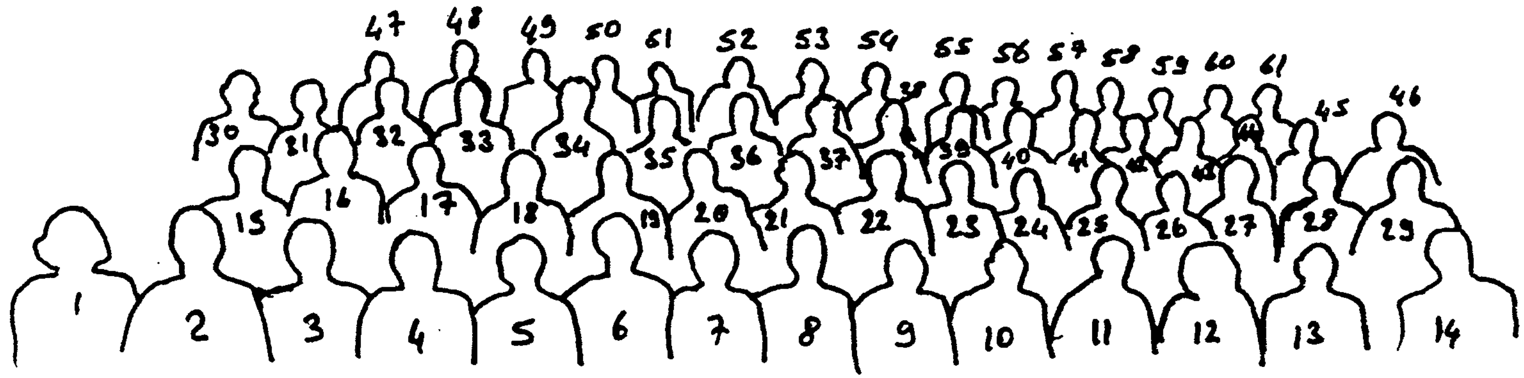

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HEAT AND MASS TRANSFER IN SEVERE NUCLEAR REACTOR ACCIDENTS

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Editor

begell house, inc.
New York, Wallingford (UK)

Library of Congress Cataloging-in-Publication Data

Heat and mass transfer in severe nuclear reactor accidents / J.T. Rogers, editor.
p. cm.
At head of title: International Centre for Heat and Mass Transfer.
Includes bibliograpical references and index.
ISBN 1-56700-059-2
1. Nuclear reactor accidents--Congresses. 2. Heat--Transmission-- Congresses.
3. Mass transfer--Congresses. I. Rogers, J. T. II. International Center for Heat and Mass Transfer.
TK9202.H37 1996
621.48'3--dc20 96-14788
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Contents

PREFACE

The International Centre for Heat and Mass Transfer was formed in 1968 with the objective of promoting and fostering international co-operation in the science of heat and mass transfer and its applications. The Centre accomplishes this mainly by organizing periodic meetings, usually at least two a year, on various specific topics in heat and mass transfer. These meetings take the form of Symposia, open meetings for which general calls for papers are issued, and Seminars, experts' meetings for which all presentations are invited. The Centre also undertakes other activities related to the field of heat and mass transfer.

Over the years, the Centre has held a number of meetings on various heat and mass transfer aspects of nuclear power. The most recent previous such meeting, a Seminar on Fission Product Transport Processes in Reactor Accidents, was held in Dubrovnik, in the then Yugoslavia in May, 1989. The Proceedings of that Seminar constituted a very useful state-of-the-art summary by invited experts on this important topic in nuclear reactor safety. The success of that Seminar prompted a decision by the Executive Committee of the Centre in 1992 to organize another experts' Seminar related to nuclear reactor safety.

In the years following the Three Mile Island and Chernobyl accidents, there has been a significant growth in research efforts on all aspects of "beyond design-basis" or severe reactor accidents, as countries with nuclear power programs realized the urgent need to ensure that such accidents did not occur again. The greater openness of communications between east and west in recent years has encouraged international efforts, including joint research programs, in this field. Considering the long-standing role of the Centre in promoting international co-operation and the key roles of heat and mass transfer phenomena in severe reactor accidents, the Executive Committee decided that it was most appropriate for the Centre to organize an experts' Seminar on Heat and Mass Transfer in Severe Nuclear Reactor Accidents.

The objective of the Seminar was to bring together, in a relaxed atmosphere, scientists and engineers involved in heat and mass transfer aspects of severe accidents in nuclear power reactors for in-depth discussion and review of the state-of-the-art in this field that would point the way to future work. The scope of the Seminar was to cover both fundamentals and applications.

The Seminar was planned and organized by an international Organizing Committee of experts in the field and was held in Cesme, Turkey, from May 21 to May 26, 1995. The Seminar was originally scheduled to be held in Dubrovnik in May, 1994, but because of the political situation in the former Yugoslavia and the necessity of re-locating the Secretariat of the Centre from Belgrade, a new site and date were selected. Eleven invited lectures and 41 invited papers were presented, two as poster papers, in four sessions. In addition, the Seminar concluded with a Panel composed of the session chairmen and others who summarized highlights of the sessions, conclusions reached and remaining issues. The Panel was aided in this endeavour by a lively discussion from the floor. Ten lectures and 38 papers are included in this volume. The volume also contains summaries of the presentations at the closing Panel as well as a summary of the discussion from the floor.

The Seminar was attended by 75 participants from 11 countries and 2 international organizations.

The Executive Committee of the Centre and the Organizing Committee of the Seminar acknowledge, with many thanks, the assistance provided to the Seminar by the sponsoring institutions: the Middle East Technical University, Ankara, the Scientific and Technical Research Council of Turkey, Ankara, the Canadian Nuclear Society, Toronto and Carleton University, Ottawa. The Seminar was organized

in co-operation with, and we are indebted to, the Turkish Atomic Energy Authority, Ankara, the International Atomic Energy Agency, Vienna, and the Nuclear Energy Agency of the Organization for Economic Co-operation and Development, Paris.

The efforts and co-operation of many people were instrumental in the organization and operation of the Seminar and the Production of this volume. Key assistance was provided by the two vice-chairmen of the Seminar: T.P. Speis, US Nuclear Regulatory Commission, Washington, D.C., USA, who also chaired the closing Panel, and V.A. Siderenko, Ministry for Atomic Energy of the Russian Federation, Moscow, Russia. Major contributions were made by, and special thanks go to, the chairmen and co-chairmen of the Sessions: D.A. Meneley, Atomic Energy of Canada Ltd., Mississauga, Ontario, Canada; L.A. Simpson, Atomic Energy of Canada Ltd., Pinawa, Manitoba, Canada; H. Unger, Ruhr-Universitat-Bochum, Bochum, Germany; M.L. Corradini, University of Wisconsin-Madison, Madison, Wisconsin USA; S.R. Kinnersly, AEA Technology, Dorchester, UK; J. Sugimoto, JAERI, Tokai-mura, Japan; K. Abe, JAERI, Tokyo, Japan.

Other members of the Organizing Committee who also assisted in the arrangements for the Seminar are: V.G. Asmolov, Kurchatov Atomic Energy Institute, Moscow, Russia; U. Brockmeier, Ruhr-Universitat-Bochum, Bochum, Germany; M. Courtaud, CEA, Grenoble, France; I. Dunbar, AEA Technology, Warrington, UK; G. Lowenhielm, Vattenfall, Sweden; D.F. Ross, USNRC, Washington, D.C., USA; K. Soda, JAERI, Tokyo, Japan.

Major support was given, both before and during the Seminar, by the Secretariat of the Centre, now located at the Middle East Technical University (METU), Ankara, Turkey, under the able direction of Faruk Arinc, the Secretary-General of the ICHMT. Special thanks are due to Dr. Arinc and to Semiha Alaybeyi of the Secretariat as well as Umit Coskun and Ilkay Yigit of METU.

Finally, my personal thanks go to Ms. Christie Egbert of the staff of the Department of Mechanical and Aerospace Engineering, Carleton University, Ottawa, Ontario, Canada for her efficient and enthusiastic assistance during the organization of the Seminar and the preparation of this volume.

J.T. Rogers
Seminar Chairman and
Editor, Proceedings

SESSION 1: OVERVIEW OF SEVERE ACCIDENTS IN NUCLEAR POWER PLANTS (IMPACTS OF TMI AND CHERNOBYL)

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TMI-2 VESSEL INVESTIGATION: RESULTS, CONCLUSIONS AND IMPLICATIONS FOR ACCIDENT MANAGEMENT STRATEGIES

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ABSTRACT. In October 1987, the U.S. Nuclear Regulatory Commission proposed that a joint international cooperative program be formed that would be sponsored by the Nuclear Energy Agency of the Organization for Economic Cooperation and Development (NEA-OECD) to conduct further investigations of the potential damage to the TMI-2 reactor vessel lower head during the accident. This proposal was accepted by NEA-OECD and the project was initiated in 1988. The program was completed five years later. This paper discusses the results from this program and the related conclusions. It specifically addresses the findings regarding the extent and type of damage to the vessel lower head and the margin of structural integrity that remained in the vessel during the accident; the conditions for potential failure (which apparently were never reached); enhanced cooling mechanisms of the debris and lower head not currently accounted in severe accident analyses; and insights for severe accident management strategies. Finally, the results/conclusions from the program have pointed to a number of areas where further studies, including appropriate experiments, can add to our understanding of the conditions under which the reactor vessel can be cooled and its integrity maintained during a severe accident. Such efforts are already underway and they will be addressed in this paper, including their specific goals and significance.

BACKGROUND

In October 1988, the Nuclear Regulatory Commission (NRC), in cooperation with 10 other countries under the auspices of the Organization for Economic Cooperation and Development's (OECD) Nuclear Energy Agency, began a joint research program to examine and analyze material samples from the lower head of the TMI-2 reactor pressure vessel. The objectives of this program, called the TMI-2 Vessel Investigation Project (VIP), were to (1) investigate the condition and properties of materials extracted from the lower head of the TMI-2 reactor pressure vessel, (2) determine the extent of damage to the lower head by chemical and thermal attack, and (3) determine the margin of structural integrity that remained in the pressure vessel.

Prior to the initiation of the VIP, the Department of Energy (DOE) had supported extensive post-accident examinations and analyses of the TMI-2 damaged core. The primary objectives of the DOE TMI-2 Accident Evaluation Program were to develop an understanding of (1) core damage progression in the upper core region, (2) heat-up and the formation and growth of the molten central region of the core, (3) relocation of approximately 20 tons of debris to the lower head, and (4) release of fission products to the reactor vessel and the containment.

The principal conclusions from the DOE program were that the TMI-2 core damage progression involved the formation of a large consolidated mass of core material surrounded by supporting crusts, the failure of the supporting crusts, and finally, the long-term cooling of a large volume of molten core material. On the basis of this program, knowledge of the end-state condition of the TMI-2 reactor vessel and core is shown in Fig. 1.

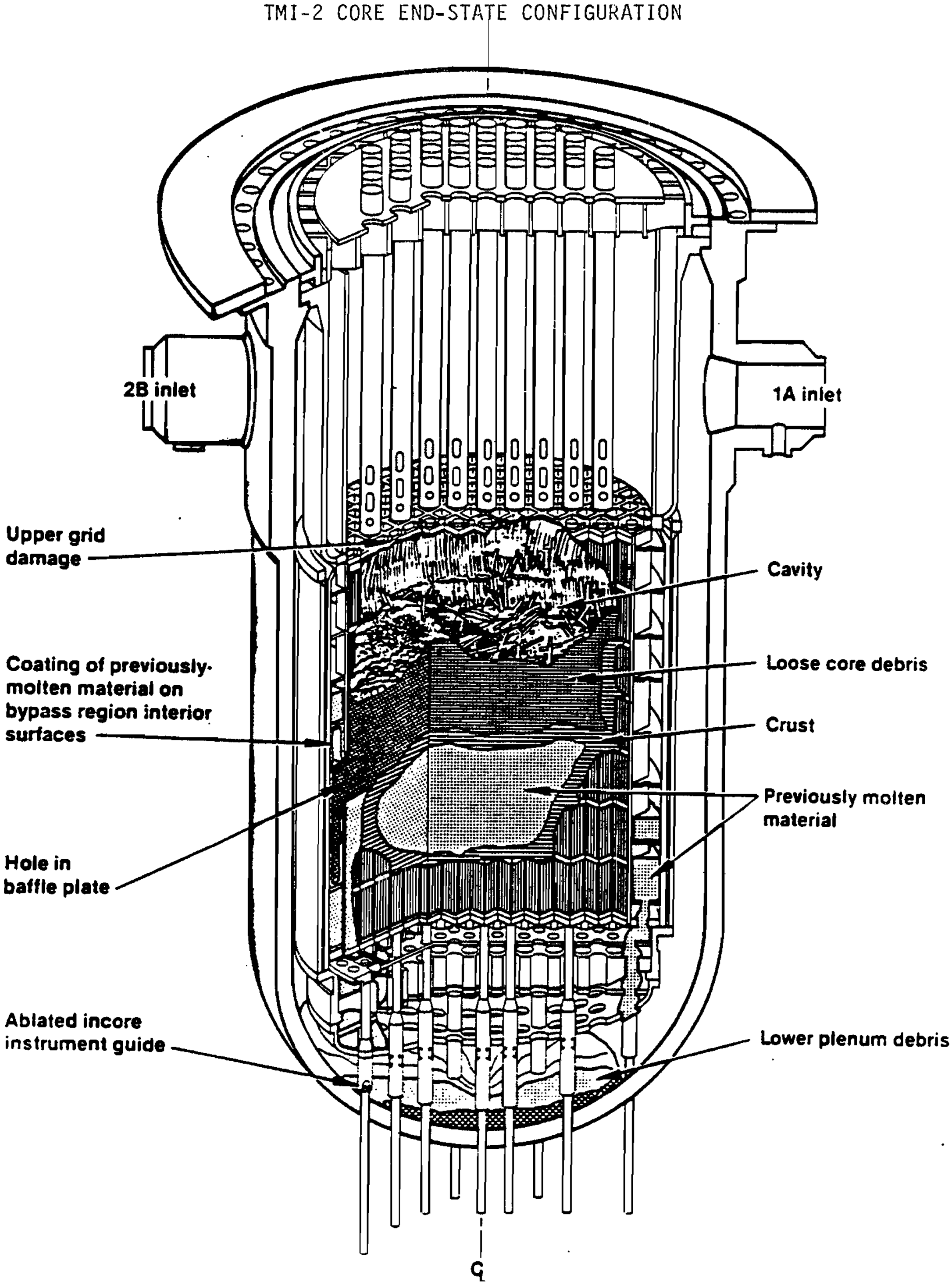

Figure 1. TMI-2 Core End-State Configuration

This information from the DOE program plus the subsequent performance of many small scale tests, both in-reactor as well as out-of-reactor (as many as 50 tests under both types of conditions), plus extensive material studies, has contributed substantially to our understanding of how severe accident evolves in a light waster reactor (LWR) (somewhat biased to a PWR). The picture which has been constructed from this understanding involves incoherent melting and downward relocation and freezing, initially of the metallic material, in a somewhat repetitive fashion for while formation of and growth of a pool of molten core material surrounded by a crust, and in the case of the TMI-2 accident scenario subsequent failure of the crust supporting the molten pool after some time following the injection of emergency core coolant system water, and finally the flow of molten material to the bottom of the vessel. Recall that our previous understanding, exemplified by the modeling in the WASH-1400 study (first extensive risk study involving LWR's) involved simple coherent melting and slumping of the core into the vessel lower head following fuel uncovery and heat-up. This TMI-2 based understanding whose main principle involves the incoherent motion (both in space and time) of molten material has profound implications for the subsequent evolution of an accident in terms of challenging both the reactor pressure vessel and the containment (e.g., conditions for steam explosions).

The TMI-2 accident finally demonstrated that at least for one severe accident scenario, the accident can be terminated and confined to the reactor pressure vessel by cooling water before the failure of the lower head. However, there was no quantitative information that could be used to determine how close the vessel was to failure.

The examinations that were performed under the OECD NEA VIP go beyond the work performed during the previous TMI-2 examinations. Specifically, the VIP plan was to obtain and examine samples of the lower head steel, instrument penetrations, and previous molten debris that was attached to the lower head and use this information to estimate the vessel margin-to-failure. The major conclusions and accomplishments of the key elements of the VIP plan are presented below.

VESSEL STEEL EXAMINATION

Argonne National Laboratory (ANL) in the United States coordinated the metallographic examinations and mechanical property tests of the vessel steel samples. All the lower head steel samples were visually examined, decontaminated, sectioned, and sent to eight of the VIP member countries for testing. The participants that examined the vessel steel samples were Belgium, Italy, Finland, France, Germany, Spain, the United Kingdom, and the United States. Examination performed by the project participants included tensile, creep, and Charpy V-notch impact tests, microhardness measurements, micro and macro photography, and chemical composition. The primary purpose of these tests was to determine the lower head steel's mechanical properties over the temperature range experienced during the accident. Optical metallography and hardness tests were performed in order to estimate the maximum temperature various portions of the lower head reached during the accident.

The results of the wide range of inspections, mechanical property determinations, and metallographic examinations of the lower head vessel samples revealed important and previously unknown facts relating to the degree of thermal attack on the lower head. Overall, these examinations revealed that a localized hot spot formed in the elliptical region on the lower head that was approximately 91.5 cm x 76 cm, as shown in Fig. 2.

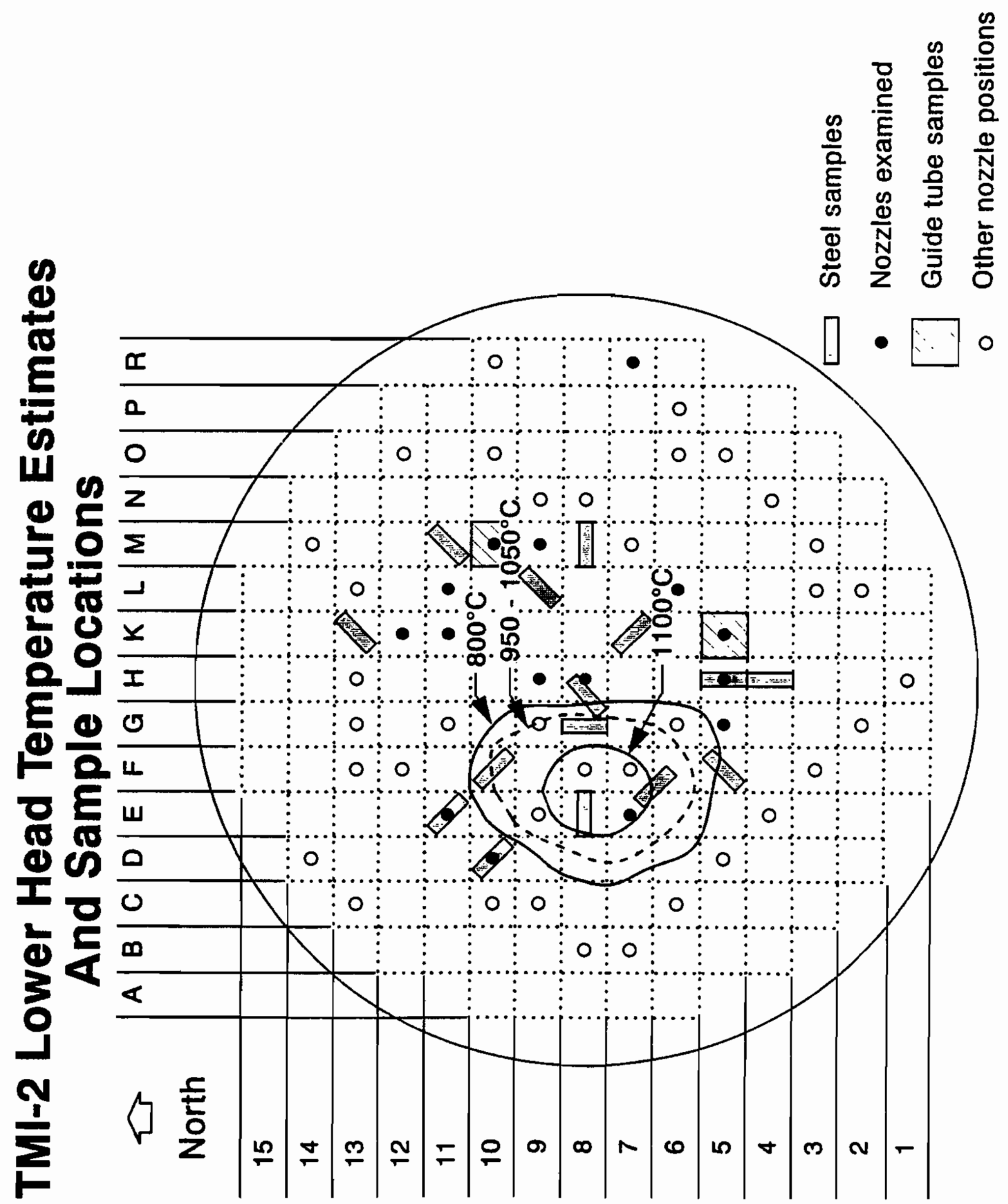

Figure 2. TMI-2 Lower Head Temperature Estimates and Sample Locations

The hot spot was located in the area where visual observations made during the defueling process indicated that the most severe nozzle damage had occurred. Metallographic examinations of samples taken from this region indicated that the inner surface of the vessel steel reached temperatures between 1075 and 1100°C during the accident. At this location, temperatures 4.5 cm into the vessel wall were estimated to be 100 ± 50° lower than the peak inner surface temperature.

Temperatures in the hot spot were considerably higher than the surrounding region of the lower head. Generally, the vessel temperature away from the hot spot did not exceed the 727°C ferrite-austenite transformation temperature for the A533B pressure vessel steel. The results of metallographic examinations could not determine an estimate f or the steel temperature away from the hot spot, only that the 727°C transition temperature was not reached.

The steel examinations were also able to provide data on the cooling rate of the lower head hot spot. Microstructural and hardness observations in the as-received state for two samples in the hot spot reflected the austenitizing head treatment and the subsequent relatively rapid cooling of this material during the accident. Cooling rates were estimated to have been in the range of 10 to 100°C/min through the transition temperature. By comparing results of the TMI-2 lower head sample examinations with results from similar metallurgical examinations of heat-treated samples from an equivalent steel, it was determined that samples in the hot spot remained at their peak temperature for as long as 30 minutes prior to being cooled.

Mechanical property tests performed on the vessel steel samples produced a wealth of high temperature mechanical property data, which not only provided information on the present condition of the lower head, but also provided input to the margin-to-failure analysis. Creep tests performed at 600 to 700°C indicated no significant differences in behavior between samples that exceeded a maximum temperature of 727°C and those that did not. Tensile tests for specimens that exceeded 727°C showed significantly higher strengths at room temperature and at 600°C when compared to those that did not exceed 727°C. The tensile tests at lower test temperatures further confirmed the hardness measurements which showed that the material from the hot spot had been austenitized and subsequently cooled rapidly.

During the sample removal effort, tears or cracks were found in the cladding of the vessel around four nozzles. Vessel steel samples containing these cracks were analyzed by ANL. It was found that the crack did not propagate into the base metal. The cracks were attributed to hot tearing of the cladding caused by differential thermal expansion between the stainless steel cladding and the carbon steel vessel that occurred during vessel cooling. Furthermore, the presence of control assembly material (Zr, Ag, Cd, and In) within the cladding tears and into the grain boundaries on the surface of some sample locations indicated that a layer of debris containing metallic material was already present on the lower head when the major relocation of ceramic molten core material to the lower head took place at 224 minutes after the initial reactor scram.

The results of a metallographic and hardness examinations could determine whether the 727°C transition temperature in the steel was exceeded. However, because microstructural and associated hardness changes in the steel do not occur below 727°C, it was not possible to estimate how far below 727°C the vessel steel temperature was away from the hot spot. Therefore, there is a large uncertainly in the actual vessel steel temperature away from the hot spot. The temperature of the vessel inner surface, in this region, during the accident could have ranged from a minimum of 327°C (normal plant operating conditions) to a maximum of 727°C.

NOZZLE EXAMINATIONS

Fourteen nozzles and two guide tube specimens were removed from the vessel by being cut off as close to the lower head as possible. Several nozzles were melted off almost flush with the vessel and could not be removed. Examinations included micro and macro photography, optical metallography, scanning electron microscope measurements, gamma scanning, melt penetration measurements, and microhardness. There were two primary purposes for these examinations. First these examinations would help to determine the extent of nozzle degradation to evaluate the thermal challenge to the lower head. Second, they would provide information on the movement of molten fuel onto and across the lower head during the relocation.

Examinations performed on the nozzles and guide tubes, conducted primarily at ANL, provided insights into the accident progression. Damage to several nozzles indicated that their end-state condition was caused by molten core material coming in contact with the nozzles at an elevation ranging from 140 to 270 mm above the lower head. Surface scale found on the nozzles below their melt-off points suggested that this molten material flowed on top of a crust of preexisting solidified fuel debris that had been cooled below its solidus temperature.

During the examinations, it was estimated that nozzle temperatures varied widely as a function of location and elevation above the lower head. They ranged from 1415°C, which is the Inconel 600 nozzle's liquidus temperature, to 1000°C at elevations of 140 mm and 64 mm above the lower head, respectively. The penetration of fuel debris downward into the nozzles was influenced by the temperature of the fuel at the time of entry, debris composition (and hence its fluidity), and the temperature of the nozzle itself.

Examination results also indicated the presence of Zr and Ag-Cd on nozzle surfaces, which interacted with the material. The presence of this material indicated that control rod material had relocated prior to the primary fuel relocation. However, the examinations were unable to determine the quantity of these materials that had relocated.

COMPANION SAMPLE EXAMINATIONS

The debris samples examined as part of the VIP were known as companion samples since they came from the same hard layer that was in contact with the lower head. Hence, they were "companions" to the lower heat steel samples. Results of the companion sample examinations made it possible to determine the debris composition and the lower head decay heat load. During the defueling process, it was discovered that the hard layer was indeed extremely hard and had to be broken into pieces for removal. However, there was virtually no adherence of the material to the lower head itself. Because the hard layer had to be broken into pieces during sample acquisition, information on the sample location was limited to identifying from which quadrant the sample was obtained.

The primary constituents of the companion samples were uranium, zirconium, and oxygen $(U, Zr)O_2$ with only small percentages (<1 wt%) of other structural material such as iron, nickel, and chromium. Control rod materials such as silver, indium, and cadmium were present in low (<0.5 wt%) concentrations. The average sample debris density was 8.2 ± 0.6 g/cm^3 with an average porosity of $18 \pm 11\%$. Based on the debris composition, it is quite probable that the molten material reached temperatures greater than 2600°C in the central core region prior to relocation. The temperature of the debris when it reached the lower head is not known, but the material reached the lower head in a molten state, and results of the examinations suggest that portions of the debris cooled slowly over many hours.

Radiochemical examinations indicated that the primary radionuclides retained in the debris bed were medium and low volatile constituents. Almost all the radiocesium, radioiodine, and radioactive noble gases volatilized from the molten core before it relocated to the lower head. Knowledge of the retained fission products is critical to calculating the debris decay heat and the resulting heat load on the lower head. Decay heat calculations indicated an overall heat load of $0.13 \pm 20\%$ W/g of debris when the relocation occurred at

224 minutes after scram and 0.096 ± 20% W/g at 600 minutes after scram. At the time of relocation, the total decay heat load was approximately 2.47 MW for the estimated 19,999 kg of material that relocated to the lower head.

MARGIN TO FAILURE ANALYSIS

The final element of the VIP, the margin-to-failure analysis, was performed to investigate mechanisms that could potentially threaten the integrity of the reactor vessel and to help improve understanding of events that occurred during the accident. Analyses addressed mechanisms that could result in lower-head penetration tube and vessel failures. Specific failure modes examined were instrument tube rupture, tube ejection, localized vessel failure, and global vessel failure.

Margin-to-failure calculations relied upon three major sources of VIP examination data: (1) nozzle examination data for characterizing melt composition and penetration distances within instrument tubes: (2) companion sample examination data for characterizing debris properties (e.g., decay heat and material composition); and (3) vessel steel examination data for characterizing peak vessel temperatures, duration of peak temperatures, and vessel cooling rate.

The margin-to-failure analyses provided significant insights into potential failure mechanisms of the TMI-2 lower head. Results of these calculations eliminated tube rupture and tube ejection as potential failure mechanisms during the accident.

Calculations indicated that the magnitude and duration of hot spot temperatures estimated in TMI-2 vessel examinations could not have been caused by an impinging jet. Rather, hot spot temperatures were due to a sustained heat load from debris on the lower head.

Because of insufficient available data, it was not possible to come up with a best-estimate quantification of the margin to failure for global or local creep rupture of the lower head. Such failures would be associated with high temperatures on the lower head coincident with high reactor coolant system pressure. However, an extensive series of analyses and calculations was performed with the best available information to try to scope the issue as described below.

The potential for the vessel to experience a global failure was evaluated for temperature distributions obtained from thermal analyses with best-estimate and lower-bound input assumptions for such parameters as debris decay heat, outer vessel heat-transfer coefficient, and the debris-to-gap heat-transfer resistance. Calculations for both of these cases indicated that global failure caused by creep rupture was predicted to occur within the first 2 hours after debris relocation because of the sustained high vessel temperatures when the RCS was repressurized. This rise in RCS pressure occurred when the plant operators closed the block valve for the power-operated relief valve at 320 minutes after reactor scram.

Localized vessel failure analyses indicated that it is possible to withstand the 1100°C hot spot temperatures for the 30-minute time period inferred from the vessel steel examinations provided that the rest of the vessel (i.e., outside the area of the hot spot) remained relatively cool. Localized calculations also indicated that the predicted time to vessel failure was reduced when a localized hot spot was superimposed on the calculated best-estimate background temperature (i.e., outside the hot spot).

Taken together, the localized and global vessel failure calculations indicated that the background vessel steel temperature behavior, which greatly depends on the heat load from the relocated debris in the lower head, was key to predicting failure from either of these mechanisms. Cool background vessel temperatures can potentially reduce structural damage and preclude global vessel failure even at high pressure and in the presence of a localized hot spot.

Thermal and structural analysis results were dominated by input assumptions on the basis of companion sample examination data, which suggested that the debris experienced relatively slow cooling over a period of many hours. However, differences between these analysis results and data from the vessel steel examinations indicated that the entire lower head cooled within the first 2 hours after debris relocation.

Although there are insufficient data to quantitatively determine the exact mechanisms that caused this cooling, scoping calculations were performed to investigate possible mechanisms that could provide this cooling. In these analyses it was assumed that the simultaneous presence of cracks and gaps within the debris provided multiple pathways for steam release (e.g., water may travel down along the gap and boil up through cracks). Results of these calculations indicated that a minimal volume of cooling channels within the debris and the vessel could supply the cooling needed to obtain vessel temperatures and cooling rates determined in metallurgical examinations. The uncertainties in the amount of debris cooling on the lower head appear to be more significant for quantifying the margin-to-failure of TMI-2 vessel than uncertainties in either the vessel failure criterion or cooling of debris as it relocates to the lower plenum. Because of these uncertainties, results of the margin-to-failure analysis should be viewed as providing insights into areas such as identifying the failure mode with the smallest margin during the TMI-2 event and emphasizing areas in which additional research may be needed in severe accident analysis.

In summary, the postulated rapid cooling of the debris on the lower head is consistent with the steel examinations showing rapid vessel cooling, consistent with energy balance calculations showing a decrease in debris internal energy between the time of debris relocation and vessel depressurization, consistent with plausible postulated gap/crack configurations within the debris, consistent with the rapid cooling assumptions utilized in the structural analysis with the vessel not failing (which was indeed the case), and only not consistent with the companion sample examinations.

CONCLUSIONS AND SIGNIFICANCE OF THE VIP FINDINGS

The principal findings and conclusions from this project, as well as the implications for safety from these findings are summarized below:

- Vessel steel examinations indicated that a localized hot spot developed in an elliptical region approximately 1 m by 0.8 m. In this region, the maximum temperature of the ferritic steel base metal near the interface with the stainless steel cladding was approximately 1100°C. The steel may have remained at this temperature for as long as 30 minutes before cooling occurred. Temperatures 4.5 cm into the 13.7 cm-thick wall were estimated to be 100 ± 50°C lower than the peak surface temperatures. Away from the vicinity of the hot spot, lower-head temperatures did not exceed the 727°C transformation temperature.

- Analyses results indicated that a localized effect, such as a hot spot, can shorten the overall vessel failure times caused by creep rupture. However, by itself, the hot spot is unlikely to cause vessel failure for the temperatures and pressures that occurred in the vessel during the TMI-2 accident.

- Without modeling enhanced cooling of the debris and lower head, the margin-to-failure scoping calculations indicated that lower head temperature distribution based upon data from companion sample examination data would have resulted in vessel failure when the reactor system was repressurized by plant operators at about 300 minutes after reactor scram.

- Even though a definitive scenario describing the movement of molten debris and the formation of a localized hot spot cannot be determined, considerable evidence indicates that a debris layer containing primarily ceramic and some metallic material insulated the lower head. The hot spot formed in a location where this layer had insufficient thickness to effectively insulate the lower head.

- The experience at TMI-2 validates the importance of accident management and perseverance in a strategy of delivering cooling water. It is also clear now as a result of the VIP that the reactor vessel provided a previously unrecognized defense-in-depth for a severe accident that was, of course, essential to success. Molten core material reaching the lower head is not synonymous with vessel failure, as is generally being assumed in probabilistic risk assessments (PRA).

Based on the VIP findings, additional research is being performed to further examine the conditions under which reactor vessel integrity is likely to be maintained during a severe accident. For example, cooling the external reactor vessel by flooding the cavity surrounding the lower part of the reactor vessel could reduce the potential for reactor vessel failure. The Nuclear Energy Agency of the OECD is sponsoring (with the NRC as a participant) the RASPLAV project in Russia to carry out an integrated program of experiments and analyses to more fully examine the conditions under which a degraded/molten core can be retained inside the reactor pressure vessel via cooling of the vessel from the outside. NRC is also initiating a research project to examine potential mechanisms and modes of in-vessel cooling of debris by water on the lower head. Finally, accident management procedures should recognize the need to control primary system pressure in order to avoid vessel creep failure during a severe accident.

Acknowledgements. This summary paper is based on the work performed in the TMI-2 Vessel Investigation Project (VIP) under the auspices of the Organization for Economic Cooperation and Development's (OECD) Nuclear Energy Agency. Substantial input to this paper was provided by Dr. A.M. Rubin and Mr. E. Beckjord of the U.S. Nuclear Regulatory Commission. Reports on more detailed technical aspects of the VIP project are documented in: J.R. Wolf et al., TMI-2 Vessel Investigation Project Integration Report, Report NUREG/CR-6197 (EGG-2734), March 1994.

REFERENCES

1. S.R. Kinnersly, et.al., "In-Vessel Core Degradation in LWR Severe Accidents: A State of the Art Report to CSNI," NEA/CSNI/R (91)12 (1991).

2. J.M. Broughton, P. Kaun, D.A. Petti, and E.L. Tolman, "A Scenario of the Three Mile Island Unit 2 Accident," Nucl. Technol., 87, pp. 34 - 53, (Aug. 1989).

3. R.W. Wright, "Melt Progression Modeling Implications of the TMI-2 Accident," in 'Fission Product Transport Processes in Reactor Accidents, Proc. of the International Centre of Heat and Mass Transfer International Seminar, Dubrovnik, May 22 - 26, 1989', J. Rogers ed., pp. 833 - 846, Hemisphere Pub. Co., New York, (1989).

4. P. Hofmann et.al., "Lessons Learned from the CORA Program," Trans 21st Water Reactor Safety Information Meeting, Bethesda, MD Oct. 25 - 27, 1993, NUREG/CP-0132.

5. R.R. Hobbins and G.D. McPherson, "Summary results of the LOFT LP-FP-2 test and their relationship to other studies at PBF and TMI-2," LOFT Madrid Conference Report (May 1990)

6. C.S. Olsen, S.M. Jensen, E.R. Carlson and B.A. Cook, "Materials Interactions and Temperatures in the Three Mile Island Unit 2 Core," Nucl. Technol., 87, pp.57 - 94, (Aug. 1989).

7. L.A. Stickler et.al., "Calculations to Estimate the Margin to Failure in the TMI-2 Vessel," NUREG/CR - 6196 (1994).

8. S.A. Hodge, J.C. Cleveland, T.S. Kress, and M. Petek, "Identification and Assessment of BWR. In-Vessel Severe Accident Mitigation Strategies," NUREG/CR-5869 (1992).

9. S. Hagen et.al., "Dry Core BWR Test CORA-33: Test Results," KfK 5261 (1994)

10. R. O. Gauntt et.al., "Metallic Core-Melt Behavior in Dry-Core BWR Accidents: The Ex-Reactor Experiments," Trans. 22nd Water Reactor Safety Information Meeting Bethesda, MD, Oct. 23 - 26, 1994, (to be published).

11. R.D. Gasser, R.O. Gauntt, and S. Bourcier, " Late Phase Melt Progression Experiment MP-1: Results and Analysis," NUREG/CR-5874 (1993).

12. R.D. Gasser et.al., "Late Phase Melt Progression Experiment MP-2: Results

13. R.O. Gauntt, "Lessons Learned from In Vessel Melt Progression Research Performed at Sandia National Laboratories," Trans. Am. Nucl. Soc., 69, pp. 317 - 318, (Nov. 1994).

14. R.C. Schmidt, private communication.

15. K. Kelkar, private communication.

16. R.C. Schmidt and R.D. Gasser, "Models and Correlations of the DEBRIS Late Phase Melt Progression Model," SAND 95-3922 (1994).

17. R.C. Schmidt and L.L. Humphries, "Late Phase Melt Progression Scoping Analysis and Sensitivity Studies for PWRs," NUREG/CR - 6296 (1995).