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THE TMI-2 ACCIDENT EVALUATION PROGRAM[1]

D. J. Osetek
J. M. Broughton
R. R. Hobbins

Idaho National Engineering Laboratory
EG&G Idaho, Inc.
Idaho Falls, ID 83415

INTRODUCTION

The accident at the Three Mile Island Unit 2 (TMI-2) reactor, now 10 years old, remains as the United States' worst commercial nuclear reactor accident. Although the consequences of the accident were restricted primarily to the plant itself, the potential consequences of the accident, should it have progressed further, are large enough to warrant close scrutiny of all aspects of the event.

The TMI-2 accident resulted in extensive oxidation and melting of the reactor core and in significant release of fission products from the fuel. At least 45% (62 metric tons) of the core melted and about 20 metric tons of molten core material relocated into the lower plenum of the reactor vessel. However, in spite of severe core damage, water in the reactor vessel prevented failure of the reactor vessel's lower head. The TMI-2 accident was similar to a station blackout or smallbreak LOCA accident sequence like those postulated in reactor safety studies. The accident progressed into the latter stages of core melt progression, that is, generally much further into the progression of a severe accident than core melt experiments have explored. As such, the accident greatly expands and enhances the current understanding of core melt progression and fission product behavior during severe accidents and provides a unique data base for assessment of severe accident computer models.

TMI-2 accident research is being conducted by the U.S. Department of Energy (DOE) to provide the basis for more accurate calculations of source terms for postulated severe accidents [1]. Research objectives supporting this goal include developing a comprehensive and consistent understanding of the mechanisms that controlled the progression of core damage and subsequent fission product behavior during the TMI-2 accident, and applying that understanding to the resolution of important severe accident safety issues. Developing a best-estimate scenario of the core melt progression during the accident is the focal point of the research and involves analytical work to interpret and integrate: (1) data recorded during the accident from plant instrumentation, (2) the post-accident state of the core, (3) results of the examination of material from the damaged core, and (4) related severe-accident research results. This paper summarizes the TMI-2 Accident

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Evaluation Program that is being conducted for the U.S. DOE and briefly describes the important results that have been achieved. The Program is divided into four parts: Sample Acquisition and Plant Examination, Accident Scenario, Standard Problem Exercise, and Information and Industry Coordination. Each of these Program parts is described in the following sections.

SAMPLE ACQUISITION AND PLANT EXAMINATION

Video inspections, debris sampling, and core boring data have provided the information that was used to reconstruct the post-accident condition of the TMI-2 core and vessel components. Figure 1 illustrates this end-state geometry. Physical examination and chemical analyses of material samples collected from various vessel regions have identified the types of interactions that occurred and the temperatures that were reached in the different parts of the core and in the relocated material. The DOE Program of core sampling is nearly complete; only a few additional samples of lower plenum debris will be collected before this part of the Program is finalized. Sample analysis by various laboratories in the U.S. and other countries is ongoing. Some results have been presented at the ANS Topical Meeting held in Washington, D.C. in November 1988 [2]. Further information will be published in 1989 and final results in 1990.

There is also a new program under way sponsored by the U.S. NRC and the Nuclear Energy Agency (NEA) of the Organization for Economic Cooperation and Development (OECD) to collect and analyze samples of steel from the inside surface of the TMI-2 vessel lower head and samples of the debris or prior molten material that is adjacent to the lower head. The results of this program will become available late in 1990. What has been learned up to this time about the post-accident condition of the TMI-2 reactor vessel and core is summarized in this section. Further details are provided in References 2-5.

Upper Plenum Region

The upper plenum assembly suffered little damage and was found to be essentially intact. Examination of the control rod drive lead screws taken from the upper plenum indicates that temperatures reached approximately 1250 K just above the central region of the core, approximately 1000 K near the core periphery, but only about 700 K at elevations above the outlet nozzles. The extent of melting and ablation of the upper grid was bi-nodal highly localized within two damage zones about 1.2 m in diameter, and did not progress significantly above the upper grid.

Core Region

Four distinct regions within the original core have been identified, as shown in Figure 1. These include: (1) a void region in the upper part of the core; (2) a region of loose, fragmented fuel debris immediately below the void region; (3) a region of consolidated, previously molten core material located in the mid-to-lower region of the core; and (4) partial to full length minimally damaged fuel rods in the lower and peripheral regions of the core.

The volume of the cavity at the top of the core was approximately 9.3 m^3. The average depth of the cavity was approximately 1.5 m; in places, maximum depth approached 2 m. Standing fuel rods remained at the core periphery in

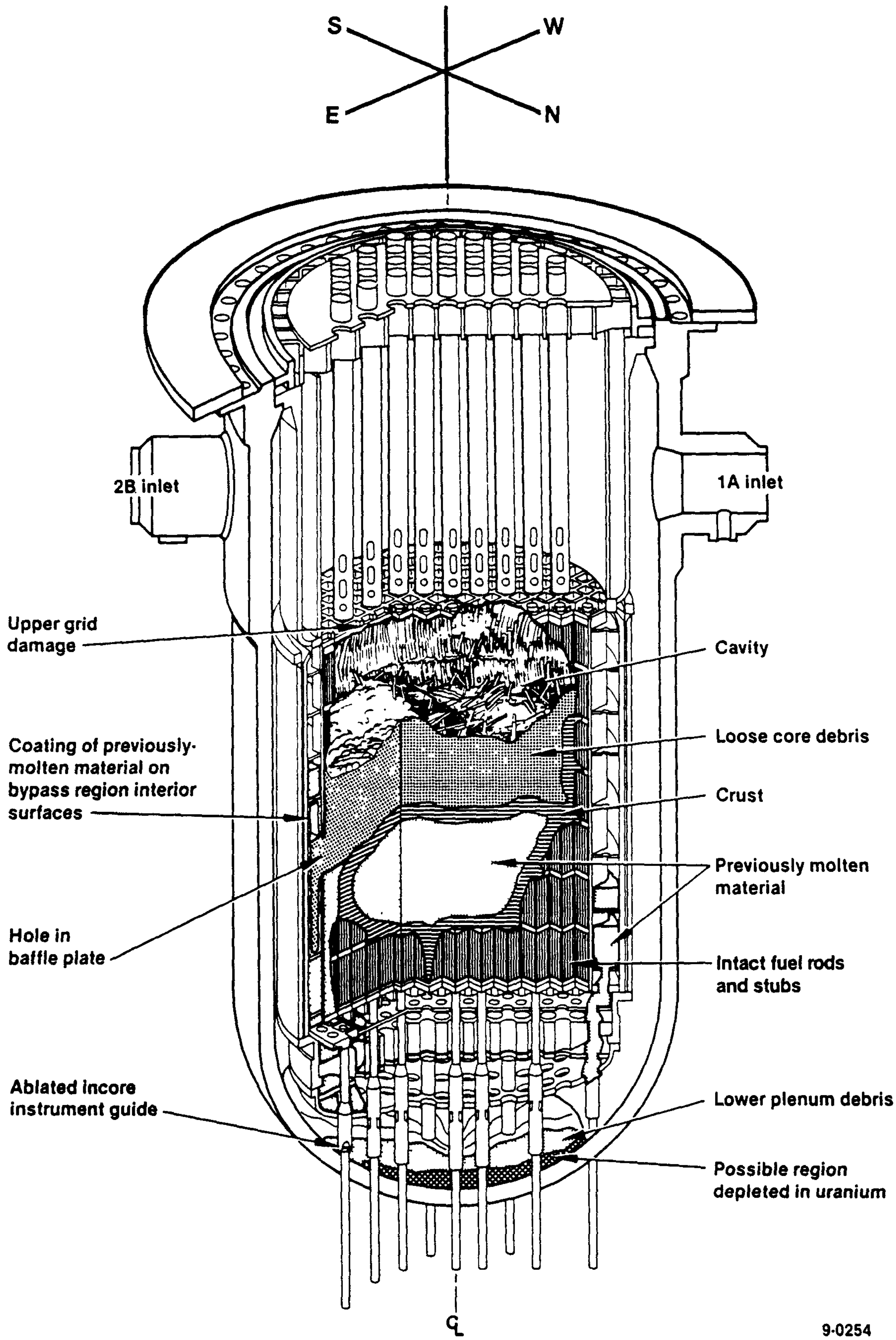

FIGURE 1. Post accident configuration of the TMI-2 core and vessel structures.

42 of the 177 fuel assemblies. In only two of these assemblies were 90% of the rods intact over the entire length; the other fuel assemblies suffered varying degrees of greater damage.

A debris bed, ranging from 0.6 to 1.0 m in depth, rested on top of the previously molten core material region located at about core mid-plane. The debris consisted of whole and fragmented fuel pellets, oxidized cladding shards, and previously molten metallic and ceramic particles. The debris was found to contain some particles of previously molten U-Zr-O, indicating peak temperatures greater than 2200 K. A few particles were previously molten $(U,Zr,)O_2$ and UO_2, indicating some regions of the upper core experienced temperatures between 2800 and 3100 K.

Beneath the upper core debris bed was a large consolidated mass, approximately 3 m in diameter, containing about 32.7 metric tons of previously molten core material. The central region of this consolidated mass was primarily previously molten ceramic $(U,Zr)O_2$ laced with previously molten metallic material. The metallic structures consisted primarily of silver and iron. A crust was identified around the consolidated region. The bottom, bowl-shaped portion of this crust, with an approximate thickness of 10 cm, was composed of previously molten metallics, primarily zirconium, silver, and iron, frozen in the flow channels between fuel rods. Standing columns of fuel pellets devoid of cladding were all that remained of the fuel rods in the lower crust. The top crust consisted of previously molten ceramic and metallic materials from 1 to 3 cm thick, with some embedded debris from the upper debris bed. The metallic material in the top crust was primarily iron and silver, with some nickel and indium.

Standing fuel rod stubs were found extending upward from the lower fuel assembly grid to the bottom of the consolidated region. These stubs varied in length from approximately 0.2 to 1.5 m. On the east side of the core, four adjacent fuel assemblies were almost completely replaced with previously molten core material, indicating that molten core material flowed downward through this region.

Core Support Assembly

In the upper core support assembly (CSA) fuel debris lay on the core former plates, between the baffle plates (vertical plates forming the peripheral boundary of the core) and the core barrel. The location of the debris varied in elevation from 1.5 m in the north quadrant of the vessel to almost zero near the southwest quadrant. The baffle plates in the east quadrant were partially melted, exposing the core barrel to potential damage. Molten core material entered the upper CSA through this hole, flowed circumferentially in the CSA, and drained into the lower CSA at nearly all locations around the core.

The lower CSA suffered little damage. Beneath several fuel assemblies in the east quadrant, resolidified material completely filled flow holes in the various plates, and columns of material between the plates suggested that a major pathway for flow of molten core material from the core into the lower plenum was in the southeast quadrant of the reactor vessel.

Lower Plenum Region

Extensive video examinations of the lower plenum were used to construct a contour map of the top surface of the lower plenum debris bed. Visual observations indicate that the size of debris particles varies widely, ranging from less than a few centimeters in diameter near the center of the vessel, to larger rock-like pieces tens of centimeters across near the vessel wall. The debris particles extend radially to the downcomer, except in the north quadrant, where an abrupt "wall" of previously molten material exists about 1.0 m from the vessel centerline.

Fission Product Distribution

The distribution of fission products in the TMI-2 reactor system has been determined by analysis of representative samples collected from the various locations in the reactor vessel and the balance of the reactor facility. The low volatility fission products that have been measured in TMI-2 samples include Ce-144, Eu-154, and Eu-155. The medium volatile species measurable are Sr-90, Sb-125, and Ru-106. The high volatiles are the noble gases, iodine and cesium; Kr-85, Cs-137, and I-129 were released from the fuel to a large extent at the temperatures found during the TMI-2 accident.

The data indicate that the low volatile fission products were not transported out of the reactor core and that only a small amount of fuel material was physically transported to the reactor coolant system. In the reactor vessel, the fuel material (and the associated low volatile fission product) distribution is similar to the overall mass distribution as might be expected.

The fission product distribution data for the medium volatiles indicate that only a small amount (3.5% of the core inventory) was released from the reactor vessel. Of this released quantity, almost all was retained in the reactor building with only 0.1% released to the auxiliary building where it was retained in the reactor coolant bleed tanks. In the reactor vessel, the distribution is again similar to the fuel material distribution which indicates that the bulk of the medium volatile fission product inventory was retained in the prior molten fuel material. The Sb-125 and Ru-106 data indicate high concentrations in the prior molten metallic portions of the lower reactor core. The metallic sample concentrations of Sb-125 are 6-20 times those found in the intact fuel.

The data for the high volatiles (Kr-85, I-129, and Cs-137) indicate significant relocation and transport of these radionuclides to the reactor coolant system and, consequently, to the reactor building atmosphere and basement. Approximately 85% of the core inventory of Kr-85 is accounted for by measurements of the containment building atmosphere.

Measurements of radioiodine concentrations in the reactor building basement, the major repository for iodine outside the reactor core, indicate that much of the core inventory of radioiodine could be contained in the reactor building basement. The data show the same retentions for iodine and cesium in the basement.

In the reactor vessel, the data indicate similar distributions in the debris for both the Cs-137 and I-129. The data for the upper core debris indicate that about 20% of the original inventories of both Cs-137 and I-129 were retained in the partially molten fuel debris. In the lower parts of the core there is evidence that suggests that some of the I-129 is found in association with metallic material.

Data from the consolidated melt region indicate an overall Cs-137 retention of 10% in prior molten material. This holdup of Cs-137 in high temperature material is surprising.

ACCIDENT SCENARIO

A postulated scenario of events that occurred during the accident was developed in the DOE Program from the post-accident core configuration, interpretation and analyses of data from on-line instrumentation, and analyses of various postulated core-damage-progression phenomena. The accident was divided into seven major periods marked in time after scram of the reactor (t=0). The reactor's coolant-system-pressure history, shown in Figure 2, provides a convenient reference with which to view these periods.

Loss of Coolant (0-100 min)

The accident was initiated by loss of feedwater to the steam generators, a "small break" loss-of-coolant accident, with limited ECCS. The resulting increase in the RCS pressure caused the pilot-operated relief valve (PORV) on the pressurizer to open and the reactor to scram. The PORV failed to close as the reactor cooling system (RCS) pressure decreased. Because of decreasing liquid inventory in the primary cooling system and continued heating by decay heat in the fuel, coolant void fraction increased sufficiently over the next hour to cause the reactor pumps to vibrate excessively from cavitation. At about 73 min, both pumps in the B-loop were shut off, resulting in reduced flow in the B-loop and loss of heat removal by the B-loop steam generator. At about 100 min, the A-loop pumps were shut off, terminating forced convection cooling of the core. The vessel's resulting liquid level is estimated to have been near the top of the core, shortly after the last coolant pump shutdown.

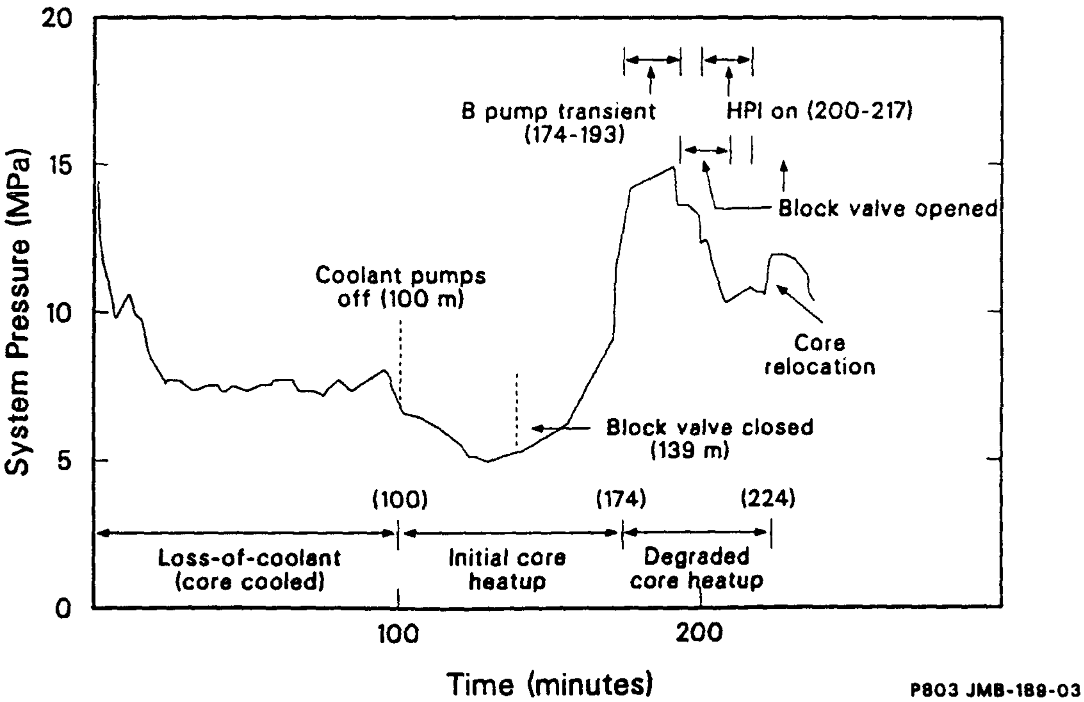

FIGURE 2. Pressure history of the TMI-2 primary coolant system during the accident.

Initial Core Damage (100-174 min)

The liquid level in the reactor vessel decreased after 100 minutes because makeup to the primary cooling system was insufficient to compensate for coolant loss through the PORV. Core uncovery began shortly after 100 min, and temperatures at the top of the core started to increase. By approximately 140 min, the core liquid level had dropped to about mid-core. The upper regions of the core heated sufficiently (1100 to 1200 K) to result in cladding ballooning and rupture. Cladding failure and release of gaseous fission products were substantiated by significant increases in containment radiation levels commencing at 139 min. At approximately this same time, the reactor operators recognized coolant was being lost through the faulty PORV and manually closed the pressurizer block valve upstream of the PORV, thus temporarily terminating coolant loss and the release of gaseous fission products to the containment.

Core temperatures continued to increase between 150-165 min. Peak core temperatures probably reached at least 1500 K between 150 and 165 min. By 165 minutes, core cooling water probably dropped to within 1 m of the bottom of the core. This drop would have resulted in downward, as well as radial, progression of core damage.

As shown in Figure 2, the RCS pressure increased approximately 5.5 MPa between 150 and 174 minutes. This increasing RCS pressure was being driven primarily by the continued progression of core damage. Zircaloy cladding oxidation at temperatures above 1500 K was probably vigorous, producing significant quantities of hydrogen that subsequently degraded heat transfer in the steam generator to the secondary system, and also rapidly heated the core above temperatures required to melt control rods (1700 K) and fuel rod cladding (2150 K).

Eutectics of Ni-Zr and Fe-Zr would be the first liquid formed as a result of interaction between the Inconel grid spacers and zircaloy cladding, probably near the center of the core. The stainless steel control rod cladding would melt as soon as temperatures exceeded approximately 1700 K, releasing molten Ag-In-Cd control material (melting point 1100 K) and allowing it to flow to the liquid/steam interface and freeze with the Ni-Zr and Fe-Zr eutectics. Analysis of lower crust samples reveals that the crust is a Zr-Ag-In-Fe-Ni metallic mixture surrounding standing columns of fuel pellets.

The process of zircaloy oxidation, melting of core metallic components, UO_2 dissolution, and relocation of molten material downward to freeze and block coolant flow channels at or near the coolant/steam interface would continue throughout this phase of the accident. This process would be expected to gradually progress outward to the periphery of the core, forming essentially a complete blockage of the central region of the core by 174 min.

Above the lower crust a region of (at least partially) molten metallics and ceramics formed. Core heatup calculations indicate peak temperatures within this region of consolidated core materials may have reached fuel melting (i.e., 3100 K) and the average temperature of the material was probably between 2600 and 2800 K.

Undamaged rod stubs below the bottom crust indicate that coolant levels did not drop below 0.5 m for any significant period of time during the accident. Water covering the bottom of the core ensured that the lower supporting crust would be cooled and probably be structurally stable throughout the accident.

The dominant release mechanism for the volatile fission products during the initial core heatup between 100 and 174 min was diffusion of gaseous fission products through the fuel matrix until the fuel was liquefied by metallic zircaloy [7].

B-loop Pump Transient (174-180 min)

Activation of the 2B coolant pump at approximately 174 min resulted in the first significant addition of coolant to the reactor vessel following the A-loop pump shutdown at about 100 min. The 2B-pump was allowed to operate approximately 19 min; however, significant flow in the B-loop hot leg was measured only for the first 15 s. The reactor cooling system experienced a rapid repressurization as a result of the pump transient, as shown in Figure 2.

The thermal-mechanical forces resulting from partial quenching of the oxidized rod remnants in the top half of the core are believed to have fragmented the oxidized cladding and fuel pellets to form the upper debris bed. The measured fission product release from the upper core debris bed was somewhat higher on average than that calculated by diffusional release from fuel particles. The debris particles experienced extensive intergranular fracturing, most likely during formation of the debris bed during the 2B-pump transient. Fuel fragmentation during a rapid reflood has been correlated with enhanced release of fission products trapped at grain boundaries [8] and was observed during the PBF severe fuel damage scoping test [9].

Degraded Core Heatup (180-200 min)

During this period, core liquid decreased as decay heat from the degraded core boiled liquid from the reactor vessel. The low thermal diffusivity of the consolidated (primarily ceramic) core material, together with the region's relatively large size, would prevent the interior from cooling, even if the reactor vessel was filled with water. Degraded core heatup calculations indicate that a pool of molten material formed in the center of the consolidated region and increased in size during this period.

Degraded Core Heatup (200-224 min)

At 200 minutes, the high-pressure injection system was actuated and emergency cooling water was injected for the next approximately 17 min. As emergency cooling water filled the reactor vessel, water began to penetrate the upper debris bed, cooling and eventually quenching the debris bed. Calculations by Kuan [10], based on estimates of the debris bed mass, initial temperature, and particle size distribution from debris bed examination data, indicate the debris bed was coolable, i.e., heat generation within it was less than dryout heat flux. Subsequent cooldown and quenching of the debris bed is estimated to have taken approximately 25 min. Thus, final quenching of the debris bed is estimated to have occurred by approximately 230 min.

The consolidated region would have continued to heat up during this period even though the reactor vessel was probably filled with water. Most consolidated core material is estimated to have been molten and to have continued to heat up; water is estimated to have covered the core; and the upper debris bed is estimated to have been nearly, if not fully, quenched.

Fission product release during this phase of the accident was probably dominated by the thermal and hydrodynamic behavior of molten corium, by volatility of fission products, and by dynamics of bubble formation, coalescence, and mobility [7]. Calculations indicate that release of volatile fission products was probably rapid (approximately 5 min), due to formation of large buoyant bubbles from small fission gas bubbles as a result of bubble coalescence and rise and of convective mixing within the molten pool. These large gas bubbles would have been trapped inside the crust until crust failure. The medium and low volatile fission products were estimated to remain as oxides or metallic phases in the melt. Little release is expected because of the low vapor pressure and low mole fraction of these materials within the pool.

Crust Failure and Material Relocation into the Lower Plenum (224-226 min)

Relocation of about 20 metric tons of molten corium into the lower plenum of the reactor vessel occurred at approximately 224 min after reactor scram. The A-loop cold leg temperature increased to a maximum in less than 20 s, while the B-loop temperature reached a maximum in approximately 1 minute. The rapid increase in cold leg temperature was probably due to the flow of steam generated in the lower plenum. The RCS pressure also increased approximately 1.8 MPa (260 psi) between 223.5 and 225.5 min. This increase was probably also caused by steam rapidly produced during this limited time period. The rapid steam production occurs as a result of heat transfer from the molten core material to the water. Nothing in recorded data or post-accident core condition suggests an energetic steam explosion occurred as the tons of molten corium relocated into the lower plenum with the reactor vessel essentially full of water.

The crust failure appears to have been in the upper half of the consolidated region near the core periphery in the southeast quadrant of the reactor vessel, and material apparently flowed downward into the lower plenum through both upper CSA and peripheral fuel assemblies. Continued heating of the molten pool could lead to melting of the supporting crust. Thermal analysis that considered a range of initial and boundary conditions was performed and indicated that the steady state thickness of the supporting crust would be 6 to 10 cm at the bottom and only about 1 cm at the top. Heat transferred from the molten pool primarily to the top crust, and this accounts for a thinner top than bottom crust. The calculated steady state crust thicknesses are consistent with observed crust thicknesses obtained from core bore samples. A 1 cm crust would probably fail from the overlying weight of debris.

Another important aspect related to core relocation is interaction between molten corium and water in the reactor vessel. Both fine particulates and large structures of previously molten debris have been observed in the lower plenum, suggesting that some degree of interaction occurred between the molten material and the coolant during the relocation into the lower plenum.

Release of volatile fission products during molten corium relocation was probably governed primarily by breakup and fragmentation of the molten stream as it penetrated water in the lower CSA and lower plenum; temperature of the molten corium; fission product concentrations and chemical species; and the high oxidation potential of the steam environment. Results of examination of ceramic samples from the central consolidated region and lower plenum debris indicate that Sb and Ru concentrate in the molten metallic debris and that both I and Cs are retained to some extent in the previously molten ceramic debris, i.e., $(U,Zr)O_2$.

RCS Recovery Period (226 min-15.5 h)

The molten corium (ceramic and metallic phases) rapidly settled onto the reactor vessel lower head and was probably not cooled significantly by water during relocation. The RCS pressure pulse between 224 and 240 min, implies that heat transfer and steam generation within the debris bed was significant for at least 15 min. Re-formed thermocouple junctions probably embedded in the debris indicate temperatures greater than 1000 K existed within the debris bed up to three days. These high temperatures and the high RCS pressure may have seriously challenged the structural integrity of the reactor vessel lower head. However, the lower head was never breached. At 15.5 h after reactor scram, forced coolant flow through the core was reestablished upon the restart of one of the A-loop primary coolant pumps.

Calculation of the thermal/mechanical response of the reactor vessel lower head was performed using a bounding range of debris bed configurations and assumed a top down quench in 20 min. These calculations indicate that thermal stress in the lower head dominated and that some limited strain may have occurred in the structure's outer regions. This analysis indicates that cooling of the inside surface of the reactor vessel lower head must have been reestablished within about 30 minutes after the relocation. Otherwise, head temperatures would have exceeded 1100 K and creep failure of the lower head would have likely occurred. Despite the fact that this relatively simple analysis establishes an upper bound on vessel temperature, it provides little insight into thermal and mechanical processes that would have challenged the integrity of the reactor vessel lower head. The actual configuration and structure of lower plenum debris and detailed metallographic examinations of the lower head are needed to reduce the uncertainties in these analyses and define the margin-of-safety of the lower head.

STANDARD PROBLEM EXERCISE

As a means of assessing and improving the computer codes that are used to describe severe reactor accidents, the DOE Program also supports a Standard Problem Exercise. A Joint Task Group has been established in collaboration with the OECD Nuclear Energy Agency (NEA) to analyze various periods of the accident. At this time, the Joint Task Group includes analysis with eight different computer codes by representatives of eleven different organizations in eight countries. A detailed report of the analysis exercise comparing calculated results with measurements from the TMI-2 accident will be published on completion of the exercise in 1990; preliminary results are given in Reference 6.

Since there remains some uncertainty in the boundary conditions, it is not unreasonable that the codes produce results (e.g., in calculated reactor coolant system pressure) that deviate from the recorded data. While the calculated results for both reactor coolant system pressure and core temperatures are qualitatively similar to the accident data, the calculated results are quantitatively quite different from the data. Thus it has been concluded that although the codes have the capability to predict severe accidents, there is need for continued refinement and development.

INFORMATION AND INDUSTRY COORDINATION

The importance of the TMI-2 accident to the world nuclear industry was recognized early by the Accident Evaluation Program. To help achieve the maximum benefit of the lessons learned from the accident, the AEP includes a special task to collect, organize, and disseminate information about the accident and the results of the plant examinations and analyses to the world nuclear community, the news media, and other interested persons. This effort has provided many useful communications over the past few years and has culminated with the organization of the ANS Topical Meeting on the TMI-2 accident in Washington, D.C. in November 1988. Over 130 papers and two panel discussions were included in the meeting. Paper summaries appear in the ANS transactions [2], and the full papers from this meeting will be published as a series of journal articles in Nuclear Technology. Expected publication dates are August-November 1989.

CONCLUSION

The accident at Three Mile Island Unit 2 resulted in severe fuel damage, much beyond that expected to occur during design-basis accidents. Most aspects of core and RCS thermal/hydraulic responses during the accident have been evaluated, starting from accident initiation at reactor scram, to establishment of coolant flow about 15.5 h after scram. The TMI-2 accident provides valuable insight into phenomena and processes controlling core melt progression and fission product behavior through the "late phase" of core melt progression.

Although not all aspects of the accident have been quantified with the degree of uncertainty desired from integral-effects experiments, the TMI-2 accident is the only source of data from a full size severe accident event for application to the resolution of technical issues. Progression of the accident to include a large scale thermal/mechanical challenge to the lower head is much beyond existing or planned experiments, both in duration and physical size. The first application of TMI-2 research results to technical issues has been the OECD/CSNI analysis exercise, which assesses severe accident analysis codes and methodologies.

Through the DOE TMI-2 research program, an inventory of core materials has been established to account for all major core components. The post-accident distribution of fission products within the plant has been determined and shows that, within the uncertainty of the determination, most fission products are accounted for. Application of these DOE research results is only beginning, and examination of reactor vessel lower head samples will be accomplished under sponsorship of the OECD/NRC cooperative research program. Thus, TMI-2 accident research will continue to provide new insight into reactor safety and severe accident phenomena for the next few years.

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