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INFORMAL REPORT

TMI-2 ACCIDENT EVALUATION PROGRAM

E. L. (Bert) Tolman
James M. Broughton
Richard K. McCardell
Sidney Langer
Richard R. Hobbins
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DISCLAIMER

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ABSTRACT

This report establishes the basis and scope of work for the Three Mile Island Unit No. 2 (TMI-2) Accident Evaluation Program sponsored by the Department of Energy (DOE). The current understanding of the TMI-2 accident is reviewed, with emphasis on the end-state distribution of important radioisotopes and the progression of core damage leading to gross core failure and fuel migration into the lower plenum regions of the reactor vessel. Unresolved technical issues related to core damage progression and fission product behavior in hypothesized reactor accidents are identified. The role of TMI-2 accident research in addressing these unresolved issues is developed and discussed. Data needs from TMI-2 and supporting analysis requirements to interpret the data are identified that will improve our understanding of the important physical processes and parameters controlling core damage progression and the resulting fission product behavior.

The sample acquisition tasks and supporting analyses to interpret the TMI-2 data will provide the technical basis for meeting the objectives of the program, which are:

1. To understand the physical and chemical state of the TMI-2 core and related structures and the external influences which affected the accident.

2. To understand what happened during the accident and to provide a qualified data base and standard problem of the TMI-2 accident to benchmark severe-accident analysis codes and methodologies.

3. To understand the relationship between the phenomena and processes controlling the accident and the important severe-accident and source-term technical issues.

4. To assure that the results of the program are effectively transferred to the nuclear industry.

SUMMARY

This document presents a description of the TMI-2 Accident Evaluation Program (AEP), to be conducted for the United States Department of Energy (DOE) by EG&G Idaho, Inc. The document includes background information to establish a perspective of the importance of the TMI-2 AEP relative to other light-water-reactor (LWR) severe accident and source term research.

The TMI-2 accident was unique in two important features relative to severe accident and source term research. First, the accident occurred in a commercial LWR under thermal-hydraulic conditions typical of a large family of hypothesized severe accidents. Second, the damage to the core has been confirmed to be more severe than the existing severe fuel damage experimental data base. Because of these unique features, the accident offers the potential of increasing our understanding of many currently unresolved severe accident and source term technical issues.

The major unresolved technical issues have been identified by extensive review studies recently completed within the technical community. These include U.S. Nuclear Regulatory Commission (NRC) review, based on the results of the Severe Accident Research Program; the Industry-sponsored Degraded Core Research Program (IDCOR); and extensive technical reviews recently completed by the American Nuclear Society, the American Physical Society, and DOE. The major technical issues identified from these reviews are summarized in Table 1 (page 3). Since damage during the TMI-2 accident was primarily limited to the core and reactor vessel, the utility of the TMI-2 data is related to those technical issues associated with in-vessel core degradation and fission product behavior prior to vessel failure. Those specific "in-vessel" technical issues for which our understanding can be significantly improved through additional TMI-2 research are identified in Table 4 (page 24) and include questions related to reactor system thermal-hydraulics, core degradation, and fission product behavior.

Application of the TMI-2 research towards resolution of the relevant technical issues will be very much dependent upon demonstrating a consistent and comprehensive understanding of the accident with respect to core damage progression and fission product release and transport. Developing this understanding is the most important program objective. Considerable progress in developing the accident scenario has been made, but additional work is necessary. The following information will be developed: more realistic temperature bounds during the initial core heatup; the extent of fuel relocation and resulting configuration of the noncoolable core regions; core relocation into the lower plenum; the physical and chemical interactions between the molten core materials and the lower plenum structures (including the reactor vessel); and the formation of a coolable configuration within the lower plenum. These questions will be resolved through additional sample acquisition and examination of the core and reactor vessel materials and supporting analysis to integrate the examination results, the plant thermal-hydraulic response as characterized by on-line instrumentation during the accident, and other independent severe fuel damage research. This will provide the desired understanding of the core damange progression and resulting fission product release, i.e., the accident scenario. The unresolved questions regarding the TMI-2 accident progression discussed above (summarized in Table 3) have almost a one-to-one correspondence to the outstanding severe accident and source term technical issues (summarized in Table 4) which have impacted the nuclear industry.

The TMI-2 Accident Evaluation Program will achieve the following objectives: (a) complete our understanding of the TMI-2 accident, including the end-state distribution of fission products; (b) apply the TMI-2 research results towards resolving the more general severe accident and source term technical issues; and (c) ensure industry coordination and participation in defining and carrying out the program. The AEP is organized into four major elements to achieve these goals. These organization elements include: (a) Examination Requirements and System Analysis; (b) Sample Acquisition and Examination; (c) Data Reduction and Qualification; and (d) Information and Industry Coordination.

The Examination Requirements and System Evaluation program element is the central, integrating element of the program with responsibility to (a) define the program goals and objectives, (b) integrate and evaluate the TMI-2 data with results from independent severe accident research programs to complete our understanding of the accident and to converge on a consistent and comprehensive accident scenario, (c) prepare and conduct a TMI-2 standard problem, and (d) utilize the TMI-2 research results toward resolution of the severe accident and source term technical issues.

Detailed tasks associated with defining the program, completing our understanding of the accident, developing the accident scenario, developing the TMI-2 standard problem, and utilizing the TMI-2 research towards resolving the severe accident and source term technical issues are presented.

The Sample Acquisition and Examination Program will provide the engineering support necessary for acquiring the plant samples and completing the examination of selected sample to provide those data necessary for developing a final accident scenario. The end-state characterization data needs have been identified and prioritized (see Table 5, page 28), and these data needs have been translated to a list of prioritized samples and/or data acquisition tasks (Table 6).

The Data Reduction and Qualification Program element will evaluate and qualify the data from on-line instrumentation that are required for the standard problem and development of the accident scenario. A centralized data base containing all of the TMI-2 research results, examination data, and supporting analysis will be developed.

The Information and Industry Coordination program element will provide for information distribution to the public and the nuclear community and coordination of the nuclear industry support in defining and supporting the program and interpreting the TMI-2 research findings.

ACKNOWLEDGMENTS

Defining the TMI-2 research program required input from a broad spectrum of reactor safety research specialists. The work and recommendations of the TMI-2 Technical Evaluation Groups have provided a basis for identifying general data needs from TMI-2. In addition, recommendations from nuclear industry representatives were utilized in developing the requirements for additional sample acquisition and evaluation work. These recommendations were coordinated by M. Fontana, H. Mitchell, I. Spiewak, and A. Buhl, of ENERGEX Associates, Inc., and are based on research results from the Industry Degraded Core (IDCOR) Research Program. Finally, the recent analyses of the RCS thermal-hydraulic behavior and core degradation progression by H. Fauske and R. Henry, of Fauske and Associates, Inc., were significant contributions that enabled the authors to strengthen portions of the document dealing with these issues.

The state of knowledge of core melt progression into the lower plenum, thermal challenge to the reactor vessel from molten core materials, and the factors that influence long-term degraded core coolability were reviewed by A. Cronenberg, of Engineering Science and Analysis, to identify unique contributions of the TMI-2 data in each of these areas. C. Allison, R. Hobbins, S. Behling, B. Tolman, D. Taylor, and J. Broughton, of EG&G Idaho, Inc., were extensively involved in interpreting the available TMI-2 data to identify how additional data and analyses could improve our understanding of the important processes of core degradation, core melt progression, reactor cooling system thermal-hydraulics, and fission product behavior during severe core damage accidents.

The initial draft of this document was submitted to the ad hoc Accident Evaluation Advisory Group (AEAG) in September, 1985; and the program was reviewed by the Group at a meeting in early October. The comments of the Group, collectively and individually, were of great assistance to the authors in focusing the program plan on areas deemed important by the Group. We therefore wish to acknowledge the contribution

of H. Fauske, Fauske and Associates, Inc.; M. Fontana, ENERGEX Associates, Inc.; R. Foulds, U.S. Nuclear Regulatory Commission; E. Kintner, GPU Nuclear Corp.; G. Thomas, Electric Power Research Institute; and E. Womack, Babcock and Wilcox, Inc., under the chairmanship of Prof. N. Todreas, of the Massachusetts Institute of Technology.

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FIGURES

TABLES

ACRONYMS

ACRR	Annular Core Research Reactor
AEAG	Accident Evaluation Advisory Group
AEP	Accident Evaluation Program
ANS	American Nuclear Society
APS	American Physics Society
CSA	core support assembly
CSNI	Committee for the Safety of Nuclear Installations
DOE	U.S. Department of Energy
ECC	emergency core cooling
EPRI	Electric Power Research Institute
ERASE	Examination Requirements and System Evaluation
FAI	Fauske and Associates, Inc.
GPU	General Public Utilities Nuclear Corp.
IDCOR	Industry Degraded Core Research Program
INEL	Idaho National Engineering Laboratory
IRG	Industry Review Group
LMFBR	liquid-metal-cooled fast breeder reactor
LOFT	Loss-of-Fluid Test Facility
LWR	light-water reactor
NRC	U.S. Nuclear Regulatory Commission
NRU	National Research Universal Reactor
ORNL	Oak Ridge National Laboratory
PBF	Power Burst Facility
PCS	primary cooling system
PORV	pilot-operated relief valve
PWR	pressurized-water reactor
RCS	reactor cooling system
RPV	reactor pressure vessel
RTD	resistance temperature detector
RV	reactor vessel
SA&E	Sample Acquisition and Examination Program
SFD	severe fuel damage

ACRONYMS (contd)

SPND	self-powered neutron detector
TEG	Technical Evaluation Group
TLD	thermoluminescent dosimeter
TMI-2	Three Mile Island Unit No. 2
TREAT	Transient Reactor Test Facility

TMI-2 ACCIDENT EVALUATION PROGRAM

1. INTRODUCTION

On March 28, 1979, the Unit 2 pressurized water reactor at Three Mile Island (TMI-2) underwent an accident which involved severe damage to the reactor core. Although the accident had minimal effects on the health and safety of the public, it did prompt a reevaluation of severe accidents, fission product source terms, and potential power reactor risks. For the past several years, the U.S. Nuclear Regulatory Commission (NRC)[1] and other independent research organizations[2-5] have labored to develop new insights regarding degraded core accidents and source terms important to nuclear safety and regulation. These research efforts have provided new information and insight, but have not conclusively defined a methodology for determining realistic source terms for severe accidents. The specification of realistic source terms for severe accidents represents a major unresolved technical issue impacting the nuclear industry today.

The TMI-2 accident is the most severe accident that has occurred in a commercial nuclear power plant. Recent evidence indicates that the TMI-2 core experienced significant fuel melting, with subsequent relocation of molten core materials into the reactor vessel (RV) lower plenum. A coolable configuration was achieved with 10 to 20 tons of core materials in the lower plenum, and the accident was terminated by on-line safety systems. A significant fraction of the more volatile fission products was released from the fuel; however, these fission products were confined in the reactor coolant and containment without significant release to the environment. The TMI-2 accident will provide a wealth of knowledge regarding core damage and fission product behavior during severe accidents. Also, the TMI-2 accident provides a unique opportunity to confirm the applicability of existing severe fuel damage and fission product data, obtained from small-scale experiments, and to benchmark current severe accident analysis codes and methodologies with data from a severe accident in a full-scale operating power reactor.

A number of technical issues have been identified[1-5] that require resolution to define realistic source terms for severe accidents. These issues and the originating source(s) are provided in Table 1. Research is being conducted world-wide to address these issues. The research methodology in general entails small-scale, separate-effects experiments and model development, together with larger, confirmatory nuclear systems experiments. These larger-scale experiments are important for confirming the capability of the reactor systems models and the applicability of the smaller-scale experiments. As previously indicated, the TMI-2 accident provides the only integral reactor systems data for a severe core damage accident; and, because of this uniqueness, it can provide key confirmatory information to qualify our understanding of important mechanisms controlling core damage progression and the resulting release and transport of fission products within the reactor coolant system (RCS).

The Department of Energy (DOE) is sponsoring the TMI-2 Accident Evaluation Program to take full advantage of this unique research opportunity. The goal of this program is to provide supporting objective evidence that can be used in conjunction with the results of world-wide severe accident and source term research to obtain significant, long-term relief from existing regulations which do not reflect or consider realistic source terms for severe accidents. To achieve this goal, the program must first develop a consistent and comprehensive physical understanding of the accident. This understanding must then be applied toward resolution of severe accident and source term issues to which the TMI-2 research is applicable. The resolution of these technical issues will contribute to the restoration of public confidence in the nuclear industry and the establishment of a sound technical basis for the desired regulatory relief. The objectives of the Accident Evaluation Program to support this overall goal are:

1. To understand the physical and chemical state of the TMI-2 core and related structures and the external influences which affected the accident.

TABLE 1. SUMMARY OF SEVERE ACCIDENT AND SOURCE TERM TECHNICAL ISSUES

Technical Issue	NRC[1]/ IDCOR[2]	ANS[3]	APS[4]	DOE[5]
I. Reactor Cooling System Thermal-Hydraulics				
1. Coupling between core degradation, RV thermal-hydraulics, and fission product behavior (integrated code)	--	--	X	X
2. RCS natural convection	--	X	X	--
3. RV natural convection	X	--	X	--
4. Modeling of emergency response	X	--	--	--
5. Essential equipment performance	X	--	--	--
6. Operator action	--	--	--	X
II. Core Damage Progression and RPV Failure				
1. Damage progression in core	--	X	X	X
2. Core slump and collapse	X	X	--	X
3. Reactor vessel failure modes	X	--	--	X
4. Hydrogen generation	X	--	--	X
5. Alpha mode containment failure[a]	X	--	--	
III. Fission Product Behavior				
1. Fission product release in RV	X	--	--	X[b]
2. Chemical form	--	-	--	X
3. Chemical reactions affecting fission product transport	--	X	X	--
4. Tellurium behavior	--	--	X	X
5. RCS fission product and aerosol deposition	X	X	--	--
6. Release of control rod materials	X	--	--	--

TABLE 1. (continued)

Technical Issue	NRC/ IDCOR	ANS	APS	DOE
7. Revaporization of fission products	X	X	X	--
8. Aerosol generation mechanisms	--	--	--	X
9. Fission product release during core-concrete interaction	X	X	X	X
10. Growth and deposition of aerosols	--	--	X	--
11. Effectiveness of suppression pools	X	--	X	X
12. In-containment volatization of iodine	--	--	X	--
IV. Containment Response				
1. Direct heating by ejected core materials	X	--	X	--
2. Heat transfer from molten core to containment/concrete	X	--	--	X
3. Hydrogen burning	X	--	--	--
4. Containment leakage	X	X	--	--
5. Containment failure mode and timing	X	X	X	X
6. Secondary containment performance	X	--	--	--

a. Containment failure as a consequence of a steam explosion in the reactor pressure vessel and penetration of the containment by a missile from the vessel.

b. Only low-volatility fission products.

2. To understand what happened during the accident and to provide a qualified data base and standard problem of the TMI-2 accident to benchmark severe-accident analysis codes and methodologies.

3. To understand the relationship between the phenomena and processes controlling the accident and the important severe accident/source term technical issues.

4. To assure that the results of the program are effectively transferred to the nuclear community.

This document reviews our current understanding of the TMI-2 accident, defines the unique severe accident and source term technical issues that can be addressed through additional TMI-2 research, and describes the elements of the TMI-2 Accident Evaluation Program (AEP) that will achieve the program objectives summarized above.

Section 2 presents an overview of our current understanding of the accident and identifies specific unresolved questions regarding the accident progression that will be addressed through sample examination and supporting analysis. Section 3 identifies specific technical issues that can, in part, be resolved from an understanding of the TMI-2 accident and prioritizes specific sample acquisition tasks that will provide dasta related to these technical issues. Section 4 identifies the four elements of the TMI-2 Accident Evaluation Program necessary to achieve the above program objectives. These elements include: (a) examination requirements and system evaluation, (b) sample acquisition and examination, (c) data reduction and qualification, and (d) industry coordination. The work scope for the first three program elements is identified and discussed in Section 4. Section 5 discusses how the TMI-2 data will be applied towards resolution of specific technical questions. Section 6 discusses the work scope of the industry coordination program element. Section 7 contains concluding remarks relative to the importance of the TMI-2 Accident Evaluation Program in developing an increased understanding of severe accidents and resulting fission product behavior.

Details of the funding resources available to the program and the partitioning of resources between sample acquisition and examination and analysis and evaluation activities are documented in the Master Funding Plan.[6]

2. CURRENT UNDERSTANDING OF THE TMI-2 ACCIDENT

The TMI-2 accident has had a profound impact on the nuclear industry, despite the fact that adverse public health effects were insignificant. Understanding the progression of this severe core damage accident and its relationship to the very small releases of radioactivity to the environment has drawn intense attention from the nuclear power community. The TMI-2 accident is the only source of full-scale, severe-accident data for addressing the outstanding technical issues. An overview of the current understanding of the accident is presented in this section. The damage state of the reactor, an accident scenario, temperature estimates in both the core debris and structural material in the upper plenum, and an accounting of fission products inventory in the plant are presented.

An understanding of the accident is being sought by a combination of methods, including: (a) interpretation of the response of online instruments during the accident; (b) visual and ultrasonic examination of the reactor vessel internals; (c) physical and chemical examination of materials removed from the reactor vessel; (d) examination of materials transported to reactor coolant system and containment system components; (e) calculations of accident damage and fission product behavior by severe-accident analysis codes; and (f) first-principle engineering calculations of specific phenomena.

2.1 Damage State Within The Reactor Vessel

The currently known damage state of the reactor core and the reactor vessel internal structures, as determined by various examinations and measurements, is depicted in Figure 1. A void now exists in the upper region of the original core, encompassing approximately one-third of the total core volume and extending to the outermost, partially damaged fuel assemblies. There is a debris bed about 1 m deep at the bottom of the core cavity. Efforts to probe down through the debris indicate the presence of a layer of hard, impenetrable material beneath the debris bed at about the mid-core elevation. Video scans of the lower regions of the reactor vessel

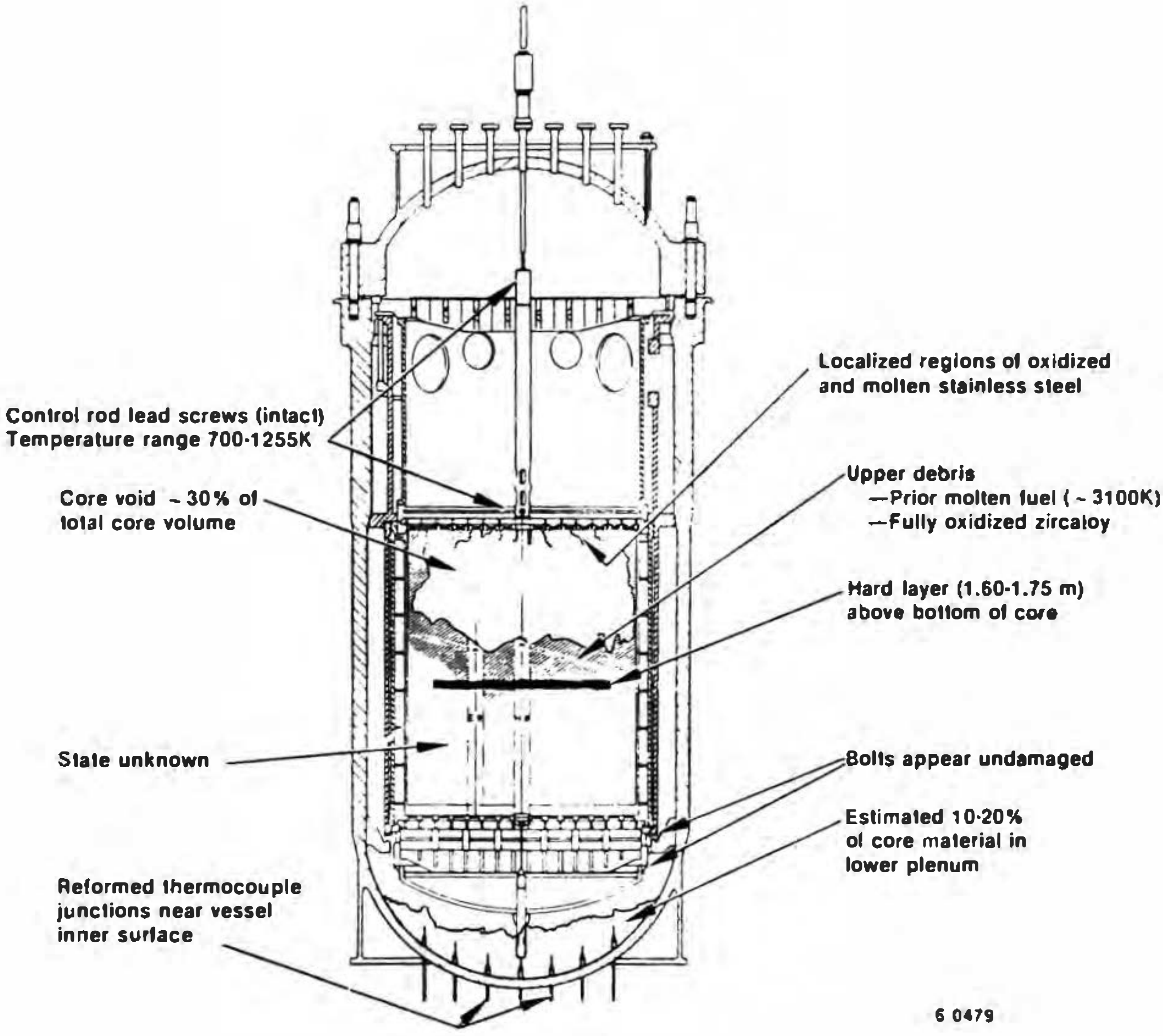

Figure 1. Known core and reactor vessel conditions.

indicate that 10 to 20% of the core material now rests on the reactor vessel lower head, about 2 m below the bottom of the original core. A photograph of the debris in the lower plenum is shown in Figure 2. Gamma measurements[7] through an instrument tube suggest that the material at the very bottom of the lower plenum may be non-fuel material (perhaps silver from melted control rods and steel from melted structural components). The particle size, as well as the texture, of the material in the lower plenum varies, ranging from uniform pea-sized gravel a few millimeters in diameter to larger pieces of lava-like material at least 200 mm in diameter. The extent of damage to the lower core support assembly is not known, since the video scans were unable to view the central regions of the lower plenum. The peripheral regions of both the lower core support structures and the reactor head do not appear to be damaged.

Particles retrieved from the core debris bed have been examined[8-9] to determine the physical, chemical, and radioisotopic characteristics of the debris. Particles greater than 1 mm dominate the size distribution in the debris, and there is a trend toward smaller particles lower in the bed. A significant depletion (up to 50%) of the zirconium content has occurred in the debris bed, and less than 10% of the silver from the control rods is accounted for in the bed. Ceramographic examination showed extensive oxidation of fuel and cladding, molten oxygen-saturated alpha phase zircaloy ($T > 2250$ K), molten UO_2-ZrO_2 ceramics ($T > 2800$ K), molten UO_2 ($T > 3100$ K), and relatively unaffected fuel ($T < 1900$ K). An average temperature of 2200 K has been roughly estimated for a debris bed sample weighing about 1 kg. Additional and larger samples of the debris bed will be taken, permitting the determination of more accurate bulk characteristics of the debris bed. The samples taken from various places in the debris bed are quite heterogeneous on a microscale (from particle to particle and even within individual particles) but are fairly uniform from sample to sample in terms of fuel structure, elemental composition, and uranium enrichment.

The damage to the underside of the core upper grid structure exhibited strong local variations, as shown in Figure 3. Areas of foamed stainless

Figure 2. Lower plenum debris (from video data).

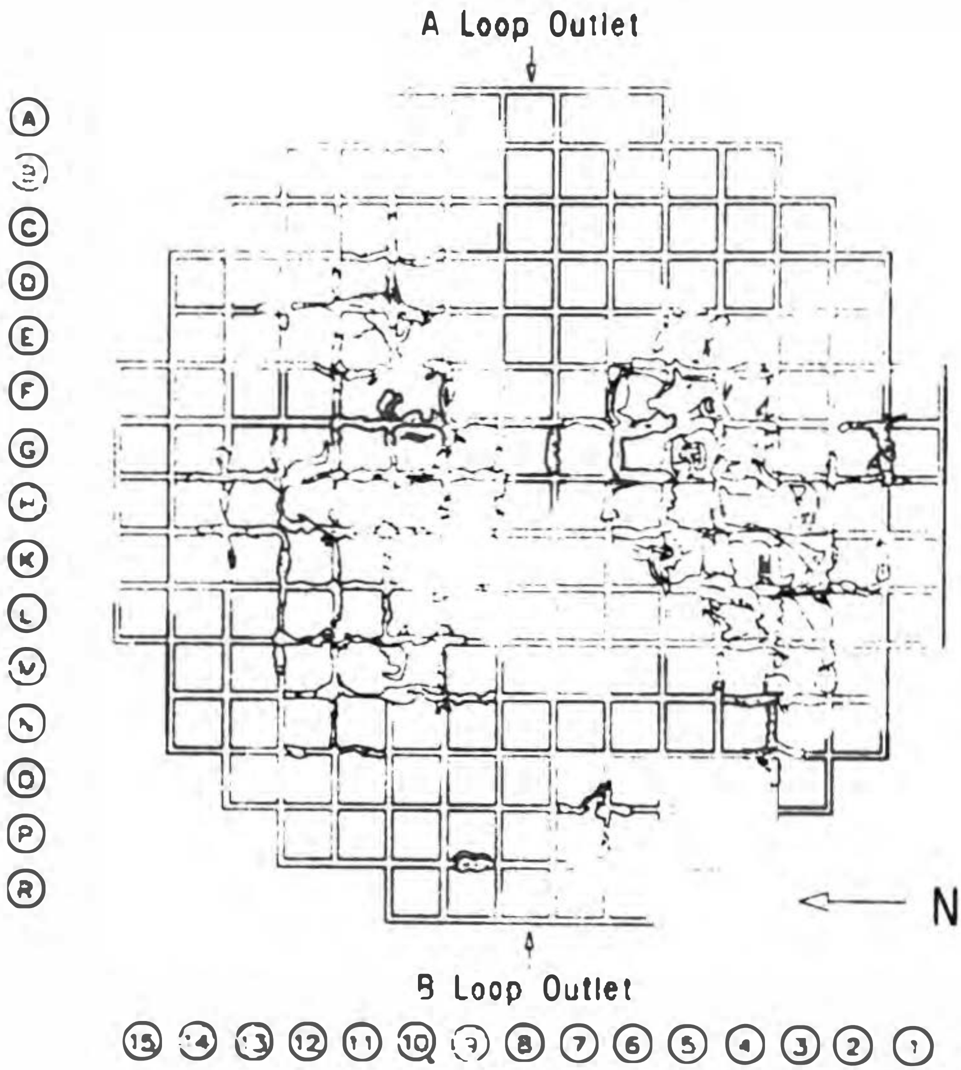

Figure 3. Damage map of TMI-2 core upper grid structure as viewed from its underside.

steel components, caused by rapid oxidation near the melting point of steel (1720 K), were adjacent to intact stainless steel components and also melted stainless steel exhibiting no evidence of foaming. These observations suggest local variations in the composition and temperature of the gas exiting the core during the high-temperature phase of the accident. Some gas streams were strongly oxidizing, others were largely absent of steam (probably containing mostly hydrogen), and some were much hotter than others. Such evidence of variations in composition and temperature of the gaseous effluent from the core suggests that there were damaged regions in the core that blocked gas flow and diverted it into channels (perhaps circuitous) through the core with different degrees of oxygen depletion due to the oxidation of metals (mainly zircaloy) by steam.

A significant axial temperature gradient existed in the upper plenum during the high-temperature phase of the accident, according to analyses of the microstructure and microhardness of two control rod drive leadscrews removed from the reactor.[10] Maximum temperatures of 1255 K near the bottom of the plenum assembly and 700 K near the top were estimated, based on analysis of the steel samples from a leadscrew in a central position. Analysis of a sample from a leadscrew near the periphery showed a maximum temperature of 1033 K near the bottom of the upper plenum. Fission product and aerosol deposition can be expected on steel surfaces at temperatures in the range from 700 to 1255 K. The plenum surfaces had been submerged for about four years before the leadscrews were removed for examination. As a result, only insoluble fission product deposition remains on the leadscrew surfaces.

2.2 Fission Product Distribution

Only about 1% of the noble gases and 3×10^{-5}% of the iodine were found to have escaped to the environment during the accident.[11] To date, measurements of fission products have been made in the core debris, in the reactor coolant, in reactor and auxiliary building sumps and tanks, and on surfaces in the reactor vessel, the reactor coolant system, and the reactor and auxiliary buildings. The estimated radioisotope accounting based on these measurements is summarized in Table 2 (Reference 12).

TABLE 2. SUMMARY OF RADIOISOTOPE ACCOUNTING FOR THE TMI-2 ACCIDENT

Plant Location	Estimated Percentage of Inventory								
	Cesium	Iodine	Tellurium	Xenon	Krypton	Strontium	Antimony	Ruthenium	Cerium
1. Fuel and core debris within the vessel	27	33	--	13[a]	13[a]	115	41	61	~100
2. Vessel internals and primary system piping	~1	<1	2	nm	nm	<1	<1	<<1	<<1
3. Primary system coolant	~10	~8	--	<1	nm	~1	nm	<<1	nm
4. Reactor and auxiliary bldg. sumps and tanks	45	41	4	nm	nm	~2	nm	<1	nm
5. Reactor and auxiliary bldg. surfaces	<1	<1	--	nm	nm	<<1	nm	nm	nm
6. Reactor bldg. atmosphere	<<1	<<1	--	nm	54	<<1	nm	nm	nm
7. Environment	nd	<<1	--	1	~1	nd	nd	nd	nd
Total accounted for (%)	83	82	6	68[b]		118	41	61	100

nm = not measured

nd = not detected above background

a. Calculation only, assumes the apparently intact fuel rods contain their initial inventory.

b. Assumes xenon retained in R.B. atmosphere to the same extent as krypton, but had decayed at time of measurement.

About 10% of iodine and cesium of the initial core inventory was found in the reactor coolant immediately following the accident; and about 40% was found in the sumps and tanks of the reactor and auxiliary buildings, together with about 4% of the core inventory of tellurium. No more than 1% of any other fission product was measured in the coolant.

Tellurium[13] has been found on leadscrew surfaces in amounts corresponding to about 2% of its core inventory, based on extrapolating results from the two leadscrews examined to the entire upper plenum surface area. This is the largest measured deposition in the RCS for any of the fission products.[14] The remaining depositions are less than 1% of core inventory. In terms of fission product chemistry, it is of interest to note that cesium was primarily found to be bound to an oxide layer on the leadscrew surfaces. Silver was found to be deposited on the leadscrews in amounts corresponding to about 1% of its inventory in Ag-In-Cd control rods, if extrapolated to the entire upper plenum surface area. Most of the fission product deposition was found near the top of the plenum assembly where the temperature was lower during the accident.

About 30% of the iodine and cesium was estimated to be in the core debris. One percent of the core inventory of strontium has been found in the reactor coolant water, and about 2% has been found in sumps and tanks. Data on tellurium in the debris bed are too few to estimate retention in core debris at this time. The debris retained approximately 115% of its strontium and practically all of its cerium. Approximately 41% of the antimony and 61% of the ruthenium inventories were retained in the core debris, and less than 1% of the core inventory of these materials has been found elsewhere. There is evidence[15] that molten stainless steel tends to concentrate available ruthenium into a separate metal phase. Measurements have not yet been performed on the fission product content of the core below the debris bed nor in the material located in the lower plenum.

2.3 Accident Scenario

The accident scenario developed here for the the initial four hours of the accident is based on the known end-state conditions of the core and reactor vessel, data from plant instrumentation recorded during the accident, the results from best-estimate analyses of the accident employing the SCDAP code,[16] and damage mechanisms deduced from severe fuel damage experiments. The important features of the accident scenario are discussed here to identify the primary mechanisms controlling core damage progression and the primary questions remaining to be resolved.

Core uncovery started between 100 and 120 min after turbine trip, which is considered the beginning of the accident. This is substantiated by the measurement of superheated steam in the hot legs at 113 min. Best-estimate predictions indicate that core temperatures were high enough to balloon and rupture the fuel rod cladding by about 140 min, releasing the noble gases and other more volatile fission products, such as iodine and cesium, in the gap between the fuel pellets and the cladding. Fission products were detected in the containment at about 143 min. These predictions also indicate that cladding temperatures rapidly increased at about 150 min, due to cladding oxidation, and quickly exceeded the zircaloy cladding melting point. The molten zircaloy dissolved some fuel; the liquefied mixture flowed down and solidified in lower, cooler regions of the core. The lowest level to which the molten material flowed was probably coincident with the coolant liquid level, which is estimated to have been into the lower one-third of the core.

By 174 min (just prior to the primary coolant pump transient, as discussed later), core temperatures had probably reached fuel melting in the central, highest-temperature regions of the core; between one-quarter and one-half of the core probably attained cladding melting temperatures with subsequent fuel dissolution and relocation. During the time period between 150 and 174 min, a relatively solid region of core materials composed of previously molten and intact fuel rods formed, as illustrated in Figure 4.a. The top of the core probably consisted of highly oxidized

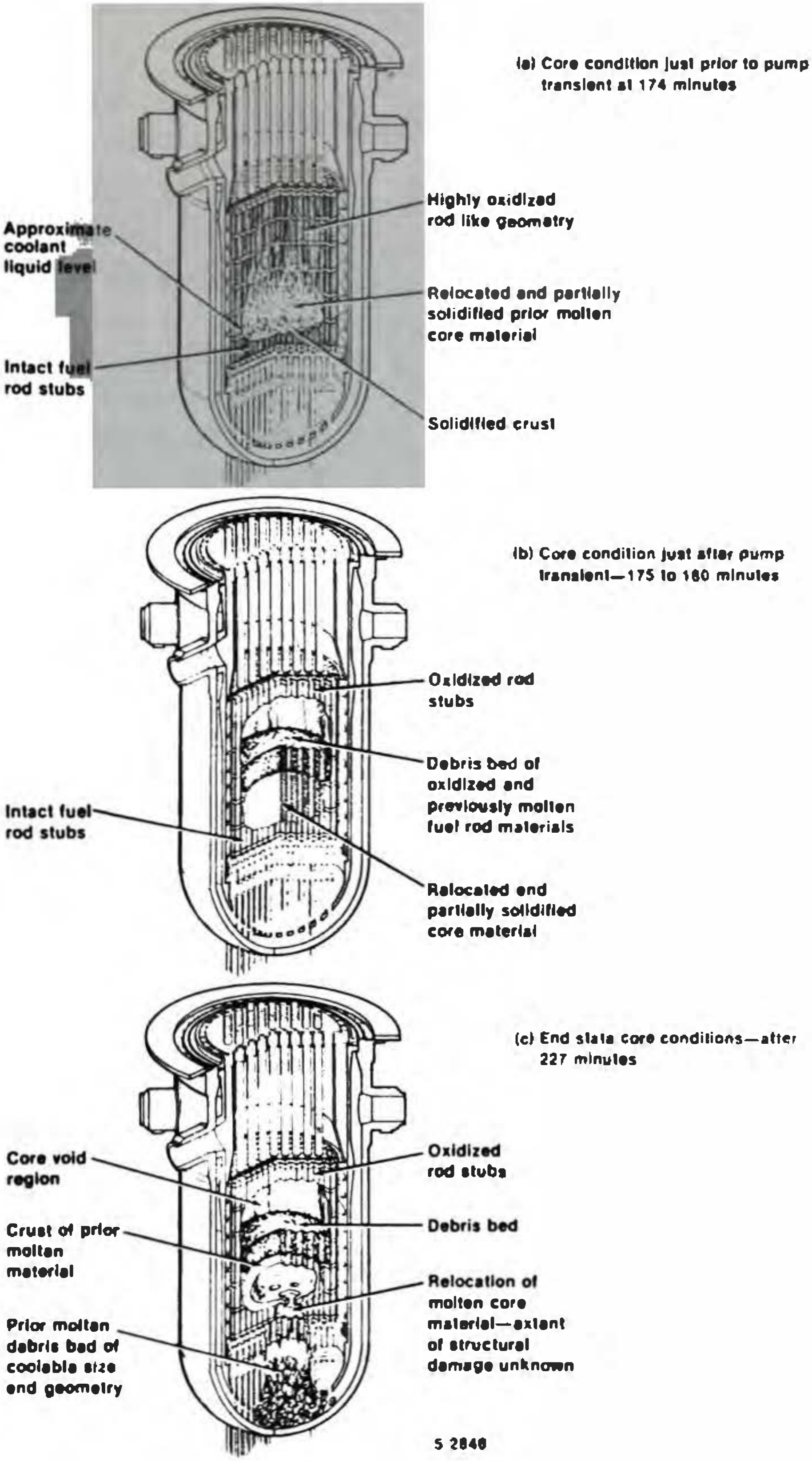

Figure 4. Hypothesized stages of the TMI-2 accident progression.

and embrittled fuel rod remnants. High-temperature molten material had not yet penetrated below about 0.75 m above the bottom of the core, since otherwise the Self Powered Neutron Detectors (SPNDs) at Level 1 and 2 (0.25 and 0.75 m above the core bottom, respectively) would have indicated anomalous behavior.[17]

The primary coolant pump transient at 174 min injected some coolant into the core. However, the extent of core cooling is not known because of the unknown flow blockage resulting from the relocated material in the lower regions of the core. Thermal and mechanical shock due to the injected coolant would result in fragmentation of the embrittled fuel rod remnants in the upper regions of the core. These fuel rod fragments could have collapsed onto the solidified surface of previously molten material, forming the rubble bed shown in Figure 4.b. Thermal calculations suggest that the zone of the relocated core materials continued to heat up even after injection of coolant into the core. These calculations are consistent with recent analysis of the in-core thermocouple alarms. The peripheral thermocouples responded to coolant injection into the core by falling back from a high-temperature alarm state, while the central thermocouples remained in their high-temperature alarm state when the core was flooded with coolant, indicating the presence of a noncoolable mass in the central part of the core even before the pump transient.

Most, if not all, of the core materials found in the lower plenum probably relocated at approximately 227 min in a molten form. This relocation was indicated by anomalous output from the Levels 1 and 2 SPNDs and by a very rapid increase of approximately 2 MPa in the primary system pressure. The increase in system pressure was apparently caused by the generation of substantial quantities of steam as the hot core material flowed into water in the lower plenum. The steam and water probably fragmented the molten material as it relocated into the lower plenum. This fragmentation may have resulted in the formation of a coolable configuration in the lower plenum. Core heatup and further core degradation were probably halted at this time by the presence of water in the lower plenum and the continued injection of water into the RCS by the high-pressure injection system. The postulated final damage configuration of the reactor core and its support structures is illustrated in Figure 4.c.

2.4 Insights from TMI-2

Our current understanding of the TMI-2 accident has provided significant insights into severe accident phenomenology:

1. The TMI-2 damage state is basically consistent with the calculations of core damage progression (such as fuel liquefaction, relocation and solidification, and the fragmentation of embrittled rods) by detailed severe core damage codes such as SCDAP.

2. Results of laboratory, severe fuel damage experiments can be used to describe the principal severe core damage phenomena.

3. The accident demonstrates that damage to the upper plenum and to the reactor vessel itself can be minimal despite very severe core damage. This implies that hot gas flow and heat transfer from the core to the upper plenum can be quite limited due to core flow blockage. Limited hydrogen production can also result from the blockage.

4. Large but very localized variations in damage to the core upper grid structure suggest core damage and that the coolant conditions within the core were highly non-uniform during the period when high-temperature gas flowed to the upper plenum.

5. Little irreversible fission product deposition in the upper plenum (generally < 1% of core inventory) occurred, which is consistent with recent in-pile test results in PBF at the INEL. Considering the temperature range experienced by the upper plenum, reversible deposition could be quite large during the accident.

6. The TMI-2 damage state and postulated accident scenario suggest a two-step core relocation process: the initial melt relocation within the core by localized candling of liquefied core materials can form a noncoolable gemoetry, ending with remelt and relocation to the lower plenum. If water is present in the lower plenum, core melt progression can be terminated with a coolable geometry in the lower plenum.

7. The present estimated releases of noble gases, iodine, and cesium from the core are consistent with our current understanding. Retention of almost all of the strontium and cerium in the core debris is consistent with calculations using NRC fission product release models;[a] so is the partial release of antimony. The significant depletion of ruthenium measured in the core debris needs to be investigated further.

8. Current evidence suggests that much of the silver inventory in control rods has disappeared from the upper core debris and may have relocated into the lower plenum.

2.5 Primary Unanswered Questions

The information concerning the accident and our current understanding of severe core damage phenomenology have led to the development of a scenario of the TMI-2 accident. To help resolve the severe accident and source term technical issues, however, the physical aspects of the accident must be quantified. Data required from examination of plant samples and on-line instrumentation include: RCS thermal-hydraulic parameters, core and structural materials peak temperatures, type and extent of materials interactions, end-state core and structural materials geometry, fission product concentration distribution and chemical forms, and aerosolization and end-state distribution of control and burnable poison materials.

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a. Based on release models described in NRC document NUREG-0772.

A number of basic issues related to the accident scenario remain to be resolved and are given in Table 3. These questions relate to RCS thermal-hydraulics, core damage progression, and fission product release from the fuel and transport and retention within the RCS. A consensus on RCS thermal-hydraulic behavior has not been achieved for the time period when the primary cooling pumps were not operating during the accident. The major uncertainties are the coolant inventory, distribution, and flow within the RCS. These are key parameters controlling core damage progression and the behavior of fission products.

The damage profile to the core upper grid structure shown in Figure 3 indicates that the flow of hot gases exiting the core was concentrated in two areas. The core geometry that created and sustained such a flow pattern and resulted in the observed damage to the upper grid needs to be determined. Also, the degree of damage to the upper plenum is much less than would have been expected for the severely damaged core as observed, and this has to be explained.

There are a number of specific questions remaining to be answered under the category of core damage progression. The processes of initial core heatup, cladding melting and fuel liquefaction, fuel melting, and the relocation and solidification of molten/liquefied core materials within the core are reasonably well understood from the KfK tests[18] and the PBF Severe Fuel Damage Tests,[19-22] but it is less well known how the burnable poison and control rods interact with the fuel rods. The subsequent heatup and relocation of molten materials into the reactor vessel lower plenum, as happened in the TMI-2 accident, remain to be understood. Damage to the TMI-2 core support assembly, instrument structures, and lower head has to be quantified and, more importantly, the formation of a coolable configuration in the lower plenum has to be fully understood.

The study of fission products in TMI-2 has focused primarily on the determination of fission product concentration within the plant. Much has been accomplished. Most areas in the fission product transport pathways have been sampled and examinations performed. The primary areas remaining

TABLE 3. UNRESOLVED TECHNICAL ISSUES RELATED TO THE ACCIDENT SCENARIO

RCS Thermal-Hydraulics

1. What was the coolant inventory as a function of time?

2. What were the flow patterns within the reactor vessel?

3. How was the core reflooded?

Core Damage Progression

1. What was the peak temperature?

2. How did the control and burnable poison rods interact with the fuel rods?

3. What was the extent of flow blockage, and how did it affect the hydrogen production?

4. How can the relocation of the core into lower plenum and the subsequent formation of a coolable configuration be understood?

5. What was the degree of damage to the core support assembly, instrument structures, and RV lower head?

Fission Product Behavior

1. What were the releases from the fuel of the less volatile fission products?

2. What were the chemical forms of the fission products?

3. What were the physical and chemical interactions that affected fission product transport?

4. How did the long-term exposure to an aqueous environment affect fission product behavior?

to be examined are the core, beneath the debris bed, the lower plenum, and the containment basement and sump. However, much remains to be done to complete the study of fission product behavior during the TMI-2 accident. For example, the following questions are still not answered: What were the chemical forms of the fission products released from the fuel, and what were the important physical and chemical mechanisms affecting fission product transport and retention within the RCS? Since the accident, the RCS has been in an aqueous environment. The long-term effects of this aqueous environment on the fission products must be understood and quantified in order to relate the present state of the fission products to their behavior during the accident. Also, in light of the severity of the TMI-2 accident in terms of both temperature and core configuration change, the release of the less volatile fission products needs to be determined.

These gaps in our current understanding of the TMI-2 accident provide direction for acquiring additional information from examination of plant samples. The acquisition and examination of these samples, especially in the core and the containment basement, together with engineering analysis, will help quantify the parameters and processes relating to the accident and eventually contribute to the resolution of the technical issues related to the accident progression, as noted in Table 3.

3. PRIORITIZATION OF DATA AND SAMPLE ACQUISITION TASKS

Prioritization of the data and sample acquisition will help focus the research and programmatic resources of the TMI-2 A&EP. The selection and examination of samples from the plant, as well as analysis and evaluation of the data and application of the research results toward resolution of the severe accident and source term technical issues, will be based, in part, upon this prioritization. First, the severe accident and source term technical issues listed in Table 1 were evaluated; and those that were judged to be significantly and directly impacted by the TMI-2 research results were identified. Then, data which can be obtained from the examination of physical samples were prioritized, based upon their applicability to both the resolution of the technical issues and the establishment of a consistent and comprehensive understanding of the accident. Additionally, the anticipated difficulty in obtaining the data was considered in setting the priorities.

The outstanding, unresolved severe-accident technical issues which have been identified by the four major severe accident and source term reviews are summarized in Table 1. These issues are grouped in four categories: RCS thermal-hydraulics, core damage progression and reactor vessel (RV) failure, fission product behavior, and containment response. The last category consists of various modes of containment failure. Since there was no containment failure during TMI-2, these issues will not be addressed by this program. Those technical issues which can be addressed directly with data from the TMI-2 accident are summarized in Table 4. The applicability of the TMI-2 research toward resolution of the technical issues is briefly summarized in this section. More detailed discussion of the data requirements and the application of the data to technical issue resolution is presented in Sections 4 and 5, respectively.

3.1 <u>RCS Thermal-Hydraulics</u>

The RCS thermal-hydraulic issues which can be directly addressed with the TMI-2 research results are natural circulation flows within the RV and the coupling of core degradation, RCS thermal-hydraulics, and fission

TABLE 4. TMI-2 RELATED SEVERE ACCIDENT AND SOURCE TERM TECHNICAL ISSUES

RCS Thermal-Hydraulics
1. Coupling among core degradation, RV thermal-hydraulics, fission product behavior, and hydrogen generation
2. RV natural circulation
Core Damage Progression
1. Damage progression in core
2. Core slump and collapse
3. Reactor vessel failure modes
4. Hydrogen generation
Fission Product Behavior
1. Release of lower-volatility fission products
2. Chemical reactions affecting fission product transport (includes chemical form)
3. Tellurium behavior
4. Vaporization and relocation of control rod materials

product behavior. The localized damage to the upper core grid, which is illustrated in Figure 3, strongly suggests that a highly localized natural convection occurred within the RV. The resultant damage to the upper plenum, however, was much less than would have been anticipated from the degree of damage to the core if an unrestricted natural circulation had been present. The mechanisms which protected the upper plenum from the high core temperatures need to be identified.

The TMI-2 accident was a severe accident with fuel melting and significant release of fission products from the fuel. As such, the data relating the interactions between core degradation, RCS thermal-hydraulics, and fission product behavior during the TMI-2 accident are appropriate for benchmarking portions of the integrated severe-accident analysis codes and methodologies. This will be accomplished via a TMI-2 standard problem which spans the accident from reactor scram until recovery of the primary cooling pumps approximately 15.5 h later.

3.2 Core Damage Progression

The TMI-2 accident is the only source of full-scale, integral-effects data on core damage, relocation of molten core materials into the lower plenum, and possible damage to the core support assembly, instrument tube penetrations, and the reactor vessel lower head. The available data from TMI-2 research appear to confirm the results from the small-scale, separate- and integral-effects experiments, such as the existence of a cohesive mass of formerly melted corium covered by loose, highly oxidized rubble, and will be invaluable in completing our understanding of the initial core heatup, materials interactions, relocation of core materials to the bottom of the core, and formation of a noncoolable configuration in the lower core region. These small-scale experiments do not encompass all phases of a severe accident. The TMI-2 accident is the only source for full-scale integral data relative to core relocation into the lower plenum, interaction of molten core materials with RV structures, and the formation of a coolable configuration in the lower plenum. Additionally, the amount of hydrogen generation may be inferred from the metallic (especially zircaloy) oxidation. Resolution of these important technical issues depends on and may be impossible without the data which presently can be obtained only from examination of the TMI-2 accident.

3.3 Fission Product Behavior

Release of fission products from the fuel, focusing primarily on the medium and low volatiles, will be addressed directly via examination of core materials. Also, analyses of the chemical forms of the insoluble species will be performed on the core and structural materials samples. However, the RCS reflood is believed to have effectively removed most of the fission products which were deposited on upper plenum and RCS structural surfaces during the initial core heatup. These fission products have been determined, based on water samples and gamma scans, to be in RCS and containment basement water or in the containment basement, either in the sludge or absorbed into the concrete walls. Their chemical form immediately following release from the fuel and during the initial transport and deposition within the RCS probably cannot be determined with certainty from examination of the end-state samples due to the long-term soaking in water. The potential reevaporization of fission products from the upper plenum cannot be determined in the absence of unwashed samples of upper plenum surfaces. Thus, the applicability of TMI-2 research results toward resolution of fission product technical issues relative to the reactor cooling system will be significantly affected by the long-term exposure of the system to water.

The uncertainty regarding tellurium behavior is related to its affinity to react with or be dissolved by metallic zircaloy and stainless steel. These reactions would effectively retain tellurium within the core and RCS except where oxidation is extensive, and the retention would not be significantly affected by the long-term exposure to water. Examination of core and RV structural samples should provide the required information regarding the mechanisms which controlled the retention of tellurium.

Vaporized control rod materials, tin (from zircaloy), and possibly the high-volatility fission products iodine and cesium are considered to be the primary sources of aerosols during a severe accident. Examination of samples from TMI-2 will provide information to quantify (a) the volatilization of control materials, (b) the distribution of those materials throughout the plant, and (c) the association of fission products with control materials. Only limited knowledge is likely to be gained from the end-state conditions regarding the mechanisms of aerosol generation, growth, and deposition within the RCS. Most materials released from the RV are now contained in the sludge found in the containment basement and in a few locations within the RCS. This sludge will be examined to determine material composition, particle size distribution, and the association of the fission products with other core and RV structural materials.

3.4 Prioritization of TMI-2 Inspections and Sample Acquisition

The basic information needs and the physical samples which can be obtained from the TMI-2 plant to provide the desired information are listed in Table 5. The basic information needs are those data which would, in general, have direct impact on resolving the high-priority technical issues and/or clarifying and quantifying the accident scenario. The physical samples have been prioritized based upon the following ranking format:

high - applicability is direct and significant

medium - applicability is less direct and/or less significant

low - applicability is indirect and/or insignificant

This ranking format is subjective and is strongly dependent upon sound engineering judgment. Additionally, availability of resources and difficulty of sample acquisition were considered. Acquisition of these samples and the subsequent measurements are considered to be adequate to address the issues listed in Table 4.

TABLE 5. PRIORITIZATION OF TMI-2 END-STATE DATA ACQUISITION

Information Needs	Physical Sample	Applicability To: Technical Issue	Accident Scenario	Weighted Priority
RCS Thermal-Hydraulics				
1. Peak temperature, material physical and chemical interactions, extent of material oxidation, effect of Ag-In-Cd and B_4C/Al_2O_3 on core damage, physical and chemical characteristics of core materials	1.1 Core bores	High	High	High
	1.2 Distinct fuel assemblies	High	High	High
	1.3 Large volume core debris	High	High	High
	1.4 Core former wall	Low	Low	Low
	1.5 Core support assembly	Medium	Medium	Medium
	1.6 Instrument structures	Medium	Medium	Medium
	1.7 PV lower head	Medium	Medium	Medium
	1.8 Fuel assembly upper-grid and end boxes	Low	Low	Low
	1.9 Fuel rod segments from upper core	Low	Low	Low
Core Damage Progression				
2. Gross structure of core and RPV internals	2.1 Video inspection	High	High	High
	2.2 Acoustic topography	High	High	High
	2.3 Acquisition of core bores	High	High	High
	2.4 Core disassembly documentation	High	High	High
Fission Product Behavior				
3. Retained fission products in the fuel, chemical form, physical and/or chemical association with core and fission product distribution throughout RCS and auxiliary piping	3.1 Core bores	High	High	High
	3.2 Distinct fuel assemblies	High	High	High
	3.3 Large volume debris structural materials,	High	High	High
	3.4 Upper plenum surfaces[a]	Low	Medium	Medium-Low
	3.5 RCS surfaces [a]	Medium	Medium	Medium
	3.6 Basement sludge [a]	Medium	High	Medium-High

TABLE 5. (continued)

		Applicability To:		
Information Needs	Physical Sample	Technical Issue	Accident Scenario	Weighted Priority
	3.7 Basement concrete wall[a] core bores	Medium	High	Medium-High
	3.8 Auxiliary piping and[a] components outside containment basement	Low	Low	Low

a. The usefulness of the data from these samples is dependent on the availability of a detailed gamma scan which will allow extrapolation to the total fission product distribution from local concentration.

Determination of the end-state gross structure of the core and RV internals is a crucial step in completing the accident scenario. Identification of the principal parameters and the understanding of the processes which controlled the formation of the end-state core and RV structure are required in the development of a more definite understanding of the core damage progression and fission product behavior.

The value of video inspections and acoustic topography in determining the gross structure of the core is readily apparent. The acquisition of core bores will provide quantifiable data concerning the location and size of voids beneath the debris bed, as well as layering of metallic and ceramic core materials. This will be based on the different characteristics of drilling through ceramic, metallic, and rodded fuel materials and void spaces. A detailed chronicle of the defueling operations will therefore be kept for future analysis. Acquisition of the above information (video and acoustic data, core bores, and defueling chronicle) is judged to be high priority.

The principal data needed to understand the details of core damage progression and RV failure are: peak temperatures, material interactions, extent of material oxidation, and physical and chemical characteristics of core materials. The proposed samples include core components (distinct fuel assemblies, core bores, large-volume debris samples, fuel assembly upper grid and end boxes, and fuel rod segments) as well as RV structures (core former wall, core support assembly, instrument structures, and RV lower head samples). The highest priority samples are the core bores, distinct (partial length) fuel assemblies, and large-volume debris samples. The resulting data will directly address issues related to core damage progression and fission product behavior. Therefore, acquisition of these samples is considered to be of higher priority than acquisition of samples from the core support assembly, instrument structures, and RV lower head.

The other samples have lower priority. For example, visual inspections completed so far indicate that the core former wall in the region of the core cavity has not been significantly damaged. If, however, significant damage is found during defueling, then acquiring a sample from the damage zone would have higher priority. The fuel assembly upper grid and end boxes and fuel rod segments from the upper core have low priority, primarily because data obtained from examination of these samples cannot be related to a specific core location, a criteria necessary to directly quantify the core damage progression.

A variety of samples, i.e. core bores, partial length fuel assemblies, etc., are required to obtain the range of data necessary to describe the phenomena of interest. For instance, acquisition of core bores and metallurgical examination of physical samples provide information on core damage. Additionally, partial length fuel assemblies from the TMI-2 core are primary sources of information on the effects on core damage from control rods and burnable poison rods. It is necessary to determine the type, extent, and spatial distribution of damage, including the radial damage gradient which existed during the accident as high temperatures progressed from the center of the core toward the core periphery. This information may be "frozen" into the peripheral assemblies remaining after the accident. Therefore, the different types of fuel assemblies, i.e., control rod, burnable poison rod, and non-control material fuel assemblies, should be sampled. The core bores should be radially and azimuthally distributed throughout the core, and the core bores should extend to the lower head.

Samples from the core support assembly, instrument structures, and the lower head should be sufficiently distributed to describe the spatial distribution of the damage, i.e. interaction between molten core and structural materials. Acquisition of these samples is not feasible until after the core volume has been defueled. Much information pertinent to the specification of these samples will be gained from the defueling operations, core bores, and engineering analysis of the interaction of molten core materials with these structures. Therefore, specification of sampling requirements for these structures will be deferred until this information is available for consideration in establishing these requirements. (Note: Acquisition of these samples is considered to be a medium priority in Table 5.) These specifications will be set forth in an addendum or revision to this document.

The primary areas where significant quantities of fission products (in addition to those already measured) are expected to be found are in core materials below the hard crust, in the peripheral fuel assemblies, and in the sludge and concrete walls of the containment basement. These samples, i.e. samples 3.1, 3.2, 3.3, 3.6 and 3.7 in Table 5, have the highest priority for acquisition and examination with respect to fission product distribution. The sampling requirements discussed above for the requisite core damage information are applicable and compatible with fission product behavior sampling requirements. The available data indicate that fission products were not retained in significant quantities on RCS surfaces either because they were not deposited there to begin with or were deposited during the core heatup and subsequently washed off during reflood. However, the sampling size was small; and steam generator surfaces were not included. Steam generator surfaces will be examined, but their priority is judged to be only moderate because the anticipated retention is low. Additional sampling of piping and system components from the containment and auxiliary building has low priority. These systems have been sampled extensively; and, based on the examination results, the retention of fission products on these surfaces is judged to be insignificant.

A simplified, prioritized list of the sampling requirements is provided in Table 6. Specification of individual samples and examinations, provided in Section 4.2, and the Sample Acquisition and Examination Plan[23] is based upon this prioritized list and the acquisition requirements discussed above. A discussion of the work scope and tasks required to complete the accident scenario follows in Section 4.

TABLE 6. SUMMARY OF PRIORITIZED SAMPLE AND/OR DATA ACQUISITION TASKS[a]

1. Video inspection of core - core bores and defueling operation.
2. Acoustic topography of core cavity after removal of loose debris and fuel rod segments.
3. Acquision of core bores.
4. Documentation of core defueling operations.
5. Core bore samples.
6. Large-volume samples of debris from the core and the lower plenum.
7. Partial length fuel assemblies from control rod, poison rod, and non-control material locations.
8. Basement sludge samples.[b]
9. Concrete samples from containment basement walls.[b]
10. Samples of the interaction zone between the core materials and lower core support assembly.
11. Samples from the interaction zone between the instrument guide tube structures and core material.
12. Samples from the interaction zone between the reactor vessel lower head surface and the lower core debris materials.
13. Samples from the interaction zone between the core former wall and core.
14. Primary cooling system surface and sediment samples from A and B loop steam generators, pressurizer, hot leg RTD thermowells, and steam generator manway and handhole covers.[b]
15. Fission product retention surfaces in upper plenum.[b]
16. Upper plenum leadscrews.[b]
17. Upper end boxes, control rod spiders, and spring from top of core.[b]
18. Fuel rod segments from debris bed.

a. This table lists the data acquisition tasks in order of descending technical priority. Actual acquisition of these data will in addition depend on sample availability, difficulty of acquisition, windows of opportunity, and allocation of resources.

b. The usefulness of the data from these samples is dependent on the availability of gamma scan which will allow extrapolation to the total fission product distribution from local concentrations.

4. PROGRAM ORGANIZATION AND WORK SCOPE TO COMPLETE ACCIDENT UNDERSTANDING

The TMI-2 Accident Evaluation Program objectives were discussed previously in the Introduction. Achievement of these objectives will ensure an adequate understanding of the physical mechanisms controlling the progression of core damage and resulting fission product behavior. This understanding of the accident scenario will provide the basis for applying our understanding of the TMI-2 accident towards resolving those technical issues summarized in Table 4.

Many questions regarding the complex thermal-hydraulic interactions that influenced the core degradation and subsequent fission product behavior during the accident remain to be resolved prior to utilizing the TMI-2 accident in a meaningful way to resolve these technical issues. These questions were briefly discussed in Section 2 and are summarized in Table 3. The scope of the TMI-2 Accident Evaluation Program to be discussed in this section will provide the additional information necessary to resolve the questions summarized in Table 3 to the extent possible and also provide the baseline for defining a consistent accident scenario.

The research approach to achieve the program objectives as discussed in the Introduction consists of the following steps:

1. Identify the data needs to characterize the damage extent to the core, lower core support assembly, instrument structures, and the reactor vessel lower head.

2. Provide the engineering support for acquiring the necessary physical samples and examination of the samples to determine their end-state condition, retained fission products, and, to the extent possible, the mechanisms leading to the end-state condition.

3. Qualify the on-line thermal-hydraulic measurements recorded during the accident that are necessary to define the RCS initial and boundary conditions as well as thermal-hydraulic response during the accident.

4. Develop a comprehensive, best-estimate accident scenario that is consistent with the TMI-2 data from the on-line instrumentation and end-state sample examinations, analysis to integrate the on-line and end-state TMI-2 data, and the results of small-scale separate- and integral-effects severe fuel damage experiments.

5. Develop a TMI-2 data base which includes the measured on-line and end-state plant characterization data, together with the supporting analysis results.

6. Define and conduct a TMI-2 standard problem to benchmark current severe accident analysis codes and methodologies.

7. Perform the necessary studies to relate the major TMI-2 research results towards resolving the severe accident and source term technical issues as discussed in Section 3, and

8. Coordinate a strong industry involvement in the acquisition, interpretation, and analysis of the TMI-2 AEP research results.

This approach provides a balanced effort encompassing evaluation of on-line instrumentation, acquisition and examination of plant physical samples, and analysis and evaluation of all available TMI-2 data.

The structure of the TMI-2 AEP to support this research approach is shown in Figure 5 and consists of four major elements: (a) Examination Requirements and System Evaluation, (b) Sample Acquisition and Examination, (c) Data Reduction and Qualification, and (d) Information and Industry Coordination.

4.1 Examination Requirements and System Evaluation Work Scope

The Examination Requirement and System Evaluation program element is the central, integrating element of the program with responsibility to (a) define the program goals and objectives, (b) integrate and evaluate the TMI-2 data (as it becomes available) with results from independent severe accident research programs to complete our understanding of the accident and to converge on a consistent and comprehensive accident scenario, (c) prepare and conduct a TMI-2 standard problem, and (d) utilize the TMI-2 research results toward resolution of the severe accident and source term technical issues identified in Section 3. Figure 6 depicts the central role of the Examination Requirements and System Evaluation program element.

The detailed tasks associated with defining the program, completing our understanding of the accident, and developing a consistent accident scenario and definition of the TMI-2 standard problem are discussed in the following sections. Utilization of the TMI-2 research to help resolve the severe accident and source term technical issues is discussed in Section 5.

4.1.1 Program Goals and Objectives

A major responsibility of the Examination Requirements and System Evaluation program element is to develop the program goals and objectives based upon the relevant severe accident and source term technical issues and the technical needs of the nuclear power industry. Establishing the program goals and objectives has been evolving since the inception of the program. The TMI-2 Technical Evaluation Group (TEG), TMI-2 Industry Review Group (IRG), and finally the TMI-2 Accident Evaluation Advisory Group (AEAG) were each convened during the course of the program to provide technical input and review. Their purpose was to assure that the program goals, objectives, and scope were properly focused. The program, as defined herein, represents the integrated effort and input from each of these consulting groups.

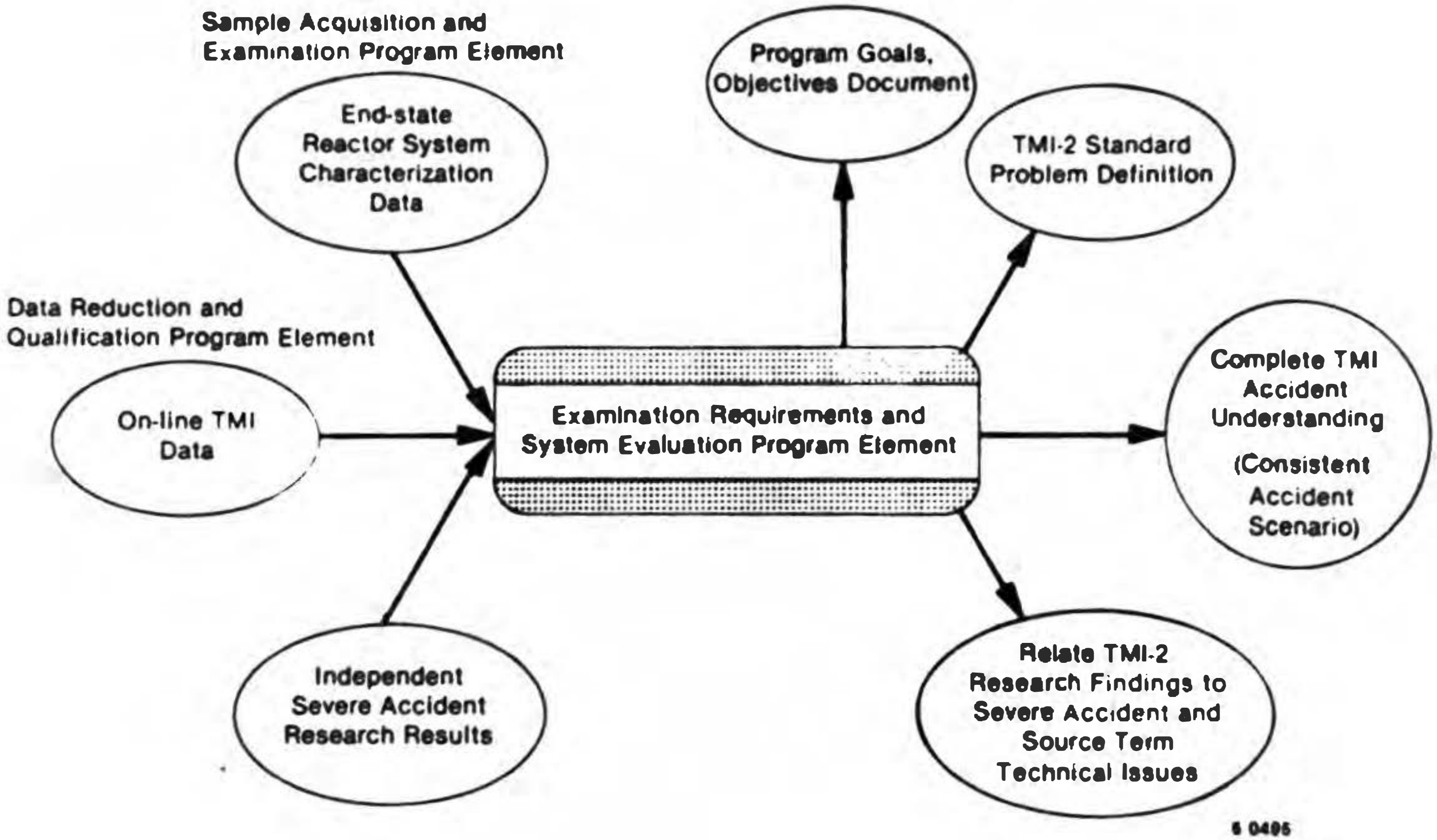

Figure 6. Central role of the Examination Requirements and System Evaluation Program Element.

The TMI-2 AEP is a dynamic program in that research results will likely dictate changes in sample examinations and/or analysis and evaluation scope and direction. The Examination Requirements and System Evaluation program element will annually review and evaluate the program based on the major TMI-2 research results. However, the TMI-2 AEP has received extensive review, both internal to EG&G and externally through the aforementioned advisory groups. Therefore, subsequent changes in the program scope and direction will only be made based upon well defined, and agreed upon, technical requirements to assure that the programmatic objectives, as stated in the Introduction, are achieved.

4.1.2 Accident Scenario Development

Development of a scenario that explains the TMI-2 core damage progression and resulting fission product behavior is the primary objective of the TMI-2 AEP. The accident scenario must be consistent with the measured RCS thermal-hydraulic response, end-state core damage data, observed end-state fission product inventory and chemistry, and the results from the small-scale, separate- and integral-effects experiments.

The current understanding of the core damage progression up to the time of the pump transient (174 min) appears to be consistent based on the TMI-2 data and our understanding of the separate-effects severe fuel damage experiments. However, our understanding of the accident progression after 174 min, particularly the details of the latter stages of core failure and fuel migration into the lower plenum, is still speculative in nature. Important questions remain to be answered related to the reactor vessel liquid level and inventory versus time and the interaction between the degraded core material and the vessel coolant, core support structures, and the reactor vessel. These questions have been noted earlier in Section 2 (Table 3). The next three sections discuss the important questions related to the RCS thermal-hydraulics, core degradation, and fission products behavior. Included are insights from recent engineering calculations to study the available TMI-2 data and a summary of additional data and analysis requirements to complete our understanding of the accident progression scenario.

Reactor Cooling System Thermal-Hydraulics. There are several major sources of uncertainty related to the RCS thermal-hydraulics. Perhaps the most important is the reactor vessel liquid level history during the initial core heatup and degradation period (100-174 min).[24,25] Additional thermal-hydraulic uncertainties include the interactions between the relocated core materials and the coolant within the reactor vessel during the initial core degradation period between 150-174 min, the period during and after the pump transient between 174-225 min, and, finally, after the core relocation event between 227-400 min.

The importance of the core liquid level versus time on the predicted core heatup is illustrated in Figure 7. Two cases are presented: Case 1 represents the best-estimate reactor vessel liquid level history based on the best-estimate plant boundary conditions coupled with engineering estimates of the core boildown.[26] These estimates were compared to independent predictions of the core liquid level versus time based on interpretation of the source range monitor data[27] and found to be generally consistent. Case 2 represents the core liquid level history that results in the SCDAP-predicted[16] core heatup and degradation to be most consistent with the TMI-2 measurements of core heatup and core degradation. Notice that the earlier core uncovery case results in a more rapid core heatup, reaching peak core temperatures exceeding 2200 K nearly 10 min earlier than Case 1. This more rapid core heatup and degradation would make a significant difference in the temperatures of the relocated core material in the lower regions of the core at the time of the pump transient which, in turn, would affect the coolability of the degraded core region and the ultimate failure mechanism of the core. Thus, any additional information to improve our estimate of the initial core coolant inventory at the beginning of the core heatup transient (100 min) and the subsequent core liquid level history will be important in developing the accident progression scenario.

Another source of uncertainty related to the reactor vessel thermal-hydraulics during the 100-174 min period is the extent of the

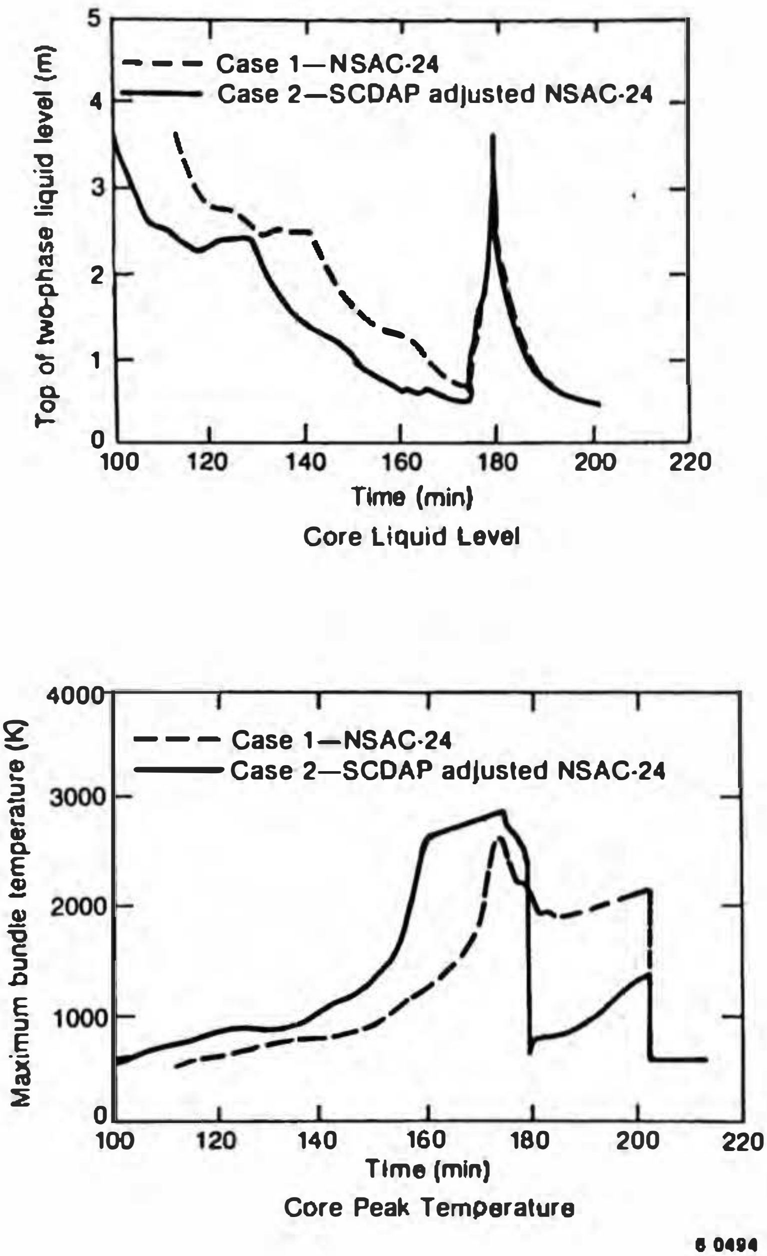

Figure 7. SCDAP-predicted core heatup vs. core liquid level history.

fuel/coolant interaction as the molten core materials relocated into the lower regions of the core and the resulting core flow blockage. The core flow blockage would result in reduced and highly multidimensional core flows which, in turn, would affect the zircaloy oxidation and hydrogen production. Differences in opinion currently exist between the NRC and IDCOR for predicting the hydrogen production after core disruption. Establishing an understanding of the initial phases of the core degradation (100-174 min) would provide a valuable baseline for addressing the basic assumptions related to the multidimensionality of core disruption, core coolant flows, and subsequent hydrogen production.

The uncertainty regarding the effectiveness of the pump transient to cool the severely degraded core must be addressed. Our current understanding of the accident suggests that the pump transient most likely resulted in shattering of the highly oxidized fuel in the upper regions of the core, thus forming the upper fuel debris bed. In addition, the affect of the pump transient on the upper plenum temperature is not understood. It is also likely that enhanced hydrogen generation and fission product release may have resulted when the upper debris bed was formed. The coolant behavior during the pump transient period also would have had a large impact on the state of the released fission products and their distribution within the reactor vessel and primary coolant system.

The last major area of uncertainty related to the RCS thermal-hydraulics is the interaction between the molten core materials and the lower plenum coolant as the molten core materials migrated from the core region into the lower plenum regions. The results of the core disruption and core material migration have yet to be studied in detail. It will be important to understand the lower plenum fuel/coolant interactions and their relationship to the ultimate coolability of the core material and challenge to the reactor vessel integrity.

Recent experimental and analytical work to interpret the TMI-2 data has provided additional insight into the reactor vessel thermal-hydraulic features during the initial heatup period of the accident. These include:

1. Testing of TMI-2-type self-powered neutron detectors (SPNDs) in the recent OECD LOFT LP-FD-02 severe core damage experiment was successful in duplicating both the initial negative currents and the large positive currents that were observed during the TMI-2 accident.[17] These large, positive currents were initiated near 1360 K and support the interpretation that severe core damage and core relocation did in fact occur in TMI-2 during the 150- to 165-min time interval. This information, together with the SCOAP predictions, supports a more rapid loss of core liquid than proposed in Reference 26.

2. Detailed mapping of the core upper grid plate damage, as shown in Figure 3, indicates two distinct, symmetric regions of oxidation and/or melt ablation. These data confirm that the reactor vessel flows and core heatup were highly localized during the initial heatup period.

3. Recent analysis of the source range detector response between 174 and 225 min indicates core relocation may have occurred prior to 174 minutes.[28]

4. SCDAP calculations[16] have provided insight into the extent of core damage versus reactor vessel liquid inventory.

5. Analyses were completed that evaluate the TMI-2 pressurizer level response during the first 16 h of the accident.[29] The results indicated that the pressurizer level measurements are valid and consistent with the overall thermal-hydraulic response of the primary cooling system. This study validates the use of the measured pressurizer liquid level as an important code benchmark parameter.

6. Analysis recently completed by Fauske and Associates, Inc.[30] has provided insight into the reactor system response during the accident. MAAP calculations simulating the first 174 min

indicate that core flow blockage significantly limits the hydrogen production and that the hydrogen significantly influences the reactor system pressurization rate. Evaluation of system pressurization response during the 180- to 220-min time period suggests that the slow system pressurization is likely due to "geometry-limited" heat transfer from the core.

The above studies have provided insight to interpret core damage progression and suggest that severe core damage and fuel relocation leading to at least partial core noncoolability occurred by as early as 174 min. Interpretation of the core relocation event (227 min) has not yet been studied sufficiently to relate the reactor cooling system measurements to hypothesized core failure mechanisms and/or core coolant interactions as the fuel migrated into the lower plenum regions.

Additional engineering studies and end-state characterization data will provide a basis for completing our understanding of the RCS thermal-hydraulics that controlled the core damage progression. These additional data needs and studies to interpret the data are summarized below.

Additional end-state data needs:

1. An important data source for assessing the reactor vessel liquid level is the visual and/or acoustic characterization of the core damage in the lower core and plenum regions. These data, together with the core sample acquisition and examination results, will allow estimates of the vessel liquid level by identifying the initial freezing locations of the core material as the fuel and zircaloy relocated downward.

2. Peak temperatures of the upper regions of the fuel rods and fuel assembly end-fittings will allow verification of the core heatup and heat transfer calculations. Because of the large temperature

gradients across the top of the core, it is important to obtain samples that can be clearly identified as to their original location.

3. The spatial temperature distribution of the upper plenum structures is necessary to evaluate code predictions of vessel natural circulation patterns and the predicted upper plenum temperatures.

4. The initial plant operating conditions, reactor cooling system boundary conditions during the transient, and on-line data recorded during the accident must be carefully characterized. These data requirements are discussed in detail in Section 4.3. The plant data provide the information necessary to benchmark engineering calculations to investigate the effect of various core degradation schemes and the interactions between the degraded core and the coolant in the reactor vessel. Uncertainties in the plant conditions that significantly affect the accident progression must be identified and bounded to the extent possible; uncertainties in the recorded time series data (reactimeter data) must be quantified. These uncertainties will provide a basis for defining and assessing bounding reactor systems calculations. The in-core instruments will also require extensive engineering evaluation to relate the measurements, which are generally outside their normal operating envelope, to the core damage progression, other on-line data, and the reactor system end-state characterization data.

Additional analytical studies:

1. Additional studies are necessary to evaluate the measured source range detector response during the 150- to 225-min time period. Additional neutronic calculations will investigate the effects of core material relocation and core and downcomer liquid levels on the source range monitor response. These calculations will

provide a basis for interpreting the anomalous source range monitor response resulting from the pump transient and relating the anomalous output to potential core damage.

2. Detailed reactor system calculations will be required to evaluate the effects of core relocation and core flow blockage on reactor vessel coolant flows, core oxidation rates (hydrogen production), core upper grid temperature, and upper plenum structure temperatures. These calculations will require a reactor systems model that includes multidimensional core degradation and vessel thermal-hydraulics to compare the predicted RCS thermal-hydraulic response for assumed core migration scenarios with the measured plant thermal-hydraulic data.

3. Three independent methods can be used to estimate the initial reactor vessel liquid level at 100 min (pump shutdown). These include (a) core boildown calculations to compare with the measured hot leg steam temperatures, (b) source range monitor response versus core liquid levels, and (c) correlation of the reactor system void fraction with measured changes in the reactor system flows when the pumps were turned off. Comparison of these three methods to estimate the initial reactor vessel liquid level at 100 min will provide a basis for identifying the uncertainty enveloped in the initial reactor vessel inventory at 100 min.

   Additional core heatup calculations are also necessary, using the uncertainty in the core liquid level and net coolant makeup to establish bounds on the core heatup rates. These calculations will be utilized to bound the core temperatures for post-174 min degraded core heatup calculations (as discussed in the next section).

4. Additional reactor system calculations are necessary to study heat transfer from various degraded core regions to help to

deduce a best-estimate degraded core geometry during the period between pump transient and the core relocation event (174 to 225 min).

5. Calculations are necessary to investigate potential core failure mechanisms and the resultant migration of molten core materials into the lower plenum regions. These studies will investigate core steam generation and condensation rates, debris bed coolability, and core material interactions with the core support structures.

Core Degradation. The core degradation in TMI-2 was very severe, as was discussed in Section 2. The core temperatures and damage in TMI-2 go beyond the current data base from separate-effects severe fuel damage experiments, thus providing a unique opportunity to improve our understanding of material interactions. In addition, the TMI-2 data offer a unique resource to evaluate multidimensional core degradation, the mechanisms affecting the core failure, and interactions of the molten materials with the lower plenum coolant and structures.

Our understanding of the core damage progression is far from complete. The initial core degradation phenomena (zircaloy oxidation, fuel liquefaction, fuel relocation, and molten fuel freezing) are becoming better understood based on separate-effects experiments. However, the importance of control rod failure on core degradation is not well understood. The formation of large molten regions of fuel material as the fuel and cladding relocate downward has not been studied previously and is likely dependent on the physical size of the core. In addition, the phenomena controlling the ultimate core failure mechanisms, migration of molten core materials, and interaction of the molten core materials with the lower plenum structures and the reactor vessel are not well quantified.

Several engineering studies have recently been performed to evaluate certain aspects of these questions. The results have provided valuable insight into the mechanisms that affect core degradation, damage to the lower plenum structures, and thermal challenge to the reactor vessel and instrument guide tubes. These studies and important results are summarized below.

1. The SCDAP analysis previously discussed has provided insight into the extent of fuel relocation during the initial heatup periods (100 to 174 min). The predicted extent of core material relocation, together with the known core damage, provides a basis for defining the configuration of the regions containing relocated core materials that formed in the lower regions of the core.

2. A recent study[31] has provided an improved understanding of the thermal response of large regions of relocated core material. The configuration was assumed to be in the form of a one-dimensional solid bed, consisting of the following layers:

    a. A bottom layer (0.4-m thick) of fused metallic mass of relocated zircaloy with the original unoxidized fuel rod stubs;

    b. A middle layer (0.5-m thick) of ceramic material comprised of oxidized zircaloy and liquefied fuel, relocated from above as well as originally in place; and

    c. A top layer (0.8-m thick) of rubble made of UO_2 and ZrO_2.

    The dimensions and the composition of the different layers were based on examinations of the TMI-2 debris bed and some assumed behavior of fuel liquefaction and relocation similar to what has been observed in severe fuel damage experiments.

    A transient thermal conduction calculation was carried out using the above configuration, assuming that 50% of the more volatile fission products had been released from the fuel. The initial temperature profile in the debris bed was based on a SCDAP calculation at the time immediately after the primary coolant pump transient at 174 min. (Average debris temperature was about 1500 K.) The upper and lower surfaces of the debris bed were assumed to radiate to heat sinks representative of the steel masses in the upper and lower plenums, initially at 1200 and 750 K respectively. No water was assumed to be present in the core.

In about 50 min, which corresponds to the time of the core relocation event at 226 min, the calculation showed that the temperatures of both the lower and upper heat sinks approached 1400 K. The surface temperatures were about 100 K higher. Most of the lower metallic layer (about 0.2-m thick) remained below the melting point of zircaloy. The ceramic region, however, was mostly molten (temperature about 2800 K). The average temperature in the rubble region was about 2600 K, much below the melting point of either ZrO_2 or UO_2.

Based on the above calculation, it appears unlikely that the noncoolable core could have melted through and relocated to the lower plenum simply from decay heat by 226 min into the accident. If the lower surface of the debris bed were indeed rich in zircaloy, the calculation shows that the temperature by 226 min was high enough for rapid oxidation; thus, a melt-through from the outside due to the heat of oxidation is a distinct possibility. Of course, mechanical failure of the lower crust is also a possible failure mode. The core examination as outlined in this document should provide a definite answer in this matter.

3. Engineering calculations by Fauske and Associates, Inc.[32] have investigated the convective heat transfer within a molten UO_2 pool. The results suggest that a molten UO_2 pool would not have produced an upper core "crust", because natural convection within the pool would result in energy transfer mostly upward

through the pool. However, if molten UO_2 were to interact with the previously molten zircaloy in the bottom crust, the pool heat transfer would be predominantly downward as a result of density differences between the UO_2 and zircaloy. This scenario would be more consistent with the upper core crust and the melt-through scenario as proposed.

4. A study has been completed to review the potential for chemical attack of the lower stainless steel core support structures and reactor vessel by the relocated core materials.[33] The core materials considered were the fuel and zircaloy oxides (UO_2,ZrO_2), control rod materials (silver, indium, cadmium), and metallic zircaloy. Conclusions of the study indicate the following:

   a. Ceramic melt materials will not chemically react with stainless steel,

   b. Silver will not chemically react with stainless steel but will dissolve approximately twice its weight in zircaloy at 1500 K,

   c. Metallic zirconium will dissolve ~1/4 its weight in stainless steel at 1400 K and seven times its weight of stainless steel at 1600 K.

These results suggest that for a higher temperature melt migration scenario the zirconium/stainless steel chemical reaction may be an important mechanism affecting the migration of the core materials within the lower plenum regions. Additional samples of the lower plenum core material and structures are needed to characterize the damage extent and material interactions and to correlate structural damage to the oxidation extent of the core materials.

5. Calculations have been completed to evaluate the potential thermal attack of the debris on the instrument penetration tubes and the reactor vessel wall.[34] The physical and thermal characteristics of the debris particles were inferred from the following TMI-2 data series:

a. Ex-core neutron flux profiles,[35]

b. Lower head video inspection,[36]

c. Wire probing of the instrument penetration tubes,[7]

d. Gamma scanning of the lower plenum,[37]

e. Hydraulic disturbance of the debris bed,[38]

f. Debris sample retrieval.[39]

Major conclusions from the thermal calculations to estimate potential failure of the instrument guide tubes and reactor vessel indicate the following:

1. Melt failure of the Inconel penetration nozzles is possible by either solid heat-generating ceramic or molten metallic debris.

2. Stable plugging of the failed (melted) penetration tubes by refreezing is predicted.

3. The stainless steel reactor vessel liner may have experienced eutectic melting by zirconium-bearing melt debris.

4. Melt failure of the lower head is not predicted.

These calculations indicate that a wide range of potential damage to the instrument tubes is possible, depending on the debris characteristics.

Samples of the debris and visual inspection of the instrument tubes and the vessel wall will be crucial to confirm our understanding of the fuel migration, damage extent to the instrument tube, and vessel wall integrity.

These studies have increased our understanding of the mechanisms likely to have controlled the core heatup leading to core failure and fuel migration into the lower plenum regions. The formation of the noncoolable core configuration and subsequent heatup rates and peak core temperatures are dependent on the core boildown scenario and the related uncertainties in the thermal-hydraulic aspects of the accident discussed in the previous section.

Additional end-state characterization data and studies to interpret the data are summarized below and will provide the basis for completing our understanding of the material interactions that controlled the core damage progression.

Additional data requirements:

1. Spatial characterization of the damage to the lower regions of the core, core former walls, lower core support structures, instrumentation guide tubes, and the lower reactor vessel head will be necessary to evaluate the damage extent and provide a basis for identifying specific samples for detailed physical examinations. Recommended measurements include video and/or acoustic data of the lower core and plenum cavities.

2. Physical measurements are required to characterize each unique core damage zone. Samples will be needed from each of the following unique core damage zones:

   a. Upper fuel debris

   b. Intact fuel assemblies (upper core)

c. Intact fuel assemblies (lower core)

d. Central, molten, noncoolable regions

e. Lower core support structures

f. Fuel debris in the lower plenum

g. Lower reactor vessel wall

h. Instrument tubes

Required physical measurements include,

a. Metallographic examination to estimate peak temperatures and, if possible, time at temperatures.

b. Chemical examination to identify material interactions and core material oxidation.

c. Radiological characterization, as discussed in the next section.

d. Physical characteristics, including particle size distribution, visual records (photographs), density, hardness, and elemental composition.

3. In addition to the end-state physical data summarized above, the RCS thermal-hydraulic data described in the previous section are necessary to correlate various assumed core degradation scenarios to the measured reactor system response. The data/calculational comparisons will provide a baseline for estimating the most likely fuel migration and long-term coolability scenarios.

Additional engineering studies:

1. Calculations are necessary to further quantify the interaction between the degraded core materials and the lower core support structures and lower plenum coolant. These calculations will provide insight regarding core heat transfer and failure mechanisms after the 227-min core relocation event.

2. Evaluation of potential core material chemical attack on the lower core crust, lower core support structures, instrument guide tubes, and reactor vessel walls will identify potential chemical reactions that may influence core material migration.

3. Longer-term analytical efforts will be required to integrate the results of the RCS thermal-hydraulics studies with the end-state TMI-2 data to identify the accident scenario that is most consistent with the end-state core and lower plenum damage. Development of a most likely accident progression scenario will provide the basis for estimating fis-ion-product release and transport during the accident.

Fission-Product Behavior. The major questions relating to fission product behavior can be grouped in two categories: category one includes those questions associated with the transient release during the high-temperature core degradation period that can be related to severe accident and source term issues; category two includes those questions related to the longer-term fission product behavior, including the end-state fission product distribution and total environmental releases.

Questions related to each of these categories, together with a brief description of the work necessary to address each of these questions, are discussed in the following two sections.

Transient Fission Product Behavior (100-400 Min). The major technical issues related to the in-vessel fission product behavior and severe accident and source term evaluation were identified in Table 1. Unfortunately, direct data relating to most of these issues are not available, due to the complex RCS hydraulic behavior which tended to confound the details of the fission product release from the fuel and subsequent transport within the reactor system. However, even in light of this limitation, the scientific community recognizes that the TMI-2 accident offers a unique source of data to improve our understanding of the specific phenomena related to in-vessel fission product behavior.[40-42] The TMI-2 specific issues related to fission product behavior were discussed in Section 3 and are summarized in Table 4.

Development of the accident scenario will be an important part of completing our understanding of the retained fission products and chemical forms of these fission products in the core material and on surfaces of the primary cooling system. The end-state TMI-2 data will be compared to separate-effects experiments through the best-estimate accident scenario and model calculations.

Fission product release and transport calculations for the entire transient have not been performed, primarily because the latter stages of core heatup leading to core failure and fuel migration are not well defined. The SCDAP[16] and MAAP[30] calculations do provide estimates of fission product release from the fuel up to 174 min and show general consistency, although the core degradation and fission product release models differ significantly in detail.

Additional calculations will be necessary to evaluate the fission product behavior within large regions of molten core material and during the fuel migration into the lower plenum regions. These fission product studies will require the core damage progression to be better defined.

<u>End-state Fission Product Distribution</u>. The most recent reviews of the end-state fission product distribution at TMI-2 are those of Owen,[12] Langer et al,[14] and Davis et al.[43] Table 2 (presented earlier) summarizes the estimated end-state fission product distribution within the TMI-2 reactor system, and the following conclusions can be made based on TMI-2 work completed to date:

1. For all of the elements examined except the noble gases, cesium, and iodine, a very large fraction of the radionuclides appear to be retained in the core material, despite fragmentation of most of the fuel rods and the apparent presence of a large quantity of once-molten core materials.

2. Permanent deposition of radionuclides on internal steel components (reactor internals and primary system piping) is small and does not appear to significantly impede radionuclide release from the primary system. This conclusion is based on analyses of components that were flushed with coolant for many months prior to examination. High radiation levels prevented early access to these components in order to confirm their behavior in the first hours and days after the accident. Without these early data, it is impossible to determine whether these components retained significant radionuclides prior to coolant flushing.

3. The reactor coolant retains a significant quantity of the cesium and iodine released from the fuel. Even if the coolant escapes the pressurized primary system piping and enters tanks or open sumps, cesium and iodine retention in water remains high.

4. Even though the reactor building contains a very large surface area that is cold (relative to the core and coolant temperatures) and consists of a variety of materials (carbon steel, stainless steel, concrete, plastics), only small amounts of radionuclides have been found on these surfaces. The high humidity conditions inside the reactor building after the accident and the resulting evaporation and condensation probably removed some surface deposits.

5. The reactor containment building atmosphere retained about 50% of the total core inventory of the noble gases. They were effectively retained for over one year, until they were deliberately vented to the environment.

6. Radionuclide releases to the environment during the TMI-2 accident were extremely small and without measurable radiological health effects.[11,44]

From the above discussions of the transient fission product behavior and end-state fission product distribution (as summarized in Table 2), it is clear that additional work is necessary to complete our understanding in each of these areas. Additional data will be necessary to complete our understanding of the end-state fission product distribution. The work described in the two previous sections to complete our understanding of the RCS thermal-hydraulics and core damage progression will be necessary to provide a basis for evaluating the consistency of the TMI-2 end-state fission product behavior data with other severe accident research results. These additional data needs and analytical tasks are summarized below.

Additional data requirements:

1. Additional core material samples are crucial in determining the end-state fission product inventory and distribution. Retained fission product concentration and chemical form data will be required for samples from each unique core damage zone, as identified in the previous section and in Table 6. Specific data requirements from each sample include:

   a. Relative activity (gamma scans),

   b. Elemental composition,

   c. Retained fission product concentrations for isotopes summarized in Table 7,

   d. Chemical forms of elements summarized in Table 7.

   Since the core temperatures may have reached values above 2600 K for several tens of minutes, characterization of the

TABLE 7. SUMMARY OF LIKELY FISSION PRODUCT CHEMICAL FORMS REMAINING IN THE TMI-2 CORE MATERIAL

<u>Cesium</u> Cs_2MoO_4 Cs_2UO_4 $Cs_2Si_4O_9$ Cs_2O Cs_2CO_3	<u>Strontium</u> SrO
<u>Tellurium</u> Tellurides of: Sn, Sb, Ag, Fe, Ni, Cr, Zr TeO_2	<u>Ruthenium</u> Ru RuO_2 RuO_2 RuO_4
<u>Antimony</u> Sb_2O_3	<u>Molybdenum</u> Mo MoO_2
<u>Barium</u> BaO	<u>Lanthanum</u> La_2O_3 LaO
<u>Silver</u> AgI	<u>Cerium</u> Ce_2O_3 CeO CeO_2
<u>Noble Gases</u>	<u>Europium</u> Eu_2O_3
<u>Iodine</u> CsI I	

low-volatility fission products, particularly strontium, barium, lanthanum, and actinides, is more important than previously estimated.

It is presently desired to obtain closure[a] on the end-state fission product distribution to within 10% for isotopes of cesium, iodine, tellurium, antimony, barium, strontium, ruthenium, lanthanum, cerium, and europium. Since examination of core material samples will be limited, extensive video data will be required to correlate and/or extrapolate the examination results to the various core damage zones. This will require extensive video data from a global perspective and closeup video data to characterize each unique damage zone in the lower core and plenum regions.

2. Additional samples of the concrete walls and remaining sludge in the reactor building basement are necessary to resolve uncertainty in the concentration of retained fission products in the basement. Even though the basement water was removed and processed (in late 1981), leaving only a few inches of residual water and sludge, radiation levels in the basement remain quite high. Although personnel have yet to enter the basement because of the high radiation, a robot vehicle has surveyed the basement and photographed the sediment deposited on the walls. Based on estimates of the activated wall surface area and the possible cesium penetration into concrete, the basement concrete surfaces could hold several percent of the total core inventory of cesium. Retention of this much cesium--and perhaps other fission products--suggests that the basement is a significant repository of fission products and requires detailed characterization.

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a. Closure in the sense that the best-estimate masses, as measured in the major fission product repositories, add up to 100% of the estimated total core inventory.

3. Additional data to characterize the primary cooling system surface deposition and core debris particulates distributed within the primary cooling system will be useful in evaluating the chemical forms of the deposited fission product species. Since interpretation of the fission product deposition on the reactor system surfaces will largely be qualitative in nature and the known concentration of the surface deposits are low, these samples are of much lower priority than either the core material or basement samples. Thus, only readily available samples of primary cooling system surfaces and debris are recommended.

Additional analytical studies required are described below:

<u>Transient Release Studies</u>

1. Development of the best-estimate accident scenario is necessary for baseline calculations to predict fission product release and transport during the core heatup period. The work described in the previous two sections related to RCS thermal-hydraulics and core degradation will be required to complete development of the accident scenario. Analysis to determine the most probable fission product chemical forms of iodine, cesium, and tellurium will be utilized in the fission product behavior calculations. In addition, sensitivity studies will necessary to assess the effects of

    - The pump transient on fission-product behavior
    - The behavior of fission products in noncoolable molten regions of core materials
    - The release of fission products during the fuel migration into the lower plenum regions.

2. Studies will be necessary to relate the estimated TMI-2 control rod behavior, based on the end-state core examinations, to independent control rod failure experiments and assess the importance of the control rods on the TMI-2 core damage and fission product behavior.

Studies to Estimate Transient vs Long-Term Fission Product Release. As discussed earlier, the transient fission product behavior during the core heatup period was not directly measurable in TMI-2 and was confounded in the longer time release measurements as the reactor system was recovered. The longer-term measurements may provide important insight into the releases prior to 400 min. The following studies will provide a basis for evaluating the initial transient release, based on the available longer-term radiological measurements.

1. A detailed chronology of long-term (>400 min) reactor system inlet and outlet flows and definition of the reactor system configuration are necessary to determine the long-term release from the reactor system.

2. Engineering studies are required to identify the relationship between the end-state fission product chemistry and the fission product chemistry during the transient (100 to 400 min). These studies will identify the long-term chemistry between the retained fission products in the fuel and the reactor vessel water.

Studies to Interpret End-State Fission Product Data

1. The end-state characterization data must be integrated to determine the best-estimate fission product distribution and inventory. This work is currently on-going and will be updated each year to reflect recently completed data acquisition and reduction work. A final report will be

completed at the end of the program that relates the known fission product inventory and distribution to our understanding of the core damage progression.

2. Work will be necessary to evaluate the chemistry and behavior of fission products transported out of the reactor core and RCS during the first days after the accident. This work will integrate the data obtained to characterize the various release pathways, the quantities of fission products that have been deposited at various locations in the RCS and auxiliary buildings, and the retained fission products within the core materials. This work would focus on the important fission product transport mechanisms, the chemistry of the TMI-2 environment, and comparisons with expected fission product behavior under similar conditions as determined from separate-effects severe fuel damage experiments.

4.1.3 TMI-2 Standard Problem

The NRC, IDCOR, and others, both foreign and domestic, have developed and assembled suites of computer codes for severe accident analysis. These codes have, in general, been developed from limited data obtained from small-scale, separate- and integral-effects experiments. Data from large-scale, integral-effects experiments are needed to benchmark and assess both the computer codes and separate-effects experiments. The TMI-2 accident is the only source of such data, even though the accident was primarily confined to the RCS and the primary pump transient was anomalous with respect to most severe accident sequences considered for licensing purposes. The confidence and credibility of the severe accident code suites will be greatly enhanced if they can satisfactorily calculate the TMI-2 accident using qualified boundary conditions and system configuration.

The TMI-2 standard problem will provide all of the necessary data to set up and perform an analysis of the TMI-2 accident, including plant configuration and geometry, operator actions and sequence of events, and the appropriate boundary conditions. The standard problem will also include a best-estimate interpretation of the accident based on the results from the core material examinations, evaluations of on-line instruments, and engineering sensitivity calculations for comparison with the code predictions.

The TMI-2 plant boundary conditions are perhaps the largest source of uncertainty in modeling the accident. These data have been extensively reviewed in previous work.[24,25,45] This work will provide the baseline for defining the reactor system boundary conditions; however, these boundary conditions will be carefully evaluated. An initial boundary condition report[25] has been published for industry review and will be updated as part of the standard problem package.

A best-estimate reactor systems model of the TMI-2 plant and accident is being developed for the standard problem utilizing the RELAP5/SCDAP/TRAPMELT code. Calculations using this model will be included with the package.

The standard problem will be defined and conducted using the CSNI[46] procedures for conducting a standard problem. This procedure requires an initial document defining the problem, with a followup workshop for analysts conducting the analysis. After the calculations are submitted, a comparison report will be issued documenting the calculational results.

The standard problem will be conducted in at least two phases. The first phase of the standard problem will include all aspects of the accident up to initiation of the pump transient at 174 min. The conditions necessary to define the standard problem from 174 to the time of forced RCS cooling (~15.5 h) are much less certain. This period of the accident will not be included in the initial standard problem. If the necessary reactor systems boundary conditions can be defined sufficiently well, the second phase of the standard problem will be conducted to cover the core relocation and RCS recovery period, from 174 min to 15.5 h.

4.2 Sample Acquisition and Examination Tasks

The sample acquisition and examination tasks are intended to provide the data necessary for understanding the end-state condition of the core and lower plenum, the important physical interactions between the core materials and reactor vessel structures, and the distribution and chemical form of the fission products within the RCS, containment, and auxiliary building systems. These data will be integrated with the on-line TMI-2 data, engineering calculations, and independent severe accident research results to identify and quantify the mechanisms controlling core damage progression and resulting fission product release, transport, and retention. As noted earlier, this process is shown in Figure 6.

The basic data needs were developed and discussed in Sections 2, 3, and 4.1; the sample acquisition tasks to provide the data were identified and prioritized in Table 6. The following sections discuss each of the sample acquisition activities to identify specific data requirements. These more detailed data requirements form the basis for specifying examination requirements for each acquisition task. The examination requirements are documented in the TMI-2 Accident Evaluation Program Sample Acquisition and Examination Plan.[23]

4.2.1 Core Bore Samples and Video Inspection of the Core (Table 6, Tasks 1, 3 and 5)

The principal information necessary to improve our understanding of the TMI-2 core damage progression and reactor vessel damage is the end-state characterization of the core and lower plenum structures, including spatial characterization of damage extent and physical characteristics of the end-state core materials. Because the core bores will provide spatial characterization data and access for video equipment into the lower core and plenum regions, the initial core bores are the highest priority samples.

The core bore samples will provide core materials from three distinct axial regions: (a) the lower core regions; (b) regions between the bottom of the core and lower elliptical flow distributor plate; and (c) the core material resting on the reactor vessel bottom head. Because of hardware limitations, separate core bores will be required for each distinct axial region.

Nine core bore locations have been chosen and prioritized, as shown in Figure 8. Prioritization of the sample locations provides a tentative guide in case drilling difficulties limit the number of samples obtainable. Prioritization is based on the following criteria:

1. Samples are required at different radial and azimuthal positions to characterize spatial differences in the core damage, materials interactions, and retained fission products.

2. Samples are required from at least one control rod fuel assembly and one burnable poison fuel assembly.

3. Utilization of the unique fuel assembly positions in which the flow holes in the lower plenum structures align.

4. Samples are obtainable only from uninstrumented fuel assemblies.

As the video data are obtained, the core bore locations may be changed to optimize the spatial characterization data possible within the limited nine locations.

Data requirements for each core bore sample consist of the following:

o Axial gamma scans and gamma spectrometry to quantify retained radioisotopes and location.

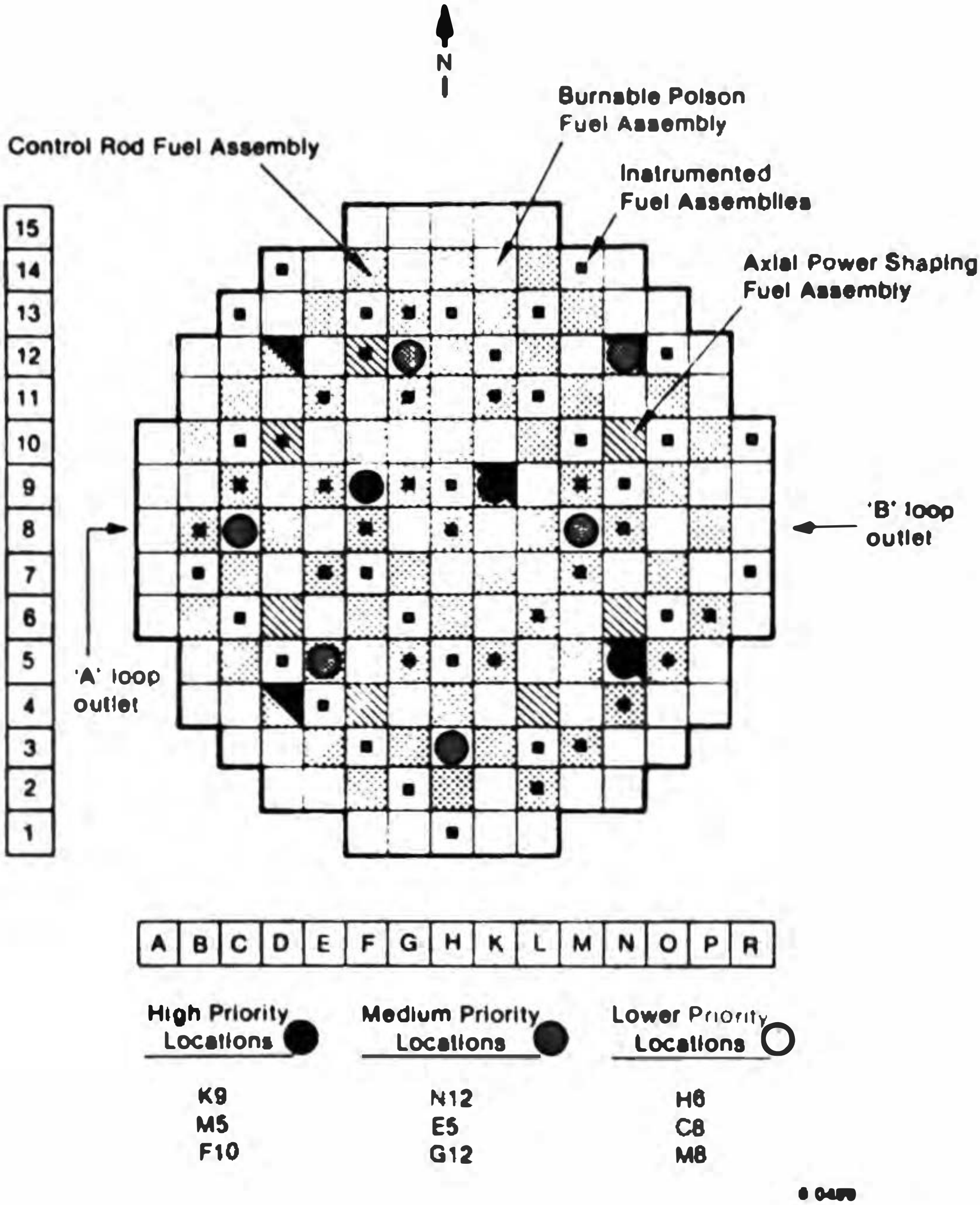

Figure 8. Proposed core bore locations.

- Correlation of the acquisition chronicle data (see Sections 3.4 and 4.2.12) with the relative gamma scan, spectrometer, and physical properties data to characterize the core material versus axial position.

- Radiochemical measurements of retained fission products and chemical forms versus axial position and material zone for those elements and compounds summarized in Table 7.

- Measurements to characterize each material zone.

    - Physical appearance,

    - Hardness,

    - Elemental and chemical composition,

    - Chemical interactions,

    - Particle size distributions, density, surface texture, porosity, ductility, grain size,

    - Peak temperatures, and

    - Oxidation state--fuel, zircaloy and other material

As the initial core bores are removed, video equipment will be utilized to visually characterize the damage in each distinct region. Global views of the damage will help correlate the core material interactions and retained fission products to larger regions of the core. In addition, detailed video data will help characterize the details of each unique damage zone. The video data are crucial for extrapolating the limited sample examination results to larger regions of the core.

4.2.2 Acoustic Topography of the Crust Below the Upper Debris Bed (Table 6, Task 2)

Topography data from acoustic scans are required to map the hard crust below the upper fuel debris bed.

4.2.3 Large Volume Samples of Debris from the Core and the Lower Plenum (Table 6, Task 6)

Grab samples from the upper core debris have been obtained and analyzed.[8,9] These small samples have provided significant physiochemical data to evaluate material interactions and fission product behavior. Eleven samples were retrieved, representing only about 0.005% of the estimated debris volume. The relatively small concentration of some fission products (particularly tellurium) has resulted in large uncertainties in the measured concentrations from the grab samples. Additional larger-volume samples are required from the upper core debris region to quantify the retained fission products.

4.2.4 Partial Length Fuel Assemblies From Control Rod, Poison Rod, and Non-Control Rod Assemblies (Table 6, Task 7)

Examination of fuel rod segments from partial length, relatively intact fuel assemblies from the lower core periphery will provide information on the axial and radial progression of core damage as well as fission product retention over a wide range of fuel rod damage. Assemblies from control rod, poison rod, and non-control material positions are required for examination. Intact rod segments will be extracted from the retrieved assemblies for detailed examination. These examinations are intended to provide information on peak fuel rod temperature, materials interactions, retained fission products, and fission product chemical form. Detailed rod segments and examination requirements will be specified based on visual inspection of each fuel assembly. The examination will emphasize changes in the relative damage across each fuel assembly and identify differences in material interactions among the fuel assembly types.

4.2.5 Reactor Building Basement Samples From Sludge and Concrete Walls (Table 6, Task 8 and 9)

The primary repositories for fission products are thought to be the reactor vessel (primarily core materials) and the containment basement, particularly the sludge and the concrete walls in addition to basement water which has already been processed. The current activity in the basement walls and sludge samples suggests significant retention (for example, several percent of the total core cesium). Independent experiments have confirmed that the concrete is an efficient absorber of cesium. Sufficient samples of the basement floor concrete walls are necessary to estimate total fission product retention in the basement walls. Sufficient samples of the basement sludge are required to estimate the total inventory in the sludge and to characterize the fission product concentrations, locations, and associated materials.

4.2.6 Primary Cooling System Surface and Sediment Samples (Table 6, Task 14)

All present experience in characterizing the plant indicates relatively small fission product deposition on the reactor system surfaces external to the reactor vessel. However, the primary cooling system surface deposition may provide the only benchmark for the fission product transport during the core degradation phase of the accident. In particular, the retained fission product chemical forms may be related to the fission product chemistry during the transient and the reactor vessel chemical environment during the accident. Analysis of the core material debris deposited in the PCS will be conducted to enhance our understanding of the plant hydraulic conditions during or shortly after the accident.

To provide these data, additional samples from readily accessible locations within the primary cooling system should include:

1. Surface deposits from the A-loop steam generator handhole cover liner.

2. Surface deposits from the B-loop RTD thermowell.

3. Surface deposits from the B-loop steam generator manway cover backing plate.

4. Surface deposits from the pressurizer manway cover backing plate.

5. Sediment samples from the steam generator tube sheet top loose debris.

6. Sediment samples from the steam generator lower head loose debris.

7. Sediment samples from the pressurizer lower head loose debris.

For each sample, the following characteristics should be determined:

- Physical appearance,
- Elemental and chemical composition,
- Particle size distributions, density, and surface texture.

In addition to the physical characteristics, radiochemical measurements of retained fission product concentrations and chemical forms are required for those elements and chemical forms summarized in Table 7.

4.2.7 Core Support Assembly Samples (Table 6, Task 10)

The extent of core support assembly (CSA) damage will be determined from visual inspection of the lower plenum and core support assembly regions through the core bore channels, as well as from selected samples of the CSA obtained during defueling. Samples of the CSA are required to determine peak temperatures and the physical and chemical interactions between molten core materials and stainless steel structures. Sample selection will be based on knowledge gained from the core bores and the follow-up video examination data.

4.2.8 Reactor Vessel Samples (Table 6, Tasks 11, 12)

The current understanding of the interactions between molten core materials and the reactor vessel, as discussed in Section 4.1, suggests that the most probable mode of vessel failure would be melting of the instrument penetration nozzles. Samples of the instrument penetration nozzles are required to determine the extent of damage to these structures and to estimate the margin to failure of the vessel. Sufficient video inspection data will be required to specify sample locations to map the damage extent to the instrument guide tubes.

The condition of the reactor vessel is not known, and our understanding of the details of the core melt progression and thermal/chemical attack on the vessel walls is not complete. However, current engineering estimates indicate only surface damage to the vessel wall. If these studies are correct, surface samples of the vessel wall may be sufficient to confirm our understanding of vessel damage. Final vessel wall sample requirements, including samples for metallurgical examination if needed, will be specified as more information is obtained during defueling.

4.2.9 Core Former Wall (Table 6, Task 13)

The core former wall appears to be basically intact in the upper regions of the core. However, below the core midplane, the extent of damage is not known. If severe damage to the core former wall becomes evident during core defueling, detailed video data of the damage zones will be necessary and selected samples of the walls will be required to determine the mode and extent of damage, temperatures, and the dominant material interactions. Sample locations will be specified as required during the core defueling.

4.2.10 Upper Plenum Surface Temperatures and Deposition (Table 6, Tasks 15 and 16)

Upper plenum surface temperatures are necessary to assess the relative importance and effect of natural convection and multidimensional flow patterns within the reactor vessel on core heatup and fission product transport/retention within the RCS. Examinations of two control rod leadscrews[10] indicate upper plenum axial temperature differences of approximately 500 K (top to bottom) and radial temperature differences (i.e., core center to periphery) of approximately 250 K. These data will be used to address the technical issues associated with RV natural circulation. However, additional samples of structural surfaces (e.g., core upper grid plate) are required to complete characterization of the retained fission products and upper plenum temperature distribution.

Based on our current knowledge of the upper plenum damage and temperature distribution, the following additional data and samples are required to characterize the upper plenum temperature and fission product distribution:

1. Relative activity mapping of the upper plenum via thermoluminescent detectors (TLDs).

2. Additional video data to characterize the axial damage extent above the core upper grid plate.

3. Additional control rod leadscrews for potential metallurgical and fission product analysis.

Examination requirements for the control rod leadscrews will be developed based on the results of relative activity mapping of the upper plenum. For example, if the relative activity mapping shows much larger spatial variation in the upper plenum activity than expected from the existing leadscrew data, additional leadscrews and other upper plenum surfaces should be examined to determine fission product deposition and temperature gradients.

4.2.11 Upper Core Samples (Table 6, Tasks 17 and 18)

Data are required to characterize the peak temperatures (and to the extent possible, the temperature history) of the upper regions of the core, core upper grid plate, and the lower structures located in the upper plenum. This information is necessary to evaluate our understanding of the multidimensional reactor vessel flow patterns during the initial core heatup phase of the accident. These data, along with engineering sensitivity calculations to study the multidimensional core degradation and reactor vessel flows, will increase our understanding of the initial core degradation, core flow blockage, and hydrogen production.

Selected samples of the upper fuel assembly structures will provide this information. Care must be taken to ensure that the original location of each sample can be clearly identified. It will be necessary to closely follow the defueling operations to ensure that unique material configurations are sampled and evaluated.

For non-fuel samples, the characterization should include the damage extent and estimate of the peak temperature (and time at temperature to the extent possible).

For samples containing fuel material, the following data are required:

- o Physical measurements to characterize
  - Appearance,
  - Elemental and chemical composition,
  - Chemical interactions,
  - Mechanical properties, density, surface texture, porosity, grain size,

- Temperatures and temperature history (to the extent possible).

o Radiochemical measurements of retained fission product concentrations for isotopes summarized in Table 7.

4.2.12 Documentation of the Core Defueling Operations (Table 6, Task 4)

The core defueling operation provides a one-time opportunity to document the global core condition and relate that to the general mechanical condition of the core materials as they are removed. This information will be valuable in developing the accident scenario. To ensure that core defueling data are available, the following requirements are identified:

1. A chronology of the defueling activities, the equipment utilized, and the results of each defueling operation should be kept.

2. In-vessel video data to identify global damage extent and consequences of the defueling operation should be taken.

3. The acquisition of targets of opportunity for unique core materials will be considered on a case-by-case basis during the defueling operations.

A summary of defueling activities will be published at the end of each fiscal year relating the defueling information to our understanding of the accident.

4.3 Data Reduction and Qualification

Since the TMI-2 accident, significant resources have been expended to interpret the operator actions, the plant sequence-of-events during the accident, and the data from selected on-line instruments. In addition, the end-state reactor systems characterization data are extensive and will increase dramatically as the core materials are removed from the reactor vessel. The TMI-2 data, however, have never been fully collected and made available through a centrally located data base. Nor have many of the data been subjected to a critical review, using a defined methodology to assign meaningful qualifying remarks and associated uncertainties. Also, inconsistencies still exist in some of the data that could have a significant impact in developing a consistent understanding of the accident progression.

The Data Reduction and Qualification program element has the responsibility to evaluate, qualify, and centralize the TMI-2 data. This section describes the approach in developing a centralized data base, discusses the types of data which will be incorporated into the data base, and identifies the data qualification requirements.

4.3.1 Data Base Overview

The data base will be a centralized data storage location for the following major data sources:

- Plant configuration data,

- Plant sequence of events and operator actions,

- Initial plant boundary conditions,

- Transient plant boundary conditions,

- On-line plant instrumentation data,

- Fission product data

- End-state examination data,

- o Results from the best-estimate TMI-2 standard problem calculations performed by EG&G Idaho,

- o Analytical studies to interpret the TMI-2 data,

- o Core defueling activities.

In addition to data storage, the data base will have the capability to manipulate selected data and some limited graphic and sorting capabilities. A schematic of the data base capability is shown in Figure 9. The data base is being designed for use on IBM Personal Computer systems. Details of the data base configuration and capabilities are documented in Reference 47.

4.3.2 <u>Data Qualification</u>

The primary purposes for reviewing and qualifying the data are (a) to identify the uncertainties in operator actions, the sequence of events, and measured on-line data for defining the TMI-2 standard problem, and (b) to provide information to improve our understanding of the accident progression and the interactions between the degraded core, RCS thermal-hydraulics response, and fission product behavior.

The basic data required to qualify for the TMI-2 standard problem are listed below:

- o RCS pressure
- o Reactor primary system temperatures
- o Reactor coolant inventory and flow rate
- o Liquid levels (reactor vessel, pressurizer, piping systems, steam generators)
- o RCS makeup and letdown flow rates
- o Operation periods of ECC injection
- o Operation periods of PORV and block valve

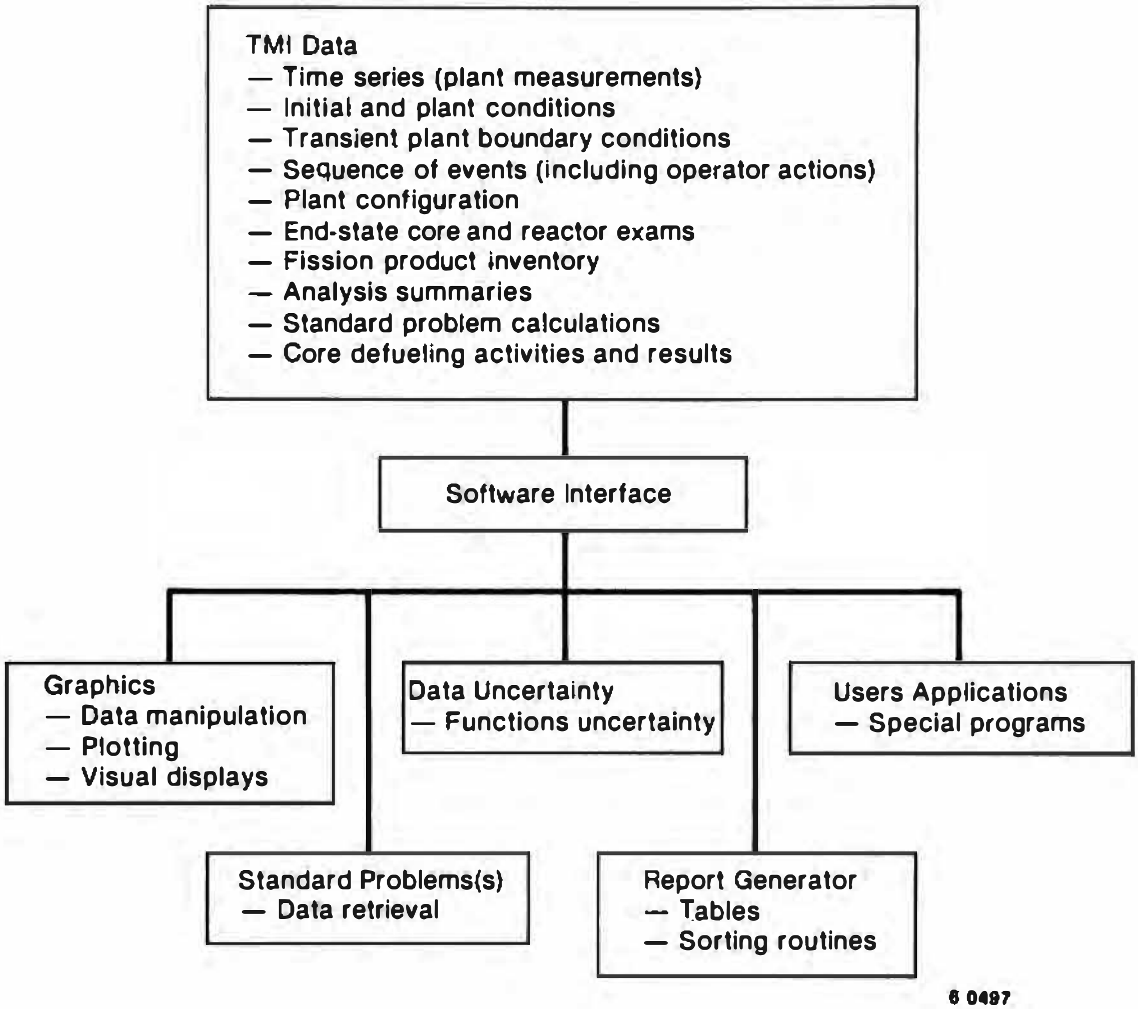

Figure 9. TMI-2 data base capability.

- In-core neutron detector and thermocouple data for estimating core damage progression and RV liquid level
- Containment radiation levels
- Source and intermediate-range detector data and supporting analysis to estimate reactor vessel liquid levels
- RCS coolant void fraction (from the Venturi meter)

These data are needed from accident initiation until stable core cooling was reestablished at about 16 h. After stable core cooling was reestablished, the events and measurements needed are those related to changes in the plant coolant inventory and fission product venting, i.e., venting of noble gases approximately one year after the accident.

The time series (reactimeter) data must be qualified to the extent that engineering uncertainties are established before utilization for the standard problem or resolving uncertainties in the accident progression. The boundary condition data must be evaluated to the extent that uncertainties that have a significant impact on the model predictions of core damage are identified.

A formal system has been adopted for determining and classifying measurement data quality. This system consists of analyzing the measurements, reviewing the results in committee, and assigning qualification levels and statements to each measurement.

In analyzing the measurements, answers to the following questions are sought: (a) are the data consistent with respect to single channel analysis criteria (range, noise limits, time response, and correlation with significant plant events and prior history)?, (b) do the data agree with other, redundant, information?, and (c) do the data agree with thermal-hydraulic theory? In the next step, a Data Integrity Review Committee (DIRC) reviews all analysis, evaluations, and comparisons performed in response to the above questions. (The DIRC will be composed of a panel of experienced persons, knowledgeable in TMI-2 data analysis.) Finally, the DIRC will assign qualification levels to each measurement.

Only the time series data will be formally qualified; this includes the reactimeter data, strip chart data, and the utility and alarm printer data. To prevent duplication of previous data qualification work, the industry is being asked to supply information related to qualification of any TMI-2 data. Where independent qualification has been completed and documented for the above data sources, the qualified data will be entered into the data base with no additional evaluation.

Qualification of the in-core instrumentation data, source range monitor output, radiation monitor data, and some of the RCSs thermal-hydraulic data will require special techniques. For most of these cases, the instrumentation was operating outside its normal ranges and must be extensively evaluated prior to relating the data to the core damage progression. The following data will require special interpretation and qualification:

Self Powered Neutron Detectors (SPNDs): The SPNDs normally provide measurements of the spatial distribution of power. The SPND output went to zero when the reactor was scrammed at the start of the accident. However, during the accident, many of the SPNDs alarmed because the output went off-scale negative, then later came back on-scale and then went off-scale positive. The initial negative behavior has been interpreted (and substantiated via independent experiments) to indicate that the detector temperature was approximately
811 K (1000°F) at the initial alarm. The subsequent positive output from th e SPNDs is judged to be a result of localized severe core damage near the det ector.

Work is underway to determine a correlation between both positive and negative SPND output and the surrounding core temperatures and damage. This correlation would allow estimates of core damage as a function of time and core damage as a function of axial core position. This confirmation in interpreting the SPND data would have important implications in interpreting the core liquid level versus time and core damage versus time.

Source Range Monitors: The two source range monitors normally provide measurement of the core gross power level. The output from one of the monitors has been evaluated and interpreted to provide core liquid level, assuming that the core geometry did not change significantly. The validity of these core liquid estimates is questionable, because it is now known the core geometry did change significantly. The postulated accident scenario provides a description of core relocaton that can be used for estimating the impact of core relocation on calculated source range monitor response. Also, the behavior of the second source range monitor has just recently been obtained and is being evaluated. Comparison and correlation of the behavior of both monitors, coupled with the data from the intermediate-range neutron monitors and the postulated accident scenario, will provide additional information on reactor vessel coolant inventory and the time and extent of core geometry changes.

Resistance Temperature Detectors (RTDs) and Core Thermocouples: RCS temperatures during the accident are indicative of the extent of core damage and the magnitude of heat transfer from the core to the RCS. RCS coolant temperatures were measured with RTDs in the hot legs of primary cooling loops and with thermocouples at the top of the core. Measured hot-leg steam temperatures during the accident should be reasonably accurate provided that radiation effects were not significant. However, the core thermocouple output was out of range of the measurement system when the temperatures exceeded about 640 K. As the core damage progressed, new thermocouple junctions were formed when the thermocouple leads melted. Experience from the PBF Severe Fuel Damage tests indicates that the new junctions provide reasonably accurate temperature measurements, but the locations of the newly formed thermocouple junctions are not known. The response of these measurement devices will be reviewed to verify the existing data and interpretations.

Venturi Meters: The primary system Venturi meters normally provide measurement of single-phase coolant flow rate. Two-phase coolant flow conditions were established in the RCS before the primary coolant pumps were turned off. The output of the Venturi meters during this time may be proportional to coolant void fraction, which increased as coolant escaped through the pilot-operated relief valve (PORV). The data from the Venturi meters will be evaluated, using methods developed from the LOFT OECD Program, to determine coolant void fraction. If coolant void fraction can be determined during this phase of the accident, then the reactor vessel coolant inventory can be more accurately estimated as a function of time, but particularly at 100 min when the pumps were shut off and the reactor vessel coolant boil down began. The uncertainty associated with the coolant inventory during the accident prior to primary coolant pump shutdown has had a significant impact on the calculated core heatup.

5. ROLE OF TMI-2 ACCIDENT EVALUATION PROGRAM IN RESOLVING TECHNICAL ISSUES

The role of the TMI-2 Accident Evaluation Program includes utilizing the TMI-2 research findings in resolving severe accident and source term technical issues that could not previously be resolved due to insufficient or nonexistent data. The TMI-2 accident resulted in relocation of molten core materials through the core boundaries and the lower core support assembly, the interaction of molten core materials with the reactor vessel lower head, and the retention of fission products within the RCS and the containment. Thus, the accident offers the potential for (a) studying the physical mechanisms that control core damage progression and relocation, (b) evaluating the effectiveness of various fission product barriers, and (c) providing information for developing and assessing the reactor systems computer models. Unfortunately, the limited TMI-2 data recorded during the accident and end-state characterization of the reactor system completed to date are not adequate for establishing a complete understanding of the core damage progression and resulting fission product behavior. Therefore, the major goal of the AEP as discussed previously is to complete our understanding of the accident and develop a consistent accident scenario. This work is discussed in Section 4. A consistent understanding of the TMI-2 accident will provide the technical basis for utilizing the TMI-2 research towards resolution of important technical issues.

The basic approach to severe accident and source term research is shown in Figure 10 and is presented to show the importance of the unresolved technical issues relative to achieving the research goal of acceptable severe accident and source term computer models. As shown in Figure 10, the technical issues, if not resolved, may significantly impact the acceptability of and confidence in the reactor system models and the degree of conservatism required for the risk consequence models.

The research approach consists of two major efforts. The first is to identify, characterize, and develop computer models describing the many individual physical parameters and processes that affect the core

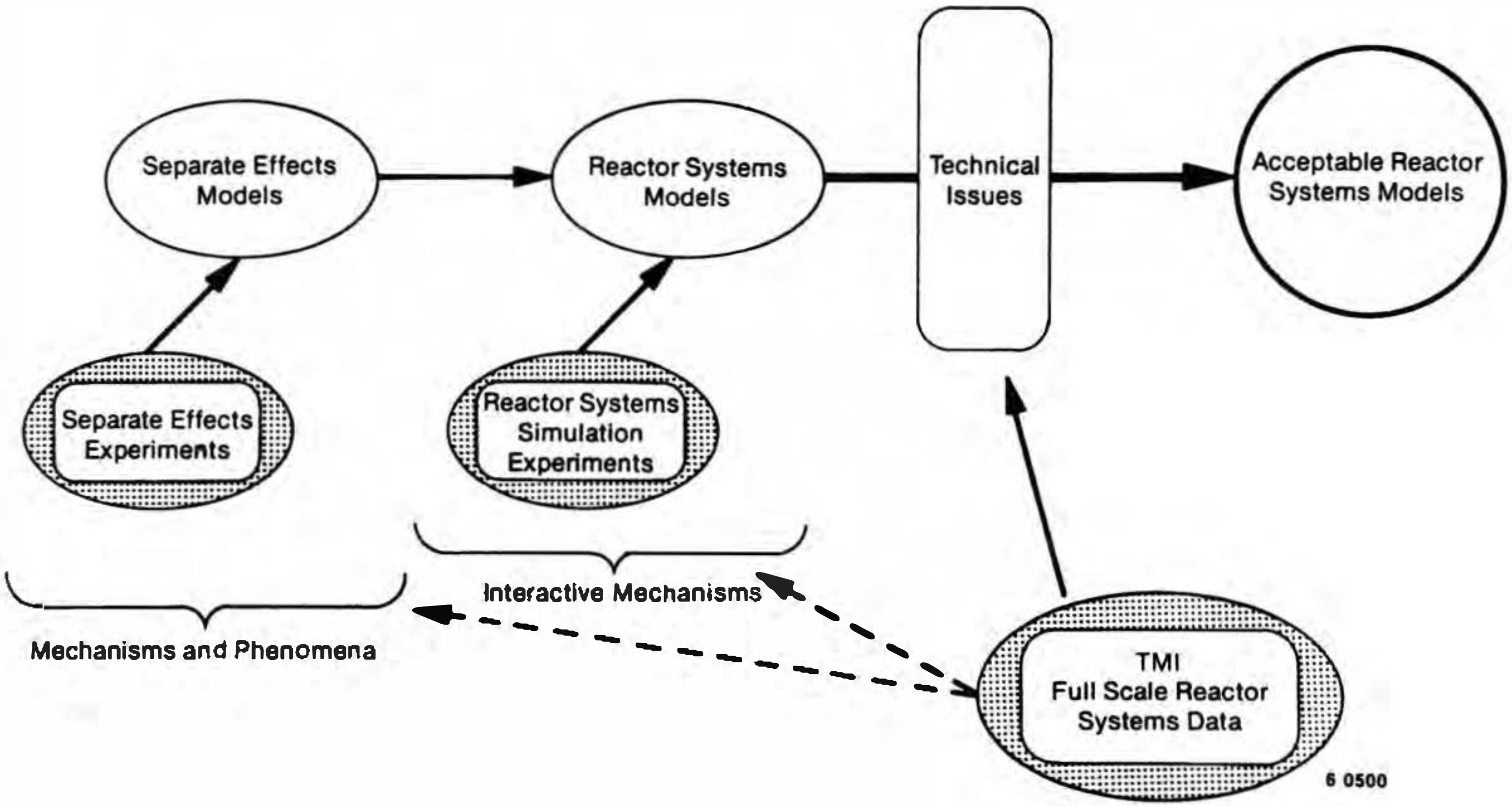

Figure 10. Severe accident and source term research methodology.

degradation and resulting fission product release and transport. This process of characterizing the important individual mechanisms entails conducting and evaluating small-scale, separate-effects experiments. The second effort in the overall research approach involves the development of reactor systems models that integrate the individual processes and mechanistic models into a comprehensive reactor system model that simulates all the important physical interactions that occur during a severe accident in a large reactor system. Development and assessment of the reactor system model necessitates conducting and evaluating large-scale, integral-reactor simulation experiments to study and confirm the important interactions among the individual mechanisms and to provide data for assessing the capability and/or conservatisms of the reactor system models.

The TMI-2 accident is unique in that it provides severe accident data not available elsewhere. As noted above, the core damage was more severe and extensive than that observed in the small-scale severe fuel damage experiments that have been performed. The data from the end-state reactor system and core material examinations, coupled with the development of a consistent understanding of the accident scenario, represent a unique research resource to extend our understanding of important technical issues associated with the in-vessel core damage progression and the resulting fission product release and transport.

The TMI-2 accident, of course, represents only one, successfully mitigated, severe accident; therefore, the TMI-2 reaearch results are not all-inclusive but must be integrated with the research results from other severe accident research programs to address and resolve the technical issues. This process is shown in figure 11 and will provide a sound basis for broad technical agreement to issue resolution. The TMI-2 Accident Evaluation Program will relate the TMI-2 research results to each of the issues in Table 4; however, an extensive integration of all available severe accident research results related to each of the issues is outside the scope of this program.

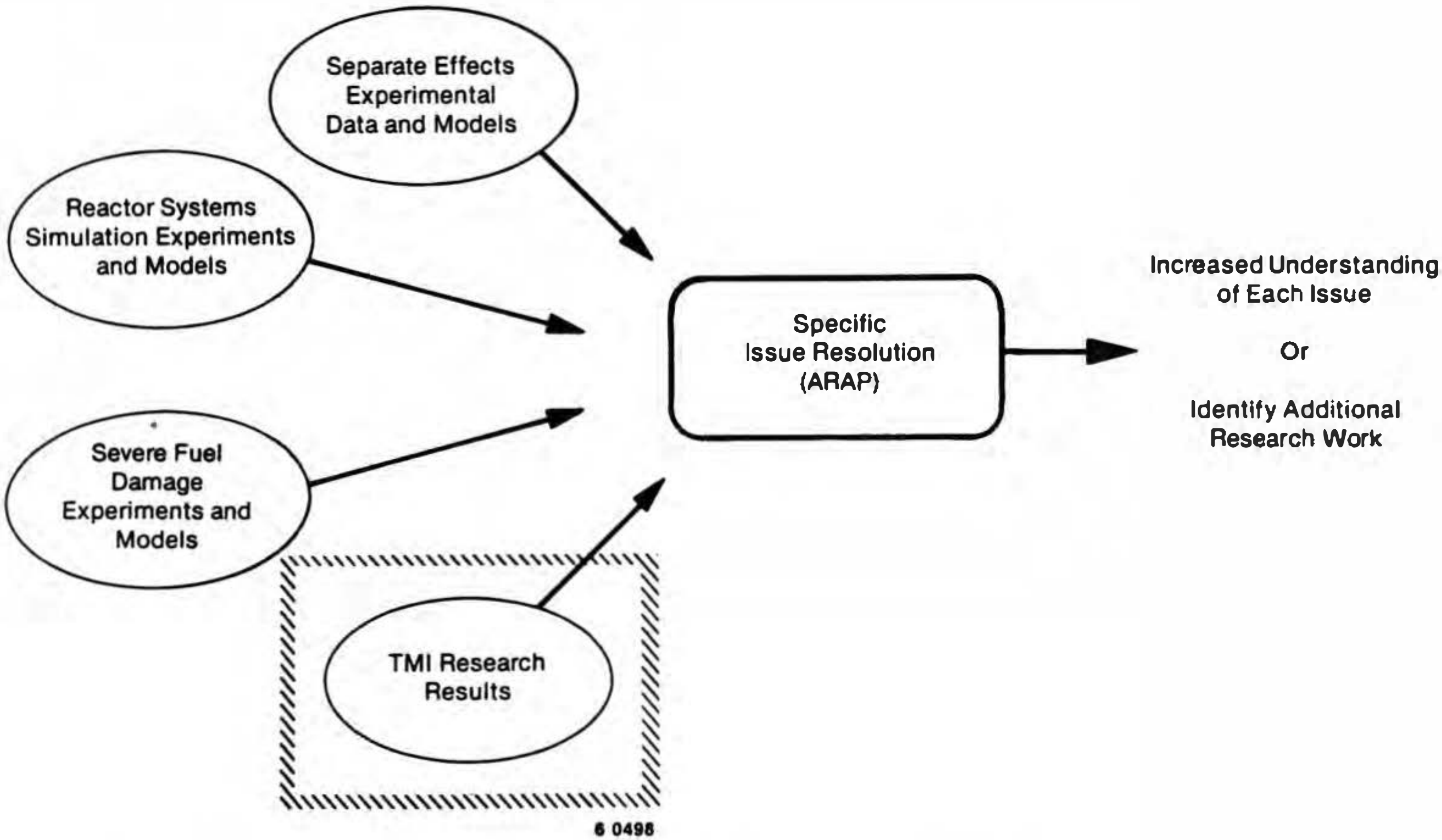

Figure 11. Data integration necessary for resolution of severe accident and source term technical issues.

The technical issues related to RCS thermal-hydraulics, core degradation, and fission product behavior for which TMI-2 data will provide important additional insights were identified and briefly discussed in Section 3 (Table 4). In the sections below, each of the issues identified in Table 4 is described in more detail, followed by a summary of our current understanding of the issue and the role of the TMI-2 Accident Evaluation Program in helping to resolve the issue.

5.1 RCS Thermal-Hydraulic Behavior

The RCS thermal-hydraulics, i.e., the time-dependent changes in the reactor system liquid inventory, coolant mass flow rates and velocities, and system pressures and temperatures, are key initial and boundary conditions controlling core damage progression, hydrogen production, and fission product release from the fuel and also transport and retention within the RCS. The steam flows within the RV control the heat transfer from the core, affect the oxidation potential, and provide the medium for transporting the released fission products from the RCS to the containment. Realistic estimates of core damage, hydrogen production, and fission product behavior require knowledge of these parameters with reasonable accuracy. The system gas composition and oxidation potential and the formation, transport, and retention of aerosols are also strongly dependent upon RCS thermal-hydraulics.

The necessary understanding of the physical processes and the capability to perform the appropriate multidimensional hydraulic calculations basically exist. The science upon which the thermal-hydraulic models are based is mature and based upon a broad and extensive experimental data base. The two primary issues needing resolution that TMI-2 data can address are the formation of natural convection heat transfer cells within the reactor pressure vessel and integrated RCS thermal-hydraulic response. These issues will be considered separately.

5.1.1 Reactor Vessel Natural Convection

Issue Description. Some calculations indicate that natural circulation steam flows in the RCS can significantly influence core heatup rates, energy transfer from the core to the upper plenum, aerosol and fission product transport from the core, the retention of fission products, and the temperatures and failure times of components in the RCS. However, the changing core geometry as core damage progresses can result in flow redistribution and/or flow restrictions within the core which cannot be calculated with certainty by existing codes.

Current Status. The effects of reactor vessel natural circulation flow redistribution on core damage progression and fission product transport and retention are not currently fully understood. For example, the IDCOR interpretation is that large flow blockages in the core region will occur and effectively limit natural circulation, thereby reducing hydrogen production and limiting the heat transfer and aerosol and fission product transport from the reactor vessel region. However, this issue remains unresolved between IDCOR and the NRC.

Relevant TMI-2 Behavior. Fuel melting did occur during the TMI-2 accident, yet damage to the upper plenum was essentially limited to the bottom of the upper core support plate. This very limited damage to the upper plenum or, conversely, existence of the high core temperatures without extensive damage to upper plenum structures would be unexpected based upon the results of calculations performed shortly after the accident. Obviously, mechanisms and/or processes occurred during the accident which effectively limited heat transfer from the core to the upper plenum.

AEP Contribution. The principal data from the TMI-2 examination, coupled with analysis, which will help resolve this issue are reactor coolant inventory versus time, core temperature distributions, and detailed characterization of the core geometry, i.e., identification of flow channels within the debris bed, cracks or flow channels through the solid prior molten region near the bottom of the core, and the extent of flow bypass around this zone of relatively solid material. These will be identified through video data and core bore examinations. Examination of upper plenum vessel internal surfaces to establish temperature distributions will also assist in resolving the issue.

5.1.2 Integrated Reactor System Response

Issue Description. The progression of a severe reactor accident involves a complex interaction of many physical and chemical processes. While analytical models have been formulated to describe most of these processes, there is a need to link these models together to produce an integrated analytical package which can accurately describe the composite accident behavior. Work is currently underway to link these models; however, there is a requirement to test the integrated analysis against data representative of a severe accident progression with the important interactive processes and appropriate materials present.

Current Status. Integrated code packages which can model the entire severe accident process are nearing completion. These include MAAP (IDCOR), and MELCOR (NRC). In addition, the NRC is developing more detailed mechanistic codes (TRAC, MELPROG, SCOAP, VICTORIA, CORCON, CONTAIN, etc.); and EPRI is developing PSCAC/CORMELT which may become integrated into a continuous model.

Relevant TMI-2 Behavior. The TMI-2 accident was a severe accident in which conditions representative of many postulated severe accidents occurred in a full-scale system composed of typical RCS materials.

AEP Contribution. A major objective is to obtain information which will allow a complete description of the relevant details of the accident progression. This will be accomplished by examination of existing plant records, as well as probing and sampling of RCS components and debris. A major objective of the AEP program is to integrate all TMI-2 examination results into a standard problem which can then be used to benchmark these codes. A detailed description of this process is given in Section 4 of this report.

5.2 Core Damage Progression and RPV Failure

The core degradation processes during a severe accident involve complex thermal, hydraulic, mechanical, and chemical interactions among the materials within the reactor vessel. A large number of in-pile and out-of-pile experiments have identified the processes that are important. In addition, computer models are becoming increasing more able to correctly model these processes.

Outstanding technical issues related to core and RV damage include the characterization of core damage mechanisms, mechanics of core slump and collapse, RV integrity, and in-vessel hydrogen generation. These issues are considered separately in the following subsections.

5.2.1 Damage Progression In-Core

Issue Description. The initial stage of damage progression in the core involves numerous complex physical, mechanical, and chemical interactions, many of which are not fully understood. The relative influence of these interactions on the mode and extent of core damage progression in a large-scale system containing representative core materials is uncertain and can have a significant influence on the subsequent progression of the accident. These initial core damage progression parameters can influence the extent of cladding oxidation later in the accident (see 5.2.4), the potential for natural circulation cooling, fission product sweepout from the core region, and the potential for formation of impervious crusts at the core boundaries which could affect coolability of the core.

Current Status. The core heatup and initial loss of core geometry are strongly influenced by oxidation of the zircaloy cladding. Because of this highly exothermic reaction, the rate of core heatup greatly accelerates at high temperatures. Numerous experiments, including out-of-pile experiments in Germany and in-pile experiments in the PBF, LOFT, NRU, and ACRR facilities, have confirmed the exponential rise in temperature of fuel rods, control rods, and other structural components due to cladding oxidation. These experiments have also characterized the influence of the oxidation of core materials on the subsequent loss of core geometry. For system conditions with only a limited degree of oxidation, the melting of core materials is a relatively smooth, candle-like process. On the other extreme, extensive oxidation forms protective oxide layers on fuel rod surfaces, resulting in only a limited change in geometry up to the melting point of the oxide layer but with ejection of molten materials into the coolant subchannels through pin-hole failures in the protective oxide layer. Similarly, the degree of oxidation influences the fragmentation and mechanical collapse of the core, with extensive oxidation leading to the formation of embrittled materials that readily shatter upon quench.

The fuel liquefaction process and resulting change in core geometry depend on the degree of cladding oxidation, melting of the metallic zircaloy, dissolution of the UO_2 by the molten zircaloy, the failure of the oxide shell which forms on the cladding outside surface, and the flow and freezing of the U-O-Zr ternary mixture as it moves downward along the rod. A large variability has been seen in the relocation of liquid U-O-Zr mixture, ranging from small molten droplets to substantial melting of the zircaloy that nearly fills the entire coolant channel. This variability appears to be a strong function of the failure time and cladding temperature.

The process controlling the loss of Ag-In-Cd control rod geometry after the liquefaction of the control material is not fully understood, and there are very few data for B_4C/Al_2O_3 burnable poison rods. It appears, however, that the liquefaction and relocation of control rod material may be controlled by some of the same processes (e.g. rapid, exothermic oxidation of the cladding in the presence of steam at high temperature) that control the fuel rod degradation. Combinations of vigorous ejection and rivulet flow of Ag-In-Cd and adjacent fuel rods have been noted, apparently due to the decrease in the difference between the system pressure and the cadmium vapor pressure. After its release from the control rods, the Ag-In-Cd may form aerosols which would affect the transport and chemistry of the fission products. The role of the B_4C/Al_2O_3 poison rods on fuel degradation is not known but may play a similar role; as mentioned previously, very few data exist on its significance in a severe accident.

Relevant TMI-2 Behavior. The TMI-2 core experienced the initial progression of core damage in a manner similar to what would be expected under many postulated severe accident scenarios. The interacting processes controlling the progression of initial core damage in TMI-2 are expected to be similar to those expected for other PWRs under comparable accident conditions.

AEP Contribution. TMI-2 data will be obtained to characterize the chemical and physical features of the remaining core debris as a function of location, both within the severely damaged core region as well as in the cooler outer regions. This will provide a spectrum of data as a function of increasing temperature and should produce an indication of how the core damage processes progressed and interacted.

5.2.2 Core Slump and Collapse

Issue Description. As the core degrades beyond the initial damage state (considered in Section 5.2.1), several phenomena become important which influence the progression of the accident and for which large uncertainties exist. These issues include the mechanics and extent of crust formation and failure, the formation and behavior of liquefied fuel (particularly with respect to interactions with core support structures, and relocation into the lower plenum), the coolability of core debris, and the rate and extent of cladding oxidation as core geometry changes become extensive.

Current Status. Data on the loss of core geometry due to fragmentation or mechanical collapse under severe accident conditions are limited. Fragmentation criteria for zircaloy cladding have been established in out-of-pile experiments. However, these experiments provide little insight into the breakup of other core materials or the characteristics of the resultant debris. Furthermore, the core geometry resulting from both cladding melting with fuel liquefaction and/or fuel rod fragmentation must be considered when determining subsequent core collapse into the lower plenum. It is not clear at present under what conditions the core debris may remain coolable. If large masses of the core remain noncoolable, they will continue to heat up and eventually remelt in their interior; and the resulting molten core materials would relocate downward once their supporting crusts fail. This process could repeat until a coolable geometry is attained or until the RV lower head is penetrated. Similarly, loose debris beds can dry out, heat up, and melt. The criteria for initial dry-out are complicated but have been characterized through numerous experiments as a function of bed characteristics and the requisite thermal-hydraulic conditions. If the relocating molten material from either the cohesive masses or loose debris cannot attain a coolable geometry within the confines of the core, the material would move downward into the lower plenum through the core support structure; or it could cause the collapse of parts of the core by the ablation and weakening of the core support structure.

The wealth of data acquired for core heatup and initial loss of geometry in the integral experiments in LOFT, PBF, and elsewhere stop short of this transition. Separate-effects experiments have been performed to evaluate the movement of molten material within loose debris beds [primarily for liquid-metal fast breeder reactor (LMFBR) materials and conditions], the ablation of structural materials by molten corium, and debris-coolant interactions. For debris-coolant interactions, the theoretical and experimental data base is extensive. However, data are currently limited that describe the transition from the initial loss of core geometry to a final coolable geometry, including the limited "dripping" of molten material into the lower plenum, as well as gross core collapse into the lower plenum.

Relevant TMI-2 Behavior. The TMI-2 core experienced significant relocation, and a large fraction of the core mass slumped and eventually relocated into the lower plenum. Crust formation appears extensive, and loss of coolable geometry apparently occurred during the progression of the accident. Thus, the TMI-2 core progressed through all of the processes associated with this issue, although the effects of restoration of coolant flow during the accident may complicate interpretation of the data.

AEP Contribution. Core bore samples will provide data on the physical and chemical properties of the core debris, as well as the spatial location of various forms of core debris (i.e., crust). Data will also provide the extent to which molten core materials interacted with the core support assembly.

5.2.3 Reactor Vessel Integrity

Issue Description. The timing and mode of RV failure following the relocation of hot core debris to the lower plenum is highly speculative, and models to describe such a failure have not been assessed with experimental data. These RV failure parameters affect the rate of ejection of core debris into the lower vessel cavity (which may contain water), and subsequently into the containment proper. Furthermore, the conditions of the core material (mass, composition, and temperature) at the time of vessel failure can have a major influence on the accident progression. The vessel failure mode and conditions of the core material are thus extremely important in assessing containment loadings due to direct containment heating or ex-vessel interactions between water in the cavity and hot core debris.

Current Status. The mode of postulated RV failure is still speculative at this time but may involve the rapid failure of vessel instrument penetrations through the lower head or the slower creep-rupture failure of the lower head or head welds. Analysis performed as part of the IDCOR program indicated that failure would most likely occur as a result of molten material on the lower head melting the instrument penetration nozzles within a few seconds to minutes. In the instance where core material enters the lower plenum as solid debris or is solidified in the lower plenum prior to contacting the lower head, the failure could take more time since the heat transfer to the lower head would be less efficient. The ORNL separate-effects experiments on creep rupture of small stainless steel vessels indicate that mechanical failure near vessel welds may occur within a few hours at temperatures elevated but significantly below the melting temperature of the vessel wall.

The experimental data base on the mode of vessel failure is limited. Some information has been obtained from experiments on the ablation of the vessel wall as molten material was ejected at high pressures.

Relevant TMI-2 Behavior. The RV in TMI-2 did not fail. However, there is evidence that a substantial fraction of the core relocated to the lower plenum and that much of this material was molten at one time. Thus, TMI-2 contains samples of core material that would have been ejected upon vessel failure and can also provide details regarding the extent of contact and interaction of molten debris with the vessel head and instrument penetrations, approximate temperature distributions in the lower head, and extent of damage to instrumentation tubes which penetrate the lower head.

AEP Contribution. Visual observations can indicate the extent of physical interaction between molten core materials and lower head structure. Samples of the core debris can provide data on mass and composition and estimates of melting temperature. Samples of instrument tubing can provide the extent of physical damage, type of materials interactions, and estimates of maximum temperature.

5.2.4 Hydrogen Generation After Core Disruption

Issue Description. Substantial agreement exists on zircaloy oxidation models. However, the most significant phenomena affecting in-vessel hydrogen generation are those associated with blockage formation by slumping fuel or any similar mechanism that inhibits access of steam to unreacted zirconium. These phenomena are not well known and represent a significant unresolved issue. Existing models show substantial disagreement in the rate, timing, and extent of hydrogen generation. These parameters can have a significant influence on containment loadings and the possibility of damage to equipment from hydrogen combustion. Data are necessary to confirm the many separate-effects models relating to the core degradation processes and also the coupled interaction of cladding oxidation and fuel/cladding dissolution and relocation. Also of importance in substantiating detailed core heatup models is the possibility of flows being channeled in such a way as to cause chimneys of hot hydrogen emanating from the core.

Current Status. The limited data available to assess the influence of core geometry changes on cladding oxidation are from in-pile experiments in PBF and small-scale, out-of-pile experiments at ORNL. Existing models used to predict such oxidation produce markedly different results due to basic input assumptions regarding the influence of core geometry changes on the flow of steam to unreacted cladding.

Relevant TMI-2 Behavior. During the in-vessel core damage progression in TMI-2, cladding oxidation occurred under conditions expected to be typical of other severe accident conditions of interest. Significant hydrogen was produced, as evidenced by combustion in the containment and noncondensible gas buildup in the RCS.

AEP Contribution. Examination of the TMI-2 core debris and determination of its composition will provide data on the extent of cladding oxidation. Data on the extent and location of flow blockages will also be useful in estimating the extent to which steam flow might have inhibited subsequent oxidation of relocated materials.

5.3 Fission Product Release and Transport

The release and transport of fission products within the RCS is a critical area of severe accident assessment. This phase represents the initial redistribution of fission products and provides the basis for subsequent source-term estimates which dictate accident consequence calculations. Several unresolved issues exist which are associated with this phase of the accident and for which TMI-2 data can provide relevant information. These issues are: release of (low-volatility) fission products during core degradation, fission product transport and chemical reactions within the RCS, tellurium behavior, and behavior of control rod materials. Each of these issues will be considered separately in the following subsections.

5.3.1 Release of (Low-Volatility) Fission Products During Core Degradation

Issue Description. Some fission products with lower volatility, including strontium, ruthenium, and lanthanum, can have significant deleterious health effects if they are released to the environment. Thus, the release of these materials from the core while the core remains in the RCS can have a significant bearing on their ultimate disposition. If significant release from the fuel occurs during this phase of the accident, the disposition of these materials in the RCS is very important, as is the potential for subsequent release from the RCS due to surface heating and/or sweepout. If most of these low volatiles remain in the melt, the potential for their release during ex-vessel core concrete interaction (in the event of RV failure) becomes important.

Current Status. Radioisotopes are categorized in Table 8 according to their volatility. This grouping is consistent with the WASH-1400[48] characterization. Possible chemical compounds of the elements are also summarized, and the boiling temperatures of the elements and compounds are shown. The high-volatility groups (i.e., chemical groups I, II, III, and IV.a) shown in Table 8 contain the noble gases, halogens, alkali metals, heavy chalcogens, and cadmium from the control rods; they are characterized by a boiling point less than 1600 K for the elemental form as well as the oxide compounds listed. The medium-volatility group is characterized by a boiling point less than 3100 K, i.e., UO_2 melting. This group includes fission product elements from the Group IV.b metals, alkaline earths, rare

TABLE 8. CORE MATERIAL AND FISSION PRODUCT VOLATILITY

WASH-1400 Group Number	Chemical Family	Element	Boiling[a] Temperature (K)	Volatility	Possible Compounds	Boiling[a] Temperature (K)	Volatility
Fission Products							
I	Noble gases	Kr[b]	121	High		--	--
		Xe[b]	166	High		--	--
II	Halogens	Br	332	High	CsBr[b]	1573	High
		I	458	High	CsI[b]	1553	High
					HI	238	High
III	Alkali metals	Rb[b]	973	High	RbI[b]	1577	High
					Rb_2O	--	--
					Rb_2O_2	1273	High
		Cs[b]	963	High	CsI[b]	1553	High
					CsOH	~1350[c]	High
					Cs_2O	--	--
					Cs_2O_2	923	High
					Cs_2UO_4[b]	--	--
IV.a	Heavy chalcogens	Se[b]	958	High	SeO_3	453	High
					SeO_2	613	High
		Te[b]	1235	High	TeO_2	~1450	High
					TeO_2O_2	--	--
					silver-telluride	--	--
					iron-telluride	--	--
					zirconium-telluride	--	--
					tin-telluride	--	--
					nickel-telluride	--	--
					chrome-telluride	--	--
					antimony-telluride	--	--
Nonfission Products							
	Control rod material	Cd	1038	High		--	--

TABLE 8 (continued)

WASH-1400 Group Number	Chemical Family	Element	Boiling Temperature[a] (K)	Volatility	Possible Compounds	Boiling Temperature[a] (K)	Volatility
Fission Products							
IV.b	Group VA metals	Sb[b]	2023	Medium	Sb_2O_3	1823	Medium
V	Alkaline earths	Sr	1657	Medium	SrO[b]	~3100[c]	Low
		Ba	1913	Medium	BaH_2	1673	Medium
					BaO[b]	~3500[c]	Low
					BaO_2	1073	High
VII	Rare earths	Eu	1870	Medium	Eu_2O_3[b]	--	--
		Sm	2051	Medium	Sm_2O_3[b]	--	--
		Pm	2733	Medium	Pm_2O_3[b]	--	-
Nonfission Products							
	Actinides	Am	2880	Medium	AmO_2[b]	--	--
	Control rod material	In	2353	Medium		--	--
		Ag	2485	Medium	AgI	1779	Medium
					silver telluride	--	--
	Structural materials	Sn	2543	Medium		--	--
		Cr	2945	Medium		-	-
		Ni	3005	Medium			
		Fe	3023	Medium	FeO	--	--
					Fe_2O_3	--	--
					Fe_3O_4	--	--

TABLE 8. (Continued)

WASH-1400 Group Number	Chemical Family	Element	Boiling[a] Temperature (K)	Volatility	Possible Compounds	Boiling[a] Temperature (K)	Volatility
Fission Products							
VI	Noble metals	Pd[b]	3413	Low	PdO	--	--
		Rh[b]	4000	Low	RhO_2	--	--
					Rh_2O_3	--	--
		Ru[b]	4173	Low	RuO_2	--	--
					RuO_4	--	--
		Mo[b]	4885	Low	MoO_2[b]	--	--
					Mo_2O_3	--	--
					MoO_3[c]	--	--
					MoO_3	~1300[c]	High
		Tc[b]	5150	Low		--	--
VII	Rare earths	Y	3611	Low	Y_2O_3[b]	--	--
		La	3730	Low	LaO	--	Medium-High
					La_2O_3[b]	4473	Low
		Ce	3530	Low	CeO_2	--	--
					Ce_2O_3[b]	--	--
		Pr	3485	Low	PrO_2	--	--
					Pr_2O_3[b]	--	--
		Nd	3400	Low	Nd_2O_3	--	--
	Actinides	Np	4175	Low	NpO_2[b]	--	--
		Pa	3505	Low	PuO_2[b]	~3570	Low
		Cm	--	Low	CmO_2[b]	--	--
	Tetravalents	Zr[d]	4650	Low	ZrO_2[b]	5273	Low

TABLE B. (Continued)

WASH-1400 Group Number	Chemical Family	Element	Boiling[a] Temperature (K)	Volatility	Possible Compounds	Boiling[a] Temperature (K)	Volatility
	Early transition elements	Nb	5015	Low	NbO_2[b]	--	Low
					Nb_2O_5	--	Low
Nonfission Products							
	Fuel	U	4091	Low	UO_2	~3450	Low
					U_3O_8	--	--
					U_4O_9	--	--
	Cladding	Zr	4501	Low	ZrO_2	5273	Low

a. Boiling temperature at 1 atm., data primarily from CRC Handbook of Chemistry and Physics, 56th Edition.

b. Probable chemical form of the fission product within the fuel

c. NAA-SR 132

d. Zirconium is both a fission product and a structural material.

e. Probable form of oxide released because of MoO_2 decomposition.

earths, actinides, and various control rod (other than cadmium) and structural materials. The volatility of elements in this classification is strongly dependent on chemical form. For example, BaO has low volatility, BaO_2 has high volatility, and BaH_2 has medium volatility. The chemical form of barium will depend primarily on fuel chemistry, the oxidizing potential in the RCS, and temperature. The low-volatility materials include fission product elements from the noble metals, rare earths, actinides, tetravalents, early transition elements, and uranium and zirconium from the fuel rods. Generally, the oxides of these elements also have low volatility; however, LaO and some of the noble-metal oxides are significantly more volatile than the element.

Fission-product volatility is dependent primarily on the chemical state of the fission products both within the fuel before release and within the primary system after release. The chemical state is, in turn, governed by the oxidation potential in the fuel and reactor coolant system. Fuel pellets initially have a stoichiometry of about $UO_{2.002}$. As the uranium is fissioned, however, the oxygen becomes partitioned among the fission products and fuel constituents, forming oxides with those having the greatest reaction potential. The oxides of the rare earths, yttrium, zirconium, and niobium, as well as some molybdenum oxides will remain in solid solution in the uranium oxide. It is obvious from Table 8 that chemical form and changes in chemical composition of the radionuclides must be considered. For example, under strong reducing conditions in a steam-starved environment, the oxides of barium and strontium may be reduced to the elements; and La_2O_3 (and the other rare earth oxides) may be reduced to LaO, which significantly enhances fission product volatility. Much greater releases of the medium- and low-volatility fission products are also possible under highly reducing conditions at temperatures up to fuel melting (i.e., ~3100 K).

Integral-effects in-pile experiments in PBF have indicated that fuel liquefaction, relocation and freezing of liquefied fuel, and fuel fragmentation and oxidation all have significant influence on fission product release from the core. The LOFT FP-2 experiment was designed to measure the release and transport of fission products and aerosols from a bundle of 100 1.7-m-long fuel rods subjected to a severe accident condition. Posttest examination of the bundle to characterize the damage state and correlate damage with fission product release is being considered. In-pile damage progression and fission product release experiments are being conducted in the NRU reactor at Chalk River, Canada. The PBF (32 1-m rods), LOFT FP-2 (100 1.7-m rods), and NRU (12 3.6-m rods) test bundles are, however, all very much smaller than the TMI-2 core (35,000 3.6-m rods); and the effects of scale have to be taken into account in order to extrapolate the results to a full-scale PWR.

Most fission product release models (CORSOR, GRASS, steam oxidation) generally calculate fission product release using correlations only as a function of time at temperature, with the exception that efforts are being made to incorporate the effects of fuel morphology and liquefaction in the GRASS code.

Experiments to measure the releases of the low-volatility fission products are difficult to perform because high temperatures are needed (T >2400 K). As a consequence, few data pertaining to the lower-volatility fission products are available, resulting in a relatively large uncertainty for the release of these fission products.

Relevant TMI-2 Behavior. Portions of the TMI-2 core achieved melting temperatures, and substantial relocation of liquefied material occurred. Thus, the core achieved temperatures similar to what would be expected in most severe accidents considered for source-term evaluations. The release of the lower-volatility fission product species from the TMI-2 core, therefore, should be representative of releases expected during many postulated severe core accidents.

AEP Contribution. Analyses will be conducted to determine the retention and chemical form of fission products in the various damaged core materials (liquefied fuel, fragmented fuel, intact fuel, molten fuel). This should yield an important contribution to understanding the relationship between fuel damage state and lower-volatility fission product release. This information, along with metallurgical estimates of temperatures, should be valuable for model development and validation and will serve as a benchmark for validating the results of small-scale experiments simulating a full-scale PWR.

5.3.2 Fission Product Transport and Chemical Reactions Within the RCS

Issue Description. There exist substantial uncertainties in the area of chemical interactions among fission products and among fission products and aerosols. Transport and deposition models which account for these complex effects have not been verified, and their predictive capability is very uncertain.

Current Status. The potential to reduce source terms by retention of fission products in the RCS has been recognized to be very important by recent studies. It has also been recognized by APS, DOE, NRC, ANS, and IDCOR that there are substantial uncertainties in this area, especially with respect to chemical interactions among fission products and structures and among fission products and aerosols. TRAP-MELT, RAFT, and other transport and deposition codes have as yet unverified models for important chemical reactions; and data relevant to this issue from experiments which represent the important parameters are limited. Recently, the influence of radiation on iodine chemistry within the RCS has been identified as a potentially significant effect, especially with respect to the disassociation of CsI leading to the formation of other chemical forms of iodine.

Relevant TMI-2 Behavior. The TMI-2 accident produced substantial fission product release under a high radiation field into a representative primary system. Thus, important chemical reactions and primary system retention mechanisms occurred which should provide additional information on the impact of radiation effects.

AEP Contribution. The information on fission product chemical forms and structural material temperatures should be useful in evaluating the potential for fission product revaporization from RCS surfaces. Measurements of deposited fission products which have remained attached to surfaces and their relationships to oxide layer structures on RCS surfaces can contribute to the understanding of the chemical environment (such as oxygen potential) during the accident.

5.3.3 Tellurium Behavior

Issue Description. This issue is actually a subset of issue 5.3.2 above. However, it has been identified separately due to the significance of tellurium release on health effects and due to the unique features of tellurium chemical behavior under the conditions of interest. The basic issue is the extent to which tellurium reacts with, and is therefore sequestered by, zircaloy and stainless steel and the release and chemical form of tellurium when these metals are subsequently oxidized. A second issue is the potential for retention of tellurium through chemical interactions with the fission product metals to form tellurides.

Current Status. Current experiments that address this issue include both out-of-pile and in-pile tests. The out-of-pile work is the measurement of fission product release from single irradiated rod segments at ORNL; in-pile work is ongoing at TREAT, PBF (Test SFD-1-4), and NRU. However, only the PBF experiment represents an integral-effects in-pile test up to the UO_2 melting point utilizing irradiated fuel, representative control materials, or bundle geometry oxidation potential due to coolant flow. To date, fission product release models treat tellurium release using gross empirical expressions which have large uncertainties. This issue has been identified by APS, DOE, and NRC as needing resolution.

Relevant TMI-2 Behavior. The behavior of fission product tellurium during the TMI-2 accident should be representative of that expected during many postulated severe core accidents.

AEP Contribution. Data from samples obtained from material in the TMI-2 RV should provide information on how much tellurium is in the vessel and core debris, the location, and materials with which it is associated as well as principal chemical compounds.

5.3.4 Behavior of Control and Poison Rod Materials

Issue Description. Aerosol generation within the RV is important, because fission product vapors may interact with the aerosols either chemically or physically and then be transported (and deposited) with the aerosol. Studies by APS, DOE, NRC, IDCOR, and ANS have recognized this issue. There are many aerosol sources in the RV (control materials, structural materials, fission products); but the mechanisms for aerosol formation from these materials, particularly control rod material, are not well understood. Specifically, the failure mode, melt behavior, and freezing and remelting characteristics, as well as separation of an alloy into its constitutive part and affinity of each part for fission products, all affect the role of control and poison rod aerosols. Understanding the process of aerosol formation involves understanding the core and structural degradation processes in the RV (such as control rod melting and expulsion of control materials when the cladding ruptures), the production of vapors, and the mechanisms and processes which control the formation of aerosols.

Current Status. Recent experiments have contributed to the understanding of the effects of control rod material on fuel degradation. Mechanistic models for the formation of aerosols from degrading core materials are available, (e.g., the SCDAP code models control rod meltdown and the RAFT code models aerosol nucleation) but have not yet been checked against an experimental data base. In-pile experiments have recently been run with control materials included in the fuel bundles (PBF, TREAT, LOFT) to investigate the effects of control material aerosols on fission product release and transport. Also, a few out-of-pile experiments have been run to investigate control rod degradation processes, interaction with fuel rods, and aerosol generation (KfK, ORNL); and more are planned (KfK, ORNL). All these experiments are, of course, small-scale, with LOFT FP-2 being the largest (100 fuel rods and 11 control rods).

Relevant TMI-2 Behavior. Prior to the accident, the TMI-2 core contained typical control rods (Ag-In-Cd) and burnable poison rods (B_4C/Al_2O_3). The accident produced temperatures which liquefied these rods, thus producing an environment relevant to severe accident behavior.

AEP Contribution. Information from the TMI-2 examinations on the distribution of Ag-In-Cd control rod materials and B_4C/Al_2O_3 burnable poison rods will be used to help identify evidence of flowing and resolidification of control rod materials, as well as important aerosol release, and thus contribute to a better understanding of aerosol formation. Direct measurements of aerosol generation mechanisms are not possible and will have to be inferred from examination of core material. The data from TMI-2 can also be used in model development and validation and as a benchmark for understanding the applicability of small-scale experiments (e.g., by measuring the ratio of Ag-In-Cd to determine the relative release of each).

6. INFORMATION DISSEMINATION AND INDUSTRY COORDINATION

The fact that a severe accident of a degree previously considered incredible took place at TMI-2 with insignificant health impact could have enhanced the public confidence in the safety of nuclear power. However, it was because this type of accident has been traditionally considered "incredible" and beyond the scope of "design basis accidents" normally considered for licensing purposes that serious questions were raised regarding the adequacy of the NRC's safety analysis criteria. As a result, the effect of the accident was precisely the opposite; the public lost confidence in the safety record of the industry, and enhanced, expensive safety features and procedures were imposed on new and operating plants. The objective of the TMI-2 Accident Evaluation Program is to obtain information from the plant and from the accident records that will provide a comprehensive understanding of the accident and its consequences which can then be used with the results of other world-wide research to address some of the important technical issues which strongly impact nuclear power today.

One of the items in the Information and Industry Coordination Program element's charter is to maintain a close relationship between the AEP and the nuclear industry. The work tasks in this element are aimed at obtaining industry input into the direction and work of the program, transferring knowledge learned from this research to the industry, coordinating the efforts of universities and consultants, and coordinating the standard problem exercise and disseminating its results.

A second work task in the Information and Industry Coordination element is to provide the public with informative and factual information from the TMI-2 Accident Evaluation Program to enhance public confidence in the safety of nuclear power plants. One specific objective is to make the public aware that, while the accident was very serious from the standpoint of plant damage, the public health and safety were not placed in jeopardy. Finally, the Industry & Information Coordination task has the responsibility of recommending to the Examination Requirements and System

Evaluation program element the addition or expansion of efforts which may be required to increase the public credibility or acceptance of the results of the Program.

6.1 Industry Coordination

Industry input, from vendors, utilities, and consultants, is vital to ensure that the issues being addressed by the AEP are those that are of interest to the nuclear power industry. In preparing this document, industry input has been used to identify the issues of major importance. Contact with industry for input to the program will be maintained throughout the program to ensure that changing concerns are considered in ongoing work.

Industry coordination also includes participation in the accident evaluation and data analysis phases of the Program. Specifically, this work task should include industry participation in:

- Examinations of plant samples,
- Evaluation of on-line plant data during the accident,
- Severe accident code calculations using the accident to benchmark the codes,
- Development and modification of the accident scenario.

Finally, industry coordination also requires the dissemination of information to the industry. As important results are developed which could impact the industry, the Information and Industry Coordination element will transmit the results as expeditiously as possible to industry. To achieve this latter objective, the information task will include:

- Periodic meetings for industry and the national laboratories to report on the current status of the program and to emphasize important results,

- Encouraging special sessions and meetings of professional societies to provide external peer review of the program results,

- Promoting the prompt publication of the results of the Program in technical literature.

6.2 Public Information

The cleanup, examination, and analysis of the TMI-2 accident continues to command a great deal of public interest. This was demonstrated when the video tapes of the inspection of the lower head became available and when molten fuel was discovered in the core debris. The Information and Industry Coordination element is responsible for the timely release of information from the examinations and analyses of the accident, using such media as press releases, video tapes, and public-oriented reports. The public releases of the results of the AEP will be coordinated with GPU and DOE.

Two specific tasks are included in the public information element:

- To prepare, for the general public, an accident scenario, along with necessary supporting data, that clearly describes what happened and why it happened,

- To provide a sufficiently complete accounting of the end-state fission product inventory. As a goal, these data will support the conclusion, based on monitoring data, that the environmental releases due to the accident were small.

6.3 Program Interface

Several organizations external to the immediate program provided services or consulting to the program. The Industrial Review Group (IRG) and the Technical Evaluation Group (TEG) were two such entities. These two groups were recently reorganized and reconstituted into the Accident Evaluation Advisory Group (AEAG) to provide a peer review of this document. It is intended that the Group be reassembled in the future on an ad hoc basis. In addition, several consultants and organizations are providing expertise in the examination and analysis tasks. The interface and coordination function for all of these organizations is provided by the Information and Industry Coordination element.

6.4 Standard Problem Exercise

The conduct of a standard (benchmark) problem exercise entails a considerable effort both in preparation and in coordination of the exercise. The pre-exercise arrangement of participants, coordination of special information requirements for each participant, publishing of the standard problem, and collection of the exercise results, as well as providing feedback to the participants, will be coordinated by the Information and Industry Coordination element.

7. CONCLUDING REMARKS

The TMI-2 accident offers unique information for studying the in-vessel behavior of a light-water reactor during a severe accident with extensive core melting, fuel migration, and interaction with the reactor vessel structures. Extensive industry assessments of the accident have reached a consensus:

> Additional end-state characterization data and supporting analysis work to interpret the data will provide a basis for completing our understanding of the mechanisms controlling the core damage progression, resulting fission product behavior, and the end-state distribution of fission products within the reactor system. Completing our understanding of these aspects of the accident is necessary to relate the TMI-2 findings towards resolution of severe accident technical issues.

The elements of the TMI-2 Accident Evaluation Program described represent a joint DOE/industry-developed approach to provide the TMI-2 data and the necessary supporting analysis to complete our understanding of the accident and to relate the major TMI-2 research findings to current severe accident technical issues.

The Examination Requirements and System Analysis (ERASE) program element is the integrating element of the program. This element will identify data needs and interpret the data as they become available. The TMI-2 standard problem will be an important part of this work. The data interpretation work will be supported by various industry groups to ensure that all viewpoints are integrated into the accident scenario development.

The Sample Acquisition and Examination (SA&E) program element will acquire the needed reactor system samples and provide the necessary examination work to characterize the samples. The examination work will also be a joint DOE- and industry-supported activity.

The Data Reduction and Qualification program element will qualify and organize the TMI-2 data into a centrally located data base. The data base will provide a valuable resource to the industry for future accessibility to the TMI-2 research results.

The Information and Industry Coordination program element will ensure that the needs of the technical community are met in an effective manner and will provide the focal point for information dissemination.

Thus, the TMI-2 Accident Evaluation Program will provide necessary and sufficient end-state characterization data and engineering support to interpret and integrate the many sources of TMI-2 data into a comprehensive accident scenario which represents a consensus understanding of the accident. Without completing this work, the TMI-2 data will be fragmented; and interpretations of the accident progression will likely differ significantly, thus reducing the effectiveness of the TMI-2 data towards resolving important reactor safety issues.

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