reached and the vessel would fail. Thus, only the quenched intermediate and lower bound cases were realistic for the TMI-2 vessel.

Calculations were performed using the ABAQUS code for the quenched, intermediate configuration (porous debris bed resting on lower head) to determine the creep rupture potential of the vessel lower head and to estimate the margin-to-failure of the lower head. Results of the calculations indicate that the vessel 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 required at 783 K. Thus, creep rupture of the vessel lower head is not expected and the margin-to-failure appears to be quite large.

The NRC/DECD Vessel Integrity Program will be fully integrated with the DOE Accident Evaluation Program. An activity list and schedule for the proposed integration have been prepared. The remaining sample examinations are designed to provide data that can be integrated into the TMI-2 accident understanding. These data include: the mechanisms responsible for the transport of the molten core material to the lower head; probable flow pathways; viscosity of the molten materials; relocation rates; and, maximum temperatures and methods of surface attack. The resulting improved accident understanding will aid in the determination of RPV safety margins and in development of improved analytical tools for severe accident mitigation and management.

Several organizations are involved in the cooperative effort to achieve the goals of the NRC/OECD Vessel Integrity Program. In addition to the NRC and the OECD partners, Argonne National Laboratory (ANL-East), the Idaho National Engineering Laboratory (INEL), General Public Utilities (GPU)-Nuclear, MPR Associates, Inc., and Power Cutting Incorporated (PCI) will be responsible for the various parts of the Vessel Integrity Program. The two sample types for the Vessel Integrity Program are lower head samples and samples of debris directly adjacent to the lower head and activities for the two sample types include: (1) vessel lower head sample acquisition, sample preparation and shipment, experimental determination of time-temperature plots of the response of archive (or equivalent) samples of the vessel lower head, metallurgical examination, and determination of the most likely temperature distribution in the lower head; and, (2) companion sample acquisition, shipment, and metallurgical examination, calculation of the creep stresses in the lower vessel head as a function of time during the accident, determination of the margin-to-failure of the lower vessel head and program integration.

Wedge shaped samples will be cut from the lower head in regions where damage to the lower head would be expected to have the highest probability of damage. In these regions, samples that contain instrument penetrations and samples that do not contain instrument penetrations, will be obtained. Samples will also be obtained for reference from a lower head region where no damage is expected. The samples will be cut about half-way through the lower head thickness.

At the present time it is believed to be necessary to obtain samples: (1) as close as possible to the area of impact on the lower head of the primary jet of molten core material; (2) toward the radial center of the lower head underneath the maximum thickness of debris; (3) in the quadrant of the lower head where a "wall" of consolidated debris similar to a lava front has been observed; and, (4) in a location of the lower head not contacted by the molten core material to act as a "control" sample.

Core debris may be fused to the lower head samples and in this event, the companion samples will be obtained with the lower head samples. The material adjacent to the lower head is a vital data source and must be preserved during sample acquisition.

The vessel lower head samples will be shipped from TMI-2 to ANL-East. MPR Associates is responsible for both the acquisition and shipment of the lower head samples to ANL-East. ANL-East has the responsibility to recieve all vessel lower head steel samples; verify that the location, size, orientation, etc., of the samples that will have been documented by MPR is consistent and reasonable; decontaminate the appropriate specimens as necessary for unrestricted handling; section the samples into metallographic and metallurgical specimens or blanks; machine into specimens as required and distribute the appropriate materials to the cooperating partner laboratories.

ANL-East also has the responsibility for the lower head archive material program which has two objectives. One is to provide a set of standards for comparison with the samples removed from TMI-2 to increase the accuracy of the estimates of the thermal history of the lower head; the other is to provide enough material for the determination of the mechanical properties of the vessel lower head during the accident.

ANL-East has the responsibility for the characterization of the vessel lower head steel mechanical properties. The objective of this work is to determine the mechanical properties of the vessel lower head under the conditions of the core-melt accident. ANL-East has the responsibility to integrate all mechanical property and metallographic studies performed by the OECD partners and the INEL; to coordinate with the Vessel Integrity Program efforts at the INEL; to participate in the preparation of a report that describes the condition of the TMI-2 lower head steel, determines the probable temperature history of the steel, and assesses the integrity of the lower head during the accident.

The INEL has the responsibility for determining the metallographic characteristics of the vessel lower head steel and to accumulate, analyze and integrate all cooperating partner results and analyses pertaining to properties, temperature and integrity of the TMI-2 lower head steel. The INEL will take the lead in preparing the final report on the reactor vessel lower head examinations and will collaborate with ANL-East as full partners in the preparation and review of this final report. The final report will describe the material condition of the reactor vessel lower head steel, its probable temperature history, and its creep and plastic deformation during the accident.

The temperature distribution calculations for the lower head clearly indicated that the composition and properties of the relocated material adjacent to the lower head must be determined before a reliable analysis of the potential creep rupture and margin-to-failure of the lower head can be performed. The objectives of the lower head companion sample examination and analysis portion of the NRC/OECD Vessel Integrity program are to: (a) develop a plan (this document) for the selection, separation, transport, examination, and analysis of materials adjacent to or near the proposed metallurgical specimens in the TMI-2 reactor vessel lower head program; (b) perform the selection, acquisition (GPU-Nuclear will acquire the samples), transport, examination and analysis according to the plan after it is approved by NRC; and, (c) determine the most likely temperature distribution that occurred in the lower head and, using this temperature distribution perform calculations of the resulting lower head stress distribution in order to assess the margin-to-failure of the TMI-2 lower head. The INEL is responsible for the NRC/OECD lower head companion sample examination and analysis portion of the Vessel Integrity Program.

The companion samples will be obtained by GPU-Nuclear using tools that are presently available for the removal of debris. If the companion samples are tightly adhered to the lower head and cannot be removed at TMI-2, then they will be removed at ANL-East, examined and shipped to the INEL. The companion samples can be shipped in the DOE cask designated the CNS 1-13C II cask. The samples will be placed in 2R containers that have been used previously for shipment between the INEL and TMI-2. The companion samples will undergo extensive metallurgical examination at the INEL including optical metallography, SEM/WDS examination and Inductively Coupled Plasma Spectroscopy. However, the thermal conductivity of the material must be determined and the methods and experience for this determination do not exist at the INEL. The metallurgists at Windscale in the United Kingdom do have the technique mastered and are presently measuring thermal diffusivity of other TMI-2 samples shipped to them. It is recommended that the U K perform these measurements for the lower head companion samples.

The purpose of the lower head analysis is to determine the creep stress and resulting plastic deformation in the steel that resulted from the temperature transient caused by the molten corium relocation to the lower head; and then to calculate the temperature and stress distribution required to cause creep rupture of the lower head and thereby establish the margin-to-failure of the TMI-2 reactor pressure vessel. To achieve this the COUPLE/FLUID code will be used with the measured thermal conductivity of the lower head companion samples and the actual geometry of the lower head debris determined during defueling used as input. The results of the COUPLE/FLUID calculations will be compared with the lower head temperature distributions determined from examination of the vessel lower head samples and the time-temperature measurements made with the lower head archive material. The best estimate of the time-dependent temperature distribution will then be determined based on these comparisons, engineering judgement, and possibly some further calculations.

The time-dependent stresses in the lower head will then be calculated using the ABAQUS code with the best estimate time-dependent temperature distribution described above and the measured physical properties of the lower head steel used as input. Finally, the temperature distribution needed to cause creep rupture of the lower head will be determined using the ABAQUS code resulting in a determination of the margin-to-failure.

The integrated NRC/OECD and DOE programs for the completion of the TMI-2 accident evaluation will provide a coordinated program that emphasizes: (a) acquisition and examination of samples to determine damage to the lower head; (b) assessment of the lower head margin-to-failure; (c) development of an understanding of the pathways and mechanisms that controlled transport of molten materials to the reactor vessel lower plenum; (d) integration of this information into the final accident scenario; (e) comparison of these data with the results of the OECD analysis exercise (standard problem exercise); and, (f) application of the understanding of the TMI-2 accident to analytical tools used for accident mitigation and management. The integrated program should improve the completeness of both the DOE Accident Evaluation Program and the NRC/OECD Vessel Integrity Program and develop the most complete and accurate understanding of the TMI-2 accident that can be obtained with the available resources.

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FIGURES

TABLES

1. INTRODUCTION

The March 1979 accident at the Three Mile Island Unit 2 (TMI-2) pressurized water reactor was the most severe accident to occur in a commercial operating power reactor in the United States. At least 45% of the core was molten during the accident and almost 20 metric tons of molten core material relocated from the core region and came to rest on the reactor pressure vessel lower head. However, the progression of the TMI-2 accident was mitigated by the presence of cooling water in the pressure vessel. Although the integrity of the reactor pressure vessel lower head was challenged by the molten core material, it did not fail. Therefore the molten core was confined[a] 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 accident and core melt progression and the associated behavior of fission products, and that can be used in the future to aid in accident management.

The United States Department of Energy (DOE) has sponsored the TMI-2 Accident Evaluation Program to take full advantage of this important 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 materials temperatures, oxidation, and interactions; to develop an accident scenario based on this understanding; and, to produce a TMI-2 Standard Problem Exercise and a computerized data base containing TMI-2 research results, examination data, and supporting analyses. The DOE Accident Evaluation Program included sample acquisition and examination for materials within the reactor pressure vessel, within the reactor coolant system, and within the containment and auxiliary buildings. Although the DOE program has or will adequately cover all of these areas with respect to the program objectives, there is no provision for acquisition of samples of the reactor vessel

a. A very small amount of core debris was transported to reactor coolant system internals by coolant flow.

lower head nor of the material directly adjacent to the vessel lower head. Since the margin-to-failure of the reactor vessel lower head is vitally important to future accident management strategies, the United States Nuclear Regulatory Commission (NRC) formed an agreement with the Organization for Economic Co-operation and Development (OECD) to investigate the TMI-2 reactor pressure vessel lower head and companion samples of debris adjacent to the lower head. This program is denoted as the NRC/OECD TMI-2 Vessel Integrity Program and this document is a plan for the accomplishment of the NRC/OECD Program.

The sponsoring partners of the NRC/OECD TMI-2 Vessel Integrity Program are: the Federal Republic of Germany, Finland, France, Italy, Japan, Spain, Sweden, Switzerland, the USNRC, and the United Kingdom.

Section 2 of this document is an overview of the post-accident conditions of the TMI-2 Core, and Section 3 is a review of the TMI-2 Accident Scenario. Analyses of the challenges to the reactor pressure vessel internals and lower head are reviewed in Section 4. Section 5 is a description of the integration of the NRC/OECD Vessel Integrity Program and the DOE Accident Evaluation Program Results. Section 6 is the plan for the accomplishment of the NRC/OECD Vessel Integrity Program.

2. POST-ACCIDENT CORE CONDITIONS

An accurate determination of the post-accident conditions of the TMI-2 Plant was required to understand not only the accident progression and fission product transport and deposition but also to understand the properties of post-accident core materials in order to defuel the reactor. This information was developed from several sources. The sources were: visual inspections of reactor vessel internals; metallurgical/radiochemical examinations of samples acquired during the course of defueling; readings from on-line instrumentation during the accident; and, experimental data developed from tests sponsored by the USNRC in various facilities including ACRR, LOFT, NRU, and PBF. The post-accident conditions of the TMI-2 Core have been well characterized and many of the early post-accident conditions have been reported.[1-5]

2.1 Plant Conditions

The major physical damage to the plant was limited to reactor pressure vessel internals. Fission products were transported by the reactor coolant system (RCS) through the stuck-open pilot-operated relief valve (PORV), an overpressure protection device mounted on top of the pressurizer, to the reactor coolant drain tank, which ruptured, and to the containment basement. Also, 5% to 7% of the core inventory of cesium and iodine and an approximately equal amount of noble gases were transported to the auxiliary building through the letdown, makeup, and purification systems. Inspections of the steam generators, reactor coolant pumps, RCS hot and cold legs, and pressurizer showed a fine layer of silt and particles throughout the RCS. The largest collections of fuel debris were identified in the pressurizer (approximately 10 kg), B steam generator upper tube sheet (approximately 50 kg), and the decay heat drop line (possibly greater than 50 kg). General conditions within the reactor and auxiliary buildings have been reported.[6-9]

2.2 Reactor Vessel Internals Conditions

During the accident, peak temperatures ranged from approximately 3100 K at the center of the core (molten UO_2) to 1255 K immediately above the core to 723 K at hot leg nozzle elevations. The upper grid structure (stainless steel) was ablated in two zones. More than 40% of the original core melted during the accident. Lower portions of three baffle plates on the east side of the core melted and some of the molten material flowed into the core bypass region. About 20,000 kg of molten materials flowed from the core and the core bypass region through the lower internals, coming to rest on the reactor vessel lower head. Figure 1 illustrates the major reactor vessel components and the post-accident configuration of the core.

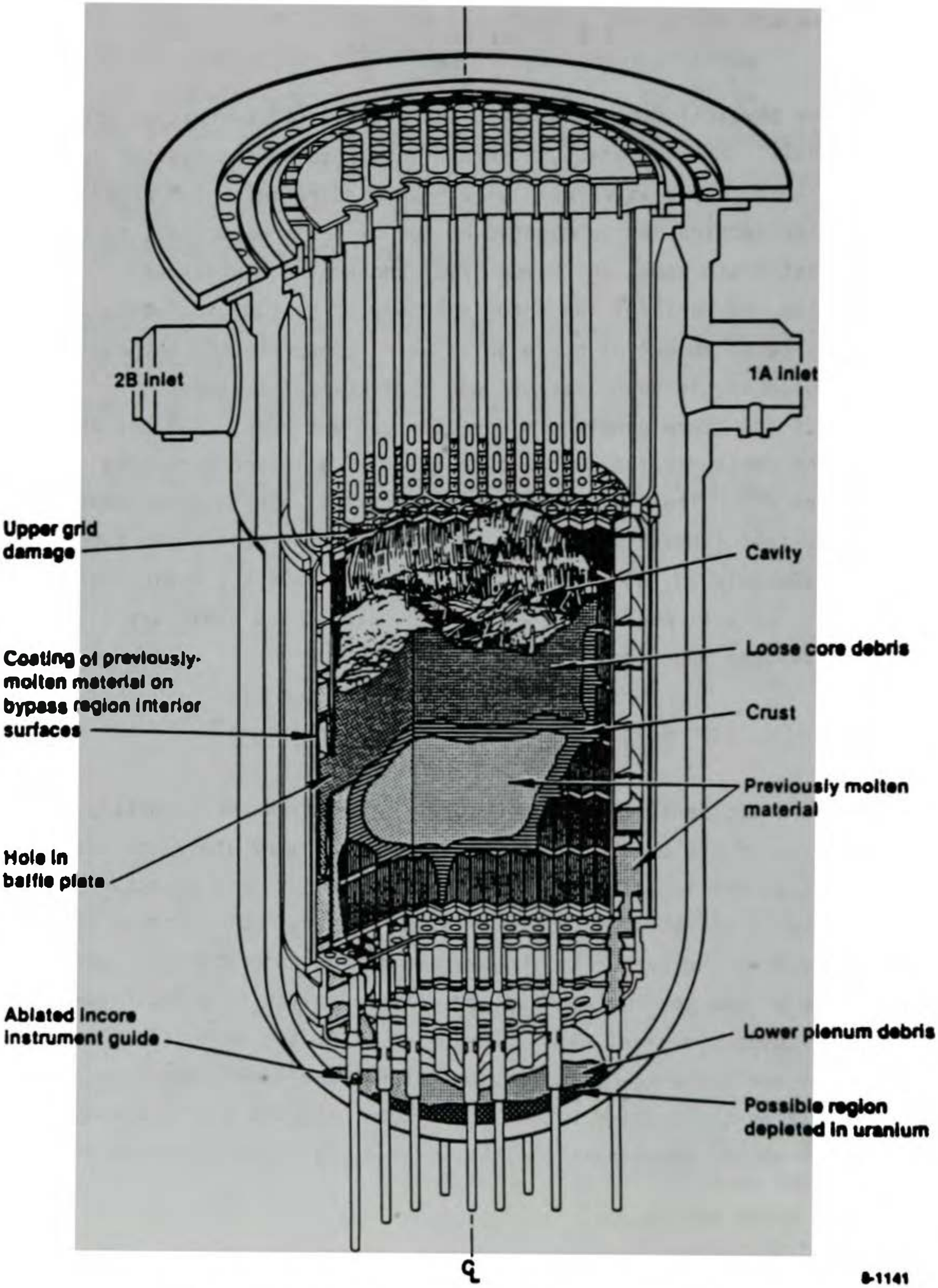


Figure 1. TMI-2 core end state configuration.

Post-accident conditions of the upper plenum assembly, the original core region, the core bypass region, the lower core support assembly (CSA), and the lower head region are described in the following subsections.

2.2.1 Upper Plenum Assembly

The upper plenum assembly, which experienced no major structural damage, was removed intact. However, two damage zones were formed on the bottom of the plenum assembly. 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 molten grid material had a foamy texture, which occurs when stainless steel oxidizes near its melting point. A molten mass close to this grid material appeared unoxidized, suggesting that some of the hot gases exiting the core were oxygen deficient. The damage to the upper grid assembly suggested that the composition and temperature of core exit gases varied significantly within the flow stream.

2.2.2 Core Region

As illustrated in Figure 1, the end-state configuration of the original core region included a core void or cavity at the top of the core region. Below that, a loose debris bed rested on a resolidified mass of material that was supported by standing fuel rod stubs that, in turn, were surrounded by portions of fuel assemblies. The previously molten, resolidified material was surrounded by a distinct crust of material in which other fragments, shards of cladding, etc., could be identified.

The core void was approximately 1.5 m deep with an overall volume of 9.3 m^3. Forty-two of the original 177 fuel assemblies were standing at the periphery of the core void. Only two of these fuel assemblies contained more then 90% of their full-length cross sections, with the majority of fuel rods intact.

The loose debris bed at the base of the cavity in the core ranged in depth from 0.6 to 1 m and consisted of whole and fractured fuel pellets, control rod spiders, end fittings, and resolidified debris totaling approximately 26,400 kg. Most of the loose debris bed consisted of shattered and resolidified material that contained regions of previously-molten U-Zr-O, indicating peak temperatures greater than 2200 K. Many of the samples obtained from the debris bed were previously molten $(U,Zr)O_2$, indicating peak temperatures greater than 2800 K. There were also a few samples of previously-molten material that were almost pure UO_2, indicating temperatures above 3100 K. Metallographic examinations of individual particles indicated that most of the debris bed remained at temperatures below 2000 K or was exposed to high temperatures for only a short time.

Beneath the loose debris bed was a large resolidified mass, approximately 3 m in diameter, 1.5 m thick at its center, and 0.25 m thick at its periphery. The resolidified mass comprised approximately 32,700 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 2800 K and possibly as high as 3100 K during the accident. It is believed that this material was fully molten. The upper crust of this mass, which consisted of the same material and which also reached 2800 K, contained fuel pellet fragments (unmelted) near the crust periphery. The lower crust consisted of previously-molten stainless steel, Zircaloy cladding, and control rod materials surrounding intact fuel pellets. The peak temperature of the lower crust material was greater than 1400 K but less than 2200 K. The resolidified mass was shaped like a funnel extending down to the lower end fitting at an assembly 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 previously-molten materials. These stubs varied in length from approximately 0.2 to 1.5 m. Longer partial fuel assemblies stood at the periphery of the resolidified mass. On the east side of the core, four adjacent fuel assemblies were nearly completely replaced with previously-molten core material; this indicated a possible relocation path of molten material into the lower CSA and core bypass region. The standing fuel assembly stubs and peripheral assemblies constituted about 44,500 kg of core debris.

2.2.3 Core Bypass Region

This region consists of vertical core baffle plates that form the peripheral boundary of the core; horizontal core former plates, to which the baffle plates are bolted; the core barrel; and the thermal shield (Figure 1). There are a number of flow holes in the baffle and core former plates (0.035 and 0.033 m in diameter, respectively) through which coolant flowed during normal operations. Visual inspection (via closed-circuit television) of baffle plates and core former plates indicated that they were in good condition, with one notable exception: on the east side of the core there was a large hole approximately 0.6 m wide and 1.5 m high, extending across the lower portion of three baffle plates. The 0.019-m-thick baffle plates and sections of three horizontal core former plates (approximately 0.032 m thick), were melted in this region. Molten core material from the core region appeared to have flowed through this hole into the core bypass region. The area behind the baffle plates contained loose debris all the way around the core region. The depth of debris within the core bypass region was approximately 1.5 m on the north side and a few millimeters thick on the southwest side. There appeared to be a resolidified crust, which varied in thickness from approximately 0.005 to 0.040 m, on the upper horizontal surfaces of the three bottom core former plates. The molten material moved down into the CSA through the flow holes (0.033m) in the core former plates. It is estimated that 4200 kg of core debris were retained in this region. No major damage to the core barrel or the thermal shield has been observed..

The 0.025-m-annulus between the core barrel and the thermal shield was inspected and only fine particulate was observed.

2.2.4 Core Support Assembly

The CSA region consists of five stainless steel structures: four flow plates and a lower grid assembly. The structures vary in thickness from 0.025 to 0.33 m with 0.080 or 0.15 m diameter flow holes. Liquified material flowed through the periphery of these structures and came to rest on the lower head. There was resolidified material (5800 kg) 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 CSA from the east side of the core. Although most of this material was seen on the east to southeast sides, many columns of resolidified material were seen all the way around the core beneath the core bypass region. The condition of the CSA structures will be better defined when this region is defueled.

2.2.5 Lower Head Region

The debris in the lower head region accumulated to a depth of 0.75 to 1 m above the lowest head elevation and to a diameter of 4 m. Inspections revealed that a large quantity of previously-molten core material rested on the lower head. The spatial distribution of these materials was neither uniform nor symmetric. Particle sizes varied from large "rocks" (up to 0.20 m) to granular particles (less than 0.010 m). Visual inspections indicated that larger rocks, especially in the northeast and southwest areas, were located towards the periphery and the debris pile was lower at the vessel center than at the periphery. Granular or "gravel-like" material was observed in the central region of the vessel. A large "cliff-like" structure formed from previously-molten core material existed in the northern region. The cliff face was approximately 0.38 m high and 1.25 m wide. It was estimated that 19,200 kg of core material relocated to the lower head region.

The only significant structural damage observed to date in the lower head region was melting of an incore instrument guide tube. Tests performed on the incore instrumentation after the accident showed that twelve of the incore thermocouples had new junctions in the lower head region.[6]

2.3 Core Material Inventory

The original core inventory included approximately 94,000 kg of UO_2 and 35,500 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 material was estimated to be 133,200 kg. The post-accident material balance is listed in Table 1. The accuracy of the numbers in Table 1 is estimated to vary from 5% for the original core region (which has been defueled) to 40% for the CSA and lower head regions (which have not yet been defueled). As each region is defueled the uncertainty of core material located in that region decreases.

TABLE 1. ESTIMATED POST-ACCIDENT CORE MATERIALS DISTRIBUTION

Core Region	Estimated Quantity (kg)	Uncertainty[a] (%)	Percent of Total Core (%)
Intact fuel assemblies (partially or fully intact)	44,500	5	33.4
Central core region resolidified mass	32,700	5	24.5
Upper core debris bed	26,600	5	19.9
Prior molten material on the lower reactor vessel head	19,100	200	14.3
Lower core support assembly[b]	5,800	40	4.3
Upper core support assembly[b]	42,000	40	3.2
Outside the reactor vessel	450	--[c]	0.3

a. The uncertainty estimates are based on defueling. Those areas which have been defueled at this time 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 flow distributor plates. The upper core support assembly is a coolant flow region outside the vertical 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.

3. THE ACCIDENT SCENARIO

A scenario of the TMI-2 accident is being developed using the data summarized in Section 2 of this report, measurements from on-line instrumentation, and supporting analytical studies[10], together with experimental data from in-pile and out-of-pile research facilities. The present accident scenario is described in the following paragraphs.

The TMI-2 accident resulted from a series of events initiated by problems in the secondary cooling system. This led to a loss of feedwater to the steam generators. Loss of secondary side cooling, in turn, led to over-pressurization of the reactor coolant system (RCS) and subsequent opening of the PORV. The PORV failed to close as the primary system pressure was reduced, thus resulting in a "small break" loss-of-coolant accident. The reactor operators, misinterpreting measurements of the RCS condition, believed that the RCS was within operating limits. Because of this misconception, the emergency core cooling system (ECCS) makeup water to the RCS was throttled to reduce the coolant injection rate. Continued coolant loss over the next 140 min left the core uncovered and allowed overheating and eventual melting of core materials.

3.1 Loss-of-Coolant Period (0 to 100 min)

The events initiating the accident and the RCS thermal hydraulic response during the first 100 min are documented in Reference 11. The primary coolant pumps provided two-phase cooling to the core during this period, thus preventing core overheating and damage. During the first phase of the accident, the amount of water in the primary coolant system decreased because the RCS makeup was insufficient to compensate for coolant loss through the PORV.

3.2 Initial Core Heating (100 to 174 min)

When the last two reactor coolant pumps were turned off, shortly after 100 min, the top of the core started to uncover and 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 min and fuel rod temperatures at the top of the core increased sufficiently (1100 K) to cause cladding rupture. At approximately this time, the operators realized that the PORV was faulty, and they closed the pressurizer block valve, thus limiting further loss-of-coolant and gaseous fission product release from the primary cooling system to the containment building. However, the block valve was cycled, open and closed, to maintain system pressure.

Rapid oxidation of the Zircaloy cladding at the top of the core began at approximately 150 min. The heat of oxidation elevated fuel rod temperatures above the cladding melting point (2125 to 2245 K depending on oxygen content) and molten cladding began dissolving some of the UO_2 fuel. Prior to this, the low-melting-temperature control materials (silver, indium, and cadmium) melted, the cadmium volatilized, and as the control rod stainless steel cladding temperature increased, the control rods failed. It is believed that the resulting molten mixture of fuel, cladding, structural (stainless steel and inconel), and control materials flowed downward and resolidified around intact fuel rods near the 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 min, as shown in Figure 2a. The first material that flowed was probably formed from the eutectic between the inconel spacer grids and the zircaloy fuel rod cladding; and, between the zircaloy guide tubes and the stainless steel cladding of the control rods at about 1500 K. Once the first liquid was formed, additional eutectic (such as 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) formed more rapidly due to enhanced atomic mobilities in the liquid as opposed to the solid state. The combination of liquid cladding, structural and control materials is then capable of flowing downward.

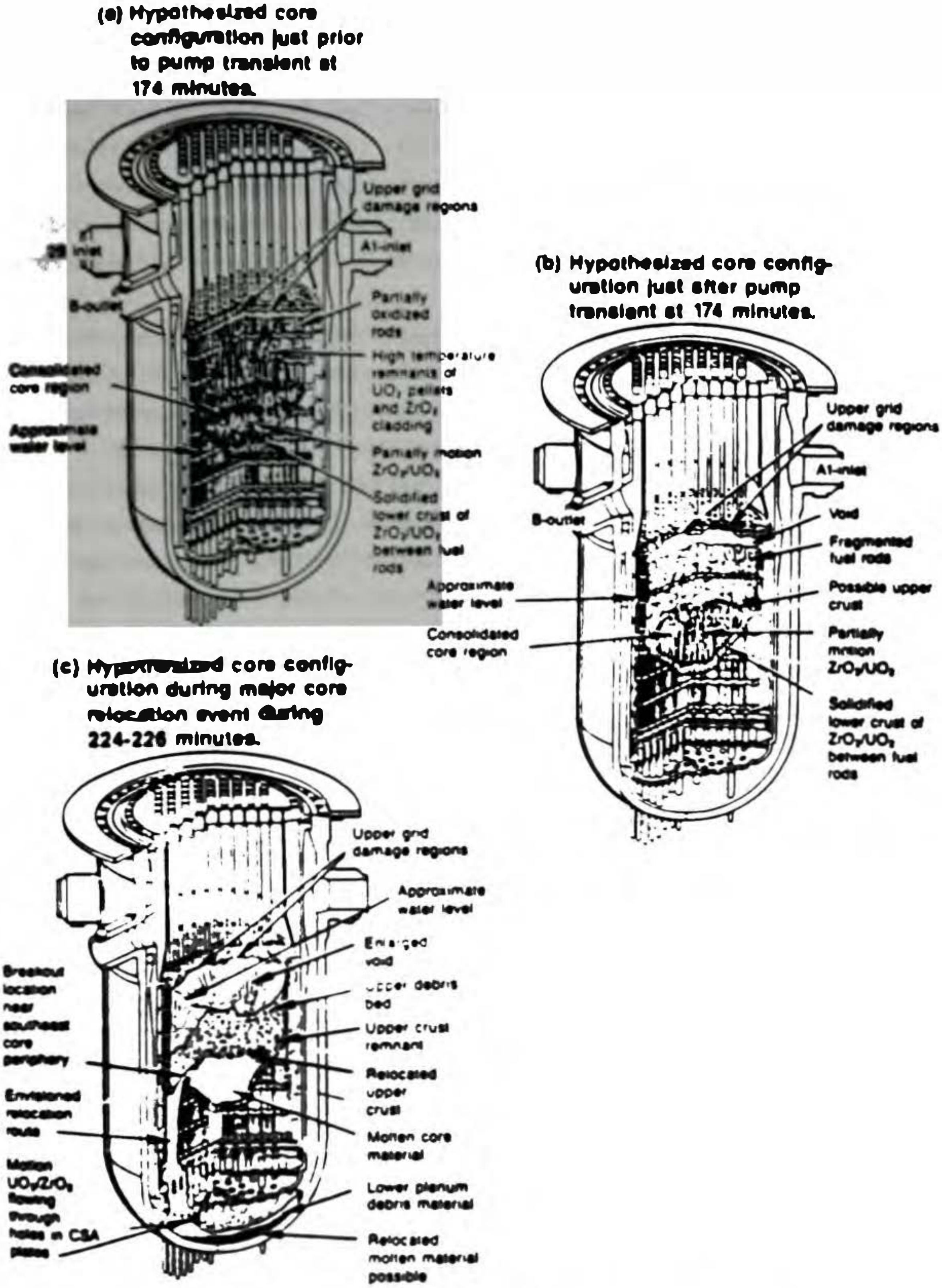


Figure 2. Hypothesized core damage progression.

3.3 Pump Transient (174 to 180 min)

At 174 min coolant was injected into the reactor vessel by operation of the B-2 primary coolant pump. The coolant flow was sufficient to have filled the reactor vessel. Thermal-mechanical interaction of the coolant with the oxidized and embrittled fuel rod remnants in the upper core regions is believed to have fragmented these standing remnants and formed the upper core debris bed. This configuration is illustrated in Figure 2b.

The interaction of the injected water with the upper debris bed during this period and the flow pattern of gas exiting the core through the upper plenum have been assessed. The observed damage pattern to the upper fuel assembly was consistent with expected flow patterns, considering the location of the exit flow orifices. Rapid oxidation within the debris bed and the subsequent interaction of the upper grid structure with the high temperature gases exiting the core at high velocity probably caused the observed limited damage. The lack of extensive melting of upper plenum structures also indicates that vigorous, long-term natural convection heat transfer probably did not occur from the core to these structures.

3.4 Degraded Core Heating (180 to 224 min)

The RCS thermal hydraulics were complicated by injection of emergency core coolant water and cycling of the pressurizer block valve. ECCS coolant injection was initiated at 200 min and subsequently filled the reactor vessel in 7 to 10 min. Studies of debris bed cooling[12] indicate that final quenching of the upper core debris bed probably occurred during the last several minutes of this time period. Effective cooling of the molten core material probably was limited to the surrounding, supporting crust material. Thus, the amount of molten material in the central region probably continued to increase because of decay heat from retained fission products and lack of coolant. Calculations simulating the accident suggest a molten pool of 20% to 45% of the original core materials was formed within the consolidated region[13] by 224 min. This is consistent with the observed molten material found in the resolidified core mass, the CSA, and the lower head regions (Table 1).

3.5 Crust Failure and Molten Core Relocation (224 to 230 min)

The primary core relocation event occurred between 224 and 226 min, probably in about 1 minute. This event was indicated by the RCS pressure monitor, self-powered neutron detectors, and the source range monitors. It is believed that failure of the supporting crust occurred in the upper region of the consolidated mass of molten core material, probably near the core periphery about 1.5 m above the bottom of the core and on the east side of the core, as shown in Figure 2c. Visual inspections conducted during core defueling indicated a flow of molten core materials into the CSA internals occurred on the east side of the core. The partitioning of flow of molten core material down through the open fuel assemblies versus flow through the melted baffle plate and the core bypass region is not obvious. Analysis of the flow of molten core materials through open fuel assemblies indicates that all of the molten core material could have relocated into the lower CSA internals and lower head in less than 1 min through only one or two fuel rod assemblies[13]. The time for baffle plate melt-through was assessed on the basis of conduction-controlled heat transfer, for the assumed geometry of semi-infinite molten debris in contact with a steel slab of finite thickness. Complete melt-through of the baffle wall thickness was calculated to occur in about 5 min.[14] Calculations were also performed for a molten jet impinging on the baffle plate. Results of the molten jet impingement calculations indicate a 3-to-26-sec baffle plate melt-through time depending upon the diameter of the jet, the superheat in the jet corium, and the relocation duration.[15]

Since the axial location of the large baffle plate melt hole (about 1.5 m) is somewhat lower than the initial elevation of the uncoolable debris bed (2.4 to 2.7 m), and since the major melt relocation event at 225 min is thought to have lasted only about 1 to 2 min, baffle plate melt-through may not directly coincide with the molten core relocation. It can be postulated that holdup of a portion of the relocating debris during downward relocation, for a period of 10 to 15 min in the region of the baffle plate melt hole, could have been responsible for melt ablation. However, partial melt debris drainage to lower elevations earlier in the accident, with a localized uncoolable blockage configuration could also have been responsible for baffle plate melt-through which could have occurred in this instance, coincident with the molten core relocation. Thus, a clear understanding does not yet exist for the timing and nature of the interaction of molten core materials with the core bypass region and fuel assemblies after failure of the supporting peripheral/upper crust.

Several hypotheses on failure of the supporting peripheral/upper crust have been proposed. These included a reduced crust thickness supporting the debris bed caused by continued melting of core materials within the consolidated mass; a stress-induced failure of the upper crust caused by the reactor system depressurization that occurred at 220 min; and a thermal/chemical interaction of the crust material with core baffle plates. It was not clear if any single failure mechanism or a combination of mechanisms was the primary cause of crust failure. However, further core material examinations and thermal/mechanical analyses of the crust should provide valuable data to focus on a most probable cause for failure of the supporting peripheral of upper crust.

The rapid relocation of such a large quantity (about 19,200 kg) of molten core material posed at least two challenges to the structural integrity of the reactor pressure vessel. The first was from the hypothetical potential for an energetic molten fuel-coolant interaction. The measured RCS system pressure confirmed that such an interaction did not occur.

The second challenge to the reactor vessel integrity was from a direct thermal interaction of molten core materials with the reactor vessel lower head. The reactor vessel lower head did not fail. Thus, a margin of vessel safety exists. Several calculations have been performed to estimate the potential for a molten fuel-coolant interaction, possible thermal ablation of the lower head by molten core jet impingement, thermal failure of the seal welds around instrument penetrations, and failure of the lower head by creep rupture under conditions of elevated temperature and RCS pressure. A discussion of the results of these calculations is contained in Section 4 of this report.

3.6 Long-Term Degraded Core Cooling (224 min to 15.5 h)

There was no evidence of a second major relocation of molten core materials into the CSA. Thus, the post-accident configuration of the core presented in Figure 1 represents a stable and coolable configuration for the materials in the core, CSA, and lower head regions. Detailed thermal analyses are in progress to investigate the long-term cooling of the consolidated mass within the core region. Preliminary results from these studies suggest that cooling of this mass took many days.

4. ATTACK OF THE REACTOR PRESSURE VESSEL

The rapid relocation of 19,200 kg of molten core material to the TMI-2 reactor pressure vessel lower head region posed a serious challenge to the integrity of the lower head. The reactor pressure vessel lower head did not fail; but, since the vessel integrity was challenged, the characterization of the condition of the lower head will provide data that is invaluable for consideration of reactor pressure vessel failure during severe accidents. The data that can be obtained from TMI-2 reactor vessel lower head examination, and examination of previouly-molten core material adjacent to the lower head, coupled with the results of calculational models of the reactor vessel lower head response to the attack will support the development of lower head damage and accident management models to predict the margin-to-failure of the reactor vessel lower head and the consequences of possible recovery options.

Several calculations have been performed to evaluate the response of the TMI-2 reactor pressure vessel to the attack resulting from molten core movement. These include: (a) the potential for an energetic molten fuel-coolant interaction (which did not occur in the TMI-2 accident); (b) the thermal ablation of the core baffle plates, former plates, and reactor lower head by the impingement of molten core materials; (c) the thermal failure of the instrument penetration tube nozzles; and, (d) creep rupture of the lower head under conditions of elevated temperature and RCS pressure. The results of these calculations are discussed in the following subsections. Details of the configuration of the reactor pressure vessel internals are described in Appendix A.

4.1 Molten Fuel Coolant Interaction Potential

A report prepared by the Steam Explosion Review Group (SERG) concluded that the occurrence of a steam explosion of sufficient energetics to lead to alpha-mode containment failure has a low probability.[16] A central feature of the quantitative responses in this assessment was a mechanistic treatment of various stages of the steam explosion sequence, with emphasis on estimation of the mass of melt that participates in the explosion. The low probability of early containment failure suggested by the SERG stems largely from the belief that the fuel mass participating in a potential steam explosion in a lower vessel plenum following molten core relocation is quite limited.

The limitation in fuel participating in a possible steam explosion was related both to the likelihood of a large, coherent pour of molten core material into a lower plenum and to the subsequent potential of molten core-coolant mixing. Estimates among the SERG members varied over a wide range (700 to 24,000 kg). These estimates can be compared with 20,000 to 40,000 kg of molten core material that must relocate to a lower plenum within a few seconds, to challenge the integrity of the reactor vessel. The molten core relocation process believed to occur during the TMI-2 accident clearly supports the SERG conclusion that an energetic in-RPV steam explosion capable of early containment failure is a low probability event.

As described in Subsection 3.5, the TMI-2 molten core relocation is believed to have started with the failure of peripheral/upper crust containing the molten core material (corium) and the subsequent pouring of this material into the lower plenum. The pouring molten core material stream(s) or jet(s) mixes with the water as it desends to the lower plenum at a rate controlled by interfacial instabilities.[17] The energetics of the vapor explosion, if it occurs, is commonly believed to be directly proportional to the extent of the corium melt-water mixing zone at the time that the vapor explosion is initiated. Once the extent of the mixing zone is determined, an upper bound to the mechanical energy release can be calculated by multiplying the thermal energy "stored" in the corium component of the mixture by an appropriate thermodynamic conversion ratio.[16]

A corium mass of 500 kg was calculated to have been available during the relocation of the TMI-2 core for participation in a steam explosion event.[18] The theoretical possibility for the failure of the TMI-2 reactor pressure vessel by a steam explosion is not evident when the 500 kg corium mass is compared with the 20,000 to 40,000 kg corium masses thought necessary to produce marginal reactor vessel lower head failure.[19] In any event, the high pressure condition in the TMI-2 vessel at the time of the core relocation most likely prevented the occurrence of a classical steam explosion altogether.[20]

4.2 Thermal Ablation of RPV Components

As discussed in Subsection 3.5, complete melt-through of the baffle wall thickness was calculated to occur in 10 to 15 min. Partitioning of the flow of corium down through the open fuel assemblies versus through the ablated hole in the baffle plates and the core bypass region has not been resolved. Examination of samples of previously-molten material from the east side of the core in the R6 assembly location would help resolve this partitioning. Since failure of the baffle plates can lead to potential corium flow into the bypass region between the baffle plates and the core barrel, corium attack of the horizontal former plates that reside between the baffle plates and core barrel was also analyzed.

4.2.1 Thermal Ablation of Former Plates

A significant amount of core debris relocated to the horizontal former plates. Estimates indicate that 4200 kg of previously-molten corium resolidified in the core bypass region, with the highest blockage of the holes in the former plates on the east side of the core. An assessment was therefore made of the migration behavior of corium during relocation through the core bypass and of the potential for melt ablation of the 0.03175-m-thick stainless steel former plates.

The core bypass volume, between the baffle plates and the core barrel, is occupied by a series of horizontal former plates at 8 different levels. Each former plate has 80 flow holes that are 0.03334 m in diameter and are aligned top to bottom. An assessment was made of the potential for lateral melt migration along the surface of the former plates versus drainage through the flow holes to characterize melt debris migration behavior within the core bypass region. A comparison of the flow area associated with the breached east-quadrant baffle plates and the flow hole area associated the 80 holes in a former plate, indicate that the breached baffle plates offer a larger hole area than the former plates. The implication of such a comparison is that lateral migration of molten corium along the surface of the former plate can be expected. Lateral melt migration is supported by the observation of debris on both the north and east quadrants of the former plates.

An estimate was also made of a characteristic drainage time for downward melt migration through the holes in the former plate. Assuming corium occupied the entire volume between two former plates, and that all holes in a former plate are available for drainage, a drainage time of about 15 s was estimated.[14] The implication of this result is that if the corium remained molten, rapid drainage of the corium through the former plate holes would have occurred. However, the fact that a significant amount of debris has been observed to remain in the core bypass region confirms that corium was frozen in the bypass region. Also, the vessel was reflooded by ECCS injection between 200 and 207 min so that water was present in the core bypass when the baffle plates failed. In-place freezing of corium in contact with coolant in the core bypass region is consistent with the observations. The freezing of corium in the core bypass region does not, however, imply that corium did not relocate to the lower plenum through the core bypass region.

An examination of debris characteristics (composition, size, surface morphology) and the extent of thermal interaction with the former plates (fused debris to former plates versus loose rubble) with the debris in the lower plenum is recommended to determine if significant differences in debris composition, mean particle size, and retained fission products exist. These data will enhance understanding of the debris cooling and potential thermal damage to vessel structural components.

4.2.2 Thermal Ablation of Reactor Vessel Lower Head

Thermal damage potential to the lower head was assessed for the configuration of coherent jet impingement of relocating corium. For this assumed configuration, the thermal response of the lower head is largely dictated by the contact time and heat transfer characteristics at the jet impingement surface. Assuming a jet diameter equal to the flow area within a single undegraded fuel assembly, the time for melt relocation as a jet was estimated to be about 75 s. This estimate is consistent with source-range monitor data which indicates that major core relocation occurred during about a 1-minute period.

Two limiting conditions were assessed with respect to jet impingement heat-transfer characteristics. The first was for a weak jet, with conduction-limited heat transfer. The second was for strong jet forces, where turbulent mixing and mass transfer effects at the impact surface lead to an enhanced convective-controlled heat transfer process. For conduction-controlled heat transfer, surface ablation of the lower head by direct heat transfer was not indicated due to the poor conductivity of the molten core material and the high thermal capacity of the lower head, which serves as an efficient and quick-response heat sink. However, calculational results for convection-controlled heat transfer indicate limited melt ablation at the surface of the lower head stainless steel liner. The calculated depth of penetration of the melt front is about 0.0127 m (compared with a head thickness of 0.1397 m) for a jet impingement time of 75 s. It would take a jet impingement time of 15 to 20 min to ablate half-way through the vessel thickness according to the calculations. Thus, for a jet impingement time of 1 to 2 min, which is believed to be the time for the relocation of the TMI-2 corium, little thermal damage to the reactor vessel lower head would result.[14]

An examination of the lower vessel head, particularly in the east-quadrant, is recommended to establish whether a jet actually channeled its way to the vessel bottom and thermally attacked the lower head. However, because little damage to the lower head is predicted (either for a strong jet or for coherent spreading of melt debris along the head surface), it may be more important to examine for melt ablation of the bottom-entry instrument guide tubes and penetration nozzles, since the limited thermal capacity of these structures make them more susceptible to melt ablation.

4.3 Instrument Penetration Tube Failure Potential

Both Babcock and Wilcox (TMI-2) and Westinghouse PWRs contain instrument penetration tubes through the lower head, which serve as entry ports for neutron flux monitors and other in-core instrumentation. Because of the large number of penetrations (52 for TMI-2) and the three-dimensional nature of thermal attack that these tubes would experience (as opposed to essentially one dimensional heat transfer in the more massive vessel head), the instrument penetration tubes could be subject to early failure and the attendant potential to duct core material from the pressure vessel to the containment.

Each bottom-entry instrument penetration is essentially a continuous tube that starts from an instrument panel located in the containment building above the top of the reactor vessel as indicated in Appendix A, Figure A. The access path is downward through the instrument tunnel and reactor cavity, turning upward below the reactor vessel, and penetrating the reactor vessel through holes in the lower head forging. The instrument penetration tubes are sealed to the reactor vessel lower head by welding brazements. In-core instruments are inserted into the RPV through these tubes and are indexed by a switching device to map the neutron flux and temperature distributions within the core. A detailed description of the instrumentation within the tubes and the geometry of the instrumentation tubes are given in Appendix A.

Figure 3 illustrates the geometry of an instrument penetration nozzle located just inside the reactor vessel lower head, which is subject to attack by molten core material. A feature contributing to the failure potential is the "fin-effect" of the tube stub surrounded by hot corium. Likewise, the temperature of the weld material at the penetration-head junction would increase faster than the vessel wall itself, since the penetration nozzel acts as a conduction path for heat transport from the corium to the welds. Consequently, the penetration junctions can be expected to fail before the lower head.

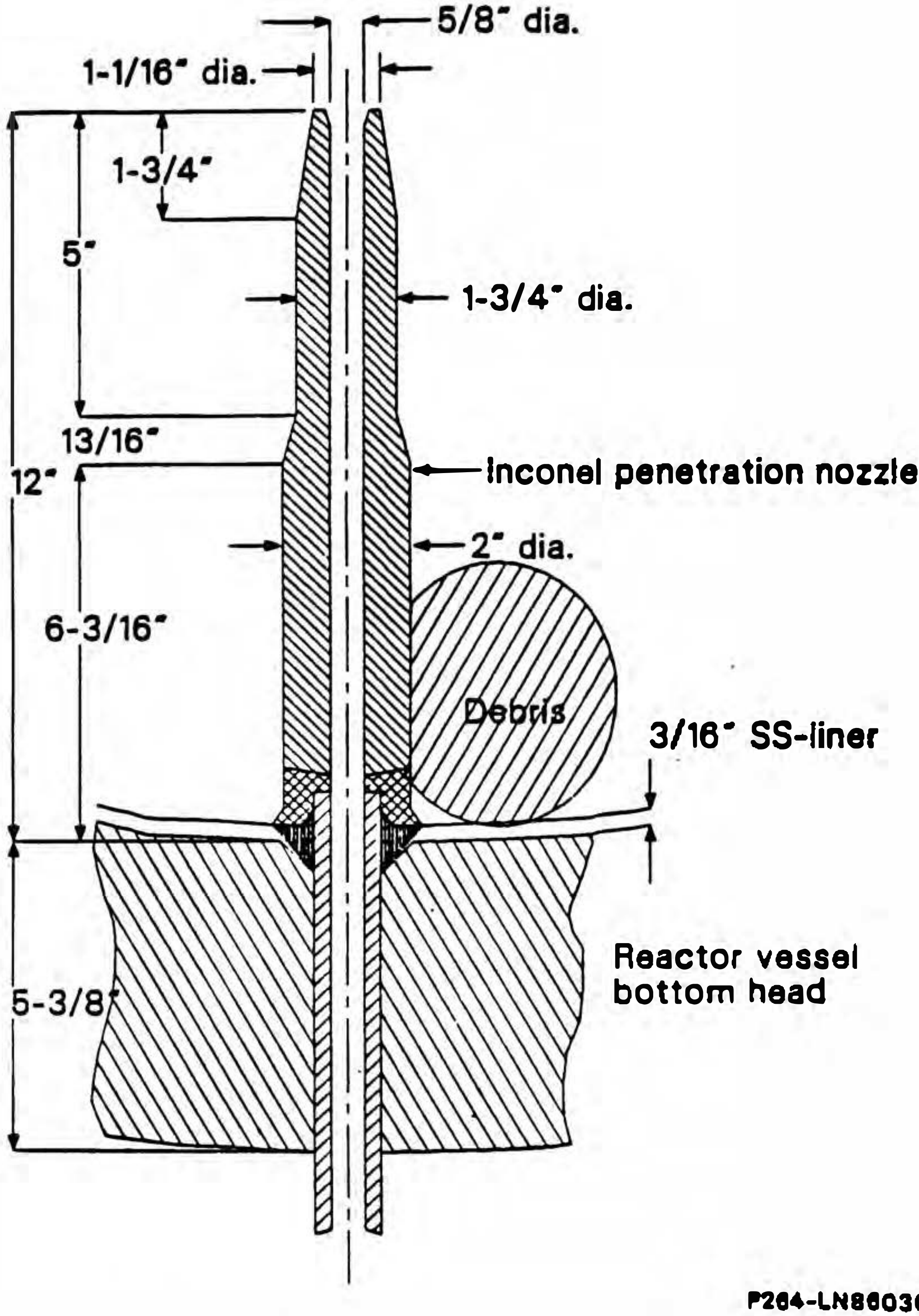


Figure 3. Instrument penetration nozzle configuration.

Thermal analyses were performed to assess the potential for failure of the nozzles prior to the observation that the R6 stainless steel guide tube had been melted.[21] The analyses indicated that melting of the nozzles could be expected if they were in contact with corium at temperatures in the range of 1600 to 1800 K, or with 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 nozzle melt failure.[21] However, corium freezing and plugging in the instrument tubes were predicted, which would prevent core material from escaping the reactor vessel.

In a separate study by Babcock and Wilcox[23] the transient heating of an instrument tube nozzle surrounded by a corium stalagmite was analyzed using an axisymmetric model in the FELCON computer code. An illustration of the model used in the calculation is shown in Figure 4. The axisymmetric geometry was used for a cylindrical column (stalagmite) of impermiable corium, surrounding an Inconel nozzle and in contact with the TMI-2 lower head. The initial temperature of the surface of the corium in contact with the lower head was taken to be 2477 K and the remainder of the corium was set at 3033 K initially. The only cooling provided was by boiling on the outside cylindrical surface of the corium stalagmite and on a 0.9525-cm-wide ring of steel where the lower head extended beyond the corium column in the model. The initial temperature of the lower head and nozzle were set at 608 K.

For these assumed conditions the FELCON-calculated results indicate that the temperature of a significant portion of the lower head would reach 1144 K or hotter. At these temperatures, the ultimate strength of the lower head is only 13,400 psi and under these conditions the lower head would have ruptured or plastically deformed during the TMI-2 pressure transient (2300 psi). Since the reactor vessel lower head did not fail during the pressure transient, it was concluded that the FELCON calculations were too conservative (i.e., the corium was cooler than 3033 K, the stalagmites were smaller than 0.254 m in outside diameter, or there was more than 14% of the surface of the lower head exposed to cooling

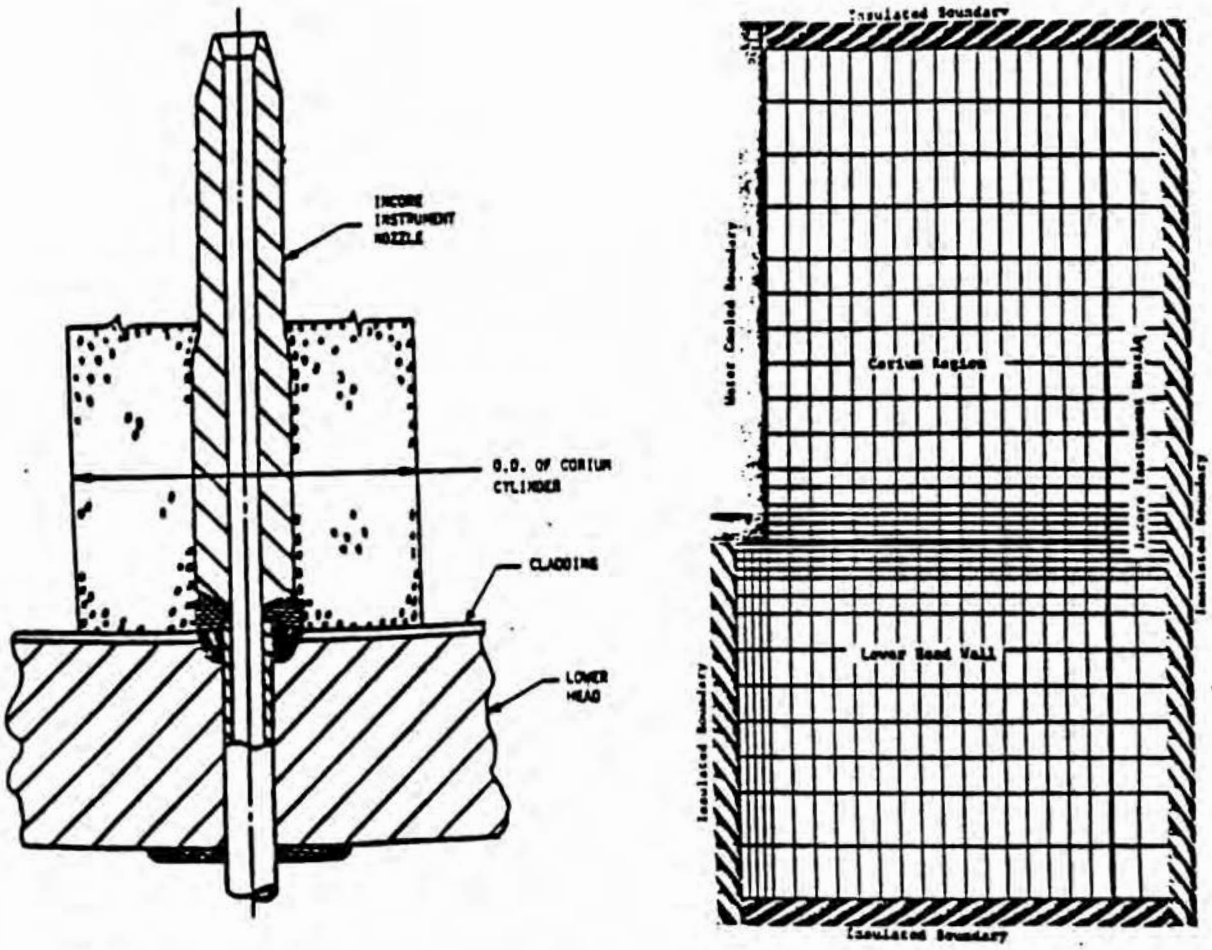


Figure 4. Model of instrument penetration calculations used in FELCON.

water). In any event, the FELCON results predict that neither the lower head nor the nozzle weld material reached their melting temperatures.

The FELCON calculations did predict that the upper part of the Inconel nozzle would reach temperatures well above the melting point. The nozzle was assumed surrounded by hot corium with no cooling, except for conduction down the nozzle and into the lower head. The centimeter or so of the nozzle attached to the lower head is close enough to the heat sink (lower head) to keep the welded portion of the nozzle from melting.

Other calculations were made using FELCON and the stalagmite geometry to study the effects of a lower corium thermal conductivity; a higher heat transfer coefficient between the corium and the lower head, and a lower heat transfer coefficient between the nozzle and the lower head. In no case did the calculated peak temperature in the head or the nozzle weld exceed 1577 K. However, since the Inconel weld melting temperature (1616 K) is less than 100 K higher than the calculated values, and since uncertainties exist in debris cooling characteristics, the possibility exists that the weld region could have suffered some thermal damage. This is especially true because corium could have entered a failed nozzle, drained to the region of the weld, and allowed two-sided thermal attack on the weld region as illustrated in Figure 5. Thus, additional analysis is presently being conducted by ESA, Inc. for various possible heat transfer conditions at the weld location.

4.4 Lower Head Creep Rupture Potential

The thermal response of the lower head was calculated using COUPLE/FLUID, a two dimensional, finite element, transient heat conduction and convection computer code.[24] Because of the wide range of physical characteristics observed in the lower plenum debris (particle size, texture, composition) and, particularly, because of the uncertainty in the composition of the material adjacent to the lower head, three assumed debris configurations were evaluated.[24] These three configurations

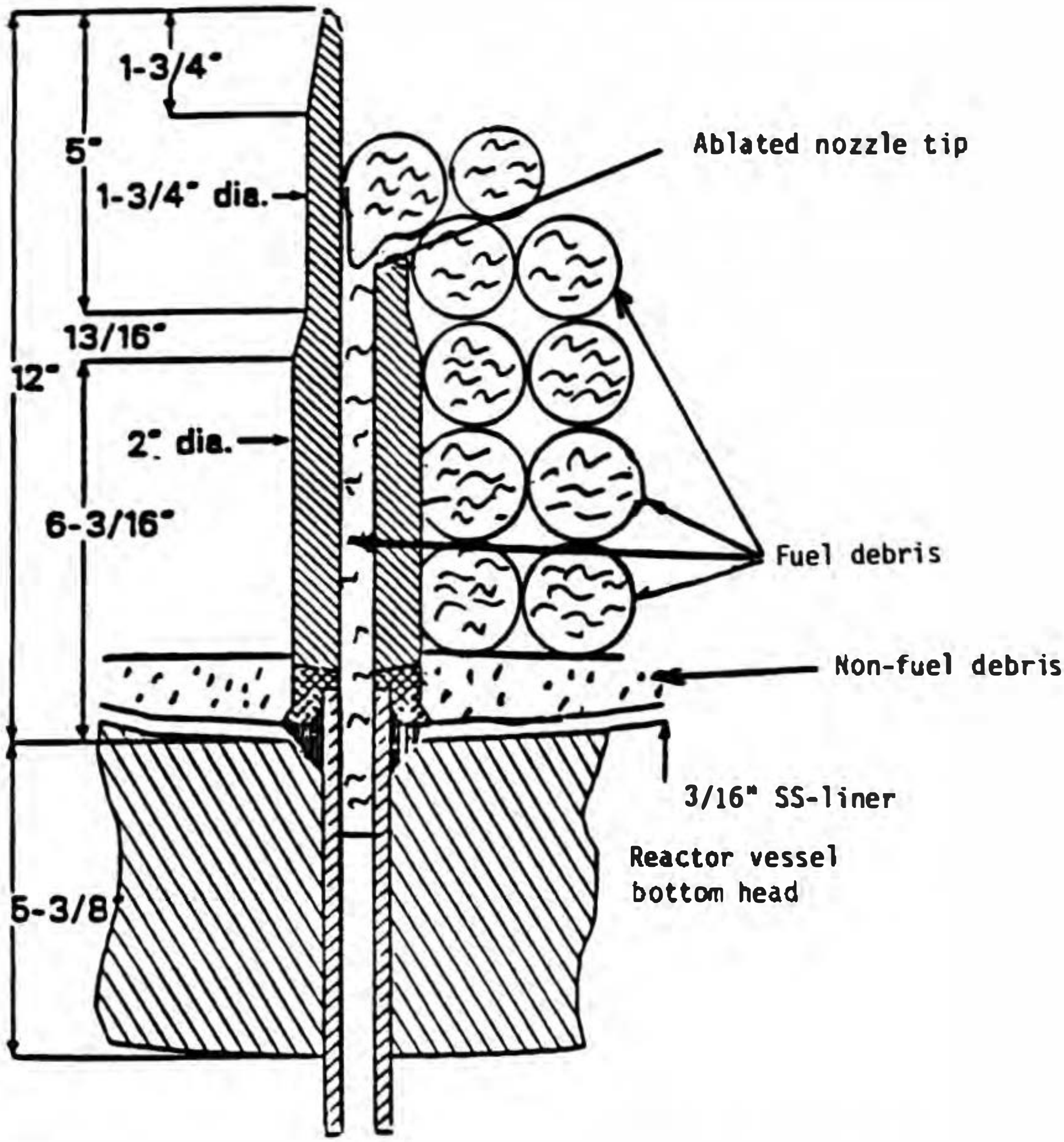


Figure 5. Double-sided attack of instrumentation penetration nozzle.

represent an upper bound, a lower bound, and an intermediate configuration shown in Figure 6.

Figure 6a depicts the configuration thought to represent an upper bound thermal challenge to the vessel. A layer of porous debris was assumed to settle onto the lower vessel head. The interstices between the debris particles were assumed to have filled with molten control rod and/or core structural materials, resulting in a consolidated (zero porosity) layer of metallic/ceramic material adjacent to the vessel. This layer would transmit heat to the lower head more rapidly than the ceramic material. A porous debris bed was assumed to reside on the frozen upper crust of the consolidated region.

Figure 6b represents a lower bound vessel thermal challenge and consists of a porous debris bed separated from the lower head by a layer of solidified control rod material that is assumed to have relocated to the lower head and frozen prior to the major core relocation. In this case, the frozen control rod layer would act as a heat sink and added thermal resistance to heat flow from the debris to the lower head.

The third configuration, thought to be an intermediate configuration between the bounding cases shown in Figures 6a and 6b, is shown in Figure 6c. The intermediate configuration consists of a porous debris bed like that of the lower bound case but with no layer of frozen control rod material between the debris bed and lower head.

In addition to uncertainty in the debris configuration, uncertainty exists relative to the coolability of the various types of debris. Analytical work to evaluate the debris cooling of the TMI-2 upper core debris indicates that for a porous debris configuration, quench times on the order of 20 min are expected.[12] However, the TMI-2 in-core thermocouple data recorded during the accident and post-accident measurements to determine the location of the thermocouple junctions both suggest that the lower plenum debris may have been at elevated temperatures for up to 2 h after the relocation event.[11] For the three debris configurations shown in Figure 6, the impact of uncertainty in coolability

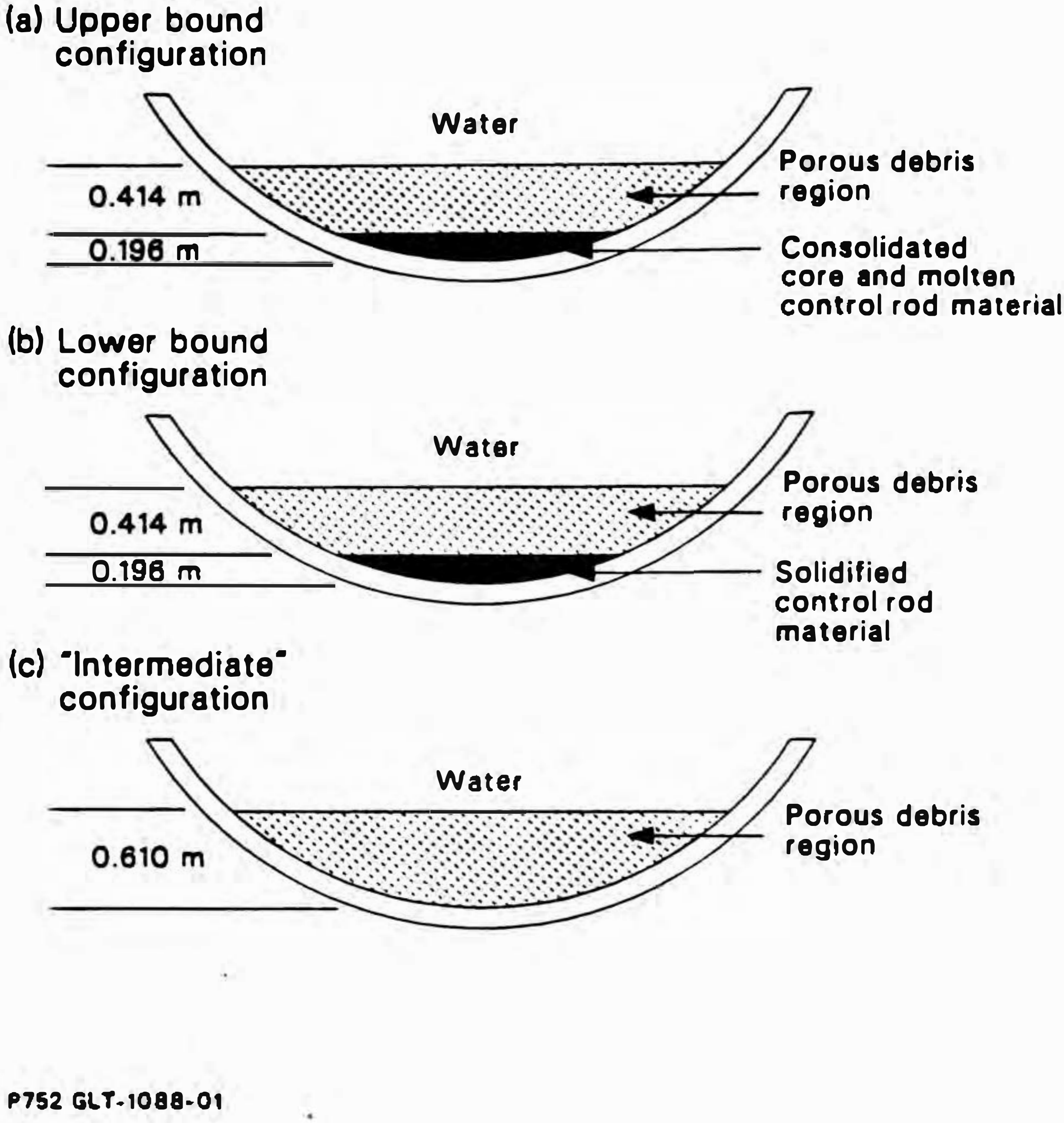


Figure 6. COUPLE/FLUID models for upper- and lower-bound and intermediate lower head region configurations.

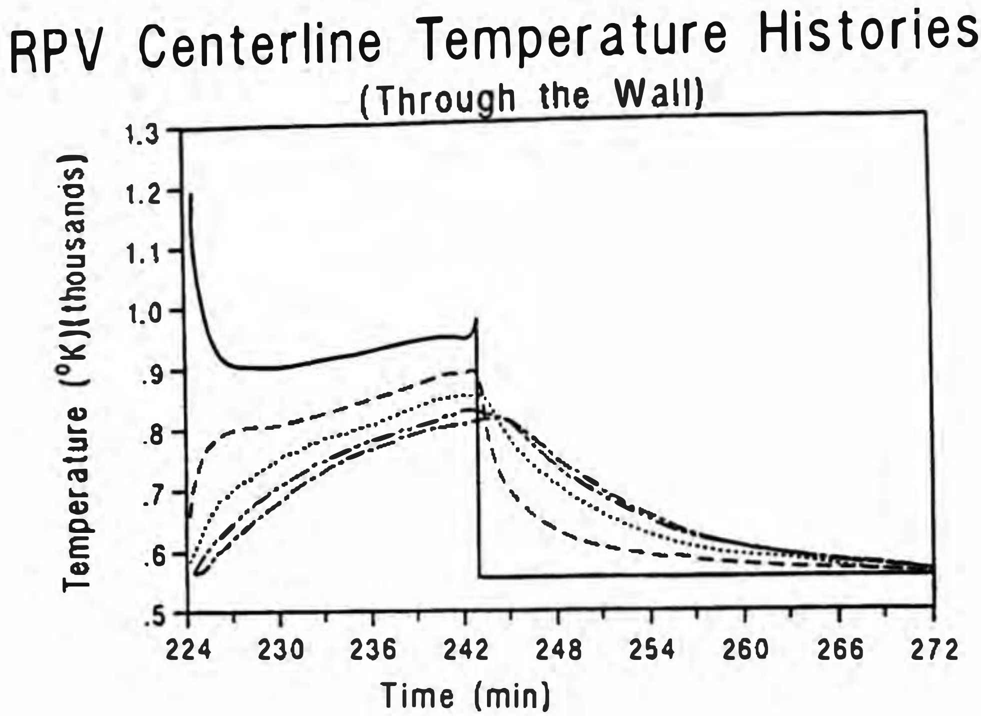
of the porous debris on the lower head thermal response was also investigated.

A lower bound for the porous debris cooling was assumed to be no coolant penetration into the debris bed. Only the upper surface of the debris bed was assumed to quench; and this quench was assumed to occur at the beginning of the temperature transient. Thus, heat transfer from the debris bed was limited by heat conduction through the porous debris and convection at the debris outer surface.

An upper bound for the porous debris cooling was approximated by assuming coolant penetration into the debris bed and complete quenching of the debris. The quench front was assumed to start at the periphery of the debris bed and move radially inward. The top surface of the debris bed was assumed to be quenched at the initiation of the transient with no further vertical quenching. The radial quench front versus time was determined by assuming the energy removal rate from the debris to be constant during the 20-min quenching period. The debris quench configuration and associated quench front position versus time based on these assumptions are shown in Figure 7.

COUPLE/FLUID calculations were performed for the upper, intermediate, and lower bound debris configurations using assumed dry and quenched porous debris cooling for each of the debris configurations. For the calculations it was assumed that the initial debris temperature was 2500 K. Results of the calculations are shown in Figure 8.[25] For all three cases of upper, intermediate and lower bound, using a dry porous debris bed, the lower head temperatures trend upward with time. For all three cases using a quenched debris bed, the lower head temperatures eventually decrease.

The variation in the operating system pressure during and after the core relocation at 225 min is shown in Figure 9. This pressure, which was measured with pressure transducers during the accident, was the major contributor to the mechanical loads on the lower head. The combined pressure on the lower head resulting from water in the RPV and the weight of the core material distributed over the lower head amounted to about 0.07



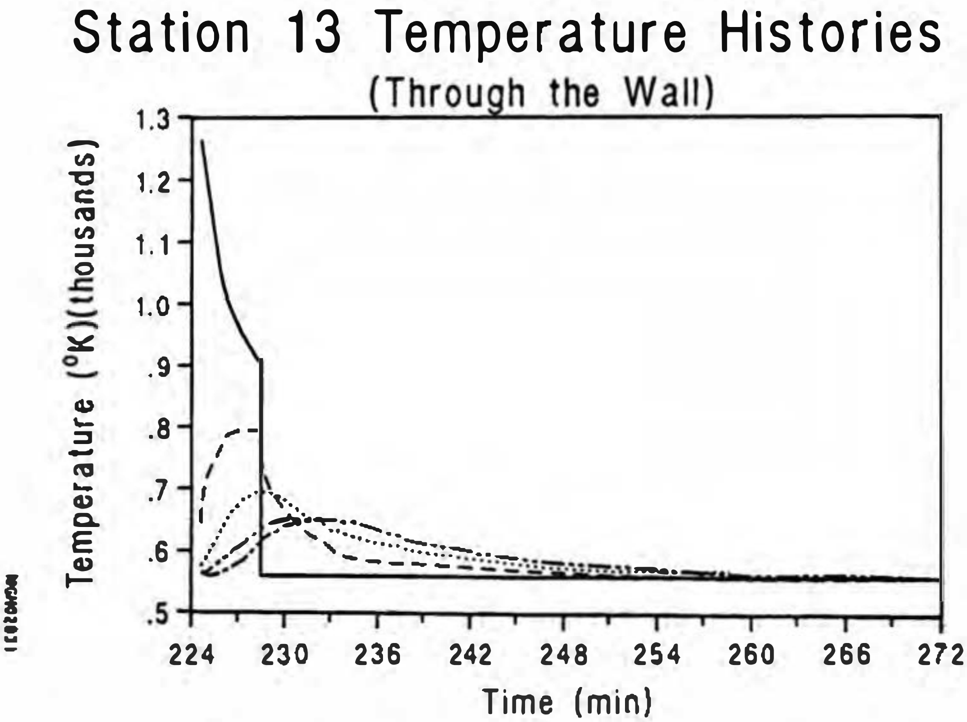


Figure 7. Assumed quench through vessel wall at central and peripheral locations used for COUPLE/FLUID calculations.

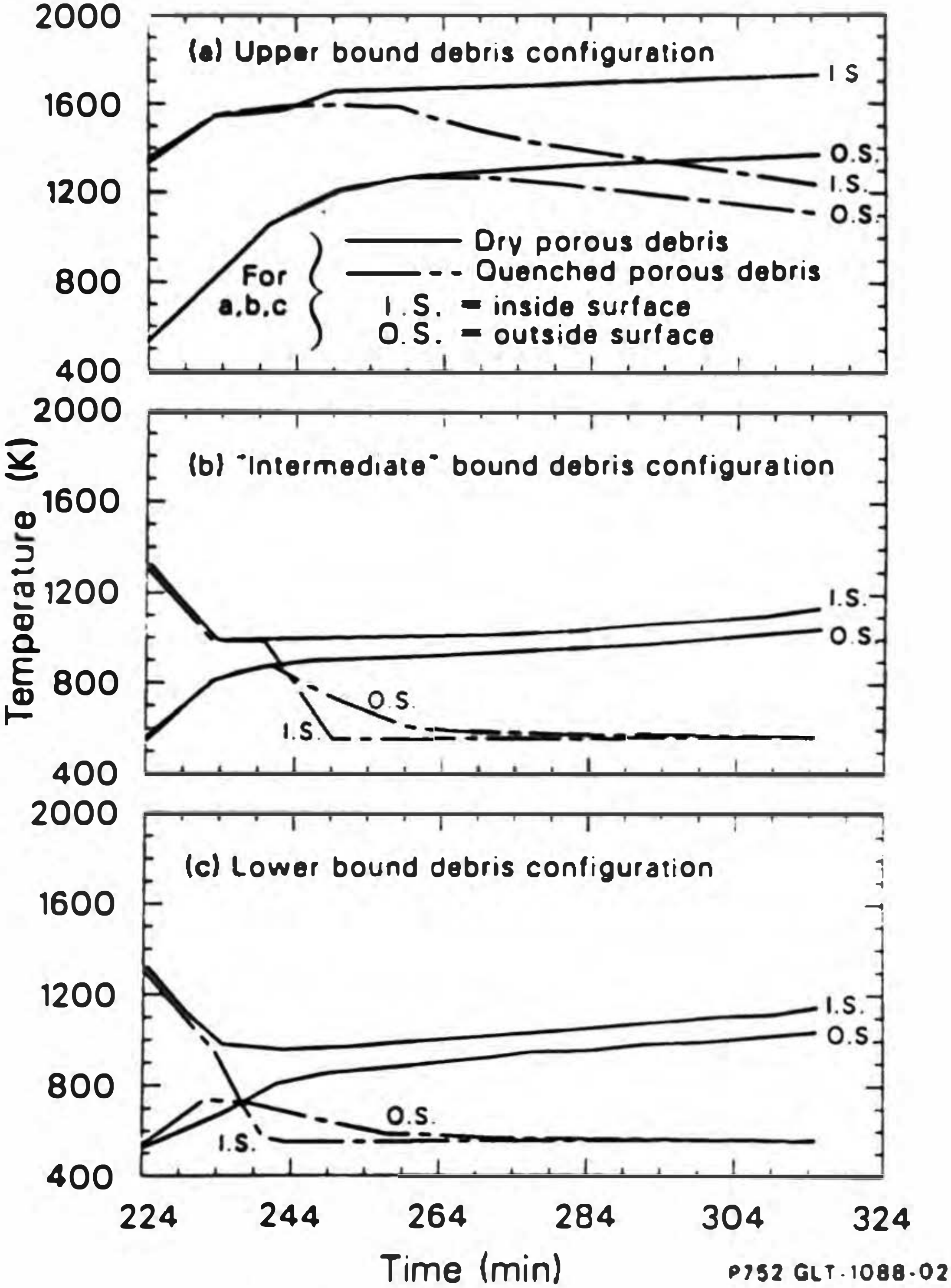


Figure 8. Results of COUPLE/FLUID calculations.

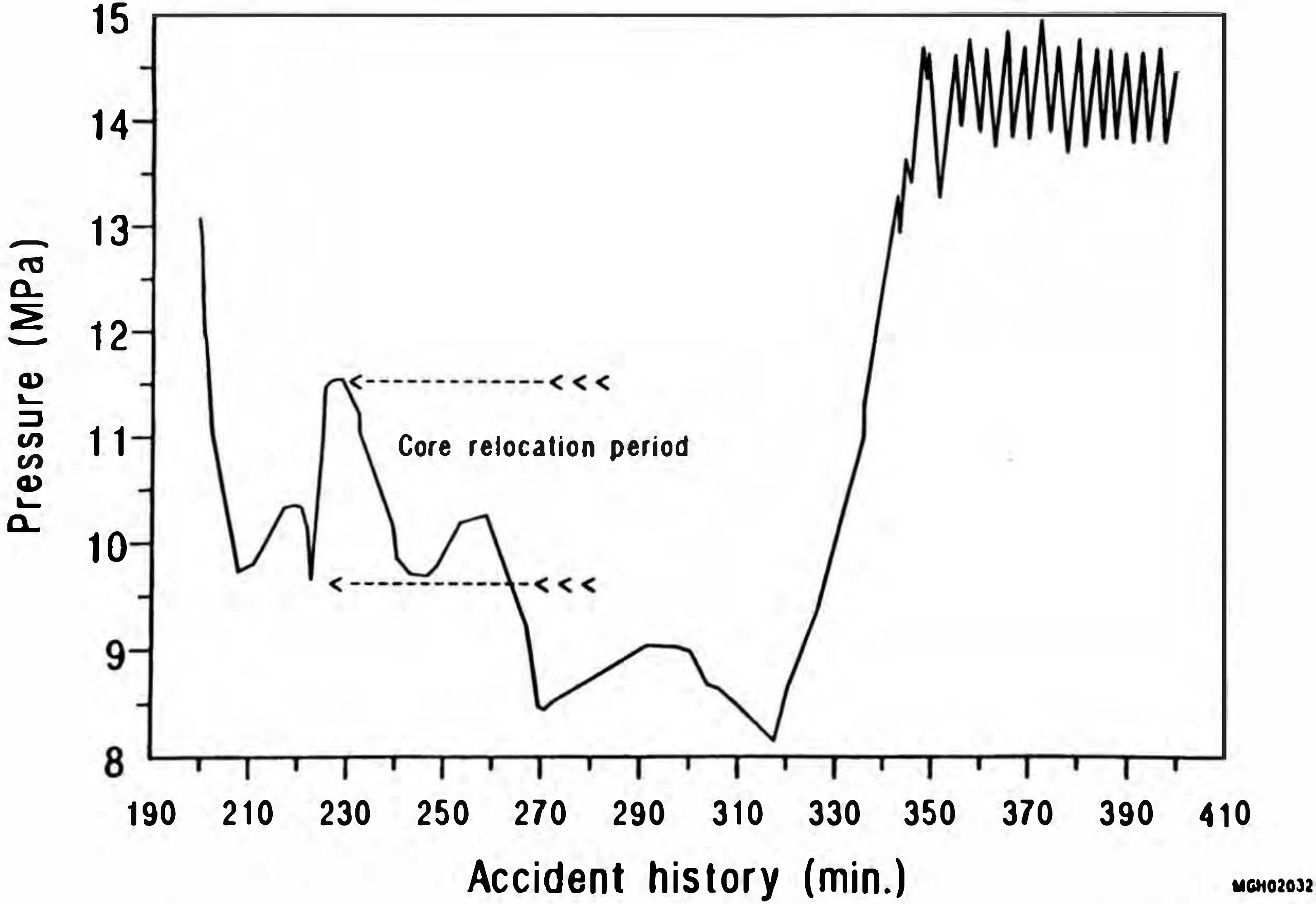


Figure 9. TMI-2 system operating pressure during and after core relocation.

to 0.14 MPA compared with operating system pressures as high as 11 MPA during the relocation period. The weight of the reactor vessel is transferred through the cylindrical portion of the RPV down to the skirt support (see Appendix A, Figure A) which is well away from the high temperature region of the lower head. Thus, the system transient pressure was the only significant force causing stress in the lower head. This type of stress is not self-limiting, i.e., it does not reach a limit as strains increase. Therefore, this load must always be carried by the lower head to maintain structural integrity of the RPV. A simple calculation of the tangential stress in the lower head, a uniform stress through the wall resulting from the system pressure, indicates a minimum stress resulting from system pressure during the transient of 74 MPA.[26]

The effects of creep on a structure's capacity are quite complex and not easily determined when temperatures are not uniformly distributed throughout the structure. The significant effects of creep begin to occur at temperatures above 644 K (700°F) for carbon steels used in RPVs. However, since ultimate strength is a temperature-dependent but not a time-dependent material characteristic, as is creep, it can be used to evaluate the TMI-2 lower head response for some of the calculated temperature distributions shown in Figure 8. If one extends the ultimate strength curve of SA533 Grade B Class 1, the reactor vessel lower head material, the ultimate strength is about 69 MPA[27] at 1144 K. This is about the minimum stress (74 MPA) induced in the lower head by operating pressure during the early stages of the core relocation when the lower head temperatures would be the highest. For all lower head debris configurations for the dry debris bed, and for the upper bound configuration with the quenched debris bed, the lower head temperatures reach 1144 K (see Figure 8, for the upper bound, quenched debris bed the temperature would reach 1144 K at about 7000 s) and therefore the lower head would be predicted to fail for these four cases.

Since failure of the reactor vessel lower head did not occur, it can be inferred that both the upper bound configuration and the dry debris bed assumption are both too conservative. Thus, a consolidated (zero porosity) previously-molten mixture of core materials most likely does <u>not</u> completely cover the lower head and the debris bed that does cover the lower head was probably quenched. Some combination of the intermediate and lower bound configurations with possibly very localized corium is now thought to be the most likely condition of the debris resting on the lower head of the TMI-2 RPV. These calculational results emphasize the need to better understand the condition of the molten core material adjacent to the lower head and to understand the mechanisms controlling breakup (debris formation) of the previously-molten core material. The calculational results also emphasize the importance of debris coolability in limiting the temperatures reached by the lower head of a RPV.

Additional calculations were performed for the quenched, intermediate configuration (porous debris bed resting on lower head) to determine the creep of the lower head for this condition in an effort to quantify the margin-to-failure of the lower head. The axisymmetric model of the lower head, skirt, and cylindrical portion of the RPV used in the calculations is illustrated in Figure 10. The ABAQUS[28] nonlinear, structural finite element code was used for the creep calculations with the lower head temperatures calculated using COUPLE/FLUID used as input to the ABAQUS code. Since the scope of the creep analysis was limited to an axisymmetric response of the lower head to the core relocation, the plate material, SA533 Grade 8 Class 1, was the primary material of concern. Elastic-plastic and creep properties for SA533 Grade 8, Class 1 material have been documented from tests up to 922 K (1200°F)[28]. Stress-strain curves, Youngs modulus and proportional limit stresses, material creep properties, and mean coefficients of thermal expansion were all derived from Reference 28 for use in the ABAQUS calculations. The ABAQUS code used the 922 K properties for temperatures above 922 K.

The results of the ABAQUS calculations[26] shown in Figures 11 through 14 illustrate four time segments of the temperature and resulting stress distribution in the axisymmetric structural analysis of the TMI-2 lower head after core relocation. Contour lines connect regions of equal temperature in the four plots on the left side of each figure while on the right side they connect regions of equal stress. The time, in seconds, after initiation of the thermal transient (time of core relocation) at

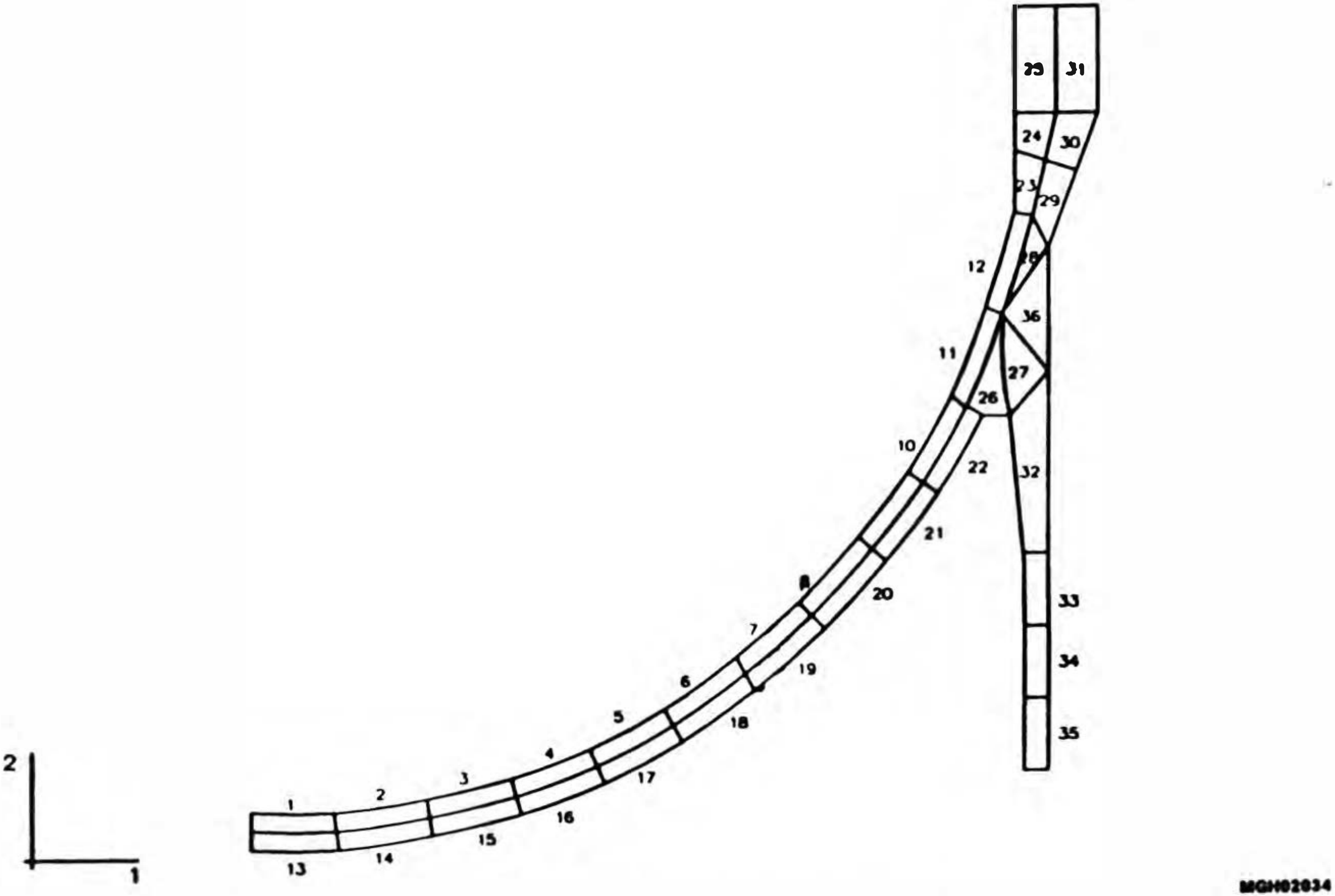


Figure 10. ABAQUS model of TMI-2 reactor pressure vessel lower head.

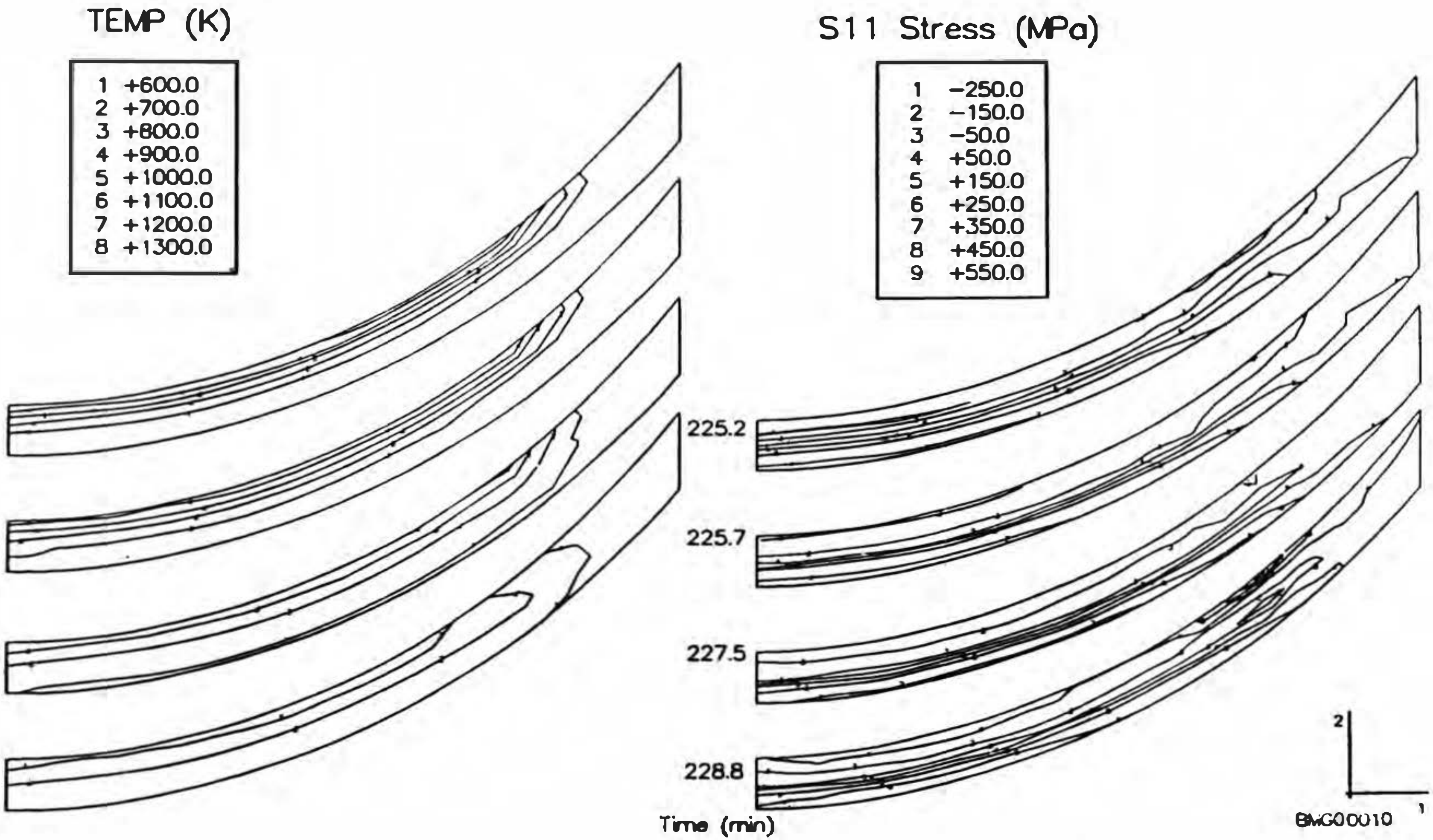


Figure 11. ABAQUS-calculated vessel inelastic response (225.2 to 228.8 min).

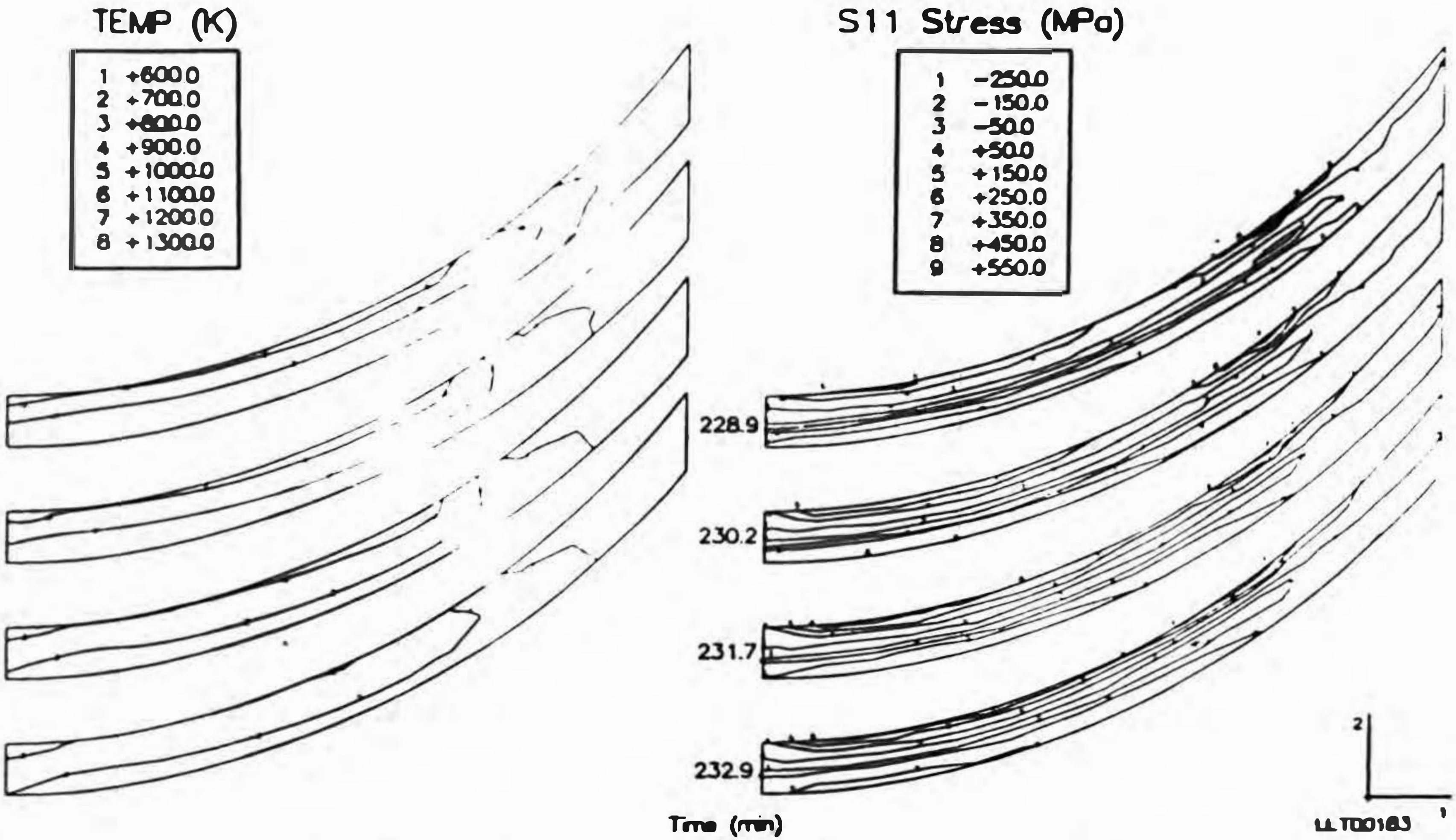


Figure 12. ABAQUS-calculated vessel inelastic response (228.9 to 232.9 min).

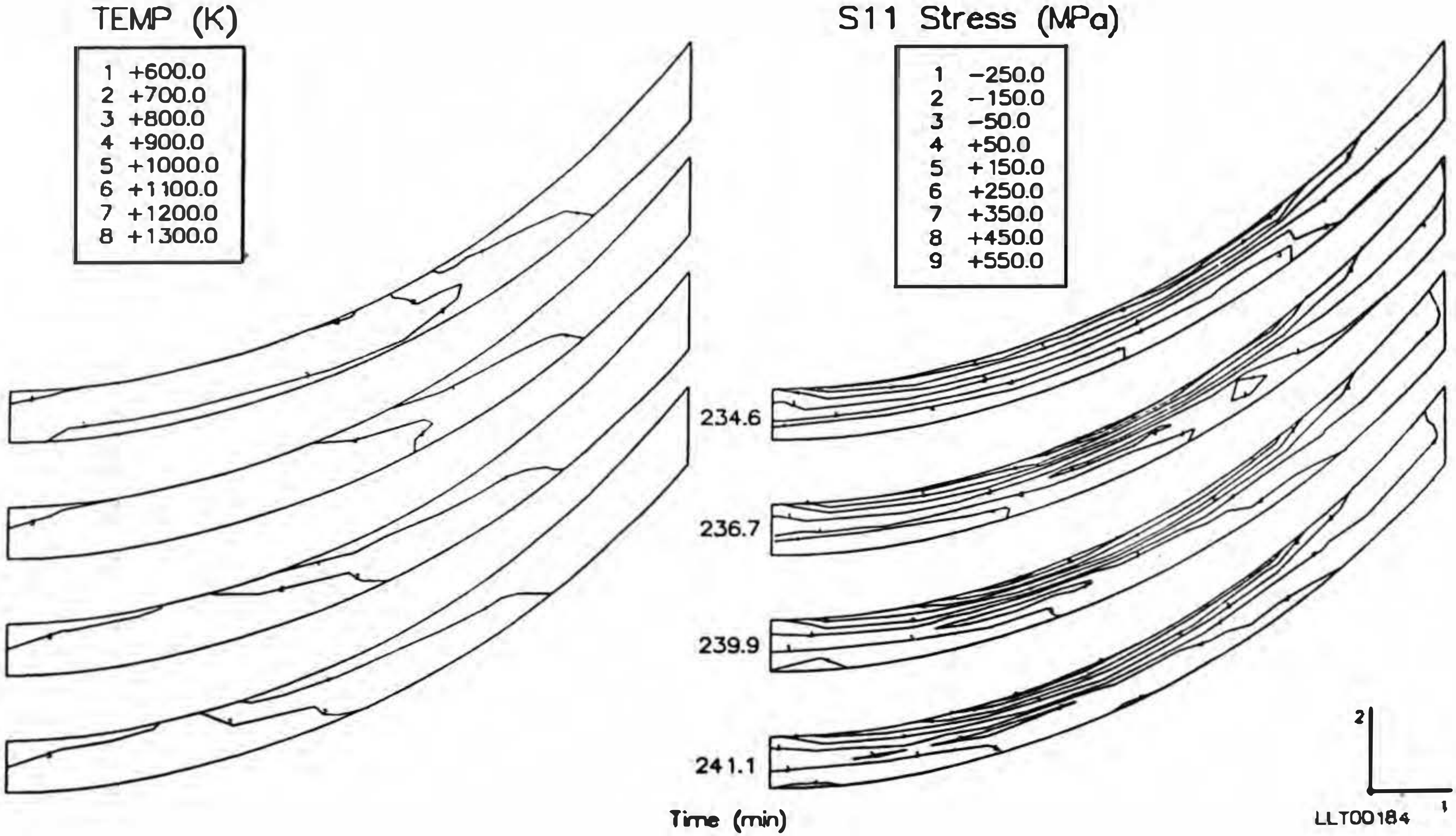


Figure 13. ABAQUS-calculated vessel inelastic response (234.6 to 241.1 min).

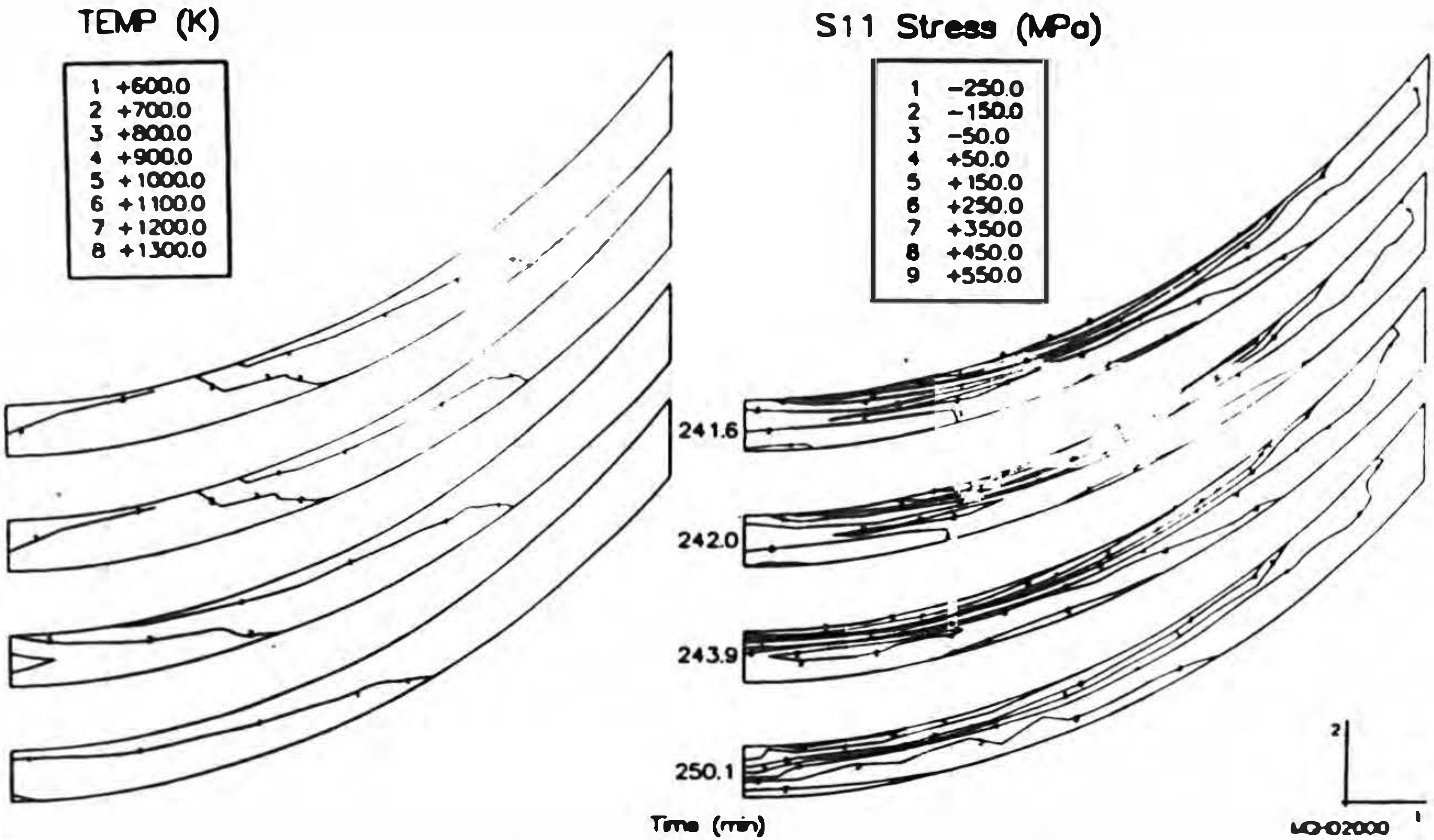


Figure 14. ABAQUS-calculated vessel inelastic response (241.6 to 250.7 min).

which these stresses were calculated, is indicated in the center of each figure. The stress component illustrated is oriented parallel to the "1" axis indicated in the lower left corner of each figure. This component of stress was selected because it was typically larger than the hoop direction component and offers the clearest picture of the stress scenario resulting from this postulated thermal transient. The four figures cover a period of about 25 s after core relocation.

As indicated in Figure 11, the unquenched temperature gradient is maintained in the vessel wall until about 225 s. The high initial gradient decreases with increasing time because of the exponential reduction of the inner wall surface temperature over this part of the transient. During this period, stress distribution through the wall consists of compression on the inside and tension on the outside as the hotter inner portion expands and is constrained by the colder outer portion. Note the high stress gradient at the midsurface, which would also indicate high shear stresses in the midsurface region. This is, essentially. a bending stress distribution through the wall during this interval.

Approximately 229 s into the transient the quench starts at the outer region of the lower head and proceeds inward. The quench is indicated by the temperature contours, shown in Figure 11 at 229 s, curling back toward the RPV centerline near the inner surface at the outer edge of the debris bed. This is in response to the cooling inner wall temperatures. This produces a localized tension-compression-tension stress gradient through the wall at the quench front. This gradient attenuates spacially towards the RPV centerline from the current quench front location but maintains its basic configuration.

In the succeeding time frames, the temperature contours migrate towards the RPV centerline as the quench front progresses towards the center of the debris bed. A high local stress gradient through the wall continues to accompany the quench front as it moves. Trailing the quench front, the localized tension-compression-tension distribution gives way to a residual bending stress which has reversed direction from the pre-quench distribution. At 1601 s, the temperature gradient is much more uniform and lower than the initial phases signaling the end of significant thermal effects on the vessel wall.

Throughout the thermal transient, note that the highest stress gradients generally occur on the inner half of the vessel wall. This might offer a signature for this postulated temperature scenario which could be identified in post-accident examination.

The overall stress gradients in the RPV lower head calculated to result from the relocation of molten core material are dominated by the thermal gradients in the lower head wall. Plastic and creep deformations occur causing redistribution of stress throughout the wall thickness. Stress gradients also undergo reversals during the temperature transient. Plastic deformations occur at various locations throughout the wall thickness at various times during the transient. Both compressive and tensile yielding occur on the inner half of the wall while primarily tensile yielding is exhibited in the outer portion. Even though plastic deformation was widely distributed, it was not very high. Maximum plastic and creep strains were each calculated to be in the 1% range. Reference 29 reports creep rupture strains at 783 K (950°F) of about 35% and elongations at ultimate strength and 783 K of about 25%. It is estimated that inclusion of physical properties for temperatures above 922 K would increase plastic deformations in the majority of the lower head wall only a small amount because of the very localized distribution of high temperatures on the inside surface of the lower head.

The ABAQUS-calculated results for the temperature input calculated with COUPLE/FLUID using a quenched, porous debris bed indicate that rupture of the lower head is not very probable. The calculated high temperatures were restricted to the inside surface of the lower head and the temperature transient was not long enough to mobilize any significant creep in the lower head wall that would lead to rupture. For this type of temperature distribution, creep only causes the high thermal compressive stresses on the inner surface to relieve rather quickly and cause the lower head wall to carry the load in its outer portions.

The full penetration weld connecting the RPV forging with the plate material in the lower head is a more probable location for the RPV to fail. This is because the welding process reduces ductility, and therefore, allowable creep strains in the head affected zone.[30] However, because this weld is much higher on the lower head and, based on the ABAQUS analysis and the postulated corium relocation time, temperatures at the weld location could not have been high enough to cause substantial creep at this location.

Another area of concern for creep rupture is around the instrument penetration tube nozzle weld. As previously suggested, failure of an Inconel nozzel could lead to the flow of hot corium down the instrument tube to the weld location leading to the potential for lower head creep failure at this very localized position.

Results of the thermal and creep analyses of the response of the lower head to the relocation of the molten core debris provide important insight relative to the understanding of the latter stages of in-vessel core damage progression leading to the attack of the RPV lower head. Further characterization of the lower plenum debris, especially the debris adjacent to the lower head, and the lower head itself will be necessary to reduce the uncertainty in the estimated mechanical response of the lower head. Physical properties of SA533 Grade B, Class 1 material need to be measured as a function of time and temperature for temperatures greater than 922 K. When these measurements are completed, samples of the RPV lower head can be compared with the measured samples to determine the temperature distribution and the creep in the RPV lower head. This data will then quantify the margin-to-failure of the TMI-2 RPV lower head and allow benchmarking of computer codes like COUPLE/FLUID and ABAQUS such that they can be used with confidence for the prediction of RPV lower head failure.

5. NRC/OECD AND DOE PROGRAM INTEGRATION

This section describes the integration of the NRC/OECD-sponsored Vessel Integrity Program (VIP) for TMI-2 and the DOE-sponsored TMI-2 Accident Evaluation Program (AEP). The objectives of the DOE-sponsored program include developing the physical understanding of the TMI-2 accident and applying this knowledge to assess severe accident scenarios and consequences. The NRC, in conjunction with the OECD, is sponsoring a new international program (VIP) to: (a) characterize the extent of lower head damage; and, (b) estimate the margin of safety to vessel failure. Integration combines the DOE and NRC/OECD results to provide a full understanding of the accident including the attack of the lower head of the TMI-2 RPV. The following subsections briefly describe the tasks covered by each of the two programs, which include tasks necessary to provide effective interconnection and integration of the programs.

5.1 Status of DOE Accident Evaluation Program

Results of the DOE AEP are included in the descriptions given in Sections 2, 3 and 4 of this report. The status of the sample acquisition and examination efforts completed or planned in the DOE AEP are listed in detail in Appendix B. The DOE AEP includes examination of samples from: (1) the reactor coolant system; (2) the ex-RCS (containment building, auxiliary building, etc.); (3) control rod leadscrews; (4) the upper core debris bed; (5) fuel rods and control rods from the core periphery; (6) core bore samples from the central, molten core region and from fuel rods, control rods and burnable poison rods from below the central molten core region; and (7) debris from the lower plenum. The DOE program of analysis and interpretation of the TMI-2 accident data will produce:

1. A description of the core damage and relocation sequence of events (updated accident scenario)

2. An assessment of the location of fission products distributed in the plant as a result of the accident (fission product distribution)

3. A computerized data base containing TMI-2 research results, examination data and supporting analysis

4. The TMI-2 Standard Problem Exercise; and,

5. A comprehensive ANS topical meeting on the TMI-2 accident.

In addition to the samples described above, other samples have been acquired and placed in archival storage including: (1) five "rocks" from the possible molten core relocation path at core position R6; (2) large and small "rocks" from the central molten core region after it was pulverized; and, (3) fuel rod assembly upper end fittings and spiders. Additional samples to be obtained by GPU Nuclear during defueling include: (1) structural and core material from the core bypass region and the core support assembly region; and, (2) large volume samples from the lower plenum region and debris samples from adjacent to the lower head. Additional sample examinations are being conducted or planned by foreign countries. Lists of the TMI-2 samples furnished to or requested by European countries, Canada, Korea and Japan for examination are listed in Appendix C. These countries are all members of a Task Group of the Committee for the Safety of Nuclear Installations (CSNI) of the OECD. Debris samples were shipped to Europe and Canada during the summer of 1987. The integration of DOE's Accident Evaluation Program and NRC/OECD's Vessel Integrity Program incorporates available results of the foreign country sample examinations into the total package of TMI-2 sample analysis and documentation.

5.2 NRC/OECD Sample Acquisition Overview

Based on the results presented in Sections 2, 3 and 4 of this report, it is clear that the TMI-2 reactor vessel lower head needs to be examined for evidence of thermal and chemical attack, and the debris adjacent to the lower head needs to be characterized so that accurate heat transfer calculations of the lower head response to the relocation of the corium can be made. Principle areas of investigation of the lower head and adjacent debris will include:

1. The east quadrant in the region where it is believed that the primary relocation of corium occurred

2. The radial center where the maximum thickness of debris occurred and,

3. The north quadrant in the regions where a wall of consolidated previously molten material has been observed.

Samples obtained at the above locations will not be through wall samples but will be thick enough to provide adequate thermal and metallurgical analysis (the ABAQUS calculations described in Subsection 4.4 predict that any plastic deformation will be on the inner portions of the lower head). The samples obtained from the east quadrant will be used to determine the response of the lower head and instrument penetration weld joints where these structures were attacked by relocated core debris and may have been attacked by jets of corium. Samples obtained at the radial center of the vessel will be used to determine the response of the lower head and instrumentation penetration welds under the deepest thickness of debris accumulated on the lower head and where there may be a layer of non-fuel materials such as silver from the control rods. Samples obtained from the region where the wall of consolidated material exists will be examined to determine the vessel response to this attack. Temperatures could have reached 1100 K in this region and temperature measurements obtained from relocated (downward from the core) in-core thermocouple junctions reached 1600 K in some areas hours after accident initiation and this could be an indication of thermal attack on the lower head.

Details of the examination of the lower head samples and the debris samples adjacent to the lower head are described in Section 6.

5.3 Integration of the NRC/OECD and DOE TMI-2 Programs

Table 2 and Figure 15 are the activity list and the schedule, respectively, for the proposed integration of the DOE and NRC/OECD TMI-2 Programs. Table 2 identifies the activities that are: (a) DOE sponsored; (b) NRC/OECD sponsored, and; (c) considered to be needed to connect the analyses of materials in the core region with the analyses of the lower head and the lower vessel debris and which are currently unsponsored. The remaining sample examinations are designed to provide data that can be integrated into the TMI-2 accident understanding such as: the mechanisms responsible for transport of the molten core material to the lower head; probable flow pathways; viscosity of the molten materials; relocation rates; and, maximum temperatures and method of surface attack. The resulting improved accident understanding will aid in the determination of RPV safety margins and development of improved analytical tools for severe accident mitigation and management.

The products of the integration of the DOE and NRC/OECD programs will include (listed chronologically):

1. TMI-2 Sample Examination

   a. DOE and NRC/OECD sample examination reports

   b. Final update of the DOE sample examination and examination work plan

2. Data Analysis

   a. An ANS Topical Meeting and Proceedings (technical papers) on the TMI-2 accident (DOE)

   b. A fission product inventory report after the completion of defueling (DOE)

TABLE 2. DOE AND NRC/OECD TMI-2 INTEGRATED PROGRAM TASKS

Task Description	Sponsor
Complete TMI-2 Sample Examination	
Core bore SEM/WDX	DOE
Lower head loose debris physical and radiochemical examinations	DOE
Complete publication of sample exam final reports	DOE
Core position R6 consolidated core material	GPU/DOE[a]
CSNI (DOE partnership) sample exams	CSNI
Lower head loose debris metallurgy	GPU/DOE[a]
Core bypass region core debris	GPU/DOE[a]
Baffle plate interaction zones	GPU/DOE[a]
Instrument guide tube interaction zones	GPU/DOE[a]
Core support assembly plate interaction zones	GPU/DOE[a]
Lower head sample acquisition	GPU/NRC/OECD
Lower head sample examination	NRC/OECD
Lower head companion sample acquisition	GPU
Lower head companion sample examination	NRC/OECD
Instrument string nozzle interaction zones	GPU/DOE[a]
Annual report	DOE
Sample disposal (DOE samples)	DOE
Sample disposal (NRC samples)	--[b]
TMI-2 Data Analysis	
TMI ANS Topical Meeting paper publication	DOE
Core damage progression scenario	DOE
Fission product inventory report	DOE
Impact of TMI-2 Lessons Learned report	DOE
Defueling and CSNI (DOE partnership) examinations results data base update	--[b]
Reactor vessel lower head data integration	NRC/OECD
Lower head examination data base date	NRC/OECD
Lower head creep rupture and margin-to-failure analysis	NRC/OECD
NRC/OECD TMI-2 VIP final reporting	NRC/OECD
TMI-2 Final Report	--[b]

TABLE 2. (continued)

Task Description	Sponsor
Standard Problem Support	
Compare CSNI results for Phase 4 and publish final report	DOE
Compare NRC/OECD examination results with standard problem calculations	--[b]

a. Sample acquisition by GPU, shipping and archiving by DOE, sponsor for examinations to be determined.

b. Sponsor to be determined.

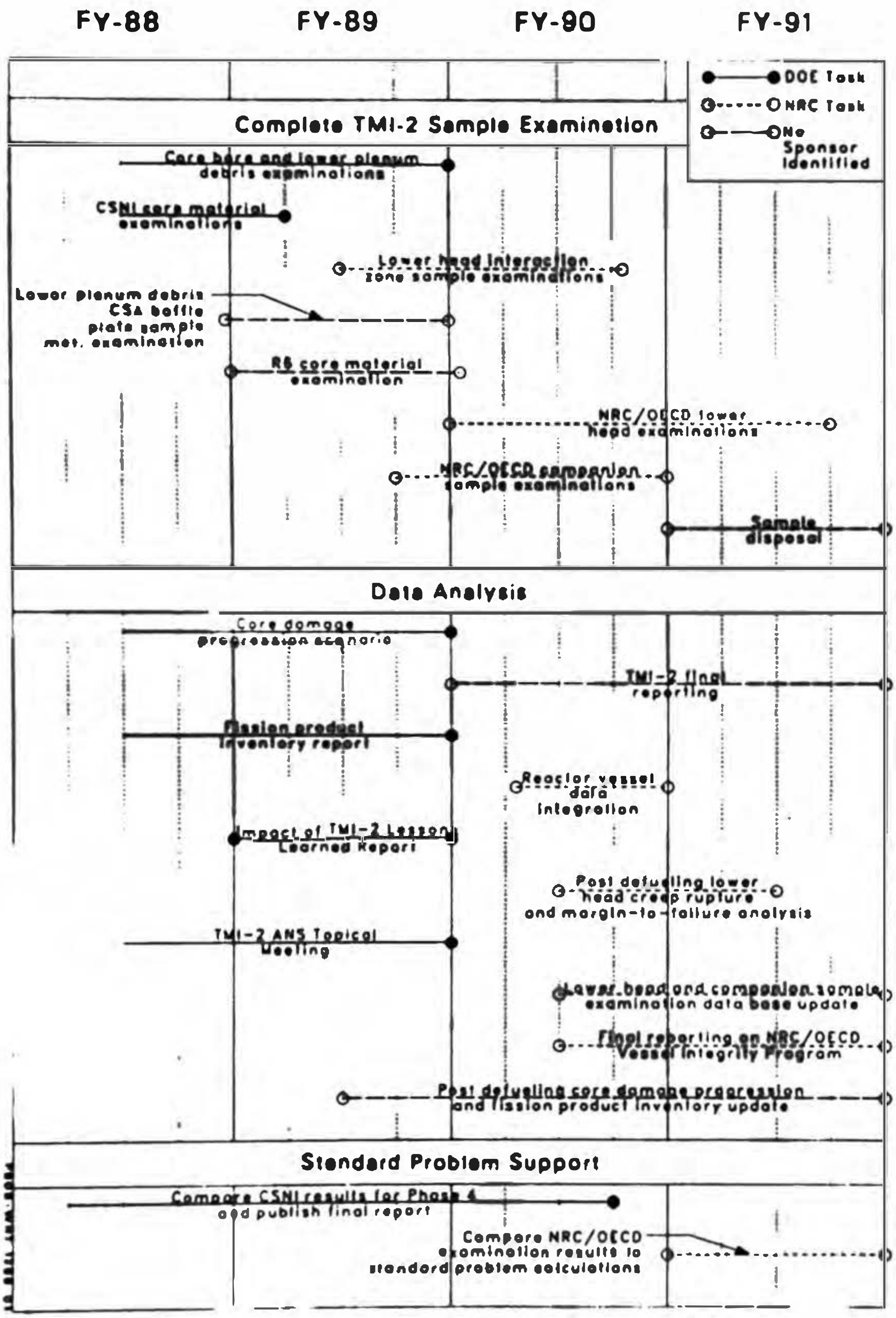


Figure 15. Integrated DOE and NRC/OECD TMI-2 program tasks.

c. An accident scenario update after defueling (DOE)

d. A resolution-of-issues topical report on core behavior safety (DOE)

e. A topical report on the reactor vessel behavior and margin-of-safety during the accident (NRC/OECD).

3. Standard Problem Support

a. A report on the OECD-sponsored TMI-2 accident standard problem exercise (DOE)

b. An incorporation of the NRC/OECD VIP results into the Standard Problem exercise (NRC/OECD).

4. Final Documentation and Reporting

a. Expansion of the accident data base to include additional results of defueling

b. Expansion of the accident data base to include the NRC/OECD lower head sample examinations (NRC/OECD)

c. Expansion of the accident data base to include the results of foreign country sample examinations

d. Final report on TMI-2 Program.

6. NRC/OECD VESSEL INTEGRITY PROGRAM

Details of the acquisition and examination of samples from the TMI-2 lower vessel head and samples of debris directly adjacent to the lower head are described in this section together with a description of the analysis of the lower head response to the temperature transient which will result in an estimate of the margin-to-failure. Several organizations are involved in the cooperative effort to achieve the goals of the NRC/OECD Vessel Integrity Program. In addition to the NRC and the OECD partners, Argonne National Laboratory (ANL)-East, the Idaho National Engineering Laboratory (INEL), General Public Utilities (GPU)-Nuclear, MPR Associates, Inc., and Power Cutting Incorporated (PCI) will be responsible for various parts of the TMI-2 Vessel Integrity Program. Cooperation among the participants is imperative to assure success of the program. Responsibilities of the various organizations are clearly specified in the following two subsections which describe: (1) lower vessel head sample acquisition, sample preparation and shipment, experimental determination of time-temperature plots of the response of archive (or equivalent) samples of the lower vessel head, metallurgical examination, determination of the most likely temperature distribution in the lower head, and program integration; and, (2) companion sample acquisition, shipment, and metallurgical examination, calculation of the creep stresses in the lower vessel head as a function of time during the accident, determination of the margin-to-failure of the lower vessel head, and program integration.

6.1 Lower Vessel Head

The objectives of the lower vessel head sample examinations are to: determine the mechanical properties and metallurgical characteristics of the lower head steel that resulted from the TMI-2 accident; deduce a scenario for and determine the temperatures of the steel in the lower head during the accident; and, assess the integrity of the lower head and margin-to-failure based on these examinations.

6.1.1 Lower Vessel Head Sample Acquisition

Wedge-shaped samples will be cut from the lower vessel head in regions where damage to the lower head would be expected to have the highest probability of damage. In these regions samples that contain instrument penetrations, and samples that do not contain instrument penetrations will be obtained. Samples will also be obtained for reference from a lower head region where no damage is expected. MPR Associates, Inc. has the responsibility for obtaining the samples and shipping them to ANL-East. MPR Associates will use metal-disintegration-machine (MDM) techniques that were designed and tested and whose operation will be overseen be Power Cutting Incorporated to obtain the wedge-shaped samples.

Three types of samples will be obtained. Type 1 samples will be cut from the central region of the lower head. Top and side views of Type 1 samples are illustrated in Figure 16 for both instrument penetration and non-instrument penetration locations. A possible sample preparation scheme for Creep/Tensile, 0.5 Compact Tensile, and Charpy V specimens is also shown in Figure 16. Type 2 samples will be cut from regions somewhat distant from the center of the lower head denoted as "hillside" regions. Type 3 samples will be cut from regions that are still further from the center of the lower head and are denoted as "steep hillside" regions. Type 2 samples are illustrated in Figure 17 and Type 3 samples are illustrated in Figure 18. The samples will not be through wall samples in order to avoid leakage from the vessel, and therefore sample cutting must be done underwater. Samples will be taken half way through the 0.1397 m thick lower head. Both the thermal ablation analysis and the creep analysis described in Sections 4.2.2 and 4.4, respectively indicate that half-wall thickness samples will adequately encompass lower head regions where both thermal ablation and plastic deformation would be expected to occur.

A TMI-2 core location diagram is shown in Figure 19 that indicates proposed lower head sample locations, lower plenum debris bed depth measurements, damage zone locations, incore instrument positions, mechanical probe locations, and locations where damage occurred to the baffle plates (the probable relocation path through the core and the core

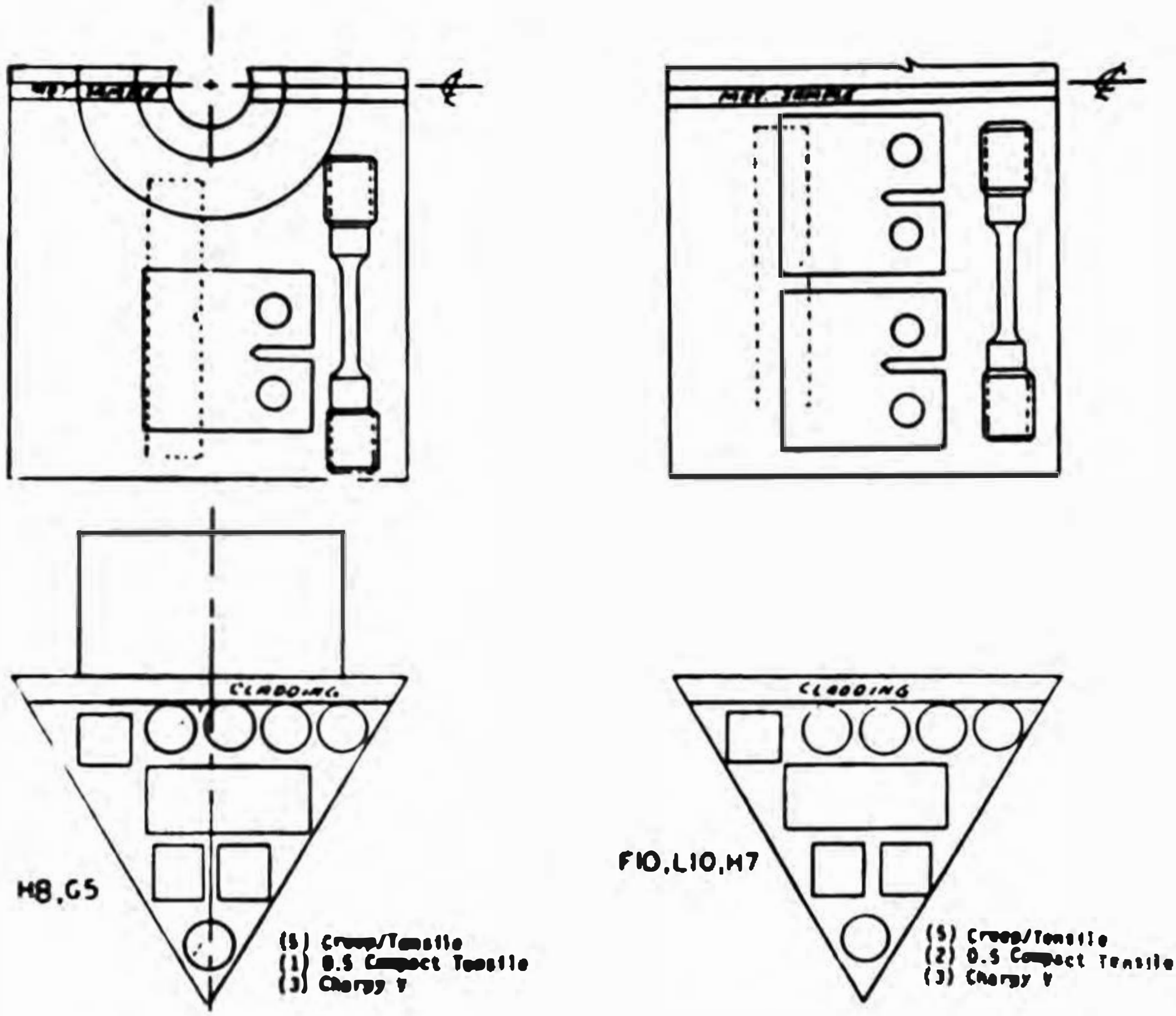


Figure 16. Central region TMI-2 lower-head samples (Type 1).

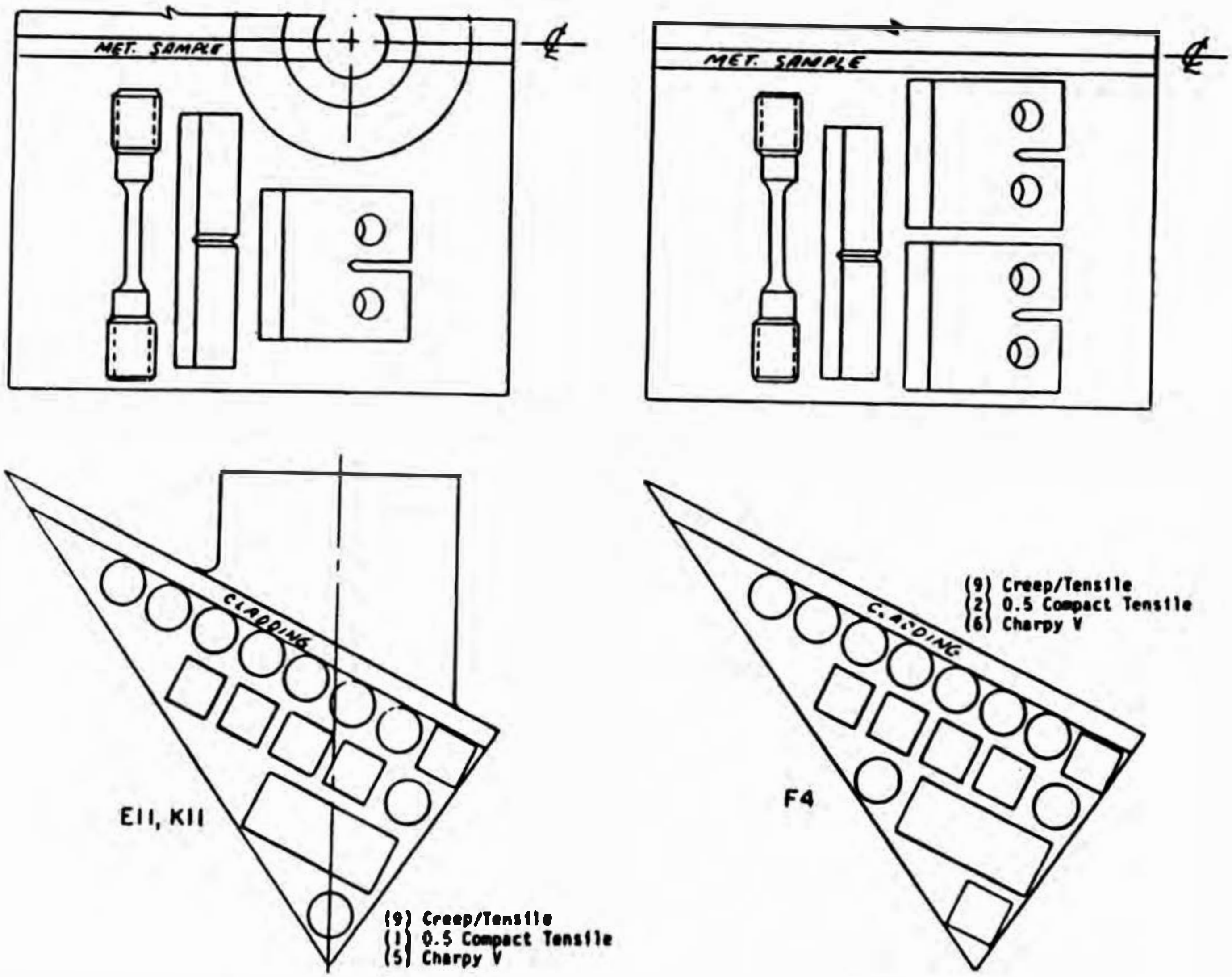


Figure 17. "Hillside" TMI-2 lower-head samples (Type 2).

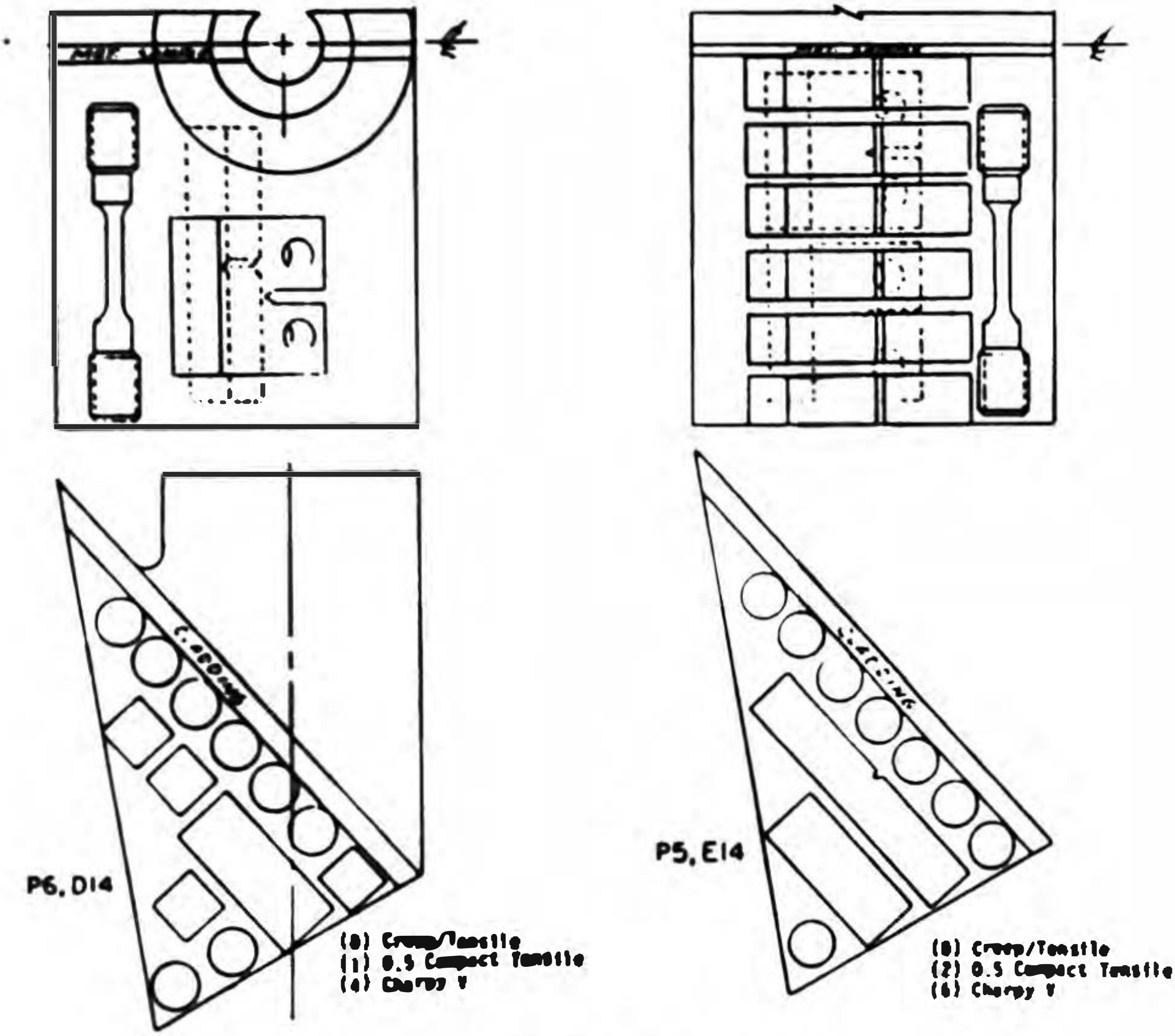


Figure 18. "Steep-hillside" TMI-2 lower-head samples (Type 3).

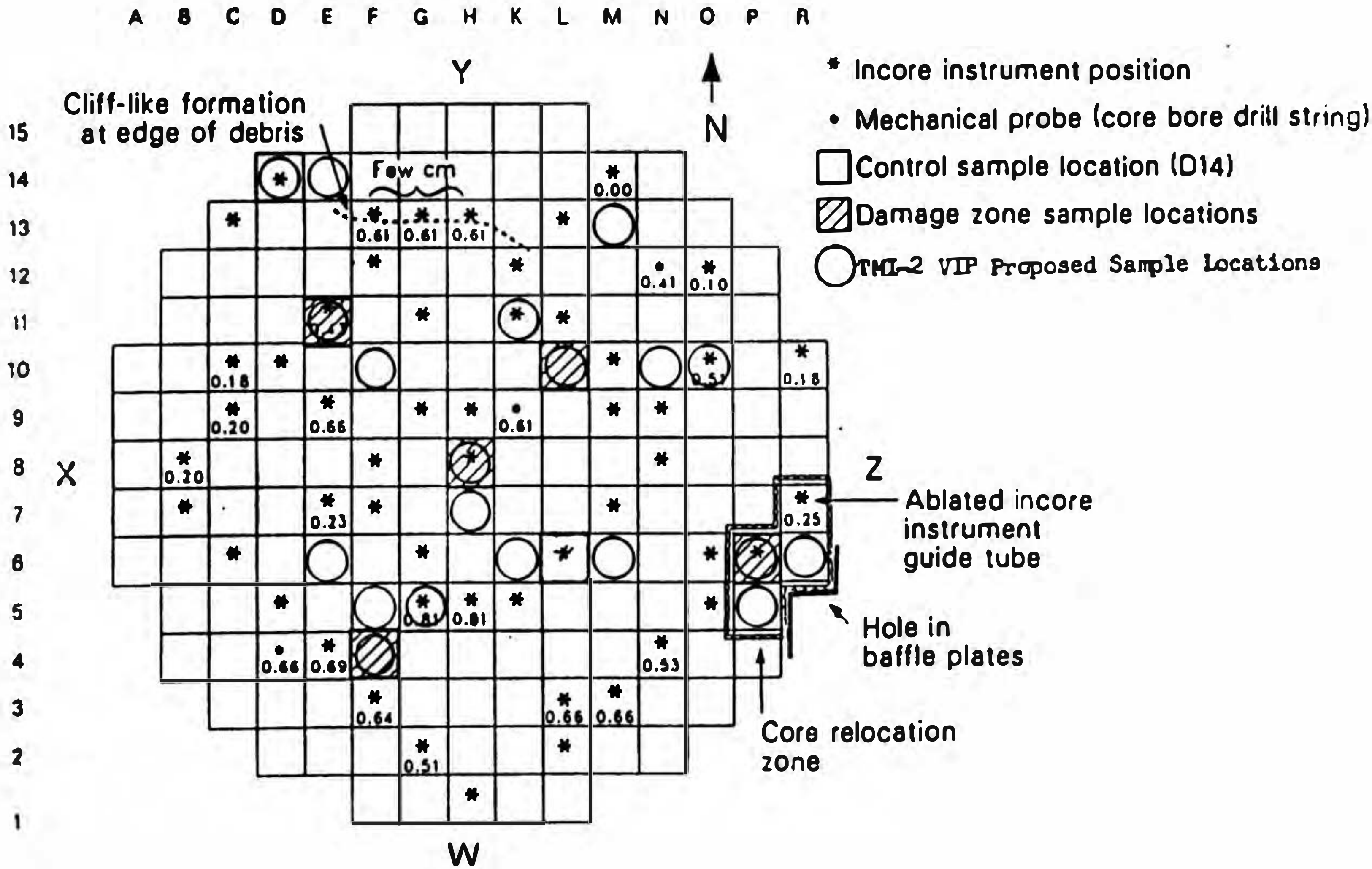


Figure 19. Proposed TMI-2 lower-head sample locations.

bypass region). A prioritized list of proposed lower head samples that corresponds to the locations shown in Figure 19 is given in Table 3, together with the rationale for the selection. The actual locations of the sample extraction will be determined after the debris is removed from the lower head region and the lower head is observed, but the proposed list of prioritized samples will be used as a guide in selecting final sample extraction locations. A time window of 30 days will be allowed for sample extraction and it is believed that a minimum of eight samples can be obtained during this window.

At the present time it is believed to be necessary to obtain samples: (1) as close as possible to the area of impact on the lower head of the primary jet of molten core material; (2) toward the radial center of the lower head underneath the maximum thickness of debris; (3) in the quadrant of the lower head where a "wall" of consolidated debris similar to a lava front has been observed; (4) in a location of the lower head not contacted by the molten core material to act as a "control" sample; (5) from locations including one or more instrument penetrations especially in the areas noted in items 1 through 4; and, (6) from other locations as may appear desirable once the debris is removed from the lower head.

The sample from item 1 above will be important to ascertain the depth of ablation (if any) from the impact of the jet of molten core material, and the extent of interaction with the stainless steel liner and carbon steel. The corollary instrument penetration sample will include one observed to be melted off. This sample will be important to determine the extent of weld failure in such a region, and the condition of the adjacent instrument penetration tube.

Samples at the radial center of the vessel lower head (item 2 above) will be used to investigate the damage to the head and instrument penetration under the thickest layer of debris accumulated on the lower head and where there may be a layer of non-fuel material composed of control and structural material.

TABLE 3. TMI-2 VESSEL INTEGRITY PROGRAM PROPOSED LOWER HEAD PRIORITIZED SAMPLE LOCATION AND RATIONALE

Sample Location	Type	Rationale
P6	Nozzle	In path of downflow jet: Ablation
D14	Nozzle	Reference nozzle; no debris
P5	Base	Pair with P6
E14	Base	Pair with D14
H8	Nozzle	Central region
E11	Nozzle	Consolidated debris; longer cooling? Short T/C junction at head
G5	Nozzle	Deep debris, sport T/C junction
K11	Nozzle	Consolidated debris, longer cooling?
H7	Base	Pair with H8
F10	Base	Pair with E11
F4	Base	Pair with G5
L10	Base	Pair with K11
F5	Base	Short T/C junction
M6	Base	Migration path
K6	Base	Migration path
E6	Base	Short T/C junction
O10	Nozzle	Nozzle submerged in melted core material
N10	Base	Pair with O10
R6	Base	Adjacent to nozzle damage
M13	Base	Edge of "lava" wall

Samples from item 3 above will be used to investigate the damage to the lower head in the area where a wall of consolidated debris has been observed and where vessel temperatures could have reached 1100 K. Temperature measurements obtained after the accident had progressed for some hours from relocated in-core thermocouple junctions reached 1600 K in some areas. This could be an indication--yet to be confirmed from metallurgical analyses--of thermal threat to the lower head.

The sample from item 4 is necessary to act as a "control" sample, where the excessive temperatures of the molten core material should not have had a direct effect on the lower vessel steel.

The metal-disintegration-machine (MDM) is expected to be able to cut through the layer of non-fuel material that may be resting on the lower head if that material does not contain any appreciable ceramic material. However, if some ceramic material is contained in the layer of material adjacent to the lower head, that material must be removed prior to cutting by MDM techniques. The INEL has the responsibility to recommend to MPR Associates (who has the responsibility of obtaining the vessel lower head samples) preferred methods of removing the ceramic or ceramic/metallic material prior to lower head sample extraction. GPU Nuclear must of course review and determine if any sample extraction technique can, in fact, be used in the TMI-2 plant.

The actual material adjacent to the lower head is a vital data source and must be preserved during sample acquisition. The material may be fused to the lower head and may contain thermochemical reaction products. If the material is not fused to the lower head, samples of material adjacent to the head will be obtained as part of the defueling process by GPU-Nuclear and this acquisition is discussed in Section 6.2. If the material is fused to the lower head, and it does contain ceramics (i.e. the MDM cutting techniques will not work), then the INEL recommends that an Abrasive Hydrojet be used to cut through the fused layer prior to sample cutting by the MDM

The Abrasive Hydrojet uses a high-pressure water stream with entrained abrasive directed through a nozzle. The Abrasive Hydrojet has been shown to cut virtually anything, including solid metal, massive ceramics, and composites, with speed and precision. Low reaction forces, no in-vessel moving parts (excluding positioning), and limited in-vessel equipment requirements make the hydrojet an attractive choice. Also, there is a pump unit already located at TMI-2, although it should have its final stage upgraded for higher delivery pressure. For this application, steel shot abrasive is recommended in place of the garnet abrasive used in early testing. The cutting efficiency of the steel shot is adequate, delivery to the nozzle is easier, and steel shot can be more easily captured and removed from the vessel. The disadvantages of the Abrasive Hydrojet are that it requires high-pressure plumbing between the pumps and the nozzle; it introduces abrasive to the reactor vessel that must be removed from the vessel; the equipment, including the pumps, is not containment ready; and, the nozzle and abrasive feed equipment are not yet located at TMI-2.

6.1.2 Lower Head Sample Shipment

MPR is responsible for the shipment of the reactor vessel lower head samples from TMI-2 to ANL-East, and ANL-East, is responsible for shipment of the samples from ANL-East to the OECD partners and to the INEL. ANL-East has the responsibility to: receive all vessel lower head steel samples; verify that the location, size, orientation, etc., of the samples that will have been documented by MPR is consistent and reasonable; decontaminate the appropriate specimens as necessary for unrestricted handling; section the samples into metallographic and metallurgical specimens or blanks; machine into specimens as required; and distribute the materials to the cooperating partner laboratories. Since the lower head samples will be decontaminated by ANL-East prior to shipment, the shipment from ANL-East will be relatively easy and inexpensive. Conversely, shipment of the contaminated lower head samples (which may have fuel-bearing material attached to them) from TMI-2 to ANL-East will be neither easy nor inexpensive. A possible method for shipment of the contaminated samples would be to use the CNS 1-13C II cask owned by DOE. A large container was designed and fabricated by the INEL for the CNS 1-13C II cask that could be used for the shipment of the contaminated lower head samples.

The large shielded container that was designed and fabricated for the CNS 1-13C II cask is illustrated in Figure 20. A bucket, which was designed to be used to remove the material to be shipped from the reactor, fits inside the shielded container. The bucket, illustrated in Figure 21, can only be used for non-fuel bearing material at the present time but it may be possible to obtain approval from the NRC to ship the reactor vessel lower head samples in this bucket if the shielded container carrier (illustrated in Figure 22) is modified so that it can be sealed. A fuel-bearing-material bucket (illustrated in Figure 23) was also designed and fabricated for use with the shielded container, but the license for the CNS 1-13C II does not presently allow its use. A view of the fitup of fuel-bearing-material bucket, the shielded container, the shielded container carrier, the CNS 1-13C II cask, and the upper and lower cask impact limiters is shown in Figure 24.

6.1.3 Archive Material Program

ANL-East has the responsibility for the Archive Material Program. The program has two primary objectives. One is to provide a set of standards for comparison with the samples removed from TMI-2 to increase the accuracy of the estimates of the thermal history of the vessel; the other is to provide enough material to determine the mechanical properties of the vessel lower head during the accident. An adequate supply of 0.127-m-thick A533-B steel (UNS K12539) will be obtained. An attempt will be made to locate archival material from the original heat used for the head. If such material does not exist, an archive plate of similar chemistry, fabrication, and heat-treatment history will be procurred.

The archive plate will first be subjected to the same fabrication heat treatment as that given to the TMI-2 lower head prior to service. Through wall variations in microstructure will be determined, and "reference" specimens will be prepared by dividing full thickness material into sections of reasonably homogeneous microstructure. These reference

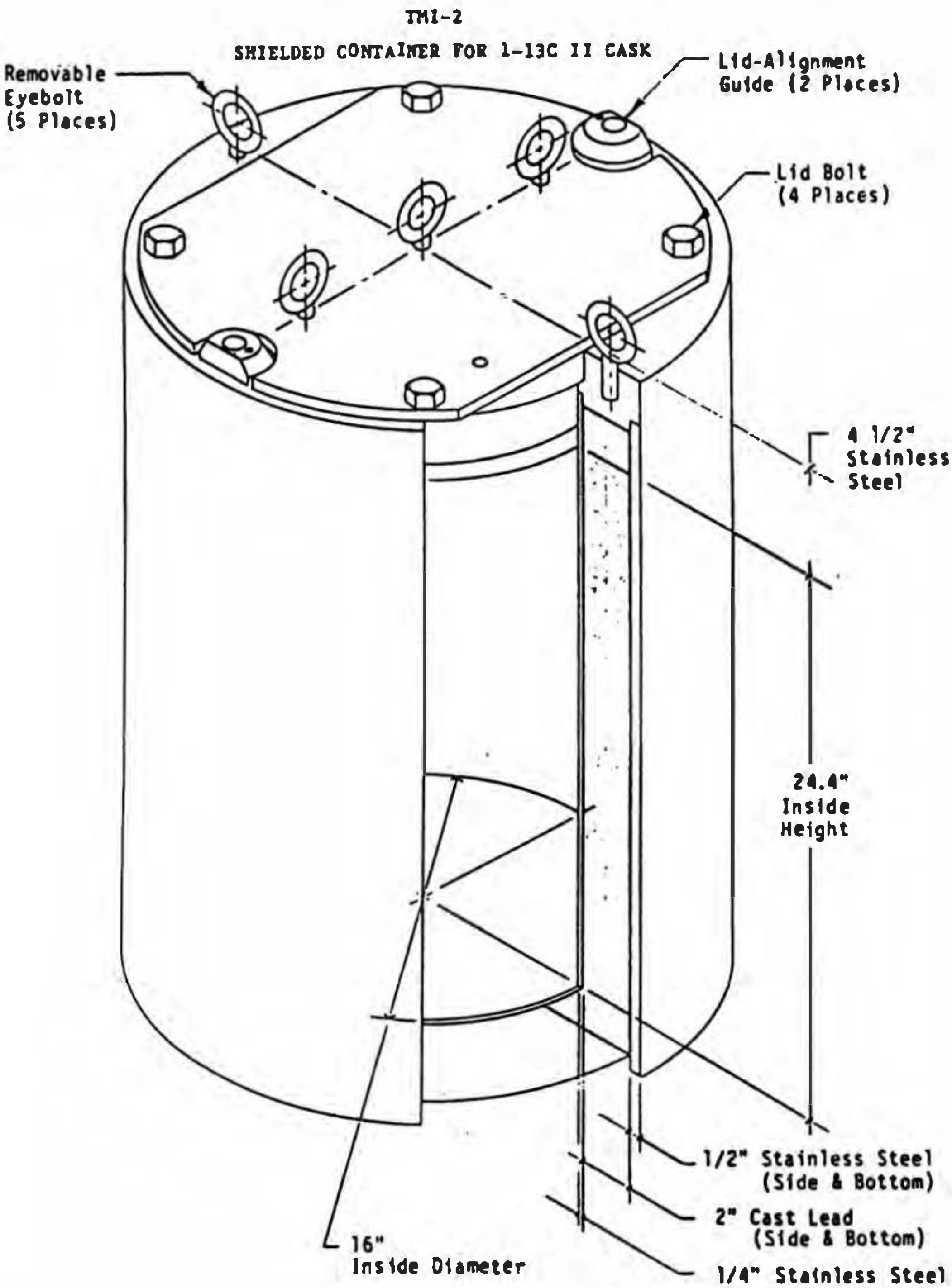


Figure 20. Large, shielded container for CNS 1-13C II cask.

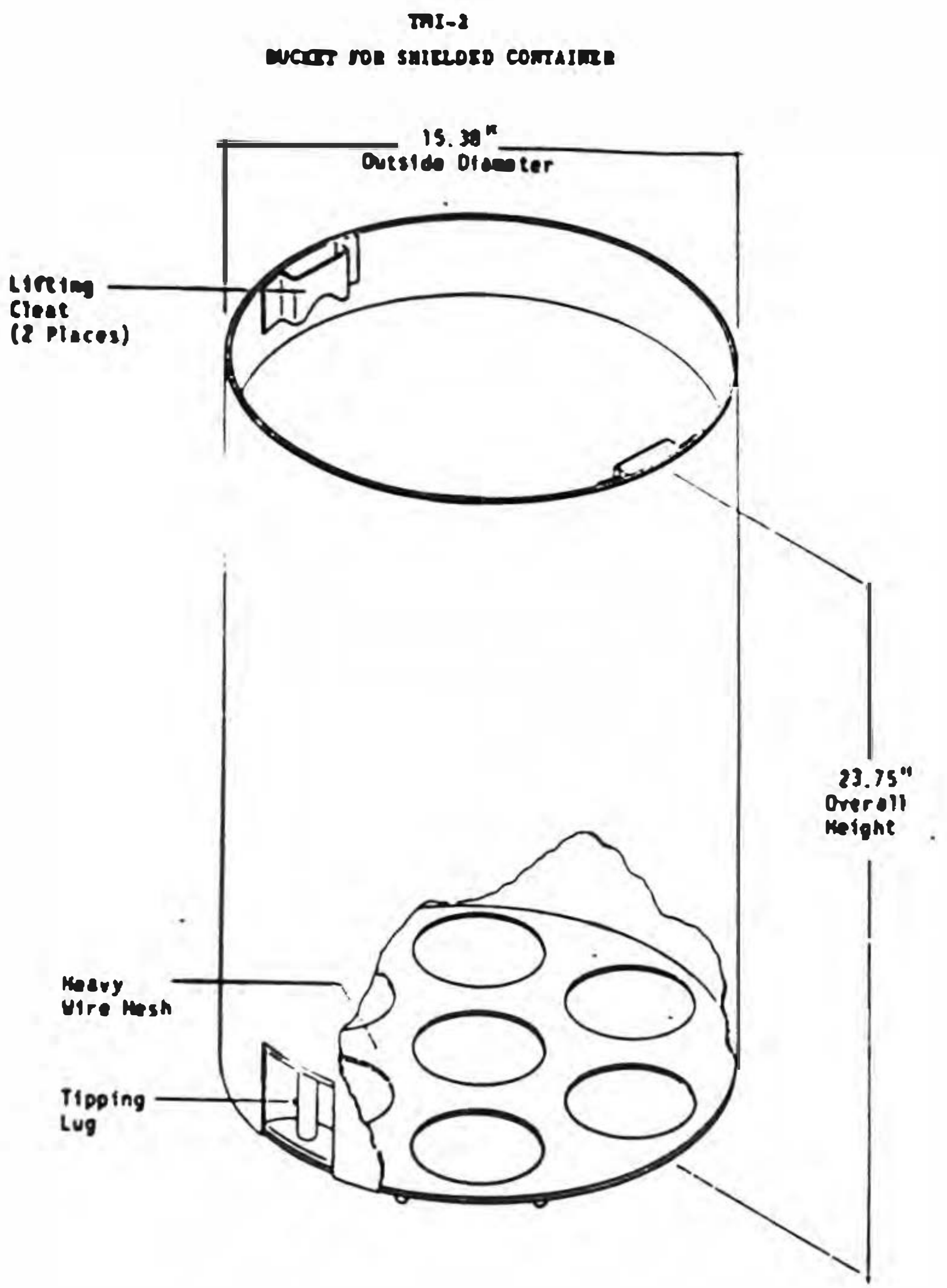


Figure 21. Bucket for use with shielded container and CNS 1-13C II cask.

TMI-2
SHIELDED CONTAINER CARRIER

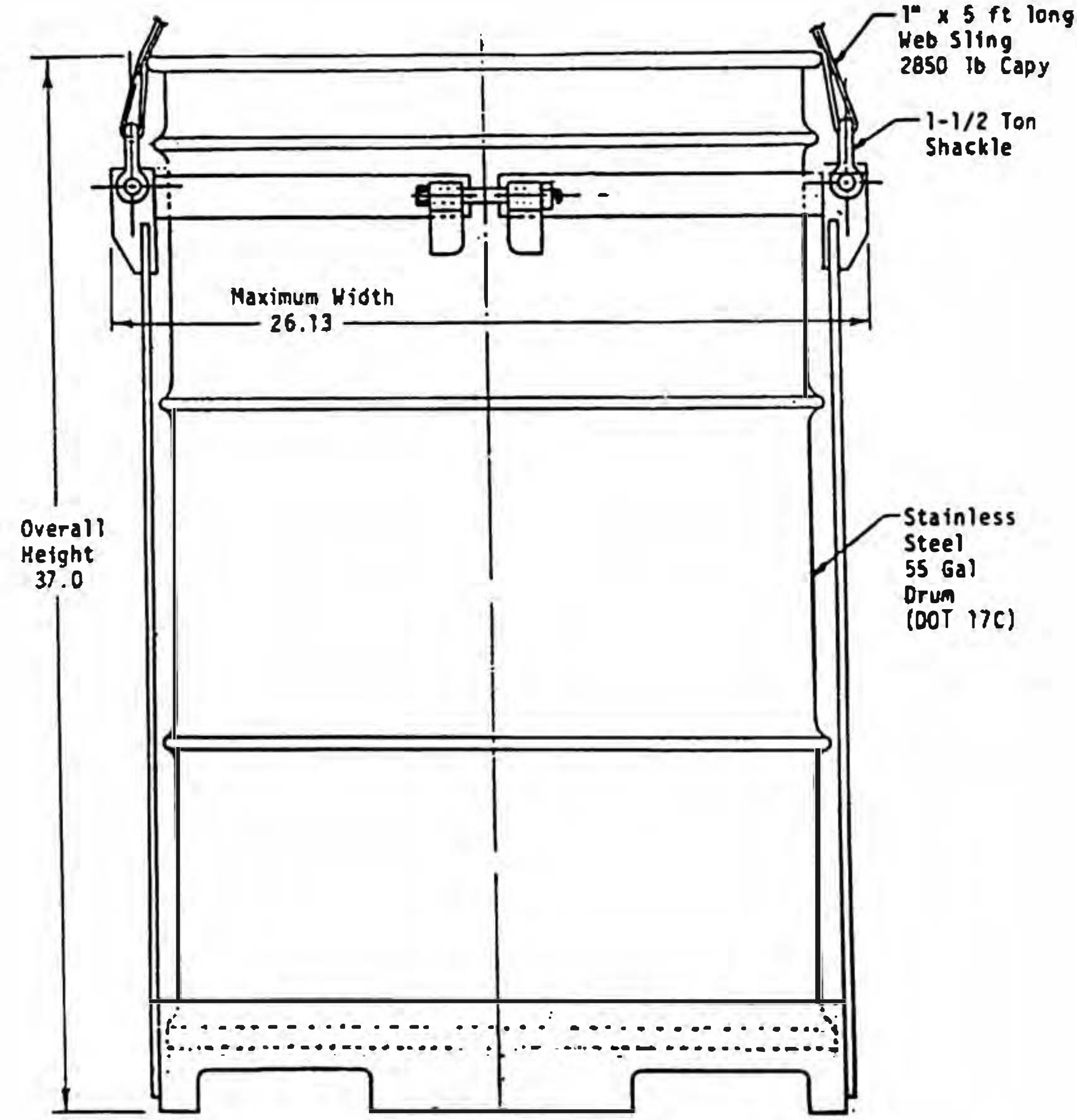


Figure 22. Shielded container carrier for use with CNS 1-13C II Cask.

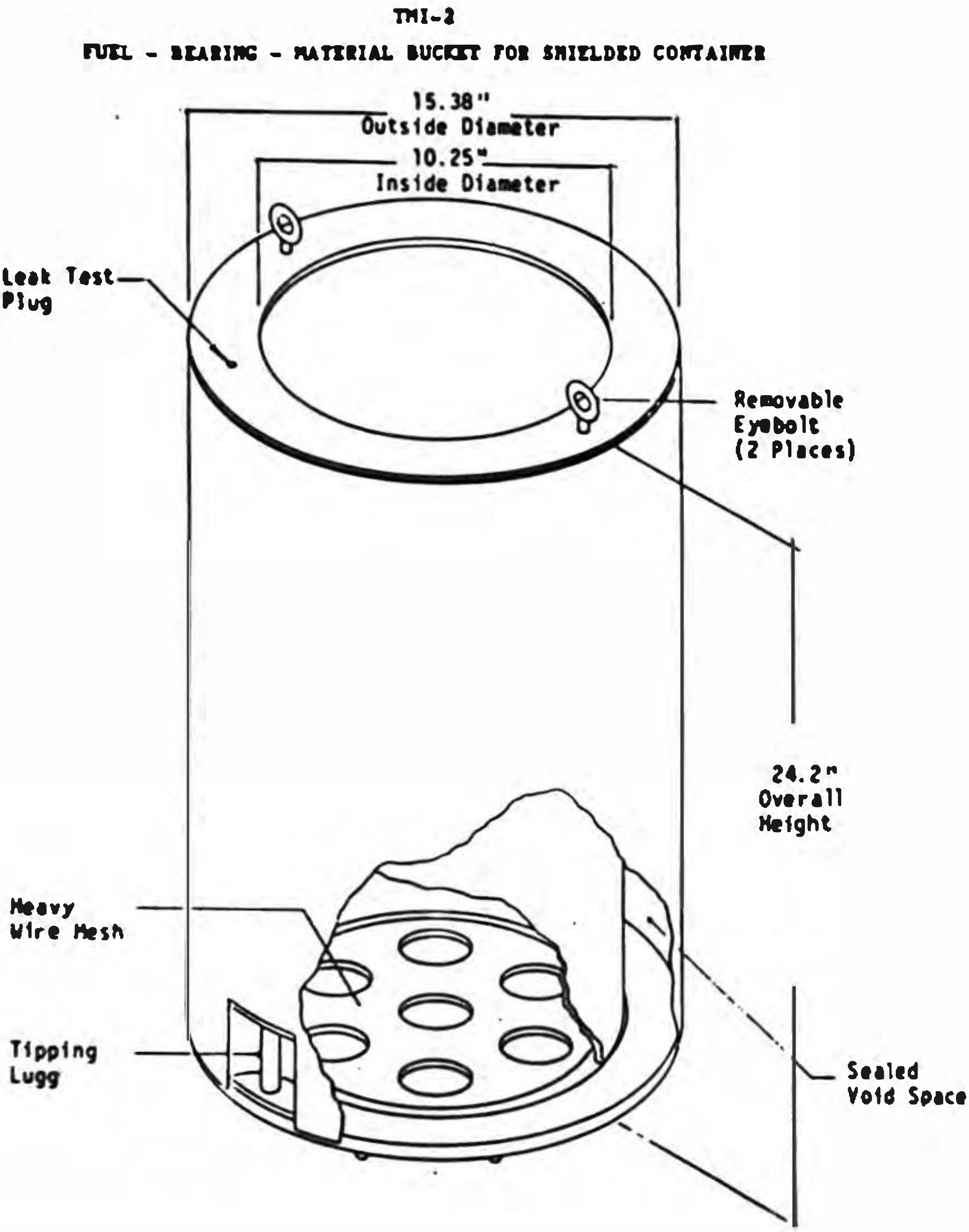


Figure 23. Fuel-bearing-material-bucket for use with CNS 1-13C II.

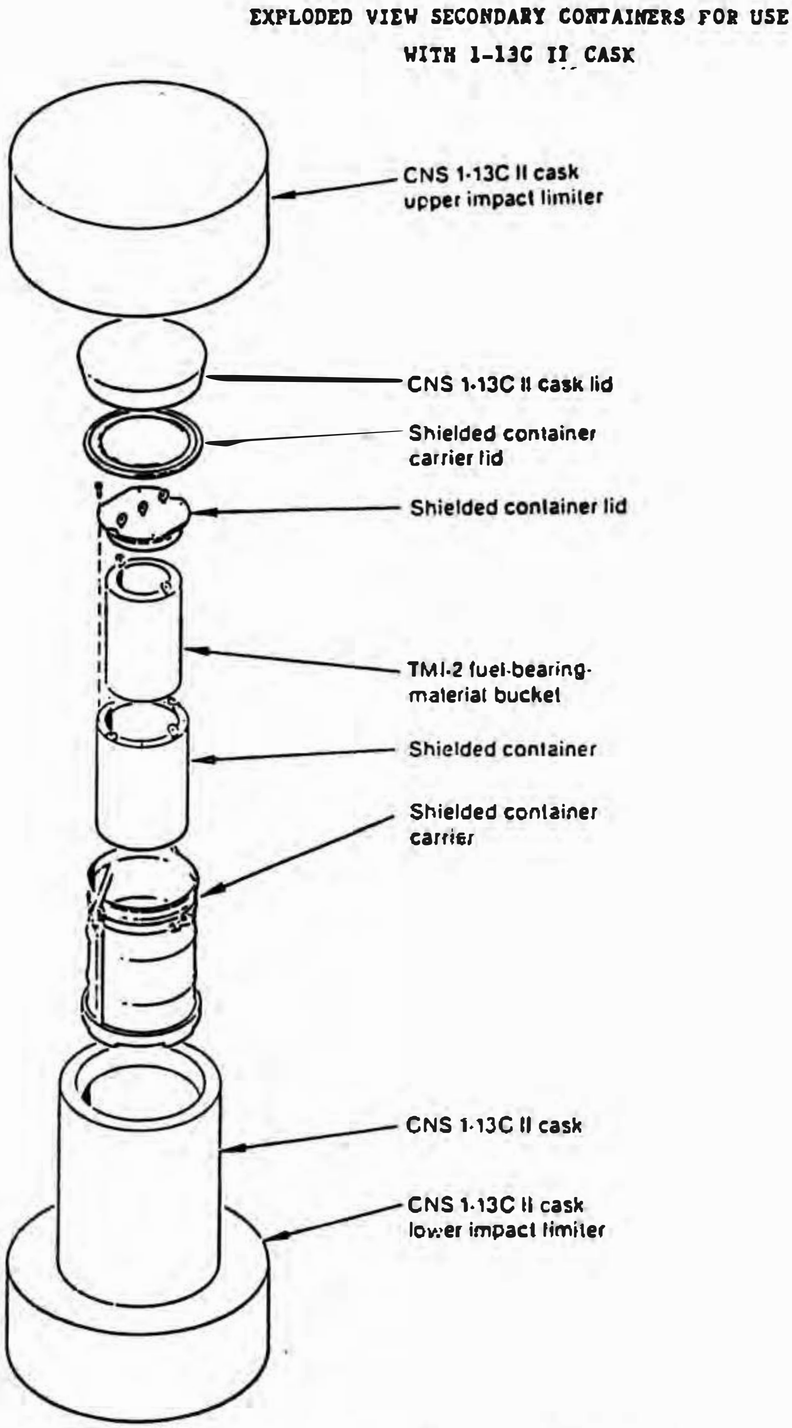


Figure 24. Fitup of fuel-bearing-material bucket, shielded container, shielded container carrier, CNS 1-13C II cask, and cask impact limiters.

specimens would then be subjected to a range of heat treatments corresponding to those postulated for the lower vessel head during the TMI-2 accident. The microstructural characteristics and hardness of these materials would then be determined and compared with those of actual samples from the vessel lower head. The results obtained will be discussed with the NRC, OECD partners, and the INEL, and the most probable thermal history from the range of those determined by previous analysis will be identified (or at least a narrower range of probable temperature histories will be identified).

The remainder of the archive material will be used to fabricate mechanical property specimens that will be tested by the OECD partners and the INEL. These laboratories are being surveyed to determine their interests and capabilities. An overall mechanical property test matrix will be developed by ANL-East and work will be apportioned to various participating laboratories so as to avoid unnecessary duplication of effort, but with sufficient cross checks to determine laboratory-to-laboratory variability.

6.1.4 Mechanical Property Characterization

ANL-East has the responsibility for the characterization of the vessel lower head steel mechanical properties. The objective of this work is to determine the mechanical properties of the vessel lower head under the conditions of the core-melt accident. These properties will be used to assess the integrity of the TMI-2 vessel and the margin-to-failure during the accident.

To accomplish this objective a series of mechanical tests will be initiated on test specimens taken from the TMI-2 lower head materials as well as on archive material that simulates the TMI-2 material. These mechanical tests will emphasize short-term stress-rupture properties, since, as noted previously, this failure mode is believed to have been the greatest threat to the integrity of the TMI-2 lower head during the accident. However, tensile, Charpy V-notch impact, fracture toughness, and hardness tests will also be conducted.

Where possible, test specimens will be obtained from both the inner surface as well as the interior thickness of the lower head for each of the samples obtained. Corresponding tests will be performed on specimens from different depths in the archive material. It is expected that the temperature ranges over which the properties must be determined will be a function of distance from the inner surface. ASTM test standards will be followed wherever possible.

The mechanical tests will be performed both at ANL and at cooperating partner laboratories. ANL-East will remove non-pressure vessel material from the TMI-2 samples (adherent, non-pressure vessel materials will undergo metallurgical examination at both ANL-East and the INEL--see Section 6.2.3), section them into blanks, machine the test specimens, and distribute these specimens to the cooperating partner laboratories for duplicate and complementary testing. The exact tests to be performed by the various laboratories will be determined on the basis of the response to the technical capabilities and interest questionnaire that has been distributed. ANL-East has the responsibility to: integrate all mechanical property and metallographic studies performed by the OECD partners; coordinate with the vessel integrity program at the INEL and exchange data with the INEL; and participate in the preparation of a report to describe the material condition of the TMI-2 vessel lower head steel, determine the probable temperature history of the steel, and assess the integrity of the lower head during the accident.

6.1.5 Metallurgical Characterization

The INEL has the responsibility for determining the metallographic characteristics of the vessel lower head steel. Metallographic studies will include full specimen profile polishing, plus more detailed studies of melt interaction zones, weld joints, weldclad-base metal interface, through-thickness microstructure (especially if a gradient is observed) and melt progression into the instrument nozzle and its penetration through the vessel wall with microhardness measurements in each region. An important aspect of this sample investigation will be to determine if the non-fuel materials have interacted in any way to create low-melting point eutectics that could more easily generate paths for containment penetrations. Metallographic examinations of the lower head steel specimens will include optical, scanning electron, and transmission electron microscopy techniques and other physical examination techniques as appropriate. It is expected that the OECD partners will perform metallographic examinations and physical property determinations of the specimens that they receive based on their interest and capabilities.

The INEL has the responsibility to accumulate, analyze and integrate all cooperating partner results and analyses pertaining to properties, temperature and integrity of the TMI-2 lower head steel and will coordinate this with the TMI-2 project at ANL-East, and exchange all data with ANL-East. Microstructure examination results of the lower head material will be compared with the archive samples, which will have a known time-temperature history, in an effort to determine the temperature transient experienced by the TMI-2 vessel lower head. An INEL internal report will be prepared describing the results of the metallographic examinations and the estimates of the lower head temperatures reached during the accident. The INEL will take the lead in preparing the final report on the reactor vessel lower head examinations and will collaborate with ANL-East as full partners in the preparation and review of this final report. The final report will describe the material condition of the reactor vessel lower head steel, its probable temperature history, and its creep and plastic deformation during the accident.

6.2 Lower Head Companion Samples

The temperature distribution in the TMI-2 reactor vessel lower head during the accident is strongly dependent on the composition and properties of the previously molten debris that relocated to the lower head region as was demonstrated by the results of the temperature calculations presented in Section 4.4. The calculations clearly indicate that the composition and properties of the relocated material adjacent to the reactor vessel lower head must be determined before a reliable analysis of the potential creep rupture and margin-to-failure of the lower head can be performed.

The objectives of the lower head companion sample examination and analysis portion of the NRC/OECD Vessel Integrity Program are to: (a) develop a plan (this document) for the selection, separation, transport, examination, and analysis of materials adjacent to or near the proposed metallurgical specimens in the TMI-2 reactor vessel lower head examination program; (b) perform the selection, acquisition, transport, examination, and analysis according to the plan after it is approved by NRC; and, (c) determine the most likely temperature distribution that occurred in the lower head and, using this temperature distribution, perform calculations of the resulting lower head stress distribution in order to assess the margin-to-failure of the TMI-2 lower head. Thus, this work will characterize the nature of the attack by the transported reactor core and structural materials on the integrity of the lower head of the TMI-2 pressure vessel. This information will support the development of lower head damage and accident management models to predict the margin-to-failure of the reactor vessel lower head and the consequences of possible recovery options. The INEL is responsible for the NRC/OECD lower head companion sample examination and analysis portion of the Vessel Integrity Program.

6.2.1 Companion Sample Acquisition

GPU-Nuclear will help select and will acquire the companion samples from regions near where the reactor vessel lower head samples will be extracted (see Section 6.1.1, Figure 19 and Table 3), provided that the material is not adhered to the vessel lower head so tightly that it cannot be broken loose. If the material is tightly adhered to the vessel lower head it will be obtained with the lower head sample and will be extracted at ANL-East, examined by ANL-East, and subsequently shipped to the INEL. Various types of tools are available at GPU-Nuclear for the removal of loosely bound materials, and the tool(s) to be used for companion sample extraction will be chosen from this inventory based on GPU-Nuclear's experience. The INEL has personnel located at the Three Mile Island Nuclear Station, who will be observing the sample acquisition as it takes place.

6.2.2 Companion Sample Shipment

The TMI-2 reactor vessel lower head companion samples will be shipped in 2R containers (16 of which are on hand at the INEL and TMI-2) that are carried by the CNS 1-13C II cask. This method of shipment of core debris samples has been successfully used many times for the DOE sample acquisition and examination program. An illustration of the 2R container is shown in Figure 25. Shipment of companion samples to the OECD partners is possible but is not funded in the present Vessel Integrity Program. Shipment of fuel debris samples to Europe has been accomplished in the TMI-2 program but is expensive and time consuming. However, if OECD partners have a special interest in companion samples and will pay for shipment of the samples, the INEL will coordinate the shipments.

6.2.3 Companion Sample Examination

Examinations of the vessel lower head companion samples will be performed to determine the composition, maximum temperature, and extent of interaction both within the samples and between the samples and the lower head steel if an interaction zone is found. The composition of the samples is important for several reasons, but the most important reason with respect to vessel lower head integrity is for the determination of the conductivity of the material. The composition of the material will establish whether it insulated the lower head from the heat source in the fuel material, as would be expected from a loose ceramic debris, or whether it provided a conduction path, as would be expected from a mixture of structural and control materials. Either of these two examples, and any combination of the two, can reasonably be expected based on findings to date in the core region, the core bypass region, and the upper part of the lower plenum (see Section 2). The lower crust of the central molten region of the core consisted of mostly metallic structural and control material frozen between standing fuel rods. If this type of material made its way

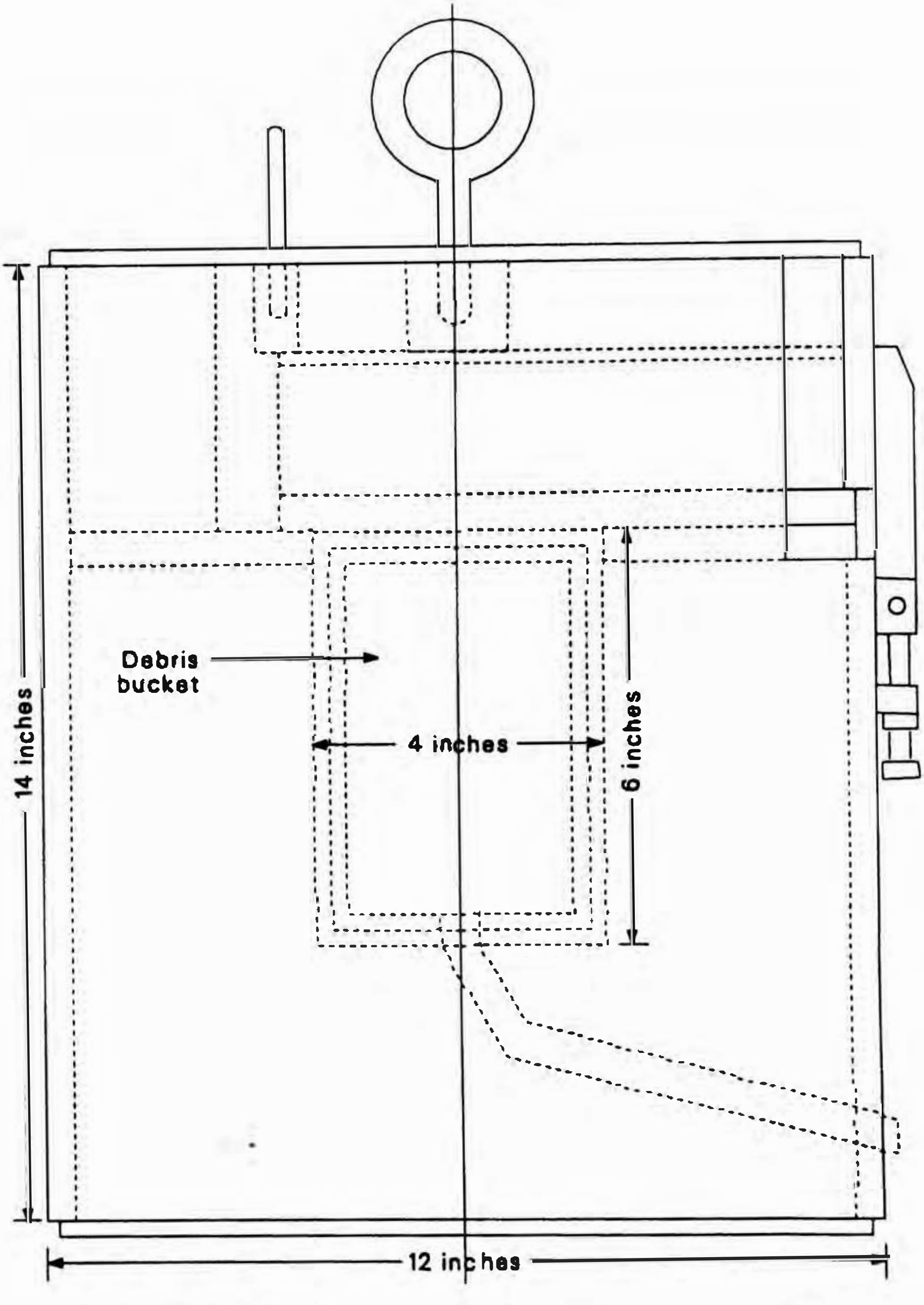


Figure 25. 2R container used for TMI-2 core debris shipment in CNS 1 13C II cask.

to the lower head first and froze, then a high conduction path from molten debris to the lower head would be expected. The material examined thus far from the lower plenum upper regions was loose ceramic debris which would insulate the lower head. Finally, material in the previously molten central region of the core was a mixture of ceramic and metallic materials.

The companion samples will be shipped to the INEL (either from ANL-East or from TMI-2) and received at the TRA hot cells. After the samples are removed from the 2R containers, which will have been shipped in the CNS 1-13C II cask, INEL will verify that the location, size, orientation, etc., that will have been recorded by either GPU-Nuclear or ANL-East, is as was specified. The samples will then be balance weighed, both black-and-white and color photographed, and their density determined by immersion techniques. The samples will then be cut into specimens and labeled. Specimens will be divided into archive, metallurgical, and bulk chemical composition specimens. The archive specimens will be stored at the INEL. Metallurgical specimens will be polished and mounts prepared for optical metallography and SEM/WDX examinations. The SEM/WDX exams will be performed at the TAN hot cell annex at the INEL. Chemical specimens will be dissolved and examined for elemental composition (17 elements), using inductively coupled plasma spectroscopy.

The thermal conductivity of the companion samples must be determined to perform more meaningful heatup calculations for the lower head. The INEL does not have capabilities in place for making the thermal diffusivity measurements required for determining thermal conductivity. This work should therefore be performed by another laboratory, where these capabilities and expertise exist. The capability for measuring thermal diffusivity exists at the Windscale Laboratory in the United Kingdom and it is probably worth the cost (to the NRC/OECD Vessel Integrity Program) of shipment of companion specimens to the UK for examination. The laboratory at Windscale is presently examining TMI-2 samples from other core positions and plans to measure the thermal diffusivity of those samples. Pacific Northwest Laboratories (PNL) has made thermal diffusivity measurements on unirradiated fuel specimens[31] and might also be able to make the measurements needed on the irradiated companion samples.

6.2.4 Analysis of Lower Head Response to Temperature Transient

The purpose of the lower head analysis is to determine the creep stress and the resulting plastic deformation in the steel that resulted from the temperature transient caused by the molten material relocation 224 min after the accident was initiated. The purpose was also to calculate the temperature and stress distribution required to cause creep rupture of the lower head and thereby establish the margin-to failure of the TMI-2 reactor pressure vessel. The calculational methods that will be used are the same as those described in Section 4.4. The time dependent temperature distribution in the lower head will be calculated using the COUPLE/FLUID computer code with the measured thermal conductivity of the lower head companion samples (described in Section 6.2.3 above) and the actual geometry of the lower head debris determined from defueling used as input. The results of the COUPLE/FLUID calculations will be compared with the lower head temperature distributions determined from examination of the vessel lower head samples and the time-temperature measurements made with the lower head archive material (see Sections 6.1.3, 6.1.4, and 6.1.5). The best estimate of the time-dependent temperature distribution will then be determined based on these comparisons, engineering judgement, and possibly some further calculations.

The time-dependent stresses in the lower head will then be calculated using the ABAQUS computer code and lower head model described in Section 4.4, with the best estimate time-dependent temperature distribution described above used as input. Based on the calculational results presented in Section 4.4, it is clear that the debris bed resting on the lower head must have quenched; otherwise, the lower head would have reached its ultimate strength because the temperature would have probably exceeded 1140 K. Thus, the quench behavior of the debris bed resting on the lower head must also be input to the ABAQUS code calculations. Finally. the temperature distribution needed to cause creep rupture of the lower head will be determined using the ABAQUS code, resulting in a determination of the margin-to-failure.

A final report incorporating the results of the lower head sample examinations, the lower head companion sample examinations, the creep rupture analysis of the lower head and the analysis of the margin-to-failure of the lower head will be prepared by the INEL.

6.2.5 Program Integration

The INEL has the responsibility for the integration of the NRC/OECD Vessel Integrity Program and the DOE Accident Evaluation Program to ensure that all of the data obtained from examination and analysis of the TMI-2 debris is incorporated in the final analysis of the TMI-2 accident and subsequently in the final accident scenario. The program integration is described in Section 5.

The integrated NRC/OECD and DOE programs for the completion of the TMI-2 accident evaluation will provide a coordinated program that emphasizes: (a) acquisition and examination of samples to determine damage to the lower head; (b) assessment of the lower head margin-to-failure; (c) development of an understanding of the pathways and mechanisms that controlled transport of molten materials to the reactor vessel lower plenum; (d) integration of this information into the final accident scenario; (e) comparison of these data with the results of the OECD analysis exercise (standard problem exercise); and, (f) application of the understanding of the TMI-2 accident to analytical tools used for accident mitigation and management. The integrated program should improve the completeness of both the DOE Accident Evaluation Program and the NRC/OECD Vessel Integrity Program and develop the most complete and accurate understanding of the TMI-2 accident that can be obtained with the available resources.

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APPENDIX A

TMI-2 REACTOR PRESSURE VESSEL INTERNALS

APPENDIX A

TMI-2 REACTOR PRESSURE VESSEL INTERNALS

This Appendix contains sketches and descriptions of some of the internals of the TMI-2 reactor pressure vessel. Figure A-1 is a schematic of the reactor pressure vessel situated in the reactor building and illustrates the location of the RPV with respect to the locations of other major components of the reactor. Figure A-2 shows the locations within the reactor pressure vessel of the major structural components. Figure A-3 shows the routing of a typical in-core instrument assembly from the seal table outside the reactor pressure vessel to the internals of the core. Figure A-4 shows the TMI-2 support assembly configuration in the lower plenum region, and Figure A-5 gives dimensions of the various components of the support assembly. The instrument penetration tubes serve as entry ports through the reactor pressure vessel lower head for neutron flux monitors, thermocouples, and other in-core instrumentation. Figures A-6 and A-7 illustrate the geometry of the instrument penetrations. The instrument assembly penetrates the reactor vessel through an Inconel penetration nozzle, about 12 in. long above the inside surface of the lower head. The penetration nozzle is sleeve-fitted into a stainless-steel guide tube (see Figure A-8), which guides the instrument assembly through the lower plenum structures (i.e., flow distributor plate, core support plate, lower grid forging, etc.) into the core region.[A-1 through A-3]

The dimensions of the Inconel nozzle are illustrated in Figure A-6, while Figure A-7 presents the geometric details of the stainless-steel guide tubes. Of importance to the lower head Vessel Integrity Program are the material and dimensional characteristics of the weld joint at the attachment junction between the Inconel nozzle and the bottom head. The weld joint designation as reported in the Final Safety Analysis report (Reference A-1) is

Weld for instrument nozzle to lower head: SB-195 E-Ni-Cr-Fe-3 (INCO 182T).

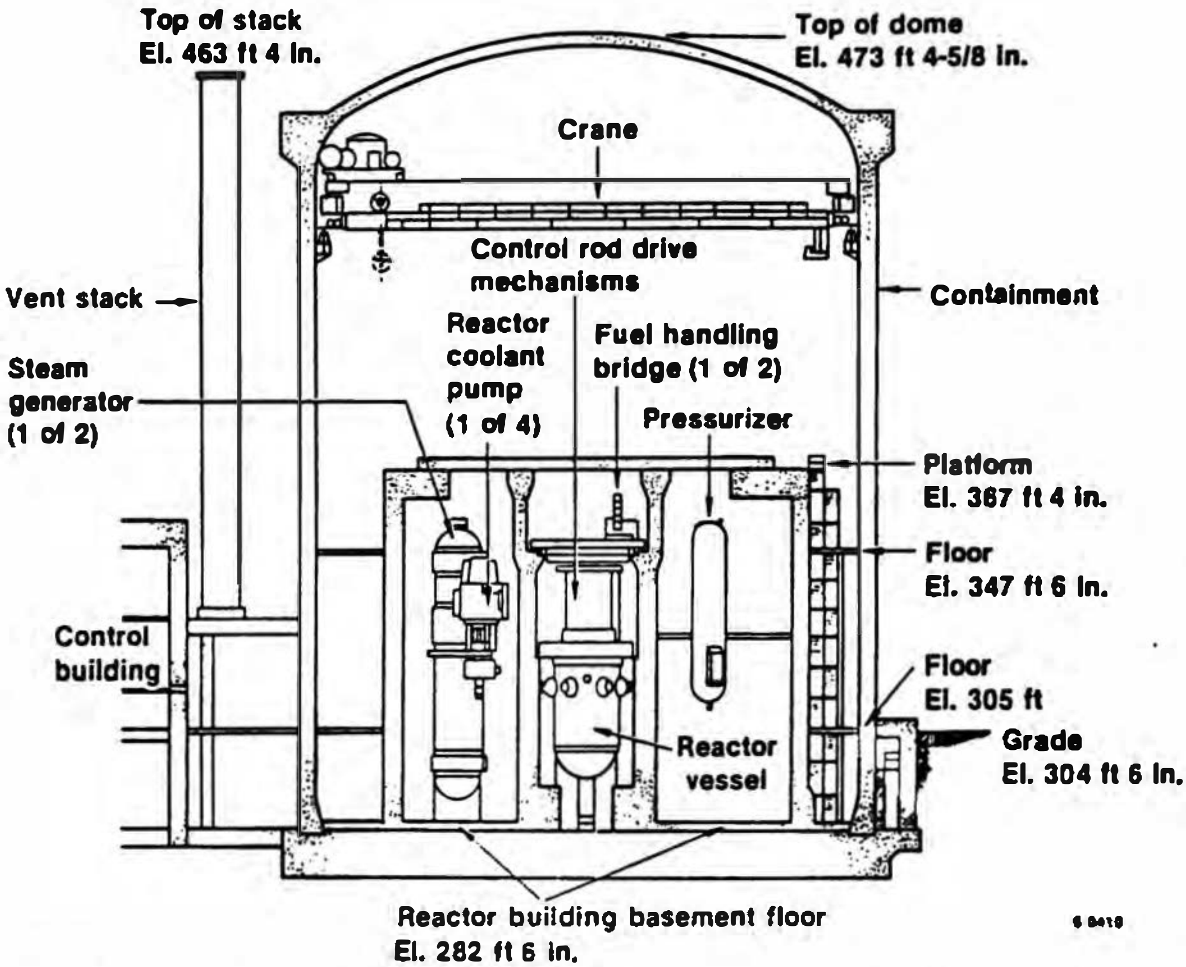


Figure A-1. TMI-2 reactor building and major components of primary cooling system.

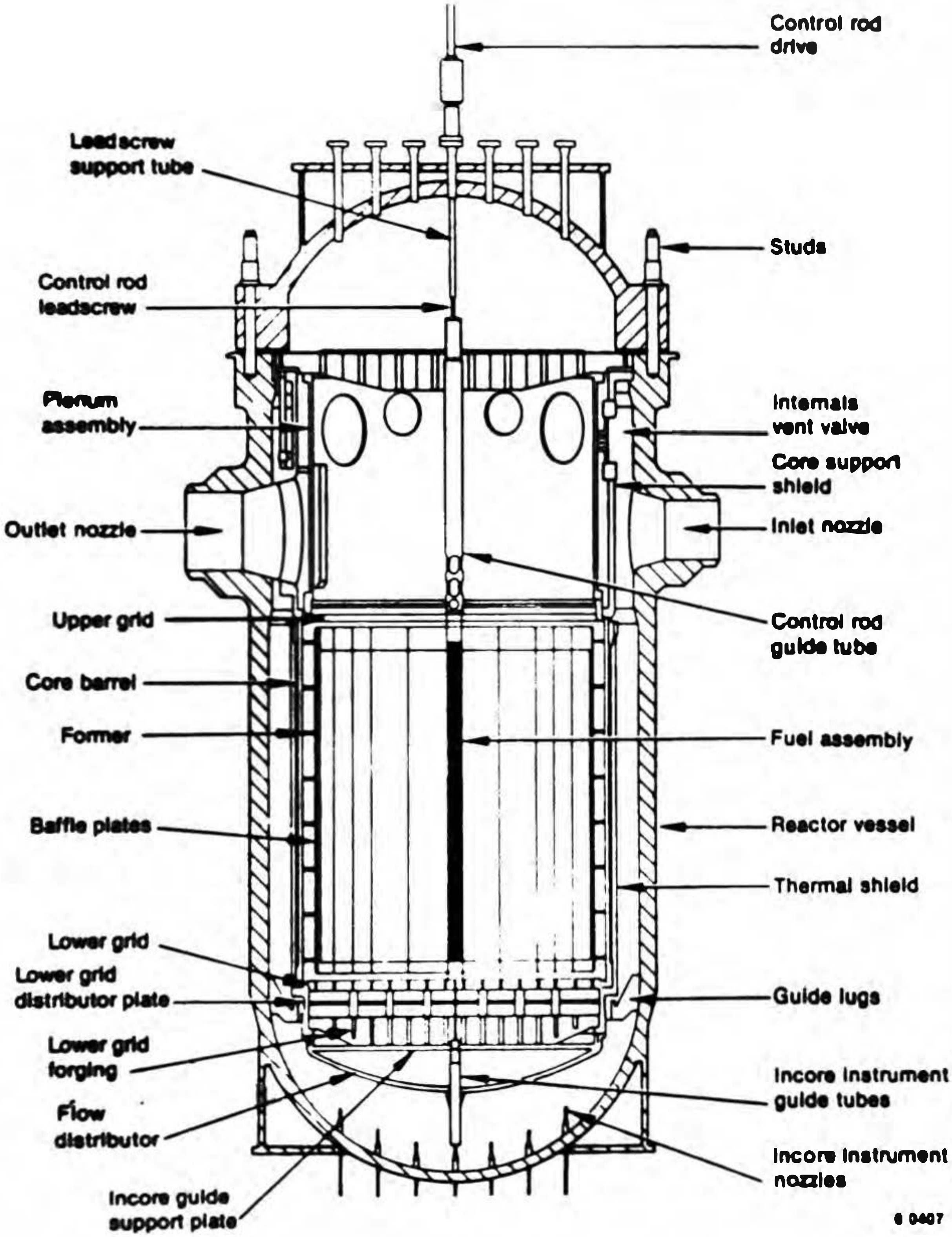


Figure A-2. General arrangement of TMI-2 reactor vessel and internals.

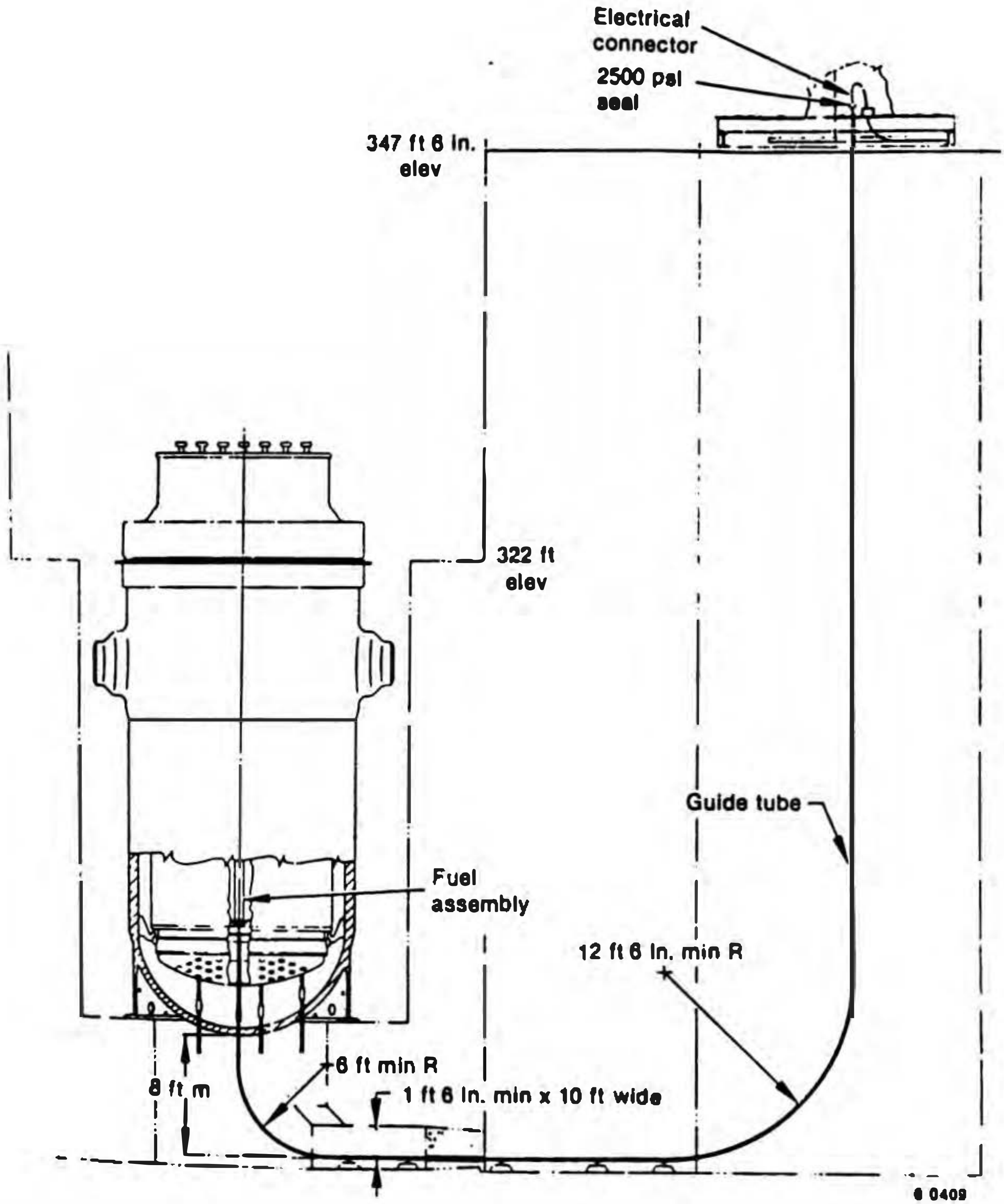


Figure A-3. Schematic of typical in-core instrument assembly.

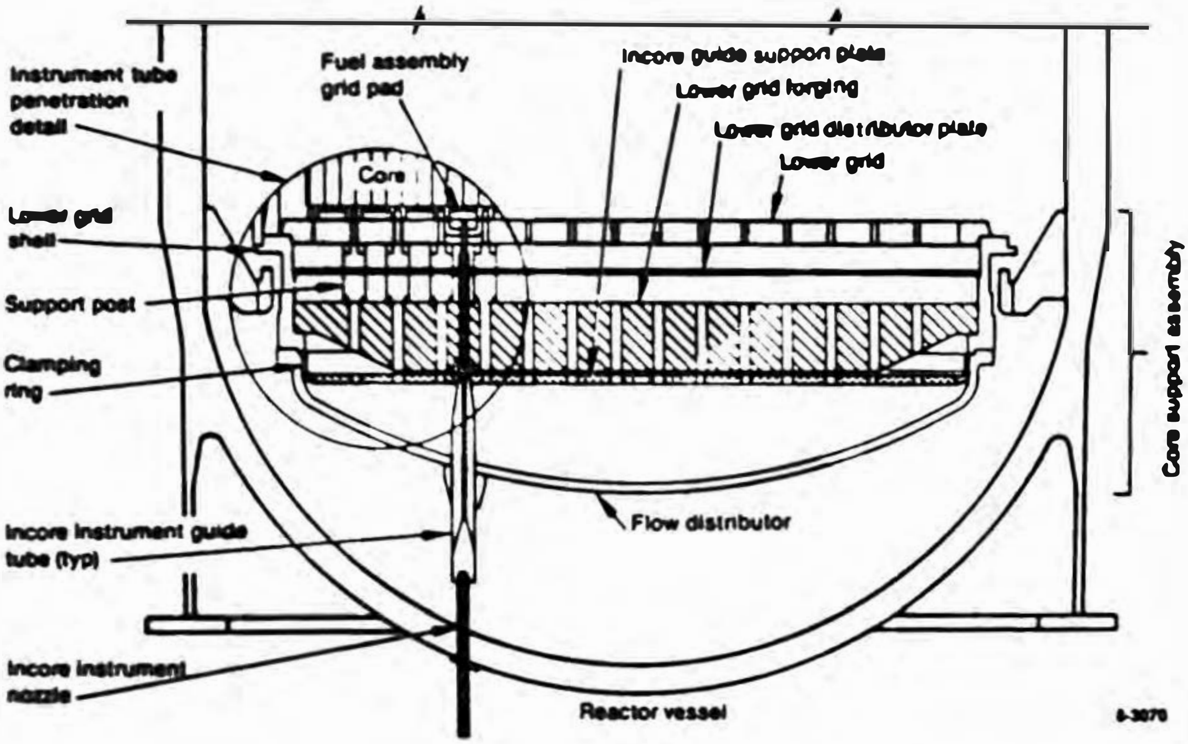


Figure A-4. TMI-2 core support assembly configuration.

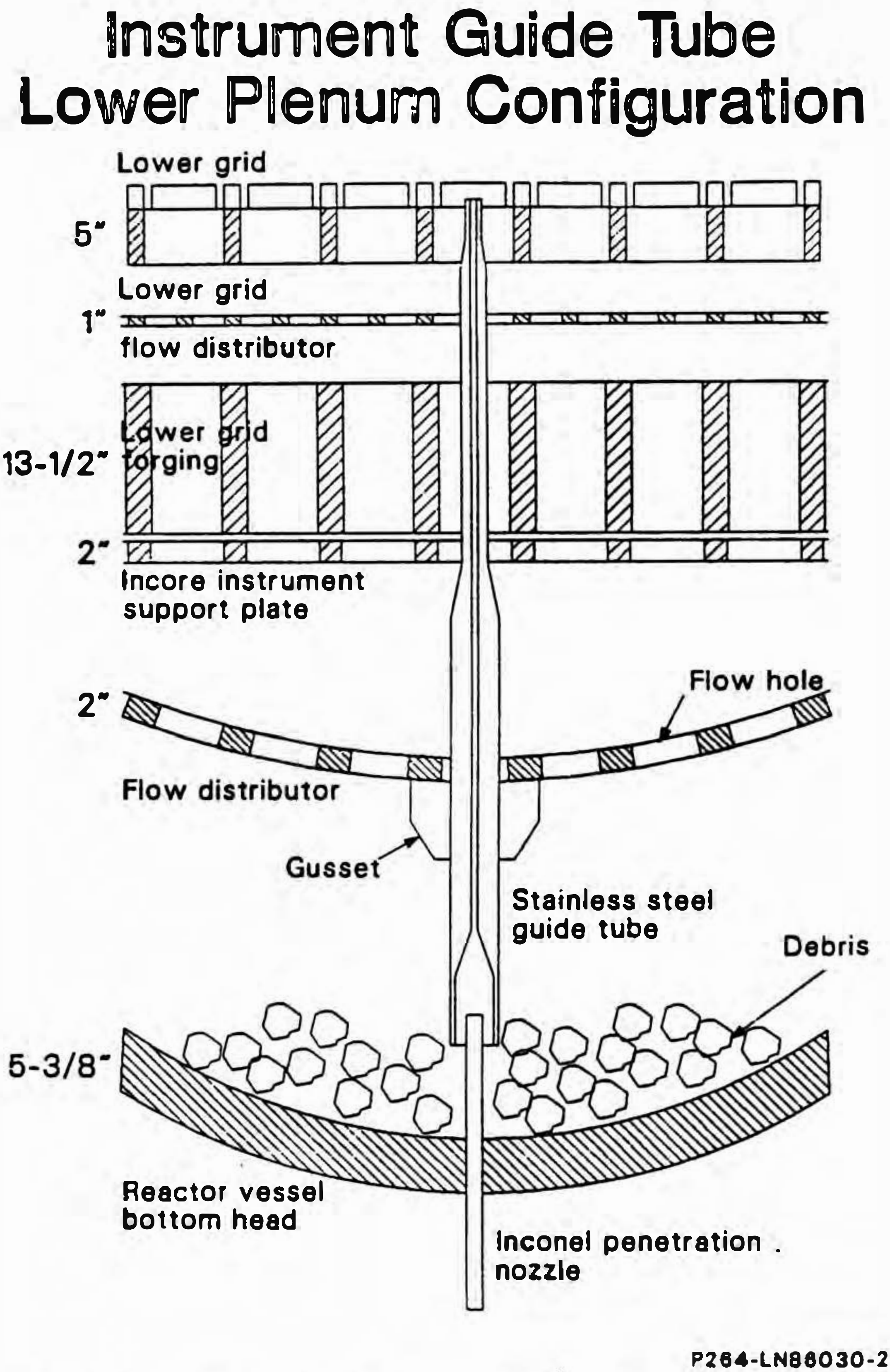
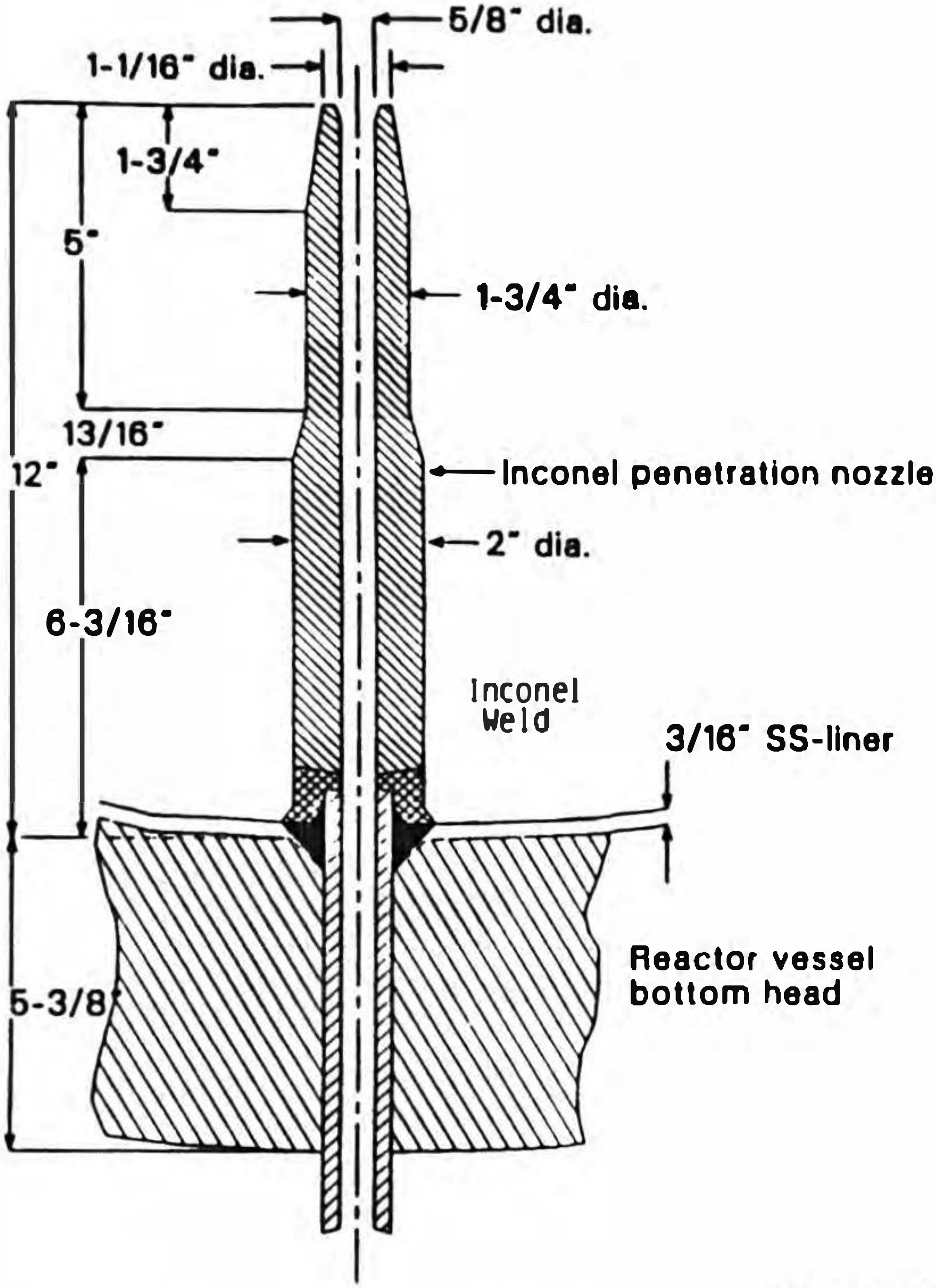


Figure A-5. Illustration of TMI-2 lower plenum region showing bottom-entry instrument penetration nozzle and guide tube.



P284-LN86030-4

Figure A-6. Dimensional characteristics of Inconel penetration nozzle.

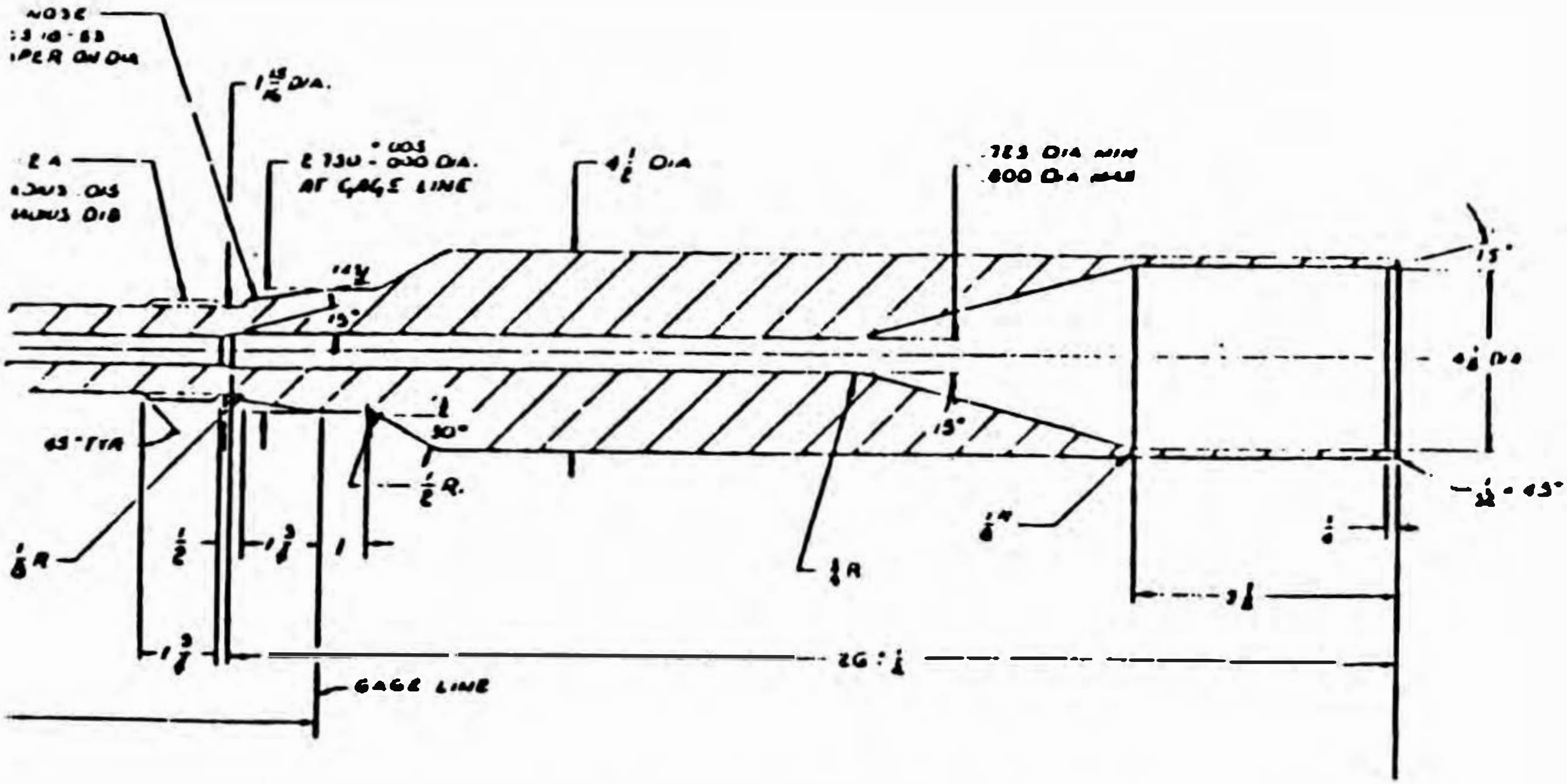


Figure A-7. Dimensional characteristics of stainless-steel instrument guide tube.

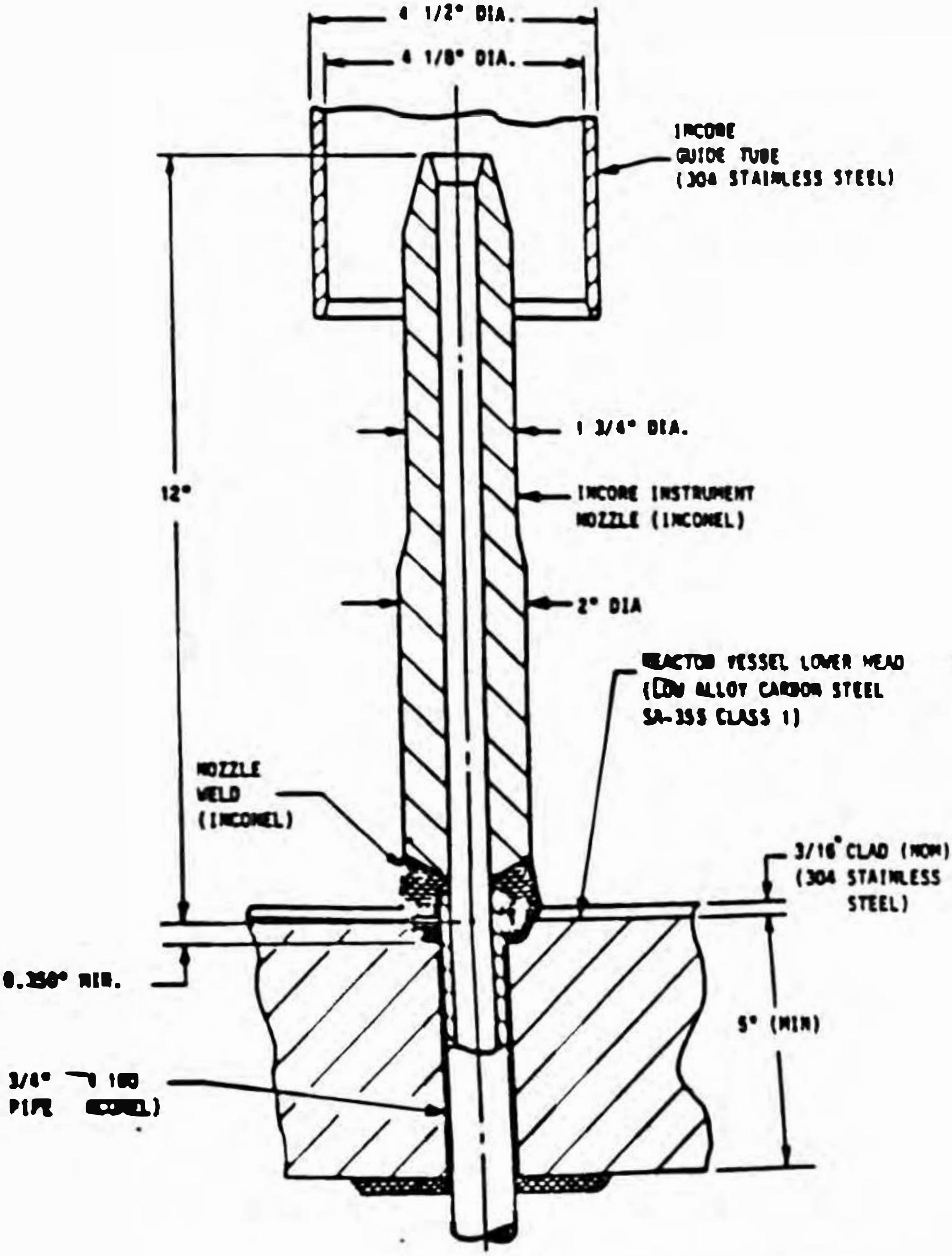


Figure A-8. Illustration of penetration nozzle with stainless-steel guide tube sleeve-fitted over tip of nozzle (Ref. 4).

The above designation is standard nomenclature for Inconel, with a 16 weight-percent chromium (Cr) composition. A phone conversation (March 8, 1988) with Mr. K. Moore (B&W: 804-385-3277), confirmed that both the internal J-weld and external bead-seal welding materials are Inconel-600 (INCO-182T).

As discussed in Reference A-4 (see page 38), Inconel-600 has a minimum ultimate strength of 80,000 psi up to 811 K (1000°F). In Reference A-4 the size of the Inconel weld was assumed to be in the range of 7 to 30 mils.

Figure A-9 shows a cross section of the instrument penetration tube. Figures A-10 and A-11 show top and side views, respectively, of the core baffle plates.

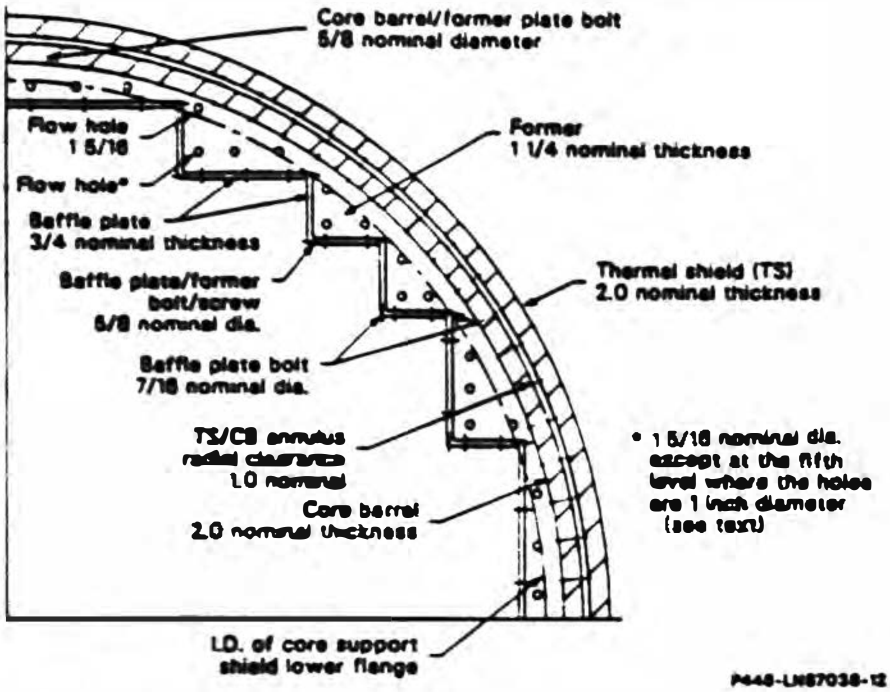


Figure A-10. Illustration of the TMI-2 core-former assembly.

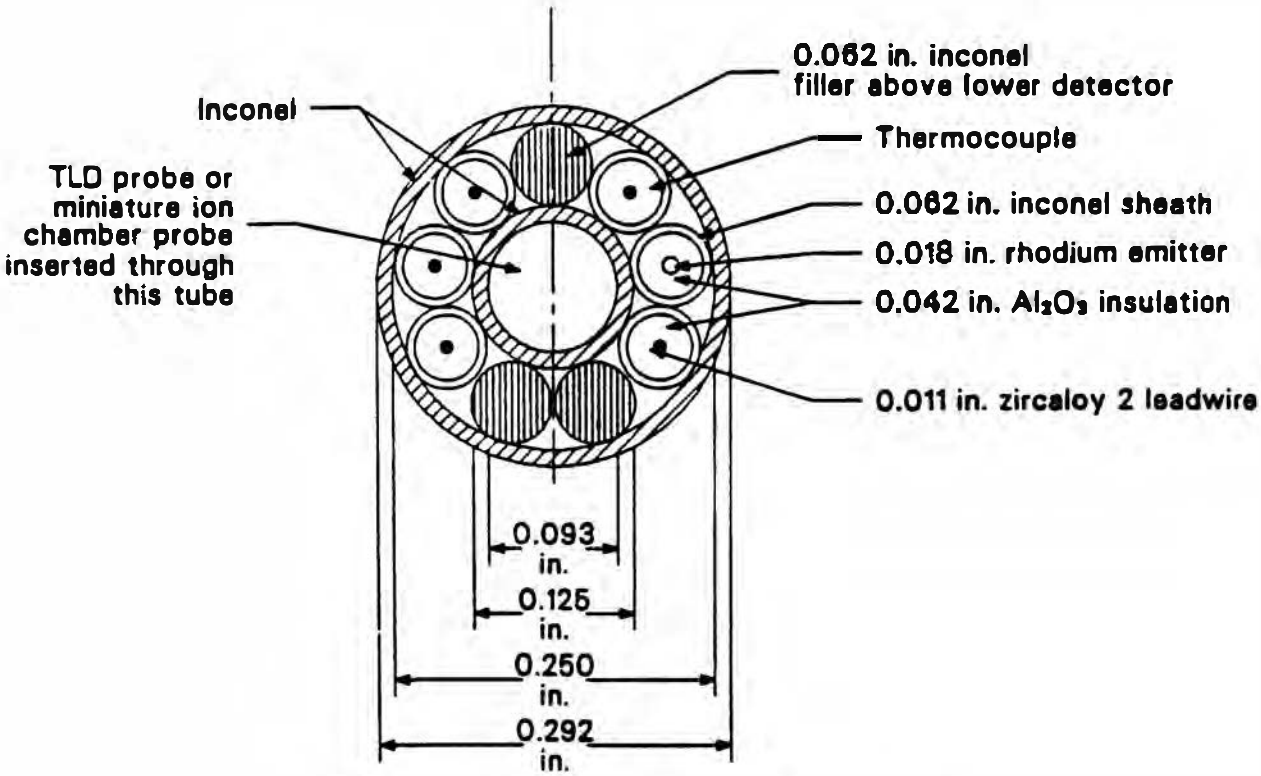


Figure A-9. Illustration of TMI-2 bottom-entry detector cross section; center hole serves as an access port for insertion of miniature ion chamber for gamma survey of lower plenum.

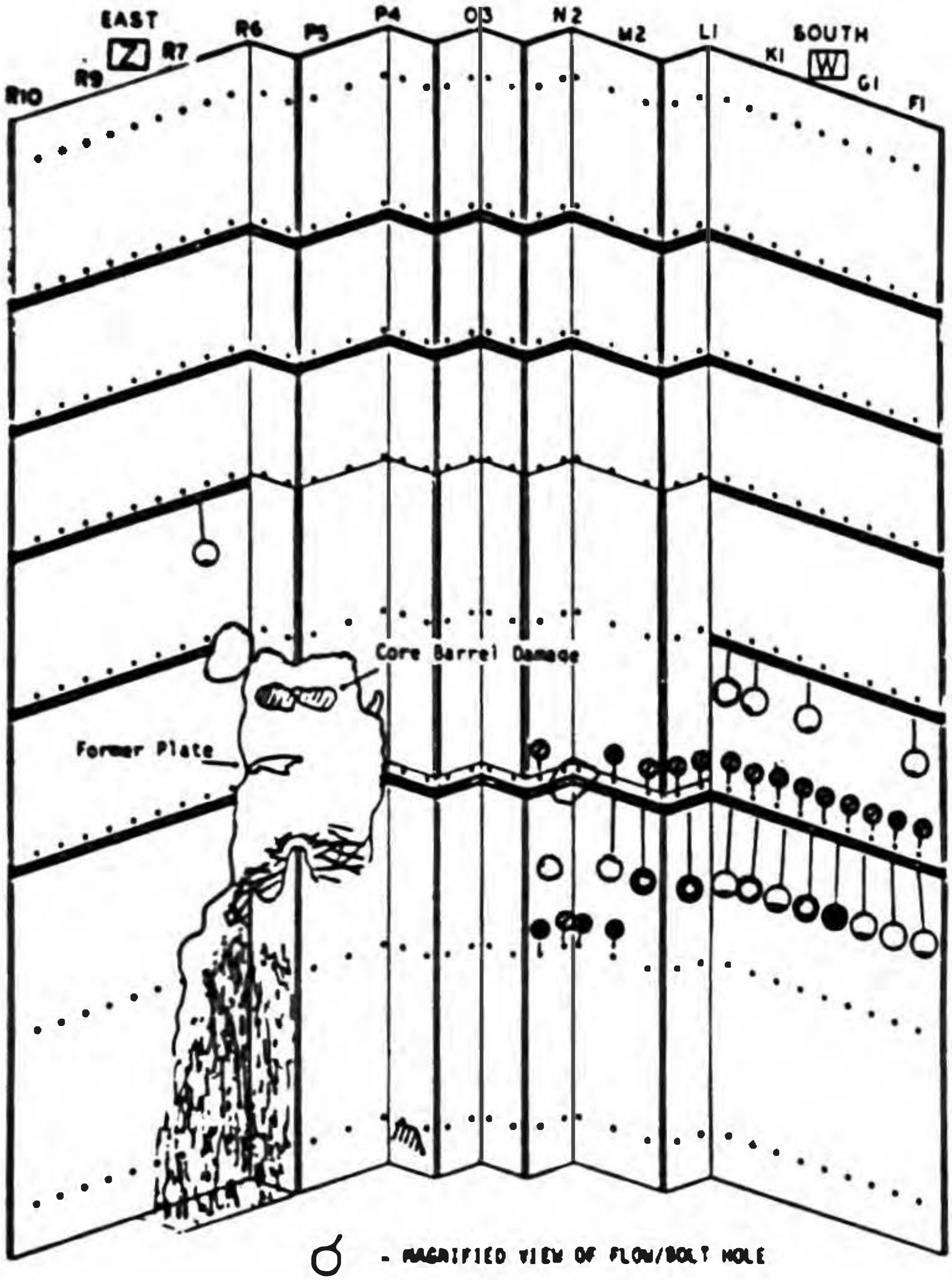


Figure A-11. Illustration of melt-through hole noted in east-quadrant baffle plates.

REFERENCES

A-1. Three Mile Island Nuclear Power Station, "Unit Number Two, License Application, Final Safety Analysis Report," Docket-50320-73, April 1974, Section 1.

A-2. Three Mile Island Nuclear Station-Unit No. 2, "Mechanical Flow Diagrams," Burns and Roe Report, April 1979.

A-3. A. W. Cronenberg, S. R. Behling, J. M. Broughton, "Assessment of Damage Potential to the TMI-2 Lower Head Due to Thermal Attack by Core Debris," EGG-TMI-7222, June 1986.

A-4. D. A. Nitti et al., "Evaluation of the Structural Integrity of the TMI-2 Reactor Vessel Lower Head," Babcock and Wilcox Report 77-1158426-00, June 1985.

APPENDIX B

STATUS OF DOE SAMPLE ACQUISITION
AND EXAMINATION PROGRAM

APPENDIX B
STATUS OF DOE SAMPLE ACQUISITION AND EXAMINATION PROGRAM

The status of the DOE Sample Acquisition and Examination Program for TMI-2 is summarized in Table B-1.

TABLE B-1. TMI-2 SAMPLE ACQUISITION AND EXAMINATION PROGRAM

Samples	Examinations	Sample Status at INEL
Ex-Reactor Coolant System		
AFHB Reactor Coolant Bleed Tanks		
Liquid	Radiochemical	Exams complete
Sediment	Physical and radiochemical	Exams complete
AFHB Makeup and Purification System Demineralizer		
Liquid	Radiochemical	Exams complete
Resin	Physical and radiochemical	
AFHB Makeup and Purification and Pump Seal Water System Filter Debris	Physical, metallurgical, and radiochemical	Exams complete
Reactor Building Basement		
Water	Physical and radiochemical	Exams complete
Sediment	Physical and radiochemical	Exams complete
Concrete	Physical and radiochemical	Exams complete
Reactor Building Air Cooler Panel Surface Deposits	Radiochemical	Exams complete
Reactor Building Upper Levels		
Free-volume air/gas	Physical and radiochemical	Exams complete
Liquid	Radiochemical	Exams complete
Concrete	Physical and radiochemical	Exams complete

TABLE B-1. (continued)

Samples	Examinations	Sample Status at INEL
Reactor Coolant Drain Tank		
Liquid	Radiochemical	Exams complete
Sediment	Radiochemical	Exams complete
Reactor Coolant System		
Surface Deposit		
Pressurizer upper head manway cover	Physical, metallurgical, and radiochemical	Exams complete
Steam generator upper head manway and handhole covers	Physical, metallurgical, and radiochemical	Exams complete
A-loop RTD thermowell	Photography and radiochemical	Exams complete
Sediment		
Pressurizer lower head	Physical and radiochemical (final exams not funded)	Preliminary exams (GPUN/Westinghouse) complete--GPUN will obtain additional samples approximately April 1988
B-loop steam generator upper tube sheet	Physical, metallurgical, and radiochemical	Exams complete
A-loop steam generator upper tube sheet	Physical and radiochemical	Exams complete
Decay heat line concretized core debris	Physical, metallurgical, and radiochemical	GPUN will obtain approximately April 1989
Reactor Vessel and Contents		

TABLE B-1. (continued)

Samples	Examinations	Sample Status at INEL
Upper core loose debris[a]	Physical, metallurgical, and radiochemical	Exams complete
Control rod leadscrews and one leadscrew support tube	Physical, metallurgical, and radiochemical	Exams complete
Lower vessel debris rocks[a]	Physical, metallurgical, and radiochemical	Exams complete
Lower vessel debris bulk samples	Physical and radiochemical (metallurgical--not funded)	GPU will obtain approximately July 1989
Lower vessel debris near lower head	Physical, metallurgical, and radiochemical--not funded	GPU will obtain approximately May 1989
Distinct Components		
Upper end fittings[a]	Photography	Exam complete
Fuel rod upper ends[a]	Physical, metallurgical, and radiochemical	Exams complete
Control rod upper ends[a]	Physical, metallurgical, and radiochemical	Exams complete
Core Bore Samples		
Consolidated region[a]	Physical, metallurgical, and radiochemical	Exam complete
Fuel rod lower ends[a]	Physical, metallurgical, and radiochemical	Exams complete
Control rod lower ends[a]	Physical, metallurgical, and radiochemical	Exams complete

TABLE B-1. (continued)

Samples	Examinations	Sample Status at INEL
Burnable poison rod lower ends[b]	Physical, metallurgical, and radiochemical	Exams complete
Instrumentation tube lower ends[b]	Physical, metallurgical, and radiochemical	Exams complete
Post-Core Bore Pulverizing of Consolidated Region (large and small rocks)[c]	Physical	Exams complete
Possible Relocation Path at Core Position R-6 (five rocks)	Physical, metallurgical, and radiochemical (not funded)	At INEL
Crust Samples from Core Bypass Region	Physical, metallurgical, and radiochemical (not funded)	GPU will obtain approximately October 1989
Core Support Assembly Samples	Physical, metallurgical, and radiochemical	GPU will obtain approximately May 1989
Lower Head Samples	Physical and metallurgical (not funded)	NRC will obtain approximately April 1989

a. Samples presently being examined by CSNI laboratories in Canada and Europe.

b. Japan has requested samples of all items superscripted with a, b, or c for examination beginning in 1990.

c. Korea has requested many of the samples superscripted with a and c for examination beginning in 1989.

APPENDIX C

TMI-2 SAMPLES FOR FOREIGN COUNTRY EXAMINATION THROUGH THE DOE PROGRAM

APPENDIX C
TMI-2 SAMPLES FOR FOREIGN COUNTRY EXAMINATION THROUGH THE DOE PROGRAM

The TMI-2 samples that are being examined by European Countries and Canada are listed in Table C-1. TMI-2 samples that will be examined by Japan and Korea are listed in Tables C-2 and C-3, respectively.

TABLE C-1. TMI-2 ACCIDENT SAMPLES FOR THE LABORATORIES OF THE OECD/CSNI JOI T TASK FORCE ON TMI-2

	Sample Number							
Sample Type	Canada	France	JRC	FRG-KFA	FRG-KFK	Sweden	Switzerland	United Kingdom
Reactor vessel	7-1	11-7-E	--	--	11-1-A	11-4-G	--	11-2-D
lower head core	11-5-F1	--	--	--	11-5-A	--	--	--
debris (rocks)	--	--	--	--	11-7-A	--	--	--
Upper core loose	HB (36 cm)	HB (77 cm)	HB (36 cm)	--	--	--	--	--
debris	HB (56 cm)	--	--	--	--	--	--	--
Upper core fuel	4 (lower 1/2)	1/2 of 2	1/2 of 5	1/2 of 2	1/2 of 5	--	--	3-94 (btm 6 in.)
rod segments	3-6 (btm 6 in.)	1/2 of 3-102	3.35 (btm 6 in.)	3-70 (btm 6 in.)	3-88 (btm 6 in.)	--	--	--
	11-3 (btm 6 in.)							
Upper core	--	--	--	3-3 (btm 6 in.)	3-7	--	--	--
control rod	--	--	--	3-13 (btm 6 in.)	3-9 (btm 6 in.)	--	--	--
segments								
Lower core fuel	G12-R8-B	--	--	--	--	--	--	K9-R14-4
rod segments	--	--	--	--	--	--	--	K9-R14-5
Lower core	--	--	--	G12-R16-2	G12-R13-4	--	--	--
burnable	--	--	--	--	G12-R16-4	--	--	--
poison rod								
segments								
Core bore core	D8-P1-F	D8-P1-E	D8-P2-B	--	D8-P1-A	D8-P3-C	G8-P11-H	D8-P3-B
sections	D8-P3-A2	G12-P1-D1	D8-P3-A1	--	G12-P1-B	G12-P1-C2	K9-P2-F	G12-P1-C1
	G12-P1-D5	--	D7-P4-E	--	K9-P1-F	--	--	K9-P1-B
	K9-P1-H	--	--	--	K9-P2-A	--	--	K9-P2-B
	--	--	--	--	D7-P4-B	--	--	--
Core bore rocks	D8-P4-B	K9-P3-H	G12-P2-E	--	D4-P1-B	G12-P10-B	G8-P8-C	B4-P2-B
	K9-P4-G	D7-P5	G12-P6-E	--	G8-P4-A	--	--	K9-P3-G
	--	--	G12-P9-B	--	G8-P5-A	--	--	--
	--	--	G12-P10-A	--	G12-P10-C	--	--	--
	--	--	N5-P1-E	--	K9-P3-B	--	--	--
	--	--	--	--	K9-P4-C	--	--	--
	--	--	--	--	K5-P1-B	--	--	--
	--	--	--	--	D7-P3	--	--	--
	--	--	--	--	D7-P8-B	--	--	--

TABLE C-2. TMI-2 SAMPLE COLLECTION FOR JAPAN--PRELIMINARY

Sample Description	Identification Number	Mass (g)	
		Total	U-235[a]
Loose debris from reactor vessel lower plenum	7-7	0.4	0.007
	11-4-B	1.3	0.025
	11-6-B	3.5	0.07
Loose debris from core cavity floor	E9-4	9.3	0.18
	H8-1	12.3	0.24
Core distinct components			
15.2 cm-fuel rod segment from core position (CP) M2 upper end	--	--	--
88.0 cm-fuel rod segment from CP C7 upper end	3-28	10	5.3
Control rod and guide tube from CP C7 upper end	--	--	--
Fuel and control rod assembly upper end fittings from CP M9	--	--	--
10.2-cm fuel rod segments from core bores	D4-R12-2	71.92	2.12
	D4-R12-4	71.92	2.12
	D4-R12-6	71.92	2.12
	D4-R12-8	71.92	2.12
	G8-R6-2	71.92	2.12
	N12-R4-2	71.92	2.12
	N12-R4-4	71.92	2.12
	N12-R4-6	71.92	2.12
	G12-R12-2	Nonfuel	--
	G12-R12-4	Nonfuel	--
	G12-R12-6	Nonfuel	--
	G12-R12-8	Nonfuel	--
	FR Seq. 6	4	2.12
10.2 cm-control rod and guide tube segments from core bores	N12-R7-2	Nonfuel	--
	N12-R7-4	Nonfuel	--
	N12-R7-6	Nonfuel	--
10.2 cm-instrument tube segments from core bores	K9-R4-2	Nonfuel	--
	K9-R4-4	Nonfuel	--
Fused-together core material	D8-P2-C	135.6	2.63

TABLE C-2. (continued)

Sample Description	Identification Number	Mass (g) Total	U-235[a]
Core bore core sections	D8-P2-D	93.7	1.79
	D8-P3-A3	35.2	0.68
	D8-P3-A4	30.2	0.76
	D8-P3-A5	22.2	0.43
	D8-P3-03	74.6	1.44
	D8-P3-04	59.8	1.16
	D8-P3-E	160.0	3.10
	G8-P11-A	93.5	1.81
	G8-P11-D	390	7.55
	G8-P11-G	209	4.04
	G8-P11-K	86.3	1.67
	G12-P1-E	60.6	1.17
	G12-P1-D4	38.8	0.75
	D7-P4-A	136.5	2.64
	D7-P4-F	147.9	2.86
	D8-P3-D3	74.6	1.44
	D8-P3-D4	59.8	1.16
	D8-P3-E	160	3.10
	G12-P1-D4	38.8	0.75
	H9/K9-P4	26	0.50
	H9/K9-P6	72.3	1.40
	H9/K9-P5	30.4	0.59
	07-P1-A2	3.8	0.074
	M11-P1	1,671	32.37
	M11-P10	62.6	1.21
Rock-size samples	D4-P2-C	6.2	0.12
	G8-P6-A	20.5	0.40
	G12-P8-8	48.5	0.94
	G12-P3-A	45	0.87
	G12-P2-D	40.5	0.78
	N5-P1-F	17.9	0.35
	N12-P1-B	0.65	0.01

a. Estimated.

TABLE C-3. TMI-2 SAMPLES FOR EXAMINATION BY THE KOREAN ADVANCED ENERGY RESEARCH INSTITUTE

Sample Identification[a]	Description	Mass (g) Total	Mass (g) U-235[b]
Lower vessel debris			
11-6-C	Large particle cross section	8.0	0.15
11-8-P1	Large particle cross section	10.1	0.19
Core distinct component			
4-b	Fuel rod segment	4.75	2.5
3-20	Fuel rod segment	8.6	1.6
3-94A	Fuel rod segment (lower 12 cm)	6	3.1
2-16B	Control rod/guide tube segment	Nonfuel	
Core bore sections			
D8-P2-E	Plug cross section (agglomerate)	123.5	2.4
K9-P1-C	Plug cross section (agglomerate/transition)	~30	0.58
K9-P2-C	Plug cross section (upper crust)	77.7	1.5
Core bore particles			
G12-P8-D	Dense debris	18	0.3
K9-P3-C	Melt-metal debris	37.7	0.7
K9-P4-A2	Porous debris (met mount)	10.5	0.2
H8/K9-P9	Porous debris	23.55	0.45

a. Each of the samples will be contained in individual aluminum sample containers which may be packaged in lead sheet to reduce the radiation field. At the time the samples are packaged, a loading diagram will be prepared which describes the packaging and location in the drums. This description will be sent with the drums to Korea.

b. Estimated.