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INFORMAL REPORT

TMI-2 ACCIDENT EVALUATION PROGRAM SAMPLE ACQUISITION AND EXAMINATION PLAN FOR FY 1987 AND BEYOND

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EG&G Idaho

Work performed under DOE Contract No. DE-AC07-76ID01570

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ABSTRACT

The purpose of this document is to update the description of the sample acquisition and examination portion of the TMI-2 Accident Evaluation Program to December 1, 1986. Additions to the previous plan (TMI-2 Accident Evaluation Program Sample Acquisition and Examination Plan, EGG-TMI-7132, January 1986) include; (a) the results of sample acquisitions and examinations and reactor disassembly activities conducted between October 1985 and December 1986, (b) a proposed TMI-2 accident sample examination program for the CSNI Joint Task Force on Three Mile Island 2 (foreign countries), and (c) a list of over 400 reference documents about the TMI-2 accident core damage, fission product release and sample acquisition and examination program. The principle findings from the recent sample acquisitions and examinations and reactor disassembly activities are as follows:

a. A complete estimate of damage and reconfiguration of the core was developed as follows:

Core Region	Percent of Core Material	
Still standing rod bundle geometry	42	
Loose debris (unmelted and previously molten core material mixture) below the cavity in the upper core region (the cavity was 26 percent of the original core volume)	25	
Previously-molten core material:	33:	
retained in core boundary		20
escaped from core boundary		13

b. The principle source of radiation in the reactor building basement is fission products absorbed by the concrete and the basement sludge contains relatively small amounts of core materials and fission products.

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1. INTRODUCTION

1.1 Purpose and Intent

The purpose of the Three Mile Island Unit 2 (TMI-2) Accident Evaluation Program Sample Acquisition and Examination (TMI-2 AEP SA&E) program is to develop and implement a test and inspection plan that completes the current-condition characterization of; (a) the TMI-2 equipment that may have been damaged by the core damage events, and (b) the TMI-2 core fission product inventory. The characterization program includes both sample acquisitions and examinations and in situ measurements. Fission product characterization involves locating the fission products as well as determining their chemical form and determining material association. The intent of the TMI-2 AEP SA&E Plan documentation is to describe the TMI-2 Sample Acquisition and Examination Plan in a manner that provides sufficient information for "stand alone" comprehensiveness.

1.2 Project Genesis

The TMI-2 Sample Acquisition and Examination will be accomplished in accordance with United States Department of Energy contractor business practices. These practices require rigorous project planning, control, and reporting to ensure that government-funded research programs are accomplished in a way that maximizes research results and the effective utilization of program resources. The TMI-2 AEP SA&E Plan will provide those assurances.

This plan is part of the EG&G Idaho, Inc., TMI-2 Programs project, which is described in the EG&G Idaho, Inc., TMI-2 Programs Division Master Plan, Revision 6, dated December 16, 1986. Included in this Master Plan is an outline of the EG&G Idaho, Inc., TMI-2 Programs Work Breakdown Structure (WBS). The SA&E program is composed of two (Level 4) elements; Sample

Acquisition (WBS No. 751400000) and Sample Examination (WBS No. 755400000). These two elements are within the TMI-2 Accident Evaluation Program (Level 2 WBS No. 75B000000).

The TMI-2 Accident Evaluation Program will accomplish the Department of Energy's program objectives of understanding the TMI-2 accident, disseminating this knowledge to the nuclear industry, and aiding in the resolution of severe accident and source term issues. The program's work is divided into four elements:

1. Examination Requirements and Systems Evaluation

2. Sample Acquisition and Examination

3. Data Reduction and Qualification[a]

4. Information and Industry Coordination.

The Examination Requirements and Systems Evaluations element is responsible for defining program scope and technical objectives, defining sample acquisition and examination data requirements, determining the accident scenario, and providing a standard problem and applying the research results to aid in the resolution of the severe accident source term issues. The Sample Acquisition and Examination element is responsible for obtaining the samples specified by the Examination Requirements and Systems Evaluation element from the TMI site, for examination of the samples, and for reporting the examination results. Data Reduction and Qualification is responsible for developing and maintaining the TMI-2 data base and for evaluating and qualifying on-line instrumentation and recorded data. Information and Industry Coordination is responsible for information transfer, coordination of review and consulting groups, interfacing with other source term research programs, and coordination of the TMI-2 standard problem exercise.

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a. Analytical and Experimental Support in Revision 4 of the Master Plan.

The tasks within the four work elements are designed to accomplish the following technical objectives:

- Identify and quantify the parameters and processes which controlled the progression of damage to the lower core support assembly, instrument penetration nozzles and guide tubes, and possibly to the reactor vessel lower head,

- Determine the plant-wide fission product behavior (source term), concentrating on release from the fuel and transport and retention in the primary cooling system,

- Provide a data base that contains the examination (and analysis) results,

- Provide a standard problem of the TMI-2 accident that includes the examination results and against which the severe accident analysis codes and methodologies can be benchmarked,

- Apply the TMI-2 accident evaluation research toward resolution of severe accident source term technical issues.

The Sample Acquisition and Examination element is specifically responsible for the collection of sample materials from the TMI-2 plant, the examination of those samples (to provide the data specified by the Examination Requirements and Systems Evaluation element), the interpretation and reporting of the examination results, and the coordination of examination activities at other laboratories. This program element is also responsible for providing engineering support for the sampling activities and for sample shipment.

1.3 Background and History

Although the March 28, 1979 accident at TMI-2 involved severe damage to the core of the reactor, it had no observable effects on the health and safety of the public in the area.[1] That such a severe core disruption accident would have no consequent health or safety effects has resulted in the questioning of earlier light water reactor (LWR) safety studies and estimates. In an effort to resolve these questions, several major research programs have been initiated by a variety of organizations concerned with nuclear power safety. The U.S. Nuclear Regulatory Commission (NRC) has embarked on a thorough review of reactor safety issues, particularly the causes and effects of core damage accidents. Industrial organizations are conducting the Industry Degraded Core Rulemaking (IDCOR) program. The U.S. Department of Energy (DOE) has established the TMI-2 Program to develop technology for recovery from a serious reactor accident, and to conduct relevant research and development that will substantially enhance nuclear power plant safety.

Immediately after the TMI-2 accident four organizations with interests in both plant recovery and accident data acquisition formally agreed to cooperate in these areas. These organizations, commonly referred to as the GEND Group--General Public Utilities, Electric Power Research Institute, Nuclear Regulatory Commission, and Department of Energy--are presently actively involved in reactor recovery and accident research. At present, DOE is providing a portion of the funds for reactor recovery (in those areas where accident recovery knowledge will be of generic benefit to the U.S. light water reactor industry) as well as the preponderance of funds for severe accident technical data acquisition (such as the examination of the damaged core). However, the core examination, rather than being an open-ended program of scientific inquiry, must be well planned and executed and must be designed to meet specific technical objectives.

The EG&G involvement with the TMI-2 accident has been continuous, initially providing technical support and consultation from the Idaho National Engineering Laboratory (INEL). In 1979, EG&G received an assignment from DOE to collect, analyze, distribute, and preserve significant technical information available from TMI-2. In 1981, the technical information assignment was expanded to include conducting research and development activities intended to effectively exploit the generic research and development challenges at TMI-2. In conjunction with this expanded assignment an organization element for Offsite Core

Examination was developed. This evolution continued, and in January 1985 DOE agreed to expand the EG&G involvement to include an evaluation of the TMI-2 accident that would develop an understanding of the accident sequence-of-events in the area of core damage and escape of core radionuclides (fission products) and materials. The TMI-2 Accident Evaluation Program document[2] implements the January 1985 agreement, defines the program required to understand the accident, and contains the guidelines and requirements for TMI-2 sample acquisition and examination.

The TMI-2 AEP SA&E Plan evolved from the requirements set forth in the TMI-2 Accident Evaluation Program document (see Reference 2). The program description provides the guidelines for the postaccident core condition and fission product inventory characterization. Examination requirements documents written previously include the GEND Planning Report 001[3] and the TMI-2 Core Examination Plan.[4] The current program description document is an extension of the preceding examination requirements documents with appropriate additions and changes to account for our enhanced understanding of the TMI-2 accident and the resultant refinements in sample and examination requirements.

The already-completed portion of this SA&E program includes in situ measurements and sample acquisition and examinations involving private organizations and state and federal agencies. It has provided the postaccident core and fission product end-state data that indicate the following:

1. Large regions of the core exceeded cladding melting (~2200 K), and significant fuel liquefaction by molten zircaloy and some fuel melting occurred with temperatures up to at least 3100 K.

2. Core materials relocated into the reactor vessel lower plenum region from the core, leaving a void in the upper core region equivalent to approximately 26% of the original core volume. Between two and twenty metric tons of core and structural materials now reside in the space between the reactor vessel bottom head and the elliptical flow distributor.

3. Fission product retention in core materials is significant, and the retention of fission products outside the core was primarily in reactor cooling system (RCS) water, water in the basement, and in basement concrete.

Significant consequences resulting from these findings include (a) increased technical interest in the TMI-2 accident because it represents a severe core damage (SCD) event in full-scale and provides evidence of a large difference between actual and predicted SCD event offsite radiation release, (b) a reconsideration of the plans and equipment for defueling the TMI-2 reactor, and (c) an expansion in the TMI-2 accident examination plan to determine the consequences of high temperature interactions between core materials and reactor vessel lower plenum structural and pressure boundary components and to determine the release from the fuel of the lower volatility fission products.

The increased technical interest induced the formation (1986) of a Joint Task Force on Three Mile Island 2 by the Committee for Safety of Nuclear Installations (CSNI) of the Organization for Economic Cooperation and Development (OECD). The task force includes nine foreign countries that indicated a desire to examine TMI-2 samples.

Section 2 of this report contains an overview of the guidelines and requirements set forth in the TMI-2 Accident Evaluation Program document, continuing with a description of what would be required to meet these guidelines and requirements, and concluding with a proposal for sample acquisition and examination tasks that can be accomplished within the available resources. Sections 3, 4, and 5 contain details of the proposed SA&E tasks. Section 6 summarizes the technical and administrative support for management of the SA&E Program. Section 7 is a summary containing the cost and schedules for the proposed SA&E program and the summary description of how the authorizing of the performance of work further subdivides the WBS and provides controls during the work accomplishment. Appendix A is a list of TMI-2 accident reference documents for use in planning and performing the SA&E program. Appendix B is a proposed TMI-2 Sample Examination Plan for the CSNI task force. Appendix C is a GPUN technical bulletin listing the core component identification marking.

2. OVERVIEW

2.1 Overview of SA&E Requirements from the Accident Evaluation Program Document

The TMI-2 Accident Evaluation Program document (see Reference 2) states that substantial contributions can be made to the resolution of SCD accident technical issues by developing an understanding of the TMI-2 accident sequence and consequences. These issues were combined into three broad technical areas: reactor system thermal hydraulics, core damage progression and reactor vessel failure, and fission product release and transport.

Table 1 in the TMI-2 Accident Evaluation Program document lists the technical issues to be addressed in TMI research. To ensure optimum results from the available program resources, the technical issues were prioritized as shown in Table 1. Two prioritization criteria were used. The first criterion is the potential of the TMI-2 sample examination data to directly enhance the understanding of each issue. Issues that could be addressed directly using data that can be obtained from TMI were prioritized as high. Low or medium priority was assigned to issues that could not be directly addressed using TMI-2 data. The second prioritization criterion is based on the relative importance of each issue to enhance the understanding of severe accident source terms. These second priorities were obtained from recommended priorities from independent industry research and from engineering judgment of the relationship of the technical issues to the environmental source term.

The sample acquisition and examination tasks will provide data to identify and quantify the mechanisms controlling core damage progression and fission product release, transport, and retention. The basic data needs, associated samples from the plant, and the overall priority of the acquisition and examination tasks are summarized in Table 2. The relative priority of the acquisition tasks is based on a subjective weighting of the associated technical issues, applicability of the TMI-2 data to the issues, and applicability of the data for establishing a consistent understanding

TABLE 1. PRIORITIZED LIST OF TECHNICAL ISSUES TO BE ADDRESSED VIA TMI RESEARCH

	Application of Data to Issue	Priority
Reactor System Thermal Hydraulics		
1. Coupling between core degradation, reactor vessel hydraulics, and fission product behavior (Integrated severe accident code)	Direct	High
2. Reactor system natural convection	Indirect	Medium
Core Damage Progression and Reactor Vessel Failure		
1. Damage progression in core	Direct	High
2. Core slump and collapse	Direct	High
3. Reactor vessel failure modes	Direct	High
4. Hydrogen generation after core disruption	Indirect	Medium
5. Alpha mode containment failure[a]	Direct	High
Fission Product Release and Transport		
1. Release of low-volatility fission products during fuel degradation	Direct, Indirect	High
2. Chemical reactions affecting fission product transport	Indirect	High
3. Tellurium behavior	Indirect	High
4. Fission product and aerosol deposition in the reactor cooling system	Indirect	Low
5. Release of control rod materials	Direct	High
6. Aerosol generation mechanisms	Direct, Indirect	High
7. Revaporization of fission products in the upper plenum	Indirect	Low
8. Core-concrete interaction	Indirect	Medium

a. Steam-explosion-accelerated missile penetration of reactor building wall.

TABLE 2. PRIORITIZED LIST OF TMI DATA NEEDS AND SAMPLE ACQUISITION TASKS

Primary Data Needs from TMI-2	Sample Data Acquisition Tasks	Prioritizaton Criteria: Technical Issue(s)[a] Priority	Prioritizaton Criteria: Data Applicability to Issue	Prioritizaton Criteria: Data Applicability for Establishing Consistent Accident Scenario	Overall Priority of Aquisition Task	Comments
1. Gross structure of core, core support structures, instrument structures, RPV lower head.	a. Video probe data through core bore channels (core and lower plenum).	High	High	High	High	a. Video inspections are high priority information needs.
	b. Topography of core and lower plenum regions.	High	High	High	High	b. Acoustic characterization of hardpan below debris bed is planned.
	c. Acquisition of core bore.	High	High	High	High	c. Qualitative data from core boring will provide valuable insights into damage.
2. Peak temperature, core and core support materials interactions, and core boundary structures.	a. Distinct fuel assembly samples.	High	High	High	High	a. Will provide data on core boundary conditions (radially), control and poison rod behavior, and fuel degradation.
	b. Core bore samples plus video charcterization to correlate with examination results.	High	High	High	High	b. Core bores are primary samples for determining temperatures, materials, and fission products vs location in the core and lower plenum.
	c. Large volume samples of core and lower plenum debris.	High	High	High	High	c. Necessary for extrapolating smaller sample material and fission product data and for debris bed characterization.
	d. Core former wall samples.	High	Medium	Medium	Medium-High	d. May not be required if intact.
	e. Core support assembly samples.	High	High	High	High	e. Extent of damage (chemical and thermal interactions) needs to be determined.
	f. Instrument structures samples.	High	High	High	High	f. Very important to assess vessel failure modes.
	g. Reactor vessel wall samples.	High	Medium	Medium	Medium-High	g. May not be required if undamaged.
	h. Fuel assembly upper grid and/or end boxes.	High	Medium	Medium	Medium	h. Judged to be important in establishing core boundary conditions.
	i. Fuel rod segments from upper core region.	High	Medium	Low	Medium	i. Important for fission product release, local oxidation.

TABLE 2. (continued)

Primary data Needs from TMI-2	Sample Data Acquisition Tasks	Prioritization Criteria: Technical Issue(s) Priority[a]	Prioritization Criteria: Data Applicability to Issue	Prioritization Criteria: Data Applicability for Establishing Consistent Accident Scenario	Overall Priority of Acquisition Task	Comments
3. Fission Product Release and Transport						
A. Retained fission products in core materials.	a. Distinct fuel assembly samples.	High	High	High	High	a. Sufficient examinations are required for characterizing the retained fission products (important high and low volatility species).
	b. Core bore samples.	High	High	High	High	b. Core bore samples are primary sources of data from core and lower plenum.
	c. Large volume samples of core and lower plenum debris.	High	High	High	High	c. Large volume samples necessary to increase detectability limit for some important radioisotopes.
B. Retained fission products on primary cooling system surfaces.	a. Upper plenum surface samples.	Medium-[c] High	Medium-Low	Medium	Medium	a. Surface deposition is important; however, only undissolvable component remains and is known to be very small. Additional data on horizontal surfaces would be used for evaluating separate effects experiments.
	b. Primary cooling surface samples. • Access covers from steam generators and pressurizer. • Sediment from steam generators and pressurizer. • RTD thermowells.	Medium-High[c]	Low	Medium	High	b. Surface deposition is important; however, only undissolvable component remains and is known to be very small. Samples from accessible locations will complete RCS inventory. Sample locations include A- and B-loop steam generators, manhole access covers (surface deposits and any accessible sediment), pressurizer, and hot leg RTD thermowells.
C. Retained fission products in containments basement.	a. Sludge samples	High[b]	Low[b]	High	High-Medium	a. Major final fission product repositories are known to be the reactor vessel and the containment basement. Uncertainty in containment inventory is still large.
	b. Basement concrete wall samples.	High[b]	Low[b]	High	High-Medium	b. Major final fission product repositories are known to be the reactor vessel and the containment basement. Uncertainty in containment inventory is still large.
D. Retained fission products in transport pathway outside the reactor cooling system (RCS) excluding the containment basement.	Non specified.[e]	High[b]	Low[b]	High	Low[d]	a. These examinations and data are primarily for definition of the accident scenario. The existing data requires more evaluation to 1) integrate the information into the accident scenario and 2) determine the need for additional samples/data.

TABLE 2. (continued)

Primary Data Needs from TMI-2	Sample Data Acquisition Tasks	Prioritizaton Criteria: Technical Issue(s)[a] Priority	Prioritizaton Criteria: Data Applicability to Issue	Prioritizaton Criteria: Data Applicability for Establishing Consistent Accident Scenario	Overall Priority of Aquisition Tasks	Comments
E. Fission product chemical form	a. Fission product chemical form from all core material samples.	High	Medium	Medium	Medium-High	a. Applicability of data obtained to date to fission product chemical form during the accident needs confirmatory evaluation.
4. Reactor system natural convection	a. Upper plenum temperature distribution	Medium	Medium	Low	Medium-Low	a. Reactor system natural convection heating was low in TMI. The confounding effect of B pump transient will make it difficult to evaluate natural convection cells in the reactor vessel.
5. In-vessel coupling of core degradation, thermal hydraulics, and fission product deposition	Data acquisition tasks 2a, 2b, 2c, 2d, 2h	High	Medium	Medium	Medium-High	a. End-state characterization data will have to be coupled with qualified online plant data and reactor systems models to define consistent accident scenarios. Coupled phenomena can only be estimated from code sensitivity calculations.

a. The priority in general applies to the technical issue grouping from Table 9 of the Sepetmber 1985 draft TMI-2 Accident Evaluation Program document.

b. Fission product retention in containment is a very high priority severe accident issue, but primarily for accidents where the core has penetrated the reactor vessel and there is significant interaction between the concrete and the molten core, with vaporization or aerosol formation directly into the containment atmosphere. The TMI-2 accident did not progress to that stage.

c. This specific technical issue is rated as medium priority for all severe accidents except the interfacing systems LOCA or "V" sequence, for which it is rated high.

d. Ranking reflects our knowledge that highest concentrations of fission products are probably in the core material and the containment basement. Also, much of this portion of the fission product pathway has already been sampled

e. This portion of the fission product transport pathway has been extensively sampled. Additional samples are not requested until a definite need is established.

of the accident. The prioritization process produced a list that assigns highest priority to samples and examinations that will provide data that directly characterize core damage progression and fission product release from the fuel. Next in relative importance are data that will characterize retained fission products in the containment basement, fission product chemical form, and structural damage within the lower plenum. The lowest priority data are those related to fission product retention in the primary cooling system and structural peak temperatures. Additional data to characterize the retention of fission products in the containment (excluding the basement) and auxiliary building transport pathways are not required at this time.

The sample acquisition tasks are listed in Table 3. This listing reflects the prioritization established in Table 2 as well as the availability of samples and the sequential need for the data to provide a consistent understanding of the accident. For instance, the core bore and associated video and acoustic information will provide data relevant to core damage progression and fission product retention in the core materials; therefore, these samples are listed before samples of the core support assembly (CSA) and lower plenum structures. Also, the CSA and lower plenum structural samples will not be available until the core has been removed from the CSA.

The basic data/measurements listed in Table 2 consist of peak temperatures, physical and chemical state of the core and structural materials, physical and chemical interactions between the fission products, core, and structural materials, the chemical form and concentrations of the retained fission products in the core and reactor coolant system, and the fission product transport pathway within the containment and auxiliary building. The measurements are required in sufficient number to map the distribution of the characteristic being measured. The data/measurements needs are reviewed including prior TMI-2 Core Examination Plan accomplishments in the following paragraphs. The items are discussed in the order of priority listed in Table 3.

TABLE 3. SUMMARY OF PRIORITIZED SAMPLE ACQUISITION TASKS

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1. Central core bore to the lower core support plate, and visual examination.

2. Central core bore to the lower head, and visual examination.

3. Large volume sample from upper debris.

4. Topography of the crust below the debris bed.

5. Mid-radius core bores to the lower plenum (3 bores).

6. Local large volume samples of debris from the core support assembly region.

7. Local large volume samples of the debris resting in the bottom of the reactor vessel.

8. Two intact, part length fuel assemblies from control rod and poison rod locations.

9. Outer-radius core bore to the lower core support plate.

10. Basement sediment samples.

11. Concrete samples from containment basement walls and floors.

12. Reactor 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.

13. Samples of the interaction zone between the core materials and the lower core support assembly.

14. Samples of the interaction zone between the instrument guide tube structures and core material.

15. Samples of the interaction zone between the reactor vessel lower head surface and the lower core debris materials.

16. Samples of the interaction zone between the core former wall and the core.

17. Fission product retention surfaces in upper plenum.

18. Upper plenum leadscrews.

19. Upper end boxes, control rod spiders, and holddown springs from top of the core.

20. Fuel rod segments from the debris bed.

Core Bore Samples (Table 3, Tasks 1, 2, 5, and 9). Core material samples are required that will allow multidimensional (axial, radial, azimuthal) interpretation of the core damage; i.e., cladding melting, fuel liquefaction and relocation, freezing of the molten core materials, and subsequent remelting and slumping of the core materials. This requirement necessitates a number of continuous axial samples of core materials through the core and lower plenum regions. Thirty core bore samples are identified: ten high, ten medium, and ten low priority samples.

The core bore removal will provide access into the lower core and plenum regions for closed-circuit television (CCTV) video probes. Acquisition of the core bores will provide access for insertion of the CCTV video camera into the center of the core and lower plenum. The CCTV will provide visual examinations of the extent of damage and guidance to possibly modify further core bore locations. The video data must be carefully keyed to reactor vessel position, and sufficient data must be taken to provide global views of the extent of damage and closeup views of the damaged core materials.

Core Debris Grab Samples (Table 3, Tasks 3, 6, 7). Grab samples from the upper core debris have been obtained and analyzed.[5] 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 samples were generally quite homogeneous, but the relatively small concentration of some fission products has resulted in relatively large uncertainties in the measured concentrations. Additional larger volume samples are required from the upper core debris region to better quantify the retained fission products, particularly tellurium, and their physical and chemical state.

Debris samples (both small localized samples and larger volume samples) will also be obtained from the loose core material resting on the reactor vessel lower head and possibly from the lower core and/or core support regions (depending on the damage conditions). This material may vary significantly from the upper debris in physical and chemical composition and structure, particle size, and retained fission product.

The physical and chemical properties of these materials in the various unique zones will be characterized. Large volume samples are required to increase the detectability of the fission products with low concentrations due to decay since the accident.

Topography of the Crust Below the Debris Bed (Table 3, Task 4). Visual and ultrasonic topography data will help characterize the frozen crust (previously molten core material) that is postulated to exist under the upper debris bed. Ultrasonic techniques similar to those used for mapping the upper core cavity will be used if practical.

Fuel Rod Segments From Distinct Fuel Assemblies (Table 3, Task 8). Examination of fuel rod segments from part length, relatively intact fuel assemblies from the core periphery will provide information on the radial progression of core damage as well as fission product retention over a wide range of fuel rod damage. Assemblies from control and poison rod positions are needed for examination. Intact rod segments will be extracted from the retrieved assemblies for detailed examination. These examinations will provide information on peak fuel rod temperature, materials interactions, retained fission products, and fission product chemical form. The core damage represented by these assemblies is representative of the damage gradient between the molten core and the relatively undamaged core former wall. Also, data on the effect on core damage of silver from control rod assemblies and of alumina from burnable poison rod assemblies will be available.

Retained Fission Products in Containment--Basement Sludge, Concrete Samples (Table 3, Tasks 10, 11). The primary remaining repositories for fission products at TMI-2 are thought to be the reactor vessel (primarily core materials) and the containment basement, particularly the sludge and the concrete walls. Sufficient samples of the basement sludge are needed to estimate the total inventory in the sludge and to characterize the fission products and the materials they are associated with. The current radioactivity in the basement and sludge samples suggests significant retention and activity from the basement concrete walls. Independent experiments have confirmed that the concrete is an efficient absorber of Cs. Sufficient samples of the basement walls and floor are necessary to estimate total fission product retention in the basement.

Fission Product Retention in Ex-Vessel Release Pathways (Table 3, Task 12). All present experience in characterizing the plant indicates relatively small fission product inventories remain in or on the surfaces of all pathways external to the reactor vessel. Additional examinations of samples from readily accessible locations are suggested to confirm these results. These include:

1. Manway/handhole covers for both A- and B-loop steam generators and sediment samples (if possible)

2. Resistance temperature detector (RTD) thermowells in the hot leg and sediment from the pressurizer.

Examinations on these samples will quantify the retained fission products, fission product chemical form, and the irreversible retention mechanisms, either physical or chemical.

Core Support Assembly Samples (Table 3, Task 13). The extent of CSA damage will be determined from visual inspection of the lower plenum and CSA regions through the core bore channels as well as from selected samples of the CSA obtained during defueling. Samples of the CSA are needed to determine peak temperatures and the important interactions between the core materials and the stainless steel structures. Sample selection will be based on knowledge gained from the core bores and the follow-up video examination data.

Reactor Vessel Samples (Table 3, Tasks 14, 15). The current understanding of the interactions between molten core materials and the reactor vessel suggests that the 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. Samples from the instrumentation penetration nozzles at the vessel center and mid-radius locations should be sufficient.

The condition of the reactor vessel is not known, and our understanding of thermal/hydraulic/mechanical details of the core melt progression and ultimate attack on the vessel walls is not complete. These data requirements will be further substantiated as defueling progresses and examination data becomes available, i.e., data from the core bores, and lower plenum volume samples. Visual examination of the vessel wall after defueling is desirable to obtain samples of the reactor vessel wall at locations other than the instrument penetrations. These data needs will be further refined from the vessel failure models as these models are developed.

Core Former Wall (Table 3, Task 16). The core former wall appears to be basically intact in the upper regions of the core. However, below the core mid-plane the extent of damage is not known. If severe damage to the core former walls becomes evident during core defueling, detailed video and acoustic mapping of the damage zones will be necessary, and samples of the walls will be needed to determine the mode of damage and the material interactions. Sample locations will be specified when the severe damage is evident.

Upper Plenum Surface Temperatures and Deposition (Table 3, Tasks 17, 18). The 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. Previous examinations of two control rod leadscrews indicate 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, in conjunction with the damage profile of the upper core support plate and structure of the debris bed, are probably sufficient to address the technical issues associated with reactor vessel natural circulation. However, additional samples of structural surfaces are needed to complete characterization of the retained fission products. The upper plenum is probably not a significant repository for fission products, so these samples and examinations are judged to be of lower priority.

2.2 Development of Sample Acquisition and Examintion Plan

Table 4 is a summary of the in situ measurements and sample acquisitions and examinations that satisfy the technical information needs identified in the TMI-2 Accident Evaluation Program document and listed in Table 2. Table 4 includes prior year sample acquisitions and examinations and in situ measurements of completeness. The Sample Acquisition and Examination Plan includes:

1. Acquisition of all samples, distinct components, and in situ measurements listed in the Future Additional Samples column under Quantity.

2. Sample examination and in situ measurement analysis of those items listed in the Proposed Future Exams column. Only the high priority tasks can be accomplished within the allocated resources. Selection was made using the examination priority list shown in Table 3.

The plans for sample acquisition and in situ measurements were developed based on the policy of retrieving samples and making in situ measurements in conjunction with the General Public Utilities (GPU) Nuclear decontamination and defueling program for the TMI-2 facility. Some decontamination and defueling program plans are currently uncertain, primarily because of budget and/or technical uncertainties. The technical uncertainties include (1) the methods and procedures for removal of the fused core and structural materials from the core and reactor vessel lower plenum regions and (2) TMI plant regions that may be selected for Interim

TABLE 4. TMI-2 ACCIDENT EVALUATION IN SITU MEASUREMENTS AND SAMPLE ACQUISITIONS AND EXAMINATIONS

Measurement/Examination Activity	Quantity: Completed Sample Acquisitions	Quantity: Completed Exams	Quantity: Future Additional Samples	Quantity: Proposed Future Exams	Priority[a]	Examiner[b]	Justification/Information
A. Reactor vessel visual examination:							Explain accident scenario and support sample selection.
1. Closed circuit television surveys	NA	5 areas[c]	NA	1 area	1	REP/AEP	Determine current conditions of molten core material escape route.
2. Sonar topography survey	NA	1 area	NA	1 area	4	REP/AEP	Core cavity dimensions after loose debris and distinct core component removal.
B. Core bore samples of fused/joined core material:							Determine condition and quantity of fused/joined core material under loose debris and between core and reactor vessel head.
1. Under loose debris	9 of 10 successful	9 partial	0	0	1, 5, 9	AEP-INEL, NRC, ANLE, CSNI	Determine retained fission product concentration and chemical form.
2. Subcore	3 unsuccessful	0	0	0	--		
C. Core distinct components:							
1. Upper core region:							
a. 6-in. fuel rod segments from core cavity periphery	6	6 (NDE only)	0	0	20	AEP-INEL, NRC-ANLE	Determine condition of unrelocated fuel rods in upper core region. In situ separation of segments.
b. Small grab samples from upper core debris	11	11	0	0	--	AEP-INEL, CSNI	Reduce uncertainty in retained fission product inventory (especially tellurium) from previous grab sample examination.
c. Large grab samples from upper core debris	6	6	0	0	3	AEP-INEL	
d. Fuel assembly upper section:							
(1) Fuel rod segments from core cavity periphery fuel assembly remnants	18	2	0	8	8	AEP-INEL, NRC-ANLE, CSNI	Study interactions between fuel rods and control or burnable poison material and variations in fuel rod damage around the core periphery. Segment separation from fuel assembly remnant will be performed in INEL hot cell.
(2) Guide tube/burnable poison rod (BPR) segments	0	0	Not Available	0	0	--	
(3) Guide tube/control rod segments	7	2	0	3	8	AEP-INEL CSNI	
(4) Instrument tub[illegible]trument string segments	1	0	0	0	19	AEP-INEL	
(5) Instrument tub[illegible]ments	0	0	0	0	19	--	
(6) Spacer grids	0	0	0	0	19	--	
(7) Upper end boxe	12	0	0	0	19	--	
(8) Holddown sprin	12	0	0	0	19	--	

Measurement/Examination Activity	Quantity: Completed Sample Acquisitions	Quantity: Completed Exams	Quantity: Future Additional Samples	Quantity: Proposed Future Exams	Priority[a]	Examiner[b]	Justification/Information
C. Core distinct components: (continued)							
e. Burnable poison rod spiders	1	0	0	0	19	--	
f. Control rod spiders	7	0	0	0	19	--	
g. Axial power shaping rod (APSR) spider surface deposit	0	0	0	0	19	--	
2. Lower core region:							Additional samples needed to characterize molten core material escape paths.
a. Fuel rod segments	69[b]	0	TBD	14[d]	TBD	AEP-INEL, NRC-ANLE, CSNI	
b. Guide tube/BPR segments	15[b]	0	TBD	8[d]	TBD	AEP-INEL, NRC-ANLE	
c. Guide tube/control rod segments	7[b]	0	TBD	7[d]	TBD	AEP-INEL, NRC-ANLE	
d. Instrument tube/instrument string segments	0	0	TBD	0	19	--	May provide information on thermocouple junction relocation.
e. Instrument tube segments	8[b]	0	TBD	1[d]	19	--	
f. Spacer grids	0	0	TBD	0	19	--	
g. Lower end boxes	1[b]	0	TBD	0	19	--	
D. Lower vessel debris:							
1. Core material samples from lower head region:							
a. Small	8	8	0	3 (CSNI)	7	REP-INEL, NRC-ANLE, CSNI	From 2 azimuthal locations via downcomer access.
b. Large	0	0	2	2	7	REP-INEL, NRC-ANLE	
2. Reactor vessel lower region gamma scans through instrument strings	1	1	0	0	--	REP-GPUN	GPUN TB 85-14.
3. Samples of loose debris in lower core support structure region	0	0	1	1	6	AEP-INEL	Character of loose debris in lower core support structure region.
E. Reactor vessel internals examinations:							
1. Control rod le[illegible]ws	3	2	TBD	0	18	AEP-INEL, AEP-B&W	Fission product transpo[illegible]h, temperature gradient, and reactor vessel natural [illegible]ulation routes.
2. Core former wal[illegible]ples	0	0	TBD	0	16		Data for isotherm maps [illegible]terials interactions at core periphery.

TABLE 4. (continued)

Measurement/Examination Activity	Quantity: Completed Sample Acquisitions	Quantity: Completed Exams	Quantity: Future Additional Samples	Quantity: Proposed Future Exams	Priority[a]	Examiner[b]	Justification/Information
E. Reactor vessel internals examinations: (continued)							
3. Leadscrew support tube lower section	1	1	0	0	Low	AEP-BCL	Characterization of surface deposits in reactor vessel dome region.
4. Core lower support structure plate samples	0	0	TBD	0	13	--	Data for isotherm maps and materials interactions along core material relocation path. Fission product inventory and materials interactions.
5. Reactor vessel lower head samples	0	0	TBD	0	15	--	Data for isotherm maps and materials interactions.
6. Lower plenum horizontal surface deposits	0	0	TBD	0	17	--	Fission product inventory data.
7. Lower plenum instrument structures	0	0	TBD	6	14	AEP-PL	Materials interactions.
F. Reactor coolant system (RCS) characterization:							
1. RCS Gamma Scans:							Capability to convert data to radionuclide and uranium abundance and location uncertain.
a. A-loop steam generator (external)	N/A	7	N/A	0	Low	GPUN/AEP	
b. Pressurizer (external)	N/A	6	N/A	0	Low	GPUN/AEP	
c. Core flood tank B	N/A	9	N/A	0	Low	GPUN/AEP	
d. Steam generator inside	N/A	0	N/A	TBD	Low	GPUN/AEP	
e. Pressurizer inside	N/A	0	N/A	TBD	Low	GPUN/AEP	
f. Pressurizer surge line	N/A	0	N/A	TBD	Low	GPUN/AEP	
g. Decay heat removal line	N/A	0	N/A	TBD	Low	GPUN/AEP	
h. Pump volutes	N/A	0	N/A	TBD	Low	GPUN/AEP	
i. Hot legs	N/A	0	N/A	TBD	Low	GPUN/AEP	
2. RCS adherent surface deposits:							Adherent fission product deposits.
a. A-loop RTD thermowell	1	1	0	0	12	INEL	
b. B-loop RTD thermowell	0	0	0	0	12	--	
c. A-loop steam generator manway cover backing plate	1	1	0	0	12	AEP-BCL	
d. B-loop steam generator manway cover backing plate	1	1	0	0	12	AEP-BCL	
e. Pressurizer manway cover backing plate	1	1	0	0	12	AEP-BCL	
3. RCS sediment:							
a. Steam generator tube [illegible] top loose debris	2	underway	0	0	12	REP-GPUN AEP-PL REP-INEL	Character of sediment in both ste[illegible]erator upper heads.
b. Steam geneator lower [illegible] loose debris	0	0	2	2	12	AEP-PL	GPU Nuclear project. Character [illegible]iment in both steam generator lower heads.
c. Pressurizer sediment	1	1	TBD	TBD	12	REP-W	Character of sediment in pressuri[illegible]wer head.

TABLE 4. (continued)

Measurement/Examination Activity	Quantity: Completed Sample Acquisitions	Quantity: Completed Exams	Quantity: Future Additional Samples	Quantity: Proposed Future Exams	Priority[a]	Examiner[b]	Justification/Information
G. Ex-reactor-coolant-system characterization:							
1. Reactor building:							
a. Liquid:							Basement liquid has been drained and decontaminated.
(1) Basement 305 ft el.	110 mL	1	0	0	Low	AEP-INEL	
(2) Basement 325 ft el.	120 mL	1	0	0	Low	AEP-INEL	
(3) Bottom open stairwell	45 mL	1	0	0	Low	AEP-INEL/ HEDL	
(4) Basement sump pit	200 mL	1	0	0	Low	AEP-INEL/ HEDL	
(5) Reactor coolant drain tank (RCDT)	120 mL	1	0	0	Low	AEP-INEL/ HEDL	
b. Sediment:							Sediment includes Susquehanna River silt as well as core fission products and materials.
(1) Basement 305 ft el.	108 g	1	0	0	10	AEP-INEL	
(2) Basement 325 ft el.	25 g	1	0	0	10	AEP-INEL	
(3) Bottom open stairwell	1 g	1	0	0	10	AEP-INEL/ HEDL	
(4) Basement sump pit	72 g	1	0	0	10	AEP-INEL/ HEDL	
(5) Reactor coolant drain tank	0.5 mg	1	0	0	10	AEP-INEL/ HEDL	
(6) Basement floor (282 ft el.):							
(a) RCDT discharge area	0	0	0	0	10	AEP-PL	
(b) Leakage cooler room, RCDT room, inside D-ring.	0	0	0	0	10	AEP-PL	
(c) Outside D-ring areas	3	2	0	0	10	AEP-SAI	
(d) Core instrument cable chase	0	0	0	0	10	AEP-PL	
(e) Sludge removal storage tank	0	0	16 (tentative)	16	10	AEP-PL	
c. Concrete bores:							
(1) Floors: 347 ft el.	8	8	0	0	Low	GPUN/AEP	
305 ft el.	6	6	0	0	11	GPUN/AEP	
282 ft el.	2	0	0	2	11	REP-GPUN AEP-INEL	GPUN has samples.

Measurement/Examination Activity	Quantity: Completed Sample Acquisitions	Quantity: Completed Exams	Quantity: Future Additional Samples	Quantity: Proposed Future Exams	Priority[a]	Examiner[b]	Justification/Information
c. Ex-reactor-coolant-system characterization: (continued)							
(2) D-ring walls: 347 ft el.	1	1	0	0	Low	GPUN/AEP	
305 ft el.	2	2	0	0	11	GPUN/AEP	
flooded region	4	1	0	TBD	11	REP-GPUN REP-INEL AEP-INEL	
(3) 3000 psi (shield) wall (flooded region)	7	1	0	TBD	11	REP-GPUN REP-INEL AEP-INEL	
(4) Block (elevator/stairwell) walls (flooded region)	6	1	0	TBD	11	REP-GPUN REP-INEL AEP-INEL	
d. Adherent surface deposits:							
(1) Air cooler panels	5	5	0	0	Low	AEP-INEL	
(2) Basement outer wall steel liner	0	0	0	0	Low	--	
2. Auxiliary and fuel handling buildings:							All equipment in the auxiliary and fuel handling buildings has been fully or partially decontaminated by flushing, filter removal, water treatment, and resin removal or treatment.
a. Liquid:							
(1) Reactor coolant bleed Tank A	125 mL	1	0	0	Low	AEP-INEL	
(2) Reactor coolant bleed Tank B	150 mL	1	0	0	Low	AEP-INEL	
(3) Reactor coolant bleed Tank C	150 mL	1	0	0	Low	AEP-INEL	
(4) Makeup and purification demineralizer B	40 mL	1	0	0	Low	AEP-ORNL	
b. Sediment:							
(1) Reactor coolant bleed Tank A	60 g	1	0	0	Low	AEP-INEL/HEDL	
(2) Makeup and purification demineralizer A (resin)	10 g	1	0	0	Low	AEP-ORNL	
(3) Makeup and purification demineralizer B (resin)	40 mL	1	0	0	Low	AEP-ORNL	

TABLE 4. (continued)

Measurement/Examination Activity	Quantity: Completed Sample Acquisitions	Quantity: Completed Exams	Quantity: Future Additional Samples	Quantity: Proposed Future Exams	Priority[a]	Examiner[b]	Justification Information
G. Ex-reactor-coolant-system characterization: (continued)							
c. Filter housing contents (filter paper, liquid, and collected solids):							
(1) Makeup and purification system							
(a) Demineralizer prefilters	both	both	0	0	Low	AEP-INEL/LANL, NRC-ANLE	
(b) Demineralizer after filters	both	both	0	0	Low	AEP-INEL/LANL, NRC-ANLE	
(2) RC pump seal water injection system filters	both	both	0	0	Low	AEP-INEL/LANL, NRC-ANLE	

a. Priority values 1 through 20 are listed in Table 3.

b. Examination responsibility is shown with the funding organization (AEP, REP, NRC, and/or GPUN) shown first (xxx/xxx indicates joint funding and/or performance responsibility), and the sample examination location shown second. Names of funding organizations are abbreviated as follows: Accident Evaluation Program, AEP; Reactor Evaluation Program, REP; Nuclear Regulatory Commission, NRC; GPU Nuclear, GPUN. Names of examination locations are abbreviated as follows: Idaho National Engineering Laboratory INEL; Argonne National Laboratory-East, ANLE; Battelle Columbus Laboratories, BCL; Westinghouse Electric Corporation, W; Science Applications International Corporation, SAI; Hanford Engineering Development Laboratory, HEDL; Oak Ridge National Laboratory, ORNL; Los Alamos National Laboratory, LANL; CSNI, Committee for the Safety of Nuclear Installations of the Organizationfor Economic Cooperation and Development (OECD). PL indicates an outside private laboratory is expected to perform the examination.

c. Completed reactor vessel CCTV surveys include the following areas: all sides of the upper core region cavity, core cavity region loose debris after dislodging core components from plenum assembly, plenum assembly, accessible areas of the downcomer and reactor vessel bottom head periphery regions, core lower region and lower core support assembly.

d. Core bore collected samples.

Monitored Storage classification leaving the area unsuitable for personnel entry and sample acquisition. The GPU Nuclear TMI-2 decontamination and defueling program includes the following:

1. An auxiliary and fuel handling building decontamination program.

2. A reactor building decontamination program.

3. A reactor building basement contamination characterization program (see K. J. Hoffstetter letter to D. M. Lake, 4240-85-0227, Reactor Building Sludge and Core Bore Samples, June 6, 1985).

4. A RCS fuel locating program (see J. D. DeVine letter to R. L. Freemerman, 4500-84-0738, Ex-vessel Fuel Locating Samples Packages, August 27, 1984).

5. A reactor vessel data acquisition program (see GPU Nuclear document TPO/TMI-117, In-Vessel Data Acquisition, September 1984).

6. The defueling program (see GPU Nuclear news release 38-85N, TMI-2 Defueling Schedule Updated, April 30, 1985).

An important part of the DOE TMI-2 Program is the Reactor Evaluation Program (REP), which supports the TMI-2 defueling program in the following areas:

1. Funding for special defueling tools and plant decontamination studies.

2. Defueling operations, which will include both sample retrieval from the reactor vessel and collection of in situ measurement data such as CCTV surveys and ultrasonic scanner topography.

The responsibility for funding the tasks outlined in Table 4 is indicated in the table and includes GPU Nuclear, the DOE Accident

Evaluation Program (AEP), the DOE Reactor Evaluation Program (REP) and the OECD. Examinations will be performed at the INEL, Argonne National Laboratory-East (ANL-E), other DOE laboratories, private laboratories (PL) or OECD/CSNI member country laboratories. Work plans were developed for the tasks summarized in Table 4 under the assumption that after the samples have been retrieved at TMI-2, the handling, packaging, and shipping activities to the INEL will be funded by the REP-supported defueling program.

The development of the TMI-2 AEP SA&E Plan included consideration of the following assumptions:

1. The total budget allowance including prior years is $20.6M from the DOE and $600K from and administered by the NRC.

2. Sample retrieval and in situ measurements will be accomplished in conjunction with GPU Nuclear's TMI-2 recovery program and with support from the DOE TMI-2 Reactor Evaluation Program in the development of special defueling tools and the collection of defueling-operation-related samples and in situ measurements.

3. Prioritization of the information needs from the sample acquisition and examination tasks is as shown in Table 3. This prioritization is based on technical needs identified and discussed in the TMI-2 Accident Evaluation Program document. These are shown in Table 2.

4. The portions of the total budget to be allocated to laboratory examination of samples is: $918K to other DOE laboratories, $1.38M to private domestic laboratories, and 2.9M to EG&G laboratories. In addition, NRC will fund about $600K for other DOE laboratory examinations.

The proposed examination plan for the core bores includes examination of nine upper core bores. Examination of the core bore samples will yield information on the condition and quantity of both the fused core materials in the lower core region and standing fuel bundles between the fused material and the core bottom. Data will also be obtained to determine fission product concentration and chemical form.

Two fuel rod segments from a part-length peripheral control rod assembly will be examined. No part-length peripheral burnable poison rod assembly was recovered during defueling. One of the fuel rod segments will be obtained from a location near a control rod position, and one from a location not near a control rod position. The control rod remnant will also be obtained. Examination of these three (two fuel rods, one control rod) rod segments will help determine the effect of the failure of the control rod on the adjacent fuel rods. Fuel rod segments from a burnable poison rod and a control rod assembly in the lower core region will be obtained and examined, by the core bore acquisition and examination program.

The large debris sample from the debris bed below the upper cavity will help reduce the uncertainty in the retained fission products (especially tellurium) that was measured from the 11 grab samples already examined. Analysis of this large sample will also help determine the homogeneity of the upper debris bed and therefore the applicability of the data from the 11 small samples to the entire debris bed.

Eight other small debris samples have been obtained from the lower vessel debris bed. Examination of these samples will indicate the fission product retention in a mixture of materials that probably contains more structural material than the upper core debris bed. A large sample of this lower vessel debris will also be obtained and examined to determine homogeneity. Also, a large sample of loose debris will be obtained from the lower core support structure region if possible.

In order to determine fission product chemical form and fission product and aerosol interaction with structural materials, samples will be obtained from both the reactor coolant system and the EX-RCS. Samples of high priority in the EX-RCS are sediment samples and concrete samples from the containment building basement walls and floor. Samples of high priority in the reactor coolant system are adherent surface deposits on the

A- and B-loop steam generator manway cover backing plates, and the pressurizer manway cover backing plate. Sediment will be obtained for examination from the steam generator lower head, the top of the steam generator tube sheet, and the bottom of the pressurizer.

The proposed TMI-2 AEP Sample Acquisition and Examination work plan is divided into the following four work package categories:

1. Reactor vessel, which includes the reactor vessel, its internal structures, and the core.

2. RCS fission product inventory, which includes the core materials and fission products now residing in the ex-vessel portion of the RCS, including the core flood tanks.

3. EX-RCS fission product inventory, which includes the core materials and fission products now residing in areas, buildings, and equipment external to the RCS.

4. Program management support, which includes personnel and services to plan, direct, and control the sample acquisition and examination program.

The three sample acquisition and examination implementation work package categories (1, 2, and 3 above) are further subdivided into sample acquisition and sample examination work packages because of the geographical separation of the respective support personnel and operations. The individual work packages provide the detailed scope of work, assumptions, products/deliverables, milestones, and prerequisites statements, logic diagrams (activity lists and schedules), and resource (labor and material) tabulations. The subdivision of the TMI-2 AEP SA&E Plan into the three TMI-2 nuclear power plant regions--reactor vessel, reactor coolant system and external to the reactor coolant system (EX-RCS)--was selected to coincide with the GPU Nuclear TMI-2 fuel location and characterization program and with the chronological separation of the core damage sequence and the offsite radiation hazard during the TMI-2 accident.

Detailed discussions of the four sample acquisition and examination work plans are contained in the next four sections of this report.

3. REACTOR VESSEL SAMPLE ACQUISITION AND EXAMINATION WORK PLAN

3.1 Introduction

The reactor vessel sample acquisition and examination work plan includes the reactor vessel, the nuclear reactor core and its support structures, the core instrument strings, including their support and ex-vessel conduit structures, and other reactor vessel (RV) internals. A diagram of the reactor vessel arrangement as it appeared before the commencement of core damage events is shown in Figure 1. A typical incore instrument assembly, including the ex-vessel conduit arrangement, is shown in Figure 2. The TMI-2 Core Support Assembly arrangement is shown in Figure 3.

The RV sample acquisition and examination work plan was developed by considering the types of data needed to help resolve the major issues discussed in Section 2. Some of the information pertinent to developing the data acquisition plan is discussed in the following paragraphs. This information includes applicable details of the TMI-2 preaccident operations including core loading details, the accident sequence including available information on the current damage state within the reactor vessel and the postaccident reactor vessel internals disassembly activities that have caused further relocation and separation of the reactor vessel internals.

3.1.1 Preaccident Operations

At accident initiation, the TMI-2 core was in the initial fuel cycle at 97% of full power with 3175 MWD/MTU average core burnup. The core loading consisted of 177 fuel assemblies and 139 rod assemblies arranged in the core positions as shown in Figure 4. The fuel assemblies were placed in the core positions with the identification marking toward the south (see Figure 5). The core position component identification marking index is provided in Appendix C.

Each of the fuel assemblies (see Figure 5) is a 15 x 15 array of 208 fuel rods, 16 zircaloy guide tubes and 1 center-position zircaloy

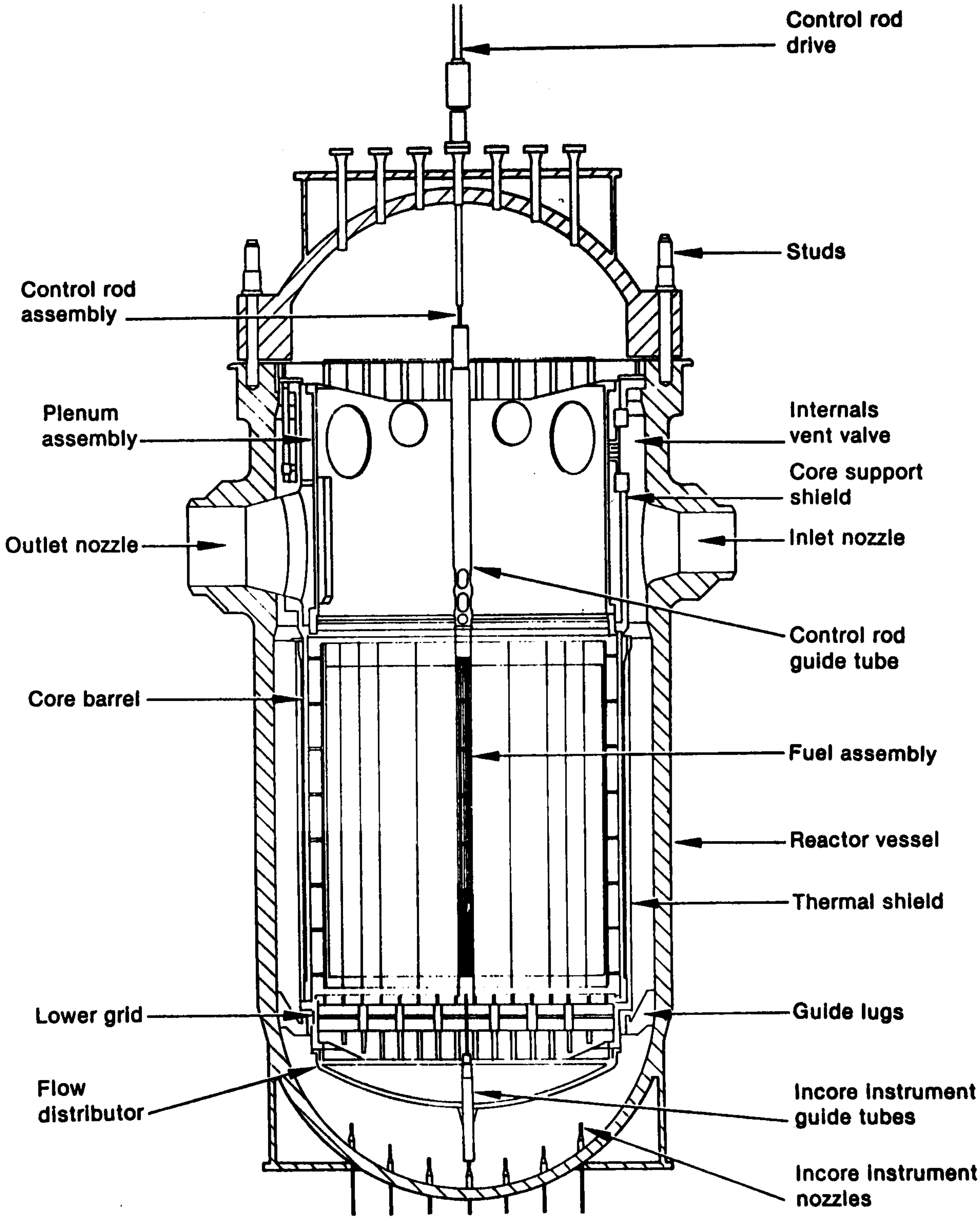


Figure 1. General arrangement of TMI-2 reactor vessel and internals.

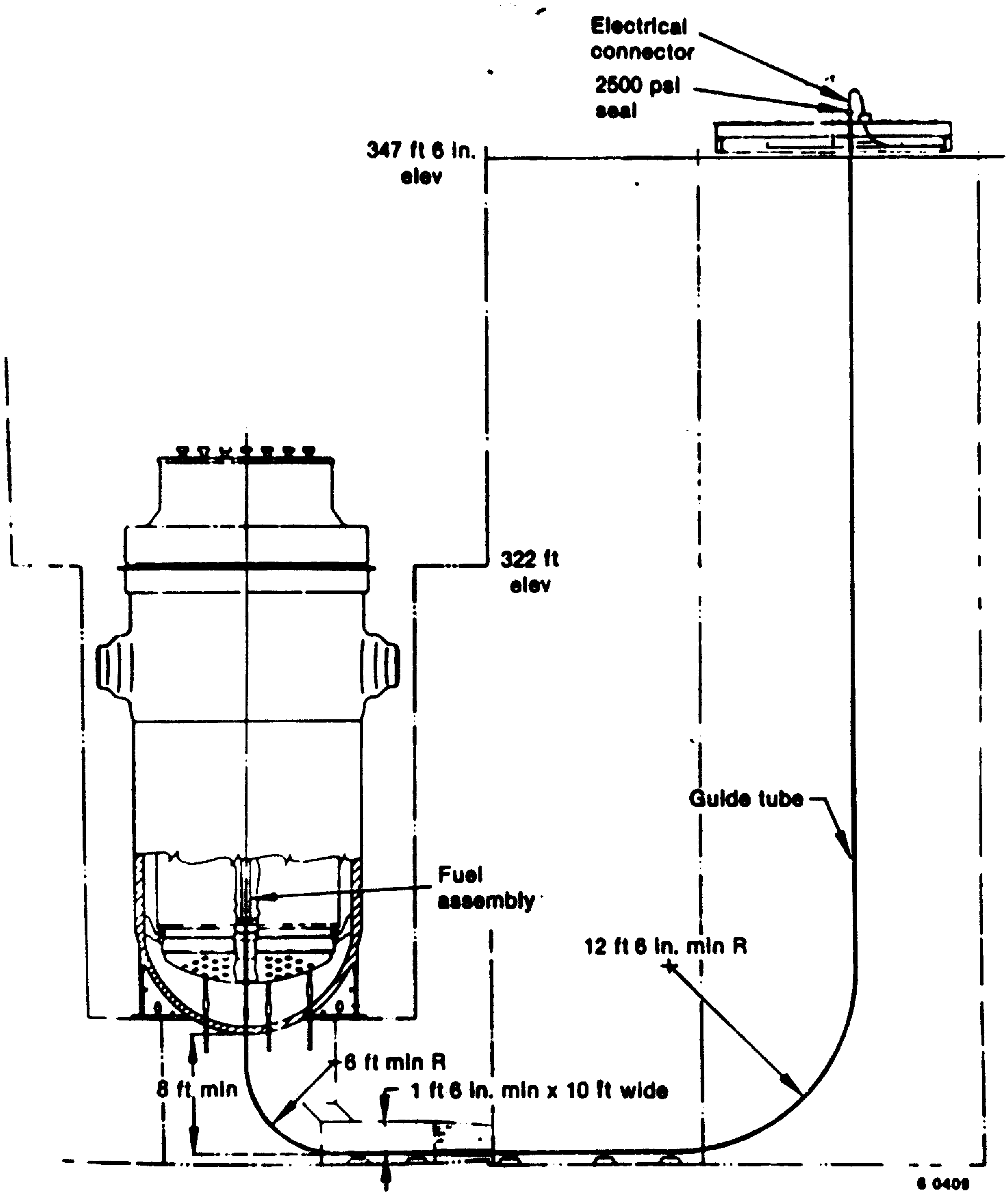


Figure 2. Schematic of typical incore instrument assembly.

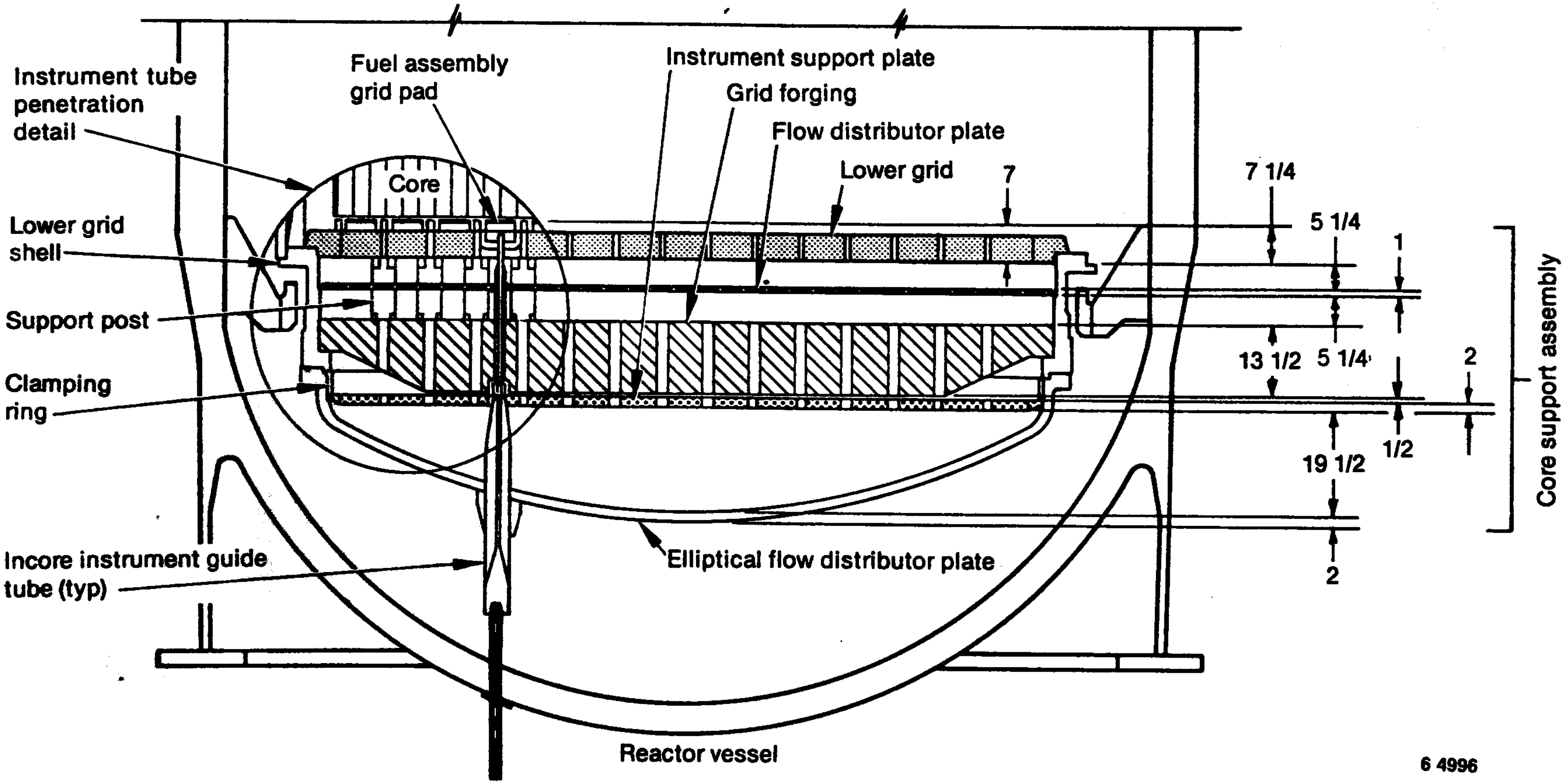
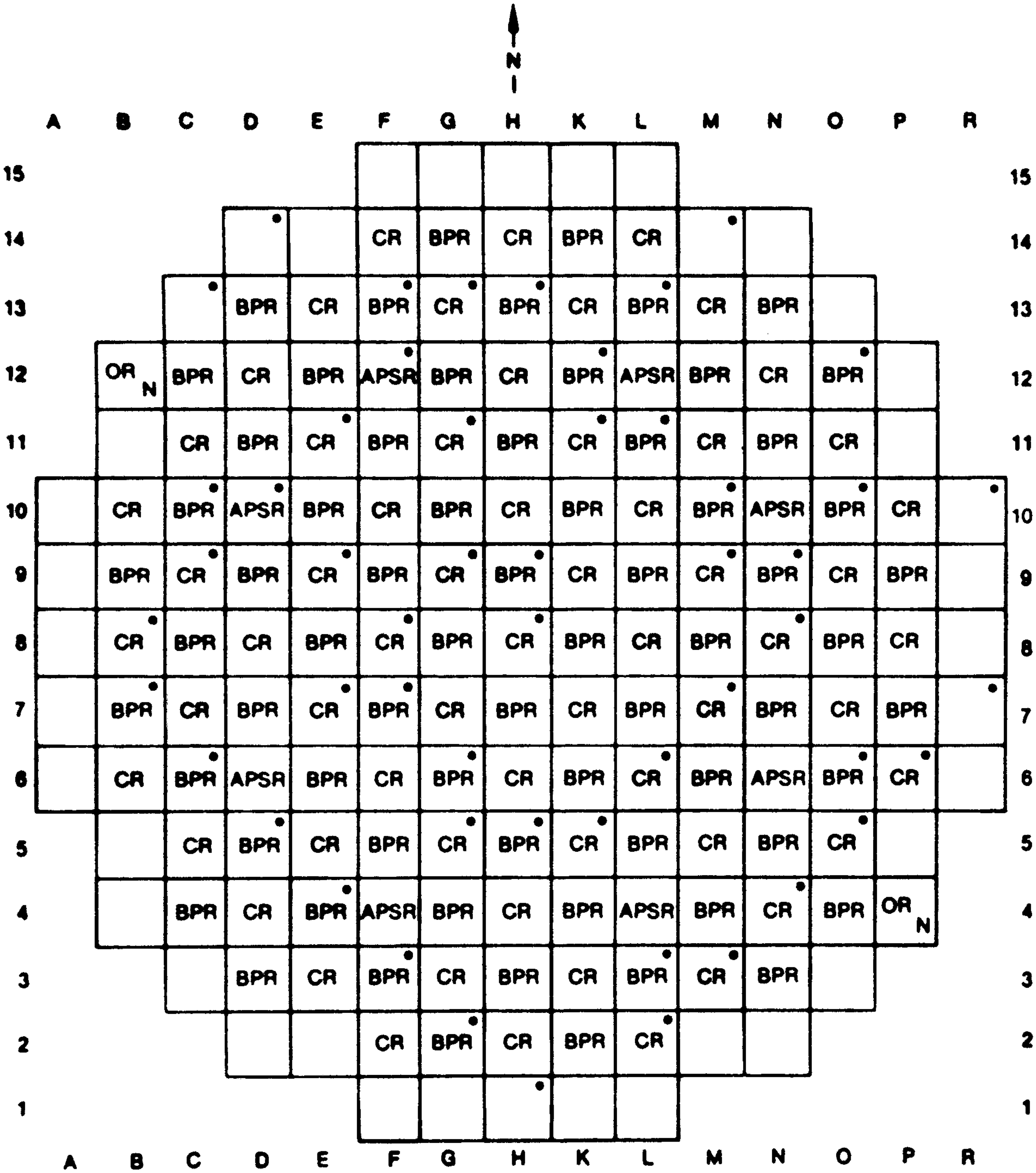


Figure 3. TMI-2 core support assembly configuration.



OR	Orifice Rod Assembly
CR	Control Rod Assembly
BPR	Burnable Poison Rod Assembly
APSR	Axial-Power-Shaping Rod Assembly
N	Primary Neutron Source

•Core Instrument String

7-0530

Figure 4. TMI-2 core loading diagram.

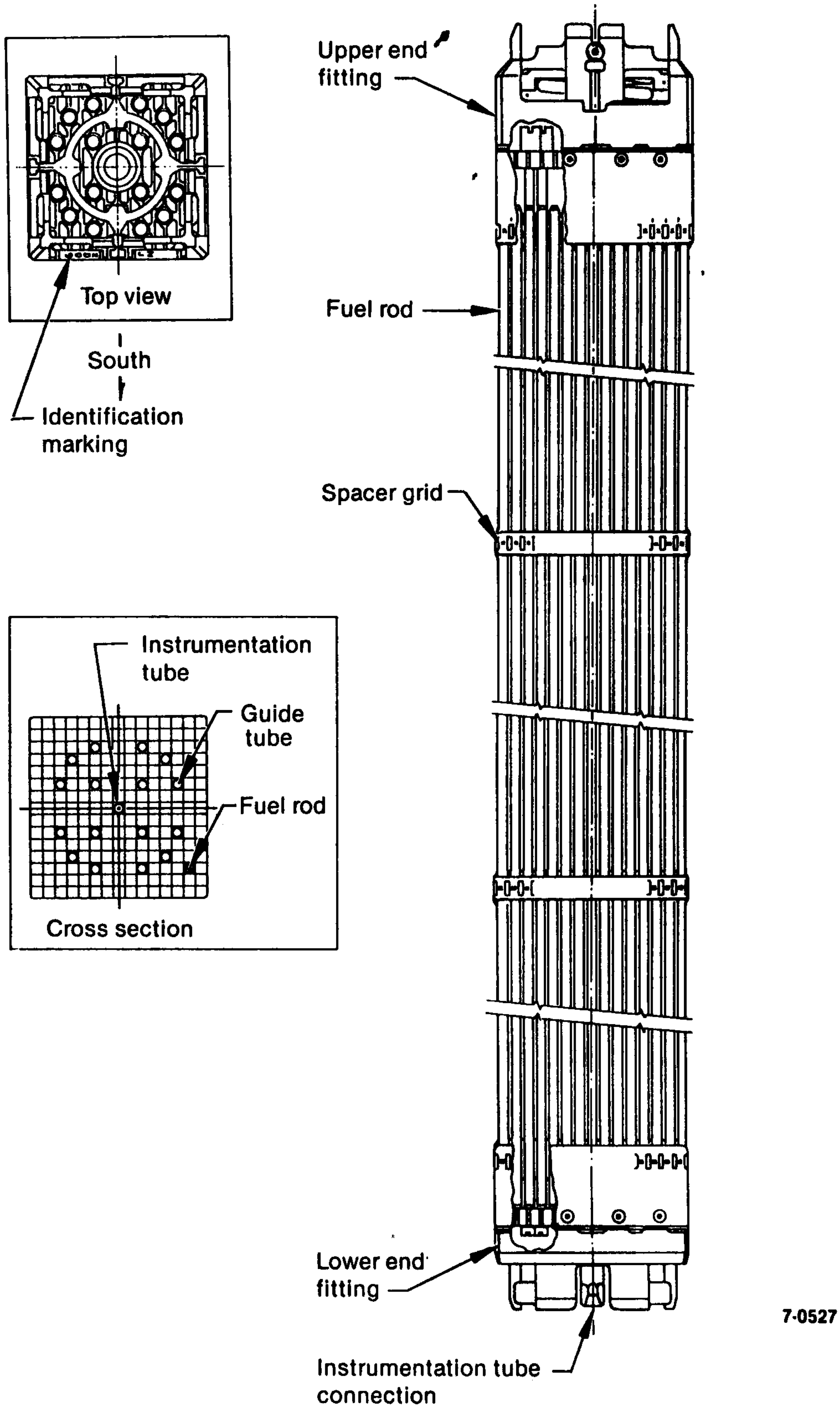


Figure 5. Side, top, and cross-sectional views of TMI-2 fuel assembly (from Reference 6).

instrument tube connected to and supported by 8 Inconel spacer grids and 304L stainless steel upper and lower end fittings. An Inconel, coil-type holddown spring is located in the upper end fitting.

All interior and two of forty peripheral core positions also have rod assemblies consisting of 16 rods connected together at the top by arms extending from a central hub. The rods fit into the fuel assembly guide tubes. The two peripheral fuel assemblies (core positions B12 and P4, next to the core former wall) contain a stationary orifice-rod assembly (see Figure 6) with 12-in.-long stainless steel rods extending into the guide tubes to restrict coolant flow, of which one in each assembly is assumed to be modified to include a neutron source rod. Interior fuel assemblies contain one of three types of rod assemblies as follows:

- Burnable Poison Rod (BPR) Assembly (see Figure 7)--The stationary BPR assemblies are located in 68 core positions as shown (BPRs) in Figure 4. Each BPR rod contains a 126-in.-long stack of Al_2O_3 (0.95)-B_4C (0.01) ceramic pellets clad in zircaloy, except for core position N13, which is assumed to contain eight rods with borated graphite instead of Al_2O_3-B_4C.

- Control Rod (CR) Assembly (Figure 8)--The CR assemblies are located in the 61 core positions shown in Figure 4. The rods contain 134 in. lengths of Ag-In-Cd clad in Type 304L stainless steel. The CR assemblies were fully inserted during the accident sequence.

- Axial-Power-Shaping-Rod (APSR) Assembly (Figure 9)--The APSR assemblies are located in the eight symmetrical core positions shown on Figure 1. Each rod contains a 36 in. length of Ag-In-Cd material clad in stainless steel. The APSR assemblies remained withdrawn at 37 in. during the accident sequence.

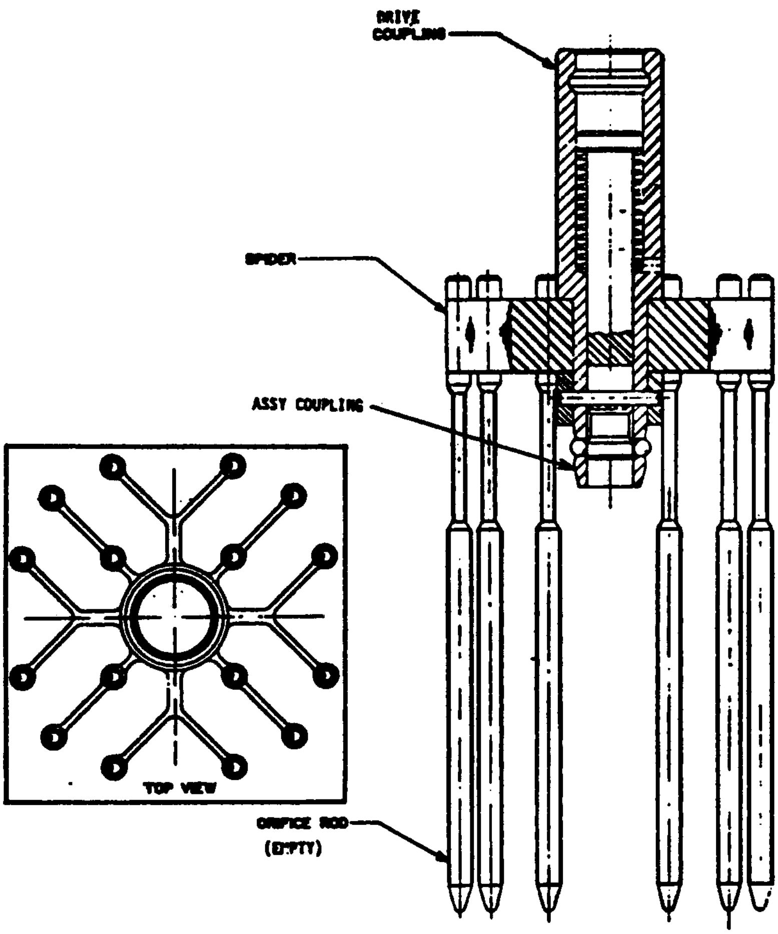


Figure 6. Orifice rod assembly (from Reference 6).

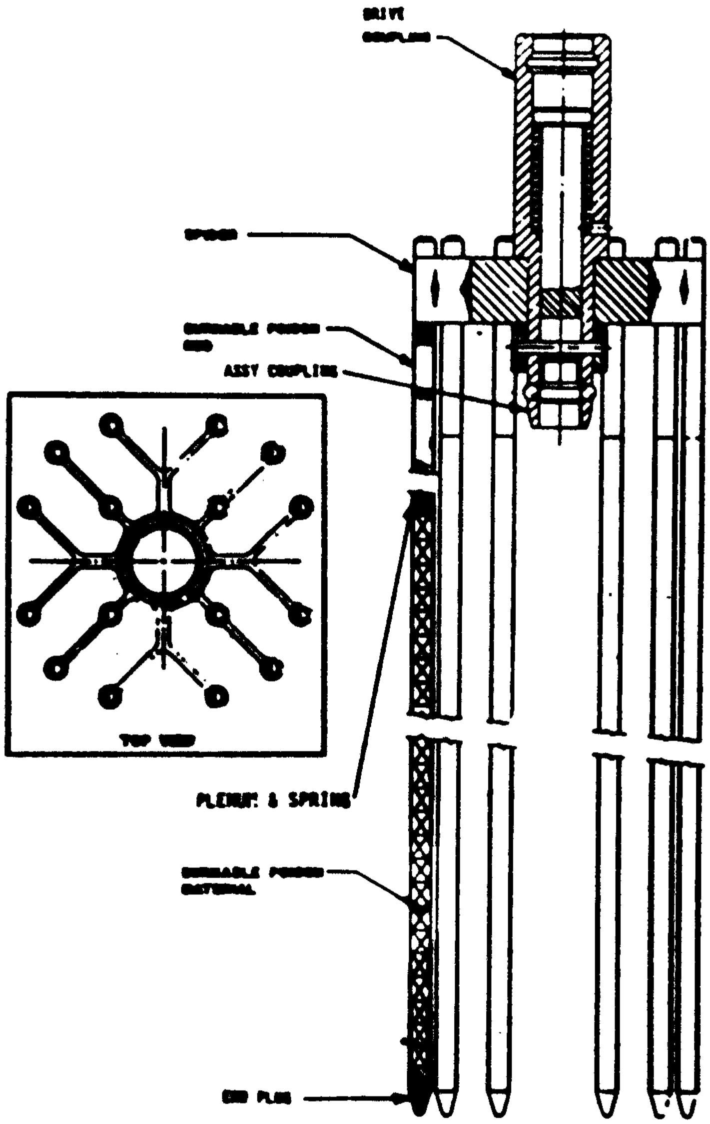


Figure 7. Burnable poison rod assembly (from Reference 6).

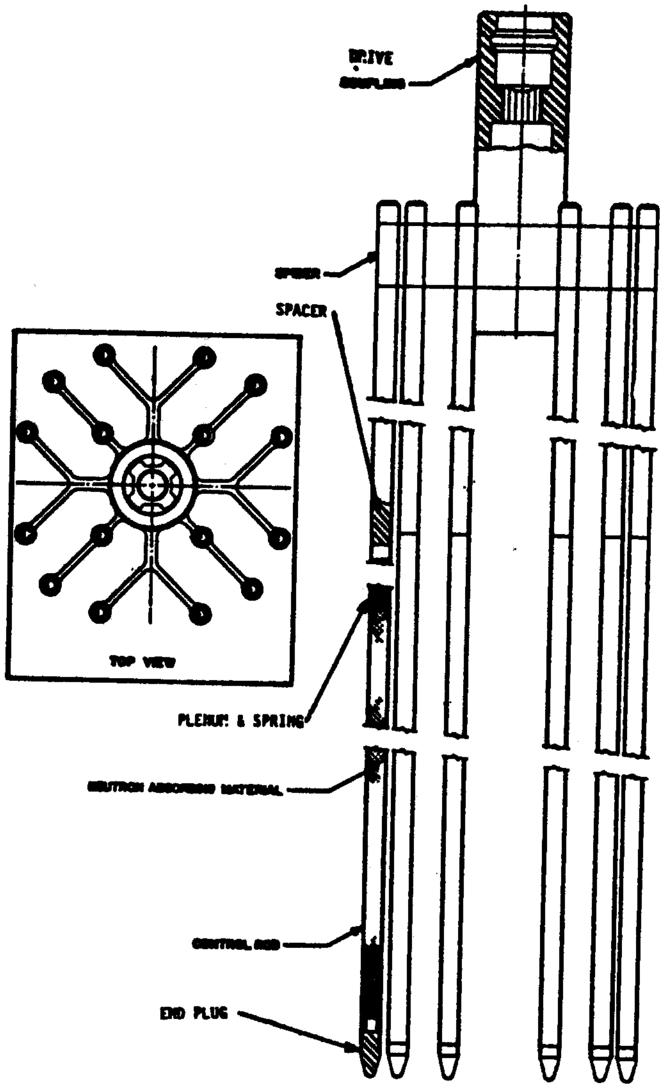


Figure 8. Control rod assembly (from Reference 6).

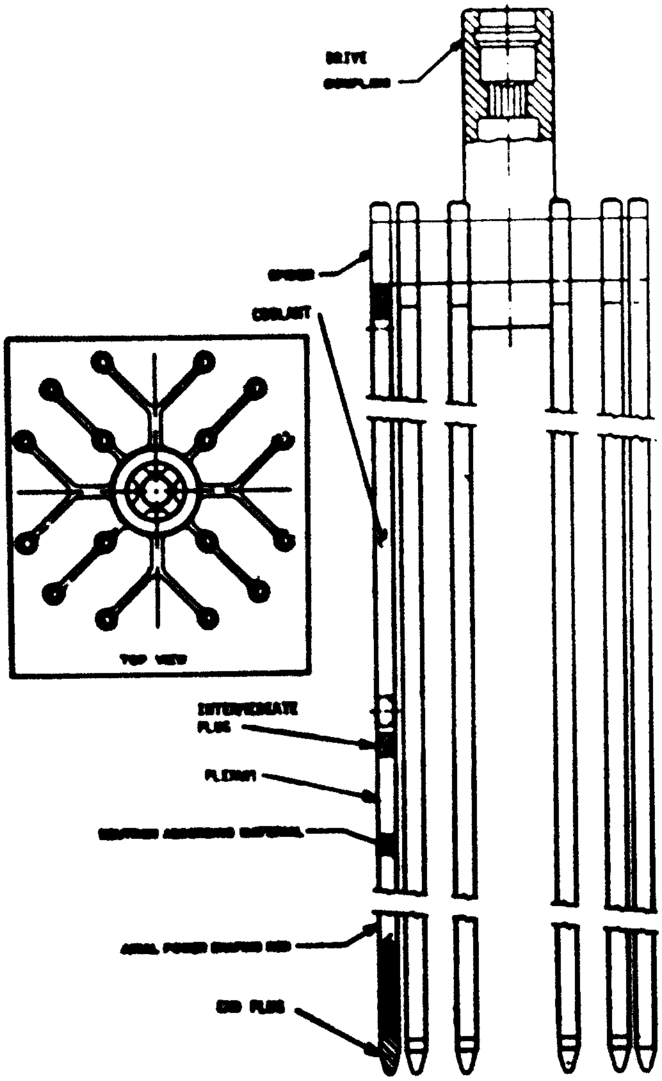


Figure 9. Axial-power-shaping-rod (APSR) assembly (from Reference 6).

3.1.2 TMI-2 Accident Sequence

Reference 7 includes the current theory of the core-component damage and relocation and the formation of the core cavity. A summary of this theory is as follows.

Core uncovery started between 100 and 113 min after turbine trip, which is considered the beginning of the core damage phase of the accident. This is substantiated by the measurement of superheated steam in the hot legs at 113 min. 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 that had accumulated in the gap between the fuel pellets and the cladding. 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; this molten U-Zr-O ternary mixture flowed down and solidified in the lower, cooler regions of the core probably at the reactor coolant liquid-vapor interface. At 150 min, the core liquid level was estimated at approximately 0.7 m, which is consistent with the lower limit of previously molten core materials in the center of the core. At this time, the high-temperature zone and most of the core damage was probably confined to the central region of the core.

By 174 min (just prior to the primary coolant pump transient), some of the fuel had been dissolved by molten cladding or melted in the central, highest-temperature regions of the core. This relocation of fuel material into the lower regions of the core probably resulted in the funnel-shaped, end-state configuration as determined from the core boring operation. Fuel rod remnants composed of oxidized cladding and the undissolved UO_2 fuel remained standing above the solid structure of relocated material. Relatively undamaged fuel assemblies existed around most of the core periphery and beneath the bottom crust of ceramic fuel rod materials. The funnel-like shape of the bottom crust was probably caused by the initial blockage of flow in the center of the core and diversion of coolant flow to the core periphery. This flow diversion enhanced the heat transfer and prevented the relocating molten core materials from flowing down to the same elevation as that at the core center.

The primary coolant pump transient at 174 min injected as much as 1000 ft^3 coolant in less than 15 s into the core. The oxidized (and embrittled) fuel rod remnants above the solid structure probably fragmented by thermal and mechanical shock due to the injected water and the fragments settled to the top of the core region where flow channels were filled forming the core cavity and a loose debris bed under the core cavity floor. The loose debris particles include agglomerates composed of oxidized cladding, unrestricted fuel pellet fragments, and previously molten fuel rod materials and are well mixed indicating agitation occurred during and/or after settling to the core cavity floor.

At 200 min, an additional high-pressure injection system (HPIS) was turned on; and increased coolant was injected in the primary cooling system. The estimate of the injection rate and the water injected by the primary cooling pump transients at 174 min, if directed entirely into the reactor vessel, would result in a covered core sometime after 200 min to provide; (1) continuous cooling to the surface of the solid structure of relocated core materials, and (2) the coolant, which would eventually quench the debris bed. Quenching of the debris bed may have been a relatively long-term and rather violent process, with water gradually penetrating the interior of the debris bed from the core periphery. The resultant steam and hydrogen (byproduct of the steam/metal oxidation process) would rapidly flow from the core into the upper plenum. This process of intermittent water ingress followed by the flow of high-velocity gases out the top of the core could vigorously mix the debris bed as well as bring molten ceramic material from the top of the solid structure up into the debris bed where the agglomerated particles were formed. Cooling at the outside surface of the solid structure would have been insufficient to prevent continued heating from decay heat and remelting of ceramic material within the interior as shown. The high-velocity steam and hydrogen flowing from the core would cause the melting and oxidation which has been observed on the underside of the core upper-grid structure. An alternate explanation for the end-state conditions observed is that the damage to the upper grid occurred prior to 174 min during core heatup and that the debris bed was mixed by the primary coolant pump after core cooling was reestablished at about 16 h into the accident.

Most, if not all, of the core materials found in the lower plenum probably relocated at approximately 224 min and were molten. The relocation of molten core materials into the lower plenum occurred primarily in the southeast quadrant of the core near the periphery through assemblies P5 and P6. Since there is no voided region in the previously molten core zone, the central portion of the top crust apparently collapsed as the material flowed into the lower plenum, leaving behind a ridge of material at the core periphery. The top of this ridge indicates the top of the crust or solid structure prior to relocation. The volume of this "sinkhole" formed by the existing ridge identified from video examinations of the core debris bed periphery and core boring at assemblies D4 and D8 and the current top of the crust is approximately 80 ft^3. This volume is essentially the same as the volume of material estimated to be in the lower plenum from the video exams and also the same as the estimated volume of fuel rod materials currently missing from the core.

The estimated end-state conditions of the TMI-2 core immediately following the accident is shown in Figure 10. This figure was constructed using the cross sections of the upper debris bed region, molten core region (agglomerate and previously molten, homogeneous ceramic), and standing fuel rods for the B through P assembly rows described previously (see Figure 4). The upper debris bed surface is approximated in the cutaway view in Figure 10; note the low point of the debris bed near the "B outlet" (east quadrant of the core). The different material structures which currently exist within the core are illustrated approximately to scale. The estimated volumes and masses of the different types of core materials are listed in Table 5.

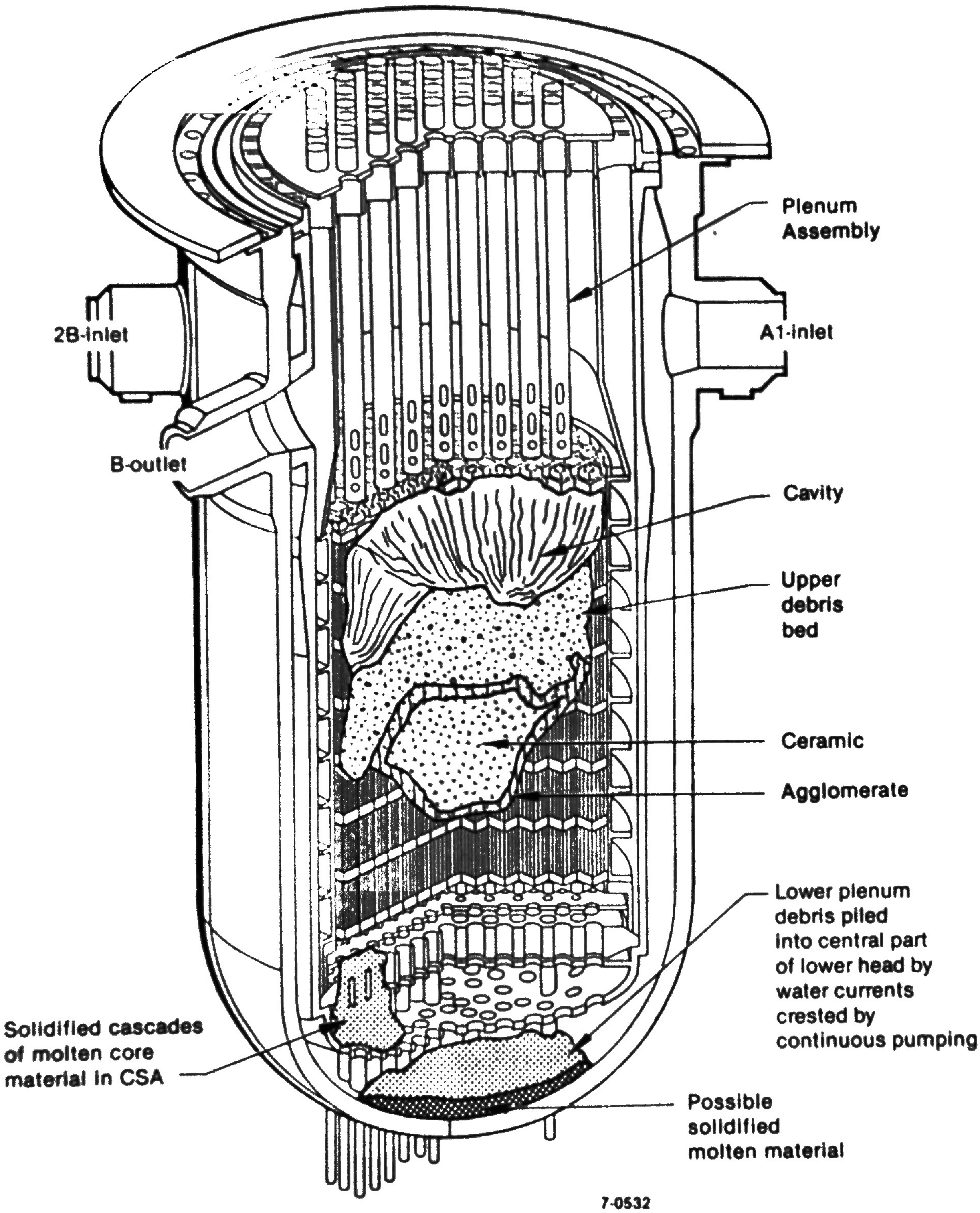


Figure 10. TMI-2 accident end-state core conditions.

TABLE 5. ESTIMATED CORE REGION VOLUMES AND MASSES AT TMI-2 ACCIDENT TERMINATIONS

Region	Estimated Volume (ft^3)	Estimated Mass (lbm)
Upper core debris	236	66,300
Molten zone	122	55,200
Standing rods	499	114,271
Lower plenum debris	74	34,800

3.1.3 Postaccident Reactor Vessel Internals Events

A series of events including precursor examinations and disassembly activities have been accomplished between the accident-sequence termination and October 1986, which affected or determined the condition of the core-cavity walls and floor. The core components have remained submerged in an ambient temperature and pressure, treated, water solution with the following target specifications:

- ph: 7.5 to 7.2
- boron: >4350 ppm
- buffer: Na OH.

No activities or examinations were attempted until personnel access inside the reactor building was reestablished in 1981. A summary of significant examination and disassembly events that have occurred follows.

3.1.3.1 Quick-Look Video Surveys. In 1982, control rod leadscrews from core positions H8, E9, and B8 were removed for possible CCTV access to the core area. The control-rod spider was still in place at B8, but was missing at core positions H8 and E9. The CCTV survey discovered a large, empty region (core cavity) in the upper-core region.

3.1.3.2 APSR Assembly--Insertion.[8] In the first quarter of CY 1983, an attempt was made to insert all eight APSR assemblies, which if successful would relocate the APSRs 37 in. downward (see Figure 4 for APSR core positions). Insertion into the core cavity depths were as follows:

Core Position	Insertion Depth (in.)
D6	0
D10	4
F4	30
F12	35
L4	8
L12	31
N6	0
N10	37

3.1.3.3 Ultrasonic Scanner Survey.[9] On August 31, 1983, an ultrasonic scanner survey was made to determine the shape and dimensions of the core cavity. The core topographical features included the following:

- The cavity extended from the upper grid plate bottom downward to approximately 7.5 ft above the core bottom and radially to the core-former wall in some places

- The core cavity volume was equivalent to approximately 26% of the original core region

- Fuel assembly remnants appeared to encircle the core cavity completely toward the upper grid plate; the maximum fuel assembly damage appeared to be on the core east side, and the least fuel assembly damage on the core west side

- The APSRs that had been inserted projected from the cavity ceiling and interfered with ultrasonic-scanner measurement of topography in the cavity-upper regions.

3.1.3.4 Reactor Vessel Head Removal.[10] In July 1984, the reactor vessel head removal, which included prerequisite uncoupling of the leadscrews from the control-rod assemblies and raising the leadscrew into the control-rod-drive mechanism (CRDM), was accomplished. The leadscrew uncoupling indicated the following:

- Thirty control-rod spiders were supported by the fuel assembly upper end fitting

- Twenty-three control-rod spiders appeared to be unsupported by the fuel assembly upper end fitting, or were missing

- Four control-rod spiders became supported by the fuel assembly upper end fitting when lowered a small distance (less than 2 in.).

3.1.3.4 Plenum-Assembly Removal.[11,12] In May 1985, the plenum-assembly removal, which included prerequisite dislodging of fuel assembly upper end fittings (see Reference 11) water jet flushing loose debris from horizontal (upward facing) surfaces (see Reference 12), and visual (CCTV) examination of the assembly, was accomplished. The dislodging of fuel assembly upper end fittings[12] indicated the following:

- Four upper end fittings (core positions D5, F3, F13, and K14) could not be dislodged

- Ten upper end fittings (core positions E4, G14, K6, L2, L13, O3, O8, O11, P8, and R6) could only be partially dislodged

- All other end fittings were missing, dislodged, or attached to their respective fuel bundles.

The water jet flushing removed loose debris "ranging in size from very fine particles to nearly fuel pellet size" from the plenum assembly, upward-facing, horizontal surfaces. Post flushing CCTV inspection indicated "some of the debris actually adhered to the plenum and could not be removed."

The CCTV examination revealed probable thermal-excursion produced damage to the plenum assembly lower surface as depicted in Figure 11.

3.1.3.5 Reactor Vessel Lower Head Region Video Surveys. In February and July 1985, the reactor vessel lower head region was partially surveyed with a CCTV camera lowered through the downcomer annulus at 13°, 63°, 167° and 245° (hole numbers 1, 4, 7, and 11, respectively) azimuthal positions and samples of the loose debris deposited on the reactor vessel lower head were collected with a remote manipulator lowered through hole numbers 7 and 11. The surveys indicated the following:

- Ten to twenty tons of probable core material had collected in the region between the reactor vessel lower head and the elliptical flow distributor

- The core material form ranged from coffee-ground size particles to a vertical-curtain appearing wall extending toward the elliptical flow distributor and appeared to be lava-like (previously molten)

- Previously molten material was hanging or attached to the elliptical flow distributor below core positions L2, L14 and N3

- The central and eastside region between-the reactor vessel lower head and the elliptical flow distributor were not surveyed.

3.1.3.6 Fuel Removal. Fuel removal commenced on November 12, 1985 and has continued through October 1986. In FY 1986 fuel removal was limited to the core cavity walls and floor and consisted of (1) upper end fittings from fuel, control rod and burnable poison rod assemblies, partial fuel assemblies and unsegregated loose debris and (2) a cumulative total weight of 51,000 pounds of the 300,000 pound core. The early fuel removal included some successful and some unsuccessful attempts to topple standing peripheral fuel assemblies onto the core-cavity floor to provide clearance for the fuel canisters, occasional unaided toppling of unstable standing peripheral fuel assemblies onto the core-cavity floor, and shear-tool

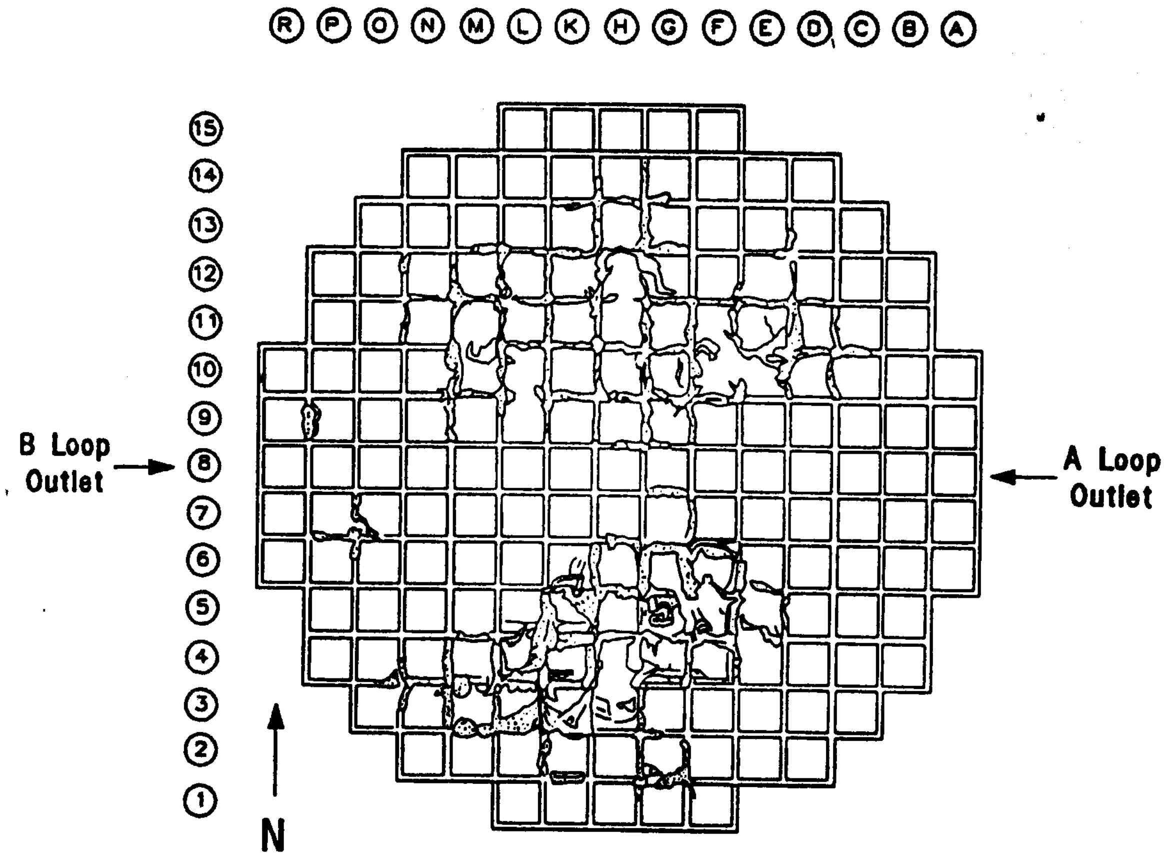


Figure 11. Damage map of the TMI-2 fuel assembly upper grid plate.

sectioning of some partial fuel assemblies lying on the floor of the core cavity. A total of 49 fuel canisters were transferred to the TMI Fuel Handling Building and 16 fuel canisters were shipped to the INEL (see Table 6).

Video surveys of the core cavity walls and floor were made in December, January, and June. Six fuel rod segments were cut from standing fuel rods at the core south (core position L1) and southeast (core positions M2 and N2) sides in December.

The fuel removal activities in FY 1986 made the following changes to the core cavity.

- The fuel assemblies still standing (June 1986) at the core periphery were reduced to 21; 15 (A6, A7, A8, A9, A10, B4, C3, D2, D14, E2, L1, L15, N14, O13 and O12) with upper end fittings and 6 (B12, E2, L1, N2, O3 and R10) without upper end fittings.

- Sufficient loose debris had been removed from the core cavity floor to expose (1) the hard crust near the 70-in. elevation above the original core bottom, and (2) a horseshoe-shaped ring of agglomerated (cemented-together rod bundle remnants) core material projecting inwards from the standing fuel rods above the hard-crust surface as shown in Figure 12. The ring extended from around the 100 in. elevation above the core bottom to the hard crust where it receded creating a cave-like geometry.

In July and August, fuel canisters D-141 and D-153 were unloaded at the INEL and the following samples of core distinct components were acquired for possible and planned future examinations:

12 fuel assembly upper end fittings
10 control or burnable poison rod assembly upper end fittings
2 burnable poison rod assembly retainers

TABLE 6. TMI-2 FUEL CANISTER CONTENTS

Canister Number	Date Loading Completed	Core Material (lb)	Partial (Upper Section) Fuel Assembly: CR	Partial (Upper Section) Fuel Assembly: BPR	Partial (Upper Section) Fuel Assembly: Perimeter	Partial (Upper Section) Fuel Assembly: No ID	Upper End Fittings: End Box and Spider Set	Upper End Fittings: End Box	Upper End Fittings: Spider	Upper End Fittings: BPR Retainer	Fuel Rods: BNDL	Fuel Rods: Individual	Debris Bucket	Unseq. Loose Debris	Location on 10/01/86	Comments
D-136	12/11/85	277				2	3	6	1			2			INEL	
D-141	12/14/85	320	1		1		5	1	1	1					INEL	
D-139	01/08/86	310					3	9		2					INEL	One BPR UEF set
D-140	01/09/86	260					4	4					1		INEL	
D-153	01/16/86	345	2				4	5		2					INEL	
D-155	01/18/86	350	1		1		5	4	3	3					INEL	
D-160	01/20/86	245					4	5	5	2	1				INEL	
D-154	01/23/86	652			1		2	3							INEL	
D-138	01/29/86	233					3	3	2				2		INEL	
D-137	02/03/86	710													INEL	
D-143	02/06/86	1485											3		TMI	
D-151	02/08/86	1594												X	TMI	
D-148	02/11/86	1542												X	TMI	
D-157	02/14/86	1398												X	TMI	
D-146	02/17/86	1528												X	TMI	
D-145	02/19/86	1514												X	TMI	
D-149	02/24/86	1550												X	TMI	
D-158	02/26/86	1558												X	TMI	
D-147	03/01/86	1551												X	TMI	
D-164	03/04/86	1349												X	TMI	
D-156	03/06/86	1627												X	TMI	
D-152	03/07/86	1367												X	TMI	
D-150	03/08/86	1505												X	TMI	
D-163	03/08/86	1422												X	INEL	
D-161	03/15/86	1414												X	TMI	
D-144	03/16/86	1355												X	TMI	
D-165	03/17/86	1457												X	INEL	
D-197	03/18/86	1333												X	TMI	
D-196	03/20/86	1343												X	TMI	
D-132	03/23/86	1418												X	TMI	
D-131	03/25/86	1293												X	INEL	
D-111	03/26/86	1429												X	TMI	
D-117	02/27/86	1343												X	INEL	
D-106	03/30/86	1483												X	TMI	
D-108	03/31/86	1468												X	TMI	
D-128	04/02/86	1358												X	INEL	

TABLE 6. (continued)

Canister Number	Date Loading Completed	Core Material (lb)	Partial (Upper Section) Fuel Assembly: CR	BPR	Perimeter	No ID	Upper End Fittings: End Box and Spider Set	End Box	Spider	BPR Retainer	Fuel Rods: BNDL	Individual	Debris Bucket	Unseq. Loose Debris	Location on 10/01/86	Comments
D-116	04/05/86	1474												X	TMI	
D-125	04/06/86	1332												X	INEL	
D-105	04/09/86	1417												X	TMI	
D-126	04/10/86	1293												X	TMI	
D-127	04/12/86	1357												X	TMI	
D-199	06/17/86	1617				1								X	TMI	
K-501P	Prior to 01/13/86	1118												X	TMI	Vacuumed material
F-432P	Prior to 08/13/86	37												X	TMI	Vacuumed material

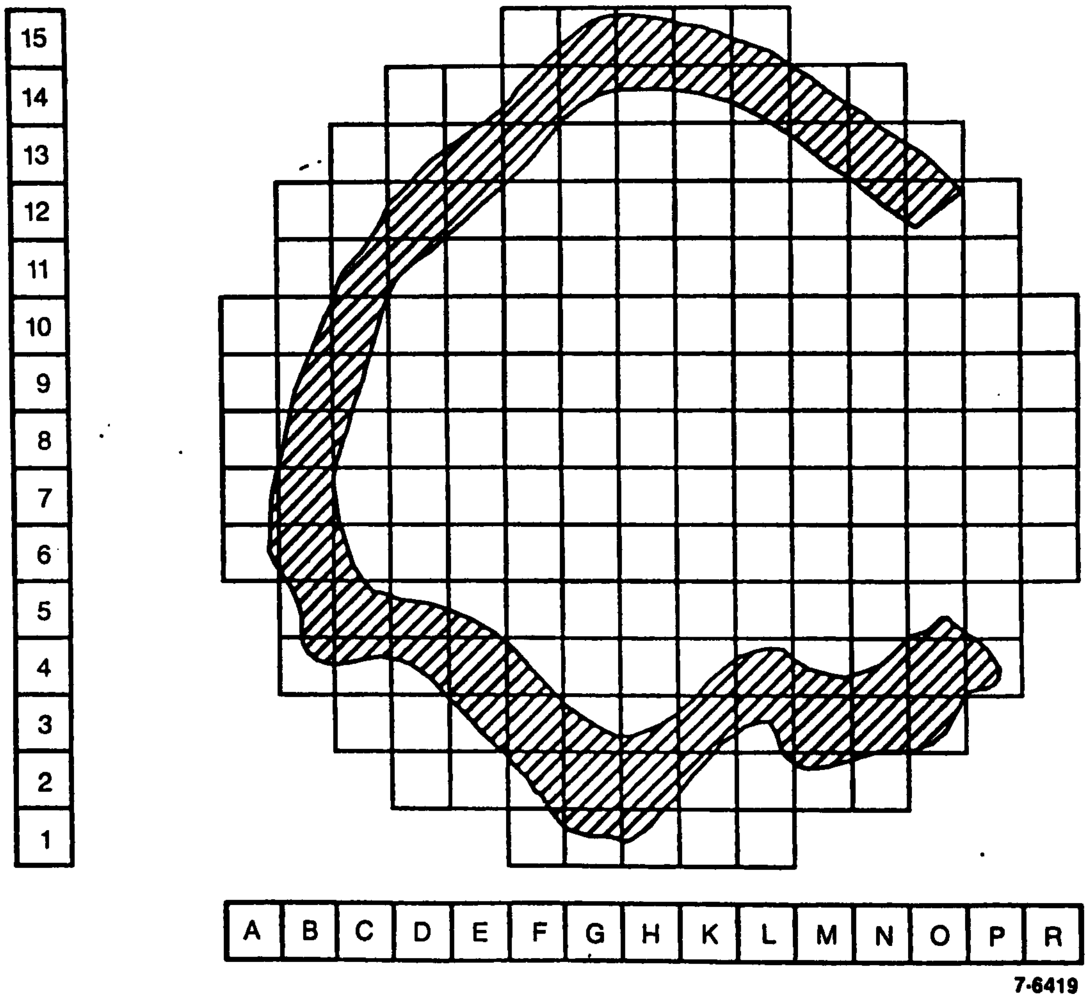


Figure 12. Estimated radial configuration of the upper ridge of agglomerate core material.

15 fuel rod upper sections

7 sets of control rod/guide tube upper sections

See Table 7 for the list of specific TMI-2 fuel assembly upper end fittings placed in temporary storage in drums in the TAN 607 Hot Shop.

3.1.3.7 Biological Growth. In January, 1986 the reactor vessel water turbidity began increasing from a biological (microorganisms) growth in the water. The source of the microorganisms was believed to be the river water, which became mixed during the accident with reactor coolant in the reactor building basement and subsequently introduced into the RCS after the contaminated basement water had been purified by the TMI-2 water cleanup system. The growth of the microorganisms was believed to be caused by (1) spillage of defueling tooling hydraulic fluid into the reactor vessel and (2) other secondary events such as increased lightings, aeration and oxygen dispersion of the reactor vessel water. Both aerobic and anerobic microorganism types were identified in the colony that evolved.

The water turbidity prevented (1) identification of most material which was loaded into the fuel canisters after January 1986 and (2) clear video surveys of surfaces and objects exposed by the loose debris removal.

In April a biological growth cleanup program commenced consisting of chemical (hydrogen-peroxide) addition to the water to kill the organisms and water filtering and feed-and-bleed operation to decrease the water turbidity. The biological growth condition continued to be a problem during the remainder of the fiscal year as hydraulic fluid spillage continued.

3.1.3.8 Core Boring. In July and August 1986, approximately 60 holes were drilled through and/or into the lower core region.

The July drilling was for the purpose of (1) acquiring lower core and reactor vessel lower head region core material samples in the as-stratified condition, and (2) visual (CCTV) inspection of the exposed lower core and

TABLE 7. TMI-2 FUEL ASSEMBLY UPPER END FITTING SAMPLE LIST

Core Position	Fuel Canister Item Number D-141	Fuel Canister Item Number D-153	Storage Drum Location Number	Storage Drum Location Basket	Description	Distinction
B8		9	4	B	Control rod fuel assembly upper end fittings	Examined leadscrew position (P6).
B10		13	3	A	Control rod fuel assembly upper end fittings	Mirror image to canister D-153 item 8 (P6).
C7	3		2	A	Control rod fuel assembly upper end fittings	Fuel and CR/GT rod examination unit.
C11	4		2	B	Control rod assembly upper end fitting	Massive damage to fuel assembly upper end fitting.
D8		3	4	A	Control rod fuel assembly upper end fittings	Core bore drill site.
G3		4	4*	B	Control rod assembly upper end fittings	Upper grid damage region periphery.
H1	11		1	B	Peripheral fuel assembly upper end fitting	Core periphery. In 12/06/85 video survey.
H8	8		1	A	Control rod fuel assembly upper end fittings	Core center position. Examined leadscrew position.
K4		1	1	A	Burnable poison rod assembly retainer	Upper grid damage region.
K15	7		1	A	Peripheral fuel assembly upper end fitting	Near mirror image to canister D-141 item 11 (H1).
L3	6		2	A	Burnable poison rod assembly retainer	Upper grid damage region periphery.
M9	8		3	A	Control rod fuel assembly upper end fittings	Adjacent to the BPR fuel assembly.
N9	4		2	A	BPR fuel assembly upper end fitting	Mating fuel assembly upper end fitting to BPR spider below.
N9	5		1	A	Burnable poison rod assembly spider	Only PBR assembly, 17-4 PH SS parts.

TABLE 7. (continued)

Core Position	Fuel Canister Item Number D-141	Fuel Canister Item Number D-153	Storage Drum Location Number	Storage Drum Location Basket	Description	Distinction
O10		6	3	B	Burnable poison rod fuel assembly upper end fitting	BPR fuel assembly upper end fitting with guide tube stubs.
P6		8	4	A	Control rod fuel assembly upper end fittings	Core east side high damage zone. Mirror image to canister D-153 item 13 (B10).

core support assembly regions and lower head region core material upper surface. Three (3.65) inch diameter holes were drilled through the lower core region at the 10 core positions (D4, D8, G8, G12, K6, K9, N5, N12, 07, and 09) shown in Figure 13 and 1.26-in. diameter holes were drilled into the lower head core material to 8 in. above the reactor vessel lower head below core positions D4, K9, and N12. The core bores and casings were loaded into TMI-2 Fuel Canisters and Shipped to INEL in September. Table 8 lists the core bore locations in the fuel canisters.

The August drilling consisted of using an approximately 2-in. diameter solid-faced bit at 48 locations within the 6 ft (73.2 in.) diameter central core region to make the lower core region fuel removal easier. Table 9 lists the core locations drill depths achieved and drill material left in the lower core region during the drilling campaign.

The core boring program produced the following information about the condition of the lower core region, the core support assembly region and the core material deposited on the reactor vessel lower head.

1. A region of previously molten core materials estimated to be approximately 122 ft^2 (about 10% of the original core volume) was confirmed to be in the lower, central region of the core. This solid structure is approximately 4-ft thick in the center of the core, 1- to 2-ft thick near the core periphery, and is roughly shaped like a bowl extending down toward the bottom of the core. Intact rod stubs exist from the bottom of the core up to the previously molten ceramic material.

   At several core bore locations, metallic inclusions appear in the upper portion of the solid, previously molten material, while in others metallic inclusions are observed near the center and/or bottom of the previously molten regions. The shapes of the metallic inclusions vary widely.

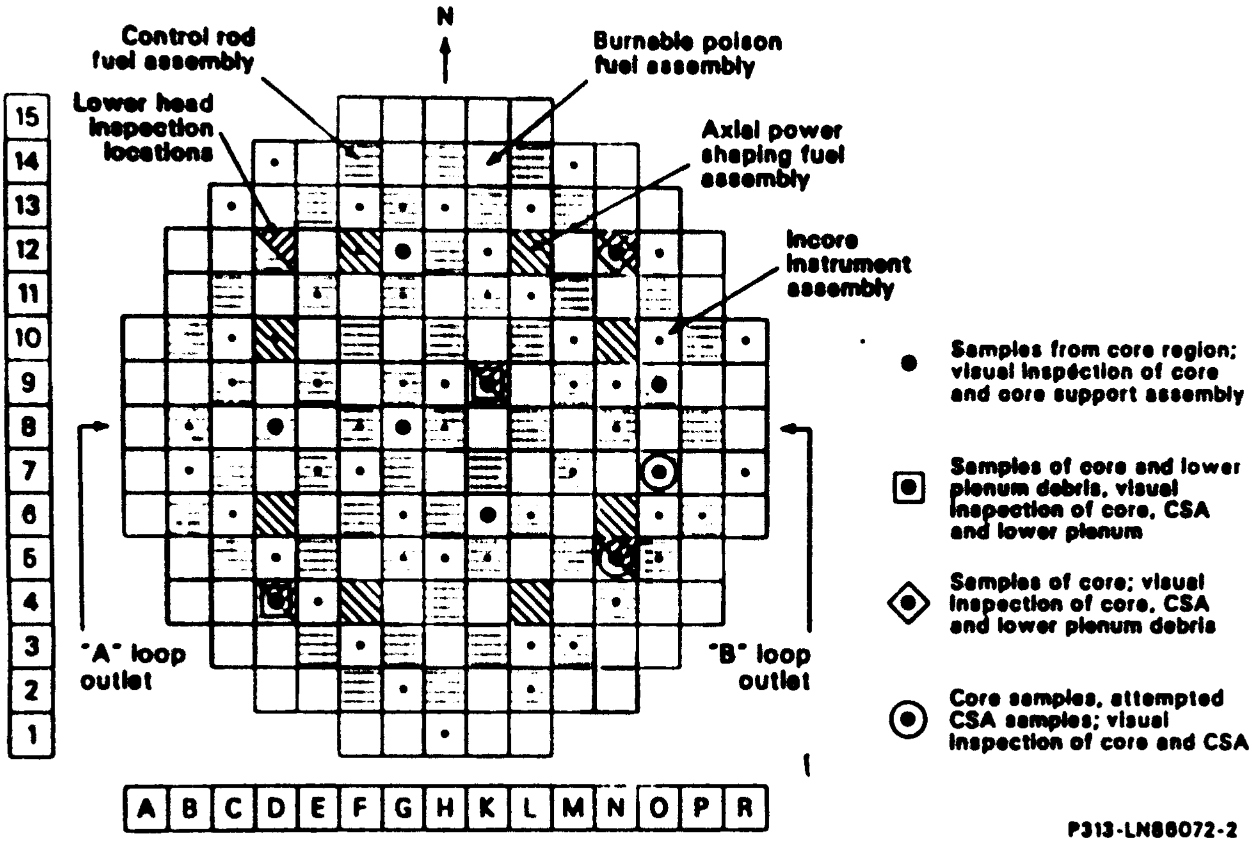


Figure 13. TMI-2 core bore locations.

TABLE 8. TMI-2 CORE BORE SUMMARY

	Segment Number				
Core Position	Core	Lower Plenum	Weight[a] (lb)	Fuel Canister Number	Canister Position
D4:	8		42	D-200	2
		[8]	0	D-200	3
D8	6		40	D-159	2
G8	3		28	D-198	1
G12	4		35	D-198	2
K6	7		0.1	D-200	1
K9:	5		24	D-159	1
		[5]	0	D-159	3
N5:	1		37.5	D-201	1
		[1]	0	D-201	3
N12	2		30	D-201	2
07:	9		23.5	D-118	1
		[9]	0	D-118	3
09	X		27.25	D-118	2
			--[a]		

	Core Material (lbs)	
Canister Number	GPUN	INEL
D-118	50	50.75
D-159	47	64.00
D-198	46	63.00
D-200	34	42.1
D-201	44	67.5
Total	221	286

a. Core material weights in the canisters are:

TABLE 9. AUGUST DRILLING OF TMI-2 LOWER CORE REGION

Core Location		
Radius (in.)	Azimuth (°)	Depth (in.)
5.2	35	0.8
	155	2.3
9.0	30	6.2
	210	46.8
	270	48.1
	330[a]	39.1
12.0	11	5.3
	311	42.3
12.2	12	0.8
	112	40.3
	127	40.0
16.25	16	6.3
	46	45.1
	284	9.0
	316	33.1
	344	5.0
18.5	31.5	37.0
	151.5	36.2
	271.5	35.7
20.5	14.5	35.8
	48.5	1.3
	74.5	42.0
	168.5	28.0
	228.5	47.9
30.0	168	34.8
35.8	175	11.4
	189	14.6
	203	20.2
	217	4.1
	231	1.3
	245	18.6
	259	16.9
	273	20.4
	287	2.8
	301	0
	315	2.2
	350	26.1
36.6	182	47.6
	196	32.7
	210	30.7
	224	3.3
	238	22.0
	266	25.3
	252	5.7
	322	4.0

a. Broke off lower 48-inch section of drill steel.

2. The primary migration path of the previously molten material into the lower plenum appears to be located on the east side of the core near the periphery, primarily at assemblies P-5 and P-6.

3. The CSA appears to be undamaged in those areas where previously molten ceramic materials have frozen in place between the CSA structural members. However, one core instrument guide tube is damaged near the lower grid flow distributor and two others were missing or covered by solidified material below the lower grid.

4. The fuel debris resting on the bottom vessel head near the center of the reactor vessel appears to be loose and relatively fine as compared with the large agglomerated debris existing near the edge of the reactor vessel in the lower plenum. The depth of vessel bottom head fuel debris was estimated to be as follows:

Core Position	Depth[a] (in.)
D4	18
K9	30
N12	12

5. The core boring produced cutting debris including sand like material, shards of fuel rod material, and fuel assembly lower end fitting plugs that (1) settled into the standing rod bundles and onto the horizontal surfaces of the lower core support assembly and reactor vessel lower head core debris or (2) obstructed holes in the core support assembly plates. Future acquisition of core material samples from below the core must be accomplished carefully to avoid or segregate the core material which relocated during the core boring campaign.

a. Depth measured after boring with possible overlay of boring debris.

3.1.3.9 Current State. The current state of the TMI-2 reactor vessel internals is shown in Figure 14. Few regions of the reactor vessel remain unexplored but important core damage progression data may exist in some of those inexplored regions as follows:

1. The northeast, east, and southeast sections of the outer two rings of lower core region fuel assemblies and the core former where the escape paths for the molten core material are believed to be

2. The northeast, east, and southeast sections of the core support assembly where escaping molten core material has solidified

3. The east side of the lower plenum (between the reactor vessel lower head and elliptical flow distributor) where previously molten core material is deposited.

4. The lower region of the core material resting on the reactor vessel lower head where a region of nonfuel core material has been predicted to be.

3.2 Purpose

In addressing the data requirements recommended in the TMI-2 Accident Evaluation Program document, a scope of work has been formulated to support these data needs while recognizing certain limitations in data acquisition inherent in the TMI-2 defueling environment. As such, the purpose of the work plan is the acquisition and examination of samples of core and noncore material from the reactor vessel during completion of defueling, along with video/acoustic documentation of the conditions of the expanding TMI-2 core void. The scope of the sample acquisition plan includes obtaining the following:

1. Additional samples of the loose debris from the lower head region below the elliptical flow distributor plate and from the lower plenum region below the core support plate.

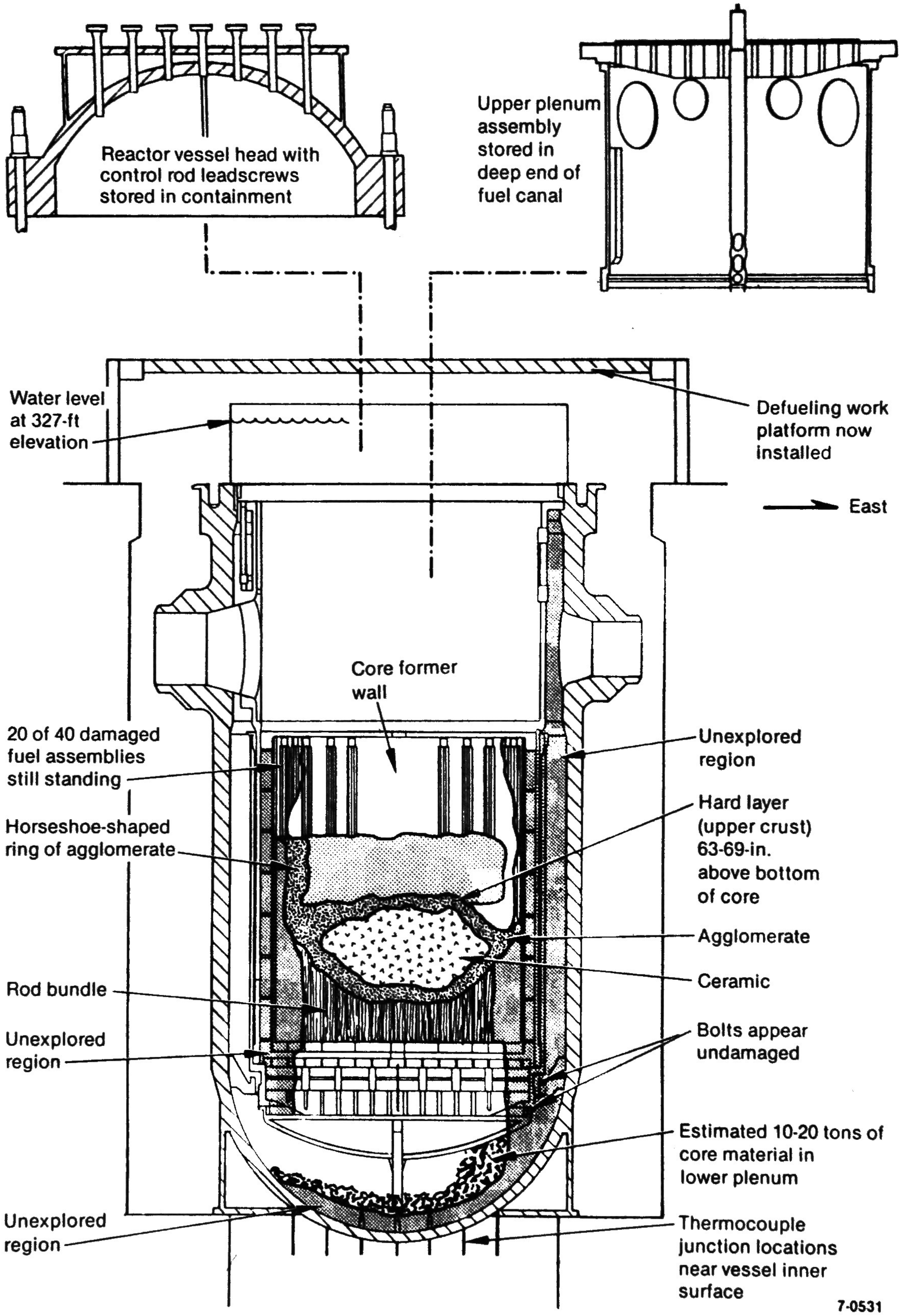


Figure 14. Core and reactor vessel conditions in October 1986.

2. Core support assembly, core former, and instrument guide tube samples.

The specific reactor vessel sample examination objectives include the following:

1. Determination of location and physical characteristics of the molten core material escape paths.

2. Determination of peak temperatures of core and structural materials.

3. Extent of material oxidation and interaction between fuel rod components and other core and structural components.

4. Extent of control rod material relocation and interaction with fuel material.

5. Spatial distribution and physical and chemical characteristics of damaged core and structural materials.

6. Distribution and retention of fission products retained in the reactor vessel and in core materials, including their chemical form and the mechanism of retention.

7. Interaction of burnable poison rod materials with fuel rod materials and the effect on core heatup.

8. The extent and type of damage to the core support assembly, and instrument tube penetrations and amount of material relocation into the lower plenum.

3.2.1 Sample Acquisition

3.2.1.1 Acquisition Equipment. The reactor vessel sample acquisition program has provided the following sample acquisition tooling and examination equipment:

Reference	Description
EGG-TMI-6834	Core Boring Equipment:
Jensen Drilling Co.	Instrumented drilling machine
EGG Drawing 419931	Lead transfer cask
EGG Drawing 419932	Drill indexing platform structure assembly
EGG Drawing 420120	Lower casing clamp hydraulic assembly
EGG Drawing 420126	Drill indexing roller platform assembly
EGG Drawing 420155	Underwater structure assembly
EGG Drawing 420170	Cask roller platform assembly
EGG Drawing 420193	Underwater structure and tilting platform assembly
EGG Drawing 420234	Middle clamp and support assembly
EGG Drawing 420235	Hydraulic control assembly
EGG Drawing 420418	Underwater structure out-of-tolerance indicator
EGG Drawing 420430	Underwater cylinder and rod end clevis
EGG Drawing 420232	REES underwater video camera manipulator assembly
Wild-Heerborg	Computer-aided theodolite indexing system
	Core Topography Equipment:
	Black and white closed-circuit video system, including camera support and articulation tooling
	Enhanced still image videotape processor, including software
	Video-recording-to-enhanced-still-image hard copy processor, including software
GEND-INF-012	Multitransducer searchlight-beam ultrasonic scanner system
EGG-TMI-6531	Loose Debris Collection Tooling:
EGG Drawing 417983	Clamshell-type.loose debris collection tool
EGG Drawing 417984	Rotating-tube loose debris collection tool
EGG Drawing 418075	Loose-debris sample handling cask
	Fuel Canister Unloading Equipment
EGG Drawing 420596	Transfer table assembly
EGG Drawing 420713	Electrical installation (transfer table control)
EGG Drawing 420535	Examination fixture assembly

Reference	Description
EGG Drawing 420429	Sample handling equipment assembly
EGG Drawing 420558	Holddown spring removal press assembly
EGG Drawing 420477	Potting system assembly (core bores)
EGG Drawing 419662	Core barrel disassembly machine (core bores)
EGG Drawing 420387	Laydown and lifting fixtures
EGG Drawing 420590	Vent and drain assembly (fuel canister)
EGG Drawing 420593	Tools and support assemblies
EGG Drawing 421144	Canister lift fixture
EGG Drawing 346945	3 Bechtel side loading debris buckets
EGG Drawing 421446	Special distinct component extraction tools
--	Special hand tools for equipment installation, operation and maintenance
N/A	Rees Model R-93-CCU CCTV camera and remote control system and Panasonic Model NV 5410 video monitor
EGG Drawing 420481	14 core bore containers
	Distinct Component Storage Drums
	Fuel/control rod cut-off band saw
TMI Spectrometer Design Data Package by A. E. Porter and D. W. Akers	TMI Mobile Gamma Spectrometer System/Scanner (for INEL core bore and distinct component examination)

3.2.1.2 Samples. The reactor vessel sample acquisition program has furnished the samples listed in Table 10 to EG&G for examinations.

3.2.2 Examination Reports/Records

The reactor vessel sample examination program has produced the following documentation:

Report Number	Title	Status
	Numerous videotape recordings of CCTV scans between 1982 and 1986. A listing of these tapes is given in Table 12.	
GEND-INF-031 Volume I	Preliminary report of TMI-2 incore instrument damage	Issued January 1984
Letter Report	The FY 1983 Examination of the Lower 3.175 m Section of the H8 Leadscrew from TMI-2	Issued December 1983 Revised March 1984

TABLE 10. REACTOR VESSEL EXAMINATION PROGRAM SAMPLES

January 1987

RV Region	Sample Description			Date Acquired	Remarks/Status
Upper Plenum:					
Core position					
B8	Control rod leadscrew			July 1982	Examined at INEL
E9	Control rod leadscrew			July 1982	Not examined, at INEL in drum
H8:	Control rod leadscrew			July 1982	Examined at INEL
	Leadscrew support tube lower 5-in. section			Early 1984	Examined at BCL
Core Cavity Floor:	Loose Debris:				
Core position	Depth (in.)	Sample Number	Weight (g)		
E9:	Surface	4	16.6	September 1983	Sample set for CSNI
	3	5	91.0	September 1983	Examined at INEL
	22	6	140.7	September 1983	Sample set for CSNI
	29	10	173.9	March 1984	Examined at INEL
	37	11	148.8	March 1984	Examined at INEL
H8:	Surface	1	70.9	September 1983	Sample set for CSNI
	3	2	126.1	September 1983	Examined by B&W
	14	7	135.9	March 1984	Sample set for CSNI
	22	3	152.7	September 1983	Sample set for CSNI
	27	8	152.8	March 1984	Examined at INEL
	30	9	153.1	March 1984	Sample set for CSNI
E11 and 12 and F11 and 12 area	--	1	49.0	May 1986	Examined at INEL
M11 and 12 and N11 and 12 area	--	2	414.9	May 1986	Examined at INEL
M8 and 9 and N8 and 9 area	--	3	2.8	May 1986	Examined at INEL
M3 and 4 and N3 and 4 area	--	4	97.8	May 1986	Examined at INEL
E4 and 5 and F4 and 5 area	--	5	10.6	May 1986	Examined at INEL
B8 and 9 and C8 and 9 area	--	6	61.3	May 1986	Examined at INEL
Reactor Vessel Lower Plenum:	Loose Debris:			July 1985	Examined at INEL
Core position	Sample Number	Description			
Southside (Hole No. 7):	7-1	1.25 x 1.25 x 1 in. rock			1 piece for CSNI
	7-6	0.25 x 0.25 x 0.125 pebble			
Southwest (Hole No. 11):	11-1	1.75 x 1.0 x 1.125 in. rock			
	11-2	1.75 x 1.25 x 1.25 in. rock			
	11-4	1.5 x 1.0 x 1.25 in. rock			
	11-5	2.75 x 2.75 x 2.0 in. rock			2 pieces for CSNI
	11-6	0.6 x 0.5 x 0.5 in. pebble			
	11-7	1.5 x 1.0 x 0.6 in. rock			

TABLE 10. (continued)

RV Region		Sample Description		Date Acquired	Remarks/Status
Core Cavity Wall:		6-in. Sections of Standing Fuel Rods:		December 1985	NDEs performed at INEL
Core Position	Rod Position	Sample Number	Axial Location Between UEF and 7th Spacer Grid		
L1:	B10:	1	Mid-span with transition end		Examined at ANL-E
		2	Adjoining section to UEF--sheared ends		Two pieces for CSNI
M2:	R15	4	Near UEF--sheared ends		Two pieces for CSNI
	R13:	5	Near UEF--sheared ends		Two pieces for CSNI
		6	Adjoining mid-span section--sheared ends		Retained at INEL
M2	L1	3	Near UEF--sheared ends		Examined at ANL-E
Core position C13?		Core instrument string (~4 feet long)		August 1984	At GPUN
Core Cavity Ceiling:					
Core position		Sample Number	Length (cm)		
C7:		Fuel rod upper ends:		December 1985	
		3-6	26.5		Examined at INEL
		3-30	40.8		Examined at INEL
		3-42	28.5		Examined at INEL
		3-61	38.0		Two 6 in. sections for CSNI
		3-70	38.0		Two 6 in. sections for CSNI
		3-94	46.8		Two 6 in. sections for CSNI
		3-14	21.5		At INEL in Shipping Tube 7
		3-18	19.3		At INEL in Shipping Tube 8
		3-20	22.0		At INEL in Shipping Tube 9
		3-28	44.5		At INEL in Shipping Tube 10
		3-35	27.0		At INEL in Shipping Tube 11
		3-88	46.0		At INEL in Shipping Tube 12
		3-89	43.9		At INEL in Shipping Tube 13
		3-98	36.5		At INEL in Shipping Tube 14
		3-102	43.8		At INEL in Shipping Tube 15
		Control rod/guide tube upper ends:		December 1985	
		3-1	36.0		Examined at INEL
		3-3	23.0		Three (1 GT) 4.5 in. long sections for CSNI
		3-7	26.4		At INEL in Shipping Tubes 20 and 21 (GT)
		3-9	15.1		At INEL in Shipping Tube 23
		3-13	17.3		At INEL in Shipping Tube 24
		3-14	23.3		Examined at INEL
		3-16	32.7		Two 4 in. sections for CSNI

RV Region	Sample Description		Date Acquired	Remarks/Status
Core Cavity Wall: (continued)				
Core position (continued)	Sample Number	Length (cm)		
C7: (continued)	Fuel Assembly Upper End Fittings:		December 1985	At INEL in drum 2A
H1:	Fuel Assembly Upper End Fitting		December 1985	At INEL in drum 1B
	Fuel Rod Upper Ends:		December 1985	
	11-2	58.5		At INEL in Shipping Tube 26
	11-3	58.5		At INEL in Shipping Tube 27
	11-5	50.8		At INEL in Shipping Tube 28
B8	CR Fuel Assembly Upper End Fittings		December 1985	At INEL in drum 4B
B10	CR Fuel Assembly Upper End Fittings		December 1985	At INEL in drum 3A
C11	CR Fuel Assembly Upper End Fitting		December 1985	At INEL in drum 2B
D8	CR Fuel Assembly Upper End Fittings		December 1985	At INEL in drum 4A
G3	CR Fuel Assembly Upper End Fittings		December 1985	At INEL in drum 4B
H8	CR Fuel Assembly Upper End Fittings		December 1985	At INEL in drum 1A
K4	BPR Assembly Retainer		December 1985	At INEL in drum 1A
K15	Peripheral Fuel Assembly Upper End Fitting		December 1985	At INEL in drum 1A
L3	BPR Assembly Retainer		December 1985	At INEL in drum 2A
M9	CR Fuel Assembly Upper End Fittings		December 1985	At INEL in drum 3A
N9:	BPR Assembly Spider		December 1985	At INEL in drum 1A
	BPR Assembly Upper End Fitting		December 1985	At INEL in drum 2A
O10	BPR Assembly Upper End Fitting		December 1985	At INEL in drum 3B
P6	CR Fuel Assembly Upper End Fittings		December 1985	At INEL in drum 4A
Core Lower Region:	10 2.4 in.-diameter Core Bores:		July 1986	Examined at INEL. Samples also furnished or retained for ANL-E and CSNI. [See Table 11 for the inventory of rocks (>1 in. in any direction) recovered from the core bores.]

Core position	Weight (lb)	Description
D4 (CR)	42	4-in. long plug of end fitting, 47-in. long rod bundle section with 13 fuel rod, 1 CR/GT and 1 instrument tube remnants, 1 2-in. core of agglomerated material and 3 rocks and some small particles from the agglomerated region.
D8 (CR)	40	33-in. long rod bundle section with 6 fuel rod, 2 CR/GT and 1 instrument tube remnants, 3 cores (1.5 to 2 in. lg.) of agglomerated material and 4 rocks and some small particles from the agglomerated region.

RV Region	Sample Description		Date Acquired	Remarks/Status
Core Lower Region: (continued)				
Core position (continued)	Weight (lb)	Description		
G8 (BPR)	28	27-in. long rod bundle section with 13 fuel rod, 1 BPR/GT and 1 instrument tube remnants, 15 rocks and some small particles from the ceramic region and a 4.5 in. long core of agglomerated material with embedded metallic.		
G12 (BPR)	35	44-in. long rod bundle section with 13 fuel rod, 2 BPR/GT and 1 instrument tube remnants, 29 rocks and some small particles from the ceramic and/or agglomerated region and a 2 in. long core of ceramic material.		
K6 (BPR)	0.1	One 6-in. long fuel rod section with sheared-ends and about 4 in. of fuel pellets.		
K9 (CR)	24	20-in. long rod bundle section with 11 fuel rod, 2 CR/GT and 1 instrument tube remnants, 2 cores (4 and 2.5 in. lg.) of agglomerated material, and 28 rocks and some small particles from the ceramic and or agglomerated region.		
N5 (BPR)	37.5	48-in. long rod bundle section with 12 fuel rod, 1 BPR/GT and the instrument tube remnants and 5 rocks and some small particles from the agglomerated region.		
N12 (CR)	30	47-in. long rod bundle section with 11 fuel rod, 1 CR/GT and the instrument tube remnants and two rocks and some small particles from the agglomerated region.		
O7 (CR)	23.5	30-in. long rod bundle section with 7 fuel rod, 1 CR/GT and the instrument tube remnants and a 2 in.-long core, 4 rocks and some small particles from the agglomerated region.		
O9 (CR)	27.25	44-in. long rod bundle section with 11 fuel rod, 2 CR/GT and the instrument tube renmants and two metallic-appearing rocks and some small particles from the agglomerated region.		

TABLE 11. TMI-2 CORE BORE PREVIOUSLY-MOLTEN CORE MATERIAL ROCK-SIZE SAMPLE DISTRIBUTION

Sample Identification	Mass (g)	Density (g/cc)	Photograph Number	Destination
D4-P2-D	27.17		86-604-5-14	ANL-E
D4-P2-A	19.70	8.93	86-604-5-10	INEL-21
D4-P2-B	19.09		86-604-5-9	CSNI-A3-UK
D4-P2-C	6.81	9.39	86-604-5-8	Archive
D4-P1-B	4.61		86-604-5-4	
D4-P1-A	1.34			
D8-P4-A	67.98	8.67	86-614-1-5	INEL-16
D8-P4-D	62.02		86-614-1-3	ANL-E
D8-P4-B	51.84		86-614-1-4	CSNI-A1-CANADA
D8-P4-C	37.63	7.77	86-614-1-2	INEL-17
G8-P10-A	268.60	8.24	86-604-1-22	INEL-11
G8-P7-A	198.00	7.35	86-604-1-25	INEL-10
G8-P9-A	163.20	7.34	86-604-1-23	INEL-23
G8-P6-B	157.80	7.62	86-604-1-24	INEL-8
G8-P5-B	120.00	7.96	86-604-1-21	INEL-12
G8-P8-A	118.50	7.40	86-604-1-26	INEL-24
G8-P4-B	60.49		86-604-1-30	ANL-E
G8-P4-A	55.14	7.45	86-604-1-28	Archive
G8-P5-A	50.06		86-604-1-27	CSNI-C5-FRG
G8-P8-C	50.03		86-604-1-34	CSNI-C6-SWITZ
G8-P8-B	39.11		86-604-1-32	
G8-P7-B	38.45		86-604-1-33	ANL-E C?
G8-P7-C	36.10	8.80	86-604-1-31	INEL-9
G8-P9-B	33.78		86-604-5-1	ANL-E C?
G8-P6-A	21.10	7.69	86-604-1-29	
G12-P9-A	132.18	7.65	86-604-1-19 and -5-22	INEL-13
G12-P4-A	90.48	7.84	86-604-1-18 and -5-20	INEL-22
G12-P8-A	82.16		86-604-5-23	ANL-E
G12-P10-A	64.28		86-604-1-20 and -5-21	CSNI-C3-JRC
G12-P2-B	60.93	8.47	86-604-5-32	INEL-14
G12-P10-B	54.65		86-604-5-24	CSNI-C4-SWEDEN
G12-P8-B	48.93	7.66	86-604-5-25	Archive
G12-P2-E	46.71		86-604-5-31	
G12-P3-A	45.41	7.70	86-604-5-26	Archive
G12-P2-D	40.93	8.33	86-604-3-3	Archive
G12-P7-A	40.75		86-604-6-34	
G12-P6-A	40.60		86-604-5-33	
G12-P4-B	38.91		86-604-5-18	
G12-P6-E	36.92		86-604-3-6	
G12-P5-A	34.87		86-604-5-17	
G12-P9-B	33.54		86-604-5-29	
G12-P2-C	30.25		86-604-6-35	
G12-P10-D	29.03		86-604-3-9	

TABLE 11. (continued)

Sample Identification	Mass (g)	Density (g/cc)	Photograph Number	Destination
G12-P10-C	28.95		86-604-3-8	
G12-P2-A	28.35		86-604-3-4	
G12-P8-C	28.18		86-604-5-19	
G12-P6-B	25.14		86-604-3-5	
G12-P9-C	24.90		86-604-5-30	
G12-P6-C	24.30		86-604-6-36	
G12-P10-E	24.18		86-604-3-7	
G12-P9-D	20.45		86-604-3-1	
G12-P6-D	19.84		86-604-3-2	
G12-P3-B	19.01		86-604-5-28	
G12-P8-D	18.04		86-604-5-27	
K9-P3-L	75.55		86-604-3-18	ANL-E
K9-P4-G	67.73		86-604-3-19	CSNI-C1-CANADA
K9-P4-E	66.30		86-604-3-20	CSNI-C2-FRANCE
K9-P4-D	61.34	6.92	86-604-3-22	INEL-7
K9-P3-A	55.80	7.56	86-604-3-24	INEL-25
K9-P4-F	46.96		86-604-3-32	CSNI-C7-UK
K9-P3-D	43.82	7.44	86-604-3-26	INEL-8
K9-P3-M	41.63	7.50	86-604-3-25	Archive
K9-P4-H	38.33	6.66	86-604-3-23	INEL-4
K9-P4-N	37.93		86-604-3-27	
K9-P3-C	37.73		86-604-3-21	
K9-P3-J	35.12		86-604-2-5	
K9-P4-L	34.58		86-604-4-35	
K9-P4-M	33.56		86-604-3-30	ANL-E C?
K9-P4-J	26.83		86-604-2-1	ANL-E C?
K9-P3-F	26.68	7.78	86-604-4-36	INEL-6
K9-P3-H	24.54		86-604-2-9	
K9-P4-B	24.53	7.42	86-604-3-28	Archive
K9-P4-A	24.33		86-604-2-3	
K9-P3-G	23.97		86-604-3-31	
K9-P3-I	23.87	7.52	86-604-2-6	Archive
K9-P3-E	23.65		86-604-2-2	
K9-P4-I	23.17		86-604-2-8	
K9-P1-B	19.98		86-604-2-4	
K9-P3-B	19.47		86-604-3-29	
K9-P4-K	19.24		86-604-4-34	
K9-P3-K	18.88		86-604-3-33	
K9-P4-C	18.85		86-604-2-7	
N5-P1-D	35.55	8.28	86-604-5-2	INEL-19
N5-P1-H	22.25	9.09	86-604-5-15	INEL-2
N5-P1-F	18.06		86-604-5-11	
N5-P1-A	10.50	7.97	86-604-5-7	INEL-18
N5-P1-E	10.34		86-604-5-13	
N5-P1-G	9.60		86-604-5-12	

TABLE 11. (continued)

Sample Identification	Mass (g)	Density (g/cc)	Photograph Number	Destination
N5-P1-B	5.97		86-604-5-5	
N5-P1-C	3.59		86-604-5-6	
N12-P1-A	145.64		86-604-5-16	ANL-E
N12-P1-B	0.66		86-604-5-3	
07-P6	76.30	5.43	86-604-1-17 and -3-17	INEL-1
07-P5	34.45		86-604-3-16	CSNI-A2, FRANCE
07-P8-B	21.80		86-604-3-13	
07-P8-C	19.98		86-604-3-10	
07-P8-A	7.24		86-604-3-11	
07-P1-A	4.48	7.61	86-604-3-14	Archive
07-P3	?		86-604-3-15	
09-P1-A	30.00	6.91	86-614-1-1	INEL-3
09-P1-B	20.44	7.22	86-604-3-12	INEL-15

TABLE 12. TMI-2 REACTOR VESSEL INTERNAL CCTV SURVEY RECORD TAPE LISTING

		Tape Record Data			
		TSA File		IRC File	
		3/4 In. Tapes			1 In. Tapes
Date Recorded	Title/Description	Number[a]	Minutes	Number	Number
07/20/82	Quick Look Press Release (core cavity)	37	60		
07/20/82	Quick Look #1 and #2 (core cavity)	22	60		
07/20/82	Quick Look #3 and #4 (core cavity)	25	60		
07/20/82	TMI-2 CCTV Excerpts of Core Internals	29	2		
07/XX/82	Quick Look #2 Enhanced (core cavity)	20	?		
07/XX/82	The Quick Look into the TMI II Unit II (devine narration)	24	18		
08/12/82	TMI-2 Quick Look 3--Edited Version (core cavity)	23	26		
10/13/83	RCS Sampling Inspection, Entry 304, Tape 1 of 2	27	?		
04/04/84	TMI-2 Core Cavity:	108	60	1A3 and 1B3	1,1A2 and 1B2
		109	60	2A3, 2B3	2,2A2 and 2B2
				3A3, 3B3	3,3A2 and 3B2
				4A3, 4B3	4,4A2 and 4B2
				5A3, 5B3	5,5A2 and 5B2
*04/06/84	TMI-2 core cavity				
04/04 and 06/84	TMI-2 core cavity-composite of horizontal scans 3, 5, 8, 10 and 14			6	
07/XX/84	TMI-2 Head Removal	34 and 44	14		
10/25/84	TMI-2 plenum and core cavity				1
10/26 and 27/84	TMI-2 plenum and core cavity				2
10/27/84	TMI-2 plenum and core cavity				3
10/29/84	TMI-2 plenum and core cavity				4
10/29 and 30/84	TMI-2 plenum and core cavity				5
10/30/84	TMI-2 plenum and core cavity				6 and 7
10/31/84	TMI-2 plenum and core cavity				8
10/31 and 11/01/84	TMI-2 plenum and core cavity				9
11/01/84	TMI-2 plenum and core cavity				10
11/01 and 02/84	TMI-2 plenum and core cavity				11
11/02 and 03/84	TMI-2 plenum and core cavity				12
11/03 and 04/84	TMI-2 plenum and core cavity				13
11/05 and 07/84	TMI-2 plenum and core cavity				14
02/20/85	Lower Vessel Examination	5 and 69	45		
02/20/85	TMI-2 Lower Head Inspection	6 and 21	11		
04/01/85	TMI-2 Core Conditions	65	10		
05/15/85	Plenum Lift Entry 613	13	59		

TABLE 12. (continued)

		Tape Record Data			
		TSA File		IRC File	
		3/4 In. Tapes			1 In. Tapes
Date Recorded	Title/Description	Number[a]	Minutes	Number	Number
07/17/85	Reactor Vessel Lower Head: Reel 1 Part 1 of 2	117	60		
	Data Management Tape 1	126	65		1
07/18/85	Reactor Vessel Lower Head: Reel 1 Part 2 of 2	118	60		
	Data Management Tape 2	127	120		
07/20/85	Reactor Vessel Lower Head: Reel 2, Part 1 of 2	115	60		
	Data Management Tape 3	128	120		2
07/22/85	Reactor Vessel Lower Head: Reel 2, Part 2 of 2	116	60		
	Data Management Tape 4	129	180		
07/23/85	Reactor Vessel Lower Head: Reel 3, Part 1 of 1	114	60		3
	Data Management Tape 5	130	150		
09/20/85	Core Void Inspection Entry 692	10	40		
09/20/85	Core Void Inspection Video Entry 693	11	7		
09/23/85	Core Void Video and CPS Interference	12	60		
11/XX/85	Conditions Inside the TMI-2 Reactor Vessel	112	13		
12/06/85	Core Cavity Walls and Floor				1
12/07/85	Core Cavity Walls and Floor				2 and 3
12/11/85	D-136 Fuel Canister Loading	61	41		
12/14/85	D-141 Fuel Canister Loading	63	35		
12/21/85	Core Cavity Walls and Floor				1
12/22/85	Entry 774 Fueled Rod Segment Acquisition: Tape 1	55	60		
	Tape 2	56	60		
	Tape 3	70	23		
12/28 and 29/85	Lower Head Video Exam Phase II	51	60		1 and 2
No Date	TMI-2 Core and Plenum Examination (JMB Narration)	1A and 1B	10		
01/08/86	D-139 Fuel Canister Loading	62	37		
01/09/86	Fuel Canister D-140 Loading	82	80		
01/16/86	D-153 Fuel Canister Loading	60	58		
01/18/86	D-155 Fuel Canister Loading	53 and 120	60		
01/20/86	D-160 Fuel Canister Loading	58	40		
01/21/86	Examination of Fuel Assembly P4, TMI-2 Defueling Entry 801	52	60		
01/23/86	Fuel Canister D-154 Loading	81	40		
01/29/86	D-138 Fuel Canister Loading	59 and 119	15		
02/02/86	D-137 Fuel Canister Loading Entry 811 and 812, Tape 1 of 2	123	60		
02/03/86	D-137 Fuel Canister Loading Entry 814 Tape 2 of 2	124	60		
02/XX/86	Biological Growth in the TMI-2 RCS	72	3		
03/19/86	D-117 Fuel Canister Loading	122	60		
04/12/86	D-128 Fuel Canister Loading	121	60		

TABLE 12. (continued)

Date Recorded	Title/Description	Tape Record Data: TSA File, 3/4 In. Tapes, Number[a]	TSA File, 3/4 In. Tapes, Minutes	IRC File, 3/4 In. Tapes, Number	IRC File, 1 In. Tapes, Number
06/01/86	Long Range Core Void Entry 932 Tape 1 of 3	75	60		
06/01/86	Long Range Core Void Entry 932 Tape 2 of 3	76	60		
06/01/86	Long Range Core Void Entry 932 Tape 3 of 3	79	60		
06/02/86	TMI-2 Core Video Core Bore Info	74	2		
06/08/86	Core Bore Locations	77 and 80	24		
06/XX/86	TMI-2 Core Video Core S/W Quadrant Core Bore Locations	78	60		
07/03/86	Core Bore Drilling Hole #1--N5 Tape 1 of 1	83	60		
07/05/86	Core Bore Location Video Hole #1--N5 Tape 1 of 2	94	50		1
07/06/86	Core Bore Location Video Hole #1--N5 Tape 2 of 2	95	60		
07/08/86	Core Bore Video Inspection Hole #2--W12 Tape 1	86	40		
07/09/86	Core Bore Video Inspection Hole #2--W12 Tape 2	87	60		1 and 2
07/09/86	Core Bore Video Inspection Hole #2--W12 Tape 3	88	60		
07/XX/86	Core Bore Location G8 Hole #3, Part I	84	60		1
07/XX/86	Core Bore Location G8 Hole #3, Part II	85	60		
07/14/86	Core Bore Video Inspection Hole #4 G12 Tape 1 of 2	92	60		
07/15/86	Core Bore Video Inspection Hole #4 G12 Tape 2 of 2	93	35		
07/16/86	Core Bore Video Inspection Hole #5 K9 Tape 1 of 3	89	60		
07/16/86	Core Bore Video Inspection Hole #5 K9 Tape 2 of 3	90	60		1 and 2
07/16/86	Core Bore Video Inspection Hole #5 K9 Tape 3 of 3	91	60		
07/19/86	Core Bore Video Inspection Hole #6 D-8 Tape 1 of 2	96	60		1
07/19/86	Core Bore Video Inspection Hole #6 D-8 Tape 2 of 2	97	60		
07/20/86	Core Bore Video Inspection Hole #7 K6 Tape 1 of 3	98	60		
07/20/86	Core Bore Video Inspection Hole #7 K6 Tape 2 of 3	99	60		1 and 2
07/20/86	Core Bore Video Inspection Hole #7 K6 Tape 3 of 3	100	60		
07/23/86	Core Bore Video Hole #8 D4 Tape 1 of 2	101	60		1
07/23/86	Core Bore Video Hole #8 D4 Tape 2 of 2	102	60		
07/24/86	Core Bore Video Hole #9 O-7 Tape 1 of 2	106	60		1
07/24/86	Core Bore Video Hole #9 O-7 Tape 2 of 2	107	60		

TABLE 12. (continued)

		Tape Record Data			
		TSA File		IRC File	
		3/4 In. Tapes			1 In. Tapes
Date Recorded	Title/Description	Number[a]	Minutes	Number	Number
07/26/86	Core Bore Video Hole #10 O-9 Tape 1 of 3	103	60		
07/26/86	Core Bore Video Hole #10 O-9 Tape 2 of 3	104	60		1 and 2
07/27/86	Core Bore Video Hole #10 O-9 Tape 3 of 3	105	34		
07/XX/86	TMI-2 Core Bore Summary--Locations D4, D8, G8, G12, K6, K9, N5 and N12	110	60		
07/XX/86	TMI-2 Core Bore Summary--Locations O7 and O9	111	9		
07/XX/86	Core Stratification Sampling	132	?		
10/12/86	TMI-2 Core Void Exam		18		

a. TMI-2 AEP TSA videocassette file number.

Report Number	Title	Status
EGG-TMI-6531-1 Revision 1	TMI-2 Core Debris Grab Sample Quick Look Report	Issued March 1984
GEND-INF-044	TMI-2 Leadscrew Debris Pyrophoricity Study	Issued April 1984
GEND-INF-031 Volume II	TMI-2 In-Core Instrument Damage--An Update	Issued April 1984
GEND-INF-012	Design and Operation of the Core Topography Data Acquisition System (initial core cavity topographic mapping)	Issued May 1984
RDD:85:5097-01:01	TMI-2 H8A Core Debris Sample Examination Final Report	Issued July 1984
EGG-TMI-6697	TMI-2 Core Debris--Cesium/Settling Test--Draft Report	Issued September 1984
Letter Report Hmb-268-84	Analysis of TMI-2 'A' Steam Generator Hot Let Resistance Thermal Detector	Transmitted November 16, 1984
GEND-INF-060 EGG-TMI-6630 (Draft)	Preliminary Report: TMI-2 Core Debris Grab Samples--Analysis of First Group of Samples	Issued July 1985
GEND-INF-052 EGG-TMI-6685 (Draft)	Examination of H8 and B8 Leadscrew from Three Mile Island Unit 2 (TMI-2)	Issued September 1985
GEND-INF-067	Examination of the Leadscrew Support Tube from Three Mile Island Reactor Unit 2	Issued March 1986
GEND-INF-075 Parts 1 and 2 EGG-TMI-6853 (Draft)	TMI-2 Core Debris Grab Samples--Examination and Analysis	Issued September 1986
EGG-TMI-7385	TMI-2 Core Bore Acquisition Summary Report	Issued September 1986
GEND-INF-074	TMI-2 Core Cavity Sides and Floor Examinations December 6, 7, 21 and 22, 1985	Draft Issued September 1986

3.2.3 Reactor Vessel Internals Sample Examination Findings

The results of the in situ sample examinations conducted to date are summarized in this section.

Core Debris Grab Samples. Examination and analysis of the 11 upper core loose debris grab samples and probing the loose debris provided the following TMI-2 accident information:

- Some particles exceeded UO_2 melting (3100 K) during the accident.
- Loose debris extends downward about 2.5 ft to a hard object 6 ft above the original core bottom and outward to at least the next-to-outside ring of fuel assemblies (approximately 20% of the core volume).
- The hard-object upper surface is relatively flat but irregular and extends to near the core periphery.
- Significant axial and radial mixing of core materials has occurred in the loose debris bed.
- The core material distribution in the loose debris indicates a depletion of lower melting temperature structural and poison materials.
- Fission product retention normalized to the measured uranium concentration is as follows:

Isotope	Abundance (percent)
Sr-90	79 to 102
Ru-106	35 to 86
Sb-125	18 to 38
I-129	10 to 28
Cs-137	6 to 32
Cr-144	90 to 130

Reactor Vessel Internals Documentation. This examination task commenced in 1982 and includes periodic surveys with CCTV and sonar (acoustic topography) devices. The findings to date include the following:

- The core topography data taken before head removal indicated that the void in the core region below the upper grid plate occupied 330 ft^3 (9.3 m^3) and extended rapidly into the peripheral row of fuel assemblies. Local variations in the nominal void radius ranged from exposed sections of core former wall to apparent standing fuel rods 12 to 14 in. inside the core former boundary. Significant quantities of core materials were suspended from the underside of the upper core support grid (1982 and 1983).

- Ablation of the plenum assembly lower grid plate and occurred in two or more mid-radius areas as shown in Figure 11 (1985).

- Downcomer and peripheral lower core support assembly structures appear to be undamaged (1985).

- Ten to twenty tons of probable core material had collected in the region between the reactor vessel lower head and the elliptical flow distributor ranging in form from coffee-ground size particles to a vertical-curtain appearing wall and appearing like lava rock (1985).

- Previously molten material was hanging or attached to the elliptical flow distributor below core positions L2, L14, and N3 (1985).

- Regions of flow channel blockage from fuel rod swelling were not observed in any regions of the still-standing fuel bundles (1985).

- Increased upper end fitting damage had occurred at fuel assemblies with burnable poison (Al_2O_3-B_4C) rods (1985).

- Still standing fuel rod regions had regions of zircaloy interaction with team (embrittlement), uranium dioxide (liquefaction) and stainless steel and Inconel (eutectics) (1985).

- Fuel assemblies were loaded into the TMI-2 core with identification markings orientated to south instead of north and without orifice rod assemblies in the peripheral fuel assemblies except at startup neutron source sites (core positions B13 and P4).

- Loose debris removal from the core cavity floor exposed a horseshoe-shaped ring of agglomerated (cemented-together rod bundle remnants) core material projecting inwards from the standing fuel rods above the hard-crust surface as shown in Figure 12. The ring extends from around the 100 in. elevation above the core bottom to the hard crust where it recedes creating a cave-like geometry (1986).

- A region of previously molten core materials estimated to be approximately 122 ft^2 (about 10% of the original core volume) was confirmed to be in the lower, central region of the core. This solid structure is approximately 4-ft thick in the center of the core, 1- to 2-ft thick near the core periphery, and is roughly shaped like a bowl extending down toward the bottom of the core. Intact rod stubs exist from the bottom of the core up to the previously molten ceramic material. At several core bore locations, metallic inclusions appear in the upper portion of the solid, previously molten material, while in others metallic inclusions are observed near the center and/or bottom of the previously molten regions. The shapes of the metallic inclusions vary widely (1986).

- The primary migration path of the previously molten material into the lower plenum appears to be located on the east side of the core near the periphery, primarily at assemblies P-5 and P-6 (1986).

- The core support assembly appears to be undamaged in those areas where previously molten ceramic materials have frozen in place between the CSA structural members. However, one core instrument guide tube is damaged near the lower grid flow distributor and two others were missing or covered by solidified core material below the lower grid (1986).

- The fuel debris resting on the bottom vessel head near the center of the reactor vessel appears to be loose and relatively fine as compared with the larger agglomerated debris existing near the edge of the reactor vessel in the lower plenum. The depth of bottom vessel head fuel debris was estimated (1986) to be as follows:

Core Position	Depth[a] (in.)
D4	18
K9	30
N12	12

Control Rod Leadscrew and Leadscrew Support Tube Examinations. The principal findings of the leadscrew and leadscrew support tube examinations were:

- Less than 2% of any core radionuclide or material was deposited on metal surfaces in the plenum assembly, with the deposited core material depleted of control rod poison material

- Upper plenum metal temperatures did not exceed the melting point (1700 K)

- Upper plenum metal temperatures ranged from 1255 K at the upper plenum inlet (center) to 755 K near the outlet

a. Depth measured after boring with possible overlay of boring debris.

- Surface deposits on the leadscrew support tube consist of a highly-adherent inner layer and loosely-adherent outer layer with a concentration of control rod poison material deposited on the inner adherent layer.

Reactor Vessel Lower Head Loose Debris. The principal findings of the reactor vessel lower head loose debris preliminary examinations are as follows:

- Material is inhomogeneous, porous, and cracked with average density of 7.2 gm/cc

- Elemental composition includes uranium (.7), zirconium (.2), iron, aluminum, chromium, nickel, and silicon

- Iron, aluminum, chromium, and nickel inclusions occur at grain boundaries

- Radioactivity concentrations occur at pore locations

- Fission product retention normalized to the measured uranium concentration is as follows:

Isotope	Abundance (percent)
Cs-137	9 to 22
I-129	0.6 to 8
Sb-125	3 to 10
Ru-106	4 to 9
Ce-144	106 to 124

Core Distinct Component Examination. The examination of the TMI-2 core distinct components commenced in FY 1987 and included segments of fuel rods and control rod/guide tubes and fuel assembly upper end (upper end boxes, spiders and BPR retainers) fittings. Preliminary findings of the examinations include the following:

- A large temperature gradient existed at the core top.

- Fuel assemblies were loaded into the TMI-2 core with identification markings oriented to south instead of north.

- Orifice rod assemblies were not loaded into the TMI-2 peripheral fuel assemblies except at start-up neutron source sites (core positions P4 and B13).

- Fuel rod upper plenums include spacer sleeves between the spring and pellet stack (instead of ZrO_2 washers) and the spring and fuel rod upper end cap.

- Silver-indium-cadmium poison material relocated upwards inside the control rods into the holddown spring region while molten indicating possible control rod cladding collapse and core exit peak temperatures during the core heatup phase.

Core Bore Examinations. The examinations of the core bores commenced in September 1986 and the preliminary findings are as follows:

- All four small diameter (1.45 in.) lower-plenum core bore tubes were empty to provide additional indication that the reactor vessel lower head core debris is loose-rock like in form where the core boring penetrated.

- The core region core boring partially recovered core material as summarized in Table 13, which includes the core material stratification estimates derived from the core boring parameters and the video survey records. At most core positions, the boring was sufficiently offset from the fuel assembly centerline to capture samples of the control or burnable poison rods in the 2.4 in. diameter bore. The loss of core material from the core bores provides an indication that the ceramic and agglomerated core material regions are frangible by milling-type tools.

TABLE 13. TMI-2 CORE BORE ACQUISITION SUMMARY--PRELIMINARY

Core Position	Weight (lb)	Stratification Summary[a,b,c] (inches from fuel rod bottom): 0 10 20 30 40 50 60 70 80 Lower End Fitting / Rod Bundle Region / Transition Region / Ceramic Region / Agglomerate Region EEERRRRRRRRRRRRRRRRRRRRRRRRRRRRRRTTTTTTTCCCCCCCCCCCCCCCCCCCCCCCCCCCCCAAAAAAAAAAAA
D4 (CR)	42	EEERRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRAAAAAAAAAAAAAAAAA 13 Fuel Rods, 1 CR/GT, 1 IT 2 in. lg. core, 6 rocks and some small particles[d]
D8 (CR)	40	EEERRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRAAAAA 6 Fuel Rods, 2 CR/GT, 1 IT 3 2 to 3 in. lg. cores, 4 rocks and some small particles[d]
G8 (BPR)	28	EEERRRRRRRRRRRRRRRRRRRRRRRRRRRTTTCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCAAAAAAAA 13 Fuel Rods 1 BPR/GT, 1 IT 15 rocks and some small particles[d] 4 in. lg. core with embedded metallic
G12 (BPR)	35	EEERRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRTTTCCCCCCCCCCCCCCCCCCCAAAAAAAAA 13 Fuel Rods, 2 BPR/GT, 1 IT 29 rocks and some small particles 2 in. lg. core of ceramic
K6 (BPR)	0.1	EEERRRRRRRRRRRRRRRRRRRRRRRRRTTTTTTTCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCC 1 6-in. Fuel Rod Section
K9 (CR)	24	EEERRRRRRRRRRRRRRRRRRRRRTTTTTCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCCAAAAAAAA 11 Fuel Rods, 2 CR/GT, 1 IT 4 in. core 28 rocks and some small particles[d] 2.5 in. core

TABLE 13. (continued)

Core Position	Weight (lb)	Stratification Summary[a,b,c] (inches from fuel rod bottom) 0 10 20 30 40 50 60 70 80
		Lower End Fitting / Rod Bundle Region / Transition Region / Ceramic Region / Agglomerate Region EEERRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRTTTTTTTTCCCCCCCCCCCCCCCCCCCCCCCCCCCCAAAAAAAAAAAA
N5 (BPR)	37.5	EEERRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRAAAAAAAAAAAAAAAAAAAAAA 12 Fuel Rods, 1 BPR/GT, 1 IT / 8 rocks and some small particles[d]
N12 (CR)	30	EEERRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRAAAAAAAAAAAAAAAAAAAA 11 Fuel Rods, 1 CR/GT, 1 IT / 2 rocks and some small particles[d]
07 (CR)	23.5	EEERRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRRAAAAAAAAAAAAAAAAAA 7 Fuel Rods, 1 CR/GT, 1 IT / 2 in. lg. core, 7 rocks and some small particles[d]
09 (CR)	27.25	EEERRRRRRRRRRRRRRRRARRRRRRRRRRRRRRRRRRRRRRRRRRRRAAAAAAAAAAAAAAAAA 11 Fuel Rods, 2 CR/GT, 1 IT / Two metallic appearing rocks and some small particles[d]

a. Stratification estimates from EGG-TMI-7315 (boring parameters and video survey records)

b. Underline identifies where material was recovered by core bores.

c. 1 type space equals 1 in.

d. Including possible fuel pellet fragments.

3.3 Detailed Work Plan

The reactor vessel sample acquisition and examination program work plan details for FY 1987 are contained in the following work packages:

Work Package Number	Work Package Title
751421300	RCS Equipment/Building Characterization
755420100	Examination of Subsurface Debris Bed Samples
755420200	Reactor Vessel Internals Documentation
755420600	Core Stratification Sample Examination
755421200	Core Distinct Components Examination
755421600	TMI-2 Lower Vessel Debris Examination
755422100	Core Sample Examination Support

Preliminary plans for acquisition and examination of samples of the core instrument string guide tubes where they penetrate the reactor vessel lower head have also been developed for FY 1988 (acquisition) and FY 1989 (examination).

Table 14 summarizes the in situ measurements and sample examinations that are involved in this work plan. The table includes the AEP-designated sample priority (1-20), the quantity of in situ measurements or samples, the TMI-2 accident information expected from the examinations, and the examination techniques, which will be used to obtain the information.

As the TMI-2 defueling program progresses it is expected that "samples of opportunity" will present themselves. Acquisition of these serendipitous samples and occurrence of unexpected observations during sample examination will modify the currently planned work scope documented herein. The work package format used in the work plan will accommodate such modifications as they occur.

Also included in this work plan are four tasks for improved examination methods development as follows:

TABLE 14. REACTOR VESSEL IN SITU MEASUREMENT AND SAMPLE ACQUISITION AND EXAMINATION PLAN SUMMARY

Measurement/Sample Description	Priority[a]	Sample Quantity	TMI-2 Accident Information	Examination Methods[b]
1. Loose Debris:			All:	
			Color, surface texture, weight, radioactivity	2, 3, 5
a. Large volume sample from upper debris bed	3	5	Particle size, density and distribution (transportability)	4, 5, 6
b. Single particle samples from lower head debris	7	8	Metal structure, grain size, core metal and oxygen distribution, peak temperature	10
c. Large volume samples from lower head region	6	2	Core metal abundance including U-235	14, 16
			Retained fission product abundance and distribution	18, 19, 20, 21, 13
2. In Situ Data Recordings:				
a. Core cavity video survey after bulk defueling	--	N/A	Core former and CSA damage, presence of core material outside the core boundary	1
b. Video survey of lower plenum after loose core debris removal	2	N/A	CSA, core instrument guides and reactor vessel lower head damage, presence of core material fused to CSA, guides or vessel lower head	1
3. Core Bore Samples:	1, 5, 9	9	Weight, sample location, and particle color, shape, size, surface texture and quantity	3, 5
a. 2.4 in. diameter cores	1, 5, 9	8	Weight, color, shape, size, surface texture, density	3, 5, 6
			Metal structure, grain size, core metal and oxygen distribution, peak temperature	10
			Core metal abundance and chemical form	14, 17
			Uranium enrichment	16
			Retained fission product abundance and distribution	18, 19, 20, 21, 13
b. Rock-size (>1 in. in any direction) pieces of previously molten material	1, 5, 9	101	Weight, color, shape, size, surface texture, density	3, 5, 6
			Metal structure, grain size, core metal and oxygen distribution, peak temperature	10
			Core metal abundance and chemical form	14, 17
			Uranium enrichment	16
			Retained fission product abundance and distribution	18, 19, 20, 21, 13
c. 4 in. long fuel rod segments	8	98	Metal structure, grain-size and interaction with other metals and chemicals, peak temperature	10
			Retained fission product abundance and distribution	18, 19, 20, 21, 13
d. 4 in. long control rod/guide tube segments	8	7	Metal structure, grain-size interaction with other metals and chemicals, peak temperature	10
			Poison material alloy depletion	14
			Captured fission product abundance and distribution	18, 19, 20, 21, 13
e. 4 in. long burnable poison rod/guide tube segments	8	6 Al_2O_3-B_4C	Metal structure, grain-size interaction with other metals and chemicals, peak temperature	10
			Captured fission product abundance and distribution	18, 19, 20, 21, 13
f. 4 in. long instrument tube sections	8	8	Metal structure and grain-size, interaction with other metals and chemicals, peak temperature	10
			Captured fission product abundance and distribution	18, 19, 20, 21, 13

TABLE 14. (continued)

Measurement/Sample Description	Priority[a]	Sample Quantity	TMI-2 Accident Information	Examination Methods[b]
4. Distinct Core Components:				
a. Fuel rod upper ends	8	2	Color, shape, length, surface texture	3
			Fission product axial distribution	7
			Component location and dimensions	8
			Metal structure, grain-size and interaction with other metals and chemicals, peak temperature	10
			Retained and captured fission product abundance and distribution	18, 19, 20, 21
b. Control rod/guide tube upper ends	8	2	Color, shape, length, surface texture	3
			Fission product axial distribution	7
			Component location and dimensions	8
			Metal structure, grain-size, interaction with other metals and chemicals, peak temperature	10
			Poison material alloy depletion	14
			Captured fission product abundance and distribution	18, 19, 20, 21
5. Core-Instrument Guide Structure Interface with Reactor Vessel	14	6	Color, surface texture, damage zone locations	3
			Internal defects (cracks, porosity, etc.)	8
			Metal structure, grain-size, interaction with other metals or chemicals, peak temperature	10, 11

a. Priority values 1 through 20 are listed in Table 3.

b. Examination methods:

1. Video surveys with electronically-enhanced still image production
2. Ion-chamber gamma detection
3. Color and/or black and white photography
4. Sieving
5. Balance weighing
6. Immersion density
7. Gamma spectrometry scanning
8. Neutron radiography
9. Autoradiography
10. Metallography with scanning electron microscopy or auger spectrometry
11. Rockwell hardness
12. Compression strength
13. Microgamma scanning
14. Inductively-coupled-plasma emission spectrometry
15. Spark source mass spectrometry
16. Delayed neutron radiochemistry
17. Bulk oxygen analysis
18. Gamma spectrometry
19. I-129 radiochemistry
20. Sr-90 radiochemistry
21. Kr-85 radiochemistry.

Task Title	Purpose
ORIGEN benchmarking	Confirm accuracy of ORIGEN prediction of uranium utilization and conversion by comparing measured TMI-2 fuel values to code predictions.
Bulk Oxygen Content	Develop a potentiometric titration method for measuring the oxygen abundance in metal bearing samples for improved determination of the core metals chemical forms.
Fission Gas Analysis	Develop INEL capability for measuring Kr-85 abundance in ceramic core materials using a known method.
Gamma spectrometry micro scanner	Develop INEL capability for measuring gamma ray emitter distribution on a microscale for comparison to SEM measurements of core metal distribution and possible clues to fission product chemical form.

3.3.1 Product

The product of the RV sample acquisition and examination work plan in FY 1987 and beyond is shown on Table 15.

3.4 Synopsis

The exploration of the reactor vessel internals was nearly completed in FY 1986. The core apparently reconfigured into four zones; the original rod-bundle-and-end-fitting geometry (43% by weight), a large (26% by volume) cavity in the upper core region, loose debris (unmelted and previously molten core material) mixture (25% by weight) and previously-molten core material (33% by weight). An estimated 40% of the previously-molten core material relocated from the core boundaries into the reactor vessel lower plenum.

The few reactor vessel regions not yet explored may contain important core damage progression data such as:

a. The actual escape pathways of the molten core material in the northeast, east and southeast sections of the lower core region.

TABLE 15. REACTOR VESSEL SAMPLE ACQUISITION AND EXAMINATION WORK PLAN PRODUCT LIST

Work Package Number	Product Item	Target Completion Date
CCTV Survey Videocassette Recordings		
REP	Interdefueling cavity debris survey	TBD
751420200	Post bulk-defueling core cavity walls and floor	TBD
751420200	Post-loose-debris-defueling of reactor vessel lower plenum region	TBD
Core Component and Material Samples		
751421600	Core debris from reactor vessel lower head central region (2 samples)	TBD
751421200	Fuel assembly lower sections (6)	September 1987
TBD	Core instrument reactor vessel penetration nozzle region (6)	September 1988
Videorecording Enhanced Still-Image Excerpts and Hardcopy Picture Albums		
755420200	Reactor vessel video survey enhanced still-image pictures for early FY-1986	February 1987
755420200	Reactor vessel video survey enhanced still-image pictures for FY-1986 and 1987	February 1988
Technical Reports		
755420100	Core cavity substrata loose debris characterization final report: Draft Final	March 1987 August 1987
755420600	Core bore examination periodic progress reports	May 1987 thru FY-1987 at six month intervals
755420600	Core bore examinations report: Draft Final	FY-1988 FY-1988
755421200	Standing fuel rod segment nondestructive examinations report: Draft Final	January 1987 July 1987

TABLE 15. (continued)

Work Package Number	Product Item	Target Completion Date
Technical Reports (continued)		
755421200	Rod (fuel and control) segment examinations--preliminary report	March 1987
755421200	Rod (fuel and control) segment examinations--final report	September 1988
755421600	Reactor vessel lower head loose debris examination report: Draft	February 1987
	Final	September 1987
TBD	Core instrument reactor vessel penetration nozzle examination report	February 1989
Improved Methods		
75542100	Summary report of accomplishment in the ORIGEN benchmarking and developing improved methods for measuring bulk oxygen abundance, Kr-85 (fission gas) abundance, and gamma emitter micro-distribution	September 1987

b. Condition of previously-molten core material solidified in the core support assembly before reaching the reactor vessel lower head region.

c. Condition of previously-molten core material now solidified underneath the elliptical flow distributor on the eastside of the reactor vessel lower head.

d. The dimensions and composition of solid possibly nonfuel core material predicted to be resting on the reactor vessel lower head central region.

In addition, the core bores may not have recovered a representative sample of the highest temperature (ceramic) material from the previously-molten core central region.

The AEP-requested (see Table 3) sample acquisition and examination tasks which cannot be satisfied for either physical or budgetary reasons include the following:

AEP Priority	Task	Reasons
2	Central core bore between the CSA and lower head	Unfused material could not be collected with the core bore
8	Intact, part length (upper end) burnable poison rod assembly	Identification marking removed during defueling
9	B-loop hot leg RTD thermowell surface deposit	Area radioactive contamination is excessive
13	Samples of the interaction zone between core materials and the lower core support assembly	Insufficient budget
15	Samples of the interaction zone between the reactor vessel lower head surface and the core materials	Insufficient budget
16	Samples of the interaction zone between core materials and the core former	Insufficient budget
17	Fission products retained on upper plenum surfaces	Insufficient budget
18	Control rod leadscrews from the upper plenum region	Insufficient budget
19	Fuel assembly upper end fitting examinations	Insufficient budget

The current strategy concerning the unsatisfied items above is to take special actions, as necessary, to acquire the samples that might be destroyed or altered by the TMI-2 defueling activities.

The sample acquisition and examination plan described herein is intended to provide sufficient data to adequately describe the TMI-2 accident scenario. Discussions with GPUN are progressing concerning completing the reactor vessel internal exploration and acquiring representative samples from the highest-temperature core material region.

In addition, special efforts are being made to determine the fission product chemical form by developing improved techniques for measuring bulk oxygen abundance and microdistribution of gamma emitters.

4. RCS SAMPLE ACQUISITION AND EXAMINATION WORK PLAN

4.1 Introduction

TMI-2 reactor coolant system piping and components are shown in Figure 15 and include the following:

- A reactor vessel containing the uranium fueled core. These are covered by a separate sample acquisition and examination work plan described in Section 3.

- Dual reactor cooling loops (A and B) consisting of the candy-cane-shaped hot legs from the reactor vessel upper plenum to the steam generator tops, two single-pass type steam generators, dual (four total) cold legs from the steam generator bottom back to the reactor vessel via the four reactor coolant pumps.

- A pressurizer connected to the cooling loops by a surge line from the A-loop hot leg to the pressurizer bottom and a spray line from the A-loop cold leg (downstream of pump RC-P-2A) to the pressurizer top.

- Dual core flooded tanks connected to the reactor vessel.

During and after the TMI-2 accident sequence that lasted until natural circulation cooling commenced (approximately 30 days after accident initiation), many events occurred that affected the character and distribution of core materials and fission products, which escaped from the reactor vessel to the reactor coolant system. The most significant events include the following:

- Fission product and a small uranium fraction release commenced in the reactor vessel at approximately 138 min after accident initiation when fuel rod rupture commenced. Reactor coolant pump operation had ceased, and the available escape paths were

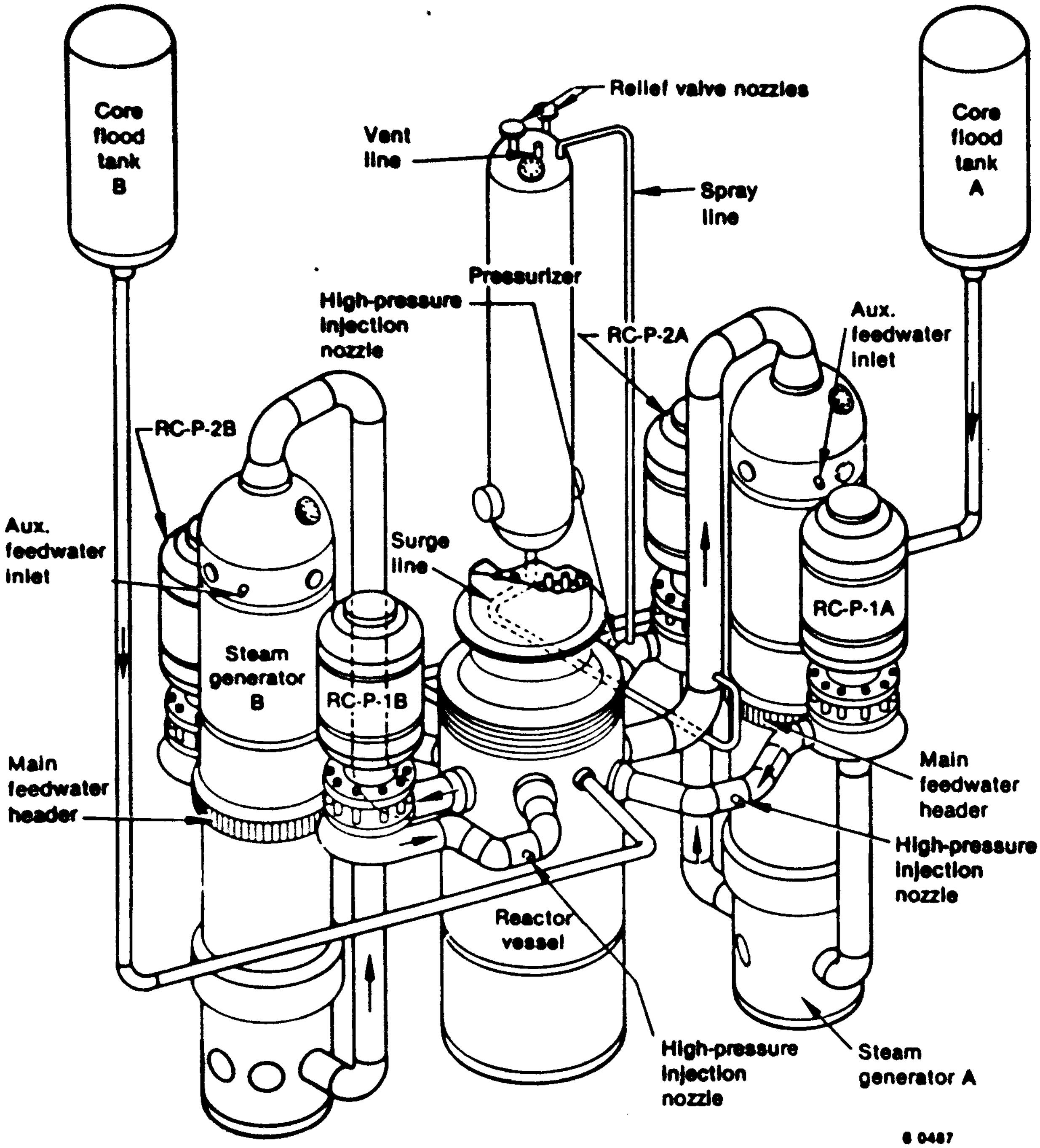


Figure 15. TMI-2 reactor coolant system piping and components.

(a) through the A-loop hot leg, the surge line, and the pressurizer because the pilot operated relief valve (PORV) was stuck open, releasing reactor coolant to the reactor basement through the reactor coolant drain tank, and (b) through the A-loop cold leg to the letdown line (downstream of reactor coolant pump RCP-P-1A).

- Reactor coolant system temperatures exceeded the coolant saturation temperature from 136 min to approximately 16 h after accident initiation in the hot legs and occasionally in the cold legs. Measured coolant temperatures did not exceed 725 K.

- The PORV/pressurizer escape path was closed at 142 min after accident initiation.

- Zircaloy-steam reaction became significant at 144 min, releasing hydrogen and other chemical reaction products into the coolant in the reactor vessel. Core material temperatures continued to rise and reached temperatures exceeding 2900 K, which could (a) generate aerosols from low volatility materials and chemical reactions, and (b) accelerate the escape of fission products for the uranium dioxide.

- A reactor coolant sample taken at 163 min contained 140 μCi/mL gross activity.

- Reactor coolant pump RC-P-2B was energized from 174 to 192 min after accident initiation. This event is believed to have reflooded the over-heated core region, fragmenting most of the standing fuel in the upper core region and creating the upper core region cavity, and caused circulation of core material particles and fission products throughout the B-loop components.

- The PORV/pressurizer escape path was reopened from 192 to 197 min and from 220 to 318 min.

- At 227 min, a significant relocation of core material from the core region into the flooded reactor vessel lower plenum region occurred, which would likely increase the escape of core material and fission products to the letdown system escape path.

- A sustained high pressure injection period commenced at 267 min and continued to 544 min.

- A reactor coolant sample taken at 283 min contained >500 μCi/mL gross activity.

- The PORV/pressurizer escape path was cycled open repeatedly during the 340 to 458 min period to prevent RCS over pressurization and was also opened from 458 to 550, 565 to 589, 600 to 668, 756 to 767, and 772 to 780 min to depressurize the RCS for core flood injection.

- Core flood tank injection probably occurred from 511 to 550 min after accident initiation. This event is believed to have caused a back flow leak path to develop from the reactor coolant system to flood tank B due to incomplete check valve reseating.

- A reactor coolant system pressurization in the 840 to 900 min period probably forced coolant and core material aerosols and volatile fission products from the reactor vessel into flood tank B.

- Forced circulation cooling of the reactor was resumed at 949 min (15 h 49 min) through the A-loop with reactor coolant pump RC-P-1A.

- Letdown flow was lost from 18 h 34 min to 26 h 30 min.

- A reactor coolant sample taken at 36 h and 15 min measured >1000 R/h on contact.

- Natural circulation cooling of the reactor commenced 30 days and 10 h after accident initiation.

- Reactor coolant water cleanup using the SDS/EPICOR-II system commenced 2 years and 106 days (07/12/81) after accident initiation and included cleanup of an equivalent of four reactor coolant system volumes of reactor coolant water.

The RCS is currently liquid-filled to the 327 ft elevation, which leaves the pressurizer and steam generator upper regions exposed to air. Prior to the liquid drawdown for defueling inadvertent injection of water with collodial suspensions of ferrous oxide, high pH and river water pollutants introduced additional contamination into the RCS and probably caused increased buildup of surface and loose deposits inside the RCS. During FY 1986, spillage of hydraulic-fluid into the reactor vessel provided sufficient nourishment to establish a microorganism community in the RCS, which increased water turbidity to eliminate observation of submerged objects and created concerns for microbiological influenced corrosion. Since April 1986, periodic treatment of the RCS water with hydrogen-peroxide to kill the microorganisms and water solution filtering and replacement has occurred to control the microorganism community.

4.2 Purpose

The purpose of the RCS sample acquisition and examination work is to retrieve and examine reactor coolant system adherent-surface and loose deposit samples. The examination objective are to determine the abundance, distribution, chemical form, and bonding characteristics of fission products and core materials deposited in the RCS and the extent to which the RCS can be decontaminated.

4.3 Accomplishments

4.3.1 Acquisition

Tooling. The RCS sample acquisition program has produced the following equipment:

Drawing/Report Number	Description/Title	Status
TBD	Germanium-crystal gamma spectrometer system, including computer softwear and point, pipe, and plane calibration sources (C. V. McIsaac, Three Mile Island Nuclear Station Unit-2 Operating Procedure for the EG&G Mobile Gamma Ray Spectrometer System--Draft)	Complete
TBD	Sodium-iodide-crystal portable gamma spectrometer system, including a Davidson Model 4106 Multi-channel Analyzer and excluding the crystal detector proper	Complete

Data. The data (gamma spectra) acquisition program has produced a cassette tape containing gamma spectra data from the following RCS regions:

Region	Spectra Quality
A-loop steam generator (external)	7 (NaI crystal)
Pressurizer (external)	6 (NaI crystal)
Core flood tank B	9 (CdTe crystal)
Miscellaneous	24

Samples. The RCS sample acquisition program has furnished the following fission product inventory samples to EG&G for examination:

TMI-2 Location	Sample Description	Date Acquired
A-Loop Hot Leg	Resistance temperature detector thermowell with 1.7 in. long by .4 in. diameter exposed surface area.	May 1984
Pressurizer Upper Head	Stainless-steel manway cover backing plate with 16 in. diameter exposed surface area.	February 1986

TMI-2 Location	Sample Description	Date Acquired
Steam Generator A Upper Head	Stainless-steel manway cover backing plate with 16 in. diameter exposed surface area.	March 1986
Steam Generator B Upper Head	Stainless-steel manway cover backing plate with 16 in. diameter exposed surface area.	March 1986
Steam Generator B Upper Tube Sheet Top	81 grams of mud-and-gravel-like sediment.	August 1986

CCTV Survey Recordings

The following videocassette recordings of RCS internal CCTV surveys have been acquired from GPUN:

Date	Object Surveyed (Tape Title)	Tape Number	Data Duration (min)
12/17/85	Pressurizer Heater Bundles Upper Bundle Grit	17	5
	Pressurizer Characterization--Entry 763 Tape 1	18	62
	Pressurizer Characterization--Entry 763 Tape 2	19	39
03/XX/86	TMI OTSG Examinations	73	10

4.3.2 Examination

The RCS examination program has produced the following reports:

Report Number	Title	Status
H. M. Burton ltr to G. R. Eidam Hmb-268-84	Transmittal of Draft Report Analysis of TMI-2 'A' Steam Generator Hot Leg Resistance Thermal Detector	Transmitted November 1984
EG&G Reactor Physics Branch ltr SCT-08-85	TMI Gamma Spectràl Data From Primary System Scanning Measurements	Completed September 1985

RCS examination activities performed by others have produced many other reports, which are listed in Appendix A.

4.3.3 Findings

The in situ measurements and sample examinations conducted to date indicate that the fractions of core materials and fission products deposited in the RCS are low, as follows:

Material/Fission Product	Estimated Abundance Fraction of Core Inventory
Uranium	Trace
Tritium	0.02
Kr-85	Negligible
Sr-90	0.01
Xe-133	Negligible
Ru-106	Negligible
Sb-125	0.001
I-129	0.012
I-131	0.11
Te-132	Negligible
Cs-134	0.008
Cs-137	0.008
Ce-144	0.0004
Plutonium	Negligible
Zirconium	Trace
Silver	Trace
Copper	Trace
Cadmium	None

Other examinations indicated that reactor coolant surfaces have an adherent surface deposit that will require removal by repeated application of decontamination solutions.

Video and borescope surveys, loose deposit sample collection and preliminary analysis indicated the following:

- Loose deposit (sediment) volumes:

Component	Description
Pressurizer	15 liters of slurry with some possible flakes of heater rod surface deposits.

Component	Description
A Steam generator:	
Tube sheet top	0.5 to 1.0 liters of solids with some greater than 1-in. long.
lower head	10 to 15 liters of slurry.
B Steam generator:	
Tube sheet top	1 to 4 liters of solids of both fine particles and small rocks up to 1/2-in. diameter.
lower head	15 to 30 liters of slurry.
A Cold Legs	20 to 30 liters of slurry.
B Cold Legs	40 to 60 liters of slurry.

- The A steam generator tube sheet top loose debris uranium composition was low (undetectable by gamma spectrometry).

- The pressurizer slurry contains from 6 to 11.2 kg of uranium by external gamma spectrometry.

Preliminary examination of the pressurizer and steam generator upper head manway cover backing plates indicated the following:

- Surface deposits range from a dull-back adherent layer in the pressurizer to a low luster tarnished surface in one steam generator and a low-luster tarnished surface with regions of adherent brownish crud in the other steam generator.

- The steam generator backing plate with the brownish crud deposits has the highest gamma activity; a factor of three greater than the other steam generator backing plate and a factor of 15 greater than the pressurizer packing plate.

4.4 Detailed Work Plan

The RCS sample acquisition and examination program work plan details are contained in the following work packages:

Work Package Number	Work Package Title
751421300	RCS Equipment/Building Characterization
755421000	RCS Fission Product Inventory Sample Examination

Table 16 summarizes the sample (RCS adherent surface and loose deposits) acquisition and examinations which are included in this work plan. The table includes the AEP-designated sample priority (1-20), the quantity of in situ measurements or samples, the TMI-2 accident information expected from the examination, and the examination techniques that will be used to obtain the information.

The product of the RCS sample acquisition and examination program work plan consists of samples of RCS surface and loose deposits and technical reports of sample examinations as follows:

Work Package Number	Product Item	Target Completion Date
741421300	A-loop steam generator:	
	lower head loose debris	February 1987
	B-loop steam generator:	
	tube sheet top loose debris	November 1986
	lower head loose debris	February 1987
	B-loop hot-leg RTD thermowell	TBD
755421000	RCS manway cover backing plate surface deposit	
	examination report: Draft	March 1987
	Final	August 1987
	B-loop steam generator tube sheet top loose	
	deposit examinations report: Draft	August 1987
	Final	December 1987

Additional reporting will be done by means of the test-and-inspection-services subcontractor's periodic progress reports and incorporation of progress-report examination data into the annual fission product inventory program updates to be prepared by the Examination Requirements and Systems Evaluation Group.

TABLE 16. RCS IN SITU MEASUREMENT AND SAMPLE ACQUISITION AND EXAMINATION PLAN SUMMARY

Measurement/Sample Description	Priority[a]	Sample Quantity	TMI-2 Accident Information	Examination Methods[b]
1. RCS Adherent Surface Deposits	12		All:	
			Color surface texture	1
a. A-loop steam generator cover backing plate		1	Total radioactivity and distribution	4
b. B-loop steam generator manway cover backing plate		1	Fission product abundance and distribution:	
			Mn-54, Co-60, Ru-106, Ag-110, Sb-125, Cs-134/137, Ce-144, Eu-154/155	5, 12
c. Pressurizer manway cover backing plate		1		
d. B-loop RTD thermowell		1	I-129	5, 10
			Sr-90	11
			Te	6
			Core material abundance and distribution:	
			Zr, Fe, Ni, Ag, In, Cd, Cr, Sn, Al, Mn, Si, Cu, Gd, Mg, Mo, Nb, B	6, 7, 8, 12
			U (includes U-235)	6, 8, 9, 12
			O	13
			Most abundant core material chemical form	15
			Decontaminatability	14
2. RCS Sediment:	12		All:	
			Volume/weight	2,16
a. Pressurizer lower head loose debris[c]		1	Particle size (transportability)	3
b. Steam generator tube sheet top loose debris		2	Color, surface texture, shape	1
c. Steam generator lower head loose debris		1	Total radioactivity	4
			Fission product abundance and distribution:	
			Mn-54, Co-60, Ru-106, Ag-110, Sb-125, Cs-134/137, Ce-144, Eu-154/155	5, 12
			I-129	5, 10
			Sr-90	11
			Te	6

TABLE 16. (continued)

Measurement/Sample Description	Priority[a]	Sample Quantity	TMI-2 Accident Information	Examination Methods[b]
			Core material abundance and distribution:	
			Zr, Fe, Ni, Ag, In, Cd, Cr, Sn, Al, Mn, Si, Cu, Gd, Mg, Mo, Nb	6, 7, 8, 12
			U (includes U-235)	6, 8, 9, 12
			O	13
			Most abundance core material chemical form	15

a. Priority values 1 through 20 are listed in Table 3.

b. Examination methods:

1. Photography
2. Balance weighing
3. Sieving
4. Ion-chamber gamma detection (including scans)
5. Germanium-crystal gamma spectrometry
6. Inductively-coupled-plasma emission spectrometry
7. Spark source mass spectrometry
8. Scanning electron microscopy with energy dispersive x-ray
9. Delayed neutron radiochemistry
10. I-129 radiochemistry
11. Sr-90 radiochemistry
12. Metallography
13. Metallography with auger spectrometry
14. Acid solution decontamination tests
15. X-ray diffraction
16. Immersion density.

c. Sample examination being performed by Westinghouse and GPUN.

4.5 Synopsis

RCS exploration during FY 1986 provided sufficient information to estimate the amount of core materials deposited in the vessels and piping. The RCS sample acquisition and examination plan is expected to satisfactorily characterize the abundance, distribution, and form of the radionuclides (fission products) and core materials deposited in the RCS and the extent to which the RCS can be decontaminated.

5. EX-RCS ACQUISITION AND EXAMINATION WORK PLAN

5.1 Introduction

The EX-RCS fission product inventory (FPI) sample acquisition and examination work plan includes the buildings and equipment outside the TMI-2 reactor coolant system that are believed to be either migration paths or destinations of core fission products or materials during and after the TMI-2 accident sequence. Figures 16 and 17 show the TMI-2 nuclear power plant site at Three Mile Island on the Susquehanna River in Middletown, Pennsylvania with its older-sister plant, TMI-1. The following site features are of special interest to the EX-RCS FPI SA&E planning:

1. Reactor Building (see Figure 18). The reactor building consists of a steel-plate-lines, reinforced concrete, cylindrical-shaped vessel designed to contain the consequences of a large-break loss-of-coolant accident including internal pressure of 60 psig at 286°F. The reactor building contains the reactor coolant system and other auxiliary equipment and extends from the 282-ft (above sea-level) elevation basement floor to the 473-ft elevation at the dome top. The site grade level is 304 ft, and the normal Susquehanna River level is 290 ft.

2. Auxiliary and Fuel Handling Buildings (AFHB). A plan view of the interconnected concrete-walled buildings is shown in Figure 19. The buildings are designed for radiation emission control because their functions include reactor coolant purification and degasification and spent fuel storage. The basement floor of both buildings is at the 280-ft elevation, with the auxiliary building penthouse roof at the 376-ft elevation and the fuel handling building roof top at the 400-ft elevation.

3. Vent Stack. The steep pipe vent stack also shown in Figure 18 extends from the 331-ft elevation to 463 ft, where gas/vapor effluent from the TMI buildings, including the reactor building and AFHB, can be released to the atmosphere.

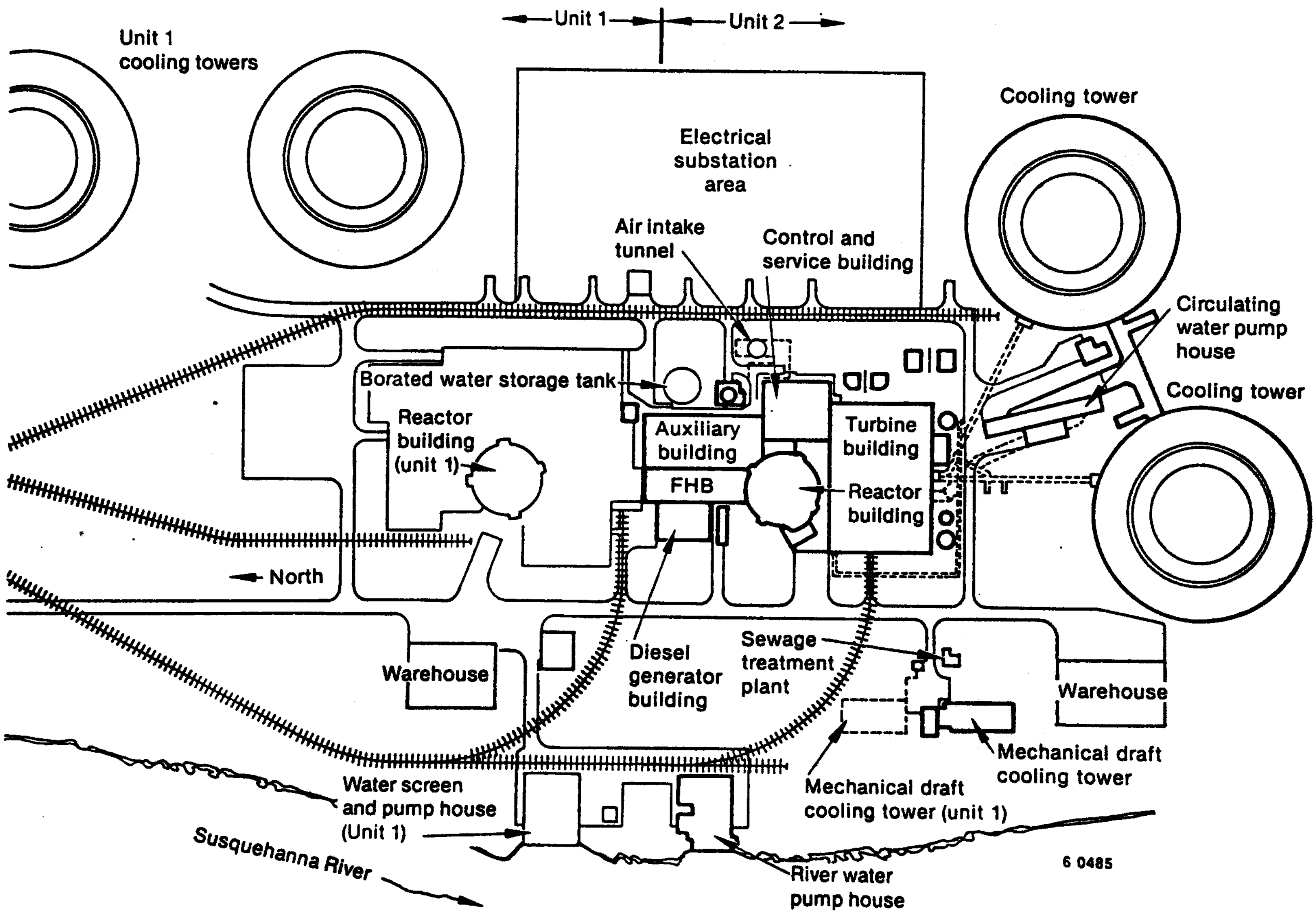


Figure 16. TMI-2 site plan.

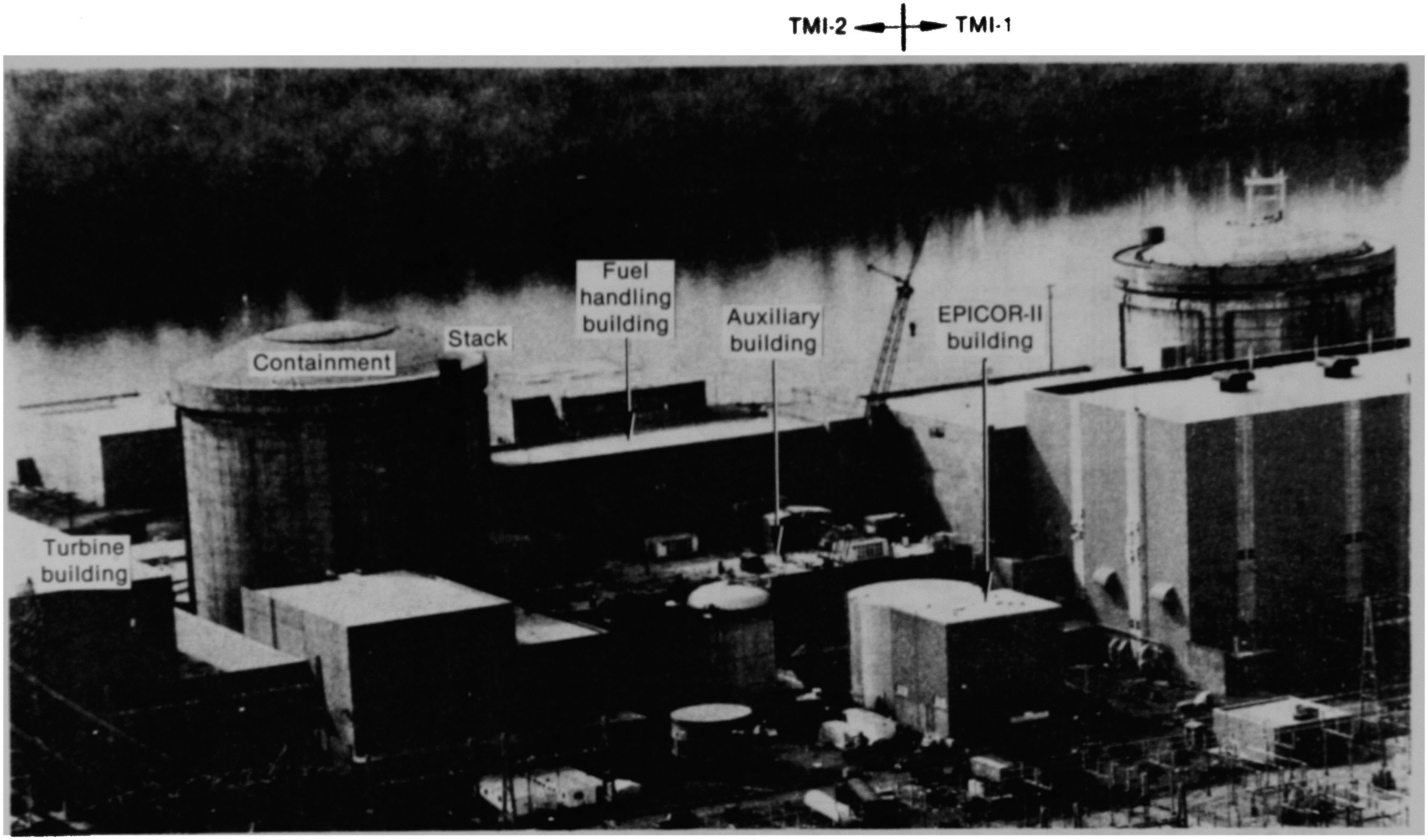


Figure 17. General building arrangement at TMI.

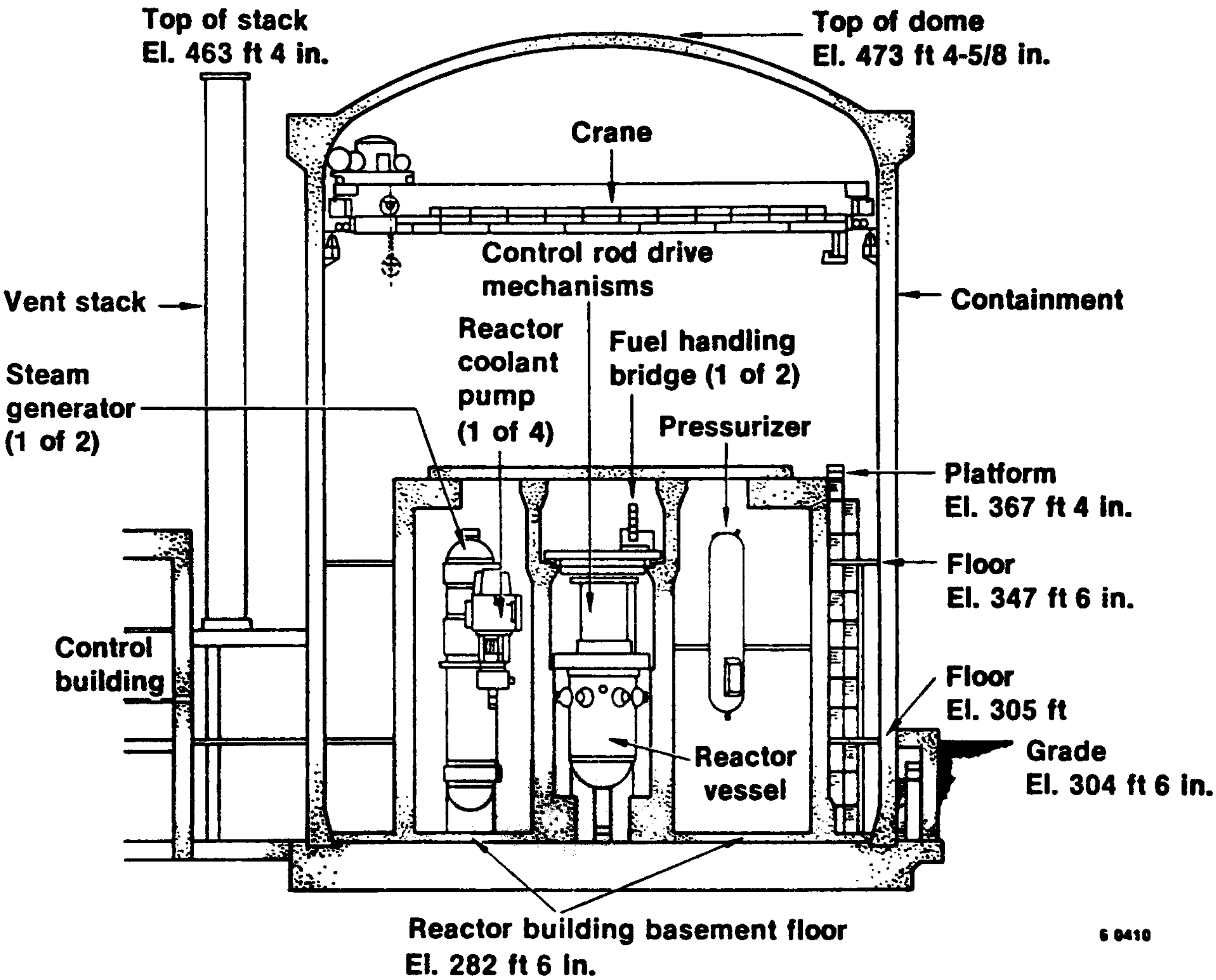


Figure 18. TMI-2 reactor building and major components of primary cooling system.

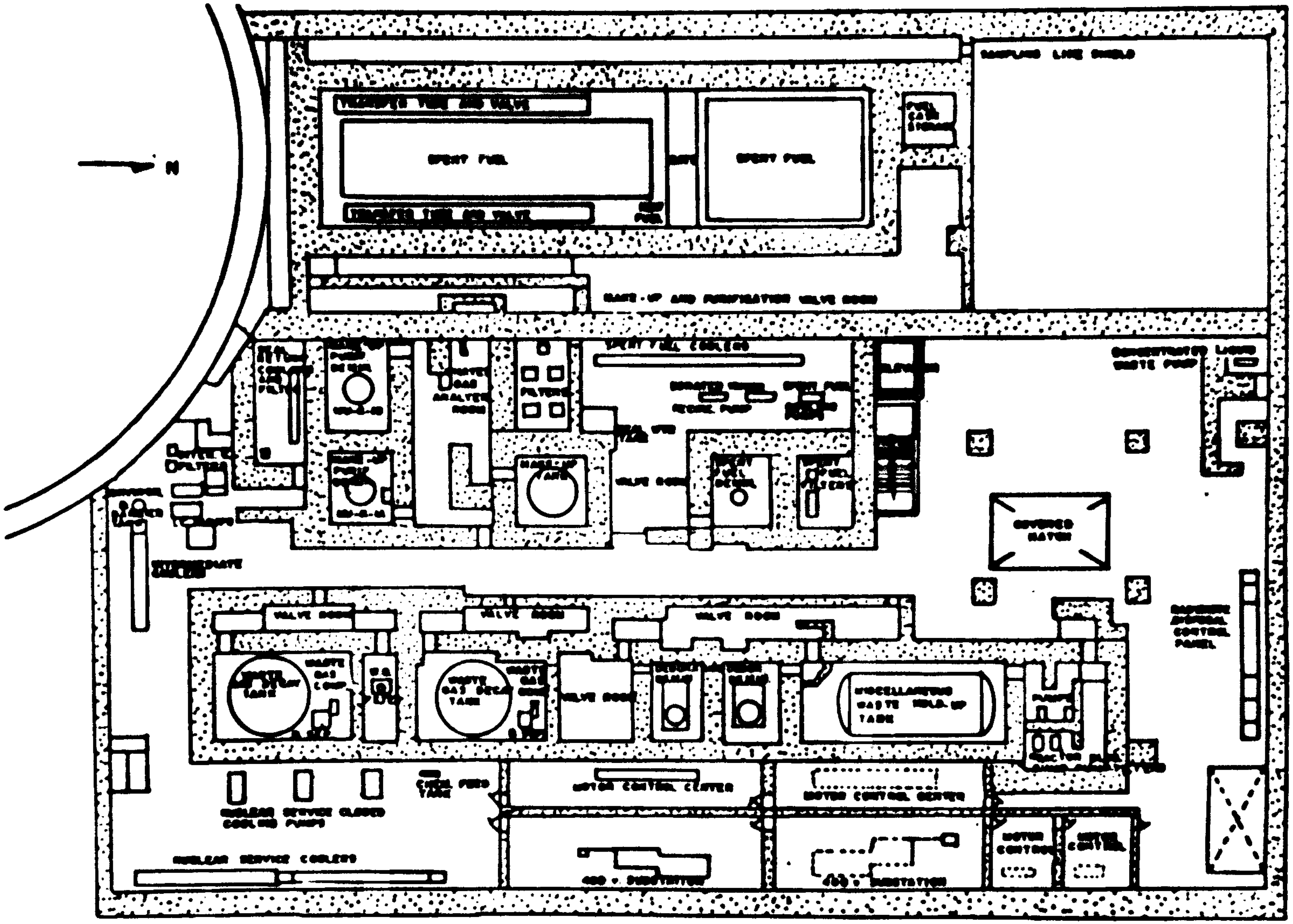


Figure 19. TMI-2 auxiliary and fuel handling buildings.

4. TMI-2 Control and Services Building. This building is connected to the AFHB by floor (liquid) drains and to the main steam system by sampling lines and extends from the 280-ft basement floor elevation to the 376-ft elevation roof top. Also, outside air is drawn in during circulation-mode ventilation of the control room.

5. TMI-1 Control and Services Building. This building is connected to the TMI-2 reactor coolant system through both reactor coolant and main steam system sampling lines. Also, outside air is drawn in during recirculation-mode ventilation of the control room.

6. Turbine Building. This building is connected to the reactor building and reactor coolant system by the main steam system and to both TMI-1 and TMI-2 control and services building by main steam system sampling lines.

7. TMI Industrial Waste Treatment System. This system filters and discharges waste water to the Susquehanna River.

TMI-2 accident studies have concluded that the fission product escape paths from the RCS during the accident sequence were as follows, in descending order of importance to the offsite radiation hazard:

1. Through the letdown system to the makeup and purification system radwaste disposal liquid system, radwaste disposal gas vent and relief systems, AFHB free volume and air exhaust system, and the vent stack to the atmosphere. Contaminated air could then be drawn into the control rooms through the HVAC and could contaminate the control room atmosphere.

2. Through the PORV/RCDT rupture disk route to the reactor building basement floor and free volume.

3. Through the PORV RCDT gas relief valve route to the radwaste disposal gas vent system, AFHB free volume and air exhaust system, and the vent stack to the atmosphere.

4. Through the RCS water sample line into the TMI-1 control and service building free volume and liquid drains and industrial waste treatment system to the Susquehanna River (believed to be very minor).

5. Through B-loop steam generator tube leaks to (a) the atmosphere via the main condenser, condenser vacuum system, the auxiliary building air exhaust discharge, and the vent stack, and (b) the Susquehanna River via the main steam system sampling lines, both control and service buildings drains, and the industrial waste treatment system (believed to be very minor).

The reactor vessel bottom and core instrument cable chase regions have not been sufficiently explored to determine whether or not an escape path from the RCS to the reactor building free volume developed through the core instrument train tubes beneath the reactor vessel. Fission products did not escape to the auxiliary building by reactor building sump pump action because the escape path was closed prior to fuel rod rupture.

After the accident sequence concluded, with commencement of core cooling by natural circulation (April 27, 1979) all fission product escape paths were controlled, including (a) the venting of reactor building radioactive gases through filter and the vent stack to the atmosphere and (b) the transport to offsite repositories of filters and ion exchange resin from the water treatment/cleaning system cleanup and decontamination of the TMI-2 liquid that became contaminated during the accident sequence. The water cleanup systems included the following:

1. The already-installed EPICOR-I system at TMI-1 for water with less than 1 μCi/mL contamination.

2. The EPICOR-II system, which was specially installed for TMI-2 accident cleanup of water with 1 to 100 μCi/mL contamination.

3. The SDS, which was specially installed in the TMI-2 AFHB spent fuel storage pool for TMI-2 accident cleanup of water with greater than 100 µCi/mL contamination.

During and after the TMI-2 accident sequence, which lasted until natural circulation cooling commenced (30 days after accident initiation), many events occurred that affected the character and distribution of fission products and core materials that escaped from the reactor coolant system. The most significant events include the following:

- Fission product and a small uranium fraction release commenced in the reactor vessel at approximately 138 min after accident initiation, when fuel rod rupture commenced. Reactor coolant circulation had ceased, and the available escape paths from the RCS were through: (a) the stuck-open PORV to the RCDT where liquid could escape to the reactor building basement floor through the rupture disk and vapor could escape through vent lines to the radwaste disposal vent gas system in the auxiliary building, and (b) the letdown line downstream of reactor coolant pump RCP-P-1A that led to either the makeup/purification or radwaste disposal systems in the auxiliary building.

- The PORV to RCDT escape path was closed 142 min after accident initiation.

- Zircaloy-steam reaction became significant at about 150 min, releasing hydrogen and other chemical reaction products into the RCS. Core material temperatures eventually reached or exceeded 3100 K, which could (a) generate aerosols from low volatility materials and chemical reactions, and (b) accelerate the escape of fission products from the uranium dioxide. Sufficient damage to the core instrument string calibration tubes probably occurred, allowing coolant to enter the calibration tubes, which extend to a "seal table" at the reactor building 347 ft elevation.

- A TMI-2 reactor coolant sample (140 µCi/mL gross activity) was taken (163 min) at the TMI-1 control and service building sampling station, introducing contaminated liquid into the liquid drains.

- Reactor coolant pump RC-P-2B was energized from 174 to 192 min after accident initiation, and this event is believed to have reflooded the overheated core region, fragmented most of the standing fuel in the upper core region, and caused circulation of core material particles and fission products throughout the RCS.

- The B-loop main steam isolation valves were opened for 7 s at 176 min, which allowed secondary coolant contaminated by primary coolant leakage through suspected B-loop steam generator tube cracks to migrate to the condenser.

- The PORV to RCDT escape path was reopened from 192 to 197 min and 220 to 318 min.

- A significant relocation of core material from the core region to the flooded reactor vessel lower region occurred at 227 min, which likely increased the escape of core material and fission products to the letdown system.

- At 234 min plus, a B-loop steam generator secondary side water sample was drawn at the TMI-2 control and services building sampling station, introducing contaminated liquid to the building sump, from where it later migrated to the Susquehanna River through the industrial waste treatment system.

- The radioactive gas escape path to the radwaste disposal gas vent system through the RCDT vent was closed at 236 min during reactor building isolation.

- Over pressure in the reactor coolant makeup tank lifted the 80-psi-set-point liquid relief valve at 266 min and discharged contaminated RCS liquid to the reactor coolant bleed holdup tanks (RCBHTs), which also overflowed and overpressured. The RCBHT over pressure lifted the 20-psi-set-point relief valves and allowed unfiltered vapor to escape to the atmosphere, via the radwaste disposal gas relief header and the vent stack. It is also believed that liquid entered the radwaste disposal gas vent header, where it would be separated and drained to the auxiliary building sump.

- A sustained high pressure injection period commenced at 267 min and continued to 544 min.

- A TMI-2 reactor coolant sample (>500 μCi/mL gross activity) was taken at 283 min from the TMI-1 sampling station, introducing contaminated liquid into the liquid drains.

- The PORV to RCDT escape path was reopened repeatedly from 340 to 458 min to prevent RCS over pressurization and opened from 458 to 550, 565 to 589, 600 to 668, 756 to 767, and 772 to 780 min to depressurize the RCS for core flood injection.

- TMI-2 control room air became contaminated (both particulate and noble gas channel alarms) at 370 min, requiring the use of personnel face masks and particulate filters until 670 min.

- A hydrogen burn occurred in the reactor building at 590 min causing a 28 psig peak pressure and actuating the reactor building spray, which injected chemically-treated (boron and sodium hydroxide) water into the reactor building for 6 min. A coincident interruption of power to the Auxiliary Building radiation monitor strip charts HP-UR-1901 and HP-UR-1902 is suspected. The HP-UR-1901 strip chart plotted the output of the RB Purge Unit area radiation monitor HP-R-3236, which was used to estimate the TMI-2 accident offsite release 4-28. The power interruption duration is estimated to be 2 h.

- Forced circulation cooling of the reactor was resumed at 949 min (15 h 49 min) through the A-loop with reactor coolant pump RC-P-1A.

- Letdown flow was lost from 18 h 34 min to 26 h 30 min.

- Overpressure in the letdown system lifted the 130-psi-set-point relief valve MU-R-3 around midnight (20 h and 30 min), allowing reactor coolant escape to the RCBHT. The RCBHT relief valves are believed to have also lifted, allowing unfiltered vapor to escape to the atmosphere, and probably allowing liquid to enter the auxiliary building sump through the radwaste disposal gas vent header. This condition lasted longer than 40 min.

- TMI-2 control room air became contaminated (particulate channel alarm) at 22 h 11 min, requiring use of personnel face masks and particulate filters for 64 min.

- An escape path was created at 24 h 35 min by opening the makeup tank vent valve MU-V-13 to the radwaste disposal gas vent header. This pathway was reopened periodically for the next several days.

- A helicopter measured 3 R/h beta gamma and 410 mR/h gamma at 15 ft above the TMI-2 vent stack at 34 h 10 min after accident initiation.

- A 100 mL TMI-2 reactor coolant sample was taken (36 h 15 min) at the TMI-1 control and services building sampling station, introducing contaminated liquid into the liquid drains. The sample radiation emission was >100 R/h at contact.

- Natural circulation cooling of the reactor commenced 30 days and 10 h (April 27, 1979) after accident initiation.

- Auxiliary building decontamination commenced 30 days (April 27, 1979) after accident initiation.

- Supplemental filters for auxiliary building venting commenced operation on May 1, 1979.

- The vent stack was capped on May 20, 1979.

- EPICOR-II cleanup of medium contamination water commenced October 1979.

- Reactor building gas cleanup and venting commenced July 28, 1980 and included reopening of the vent stack.

- SDS/EPICOR-II cleanup of the high-contamination water commenced July 12, 1981, and included cleanup of an equivalent of four reactor coolant system volumes of reactor coolant water. Reactor building basement water cleanup was completed in May 1982.

- Reactor building decontamination commenced in March 1982.

An estimated 643,000 gallons of contaminated water collected in the reactor building basement between accident initiation and September 1981, when SDS cleanup of the water commenced. The steadily increasing depth of waste in the basement at key accident-sequence events was as follows:

Time After Accident Initiation	Event	Basement Water Depth[a]
227 min	Major core material relocation to reactor vessel lower plenum region	10 in.
15 h 40 min	Commence sustained forced-circulation cooling of core	2 ft 8 in.
30 days 10 h	Commence natural circulation cooling of core	4 ft 3 in.
910 days (09/23/81)	Commence SDS cleanup of RB Basement	8 ft 6 in.

The basement water is believed to have composed of the following sources on 09/23/81 (see Reference 1):

Water Source		Percent
Reactor Coolant System:	First 72 h of accident	41
	Next 907 days	28
Reactor Building Spray System		3
Susquehanna River		28

The spray system water contained boron and sodium hydroxide chemicals, and the river water (from leaks in the river water cooling system) silt was composed of the following major elements in order of concentration: Fe, Si, Mn, Pb, Ca, K, S, Al, Ba, Na, and Ti.

The event sequence shows a chronological separation of the core damage events and the offsite radiation release. The core damage probably ended about 4 h and 30 min after accident initiation, when the high pressure injection refill of the RCS commenced. The probable initiation of the offsite radiation hazard coincident with the measurement of TMI-2 control room air contamination was 6 h and 10 min after accident initiation. The

a. Assumes linear relationship of gallons of water to water depth and 643,000 gallons equals 8 ft 6 in. water depth.

control room air is believed to have been contaminated by the outside air. The offsite radiation release continued for several days until the makeup tank venting through valve MU-V-13 was no longer necessary.

The measurements of the offsite radiation source characteristics showed that noble gases were the dominating contributor to the offsite source-term and that cesium and iodine contribution was negligible. This observation indicates that effectively all of the nongaseous fission products (cesium, iodine, strontium, etc.) inventory was retained by the TMI-2 buildings and equipment during the TMI-2 accident sequence.

The TMI-2 EX-RCS buildings and equipment are still being decontaminated. The decontamination process commenced April 27, 1979 30 days after accident initiation. All fluid systems have been flushed, fluid and gas filters removed, fluid treatment resin beds removed or decontaminated, and TMI-2 accident liquid effluent decontamination. The decontamination has not yet reduced radiation to personnel-entry levels in the following areas:

1. The reactor building basement, which includes the letdown coolers, the RCDT, sediment containing fission products and core materials, and concrete, which has absorbed fission-product contaminated liquid.

2. The reactor building D-ring compartment, which contains the RCS B-loop.

3. The fuel handling building makeup and purification valve room, which contains the letdown system block orifice and piping.

The above conditions create a condition where (a) samples that are representative of, or traceable to, conditions which existed during the accident are no longer numerous and (b) sample acquisition from contaminated personnel exclusion areas is limited to what can be obtained with remote-operated hand tools, and robots.

5.2 Purpose

The purpose of the EX-RCS sample acquisition and examination program is the retrieval and examination of reactor building basement sediment and absorber (concrete) samples. The examination objectives are to complete the EX-RCS search program for fission-products and core materials which escaped from the RCS during and following the TMI-2 accident. The specific examination objectives are to determine the following:

- Abundance and distribution of fission products and core materials in EX-RCS buildings and equipment that are judged to be inadequately surveyed.

- Current condition of the fission products and core materials that are found.

5.3 Accomplishments

5.3.1 Introduction

The EX-RCS search program for the escaped radionuclides (fission products) and core materials has been a continuous effort since and including the day (March 28, 1979) of the accident. The expansion of the TMI-2 Core Examination Plan to a TMI-2 Accident Evaluation Program has resulted in resumption of an EX-RCS sample acquisition and examination work (search program) plan. The approach to developing a productive search program was to evaluate the current completeness of the search program by locating buildings and equipment, which had not yet or only partially been inventoried for fission products. The evaluation developed the following:

1. A preliminary map (Figure 20) showing schematically the equipment, buildings, and areas where fission products may be present.

2. A preliminary matrix chart (Table 17) showing the extent of the already completed TMI-2 accident fission product search program.

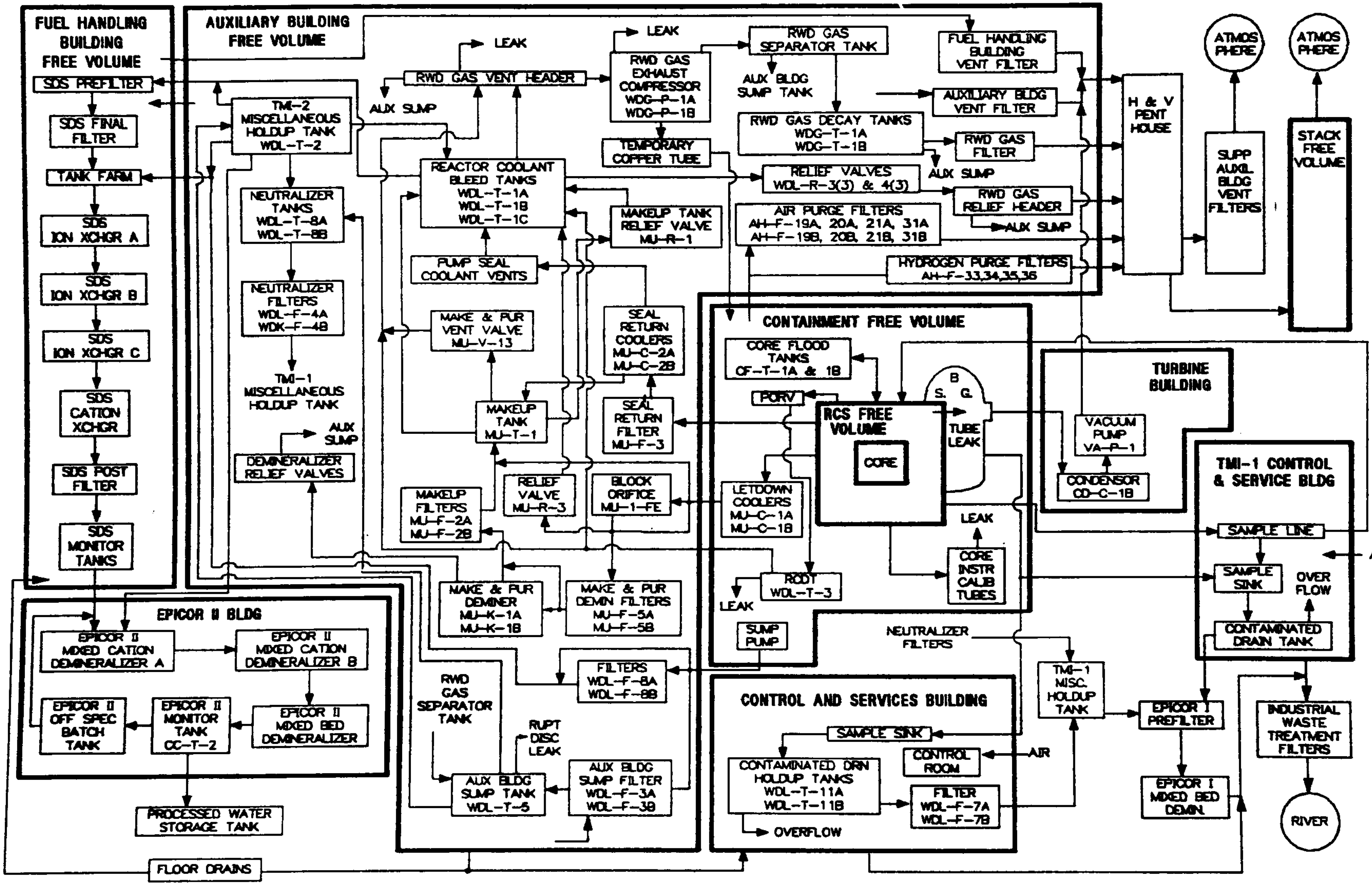


Figure 20. TMI-2 radioactive material location map.

TABLE 17. MATRIX TABLE OF COMPLETED FISSION PRODUCT INVENTORIES[b,c]

Location	Area Radiation Emission Mapping		Radiochemical Composition Examinations					Chemical Composition Examinations					
	γ and β	γ spectra	Liquid	Gas	Solids and Sediment	Surface Deposit	Absorber	Liquid	Gas	Solids and Sediment	Surface Deposit	Absorber	Miscellaneous Information
Core and RV Lower Plenum	X	--	X	NA	X (.2)	--	NA	X	NA	X (.2)	--	NA	--
RV Upper Plenum	X	--	X	NA	X (.5)	X (.5)	NA	X	NA	X (.5)	X (.5)	NA	a
Steam Generators	X	X	X	NA	--	--	NA	X	NA	--	--	NA	a
Pressurizer	X	X	X	NA	--	--	NA	X	NA	--	--	NA	a
RCS Piping, Pumps and Valves (PP&V)	X	--	X	NA	--	X (Partial)	NA	--	NA	--	X (Partial)	NA	a
Core Flood Tanks	X	--	--	--	--	--	NA	--	NA	--	--	NA	a
Core Flood Piping	X	X	--	NA	--	--	NA	--	NA	--	--	NA	--
M&P Letdown Coolers	--	--	--	NA	--	--	NA	--	NA	--	--	NA	(One may be blocked)[a]
RCS Drain Tank	--	--	X	NA	X	--	NA	--	NA	X (Partial)	--	NA	a
Containment Building Free Volume:													
1. 347 ft to Dome	X	X	NA	X	NA	X	X (Partial)	NA	X	NA	X	--	a
2. 305 ft to 347 ft	X	X	NA	X	NA	X	X (Partial)	NA	X	NA	--	--	(Partial)[a]
3. Bsmnt. ft to 305 ft	--	--	X	X	X	X	--	X (Partial)	X (Part L)	--	--	--	10 in. water depth at 227 min
Core Instrument Tubes	X (Seal Table)	--	--	--	--	--	--	--	--	--	--	--	--
M&P Block Orifice	X	X	--	NA	--	--	NA	--	NA	--	--	NA	a
M&P Demineralizer Filters (5A and 5B)	1 of 2	Post Filter	--	NA	--	1 of 2	NA	--	NA	--	--	NA	a
M&P Demineralizers	X (Partial)	--	X (Partial)	NA	X	--	NA	X (Partial)	NA	X	--	NA	a
Makeup Filters (2A and 2B)	1 of 2	Post Flush	--	NA	X	--	NA	--	NA	X	--	NA	a
Makeup Tanks	Post Filter	Post Flush	--	--	--	--	NA	--	--	--	--	NA	a

TABLE 17. (continued)

Location	Area Radiation Emission Mapping		Radiochemical Composition Examinations					Chemical Composition Examinations					Miscellaneous Information
	γ and β	γ spectra	Liquid	Gas	Solids and Sediment	Surface Deposit	Absorber	Liquid	Gas	Solids and Sediment	Surface Deposit	Absorber	
MUPS PP&V	Post Flush	Post Flush	X	NA	--	--	NA	X	NA	--	--	NA	a
Pump Seal Return Filter (F-3)	X	X	--	NA	--	--	NA	--	NA	--	--	NA	a
Pump Seal Return Coolers	X	X	--	NA	--	--	NA	--	NA	--	--	NA	a
Pump Seal Injection Filters (F-4a and 4b)	X	--	--	NA	X	--	NA	--	NA	X	--	NA	a
Pump Seal PP&V	--	--	--	NA	--	--	NA	--	NA	--	--	NA	--
React Building Sump Filters(8A and 8B)	--	--	--	--	--	--	--	--	--	--	--	--	Prerod burst contamination
RCS Liquid Waste PP&V:													
1. Reactor Building	--	--	--	--	--	--	NA	--	--	--	--	--	--
2. Auxiliary Building	--	--	--	--	--	--	NA	--	--	--	--	--	--
RCS Bleed Holdup Tanks													
1. WDL-T-1A	X	X	X 12/79	--	X	--	NA	X (Partial)	--	X	--	NA	8/20/81, start flushing[a]
2. WDL-T-1B	X	--	X 1/80	--	X	--	NA	X (Partial)	--	X	--	NA	a
3. WDL-T-1C	X	X	X 2/80	--	X	--	NA	X (Partial)	--	X	--	NA	a
Auxiliary Building Sump Filters (3A and 3B)	--	--	--	--	--	--	--	--	--	--	--	--	Post core damage contamination
Auxiliary Building Sump Tank	X	--	X 2/80	--	--	--	NA	X 2/80	--	--	--	NA	Post core damage contamination[a]
Miscellaneous Waste Holdup Tanks	X	--	X	--	--	--	NA	X	--	--	--	NA	Post core damage contamination[a]
Neutralizer Tanks	--	--	--	--	--	--	--	--	--	--	--	--	Post core damage contamination[a]
Neutralizer Filter (4A and 4B)	9-25-	--	X	--	--	--	--	X	--	--	--	--	Post core damage contamination

TABLE 17. (continued)

Location	Area Radiation Emission Mapping		Radiochemical Composition Examinations					Chemical Composition Examinations					Miscellaneous Information
	γ and β	γ spectra	Liquid	Gas	Solids and Sediment	Surface Deposit	Absorber	Liquid	Gas	Solids and Sediment	Surface Deposit	Absorber	
Auxiliary Building Radwaste Disposal Systems P&PV	--	--	X	--	--	--	--	--	--	--	--	--	--
M&P Relief Valve Header	--	--	--	--	--	--	--	--	--	--	--	--	--
Reactor Coolant Process Gas Decay Tanks	--	--	--	X	--	--	NA	--	--	--	--	NA	--
Reactor Coolant Process Gas Exhaust Compressor	--	--	NA	--	--	--	NA	--	--	--	--	NA	--
Reactor Coolant Process Gas Exhaust Filter	--	--	NA	NA	--	--	?	NA	NA	--	--	NA	--
Reactor Coolant Process Gas Ducting and Valves	--	--	NA	X	--	--	NA	NA	--	--	--	NA	--
Auxiliary Building Ventilation Filter	--	--	NA	X	--	--	?	NA	--	--	--	NA	--
Fuel Handling Building Ventilation Filter	--	--	NA	--	--	--	?	NA	--	--	--	NA	--
Auxiliary Building Ventilation Ducting, Valves and Compressor	--	--	NA	X	--	--	NA	NA	--	--	--	NA	--
Reactor Building Air Purge Filters	--	--	NA	--	--	--	NA	NA	--	--	--	NA	--
Reactor Building Air Purge Ducting, Valves, and Compressor	--	--	NA	--	--	--	?	NA	--	--	--	NA	--
RB Air Purge Duct, Valves and Compressor	--	--	NA	--	--	--	NA	NA	--	--	--	NA	--
Auxiliary Building Free Volume													
1. 328 ft to roof	--	--	--	X	--	--	--	--	--	--	--	--	--
2. 305 ft to 328 ft	--	--	--	X	--	--	--	--	--	--	--	--	--
3. 281 ft to 305 ft	--	--	X	X	--	--	--	X	--	--	--	--	--

TABLE 17. (continued)

Location	Area Radiation Emission Mapping		Radiochemical Composition Examinations					Chemical Composition Examinations					Miscellaneous Information
	γ and β	γ spectra	Liquid	Gas	Solids and Sediment	Surface Deposit	Absorber	Liquid	Gas	Solids and Sediment	Surface Deposit	Absorber	
Fuel Handlg Building Free Volume	--	--	X	--	--	--	--	X	--	--	--	--	--
Process Gas Separator Tank	--	--	--	--	--	--	NA	--	--	--	--	NA	? NUREG-0600, p. II-3-11
Vent Stack Free Volume	--	--	NA	X	--	--	--	NA	--	--	--	--	Capped 5/20/79 through 6/28/80
Auxiliary Building Supplemental Filters	--	--	NA	--	--	--	?	NA	--	--	--	--	5/1/79 through 6/28/80
Contaminated Drain Tank	X	--	X	NA	--	--	NA	X	NA	--	--	NA	a
Contaminated Drain Tank Filters	--	--	--	--	--	--	--	--	--	--	--	--	--
Control and Service Buildings Radwaste Disposal System PP&V	--	--	--	--	--	--	--	--	--	--	--	--	--
Reactor Coolant Sample Line and Valves	--	--	X	--	--	--	--	--	--	--	--	--	Until 6/17/80
TMI-1 Contaminated Drain Tank	--	--	--	--	--	--	--	--	--	--	--	--	--
TMI-1 Control and Services Building Free Volume	--	--	--	--	--	--	--	--	--	--	--	--	--
TMI-1 Miscellaneous Holdup Tanks	--	--	--	--	--	--	--	--	--	--	--	--	--
Industrial Waste Treatment Filters	--	--	--	--	--	--	--	--	--	--	--	--	--
Industrial Waste Treatment PP&V	--	--	X	--	--	--	--	--	--	--	--	--	--
EPICORE I Prefilter	--	--	--	--	--	--	--	--	--	--	--	--	
EPICORE I Demineralizers	--	--	--	--	--	--	--	--	--	--	--	--	
EPICORE I PP&V	--	--	--	--	--	--	--	--	--	--	--	--	
EPICOR II Demineralizer A	--	--	--	--	--	--	--	--	--	--	--	--	

TABLE 17. (continued)

Location	Area Radiation Emission Mapping: γ and β	Area Radiation Emission Mapping: γ spectra	Radiochemical Composition Examinations: Liquid	Radiochemical: Gas	Radiochemical: Solids and Sediment	Radiochemical: Surface Deposit	Radiochemical: Absorber	Chemical Composition Examinations: Liquid	Chemical: Gas	Chemical: Solids and Sediment	Chemical: Surface Deposit	Chemical: Absorber	Miscellaneous Information
EPICOR II Demineralizer B	--	--	--	--	--	--	--	--	--	--	--	--	--
EPICOR II Mixed Bed Demineralizer	--	--	--	--	--	--	--	--	--	--	--	--	--
EPICOR II Monitor Tank	--	--	X ?	--	--	--	--	--	--	--	--	--	--
EPICOR II Off-Spec Batch Tank	--	--	--	--	--	--	--	--	--	--	--	--	--
EPICOR II Piping, Pumps and Valves	--	--	--	--	--	--	--	--	--	--	--	--	--
SDS Prefilter	--	--	--	--	X	--	--	--	--	X (TRU)	--	--	--
SDS Final Filter	--	--	--	--	X	--	--	--	--	X (TRU)	--	--	--
SDS Tank Farm	--	--	--	--	--	--	--	--	--	--	--	--	--
SDS Ion Exchanger A	--	--	--	--	X	--	--	--	--	X (TRU)	--	--	--
SDS Ion Exchanger B	--	--	--	--	X	--	--	--	--	X (TRU)	--	--	--
SDS Ion Exchanger C	--	--	--	--	X	--	--	--	--	X (TRU)	--	--	--
SDS Post Filters	--	--	--	--	--	--	--	--	--	--	--	--	--
SDS Monitor Tanks	--	--	X ?	--	--	--	--	--	--	--	--	--	--
SDS Piping, Pumps and Valves	--	--	--	--	--	--	--	--	--	--	--	--	--
Atmosphere	X	X	NA	X	NA	--	--	NA	--	NA	--	--	--
Susquehanna River	--	--	X ?	--	--	--	--	--	--	--	--		--

a. Science Applicat[illegible]nc. (SAI) history.

b. X indicates recor[illegible]sts of in situ measurement or sample examination.

c. X (.X) indicates [illegible]ion of equipment, building or area inventoried.

3. Knowledge that many other organizations have participated in the planning and performance of the EX-RCS fission product inventory program and that most building areas and equipment have been decontaminated so that samples that are representative of or traceable to conditions that existed during the accident are no longer numerous.

5.3.2 Acquisition

Tooling. The EX-RCS sample acquisition program has developed and provided the following sample acquisition tooling:

Drawing/Report Number	Description/Title	Status
TBD	Electrically-Operated, Vacuum-Actuated, Remote-Operated Liquid/Sediment Sampler	Complete

The gamma spectrometer equipment listed in Subsection 4.3.1 will also be used in the EX-RCS fission product inventory program.

Samples. The EX-RCS sample acquisition program has furnished the following fission product inventory samples to EG&G for examination:

TMI-2 Location	Sample Type	Quantity	Date Acquired
Reactor coolant bleed tank A	Liquid (filtered)	125 mL	December 1979
Reactor coolant bleed tank B	Liquid (filtered)	150 mL	January 1980
Reactor coolant bleed tank C	Liquid (filtered)	150 mL	February 1980
Reactor coolant bleed tank A	Solids (sediment)	60 g	August 1981

TMI-2 Location	Sample Type	Quantity	Date Acquired
Makeup and purification demineralizer prefilter (MU-F-5B)	Solid debris filter w/some filter paper remaining	2 g 204 g (filter paper, liquid, and collected solids)	February 1981 1982
	Vacuum collected debris	Small	March 1982
Makeup and purification demineralizer prefilter (MU-F-5A)	Vacuum collected debris	Small	March 1982
Makeup and purification demineralizer after filter (MU-F-2A)	Filter	406 g (filter paper, liquid, and collected solids)	March 1982
	Vacuum collected debris	436 g (filter paper, liquid, and collected solids)	March 1982
Makeup and purification demineralizer after filter (MU-F-2B)	Filter	206 g (filter paper, liquid, and collected solids)	March 1982
	Vacuum collected debris	Small	March 1982
Makeup and purification demineralizer A (MU-K-1A)	Solid (resin)	10 g	April 1983
Makeup and purification demineralizer B (MU-K-1B)	Slurry (liquid and resin)	12 samples (80 mL total w/40 mL solids)	April 1983
Pump seawater injection filter (MU-F-4A)	Filter	83 g (filter paper, liquid, and collected solids)	March 1982
	Vacuum collected debris	80 g	March 1982

TMI-2 Location	Sample Type	Quantity	Date Acquired
Pump seawater injection filter (MU-F-4A)	Filter	Not measured	March 1982
	Vacuum collected debris	Small	March 1982
Reactor building basement:			
305 ft floor elevation under south equipment hatch (entry 10)	Liquid and sediment	110 mL with 108 g filtered solids	May 1981
305 ft elevation in the open stairwell	Liquid and sediment	120 mL with 25 g filtered solids	September 1981
Bottom of open stairwell	Slurry	45 mL with 1 g solids	June 1982
Basement sump pit	Liquid and sediment	200 mL with 72 g filtered solids	August 1983
Reactor coolant drain tank (WDL-T-3)	Liquid and sediment	120 mL with 0.5 mg filtered solids	December 1983
Reactor building air coolers	Access panels, 30 x 40 in.	5	August 1983
Reactor building basement:			
Impingement area	Slurry (liquid and sediment)	75 mL with 8 g solids	September 1985
	Slurry (liquid and sediment)	160 mL with 36 g solids	September 1985
3000 psi uncoated support wall--8.3 ft above floor	Concrete core bore A1440	1 in. diameter by 1/2 in. long	November 1985
5000 psi coated D-ring wall--2.8 ft above floor	Concrete core bore A1439	1 in. diameter by 7/8 in. long	November 1985
Elevator uncoated block wall--8.2 ft above floor	Concrete core bore SUB-2	1 in. diameter by 1-1/2 in. long	February 1986

Table 17 identifies the locations of many other in situ measurements and sample acquisitions and examinations which have been accomplished since 4:00 a.m. on March 28, 1979 to locate and characterize the fission products that escaped from the RCS during the accident.

5.3.3 Examination

The EG&G-controlled fission product inventory support program has produced the following reports:

Report Number	Title	Status
GEND-INF-011	First Results of the TMI-2 Sump Samples Analyses Entry 10	Complete July 1981
GEND-INF-011 Volume II	Reactor Building Basement Radionuclide Distribution Studies	Complete October 1982
GEND-INF-011 Volume III	Reactor Building Basement Radionuclide and Source Distribution Studies	Complete October 1982
GEND-INF-039	Final Analysis on TMI-2 Reactor Coolant System and Reactor Coolant Bleed Tank Samples	Issued June 1983
GEND-042	TMI-2 Reactor Building Source Term Measurements: Surface and Basement Water and Sediment	Complete October 1984
EGG-TMI-6181	Interim Report on the TMI-2 Purification Filter Examination	Complete February 1983
EGG-TMI-6580	TMI Particle Characterization Determined from Filter Examinations	Draft Complete September 1984
GEND-INF-041	Radionuclide Mass Balance for the TMI Accident: Data Through 1979 and Preliminary Assessment of Uncertainties	Complete November 1981
GEND-INF-054	Results of Analyses Performed on Concrete Cores Removed from Floors and D-Ring Walls of the TMI-2 Reactor Building	Issued June 1984

Report Number	Title	Status
H. M. Burton (EG&G) letter to B. K. Kanga (GPU) Hmb-207-83	Purification demineralizer Resin Samples	Issued June 22, 1983
K. L. Wright (SAI) letter to E. R. Eidam (GPUN)	Radioanalytical Report (reactor building basement sediment sample examination)	Issued August 11, 1986

Reports by others which describe and/or evaluate the EX-RCS fission product inventory investigation program are listed in Appendix A.

It appears that the greatest offsite radiation release occurred during the following periods:

- 20 to 92 h after accident initiation, due to probable noble gas dominated fission product escape from the vent stack via the letdown and radwaste disposal gas vent and relief systems.

- 6 to 11 h after accident initiation, due to probable noble gas dominated fission-product escape from the vent stack via the letdown and/or radwaste disposal gas vent and relief systems.

Other findings include the following:

1. The reactor building sump to auxiliary building liquid escape path was closed prior to fission product escape from the fuel rods.

2. Most TMI-2 EX-RCS buildings and equipment have been completely or partially decontaminated by flushing, water treatment, contaminated filter removal, and water treatment resin removal.

Examination and testing of reactor building basement sediment and concrete bore samples and thermoluminscent detector mapping of basement radiation indicates the following:

- The sediment on the basement floor is not the principle source of radiation in the basement and contains only small quantities of core materials.

- The basement concrete is the principle source of radiation in the basement with radioactive contamination (cesium and strontium) penetrating throughout the porous concrete block and about 1/4-in. deep (90% of radioactivity) in the high density concrete walls and floor with less contamination in coated (painted) areas.

- The radioactive contamination of the basement walls is concentrated near the water level (5.5 to 8.5 above the floor in the flooded basement after the accident.

- It may be possible to remove most of the radioactivity from the concrete by leaching in a borated water solution.

5.4 Detailed Work Plan

The EX-RCS sample acquisition and examination program work plan details are contained in the following work packages:

Work Package Number	Work Package Title
741421300	RCS Equipment/Building Characterization
755420300	EX-RCS Fission Product Inventory Sample Examination

Table 18 summarizes the sample (reactor building basement sediment and concrete bores) acquisition and examinations, which are included in this work plan. The table includes the AEP-designated sample priority (1-20), the number of samples, the TMI-2 accident information expected from the examination plan, and the examination techniques which will be used to obtain the information.

Other EX-RCS fission product sample examinations that were considered include the following:

TABLE 18. EX-RCS SAMPLE ACQUISITION AND EXAMINATION PLAN SUMMARY

Measurement/Sample Description	AEP Priority[a]	Number of Samples	TMI-2 Accident Information	Examination Methods[b]
1. RB basement concrete absorption:	11	19	Surface condition (color, texture)	1
5000-psi (D-ring) wall bores		4		
3000-psi (shield) wall bores		7	Fission product abundance and distribution in concrete:	
Block (elevator/stairwell) bores		6	Depth of fission product penetration	11, 12, 5
Floor bores		3	Fission product abundance and distribution:	
			Mn-54, Co-60, Ru-106, Ag-110, Sb-125, Cs-134/137, Ce-144, Eu-154/155	6
			I-129	6, 7
			Sr-90	8
			Uranium abundance and distribution	10

a. Priority values 1 through 20 are listed in Table 3.

b. Examination Methods: (1) Photography, (2) Balance weighing, (3) Dimensional (length and diameter) measurements, (4) Sieving, (5) Mill or grind concrete bore into eight power samples and dissolve, (6) Germanium-crystal gamma spectrometry, (7) I-129 radiochemistry, (8) Sr-90 radiochemistry, (9) Inductively-coupled-plasma emission spectrometry, (10) Delayed neutron radiochemistry, (11) Ion-chamber gamma detector (including scans), (12) Autoradiograph maps (photographic paper in contact with outside surface).

Sample Description	AEP Priority	Sample Quantity
1. Reactor building basement sediment from the elevator and sump well floor depressions	10	2 one kg samples
2. Reactor building basement wall liner adherent surface deposit	Low	2
3. Equipment internal deposits:	Low	
a. Reactor coolant drain tank		
• Sediment (only 9 mg was collected and examined		1
• Adherent surface deposit		1
b. Letdown coolers:		
• Sediment		2
• Adherent surface deposits		2
c. Letdown block orifice:		Entire orifice
• Sediment		
• Adherent surface deposits		

The impact of not examining these samples is judged to be minimal for the following reasons:

1. The other 12 reactor building basement floor sediment samples will provide sufficient data to assess the abundance of fission products and core materials in the basement sediment.

2. The basement floor sump well was already sampled by collecting a liquid/suspended-solids sample during sump-pump-recirculation agitation of the sump contents, and only small quantities of fission-products and core materials were found in the samples.

3. A prior reactor coolant drain tank sediment sample collection with remote-operated hand tools indicated the RCDT contains very little sediment, fission products, or core materials.

4. The letdown line sediment and adherent deposits are believed to be small due to continual flushing action during the accident sequence. If suspected plugging of one letdown cooler is

confirmed, the importance of letdown cooler sediment samples will be reconsidered. A pin-hole-type gamma camera survey of the block orifice indicated the block orifice does not contain as much fission product contamination as the nearby bypass line plumbing, which is inconsistent with the suspicion of block orifice plugging that had been the basis for considering acquisition and examination of the block orifice.

5. The TMI-2 accident sequence history information is not obtainable from the letdown system retained fission product and core material characterization because of postaccident flushing and the inability to segregate the sediment chronologically. The solids, which became suspended by the force circulation of reactor coolant through the reactor coolant system, which commenced about 16 h after accident initiation, would dominate the deposits in the letdown system and would not be traceable to chronological details of the accident sequence of events.

6. The location and abundance of fission products and uranium in the letdown system and RCDT plumbing can be determined adequately using pin-hole-type gamma camera surveys, thermoluminescent detector strings, and portable gamma-spectrometer detectors.

The product of the EX-RCS sample acquisition and examination program work plan consists of samples of reactor building basement wall and floor concrete and technical reports of sample examinations as follows:

Work Package Number	Product Item	Target Completion Date
751421300	a. Reactor building basement wall and floor concrete bores sample acquisition:	
	• 5000 psi (D-ring) wall	April 1987
	• 3000 psi (shield) wall	April 1987
	• Block (elevator/stairwell) wall	April 1987
	• Floor (locations to be determined)	April 1987

Work Package Number	Product Item	Target Completion Date
755420300	a. Reactor building basement wall and floor concrete examination report: Draft	September 1987

5.5 Synopsis

The additional in situ (gamma detector and thermoluminescent detector strings) measurements and sample (sediment and concrete bore) acquisition and examinations accomplished by GPUN in FY 1986 significantly improved the exploration and characterization of the reactor building basement for core fission products and materials. Figure 21 is a map of the basement with symbols showing locations of in situ measurements and sample acquisitions made since the accident. The EX-RCS fission product inventory sample acquisition and examination plan described herein for FY 1987 is designed to improve the accuracy of the estimates of the quantities and locations of core fission products and fuel (uranium), which were deposited in the reactor building basement concrete.

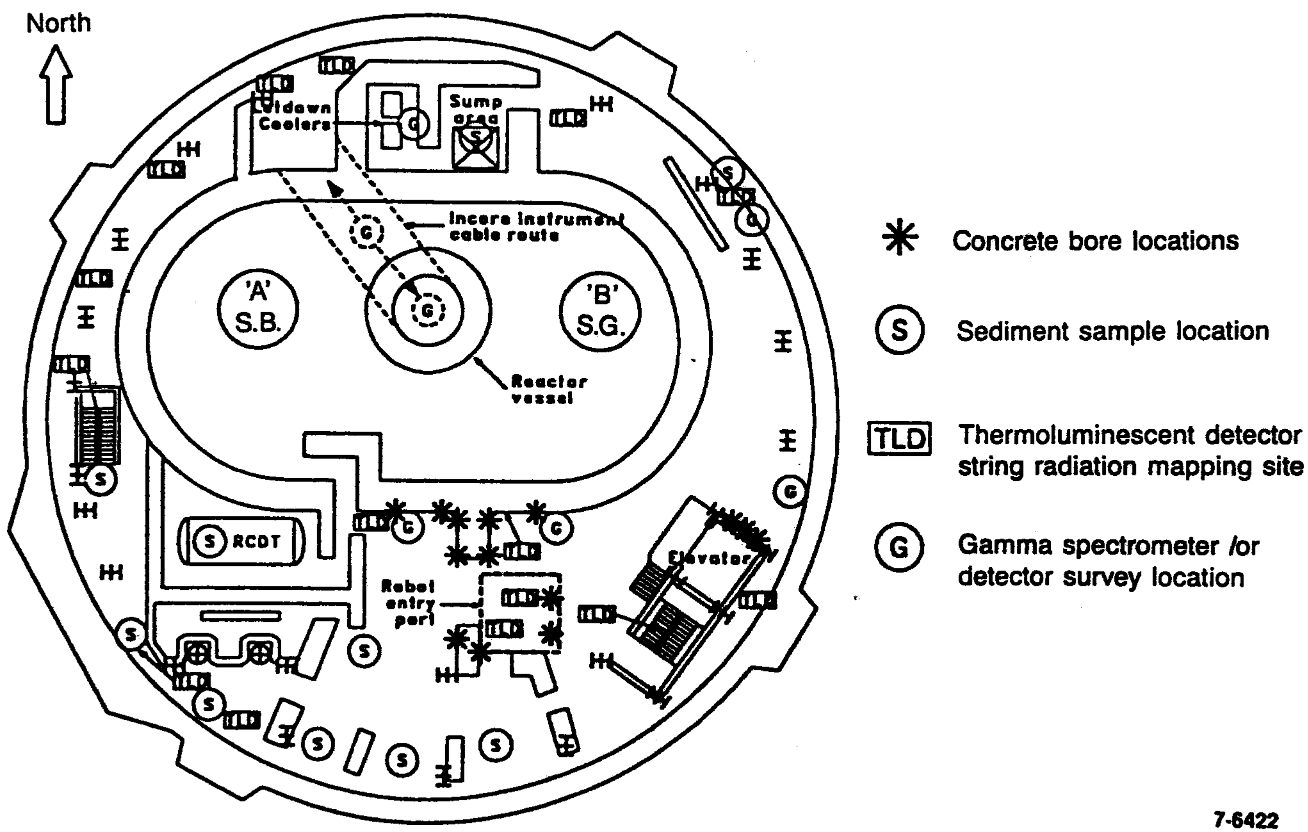


Figure 21. TMI-2 reactor building basement--FPI sample locations.

6. SAMPLE ACQUISITION AND EXAMINATION PROJECT MANAGEMENT SUPPORT WORK PLAN

6.1 Purpose

The TMI accident evaluation program sample acquisition and examination project management support provides the following:

a. Recruitment, maintenance, and supervision of a clerical and technical support staff.

b. Planning, technical direction, control, and documentation for the TMI-2 Accident Evaluation Program in situ measurements and sample acquisition and examinations.

c. Planning, technical direction, control, documentation, and maintenance of related support equipment (both hardware and software).

The documentation support includes periodic (weekly, monthly, annually) report contributions and formal status and technical presentations to EG&G, DOE, and special review and technical society groups.

6.2 Accomplishments

Visible products of the management support include: the periodic status reports that have emanated from the project since the creation (1981) of the EG&G-operated TMI Unit 2 Technical Information and Examination Program, special reports and master task subcontracts with private laboratories for TMI-2 sample examination support.

Special reports which have been published are as follows:

Report Number	Description/Title	Status
EGG-TMI-6169	TMI-2 Core Examination Plan	Revised July 1984

Report Number	Description/Title	Status
PF-NME-84-005	Participating Laboratories Survey	Completed September 1984
J. L. Mayberry letter to Distribution JLM-1-85	Core Sample Acquisition and Examination Work Plan	Draft Issued January 1984 for internal review
R. C. Schmitt letter to Distribution RCS-1-85	TMI-2 Core Examination Plan Evaluation	Draft issued January 1985 for internal review
EGG-TMI-7132	TMI-2 Accident Evaluation Program Sample Acquisition and Examination Plan	Issued January 1986
EGG-TMI-7121	TMI-2 Accident Evaluation Program Sample Acquisition and Examination Plan--Executive Summary	Issued January 1986
M. L. Russell letter to Distribution MLR-7-86	TMI-2 Accident Reference Document Listing	Issued June 1986
R. K. McCardell letter to J. Royen RKM-18-86	Draft TMI-2 Sample Examination Plan (for CSNI Members)	Issued June 6, 1986

Master task subcontracts which have been placed with private laboratories for TMI-2 sample examination support include the following:

Subcontract Number	Title	Status
C86-130969	Master Task Subcontract Between EG&G Idaho, Inc. and Battelle Columbus Division	Distributed April 1986
C86-130970	Master Task Subcontract Between EG&G Idaho, Inc. and General Electric Company	Distributed April 1986
C86-130971	Master Task Subcontract Between EG&G Idaho, Inc. and Babcock & Wilcox	Distributed May 1986

The current clerical and technical staff organization, shown in Figure 22, includes senior technical personnel with severe core damage accident and/or experiment and postaccident/experiment sample acquisition and examination experience. A geographic separation of staff members requires special supervisory skill applications to develop and maintain a cohesive team effort. The organization arrangement identifies the individual acquisition and examination project responsibilities and is intended to also:

1. Identify one individual to function as a coordinator and spokesperson for each of the four areas of TMI-2 sample acquisition and examination responsibilities at INEL.

2. Sustain continuity of individual examination task responsibility.

3. Distribute the staff support among all four INEL work assignment categories.

6.3 Detailed Work Plans

The management support work plan details are contained in the following work package:

Work Package Number	Work Package Title
7554PM00	Sample Acquisition, Handling, and Examination Project Management

The deliverable products of the management support work plan are as follows:

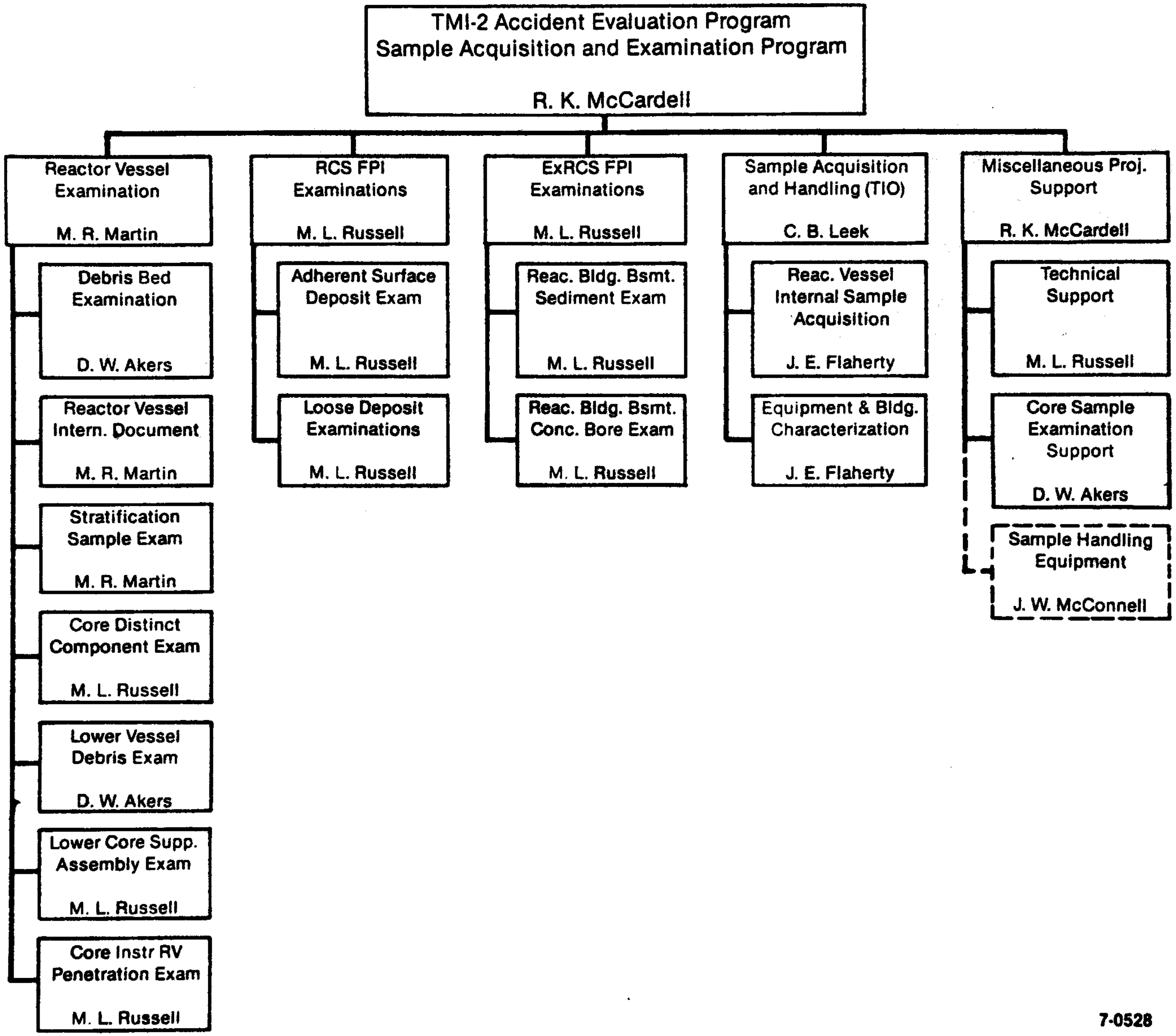


Figure 22. TMI-2 AEP SA&E project organization chart.

Product	Target Completion Date
a. <u>TMI-2 Accident Evaluation Program Sample Acquisitions and Examination Plan for FY 1987 and Beyond</u>	
• First Draft	January 1987
• Final	February 1987
• Annual update	December

7. SUMMARY

The material presented in the previous sections is intended to accomplish the following:

1. Explain the development of the examination plan for the severe core damage accident issues set forth in the TMI-2 Accident Evaluation Program document from sample selection to final reporting of the sample examination results.

2. Provide a perspective of the status of the TMI-2 accident investigation by identifying the examination program accomplishments in prior years.

3. Be flexible to accommodate new findings, information, and knowledge that may become available from either this examination plan, the GPU Nuclear defueling program, or any SCD research program.

4. Develop a TMI-2 accident examination program manual which can be (1) revised annually as new findings cause redirection, and (2) used for referenced by the analysts performing the studies needed to develop the understanding of the TMI-2 accident sequence and its radiological consequences.

The proposed financial plan for the SA&E Plan is shown in Table 19 and the companion schedule of activities is shown in Table 20. The list of work package numbers and titles on Table 19 identifies the entire work breakdown structure for the SA&E plan. In brief, the SA&E Plan work breakdown structure provides the following:

1. Acquisition of the samples listed in Table 4 in the Future Additional Samples column. For FY 1987 this includes two large-volume samples of core debris from the reactor vessel lower head region and possible peripheral fuel assembly lower sections at molten-core-material escape path locations.

TABLE 19. TMI-2 AEP SAMPLE ACQUISITION AND EXAMINATION WORK BREAKDOWN STRUCTURE AND FUNDING PLAN

		Cost by FY-BO (in 000)					
Task	Work Package Number	FY-1985 Actual	FY-1986 Actual	FY-1987	FY-1988	FY-1989	Total
Sample Acquisition:							
Technical Coordination	Complete	$ 7	$ 0	$ 0	$ 0	$ 0	$ 7
Project Management	Complete	37	0	0	0	0	37
RV Internals	Complete	52	36	0	0	0	88
RTD Thermowells	Complete	7	0	0	0	0	7
Incore Instruments	Complete	70	0	0	0	0	70
Fuel Rod Segments	Complete	107	83	0	0	0	190
Core Bores	Complete	1,679	1,753	0	0	0	3,432
Leadscrews	Complete	16	(5)	0	0	0	11
RCS Characterization	Complete	97	27	0	0	0	124
Discrete Core Components	Complete	18	34	0	0	0	52
RCS Equipment/Building Characteriztion	751421300	0	26	75	75	0	176
AEP Reserve	751421000	0	0	347	159	0	506
Acquisition Total	N/A	$ 2,090	$ 1,954	$ 422	$ 234	$ 0	$ 4,700
Sample Examination:							
Project Management	75542PM00	$ 114	$ 495	$ 549	$ 362	$ 0	$ 1,520
Debris Bed Samples	755420100	411	139	122	0	0	672
RV Internals Documentation	755420200	52	141	159	0	0	352
EX-RCS FP Inventory[a]	755420300	8	49	73	10	0	140
Lower Vessel Debris	Complete	1	0	0	0	0	1
Fuel Rod Segments	Complete	7	0	0	0	0	7
Core Bores	755420600	8	353	1,915	641	0	2,917
Leadscrews	Complete	153	6	0	0	0	159
Leadscrew Support Tube	Complete	62	0	0	0	0	62
RCS FP Inventory	755421000	19	59	442	0	0	520
Discrete Core Components	755421200	8	894	393	317	0	1,612
Lower Vessel Debris	755421600	0	196	320	0	0	516
Core Former Wall	Deleted	0	0	0	0	0	0
Core Support Assembly	Deleted	0	0	0	0	0	0
Core Sample Examination Support	755422100	0	0	174	0	0	174
RV Instrument Penetration	Out Year	0	0	0	0	250	250
RV Lower Head	Deleted	0	0	0	0	0	0
CSNI Samples	755422200	0	5	59	0	0	64
Examination Total	N/A	$ 843	$ 2,337	$ 4,206	$ 1,330	$ 250	$ 8,966
Acquisition and Examination Total	N/A	$ 2,933	$ 4,291	$ 4,628	$ 1,564	$ 250	$13,666

TABLE 19. (continued)

		Cost by FY-BO (in 000)					
Task	Work Package Number	FY-1985 Actual	FY-1986 Actual	FY-1987	FY-1988	FY-1989	Total
Related Capital Equipment:							
Stratification Sample Acquisition Equipment	Complete	$ 1,710	$ 27	$ 0	$ 0	$ 0	$ 1,737
Core Canister/Sample Handling Equipment	Complete	455	637	0	0	0	1,092
Image Processing and Documentation	Complete	259	0	0	0	0	259
Core Topography	Complete	377	174	0	0	0	551
Total RCE		$ 2,801	$ 838	$ 0	$ 0	$ 0	$ 3,639
Other DOE Labs		$ 115	$ 0	$ 0	$ 0	$ 0	$ 0
Costs Prior to 1985		$ 0	$ 0	$ 0	$ 0	$ 0	$ 1,966

a. FY-1985 work was RTD thermowells.

TABLE 20. TMI-2 AEP SAMPLE ACQUISITION AND EXAMINATION PLAN--SCHEDULE SUMMARY

	Schedule		
Activity Description	FY-1987	FY-1988	FY-1989
RCS Equipment and Building Characterization (sample acquisition)	XXXXXXXXXXXX	XXXXXXXXXXXX	
AEP Reserve	XXXXXXXXXXXX	XXXXXXXXXXXX	
SA&E Program Management	XXXXXXXXXXXX	XXXXXXXXXXXX	
Subsurface Debris Bed Sample Examination	XXXXXXXXXXXX		
Reactor Vessel Internals Documentation	XXXXXXXXXXXX		
EX-RCS Fission Product Inventory Sample Examination	XXXXXXXXXXXX	XXXX	
Core Bore Sample Examination	XXXXXXXXXXXX	XXXXXX	
RCS Fission Product Inventory Sample Examination	XXXXXXXXXXXX		
Discrete Core Component Examination	XXXXXXXXXXXX	XXXXXXXXXXXX	
Lower Vessel Debris Examination	XXXXXXXXXXXX		
Reactor Vessel Instrument Penetration Sample Examination			XXXXXXXXXXXX
Core Sample Examination Support	XXXXXXXXXXXX		
Sample Shipment to CSNI	XXXXXXXX		

Examination of the samples listed in the Proposed Future Exams column of Table 4. For FY 1987 this includes: completing the examination of nine core bores including fourteen fuel rod segments, four burnable poison rod/guide tube segments, and nine control rod/guide tube segments, two upper core region fuel rod segments, two upper core region control rod/guide tube segments, two large samples of core cavity floor loose debris, three RCS manway cover backing plates, a B-loop steam generator tube sheet top loose sediment sample, other RCS and piping loose sediment, the B-loop RTD thermowell and several reactor building basement wall and floor concrete bores. Examination of the remaining proposed samples is planned for FY 1988 and 1989.

The TMI-2 AEP will evaluate the availability of and pursue other resources to examine all the samples listed in the Future Additional Samples column of Table 4. Potential resources include the NRC, OECD/CSNI,[a] and domestic fuel suppliers.

A cost breakdown showing the proposed proportions of examination activities to the INEL, private laboratories, and other DOE laboratories is shown in Table 21.

Further subdivision of the WBS occurs during the process of authorizing the performance of work. INEL staff support and equipment and facilities operations are authorized using a system of work releases for nonunion supported activities and site work releases for union-supported activities. Work release documents include the Work Breakdown Structure account number, detail work scopes, schedules, and cost estimates. Site work release operations include step-by-step work procedures and quality assurance and operational safety organization approval and surveillance.

Offsite (non-DOE) support for services and/or equipment is obtained in two steps. First, the project authorizes the support with a requisition

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a. Organization for Economic Cooperation and Development, Committee on the Safety of Nuclear Installations.

TABLE 21. COST BREAKDOWN OF TMI-2 ACCIDENT EVALUATION PROGRAM SAMPLE EXAMINATION

Task	Funding ($ x 1000) INEL	Private Laboratories	Other DOE Laboratories
1. Subsurface debris bed samples	260.0	--	50[a]
2. Ex-reactor coolant system fission product inventory	--	399	--
3. Core bores	1,140.1	--	666.7 500[a]
4. Reactor coolant system fission product inventory	18	375.4	--
5. Distinct core components	740	--	251
6. Lower Vessel Debris	600	--	50[a]
7. Instrument tube penetrations	40	210	--
Totals	2,798.1	984.4	1,517.7

a. Work performed at ANL-E and funded by NRC.

which includes the WBS account numbers, work scope/equipment technical specifications, and quality assurance requirements and subcontracts organization then adds the federal-contract-regulation terms and conditions stipulations and obtains a qualified supplier to perform the work.

Other DOE laboratory support services are authorized with a requisition for services and/or equipment and/or a letter request to DOE with the appropriate work scope description. The finance transaction is conducted by DOE transfer of funds from the INEL cost account to the other laboratory cost accounts.

As work is performed, a comprehensive planning and budgets system provides cost and performance information using the work release, site work release, and requisition charge numbers as the basic accounting level. INEL labor charges are reported weekly, and nonlabor charges are reported monthly.

8. REFERENCES

1. C. V. McIsaac and D. G. Keefer, TMI-2 Reactor Building Source Term Measurements: Surfaces and Basement Water and Sediment, GEND-042, October 1984.

2. E. L. Tolman et al., TMI-2 Accident Evaluation Program, EGG-TMI-048, February 1986.

3. GEND Planning Report 001, June 1980.

4. J. O. Carlson, ed., TMI-2 Core Examination Plan, EGG-TMI-6169, July 1984.

5. D. W. Akers et al., Preliminary Report: TMI-2 Core Debris Grab Samples--Analysis of First Group of Samples, GEND-INF-060 Volume 1, July 1985.

6. TMI-2 Accident Core Heat-Up Analysis, NSAC-25, Nuclear Associates International and Energy Incorporated, June 1981.

7. E. L. Tolman et al., TMI-2 Core Bore Acquisition Summary Report, EGG-TMI-7385, September 1986.

8. R. W. Garner et al., An Assessment of the TMI-2 Axial Power Shaping Rod Dynamic Test Results, EG&G Idaho, Inc., GEND-INF-038, April 1983.

9. L. S. Beller and H. L. Brown, Design and Operation of the Core Topography Data Acquisition System for TMI-2, GEND-INF-012, EG&G Idaho, Inc., May 1984.

10. P. R. Bengel, TMI-2 Reactor Vessel Head Removal, GEND-044, GPU Nuclear Corp., September 1985.

11. V. R. Fricke, Results of End Fitting Separation in Preparation for Plenum Jacking, TPB-84-2, GPU Nuclear Corp, November 1984.

12. D. D. Wilson, TMI-2 Reactor Vessel Plenum Final Lift, GEND-054, GPU Nuclear Corporation, January 1986.

APPENDIX A

TMI-2 ACCIDENT REFERENCE DOCUMENTS HISTORY (PRELIMINARY)

APPENDIX A

TMI-2 ACCIDENT REFERENCE DOCUMENTS HISTORY (PRELIMINARY)

Appendix A contains a list of TMI-2 accident reference documents for use in planning and performing the TMI-2 AEP Sample Acquisition and Examination program. It is intended that the list have the following features:

a. Completeness in regards to (1) all information that has been published about the planning of the sample acquisition and examination program and (2) all information that has been published about the core damage and fission product inventory release during and following the accident.

b. Identification of SA&E and other individuals holding of the listed documents.

c. An annual update.

For convenience the list is arranged chronologically by date of document issue and in the following information categories:

1. TMI-2 accident general information

2. Reactor vessel information

3. Reactor coolant system information

4. EX-RCS information including the general fission product inventory information.

TMI-2 ACCIDENT EVALUATION PROGRAM SAMPLE ACQUISITION AND EXAMINATION REFERENCE DOCUMENTATION LIST--PRELIMINARY (December 1986)

Information[a] Category	Report Number	Publication Date	Title	Author Name	Author Company[b]	SA&E[d] Custodian
1-1	Docket 50-320	00/71?	Three Mile Island Nuclear Station Unit 2--Final Safety Analysis Report:	Unidentified	MEC, JCPL and PEC	MLR
			Volume 1: Introduction, General Description of Plant and Site Characteristics			MLR
			Volume 2: Design Criteria--Structures, Components, Equipment and Systems			MLR
			Volume 3: Containment Structure Analyses Appendixes			MLR
			Volume 4: Reactor and Reactor Coolant System			MLR
			Volume 5: Engineered Safety Features			MLR
			Volume 6: Instrumentation and Controls			MLR
			Volume 7: Electric Power and Auxiliary Systems			MLR
			Volume 8: Steam and Power Conversion System, Radioactive Waste Management and Radiation Protection			MLR
			Volume 9: Conduct of Operations, Initial Tests and Operation and Accident Analysis			MLR
			Volume 10: Quality Assurance and Responses to Additional Information Requirements			MLR
			Volume 11: Responses to USAEC Request for Additional Information--First Round Questions			MLR
			Volume 12: Responses to USAEC Request for Additional Information--Second Round Questions			MLR
1-2	No Number	06/79	Second Interim Report on the Three Mile Island Station Unit 2 (TMI-2) Accident	Unidentified	MEC	
1-3	Burns & Roe R-008	06/79	Listing of All Reactor Building Electrical and Instrument Equipment	Lane	B&R	(Knauts)
1-4	GQL-0924	07/79	Third Interim Report on the Three Mile Island Station Unit 2 (TMI-2) Accident	J. G. Herbein	MEC	
1-5[c]	NUREG-0578	07/79	TMI-2 Lessons Learned Task Force--Status Report and Short-Term Recommendations	Unidentified	NRC-ONRR	MLR
1-6	No Number	07/79	Planning Study for Containment Entry and Decontamination	Not Identified	BNI	
1-7[c]	NUREG-0600	09/79	Investigation into the March 28, 1979 Three Mile Island Accident by Office of Inspection and Enforcement	Unidentified	NRC-IE	MLR
1-8[c]	No Number	10/79	Report of the President's Commission on the Accident at Three Mile Island--the Need for Change: The Legacy of TMI	J. G. Kemeny, et al.	President's Commission	MLR
1-9	NUREG-0585	10/79	TMI-2 Lessons--Learned Task Force--Final Report	Unidentified	NRC-ONRR	MLR
1-10	No Number	12/79	Summary Technical Plan for TMI-2 Decontamination and Defueling		MEC	
1-11[c]	NUREG/CR-1250	01/80	Three Mile Island A Report to the Commissioners and the Public--Volume II	M. Rogovin	NRC-SIG	
1-12[c]	NSAC-80-1	03/80	Analysis of Three Mile Island--Unit 2 Accident	Unidentified	NSAC	MLR
1-13	Heat Transfer Engineering Volume 1, No. 3	03/80	The Accident at Three Mile Island	J. G. Collier L. M. Davies	UKAEA	MLR

Information Category[a]	Report Number	Publication Date	Title	Author: Name	Author: Company[b]	SARE[d] Custodian
1-14	TMI-II-RR-6	04/80	TMI-2 Recovery Quarterly Progress Report for the Period Ending 3/31/80		MEC	
1-15[c]	GEND-001	06/80	GEND Planning Report	Unidentified	GPUN	MLR
1-16	NUREG-0660 Volume 1 and 2	08/80	NRC Action Plan Developed as a Result of the TMI-2 Accident	Unidentified	NRC	MLR
1-17	GEND-002	10/80	Facility Decontamination Technology Workshop November 27-29, 1979	Sponsored by DOE and EPRI	TIO	MLR
1-18[c]	NUREG-0683	00/81	Final Programmatic Environmental Impact Statement Related to Decontamination and Disposal of Radioactive Wastes Resulting from March 28, 1979 Accident--Three Mile Island Station Unit 2	Unidentified	MEC, JCP&LC, PEC	MLR (Draft)
1-19	R-81-002	01/81	Catalog of Data Collected During the TMI-2 Accident	Monday	TEC	(Knauts)
1-20	GPU-TDR-044	02/81	Annotated Sequence of Events, March 28, 1979	T. L. Van Witbeck J. Putnam	GPUN	MLR
1-21	GPU-TDR-261	05/81	Annotated Sequence of Events, March 29, 1979 Through April 30, 1979--TMI-2	R. Smith	GPUN	MLR
1-22	GEND-INF-022	08/82	Status of TMI-2 Instruments and Electrical Components	Hezbert	EG&G	(Knauts)
1-23	NUREG-0900	01/83	Nuclear Power Plant Severe Accident Research Plan	Unidentified	NRC-ONRR	MLR
1-24	EGG-TMI-6169	04/83	TMI-2 Core Examination Plan--Review Copy	D. E. Owen, et al.	EG&G	MLR
1-25	SD-WM-TI-067	04/83	Analysis of Three Mile Island Unit 2 Reactor Cooling System Transients	J. O. Henrie A. K. Postma	RI-RHO	MLR
1-26	EGG-TMI-6169	12/83	TMI-2 Core Examination Plan	D. E. Owen, et al.	EG&G	MLR
1-27[c]	No Number	04/84	Report on TMI-2 Technical Data Acquisition Program Plan	J. C. Cunnane, et al.	BCL	MLR
1-28	GEND-40	06/84	Final Report on the In Situ Testing of Electrical Components and Devices at TMI-2			
1-29	EGG-TMI-6169	07/84	TMI-2 Core Examination Plan	J. O. Carlson, Editor	EG&G	MLR
1-30[c]	PF-WME-84-005	10/84	TMI-2 Core Sample Acquisition and Examination Project--Participating Laboratories Survey	M. R. Martin S. O. Peck	EG&G	MLR
1-31	RCS-1-84 (Letter)	01/85	TMI-2 Core Examination Plan Evaluation	M. L. Russell	EG&G	MLR
1-32	NP-3810-SR[d]	01/85	Joint TMI-2 Information and Examination Program--EPRI Participation and Support	TBD	EPRI	MLR
1-33	No Number	06/85	Needs for Results of TMI Data Acquisition and Analysis Program	M. H. Fontana, et al.	EAI	
1-34[c]	GEND-50	07/85	TMI Technical Information and Examination Program Instrumentation and Electrical Summary Report	R. D. Meininger	EG&G	MLR[d]
1-35	No Number	07/85	The Impact of TMI-2 on Future Licensing	W. F. Pasedag A. K. Postma	NRC NRC	MLR
1-36	No Number	09/85	TMI-2 Accident Evaluation Program--Draft	Unidentified	EG&G	MLR
1-37	N/A	10/85	TMI-2 Programs Division Master Plan Rev. 4	Unidentified	EG&G	RKM
1-38[c]	EGG-TMI-7132	01/86	TMI-2 Accident Evaluation Program Sample Acquisition and Examination Plan	M. L. Russell, et al.	EG&G	MLR
1-39[c]	EGG-TMI-7121	01/86	TMI-2 Accident Evaluation Program Sample Acquisition and Examination Plan--Executive Summary	M. L. Russell, et al.	EG&G	MLR
1-40	EGG-TMI-7048	02/86	TMI-2 Accident Evaluation Program	E. Tolman, et al.	EG&G	MLR

Information[a] Category	Report Number	Publication Date	Title	Author		SA&E[d] Custodian
				Name	Company[b]	
1-41	CONF-8510166	04/86	Proceedings of the First International Information Meeting on TMI-2 Accident (10-21-85)	Compiled by S. R. Langer	EG&G	MLR[d]
1-42[c]	DOE/NE/34109--TI	04/86	USDOE and GPUNC R&D Activities on TMI-2, Annual Report for 1985	Not Identified	GPUNC EG&G	MLR[d]
1-43[c]	GEND-055	04/86	USDOE TMI R&D Program 1985 Annual Report	G. R. Brown, Editor	EG&G	MLR
1-44	TPO/TMI-186	07/86	Planning Study on a Strategy for Recovery Program Completion and Postconfiguration		GPUN	
1-45	EG&G Letter PJG-89-86	12/86	TMI-2 Programs Division Master Plan Revision 6	P. J. Grant	EG&G	MLR
2-1	No Number	08/69	RV Internals Drawings--(Reduced Size)	N/A	B&W	MLR
2-2	TRG-71-37 Rev. 1	01/76	Internals Fabrication and Trial Fitup--Photographs	Unidentified	B&W	MLR
2-3	B&W FMD-79-269	04/79	In-Core Thermocouple Data	Walton	B&W	(Knauts)
2-4	No Number	04/79	In-Core Instrument Resistance Measurement	H. Warren		
2-5	TDR-049	08/79	TMI-2 Postaccident Criticality Analyses	E. W. Barr, et al.	GPUN	
2-6	GOV-00-0095	10/79	Technical Staff Analysis Report on Chemistry to President,s Commission on the Accident at Three Mile Island	R. E. English Task Force		MLR
2-7	ORNL/CSD/TM-106	12/79	Criticality Analyses of Disrupted Core Models of TMI-2	R. N. Westfall, et al.	ORNL	
2-8	LA-8041-MS	03/80	TMI-2 Decay Power: LASL Fission Product and Actinide Decay Power Calculations for the President's Commission on the Accident at Three Mile Island	T. R. England W. B. Wilson	LASL	
2-9	B&W-80-78	10/80	Catalog of TMI-2 Data (loose parts monitor, SPND, core TC)		B&W	(Knauts)
2-10	NSAC-24	01/81	TMI-2 Accident Core Heat-up Analysis	TBD	NSAC	
2-11	SAI-DP-245-22	01/81	TMI-2 Instrument History Folder Users Guide	Mayo Zigler	SAI	(Knauts)
2-12	GEND-016	05/81	Accountability Study for TMI-2 Fuel	P. Goris D. D. Scott	HEDL	MLR
2-13	GEND-007	05/81	Three Mile Island Unit 2 Core Status Summary A Basis for Tool Development for Reactor Disassembly and Defueling	D. W. Croucher	EG&G	MLR
2-14	NSAC-25	06/81	TMI-2 Accident Core Heat-up Analysis A Supplement	D. Coleman C. Shaffer	NAI EI	MLR
2-15	TPO/TMI-005	06/81	Technical Plan for Reactor Disassembly and Defueling	Unidentified	BNI	MLR
2-16	No Number	08/81	Characterization of TMI-2 Postaccident Primary Coolant	J. E. Cline, et al.	SAI	
2-17	Twelfth Information Meeting on Reactor Noise Accident Analysis	10/81	Postaccident Reactor Diagnostics at TMI-2	Mayo	B&W?	(Knauts)
2-18	GEND-17	12/81	Analysis of the SPND Measurement System Response to Temperature During TMI-2 Accident	N. Wilde J. L. Morrison, Jr.	EG&G	
2-19	GEND-INF-017-7	01/82	Field Measurements and Interpretation of TMI-2 Instrumentation: YM-AMP-7023 and 7025 (loose parts monitor)	Jones, et al.	TEC	(Knauts)
2-20	GEND-INF-017-11	04/82	Field Measurement and Interpretation of TMI-2 Instrumentation: NI-Amp-2 (source range amplifier)	Jones, et al.	TEC	(Knauts)
2-21[c]	NSAC-28	05/82	Interpretation of TMI-2 Instrument Data	Numerous (from B&W, CEI, ORNL, TEC & NSAC)	NSAC	MLR

Information Category[a]	Report Number	Publication Date	Title	Author Name	Author Company[b]	SA&E Custodian[d]
2-22	ANS WAM Trans. Volume 43 (p. 5)	11/82	TMI-2 Core Examination: First Results	D. E. Owen M. R. Martin	EG&G	
2-23	GEND-20	11/82	Examination Results on TMI-2 LPM Charge Converters YM-AMP-7023 & 7025			
2-24	SAI-139-83-01-RV	12/82	Analysis of Three Sections of TMI-2 H-8 Leadscrew by Collimated Gamma Spectroscopy	J. A. Daniel, et al.	SAI	
2-25	TPO/TMI-026	12/82	Quick Look Inspection Results		GPUN	
2-26	GEND-INF-031 ED-E3-82-015	01/83	Preliminary Report of TMI-2 In-Core Instrument Damage	M. E. Yancey N. Wilde	EG&G EG&G	MLR
2-27	EGG-TMI-5966	02/83	TMI-2 Core Examination Program: INEL Facilities Readiness Study	T.B. McLaughlin	EG&G	MLR
2-28	NUS-TM-346	03/83	TMI-2 Preliminary Plenum Surface Area and Estimated Cesium Retention	D. W. Tonkay	NUS	
2-29	GEND-30, Volume I	04/83	TMI-2 Quick Look and CRDM Uncoupling			
2-30	GEND-30, Volume II	04/83	Quick Look Inspections: Results			
2-31	GEND-INF-038	04/83	An Assessment of the TMI-2 Axial Power Shaping Rod Dynamic Test Results	R. W. Garner, et al.	EG&G	MLR
2-32	LWR Severe Accident Meeting-Cambridge, MA	08/83	TMI-2 Core Examination	R.R. Hobbins, et al.	EG&G	
2-33	GEND-INF-023 Volume V	09/83	Analysis of TMI-2 Reactor Coolant System Transients	J. O. Henrie		
2-34	4550-83-0412	09/83	TMI-2 Core Radial and Axial Power History Data	G. R. Eidam	GPUN	MLR
2-35	TPO/TMI-097 SAI-83/1083	11/83	Analyses of the H-8, B-8 and E-9 Leadscrews from the TMI-2 Reactor Vessel	J. A. Daniel, et al.	SAI	MLR
2-36	TPO/TMI-102	12/83	Planning Study on Method for Measuring Fuel Materials Collected in Lower Region of Reactor Vessel		GPUN-TPO	
2-37	TPO/TMI-080	12/83	Planning Study on Plenum Disposal		GPUN	
2-38	GEND-INF-062	00/84	TMI-2 Reactor Vessel Head Removal	P. R. Bengel, et al.	GPUN	
2-39[c]	EPRI-NP-3407	01/84	Initial Examination of the Surface Layer of a 9-Inch Leadscrew Section Removed from TMI-2	G. M. Bain G. O. Hayner	B&W	MLR
2-40	No Number	01/84	The Sequence of Core Damage in TMI-2 (Accident Review Workshop at Harrisburg, PA	M. L. Picklesimer		
2-41	TPO/TMI-103	02/84	Chemical Analyses and Test Results of Sections of the TMI-2 H8 Leadscrew	K. J. Hofstetter, et al.	GPUN	
2-42	EGG-TMI-6531-1	03/84	TMI-2 Core Debris Grab Sample Quick Look Report	D. W. Akers R. L. Nitschke	EG&G EG&G	MLR
2-43	GEND-INF-031 Volume II	04/84	TMI-2 In-Core Instrument Damage--An Update	M. E. Yancey, et al.	EG&G	MLR
2-44	GEND-INF-044	04/84	TMI-2 Leadscrew Debris Pyrophoricity Study	R. L. Clark, et.al.	PNL	MLR
2-45	GEND-INF-012	05/84	Design and Operation of the Core Topography Data Acquisition System for TMI-2	L. S. Beller H. L. Brown	EG&G EG&G	MLR
2-46	NP-3509	06/84	Use of Pressurized Water to Decontaminate TMI-2 Leadscrew Sections	H. R. Gardner Pr. Inv.	Quadrex	MLR
2-47	EGG-TMI-6630	06/84	Draft Preliminary Report: TMI-2 Core Debris Grab Samples--Analysis of First Group of Samples	D. W. Akers, et al. B. A. Cook	EG&G EG&G	MLR

Information Category[a]	Report Number	Publication Date	Title	Author Name	Author Company[b]	SAME[d] Custodian
2-48	ANS Top. Meeting on FP Behavior and Source Term Research	07/84	TMI-2 Leadscrew Radionuclide Deposition and Characterization	K. Vinjamuri, et al.	EG&G	
2-49[c]	RDD:85:5097-01:01	07/84	TMI-2 H8A Core Debris Sample Examination--Final Report	G. O. Hayner	B&W	MLR
2-50[c]	EGG/TIO-M-00284	08/84	Radionuclide Distribution in TMI-2 Reactor Building Basement Liquids and Solids	J. T. Horan	EG&G	MLR
2-51[c]	EGG-M-11984	08/84	TMI-2 Core Debris Analytical Method and Results	D. W. Akers B. A. Cook	EG&G	MLR
2-52	EGG-TMI-6697	09/84	Draft Report: TMI-2 Core Debris--Cesium Release/Settling Test	D. W. Akers D. A. Johnson	EG&G	MLR
2-53	NUS-TMI-3467	09/84	TMI-2 Preliminary Plenum Surface Areas and Estimated Cesium Retentions	D. W. Tokay	NUS	
2-54	Twelfth WRSR Meeting	10/84	TMI-2 Core Debris Sample Analysis	B. A. Cook D. W. Akers	EG&G	
2-55	No Number	10/84	Mechanisms for Anomolous Signal Outputs from Self-Powered Neutron Detectors	C. P. Cannon	HEDL	
2-56[c]	TPB-84-2	11/84	Results of End Fitting Separation in Preparation for Plenum Jacking	V. R. Fricke	GPUN	MLR
2-57[c]	TB-84-1	11/84	Plenum Inspection Results	V. R. Fricke	GPUN	MLR
2-58	BAW-1855	12/84	TMI-2 Planning Study for Core Support Assembly Defueling	TBD	B&W	
2-59	TPB-84-7	12/84	Plenum TLD Data	F. Augustine	GPUN	MLR
2-60[c]	TPB-84-8	12/84	Core Debris Bed Probing	V. R. Fricke	GPUN	MLR
2-61	(ANS Winter Meeting)	11/85	TMI-2 Core Conditions and Postulated Accident Scenario	J. M. Broughton, et al.	EG&G	
2-62[c]	0362-1626/85/1022-0035 Annual Rev. Energy 1985 10:35-52	00/85	The Three Mile Island Unit 2 Core: A Postmortem Examination	R. S. Denning	BCL	MLR
2-63	TPB-85-5	02/85	Source Term (Radiological Characterization of Fuel Debris Grab Samples)	R. Rainish	GPUN	MLR
2-64[c]	TPB-85-6	02/85	Reactor Lower Head Video Inspection	V. R. Fricke	GPUN	MLR
2-65[c]	TPB-84-8 Rev. 1	02/85	Core Debris Bed Probing	V. R. Fricke	GPUN	MLR
2-66	TPB 85-6 Rev. 1	03/85	Reactor Lower Head Video Inspection		GPUN	
2-67	NACES Symposium Corrosion 85	03/85	Initial Examination of Decontamination Barrier on TMI-2 Leadscrew	V. F. Baston	GPUN	
2-68	DGK-5-85 (Letter)	04/85	In-Core Instruments Tube Probing	D. G. Keefer	EG&G	MLR
2-69	TPO/TMI-165	04/85	Determination of Fuel Distribution in TMI-2 Based on Axial Neutron Flux Profile	A. J. Baratta B. Bendini	PSU	
2-70	TPB-85-11	04/85	Gamma Scanning of In-Core Detectors		GPUN	
2-71	GEND-INF-059 HEDL-7484	05/85 04/85	Solid-State Recorder Neutron Dosimetry	R. Gold, et al.	HEDL	MLR MLR*
2-72	GEND-INF-060 Volume II	05/85	TMI-2 H8A Core Debris Sample Examination Final Report	G. O. Hayner	B&W	
2-73a	TPB-85-19	08/85	Core Condition Summary	F. Augustine	GPUN	MLR
2-74[c]	TPO/TMI-175	06/85	Analysis of Gamma Scanning of In-Core Detector #18 (L-11) in Lower Reactor Vessel Head	R. Rainish	GPUN	MLR
2-75	TPB-85-14	06/85	Analysis of Gamma Scanning of In-Core Detector #18 (L-11) in the Lower Reactor Vessel Head	R. Rainish	GPUN	MLR

Information Category[a]	Report Number	Publication Date	Title	Author Name	Author Company[b]	SAAE[d] Custodian
2-76[c]	TPB-85-015	06/85	Plenum Underside Damage	G. Worku	GPUN	MLR
2-77	EGG-TMI-6853 Parts 1 and 2	07/85	Draft Report: TMI-2 Core Debris Grab Samples--Examination and Analysis	D. W. Akers, et al.	EG&G	MLR
2-78	GEND-INF-060 Volume 1	07/85	Preliminary Report: TMI-2 Core Debris Grab Samples--Analysis of First Group of Samples	D. W. Akers, et al.	EG&G	MLR
2-79	TPB-85-20	07/85	Hydraulic Disturbance of the Debris in the Bottom Head of the TMI-2 Reactor Vessel	V. R. Fricke	GPUN	MLR
2-80[c]	RDD:85:4240-01:01	07/85	Grain Size Examination of TMI-2 Fuel Pellets	G. O. Hayner	B&W	MLR
2-81	(ACS Symposium)	05/85	TMI-2 Core Debris Chemistry and Fission Product Behavior	D. W. Akers	EG&G	MLR
2-82[c]	RE-E-85-004	08/85	Reevaluation and Analysis of the TMI-2 Core Damage Sequence	D. J. N. Taylor	EG&G	MLR
2-83	RE-E-85-005	09/85	Laboratory Testing of TMI-2 Self-Powered Neutron Detector	D. J. N. Taylor	EG&G	
2-84	TPO/TMI-138 Rev. 1	09/85	Data Report-Reactor Characterization Vol. I		GPUN	
2-85	GEND-INF-052 EGG-TMI-6685	09/85	Examination of H8 and B8 Leadscrews from Three Mile Island Unit 2 (TMI-2)	K. Vinjamuri, et al.	EG&G	MLR
2-86[c]	GEND-044	09/85	TMI-2 Reactor Vessel Head Removal	P. R. Bengel, et al.	GPUN	MLR
2-87[c]	TPO/TMI-117 Rev. 1	09/85	In-Vessel Data Acquisition Technical Plan	T. L. Ott J. A. Weissburg	GPUN	MLR
2-88[c]	TB-85-21 Rev. 1	10/85	Lower Head Core Debris Samples	G. Worku	GPUN	MLR
2-89	(ANS Winter Meeting)	11/85	Elemental and Radionuclide Content of TMI-2 Core Debris Grab Samples	D. W. Akers R. R. Hobbins	EG&G	MLR
2-90[c]	Corrosion/85, Paper #120, NACE, Houston, TX	12/85	Examination of Core Debris Samples from the Three Mile Island Unit 2 Reactor	G. O. Hayner, et al.	B&W	MLR
2-91	NP-4292	01/86	Simulation of the TMI-2 Accident Using the MAAP Modular Accident Analysis Program Version 2.0	M. A. Kenton, et al.	FAI	MLR[d]
2-92[c]	GEND-INF-073	01/86	TMI-2 Defueling Tools Engineering Report	Not Identified	B-NARD	MLR
2-93[c]	TB-86-004	01/86	Visual Examination of the Core Void Periphery	G. Worku	GPUN	MLR
2-94[c]	TPO/TMI-173	01/86	TMI-2 Reactor Vessel Plenum Final Lift-Data Report	D. C. Wilson	GPUN	
2-95[c]	TB-86-01 Rev. 1	02/86	Fuel Rod Segment Sampling	G. Worku	GPUN	MLR
2-96[c]	TMI-86-001 (Draft)	02/86	TMI-2 Self-Powered Neutron Detector Data Interpretation	D. J. N. Taylor	EG&G	MLR
2-97	EGG-TMI-7174	03/86	TMI-2 Source and Intermediate Range Neutron Flux Monitors Data Report	R. D. McCormick	EG&G	MLR
2-98[c]	GEND-INF-067	03/86	Final Report on the Examination of the Leadscrew Support Tube from Three Mile Island Reactor Unit 2	M. P. Failey, et al.	BCL	MLR
2-99	Progress in Nuclear Energy, Vol. 17, No. 2 pp. 141-174	04/86	Reactor Disassembly Activities at Three Mile Island	W. M. Burton R. L. Freemerman	EG&G BNI	MLR
2-100	No number?	04/86	Extended Analysis of Source Range Monitor to Evaluate Core Relocation During the TMI-2 Accident	A. J. Baratta H. Y. Wu	PSU	
2-101	TB-84-08 Rev. 2	04/86	Core Debris Bed Probing	G. Worku	GPUN	MLR
2-102	TB-85-19 Rev. 1	04/86	Core Conditions Summary	?	GPUN	
2-103[c]	TB-85-21 Rev. 3	05/86	Lower Head Core Debris Samples	G. Worku	GPUN	MLR
2-104[c]	TB-86-33 Rev. 0	06/86	Dewatering Canisters in Preparation for Offsite Shipment	R. Rainish	GPUN	MLR

Information Category[a]	Report Number	Publication Date	Title	Author Name	Author Company[b]	SA&E Custodian[d]
2-105[c]	TB-86-35 Rev. 0	07/86	Core Stratification Sampling Program	G. Worku	GPUN	MLR
2-106[c]	TB-86-35 Rev. 1	07/86	Core Stratification Sampling Program	G. Worku	GPUN	MLR
2-107[c]	TB-86-35 Rev. 2	07/86	Core Stratification Sampling Program	G. Worku	GPUN	MLR
2-108[c]	TB-86-35 Rev. 3	08/86	Core Stratification Sampling Program	G. Worku	GPUN	MLR
2-109[c]	RDD:86:522-01:01	08/86	TMI-2 H8A Core Debris Melting Point Study	G. O. Hayner R. E. Womack	B&W	MLR
2-110[c]	TB-86-33 Rev. 1	09/86	Dewatering Canisters in Preparation for Offsite Shipment	R. Rainish	GPUN	MLR
2-111[c]	EGG-TMI-7385	09/86	TMI-2 Core Bore Acquisition Summary Report	E. L. Tolman, et al.	EG&G	RKM
2-112	GEND-INF-075 Parts 1 and 2	09/86	TMI-2 Core Debris Grab Samples--Examination and Analysis	D. W. Akers, et al.	EG&G	MLR
2-113[c]	TB-86-39	10/86	Estimated Specific Activities and Radionuclide Inventories of Neutron Activated RPV Internal Components	R. Rainish	GPUN	MLR
2-114[c]	TB-86-42	10/86	Results of the October, 1986 Core Void Region Video Inspection	V. R. Fricke	GPUN	MLR
2-115[c]	TB-85-19 Rev. 2	10/86	Core Conditions Summary	G. Worku	GPUN	MLR
2-116[c]	ANL-E Letter Dated 10/01/86	10/86	Results of Auger and Microprobe Examinations of Five Samples Taken from TMI-2 Upper Plenum	J. E. Sanecki	ANL-E	RKM
2-117[c]	TB-86-43 Rev. o	11/86	Impact of Core Drilling Operations on Defueling Platform Dose Rates (also RCS coolant conditions)	R. Rainish	GPUN	MLR
2-118[c]	EGG-TMI-7402	11/86	Core Relocation in the TMI-2 Accident	P Kuan	EG&G	MLR
2-119[c]	EGG-M-25986	11/86	TMI-2 Lower Vessel Debris Examinations	D. W. Akers, et al.	EG&G	MLR
2-120[c]	TB-86-45	12/86	Crust Breaking Via Core Drilling	S. Bokharee	GPUN	MLR
3-1[c]	NPGD-TM-557	07/80	Compilation of Chemistry Results for TMI-2 Reactor Coolant System	J. H. Hicks	B&W	MLR
3-2	ANL/LWR/SAF 80-4	10/80	Analysis of Thermal-Hydraulic Behavior During TMI-2 LOCA	J. C. M. Leung	ANL	
3-3	No Number	09/81	Characterization of TMI Unit 2 Postaccident Primary Coolant	J. E. Cline, et al.	SAI	
3-4	GEND-018	11/81	Nondestructive Techniques for Assaying Fuel Debris in Piping at Three Mile Island Unit 2	K. Vinjamuri, et al.	EG&G	MLR
3-5	Nuclear Engineer Design 69	00/82	Post Facta Analysis of the TMI-2 Accident (1): Analysis of Thermal-Hydraulic Behavior by Use of RELAP4/MOD6/V4/J2	F. Tanabe, et al.	?	
3-6	GEND-INF-017-5	01/82	Field Measurements and Interpretation of TMI-2 Instrumentation CF-2-LT2 (Flood Tank Level)	Jones, et al.	EG&G	(Knauts)
3-7	GEND-INF-017-4	01/82	Field Measurements and Interpretation of TMI-2 Instrumentation CF-2-LT4	Jones, et al.	TEC	(Knauts)
3-8	GEND-INF-026	08/82	Static In Situ Testing of Axial Power Shaping Rod and Shim Safety Control Rod Mechanisms	Soberano, et al.	UE&C	(Knauts)
3-9[c]	GEND-INF-024	11/82	Review of TMI-2 Resistance Temperature Detectors, Accident Data and In Situ Testing	J. W. Mock	?	MLR
3-10[c]	EPRI-NP-2722	11/82	Characterization of the Contamination in the TMI-2 Reactor Coolant System	J. C. Cunnane S. L. Nicolosi	BCL	
3-11[c]	No Number	03/83	Report on Analysis of Two TMI-2 'Quick-Look' RCS Fluid Samples	T. L. Hardt J. E. Bullard	GPUN	MLR

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3-12[c]	ORNL-AMS8304R1 Volume II	09/83	Status of TMI-2 Primary RTDs During and After the Accident	H. M. Hashemian K. E. Holbert	AMS AMS	MLR
3-13[c]	TPO/TMI-051	04/84	Planning Study and Characterization of Fuel Debris in TMI-2 (Reactor Coolant System)	S. A. Bokharee	GPUN	MLR
3-14[c]	TPO/TMI-124	08/84	Ex-Vessel Fuel Characterization	S. A. Bokharee	GPUN	MLR
3-15	EPRI-NP-3804	11/84	Gamma-Ray Spectrometer System for High-Radiation Fields	G. R. Laurer, et al.	NYU-MC	MLR
3-16	Hub-268-84 (Letter) GEND-INF-014 (not published)	11/84	Analysis of TMI-2 "A" Steam Generator Hot Leg Resistance Thermal Detector	D. W. Akers, et al.	EG&G	MLR
3-17	TPB-84-5	12/84	OTSG "A" External Measurements	C. H. Distenfeld	GPUN	MLR
3-18	TPB-84-6	12/84	Ex-Vessel Fuel Generic Survey Results	C. H. Distenfeld	GPUN	MLR
3-19	Letter	12/84	List of Materials Wetted by the Reactor Coolant System at TMI-2	E. J. Bateman L. H. Lilien	B&W GPUN	
3-20	TPO/TMI-122	01/85	Reactor Coolant System Sample Results	C. S. Orland P. M. Holt	GPUN GPUN	MLR
3-21	TPB-85-7	02/85	Fuel Deposition in the "B" Core Flood Tank System	C. H. Distenfeld	GPUN	MLR
3-22	No Number	03/85	Reactor Coolant System Debris Transport		MPR	
3-23	DGK-10-85 (Letter)	05/85	Status of Primary Systems Gamma Scans	D. G. Keefer	EG&G	MLR
3-24	ACS Symposium	05/85	Thermal Hydraulic Features of the TMI Accident	E. L. Tolman	EG&G	
3-25	4550-85-0197 (Letter)	07/85	"A" D-Ring Gamma Camera Scans	C. H. Distenfeld	GPUN	MLR
3-26	TPB-85-018	07/85	"A" D-Ring Gamma Camera Scans	J. Greenborg	GPUN	MLR
3-27[c]	GEND-INF-064 ORNL/M-98[d]	09/85	Postaccident Examination of Platinum Resistance Thermometers Installed in the TMI-2 Reactors	R. M. Carroll R. L. Shepard (R. C. Strahm, APPIII)	ORNL ORNL EG&G	MLR
3-28	STC-08-85 (Letter)	09/85	TMI Gamma Spectral Data from Primary System Scanning Measurements	S. T. Croney	EG&G	MLR
3-29[c]	TB-85-37	11/85	RCS Contamination Radiation Model	A. Yahashi	GPUN	MLR
3-30[c]	TB-86-02	01/86	Physical/Radiological Inspection and Sampling of the Pressurizer	H. P. Wood	GPUN	MLR
3-31	EGG-TMI-7100	01/86	Analysis of TMI-2 Pressurizer Level Indications	J. L. Anderson	EG&G	MLR
3-32[c]	NP-4292	01/86	Simulation of the TMI-2 Accident Using the MAAP Modular Accident Analysis Program Version 2.0	M. A. Kenton, et al.	FAI	MLR[d]
3-33[c]	TB-86-13	02/86	Gamma Analysis of Pressurizer Sample	T. E. Cox	GPUN	MLR
3-34[c]	TB-86-16	03/86	Alpha Measurements and Surface Scrape Samples of Pressurizer Manway Diaphragm	M. Lambert	GPUN	MLR
3-35[c]	TB-86-19	03/86	Analysis of TLD String Drop Data--Pressurizer Internal Surfaces	A. Yahashi	GPUN	MLR
3-36[c]	TB-86-24	04/86	OTSG-A Upper Tube Sheet Debris Samples	P. J. Babel	GPUN	MLR
3-37[c]	TB-86-23	04/86	Examination of "A" & "B" Steam Generators for Dislocated Fuel	H. P. Wood	GPUN	MLR
3-38[c]	No Number (draft)	04/86?	Adherent Activity on TMI-2 Internal Surfaces	V. F. Baston K. J. Hofstetter	GPUN	MLR
3-39[c]	TB-85-10a	08/86	A Reevaluation of Fuel in the Pressurizer	C. H. Distenfeld	GPUN	MLR

Information Category[a]	Report Number	Publication Date	Title	Author Name	Author Company[b]	SA&E[d] Custodian
3-40	No Number	08/86	Preliminary Compilation of TMI-2 Water Processing Reference Data	V. F. Baston	PSI	MLR
3-41[c]	TB-86-37	09/86	Deposition of Fuel on the Inside Surfaces of the RCS	J. Greenborg	GPUN	MLR
3-42	PSI-TR-86/011	09/86	Chemical Behavior of Selected Radionuclides in the TMI-2 Reactor Coolant System Fluid	V. F. Baston	PSI	MLR
3-43[c]	EGG-TMI-7324	09/86	Determination of Void Fraction from Source Range Monitor and Mass Flow Rate Data	R. D. McCormick	EG&G	MLR
3-44[c]	TB-86-44	11/86	"B" Steam Generator Tube Sheet Fuel Estimate	J. Greenborg	GPUN	MLR
3-45[c]	TB-86-49	12/86	Impacts of Core Drilling on the Reactor Coolant System	R. Rainish	GPUN	MLR
4-1	No Number	No Date (1979)	Three Mile Island Nuclear Station Unit 1 and 2 Radioactive Effluent Release Report for January 1-June 30, 1979	Unidentified	TBD	
4-2	Memorandum	04/79	Preliminary Estimates of Radioactivity Releases from Three Mile Island	L. H. Barrett	NRC	
4-3	PGC-TR-171 *	04/79	Interim Report on the Three Mile Island Nuclear Station Offsite Emergency Radiological Environmental Monitoring Program	Unidentified	?	
4-4	EML-357	05/79	Radiation Measurements Following the Three Mile Island Reactor Accident	K. M. Miller	EML	
4-5	Letter	05/79	Monitoring Activities of the Department of Health, Education and Welfare in Support of the Three Mile Island Nuclear Incident for the Period March 28-April 15, 1979	J. C. Villforth	HEW	
4-6[c]	NUCON 6MT611/04	05/79	Analysis of the Absorbers and Absorbents from Three Mile Island Unit #2	Unidentified	NCSI	MLR
4-7	No Number Memo Report	05/79	Water Inventory as of 0800, 3/30/79	S. Lamana	MEC	
4-8[c]	NUREG-0558	05/79	Population Dose and Health Impact of the Accident at the Three Mile Island Nuclear Station	ADHOC Population Dose Assessment Group	NRC, EPA and HEW	(Langer)
4-9	No Number--GPU Micro #COR-0466.00	05/79	Isotope Inventory Balance	J. D. Phinney	B&W	
4-10	No Number--GPU Micro #PC-0001.02	05/79	Bleed Tanks and Pressurizer Sample Results	J. D. Phinney	B&W	
4-11	No Number--GPU Micro #PC-0001.02	05/79	Strontium and Gamma Isotopic Analyses	J. D. Phinney	B&W	
4-12	No Number--GPU Micro #COR-468	05/79	Reactor Coolant Sample and "A" Bleed Tank Sample Result	J. D. Phinney	B&W	
4-13	Memorandum	06/79	Radioactive Gases Released from TMI on the Morning of March 30, 1979	C. O. Gallina to A. F. Gibson		
4-14	NUCON 6MT611/09	06/79	Analysis of the Absorbers and Absorbents from Three Mile Island Unit 2	Unidentified	NCSI	
4-15	WAPD-RC/E(TR)-160	06/79	Summary Report of Radiological Assistance Team Actions: Three Mile Island Accident	Unidentified	BAPL	
4-16	NSSR CONFerence	06/79	Report on Preliminary Radioactive Airborne Release and Charcoal Efficiency Data: TMI-2	J. T. Collins, et al.	?	

Information Category[a]	Report Number	Publication Date	Title	Author Name	Author Company[b]	SARE Custodian[d]
4-17	Unpublished	06/79	Plan for Decontamination of Auxiliary and Fuel Handling Buildings	J. F. Remark, et al.	?	
4-18	No Number	07/79	Harrisburg PA Milk Results	Unidentified	EPA-EMSL-LV	
4-19	No Number	07/79	Harrisburg PA Water Results	Unidentified	EPA-EMSL-LV	
4-20	GPU-TDR-TMI-103	07/79	Primary and Secondary Coolant Analysis	R. V. Furio P. A. Zanis	GPU	
4-21	NUREG/CR-0913	07/79	Generation of Hydrogen During the First Three Hours of the Three Mile Island Accident	R. K. Cole	SNL	
4-22	TDR-TMI-116	07/79	Assessment of Offsite Radiation Doses from the Three Mile Island Unit 2 Accident	Unidentified	PL&G	
4-23	No Number (draft report)	07/79	Estimate of External Whole-Body Radiation Exposure to Population Around TMI Nuclear Power Station	S. P. Hull	BNL	
4-24	NUREG-0591	08/79	Environmental Assessment, Use of EPICOR-II at Three Mile Island Unit 2	Unidentified	NRC	
4-25	ORNL/TM-7044	08/79	Involvement of ORNL Chemical Technology Division in Contaminated Air and Water Handling at TMI	R. E. Brookshank L. J. King	ORNL	
4-26	No Number	08/79	Effluent Releases from TMI Units 1 and 2 for First and Second Quarter	Not Identified	P-GC	
4-27	B&R-GPU-R-026	10/79	Reactor Building Free Volume Calculation	A. S. Dam	B&R	
4-28	No Number	10/79	Report of the Task Group on Health Physics and Dosimetry to President's Commission	J. A. Auxier, et al.		(Langer)
4-29	No Number	10/79	Technical Staff Analysis Report on Transport of Radioactivity from the TMI-2 Core to the Environs to President's Commission on the Accident at Three Mile Island	N. Lauroski		
4-30	NUCON 6MT611/13	10/79	Summarized Postaccident TMI Unit 2 HVAC Absorber Systems Sample Data	Unidentified	NCSI	
4-31	No Number	11/79	TMI-2 Power History, Isotopic Analysis, LOR-2, Version 2		B&W	
4-32	GPU-TDR-059	02/80	Offsite Radiation Release	K. Woodard	GPUN	(Knauts)
4-33	ORNL-TM-7081	02/80	Postaccident Cleanup of Radioactivity at the Three Mile Island Nuclear Power Station	R. E. Brookshank W. J. Armento	ORNL	MLR
4-34	GPU-TDR-073	02/80	Deposition Activity at the 347' Elevation from Gamma Measurements in Penetration R-626	S. R. Blazo E. Walker	GPUN	
4-35	EPRI-NP-1389	04/80	131-I Studies at TMI-2	J. E. Cline, et al.	SAI	
4-36	R-80-012	05/80	NSAC EPRI Origen Code Calculation of TMI-2 Fission Product Inventory	R. G. Canada	TEC	
4-37	NUREG-0062 Volume 1 and 2	05/80	Final Environmental Assessment for Decontamination of the Three Mile Island Unit 2 Reactor Building Atmosphere--Final NRC Staff Report	Unidentified	NRC	
4-38	GPU-TDR-112	05/80	Postaccident Sampling and Analysis of the TMI-2 Reactor Building Atmosphere	J. Tate T. C. Menzel	GPUN	
4-39	CONF-800403 ANS/ENS Symposium	06/80	Fission Product Release from the Fuel Following the TMI-2 Accident	W. N. Bishop, et al.	TBD	

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				Name	Company[b]	
4-40	TIO-11277	07/80	Compilation of Chemistry Results for TMI-2 Reactor Coolant System			
4-41	GPU-TDR-182	09/80	Reactor Building Purge--Analysis of the Measurement of Vented Activity	P. J. Babel T. C. Menzel	GPUN	
4-42	No Number	09/80	Environmental Radioactivity at the TMI Venting Phase	Unidentified	EPA	
4-43	GPU-TDR-162	09/80	Postaccident Sampling and Hazardous Gas Analysis of TMI-2 Reactor Building Atmosphere for Support of Reactor Building Entry	J. W. Langenbach	TBD	
4-44	SAI-139-80-573-LJ	10/80	Meas. of 129-I and Radioactive Particulate Concentrations in TMI-2 Containment Atmosphere During and After the Venting	J. E. Cline, et al.	SAI	
4-45	GPU-TDR-071	10/80	Postaccident Plateout Measurements of the Hydrogen Recombiner Spool Piece	J. Tate T. C. Menzel	GPUN	
4-46	TIO 1225	10/80	Auxiliary Building Sump Sample Analytical Results	L. C. Rogers	B&W	
4-47	No Number	11/80	Metropolitan Edison's Environmental Monitoring Activities Conducted During the Krypton-85 Venting at Three Mile Island Unit 2	W. E. Reithle	GPUN	
4-48	Trans of ANS Volume 35	11/80	The EPA's Radiation Monitoring and Surveillance Activities During the Purging of TMI-2	E. W. Bretthauer, et al.	US-EPA	
4-49	No Number (ANS Trans Volume 35)	11/80	Monitoring Krypton-85 During TMI-2 Purging Using the Penn State Noble Gas Monitor	W. A. Jester A. J. Baratta	PSU-NED	
4-50	ANS Trans Volume 35	11/80	A Citizen's Radiation Monitoring Program for the TMI Area	A. J. Baratta, et al.		
4-51	ANS Trans Volume 35	11/80	The Management of KR-85 by a Community Monitoring Program	M. A. Reilley		
4-52[c]	LMF-70	01/81	Characterization of an Aerosol Sample from TMI Reactor Auxiliary Building	J. B. Knauer G. M. Kanapilly	ORNL LB&ERI	(Akers)
4-53[c]	RE-P-81-015	01/81	Preliminary Investigation of Feasibility of Gamma Spectra/Neutron Counting Techniques to Locate and Characterize TMI-2 RCS Fuel Debris	C. V. McIsaac	EG&G	(Akers)
4-54	CONF-801038	02/81	Studies of Airborne Iodine at TMI-2	J. E. Cline, et al.	SAI	
4-55	CONF-8010138	02/81	Investigations into the Air Cleaning Aspects of the TMI Accident	R. R. Bellemy	TBD	
4-56	FDA 81-8142	02/81	Use of Photographic Film to Estimate Exposure Near the Three Mile Island Nuclear Power Station	R. E. Shuping		MLR Abstract
4-57[c]	GEND-013	03/81	TMI-2 Reactor Building Purge--KR-85 Venting	L. J. Kripps	TBD	
4-58[c]	GEND-009	04/81	Measurements of 129-I and Radioactive Particulate Concentrations in the TMI-2 Containment Atmosphere During and After Venting	J. E. Cline, et al.	SAI	MLR
4-59[c]	GEND-005 EGG-PHYS-5337	05/81	Characterization of the Three Mile Island Unit-2 Reactor Building Atmosphere Prior to the Reactor Building Purge	J. K. Hartwell, et al.	EG&G	MLR
4-60	SAD 527-A Rev. 2	05/81	Division III System Design Description for Submerged Demineralizer System for TMI Unit II Recovery		GPUN	
4-61[c]	GEND-INF-001	06/81	Quick Look Report Entry 1 TMI-2 July 23, 1980	Not Identified	BNI GPUN	MLR
4-62[c]	GEND-INF-011	07/81	First Results of TMI-2 Sump Samples Analyses--Entry 10	D. H. Meikrantz	EG&G	MLR

Information Category[a]	Report Number	Publication Date	Title	Author: Name	Author: Company[b]	SA&E[d] Custodian
4-63	GPU TDR-055	07/81	Pathways for Transport of Radioactive Material Following the TMI-2 Accident	L. Flaherty J. Paridiso	EI GPUN	MLR
4-64	GEND-INF-008	07/81	Quick Look Report on HP-RT-211 Multivalued Behavior	J. W. Mock, et al.	EG&G, UE&C and SNL	MLR
4-65	SAI-139-081-07-RV	08/81	Gamma Scan of TMI-2 Reactor Coolant Bleed Tanks	J. E. Cline, et al.	SAI	
4-66[c]	GEND-INF-005	08/81	Quick Look Report Entry 5, TMI-2 December 11, 1980	Not Identified	BNI GPUN	MLR
4-67[c]	GEND-INF-006	08/81	Quick Look Report Entry 6, TMI-2 February 3 and 5, 1981	Not Identified	BNI GPUN	MLR
4-68[c]	GEND-INF-007	08/81	Quick Look Report Entry 7, TMI-2 March 17, 19 and 20, 1981	Not Identified	BNI GPUN	MLR
4-69	No Number	09/81	Analysis of TMI-2 Paint Chip Samples	Unidentified	SAI	
4-70	TIO-6565	09/81	Curie Estimate Basement Sample	C. V. McIsaac	EG&G	
4-71[c]	SAI-139-81-02-RV	09/81	Radionuclide Mass Balance of TMI-2 Accident	J. A. Daniel	SAI	(Akers)
4-72	SAI-139-81-01	09/81	Measurement of Surface Contamination Levels on Designated Floor Areas on Elevation 305', TMI-2 Reactor Building	E. D. Barefoot, et al.	SAI	MLR
4-73[c]	GEND-006	10/81	Color Photographs of the Three Mile Island Unit 2 Reactor Containment Building: Volume I--Entries 1, 2, 4, 5, 6	G. R. Eidam J. T. Moran	GPUN	MLR
4-74	GEND-14	10/81	Examination Results of TMI Radiation Detector HP-R-211			
4-75	SAI-139-81-06-RV	11/81	Measurement of Contamination on Containment Coolers C, D and E and Surface Contamination on a Designated Floor Area on Elevation 305 ft, TMI Reactor Building	D. S. Cameron, et al.	SAI	
4-76	NSAC-30	11/81	Iodine-131 Behavior During the TMI-2 Accident	C. A. Pelletier	SAI	
4-77	GEND-INF-017 Volume I	11/81	Field Measurements and Interpretations of TMI-2 Instrumentation: CF-1-PT3	J. E. Jones, et al.	TEC	
4-78	GEND-INF-017 Volume II	11/81	Field Measurements and Interpretations of TMI-2 Instrumentation: CF-1-PT4	J. E. Jones, et al.	TEC	
4-79	RE-P-82-007	01/82	TMI-2 Monthly Report for January, 1982	C. V. McIsaac D. D. Simpson	EG&G	
4-80	GEND-INF-017-3	01/82	Field Measurements and Interpretation of TMI-2 Instrumentation HP-R-211 (Radiation Monitor)	Jones, et al.	TEC	(Knauts)
4-81[c]	GEND-INF-019	01/82	Estimated Source Terms for Radionuclides and Suspended Particulates During TMI-2 Defueling Operations			
4-82	GEND-INF-017-6	01/82	Field Measurements and Interpretation of TMI-2 Instrumentation: IC-10-DPT (CRDMs bypass flow)	Jones, et al.	TEC	(Knauts)
4-83	GEND-INF-017-8	01/82	Field Measurements and Interpretation of TMI-2 Instrumentation: HP-R-212 (Radiation Monitor)	Jones, et al.	TEC	(Knauts)
4-84	GEND-INF-017-9	01/82	Field Measurements and Interpretation of TMI-2 Instrumentation: HP-R-213 (Radiation Monitor)	Jones, et al.	TEC	(Knauts)
4-85	LRC-5266 TIO-10773	02/82	Analysis of TMI-2 Makeup Filter MU-F-513 Debris	V. B. Subrahmanyam	B&W	
4-86[c]	GEND-INF-009	02/82	Pre-Decontamination Gamma-Ray Surface Scans in TMI-2 Containment Building 305' Elevation	E. D. Barefoot, et al.	SAI	MLR
4-87	7132-82-167	03/82	Investigation of TMI Hydrogen Phenomena of March 28, 1979	TBD	GPUN	

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				Name	Company[b]	
4-88	SAI-139-82-05-RV	04/82	Pre- and Postdecontamination Gamma-Ray Scans of TMI-2 Containment Surfaces, Elevations 305 and 347 Feet	E. D. Barefoot, et al.	SAI	
4-89	GEND-INF-017 Volume 10	04/82	Field Measurements and Interpretation of TMI-2 Instrumentation: HP-R-214	J. E. Jones, et al.	TBD	
4-90[c]	GEND-INF-021	05/82	Analysis Data on Samples from the TMI-2 Reactor Coolant System and Reactor Coolant Bleed Tank	R. L. Nitschke	EG&G	
4-91	ED-E3-82-017	06/82	Current Status and Accident Presentation of Containment Air RTD's	Mock	EG&G	(Knauts)
4-92	GEND-INF-023 Volume I	06/82	Investigation of Hydrogen Burn Damage in the Unit 2 Reactor Building	N. J. Alvarez, et al.		
4-93	RE-P-82-067	07/82	TMI-2 Monthly Report for July, 1982	C. V. McIsaac O. D. Simpson	EG&G	
4-94[c]	GEND-INF-023 Volume II	08/82	Estimated Temperatures in the TMI-2 Containment Building During the 1979 Accident	H. W. Schutz P. K. Nagata	EG&G	
4-95[c]	GEND-015	08/82	Characterization of EPICOR-II Prefilter Liner 16	J. D. Yesso, et al.	BCL	MLR
4-96	SAI-139-82-12-RV	09/82	Preliminary Radioiodine Source Term and Inventory Assessment for TMI-2	C. A. Pelletier, et al.	SAI	
4-97	GEND-INF-011, Volume II	10/82	Reactor Building Basement Radionuclide Distribution Studies	T. E. Cox, et al.	EG&G	
4-98	SAI-139-82-14 RV	10/82	Characterization of Contaminants in TMI-2 Systems--Interim Report	J. A. Daniel, et al.	SAI	MLR
4-99[c]	RE-P-82-095	10/82	Calibration of Two Surface Samplers for Collecting (12/81 and 3/82) Samples from TMI-2 RB Concrete and Steel Surfaces	C. V. McIsaac	EG&G	(Akers)
4-100[c]	RE-P-82-111	11/82	Estimated Exposure Rates and Inventories for TMI Makeup and Purification System Demineralizers A and B	D. E. Wessol, et al.	EG&G	(Akers)
4-101	GEND-INF-029 Volume I	11/82	TMI-2 Pressure Transmitter Examination Program Year End Report: Examination and Evaluation of Pressure Transmitters CF-1-PT3 and CF-2-LT3	F. T. Soberano		
4-102	ANS Meeting	11/82	Characterization of Fission Product Deposition in the TMI-2 Reactor Coolant and Auxiliary Systems	J. A. Daniel J. C. Cunnane	TBD	
4-103	ANS Winter Meeting	11/82	Processing of the TMI-2 Reactor Building Sump and the Reactor Coolant System	K. J. Hofstetter C. G. Hitz	GPUN	
4-104	ANS Winter Meeting	11/82	Fission Product Transfer in the TMI-2 Purification System	T. E. Cox	EG&G	
4-105	GEND-19	11/82	Examination Results of TMI Radiation Detector HP-R-213			
4-106	TPO/TMI-027	11/82	Data Report on Reactor Building Basement--History and Present Conditions	TBD	GPUN	
4-107	RE-P-82-124	12/82	Gamma Scans of TMI-2 Makeup and Purification Filters and Associated Vacuum Filters	R. L. Nitschke	EG&G	(Akers)
4-108	HEDL-7285 HEDL-SA-2834	12/82	Fuel Content of TMI-2 Unit 2 Makeup Demineralizers	J. P. Neece, et al.	HEDL TIO-16217	
4-109	TIO-15810	12/82	Analysis of TMI-2 Reactor Coolant Bleed Tank "A" Sludge Sample	J. J. McCown	W	
4-110	GPU-TDR-031	12/82	Reactor Building Radiation Characterization	R. Gardner	GPUN	(Knauts)

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4-111	LA-9622-MS	01/83	TMI-2 Fission-Product Element and Isotopic Inventories	T. R. England W. B. Wilson	TBD	
4-112	TPO/TMI-034	01/83	Technical Plan for Sludge Removal from Elevation 282'6"			
4-113	TPO/TMI-035	01/83	Technical Plan for TMI-2 Core Accountability	TBD	GPUN	
4-114[c]	GEND-INF-019 Volume II	02/83	Estimated Source Terms for Radionuclides and Suspended Particulates During TMI-2 Defueling Operations	P. G. Voilleque, et al.	SAI	MLR
4-115[c]	EGG-TMI-6181	02/83	Interim Report on the TMI-2 Purification Filter Examination	R. E. Mason, et al.	EG&G, LANL & ANL	MLR
4-116[c]	GEND-031	02/83	Submerged Demineralizer System Processing of TMI-2 Accident Waste Water	H. F. Sanchez, et al.	TBD	
4-117	NUS-TM-347	03/83	Pipe Volume Identification for Systems Utilized in Radioactive Material Transfer During the TMI-2 Accident	D. W. Tonkay	NUS	
4-118	NUS-TM-348	03/83	Shapes and Volumes of Components in the TMI-2 Makeup and Purification System and the Reactor Building Basement	R. J. Davis	NUS	
4-119[c]	GEND-028	03/83	Preliminary Radioactive Source Term and Inventory Assessment for TMI-2	C. A. Pelletier, et al.	TBD	
4-120[c]	TPO/TMI-043 Rev. 0	03/83	Radioactive Waste Management Summary Review	G. Worku	GPUN	MLR
4-121	83-095 (Letter)	03/83	TMI-2 Curie Inventory	J. A. Daniel	SAI	
4-122	EPRI-2922	03/83	Characterization of Containments in TMI-2 System--Interim Report		EPRI	
4-123	GEND-INF-023 Volume IV also RHO/GR/SA/BP	03/83 10/82	Analysis of the Three Mile Island (TMI-2) Hydrogen Burn	J. O. Henrie A. K. Postma	RI	
4-124[c]	GEND-027	04/83	Characterization of EPICOR-II Prefilter Liner 3	N. L. Wynhoff V. Pasupathi	BCL	MLR
4-125	TPO/TMI-027	04/83	Reactor Building Basement--History and Present Conditions	G. Worten, et al.	GPUN & BNI	
4-126[c]	GEND INF-032 Volumes I and 2	04/83	Radionuclide Mass Balance for the TMI-2 Accident: Data Base System and Preliminary Mass Balance	M. I. Goldman, et al.	NUS	MLR
4-127	Manuscript Submittal to Nuclear Technology	04/83	Engineering Analysis of Letdown Data Taken During Primary Coolant Cleanup at Three Mile Island	K. J. Hofstetter, et al.	GPUN & ORNL	
4-128[c]	NUS-4350-Draft	04/83	TMI-2 Technology Transfer: Task 2--Demonstration Utilization of Source Term Data	R. R. Sherry, et al.	NUS	MLR
4-129	NUS-TM-349	04/83	EPICOR II and SDS Influent Sources and Concentrations	T. Lookabill	NUS	
4-130	GEND-INF-030	04/83	Analysis of Air Temp Measurements from TMI-2 Reactor Building	Fryer	EG&G	(Knauts)
4-131	SD-WM-TI-067	05/83	Analysis of TMI-2 Transients	J. O. Henrie A. K. Postma	RI	
4-132[c]	GEND-INF-011 Volume III	06/83	Reactor Building Basement Radionuclide Distribution Studies	T. E. Cox, et al.	EG&G	MLR
4-133[c]	NUS-TM-352	06/83	Information on Reactor Building Surface Contamination	R. J. Davis	NUS	MLR
4-134	Third Symposium on Separation Science and Technology for Energy Application	06/83	The Use of the Submerged Demineralizer System at Three Mile Island	K. J. Hofstetter C. G. Hitz	GPUN	

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4-135[c]	GEND-INF-039	06/83	Final Analysis on TMI-2 Reactor Coolant System and Reactor Coolant Bleed Tank Samples	T. E. Cox, et al.	EG&G	MLR
4-136[c]	Hmb-207-83 Letter	06/83	Purification Demineralizer Resin Samples	A. P. Malinauskas	ORNL	MLR
4-137	LA-9795-MS	08/83	NDA Measurement of Demineralizers at TMI-2	J. R. Phillips	LANL	
4-138[c]	NUS-4432 Volume 2	09/83	Radionuclide Mass Balance for the TMI-2 Accident: Data through 1979 and Preliminary Assessment of Uncertainties--Appendix C TMI-2 Mass Balance: Data Base	R. J. Davis, et al.	NUS	MLR
4-139[c]	NUS-TM-351	09/83	Information on Radioactivity in Solids for Inclusion in the TMI-2 Mass Balance Data Base	R. J. Davis	NUS	MLR
4-140[c]	NUS-TM-354	09/83	TMI-2 Mass Balance Chronology Extension Calculations	D. W. Tonkay E. A. Vissing	NUS	MLR
4-141[c]	GEND-033	09/83	The Use of Multi-Element Beta Dosimeters for Measuring Dose Rates in the TMI-2 Containment Building	R. I. Scheppelz	PNL	MLR
4-142[c]	NUS-TM-350	09/83	Information on Gaseous Radioactivity for Inclusion in the TMI-2 Mass Balance Data Base	R. J. Davis	NUS	MLR
4-143[c]	GEND-INF-023 Volume III	09/83	Data Integrity Review of Three Mile Island Unit 2 Hydrogen Burn Data	J. K. Jacoby, et al.	EG&G	MLR
4-144	HEDL-TC-2492	10/83	Solid State Track Recorder Neutron Dosimetry Measurements for Fuel Debris Location in the Three Mile Island Unit-2 Makeup and Purification Demineralizer	F. H. Ruddy, et al.	HEDL	
4-145[c]	GEND-037	10/83	Surface Activity and Radiation Field Measurements of the TMI-2 Reactor Building Gross Decontamination Experiment	C. V. McIsaac	EG&G	
4-146	HEDL-7285	10/83	Fuel Assessment of Three Mile Island Unit-2 Makeup Demineralizers by Compton Recoil Continuous Gamma Ray Spectroscopy	McNeece, et al.	HEDL	
4-147	GPU-TDR-082	12/83	Airborne Recontamination Studies	Tarpenian Furio	GPUN	(Knauts)
4-148[c]	EGG-TIO-M00784	00/84	TMI-2 Plant Demineralizer Sample Analysis	J. D. Thompson	EG&G	MLR
4-149	TPO/TMI-107	02/84	Evaluation of Concrete Borings from Reactor Building	Unidentified	GPUN & BNI	
4-150	GEND-INF-049	03/84	Examination Results of TMI Radiation Detector HP-R-212	G. M. Mueller		
4-151	TPO/TMI-110	03/84	Data Report on Underhead Data Acquisition Program		GPUN	
4-152	GEND-INF-029 Volume II	04/84	TMI-2 Pressure Transmitter Examination and Evaluation of CF-1-PT2, CF-2-LT1 and CF-2-LT2	M. E. Yancey R. C. Strahm	EG&G	
4-153	GEND-INF-013	05/84	TMI-2 Purification Demineralizer Resin Study	J. D. Thompson T. R. Ousterhoudt	TBD	
4-154[c]	GEND-INF-054	06/84	Results of Analyses Performed on Concrete Cores Removed from Floors and D-Ring Walls of the TMI-2 Reactor Building	C. V. McIsaac, et al.	EG&G	
4-155	ANS Symposium	07/84	TMI-2 Radiocesium Behavior	R. A. Lorenz, et al.	ORNL	
4-156	EGG-TMI-6580	09/84	TMI Particle Characterization Determined from Filter Examinations--Draft	C. S. Olsen, et al.	EG&G, LANL & ANL-E	MLR
4-157	EPRI-NP-3694	09/84	Characterization of Contaminants in TMI-2 Systems	J. A. Daniel, et al.	SAI	MLR
4-158	CONF-840914-22	09/84	Tellurium Behavior During and After the TMI-2 Accident	K. Vinjamuri, et al.	EG&G	MLR

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4-159[c]	EGG-TMI-6580	09/84	TMI Particle Characterization Determined from Filter Examinations	C. S. Olsen, et al.	EG&G, LANL, ANL-E	MLR
4-160[c]	EGG-TMI-6701	09/84	Tellurium Release and Deposition During the TMI-2 Accident	K. Vinjamuri, et al.	EG&G	MLR
4-161[c]	GEND-042	10/84	TMI-2 Reactor Building Source Term Measurements: Surfaces and Basement Water and Sediment	C. V. McIsaac D. G. Keefer	EG&G	MLR
4-162[c]	GEND-INF-047 NUS-4432, Volume 1	11/84	Radionuclide Mass Balance for the TMI-2 Accident: Data through 1979 and Preliminary Assessment of Uncertainties	R. J. Davis, et al.	NUS	MLR MLR
4-163[c]	EGG-10282-1009	12/84	Airborne Cloud Tracking Measurements During the Three Mile Island Nuclear Station Accident	R. H. Beers, et al.	EG&G-EM	MLR
4-164[c]	No Number	12/84	Characterization of TMI-2 Auxiliary Building Sump and Sump Tank Radwaste	J. A. Wilson, et al.	B&W	MLR
4-165[c]	TPO/TMI-125 (Volumes 1 and 2)	01/85	Data Report on Reactor Building Radiological Core Characterization	?	GPUN	
4-166[c]	TPB-85-10	04/85	Estimates of TMI-2 Letdown Demineralizer Resin Retained and Eluted Fission Products and Fuel	T. E. Cox	GPUN	MLR
4-167	EPRI-NP-3975[d]	04/85	Analysis of the Hydrogen Burn in the TMI-2 Containment	R. G. Zalosh	FMRC	MLR[d]
4-168	Nuclear Technology Volume 69	04/85	Circulation within the Primary System at TMI-2 with Lowered Coolant Level and Atmospheric Conditions	V. F. Baston, et al.	PSI	
4-169	TPO/TMI-050	05/85	Planning Study on System Options and Requirements for Locating Fuel in TMI-2		GPU-TPO	
4-170	CONF-850417-18[d] ACS Symposium	05/85	Cleanup of TMI-2 Demineralizer Resins	W.D. Bond, et al.	GPUN	MLR
4-171[c]	ACS Symposium	05/85	TMI-2 Reactor Building Source Term Measurements	C. V. McIsaac D. G. Keefer	EG&G EG&G	MLR
4-172	TB-85-08 Rev. 1	05/85	Reactor Building Basement Fuel Estimate	TBD	GPUN	
4-173	EPA-600/4-85-042	06/85	Monitoring the Venting of Three Mile Island: Report of an Evaluation Workshop	Not Identified	USEPA-EMSL	LIB
4-174[c]	TPO/TMI-043 Rev. 4	08/85	Radioactive Waste Management Summary Review	I. Igarashi	GPUN	MLR
4-175[c]	GEND-INF-063	08/85	Analysis of the TMI-2 Dome Radiation Monitor	M. B. Murphy, et al.	SNL	MLR
4-176	EGG-PBS-6798	08/85	TMI-2 Isotopic Inventory Calculation	B. G. Schnitzler J. B. Briggs	EG&G	
4-177[c]	TPO/TMI-176	09/85	Cesium Elution of Makeup and Purification Demineralizer Resins	K. J. Hofstetter	GPUN	MLR
4-178[c]	ENERGEX R85-009	09/85	Review of Severe Accident Issues Which Relate to Fission Product Behavior at TMI-2	H. A. Mitchell, et al.	EAI	MLR
4-179[c]	TB 85-35	10/85	Robotic Sediment Sampling	R. Brosey	GPUN	MLR
4-180[c]	TB 85-33	11/85	Makeup Pump Room Reactor Fuel Quantification	P. Babel	GPUN	MLR
4-181[c]	TB 85-34	11/85	Urgent D-Ring Decontamination Problems and Technique Alternatives	H. P. Wood	GPUN	MLR
4-182[c]	TB 85-041	12/85	Preliminary Radiological Surveys of Concrete Cores Removed from Reactor Building Basement Walls	H. P. Wood	GPUN	MLR
4-183	GEND-INF-029 Volume III	12/85	Examination and Evaluation of TMI-2 Transmitters CF-1-PT4 and CF-2-LT4	M. E. Yancey R. C. Strahm	EG&G	

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4-184	ACS Symposium Series 293	00/86	Water Chemistry: The Three Mile Island Accident Diagnosis and Prognosis	K. J. Hoffstetter V. F. Baston	GPUN PSI	
4-185[c]	TB-85-35 Rev. 1	01/86	Robotic Sediment Sampling	T. E. Cox	BNI	MLR
4-186[c]	TB-86-08	02/86	Makeup Tank Room (AX116) Fuel Quantification	P. Babel	GPUN	MLR
4-187[c]	TB-86-10	02/86	"B" Steam Generator TLD Characterization	B. H. Brosey M. W. Lambert	GPUN	MLR
4-188[c]	TB-85-11	02/86	Autoradiography of Concrete Cores	C. M. Davis	GPUN	MLR
4-189[c]	TB-86-5	02/86	Reactor Building Concrete Core Samples	T. E. Cox	BNI	MLR
4-190[c]	TB-86-06	02/86	Makeup Suction Valve Room (FH-001) Fuel Quantification	P. J. Babel	GPUN	MLR
4-191[c]	TB-86-07	02/86	Makeup Discharge Valve Rooms (FH-003A and FH-003B) Fuel Quantification	P. J. Babel	GPUN	MLR
4-192	TB-86-21	03/86	Makeup Valve Room (FH-101) Fuel Quantification	P. J. Babel	GPUN	MLR
4-193[c]	TB-86-20	03/86	Radiation Mapping System	R. D. Shauss	GPUN	MLR
4-194[c]	TB-86-5 Rev. 1	04/86	Reactor Building Concrete Core Samples	P. J. Babel	GPUN	MLR
4-195[c]	TB-86-18	04/86	RB Basement Thermoluminescent Dosimeter (TLD) Comparison Study	B. Brosey	GPUN	MLR
4-196[c]	TB-86-26	05/86	Letdown Cooler Room Fuel Quantification	P. J. Babel	GPUN	MLR
4-197[c]	TB-86-27	05/86	Preliminary Leaching Data for Concrete Bores	P. J. Babel	GPUN	MLR
4-198[c]	TB-86-30	06/86	Reactor Building Basement Wall & Floor Gamma Measurements	B. Brosey	GPUN	MLR
4-199[c]	TB-86-05 Rev. 2	06/86	Reactor Building Concrete Core Samples	B. H. Brosey	GPUN	MLR
4-200[c]	TB-86-31 Rev. 0	06/86	Volume of Sediment Inside Secondary Shield	W. P. Wood	GPUN	MLR
4-201[c]	TPO/TMI 043 Rev. 5	06/86	Radioactive Waste Management--Summary Review	H. Igarashi	GPUN	MLR
4-202[c]	No Number	08/86	Radioanalytical Report (RB Basement Sludge)	Not Identified	SAI	MLR
4-203	ACS Symposium, ACS-293	08/86	TMI-2 Reactor Coolant System Radionuclide Accumulation Rates	V. F. Baston K. J. Hoffstetter	PSI GPUN	
4-204[c]	EGG-2407	09/86	Fission Product Inventory Program FY-85 Status Report	S. Langer, et al.	EG&G	MLR
4-205[c]	None	09/86	Radiation and Health Effects--A Report on the TMI-2 Accident and Related Health Studies	V. H. Behling J. E. Hildebrand	GPUN	RKM
4-206	TB-86-38 Rev. 0	09/86	Summary of Fuel Quantities External to the Reactor Vessel		GPUN	
4-207[c]	TB-86-30 Rev. 3	10/86	Reactor Building Basement Wall Gamma Measurements	B. H. Brosey	GPUN	MLR
4-208[c]	TB-86-41	10/86	Cerium 144 as a Tracer for Fuel Debris	J. Greenborg	GPUN	MLR
4-209	ACS Symposium	10/86	Fission Product Behavior in the TMI-2 Core: Preliminary Evaluation of Transport and Chemistry	D. W. Akers, et al.	EG&G	MLR
4-210[c]	TB-86-30 Rev. 4	11/86	Reactor Building Basement Wall Gamma Measurement	B. H. Brosey	GPUN	MLR
4-211[c]	TB-86-5 Rev. 3	11/86	Reactor Building Concrete Core Samples	R. E. Lancaster	GPUN	MLR
4-212[c]	TB-86-5 Rev. 4	12/86	Reactor Building Concrete Core Samples	R. E. Lancaster	GPUN	MLR
4-213[c]	TB-86-48	12/86	Cleanup Filters (WDL-F6A and B and WDL-F9A and B) Fuel Quantification	P. J. Babel	GPUN	MLR
4-214[c]	TB-86-47	12/86	Decay Heat Vaults (AX-501 and AX-502) and RB Spray Vaults (AX-503 and AX-504) Fuel Quantification	P. J. Babel	GPUN	MLR
4-215[c]	TB-86-46	12/86	Assessment of RB Basement Postdecontamination Exposure Rates	B. H. Brosey	GPUN	MLR
4-216[c]	TB-86-27 Rev 2	12/86	One Hundred Twenty-Five Day Leaching Data for Basement Concrete Cores	C. H. Distenfeld	GPUN	MLR

Information Category[a]	Report Number	Publication Date	Title	Author Name	Author Company[b]	SAMG[d] Custodian
4-217	Report	No Date	EPA Monitoring After the Three Mile Island Reactor Incident	Unidentified	EPA	
4-218	NUCON 6NT611/12	No Date	Iodine-131 Removal Efficiency Determination of Absorbent Samples	Unidentified	NCSI	
4-219	Letter	No Date	Summary of Radiological Assistance Team Actions Three Mile Island Incident	M. B. LeBoeuf		
4-220F	ORNL/TM-9666 Draft	No Date	The Behavior of Fission Product Cesium in the TMI-2 Accident	R. A. Lorenz, et al.	ORNL	MLR

a. Information Categories:

1-General TMI-2 Accident;
2-Reactor Vessel Region Examination;
3-RCS Region Fission Product Inventory Examinations, and
4-EX-RCS Fission Product Inventory Examinations (includes General FPI [Source Term] References).

b. Company list

A&MS	Analysis and Measurement Services (Knoxville, TN)
ANL	Argonne National Laboratory
ANL-E	Argonne National Laboratory--East (Chicago area)
B&R	Burns & Roe Co.
B&W	Babcock & Wilcox Co.
BAPL	Bettis Atomic Power Laboratory
BCL	Battelle Columbus Laboratory
BNI	Bechtel National, Inc.
BNL	Brookhaven National Laboratory
EAI	ENERGEX Associates, Inc.
EG&G	Edgerton, Germeshausen and Grier, Inc.--Idaho
EG&G-EM	Edgerton, Germeshausen and Grier, Inc.--Energy Measurements (Las Vegas, NV)
EI	Energy Incorporated
EML	Environmental Measurements Laboratory
EPA	United States Environmental Protection Agency
EPA-EMSL-LV	United States Environmental Protection Agency Environmental Monitoring and Support Laboratory--Las Vegas, NV
EPRI	Electric Power Research Institute
FAI	Fauske and Associates Inc.
FMRC	Factory Mutual Research Corp. (Norwood, MA)
GPUN	General Public Utilities Nuclear
HEDL	Hanford Engineering and Development Laboratory
HEW	United States Department of Health, Education and Welfare
JCPL	Jersey Central Power and Light Co.
LANL	Los Alamos National Laboratory
LASL	Los Alamos Scientific Laboratory

Information[a] Category	Report Number	Publication Date	Title	Author Name	Author Company[b]	SA&E[d] Custodian

b. Company list (continued)

LB&ERI	Lovelace Biomedical & Environmental Research Institute, Albuquerque, NM
MEC	Metropolitan Edison Co.
MPR	MPR Associates
NAI	Nuclear Associates International
NCSI	Nuclear Consulting Services, Inc.
NRC	United States Nuclear Regulatory Commission: IE Office of Inspection and Enforcement; SIG Special Investigation Group; ONRR Office of Nuclear Reactor Regulation
NSAC	Nuclear Safety Analysis Center
NUS	Nuclear Services Corporation
NYU-MC	New York University Medical Center
ORNL	Oak Ridge National Laboratory
P-GC	Porter-Gertz Consultants
PEC	Philadelphia Electric Co.
PL&G	Pickard, Lowe & Garrick
PNL	Pacific Northwest Laboratories
PSI	Physical Sciences Incorporated
PSU-NED	Penn State University--Nuclear Engineering Dept.
Quadrex	Quadrex Corporation
RI	Rockwell International
RI-RHO	Rockwell International--Rockwell Hanford Operations
SAI	Science Applications Incorporated (Rockville, MD)
SNL	Sandia National Laboratories
TEC	Technology for Energy Corporation (Oak Ridge, TN)
TIO	EG&G Technical Integration Office
UE&C	United Engineers and Constructors
UKAEA	United Kingdom Atomic Energy Authority
W	Westinghouse

c. The publication's list of References has been used in generation of this list.

d. The document is in microfiche form.

APPENDIX B

TMI-2 SAMPLE EXAMINATION PLANS FOR CSNI

APPENDIX B

TMI-2 SAMPLE EXAMINATION PLAN FOR CSNI

PURPOSE

The purpose of this report is to identify a program of TMI-2 sample examinations for review and agreement by the Committee for the Safety of Nuclear Installations (CSNI) Joint Task Force on Three Mile Island 2. The proposed examination program (a) would be conducted by the CSNI member countries and (b) is limited to the samples that are expected to be available for shipment in late 1986. This report completes the assignment given to EG&G Idaho for a "strawman plan of examinations of samples to be included in the first shipment taking into account the interests and the experimental facilities of the organizations concerned."[a] Since the products of the examination program are examination results reports, suggestions for report format and extent are included.

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a. OECD/NEA/CSNI document SEN/SIN(86)18, Summary Record of the First Meeting held at EG&G Idaho Offices, Idaho Falls, USA on April 28 and 29, 1986.

SUMMARY

Table B-1 is a summary showing the list of sample types expected to be available for the first shipment, the samples requested by the CSNI member countries, and the quantity of samples proposed by EG&G for examination by the CSNI member countries. The proposed sample examination assignments were developed using the following criteria:

1. Member country requests for samples

2. Quantity of samples available

3. Member country gross national product for equitable sample examination distribution

4. Member country examination capabilities, facilities, and experience.

The member country requests for samples were developed from either (a) the requested CSNI member country requirements for types and number of samples,[a] types of measurements they would be able to make, experimental techniques they would use, etc., or (b) the original questionnaire responses. Sample requests have been received from the following countries or organizations since the April meeting:

- Canada

- France

- Germany

- JRC

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a. Ibid.

TABLE B-1. SAMPLE REQUESTS AND PROPOSALS

Sample	France	Germany	Canada	Japan	JRC	Sweden	Switzerland	United Kingdom
Core Bores	1/3[a]	3/2	4/3	4/4	1/1	1/1	1/5 including 2 lower core fuel rod segments	7/7 including 2 lower core fuel rod segments
Fuel Rod Segments (Peripheral)	1/1 (1/2 of #5)[b]	4/1 (#2, 1/2 to KFK 1/2 to KFA)	2/1 (1/2 of #4)[b]	2/1 (1/2 of #4)[b]	1/1 (1/2 of #5)[b]	--	--	--
Fuel Assembly Rod Segments and Control Rods (cr)	1/1	3/2 fuel rods, 2 cr/guide tubes and 1 bpr/guide tube	3/1 fuel rod	3/3 fuel rods and 1 cr/guide tube	--	--	--	7/2 upper core fuel rod segments
Upper Core Loose Debris	--	1/1-H8 surface	1/2-H8 56 cm depth and E9 surface	2/1-E9 56 cm depth	1/1-H8 36 cm depth	--	--	7/1-H8 77 cm depth
Lower Core Loose Debris	--	0/1, 11-5 FP gradient	2/2, 11-5 FP gradient and 7-1	--	--	--	--	1/1
Burnable Poison Rod Spider Upper End Boxes, and Springs	--	1/0	--	3/3	--	--	--	--

a. The number of samples requested by the CSNI member country and the number of samples suggested by EG&G is indicated with the CSNI request first (e.g., 1/3, one sample requested and three samples proposed by EG&G for CSNI examination).

b. This refers to 1/2 of the fuel rod segment.

- Japan

- Sweden

- United Kingdom.

Specific sample descriptions, TMI-2 Accident Evaluation Program information needs, and suggested examination methods are defined in the following section.

SAMPLE EXAMINATION PROGRAM

The planned TMI-2 sample examination program requires a number of nondestructive, metallurgical and radiochemical examination techniques to obtain information relevant to the TMI accident. Visual and photographic examinations are used to determine the shapes, sizes and types of particles in each sample. Neutron radiography is used to determine the internal distribution of UO_2 and control materials (i.e., Ag, In, and Cd) in fuel and control rods. Particle size distribution analyses (seiving and SEM) are used to evaluate the mobility (aerosol and/or hydrosol) of the particulate debris.

The metallurgical examinations provide data in the form of optical photographs, SEM backscatter and secondary electron images, X-ray spectra from the SEM/EDS examinations, and quantitative elemental compositions from scanning auger spectroscopy (SAS) and electron microprobe. Area fractions of the major components, grain sizes and the composition of material in the grain boundaries are needed. Temperature estimates (peak and average) are determined from the U-Zr-O ratios measured by SAS. Estimates of temperature are important to many aspects of the accident (e.g., fission product release behavior is extremely temperature dependent).

The radiochemical examinations provide quantitative data on the elemental composition and radionuclide content of individual particles, particle size groups and bulk samples. Elemental composition analysis is performed by inductively coupled plasma (ICP) spectroscopy for 18 core constituents (i.e., Ag, Al, B, Cd, Cr, Cu, Fe, Gd, In, Mn, Mo, Ni, Nb, Si, Sn, Te, U and Zr) important to understanding the TMI accident. Radionuclide analyses are performed by gamma spectroscopy (Co-60, Ru-107, Sb-125, Cs-134, Ce-144, Eu-154 and Eu-155), liquid scintillation analysis for beta emitters (i.e., Sr-90) and neutron activation analysis for fissile material and I-129 content.

Thermal ionization mass spectroscopy (TIMS) is used to determine the burnup of fuel samples to better define fission product retention as functions of neutron flux and uranium content. These data are needed to obtain information on fission product retention as functions of chemical composition and temperature. Evidence of correlations between elemental content (i.e., structural material) and radionuclide concentration has been found.

Other examinations proposed to CSNI that are being performed by U.S. laboratories include: (a) fuel melt experiments (to 3000 K) which can be used to better evaluate the fission product release behavior of the TMI fuel; and, (b) micro gamma spectrometry analysis which can be used to define radionuclide location in small samples for correlation with metallurgical analyses.

Core Bores

Sample Description

Core material samples will be obtained to determine axial, radial and azimuthal properties of the lower core region and of the material on the lower reactor head. Currently, 18 core bores will be removed and scanned using a hot cell gamma spectrometry system. Samples will then be removed based on these results. Samples retrieved from the upper vessel will be 6.3 cm in diameter by 150-210 cm long. Analysis samples will be approximately 1 cm thick.

Information Needs

The purpose of the CSNI examination of the core bores is to evaluate the physical, metallurgical, and radiochemical characteristics of the upper vessel region of the core to obtain information about the TMI accident. Information needs are:

- The nature of material stratification and relocation, materials interaction and prior peak temperatures of the materials

- Radionuclide retention in the debris and to determine the effects of maximum temperature and chemical composition on retention

- The abundance, location, and chemical form of retained fission products

- The location, extent, and chemical form of core constituents in homogeneous debris and at interaction zones

- Prior peak and average temperatures of the samples.

Specific information needs are:

1. Nondestructive information

   Photovisual characterization (color and surface texture,) of the samples is necessary for comparison with work being performed at other laboratories.

2. Metallurgical

   a. Analysis of a representative sampling of the available material for comparison with previous results should be performed. Suggested methods include: optical metallography, SEM/WDX, SAS, and electron microprobe.

3. Radiochemical

   a. Evaluate the retention of fission products in the debris as functions of burnup and chemical composition. Suggested methods include: gamma spectroscopy, radiochemistry, TIMS, electron microprobe and ICP.

   b. Analyze representative bulk and individual particle samples of the debris for radionuclide (gamma emitters, I-129, Sr-90, radiotellurium), elemental, and fission gas content for comparison with previous analyses. Suggested methods include: gamma spectroscopy, ICP, radiochemistry, TIMS, electron microprobe and high temperature furnaces.

c. Evaluate radiotellurium retention in bulk samples and individual particles. Suggested methods are TIMS, SIMS, and electron microprobe.

Fuel Assembly and Control or Burnable Poison Rod Upper End Fittings

Sample Description

The upper end fittings were loaded into TMI-2 fuel canister D-141 in December, 1985. Nondestructive examinations, which will be performed at INEL consist of confirming or obtaining the end fitting identification marking and photography to determine which end fittings may be of most interest. The end fitting inventory is expected to be as follows:

1. Fuel assembly NJ00Q8 upper end fitting with approximately 18 in. of guide tubes remaining and control rod assembly C179 spider and control rod remnants from core position C7.

2. Fuel assembly NJ00P2 upper end fitting and control rod assembly C157 spider from core position E13. Both end fittings are partially ablated.

3. Fuel assembly NJ00R1 upper end fitting and control rod assembly C167 spider from core position M9.

4. Three unidentified sets of fuel assembly upper end fittings and control or burnable poison rod assembly spiders. Fuel assembly upper end fittings are partially ablated.

5. Two unidentified, partially ablated, fuel assembly upper end fittings.

6. One identified fuel assembly upper end fitting with approximately 18 in. of guide tubes remaining.

Video surveys of TMI-2 and the LOFT FP-2 test fuel assembly upper end fittings indicate that agglomerated core material nuggets may be deposited on the upper end fittings.

Information Needs

The objective of the CSNI Examination of the fuel and control or burnable poison rod assembly upper end fittings would be to characterize the end fittings and surface deposits for the following TMI-2 accident information:

- The abundance, location, and chemical form of retained fission products.

- The abundance, location, and chemical form of core region materials that have relocated to the upper end fitting,

- The prior peak temperature,

- The melting temperature and other physical properties of prior-molten, agglomerated core material nuggets now attached to the upper end fittings.

The CSNI examination program should include nondestructive, metallurgical, radiochemical and other methods to acquire the following information:

1. Nondestructive examination information:

   a. External appearance (color, surface texture, blemishes, and deformities) of separated components or regions exposed by removing end fitting components.

      Suggested method: color photography including magnification

b. Weight of separated components to determine quality of upper end fitting material relocated during accident.

Suggested method: balance weighing.

2. Metallurgical examination information:

a. Extent of surface deposition and material interactions, peak temperatures, fission product location, and agglomerated composition.

Suggested methods:

- Optional metallography/microscopy of metallographic specimens from the base material, uniform surface deposits, core material agglomerate surface deposits and material interaction zones

- SEM-EDX and/or SEM-WDX examination of metallographic specimens

- SEM examination of core material agglomerate and material interaction zone metallographic specimens

- SIMS/microprobe examination of surface deposits and material interaction zones for fission product location.

3. Radiochemical examination information

a. Gamma-emitting fission product abundance and location.

Suggested methods:

- Gamma spectrometer analysis of dissolved samples from (1) upper end fitting uniform surface deposits and (2) core material agglomerates,

- Micro gamma spectrometry of metallographic samples of uniform surface deposits, core material agglomerates, and material interaction zones.

b. Location and abundance of metallic elements.

Suggested methods: ICP or AAS analysis of dissolved specimens of uniform surface deposits, core material agglomerates or material interaction zones.

c. Sr-90, I-129 and tellurium retention in uniform surface deposits, core material agglomerates and material interaction zones.

Suggested methods:

- Chemical separation and liquid scintillation counter processes for Sr-90,

- Chemical dissolution, separation, low-energy photon spectrometry or irradiation and neutron activation analysis for I-129,

- Tellurium chemistry (TIMS, SIMS).

4. Other Examinations

a. Melting temperature of core material agglomerates.

Suggested method: melting temperature test.

The CSNI laboratories are encouraged to use alternate or supplemental examination methods to derive the TMI-2 accident information from the upper end fittings. One test that might furnish valuable information is a test of agglomerated core material at the melting temperature and atmospheric pressure for fission product and core material release to study the effects of temperature on the core decomposition process. Similar examinations are being performed at the Battelle Columbus Laboratories.

Peripheral Fuel Rod Segments

Sample Description

The peripheral fuel rod segments are six segments of peripheral fuel assembly rods which were removed from the span of standing fuel rods between the fuel assembly upper end fitting and the next (below) spacer grid on December 22, 1985. The fuel rod segments which were removed with a shear tool, are about 6 inches long and have deformed ends. A nondestructive examination program consisting of neutron radiography, gamma spectrometer scans and photography has been performed at INEL. Four of the eleven fuel rod segment deformed ends were removed to obtain fuel rod cross section photographs. A brief description of the fuel rod segments is as follows:

- Segment 1, core position L1, fuel rod position B10 (segment with transition gone at lower end and containing approximately 7 fuel pellets)--no end removal

- Segment 2, core position L1, fuel rod position B10 (Segment 1 adjoining section with the upper 7-1/2 fuel pellets, a zircaloy-sleeve spacer and a possible remnant of the pellet stack holddown spring)--cut at the lower end of the top surface of the lowest fuel pellet.

- Segment 3, core position N2, fuel rod position N1 (segment includes upper 6-1/2 fuel pellets, a zircaloy-sleeve spacer and possible cladding-internal component interaction zones)--removed upper end at fuel pellet stack upper surface, removed lower end at top surface of the lowest fuel pellet and separated components (fuel pellets, insulating washers, sleeve spacer and cladding) from amputated deformed ends. This fuel rod was close to or at the perimeter of the core cavity.

- Segments 4 (core position M2, fuel rod position R15 with 11-1/3 fuel pellets) and 5 (core position M2, fuel rod position R13 upper region including upper 11-1/2 fuel pellets)--no end removal. This fuel rod was adjacent to the core former wall.

- Segment 6, core position M2, fuel rod position R13, (Segment 5 adjoining section with 6-1/2 fuel pellets)--removed lower end at the top surface of the bottom 1/2 pellet. This fuel rod was adjacent to the core former wall.

Information Needs

The objective of CSNI examination of the standing fuel rod segments is to characterize fuel rod material and retained fission products in the upper core region for derivation of the following TMI-2 accident information:

- The abundance, location and chemical form of retained fission products

- The location, extent and chemical form of fuel rod cladding and internal component interaction zones

- The prior peak temperature.

The CSNI examination program should include nondestructive, metallurgical, and radiochemical methods to acquire the following information:

1. Nondestructive examinations information:

   a. External appearance (color, surface texture, blemishes, and deformities) of internal components separated from the cladding.

      Suggested method: color photography and magnification

2. Metallurgical examination information:

   a. Extent of oxidation, material interactions and gross surface depositions, peak temperature and fission product location.

      Suggested methods:

      - Optical metallography/microscopy, of metallographic specimens of the rod cross sections

      - SEM-EDX and/or SEM-WDX examination of cladding-internal component interaction zones and cladding

      - SAS examination of cladding-internal component interaction zones for oxygen location and concentration

      - SIMS/electron microprobe examination of all components for fission product location.

3. Radiochemical examination information:

   a. Gamma-emitting fission product abundance and location.

      Suggested methods:

      - Gamma-spectrometer analysis of dissolved samples of fuel pellets, cladding-internal component interaction zones, cladding external and internal surface deposits,

- Micro-gamma spectrometry of metallographic samples.

b. Location and abundance of metallic elements.

Suggested methods: ICP or AAS analysis of dissolved specimens of cladding, internal component interaction zones, and cladding surface deposits

c. Sr-90, I-129 and tellurium retention in fuel pellets, cladding-fuel pellet interaction zones and cladding external and internal surface deposits.

Suggested methods:

- Chemical separation and liquid scintillation counter process for Sr-90,

- Chemical dissolution, separation, low-energy photon spectrometry or irradiation and neutron activation analysis for I-129,

- Tellurium chemistry.

d. U-235 depletion (burnup) of fuel.

Suggested method: TIMS, alpha spectrometry.

The CSNI laboratory is encouraged to use alternate or supplemental examination methods to derive the TMI-2 accident information from the fuel rod segments.

Fuel Control, and Burnable Poison Rods from Remnant Fuel Assemblies

Sample Description

The rods will be obtained from the upper end remnants of control rod fuel assembly C7 and a to-be-identified fuel assembly which were loaded into TMI-2 fuel canister D-141 in December, 1985. It is suspected that the unidentified fuel assembly is not a burnable poison rod fuel assembly. A nondestructive examination program consisting of neutron radiographs, gamma spectrometer scans, and photography will be performed at INEL to determine which rods are of the most interest. The rods may be up to 24 in. long. The remnant fuel rods will be withdrawn from the upper end fitting and the guide tube poison rod section will be separated from the upper end fitting by cutting or shearing both the control rod and its guide tube which are expected to be fused together.

Information Needs

The objectives of the CSNI examination of the remnant fuel assembly rods is to characterize the fuel rod, control or burnable poison rod, and guide tube material and retained fission products in the upper core region for derivation of the following TMI-2 accident information:

- The abundance, location, and chemical form of retained fission products,

- The location, extent and chemical form of fuel rod cladding and internal component interaction zones,

- The location, extent and chemical form of control or burnable poison material, cladding and guide tube interaction zones,

- The extent and chemical form of control or burnable poison material interaction with adjacent fuel rods,

- The prior peak temperature of the control, burnable poison or fuel rods.

The CSNI examination program should include nondestructive, metallurgical and radiochemical methods to acquire the following information:

1. Nondestructive examination information:

   a. External appearance (color, surface texture, blemishes, and deformities) of internal components separated from the cladding or guide tubes.

   Suggested method: color photography and magnification

2. Metallurgical examination information:

   a. Extent of oxidation, material interactions and gross surface depositions, peak temperatures and fission product location.

   Suggested methods:

   - Optical metallography/microscopy of metallographic specimens of the rod cross sections.

   - SEM-EDX and/or SEM/WDX examination of fuel rod cladding, guide tube-cladding-poison component interaction zones, and if found, burnable poison pellets.

   - SAS examination of fuel rod cladding-internal components interaction zones, fuel rod cladding and guide tube-cladding-poison component interaction zones,

- SIMS/microprobe examination of all components for fission product location.

3. Radiochemical examination information:

   a. Gamma-emitting fission product abundance and location.

      Suggested methods:

      - Gamma spectrometer analysis of dissolved samples of fuel pellets, fuel rod cladding-internal component interaction zones, guide tube cladding poison component interaction zones, fuel rod cladding external and internal surface deposits, and guide tube external surface deposits,

      - Micro-gamma spectrometry of metallographic samples from both fuel rods and guide tube-poison rods.

   b. Location and abundance of metallic elements.

      Suggested method: ICP or AAS analyses of dissolved specimens of fuel rod cladding internal component interaction zones, fuel rod cladding, guide tube-cladding-poison component interaction zones, and fuel rod cladding external and internal surface deposits and guide tube internal surface deposits and guide tube external surface deposits.

   c. Sr-90, I-129 and tellurium retention in fuel pellet, fuel rod cladding-fuel pellet interaction zones, cladding internal and external surface deposits and guide tube external surface deposits.

      Suggested methods:

- Chemical separation and liquid scintillation analysis for Sr-90,

- Chemical dissolution, separation, low-energy photon spectrometry irradiation, and neutron activation analysis for I-129,

- Tellurium chemistry.

d. U-235 depletion (burnup) of fuel.

Suggested method: TIMS, alpha spectrometry.

The CSNI laboratory is encouraged to use alternate or supplemental examination methods to derive the TMI-2 accident information from the fuel rod segments.

Upper Vessel Core Debris

Sample Description

Upper vessel debris obtained to date is particulate material of which more than 80% is larger than 1 mm in diameter. Eleven samples were obtained from core locations H8 (central) and E9 (mid-radius) at depths ranging from the surface to 90 cm into the debris bed. The debris samples were sieved into 9 particle size groups from 4 mm to 30 micrometers in size. Table B-2 lists available material and particle sizes of the individual samples. Approximately 30% of each sample was removed for analysis.

Available Sample Analysis Results

Each of the upper debris samples was characterized by EG&G. All information from these evaluations will be made available to the CSNI members. Principal examinations performed to date are: photovisual,

TABLE B-2. AVAILABLE UPPER VESSEL DEBRIS

Particle Size Fraction (μm)	Sample Weight (grams)							
	H8 Surface[a]	H8 56 cm[b]	E9 Surface	E9 8 cm[b]	E9 56 cm[b]	H8 36 cm[b]	H8 70 cm[b]	H8 77 cm[b]
>4000	8.4	42.5	11.0	46.4	38.7	4.9	18.9	3.8
1680-4000	18.5	34.3	--[c]	9.3	32.9	39.6	49.6	35.9
707-1000	10.4	12.4	--[c]	4.2	9.2	28.9	19.3	20.4
297-707	2.1	4.2	--[c]	0.3	4.0	9.0	6.1	10.8
149-297	0.6	0.9	--[c]	--[c]	0.6	6.5	5.0	16.9
74-149	0.3	0.5	--[c]	--[c]	0.4	0.6	0.9	6.0
30-74	0.1	0.1	--[c]	--[c]	0.1	0.2	0.6	3.5
<30	--[c]	0.01	--[c]	--[c]	--[c]	0.01	0.4	0.6

a. Quantities less than 1 gram may have been expended during analysis. Samples listed below are either not available or were used for experimental purposes:

1. H8 - 8 cm
2. E9 - 74 cm
3. E9 - 94 cm

b. Indicates depth into the debris bed.

c. Not measurable.

density, optical metallography, SEM, SAS, gamma spectroscopy, I-129 content, Sr-90 content and elemental composition. Principal results of the debris sample examinations are:

- Estimated average peak temperature for the debris bed is about 2200 K.

- The presence of layered prior molten particles indicates either a prolonged candling sequence or multiple temperature ramps occurred during the fuel liquefaction/relocation period of the accident.

- Significant core constituent depletion (less than or equal to 50% for Zr) occurred in the majority of the debris analyzed indicating significant relocation of some core constituents (i.e., Ag, In, Cd, and Al).

- The majority of particles and aliquots examined contained a mixture of most core constituents, indicating significant mixing and disruption of the original core.

- Significant amounts (70-80% of all Cs-137 and I-129 have relocated from the debris bed.

- Evidence indicates that Ru-106 and Sb-125 have been significantly released from the fuel and may have become associated with structural material components.

- Radionuclides were released from the fuel in the approximate order of volatility of the elemental constituents, except for Ru-106 and Sr-90. Ru-106 release is higher and Sr-90 release is lower than expected probably due to the formation of higher or lower volatile oxide forms.

Information Needs

The purpose of the CSNI examination of the upper vessel debris is to evaluate the physical, metallurgical and radiochemical characteristics of the upper vessel region of the core to obtain information about the TMI accident. Information needs are to evaluate:

- The nature of material stratification and relocation, materials interaction and prior peak temperatures of the materials,

- Radionuclide retention in the debris and to determine the effects of maximum temperature and chemical composition on retention.

- The abundance, location and chemical form of retained fission products,

- The location, extent and chemical form of core constituents in homogeneous debris and at interaction zones,

- Prior peak and average temperatures of the debris bed.

Specific information needs are:

1. Physical information

   Photovisual characterization of the samples is necessary for comparison with work being performed at other laboratories.

2. Metallurgical

   a. Analysis of a representative sampling of the available material for comparison with previous results should include characterization of the large particle size groups (>1 mm) which do not exhibit prior molten behavior. Previous examinations focused on prior molten material and nonrepresentative specimens. Suggested methods include: optical metallography, SEM/WDX, SAS, and electron microprobe.

   b. Upper debris smaller than 1 mm should be characterized as a function of particle size to better determine the composition of these materials and to evaluate potential mechanisms for the formation of aerosols. Suggested methods include: SEM EDX/WDX, electron microprobe, and SAS.

3. Radiochemical

a. Evaluate the retention of fission products in the debris as functions of burnup and chemical composition. Suggested methods include: gamma spectroscopy, radiochemistry, TIMS, electron microprobe, and ICP.

b. Analyze representative bulk and individual particle samples of the debris for radionuclide and fission gas content for comparison with previous analyses. Suggested methods include: gamma spectroscopy, radiochemistry, TIMS electron microprobe and high temperature furnaces.

c. Evaluate radiotellurium retention in bulk samples and individual particles. Suggested methods are TIMS, SIMS, and electron microprobe.

Lower Vessel Core Debris

Sample Description

Lower vessel debris already received or to be acquired are prior molten material fragments ranging up to 6 cm in diameter and two particulate material samples. Eleven fragments were recovered from the southern and southwestern quadrants of the debris layer resting on the lower reactor head. Table B-3 lists the dimensions, weights, matrix density, and radiation fields associated with each of the molten material fragments. Approximately 0.5 cm thick wafers were then removed from the center of the particles for analysis. Both the bulk samples and portions of the individual wafers are available for analysis. The two particulate samples will be acquired in late FY 86.

Available Sample Analysis Results

Each of the wafers removed from the bulk samples was characterized by EG&G. All information from these evaluations will be made available to the

TABLE B-3. TMI LOWER VESSEL PARTICLES

Particle Number	Size (in.)	Radiation Measurements Beta/Gamma (R/hr)	Gamma (R/hr)	Dry Weight W (g)	Matrix Density (g/cc)
7-1	1.2 x 1.0 x 0.8	13	1.6	50.1	6.57
7-6	0.4 x 0.2 x 0.2	1.2	0.13	1.0	--[a]
7-7	0.2 x 0.2 x 0.1	0.8	0.10	0.4	--[a]
11-1	1.5 x 0.7 x 0.6	12	1.2	39.7	8.25
11-2	1.8 x 1.3 x 1.2	26	3.0	123.9	6.94
11-4	1.8 x 1.0 x 1.0	25	2.9	107.1	6.75
11-5	2.5 x 2.5 x 2.2	>50	7.5	553.9	6.60
		42[a]	5.5[a]	--	--
11-6	0.7 x 0.7 x 0.4	4	0.5	12.7	6.7
11-7	1.6 x 1.2 x 1.1	30	3.2	118.8	7.18
11-10	0.4 x 0.2 x 0.2	0.7	0.1	0.6	--[a]
11-11	0.5 x 0.5 x 0.5	3.5	0.32	5.5	--[a]

a. Reading taken at 10 in. At 8 in. the detector was offscale (>50 R/h β/γ).

CSNI members. Principal examinations performed were: photovisual, density, optical metallography, SEM EDX, SAS, gamma spectroscopy, I-129 content, Sr-90 content, and elemental composition. Principal results of these examinations are:

- All samples are generally homogeneous prior molten material.
- All samples are from 50-70% uranium and similar in composition to the upper debris samples.
- Estimated retentions of Cs-137 and I-129 range from 9-22% and 0.6 to 8% respectively.
- Corresponding concentration gradients for Cs-137 and I-129 were observed across particle 11-5-C. The concentration gradients were approximately a factor of 40 for cesium and 15 for iodine.

- Metallurgical examination indicates the presence of a fragment of fuel in a zone of a prior molten fuel/zircaloy mixture.

- There are metallic inclusions at the grain boundaries. The principal components are core structural components (Fe, Cr, etc.)

- Retentions of Sb-125 and Ru-106 are significantly lower that the observed retention in the upper debris bed.

Information Needs

The purpose of the CSNI examination of the lower vessel debris is to evaluate the physical, metallurgical, and radiochemical characteristics of materials relocated to the lower vessel of the reactor to obtain information to better define the accident. Information needs are to evaluate:

- The nature of material stratification and relocation, materials interaction and prior peak temperatures of the materials,

- Radionuclide retention in the debris and to determine the effects of maximum temperature and of chemical composition on retention.

- The abundance, location, and chemical form of fission products.

- The location, extent, and chemical form of core constituents in homogeneous debris and at interaction zones and grain boundaries.

- Prior peak and average temperatures of the debris.

Specific Information needs are:

1. Physical information

   a. Photovisual characterization of both fragment and particulate samples is necessary for comparison with work being performed at other laboratories.

2. Metallurgical

   a. Fragments and particulate analysis of a representative sampling of the available material for comparison with previous results. Examinations should focus on material interaction, deposition in the grain boundaries and peak temperatures. Suggested methods include optical metallography, SEM EDX/WDX, SAS, electron microprobe and SIMS.

   b. Particle 11-5-C exhibits a fission product gradient across the particle. Characterization of chemical composition, oxygen content, temperature, and microstructure is requested to determine the cause of the gradient. Suggested methods include: SEM EDX/WDX, SAS, electron microprobe and SIMS.

3. Radiochemical

   a. Evaluate the retention of fission products (including fission gases) in the debris as functions of burnup and chemical composition. Suggested methods include: gamma spectroscopy, radiochemistry, TIMS.

   b. Evaluate the observed Cs-137 and I-129 gradients across particle 11-5-C us functions of chemical composition. Suggested methods include electron microprobe, SAS, SIMS, gamma spectroscopy, all high temperature frames.

APPENDIX C

TMI-2 CORE POSITION COMPONENT IDENTIFICATION MARKING

(GPUN Technical Bulletin 86)

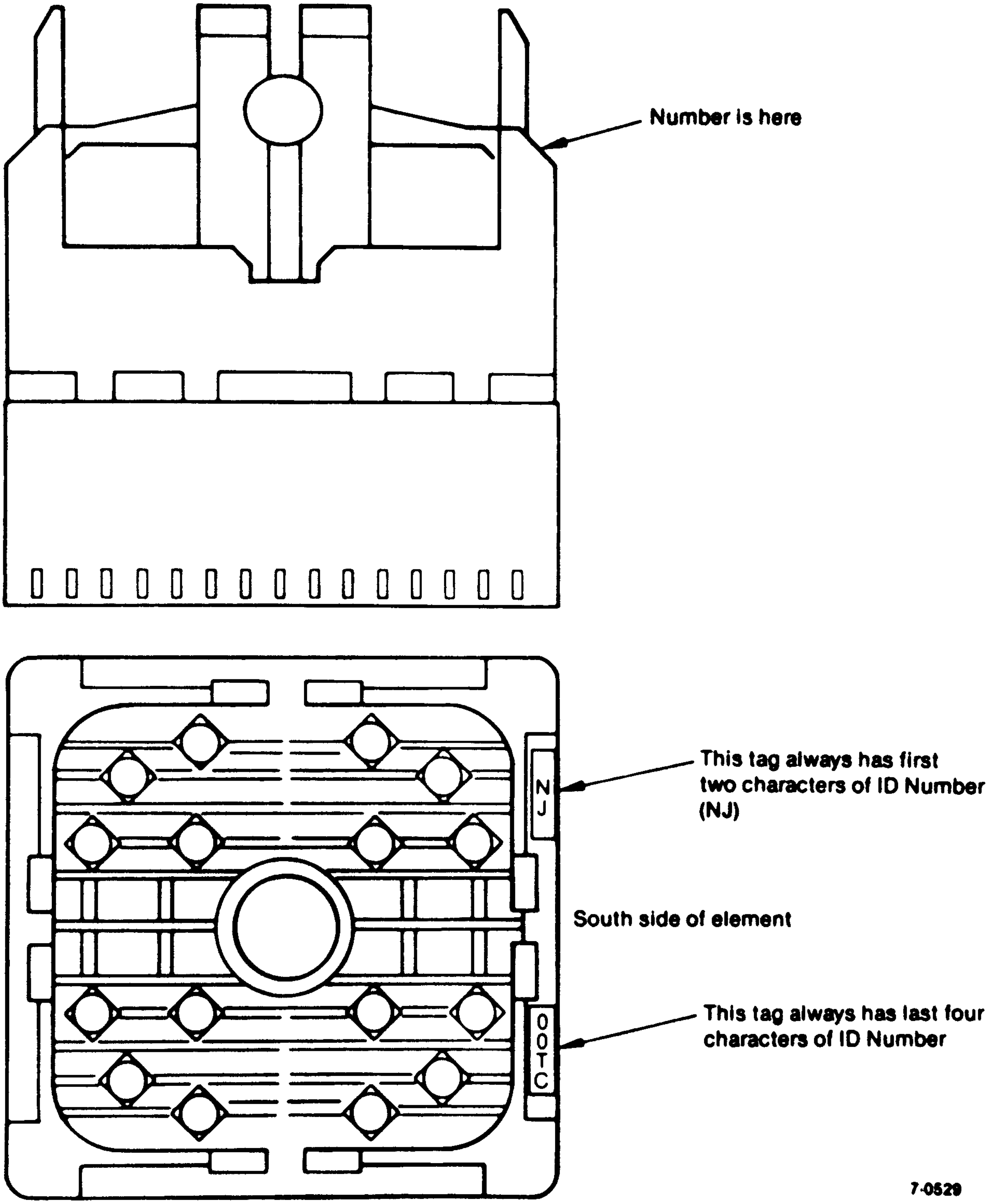

Figure C-1. Fuel element ID number orientation.

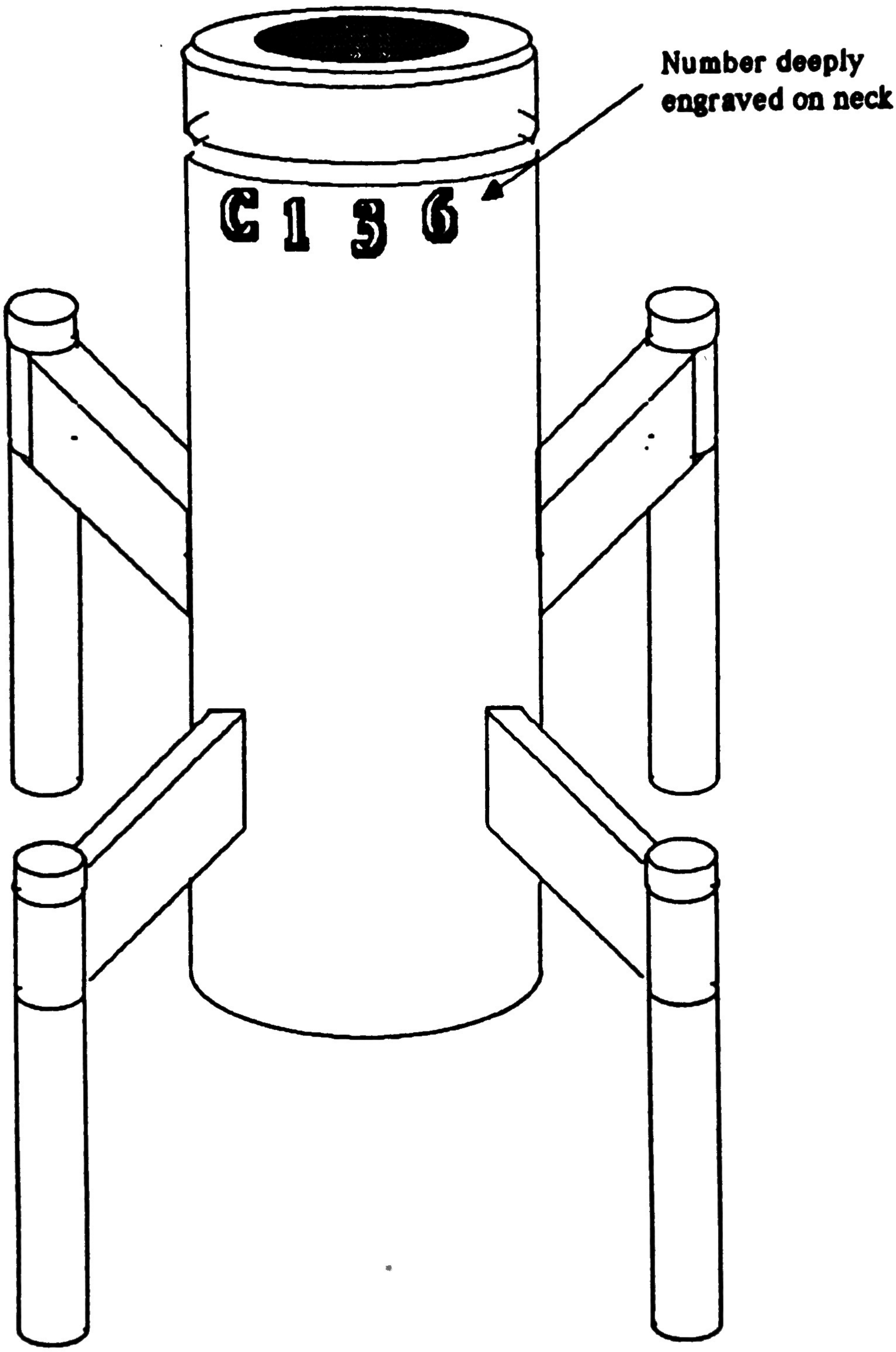

Figure C-2. Control element ID number orientation.

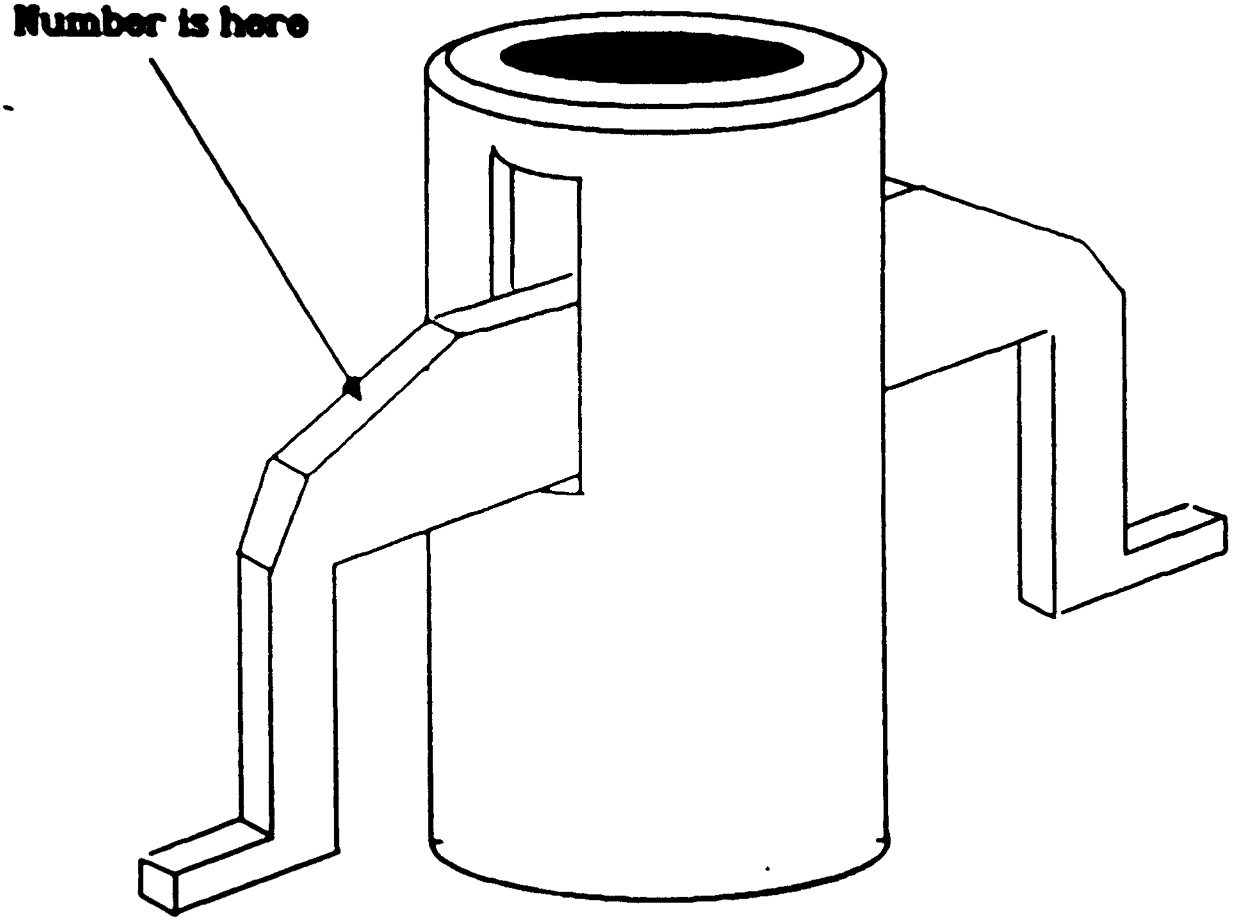

Figure C-3. Burnable poison rod assembly retainer.

SUBJECT: IDENTIFICATION NUMBERS ON CORE COMPONENTS

REFERENCES:

SUMMARY:

Attached is a summary of the identification numbers for fuel elements, control elements, and BPRA retainers. Also included are figures showing the location of these numbers on the component, and the original orientation of the component in the core grid array.

DISCUSSION:

This information was extracted from TMI-2 start-up information and was provided by Joe McCarthy of GPUN Fuel Management in Parsippany. It is provided for purposes of identifying the original location and orientation of components picked up during defueling.

IMPLICATIONS & USE:

This information can be used by defueling operators and others to determine, for any components in the core, where it was located before the accident and how it was oriented in that location. For items which have the identifying mark, visual observation will allow a more complete accounting of canister or sample container contents for the permanent record.

ATTACHMENTS:

- Table 1. Fuel Element Identification vs. Core Grid Position
- Table 2. Control Element Identification
- Table 3. BPRA Retainer Identification
- Figure 1. Fuel Element ID Number Orientation
- Figure 2. Control Element ID Number Orientation
- Figure 3. Burnable Poison Rod Assembly Retainer ID

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APPROVED BY: R. H. Fillnow 8621

MANAGER, P P & A

ID of Fuel Element	Core Grid Location	ID of Fuel Element	Core Grid Location	ID of Fuel Element	Core Grid Location
NJ00E3	M7	NJ00QP	L4	NJ00RS	F7
NJ00PM	M4	NJ00QQ	F12	NJ00RT	K8
NJ00PN	E4	NJ00QR	M3	NJ00RU	K4
NJ00PP	E12	NJ00QS	E3	NJ00RV	L11
NJ00PQ	N11	NJ00QT	M13	NJ00RW	F3
NJ00PS	N12	NJ00QU	N10	NJ00RX	E6
NJ00PT	D12	NJ00QV	G3	NJ00RY	M10
NJ00PU	N4	NJ00QW	L10	NJ00RZ	N9
NJ00PV	D4	NJ00QX	G7	NJ00S0	G4
NJ00PW	C5	NJ00QY	H6	NJ00S1	D9
NJ00PX	O11	NJ00QZ	F8	NJ00S2	H3
NJ00PY	C11	NJ00R1	M9	NJ00S3	L9
NJ00PZ	E13	NJ00R2	G5	NJ00S4	F9
NJ00Q0	O5	NJ00R4	K5	NJ00S6	M8
NJ00Q1	N6	NJ00R5	O9	NJ00S7	F11
NJ00Q2	D10	NJ00R6	F6	NJ00S8	M6
NJ00Q3	L12	NJ00R7	O7	NJ00S9	O6
NJ00Q4	D6	NJ00R8	E11	NJ00SA	L13
NJ00Q5	F4	NJ00R9	M11	NJ00SB	K12
NJ00Q6	M5	NJ00RA	H12	NJ00SC	C10
NJ00Q7	K3	NJ00RB	N8	NJ00SG	N5
NJ00Q8	C7	NJ00RC	D8	NJ00SH	H7
NJ00Q9	K13	NJ00RD	H4	NJ00SJ	H9
NJ00QA	G13	NJ00RE	P7	NJ00SK	L7
NJ00QB	C9	NJ00RF	B9	NJ00SL	K10
NJ00QC	K11	NJ00RG	L3	NJ00SM	G10
NJ00QD	K7	NJ00RH	F13	NJ00SN	C6
NJ00QE	G9	NJ00RJ	K14	NJ00SP	P9
NJ00QF	H10	NJ00RK	G14	NJ00SQ	G2
NJ00QG	E7	NJCORL	B7	NJ00SR	L5
NJ00QH	G11	NJ00RM	K2	NJ00SS	G12
NJ00QJ	E9	NJ00RN	F5	NJ00ST	O8
NJ00QK	F10	NJ00RO	L8	NJ00SU	O10
NJ00QL	L6	NJ00RP	G8	NJ00SV	N7
NJ00QM	E5	NJ00RQ	G6	NJ00SW	D7
NJ00QN	K9	NJ00RR	H8	NJ00SX	H13

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ID of Fuel Element	Core Grid Location	ID of Fuel Element	Core Grid Location
NJ00SY	C8	NJ00U1	B11
NJ00SZ	E8	NJ00U2	E2
NJ00T0	E10	NJ00U3	R9
NJ00T1	H5	NJ00U4	B12
NJ00T2	H11	NJ00U5	E14
NJ00T3	K6	NJ00U6	P5
NJ00T4	D11	NJ00U7	P11
NJ00T5	D5	NJ00U8	O3
NJ00T6	M12	NJ00U9	N3
NJ00T7	C4	NJ00UA	B10
NJ00T8	D13	NJ00UB	B8
NJ00T9	O12	NJ00UC	A8
NJ00TA	D3	NJ00UC	B6
NJ00TB	C12	NJ00UD	R8
NJ00TC	A9	NJ00UE	H14
NJ00TD	D14	NJ00UF	H2
NJ00TE	N14	NJ00UH	N13
NJ00TF	A6	NJ00UJ	O4
NJ00TF	C13	NJ00UK	L2
NJ00TG	F15	NJ00UL	L14
NJ00TH	R6	NJ00UM	P8
NJ00TJ	L15	NJ00UN	P10
NJ00TK	R10	NJ00UO	M2
NJ00TL	R7	NJ00UP	P6
NJ00TM	G1	NJ00UQ	F14
NJ00TN	F1	NJ00UR	F2
NJ00TQ	L1	NJ00US	G15
NJ00TR	D2	NJ00UT	B4
NJ00TS	C3	NJ00UU	H1
NJ00TT	O13	NJ00UV	K15
NJ00TU	M14	NJ00UW	K1
NJ00TV	P4	NJ00UX	B5
NJ00TW	A10	NJ00UY	H15
NJ00TX	P12		
NJ00TY	A7		
NJ00TZ	N2		

Control Element	ID of Cont. Element	Core Grid Location	Control Element	ID of Cont. Element	Core Grid Location
SU SOURCE	03AT	B12	BPRA	B166	O4
SU SOURCE	03AU	P4	BPRA	B167	O6
APSRA	A017	D10	BPRA	B168	D3
APSRA	A018	F12	BPRA	B169	D13
APSRA	A019	L12	BPRA	B170	O12
APSRA	A020	N10	BPRA	B171	F3
APSRA	A021	N6	BPRA	B172	O8
APSRA	A022	L4	BPRA	B173	C10
APSRA	A023	F4	BPRA	B174	C6
APSRA	A024	D6	BPRA	B175	N3
BPRA	B139	G10	BPRA	B176	G12
BPRA	B140	P9	BPRA	B177	C8
BPRA	B141	K6	BPRA	B178	O10
BPRA	B142	H9	BPRA	B179	K12
BPRA	B143	H7	BPRA	B180	N7
BPRA	B144	K8	BPRA	B181	N9
BPRA	B145	G6	BPRA	B182	F13
BPRA	B146	F7	BPRA	B183	D9
BPRA	B147	G2	BPRA	B184	H13
BPRA	B148	P7	BPRA	B185	G4
BPRA	B149	G8	BPRA	B186	H3
BPRA	B150	L9	BPRA	B187	E8
BPRA	B151	K14	BPRA	B188	H11
BPRA	B152	K2	BPRA	B189	H5
BPRA	B153	B7	BPRA	B190	N13
BPRA	B154	K10	BPRA	B191	M10
BPRA	B155	F9	BPRA	B192	L5
BPRA	B156	L7	BPRA	B193	M4
BPRA	B157	G14	BPRA	B194	M6
BPRA	B158	B9	BPRA	B195	E6
BPRA	B159	M8	BPRA	B196	N11
BPRA	B160	D7	BPRA	B197	D5
BPRA	B161	K4	BPRA	B198	E4
BPRA	B162	L3	BPRA	B199	L11
BPRA	B163	C12	BPRA	B200	D11
BPRA	B164	C4	BPRA	B201	F11
BPRA	B165	L13	BPRA	B202	F5

Control Element	ID of Cont. Element	Core Grid Location	Control Element	ID of Cont. Element	Core Grid Location
BPRA	B203	N5	CRA	C156	C11
BPRA	B204	E10	CRA	C157	E13
BPRA	B205	M12	CRA	C158	M13
BPRA	B206	E12	CRA	C159	O11
CRA	C123	H8	CRA	C160	O5
CRA	C124	B8	CRA	C161	M3
CRA	C125	H14	CRA	C162	E3
CRA	C126	P8	CRA	C163	C5
CRA	C127	H2	CRA	C164	E9
CRA	C128	E11	CRA	C165	G11
CRA	C129	M11	CRA	C166	K11
CRA	C130	M5	CRA	C167	M9
CRA	C131	E5	CRA	C168	M7
CRA	C132	B10	CRA	C169	K5
CRA	C133	F14	CRA	C170	G5
CRA	C134	L14	CRA	C171	E7
CRA	C135	P10	CRA	C172	C9
CRA	C136	P6	CRA	C173	G13
CRA	C137	L2	CRA	C174	K13
CRA	C138	F2	CRA	C175	O9
CRA	C139	B6	CRA	C176	O7
CRA	C140	F8	CRA	C177	K3
CRA	C141	H10	CRA	C178	G3
CRA	C142	L8	CRA	C179	C7
CRA	C143	H6	CRA	C180	G9
CRA	C144	D8	CRA	C181	K9
CRA	C145	H12	CRA	C182	K7
CRA	C146	N8	CRA	C183	G7
CRA	C147	H4			
CRA	C148	F10			
CRA	C149	L10			
CRA	C150	L6			
CRA	C151	F6			
CRA	C152	D12			
CRA	C153	N12			
CRA	C154	N4			
CRA	C155	D4			

Control Element	ID of Retainer	Core Grid Location
SU SOURCE	L004	B12
SU SOURCE	L005	P4
BPRA	L007	B7
BPRA	L008	B9
BPRA	L009	C4
BPRA	L010	C6
BPRA	L011	C8
BPRA	L012	C10
BPRA	L013	C12
BPRA	L014	D3
BPRA	L015	D5
BPRA	L016	D7
BPRA	L017	D9
BPRA	L018	D11
BPRA	L019	D13
BPRA	L020	E4
BPRA	L021	E6
BPRA	L022	E8
BPRA	L023	E10
BPRA	L024	E12
BPRA	L025	F3
BPRA	L026	F5
BPRA	L027	F7
BPRA	L028	F9
BPRA	L029	F11
BPRA	L030	F13
BPRA	L031	G2
BPRA	L032	G4
BPRA	L033	G6
BPRA	L034	G8
BPRA	L035	G10
BPRA	L036	G12
BPRA	L037	G14
BPRA	L038	H3
BPRA	L039	H5
BPRA	L040	H7
BPRA	L041	H9
BPRA	L042	H11
BPRA	L043	H13
BPRA	L044	K2
BPRA	L045	K4
BPRA	L046	K6
BPRA	L047	K8
BPRA	L048	K10
BPRA	L049	K12
BPRA	L050	K14
BPRA	L051	L3
BPRA	L052	L5
BPRA	L053	L7
BPRA	L054	L9
BPRA	L055	L11
BPRA	L056	L13
BPRA	L057	M4
BPRA	L058	M6
BPRA	L059	M8
BPRA	L060	M10
BPRA	L061	M12
BPRA	L062	N3
BPRA	L063	N5
BPRA	L064	N7
BPRA	L065	N9
BPRA	L066	N11
BPRA	L067	N13
BPRA	L068	O4
BPRA	L069	O6
BPRA	L070	O8
BPRA	L071	O10
BPRA	L072	O12
BPRA	L073	P7
BPRA	L074	P9